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Status of non-electric nuclear heat applications:

Technology and safety

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FOREWORD

Nuclear energy plays an important role in electricity generation, producing 16% of the world's electricity at the beginning of 1999. It has proven to be safe, reliable, economical and has only a minimal impact on the environment. Most of the world's energy consumption, however, is in the form of heat. The market potential for nuclear heat was recognized early. Some of the first reactors were used for heat supply, e.g. Calder Hall (United Kingdom), Obninsk (Russian Federation), and Agesta (Sweden). Now, over 60 reactors are supplying heat for district heating, industrial processes and seawater desalination. But the nuclear option could be better deployed if it would provide a larger share of the heat market. In particular, seawater desalination using nuclear heat is of increasing interest to some IAEA Member States.

In consideration of the growing experience being accumulated, the IAEA periodically reviews the progress and new developments in the field of nuclear heat applications. This publication summarizes the recent activities among Member States presented at a Technical Committee meeting in April 1999. The purpose of the meeting was to provide a forum for the exchange of up to date information on the prospect, design, safety and licensing aspects, and development of non-electrical applications of nuclear heat for industrial use. This mainly included seawater desalination and hydrogen production.

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EDITORIAL NOTE

CONTENTS

SUMMARY

1. INTRODUCTION

Since the early days of nuclear power development, the direct use of heat generated in reactors has been widely practised and expanding. In addition to the forerunners, UK and Sweden, many other countries have found it convenient to apply nuclear heat for district heating or for industrial processes, or for both, in addition to electricity generation [1, 2]. They include Bulgaria, Canada, China, the Czech Republic, Germany, Hungary, India, Japan, Kazakhstan, the Russian Federation, Slovakia, Sweden, Switzerland and Ukraine. Though less than 1% of the heat generated in nuclear reactors worldwide is at present used for district and process heating, its operating experience exceeds about 600 reactor-years[1] and there are signs of increasing interest in these applications[2].

The direct use of nuclear heat is nothing new. After all, the result of the nuclear fission process is the generation of heat within the reactor. The heat is removed by the coolant circulating through the core, that can then be applied to the generation of electricity or used in providing hot water or steam for industrial or space heating purposes. There are, however, substantial differences between the properties and applications of electricity and of heat, as well as between the markets for these different forms of energy. These differences as well as the intrinsic characteristics of nuclear reactors are the reasons why nuclear power has predominantly penetrated the electricity market and had relatively minor applications as a direct heat source.

About 33% of the world's total energy consumption is currently used for electricity generation. This share is steadily increasing and is expected to reach 40% by the year 2015. Of the rest, heat consumed for residential and industrial purposes and the transport sector constitute the major components, with the residential and industrial sectors having a somewhat larger share. Practically the entire heat market is supplied by burning coal, oil, gas, or wood.

Overall energy consumption is steadily increasing and this trend is expected to continue well into the next century. Conservation and efficiency improvement measures have in general reduced the rate of increase of energy consumption, but their effect is not large enough to stabilize consumption at current values.

A modest increase in the generation of nuclear electricity is expected during the next two decades. In the transport sector, practically no application of nuclear energy is foreseen, except indirectly through the increased use of electricity.

The heat market is an open challenge. Though nuclear energy has been used to supply a portion of the heat demand, it has not yet achieved significant penetration. How far and how fast it could capture part of this market will depend mainly on how the characteristics of nuclear reactors can be matched with the characteristics of the heat market, in order to successfully compete with alternative energy sources.

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[1] A list of nuclear plants providing heat for non-electrical products is given as Tables I, II and III. The IAEA is extending the current Power Reactor Information System (PRIS) database to include the information of nuclear heat application at these plants. Phase 1 of this extension is in progress to accommodate information at co-generation nuclear power plants, since the information of the nuclear energy source is readily available in the current PRIS database. Out of about 60 nuclear reactors providing heat for non-electrical products, about 50 plants have provided the IAEA with the design characteristics data. Upon completion of Phase 1 of PRIS extension, the inclusion of heat-only reactors and some of HTGRs will be planned.

[2] Planned projects known to the IAEA are listed in Tables IV, V and VI.

TABLE I. OPERATING NUCLEAR HEATING PLANTS

Country	Code	Unit name	Location	Application[a]	Phase	Start of operation Power	/ Heat	Power (MW(e) net)	Heat output (MW(th))	°C at Interface (Feed/Return)	Remarks
Bulgaria	BG-5	Kozloduy 5	Kozloduy	E, DH	Commercial	1988.9	1990?	953	20	150/70	
Bulgaria	BG-6	Kozloduy 6	Kozloduy	E, DH	Commercial	1993.12	1990?	953	20	150/70	
China		NHR-5	Beijing	DH	Experiment	1989	1989	0	5	90/60	
Hungary	HU-2	PAKS 2	Paks	E, DH	Commercial	1984.11		433	30	130/70	
Hungary	HU-3	PAKS 3	Paks	E, DH	Commercial	1986.12		433	30	130/70	
Hungary	HU-4	PAKS 4	Paks	E, DH	Commercial	1987.11		433	30	130/70	
Russia		Research reactor	Obninsk	DH	Commercial	1954	1976	0	10-20	130/70	
Russia	RU-141	Bilibino 1	Bilibino	E, DH	Commercial	1974.4	1974.4	11	29	150/80	
Russia	RU-142	Bilibino 2	Bilibino	E, DH	Commercial	1975.2	1975.2	11	29	150/80	
Russia	RU-143	Bilibino 3	Bilibino	E, DH	Commercial	1976.2	1976.2	11	29	150/80	
Russia	RU-144	Bilibino 4	Bilibino	E, DH	Commercial	1977.1	1977.1	11	29	150/80	
Russia	RU-9	Novovoronezh 3	Novovoronezh[b]	E, P,DH	Commercial	1972.6	1972.6	385	38	105/70	
Russia	RU-11	Novovoronezh 4	Novovoronezh	E, P,DH	Commercial	1973.3	1973.3	385	38	105/70	
Russia	RU-96	Balakovo 1	Balakovo	E, DH	Commercial	1986.5	1986.5	950	233	130/70	
Russia	RU-97	Balakovo 2	Balakovo	E, DH	Commercial	1988.1	1988.1	950	233	130/70	
Russia	RU-98	Balakovo 3	Balakovo	E, DH	Commercial	1989.4	1989.4	950	233	130/70	
Russia	RU-99	Balakovo 4	Balakovo	E, DH	Commercial	1993.12	1993.12	950	233	130/70	
Russia	RU-31	Kalinin 1	Udomlya	E, P,DH	Commercial	1985.6	1985.6	950	93	150/70	
Russia	RU-32	Kalinin 2	Udomlya	E, P,DH	Commercial	1987.3	1987.3	950	93	150/70	
Russia	RU-12	Kola 1	Polyarnie Zory	E, P,DH	Commercial	1973.12	1973.12	411	29	130/70	
Russia	RU-13	Kola 2	Polyarnie Zory	E, P,DH	Commercial	1975.12	1975.12	411	29	130/70	
Russia	RU-32	Kola 3	Polyarnie Zory	E, P,DH	Commercial	1982.12	1982.12	411	58	130/70	
Russia	RU-33	Kola 4	Polyarnie Zory	E, P,DH	Commercial	1984.12	1984.12	411	29	130/70	
Russia	RU-21	Belojarsk 3 (BN-600)	Zarechny	E, P,DH	Commercial	1981.11	1981.11	560	198	130/70	
Russia	RU-15	Leningrad 1	Sosnovy Bor	E, P,DH	Commercial	1974.11	1974.11	925	29	130/70	
Russia	RU-16	Leningrad 2	Sosnovy Bor	E, P,DH	Commercial	1976.2	1976.2	925	29	130/70	
Russia	RU-34	Leningrad 3	Sosnovy Bor	E, P,DH	Commercial	1980.6	1980.6	925	29	130/70	
Russia	RU-35	Leningrad 4	Sosnovy Bor	E, P,DH	Commercial	1981.8	1981.8	925	29	130/70	
Russia	RU-17	Kursk 1	Kurchatov	E, P,DH	Commercial	1977.10	1977.10	925	148	130/70	
Russia	RU-22	Kursk 2	Kurchatov	E, P,DH	Commercial	1979.8	1979.8	925	148	130/70	
Russia	RU-38	Kursk 3	Kurchatov	E, P,DH	Commercial	1984.3	1984.3	925	204	130/70	
Russia	RU-39	Kursk 4	Kurchatov	E, P,DH	Commercial	1986.2	1986.2	925	204	130/70	
Russia	RU-23	Smolensk 1	Desnogorsk	E, P,DH	Commercial	1983.9	1983.9	925	201	130/70	

[a] E: Electricity (Power), P: Steam supply for process heat, DH: Steam/hot water supply for heating (partly consumed at the end?).
[b] Unit 1 was taken out of operation in 1988. Unit 2 was taken out of operation in 1990.

TABLE I. (cont.)

Russia	RU-24	Smolensk 2	Desnogorsk	E, P,DH	Commercial	1985.7	1985.7	925	201	130/70	
Russia	RU-67	Smolensk 3	Desnogorsk	E, P,DH	Commercial	1990.1	1990.1	925	201	130/70	
Slovakia	SK-13	Bohunice 3	Bohunice/Trnava	E, DH	Commercial	1985.2	1987	408	240	150/70	
Slovakia	SK-14	Bohunice 4	Bohunice/Trnava	E, DH	Commercial	1985.12	1987	408	240	150/70	
Switzerland	CH-1	Beznau 1	Beznau	E, DH	Commercial	1969.9	1983	2×360	80	128/50 (water)	
Switzerland	CH-2	Beznau 1	Beznau	E, DH	Commercial	1971.12	1984	2×360	80	128/50 (water)	
Ukraine	UA-27	Rovno 1	Kuznetzovsk	E, DH	Commercial	1981.9	1981.1	363	58	130/70	
Ukraine	UA-28	Rovno 2	Kuznetzovsk	E, DH	Commercial	1982.7	1982.1	377	58	130/70	
Ukraine	UA-29	Rovno 3	Kuznetzovsk	E, DH	Commercial	1987.5	1987.1	950	233	130/70	
Ukraine	UA-54	Zaporozhe 1		E, DH	Commercial	1985.12	1985	950	232	150/70	
Ukraine	UA-56	Zaporozhe 2		E, DH	Commercial	1986.02	1985	950	232	150/70	
Ukraine	UA-78	Zaporozhe 3		E, DH	Commercial	1987.3	1985	950	232	150/70	
Ukraine	UA-79	Zaporozhe 4		E, DH	Commercial	1988.4	1985	950	232	150/70	
Ukraine	UA-126	Zaporozhe 5		E, DH	Commercial	1989.10	1985	950	232	150/70	
Ukraine	UA-127	Zaporozhe 6		E, DH	Commercial	1996.9	1985	950	232	150/70	
Ukraine	UA-44	South Ukraine 1	Yuzhnoukrainsk	E, DH	Commercial	1983.10	1976?	950	151	150/70	
Ukraine	UA-45	South Ukraine 2	Yuzhnoukrainsk	E, DH	Commercial	1985.4	1976?	950	151	150/70	
Ukraine	UA-48	South Ukraine 3	Yuzhnoukrainsk	E, DH	Commercial	1989.12	1976?	950	232	150/70	

TABLE II. OPERATING NUCLEAR DESALINATION PLANTS[a]

Country	Code	Unit name	Location	Application[b]	Phase	Start of operation Power / Heat		Power (MW(e) net)	Water Capacity (m^3/day)	Desalination Process	Remarks
Japan	JP-23	Ikata-1	Ehime	E, WD	Commercial	1977.6	1976	538	2000	MSF	
Japan	JP-32	Ikata-2	Ehime	E, WD	Commercial	1982.3	1976	538			
Japan	JP-15	Ohi-1	Fukui	E, WD	Commercial	1979.3	1973	1120	3900	MSF	
Japan	JP-19	Ohi-2	Fukui	E, WD	Commercial	1979.12	1976	1120			
Japan	JP-45	Genkai-3	Fukuoka	E, WD	Commercial	1994.3		1127	1000	MED	
Japan	JP-46	Genkai-4	Fukuoka	E, WD	Commercial	1997.7		1127			
Japan	JP-	Takahama	Fukui	E, WD	Commercial				1000	MED	
Japan	JP-33	Kashiwazaki-Kariwa 1	Niigata	E, WD	Commercial	1985.9		1067	1000	MSF[c]	
Kazakhstan	KZ-10	BN-350[d]	Aktau	E, WD, P, DH1	Commercial	1973	1963?	70	120,000	MED/MSF	

[a] All nuclear desalination plants with the exception of Aktau/Kazakstan are for on-site water supply. Some desalination plants were first operated with conventional energy.

[b] E: Electricity (Power), WD: Water desalination (Unless otherwise mentioned, seawater desalination), P: Industrial process heat.

[c] This desalination facility was not put into service after construction, because other fresh water resources were made available.

[d] BN-350 was closed in 1999. The desalination unit is still in service with heat from the alternative boiler.

TABLE III. NUCLEAR PROCESS HEAT PRODUCTION PLANTS

Country	Code	Unit name	Location	Application[a]	Phase	Operation Period (Start-close) Power / Heat		Power (MW(e) net)	Heat delivery (MW(th))	Product and production	°C at Interface (Feed/Return)	Remarks
Canada	CA-8	Bruce-1	Bruce	E, P	Operation suspended	1977.9–1997.10	1981-	848		D2O, Agroindustry		
Canada	CA-9	Bruce-2	Bruce	E, P		1977.9–1995.10		848				
Canada	CA-10	Bruce-3	Bruce	E, P		1978.2-–998.4		848				
Canada	CA-11	Bruce-4	Bruce	E, P		1979.1–1998.3		848				
Germany	DE-10	Stade	Stade	E, P	Commercial	1972.5–		640	30	Salt refinery	190/100	
Japan		HTTR	O-arai	P	Power up	1998 Critical		30	30	in planning		
Russia	RU-21	Belojarsk 3 (BN-600)	Zarechny	E, DH, P	Commercial	1981.11	1981.11	560	42		2730	Feed Enthalpy (Kj/Kg)
Russia	RU-12	Kola 1	Polyarnie Zory	E, DH, P	Commercial	1973.12	1973.12	411	4.2		2700	(Kj/Kg)
Russia	RU-13	Kola 2	Polyarnie Zory	E, DH, P	Commercial	1975.12	1975.12	411	4.2		2700	(Kj/Kg)
Russia	RU-32	Kola 3	Polyarnie Zory	E, DH, P	Commercial	1982.12	1982.12	411	4.2		2700	(Kj/Kg)
Russia	RU-33	Kola 4	Polyarnie Zory	E, DH, P	Commercial	1984.12	1984.12	411	4.2		2700	(Kj/Kg)
Russia	RU-17	Kursk 1	Kurchatov	E, DH, P	Commercial	1977.10	1977.10	925	16		2743	(Kj/Kg)
Russia	RU-22	Kursk 2	Kurchatov	E, DH, P	Commercial	1979.8	1979.8	925	16		2743	(Kj/Kg)
Russia	RU-38	Kursk 3	Kurchatov	E, DH, P	Commercial	1984.3	1984.3	925	16		2743	(Kj/Kg)
Russia	RU-39	Kursk 4	Kurchatov	E, DH, P	Commercial	1986.2	1986.2	925	16		2743	(Kj/Kg)
Russia	RU-15	Leningrad 1	Sosnovy Bor	E, DH, P	Commercial	1974.11	1974.11	925	19		2712	(Kj/Kg)
Russia	RU-16	Leningrad 2	Sosnovy Bor	E, DH, P	Commercial	1976.2	1976.2	925	19		2712	(Kj/Kg)
Russia	RU-34	Leningrad 3	Sosnovy Bor	E, DH, P	Commercial	1980.6	1980.6	925	19		2712	(Kj/Kg)
Russia	RU-35	Leningrad 4	Sosnovy Bor	E, DH, P	Commercial	1981.8	1981.8	925	19		2712	(Kj/Kg)
Russia	RU-9	Novovoronezh 3	Novovoronezh	E, DH, P	Commercial	1972.6	1972.6	385	29		2700	(Kj/Kg)
Russia	RU-11	Novovoronezh 4	Novovoronezh	E, DH, P	Commercial	1973.3	1973.3	385	29		2700	(Kj/Kg)
Russia	RU-20	Novovoronezh 5	Novovoronezh	E, P	Commercial	1981.2	1981.2	950	17		2700	(Kj/Kg)
Russia	RU-23	Smolensk 1	Desnogorsk	E, DH, P	Commercial	1983.9	1983.9	925	19		2743	(Kj/Kg)
Russia	RU-24	Smolensk 2	Desnogorsk	E, DH, P	Commercial	1985.7	1985.7	925	19		2743	(Kj/Kg)
Russia	RU-67	Smolensk 3	Desnogorsk	E, DH, P	Commercial	1990.1	1990.1	925	19		2743	(Kj/Kg)
Switzer-land	CH-4	Goesgen	Goesgen	E, P	Commercial	1979.11–		970	25	Cardboard	220/100	

[a] E: Electricity (Power), P: Industrial process heat

TABLE IV. NUCLEAR DESALINATION PROJECTS

Country	Plant name	Location	Application	Start of operation Reactors / Desal.	Phase	Power (MW(e))	Water Capacity (m^3/day)	Remarks
India	Kalpakkam 1,2	Tamil Nadu	Electricity/ Desalination	Reactors:1984–86 RO:2000/2001 MSF: after 2001	Connection of desalination system	2 × 170	6 300	Hybrid MSF / RO

TABLE V. OPERATING NUCLEAR PROCESS HEAT PRODUCTION PLANTS

Country	Plant name	Location	Application	Start of operation Reactors / Heat	Phase	Power (MW(e))	Heat delivery (MW(th))	°C at Interface (Feed/Return)	Remarks
Canada[a]	Bruce-A	Bruce	Process Heat	1977–87 1981	Laid up in 1998	4 × 848 4 × 860	5350		D2O production and six industrial heat customers
Germany	Stade	Stade	Electricity/ Process Heat	1983	Commercial	640	30	190/100	Salt refinery
Japan	HTTR	O-arai	Process Heat	1998	Commissioning	0	30	950/395	Experiments for HTR technology developmt.
Switzerland	Goesgen	Goesgen	Electricity/ Process Heat	1979 1979	Commercial	970	25	220/100	Cardboard factory

[a] Unit 2 of Bruce A was taken out of service in 1995, units 1, 3 and 4 will be taken out of service in spring 1998.

TABLE VI. NUCLEAR PROCESS HEAT PRODUCTION PROJECTS

Country	Plant name	Location	Application	Start of operation Reactors / Heat	Phase	Power (MW(e))	Heat delivery (MW(th))	°C at Interface (Feed/Return)	Remarks
China	HTGR-10	Beijing	Electricity/ Process Heat	Criticality 1999?	Construction	5	10	700–950/250	Experiments for HTR technology development.
Russia	VGM		Process Heat		Design	0		750–950	

- **Characteristics of the heat market**

Transport of heat is difficult and expensive. The need for a pipeline, thermal isolation, pumping, and the corresponding investments, heat losses, maintenance, and pumping energy requirements make it impractical to transport heat beyond distances of a few kilometers or, at most, some tens of kilometers. There is also a strong size effect. The specific costs of transporting heat increase sharply as the amount of heat to be transported diminishes. Compared to heat, the transport of electricity from where it is generated to the end user is easy and cheap, even to large distances measured in hundreds of kilometers.

The residential and the industrial sectors constitute the two major components of the overall heat market. Within the residential sector, while heat for cooking has to be produced directly where it is used, the demand for space heating can be and is often supplied from a reasonable distance by a centralized heating system through a district heating transmission and distribution network serving a relatively large number of customers.

District heating: District heating networks generally have installed capacities in the range of 600 to 1200 megawatt-thermal (MW(th)) in large cities, decreasing to approximately 10 to 50 MW(th) in towns and small communities. A potential market for district heating only appears in climatic zones with relatively long and cold winters. In western Europe, for example, Finland, Sweden, and Denmark are countries where district heating is widely used. The annual load factors of district heating systems depend on the length of the cold season when space heating is required, and can reach up to about 50%, which is still way below what is needed for base load operation of plants. Also, to assure a reliable supply of heat to the residences served by the district-heating network, adequate backup heat generating capacity must be provided. This implies the need for redundancy and generating unit sizes corresponding to only a fraction of the overall peak load. The temperature range required by district heating systems is around 100 to 150 °C.

In general, the district heating market is expected to expand substantially. Not only because it can compete economically in densely populated areas with individual heating arrangements, but also because it offers the possibility of reducing air pollution in urban areas. While emissions resulting from the burning of fuel can be controlled and reduced up to a point in relatively large centralized plants, this is not practical in small individual heating installations fuelled by gas, oil, coal, or wood.

Industrial processes: Within the industrial sector, process heat is used for a very large variety of applications with different heat requirements and with temperature ranges covering a wide spectrum. While in energy intensive industries the energy input represents a considerable fraction of the final product cost, in most other processes it contributes only a few percent. Nevertheless, the supply of energy has an essential character, i.e., all industrial users need the assurance of energy supply with a very high degree of reliability and availability, approaching 100% in particular for large industrial installations and energy intensive processes.

Regarding the power ranges of the heat sources required, similar patterns are found in most industrialized countries. In general, about half of the users require less than 10 MW(th) and another 40% between 10 and 50 MW(th). There is a steady decrease in the number of users as the power requirements become higher. About 99% of the users are included in the range of less than 300 MW(th), which accounts for about 80% of the total energy consumed. Individual large users with energy intensive industrial processes cover the remaining portion of the industrial heat market with requirements up to 1000 MW(th), and exceptionally even more. This shows the highly fragmented nature of the industrial heat market.

The possibility of large scale introduction of heat distribution systems supplied from a centralized heat source — which would serve several users concentrated in so-called industrial parks — seems rather remote at present, but could be the trend on a long term. Contrary to district heating, the load factors of industrial users do not depend on climatic conditions. The demands of large industrial users usually have base load characteristics.

The temperature requirements depend on the type of industry, covering a wide range up to around 1500°C. The upper range above 1000°C is dominated by the iron/steel industry. The lower range up to about 200 to 300°C includes industries such as seawater desalination, pulp and paper, or textiles. Chemical industries, oil refining, oil shale and sand processing, and coal gasification are examples of industries with temperature requirements of up to the 500 to 600°C level. Refinement of coal and lignite, and hydrogen production by water splitting are among applications that are renewing the interest and they require temperatures between 600 and 1000°C.

- **Characteristics of nuclear heat sources**

There is plenty of experience of using nuclear heat in both district heating and in industrial processes, so the technical aspects can be considered well proven. There are no technical impediments to the application of nuclear reactors as heat sources for district or process heating. In principle, any type and size of nuclear reactor can be used for these purposes.

Potential radioactive contamination of the district heating networks or of the products obtained by the industrial processes is avoided by appropriate precautions, such as intermediate heat transport circuits with pressure gradients, which act as effective barriers. No incident involving radioactive contamination has ever been reported for any of the reactors used for these purposes.

Regarding the temperature ranges, light and heavy water reactors can provide temperatures of up to about 300°C, liquid metal cooled fast reactors up to 540°C, advanced gas cooled reactors up to 650°C, and high temperature gas cooled reactors up to about 1000°C.

For applications to district or process heating, there are basically two options: Co-generation of electricity and heat, and heat-only reactors. Co-generation has been widely applied, while there is not much experience in heat-only reactors. In principle, any amount of heat can be extracted from co-generation reactors, subject to design limitations. Co-generation plants, when forming part of large industrial complexes, can be readily integrated into an electrical grid system to which they supply any surplus electricity generated. In turn, they would serve as a backup for assurance of electricity supply. This means a high degree of flexibility. Heat-only reactors, on the other hand, have only one objective, as they are not intended for generating electricity.

The availability of nuclear reactors is, in general, similar to fossil fuelled power plants. As shown by experience, availability factors of 70% to 80% or even 90% can be achieved. The frequency and duration of unplanned outages can be kept very low with good preventive and predictive maintenance. Availability and reliability of a reactor, however, can never reach the nearly 100% levels required by most large heat users. Multiple unit co-generation power plants, modular designs, or backup heat sources are suitable solutions for redundancy.

Nuclear reactors are capital intensive. The influence of the fixed cost component is predominant in the final cost of energy. Therefore, base load operation with load factors as high as achievable is needed for competition with alternative sources. This is only possible when the demand of the heat market to be supplied has base load characteristics, or when the combined electricity and heat market enables overall base load operation of a co-generation plant.

Nuclear reactors can be technically proven, safe, reliable and environmentally clean energy sources, but for commercial deployment they also have to be economically competitive with alternative energy sources. Compared to fossil fuelled sources, nuclear reactors are characterized by higher investment costs compensated by lower fuel costs. The penetration of nuclear power into the electricity market would not have been possible without having fulfilled the condition of economic competitiveness. Even with prevalent low fossil fuel price levels, nuclear power has retained its competitive position in most parts of the world. Should fossil fuel prices increase, as is expected to occur, the economically competitive position of nuclear power, both for electricity generation and for heat supply, will improve.

Due to the size effect, nuclear economics are, in general, improved for larger units. This has led to the development and predominant deployment of large size reactors in industrialized countries with very large interconnected electrical grid systems. Nevertheless, there has been and there continues to be a market for small and medium sized power reactors (SMRs). Various innovative designs of SMRs have been intensively developed to overcome the so-called scale-demerit for better economic competitiveness.

Siting of nuclear plants is another major issue. For co-generation or heat-only reactors, closer location to the load centers has even a stronger incentive. The NIMBY (not in my back yard) syndrome, however, is an important factor affecting site selection. It promotes a trend to choose remote but accessible locations, in order to avoid potential conflicts and opposition. Remote siting far from densely populated areas makes it also easier to comply with regulatory requirements. Advanced reactor designs, in particular in the SMR range with improved safety features, could be perceived as acceptable for close siting by the public. They also could more easily meet regulatory requirements and could maintain heat transmission costs at reasonable levels.

In nuclear power, unlike in many industrial undertakings, the long term viewpoint is predominant. The planning, design, project preparatory activities, and licensing takes years to be completed for any nuclear reactor. Reactors are designed and built to last for about 40 years or more, and to achieve the economic benefits expected, they have to be operated with high load factors during their economic lifetime. There are also infrastructure requirements, which require time and considerable development efforts, if not already available. These efforts are only justifiable under a long term perspective directed to a nuclear programme.

- **Prospects for nuclear heat applications**

The technical viability of employing nuclear heat sources for district heating or for industrial processes has existed since the very start of nuclear development. A substantial penetration into the commercial heat market, however, has not yet taken place. Prospects will mainly depend on where and how the demand characteristics of the heat market can be matched by what nuclear reactors are able to offer.

District heating: For the district heating market, co-generation nuclear power plants are one of the supply options. In the case of medium to large nuclear reactors, due to the limited power requirements of the heat market and the relatively low load factors, electricity would be the main product, with district heating accounting for only a small fraction of the overall energy produced. These reactors, including their siting, would be optimized for the conditions pertaining to the electricity market, district heating being, in practice, a by-product. Should such power plants be located close enough to population centers in cold climatic regions, they could also serve district heating needs. This has been done in Russia, Ukraine, the Czech Republic, Slovakia, Hungary, Bulgaria, and Switzerland, using up to about 100 MW(th) per power station. Similar applications can be expected for the future wherever similar boundary conditions exist.

For small co-generation reactors corresponding to power ranges of up to 300 MW(e) and 150 MW(e), respectively, the share of heat energy for district heating would be larger. But electricity would still be expected to constitute the main product, assuming base load operation, for economic reasons. The field of application of these reactors would be similar to the case of medium or large co-generation reactors. Additionally, however, they could also address specific objectives, such as the energy supply of concentrated loads in remote and cold regions of the world.

Heat-only reactors for district heating are another option. There are several designs being pursued. Such applications have been implemented on a very small scale (a few MW(th)) as experimental or demonstration projects. Construction of two units of 500 MW(th) initiated in the 1980s in Russia and suspended in 1990 was resumed in 1996. Application of a 200 MW(th) unit has been evaluated in China. Clearly the potential applications of heat-only reactors for district heating are limited to reactors in the very small size range. These reactors are designed for siting within or very close to population centers so that heat transmission costs can be minimal. Even so, economic competitiveness is difficult to achieve due to the relatively low load factors required, except in certain remote locations where fossil fuel has a big penalty of high transportation costs, and the winter is very cold and long.

In summary, the prospects for nuclear district heating are real, but limited to applications where specific conditions pertaining to both the district heating market and to the nuclear reactors can effectively be met. The prospects for co-generation reactors, especially in the SMR range, seem better than for heat-only reactors, mainly because of economic reasons.

Industrial processes: The characteristics of the market for process heat are quite different from district heating, though there are some common features, particularly regarding the need for minimal heat transport distance. Industrial process heat users, however, do not have to be located within highly populated areas. Many of the process heat users, in particular the large ones, can be and usually are located outside urban areas, often at considerable distances. This makes joint siting of nuclear reactors and industrial users of process heat not only viable, but also desirable in order to drastically reduce the heat transport costs.

For large size reactors, the usual approach is to build multiple unit stations. When used in the co-generation mode, electricity would always constitute the main product. Such plants, therefore, have to be integrated into the electrical grid system and optimized for electricity production. For reactors in the SMR size range, and in particular for small and very small reactors, the share of process heat generation would be larger, and heat could even be the predominant product. This would affect the plant optimization criteria, and could present much more attractive conditions to the potential process heat user. Consequently, the prospects of SMRs as co-generation plants supplying electricity and process heat are considerably better than those of large reactors.

Several co-generation nuclear power plants already supplied process heat to industrial users. The largest projects implemented are in Canada (Bruce, heavy water production and other industrial/agricultural users) and in Kazakhstan (Aktau, desalination). There are also medium and limited size of co-generation nuclear power plants in operation. Other power reactors which currently produce only electricity, could be converted to co-generation. Should there be a large process heat user close to the plant interested in receiving this product, the corresponding conversion to co-generation would be technically feasible. India is now connecting a seawater desalination process to an existing PHWR for converting it to a co-generation plant. It involves, however, re-licensing procedures and additional costs. Some such conversion projects could be implemented but, in general, prospects for this option seem rather low.

Installing a new nuclear co-generation plant close to existing and interested industrial users has better prospects. Even better would be a joint project whereby both the nuclear co-generation plant and the industrial installation requiring process heat are planned, designed, built, and finally operated together as an integrated complex.

Current and advanced light or heavy water reactors offer heat in the low temperature range, which corresponds to the requirements of several industrial processes. Among these, seawater desalination is presently seen as the most attractive application. Other types of reactors, such as liquid metal cooled fast reactors and high temperature gas cooled reactors can also offer low temperature process heat, but in addition, they can cover higher temperature ranges. This extends their potential field of application. Indonesia is evaluating a possibility to use heat from an HTGR for desalination. These reactors still require substantial development in order to achieve commercial maturity. Should they achieve economic competitiveness as expected, their prospects seem to be promising in the medium to long term, especially for high temperature industrial applications.

Heat-only reactors have not yet been applied on an industrial/commercial scale for the supply of process heat. Several designs have been developed and some demonstration reactors have been built. Economic competitiveness seems to be an achievable goal according to many studies which have been performed. The Moroccan study to use a 10 MW(th) heating reactor of Chinese design for seawater desalination demonstration plant in Tan-Tan will be a good milestone in the application. The potential market for such heat-only reactors would be limited to the very small size range, i.e. below about 500 MW(th).

The prospects for applying nuclear energy to district and process heating are closely tied to the prospects of deploying SMRs. A recent market assessment for SMRs found that 70 to 80 new units are planned in about 30 countries up to the year 2015. It was also found that about a third of these units are expected to be applied specifically to nuclear desalination. Of the rest, a substantial share could supply heat in addition to electric energy, while a few are expected to be heat-only reactors.

- **Nuclear safety and licensing aspects**

The overall safety and licensing issues associated with an integrated facility consisting of a nuclear energy system coupled to a heat utilization unit such as desalination or district heating system, are primarily those associated with the nuclear plant itself. Nevertheless, the safety and licensing of the integrated system must be addressed and some specific characteristics such as siting and coupling of the reactor with the heat utilisation unit, require particular consideration from a safety point of view.

Specific aspects could include, for example, in case of desalination plants, the potential for introduction of radioactivity into the potable water being produced by the facility, the possibility of interaction effects between the nuclear plant and the desalination plant, environmental issues arising from the discharge of concentrated brine from the facility, the potential impact of shared resources such as intake and outfall structures, and the “backfitting” of desalination systems with already existing nuclear plants. There may also be issues that arise if siting of the facility near population centers is considered.

The current safety approach based on the defence in depth strategy, has been shown to be a sound foundation for the safety and protection of public health, and gives the plant the capability of dealing with a large variety of sequences, even beyond the design basis, however the safety implication of transients that can be imposed on the nuclear plant by the desalination or district heating units should be carefully assessed.

The existing IAEA standards contain requirements and recommendations that are generally applicable also to nuclear plants to be used for co-generation or non electrical applications and provide a reference for a safe design and for a complete safety assessment.

2. OVERVIEW OF NUCLEAR HEAT APPLICATION

Twenty-three participants from 11 Member States attended the meeting, including two representatives from the IAEA, to update relevant information on advances in nuclear heat applications for non-electrical products since the previous meeting on the subject in 1997 [1]. Nineteen papers in total were brought to the meeting. The contribution was made in three groups:

- Low temperature applications with 9 papers including one overview paper by the IAEA.
- High temperature applications with 7 papers.
- Nuclear safety and licensing with 3 papers including one overview paper by the IAEA.

This section highlights key points addressed in each group.

2.1. Applications at low temperatures

Low temperature applications includes seawater desalination, district heating, agro-industrial applications, etc. Papers from Canada, China, Egypt, India, Morocco, the Russian Federation and Tunisia mainly focused on application to seawater desalination.

Seawater desalination is one of the main applications of low temperature nuclear heat. The steam from a nuclear reactor could be used for thermal desalination, namely in multi stage flash (MSF) and multi-effect distillation (MED). Warmed seawater from the condenser outlet could be used as preheated feed for a seawater reverse osmosis (SWRO) plant. Higher water temperature increases membrane water flux and hence higher throughput and a reduced specific energy consumption.

Participants from Morocco, Tunisia, Egypt and Indonesia presented their future needs for potable water requirements at some specific sites. Nuclear desalination technology could be the potential solution to meet these needs. Several technologies and designs had been presented that could be adopted as per the requirements by the users.

The participants showed strong interest in the new ongoing demonstration projects such as a nuclear heating reactor coupled with an MED-desalination plant in Morocco with China, and a hybrid MSF-RO nuclear desalination project of India as well as R&Ds on coupling of nuclear heating and desalination technologies. Also, designs have been developed by Russia on floating nuclear reactor/cum desalination plants based on both RO & MED processes. Indonesia has undertaken a feasibility study for a plant using HTGR coupled with desalination units.

Following are selected key features presented by the participants.

Canada (AECL) carried out four major studies in the field of co-generation and heat applications in the past few years. The most recent work, which addresses Canada's commitment to reduce greenhouse gas emissions, discussed demand, supply, economics and gas reduction factor. The second study examined possibilities for developing an integrated energy centre (Bruce Center in Ontario) in a cascade fashion, from electricity production to hydrogen, transportation fuels, and finally to low grade heating for agriculture. The third study described application of nuclear-generated steam to extraction of oil from oil sands. The fourth study presented a technical and economic evaluation of seawater desalination using nuclear heat.

China (INET) has been developing a commercial sized heating reactor (NHR-200) based on the experience gained by NHR-5. The first installation project of an NHR-200 in the Daqing Oil Field has been dormant due to institutional factors since the construction permit was issued in 1996.

In **Egypt** feasibility studies and relevant R&Ds are ongoing in the field of nuclear desalination. Preliminary economic assessment of potable water production by various energy sources and desalination processes proposed for the El-Dabaa site has been carried out.

India (BARC) is continuing its civil work for the Indian demonstration programme on nuclear desalination at Kalpakkam for connecting an MSF-RO hybrid system to existing PHWR plants. The commissioning is scheduled for 2001/2002.

In view of the prospect of synthetic fuel production needs, **Indonesia** has an incentive of applying high temperature heat seems more attractive to the country. A study on a possibility of a demonstration plant to be used for seawater desalination at low temperature as a first step to develop nuclear technology in the country is being assessed.

The **Russian Federation** is developing a floating power unit for cogeneration (electricity and water) at the OKB Mechanical Engineering. The core is designed to meet the latest safety criteria (KLT-40 to KLT-40S) and the manufacturing of key components including a reactor vessel was initiated. Construction work at PEVEK in northern Russia is scheduled to start in 2005 if financially feasible.

Tunisia envisages a nuclear desalination plant with water production capacity of about 60 000 m^3/d, based on the results of assessing the seawater desalination needs after the year 2000, particularly in southern Tunisia.

Morocco foresees continuation of its project plan at Tan-Tan after the completion of the pre-project study of its demonstration plant at the site in October 1998 jointly with China.

2.2. Applications at high temperatures

High temperature applications include synthesis gas reforming and hydrogen production.

For the possible high temperature applications from HTGR heat, a total of six papers were presented. It was generally well understood and agreed by the participants that the high temperature applications by HTGRs would be promising in the future, not only from the view point of the global environmental issues of the CO_2 emission, but also stable energy supply alternative to the fossil fuels. On the other hand, it was recognized that the technologies are not well developed so far and no economic prospect is yet clear, thus further research and development is required before commercial deployment.

It was reconfirmed that HTGR provides a possible opportunity to convert nuclear energy to other useful forms of energy in much higher thermodynamic efficiency, compared to simply producing steam and generating electricity via a Rankine cycle. The HTGR heat also enables a broad variety of applications not only in the electricity sector but also in the transportation (fuel) and industrial heat sectors. HTGR heat applications can therefore find markets in various niches.

China (INET) is continuing a conceptual design for a steam reformer for its HTR-10, which is now under construction. Several modifications are considered:

(1) Decreasing the pressure of the working He, to a level still higher than the secondary helium for safety reasons;
(2) Increasing the temperature by operating the reactor core at 950ºC and increasing the secondary helium outlet temperature to 905ºC;
(3) Increasing the heat transfer both on the helium side by extended surface of the reactor tube and on the catalyst side;

A computer code was developed for the analysis of the reformer.

Japan (JAERI) is proceeding with the R&D programme on steam/CO_2 reforming as the first priority candidate to be coupled to the HTTR. The reforming system shall be designed with emphasis on featuring hydrogen production. The hydrogen will be used as an energy carrier for reducing CO_2 emission.

JAERI is now constructing an out of pile simulation facility, in which the helium gas is heated electrically, in order to test their steam reformer. Construction will be completed by the year 2000 and thereafter the tests will be performed until 2004. The out of pile tests will include component level tests such as hydrogen/tritium permeation tests.

The construction of the HTTR hydrogen production system hopefully will start in 2004 and is planned to be in operation in 2006. JAERI also tries to improve the heat transfer characteristics of the reformer tubes by adding disc fins to extend the external surface of the tube, which will be applied hopefully to the HTTR system. In addition, JAERI continues the development of the IS process for splitting water. A closed laboratory scale thermo-chemical cycle was successfully operated and a larger scale loop with a new engineering facility will operate after 2000.

In view of the industrial sector’s interest in applying gas turbines for electricity and process heat, the R&D plans for HTTR are expected to be shifted to the commercial application from the basic research on the long term basis.

Israel presented a proposal for a new type of reformer heated by condensing vapors of e.g. Na on the outside of the tube or using a Na heat pipe directly to the internal catalyst bed. This method allows higher heat transfer and favorable isothermal conditions. The Na can also be used as a safety means in case of operational failure in the reformer plant by rejecting the heat to an emergency condenser using natural circulation of the Na. The possibility of reforming biogas was emphasized in the presentation, so that any carbon ending as CO_2 does not contribute to a net emission. If the biogas is generated by processing of urban wastes or agriculture wastes, economy of the entire system could be quite favorable

The **Russian Federation** (OKBM) is continuing an international program together with GA (USA), Framatome (France) and Fuji Electric Company (JAPAN) to develop a conceptual design for a new generation of NPP model GT-MHR which uses weapon grade plutonium with a passively safe reactor. The first module can be constructed in 2007. The reactor will be 600 MW(th) and the power conversion efficiency up to 47%. This development is expected to be useful also for non-electrical applications.

Indonesia is assessing their needs for the application of nuclear process heat including electricity production as well as their development plan to be done mainly at BATAN, in cooperation with other countries. Considering the present technology level, the first demonstration plant will be applied to seawater desalination and electricity production. A wide range of R&D plans is envisaged in the field of reactor technology, economy, financing, site and environment studies, etc. It was reconfirmed that a market for the process heat applications by HTRGs is apparently growing. One of the main targets of the program is ultimately to produce synthetic fuels to meet the increasing demand. This application requires HTGR heat and steam/CO_2 reforming.

2.3. Safety and licensing aspects

The group was represented by the IAEA, France and South Africa. The discussion on the papers was very stimulating and provided some useful indications on safety and licensing aspects of existing HTGR design and for future activity of the IAEA in the area of defence in depth and application of the IAEA safety standards to HTGRs and NPPs used for non-electrical applications.

The **IAEA** continues its main activity related to the safety of nuclear desalination, and is preparing to revise the IAEA Safety Standards for Design. The current safety approach based on the safety objectives and defence in depth strategy, including the applicability of the IAEA Safety Standards to specific nuclear plants or applications, are being reviewed. The main safety issues connected with the coupling of a nuclear plant with a desalination or heat utilization unit are also being addressed.

France reviewed the current CEA/DRN safety approach to the design and safety assessment of future nuclear installations elaborating, in particular, on the application of defence in depth and the verification of its correct implementation.

The CEA/DRN approach is mainly founded on the three following essential principles:

- The allowable radiological limits for operators, public and environment should be the same for all nuclear installations.
- The safety assessment for all application should be based on the integral adoption of the defence in depth strategy.
- The safety approach should integrate the specific characteristics of each application.

The Council for Nuclear Safety of **South Africa** (CNS) is reviewing the conceptual design of the Pebble Bed Modular Reactor (PBMR). The specific features of the reactors and their implication on licensing were underlined. The approach to the licensing of PBMR will be similar to that followed in the past for the reactors in Koeberg. This included a quantitative risk assessment to demonstrate compliance with the CNS fundamental objectives expressed in probabilistic terms. The assessment of the safety of the PBMR presents additional difficulties because, for this reactor, a complete 'off-the shelf' package of rules to define the design basis is not available and should be developed.

3. SUMMARIES OF THE CONCLUSIONS OF THE MEETING

Evaluation of the papers presented and relevant discussions by participants led to the following conclusions.

3.1. Low temperature applications of nuclear heat

Seawater desalination is one of the main applications of low temperature of nuclear heat. The steam from nuclear reactors could be used for thermal desalination, namely in MSF and MED. Hot sea water from the condenser outlet of the nuclear plant could be used as preheated feed for the SWRO plant since high temperature increases membrane water flux and hence throughout at a reduced specific energy consumption.

Participants from Morocco, Tunisia, Egypt, Indonesia presented their future needs for potable water requirements at some specific sites. Nuclear desalination technology could be the potential solution to meet these needs. Several technologies and designs had been presented and these could be adopted as per the requirements by the users.

The participants showed their keen interest in the new ongoing projects such as the nuclear heating reactor coupled with MED-desalination demonstration plant of China and Morocco, hybrid MSF-RO nuclear desalination demonstration project of India as well as R&D on coupling of nuclear heating and desalination technologies. Also designs have been carried out by Russia on the nuclear floating reactor combined with desalination plants based on both RO&MED processes. Indonesia has undertaken a feasibility study for a plant using HTGR coupled with desalination units.

It is recognized that

— the IAEA is playing a major role by co-ordinating all these programmes and activities through and specialized technical meeting on this matter.
— Interregional technical co-operation projects carried by the IAEA are an important forum for the exchange of experience between supplies and users.

It was understood among participants that the following elements would facilitate further application of nuclear heat for non-electric products.

— To promote effectively nuclear desalination technology, the suppliers should improve their technology by implementation and operation before transferring it to users.
— The nuclear desalination technology should be developed in full respect of safety procedures and requirements, with the main objective of reducing the cost of the product water by simplifying the design.
— Public confidence and acceptance should be ensured for all aspects of this technology.
— Nuclear culture is necessary to implement a nuclear desalination programme for the potential user countries.
— An international effort should be continued to co-ordinate activities of nuclear desalination programmes of Member States.
— The international effort to promote the implementation of regional and interregional programmes should be made, aiming at collecting and compiling the experience data, communicating procedures and requirements with more frequency.
— An international effort should be continued to contribute in all aspects to the implementation of a first nuclear desalination plant as an international model.
— Experience on the use of nuclear energy for other non electrical applications such as district heating is relevant and should be promoted and developed.

3.2. High temperature applications of nuclear heat

The high temperature applications by HTGRs would be promising as a stable energy supply alternative to the fossil fuels while contributing to the reduction of CO_2 emission for the global environment.

HTGRs also provide a possible opportunity to convert nuclear energy to other useful forms of energy in much higher thermodynamic efficiency, compared to simply producing steam and generating electricity via a Rankine cycle. The HTGR heat also enables a broad variety of applications other than in the electricity sector, such as in the transportation (fuel) and industrial heat sectors.

However, further research and development is required for commercial deployment of HTGRs, including the economic assessment. Experience to be obtained from HTTR in Japan, HTR-10 in China and other out of pile tests will be effectively shared by the interested people.

3.3. Safety and licensing

In the discussion on the future of nuclear power, the technical, safety, social and economic aspects shall be taken into account.

Nuclear energy should be recognised as an essential factor for the worldwide sustainable development. The potential of non-electric applications, discussed during the meeting, is an essential part of this demonstration. On the other hand, safety implementation and perception (i.e. public acceptability) become major concerns for all the countries currently engaged or interested by nuclear energy. The competitiveness of this energy source remains the key issue for its concrete implementation. Within this context, the harmonisation of safety approaches should be pursued both for the design and the safety assessment of future plants. This harmonisation is recognised as an essential step to achieve common and understandable positions and for a successful implementation.

It is also recognised that the specific features for plants involved in non electrical applications will lead to take into account:

- issues induced by site locations;
- specific aspects linked to the plant sizes;
- the innovative character of new concepts, being conscious that simple architecture can help to achieve an adequate safety level.

The IAEA effort for the revision of the entire collection of the Safety Standards is recognised as a unique chance to achieve a large harmonisation of nuclear installations safety philosophy.

The South African PBMR licensing project, currently under way, is conducted by CNS respecting prevailing international norms and practices. This is a very interesting occasion to prepare and implement the licensing procedure for an innovative HTGR power plant.

An analogous project would be launched on the GT MHR concept, the design of which is ongoing in an international context.

Strong interest is recognised for these projects and for enlarged exercises involving countries interested by the characteristics and the performances of these HTGR concepts (e.g. Japan, Indonesia, France).

Worldwide safety authority activities (e.g. TSO in Europe) as well as the availability of utilities requirements (e.g. ALWR in USA, and EUR in Europe) developed within the frame of evolutionary and innovative concepts, are also an occasion to identify and synthesise the main guidelines for future safety recommendations and requirements.

Innovative concepts and innovative nuclear energy utilisation suggest specific needs for continuing and strengthening efforts to:

— Continue training and education activities to spread safety culture toward the countries interested by the nuclear energy development.
— Continue the activity on interpretation and implementation of defence in depth levels, for example for the HTGR concepts.
— Implement a co-ordinated effort to investigate the safety aspects and licensing issues on a specific and well defined project: e.g. JAERI–HTTR with the corresponding heat application installations.
— Implement a co-ordinated discussion on all the technical aspects of the safety approach, namely the methodologies for the plant assessment, the component classification methodologies, etc. This will create the basis for a useful discussion between the designers/suppliers/utilities and the safety authorities.

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LOW TEMPERATURE APPLICATIONS

CANDU CO-GENERATION OPPORTUNITIES

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Abstract

Modern technology makes use of natural energy "wealth" (uranium) to produce useful energy "currency" (electricity) that can be used to society's benefit. This energy currency can be further applied to help solve a difficult problem faced by mankind. Within the next few years we must reduce our use of the same fuels which have made many countries wealthy — fossil fuels. Fortunately, electricity can be called upon to produce another currency, namely hydrogen, which has some distinct advantages. Unlike electricity, hydrogen can be stored and can be recovered for later use as fuel. It also is extremely useful in chemical processes and refining. To achieve the objective of reducing greenhouse gas emissions hydrogen must, of course, be produced using a method which does not emit such gases. This paper summarizes four larger studies carried out in Canada in the past few years. From these results we conclude that there are several significant opportunities to use nuclear fission for various co-generation technologies that can lead to more appropriate use of energy resources and to reduced emissions.

INTRODUCTION

The most recent work addresses Canada's commitment to reduce greenhouse gas emissions [1]. Demand, supply, economics, and gas reduction factors are discussed. The potential impact of efficient transportation with low emissions is quantified, based on both existing and envisaged technological advances, utilizing electrolytic hydrogen and fuel cells. The second study is aimed at possible future development of flexible and market-driven groups of industries which can be called "energyplexes". The specific case studied is the Bruce Energy Centre in Ontario [2]. Beginning from a base of existing large scale installations, the study examines possibilities for developing an integrated energy centre in a "cascade" fashion, from electricity production to hydrogen, transportation fuels, and finally to low-grade heating for agriculture. The economics of this energyplex centre are discussed. The third study describes application of nuclear-generated steam to extraction of oil from the Athabasca oil sands in Western Canada [3]. Novel extraction technology and comparative economics of nuclear vs fossil steam are discussed. The motivation for this work came from the very large commitment of natural gas which otherwise would be needed to extract this oil. The fourth study presents a technical and economic evaluation of seawater desalination using nuclear heat [4], and innovative reverse osmosis technology, at a site in Indonesia. This study found favorable economic and technical results, even under these difficult conditions.

TRANSPORTATION FUELS FOR GREENHOUSE GAS REDUCTION

The simple concept underlying this work uses nuclear power plants to generate electricity for the grid, for use in either central or distributed hydrogen production. Hydrogen is produced by electrolysis. By-products include heavy water and oxygen. Deuterium can be extracted from the hydrogen stream. Heavy water is essential for operation of CANDU reactors, so that there is a very substantial additional economic synergy between the application of nuclear energy and hydrogen production in this case. AECL has developed experience in handling hydrogen because hydrogen is central to processes that offer cheaper heavy water for CANDU reactors. There is, of course, a wider interest in the expanded use of hydrogen. It is the obvious chemical gateway through which electricity from virtually inexhaustible sources can supplement and provide the liquid fuels and petrochemicals that now come from conventional oil. There is a new way to promote use of that gateway [5] by adding the Combined Electrolysis and Catalytic Exchange (CECE) process to electrolysis and improving the economics with byproduct heavy water. It this were widely adopted, the price of heavy water could fall by over a half, which would have a noticeable effect on the price of electricity from CANDU reactors, providing positive feedback around a circle of electricity-electrolysis-hydrogen.

A reactor design appropriate for this application is the CANDU 6, which has been exported, so far, to four foreign countries and is operating in Canada. The lifetime average capacity factor of plants using this reactor design is about 85%. Canada is one of the largest exporters of nuclear plants in the world and competes successfully against oil, coal, gas, as well as against other nuclear designs.

Nuclear fuel supply to support the nuclear-hydrogen option is very large at the present time and effectively infinite in the long term [6, 7]. The supply of nuclear fuel is adequate to energize any envisaged hydrogen production program.
With respect to Green House Gas (GHG) emissions, CANDU reactors already have reduced emissions in Canada by over 1000 Mt/CO_2 cumulative to date, and continue to avoid emissions by about ~100 Mt/y.

Economic analysis

In Figure 1 we compare estimates and trends for future generating costs from several alternate sources. These values are derived directly from The European Renewable Energy Study [8] and from actual CANDU generating costs [1] without taking credit for the reduction of nuclear plant cost which is expected in the future. At the same time the cost estimates for the renewable energy options assume significant technological improvements.

The European data used here are of interest because there is already significant wind-energy deployment and experimentation with "competitive" but regulated energy markets, and on market-share targets and subsidies for wind farms. Regardless of the absolute magnitudes, which show significant cost increases for use renewable sources, the relative estimates in Figure 1 also clearly show that 20 to 30 years are needed for renewable energy costs to decline significantly. Nuclear energy costs are known accurately today.

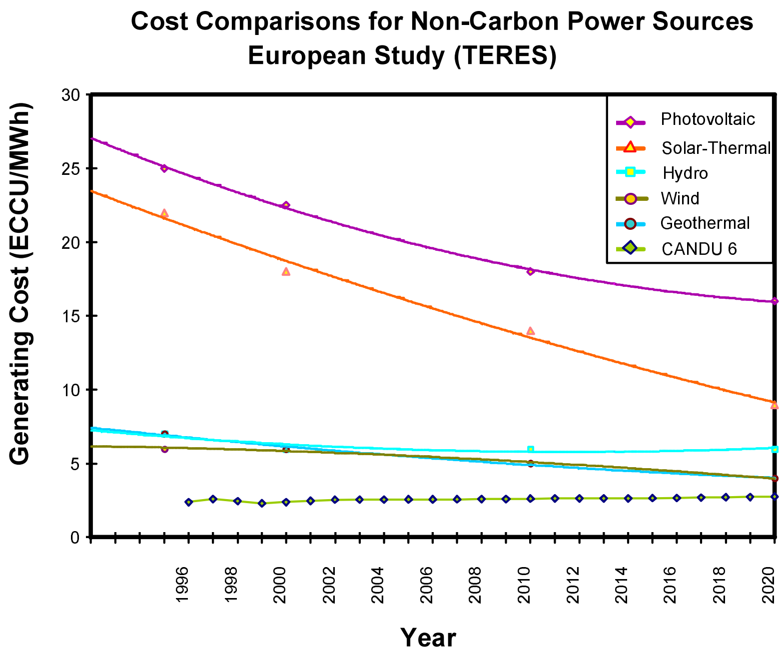

Figure 1. Comparative current and projected electrical generation costs [8].

The proposed non-carbon bridge to the future utilizes a balance of renewable and nuclear energy sources and hydrogen, coupled with the management of emissions. The portfolio of electric energy sources is robust in the sense that nuclear is used for the large blocks of base load power, solar wind and other sources contribute as available, and hydrogen serves for energy storage.

It took nearly a century to build the world's present electricity production systems and energy-use patterns: it was an immense enterprise, and it is unreasonable to expect all of this to be changed quickly. A bridge should be built to enable changes to occur gradually over the next 20 to 50 years, without causing significant economic disruption.

Hydrogen generation and end use alternatives

(a) Central hydrogen generation

One non-carbon concept is to generate hydrogen using a central plant to take advantage of the economies of scale. Hydrogen then can be distributed by tanker or pipeline as a compressed gas or in a chemically combined form such as ammonia to the local distribution sites for final distribution and use. To achieve essentially zero emissions, the preferred hydrogen production process is electrolysis of water. Hydrogen produced in this manner is sensitive to the assumed cost of bulk electricity, which is typically about 25 to 33% of the total H_2 cost. The costs have been carefully analyzed and show that hydrogen can be produced economically in advanced electrolytic equipment [9] at about 70% efficiency, so that the energy required is of the order of ~50 kWh(e)/kg H_2. By-product deuterium gas production does not require significant additional energy input.

CANDU 6 plants normally achieve an 80% operating capacity factor. Calculations for a commercial-scale electrolysis plant show that a 690 MW CANDU 6 reactor can simultaneously co-generate about 95 Mg/y of D_2O at that capacity factor. Since annual consumption resulting from heavy water leakage is less than 5 Mg/y, the system is much more than self-sufficient in heavy-water production. The annual net production is:

One CANDU 6 @ 690 MW(e) => 90 Mg/y D_2O + 97 Gg/y H_2.

At today's prices, and to give perspective on the order of magnitude, the hydrogen is valued at about \$800/Mg, using electricity generated at ~2.5 cents/kWh(e), and is hence potentially worth ~\$115 M/y revenue at the wholesale site. The D_2O by-product is worth another ~\$25 M/y at an assumed market price of \$250/kg.

(b) Embedded hydrogen generation

Local small scale electrolysis units, which eliminate the need for a central facility, have been examined as a second option [10]. Electrolysis units normally are modular, so specific production cost varies only slightly with capacity. Local generation avoids transportation costs and large storage containers. Electricity can be supplied either from a grid with central power plants or from local renewable (solar or wind) power.

It may be desirable to have a local small scale capability for H_2 production. The extra cost of compression locally is also small, say, 10% of the generating cost. As will be seen, the local generation need is quite small. Ideally hydrogen would be generated during off-peak periods of the electricity supply system, thereby better utilizing electrical generating capacity. This mode is especially suitable for nuclear units because of their small incremental cost of fuel. The pricing structure for electricity could be designed to encourage off-peak use.

Synergism with hydrogen in transportation

At present electricity represents a negligible ~0.2% of fuel consumption in transportation, which is the unchallenged domain of oil and gas. Transportation in Canada alone represents some 15% of the GDP, with a total energy use ~1.5 E/J/y causing emissions of about 150 Mt CO_2/y, which are projected to rise, as gasoline use rises, to about 200 Mt CO_2/y by 2020. Canada's fleet of personal vehicles consisted of about 15.5 million cars and light trucks in 1995 (NRCan, 1998). These vehicles alone generated 91 Mt of GHGs expressed as CO_2 equivalent in 1995.

Extensive analysis has been undertaken on the costs and benefits of hydrogen as a transportation fuel [10]. In principle, it is very attractive and simple. Hydrogen is abundant and it can be burned either as a raw gas or with a carrier. It also can be used to feed mobile on-board fuel cells to recombine with oxygen and produce water and energy. There is a variety of possible vehicles, e.g., an internal combustion engine (ICE) fed by methanol, natural-gas, or hydrogen, combined with batteries, fuel cells, conventional combustion or both. Emissions for each choice are shown in Figure 2.

For hydrogen production, as for everything else, we must consider the entire process from production to end use, so the carbon dioxide emissions using conventional reforming and electrolysis based on fossil-fuel sources are comparable with or even greater than those from simply using gasoline. Thus there is, significantly, no real advantage to producing hydrogen in this way to reduce emissions. One might as well burn natural-gas or propane.

The lowest carbon-dioxide emitting vehicles are the hybrid and fuel cell vehicles. The achievable efficiency of the process is about 50% at the vehicle level (from combustion to motion). This efficiency gives about a factor of three improvements in the 'equivalent' fuel economy, to approximately 2.75 l/100 km (80–90 mpg) relative to today's vehicles.

Although some further development of efficient cars and other motive systems remains to be done, there is a way forward that will allow growth of personal and public transportation systems at the same time as allowing drastic reduction of greenhouse gas emissions.

DESIGN FOR AN ENERGY COMPLEX

The Bruce Energy Center (Figure 3) has been developed over the past 20 years next to the Bruce Nuclear Power Development (8 × 850MW(e) units) on the Eastern shore of Lake Huron in Ontario. The initial development focused primarily on agriculture-based industry, on the premise of available and affordable steam generated by the Bruce "A" units (with additional supply backup from fossil-fired boilers).

A sustainable development model now being developed represents an initiative to complete the larger mission of the Bruce Energy Center. This development is intended to demonstrate commercial application of "closed-loop" and integrated systems, the introduction of nuclear hydrogen and absorption of CO_2.

Sustainable development model

1) Cogeneration of electricity and process steam using a Nuclear fission reactor
2) A menu of feedstocks ranging from farm produced carbo-hydrates and solid wastes to low-grade carbon sources and carbon dioxide,
3) A series of state of the art processing, synthesizing and refining processes, and
4) End products that have markets and in their own right have environmental value-added.

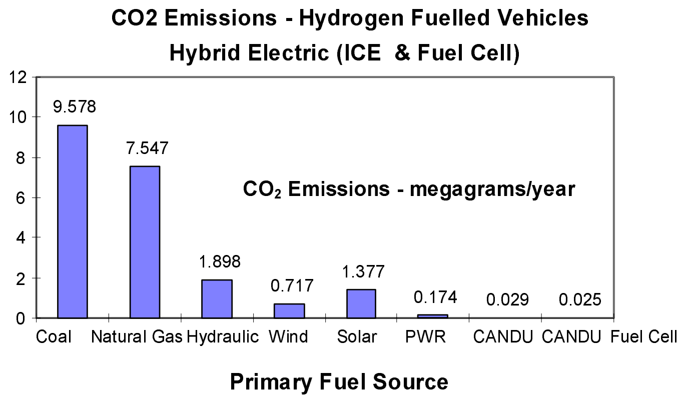

Figure 2. Emission levels of hydrogen-fuelled vehicles.

Consistent with the premise and vital to the economic viability is the integration of processes i.e. carbon dioxide from fermentation processes used to produce ethanol is combined with electrolytic hydrogen to form methanol ($CO_2 + 3H_2 \rightarrow CH_3OH + H_2O$). Similarly, the cascading use of thermal energy is applied to maximize benefits, i.e. steam is used first for ethanol distillation, secondly for greenhouse operation and finally for aqua-culture. One potential sequence of products is shown in Figure 4; heavy water production and desalination processes have been added to the original plan shown in Reference 2.

Figure 3. Bruce Nuclear Power Development (8 × 850MW(e) CANDU units, heavy water production, greenhouse complex using nuclear steam).

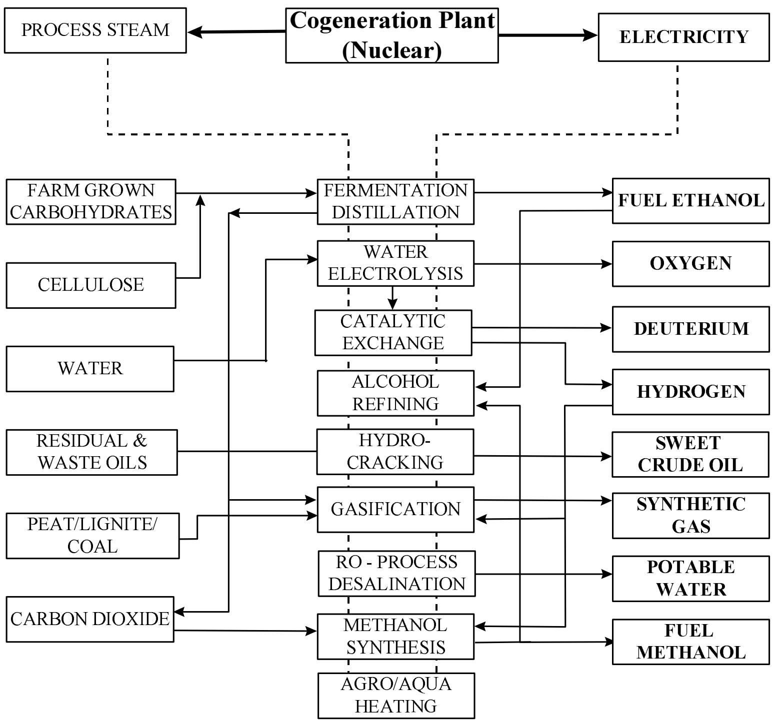

Figure 4. Illustration of nuclear cogeneration with integrated processes and cascading use of thermal energy.

The relationship of the development to Bruce "A" generating station, and the demand/supply profile of Ontario Hydro are crucial to the economic viability of the electrolysis-based first stage. It will be necessary to generate electrolytic hydrogen and oxygen at lower prices than is possible by conventional methods. Many factors, site specific to the Bruce Nuclear Power Development and the Bruce Energy Center, will influence the operation of this facility. Use of off-peak electricity would strongly improve facility economics.

Methanol synthesis

The initial CO_2 to combine with the hydrogen will be taken from the existing 20 000 litre/y Commercial Alcohol Inc. fermentation ethanol plant at the Bruce Energy Center. This plant provides approximately 15 000 Mg/y of useable CO_2 and this, in turn, provides the absorption of 27% of H_2 production. The remaining 73% of CO_2 would be supplied by a planned 100 million litre/year fermentation plant.

Planned industries

The development company is reviewing plans for further projects as follows:

- 600 Mg/day municipal solid waste separation plant,
- 1000 Mg/day cellulose conditioning plant,
- 75 Mg/day medium density fibre board plant,
- 100 Mg/day organic waste processing plant,
- 36 Mg/day plastic extrusion plant.
- 100 Mg/day ammonia processing plant.
- 1300 Mg/day ETBE/MTBE/Ethers synthesis plant.
- 20 000 litre/day waste/residual oil upgrading plant
- Four(4) 4-acre greenhouse c/w common packing and administration, and
- 200 ,000 lb./year high density aqua-farming plant.

These components will be developed to provide a reduced-cost feed stock for downstream industry while providing a good level of profit for each of the individual process activities.

50 MW(e) Electrolysis at $.02 per kwh,
hydrogen @ $2 000 /Mg
Costs in Canadian Dollars

Capital cost		
Land, bldg.	1 500 000	
Equipment (256 cells)	28 400 000	
Annual revenue	20 100 000	
Annual operating cost		
electricity	13 400 000	
other (int., amort., tax	5 500 000	
Annual net revenue	1 200 000	
Revenue products		
60 000 Mg oxygen at $50 per Mg		3 000 000
7 Mg D2O at 300 k$ per Mg		2 100 000
7500 Mg hydrogen at $2,000 per Mg		15 000 000

Figure 5. Example costs of electrolytic hydrogen production.

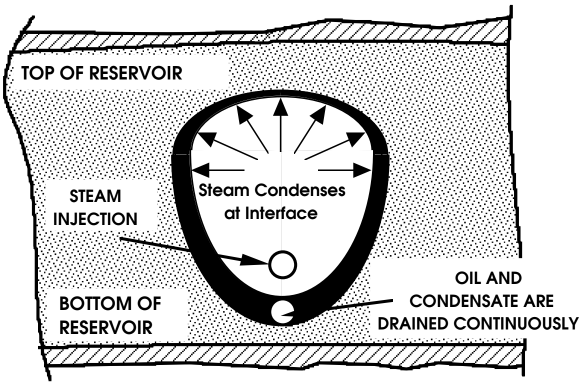

Figure 6. Steam injection to, and recovery from, the oil-bearing formation.

Thi model for sustainable development using hydrogen for recycling of CO_2 and fuelled by a non-carbon, inexhaustible energy source requires several levels of entrepreneurial innovation. There are no significant technological barriers. The main impetus must be the recognition of a need for change. When that need is recognized it is obvious that a large number of beneficial and economical processes can be developed from application of nuclear fission heat.

OIL SANDS TECHNOLOGY

The usual assumption concerning the use of extracted oil is that it is intended for burning in one way or the other. In the short term this undoubtedly will be the case. But in the longer term oil can be part of the transition to a sustainable system, in which its use is limited to applications such as chemical feedstocks.

In the last twenty years the technology used in extraction of oil from Canada's oil sands has advanced in many areas. One of these technologies, **steam assisted gravity drainage or SAGD**, stands out as a low cost and applicable method for all oil sand depths below 30 metres of overburden. It has been established [11] that the supply cost of bitumen produced from a SAGD operation is well below the historic bitumen plant gate selling price of between ten and fifteen dollars per barrel.

The steam assisted gravity drainage approach to the thermal recovery of heavy oil depends on long horizontal wells placed at the base of the reservoir. Steam is introduced into the base of the reservoir. Because of the low density of gaseous steam it rises in the reservoir and heats the formation. The heated oil and water (both condensed steam and heated formation water) in the formation drain down to the horizontal well from which they are pumped to surface. Figure 6 shows the mechanism by which the process proceeds within the reservoir. The steam is injected into the reservoir at a pressure slightly above the natural pressure of the formation, from another horizontal well. The process is ineffective with vertical producing wells because of the relatively low flows that can be achieved under these conditions. However, with long horizontal wells, economic production rates can be achieved. SAGD technology is the key to the production process. A fundamental Steam Assisted Gravity Drainage (SAGD) production unit consists of two parallel horizontal wells vertically spaced 5 metres apart at the bottom of the oil-bearing zone. Steam is injected into the upper well and production comes from the lower well. Steam, injected just below formation fracture pressure, heats the otherwise immobile bitumen which flows by gravity to the lower well. The key features of the extraction facility are shown in Figure 7. The development of the SAGD technology has assured that bitumen can be produced in the Fort McMurray area of North-eastern Alberta at a competitive price at any volume that the market can absorb.

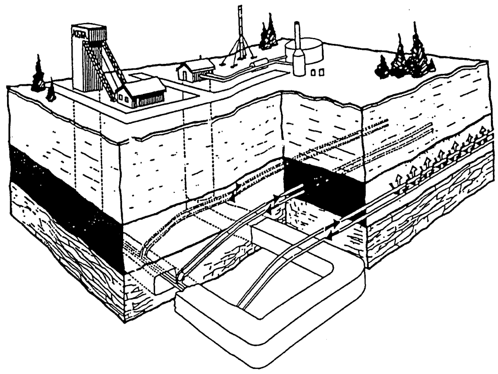

Figure 7. Arrangement of SAGD injection and extraction wells.

Magnitude of the resource

A reservoir screening study [12] determined the magnitude of the bitumen resource which can be economically exploited using the SAGD technology. This study indicates that at least 28 commercial recovery projects could be supported, each producing at least 130×10^3 barrels (20.6×10^3 m^3) of bitumen per day for 30 years. The ultimate potential of this recovery method is so large that it is unlikely that any other known technology will be needed.

Energy supply alternatives

Extraction of bitumen requires energy input, mostly for boiling water. Currently natural gas is the fuel of choice. The twenty-eight, 30 year SAGD recovery schemes would produce a total of at least 40 billion barrels (6.35×10^9 m^3) of bitumen. More than 42 EJ of energy would be needed to generate the required steam. An additional 42 EJ of natural gas would be required to produce the hydrogen required to upgrade the produced bitumen to synthetic crude oil (SCO) standards. The combined requirements for steam generation and hydrogen production represents 60 percent of the remaining ultimate potential of natural gas in Alberta. Clearly fuels other than natural gas must be used if the ultimate potential of oil sands is to be realised. Nuclear fission is an obvious alternative, and can produce steam at the pressure required for SAGD recovery.

Cost of steam generation and distribution

Three different fuel cases are considered as outlined above. Only the cost of steam generation is considered to be dependent on the fuel source. The costs of water treating were considered to be the same despite the fact that the lower generation pressures and temperatures in the CANDU cases should place less stringent constraints on the boiler feed water. The lower distribution pressure for the CANDU cases will require larger diameter pipe but with a lower wall thickness and less insulation. For the purposes of this study it is assumed that these changes in the distribution system will compensate and that the cost will be the same. As the SAGD process only utilises the latent heat of the injected steam, the mass of steam required at a pressure of 15 MPa is increased by a factor of 1.12 for the natural gas cases to achieve the equivalent heat injection at the design basis of 3.43 MPa.

Steam cost of supply

Based on the cost data developed in the previous section, the cost of supplying steam to an in-situ SAGD operation can be computed. Calculated supply costs do not include allowances for income taxes, royalties or finance charges. They are computed at several discount rates to account for the time value of money. Steam supply costs were determined for the different cases using both natural gas and nuclear steam. The supply costs were computed at discount rate of eight percent.

Electricity costs were not generally included as part of the cost of generating steam. One exception was the additional electrical power require to boost the boiler feed water to 15 MPa required in the natural gas cases. In the nuclear power cases, it was assumed that part of the electrical power requirements for the SAGD operation were generated by the reactor. This power generated by the reactor for SAGD operations was treated as a credit for the cases involving nuclear generation.

Summary of oil sand studies

Based on the work completed during this study, a number of conclusions can be drawn.

1. The CANDU nuclear reactor is a viable alternative for supplying steam to in-situ oil sand projects based on SAGD technology for the recovery of bitumen when natural gas price exceeds $2.50 Canadian per gigajoule.

2. Because of the risk associated with high initial capital investment, it is unlikely that the nuclear option will be seriously considered until the following conditions are met:

   a) An existing SAGD bitumen facility (production >100 000 BPD) based on the use of natural gas as fuel is in production in an area not exceeding 10 000 hectares
   b) The economics of implementing the CANDU option can be justified on the bases of reducing the operating cost of this large facility.
   c) The public is convinced of the environmental advantages of using a nuclear reactor over other alternatives such as coal- and bitumen-based fuels

3. As the development of the Canadian oil sand resource matures, energy requirements to produce the in-situ steam and hydrogen for upgrading become so large that the natural gas found in Alberta will not be sufficient to meet the extraction energy requirement. Uranium, coal and fuel derived from bitumen are the only resources large enough to provide viable alternatives.

4. A CANDU nuclear reactor would provide sufficient steam so that the oil sand resource within a reasonable distance (5–6 km) to transport steam using a surface distribution system would be exploited within 30 to 40 years.

5. The production of electrolytic hydrogen for upgrading of bitumen is a viable option to natural gas at low electrical power rates (below $15/MWH).

6. Should limits be set on the levels of carbon dioxide emissions to meet Canada's commitment to the Kyoto protocol, the use of nuclear energy to develop the oil sands with SAGD technology can be a feasible alternative. Reference 13 concludes that a large amount of future CO_2 emission can be eliminated by coupling the SAGD process with nuclear steam and electricity production.

WATER DESALINATION USING NUCLEAR FISSION HEAT

The need for potable water has been amply demonstrated [14, 15]. The IAEA has done a considerable amount of work in the area of nuclear desalination [16, 17]. The main drawback of desalination, however, is that it is an energy intensive process. Therefore, the increasing global demand for desalted water creates a tremendous collateral demand for new sources of electrical power.

In addition to providing a means of meeting regional electricity demand, the CANDU nuclear reactor can also serve as an energy source for a reverse osmosis (RO) seawater desalination plant. In conjunction with the use of electrical energy, waste heat from the reactor is used in the desalination plant to improve the efficiency of the RO process. This is done by using condenser cooling water as a source of preheated feedwater for the RO system. The system design also makes use of advanced feedwater pretreatment and sophisticated design optimization analyses. The net result is improved efficiency of energy utilization, increased potable water production capability, reduced product water cost and reduced environmental burden. This approach to the integration of a seawater desalination plant has the advantage of maximizing the benefits of system integration while at the same time minimizing the impacts of physical interaction between the two systems. Consequently, transients in one plant do not necessarily have adverse effects on the other.

The CANDESAL design concept

Only two commercially-available and proven technologies are available. These are multi-effect distillation (MED) and reverse osmosis. More detailed preliminary studies showed that in order to match the required thermal conditions for MED, changes were required to the reactor's balance of plant design that were both expensive to implement and led to reduced electrical generating efficiency. Moreover, the loss in electrical generating capability was such that the overall water and electrical production capacity was not as great as that which could be achieved using RO combined with the standard CANDU design.

The use of ultrafiltration (UF) pretreatment provides high quality feed water to the RO process. This serves to protect the RO membranes and enhance their performance, thereby reducing the total number of RO membranes required and increasing their lifetime. The result is reduced plant capital cost and reduced demand for membrane maintenance and replacement.

The RO process depends on a complex set of relationships between a variety of operating parameters including the preheated feedwater temperature, feedwater analysis, RO system operating pressure, membrane feed flow rate, recovery, permeate quality and flow rate, and brine concentration and flow rate. A comprehensive design optimization based on integrated system performance analyses is carried out to establish the best balance of design features and performance characteristics to achieve specified performance objectives and reduced water production costs.

This approach to the coupling of seawater desalination systems with nuclear reactors has the advantage of maximizing the benefits of system integration while at the same time minimizing the impact of physical interaction between the two systems. This is extremely important, since there must be a high degree of assurance that unanticipated operating transients in the desalination unit do not have an adverse impact on either reactor safety or operational reliability. Conversely, it would be undesirable to have reactor upsets, whether anticipated or not, that would require shutdown of the water production plant.

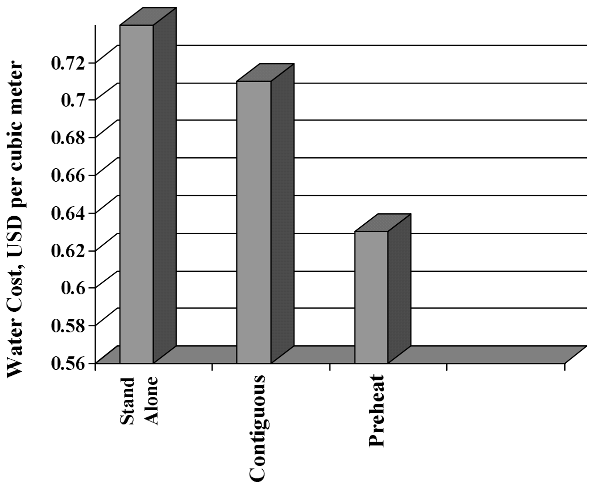

Figure 8. Effect of nuclear preheat on desalinated water cost, Indonesia site.

As the CANDESAL nuclear desalination/ Cogeneration system design concept has evolved, it has developed in a direction which allows use of a standardized off-the-shelf CANDU reactor without modification, while at the same time accruing significant benefits from systems integration due to improved performance characteristics and energy utilization.

Results

A detailed engineering and economic study has been made for a site in Indonesia [18], including the effects of several parameters such as feedwater pressure, feedwater temperature, and salt concentration. The results of the economic assessment are summarized briefly in Figure 8.

The results of this preliminary design evaluation study show the technical, performance and economic characteristics of a large scale RO seawater desalination plant coupled with a CANDU 6 nuclear power plant, operating under conditions typical of a site in Indonesia. Based on site characteristics for the site supplied by Indonesian participants in the study, a reference design configuration was developed and described for a reverse osmosis nuclear desalination system that meets those conditions.

The Muria reference design consists of ten identical RO trains operating in parallel. Each train is capable of producing slightly more than 24 000 m3/d at a 29(C reference seawater temperature. The total water production capacity of the plant is thus in excess of 240 000 m3/d, with potable water quality consistent with World Health Organization standards, having a total dissolved solids content of less than 500 ppm.

The estimated capital cost of the plant is on the order of US$236 million, with a cost of water production of about US$0.63/$m^3$ based on standard economic assumptions used by the IAEA in their economic analyses. The cost of water from a stand-alone RO plant under these same conditions is about US$0.74, or about 17% higher.

Based on the results of this work it can be concluded that a nuclear desalination facility based on the integration of the CANDU 6 reactor with a reverse osmosis desalination plant can be configured to operate effectively and efficiently even under the high seawater salinity and temperature conditions prevailing in Indonesia. Such a plant can provide for the cogeneration of water and electricity, using waste heat from the electrical generation process to improve the efficiency of the water generation process. Such a plant is based on currently available, proven technologies and could be implemented on request. The innovative approach taken to the application of RO technology leads to performance and economic advantages that represent significant improvements over other alternatives currently available.

CONCLUSIONS

There can be no firm conclusions at this time because the gigantic task of replacing the world's existing primary energy resources with new ones has only begun. However, some useful guidelines might be found in the work done so far.

It is abundantly obvious that nuclear fission energy is the best candidate for supply of at least a major part of the world's large scale energy supplies. It is a mature, viable, and economic technology. The most important remaining developments of this technology are reduction in electricity generating cost and the adoption of advance nuclear fuel cycles.

Several mature associated technologies (electrolysis, motive-fuels production, motive-fuels handling, heavy-oil extraction, desalination, heavy water production, etc.) already exist. It seems that there are no major technological barriers in the way of a bright and sustainable energy future.

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PROSPECT FOR THE APPLICATION OF NHR IN CHINA

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Abstract

Space heating is a very important field in the non-electrical application of nuclear energy. The nuclear heating reactor has been developed by the Institute of Nuclear Energy Technology, Tsinghua University. A series of advanced features are adopted in the design for the NHR to achieve a higher standard of safety. Therefore any off-site emergency actions such as sheltering, evacuation, relocation and decontamination are not needed. The NHR can be used in district heating, seawater desalination, providing low-pressure steam for industrial process and as a heat source for agriculture and breed aquatics. The NHR R&D program, the present status and the prospect based on the analysis from the points of safety, economy and society will be discussed in this paper.

1. Retrospection of R&D on the NHR

In order to solve the problems related to the energy shortage, environmental pollution and the overburdened transportation systems in China and also to expand the utilization fields of nuclear energy except electricity generation, the study of using a simpler reactor for district heating was started in the Institute of Nuclear Energy Technology (INET), Tsinghua University. In 1983 and 1984, INET conducted successful tests of nuclear district heating using the existing pool type research reactor. After that, the several reactor types that can be used for heating were compared and vessel type nuclear heating reactor (NHR) was selected to be as the main research direction.

During the Seventh Five-year Plan (1985~1990), the research and development works on the key technology, such as natural circulation, integrated design, passive residual heat removal and hydraulic control rod driving system, were completed. A 5MW test NHR (NHR-5) was begun to construction on March 1986 and put into operation in 1989. Since then, the reactor had been successfully operated for district heating and some special experiments. At the same time, the development of NHR-200 as industrial demonstration had already been included as a project of the Eighth Five-year Plan as well as the State Development Plan (1991~2000).

On the basis of the success of NHR-5, a commercial sized NHR with output of 200MW thermal power (NHR-200) has been developed by INET. On Oct. 1992, the Daqing Oilfield Administration, as the applicant of the first NHR-200 demonstration plant, applied to construct the “Daqing Oilfield 200MW Nuclear Heating Plant”. After argumentation by the experts, the application was approved by the State Plan Committee on Jun. 1993. The feasibility study report was passed the evaluation by the China International Project Consultation Company and the State Development Bank on Dec. 1993 and Jul. 1994, respectively. The siting report and the environmental impact assessment report was approved by the respective authorities. On August 1995, the State Council confirmed to carry the project into execution.

On the basis of the basic design, the application for the construction permit of the first NHR-200 demonstration plant was submitted to the Chinese National Nuclear Safety Administration on Nov. 1995 with the PSAR. From then on the licensing procedure started formally and lasted about one year. Finally, the applicant got the construction permit on Dec. 1996.

2. Main conclusion resulted from NHR-5

The initial fuel loading of the NHR-5 was achieved on Oct. 9, 1989. The experimental reactor went to the first criticality on Nov. 3, 1989. And the rated power was reached on Dec. 16, 1989. Since then, the reactor had been successfully operated for district heating until the main objectives were actualized. The main items of experiments conducted on NHR-5 after the commissioning are summarized in Table I.

Table I. Main test items at NHR-5

Test item	Description	Main results
Positive reactivity introduce	2mk introduce by control rod withdrawn	Power peak reached 1.18 times
ATWS	Isolating all of heat sink Position of control rods not change	Reactor power decreased to dissipation level automatically
LOCA	Nature circulation interrupted in RPV by drawn water from primary circuit	Residual heat can be removed by condensation on primary heat exchanger
Heat & electricity co-generation	Third circuit replaced by steam circuit, cooling water of condenser used for heating	Test is successful. Whole facility is operated smoothly.
Refrigeration	Reactor and steam generator coupled with a LiBr refrigeration cycle device	Chilled water with temperature of 7°C can be provided
Seawater desalination	Reactor and steam generator coupled with a small MED device	Result shows that the design is successful

The conclusion from the results of the operation and experiments can be summarized as follows:

- The NHR-5 achieved a high availability of 99% (Factual operation days divided by planned operation days). There were four times unplanned scram, in which two were caused by loss of offsite power, one was caused by failure of air compressor and the other one was caused by operator mistake in in-service inspection on the instrument power source. The longest shutdown time caused by unplanned scram was five hours.
- The measures against radioactivity contamination were very effective. The detecting data showed that the radioactive level in intermediate circuit and heating grid was same with the one of local background level.
- The rad waste produced in the three years could be disregarded. The amount of wastewater was $8.5m^3$ with β radioactive level of 14Bq/l. The radioactive level of gaseous influent at the outlet of the stack was as low as the background.

• The average professional collective dose was 5.67mSv-man per year. The environmental surveillance proved that there was no influence on the environment in around region by the operation of NHR-5.

3. The first NHR-200 demonstration plant

The first NHR-200 demonstration plant is designed for Daqing Oilfield, which is located in the northeast of China. The planned site is 105km apart from Qiqihaer in northwest and 160km apart from Harbin in southeast. The center of the site is at 124°53′15″ of east-longitude and 46°35′28″ of north-latitude [1].

The area of the site is about 300m × 400m. It is divided into three regions according to its functions, namely main installation region, auxiliary installation region and management buildings. The main installation region includes all of buildings that would be contaminated, such as the reactor building, drainage pool and stack. The position for the second reactor building is also reserved in this region. The auxiliary installation region is a non-radioactivity one, including make-up, circulating water system, transformer station and so on. The total area is considered and arranged as the requirements of 2 × NHR-200 plants. The heating grid is constituted by NHR-200 plant and 9 sub-heating-stations. The sketch map of the heating grid is shown in Fig. 1.

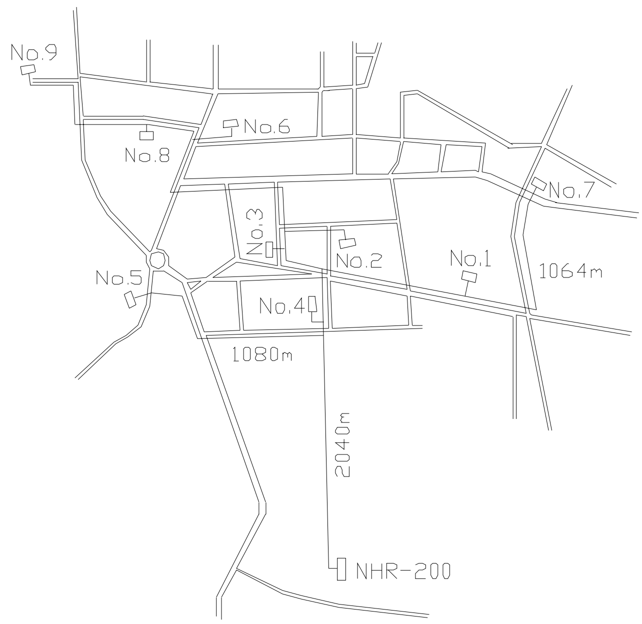

Fig.1 Sketch map of the heating grid

In Daqing Oilfield, the lowest temperature was reached to –36.2°C and heating period is longer. Up to 1996, there are about 14 million square meters of building area and more than one million tons of oil and one hundred million cubic meters of gas per year were consumed by heating. In the basic design, 3.8 million square meters of building area, in which the heating load is 319MW, will be heated by NHR-200. NHR-200 can supply 196MW as basic load and oil-fired boilers will share others.

Heating is needed for 181days in Daqing area. NHR-200 can produce a quantity of heat of 3.13×10^6GJ. The self-consumption of the heat energy by NHR-200 plant is 2% of the rated power. Therefore, NHR-200 can provide 3.07×10^6GJ to the grid as basic load. Using NHR-200 as heating source will save about one hundred thousand tons of oil per year.

In order to optimize the property of NHR-200, some existing oil-fired boilers are remained to regulate the peak load of heating. They need 1.34×10^4 tones of oil to provide 123kW for heating together with NHR-200 per year when outside temperature is lower than –6.97°C. Comparison with no boilers to regulate the peak load of heating, the heating area increases by 64% and the 85.4% of the total quantity of heat is provided by NHR-200 during the heating period.

In order to further evaluate the economy of NHR-200, the economic analysis was carried depends upon the budgetary estimate and the heating cost of the oil-fired boilers. It is required that the analysis result should be compared with heat price that is given by the replaced oil-fired boilers.

The cost can be divided into two parts: the cost of the nuclear heating plant and the cost of the heating grid. The items taken into account in the cost are fuel, material, electricity, water, personnel, maintenance, depreciation, management and capital cost. They are summarized into three parts: investment cost, O&M cost and fuel cost. The decommissioning cost is not included in and will be taken after the loan reimbursement. The cost composition is listed in Table 2. The specific cost is 31.54 RMB¥/GJ at the condition of average interest rate of 7.65%.

Table 2. Proportional costs of NHR-200 plant

Investment	61.74%
O&M	29.02%
Fuel	9.24%

In order to reflect the actual status, three oil-fired boiler plants are selected to calculate the heating cost. And three representative oil prices are selected to compare the heating cost. The heating costs of the oil-fired boiler plants are listed in Table 3. The result can be given through specific cost comparison of NHR-200 and oil-fired boilers at the different oil price (see Table 4). The oil price in Table 3 and Table 4 are the CIF (cost, insurance and freight) and the exchange rate between Chinese RMB and US Dollar is 1US$= 8.3RMB¥. The heating cost of NHR-200 is lower than that of oil-fired boiler. In addition, the sensitive analysis shows that the heating cost of NHR-200 will be 33.2 RMB¥/GJ if the investment increase with 10% and that the heating cost will be decrease with 5% if the interest rate drops per 1% [2].

Table 3 Oil-fired boiler plants heating costs

Oil price*		Cost, RMB¥/GJ			
RMB¥/tonne	US$/Barrel	No. 1	No.2	No. 3	Ave.
830	14.06	37.87	33.44	31.39	34.23
958	16.23	41.71	37.28	35.24	38.08
1,280	21.68	51.39	46.96	44.91	47.76

* The oil price means CIF (cost, insurance and freight).

Table 4 Cost comparison between NHR-200 and oil-fired boiler plant

		NHR-200	Oil-fired boiler plant		
Oil price*	RMB¥/tonne	/	830	958	1,280
	US$/Barrel		14.06	16.23	21.68
Fuel cost , RMB¥/GJ		2.9	25.5	29.4	39.3
Production cost, RMB¥/GJ		19.2	32.6	36.5	46.2
Total cost, RMB¥/GJ		31.54	34.23	38.08	47.76

4. Prospect

Coal is the most important primary energy and the coal annual consumption has be 1.2×10^9 tones in China since 1993, which is about 30% of the total coal consumption in the world and about 75% of total primary energy consumption in China. Because of the distribution, exploiting process, and itself quality of the coal, some serious problems have been encountered duo to a massive coal burning.

Coal resource is rich in northwest and poor in east and south. In east, coal reserves is about 23.2%, annual output is about 50% and annual consumption is 73% of the total in China. It is estimated that the annual output will decrease to 40% and consumption will retain at 73% in 2010. Therefore, several hundred million tones of coal will be transported at that time. In addition, long distance transportation must increase the cost of the coal and make the overburdened transportation system extremely.

Except the problems on production and transportation, the coal exploitation results ground surface caving in, a mass of gangue, and a large quantity of wastewater. Moreover, unrestricted use of coal has resulted in serious influence on environment. CO_2 is the main greenhouse gas. Burning of fossil fuel is the primary contributor, in which coal as solid fuel is two times of liquid and gaseous fuels such as oil and gas for the concentration of CO_2. It is estimated that CO_2 produced by coal burning is 85% of the total released into atmosphere in China. SO_2 is the secondary byproduct by coal burning. In China, total amount of 1.8×10^7 tones of SO_2, in which about 80~90% was produced by coal, was released in 1994. It is the primary reason of acid rain. The third main source of environment pollution produced by coal is soot. The quantity of coal-soot released that is more than 70% of the total countrywide is the main reason of the floating pollution at atmosphere. In addition, the residue after coal burning, of which more than 2.5×10^8 tones will be let off countrywide, is also to pollute environment [3]. Even for radioactivity, the level produced by coal burning is higher than the one by nuclear heating.

The purpose of developing nuclear heating is to replace fossil fuel with nuclear fuel and to provide a possible solution for the serious problems above mentioned. For the NHR, it should be located in the vicinity of the user duo to the consideration of heat transportation and economy. Therefore, the more stringent safety requirements for NHR-200 are adopted in design.

Based on the existing general criteria for NPP and the specific characteristics of the NHR-200, no off-site emergency actions such as sheltering, evacuation, relocation and decontamination is the fundamental safety requirement for the NHR in case of all credible accidents [4]. In other words, the radioactive release from the NHR-200 has to be reduced to such a low level that off-site emergency actions are not needed. It means that for the plant site a maximum individual resulted from an activity should be not larger than 5mSv. The results of safety analysis indicates that a release of 3.7×10^{13} Bq ^{131}I can be the limitation for the maximum credible accident [5]. Correspondingly, more rigorous requirements in terms of fuel element damage, DNBR and fuel temperature limits are specified. The reactor core has to be covered by coolant in all loss of coolant accident (LOCA) cases. In addition, a 250m in radius non-residential area and a 2km in radius area of restricted development surrounding the site are required.

To meet the requirement of the general safety objective for the NHR-200 a series of technical measures [6] are adopted in design. The main technical and safety features can be briefly summarized as follows:

• Integrated design. That the core will be always covered by coolant is one of the fundamental design criteria for the NHR-200. The integrated arrangement ensures that the possibility of large LOCA caused by the breaking of main pipe can be excluded.

• Natural circulation. Both normal operation and shutdown cooling, natural circulation is the only one operation pattern in the primary circuit. The residual heat removal system, which is composed by two independent trains, is the most important safety system. The residual heat is also removed by a passive pattern, natural circulation.

• Hydraulic control rod driving. It meets the requirement of "fail-safe" principle i.e. control rods will drop into the core automatically in case of lose of power supply, depressurization of RPV, pipe break and pump shut down events. In addition, a boric acid injection system as a second shutdown system will be operated when the event of anticipated transient without scram (ATWS) occurs.

• Self-pressurized performance. The primary pressure is maintained by the saturate steam pressure corresponding the core outlet temperature and a certain inventory of nitrogen.

• Effective isolating from radioactivity. The multi-barriers, including fuel cladding, primary pressure boundary, steel containment and a secondary containment, compose the multi-defenses against the release of radioactive substance. In addition, the nuclear heat supply system is composed by triple loops, i.e. the primary circuit in the RPV, an intermediate circuit and the heating grid. The working pressure in the intermediate circuit is higher than that in the heating grid and that in the RPV, so that the pollution of radioactivity can be prevented from and the safety of the heating grid can be ensured.

As a result, NHR-200 is satisfied not only in economy but also in the radiation safety. For the environmental protection, NHR-200 is more attractive. In case of the same output thermal power of 200MW, same heating quantity of 3.07×10^6GJ, burnup of 3×10^4MWd/tu, the efficiencies of oil-fired boiler and coal-fired boiler are 80% and 70% respectively, the energy consumption comparison for the NHR-200, oil-fired boiler and coal-fired boiler are listed in Table 5.

Table 5 Energy consumption unit: tonne

Energy needed	NHR-200	Oil-fired boiler	Coal-fired boiler	Difference
Nuclear fuel*	~1	0	0	+1
Oil	1.34×10^4	10.76×10^4	/	-9.42×10^4
Coal (2.09×10^7J/kg)	2.7×10^4	/	25×10^4	-22.3×10^4

* With enrichment of 3%.

It can be resulted from the table that one tonne of nuclear fuel consumed by NHR-200 can instead of about one hundred thousand tones of oil or more than two hundred thousand tones of coal. Therefore, environment will benefit greatly from using nuclear fuel to replace coal or oil as heating source (see Table 6). With the same thermal power, an NHR-200 can reduce the release of 3.85×10^5 tones of CO_2, 6×10^3 tones of SO_2, 1.6×10^3 tones of nitrogen oxide, 5×10^3 tones of soot and 5×10^4 tones of residue

[7]. From the point of view of environmental protection, it can be considered that nuclear is a clean energy resource.

Table 6 Environment impact by different heating plants, tonne/year

	Fossil-fired boiler			NHR-200
	Oil	Coal	Gas	
Thermal power	200MW	200MW	200MW	200MW
Carbon dioxide	200 000	385 000	204 600	0
Sulphur oxide	1800	6000	146	0
Heavy metal	0.2	0.4	0.2	0
Nitrogen oxide	619	1,600	807	0
Radioactivity(mSv/ person·year)	0.001	0.013	0.001	0.0004
Soot	49.2	5000	30.9	0
Residue	1.5	50 000	0.5	0
High level rad waste	0	0	0	~1
Transportation for fuel	100 000	250 000	$10^8 m^3$/year	~1

Consequently, nuclear heating is helpful to environmental protection because no baleful gases, such as CO_2, SO_2 etc., will be released during the process. The level of radioactivity that is about one thirtieth of the core-fired boilers is lower than the value envisioned by people. The NHR-200 as a safe, clean and economic energy resource could be widely used in district heating, air conditioning, seawater desalination, providing process steam, and supplying heat source for agriculture and breed aquatics. Some cities and large enterprises have shown that they are very interested in introducing the NHR-200 into their local energy system. The NHR-200 is also helpful to ensure that the economy of China will be developed continuable in 21 century.

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NUCLEAR DESALINATION IN EGYPT: ACTIVITIES AND PROSPECTS

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Abstract

The main source of freshwater resources in Egypt is the River Nile. The Egyptian share of the Nile water was limited to 55.5×10^9 m^3/year in the Nile Water Treaty concluded with Sudan in 1959. Due to the rapid population growth, the annual per capita freshwater resources declined from 2560 m^3 in 1955 to 970 m^3 in 1995. Consequently, desalination plants of various sizes and technologies have been introduced to Egypt in the past three decades. The Egyptian desalination inventory increased from less than 2000 m^3/day in 1970 to almost 175 000 m^3/day in 1997, of which 54% was seawater desalination. The energy-intensive seawater desalination technologies are expected to play an increasing role in mitigating future potable water deficit in Egypt. Egypt has been considering for a number of years the introduction of nuclear energy to meat the combined challenge of increasing electricity and water demand on one hand and the limited primary energy and water resources on the other hand. In this regard, Egypt has been carrying a number of national, regional and international activities. This paper presents an overview of the Egyptian activities in the field of nuclear desalination including, feasibility studies and R&D activities. The results of recent studies are presented regarding: quantification of seawater desalination market in Egypt and preliminary economic assessment of potable water production by various combinations of energy sources and desalination processes proposed for El-Dabaa site.

1. INTRODUCTION

Egypt is located in the extreme northeast of Africa and consists of approximately one million square kilometers of land. As shown in Figure 1, the country can generally be organized into four major areas, namely: (i) The Nile Valley and Delta; (ii) The Western Desert; (iii) The Eastern Desert; and (iv) Sinai Peninsula. For the most part, Egypt lies within the temperate zone, and the bioclimate varies from arid to extremely arid. The River Nile is the main source of renewable fresh water. Owing to halt in Upper Nile conservation projects, Nile water cannot exceed 55.5-billion m^3/year fixed by the 1959 Agreement with the Sudan, at least in the foreseeable future. Other conservation projects within Egypt and the development of rechargeable groundwater resources are expected to have limited impact on the Egyptian renewable fresh water resources as can be seen in Table 1.

TABLE 1. PROJECTED DEVELOPMENT OF THE EGYPTIAN FRESH WATER RESOURCES

Year	Fresh Water Resources, billion m^3/year			Total
	Surface	Ground	Others	
1997	55.5	3.75	5.85	65.10
2002	55.5	4.90	7.00	67.40
2007	55.5	4.90	7.00	67.40
2012	55.5	4.90	7.00	67.40
2017	55.5	4.90	7.00	67.40

Source: Reference [1]

Due to the rapid population growth, the annual per capita freshwater resources (SWC) declined from 2560 m^3 in 1955 to 835 m^3 in 1997. This indicates that Egypt has been declining from a water abundant country (SWC >1667 m^3/capita) to a water stressed country (SWC < 1000 m^3/capita). Consequently, desalination plants of various sizes and technologies have been introduced to Egypt in the past three decades. The Egyptian contracted land based desalting plants rated more than 100 m^3/day per unit increased from less than 2000 m^3/day in 1970 to almost 175,000 m^3/day in 1997, of which 54% was seawater desalination [2]. The development of the Egyptian desalination inventory by process and feedwater type is shown in Table 2.

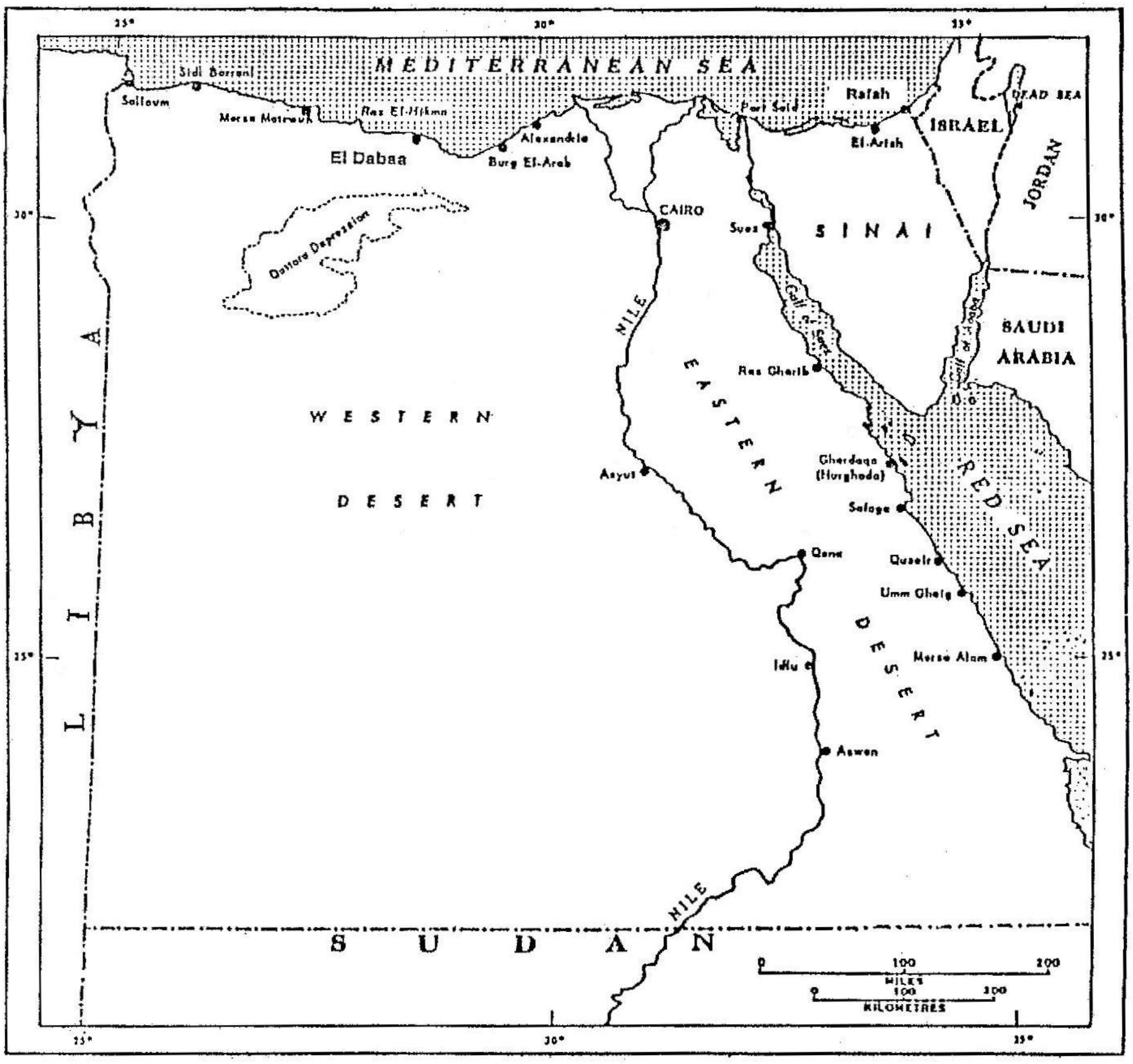

FIG. 1. General map of Egypt.

TABLE 2. DEVELOPMENT OF THE EGYPTIAN DESALINATION INVENTORY, m^3/d

Operation Year	Seawater					Brackish Water				TOTAL
	MED	MSF	VC	SWRO	Total	MED	ED	BWRO	Total	
1972	0	200	0	0	200	957	492	0	1449	1649
1977	0	3800	0	0	3800	957	1604	1300	3861	7661
1982	0	3800	0	0	3800	957	6228	5009	11194	14994
1987	120	4540	3740	3205	11605	957	22559	10999	34515	46120
1992	420	5790	4660	14103	24973	957	25269	22519	48745	73718
1997	1620	11382	7850	43309	64161	957	31749	39403	72109	136270

Source: Wangnick Consulting GMBH, "1998 IDA Worldwide Desalting Plants Inventory", Report No. 15, June 1998

Table 3 shows the future seawater desalination inventory as projected by two studies, namely the Academy of Scientific Research and Technology study (ASRT) and a study carried out for the International Atomic Energy (IAEA). In the ASRT study [1], the future deficit in potable water supply was estimated based on developmental plans for other consuming sectors and projected potable water demand. It was also assumed that, part of the future deficit would be covered through: (i) redirecting part of the irrigation water probably by changing the crop pattern and/or utilization of sewage water for irrigation; (ii) rationalization.

of potable water consumption and reduction of distribution losses; and (iii) desalination of brackish water. Hence, seawater desalination was assumed to cover only 10% of the deficit in potable water supply. The details of the methodology have been explained in an earlier publication [3].

In the IAEA study [4], the projections were based on historical records of installed seawater desalting capacity [5], known orders for new capacities to be installed over the next few years and the experience of the specialists. In the case of Egypt, the extrapolation of historical data, carried out in the IAEA study, could only be true if the overall surplus pattern prevails. Therefore, the results of that study as far as Egypt is concerned may be considered as an Optimistic Scenario, where no deficit in potable water supply is foreseen. In fact recent IDA accounts of operational desalination plants inventory in Egypt [2], indicates that the actual 1997 figures are more than 50% higher than the IAEA study as shown in Table 2. The energy-intensive seawater desalination technologies are expected to play an increasing role in mitigating future potable water deficit in Egypt.

Published data [6-8] on total primary energy supply (TPES) in Egypt during the period 1972-1997 are shown in Table 4. Analysis of this data indicates that most of primary energy consumed in Egypt has been of hydrocarbon sources (oil and gas). Due to the limited hydro and coal resources in Egypt the share of hydrocarbon energy in TPES increased from 90.2% in 1972 to 94.9% in 1997. However, over the same period oil share of TPES has declined from 89.3% to 63.7 and gas share of TPES has increased from 0.9% to 31.2%. This reflects the fact that while oil reserves in Egypt are being depleted, gas reserves has been increasing with new discoveries. Unless new discoveries are made crude oil reserves will be depleted 10 years, natural gas reserves in 60 years and coal reserves in 17 years. Hydropower is nearly fully utilized. There is a potential for solar and wind energies but the technology for large-scale electricity production is not yet economic.

TABLE 3. PROJECTED SEAWATER DESALINATION MARKET IN EGYPT

Year	Potable Water, Mm^3/d			Desalination, 1000 Mm^3/d	
	Supply	**Demand**	**Deficit**	**IAEA Study[4]**	**ASRT Study[1]**
1997	10.246	9.715	-0.531	41,700	-
2002	7.232	10.783	3.551	70,900	355,100
2007	5.863	11.791	5.890	120,400	589,000
2012	4.493	12.753	8.260	204,400	826,000
2017	3.123	13.673	10.550	347,200	1,055,000

Source: Reference [1]

TABLE 4. DEVELOPMENT OF TOTAL PRIMARY ENERGY SUPPLY, '000 toe

Year	TPES, '000 toe					Structure of Supply, %			
	Coal	**Oil**	**Gas**	**Hydro**	**Total**	**Coal**	**Oil**	**Gas**	**Hydro**
1972	341.3	7132.1	72.9	443.7	7990.0	4.3	89.3	0.9	5.6
1977	630.6	10037.8	353.0	777.2	11798.6	5.3	85.1	3.0	6.6
1982	717.8	14876.4	2035.0	900.8	18530.0	3.9	80.3	11.0	4.9
1987	768.8	20932.7	4713.9	744.6	27160.0	2.8	77.1	17.4	2.7
1992	735.0	22594.2	7985.6	834.2	32149.0	2.3	70.3	24.8	2.6
1997	875.0	23721.8	11608.3	1051.9	37257.0	2.3	63.7	31.2	2.8

Source: References [6-8]

The total final energy consumption (TFC) increased in this period at an average annual growth rate of 6.1%. In the same period, electricity consumption increased at an average annual growth rate of 8.3%. This is reflected in the continuous increase in electricity share in TFC from 9.9% in 1972 to 16.4% in 1997. The development of electricity generation by source, over the period 1972-1997 is depicted in Table 5.

Due to the increasing demand for electricity and the limited hydro resources, the demand has been met primarily through installation of fossil fuel thermal power plants. This led to the decline of the hydro share in total electricity generation from 64% in 1972 to about 21% in 1997. The utilization of natural gas in electricity generation increased dramatically over the past two decades [6-8]. In 1997, the total installed generating capacity was 13.3 GW. The demand for both primary energy and electricity is expected to continue at higher growth rates in the future due to the ambitious governmental plans aiming at increasing the gross domestic product (GDP) at an average annual growth rate of 8% over the next 20 years. The projected installed capacities for three scenarios of economic growth are shown in Figure 2.

Nuclear Energy can play an important role in providing future energy needs, ensuring better energy mix, and saving fossil fuel for future generations to be used mainly for chemical industries. Generally, wherever there is a need for potable water to supply a population center, an industry, or both, there is also a need for energy, particularly electricity. Nuclear reactors providing electricity and/or heat to a desalination plant can in principle, also supply the electricity market. A nuclear reactor designed to meet the combined energy demand resulting form water desalination and the electricity market, would be larger and consequently takes advantage of economics of the scale.

TABLE 5. DEVELOPMENT OF ELECTRICITY GENERATION BY SOURCE

Year	Electricity Generation, GWh				Structure of Generation, %		
	Oil	Gas	Hydro	Total	Oil	Gas	Hydro
1972	2884	0	5159	8043	35.9	0.0	64.1
1977	4607	488	9037	14131	32.6	3.5	63.9
1982	9717	2385	10474	22576	43.0	10.6	46.4
1987	18751	10436	8658	37845	49.5	27.6	22.9
1992	18037	19241	9700	46978	38.4	41.0	20.6
1997	14235	32950	12225	59410	24.0	55.5	20.6

Source: References [6-8]

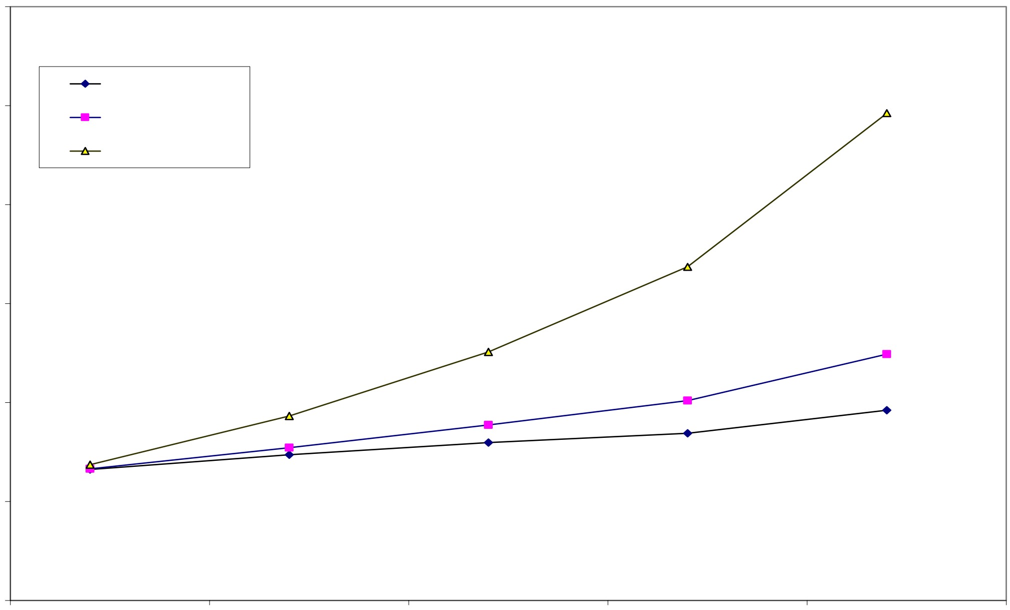

Fig. 2. Projections of future generating capacities.

2. DESALINATION PROCESSES

Previous work [9-11] identified RO, MSF and MED as the most interesting large-scale water production processes. All are proven by experience and are commercially available. While RO requires electricity, MSF and MED use mainly thermal energy and to a lesser extent electrical energy. RO has the lowest energy consumption and the largest development potential [12], particularly when the RO feed water is preheated in the power plant condenser.

The MSF process is the most widely used desalination process, especially in the Middle Eastern countries, where oil is plentiful. Inherently, however, the MED process has lower energy consumption, lower investment cost, and appears to be less sensitive to corrosion, scaling and fouling than the MSF process [10,11]. In addition, its operational flexibility in partial load is higher. Therefore, MED and RO were selected as the preferable technologies for nuclear desalination.

3. ENERGY SOURCES

A set of criteria utilized in a recent IAEA study [12] was used for initial screening of 'available reactors'. In a second stage of the selection process, only Pressurized Water Reactors (PWRs) and Pressurized Heavy Water Reactors (PHWRs) were considered for economic and/or safety reasons. This resulted in the following list of reactors which could be considered for nuclear desalination, and which appear to offer the best prospects for near term commercial deployment in Egypt:

(1)	Large Size PWR	(900–1000 MW(e))	Various designs
(2)	Medium Size PWR	(400–800 MW(e))	AP-600 (USA)
(3)	Medium Size PHWR	(400–800 MW(e))	CANDU-3 (Canada)
			CANDU-6 (Canada)
(4)	Small Size PWR	(25–300 MW(e))	NP-300 (France)

All operate in the co-generation mode, (electricity and heat), that was found to be more economic than heat-only options [10, 11].

4. ECONOMIC ANALYSIS OF DESALINATION OPTIONS

To estimate the cost of power and water for the nuclear and fossil desalination option, the IAEA Co-generation/Desalination Economic Evaluation (CDEE) spreadsheet program [14] was used. The details of this methodology are described in several IAEA publications [10, 13].

Assumptions included 8% real interest rate and US $ 15.5/bbl crude oil price with 2% real escalation. The economic assessment was performed using cost data for nuclear energy sources provided by the vendors in 1991 during previous IAEA study [10], restated in 1996 US $. The operation reference date was assumed to be January 1, 2017 to tie in with the projected water demands developed for the year 2017. A 10% increase in the base construction cost has been included to allow for the added costs of installing such plants in Egypt.

The resulting levelized water costs in US $/m^3 at the desalination plant outlet in various sites are summarized in Table 6. It is clear from the Table that under the assumptions made for the various combinations in each site; the most economic nuclear and fossil options are CANDU-6/RO and Combined Cycle/RO, respectively. The results of the economic evaluation confirm the results obtained during the IAEA generic study [10] and the North African regional study [11] that, the levelized water costs for both the nuclear and fossil options were in the same range. Higher oil prices and/or lower interest rates favors the nuclear options. The most economic desalination process is the contiguous RO with preheat.

5. CURRENT ACTIVITIES

During the 1960s, the feasibility of using nuclear reactors for seawater desalination was investigated by the International Atomic Energy Agency (IAEA) as well as by individual member states including Egypt. However, in the 1970s the interest in nuclear desalination became less than the other applications such as electricity generation, district heating and industrial uses of process heat.

TABLE 6. SUMMARY OF LEVELIZED WATER COSTS

ENERGY SOURCE	POWER (MWe)	COST OF DESALTED WATER IN VARIOUS SITES, US $/m^3					
		Site I (172 000 m^3/d)		Site II (108 000 m^3/d)		Site III (87 000 m^3/d)	
		MED	**RO**	**MED**	**RO**	**MED**	**RO**
I-Nuclear							
NP-300	300	1.140	0.840	1.172	0.876	1.376	0.901
CANDU-3	450	1.113	0.779	1.203	0.814	1.270	0.839
AP-600	600	**1.026**	0.765	**1.070**	0.784	**1.100**	0.796
CANDU-6	660	1.087	**0.750**	1.133	**0.768**	1.165	**0.780**
II-Fossil							
Combined cycle	**350**	**0.900**	**0.784**	**0.918**	**0.770**	**0.930**	**0.784**
Steam turbine	600	1.042	0.795	1.055	0.813	1.069	0.832

Bold numbers are the most economic nuclear and fossil coupling options

Towards the end of the 1980s, renewed interest in utilizing nuclear energy for seawater desalination, as well as, interest in nuclear reactors in the small and medium range, were indicated by a number of developing countries including Egypt. As a result, Egypt participated in a number of international studies in the field of nuclear desalination [9–12] and carried out with the IAEA technical assistance a feasibility study on the introduction of SMPRs to Egypt [13].

The main concerns regarding the nuclear power/desalination are: (a) The viability if the nuclear option; (b) The economic competitiveness of the nuclear option; and (c) The safety issues. To address these concerns, Egypt has adopted an integrated approach that builds on previous national and international experience. In this regard, Egypt has initiated, with the IAEA technical assistance, a number of inter-related projects. In the following Sub-sections these projects are briefly reviewed.

5.1. Comparative assessment of strategies and option for electricity generation in Egypt up to 2020

The Egyptian Ministry of Electricity and Energy (MEE) approached the International Atomic Energy Agency (IAEA) for assistance in carrying out a comparative assessment study of the MEE strategies and options for meeting the increasing demand for electricity. The IAEA approved the project EGY/0/015 to provide the required technical assistance. It was agreed that, the main tool for the comparative assessment would be the DECADES package, developed by the IAEA and other international organizations to simulate the environmental and economic impacts of the various energy chains.

The main objective of the study is to determine the optimal electricity generation mix up to the year 2020, including nuclear and renewable (solar and wind) energies. In this respect, the following tasks are addressed:

1. Simulation and assessment of Renewable Energy Options (REO), in particular thermal/solar and wind plants.
2. Simulation and assessment of Grid Inter-Connection (GIC) with neighboring countries.
3. Simulation and assessment of Independent Power Producers (IPP).
4. Investigate the possible role of nuclear power in the electricity generation mix.

To date, the country specific database has been compiled through a number of Expert Missions. Because the DECADES package does not include models for renewable energy, grid inter-connection or independent power producers, work is currently progressing with the assistance of the IAEA to develop these models. The study is expected to be completed by the end of 2000.

5.2. Investigation of feedwater preheating effect on RO performance

Although the relationship between feedwater temperature and membrane permeability is well known, the idea of utilizing the condenser's cooling water as a source of feed for RO systems did not appear until early 1994 [14]. This concept has been adopted and investigated by the IAEA in all subsequent studies [11, 12]. These and other studies [1, 3] have shown that there is a potentially significant economic and performance benefit through the combined effects of feedwater preheating and system design optimization. These conclusions have been drawn, however, from analyses and preliminary design studies without any experimental validation.

Experimental validation is of extreme importance in the confidence building process, particularly when other experts [12] argue that elevated temperatures may result in higher product water salinity, more rapid membrane fouling, greater membrane compaction, reduction in membrane lifetime and that saving in total water cost by elevating temperature from 15~18 C to 30 C would be in the range of 3% only.

In view of the possible role of RO desalination technology in any future Egyptian nuclear desalination program and the need to validate the concept of RO feedwater preheating, NPPA has decided to construct an experimental RO facility at its site in EI-Dabaa, with the following objectives:

1. *Overall:* to investigate experimentally whether the projected performance and economic improvements of preheated feedwater can be realized in actual operation. The intent is to simulate as closely as possible performance characteristics that would be expected to occur in commercial large-scale RO seawater desalination plant.

2. *Short term (3 years):* to study the effect of feedwater temperature and pressure on RO membrane performance characteristics over a range of temperatures (20-45 C) and pressures (55-69 bar). The intent is to gather data on all aspects of system operation, utilizing membranes from three different manufacturers, so that sufficient data analysis is possible to determine if the performance and economic benefits suggested by the analytical models can in fact be demonstrated by experiments, and to determine the possible differences in results due to materials and type.

3. *Long term:* to study the effect of feedwater temperature and pressure on RO membrane performance characteristics *as* a function of time. The intent is to select one of the membranes used during the short4erm program for extended study to investigate possible reduction in membrane lifetime due to effects such as increased fouling or membrane compaction.

The results of this experimental work could have a strong influence on how the international nuclear desalination community perceives the value/benefit of feedwater preheating, and hence there is a common international interest in this project. Therefore, NPPA proposed in this research project as part of the IAEA coordinated research project "optimization of the coupling of nuclear reactors and desalination systems". In May 1998, the IAEA agreed to fund the project and a research contract was conclude with NPPA (10244/RO).

To date the following tasks have been completed: (a) Design of the experimental facility; (b) Development of detailed experimental program; and (c) Preparation of the technical specifications and tender documents. Call for bids will be made in June 1999 and construction could start in August.

5.3. Technical and economic feasibility of nuclear power/desalination plants in Dabaa

Due to the limited freshwater and primary energy resources in Egypt, nuclear power could play an important role in meeting future needs of electricity and potable water. In this regard, Egypt has already qualified El-Dabaa as a site for the first nuclear power plant. Egypt has also participated in an earlier regional project evaluating feasibility of nuclear desalination in North Africa [11] and carried out a preliminary study [1]. The country is now at the point of making a decision as to whether to go forward with nuclear desalination. Therefore, this TC Project (EGY/4/040), was requested by NPPA and approved by the IAEA Board of Governors in November 1998.

The main objective of the project is to provide the decision-maker with all the necessary information regarding the technical and economical feasibility and viability of the nuclear option for electricity generation and seawater desalination. The project is expected to provide the following information:

- Analysis of the Egyptian economic and re-establishment of the future needs for both electricity and water.
- Quantification of water needs in El-Dabaa area.
- Technical (including safety) and economic evaluation of Nuclear Desalination systems.
- Local participation capabilities and the impact of the Egyptian developmental efforts.
- Financing requirements and methods.
- The necessary conditions for launching Nuclear Desalination program.

The project was initiated in January 1999 with a kick-off meeting. The purpose of the meeting was to confirm the object and scope of the project, completion time, responsibilities, assistance required from the Agency and to prepare an Action Plan for the project. The project will be completed by the end of 2000.

5.4. Establishing a quality assurance program for NPPA

The Nuclear Power Plants Authority (NPPA) has carried out an extensive site qualification program and an infrastructure-upgrading program, so that it can commence without delay with the execution of the Nuclear Power/Desalination project whenever the decision is taken. In this regard, NPPA has started the procedures to obtain a site permit to construct the first nuclear plant on El-Dabaa Site. One of the essential documents, which will be submitted to the licensing Authority, is the Quality Assurance (QA) Manual. NPPA has already started a QA program and the preparation a QA Manual, which will be applied in all its activities. To finalize this, this TC Project (EGY/4/042), was requested by NPPA and approved by the IAEA Board of Governors in November 1998. The main objective of the project is to:

- Finalize NPPA QA manual up to the international standards.
- Enable NPPA staff to plan for full implementation of QA program.

The project which is expected to be completed by the end of 2000, will be carried out through the following activities:

- Analyzing NPPA activities in various phases of the first NPP project.
- Finalizing the draft QA Manual prepared by NPPA quality staff.
- Preparing QA sub manuals for two NPPA departments (Design and Procurement).
- Preparing procedures for:
  - ♦ Document Control.
  - ♦ Procurement.

6. CONCLUSIONS

In view of the limited resources of primary energy and fresh water resources, and the increasing demand for electricity and water, nuclear energy could play an important role in generating electricity and desalting water. A number of inter-related projects is currently in progress to address concerns regarding viability, competitiveness and safety of nuclear option.

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HYBRID MSF-RO NUCLEAR DESALINATION DEMONSTRATION PROJECT, KALPAKKAM — DESIGN OF PREHEAT RO SYSTEM

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Abstract

The Nuclear Desalination Demonstration Project (NDDP), Kalpakkam envisages setting up a 6300 m^3/d capacity hybrid MSF-RO plant based on the indigenous design. The plant will meet the process water requirements of MAPS and will augment the drinking water sources of the nearby complex. The hybrid design has the following advantages; (i) operation of RO plant using grid electricity during reactor shut down thus increasing the desalination plant availability; (ii) economic production of process water for steam generation for the power station after minor polishing of the product water from MSF plant; (iii) blending the products of MSF and RO plants to make drinking quality water; (iv) to utilize the cooling seawater from the reject stages of MSF plant as feed for the RO plant. This would either increase the RO product output or reduce the operating pressure of RO plant thereby improving the economics. This paper describes the basic process flow sheet of SWRO plant, part of the hybrid MSF-RO desalination plant. The SWRO plant consists of pretreatment system, high pressure pumping system with energy recovery turbine, RO modules and membrane cleaning system. The product water from SWRO plant having TDS of about 400 ppm will be mixed with product water of the MSF plant containing 10 ppm TDS. The projection of the permeate out put and TDS for a few SWRO commercial modules at different sea water temperatures are carried out to select the optimum conditions of the SWRO plant. The role of energy recovery from RO reject stream is also investigated. As the seawater TDS at our site is around 35,000 ppm it has been proposed to study the feasibility of using reject stream from RO plant as feed for the MSF plant in future.

Introduction

The Nuclear Desalination Demonstration (NDDP) Project at Kalpakkam consists of 1800 m^3/day (0.4 MGD) Seawater Reverse Osmosis (SWRO) plant and 4500 m^3/day (1 MGD) Multi Stage Flash (MSF) plant making a total production of 6300 m^3/day (1.4 MGD) water. The plant has been designed indigenously from the experience gained from MSF and SWRO pilot plants at Trombay. The steam required for the brine heater of the MSF plant would be drawn from 170 MW(e) Pressurised Heavy Water Reactor (PHWR), Madras Atomic Power Station (MAPS).

An MSF pilot plant of 425 m^3/day capacity had been earlier designed, constructed and operated . Similarly a small RO plant has been designed, constructed and commissioned. From the operation of these MSF and RO pilot plants, considerable experience have been gained and is utilized for the design of the 1.4 MGD plant. The MSF plant at Kalpakkam needs about 21 tonnes/h of saturated steam at 3 bar pressure. Sea water supply will be from the outlet streams of MAPS which will be at a slightly higher temp (32–34°C). Higher sea water temperature will be advantageous for SWRO plant since water flux of the membrane is about 2.5 per cent higher per degree temperature rise at a fixed pressure. The hybrid MSF-RO plant aims in reducing operation and maintenance (O & M) cost of the product water. A part (1000 m^3/day) of the high purity water (TDS = 10–20 ppm) from MSF plant is utilized as process water after minor polishing and the remaining water from MSF plant is mixed with 400 ppm of water of SWRO plant to produce drinking water (TDS — 150 ppm) with minimum post treatment of water produced either from MSF or SWRO plant. Both the plants have common sea water intake and outfall facilities. It is also possible to use cooling sea water from reject stage of the MSF plant as feed to RO plant. This will give a high throughput as its temperature is 6–8°C higher than ambient.

Salient Features of NDDP

The flowsheet of 6300 m^3/day desalination plant is given in Fig. 1. Steam for MSF plant (21 t/h) and electrical power for both MSF (0.6 MW(e)) and SWRO plant (0.5 MW(e)) are drawn from MAPS.

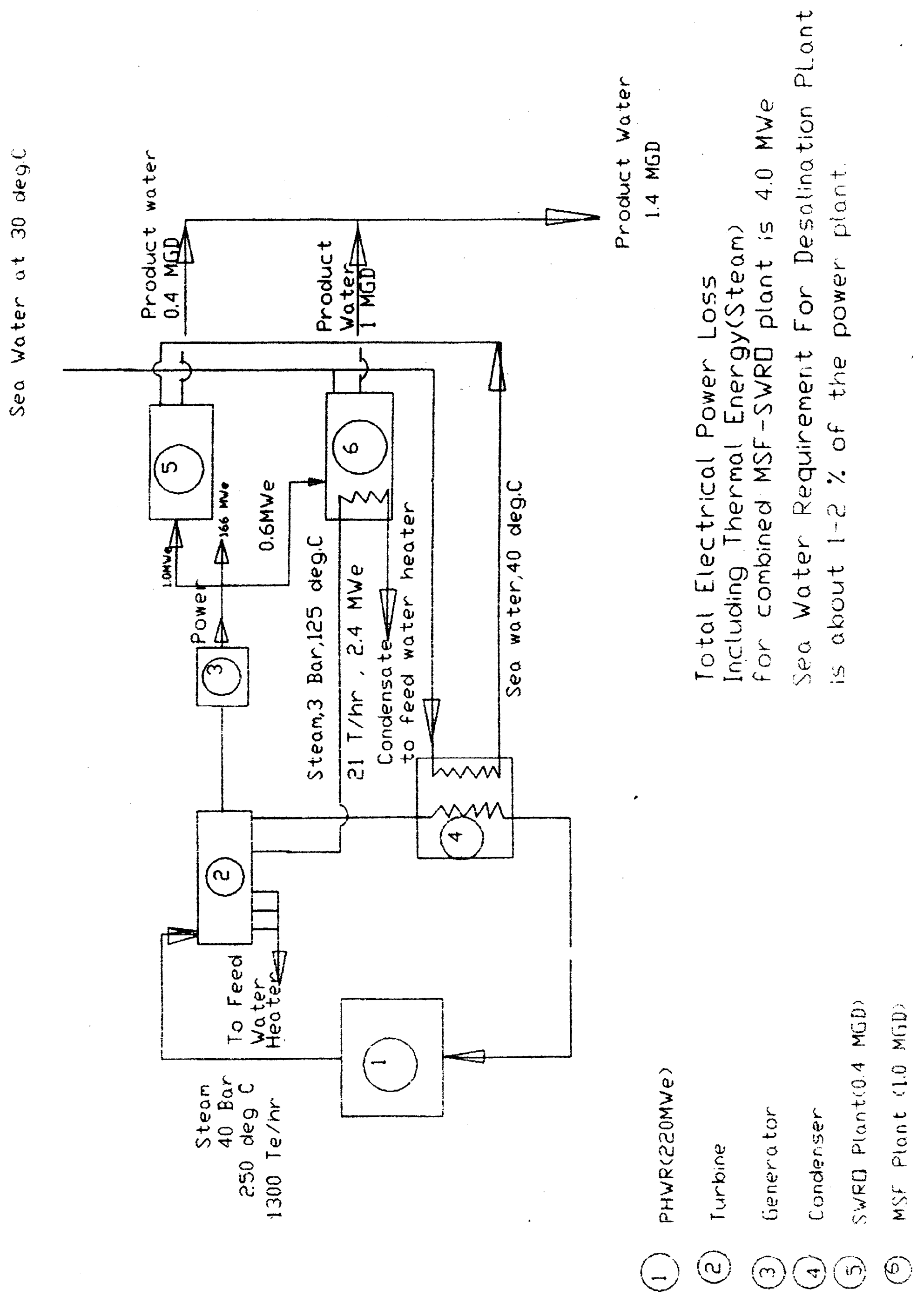

FIG. 1. Flow diagram of 1.4 MGD MSF-SWRO desalination plant coupled to 170MW(e) nuclear power station.

SWRO Plant

The process flowsheet of the 1800 m^3/day SWRO plant is shown in the Fig. 2. It consists of pretreatment system, high pressure pumping and module system, post treatment and auxiliary system. In seawater reverse osmosis plant, pretreatment of seawater is an important step to have optimum membrane life. Seawater composition at MAPS is given in Table 1.

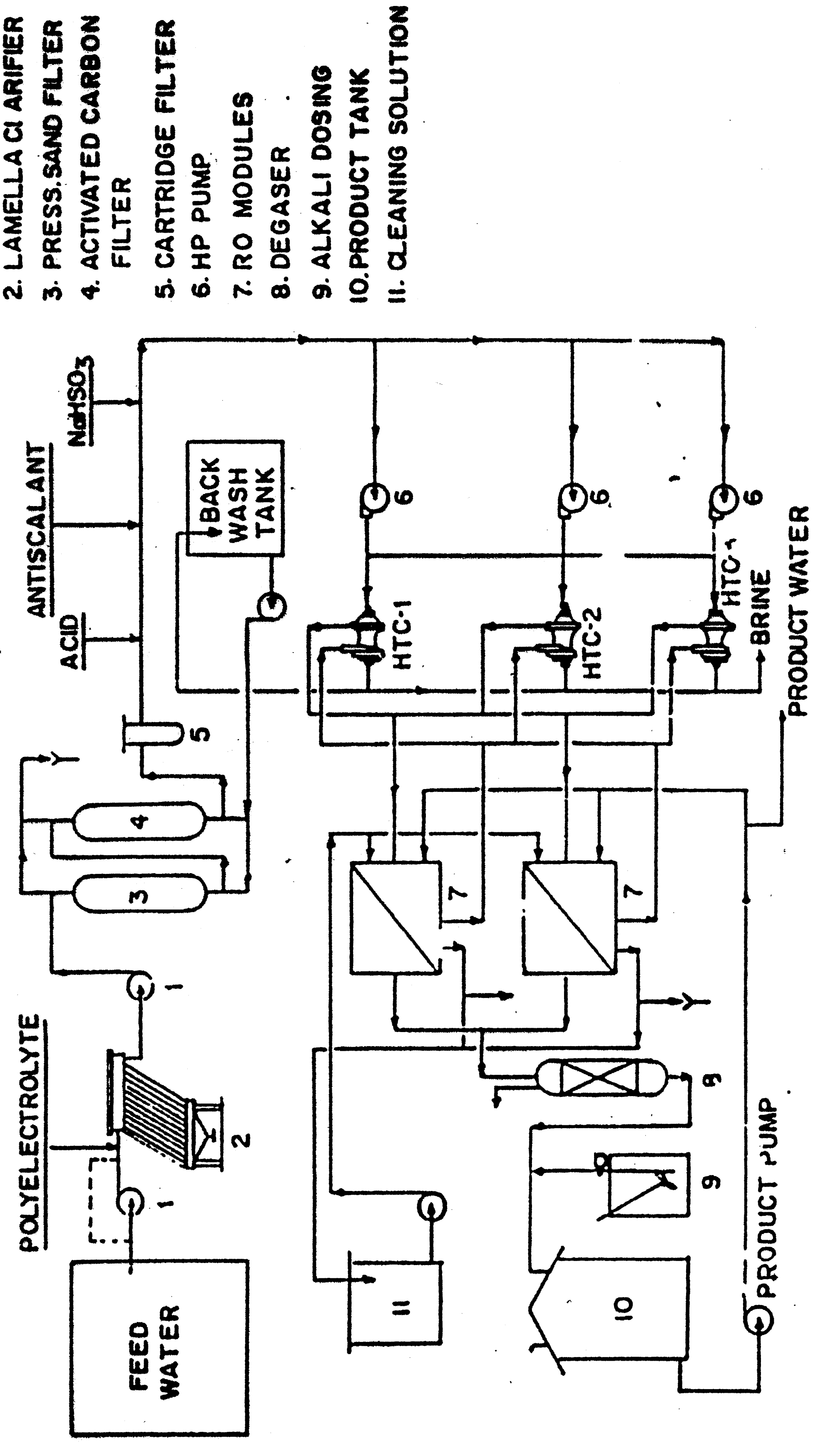

FIG. 2. Flowsheet for 1800 M^3/day (0.4 MGD) seawater RO plant at Kalpakkam.

Table 1. Sea water composition at Kalpakkam

pH	:	8.1
TDS by wt	:	35600
Total solids (ppm) by wt	:	36012
Suspended solids (ppm)	:	410
Total hardness (ppm $CaCO_3$)	:	6300
Sodium (ppm)	:	10556
Calcium (ppm)	:	400
Magnesium (ppm)	:	1272
Potassium (ppm)	:	380
Total alkalinity (ppm) $CaCO_3$	:	138
Chloride (ppm)	:	18981
Sulfate (ppm)	:	2650
Fluoride (ppm)	:	1.3
Iron (ppm)	:	0.1
Silica (ppm)	:	0.8

*Pretreatment section:*It consists of disinfection using chlorine, clarification (coagulation, flocculation and sedimentation) to remove major suspended load, media filtration to further reduce the suspended impurities, activated carbon filtration to reduce organic load and to remove chlorine, reduction of alkalinity by pH adjustment, addition of scale inhibitor, complete removal of chlorine by addition of sodium bisulfite and microcartridge filtration for removal of very fine particles. To minimise the scaling on the membrane, antiscalant, SHMP is also added.

High pressure pumping & RO modules: The RO modules are arranged in two parallel trains; each train producing 37.5 m^3/h. product. Each train is fed with pretreated seawater of 110m^3/h. at a pressure of 55 kg/cm^2. The membrane elements consist of TFC spiral wound polyamide membrane. (Element type is 8040 i.e. 8 inch diameter and 40 inches long and element capacity is 22 m^3/day). Total product recovery is 35%. Each train has 14 pressure vessels with 6 elements in each. Thus it makes a total of 28 pressure vessels and 168 elements in the plant. The technical specifications of the membrane & modules are given in the Table 2. Both the trains of RO modules together will produce 75 m^3/h. product water of 400 ppm average TDS.

Each train is fed with seawater by high pressure centrifugal pump coupled with Energy Recovery Turbine (ERT) and motor on the same shaft. There are total 3 pump sets two operating and one standby. Each pumpset has 110 m^3/h capacity @ 55 kg/cm^2 pressure. About 33% of the energy can be saved by use of the ERT. Both the pump and the ERT is made of second generation duplex stainless steel alloy 2205. Inflow to ERT is 70 m^3/h. at 53 kg/cm^2.

Post-treatment system: It consists of forced draught degasser to strip off the CO_2 and lime dosing for final pH and alkalinity adjustment.

Auxiliary system: It consists of module flushing system and membrane cleaning system. Cleaning chemicals are citric acid at pH of 3.5–4.0 adjusted with NH_4OH to remove carbonates and hydroxides followed by alkaline EDTA solution with 1% tri sodium phosphate for removal of sulfates, organic and biological slimes.

Table 2. Technical specifications of 1800 m^3/day SWRO plant

1.	Water produced per train	:	37.5 m^3/h.
2.	Number of trains	:	2
3.	Water produced from both trains	:	75 m^3/h.
4.	Seawater flow per train	:	110 m^3/h
5.	Seawater TDS	:	35000 ppm
6.	Feed water pH	:	6.5
7.	Number of modules/train	:	14
8.	Number of elements/module	:	6
9.	Total number of modules in the plant	:	28
10.	Total number of elements in the plant	:	168
11.	Membrane characteristics:		
	(a) Type	:	TFC spiral would element
	(b) Model	:	8” dia. × 40” long (8040 type)
	(c) Element capacity	:	22 m^3/day/element
	(d) Solute rejection of membrane	:	99.6% at standard test conditions
	(e) Flux decline coefficient per year	:	7%
	(f) Solute passage increase factor/a.	:	10%
12.	Design temperature	:	32°C
13.	Op. pressure	:	55 bar

Cost

The investment for the 1800 m^3/d SWRO plant and the cost of water are given in Table 3. The basis of calculation is mentioned in Table 4.

Table 3. Calculation of investment and water cost of SWRO plant

Plant capacity	:	1800 m^3/day
Investment, $	:	2 941 176
Plant factor	:	90
Power cost, $/kWh	:	.058
Power consumption kWh/m^3	:	6.10
Maintenance (as % of investment)	:	3
Fixed cost, $/$m^3$	:	0.62
Power cost, $/$m^3$	:	0.35
Chemical cost, $/$m^3$	:	0.03
Membrane replacement cost, $/$m^3$	:	0.14
Maintenance, $/$m^3$	:	0.15
Water cost, $/$m^3$	:	1.29

Table 4. Basis of calculation

Plant life, a.	:	25
Interest rate	:	12%
Membrane life, a.	:	3
Membrane cost, $/element	:	1500
Membrane elements	:	168
Inhibitor cost, $/kg	:	0.58
Inhibitor dosing, ppm	:	10
Acid cost, $/kg	:	0.09
Acid dosing, ppm	:	103

Water composition: The composition of permeate water produced from SWRO plant without post treatment is given in the Table 5 and the composition of water meant for drinking after addition of dosing chemicals is given in the Table 6.

Table 5. Projected permeate water composition without post treatment

TDS, ppm	:	336
Calcium, ppm	:	0.95
Magnesium, ppm	:	3.03
Sodium, ppm	:	120.43
Potassium, ppm	:	5.42
Strontium, ppm	:	003
Bicarbonate, $HCO'_{3,}$ ppm	:	1.52
Sulfate, SO_4'', ppm	:	6.81
Chloride, C'	:	195.13
Fluoride, F'	:	0.03
Carbondioxide	:	42.08
Ph	:	4.9
Langelier Saturation Index, LSI	:	-6.4
Ryzner Index, RI	:	17.6

Table 6. Water composition after post treatment

Dosing Chemicals ,$Ca(OH)_2$	:	36 ppm
pH after dosing $Ca(OH)_2$	:	8.43
Ca,ppm	:	20.4
Mg, ppm	:	3
Na, ppm	:	120
K, ppm	:	5.4
Sr, ppm	:	0.03
CO_3'', ppm	:	.8
HCO_3', ppm	:	58.3
SO_4'', ppm,	:	6.8
Cl', ppm	:	195.1
Total TDS	:	410
Langelier Saturation Index (LSI)	:	0.1
Ryzner Index (RI)	:	8.2

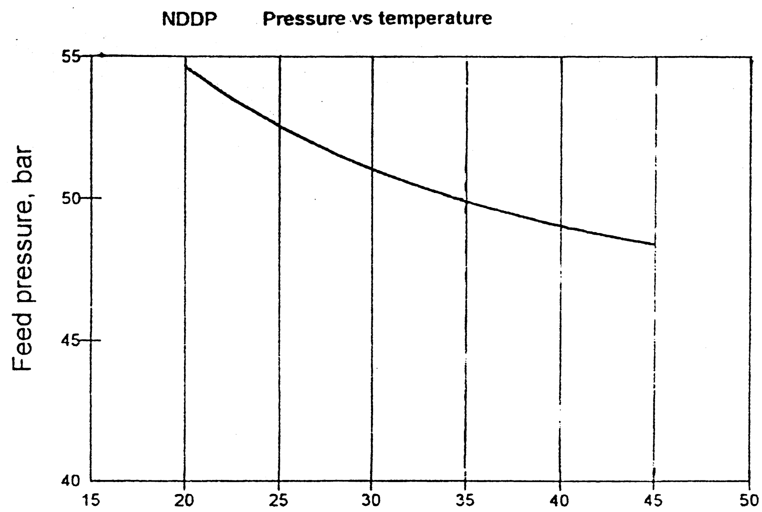

FIG. 3. Feed water temperature, C.

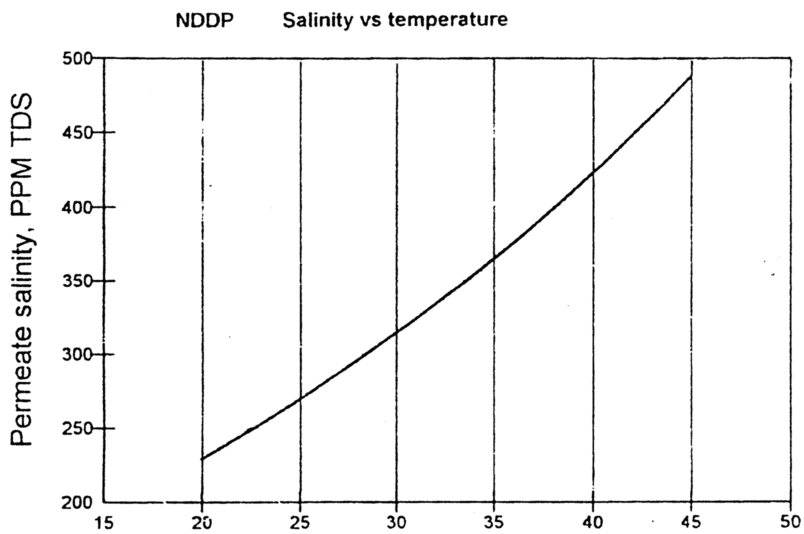

FIG. 4. Feed water temperature, C.

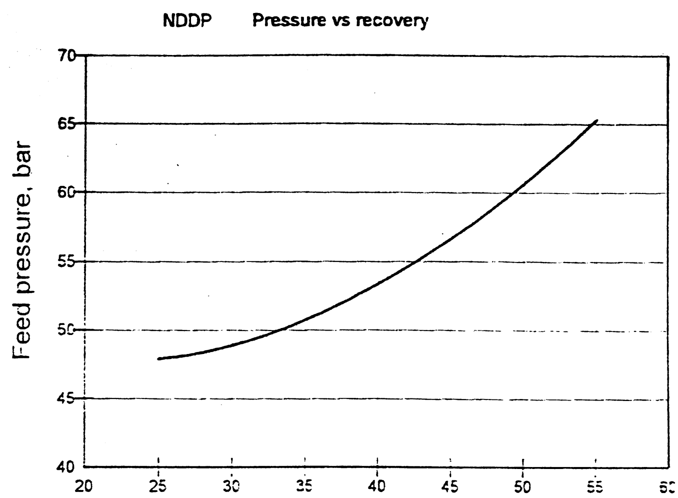

FIG. 5. Permeate recovery, %.

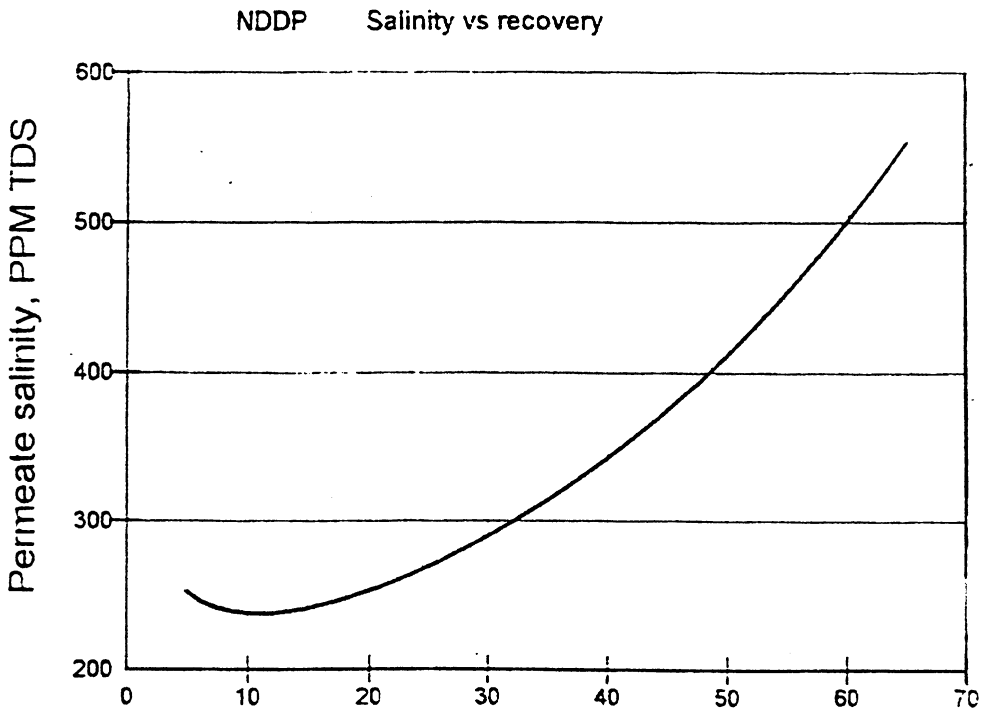

FIG. 6. Permeate recovery, %.

Elevated temperature operation: The projections made for reduction in applied pressure with increase in feed temperature as well as slight increase in permeate salinity is shown in Figs 3&4. Figures 5& 6 show increase in applied feed pressure and permeate salinity with increase in permeate recovery.

Reduction of energy consumption (future plan): SWRO has become more economical after commercialization of Energy Recovery Units. At present commercial ERUs are based on conversion of power through shaft and energy consumption is approximately 6–6.5 kWh/m^3 of product water. Introduction of Work Exchanger Energy Recovery Unit (WEER) can bring down energy consumption below 4 kWh/m^3 of water. In principle, the work exchanger transfers the fluid pressure in the reject stream to fluid pressure in the feed stream across a piston. NF followed by RO can be used for overall increase in recovery and hence will reduce the energy consumption.

PRE-PROJECT STUDY ON A DEMONSTRATION PLANT FOR SEAWATER DESALINATION USING A NUCLEAR HEATING REACTOR IN MOROCCO

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Abstract

This paper gives in the first part detailed information on the pre-project study on a demonstration plant for seawater desalination using heating reactor implemented by both Moroccan and Chinese sides. The main findings of the pre-project study are given in the second part.

1. INTRODUCTION

Knowing that prior studies carried out by IAEA have revealed that the use of nuclear energy for the desalination of sea water is technically feasible and may compete with fossil energy, Morocco has decided to carry out a specific study for Tan-Tan site, which will require 8 000 m^3/d of desalinated water by the year 2000.

To that end, in the framework of the co-operation between China and Morocco states, both sides with the assistance of IAEA, have implemented jointly a pre-project study adopting a 10 MW(th) nuclear heating reactor NHR-10, as its heat source and the vertical tube high temperature multi-effect evaporation process (MED-VTFE).

2. MAIN CHARACTERISTICS OF PRE-PROJECT STUDY

In order to perform this project, Morocco and China have established the two following documents:

(1) Agreement between the Moroccan Ministry of Energy and Mines and the China State Science and Technology Commission for cooperation in the pre-project study.
(2) Proposal for pre-project study.

The agreement was signed by authorities of both sides on 20 September 1996 in Rabat. The period of this agreement is 18 months starting from the mentioned date. The Moroccan side has appointed a technical committee including the representatives of the Moroccan departments involved in this project.

The objectives of a demonstration plant are:

– To build up technical confidence in the utilization of a nuclear heating reactor for desalination of seawater,
– To establish a database for reliable extrapolation of water production costs for a commercial nuclear desalination plant of the same type (200 MW(th); 160 000 m^3/day).

3. OBJECTIVES OF THE PRE-PROJECT STUDY

The basis task of the pre-project study is to indicate the possible location of the plant, lay down the technical basis for the reactor and the desalination plant, expound and verify its technology, safety, reliability, economy and availability. Its major objectives are as follows:
(1) Specify the concept design of the reactor and the desalination plant in the demonstration plant, and expound and verify its technical feasibility.
(2) Expound and verify the safety of the demonstration plant and the international standards.

(3) Estimate the investment of the project, assess the cost of the fresh water produced to expound and verify its economical feasibility and application prospects.

(4) Made a comprehensive conclusion on the necessity, safety, technical feasibility and economical viability to provide decision maker with the feasibility for the establishment of such a demonstration plant.

4. GENERAL DESCRIPTION OF THE NUCLEAR SEAWATER DESALINATION DEMONSTRATION PLANT

As the heat source for the desalination plant, the NHR-10 reactor provides the desalination plant with 105–130°C saturated steam via the primary circuit, the intermediate circuit and the steam supply circuit. In the VTFE first effect, the steam is cooled and condensed by seawater, and then flows back to the steam generator as feed water. The seawater which is heated by steam in the first effect becomes secondary steam which goes into the next effect. This evaporation-condensation process repeats further until the last effect, where the produced steam will be cooled and condensed by the feed seawater to the desalination plant. The feed seawater, taking the counter flow pattern, flows through every effect all the way, wherein pre-heated, into the first effect. Through evaporation at every effect, the salt content of the seawater becomes increasingly higher. At last, the seawater with high salt content will be discharged into the sea. The fresh water coming from one effect to the next one will flash thus increasing the heat recovery. To maintain the heat transfer efficiency of the evaporation-condensation process, vent facilities are designed for removal of non-condensable gases. Figure 1 gives the schematic diagram of nuclear desalination demonstration plant.

4.1. Nuclear steam supply system

Being different from nuclear power plants, the NHR-10 nuclear steam supply system consists of the reactor coolant circuit, intermediate circuit and the steam supply circuit. The four main heat exchangers in the reactor pressure vessel are divided into two groups, with two heat exchangers of each group being parallely connected to an intermediate circuit. There are two independent intermediate circuits, each having a steam generator. Steam flows from the two steam generators can be, alone or parallely, led to the steam inlet of the first effect of the two VTFE systems. Condensate of the live steam is sent back as feed water to the steam generator by condensate pumps. This simplified interface design can increase operational flexibility and thus enhance the operation availability.

The multi-barrier design, include active and passive barrier, is to assure that the fresh water produced shall not be radioactively polluted.

4.2. ***The vertical tube foamy flow evaporation system***

The VTFE system is a high temperature desalination process. Its direct coupling with the nuclear steam supply system is one of the best interface approaches. This coupling can simplify system design and operation.

The VTFE system consists of two units with each unit consisting of four towers. Each tower includes seven effects. Towers are connected by pipings. In every desalination unit there are 28 effects. The designed capacity of each unit is 4,000 m^3/d, thus the total design capacity of the water production plant is 2 × 4 000m^3/d. Figure 2 shows the arrangement of the 8 desalination towers. The first two towers containing effects 1–14 have a diameter of 3.0 m, and second two towers containing effects 15–28 have a diameter of 3.4 m. All towers have a height of 34 m.

The pre-heater in each effect is a long, vessel-tube type heat exchanger. Its vessel side is connected to the vessel side of the evaporator. The residual 15–20% heat of the steam after going through the evaporator is used to pre-heat the seawater.

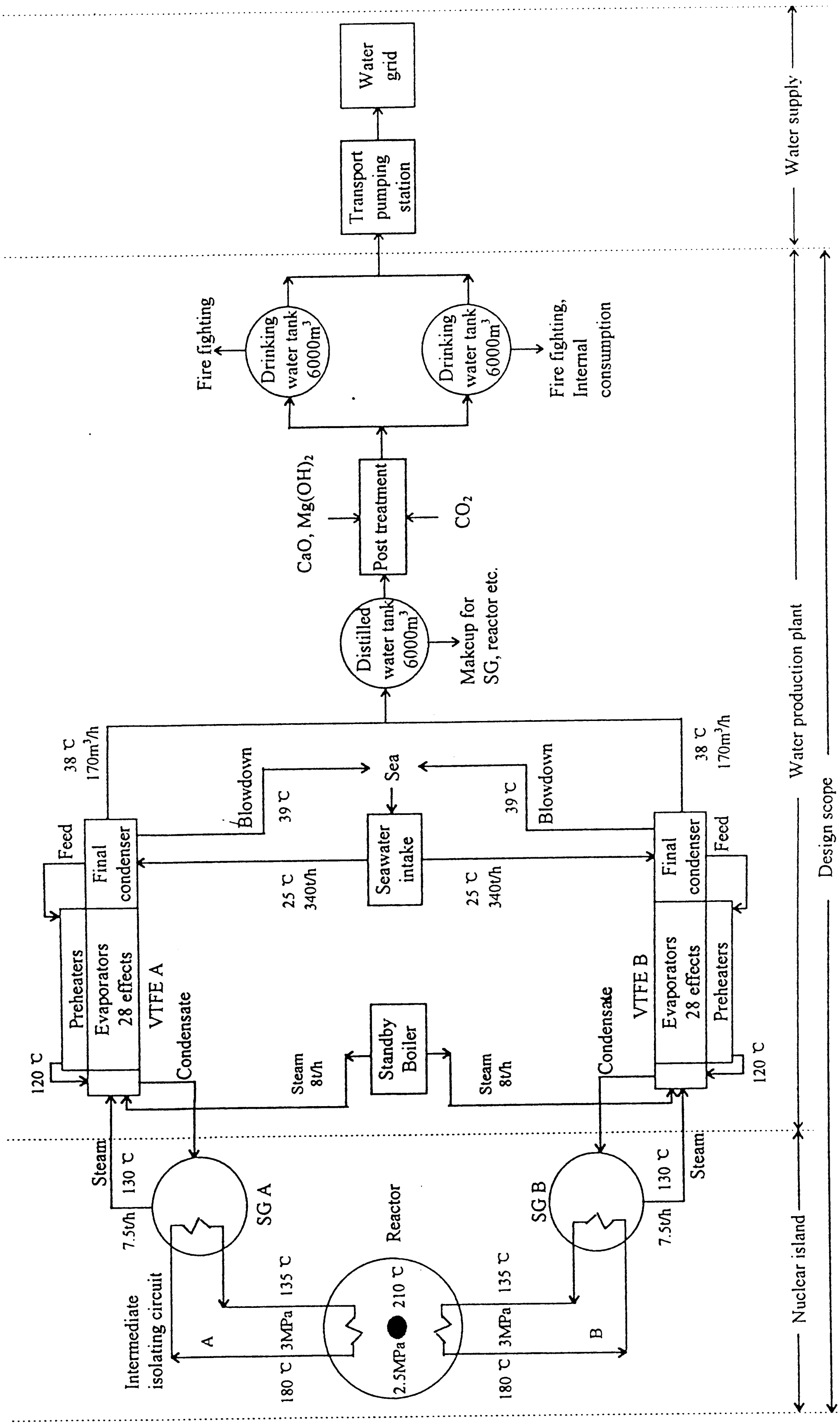

FIG. 1. Schematic diagram of nuclear desalination demonstration plant.

The feed seawater of the desalination plant is pre-treated to prevent from fouling on heat transfer surfaces, and the produced fresh water is to be post-treated, so that it meets drink water standards.

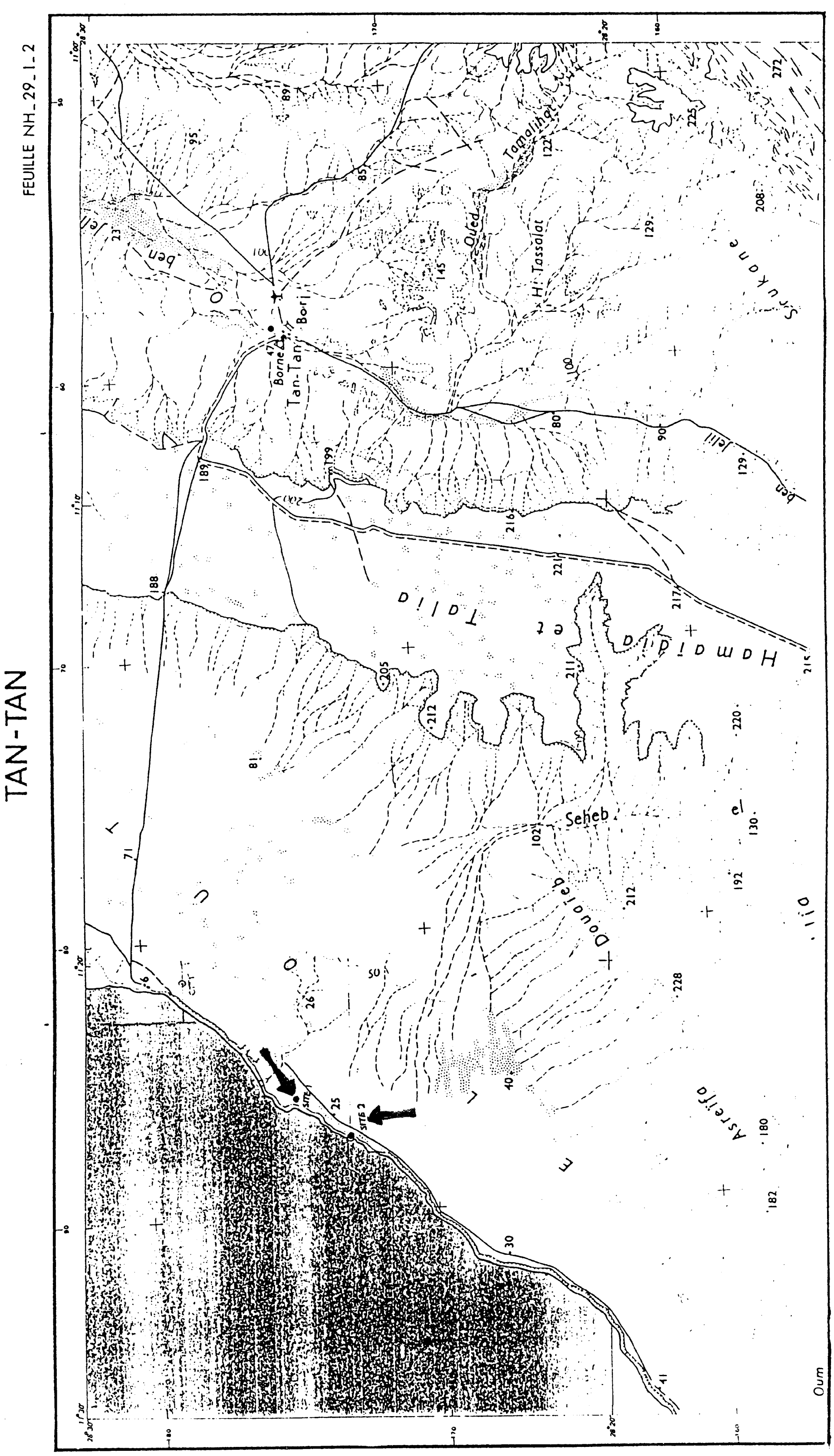

FIG. 2. The locations of proposed sites.

The key design parameters of the nuclear seawater desalination plant are listed in Table 1.

TABLE 1. GENERAL DESIGN PARAMETERS OF NDDP

Reactor thermal power	MW	10
Pressure	Mpa	2.5
Reactor outlet/inlet temperature	°C	210/180
Intermediate circuit		
pressure	Mpa	3.0
water temperature at steam generator outlet/inlet	°C	135/180
Steam temperature at outlet	°C	130
Steam flow rate	kg/s	4.37
Seawater desalination plant		
design capacity	m^3/d	8 000
process		VTFE
number of units		2
number of effects of every unit		28
capacity per unit	m^3/d	4 080
GOR		21.6
seawater inlet temperature	°C	25
flow rate	t/h	2 × 340
water quality (TDS)	ppm	20
Reactor operation cycles: first cycle	EFPa	5.89
second cycle	EFPa	1.92
Maintenance period of the desalination plant	a	3
Planned reactor outage	%	3
Unplanned reactor outage	%	4
Planned desalination plant outage: Arranged at the time with reactor planned outage	-	-
Unplanned desalination plant outage	%	4
Total availability	%	89
Reactor refueling cycle	EFPa	7.81
Reactor—desalination plant cooperative maintenance plan	a	3+3+2

5. DESIGN FEATURES

Since the reactor safety is of special importance to the public, the environment and the water quality, it must be specially addressed and emphasized in the design to make the reactor safety reaching a considerably high level. Safety analysis and the safety tests on the NHR-5 have shown this high level safety.

The NHR-10 is a vessel type light water reactor featured by integral arrangement, natural circulation, self-pressurisation and passive safety, with the NHR-5 as its proto-type.

The NHR-10 reactor is the heat source specifically designed for desalination.

5.1. Reactor safety

The NHR-10 is designed to have relatively strong negative temperature coefficient of reactivity, which inherently suppress abnormal power and temperature transients. Because the NHR-10 has low power densities, it has a relative big margin against the DNB. Therefore the integrity of fuel claddings will be ensured.

A series of measures have been taken in the design to make the reactor not sensitive to LOCA accidents. Under all design basis LOCA accidents, the reactor core remains flooded and cooled. The core-melt frequency of the NHR-10 should be lower than that of the present PWR power plants (10^{5}/a) by at least two orders of magnitude.

The NHR-10 is designed to have a big coolant inventory (~ 2.8 m^3/MW(th)) so that pressure transients during any accidents are rather gentle owing to a great thermal inertia in the primary system. The integrity of the coolant pressure boundary is maintained.

The multi-barriers for radioactivity confinement are designed for the NHR-10. They are: fuel claddings, the coolant pressure boundary and the containment as well as the well-sealed reactor building. This provides complete and effective barriers to radioactivity releases in all cases.

The above described safety features can assure the overall safety goals to be met; protection of the public safety, no pollution of the environment and protection of the investment.

5.2. Safety in fresh water production

In comparison to using conventional energy sources, radioactive pollution in water production processes should be prevented when using nuclear energy for seawater desalination. Sufficient consideration is given in coupling the NHR-10 and the VTFE systems.

Reliable multi-barriers are designed in the NHR-10 desalination system. There is in the reactor system an intermediate circuit whose pressure is higher than the primary circuit. This circuit separates the reactor coolant circuit and the steam supply circuit via the main heat exchangers and the steam generators. The steam supply circuit itself is separated from the radioactive reactor coolant to enter the fresh water systems, unless all the three barriers fail simultaneously, overlapped the failures of the pressure barrier of the intermediate circuit and isolation.

In addition, such means as on-line radioactivity monitoring or sampling analysis have been adopted in the intermediate circuit, the steam supply circuit and the fresh water production systems, in order to detect abnormal conditions and to take corresponding actions.

5.3. Common site

The NHR-10 reactor and the water plant have a common site. Because of the high level reactor safety, the site can be near the water users. This common site approach has the following advantages:

(1) Sharing on staffs and facilities for management, operation and maintenance,
(2) Sharing on infrastructure and common facilities, and
(3) Minimizing fresh water pipings.

This is beneficial to enhance the economy of the water plant.

5.4. Site selection and environment evaluation

Through a joint site survey by the experts from IAEA, INET as well as Morocco side, two candidates of sites were proposed. The locations of two proposed sites are shown in Fig. 2.

For proposed two sites it is not expected to encounter prohibitively adverse conditions which would preclude the construction and safe operation of the nuclear desalination plant. The influence on the environment and population are acceptably minor.

6. ECONOMIC ASSESSMENT

The total base investment cost of the project based on the price level in January 1998 is 38.00 MUSD where the 10 MW(th) nuclear heating reactor base investment cost is 22.50 MUSD, and the 2×4000 m^3/d VTFE desalination system base investment is 15.50 MUSD. The national participation taken into account is around 40 %. As the scale of the NHR-10 is very small, so the specific investment cost much higher than that of the commercially sized heating reactor.

The main parameters used in calculating the production water cost include construction lead time, economic life, load factor, discount rate, capital charges, interest during the construction period, nuclear fuel cycle cost, operation and maintenance cost etc., excluding the cost related to water storage, transportation and distribution.

The sensitivity analysis indicates that the changes in values of the main parameters affect the water cost according the methodology for the economic evaluation of cogeneration/desalination options (E.E.C.D) of IAEA. Calculations were performed for the following cases:

Different discount rate,

Different load factor assumptions,

Advanced or delayed construction lead time,

Different import duties and taxes in Morocco.

For the purpose of economic study of large commercial nuclear heating reactor desalination plant, the construction cost and the potable water cost of a 200 MW(th) heating reactor desalination plant are given in the Table 2.

The calculation results indicate that such nuclear desalination plant should be economically competitive with fossil one.

Table 2 shows an economic data summary for the reference cases of the 10 MW(th) and 200 MW(th) nuclear desalination plant.

7. CONCLUSIONS

- The design of 10 MW(th) nuclear seawater desalination demonstration plant, based on existed 5 MW(th) nuclear heating reactor and high temperature MED desalination technology is advanced and realistic. The nuclear heating reactor is possessed of good inherent and passive safety features.

A desalination system coupling with nuclear heating reactor can ensure the produced water from radioactivity contamination. There is no technical hinders that cannot be overcome during the implementation of this project.

TABLE 2. ECONOMIC DATA SUMMARY OF REFERENCE CASE

10 MW nuclear desalination plant		
Base construction cost of NHR-10	MUS$	22.50
Base construction cost of 2 × 4000 m^3/d MED water plant	MUS$	15.50
Interest rate during construction period	%	6.5
Water plant capacity	m^3/d	8000
Construction lead time	month	36
Economic life time	year	30
Load factor	%	89.6
Discount rate	%	10
Levelized water production cost	US$/$m^3$	2.79
200 MW nuclear dsalination plant		
Base construction cost of NHR-200	MUS$	97.28
Base construction cost of 4 × 43200 m^3/d MED water plant	MUS$	163.8
Interest during construction period	%	6.5
Water plant capacity	m^3/d	160000
Construction lead time	month	42
Economic life time	year	30
Load factor	%	91.5
Discount rate	%	10
Levelized water production cost	US$/$m^3$	0.998

- For the proposed two sites in Tan-Tan along the coast it is expected that there are no such conditions, which could preclude the construction and safe operation of the nuclear desalination plant. The impact on the environment and population is negligible.
- The producing cost of potable water is a little bit high due to small scale of the 10 MW(th) nuclear desalination plant and high discount rate. The price is around 2.79 USD/m^3.

The estimation of producing cost for a 200 MW nuclear desalination plant, which is in a economic scale with a capacity of 160 000 m^3/d, is about 0.95-1 USD/m^3. Therefore, it has good prospects in Morocco in the light of mitigating the problem of energy shortage and lack of fresh water resources.

THE SEAWATER DESALINATION NEEDS OF TUNISIA AFTER THE YEAR 2010

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Abstract

The supply of drinking water for north and Tunisia is guaranteed from surface water resources in the north and other subsurface resources. These resources will satisfy the water demand in this region until the year 2010 and 100 000 m^3/d by the year 2015. In the south of Tunisia, the water supply comes from local subsurface resources, including the lake water of the chotts. Maximum exploitation of these lakes, whose average salinity exceeds 2g/L, has already been reached. Therefore, non-conventional resources such as desalination have become unavoidable if the water quality is to be improved and the resources are to be maximized. The needs of this region will reach 80 000 m^3/d by the year 2010. This deficit can only be met by the desalination of seawater. At present, about 60 000 m^3/d of water is desalinated in the country using the reverse osmosis process and electric energy.

1. WATER RESOURCES

Rainfall in Tunisia varies between 200 and 1000 mm. the potential water resources amount to 4.4×10^9 m^3, of which 60% are waters. Of these resources, 68% have been tapped. It is expected that by the year 2010 all these water resources will have been exploited. Only 50% of these resources have a salinity of less than 1.5 g/L, while 30% have more than 4 g/L. It is worth noting that in 1994, 110×10^6 m^3 of waste water were treated, of which only 23% were used for irrigation.

2. WATER DEMAND

The water demand in 1994 reached 2.2×10^9 m^3, 80% of which were used for irrigation and the rest as potable water. For the latter, the average domestic consumption is less than 100 L/d, whereas the average tourist consumption is 550 L/d. the increase in potable water was about 4% per year during the period 1992–1996. Water consumption as a function of use (Fig. 1) is as follows: 67% for domestic consumption; 15% for collective consumption (administration, commerce, etc.); 12% for industrial consumption; and 6% for tourist consumption.

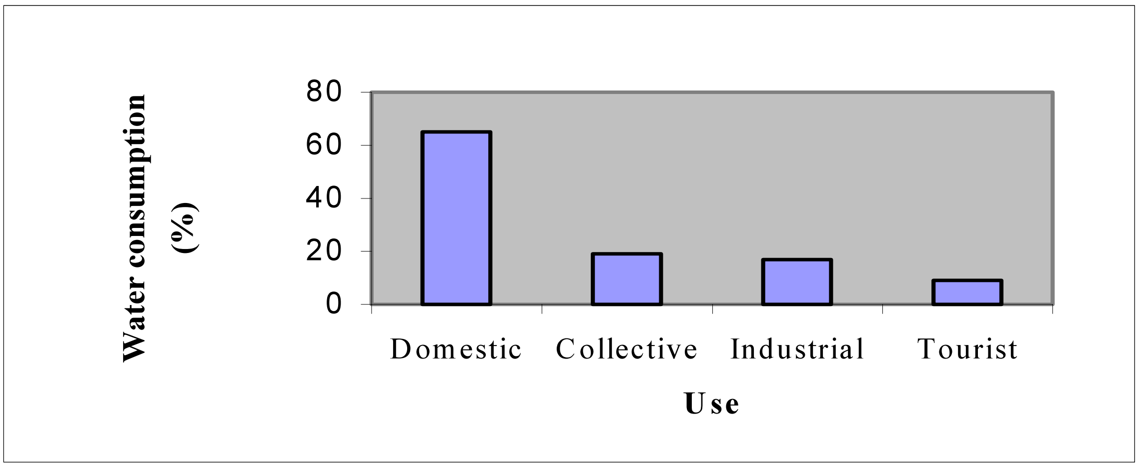


FIG. 1. Water consumption versus use.

3. DEMAND-RESOURCE BALANCE IN THE SOUTH OF TUNISIA

3.1. Introduction

Potable water in north and central Tunisia is supplied from surface waters in the north and from groundwater resources in the south. These resources will meet the demand until the year 2010. In the south of the county, notably in the regions of Gabès, Medenine and Tataouine, water is supplied from local ground water resources. These water resources, with an average salinity of more than 2 g/L, have already reached their maximum exploitable capacity. Therefore, non-conventional resources such as desalination have become unavoidable if the water quality is to be improved and the resources are to be maximized.

3.2. Water demand

The south of Tunsia defined as the counties of Medenine and Tataouine, and the regions of greater Gabès and Mareth (see map). In 1995, these areas had 700 000 inhabitants; this is expected to reach 1 050 000 in the year 2010 water demand will double between 1995 and 2010. Details of the water demand (1990–2010) are given in Table I.

TABLE I. WATER DEMAND

	1990	1995	2000	2005	2010
Population (103)	603.5	694.1	797.6	914.3	
1048.1					
Specific consumption (L/d)	67	70	73	76	79
Total consumption (m^3/d)	40.435	48.588	58.226	69.487	82.802
Distribution losses (%)	28.0	27.2	26.3	25.4	24.6
Requirements (with losses) (L/s) (peak)	910	1081	1280	1509	1779
Tourist requirements (L/s) (peak)241	293	353	421	474	
Total for Gabès (L/s) (peak)		512	598	719	861
Total for Medenine and Tataouine (L/s) peak)		862	1035	1211	1392
Total requirements (L/s)(peak)	1151	1374	1633	1930	2253

3.3. Water resources

3.3.1. Gabès region (Table II)

This region is supplied mainly from wells at El Fejij, which is located 40 km from Gabès, and also from local water resources in Mnara and Boulbaba. The salinity of these resources, which are estimated to be 850 L/s during daily peaks, is 3.2 and 2.8 g/L for Chott El Fejij and Manara and Boulbaba, respectively.

A 22 500 m^3/d reverse osmosis desalination station (three lines) has been in service since January 1996, supplying the Gabès region with potable water; the salinity and 260 L/s with 0.5 g/L. In 1998, a fourth line will be put into service, which will increase the station capacity to 30 000 m^3/d. The available resources will be 663 L/s, of which 347 L/s will be desalinated water (Table III).

TABLE II. WATER RESOURCES (L/s) IN THE SOUTH OF TUNSIA

	1995	1998	1999	2010
Gabès	710	663	663	663
Medenine	815	815	1093	1093
Tataouine				
Total	1525	1478	1756	1756

TABLE III. DEMAND-RESOURCE BALANCE THE SOUTH OF TUNISIA

	1995	1998	2000	2010
Demand (L/s)	512	562	598	861
Resources (L/s)	710	663	663	663
Balance (L/s)	+198	+101	+65	-198
(m^3/d)				17
100				

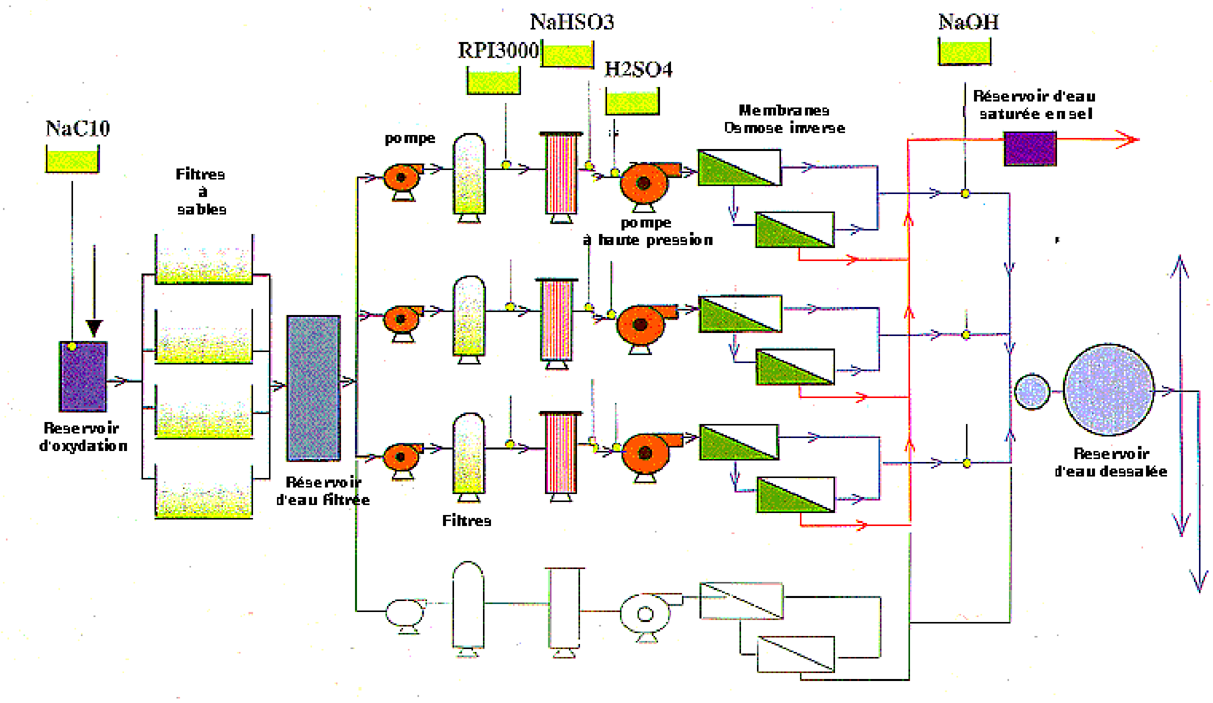


FIG. 2. Gabes desalination station.

3.3.2. Medenine and Tataouine counties

Potable water is transport to this region via a supply system called ‘adduction du sud Tunisien’. This system ensures the transport of groundwater from Zeuss-Koutin, south of Gabès and El Ababsa to various towns in the two counties. Local wells at Ghomrassene, Tataouine and Ben Guerdane also reinforce these resources. The overall capacity is evaluated at 815 L/s during daily peaks (including the wells at El Maouna) Table IV). The overall salinity of these resources is relatively high, and can reach 2.5 g/L, especially during peak periods.

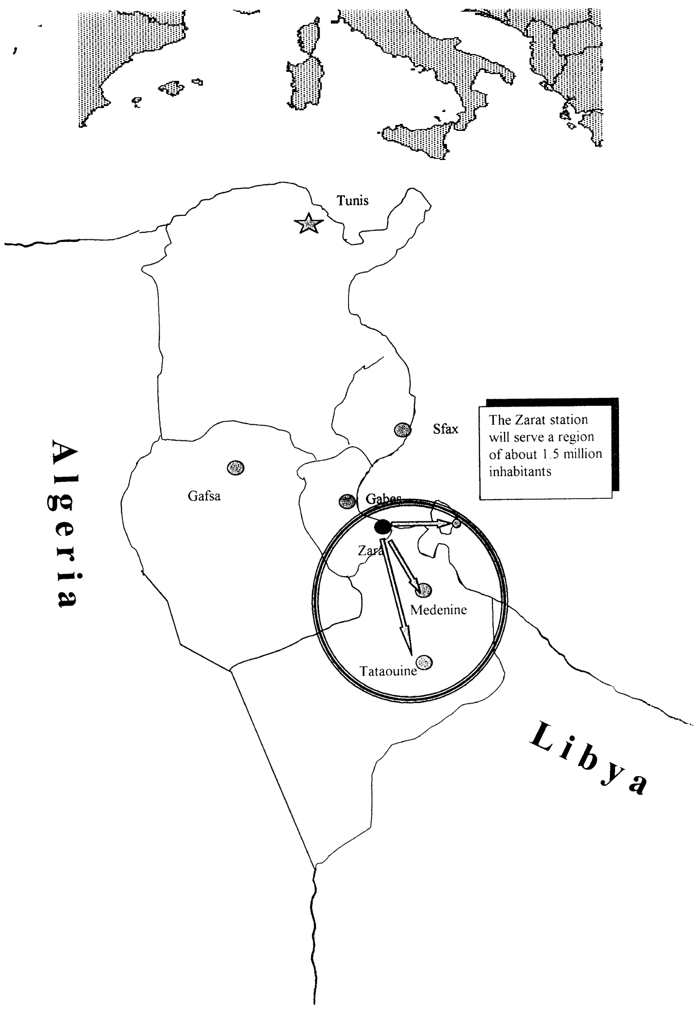
TABLE IV. DEMAND-RESOURCE BALANCE IN THE MEDENINE AND TATAOUINE COUNTIES

	1995	1998	2000	2010
Demand (L/s)	862	962	1035	1392
Resource (L/s)	815	1093	1093	1093
Balance (L/s)	+47	+131	+58	-299
(m^3/d)				25 800

To increase the resources and to improve the water quality, two reverse osmosis desalination stations, each with a capacity of 12 500 m^3/d, are already in operation in Jerba and Zarzis. They will desalinate brackish water, with 6 g/L drawn from the Miopliocene reserve. Therefore, the water resources of this region increase to 1093 L/s during peak periods.

4. CONCLUSION

The seawater desalination need in the south of Tunisia will reach about 60 000 to 80.000 m^3 /d between 2010 and 2015. A nuclear desalination plant could be planed to supply this region in electricity and potable water with ND capacity of 60 000 m^3 /d to 80 000 m^3 /d. The identified Zarat site or Skira site could be selected to shelter this plant (map1).



Map 1.

PROSPECTS FOR THE UTILIZATION OF SMALL NUCLEAR PLANTS FOR CIVIL SHIPS, FLOATING HEAT & POWER STATIONS AND POWER SEAWATER DESALINATION COMPLEXES

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Abstract

Small power nuclear reactor plants developed by OKB Mechanical Engineering are widely used as propulsion plants in various civil ships. Russia is the sole country in the world that possesses a powerful ice-breaker and transport fleet which offers effective solution for vital socio-economic tasks of Russia's northern regions by maintaining a year-round navigation along the Arctic sea route. In the future, intensification of freighting volumes is expected in Arctic seas and at estuaries of northern rivers. Therefore, further replenishment of nuclear-powered fleet is needed by new generation ice-breakers equipped with advanced reactor plants. Adopted progressive design and technology solutions, reliable equipment and safety systems being continuously perfected on the basis of multiyear operation experience feedback, addressing updated safety codes and achievement of science and technology, allow the advanced propulsion reactor plants of this type to be recommended as energy sources for floating heat and power co-generation stations and power-sea water desalination complexes.

1. EXPERIENCE AND PROSPECTS OF SMALL REACTOR PLANTS UTILIZATION FOR CIVIL SHIPS IN RUSSIA

In the history of the Russian Arctic regions exploration and development this year (1999) is notable by three remarkable anniversaries, viz.: a century of the Russia's ice-breaker fleet, 60th anniversary of Murmansk shipping company — ice-breakers operator and 40th anniversary of civil nuclear-powered fleet, which history originated with the first nuclear ice-breaker "Lenin".

Prospects of economic activity development in regions adjoining the Russia's Arctic seas coast seem to be problematic without the intensive use of nuclear-powered ice-breaker/cargo fleet, that proved for a short time its indisputable advantages compared to other type (conventional) ice-breakers. Due to the use of nuclear-powered ice-breakers a cargo traffic volume was increased along the Arctic sea route, so that it in a factor of about 10 exceeds a traffic volume in the remainder (foreign) part of Arctic. In future, as projects of abundant Arctic oil and natural gas fields development would be realized and cargo flows increase between Europe and Asia, a freight traffic will build-up (see Fig. 1) and consequently a role of nuclear-powered fleet will respectively become more vital. Due to this reason a challenge exists for the coming years to preserve the existing potential of nuclear-powered ice-breaker fleet and then to develop it further.

The nearest task in this field is to extend lifetime of the propulsion reactor plants, that would permit to continue the nuclear ice-breakers operations in the Arctic seas and to obtain a time reserve needed for design and construction of new generation nuclear ice-breakers. These are, first of all, ice-breakers for shallow areas of Arctic coast and Siberian river estuaries, and capital ice-breakers for year-round running of cargo ships along traditional ways of the Arctic route.

The Russia's civil nuclear-powered fleet currently consists of seven nuclear ice-breakers and one cargo (lighter carrier) ship. The ice-breaker "Lenin" is already removed from operation. The nuclear-powered ships and their reactor plants main performance indicators over the period from 1970 till 1998 are summarized in Table 1.

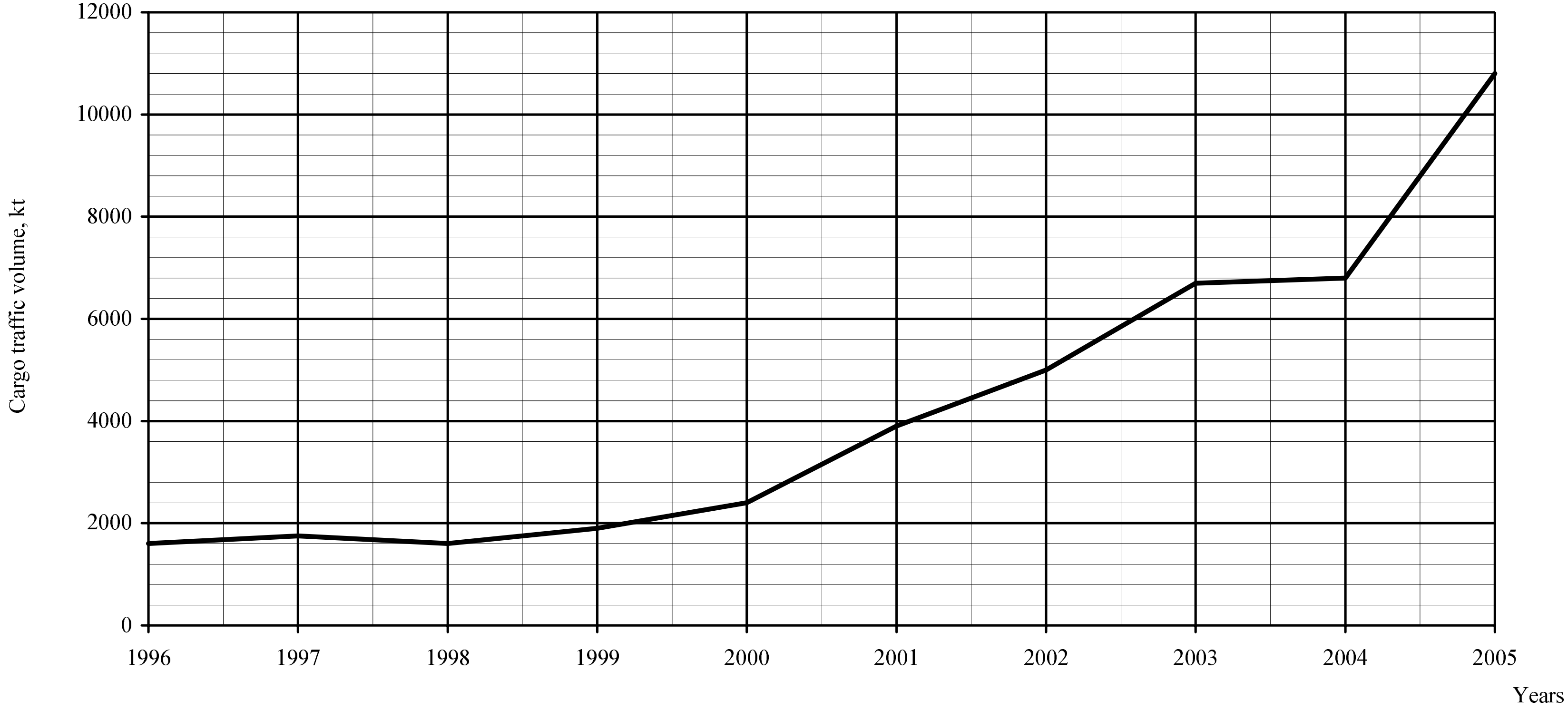


Fig. 1. Cargo traffic volume variation with time at Arctic sea route.

TABLE 1. NUCLEAR-POWERED SHIPS PERFORMANCE INDICATORS OVER THE PERIOD FROM 1970 TO 1999

Characteristics	Name of ship									
	i.-br. "Lenin"	i.-br. "Arctica"	i.-br. "Sibir"	i.-br. "Russia"	i.-br. "Sov. Souz"	i.-br. "Taymir"	i.-br. "Vaygach"	i.-br. "Yamal"	l.-cr. "Sev-morput"	i.-br. "50th celebration of Victory"
1. Year of commissioning	1970	1975	1977	1985	1989	1989	1990	1992	1988	under construction
2. Averaged duration of operation per year, days	230	235	232	208	273	300	288	298	275	
3. Total reactor operating record from power startup, h	106736 / 106384	131669 / 132321	94785 / 94043	66779 / 66118	58958 / 58522	62164	56311	35704 / 35081	68210	
4. Total energy produced from power start-up, $\times 10^3$ MWt·h	6523 / 6398	8680 / 7978	6095 / 6934	4630 / 4894	3899 / 3875	4918	4811	2062 / 2601	3744	
5. Distance sailed, miles 1) total;	654400	878599	740786	424311	351743	335919	258053	229819	258107	
2) incl. through ice	560600	773203	472787	391182	301159	330817	232363	204648	88234	
6. Number of ships conducted	3700	2913	1711	1212	501	984	680	572	–	

i.-br. — ice-breaker
l.-cr. — lighter-carrier

The total operating record of the reactors exceeds 160 reactor-years, while that for individual equipment items in some operating reactors exceeds 130 thou h and continues to increase further. During that period no incidents involving chain reaction control violation or inadmissible release of radioactivity were identified [1].

During the entire reactor plants lifetime designers of systems and equipment carried out supervision for their operation. Every event of failure or deviation from normal operating conditions was thoroughly analyzed. Simultaneously, new technical solutions are tested and plants operation modes are optimized. Resulting from the consistent conduction of planned activity on perfection of equipment and systems, and optimization of operating modes a specified useful lifetime of the plant's key equipment has been increased from 25-30 thou h up to 100–150 thou h.

A great complex of work is currently being performed that is associated with examination of equipment and piping of nuclear ice-breaker "Lenin", that have already exhausted its specified life. Results of the comprehensive study will allow a justified decision to be made about further extension of the reactor plants lifetime, eventually up to the ship service life value. Considerable organizational efforts are currently applied by design and operating enterprises to attain this goal.

Multiyear data of laboratory monitoring of snow, soil and vegetation samples around areas where the nuclear ships are stationed at does not reveal their environmental effect. Mean-annular exposure dose to personnel does not exceed 0.5 rem.

Level of safety and environmental cleanliness of last modification propulsion reactor plants meets all the requirements imposed by effective domestic and international safety guides, eliminates any restrictions on their deployment areas. Commercial cruises of the nuclear ice-breakers with foreign tourists on board to the North Pole corroborate a sufficiently wide recognition of the ships safety and reliability.

Furthermore, Russian design organizations basing on the solid accumulated experience, proceed to develop prospective reactor plants for future nuclear ice-breakers, that will meet the actual requirements of the Customer in terms of their safety, service life, useful lifetime, mass and other technical characteristics.

As the nuclear ice-breakers service life exhaust their decommissioning will become more and more urgent problem. Russia's Ministry for nuclear power (MinAtom) and Ministry for transport (MinTrans) currently carry out preparatory activities to solve this problem. Positive multiyear experience of the propulsion reactor plants fault-free operation gives grounds to recommend KLT-40 - type NSSS as a source of energy for heat and power supply and for seawater desalination purposes.

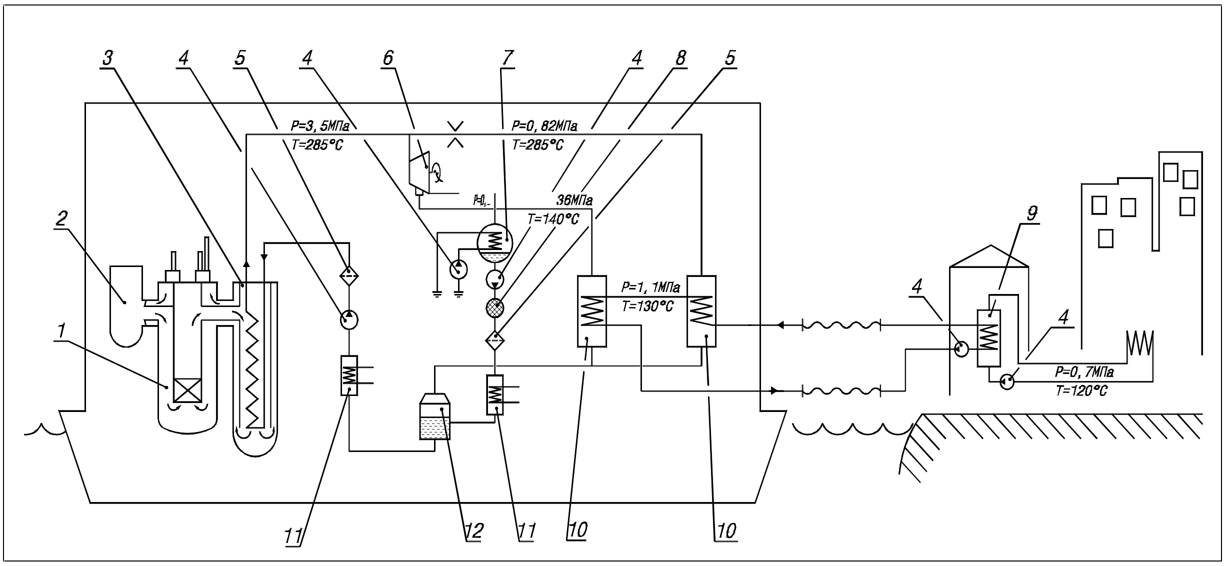
2. PROSPECTS OF SMALL NUCLEAR REACTOR PLANTS USE FOR FLOATING HEAT AND POWER CO-GENERATION STATIONS

The extreme northern and similar remote regions of Russia occupy more than half of the country's territory, where the major portion of mineral and energy resources are located, including oil, natural gas, nickel, gold, diamonds and rare metals. However, majority of these regions are not provided with a centralized power supply systems, have no fuel-energy resources expedient for effective utilization, while fossil fuel delivery there entails great difficulties and expenses. Therefore, application of NPP, especially floating ones, becomes justifiable and prospective for these regions.

Investigations performed have shown that in the indicated regions of Russia there are tenths of places where a need of nuclear heat and power plants exists just now or will appear in the nearest future. Several sites for potential deployment of floating nuclear heat and power plants are currently studied, viz.: at Peveck (Chuckot autonomous district), Norilsk industrial region (RAO "Norilsky nickel"), coastal areas of Kamchatka peninsula and Far East, where small decentralized power sources are basically used now.

In Russia the design of floating power unit (FPU) with KLT-40 - type reactor plants is currently nearing completion for a leading nuclear heat and power co-generation plant (NHPP) to be deployed at Peveck [2].

Manufacturing of the most labor-consumption equipment for the reactor plants is already underway. The floating power unit is intended to generate electricity and heat, being a constituent of the nuclear co-generation station. The FPU includes two KLT-40S reactor plants and two steam turbines with electric generators, combined in two individual units of 35 MW(e) capacity each. The turbines have steam extraction bleed-off for heating a feed water and intermediate circuit water supplied into a related heating grid. Principal flow diagram of the FPU together with a coastal heating circuit is shown in Fig. 2.

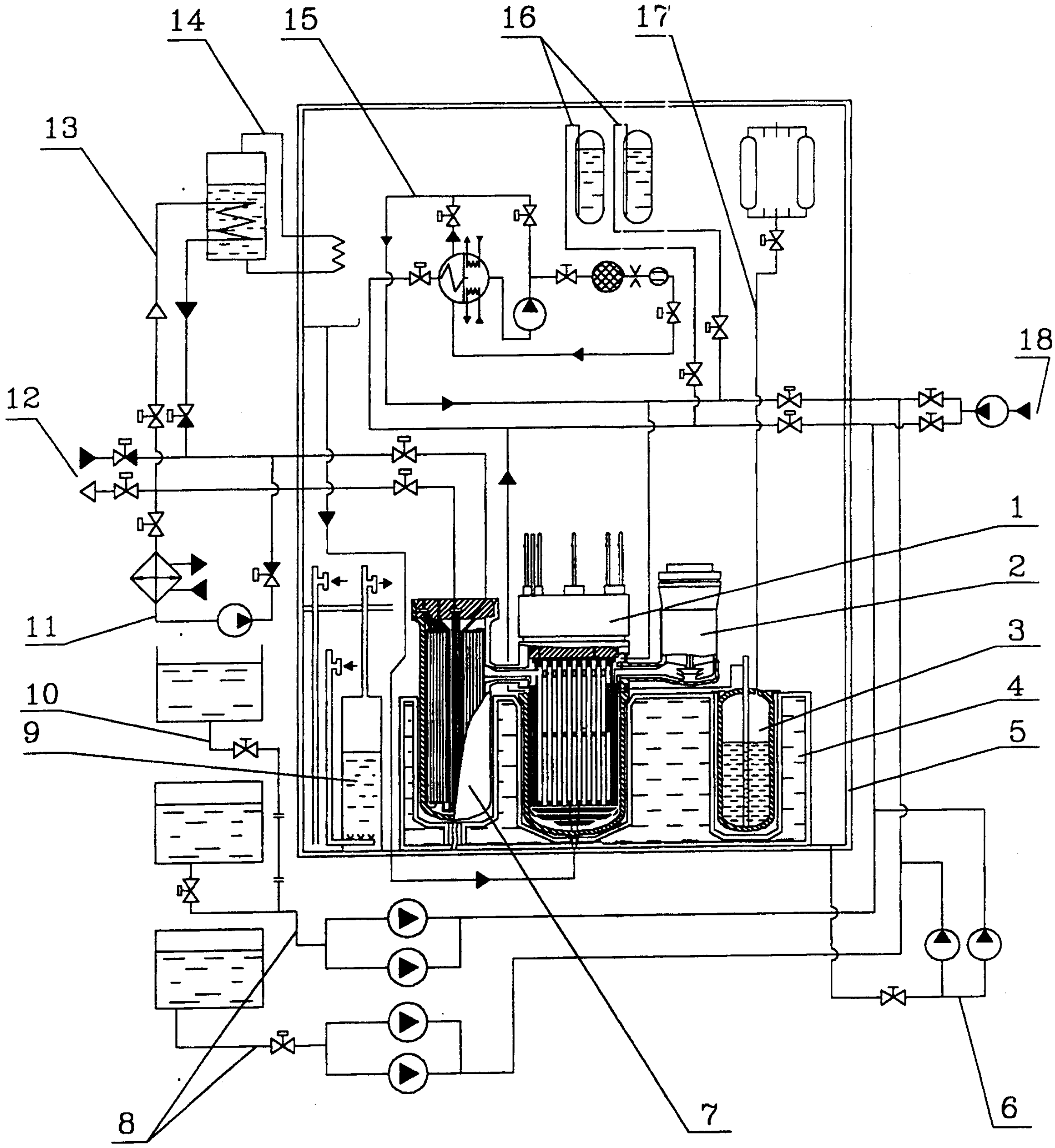


1 – Reactor
2 – Primary pump
3 – Steam generator
4 – Feed water pump
5 – Mechanical filter
6 – Turbine generator
7 – Condenser
8 – Ion exchange filter
9 – Feed water heater
10 – Intermediate circuit heat exchanger
11 – Feed water heater
12 – Deaerator

FIG. 2. Schematic plow diagram of NCGP with the shore heat supply system.

The KLT-40S nuclear reactor plant has been developed based on equipment used in the ice-breakers, with reliance on shipbuilding technologies and engineering solutions proven by many-year operating experience under the most severe navigation conditions in Arctic.

PWR as the most widely used and proven reactor type is used in KLT-40S reactor plant. Its steam generator is of once-through coiled type, reactor coolant pump is canned centrifugal two-speed circulator. The reactor plant components, viz.: reactor, 4 steam generators and 4 reactor coolant pumps are joined in a steam-generating unit by nozzles with the same robustness as the reactor pressure vessel, thus forming four circulation loops. The reactor plant is enclosed into a protective shell (containment), which in turn is located within a protective enclosure. The reactor plant is equipped with engineered (active) and passive safety systems. Fig. 3 shows principal system configuration of the KLT-40S reactor plant.



1 – Reactor
2 – Reactor coolant electric pump
3 – Pressurizer
4 – Metal-water shielding tank
5 – Protective shell
6 – Recirculation system
7 – Steam generator
8 – ECCS
9 – Bubbler system for pressure Suppression in protective shell
10 – Liquid absorber removal system
11 – Emergency shutdown cooling system
12 – To STP
13 – Passive emergency shutdown cooling system
14 – Protective shell pressure suppression emergency condensation system
15 – Primary circuit purification and cooldown system
16 – Passive ECCS
17 – Pressurization system
18 – From STP

Fig. 3. Nuclear steam supply system with KLT-40C reactor.

The design of KLT-40S reactor plant is developed in conformity with the latest general regulatory provisions for nuclear safety - OPB-88/97, Rules for nuclear safety of reactor plants for NPP-PBYa RU AS-89, Radiological safety regulations NRB-99, the Russian Federation law "On radiological safety of population", Rules for nuclear ships classification and construction of the Russia's maritime Register, IAEA safety guides etc. Presently a site for the floating nuclear co-generation plant is being licensed in the Russia's regulatory body (Gosatomnadzor).

Inherent self protection properties of the reactor, maximal utilization of passive and self-actuated safety features ensure the reactor resistance against any errors of personnel and failures of equipment. Exposure dose to population under normal operation conditions at a distance of 1 km from the plant is about 0.01 m rem per year. Evacuation of local population during accidents in the plant is not needed.

The steam turbine plant (STP) is used to generate electric energy and to heat water in the intermediate circuit of the related heating system. The STP consists of a steam turbine, double-section horizontal condenser and electric generator. The intermediate circuit includes heating grid water heaters, where steam from intermediate bleed-offs of the turbine is used as a heating coolant, and a peak heating grid heater, where steam from the main steam line is used.

3. PROSPECTS OF SMALL REACTOR PLANTS UTILIZATION FOR FLOATING POWER-SEA WATER DESALINATION COMPLEXES.

World demands in fresh water is growing steadily as population increases and industrialization level rises. That is the reason why sea water desalination facilities are being developed, including those used nuclear energy, particularly for utilization in remote coastal regions.

On the basis of floating nuclear power unit a number of conceptual designs of dual-purpose power-desalination complexes for production of electricity and fresh water has been developed.

Facilities with multistage horizontal-tube film evaporators, developed by Sverdlovsk Research Institute for Chemical Machinebuilding (Ecaterinburg, Russia) [3], are used in the designs as distillation desalination plants. Facility proposed by "Candesal" company, based on utilization of highly-permeable membranes manufactured using technology of Dow Film Tec [4], is adopted as reverse-osmosis desalination plant. The power plant design options use condensation and back-pressure turbines.

3.1. Power-seawater desalination complexes based on distillation desalination technology

During analysis of technical and economic characteristics the following configurations of power-sea water desalination complexes were considered:

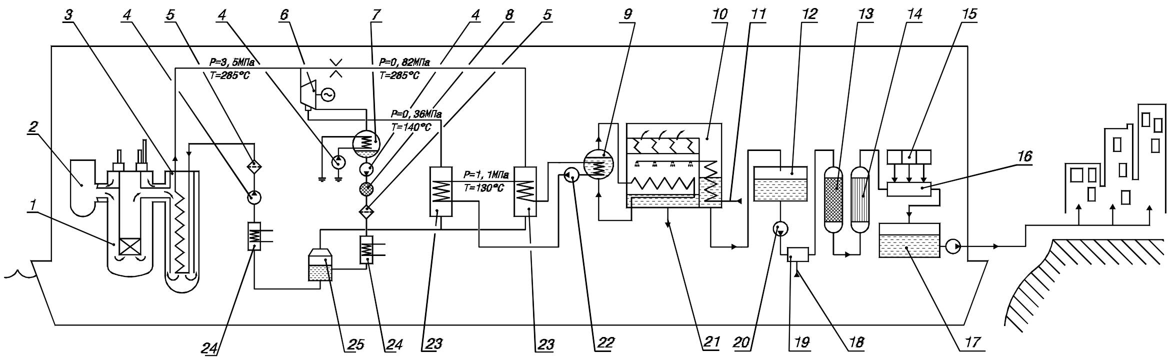
- use of steam extracted from condensation turbine bleed-offs for distillation desalination facilities (Fig. 4);
- use of a back-pressure turbine and heat extracted from a condenser for distillation desalination facilities (Fig. 5).

Heat and flow diagram of the complex to the first design option is similar to that of the floating nuclear heat and power co-generation station. Its distinctive feature is in the use of steam extracted from turbine bleed-off for sea water desalination, rather than for heating purposes.

3.2. Power-seawater desalination complexes based on reverse-osmosis technology

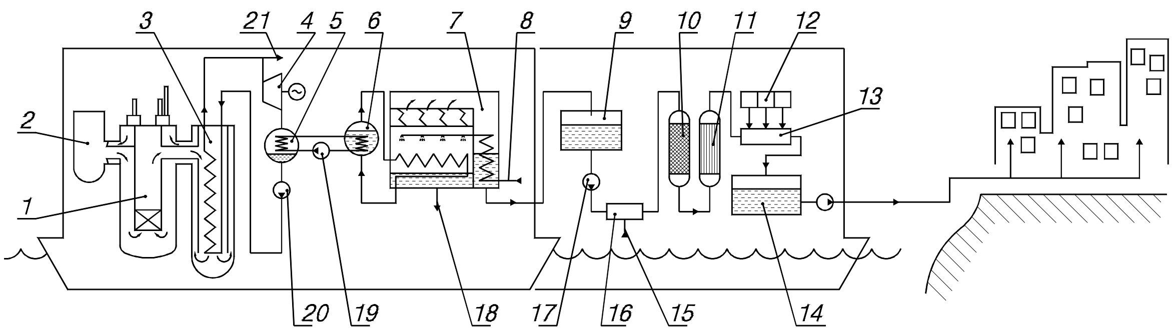
For the given type desalinators a configuration with a condensation turbine are considered for electricity generation and with turbine condenser waste heat utilization for heating of seawater to be desalted.

The complex includes FPU and reverse-osmosis desalination facilities, located on a shore or in a special vessel. Results of analysis made for comparison, of the design options with two KLT-40S reactors are summarized in Table 3.



1 – Reactor
2 – Reactor coolant pump
3 – Steam generator
4 – Secondary circuit electric pump
5 – Mechanical filter
6 – Turbo-generator
7 – Condenser
8 – Ion exchange filter
9 – Steam generator
10 – Distillation desalination plant
11 – Sea water inlet
12 – Distillate intake tank
13 – Water enrichment facility
15 – Plant for water fluorine, chlorine treatment and stabilization
16 – Mixer
17 – Potable water tank
18 – H_2CO_3 solution
19 – Mixer
20 – Potable water preparation plant pump
21 – Evaporated sea water brain
22 – Intermediate circuit electric pump
23 – Intermediate circuit heat exchanger
24 – Feed water heater
25 – Deaerator

Fig.4. Schematic flow diagram of nuclear desalination complex using turbine steam extraction line.



1 – Reactor
2 – Reactor coolant pump
3 – Steam generator
4 – Turbo-generator
5 – Condenser
6 – Steam generator
7 – Distillation desalination plant
8 – Sea water inlet
9 – Distillate intake tank
10 – Water enrichment facility
11 – Running water sorbent containing filter
12 – Plant for water fluorine, chlorine treatment and stabilization
13 – Mixer
14 – Potable water tank
15 – H_2CO_3 solution
16 – Mixer
17 – Potable water preparation plant pump
18 – Evaporated sea water brain
19 – Intermediate circuit electric pump
20 – Secondary circuit electric pump
21 – To condensation turbine

Fig. 5. Schematic flow diagram of nuclear desalination complex with back-pressure turbine.

TABLE 2. SUMMARIZES BASIC TECHNICAL CHARACTERISTICS OF THE FLOATING POWER UNIT.

Characteristic	Value
1 Number of reactor plants	**2**
2 Type of reactor plant	**PWR KLT-40**
3 Thermal power, MWt	**2 × 148**
4 Steam-generating capacity, tons/hour	**2 × 240**
5 Steam pressure at steam generator outlet, MPa	**3,8**
6 Steam temperature at steam generator outlet, ^{0}C	**290**
7 Feed water temperature, ^{0}C	**170**
8 Installed turbine generator plant electric power, MWe	**2 × 35**
9 Rated electric power in turbine steam-extraction operation mode, MWe	**2 × 30**
10 Electric power in turbine condensing operation mode, MWe	**2 × 32,5**
11 Heat to heat supply system, GCal/h	**2 × 25**

TABLE 3. POWER-SEAWATER DESALINATION COMPLEXES OUTPUT DATA.

Type of flow diagram	Type of desalination plant	Type of turbine	Maximum output	
			Fresh water, m^{3}/d	Electric power, MW
Option 1	**Distillation desalination plant**	**Condensation with steam extraction for DOU desalination facilities**	**33000 - 0**	**58 - 5**
Option 2	**Distillation desalination plant**	**Back-pressure**	**147000 - 44000**	**24.8 - 5**
Option 3	**Reverse osmosis (input water is heated in condenser)**	**Condensation**	**285000 - 0**	**0 - 65**

CONCLUSIONS

(1) Small power nuclear reactor plants are widely used in Russia for nuclear ice-breakers and cargo ships, which operations for a long time provide life sustenance and economic development of Russia's regions at Extreme North and Far East. They have a real prospects of further utilization.

(2) Successful experience of small power propulsion reactor plants operation in nuclear ice-breakers and other civil ships gives grounds to recommend them as energy sources for heat and power co-generation stations and power-seawater desalination complexes.

(3) Based on the advanced propulsion nuclear steam supply system KLT-40S a leading co-generation nuclear station with floating power unit is currently being created in Russia, for deployment at port of Peveck in Chuckot national district.

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A STUDY ON THE INTRODUCTION OF A DEMONSTRATION PLANT FOR THE CLEAN AND EFFICIENT PRODUCTION OF WATER AND ELECTRICITY IN INDONESIA

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Abstract

BATAN interested to develop a nuclear reactor that has high degree of safety and broad spectrum of process heat application including for electricity to be introduced in Indonesia and in the Asia Pacific Region. Having nuclear reactor technology that meet those requirements a flexible direct marketing in the short and long term with a minimum adaptation could be done. Nuclear process heat applications such as for synthetic fuel production require supply of energy at high temperature levels up to 950°C in order to achieve a sufficiently high reaction rate. The high temperature reactor is the only reactor system working in this temperature range. However, as the first step of development of the reactor technology in Indonesia, introduction (in the near time) of demo plant for the clean and efficient production of water and electricity may be is a realistic strategy. Therefore, the study is focused on the high temperature gas cooled reactor technology coupled with the proven desalination plant. The scope of study to be done including reactor technology, safety system evaluation, fuel cycle technology, waste management and decommissioning, economic and financing, licensing process, project development, site and environmental study, and plant layout. BATAN and related parties are preparing the study. In this report the scope of study and its progress is briefly presented.

INTRODUCTION

The parties — National Nuclear Energy Agency (BATAN), Institute for Safety Research and Reactor Technology, Germany (FZJ), and Blaauwendraad Management Consultants, Netherlands (BMC) — interested to develop a nuclear reactor that has high degree of safety and broad spectrum of process heat application including for electricity to be introduced in Indonesia and in the Asia Pacific Region.

In Indonesia, since 1993/1994 about 20% of petroleum fuel demand, mainly from transportation sector, cannot be supplied locally due to capital investment problems in expanding the capacity of refinery plant. In the mid-long period the deficit will increase due to declining of production capacity of the oil field that natural gas field should be exploited to cover transportation fuel demand [1, 2]. Nuclear process heat applications such as for synthetic fuel production require supply of energy at high temperature levels up to 950°C in order to achieve a sufficiently high reaction rate. The high temperature reactor (HTR) is the only reactor system working in the temperature range for synthetic fuel production. However, Batan recognized that introduction (in the near time) of demonstration plant for production of water and electricity may be is a realistic strategy as the first step of development of the reactor technology in Indonesia especially from technology transfer and public acceptance point of view.

Batan, FZJ, and BMC are preparing a study to introduce co-production plant for water and electricity as a demonstration plant in Indonesia using HTR as a heat source coupled with a desalination plant. In this report a draft of scope of study, continuing reviewed by the parties, is presented. The contribution of each member to the study is still under discussion and may be open for the new member.

SCOPE OF STUDY

1. Reactor technology

1.1. Selection of technology

Review on the background of High Temperature Reactor specifying the various conceptual differences with particular emphasis of the various fuel element types: block, pin-in-block, and pebble. Specification of the technological "maturity" for direct marketing in the world, in the Asia Pacific

Region and in Indonesia, and description of key elements and design features of the reference plant the above objective could be obtained.

- *Key elements* indicating simplified safety system with reduced reliance on operator (inherently safe and catastrophic/severe accident free), reliable equipment, economics, defense-in-depth (by multibarriers and application of proven technology), standardization of the design, and easy maintenance & reduced occupational exposure.
- *Design features* briefly explaining various merit of the selected power range, fuel geometry, type energy conversion, core geometry, fuel type, fuelling/ refueling strategy, and type of cooling.

The result may be: The Pebble Bed HTR from Siemens/ABB-HTR GmbH is the most advanced system with respect to technological maturity because of the following reasons:

(a) The detailed engineering is ready, the procurement can be started
(b) The concept of the plant has passed a licensing procedure in FRG and
(c) The Reactor Safety Commission of FRG has given a positive recommendation with respect to a licensing procedure in Germany.

Another important aspect that the vendor industry in Germany offers the "excess to the total know-how" to interested institutions, since this industry went into hibernation with a complete stop of all marketing efforts.

1.2. Conceptual design

Review of the various conceptual designs of the various concept of high temperature reactor, including:

- The various processes for production of electricity and drinking water.
- *Design parameters* explaining reactor thermal power, core dimensions reactor inlet temperature, reactor outlet temperature and energy conversion system design parameters, refueling interval, heavy metal load and enrichment, and design life.
- *Technical study and calculation* involving core neutronic calculations, primary loop purification study, temperature distribution study, shielding calculation, PCU performance, simulator, building seismic study, single line diagram, rnanifold & support, plant electrical, I&C system, fuel and de-fuel system, heat removal system, interrelations between system functions and their support systems, auxiliary system, etc.
- *System main component identification and layout* involving system/ component list(s), system/ component specs (requirement), plant layout including classification of systems, typical drawing and/or sketches, etc.

The conceptual design of a "Pebble Bed HTR with Steam Turbine" for the production of electricity and process heat, respectively drinking water is available in the know how of HTR GmbH.
The available conceptual design resulting from proper consideration will be used as a basis for the Technical Bid Document.

2. Safety system evaluation

- Evaluation on the safety requirements including safety philosophy, criteria, features and standards for future nuclear energy technology, on the various possibilities to realize this requirements and identification of the most important system and component to "produce" safety.
- Estimate safety evaluation concerning the probabilistic safety assessment (PSA) level-1 of the plant and compared to the PSA level-1 of RSG-GAS (multi purpose reactor — GAS).

The result may be: (1) The coated particle is a very important component because of its high temperature resistance, its high retention capability for normal operation and for accident conditions and the stable behavior at final storage, (2) The pebble type fuel element provides the possibility of continuous fuelling, avoiding risky excess reactivity, and (3) The principle of modularization of the core ensures the avoidance of core meltdown accidents.

3. Fuel cycle technology

Description and strategy development of the technology of the coated particle and the pebble type fuel element with respect to fabrication, application and performance, and intermediate as well as final disposal.

Evaluation on the existing fuel fabrication plants in Japan and in China for possibly fabrication of the first core and assistance for future Indonesian fuel fabrication plant.

4. Waste management and decommissioning

Short Description of the experiences from AVR and THTR on waste management and de-fuelling, as well as intermediate storage of fuel elements, as well as the secure enclosure of the closed plants and short mentioning of the future plans for the decommissioning.

Including to the above description is the description of the strategy development of back-end of the fuel and recommendation to the solution using appropriate technology.

5. Economy and financing

Evaluation on the economy of future pebble bed HTRs on the background of the experiences in Germany, but also taking into account the approach of "series production" in a global scale proposed by ESKOM in South Africa, in terms of electricity generation cost and water production cost to other system.

Discussion of various methods of financing the project for the first unit, called Research and Demonstration HTR, as well as for larger number of to be produced modules for larger power stations in Indonesia and in the Asia Pacific Region.

The evaluation and discussion cover the following cases:

- Procurement/manufacture & costing study for obtaining manufactures/suppliers list(s), request for quotation, quotation/tender adjudication, component re-costing of German design, etc.
- Project investment cost for implementing first of a kind (FOAK or prototype) plant program
- Financing plan for the project
- Life cycle cost analysis for an Nth of a kind (NOAK) plant, built in Indonesia
- Market potential
- Inventarisation of possible input by the Indonesian industry

6. Licensing process

Short description on the possibilities for a licensing procedure in Indonesia. The background for that are the licensing processes for the three existing research reactors.

The description covers the assessment of the licensability of the plant in Indonesia in the cases of:

- Applicability of Indonesian nuclear regulation for introducing the HTGR
- Source terms of the HTR-module installation
- Health physics
- Etc.

7. Project development

7.1. Organisation

(1) Evaluation on the needed qualified personnel to fulfill the tasks of an architect engineer for the planning, procurement, construction and get into operation of the first plant, the Research and Demonstration HTR.
(2) Evaluation on the possibilities of co-operation and support from industry in Indonesia with particular stress on the need of an increasing local content, for the first plant, as well as for the establishment of a series production.
(3) Identification of project oriented R&D needs in support for the first plant.
(4) Evaluation on the organization of the know-how transfer with the total access to the know-how with the German Vendor Industry.
(5) Development of a marketing strategy for Indonesia and for export into the Asia Pacific Region.
(6) Evaluation on the possibilities to gain assistance by international co-operation.

7.2. Project management and schedule

- Short description of the establishment of a project management in industrial scale including its organization and responsibility and project management plan.
- Development of a schedule for the planning, engineering, procurement, construction and commissioning, and operation for the 1st plant and the research and demonstration HTR, as well as a series production.
- Preparation of a proposal for further planning and next steps.

7.3. Quality assurance program

Evaluation on the establishment of a QAP in Indonesia.

8. Site and environmental aspect study

Collection of the relevant data about the foreseen site and its condition on the particular consideration of earthquake requirements and environmental impact requirement through the following activities:

(a) Select and evaluate the available site data at Ujung Lemahabang as the best candidate site, which appropriate to the proposed plant location, as the input data for preparing the site data information (SDI) and environmental impact analysis report (EIAR)
(b) In order to prepare only the appropriate data/information pertinent to the proposed site for the project to be used as input data for SDI and EIAR development, series of work including, but not limited to the following:

(1) Identification of aspects of the proposed site that could be needed for the development of SDI and EIAR,
(2) Review for further identification of data/information relevant to each aspect identified in point 1) above that might be needed for the development of SDI and EIAR,
(3) Evaluate the identified data/information in 2) for their validity, appropriateness, correctness,
(4) Develop a report containing the appropriate data/information pertinent to the development of SDI and EIAR of the project in the later stage,
(5) Present and discuss the report in the joint meeting Batan-BMC-FZJ.

Source of data/information: this study is designed to utilize as much as possible all available data/information from previous study especially, they that already in Batan possession. Other data/information could be drawn from databases of competent organization such as BMG, ITB, Volkanologi, PPGL.

9. Plant layout

Evaluation on the needed work and needed qualified personnel for the plant layout, planning and construction under particular consideration of the site and environmental aspects.

Propose the appropriate layout of the plant in order to determine the size of the project area. (a typical layout of a single unit plant).

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HIGH TEMPERATURE APPLICATIONS

PROSPECTS OF HTGR PROCESS HEAT APPLICATION AND ROLE OF HTTR

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Abstract

At Japan Atomic Energy Research Institute, an effort on development of process heat application with high temperature gas cooled reactor (HTGR) has been continued for providing a future clean alternative to the burning of fossil energy for the production of industrial process heat. The project is named "HTTR Heat Utilization Project", which includes a demonstration of hydrogen production using the first Japanese HTGR of High Temperature Engineering Test Reactor (HTTR). In the meantime, some countries like China, Indonesia, Russia and South Africa are trying to explore the HTGR process heat application for industrial use. One of the key issues for this application is economy. It has been recognized for long time and is still now that the HTGR heat application system is not economically competitive to the current fossil ones, because of the high cost of the HTGR itself. However, the recent movement on the HTGR development, as represented by South Africa Pebble Beds Modular Reactor (SA-PBMR) Project, has revealed that the HTGRs are well economically competitive in electricity production to fossil fuel energy supply under a certain condition. This suggests that the HTGR process heat application will be also possible to get economical in the near future. In the present paper, following a brief introduction describing the necessity of the HTGRs for the future process heat application, Japanese activities and prospect of the development on the process heat application with the HTGRs are described in relation with the HTTR Project. In conclusion, the process heat application system with HTGRs is thought technically and economically to be one of the most promising applications to solve the global environmental issues and energy shortage which may happen in the future. However, the commercialization for the hydrogen production system from water, which is the final goal of the HTGR process heat application, must await the technology development to be completed in 2030's at the earliest. The HTTR heat Utilization Development Project is hoped to contribute to the technology development, cooperating with other countries of interest.

1. Introduction

The ultimate potential offered HTGRs derives from their unique ability to provide heat at high temperatures (e.g. in the range from 500℃ to 1000℃) for endothermic chemical processes. Heat from HTGRs can be used for production of synthesis gas and/or hydrogen and methanol by steam/methane reforming, production of hydrogen by high temperature electrolysis of steam and by thermochemical splitting of water, production of methanol by steam or hydrogasification of coal, and for processes which demand lower temperatures, such as petroleum refining, sea water desalination, district heating, and generation of steam for heavy oil recovery and tar sand mining. The application of such nuclear heat as above, referred herein as "process heat application", can make a significant contribution to resolve global environmental problems which result from burning fossil fuels.

This paper introduces the Japanese research and development activities on the process heat application as well as industries' prospect of the commercialization.

2. Research and development activities in Japan

(1) History

The history of the research and development activities on HTGRs dates from about 30 years back. At that time, the possibility of direct steel making was sought by utilizing the heat from HTGRs. Then, the VHTR (Very High Temperature Reactor) Project was initiated in 1969 at JAERI, Japan, including research and development (R&D) which covers all fields necessary for the reactor design and construction of the VHTR as well as process heat application systems. However, there was no urgent or strong commercial demands coming up afterwards, although the essential needs of the HTGRs were well understood for the future. Thus the Project was reviewed by the Government to shift it to more basic research for the future rather than immediate development for commercial use.

In accordance with this review, Atomic Energy Commission of Japan issued in 1987, the revision of Long term Program for Development and Utilization of Nuclear Energy, recommended that Japan should proceed with the development more advanced technologies for the future, in parallel with existing nuclear systems. The Long term Program emphasized that the HTGR is considered as one of the most promising nuclear reactors to improve the economy and to extend the application of nuclear energy. In conclusion, the construction of the High Temperature Engineering Test Reactor (HTTR) was decided to establish and upgrade HTGR technology basis as well as to be used as a tool for innovative basic research in the field of high temperature engineering. It should be noted that the HTTR is neither an experimental nor prototype rector for commercial HTGR, but a test rector for the future.

According to the revised Long term Program for Development and Utilization of Nuclear Energy, the construction of the HTTR was initiated in 1991 and is now in the stage of commissioning tests. The first criticality of the HTTR was attained in November 1998, followed by the rated power operation about one year later. The R&D on the process heat application has been also continued to couple the system to the HTTR.

(2) Current status

In Japan, the basic study on HTGR process heat application had been made for longer than 20 years mainly at JAERI, Japan. The current R&D activities have successively followed the past basic studies in larger engineering scale tests within a framework of the HTTR Heat Utilization Project, including international cooperation. Some IAEA Member States are cooperating in the design and evaluation of potential HTTR heat utilization systems within a frame of the IAEA Coordinate Research Programme (CRP). Countries participating in this CRP include China, Israel, Germany, Russia, Indonesia, Japan and the USA. In the CRP, the processes being assessed are selected by the CRP participants according to their national interest depending on status of technology, economic potential, environmental considerations, and other factors. The following are being examined:

(1) Steam reforming of methane for production of hydrogen and methanol
(2) Thermochemical water splitting for hydrogen production (IS Process)
(3) High temperature electrolysis of steam for hydrogen production
(4) Gas turbine for electricity generation
(5) CO_2 reforming of methane for production of hydrogen and methanol
(6) Combined coal liquefaction and steam generation

Among them, the steam reforming of methane was selected as the first candidate to be demonstrated at the HTTR for production of hydrogen (and methanol). In the next to the steam reforming system, the system of either the gas turbine or the thermochemical water splitting (IS process) is the highest priority candidates for the HTTR test.

(i) Steam reforming system [1–6]

The steam reforming system was selected as the first candidate of the HTTR heat utilization demonstration test, because its technology is proven in the non-nuclear application with fossil energy sources so that an early coupling to the HTTR is possible. Also, the technology obtained through the development of the HTTR steam reforming system would be applicable to other possible process heat application systems such as CO_2 reforming and thermochemical hydrogen production systems of e.g. IS process.

For this purpose, an extensive effort has been continued in the system design and R&D for improving the efficiency of the hydrogen productivity. An out-of-pile mock up test is prepared at present in prior to the coupling. The first HTTR demonstration will be hopefully shown in 2004 or a little later. The details are available in the separate papers shown in the references.

(ii) IS Process [1–4, 7–10]

In the steam reforming hydrogen production system mentioned above, methane is used as a feed gas together with water to produce hydrogen, thus the emission of CO_2 is unavoidable. It is generally understood that the final goal of the hydrogen production system using HTGRs is to produce it from water without emission of CO_2. For this purpose, the thermochemical IS process was studied in a small scale laboratory experiment. In the experiment, a closed-cycle continuous operation in a steady state for 48 hours was successfully achieved at JAERI. Then the development activity will be shifted to more engineering system development using a larger scale facility at a hydrogen production rate of std. 50 litters an hour. The coupling to the HTTR will be possible at earliest in 2010. The details are available in the separate papers shown in the references.

3. Industries prospect for commercialization

In parallel to the HTTR activities, Japanese nuclear industries' group is trying to explore the development of the HTGRs. In their tentative survey, it was recognized that the HTGRs with gas turbine may be possible to be economically competitive to the current LWRs, as is planned to be built in South Africa. Then, they will recommend that the HTGRs shall be commercialized in Japan first for electricity generation with gas turbine system, hopefully in 2010's. In response to this recommendation, some feasibility studies or design works will be initiated soon among them in cooperation with JAERI.

With respect to the process heat application by HTGRs, it is concluded that its application will be promising in the future from not only the global environmental point of view, but also the conservation of fossil fuels resources. In this regard, the hydrogen production from water, not from fossil fuel, is recognized as the ultimate goal of the HTGR process heat application. Under this understanding, the development study on the thermochemical water splitting for hydrogen production like the "IS Process" is evaluated to be the highest priority. On the other hand, it was found that no direct or urgent demand for the nuclear process heat application exists at present in Japan. Furthermore, technologies are assessed to be not matured or well developed in the thermochemical water splitting system, still now. It is therefore directed that the current development study underway at JAERI be continued with focus on technical development of materials to be used at chemical reaction equipment and on improvement of efficiency and economy. The first pilot plant in Japan for the process heat application with the HTGR is foreseen in 2030's at the earliest.

It is also pointed out that the potential needs for the process heat application exist in some overseas countries like China, Indonesia, the Russian Federation and South Africa. The commercialization will be coming up earlier in these countries than in Japan. Therefore, some industries are looking to their countries, trying feasibility studies.

4. Economic aspect

One of the biggest problems is absolutely economy for the commercial process heat application systems. It is said that the economy of the total system is made worse by capital cost for HTGRs which supply heat to the heat application system via intermediate heat exchanger, because it is presumed that the cost fraction of heat application system downstream intermediate heat exchanger and hot duct is relatively small in comparison with HTGRs themselves. According to a private communication, a German simple estimation suggests the fraction is less than one-third, maybe one-fifth. The operation and maintenance cost will be relatively small in comparison with the current fossil fuel systems. Thus, the economy improvement of the reactor is inevitable for the success of the commercial plant.

It is obvious that the HTGR safety is achieved by a core with low power density. In comparison to the current LWRs, the power density is less than one-tenth in the HTGRs. Such low power density yields the inherent safety aspect, whereas it requires more capital due to the scale demerit. For example, the size of the pressure vessel of the HTTR with 30 MW thermal output is as large as that of medium size of LWR with 500 MW electrical output. The HTGRs are apparently disadvantageous in economy in comparison with the current LWRs.

On the other hand, the inherent safety aspect in the HTGRs could make it possible that no or quite limited engineering safeguards of reactor grade quality are needed. The only safety elements in the entire system are the fuel element and graphite core components which can be checked in running operation, while the safety of LWRs with high power density is ensured by extensive, active and passive safeguards and the reactor grade quality of the components and materials. Sophistication and expensive reactor grade quality is particularly required for all components of LWRs, but, in the case of HTGRs, ultimately only for the fuel element and graphite core components. Thus, the HTGRs would provide a new, qualitatively different safety, resulting in decreasing the cost. This saftey aspect can also make the heat application system designed in a general industrial safety grade, not nuclear grade, resulting in the significant cost reduction of the system. JAERI is now under developing a new safety philosophy applicable to the future commercial heat application systems, including countermeasures against possible fire or explosion by combustion gasses like methane and hydrogen.

It is also true that the economy of the process heat application system with the HTGRs depends on the productivity efficiency and availability of the system. At JAERI the study for improving the efficiency is taken by developing e.g. in the case of thermochemical IS process, innovative membrane technologies such as high temperature hydrogen separation membranes made of ceramics.. As for availability, the HTTR demonstration test will give the answer. In conclusion, it can be said that the economy improvement of the reactor can solve the problem of the overall economy of the nuclear process heat application systems. The planned commercial development of the gas-turbine HTGRs can be, therefore, regarded as a primary step for the future process heat application system development.

5. CONCLUDING REMARKS

Under an understanding that HTGRs can play an important role to expand the nuclear heat application to chemical industries against the current environmental issue of the CO_2, JAERI proceeds with the development of the nuclear process heat application systems coupling to the HTTR. Global eyes are kept by not only nuclear persons of interest but also the public upon the development of the HTTR heat utilization system, since its successful achievement may enhance the possibility to solve the environmental issue of CO_2 emission as well as a possible energy crisis which might happen in the future.

Finally it should be emphasized that an overall support and understanding from the overseas countries of concern are needed and wished for the success of the Project. The Project is highly expected to contribute so much to promoting international cooperation on the development of HTGRs and its process heat application.

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DESIGN OF THE STEAM REFORMER FOR THE HTR-10 HIGH TEMPERATURE PROCESS HEAT APPLICATION

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Abstract

The 10MW High Temperature Reactor Test Module (HTR-10) is being constructed now. It plans to be operated in 2000. One of its objectives is to develop the high temperature process heat application. The methane steam reformer is one of the key-facilities for the nuclear process heat application system. The paper describes the conceptual design of the HTR-10 Steam Reformer with He heating, and the design optimization computer code. It can be used to perform sensitivity analysis for parameters, and to improve the design. Principal parameters and construction features of the HTR-10 reformer heated by He are introduced.

1. Introduction

The High temperature gas cooled reactor HTR-10 now is being constructed in the Institute Nuclear Energy Technology of Tsinghua University. It plans to be operated in 2000. Than some inherent safety experiments will be performed in the following 2 years. After these experiments the operating temperature will be increased to 900℃ (or 950 ℃). At the beginning of the 21 century, about 2 to 3 years will be used to install a nuclear process heat application system.

The methane-steam reformer heated by He (or N_2) is one of the key-facilities for the nuclear process heat application system. Generally speaking, the nuclear heating reformer is a tube-shell type reformer. In comparison with a fossil fuel reformer its heat transfer efficiency is lower, reforming transform ratio is smaller and price is higher. So in order to make it commercial competitive, it is necessary to optimize the design to increase the productivity and decrease the cost of the reformer.

REFORM computer code is a design optimization computer code for the HTR-10 steam-methane reformer heated by He or N_2. It can be used to perform sensitivity analysis for different parameters, and to improve the design.

Principal parameters and construction features of the HTR-10 reformer by He (or N_2) are introduced.

2. Principal considerations for the reformer design

In Comparison with a fossil fuel reformer, the high temperature nuclear heat reformer has a series of disadvantages. For example: The forced convection heat transfer used in the nuclear heat reformer is less efficient than the radiation heat transfer used in a ordinary reformer, so the reforming temperature is lower about 50-100 ℃ than in ordinary reformer. From safety consideration the pressure of the high temperature process heat application system must be higher than the pressure of the intermediate heat exchanger (IHX) secondary side.

As is known to all, according to the dynamical balance there are two principal reactions for the reforming process:

$$CH_4 + H_2O \rightarrow CO + 3H_2 \quad -206\, kJ/mol \qquad (1)$$
$$CO + H_2O \rightarrow CO_2 + H_2 \quad +42\, kJ/mol \qquad (2)$$

Increase of pressure or decrease of temperature results in decrease of H_2 productivity.

In order to increase the H_2 productivity following design considerations are adopted:

- To decrease the work pressure of the catalytic bed, To make $P_p \approx P_{He}$ (or P_{N2})
- To increase the H_2O/CH_4 ratio in the raw gas low as possible.
- To preheat the process gas by residual heat of the production gas, to increase the process gas temperature at the inlet of the catalytic bed.
- To intensify heat transfer at out side of the catalytic bed.
- To increase the temperature difference between the H_2 (or N_2) and process gas.
- To improve heat transfer features of the fixed catalytic bed.

2.1 To decrease work pressure of the catalytic bed

Decrease of the reaction pressure can increase the transformation ratio of the CH_4. From view point of safety, it requires that the He (or N_2) can't be leaked into the process gas when the catalytic tube is broken. So the pressure of the process gas should be higher than gas pressure in the IHX secondary side. For HTR-10 the gas pressure in the IHX secondary side is 3.2MPa. So the process gas pressure at the outlet of catalytic bed is selected 3.4MPa. The pressure drop in the catalytic bed is about 0.6-0.8MPa. So the work pressure of the reformer should be selected about 4.0MPa.

The counter flow mode is adopted in the IHX for the He (or N_2) and the process gas. The pressure difference between the two gases at the inlet of the catalytic bed is about 0.8-1MPa, and at the outlet of the catalytic bed—only 0.2MPa, so that the pressure difference between the two gases can be kept minimum at the hottest pipe section.

2.2 To increase the H_2O/CH_4 ratio

In the transform process of the CH_4, existed at the same time the carbon separation effect:

$$CH_4 \Leftrightarrow C + 2H_2 \quad -82.5\, kJ \qquad (3)$$
$$2CO \Leftrightarrow CO_2 + C \quad +172.5\, kJ \qquad (4)$$
$$CO + H_2 \Leftrightarrow C + 2H_2O \quad +122.7\, kJ \qquad (5)$$

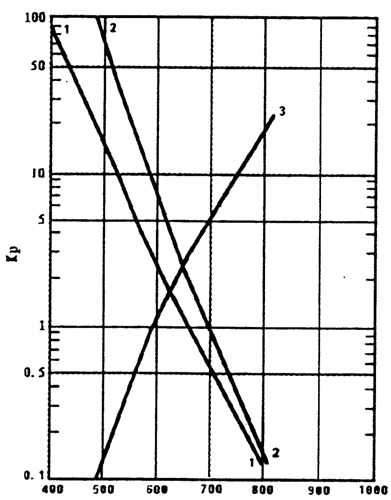

Fig.1 The balance constants of the reaction (3)~(5)

$$1: CO + H_2 = C + H_2O \qquad K_p = \frac{P_{H_2O}}{P_{CO} \cdot P_{H_2}}$$

$$2: 2CO = C + CO_2 \qquad K_p = \frac{P_{CO_2}}{P_{CO}^2}$$

$$3: CH_4 = C + 2H_2 \qquad K_p = \frac{P_{H_2}^2}{P_{CH_4}}$$

The balance constants of the equation (3)-(5) and their dependence on temperature are given in the Fig.1. For different reaction the effects of temperature and pressure on the carbon separation are different. In order to avoid carbon separation in the above three reactions proper steam quantity should be selected.

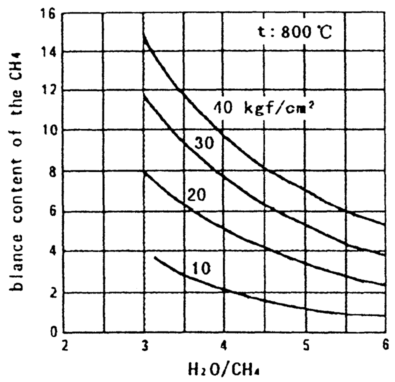

Fig.2 The effect of H_2O/CH_4 ratio on balance content of the CH_4

The effect of H_2O/CH_4 ratio on the balance content of the CH_4 in the CH_4 –steam transform process is given in Fig.2. From Fig.2 it can be seen that increase H_2O/CH_4 ratio is to the benefit of transform reaction. But too big H_2O/CH_4 ratio results decrease of capability of the apparatus and increase of energy consumption. So the proper H_2O/CH_4 ratio should be selected.

2.3 To increase the CH_4 –steam reforming temperature

Reforming reaction (1) is a strong endothermic reactor. Increase of the temperature is of benefit to increase of H_2 productivity.

If the He temperature at the outlet of the reactor core is increased to 950 ℃ , IHX secondary side outlet temperature is increased to 905℃, the He (or N_2) temperature at the outlet and inlet of the reformer can be increased to 600/890℃. Both the increase of reforming temperature and the temperature difference between two side gases can improve the productivity.

Series connection operating mode for steam reformer (SR) and steam generator (SG) can increase the He (or N_2) temperatures at inlet and outlet of reformer. Consequently, it can also increase the temperature difference between the He (or N_2) and the process gas, and increase the productivity per unit power.

2.4 To intensify heat transfer on the He (or N_2) side

Construction of the tube-shell type reformer requires the forced convection mode on outside of the heat transfer tube. Efficiency of forced convection heat transfer for gas is lower than that of radiation heat transfer. In order to increase heat transfer efficiency it is necessary to increase velocity and turbulence of the gas flow.

Analysis shows that use of cross flow can improve heat transfer more than concentric pipe. But at the same time, the hydraulic resistance will be more.

Application of ribbed construction on the outside surface of the catalytic pipe can effectively increase the gas heat transfer coefficient. In our case, the total heat transfer coefficient can be increased to about 30% .

2.5 To improve the heat transfer features of the catalytic pipes

Dynamics analysis of the CH_4-steam transform shows that, existing catalyst, relatively high reaction velocity can be achieved under temperature 600-800 ℃ . In this case reforming reaction is controlled mainly by heat transfer and balance dynamics.

Use of increasing balance transformation rate and to intensify heat transfer outside catalytic pipes to increase transformation rate have been analyzed above. Analysis of heat transfer features shows the thermal resistance of the catalytic bed cannot be neglected. So improvement of catalytic bed heat transfer features can also considerably increase transformation rate.

Heat transfer processin the catalytic bed is very complicated. It includes conventional heat transfer between solid particles and gases, heat conductivity in the solid particles and gases, and radiation heat transfer in the catalytic bed. All these processes effect on He final results of heat transfer. Therefore, usually empirical formulas are used for catalytic bed heat transfer calculation. In our design effective conductivity of the catalytic bed is used to character its heat transfer features:

$$\frac{\lambda_e}{\lambda} = \frac{\lambda_e^0}{\lambda} + (\alpha\beta)\,\mathrm{Re}\,\mathrm{Pr}$$

There

λ is gas heat conductivity

λ_e is solid particle heat effective heat conductivity

λ_e^0 is catalytic bed heat conductivity when the gas velosity=0;

α is ratio of sectional mass transfer velocity to the mass transfer velocity in the flow direction;

β is coefficient characteristic effects of particle diameters and distances between particles.

The effective heat conductivity of the catalytic bed depends on empty bed velocity, porosity of the bed, construction of the catalyst etc. Increase of the empty bed velocity can effectively improve heat transfer of the catalytic bed. But this means to increase the production intensity.

As mentioned above, catalytic bed heat transfer is a complicated process. Necessary experiments and verification are required.

3. Computer Code REFORM

Methane-steam reformer is a counter-flow mode, tube-shell type, gas-solid fixed-bed reaction-heat exchanger. In the reformer heating gas is coming from the secondary side of the Intermediate Heat-exchanger (IHX). Main reactions occurred in the reformer are the equations (1) and (2). The computer code REFORM is used for reformer design calculation and parameter selection.

3.1 Main equations of the fixed-bed reaction apparatus

• The mass conservation equation:

$$F\,dx = r_1\,dw \qquad F\,dy = r_2\,dw$$

There

F is entering rate of the raw gas, kmol/s;
x is methane transformation rate;
y is CO_2 formation rate;
w is weight of the catalyst, kg;
r_1 is reaction rate of the equation (1);
r_2 is reaction rate of the equation (2).

• The energy conservation equation:

$$\Sigma m_i C_{pi} dT = u_1 A_1 (T_{s1} - T)dZ + u_2 A_2 (T_{s2} - T)dZ - Fdx\Delta H_1 - Fdy\Delta H_2$$

There

ΔH_1 and ΔH_2 is reaction heat of the reactions (1) and (2), respectively;
m_i is mass of the i component, kmol/s;
C_{Pi} is specific heat (at constant pressure) of the i conponent kJ/kmol • K;
T is process gas temperature in the catalytic tubes, K;
T_{S1}, T_{S2} is heating gas temperatures, K;
u is heat transfer coefficient, kJ/m^2.s.K;
A_1, A_2 is heat transfer area per length of the tube, m^2/m;
Z is length of the tube (beginning at the inlet), m.

• The momentum conservation equation

$$dp = f'\left(\frac{\rho u^2}{d_s}\right)\left(\frac{1-\varepsilon_B}{\varepsilon_B}\right)dZ$$

There

ε_B is porosity of the catalytic bed;
ρ is density of the fluid;
u is average empty bed velocity;
d_s is equivalent diameter of the particle.

3.2 The main hypotheses

- The code REFORM is developed on the basis of quasi-homogeneous phase model. In other words, the solid particles and the fluid are considered as a homogeneous phase system.
  - Only one dimension axial flow is considered. That means the fluid temperature and concentration in a cross section are equal.
  - The fluid integral moving model is used in the code.
  - The ideal gas equations can be used for the mixed gas.
  - The temperature distribution of the heating gas along the tube is linear.

3.3 The reformer parameters analyses

Using the computer code REFORM, relationship between the methane transformation ratio and the heat transfer, optimized reforming power and effect of reactor outlet temperature on the transformation ratio are analyzed.

From analyses, it is suggested that for the HTR-10, the core outlet temperature should be raised up to 950 °C and the helium-helium IHX would be used. Series connection operating mode can be adopted for the process heat application system. The parameters can be selected as follows: Reformer power =2.5MW, steam generator power=2.5MW, reforming temperature ≈ 840 °C. In this case the transformation ratio is about 0.7. The process heat application system is show in Fig.3.

4. Construction and Main Parameters of the HTR-10 Reformer Heated by He gas

Following design parameters and geometrical dimensions of the reformer are also selected by the same analyses as mentioned above.

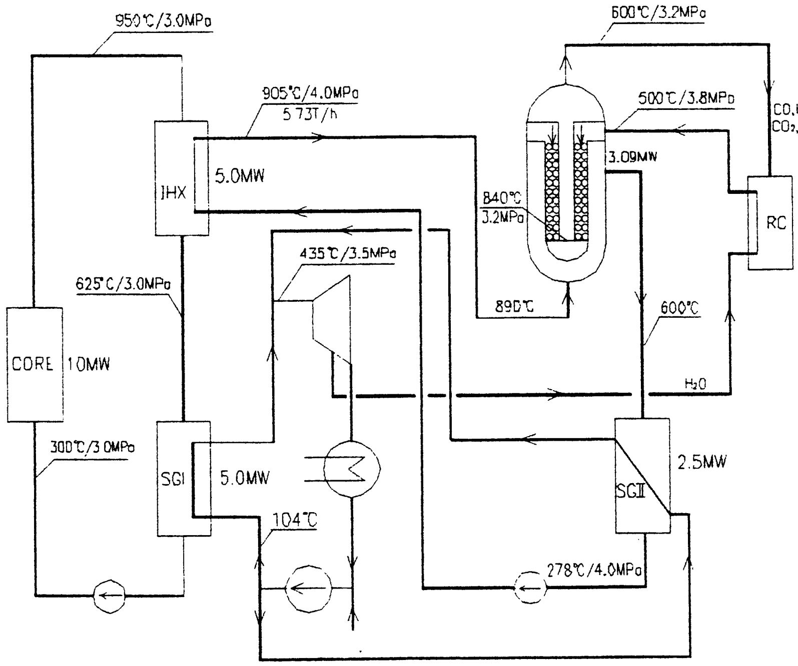

Fig.3 HTR-10 Nuclear process heat application system

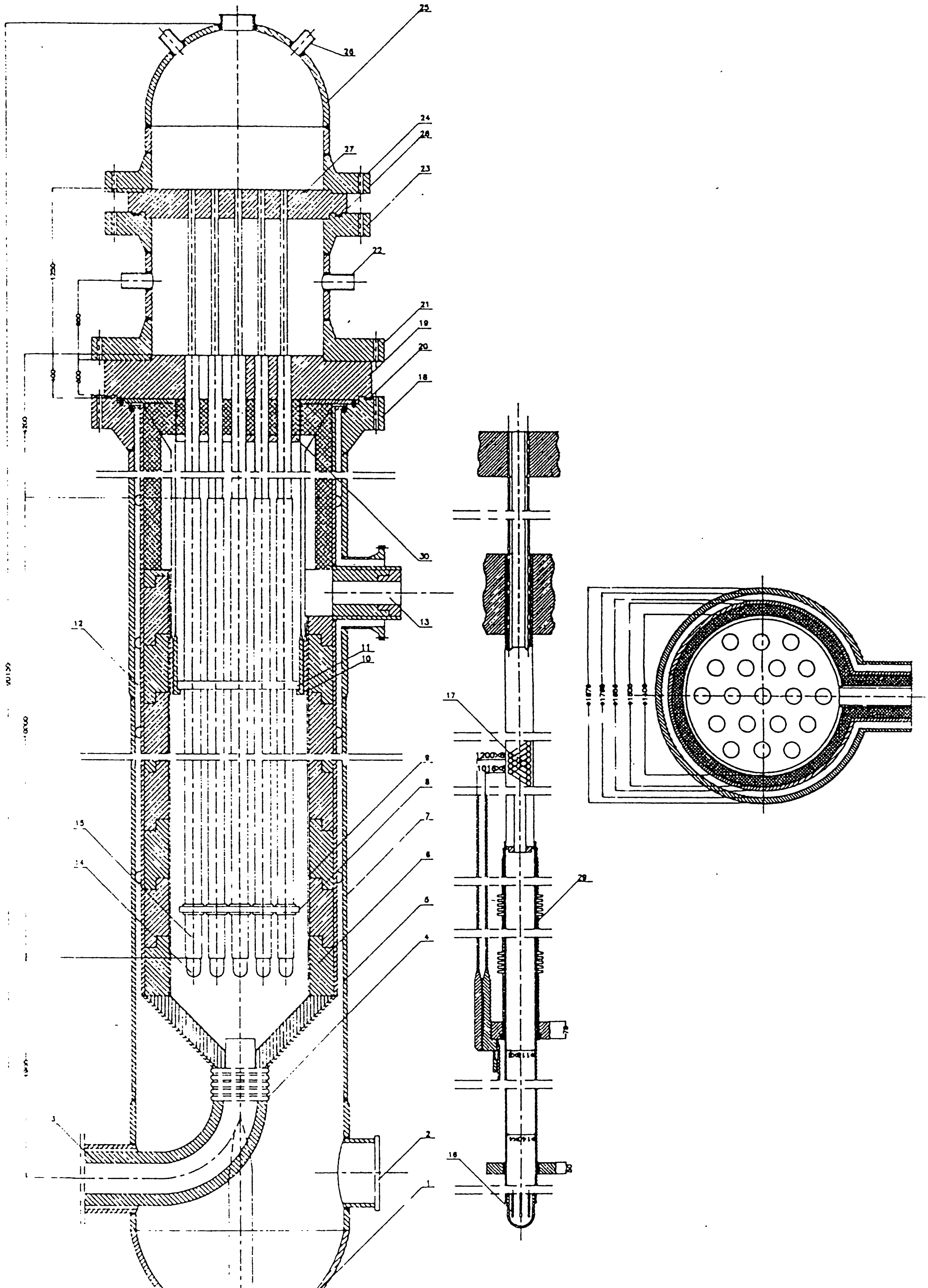

Fig. 4 The construction of the reformer

4.1 Main parameters

Power of the reformer	3.09MW
Work pressure of the helium	3.0MPa
He flow rate	5.73T/hr.
He temperature at the outlet/inlet	600/890 ^oC
Composition of the process gas	
Raw gas	CH_4/H_2O=1:4
Product gas	CH_4 : 3.11% CO: 6.32%
	CO_2 : 5.74 %; H_2 : 42.89 %
	H_2O: 41.93 %。
Process gas flow rate	1.57 kg/s
Process gas temperature at the outlet/inlet of the catalytic bed	840/500 ^oC
Process gas pressure at the outlet/ inlet of the catalytic bed	3.7/3.4MPa

4.2 Construction parameters of the reformer

Parameters of the tube-bundle	
Number of the reformer tube	30
Diameter of the tube bundle	1400mm
Diameter of the catalytic tube	Φ 116x8mm
Active section length of the catalytic tube	10140mm
Diameter of the helium coat tube	Φ 140x4
Diameter of the centre guide tube	3xΦ18x1.5
Parameters of the vessel	
Working pressure	4.4MPa
Working temperature	350 ^oC
Outside diameter, mm	2276 mm
Height, mm	20010mm
Parameters of the regenerator	
Working pressure	4.4 MPa
Outlet/inlet temperature	
Raw gas	320/500 ^oC
Product gas	600/400 ^oC
Diameter of the spiral tube	Φ18x1.5
Number of the tubes	3x30
Outside diameter of the helical tube	Φ54
Height	3.72m

4.3 Construction figures

The construction of the reformer is shown in Fig.4.

DEVELOPMENT PROGRAM ON HTTR HEAT APPLICATION SYSTEMS AT JAERI

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Abstract

The High Temperature Engineering Test Reactor (HTTR), which is a Japanese High Temperature Gas-cooled Reactor (HTGR) with 30MW thermal output at 950°C of the coolant outlet temperature, was constructed at Oarai Research Establishment of Japan Atomic Energy Research Institute (JAERI). The HTTR has attained the first criticality on November 1998. In JAERI, a hydrogen production system was selected as a heat utilization system of the HTTR. The development program on the HTTR hydrogen production system consists of two parts ; one is to establish technologies connecting the hydrogen production system with the HTTR, the other is to establish technologies producing hydrogen from water by using nuclear heat. Finally, hydrogen can be produced from water by using nuclear heat supplied by the HTTR. In the hydrogen production system connected to the HTTR at first, JAERI selected a steam reforming process because its technology had matured. The HTTR hydrogen production system adopting the steam reforming process is being designed to produce hydrogen of about 3800 Nm^3/hr by using nuclear heat (10MW, 905°C) supplied from the HTTR. The safety principle and criteria are also being investigated for the HTTR hydrogen production system. A facility for an out-of-pile test prior to the demonstration test with the HTTR hydrogen production system is under manufacturing to carry out tests of safety, controllability and performance. The out-of-pile test facility simulates key components downstream an intermediate heat exchanger of the HTTR hydrogen production system on a scale of 1 to 30. The tests will be started in 2001 and continued for 4 years or longer. In parallel to the tests, a hydrogen/tritium permeation test and a corrosion test of a catalyst tube of a steam reformer are being carried out to obtain data necessary for the design of the HTTR hydrogen production system. A kind of thermochemical methods called IS process is under studying to produce hydrogen from water by using nuclear heat. Stable hydrogen production of 0.001 Nm^3/hr has been successfully demonstrated for 48 hours in a laboratory scale experiment. The study has been just started to head for the next engineering step aiming at 0.05 Nm^3/hr from this year. After the demonstration test of the steam reforming process, the steam reforming process is planed to be replaced with the IS process in the HTTR. The research and development on the nuclear heat application systems was consigned by Science and Technology Agency since January in 1997. In the presentation, an overview of the HTTR heat application systems at JAERI is described with an emphasis of technical subjects to be solved for commercialization as well as technical achievement obtained so far.

1. INTRODUCTION

Consumption of a huge amount of fossil fuels due to invention of a steam engine in the industrial revolution has caused an enhanced global warming. In order to relax the global warming issue, that is, to reduce CO_2 emission, new energy resource/carrier and technologies for those are required to be developed at present. Nuclear energy can satisfy a large amount of energy demands without significant CO_2 emission. The most of nuclear energy are used for generating electricity. The ratio of electricity to the secondary energy demands, however, is small as about 21% in Japan 1995, and is predicted to be only 23% in 2010. High Temperature Gas cooled Reactors (HTGRs) having many outstanding safety features can produce a very high outlet temperature as high as 1000°C. Therefore HTGRs can generate electricity with high thermal efficiency of about 50%. In addition, the nuclear heat with high temperature such as a 1000°C is possibly used in the non-electric field, because it can cover the region of 60–70 % to all kinds of industries using heat. Thus, HTGRs can provide nuclear energy to both electric and non-electric fields and can contribute to reduction of CO_2 emission.

Japan Atomic Energy Research Institute (JAERI) has constructed a 30-MWt HTGR with a reactor outlet coolant temperature of 950°C, named HTTR (High Temperature engineering Test Reactor), to develop technology and to demonstrate effectiveness of high-temperature nuclear heat utilization. The first criticality of the HTTR has been just achieved in November 10, 1998, and the power rise test is being performed in the HTTR, then reactor performance tests, safety demonstration tests, and high temperature irradiation tests will be carried out [1]. After the tests, a high-temperature nuclear heat utilization system is planned to be coupled with the HTTR from around 2002.

A hydrogen production system is considered as one of the leading nuclear heat utilization systems in non-electric field because hydrogen has superior characteristics as energy carrier and its demand is expected to increase in near future. If combustion of fossil fuels would not be allowed because of reducing CO_2 emission, an alternative fuel to obtain heat energy with high temperature is only hydrogen which burns to become water. Research and development (R&D) on the hydrogen production systems are as follows;

(1) R&D of connection technologies between a nuclear plant and a chemical plant,
(2) R&D on hydrogen production by water splitting method with nuclear heat.

In the first item, JAERI is designing a hydrogen production system to connect to the HTTR [2]. A steam reforming process with natural gas was selected for the the HTTR hydrogen production system, because its technology has already matured in existing fossil-fired plants, and are applicable to other hydrogen production systems and chemical plants. In the second item, JAERI is carrying out R&D on an iodine sulfur (IS) process of a kind of thermochemical method for hydrogen production by means of water splitting [3]. The basic study on the IS process has just succeeded in a laboratory scale experiment, then the study goes forward to a next step from this year.

The overview of the development program is described on the HTTR heat application systems at JAERI in Japan.

2. DEMONSTRATION PROGRAM OF HYDROGEN PRODUCTION IN HTTR

Figure 1 shows the development schedule of the demonstration program of the hydrogen production at JAERI. The development program for the HTTR hydrogen production system with the steam reforming process consists of the development of the HTTR hydrogen production system itself, an out-of-pile test, and two component tests. The dotted lines in Fig.1 shows that its execution will be decided after taking the check and review (C&R) in around 2000. The design and safety studies of the HTTR hydrogen production system with the steam reforming process are carried out until about 2001. The facility of the out-of-pile test for the HTTR hydrogen production system is designed and constructed in 1996–2000, then the out-of-pile test is performed until 2004 or longer. In the component tests, two experimental apparatus were completed in 1997, and the hydrogen/tritium permeation test and the corrosion test are being carried out in 1998–2000. After the C&R of the HTTR hydrogen production system, the construction of the HTTR hydrogen production system will be started from 2004 and the demonstration test from around 2006. In parallel, a facility of a close cycle test for the IS process is being designed and manufactured from this year. The test will be started from 2001 to 2004. Passing through a next bench-scale test, the steam reforming process of the HTTR hydrogen production system is planed to be replaced with the IS process.

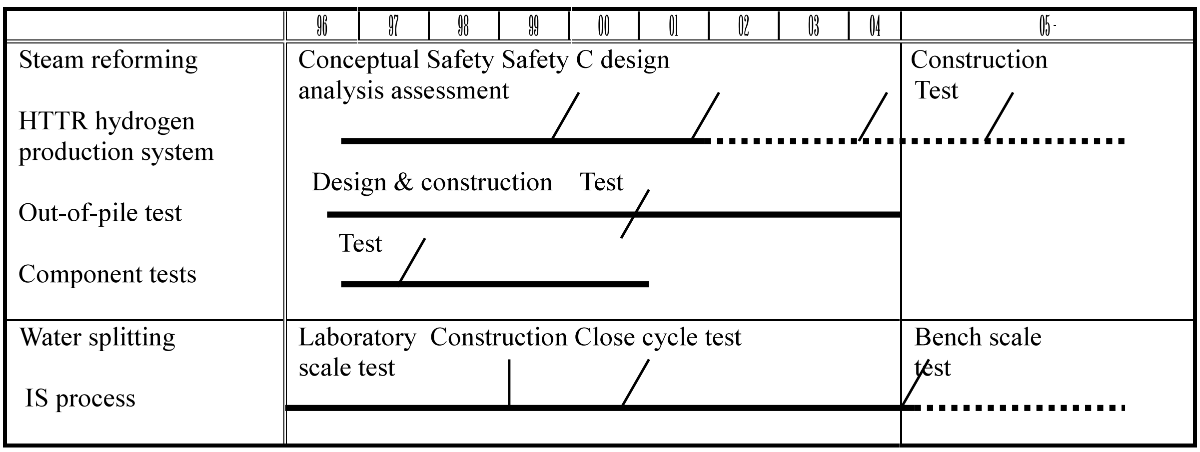

Fig.1 Schedule of demonstration program on HTTR heat application systems at JAERI.

The HTTR hydrogen production system with the steam reforming process is designed to utilize the nuclear heat effectively and achieve hydrogen productivity competitive to that of a fossil-fired plant with operability, controllability and safety acceptable enough to commercialization. Figure 2 shows an arrangement of the main components. The HTTR reactor supplies nuclear heat of 10MW with 950°C to the IHX in the reactor cooling loop, and then the nuclear heat is transferred from the IHX to the secondary helium loop to be utilized for the production of hydrogen. Due to heat loss along the secondary helium piping from the IHX to a steam reformer (SR), the secondary helium temperature is reduced to 880°C at the SR inlet, whereas the IHX outlet temperature is 905°C. Design specifications of the HTTR hydrogen production system is shown in Table 1. The key components, such as the SR and a steam generator (SG), and their arrangement were designed to achieve hydrogen productivity competitive to that of a fossil-fired system with operability, controllability and safety acceptable enough for commercialization [2].

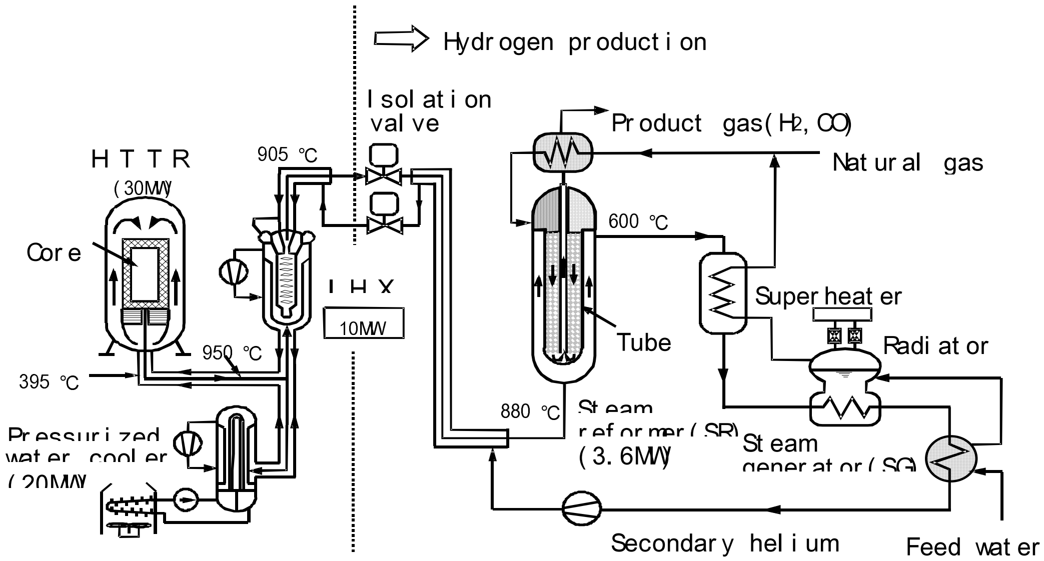

FIG. .2. Flow scheme of HTTR steam reforming hydrogen production system.

Table 1. Design specifications of the HTTR and out-of-pile steam reforming hydrogen production systems

Items	HTTR system	Out-of-pile test system
Pressure ; Process gas/Secondary helium gas	4.5/4.1 MPa	
Inlet gas temperature of steam reformer Process gas/Secondary helium gas	450/880°C	
Outlet gas temperature of steam reformer Process gas/Secondary helium gas	600/600°C	600/650°C
Natural gas feed	1300kg/hr	43kg/hr
Helium gas feed	8700kg/hr	330kg/hr
Steam/carbon ratio	3.5	2–4
Hydrogen production rate	3800 Nm^3/hr	110 Nm^3/hr
Heat source	Reactor (10MW)	Electric heater (380kW)

Although the hydrogen production system by steam reforming is matured in fossil-fired plants, some safety-related technology should be developed for coupling with HTGRs as well as the HTTR in the following.

(1) Mitigation of thermal disturbance to reactor

The SG supplies steam to the SR, and can also stabilize the inlet temperature of the IHX in the secondary helium coolant loop. Even if the helium gas temperature at the SR outlet, that is, the SG inlet is increased by some thermal disturbance such a malfunction in the rocess gas line, the helium gas temperature at the SG outlet can be kept constant at the saturation temperature of steam by controlling the pressure in the SG. This performance of the SG working as an absorber of thermal disturbance can make it possible that the nuclear reactor is stopped according to a normal operation procedure but not with a reactor scram for some malfunction or accident at the heat utilization system. We aim to limit the temperature fluctuation of the secondary helium gas within 10 K at the SG outlet, because the temperature rise above 15 K compared with the normal temperature at the reactor inlet causes the HTTR reactor scram.

(2) Assurance of structural integrity of catalyst tube

(a) Control of pressure difference between helium and process gases at catalyst tube

The catalyst tube in the SR is a component important to safety, because it forms a pressure boundary between helium and process gases. In design of the catalyst tube, its wall thickness is decided considering both outer pressure of helium gas at 4.1MPa and inner pressure of process gas at 4.5MPa to assure the structural integrity in all conditions such as not only normal startup and shutdown but also malfunction and accident at the heat utilization system. This design, however, makes the wall thickness too large; for example, the wall thickness becomes about 130mm and an inner diameter 128mm using Alloy 800H. To realize the reasonable wall thickness such as 10mm order, it must be decided considering the pressure difference between helium and process gases, and a control system is required to keep the pressure difference within an allowable value. In concrete terms, the control system makes the process gas pressure follow pressure change of helium gas.

(b) Estimation of hydrogen embrittlement and corrosion of catalyst tube

The catalyst tube is designed to be made of Hastelloy XR, which is a nickel-base, helium corrosion — and heat-resistance super alloy developed for the HTTR by the JAERI. It is necessary to examine characteristics of corrosion due to metal dusting and oxidation and strength reduction due to hydrogen embrittlement. The corrosion test is described later.

(3) Estimation of tritium permeation

Tritium produced in the HTTR core flows with the primary helium gas coolant to the IHX, then permeates through the Hastelloy XR tube of the IHX to the secondary helium gas coolant and through the Hastelloy XR tube of the SR, at last, mixes with the process gas. Therefore tritium concentration in the process gas must be estimated because tritium cannot be perfectly removed by a purification system in the HTTR. The hydrogen/tritium permeation test is described later.

3. OUT-OF-PILE TEST

The main objectives are investigation of transient behavior and establishment of operation and control technology, focussing on establishment of the safety-related technology described previously, as well as design verification of performance of high temperature components, such as the SR and SG.

The test facility has an approximate hydrogen production capacity of 110Nm^3/h and simulates key components downstream the IHX of the HTTR hydrogen production system on a scale of 1 to 30[4]. Design specifications of this test system is also shown in Table 1. Figure 3 shows a schematic flow diagram of the test facility. An electric heater with 380kW is used as a heat source instead of the nuclear heat to heat helium gas up to 880°C at the SR inlet of the same conditions as the HTTR hydrogen

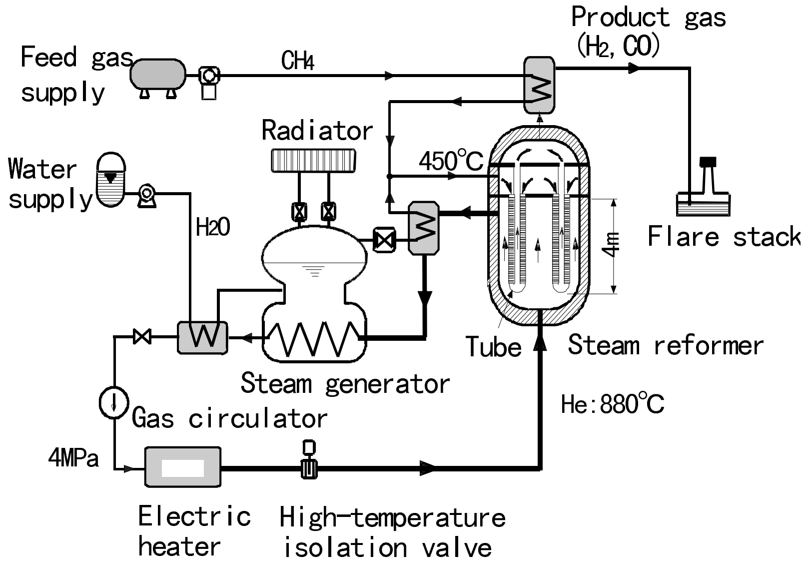

FIG. .3. Flow scheme of out-of-pipe test system.

production system. The process gas pressure is controlled by a control valve installed downstream the SR, monitoring the pressure difference between helium and process gases.

In the fossil-fired plant, the process gas receives heat from combustion air of about 1200°C by heat radiation, and the heat flux at the outer surface of the catalyst tube reaches 70 000–87 000 W/m^2. In order to achieve the same heat flux as that of the fossil-fired plant, it is very important to promote heat transfer of helium gas by forced convection because the temperature of helium gas the temperature of heat source is too low compared with that of the fossil-fired plant. So, disc-type fins, 2mm in height, 1mm in width and 3mm in pitch, are arranged around outer surface of the catalyst tube in the test facility to increase a heat transfer coefficient of helium gas by 2.7 times, 2150W/m^2K with the fins, and a heat transfer area by 2.3 times larger than those of smooth surface, respectively. As the result, the heat transfer performance of the catalyst tube in the test facility becomes competitive to that of the fossil-fired plant.

The test plan consists of three categories; (i) normal startup/shutdown test, (ii) safety-related test, and (iii) high-temperature component test. The objective of the normal startup/shutdown test is to optimize a feed ratio of steam to natural gas according to change of the temperature and pressure of helium gas supplied from the HTTR to restrain the above fluctuation within allowable range. The objective of the safety-related test is to establish the safety-related technology dealing with malfunction and accident at the process gas line. The emergency shutdown method of the hydrogen production system will be also established to assure the safety, especially structural integrity of the catalyst tube, by the experiment. Thermal and hydraulic performance of the SR and SG is clarified in the high-temperature component test. The SR is investigated focusing on chemical reaction characteristics which is very important to predict transient behavior and hydrogen productivity of the hydrogen production system. The pressure controllability, transient behavior of temperature of helium gas and steam, steam production rate and natural convection of steam and condensed water will be investigated in detail.

4. COMPONENT TESTS

In parallel to the out-of-pile test described above, the corrosion test and hydrogen/tritium permeation test are being carried out in small apparatus to establish the safety-related technology, to obtain detailed data for a safety review of the HTTR hydrogen production system, and to develop a calculation code of hydrogen/tritium permeation.

The objective of the corrosion test is to estimate the effect of corrosion, oxidation and hydrogen embrittlement on strength reduction of Hastelloy XR. Metallography and material tests are in progress on strength and creep of test specimens exposed in the corrosive gases such as CH_4, CO, H_2O, and H_2 with temperature up to 900°C.

Tritium produced in the HTTR core permeates into the hydrogen production system, on the other hand hydrogen in the product gases also permeates into the HTTR core to the opposite direction of the tritium permeation. The aims of the hydrogen/tritium permeation test are to obtain the data of permeation coefficient in the very low tritium partial pressure less than 10 Pa, to examine the effect of an isotope of hydrogen simultaneously existing in the gas which is called mutual diffusion, and the effect of protection for hydrogen/tritium permeation by the coating film on the reforming tube such as oxidation film, calorizing film and so on [5]. In the mutual diffusion test, deuterium is used instead of tritium as the isotope of hydrogen. The data on hydrogen permeation of the Hastelloy XR has been obtained as a standard data.

5. Hydrogen Production by Water Splitting

Hydrogen production from water using HTGRs is considered as an ideal method because any CO_2 emission is not expected from the system. JAERI has been conducting basic studies on the IS process, which is a kind of thermochemical methods for hydrogen production by water splitting, as a next heat utilization system of the HTTR following the steam reforming system described above. The IS process produces hydrogen by absorbing a high temperature heat with 800–900°C supplied from HTGRs. The IS process is composed of the following three chemical reactions. The details of the IS process is shown in Fig.4.

$$2H_2O + SO_2 + I_2 \rightarrow H_2SO_4 + 2HI \quad (1)$$
$$H_2SO_4 \rightarrow H_2O + SO_2 + (1/2)O_2 \quad (2)$$
$$2HI \rightarrow H_2 + I_2 \quad (3)$$
$$H_2O \rightarrow H_2 + (1/2)O_2 \quad (4)$$

The process works like a chemical engine to produce hydrogen by absorbing high temperature heat through endothermic decomposition of sulfuric acid and dissipating low temperature heat through exothermic reaction. The IS process has attractive features such that all the process chemicals are used in its fluid phase and the endothermic sulfuric acid decomposition reaction proceeds stoichiometrically with large entropy change. The IS process was proposed and studied by General Atomic Co. [6] and has been studied also in Japan [3], Germany [7], and Canada [8].

JAERI's effort has so far been concentrated on demonstrating the continuous hydrogen production by connecting the three chemical reactions. By acquiring physico-chemical data relating to the step of reaction (1), it could be realized the continuous and stoichiometric production of hydrogen and oxygen with stable rate [9]. Figure 5 shows a result of the laboratory scale experiment carried out. Stable production of hydrogen of 0.001Nm^3/h and oxygen of its half from water has been successfully made in a closed cycle process operated continuously for 48 hours. In Fig.5, to show the stability of the process solution, the fluctuation of the iodine concentration in sulfuric acid solution is also shown to be small.

At present, a demonstration using a scaled-up glass apparatus is under study, where the operating condition will be modified to realize the more efficient hydrogen production in a liquid phase separator placed between Reaction (1) and Reactions (2), (3) at elevated temperature condition (0°C to 95°C) to achieve better separation of HI and H_2SO_4. Also, an introduction of advanced separation technologies, which includes ceramic hydrogen separation membranes for efficient HI decomposition, is under study to improve the process scheme. In parallel with these process studies, studies on materials of construction for further scaling up are in progress to meet the corrosive process conditions such as boiling sulfuric acid and SO_2-SO_3-H_2O-O_2 gaseous mixture at 800°C. As for the boiling sulfuric acid condition, iron-silicon alloys and silicon impregnated SiC are the promising candidates from the viewpoint of corrosion resistance. A closed-cycle test as an next step in which hydrogen of 0.05 Nm^3/h is produced has been started from this fiscal year.

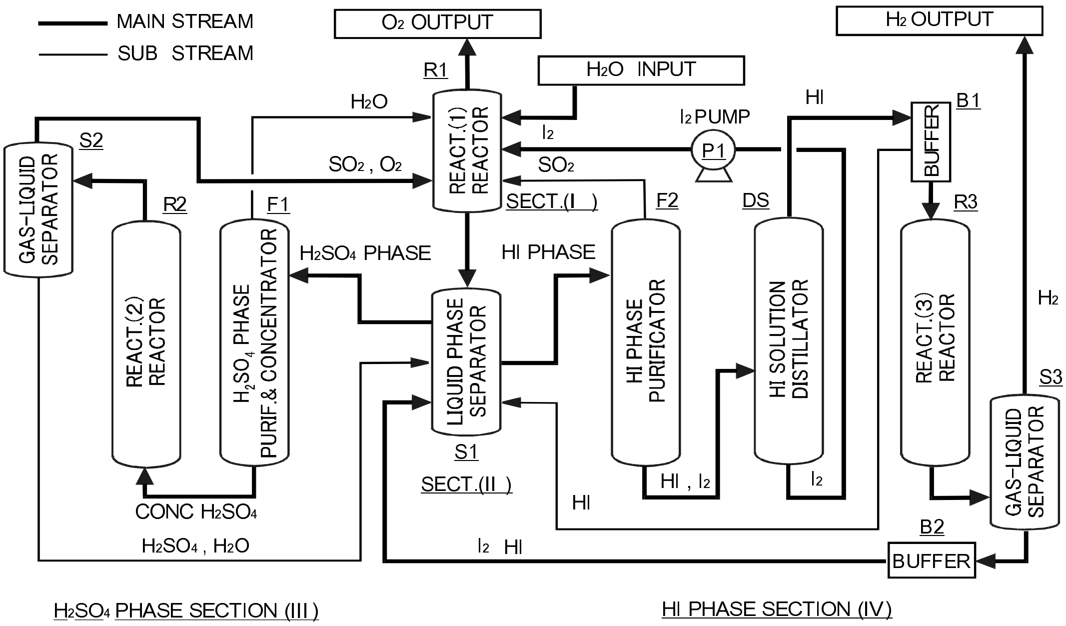

FIG. 4. Schematic diagram of IS process.

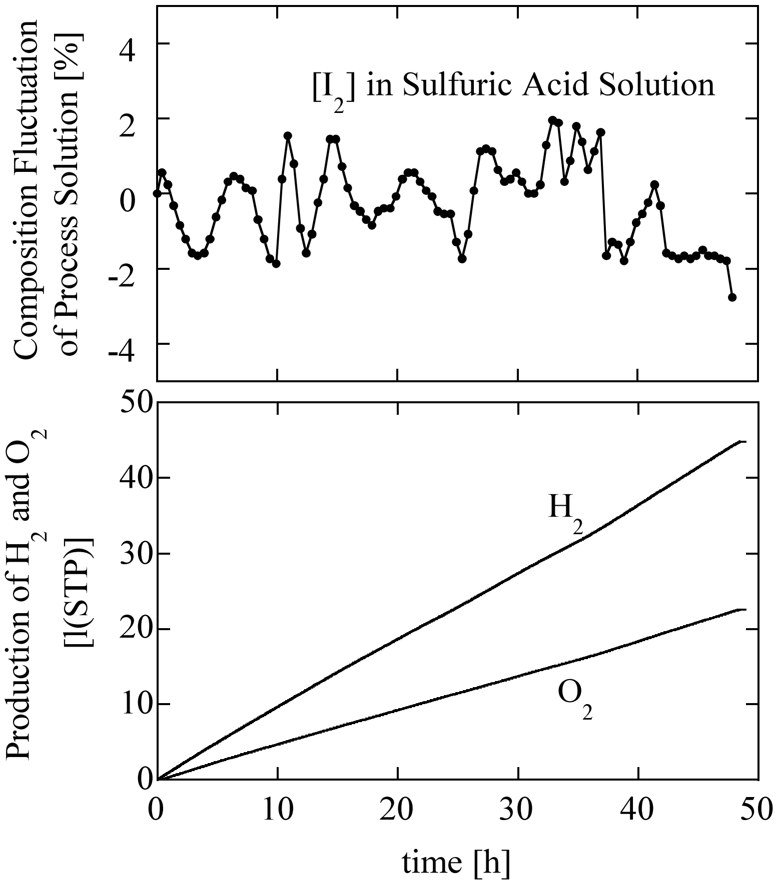

FIG.5. Results of stable hydrogen production operated continuously for 48 hours in laboratory scale test of IS process

6. CONCLUDING REMARKS

Under an understanding that HTGRs can play an important role to expand the nuclear heat application to chemical industries against the current environmental issue of the CO_2 , JAERI proceeds with the development of the nuclear process heat application system coupling to the HTTR. Global eyes are kept by not only nuclear persons of interest but also the public upon the development of the HTTR heat application system, since its successful achievement may enhance the possibility to solve the environmental issue of CO_2 emission as well as a possible energy crisis which might happen in the future.

Finally it should be emphasized that an overall support and understanding from the overseas countries of concern are needed and wished for the success of the Project. The Project is highly expected to contribute so much to promoting international cooperation on the development of HTGRs and its process heat application.

The R&D of the nuclear heat application systems was consigned by Science and Technology Agency.

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HIGH TEMPERATURE NUCLEAR HEAT FOR ISOTHERMAL REFORMER

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Abstract

High temperature nuclear heat can be used to operate a reformer with various feedstock materials. The product synthesis gas can be used not only as a source for hydrogen and as a feedstock for many essential chemical industries, such as ammonia and other products, but also for methanol and synthetic fuels. It can also be burnt directly in a combustion chamber of a gas turbine in an efficient combined cycle and generate electricity. In addition, it can be used as fuel for fuel cells. The reforming reaction is endothermic and the contribution of the nuclear energy to the calorific value of the final product (synthesis gas) is about 25%, compared to the calorific value of the feedstock reactants. If the feedstock is from fossil origin, the nuclear energy contributes to a substantial reduction in CO_2 emission to the atmosphere. The catalytic steam reforming of natural gas is the most common process. However, other feedstock materials, such as biogas, landfill gas and CO_2-contaminated natural gas, can be reformed as well, either directly or with the addition of steam. The industrial steam reformers are generally fixed bed reactors, and their performance is strongly affected by the heat transfer from the furnace to the catalyst tubes. In top-fired as well as side-fired industrial configurations of steam reformers, the radiation is the main mechanism of heat transfer and convection heat transfer is negligible. The flames and the furnace gas constitute the main sources of the heat. In the nuclear reformers developed primarily in Germany, in connection with the EVA-ADAM project (closed cycle), the nuclear heat is transferred from the nuclear reactor coolant gas by convection, using a heating jacket around the reformer tubes. In this presentation it is proposed that the helium in a secondary loop, used to cool the nuclear reactor will be employed to evaporate intermediate medium, such as sodium, zinc and aluminum chloride. Then, the vapors of the medium material transfer the heat to the reformer and condense on its walls. Three configurations can be conceived. The vapors of the sodium are condensed on the outside surface of the reformer tube and the liquid is then drained into a pool-boiling-He/Na heat exchanger. Another option is internal heating of an annular catalyst bed reformer with a sodium heat pipe, for instance. The third option is reformer tubes immersed inside the boiling medium. In all cases means for the enhancement of heat transfer can be applied. The use of condensing metal vapors increases the heat transfer compared to the convective reformer, resulting in more compact reformers. The safety aspects of this approach are twofold: (a) Additional physical separation between the nuclear reactor coolant and the chemical plant. This will enable to reduce the operating pressure inside the reformer, thus reducing the working temperature and increasing the extent on the reaction and the CH_4 conversion. (b) In case of a failure in the chemical plant, the liquid metal can be used as a safety buffer. In this event, the vapors of the medium material are diverted and condensed in an emergency condenser and returned to the pool boiling by natural circulation, avoiding the need to reduce immediately the power rating of the nuclear reactor or even its complete shutdown.

1. INTRODUCTION

The high temperature gas-cooled reactors (HTGR) can provide heat at a temperature range of 950–1000°C. The heat is delivered by helium, which is used as the coolant for the reactor in the primary loop, usually through an intermediate heat exchanger to a secondary loop. These high temperatures can be used to endothermic chemical processes, such as the reaction between hydrocarbon and steam or CO_2 (known as reforming), to produce a mixture of hydrogen and carbon monoxide (synthesis gas).

The medium range of temperatures can be further used for production of steam for the reforming process and for the generation of electricity. Finally, the low temperatures can be exploited for process heat, space heating, refrigeration and water desalination.

This range of potential uses can make the HTGR an efficient and important energy source, especially due to its inherent safety features.

So far, nuclear reformers have been developed primarily in connection with the transportation of nuclear heat [1], aimed at working in a closed loop with a matching methanator where the reaction is reversed and useful heat is released at the customer site.

The current reformers are proposed to use the nuclear heat in an open loop mode for the production of fuels and chemicals. Work in this direction is presently performed in Japan [2] and other places. One of the main drawbacks of the nuclear reformer heated by the reactor coolant is the relative lower heat transfer coefficient compared to an industrial reformer and lower conversion of the hydrocarbon feed (efficiency of the reaction), because of the relative high working pressure.

The present paper proposes a method to increase both the heat transfer coefficient and the conversion with additional safety and flexibility in the interfacing between the chemical reformer and the nuclear reactor because of the relative high working pressure.

2. THE REFORMING PROCESS

2.1. Steam reforming

Catalytic steam reforming of hydrocarbon feedstock is a basic process in the chemical industry and is used for the manufacture of hydrogen, ammonia and methanol [3]. The basic reaction is:

$$C_nH_m + nH_2O \rightarrow nCO + \left(\frac{m}{2} + n\right)H_2$$

Methane is the most common feedstock. The reaction is:

$$CH_4 + H_2O \rightarrow CO + 3H_2$$

$$\Delta H^o_{25^oC} = 206kJ/mole = 49.3kcal/mole$$

The reaction is highly endothermic and therefore is conducted in the industry in a number of parallel long tubes at a temperature range of 750–850°C and pressure of 10–30 atm.

In general, the reaction is not kinetically controlled and equilibrium can be reached at every point in the reformer. The limiting parameter is the heat transfer into the reaction.

There are several potential side reactions: the water gas shift (WGS), which is an exothermic reaction:

$$CO + H_2O \rightarrow CO_2 + H_2$$

$$\Delta H^o_{25^oC} = -41.2kJ/mole$$

and several undesirable side reactions, like solid carbon deposition in the catalyst bed, which causes its activity reduction, disintegration (powdering) and increases the pressure drop in the reactor:

$$CH_4 \rightarrow C + 2H_2$$

$$2CO \rightarrow C + CO_2$$

$$CO + H_2 \rightarrow C + H_2O$$

$$CO_2 + 2H_2 \rightarrow C + 2H_2O$$

Increasing the steam to carbon ratio much higher than the stoichiometric favors hydrogen production and reduces the carbon formation. Ratios of 3–4 are typical in the industry. However, high ratios result in an energetically inefficient process. The equilibrium compositions for methane reforming in the case of molar steam to methane ratio of 3:3 reveal that if, for instance, 96–98% CH_4 conversion is desired, the following equilibrium conditions are required:

CH_4 conversion (%)	Temperature (°C)	Pressure (atm)
98	650	1
98	760	5
98	840	10
98	970	40
96	620	1
96	740	5
96	790	10
96	920	40

It is therefore desirable for nuclear reformers, where the energy source temperature is limited to only 950–1000°C, to operate at lower pressures. The range between 5–10 atm should be considered.

The industrial tubular reformers are mostly top- or wall-fired box type units. In the top-fired reformers, the heat is provided through radiation from long burner flames (typical 2–2.5-m length) and the hot flue gases. In the wall-fired reformers a row of tubes is heated mainly by the radiant side wall.

The nuclear reformers, developed so far, are mostly heated by hot helium gas flowing in a heating jacket around the reformer tube and the heat transfer is primarily convective.

2.2. CO_2 reforming

In the past, this reaction was used on a relatively small scale, mostly to produce synthesis gas with low H_2/CO ratio for industrial processes. In recent years, considerable attention has been paid to this process [4], due to the following potential applications:

(a) A promising technology for the utilization of the most common greenhouse gases.

(b) The availability of large amounts of CH_4 and CO_2 mixtures worldwide (naturally contaminated gas wells and biogas), which are not exploited and piped because separation of CO_2 is not economical.

The reaction of CO_2 reforming of methane can be described as:

$$CO_2 + H_2 \Leftrightarrow 2CO + 2H_2$$

$$\Delta H^o_{25^oC} = 247 kJ / mole$$

and the carbon formation situation is even more emphasized than in the case of steam reforming. Nevertheless, catalysts based on noble metals, such as Ru, Rh, Pd, Ir and Pt, show high selectivity for carbon-free operation.

3. POTENTIAL APPLCIATION FOR NUCLEAR REFORMERS

High temperature heat from HTGR provides a variety of potential applications, if used to reform hydrocarbon feedstock; among these are:

(a) Steam and CO_2 reforming of methane and direct combustion of the product synthesis gas in a gas turbine to generate electricity. The nuclear heat contributes about 25% of the calorific value of the product gas compared to the feed, as can be seen in Fig. 1. The HTGR can be used to process biomass or waste through the reforming of the biogas produced by anaerobic digestion of the biomass (see Fig. 2), or through direct pyrolysis of the biomass at 600–700°C and further reforming of the biomass fuels produced during pyrolysis as volatile material. Finally, the product synthesis gas is combusted in a gas turbine to generate electricity at high efficiency.

(b) Production of liquid fuels from the synthesis gas, such as methanol, synthetic gasoline or Diesel oil through the Fischer-Tropsch process. The methanol is especially important not only as a direct fuel but also as a starting material to other automotive fuel additives, turbine fuel, other alcohol substances and fuel for electric cars that use fuel cells.

(c) Production of various commercial chemicals from the synthesis gas, such as formaldehyde, acetic acid, methyl acetate, ethylene, vinyl acetate, and many others.

(d) Production of hydrogen for (i) fuel and hydrogenation processes, e.g., upgrading of heavy oil residues, and (ii) for the production of chemicals like ammonia.

With this variety of potential applications, the nuclear reformer and the HTGR can be employed to provide energetic supply, not only to the electrical sector but also to the transportation and industrial sectors.

4. THE ISOTHERMAL NUCLEAR REFORMER

As described above, the future HTGR limiting temperatures are perhaps 950–1000°C at the primary helium loop and 900–950°C at the secondary one. The convective heat transfer from the helium gas to the reformer tube requires enhancement and the process working

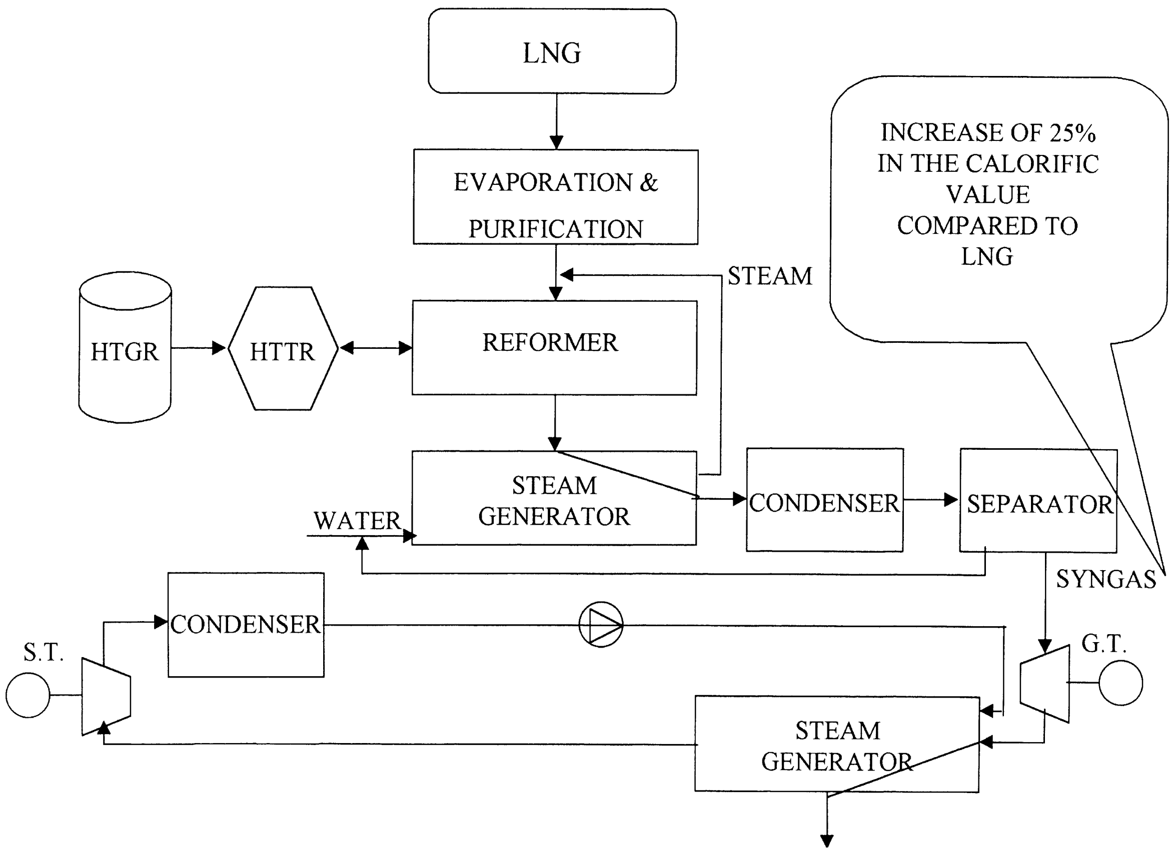

FIG. 1. HTGR heat for reforming of LNG and direct combustion of the SYNGAS in a combined cycle.

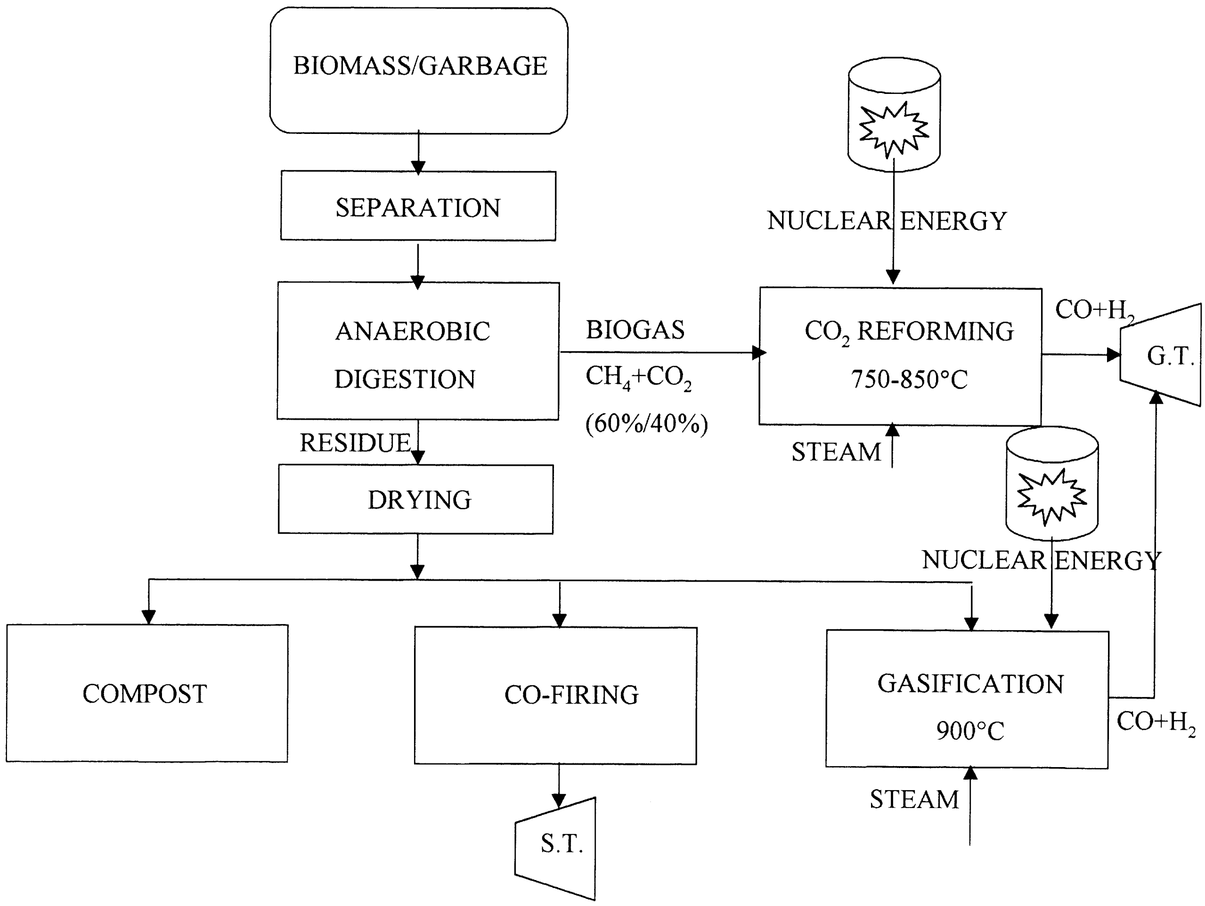

FIG. 2. Nuclear processing of biomass/garbage.

pressure in the reformer has to be slightly higher than the helium (usually 40 atm) for safety reasons (to avoid helium leaking out in the eventuality of failure in the reformer tube).

These requirements are in contradiction with the thermodynamics of the reforming process, where a lower pressure in the reformer and higher heat transfer characteristics are needed. The isothermal reformer offers a possible solution to these problems.

The isothermal reformer is based on the concept that the helium is used to evaporate tubes at a constant temperature (see Fig. 3). The intermediate medium can be molten salts, such as zinc chloride (which melts at 290°C and boils at 732°C) or tin chloride ($SnCl_2$) (which melts at 247°C and boils at 623°C), and liquid metals, such as sodium, cadmium and zinc. This concept improves the heat transfer substantially. The condensation temperature can be selected according to the process conditions. For instance, if 96% CH_4 conversion and 5 atm are selected, the condensation temperature can be 780–790°C. Figure 3 shows also two additional options: (i) heat pipes that can be utilized to heat internally the catalyst bed, and (ii) reformer tubes immersed in the boiling medium that is heated by the helium from the HTGR. The helium leaving the reformer is cooled further in a steam generator that produces steam for the reforming process and extra steam for generation of electricity or for operation of mechanical equipment. This configuration of the reformer increases the heat transfer characteristics while its operating pressure can be reduced, since there is no direct contact with the helium loop and the boiling medium serves as an additional safety barrier.

An additional safety feature provided by the boiling medium is in the case of process failure in the reformer plant, i.e., a problem in the feed supply. In this eventuality, the power level of the HTGR must be reduced or even completely interrupted. This is an undesirable

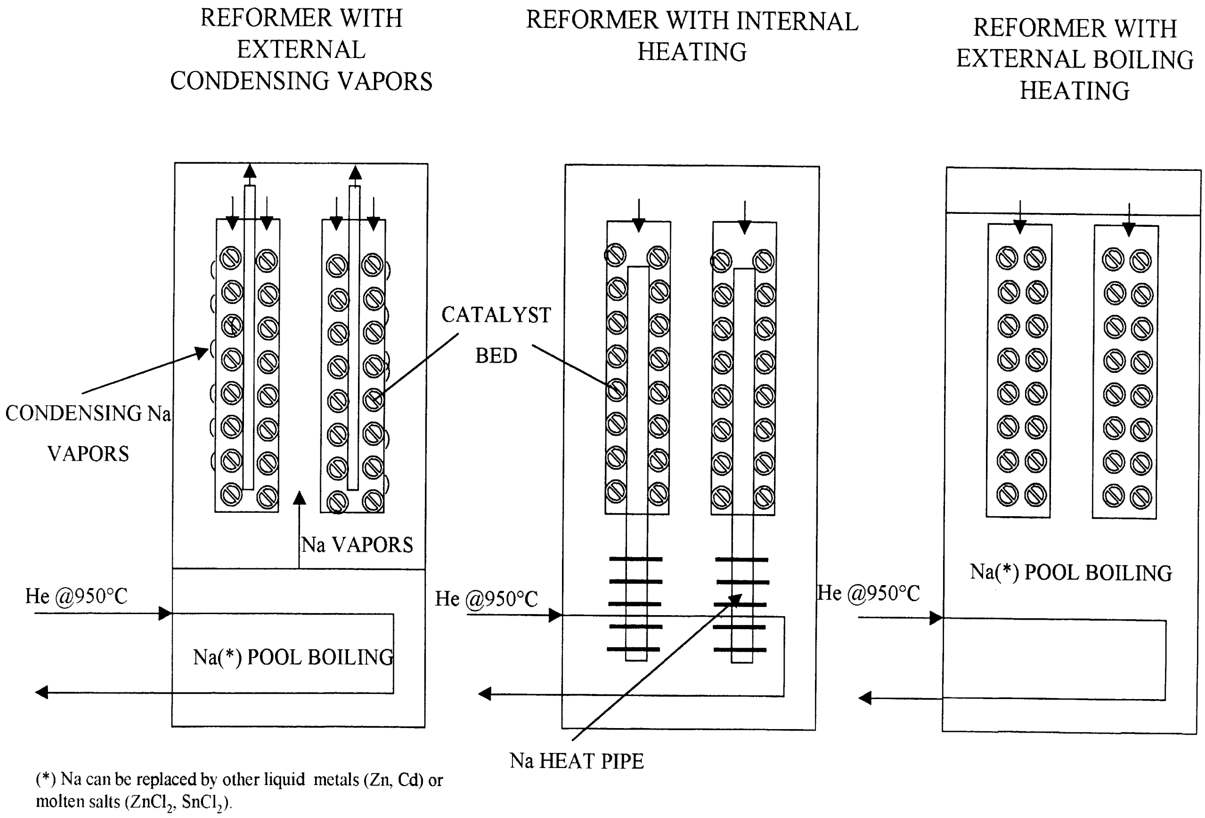

FIG. 3. Pseudo-isothermal reformers.

situation, particularly if the failure in the reformer plant is temporary. In this case, the vapors of the medium are diverted to an emergency condenser, as shown in Fig. 4. The access nuclear heat is rejected outside and the liquid medium is returned to the boiling enclosure by natural circulation. The liquid in the boiling space can be even subcooled in this situation, to enable a colder return of the helium to the HTGR.

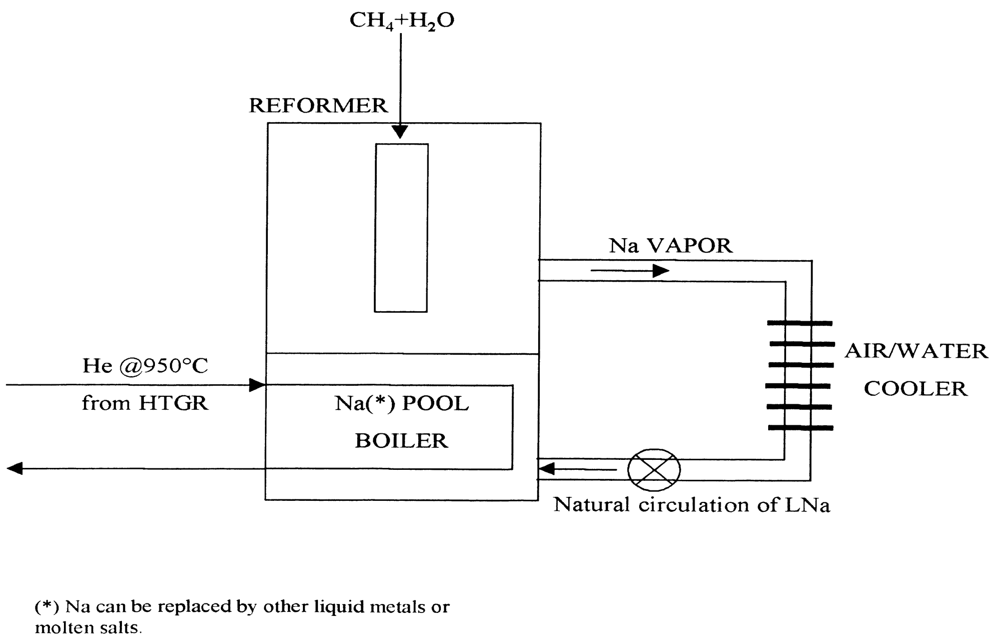

FIG. 4. Safety option for the isothermal reformer.

Finally, Fig. 5 shows an additional candidate concept for an isothermal reformer. In this figure the secondary helium loop and the He/He IHX are discarded. Instead, an intermediate medium boiler is introduced. This boiler is heated by the primary helium loop to evaporate the selected medium. The vapors are condensed on the reformer tubes, as previously described. Then, the liquid is further cooled to preheat the feed of the reformer and generate process steam, and finally pumped back to the evaporator. This configuration must be further analyzed from a nuclear safety viewpoint.

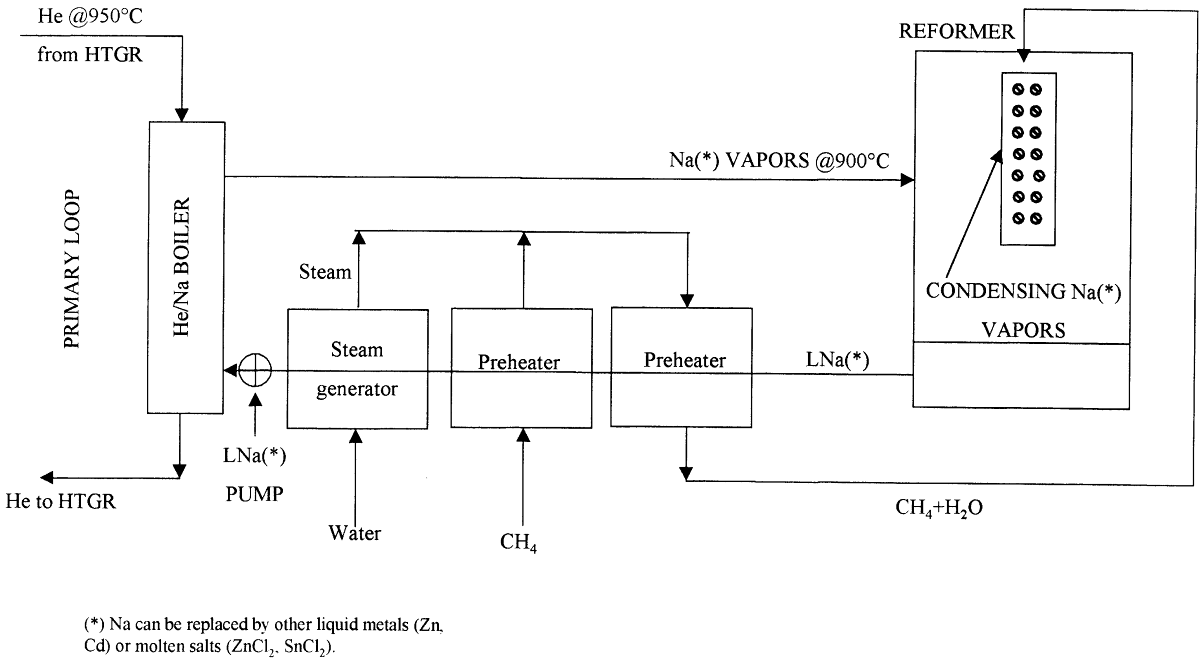

FIG. 5. Isothermal reformer with HTGR's primary He loop and liquid metal or molten salt secondary loop.

5. CONCLUSIONS

The HTGR as a heat source for the reforming process offers the opportunity to use nuclear heat not only for the electricity sector, but also for the production of fuels for transportation and chemicals for the industry. The HTGR also confers the possibility to process low-grade organic matters and convert them to clean fuel. It can process biomass and biogas, thus supplying fuel with potentially null CO_2 emission.

The nuclear reformers require modified and innovative solutions to achieve higher conversion and thermal efficiencies of the reforming process.

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POTENTIAL FOR POWER UTILIZATION OF WEAPONS GRADE PLUTONIUM IN GT-MHR NUCLEAR REACTOR

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Abstract

A new generation modular helium-cooled reactor (600MW(th)) is being designed by the Russian enterprises with participation of General Atomics (USA), Framatom (France) and Fuji Electric (Japan). The design pursues high conversion efficiency, higher safety potential and capability of disposing weapon-grade plutonium. The core outlet temperature is designed as 850°C. The paper summarizes other key design characteristics and provides illustrated structures of the reactor system.

1. INTRODUCTION

Certain success has been attained by now in various fields of power machine building industry (viz.: large gas turbine technology, EM bearings and ultra-compact HX) as well as in nuclear reactor technology (meeting the most strict safety requirements). It makes possible to originate development of modular helium-cooled reactor coupled with a gas turbine (GT-MHR), which ensures:

- Higher energy conversion efficiency (about 50%) than that in existing power reactor systems
- Cheaper electricity as compared to that generated by existing NPPs or conventional power plants
- Elimination of core melting event in any emergency situation
- Minimal thermal and radiological impacts to the environment

600MW(th) GT-MHR power unit is being designed by Russia's MinAtom enterprises and RRC KI together with General Atomics (USA), Framatom (France) and Fuji Electric (Japan).

2. BASIC CONCEPTUAL PRINCIPLES OF REACTOR PLANT FOR WEAPON-GRADE PLUTONIUM DISPOSITION

- GT-MHR is a new generation NPP with passively safe reactor, relied upon proven technologies. GT-MHR reactor basic design characteristics are introduced in Table1.

- GT-MHR completely meets weapon-grade plutonium disposition objectives due to, primarily, high burnup of initially loaded Pu (to 90%) and greater amount of electricity generated from unit of Pu mass as compared to other reactor types.

Table 1 GT-MHR reactor basic design characteristics

Parameter, unit	Value
Thermal capacity, MW	**600**
Power conversion efficiency, %	**up to 47**
Helium coolant temperature (inlet/outlet), ^{0}C	**490/850**
Helium pressure, MPa	**ab 7.0**
Core diameter (inner/outer), m	**2.64/4.84**
Core height, m	**8**
Fresh Pu load, kg	**ab 750**
Discharge Pu-239 burnup, %	**90**
Number of fuel cycles for core life	**3**
Annual Pu load, kg/yr	**250**

Spent fuel is completely meets the requirements for disposal in deep geological formations. Annual consumption of Pu in one GT-MHR module is about 250 kg.

- Beginning from 1994 MinAtom and General Atomics, and later on Framatom and Fuji Electric, joined together in cooperative program for development of GT-MHR. Nearest goal of GT-MHR international program is to develop preliminary and detailed designs and to conduct needed experimental work. Further goal is to construct a prototype plant in Seversk.

Conceptual Design 600 MWth GT-MHR power unit was completed by Russia's MinAtom enterprises and RRC KI together with General Atomics (USA), Framatom (France) and Fuji Electric (Japan):

- GT-MHR reactor assembly has modular configuration (Fig. 1). The reactor core is enclosed in a protective steel pressure vessel connected by cross duct to power conversion system (PCS) vessel. The vessels outer diameters are 8.4 m and 8.5 m respectively. Modules are located in underground pressure containment vessel

- GT-MHR uses Pu fuel in a form of small particles with multilayer coating (Fig. 2). Fuel blocks configuration is similar to that used in FSV reactor (USA). About 20 mill fuel particles are contained in a standard fuel block

- Annular-type core is composed of 1020 hexagonal prismatic fuel blocks stacked in 102 columns by 10 blocks each (Fig. 3). One third portion of fuel blocks is reloaded every year

- The core has negative temperature reactivity coefficient at any of operating temperatures

- Power conversion system realizing a closed gas turbine cycle is completely housed within PCS vessel. Turbomachinery consists of generator, gas turbine, two sections of compressor, fixed to a single vertical shaft supported onto EM bearings. PCS incorporates three compact heat exchangers, viz.: highly effective recuperator and water cooled precooler and intercooler. PCS flow diagram is depicted on Fig. 4

- HTGR technology gives a solution of the non-proliferation task through utilization of small fuel particles with multilayer coatings. This form of nuclear fuel is not suitable for military purposes pending its utilization for power generation.

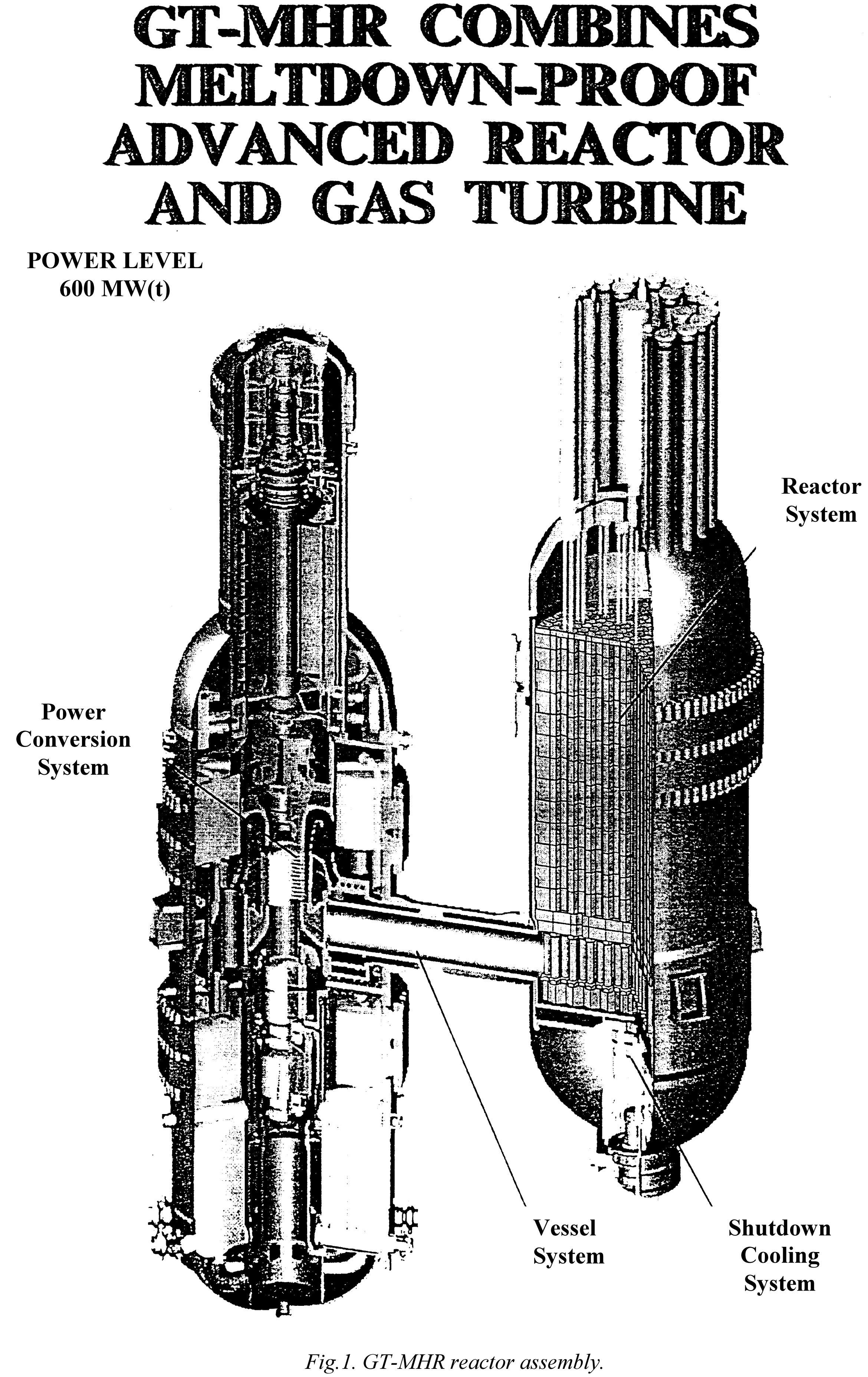

Fig.1. GT-MHR reactor assembly.

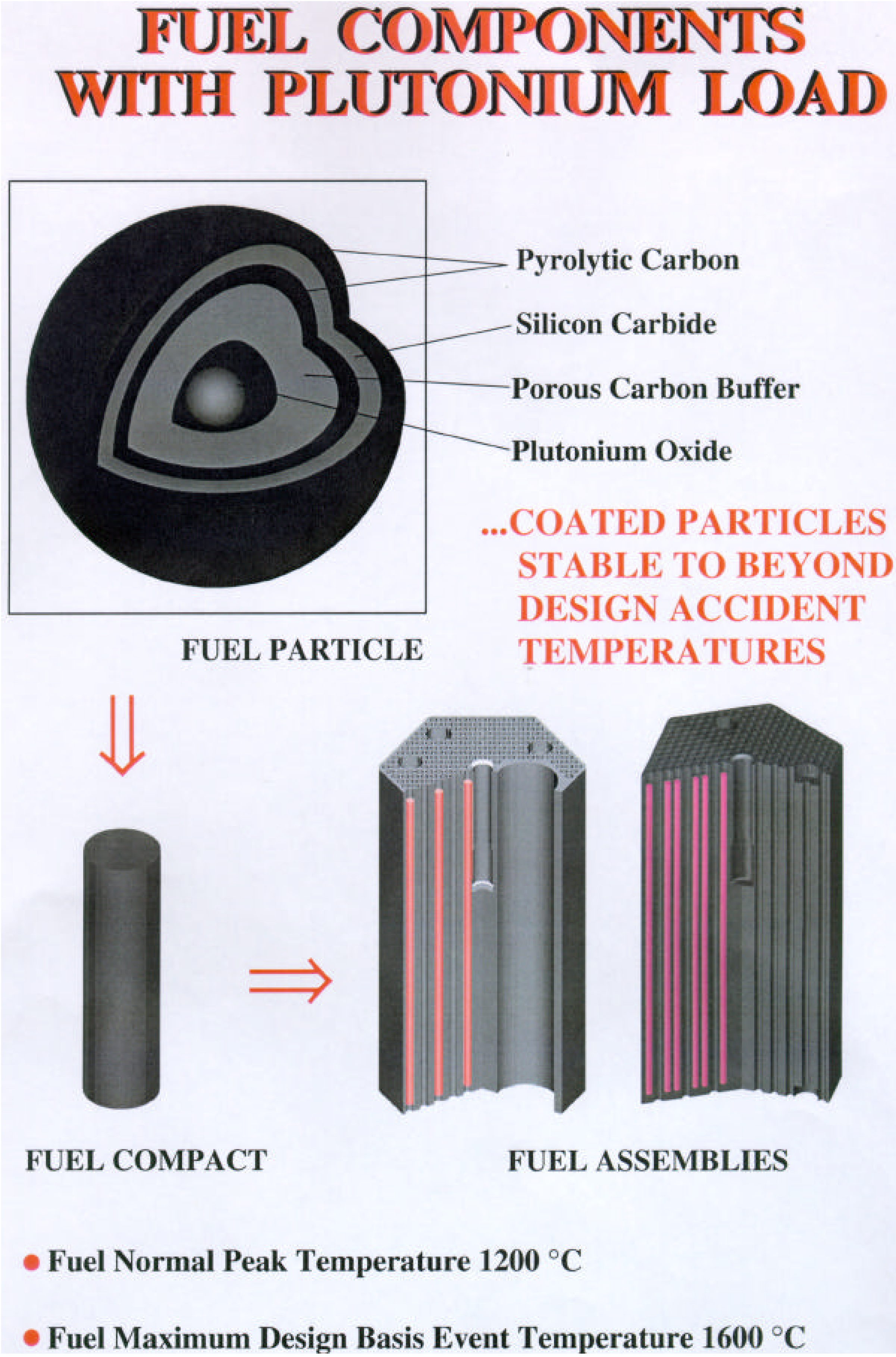

Fig. 2. GT-MHR fuel components.

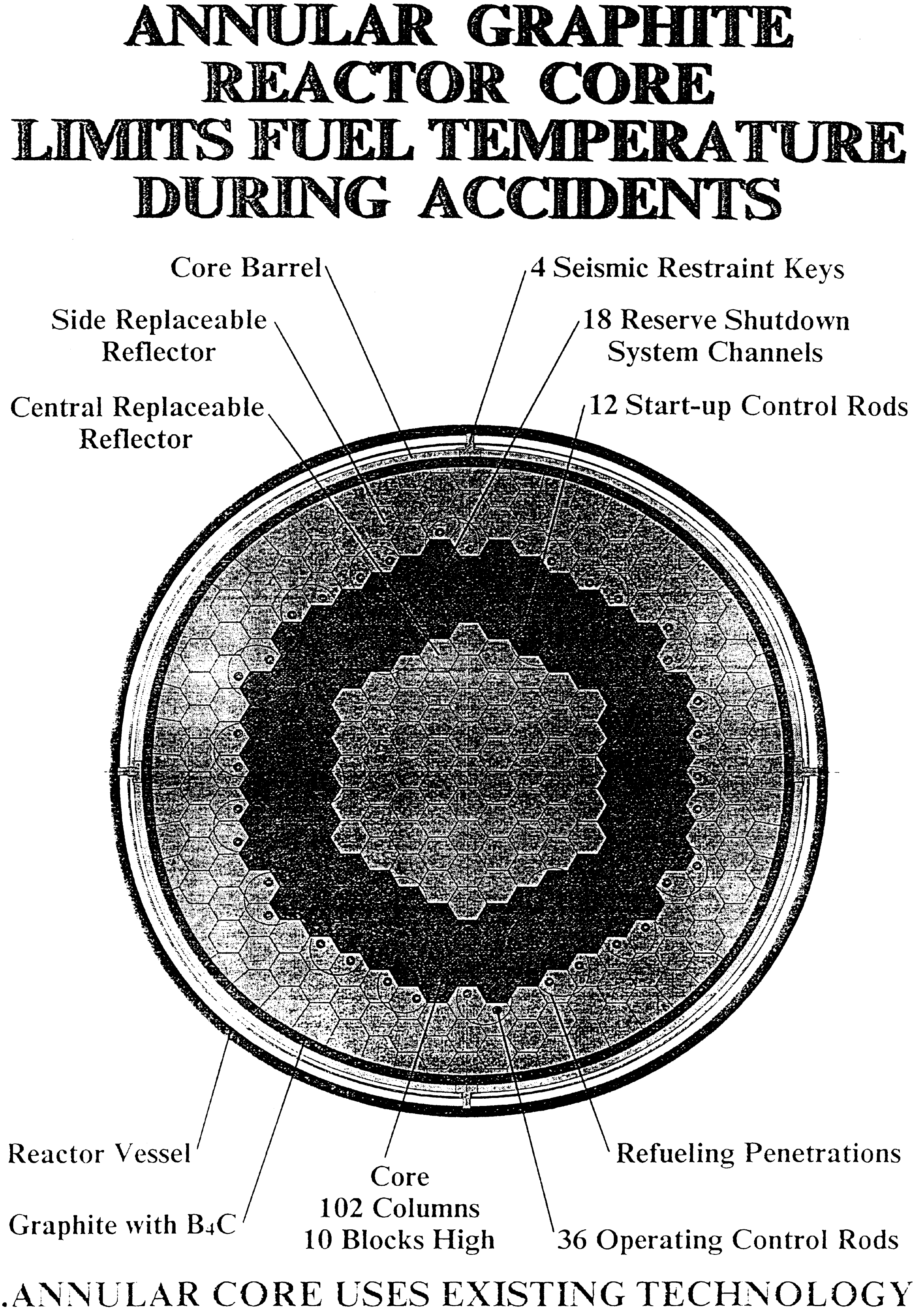

Fig. 3. GT-MHR fuel blocks.

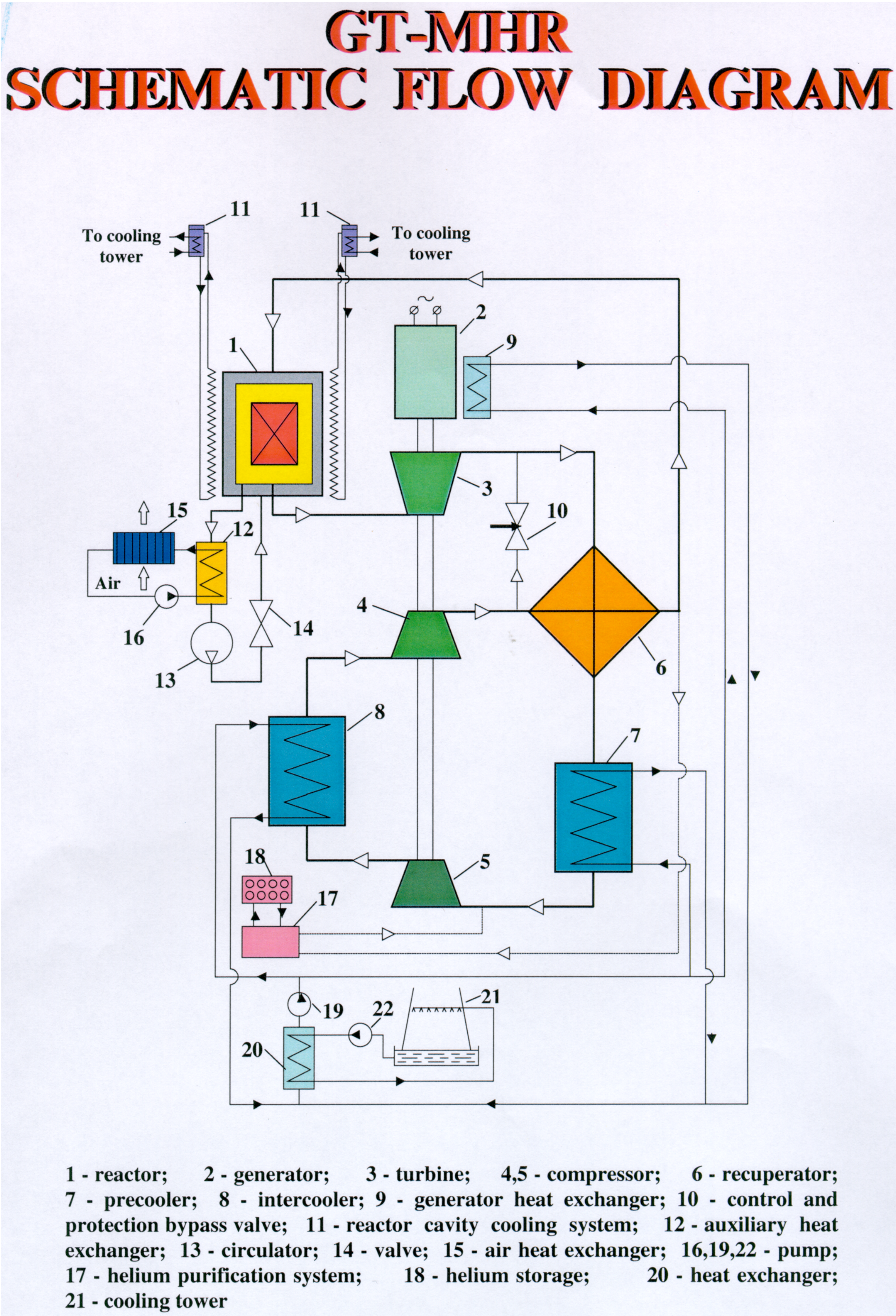

Fig 4. GT-MHR schematic flow diagram.

3. CONCLUSIONS

- GT-MHR reactor plant is capable of efficient utilizing of weapons-grade plutonium as fuel for generation of electricity and other purposes, providing high power conversion efficiency and minimal impact to the environment.
- GT-MHR construction in Seversk will allow existing environmental and social problems of SCC and region in general to be solved. The choice of a plant site in Seversk with developing infrastructure and experienced specialists is the weighty input in design realization.
- Implementation of the Programme substantially depends on international support by governments and private industrial firms.

SAFETY AND LICENSING ASPECTS

IAEA ACTIVITY RELATED TO SAFETY OF NUCLEAR DESALINATION

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Abstract

The nuclear plants for desalination to be built in the future will have to meet the standards of safety required for the best nuclear power plants currently in operation or being designed. The current safety approach, based on the achievement of the fundamental safety functions and defence in depth strategy, has been shown to be a sound foundation for the safety and protection of public health, and gives the plant the capability of dealing with a large variety of sequences, even beyond the design basis. The Department of Nuclear Safety of the IAEA is involved in many activities, the most important of which are to establish safety standards, and to provide various safety services and technical knowledge in many Technical Co-operation assistance projects. The department is also involved in other safety areas, notably in the field of future reactors. The IAEA is carrying out a project on the safety of new generation reactors, including those used for desalination, with the objective of fostering an exchange of information on safety approaches, promoting harmonization among Member States and contributing towards the development and revision of safety standards and guidelines for nuclear power plant design. The safety, regulatory and environmental concerns in nuclear powered desalination are those related directly to nuclear power plants, with due consideration given to the coupling process. The protection of product water against radioactive contamination must be ensured. An effective infrastructure, including appropriate training, a legal framework and regulatory regime, is a prerequisite to considering use of nuclear power for desalination plants, also in those countries with limited industrial infrastructures and little experience in nuclear technology or safety.

1. INTRODUCTION

The general approach to safety of nuclear reactors supplying heat or electrical power to desalination plants is equivalent to the approach used for nuclear power plants producing of electricity. The nuclear plants for desalination to be built in the future will have to meet the standards of safety required for the best nuclear power plants currently in operation or being designed, and for this reason the safety aspects are common with those related to new generation reactors for which a dedicated programme exists at the IAEA. Most of the general safety considerations reported in this paper have been discussed and analysed during the development of this programme. Some specific characteristics of desalination plants such as siting and coupling which require particular consideration from a safety point of view, and further safety studies will be needed when the type and size of the reactor are determined.

2. GENERAL SAFETY ASPECTS OF NUCLEAR POWER PLANTS

There are three safety objectives from which all safety principles and requirements are derived:

General Nuclear Safety Objective: *To protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards.*

Radiation Protection Objective: *To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept* *below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.*

Technical Safety Objective: *To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.*

The safety objectives shall be achieved through the application of the defence in depth strategy that will continue to be the overriding approach for ensuring the safety of workers and the public, and for protecting the environment. This strategy is effective in compensating for human and equipment failures, both potential and actual. The concept is based on several levels of protection, including successive barriers that prevent the release of radioactive material to the environment. However, its efficacy depends on rigorous implementation.

Levels of Defence in Depth (From INSAG-10)

Levels of defence	Objective	Essential means
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features
Level 3	Control of accidents within the design basis	Engineered safety features and accident procedures
Level 4	Control of severe plant conditions including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

This implies a determined effort to make the defence effective at each level, particularly for accident prevention and accident mitigation. There is not a unique way to implement defence in depth, since there are different designs, different safety requirements in different countries, different technical solutions and varying management or cultural approaches. Nevertheless, the strategy represents the best general framework to achieve safety for nuclear power plants and, thus, nuclear powered desalination plants. In general, strong implementation of defence in depth requires a determined and constant effort from the design phase, to construction and operation in order to provide graded protection against a wide variety of transients, abnormal occurrences and accidents, including human error and equipment failures within the plant, and events initiated outside the plant.

2.1. Design basis approach and severe accident treatment

Operating nuclear plants are largely designed according to the design basis accidents approach. This means that the plant is deterministically designed against a set of hypothetical accident situations according to well established design criteria in order to meet the radiological targets. The current design basis approach has been shown to be a sound foundation for the safety and protection of public health, in part because of its broad scope of accident sequence considerations, and because of its many conservative assumptions which have the effect of introducing highly conservative margins into the design that, in reality, give the plant the capability of dealing with a large variety of sequences, even beyond the design basis.

The deterministic approach is complemented by probabilistic evaluations with the main purpose of verifying that the design is well balanced and there are not weak areas or systems which could allow for the possibility of risky sequences. Often, probabilistic targets for core damage frequency and for containment performance are established. Experience and analysis have shown, however, that some sequences beyond the design basis (i.e. severe accidents) may need to be considered explicitly in the design, providing it with additional safety features to further prevent and mitigate such severe sequences. In this regard probabilistic safety assessment is recognized as a very efficient tool for identifying those sequences and plant vulnerabilities that require specific design features (elimination by design of the most challenging sequences to the containment). This, together with an effective containment system including good control of potential containment by-pass, ensure minimum radiological impact, with an extremely small chance of any off-site radioactive releases. For a nuclear powered desalination plant, the design basis may need to also include some transients or abnormal occurrences that might originate in the desalination unit itself.

3. SPECIFIC SAFETY ASPECTS OF DESALINATION PLANTS

The total power (electrical and thermal to supply potable water to a medium sized town) varies from a few to several hundred megawatts, and thus any proposed reactor falls into the small or medium sized category. Larger sizes would be required for the combined production of water and electrical power.

The nuclear power plants used for water desalination have several characteristics that are similar to those power plants used for district heating reactors (e.g. siting, power size, possibility of combined production), and the experience gained with these plants should be considered in designing nuclear powered desalination plants.

3.1. Coupling

The overall safety of an integrated complex composed of a nuclear reactor plant coupled to a desalination plant is predominantly dependent on the safety of the nuclear reactor plant and the effect of coupling, or rather the interaction between the desalination plant and the nuclear plant. This interaction should be analysed in various coupling situations to assess its effect on the safety of the reactor and on the overall nuclear desalination system, either in normal operation or in an accident situation.

Coupling will not pose any new safety concern if desalination uses only electrical power.

In thermal processes, the energy to be supplied is mainly low temperature process steam or water. Coupling is accomplished via a heat transfer circuit. Since radioactivity exists in the primary steam or hot water, the risk of contamination of product water exists and must be avoided. This can be done by adding intermediate loops maintained at values of pressure such that any leakage would not produce transfer of contamination to the distributed water. These simple measures, together with appropriate instrumentation and monitoring should be effective in preventing contamination of the distributed water. They do not seem to present any particular technical difficulty.

All the information available from the operating experience accumulated on an existing plant (Aktau, Kazakstan) and from conventional desalination plants will also provide a valuable source of information for design and operation purposes. Operational transients in a desalination plant would have direct feedback into the reactor system. Such transients could have safety implications and need to be assessed.

3.2. Siting

For obvious reasons, the siting of a nuclear powered desalination plant raises some safety concerns, mainly because of the site selection restraints. The plant has to be built on a coastal site and near to populated areas to limit the cost of potable water distribution. The choice of site raises problems related to oceanography (tides, plant elevation) and very often to seismicity (frequent presence of faults on coasts).

The proximity of the nuclear desalination complex to population centres and its implication on the design and to the emergency planning and water supply should be examined.

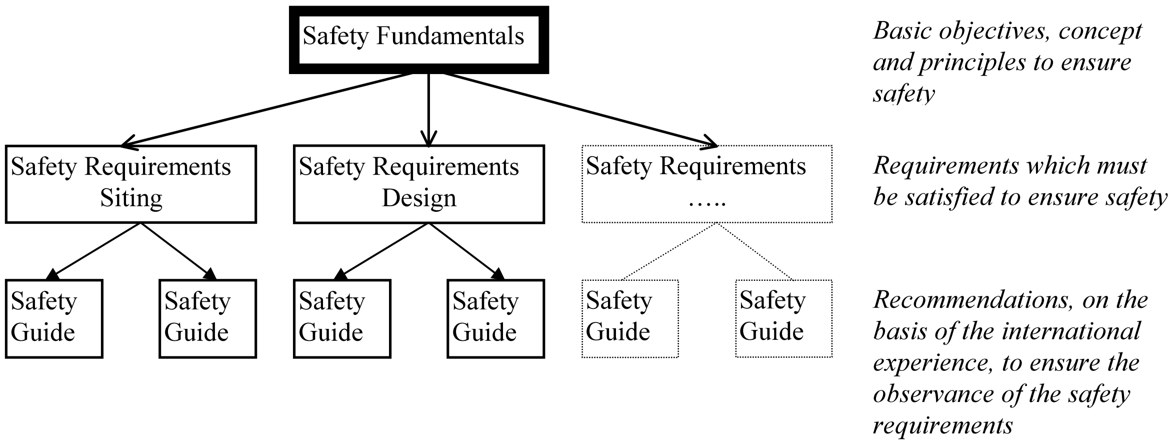
If the site is in a remote area an important aspect to consider is the availability of adequate external electric power grid or supply for safe operation of the nuclear plant.

4. THE ROLE AND ACTIVITIES OF THE IAEA

The Department of Nuclear Safety is involved in many activities, the most important of which are to establish safety standards, and to provide various safety services and technical knowledge in many Technical Co-operation assistance projects. The department is also involved in other safety areas, notably in the field of future reactors. The newly established Convention on Nuclear Safety was developed under the auspices of the IAEA.

The Agency produces many documents related to nuclear safety, the most important of which are those now to be included in the Safety Standards Series (SSS), formerly the Safety Series, which included the NUSS programme. The SSS will comprise three levels: Fundamentals, Requirements and Guides.

Hierarchy of the IAEA Safety Standards Series



4.1. Safety Fundamentals (SFs)

Currently, there are three SF documents, but in the long term aim is to combine these into a single document. These are the first documents in the hierarchy; they present basic objectives, concepts and principles to ensure safety in the development and application of atomic energy or radioactive material for peaceful purposes. The SF documents constitute the reasons why activities must fulfil certain requirements; they do not state what these requirements are, they are self-sufficient and do not include a list of references. In the SF on Safety of Nuclear Installations (SS-110) there are 25 fundamental principles grouped into four main areas, related to the Legislative and Regulatory Framework,the Management of Safety, the Technical Aspects of Safety and the Verification of Safety.

4.2. Safety Requirements (SRs) and Safety Guides (SGs)

Supporting the SFs are Requirements (formerly termed Codes, Standards or Regulations). In the nuclear safety area there will be six main areas: Governmental Organizations, Siting, Design and Operation of thermal neutron nuclear power plants, Quality Assurance and the Research Reactor Series which has two SR documents. All the existing NUSS codes (except QA, which was published in October 1996) are now subject to a comprehensive revision process, which is being overseen by the Nuclear Safety Standard Safety Committee (NUSSAC). This revision will ensure that all the relevant principles in the SF are systematically addressed, thus enabling a coherent set of documents to be produced. The SRs will set out in more detail what is required of Member States to ensure safety in a particular area, and they are governed by the content of the SFs. SRs do not generally present recommendations on or explanations of how to meet the requirements. This more detailed aspect is covered by the third level in the hierarchy, namely, the Safety Guides. The SGs present recommendations on the basis of international experience, of the measures to be followed to meet the requirements set out in the SR documents.

4.3. The IAEA Safety Standards for the Design

Table 1 shows as an example the existing (left side) and the future (right side) structures of part of the Safety Standards for the Design. Some publications will be merged and some, as the Safety Guide on Safety Assessment, will be prepared ex-novo.

The whole process for the revision of the Safety Standards for the Design, comprises the revision of the Code and 15 Safety Guides and the preparation of two additional new Safety Guides. The process will be concluded by the year 2001.

4.4 Specific requirements applicable to nuclear desalination and other heat utilization applications

As previously mentioned all of the current safety standards (NUSS and standards for research reactor) are being subjected to a comprehensive review and revision process. Since the safety requirements applicable to nuclear desalination, district heating and other heat utilization applications will, in general, be those applicable to nuclear power plants, it should be possible during this revision process to incorporate within the revised documents any new or unique requirements specific to these systems. For example the Requirements for the Design which is intended to replace the existing “Code on the Safety of Nuclear Power Plants, Design (50-C-D Rev. 1)” will incorporates the following new requirement:

> **Power plants used for co-generation, heat generation or desalination.**
> *Nuclear power plants coupled with heat utilization units (e.g. district heating) and/or water desalination units shall be designed to prevent transport of radioactivity from the nuclear plant to the desalination or district heating unit during any condition of normal operation including anticipated operational occurrences, design basis accidents and selected severe accidents.*

The above requirement represents the only one that was specifically added for nuclear power plants used for desalination, heat production or desalination. This underlines once more that there are no major differences, from a safety point of view, from these nuclear plants and those used for electricity generation only.

TABLE 1. EXAMPLE OF EXISTING AND FUTURE STRUCTURES OF THE SAFETY STANDARDS FOR THE DESIGN

•50-C-D (Rev. 1) Code on the Safety of Nuclear Power Plants: Design	**•Requirements for the safety of nuclear power plants: Design**
•50-SG-D1 Safety Functions and component classification for BWR, PWR and PTR	
•50-SG-D2 Fire protection in Nuclear power plants	**•Fire protection in Nuclear power plants**
•50-SG-D4 Protection against Internally Generated Missiles and their Secondary Effects in Nuclear Power Plants	**•Protection against Internally Generated Missiles and their Secondary Effects in Nuclear Power Plants**
•50-SG-D5 (Rev. 1) External Man-induced Events in relation to Nuclear Power Plant Design	**•External Man-induced Events in relation to Nuclear Power Plant Design**
•50-SG-D9 Design Aspects of Radiation Protection for Nuclear Power Plants	**•Design Aspects of Radiation Protection for Nuclear Power Plants**
•50-SG-D11 General Design Safety Principles for Nuclear Power Plants	
•50-SG-D15 Sysmic design and qualification for nuclear power plants	**•Sysmic design and qualification for nuclear power plants**
	•<u>Safety Assessment and Verification</u>
CODE + 7 GENERAL SAFETY GUIDES	**REQUIREMENTS + 6 GENERAL SAFETY GUIDES**

Figure 1 shows how the concept of the Safety Assessment has been included in the revised Safety Standards for the Design. The Safety Assessment is a tool for the design and represents a relevant part of the design process since the very initial stage. The same methods (deterministic, probabilistic) for the Safety Assessment can be used by the Designers, Reviewers and Regulatory Bodies.

4.5. Current experience accumulated on research reactors

Nuclear desalination plants have been proposed for various Member States, in particular, those that are located in arid areas of Africa, Asia and elsewhere. Many of these countries have no experience at all with nuclear reactors, while a few have one or more research reactors.

Reviewing the experience gained with research reactors in several developing countries the following points can be made that may be applicable to a desalination project:

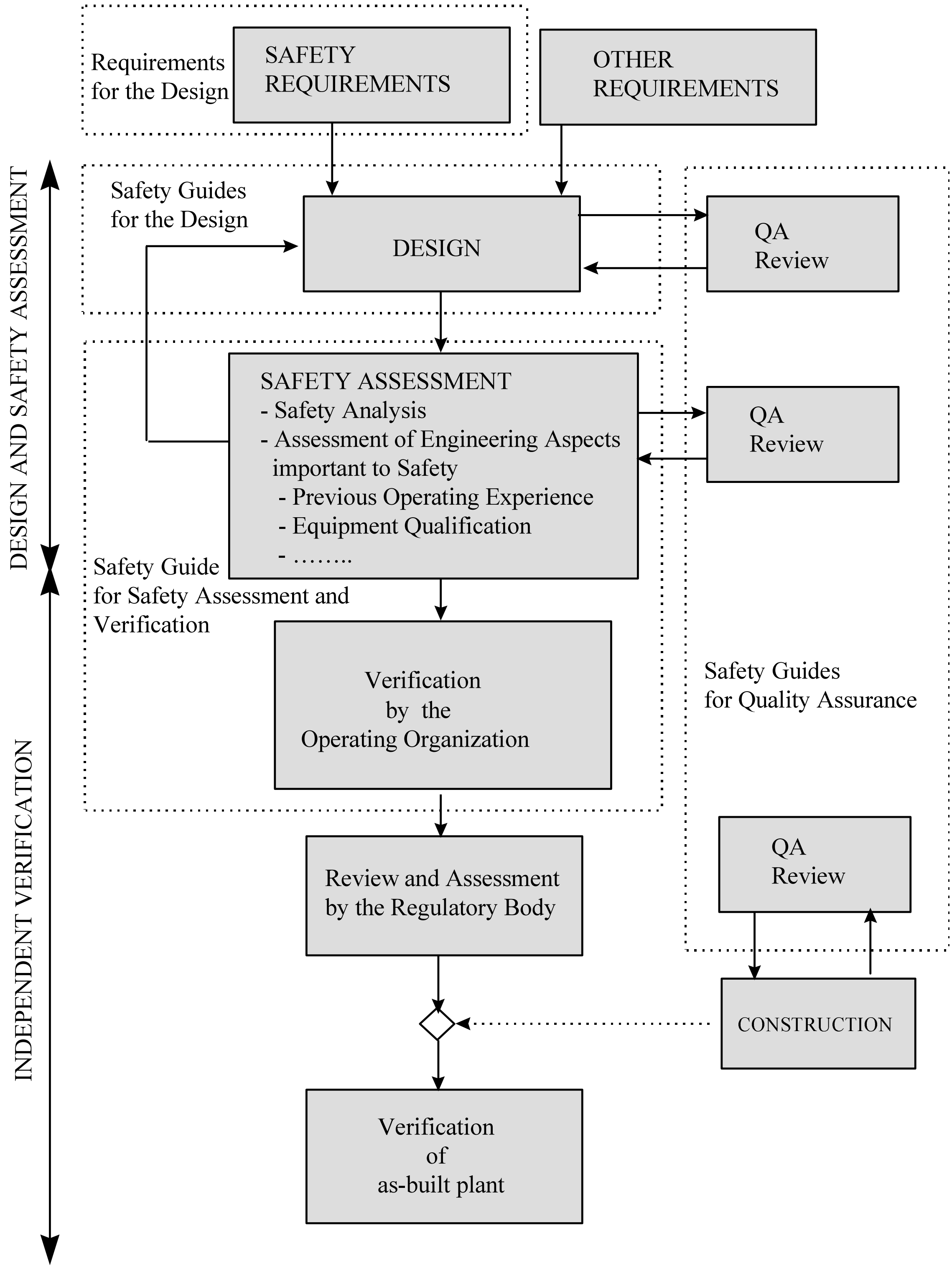


FIG. 1. *Areas covered by the IAEA Safety Standards for the Design of NPPs*

(1) Experience with a research reactor facility may be quite useful as it usually means that the country already has: a nucleus of a regulatory authority; some infrastructure in radiation protection and waste effluent control related to nuclear reactors; group of knowledgeable personnel in the areas of reactor operations and maintenance; programmes for the training of personnel; and experience with Agency sponsored projects.

(2) A research reactor facility (especially a larger reactor) can be used to simulate or experiment with some of the processes associated with a desalination plant, and can also be used as a school for training the new staff needed for the new project.
(3) Developing countries vary greatly in their political stability, economic wealth, technological infrastructure, logistical infrastructure, and general technical and safety related attitudes.

While gaining experience with a research reactor is expected, in general, to be useful as a first step before introducing nuclear power (or desalination), this same experience can shed light on the deficiencies that may undermine the prospects for such a project unless, in particularly serious cases, adequate international support can be provided.

4.6. IAEA activities on new generation nuclear power plants that provided input to the revision of the safety standards for the design

The IAEA activities on the safety of new generation reactors, which were formally initiated after the Conference on the Safety of Nuclear Power: Strategy for the Future held in September 1991, are being carried out under the project Safety Approaches to the New Generation of Nuclear Power Plants. The main objective of this project is to foster an exchange of information on safety approaches to new generation nuclear power plants with a view to promoting harmonization among Member States and contributing to the development and revision of safety standards and guidelines for nuclear power plant design. It is expected that the new standards will have an impact on the design of all nuclear power plants, including those for desalination, constructed in the coming years.

In June 1995, following INSAG's review and comments, the Agency published a technical document, Development of Safety Principles for the Design of Future Nuclear Power Plants (IAEA-TECDOC-801). This documents provided a basis for the development of safety objectives and principles for new generation nuclear power plants and for the revision of safety standards. The key proposal is that severe accidents beyond the existing design basis will be systematically considered and some of them explicitly addressed during the design process for future reactors. The document also emphasizes the need to further lower the risk of any serious radiological consequences and to ensure that the potential need for prompt off-site protective actions can be reduced or even eliminated (good neighbour concept).

Additional effort has been made to prepare a technical document on the implementation of defence in depth for new generation Nuclear Power Plants. The work was based on the report on defence in depth prepared by INSAG, and the main objective was to bring together the relevant aspects of existing publications on both defence in depth and future reactor designs, and then to apply recent defence in depth formulations specifically to ongoing developments in future plant designs.

Particular attention has been focused on identifying and addressing those factors that have the potential to affect multiple levels of defence in depth. This provides high confidence that appropriate actions will be taken to ensure the effectiveness of the defence in depth concept against failures that have the potential to impact multiple levels of defence in depth. (Human failure, internal and external hazards, etc.).

The report provides a good general framework for a safety evaluation and also gives some indication as to how the defence of each level could be enhanced.

4.7. Current activity on safety aspects of nuclear desalination

The activity is being carried out in accordance to the resolution of the General Conference that in the "Plan for producing potable water economically" of September 1998 stated:

The General Conference urges the Director General to continue the Agency's work regarding the safety aspects of desalination using nuclear energy.

The work on safety is being carried out in very tight co-ordination with the work on technological aspects of the Department of Nuclear Energy. The main task consists of preparing a technical document on safety aspects of nuclear power plants coupled with sea water desalination and/or other heat utilization units. In this document, now in draft form, the main safety and licensing aspects and issues are identified and addressed in detail and they are briefly mentioned below.

- Coupling of the reactor with the desalination unit
  - Single and multi-purpose plants
  - Various coupling situations
    - Thermal and/or electrical
  - Potential of transfer of contamination from the nuclear plant to the potable water
    - Scenarios to be addressed
    - Addition of intermediate loop as preventive measure
  - Sharing of resources between the NPP and the desalination unit (intake and out-fall structures)
  - Brine discharge (environmental issues)
- Operational transients and accidents
  - They have to be considered in the safety analysis but do not seem to pose particular safety concerns
- Water quality and monitoring
  - Radioactivity content shall meet the national and international standards

A continuos monitoring may be difficult because of the very low levels of radioactivity and a batch monitoring will be necessary

- Availability of product water
  - High availability target will require water storage and energy backup supply
- Siting
  - Proximity to population centres will result in no need for planned evacuation
- Licensing
  - Need of an effective legislative framework for the regulation of nuclear facility
  - Need of independent Regulatory Body
  - Need of well developed safety culture

5 CONCLUSIONS

The safety, regulatory and environmental concerns in nuclear powered desalination are those related directly to nuclear power plants, with due consideration given to the coupling process.

It is expected that any reactors used for desalination purposes will be designed, constructed and operated in accordance with internationally recognized safety standards.

IAEA missions to operating nuclear power plants coupled to heat production and desalination plants have not revealed any serious specific safety concerns related to the interaction of the nuclear plant with the heat distribution plant or desalination plant, but they have shown that any safety concerns are related to the reactor itself.

Nuclear safety and environmental considerations in nuclear desalination are those arising from the use of nuclear reactors as energy sources.

Nuclear safety and regulatory actions should be based on relevant IAEA safety standards.

The most serious concern, as experience with research reactors has shown, arises from the fact that very often countries that need water are developing countries, with limited industrial infrastructures and little experience in nuclear technology or safety.

An effective infrastructure including appropriate training, a legal framework and a regulatory regime, is a prerequisite to considering use of nuclear power for desalination plants.

Another relevant aspect is the social and political instability of some countries where nuclear facilities could be possible targets of external attack; the plant would require comprehensive physical protection arrangements.

With respect to existing international safety standards and guides, they also seem to be appropriate covering desalination plants. There seems to be no need to prepare any specific guidance for the safety of nuclear powered desalination plants.

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THE CURRENT CEA/DRN SAFETY APPROACH FOR THE DESIGN AND THE ASSESSEMENT OF NON-ELECTRICAL APPLICATIONS OF NUCLEAR HEAT

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Abstract

This paper presents the basis of the safety approach currently implemented by the Commissariat à l'Énergie Atomique — Nuclear Reactor Directorate (CEA/DRN), both for the design and the assessment of innovative systems and future nuclear installations. It is considered that the described approach is applicable to the plants build for non-electrical applications of nuclear heat. This is typically the case of Nuclear Desalination Installations. This approach is the result of the experience maturated, within the context of the CEA/DRN Innovative Programme, through practical applications over several future concepts (both fission and fusion plants). The background of this experience is structured coherently with the European Safety Authorities recommendations, the European Utilities Requirements (EUR) and the "fundamental safety objectives" defined by the IAEA. The Defence In Depth principle and its application, by means, among others, of the barrier concept, remains the basis of the safety design process of future nuclear installations. Its adequacy is checked through the safety assessment. The methodology for Lines of Defence (LOD) implementation as well as the one for the LOD architecture assessment is shown and motivated. The document shows that the clear and unambiguous definition of the safety approach provides an essential base for the organisation of the design tasks, being sure that the safety aspects are correctly taken into account and implemented, and for an adequate safety assessment of the final design, both from qualitative point of view as well as for the quantitative safety analysis.

1 - INTRODUCTION AND PURPOSE OF THE PAPER

The purpose of this paper is to make general considerations about the top-tier safety related requirements implemented by the CEA/DRN for future nuclear installations design and/or assessment. It is considered that these requirements are applicable to the plants implemented for non-electrical applications of nuclear heat. This is typically the case of Nuclear Desalination Installations.

The approach content is the result of the experience maturated, within the context of the CEA/DRN innovative programme, with practical applications over several future concepts both for fission reactors ("power" and "research") and fusion plants. The details of this approach are still under discussion at the CEA for formal acceptance.

Hereafter, two main items are discussed:

⇨ the safety approach for the design and operation with, in particular, the adoption of the Defence In Depth principle and the rationale for the definition of the safety objectives ;

⇨ the approach for safety assessment to verify that the Defence In Depth principle is correctly implemented and that the safety objectives are met.

It is considered that the clear definition of the content of the first item can provide **an essential base for the organisation of the design tasks**, being sure that the safety aspects are correctly taken into account and implemented. Concerning the safety objectives, i.e. the allowable radiological consequences, the harmonisation is suggested between the different types of nuclear installations.

The second item is fundamental to organise **an adequate safety assessment of the final design**, both from qualitative point of view (e.g. the compliance of the " safety related architecture " or the correct implementation of human factor related requirements) as well as for the quantitative safety analysis (e.g. the definition of the rules for the analysis). Within this frame, considering the complexity of the plants " safety related architecture ", the need for the coherency with the defence in depth concept and with the probabilistic objectives currently adopted, leads to suggest the notion of **the Lines of Defence (LOD)** shortly developed below.

2 - THE “ BASIC NUCLEAR INSTALLATIONS ”

The French formal definition for the “ Basic Nuclear Installations ” (BNIs ; i.e. nuclear plants and facilities) can be summarised as follows :

“One Installation is classified as Basic Nuclear Installation (Installation Nucléaire de Base - INB in French) due to its potential for a “ nuclear risk ” : quantities of fissile material, of fission products, of radioactive elements, of radiation beams. In these conditions the “ nuclear risk ” becomes the main guideline for the risk analysis. Within the BNI perimeter the safety analysis shall then take into account all the failures and all the hazards that can interact with the radioactive materials” (IGSN-CEA, 1998).

The nuclear plants integrated into the nuclear desalination installations, clearly enter within the frame of this definition.

3 - THE CEA/DRN POSITION ON THE SAFETY REQUIREMENTS AND OBJECTIVES FOR FUTURE INSTALLATIONS BOTH FOR ELECTRICAL AND NON ELECTRICAL APPLICATIONS OF NUCLEAR HEAT

One of the major missions of the CEA/DRN is the co-ordination and the partial achievement of the CEA Innovative Programme. Among the corresponding tasks, the CEA/DRN shall :

- define the safety requirements, the safety objectives and the methodology for the assessment of the innovative reactors and/or specific nuclear installations (e.g. : fission reactors and fusion plants, etc.) ;
- achieve the safety analysis for innovative reactors and/or specific nuclear installations (e.g. desalination plants).

These safety-related missions are conducted in a context characterised by a large spectrum of “ boundary conditions ” :

- several innovative concepts are considered simultaneously : PWR (low pressure PWR, Supercritical Reactor), BWR (ESBWR), Small and Medium Reactors (SMR - modular integral concepts), HTGR, FBR (liquid metal or gas cooled), experimental reactor (Reactor Jules Horovitz - RJH), fusion reactors (ITER, DEMO) ; Hybrids systems, for Partitioning and Transmutation, are also considered (i.e. the Accelerator Driven Systems)
- several energy utilisation are considered : electricity, cogeneration, district heating, desalination, etc. ;
- all these items are characterised by a great diversity in term of deepness for the possible/needed degree of assessment (e.g. as a function of the available documentation).

Within this context, for the CEA/DRN, it is essential to elaborate a coherent and unambiguous approach applicable to all the above concepts and integrating all the evoked conditions. The one implemented by the CEA/DRN is founded on **three essential principles** :

- The **safety objectives of the BNIs should be the same** in terms of doses applicable respectively to the operators, the public and the environment (i.e. the radiological consequences) and that for all the plant conditions : normal, incidental and accidental. The adoption of ALARA shall be considered.
- It is considered that, for all the concerned nuclear installations, the implementation of an adequate safety related architecture and the organisation of the safety assessment should be achieved using analogous and comparable approaches based on **the integral adoption of the defence in depth** (i.e. all the levels should be considered).
- These approaches should be **able to integrate the peculiar characteristics of each installation**. This will be made in coherency with the defence in depth concept : the number and the quality of the required “ defences ” are function of the potential internal and external hazards, and consequences of failures.

4 - GENERAL NUCLEAR SAFETY APPROACH FOR DESIGN AND OPERATION : APPLICATION OF THE DEFENCE IN DEPTH PRINCIPLE.

The **Defence In Depth concept** (as defined by INSAG 75-3) and its application, by means, among others, of the **barrier principle**, remains the basis of the future plants safety design process. Defence In Depth shall be applied both to design and operation. Its appropriateness is checked through the **safety assessment**.

a) Defence In Depth

As discussed above, the levels of Defence In Depth should be systematically applied :

1st level : **Prevention** : Quality in design and achievement, prevention of nonconformity.

2nd level : **Surveillance, detection and control :** Quality of operation, keeping the facility within authorised limits.

3rd level : **Safety systems and Protection systems design :** Postulate of all plausible incidents and accidents and implementation of means to limit the effects of these accidents to acceptable levels.

4th level : **Accident management and containment protection, limitation of consequences :** Prevention of deterioration of accidental conditions and limitation of severe accident consequences.

5th level : **Response outside the site :** Limitation of radiological consequences for populations in case of significant releases.

b) The barrier principle

The number of barriers depends on the risk, and aims at the implementation of a " progressive " defence architecture[1]. More particularly, it is important to render improbable the events which are likely to affect more than one barrier at a time. Besides, the means allowing the operator to **know constantly the state of the barriers** shall be implemented. Finally, it is stressed that the correct implementation of the different levels of defence in depth should aim at guarantee the **defence homogeneity**[2] **and exhaustiveness**[3]**.**

Nota bene : *These two objectives as well as the principle for a "progressive defence architecture" are essential to achieve an acceptable safety architecture*

Examples for the design options selection are given in Appendix 1.

5 - GENERAL APPROACH FOR THE NUCLEAR SAFETY IMPLEMENTATION

5.1 - Design Basis Conditions

The conventional **Design Basis Conditions** (DBCs) are characterised by a **Postulated Initiating Event** (PIE) which occurs when the facility is in a given **initial state**[4].

The PIEs should be classified by category, based on their estimated occurrence frequency (f: frequency per reactor and per year). By analogy with the PIEs, a similar categorisation is chosen for the DBCs. The plant safety design and assessment is based on the analysis of a set of envelope representative DBCs. As an example, the following table summarises the categorisation used in France and recall the "qualitative acceptance requirements".

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1 The lack of progressiveness means the existence of " short " sequences for which, upstream from the initiator, the failure of a barrier entails a major increase, in terms of consequences, without any possibility of salvaging the situation at an intermediate stage.

2 The homogeneity means that no initiator family participates in an excessive manner to the global probability of the major accident ; this might also be reached within a same Postulated Initiating Event (PIE) family.

3 The exhaustiveness should be a target during the phase for the identification of the initiating events.

4 The initial states - steady states and operational transients - should be defined by the designer.

Table 1. Conventional Design Basis Conditions categorisation

DBC Category	Definition	Occurrence frequency (events / reactor · year)	Qualitative acceptance requirements
I	**Normal Operation conditions**	list of operational events to be defined explicitly	Plant conditions within the limits of the technical specifications and defined in the general operating rules
II	**Incident conditions**	$10^{-2} \sim < f$	Consequences must remain extremely limited
III	**Accident conditions**	$10^{-4} < f < 10^{-2}$	Consequences must remain sufficiently limited
IV	**Hypothetical accident conditions**	$f < 10^{-4}$	Consequences must remain acceptable

5.2 - Design Extension Conditions

To complete the deterministic approach, the compliance with the forth level of Defence In Depth imposes, on one hand, the integration of a possible **lack of exhaustiveness** in the deterministic assessment and, on the other hand, the demonstration of **the potential of the facility for the prevention, control and limitation of the consequences of " severe accidents**[5]" (cf. section 4 : the 4th level of the Defence in Depth).

The European Technical Safety Organisations TSOs (TSO, 1997) elaborated a set of recommendations for the treatment of severe accidents for future PWRs. Coherently with the principles evoked in section 3, CEA/DRN considers that these top-tier requirements are broadly applicable to all the future BNIs :

" ...For future designs, accidents beyond the "classical" deterministic design basis have to be considered at an early stage of the design to obtain a significant reduction of the probability of occurrence of severe accidents. Accident situations which would lead to large early releases have to be "practically eliminated". For the remaining accident situations..., the potential release of radioactive materials outside the containment system should be small, so that only limited protective measures are required.

The accident scenarios to be considered early in the design process should be all those accidents, even of very low probability, that can be judged as physically plausible. ...

Severe accident sequences should be either "practically eliminated" if sufficient preventive design and operation provisions are taken, or "dealt with"... There is a need to develop adequate guidance to clearly establish when "sufficient design and operation provisions" have been taken to practically eliminate a severe accident sequence[6]*.*

Evaluation of severe accidents should be performed using a "best-estimate" approach together with a quantification of the uncertainties to determine, for representative scenarios, a spectrum of the possible outcomes.

Systems and components provided only for severe accidents, should not require the same conservative analysis and requirements that are necessary for those developed to cope with "classical" design

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5 Considering the needs for taking into account the different reactors specificity (e.g. no core melting (?) for the gas cooled reactors core) **severe accident configurations are identified as highly hypothetical accidental conditions or envelope situations, studied to design and to implement the LODs needed for the 4th level of the defence in depth (see also section 6.3)**.

6 Referring TSO, on one hand the " practical elimination " of such accident sequences shall be matter of judgement and each type of accident sequences has to be assessed separately and, on the other hand, their practical elimination cannot be demonstrated by the compliance with a general " cut-off " probabilistic value.

basis accidents;.... There should be a high confidence that necessary equipment will survive severe accident conditions for the period that it is needed to perform its intended function. "

In order to answer these requirements, the conventional **DBCs** shall be completed by the integration of accidental situations generated by multiple failures (e.g. : the total loss of the redundant systems) or the severe accidents. These situations studied for the prevention, control and limitation of consequences are qualified as **Design Extension Conditions (DECs)**[7]. Their analysis may lead to design additional systems and/or to adapt existing systems in order to guarantee that the associated safety objectives are verified ; two types of accidents shall be considered :

- the **Complex sequences** for the **prevention of core meltdown and/or severe accidents** in the facilities. The analysis of these sequences may lead to design additional systems and/or to adapt existing systems to guarantee that the associated safety objectives are verified. For these situations, additional failures will not be taken into account. From the safety point of view, the designer should aim at meeting the conventional design basis targets (4^{th} category ; cf. table 1).
- the **Severe accidents** for the prevention of unacceptable releases (see footnote N° 6). Specific objectives are established ; referring TSO, they correspond to a " design release ", outside the containment, "*so that only limited protective measures are required*". As for the situations above, additional failures will not be taken into account. Nevertheless, the " *quantification of the uncertainties to determine, for representative scenarios, a spectrum of the possible outcomes* " (TSO, 1997) is also an essential step.

For each of these families, a limited number of situations shall be identified and taken into account to design the **safety related architecture**.

For other accidental situations, the release of which is estimated as unacceptable for the environment, it is necessary to demonstrate that : they are either excluded by design, or that their occurrence frequency allows them to be rejected in the **Residual Risk**. These situations will not be analysed.

This part of the approach meets the recommendation for the exclusion of any " cliff edge effect[8] " leading to an early release, greater than the design release (i.e. a non-admissible source term).

In conclusion, a BNI whose mobilisable inventory is greater than the one " unacceptable for the environment ", shall answer the requirements evoked above.

The global approach is summarised in Fig. 1. The whole sets DBCs and DECs, taken into account for the design of the global safety related plant architecture, are qualified of **Plausible Plant Conditions (PPCs)**. The acceptance requirements are recalled.

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7 The Design Extension Conditions (DECs) is the terminology suggested in the European Utility Requirements document (EUR, 1998). The same concepts are grouped within the Risk Reduction Categories (RRC) of the European Pressurised Reactor (EPR) .

8 This risk corresponds to the **mobilisation of a potentially unacceptable source term with the simultaneous loss of the containment** (release higher than the design release).

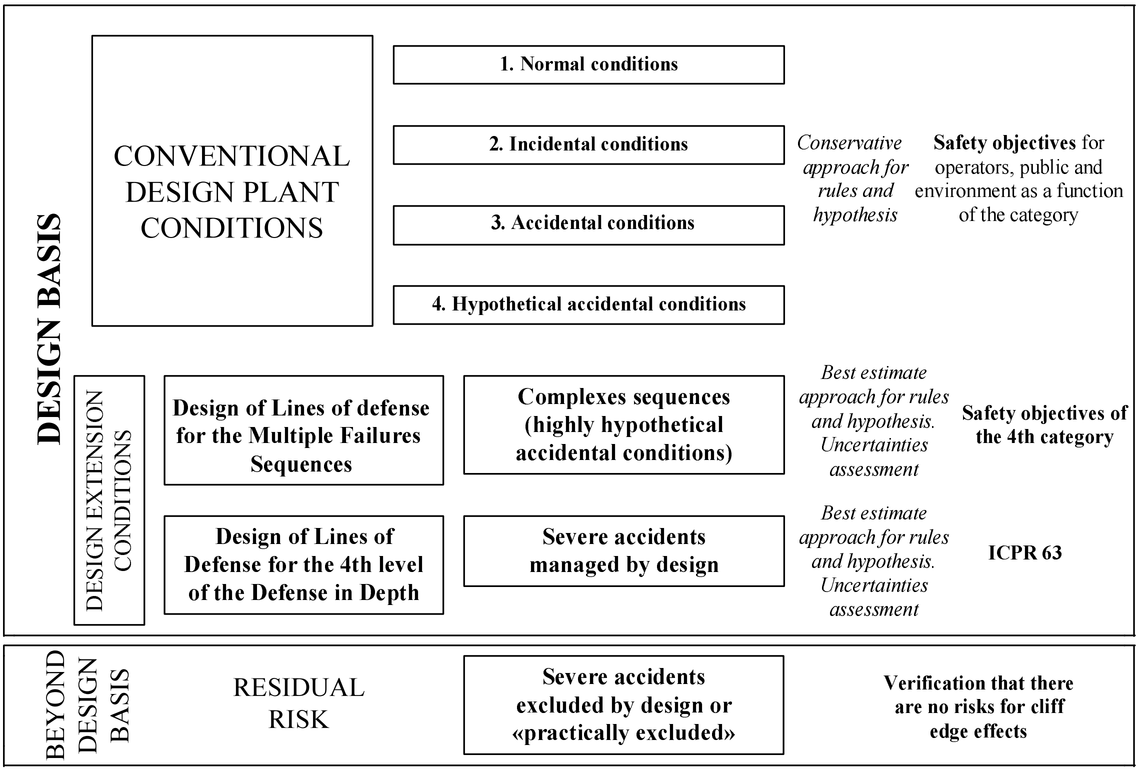

FIG. 1. Safety approach for plant design and assessment.

6 - SAFETY OBJECTIVES

6.1 – Safety objectives in terms of radiological consequences

The basis for the suggested approach is, first of all, the coherency with the "fundamental safety objectives" defined by the IAEA (Safety Series N°110 - 1993).

The Safety Objectives in terms of "allowable radiological consequences" define the criteria taken into account at the design level " *to protect individuals, society and the environment from harm ... against radiological hazards* " (cf. IAEA - 1993). It is considered that the risk[9] shall be minimised and it shall be homogeneous[10].

As discussed above, PPCs are categorised according to the corresponding estimated frequency. For the different reference incidents and accidents with radiological significance, appropriate **technical criteria** have to be respected (i.e. for each category). For DBCs, these radiological consequences shall satisfy the qualitative acceptance requirements shown on table 1. For DECs, the retained objectives shall lead to guarantee that "*only limited protective measures are required*" (cf. TSO). As pointed out on section 3, the retained values **should be the same for all the future BNIs including the Nuclear Desalination Plants**.

Finally, some events are rejected into the Residual Risk category if sufficient design and operation provisions are taken, so that it can be clearly demonstrated that it is possible to " practically eliminate " this type of accident situations.

Figure 2 summarises the allowable/non allowable risk domain. A tentative qualitative comparison with the INES Scale classification is presented.

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9 The risk is defined as the « product » between the sequence occurrence frequency and the consequences.

10 Higher radiological consequences can be deemed tolerable for categories of lower estimated frequency.

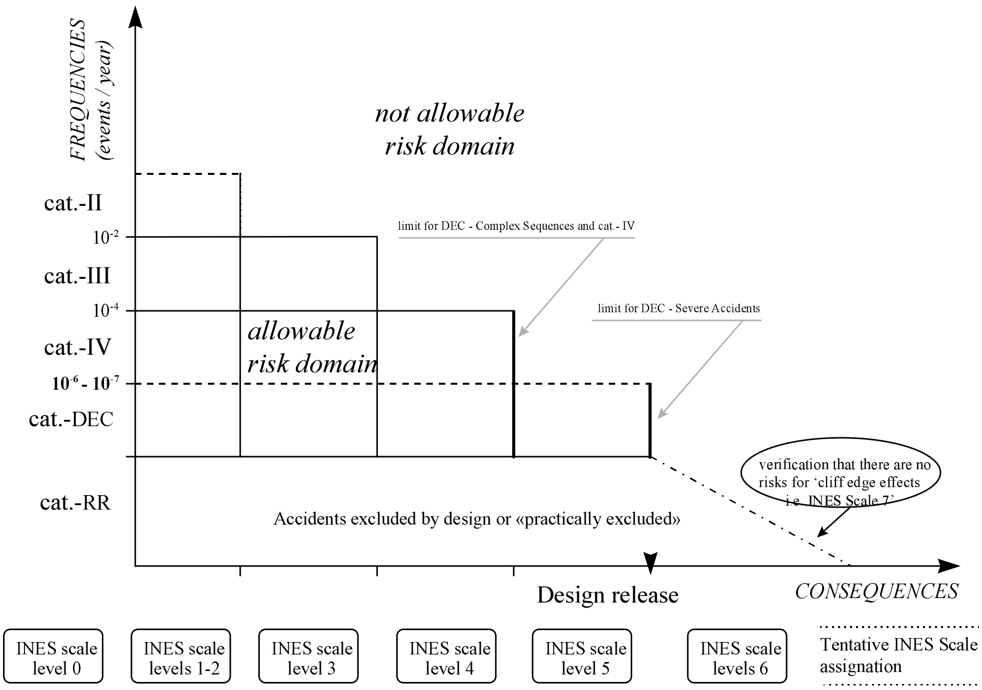

FIG. 2. Allowable/not allowable risk domains.

6.2 – Use of probabilistic safety studies and probabilistic objectives

Probabilistic safety studies could possibly be used as a design and analysis tool, in the different stages of the project. For the future and innovative concepts, the main difficulty could comes from the lack of reliable data. On the other hand, the implementation of probabilistic approach needs the definition of clearly understandable probabilistic objectives.

For future fission plants the reduction of the probability of core meltdown (i.e ; the severe accident), from that for present reactors, should be a target in itself. A design target of 10^{-6}/ reactor · year should be set for **internal events** ; this target should be compatible with those set internationally, in particular by INSAG 75-3 (10^{-5} / reactor · year, with all types of failures and events taken into account). This target should be applicable to the future BNIs for the " severe accidents ". On the other hand, for the design release, a global probabilistic target of 10^{-6}/reactor · year, for **all initiating events** is retained for fission plants. This target should also be applicable to the future BNIs.

Before to enter within the detail of such objectives, it is essential to discuss the notion of Lines of Defence.

6.3 - The notion of "Lines of Defence"

The term **Line of Defence (LOD)** is used for : any inherent characteristic, equipment, system, etc., implemented into the safety related plant architecture and any procedure foreseen coherently with the General Rules for Plant Operation (e.g. human actions : preventive, protective, etc.), the objective of which is to accomplish a given safety function. According to its nature and frequency, the event considered can itself correspond to the disappearance of a line of defence (e.g. a breach). This notion is coherent with the " **effective defences** " evoked by IAEA (Safety Series N°110 : The Safety of Nuclear Installations - 1993) in the definition of the General Nuclear Safety Objective : " *To protect individuals, society and the environment from harm by **establishing and maintaining in nuclear installations effective defences** against radiological hazards* ". The idea on which the proposed concept is based is that, following a PIE, the passage from the normal plant condition to the degraded condition results from the failure of **inherent characteristics**[11] **and/or equipments**

[11] A detailed analysis shows that even the "inherent characteristics" are characterised by a given "reliability". Generally speaking, this reliability, is defined as the probability to fail the mission requested to achieve a generic safety function. This reliability depends on environmental, physical, nuclear or chemical phenomena.

and/or procedures implemented to cope the PIE, i.e. which achieve the requested mission and fulfil the corresponding safety function.

According to the concept of **defence in depth**, the **safety design** leads to practically implement these **lines of defence** organising the global architecture and their interactions. The plant **safety assessment**, both for normal and abnormal plant conditions, leads to identify and count these lines, verifying they are able to fulfil the requested missions (performances) with the needed reliability. In order to define this concept more precisely, two types of LOD are considered : **the strong lines** (called a) and the **average lines** (called b). The strong lines of defence (**a**) correspond to inherent features and protection or safeguard systems designed to meet high reliability performances. Their probability of failure is of the order of 10^{-3} per year or per demand. The average lines (**b**), do not meet the same design or implementation requirements and they show a lower reliability (e.g.: lesser safety margins, operator actions). Their failure probability per year or per demand must nevertheless be of the order of 10^{-1}.

6.4 – The LOD method for the design and the assessment

These global objectives are essential but they cannot be directly used in the design and it is necessary to translate them, in a practical manner, into intermediate objectives that can guide the designer in the definition of the needed inherent characteristics, equipment, procedures (i.e. the **number** and the **quality of LOD**) and in the architecture of the concept. These intermediate objectives might also give margins destined to cover the uncertainties of the probabilistic approach. Moreover, as a design goal, accident situations which would lead to large early releases have to be practically eliminated, i.e. when they cannot be considered as physically not plausible, design provisions have to be taken to design them out (this corresponds to the rejection of these failure configurations in the Residual Risk). Referring TSO, the " practical elimination " of such accident sequences is a " *matter of judgement* " and each type of accident sequences has to be separately assessed ; moreover, *due to the limited knowledge on some physical phenomena and due to the large variety of these types of accident sequences, their practical elimination cannot be demonstrated by the compliance with a general "cut-off " probabilistic value*.

As showed on section 6.2, in order to show our concern in improving the margins, it is suggested to retain a probabilistic objective for the severe accident, due to initiators of internal origin, of 10^{-6} per reactor per year ; this value become a **fraction of 10^{-7} per reactor, per year, per family of initiators and per safety function.** Moreover, some sequences and the risk of cliff edge effects must be excluded by design. As discussed above, the practical elimination cannot be demonstrated by the compliance with a general " cut-off " probabilistic value ; nevertheless it can be considered that the corresponding frequency should be **much lower than 10^{-7}/reactor year**. In our approach it is considered that the **"** *matter of judgement* " evoked for the demonstration of the practical elimination, could be satisfied using the LOD notion, designing an architecture which implement an adequate number of effective LOD. In parallel to these objectives, it must be emphasised that, in as much as possible, the design of the facility must aim at **defence homogeneity, progressiveness and exhaustiveness**. Taking into account both the probabilistic objective and the homogeneity and progressiveness principles, leads us to recommend an architecture in coherence with Table 2.

During an effective LOD methodology application, the individuation of the globally needed LOD number is obtained by means of the aid of the Table 2. For a better understanding the Figure 3 can also be seen. This figure explains, by means a graphic, the same rationale.

It is important to note that the recommended defence architecture of table 2/fig. 3 is structured in accordance with the " classical " frequency categorisation normally presented on the risk domain.

In summary, the application of the method is based on the adoption of the following rule: " For each Plausible Plant Condition, and for each safety function the failure of which entails potential consequences higher than those allowed, the number of LODs implemented to accomplish the function should be at least equal to this recommended into the safety related defence architecture (cf. table 2). The counting of these lines, and the comparison with a predefined allowable architecture, allows to verify that the global risk of the "PPC + failure of the lines of defence" sequence, is acceptable ".

Table 2 - Defence architecture - position of the Lines of Defence (LOD) between the initial design condition and the operating situation of which the limits must be respected

		Category of the initial design condition			
		2	3	4	DEC
Category of the	2	**b**			
resulting	3	**b**	**b**		
operating	4	**a**	**a**	**a**	
situation	DEC	**b**	**b**	**b**	**b**
Total number of LODs to be implemented to prevent severe accident		**2a+b[a]**	**2a**	**a+b**	**b**
Total number of LODs to practically exclude a given sequence		**> 2a+b**	**> 2a**	**> a+b**	**> b**

Note: " It is understood that, for a given initial design condition, in case of a potential source term lower than that considered as unacceptable, but nevertheless superior to that allowed in the category of initial design conditions, the objectives, in terms of number of lines of defence, must be compatible with respecting the radiological consequences of the different occurrence categories (cat. I to IV; see Table 2) ".

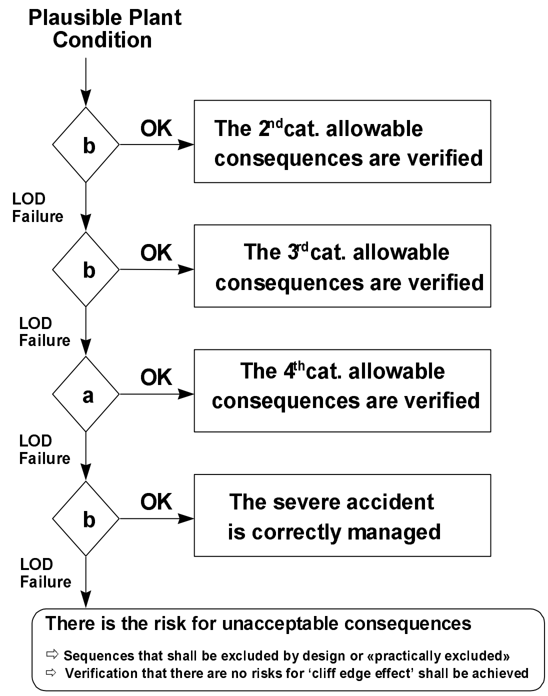

FIG. 3. Example of LOD implemention for a cat_{freq}-II plausible plant condition (PPC), versus a safety function whose failure leads to unacceptable consequences.

[a] Note that [b+b = a] if the independence between the different lines is effective.

The practical systematic methodology implementation is summarised through two main steps : the design's review for the full set of PIEs, with particular emphasis on the systematic identification of all safety functions ; the systematic analysis of provided LODs for all the corresponding PPCs and the comparison with the suggested safety related architecture.

Such approach reaches, as output, design recommendations and can allows fruitful and direct discussion between the designer and the analyst. The generality of the methodology allows its application to support the preliminary safety assessment during the first phases of plant conception and design, as well as in assessing already defined innovative concepts.

7 - CONCLUSIONS

It is considered that the definition of clear and unambiguous Design Requirements is essential to provide **a common base for the organisation of the design and the assessment tasks** for future nuclear installations. The discussed safety approach is the result of the experience maturated, within the context of the CEA/DRN Innovative Programme through practical applications over several future concepts, both for fission and fusion reactors. The background of this experience is structured coherently with the European Safety Authorities recommendations, the European Utilities Requirements (EUR) and the "fundamental safety objectives" defined by the IAEA.

The CEA/DRN approach is founded on **three essential principles** :

- The allowable radiological values applicable for the plant operators, the public and the environment **should be the same for all the future BNIs including the Nuclear Desalination Plants**.
- For all these BNIs, the implementation of an adequate safety related architecture and the organisation of the safety assessment should be achieved using analogous and comparable approaches based on **the integral adoption of the defence in depth**.
- These approaches should be **able to integrate the peculiar characteristics of each installation**.

The main concern of the paper is the complete applicability of the **Defence In Depth principle** (i.e. all the levels) for the safety design process of all future nuclear installations. The notion of **Lines of Defence** is used to concretely help the implementation of the corresponding principles, and to assess the pertinence of the chosen design options :

- the **safety design** leads to practically implement these **Lines of Defence** organising the global architecture and their interactions ;
- the **safety assessment**, both for normal and abnormal conditions, leads to identify and count these lines, verifying they are able to fulfil the requested missions (performances) with the needed reliability.

Such approach reaches, as output, design recommendations and can allows fruitful and direct discussion between the designer and the analyst.

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APPENDIX 1

“ The implementation of the defence in depth for the design option selection ”

The different levels of Defence In Depth should be systematically applied :

1st level : Prevention : Quality in design and achievement, prevention of nonconformity

a ▫ Facility with an excellent inherent resistance so as to reduce risks of failures or internal hazards ;
b ▫ In-depth study of all the predictable plant conditions both normal and off-normal ;
c ▫ Appropriate and clearly defined safety criteria ;
d ▫ Facility with a reduced potential for consequences following off normal sequences;
e ▫ Use of well established and qualified rules and codes which provide, at all times, sufficient margins related to the limits defined to ensure the proper behaviour of the facility ;
f ▫ Margins in order to avoid constantly using the systems designed in case of off-normal operating conditions ;
g ▫ Choice of materials and fabrication controls in compliance with the design and construction rules;
h ▫ Compliance to the existing quality assurance rules ;
i ▫ Integration of the safety culture in the design team and in the operating team.

2nd level : Surveillance, detection and control : Quality of operation, keeping the facility within authorised limits

- Measures anticipating failure : periodic surveillance program which recognises the off-normal evolution of the most important material, which could, in the long run, lead to failures (periodic controls and tests). The maintenance policy will therefore be based on:
  - a preventive programme (periodic controls)
  - an on line control of the operating parameters
  - a predictive programme for the equipment whose reliability is of vital importance and for which experience feedback does not allow to determine systematic replacement times.
- Regulation, control and protection systems which are sufficiently reliable, in order to detect a nonconformity or a failure as soon as it occurs, and to stop an off-normal evolution before the materials are solicited beyond the foreseen conditions.

3rd level : Safety systems and Protection systems design : Postulate of all plausible incidents and accidents and implementation of means to limit the effects of these accidents to acceptable levels.

- Postulate of all plausible incidents and accidents.
- Implementation of means (inherent characteristics, engineered safety features and protective systems, procedures) to limit the effects of these accidents to acceptable levels.
- The choice of postulated incidents and accidents should be done at the beginning of the project, with the aim to integrate perfectly the needed systems to the whole facility.

4th level : Accident management and containment protection, limitation of consequences : Prevention of deterioration of accidental conditions and limitation of severe accident consequences.

- Integration of multiple failures (e.g. postulate of loss of redundant safety systems).
- Implementation of additional measures (inherent features, adapted materials, procedures) to be able to face operating conditions which have not been treated by the first three levels but are likely to lead to significant releases, and to limit the consequences of severe accidents.

Nota bene : the notion of multiple failures leading to highly hypothetical sequences is not systematically integrated by the designers. To take into account these sequences is essential to achieve a correct severe accident prevention. Moreover the principle for the implementation of additional dedicated measures to cope with such a sequences is essential.

5th level : Response outside the site : Limitation of radiological consequences for populations in case of significant releases.

CURRENT STATUS OF THE PBMR LICENSING PROJECT

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Abstract

The CNS is currently reviewing the PBMR conceptual design from a licensibility point of view. The PBMR concept is based on a High Temperature Gas Cooled Reactor - pebble bed reactor type. It is anticipated that the PBMR design will rely on inherent safety characteristics to contain fission products within fuel over the full range of design basis events. This feature combined with the high temperature integrity of the fuel and structural graphite, allows the safe use of a high coolant temperature, which allows consideration of the future development of this reactor for non-electrical applications of nuclear heat for industrial use. The CNS licensing approach requires that the licensing and design basis of the plant should respect prevailing international norms and practices and that a quantitative risk assessment should demonstrate compliance with the CNS fundamental safety standards. The first stage of the licensing process is now ongoing; this is a pre-application phase, which will result in a statement on licensibility being issued. Identification of the specific documentation requirements and information needed is required across every step of the licensing process. Top level regulatory requirements have been established for the PBMR. They include the CNS fundamental safety standard and basic licensing criteria, which describes requirements on licensees of nuclear installations regarding risk assessment and compliance with the safety criteria and define classification of licensing basis events.

INTRODUCTION

The PBMR licensing project started in South Africa in July 1997 when ESKOM, the National Electricity Utility, "opened" a project to start investigation into the licensing of the South African high temperature gas-cooled Pebble Bed Modular Reactor (PBMR) concept. The South African nuclear regulatory authority, the Council for Nuclear Safety (CNS) is currently at the first licensing stage of the safety review of the PBMR. Formal application for PBMR stage license was received by the CNS from ESKOM at the end of July '98, where ESKOM requested for the issuance of the licence defining and approving the safety bases for a proposed PBMR. The CNS is evaluating now the acceptability of the safety bases for the proposed PBMR concept.

The PBMR concept is based on High Temperature Gas Cooled Reactor — pebble bed reactor type. The proposed PBMR design reference is German "HTR Module" design, which was reviewed by German Regulatory Authorities and declared "licensable".

Central features that provides inherent (passive) safety of this type reactor design is a nuclear fuel in the form of spheres with the coated uranium particles inside, which are distributed in the graphite matrix covered by an unfuelled graphite coat. This inherent safety is based on the ability of the coated particles to retain all key radionuclides as long as a maximum fuel temperature of about 1 600^{o}C is not exceeded. Therefore, this reactor design has the possibility to exclude reliance on any active safety systems inside the primary circuit for postulated accident scenarios. A few changes to this "original" reference design have been made in Eskom's design [1], to increase the electrical power to about 110 MW(e). The main differences are direct Brayton cycle, higher temperature and an annular core. Helium, which is chemically inert and cannot be activated, is used as a coolant and medium driving the turbo generator. This, combined with the high temperature integrity of the fuel and structural graphite, allows the use of a high coolant temperature about 900^{o}C.

The possibility to achieve high temperature and passive safety in the PBMR design allows consideration of future development of this reactor for non-electrical applications of nuclear heat for industrial use. A heat source provided by a Pebble Bed Nuclear Reactor could be used for district heating, seawater desalination, heavy water production, hydrogen production and other industrial applications. Therefore the current status of the licensing of the PBMR (still with electrical power production) is presented.

CNS LICENSING APPROACH

South Africa operates the Koeberg Nuclear Power Plant (NPP) with two Framatome design PWR units, 900 MW(e) each, commissioned in 1983 and 1984 and situated near Cape Town. A 20 MW ORNL swimming pool research and isotope production reactor is also licensed to operate. The majority of previous experience of the CNS has been based mainly on licensing of the PWR reactor type. To acquire additional expertise in the HTGR area the CNS approached international companies for specialised consultant services to back up its existing technical specialists. The CNS is currently also investigating the necessary additional staff complement.

The Licensing approach, which is similar with one used for the Licensing of the Koeberg Reactors has been adopted in respect of the PBMR. It requires that the licensing and design basis of the plant should respect prevailing international norms and practices. It also requires that a quantitative risk assessment should demonstrate compliance with the CNS fundamental safety standards. The establishment and maintenance of an adequate emergency plan is required based on the outcome of the safety assessment process. Conditions of licence will be established on the outcome of the assessment including requirements for General Operating Rules.

From July 1997 to date, discussions with ESKOM took place to plan the overall Licensing process to be adopted. These include all the activities required to develop the licensing basis, perform the safety assessment, issue and update the nuclear licence for the PBMR plant during all stages of the project and plant life cycles: design, construction and installation, commission, testing, operation and decommission of the plant. It is subdivided into the following main phases:

- A Pre-Application Phase, which is a current first licensing stage.
- An Application Phase, when the PBMR licence will be granted and updated for every sub-stage of the licensing process e.g. construction, start-up, commission and operating.
- Operation Phase.
- Decommission Phase.

LICENSING PROJECT MANAGEMENT

The detailed licensing project specifications including the licensing processes, project scope and project organisation are documented in the CNS PBMR Project Management Manual, which was officially issued at the end of the last year. The overall Project is managed by the CNS PBMR Project Manager.

In order to ensure an "orderly" and efficient Licensing process the various activities have been grouped in Sub- Projects. Each Sub-Project is in turn managed by a Technical Project Leader. Four Sub projects have been identified this far:

- Project A: Licensing Basis
- Project B: Risk Assessment
- Project C: General Operating Rules & Plant Design Engineering
- Project D: Pre-operational testing programmes.

Agreement for licensing fees, which covers CNS expenditure concerning overseas consultants, CNS local staff and optional local consultants, has been agreed with ESKOM. Agreement with overseas consultants have been finalised for support in the following disciplines:

ESI (Germany) - Radiation Safety Engineering and Fuel
NNC/BEI (UK) - Materials
Framatome (France) - Design Criteria and Rules and Nuclear Engineering.

An agreement with US General Atomics is expected to be signed in a few weeks time. The main licensing activities are currently being carried out by consultants, which are being shadowed by CNS staff.

The CNS has established quality management requirements, which apply at each stage of the project, and a few initial audits on ESKOM activities have been already performed to ensure that the PBMR project follows the required processes.

Regular Licensing meetings are held with ESKOM. During these meetings, the project status, licensing progress review, problems raised and ways to resolve them are discussed.

CURRENT STATUS OF THE LICENSING PROCESS

As was already mentioned the project is currently in a pre-application phase of the licensing process, which was described in the CNS paper presented last year at the TCM " Safety Related Design and Economic Aspects of High Temperature Gas Cooled Reactors", here at INET [2].

The main activities during this stage include:

a) establishment of the licensing basis,
b) safety review and assessment of the PBMR Safety Analysis Report (SAR) Rev 0 and supporting documents,
c) issue of the licensibility statement.

The documentation forming the PBMR licensing basis is indicated in Figure 1.

The first two documents have been developed by the CNS and they define “Top Level Regulatory Requirements to the PBMR”. The CNS Risk Criteria [3] describes the requirements on licensees of nuclear installations regarding risk assessment and compliance with the safety criteria and it is applicable to any nuclear installation licensed in the South Africa. The second document has been developed specific for the PBMR. It is currently under review and will be officially issued during the next month. The document "CNS PBMR Basic Licensing Criteria” is the CNS Licence Document, which sets out the safety criteria of the CNS and the requirements on licensees to demonstrate compliance with these criteria. These criteria apply to events or combinations of events, which lead or could lead to exposure of either the plant personnel or members of the public. Three groups of event are considered which are defined in terms of the annual frequency of occurrence, ranging from those leading to exposures which are considered to be part of normal operation and those which are less likely to occur. For the three groups considered, safety requirements and numerical safety criteria are provided (See Table 1). Amongst the safety criteria specified in addition to the numerical criteria, are the application of the ALARA principle and the principle of defence-in-depth. The application of the ALARA principle involves, through the application of one of a range of simplistic to complex techniques, the choice of the option that gives the optimum level of safety among the feasible alternative safety options. In addition to this, the most important technical principle is that of defence-in-depth. This principle requires that there should be layers (structures, components, systems, procedures, or a combination thereof) of overlapping safety provisions. Accident prevention and accident mitigation are natural consequences of the defence-in-depth principle.

Unfortunately unlike the LWRs, e.g. Koeberg NPP, broad international consensus has not been developed in terms of internationally acceptable general design criteria and design rules for the PBMR. No international "off the shelf" package is available for defining the design basis of the PBMR. Of course some rules and criteria were developed during the licensing of HTGRs in Germany, USA have investigated design bases for MHTGR Project, etc., but because there is no commercial HTGR plant existing in the world this work has not yet been finished. Therefore the CNS requested ESKOM to establish and document the General Design Criteria (GDC) and associated design rules for the PBMR using existing documents as guidelines.

Identification of specific documentation requirements and information needed is required across every step of the licensing process and because of the lack of expertise in the development of this documentation for PBMR. ESKOM were advised to use existing documents, mainly from USA and Germany, as guidance for development not only of the general design criteria, but also for a set of others needed for development of the licensing basis. The CNS also requested ESKOM to develop and submit for CNS review, documents describing processes and methodologies used to develop licensing basis documents. Many licensing documents concerning the HTGR have been requested and received by the CNS from US NRC and forwarded to ESKOM.

The CNS itself is currently developing a PBMR Safety Assessment Review Guide for the guidance of CNS reviewers in performing the safety review of application to construct or operate the PBMR nuclear power plants and the review of applications to approve standard design for PBMR nuclear power plants. It also serves the purpose of assuring the quality and uniformity of staff safety reviews. The PBMR Review Plan is being developed by, inter alia, considering the light water reactor edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-0800, customised to the needs of the PBMR. It was clear that the manpower effort required to undertake a complete, thorough review and modification of NUREG 0800 for application to the PBMR would be substantial, but it was decided that such an effort would be worthwhile. Therefore a significant number of items, which covered a representative range of technologies including fuel and its storage, core, residual heat removal, fission products, radioactive waste management, radiation protection, accident analysis, specific components and structural materials, were completely modified.

The proposed PBMR design is quite different from any other plant operating anywhere in the world today. Thus, although a standard review plan can be produced for this

Table 1: PBMR Basic Licensing Criteria

EVENT	SAFETY REQUIREMENTS	SAFETY CRITERIA
CATEGORY A Category A events (or combinations of events) are those which lead to exposure and which occur with a frequency not less than one in one hundred years (10^{-2} y^{-1}). Such events are treated as part of Normal Operation.	The design shall be such to ensure that under anticipated conditions of Normal Operation, which includes exposures resulting from minor mishaps and misjudgements in operations, maintenance and decommissioning, there shall be no radiation hazard to the workforce and members of the public.	Limitation of individual radiation dose to : **20 mSv** to plant personnel and **250 □Sv** to members of the public All doses shall be kept **ALARA** The principle of d**efence-in-depth** shall be applied
CATEGORY B Category B events (or combinations of events) are those which lead to exposure and which occur with a frequency of between one in one hundred years (10^{-2} y^{-1}) and one in one million years (10^{-6} y^{-1}).	The design must be such to prevent and mitigate potential equipment failure or withstand externally or internally originating events which could give rise to plant damage leading to radiation hazards to plant personnel and members of the public in excess of the safety criteria. The analysis performed to demonstrate compliance with this requirement must be conservative.	Limitation of individual radiation dose to : **500 mSv** to plant personnel and **50 mSv** to members of the public All doses shall be kept ALARA The principle of defence-in-depth shall be applied
CATEGORY C Category C events (or combinations of events) are all possible events that could lead to exposure with the exception of those which are treated as part of Normal Operation. As such, Category C events will include Category B events as well as events which occur with a frequency of less than 10^{-6}.	The design shall be demonstrated to respect the CNS risk criteria for plant personnel and members of the public The analysis performed to demonstrate compliance with this requirement must use best estimate data with a supporting uncertainty analysis	Limitation of risk to the values set by the CNS risk criteria. Plant Personnel 5×10^{-5} peak individual risk 10^{-5} average risk Members Of The Public 5×10^{-6} peak individual risk 10^{-8} population risk per site a bias against larger accidents **and** ALARA and defence-in-depth Principle

plant it must be recognised that its development will continue. There are specific aspects that will require additional consideration when reviewing the safety case, because the PBMR is designed to operate at temperatures well in excess of those experienced in other commercial plant, has a fuel embedded in graphite spheres with no metallic cladding, has a helium coolant with no specified impurity limits as yet, and is a direct cycle plant. Thus there are many potential material selection problems and this CNS document will provide guidance as to what additional information the reviewer must seek from the applicant to enable him to complete his review with the necessary level of confidence.

Documentation presented on Figure 1 below the “Top Level Regulatory Requirements” is being developed by ESKOM and fall under the CNS review and acceptance process. The Estimate PRA planed to be submitted in July of this year. The PBMR Safety Analysis Report will be the main document that demonstrates how the PBMR meets the required nuclear safety criteria. The guidance on format, scope and content of Safety Analysis Report was provided to ESKOM and will be discussed in detail during the next month. Revision 0 of the SAR is planed to be submitted at the end of October of this year.

Up to now the first issue of the following documents have been submitted by ESKOM:

- Licensing Manual, containing ESKOM's "Overall Licensing Philosophy for PBMR"
- Glossary of Terms
- PBMR Technical Description
- Proposed General Design Criteria for PBMR
- Proposed list of events/accidents.

These documents have been reviewed and assessed by the CNS. A few briefings and workshops between the CNS and ESKOM have been organised to improve and speed up the review process. Some deficiencies and concerns have been found and forwarded to ESKOM. The main concerns are caused by limited data availability and operational experience feedback for this type of reactor. The CNS has issued a safety evaluation report on the PBMR technical description.

Development of the radiation protection programme has started. It will include:

Staffing and training needs, surveillance requirements, controlled zone classification, demarcation, access control requirements, external and internal dosimetry requirements, protective clothing, change room and laundry specifications, respiratory programme, instrumentation requirements, radioactive material control, decontamination programme, ALARA programme, shielding requirements, computer system requirements, procedures, reporting and record retention, etc.

An emergency plan basis will be produced by ESKOM in the middle of this year.

According to national environmental legislation (Environment Conservation Act, 1989), for any industrial construction an Environmental Impact Assessment (EIA) shall be performed and accepted by Department of Environmental Affairs and Tourism (DEAT). It has been agreed between DEAT and the CNS that CNS will provide technical support to DEAT in evaluation of the radiation impact of the PBMR plant. ESKOM is currently busy preparing an EIA report and two public hearings are already planned for June and September of this year.

PBMR LICENSING BASIS

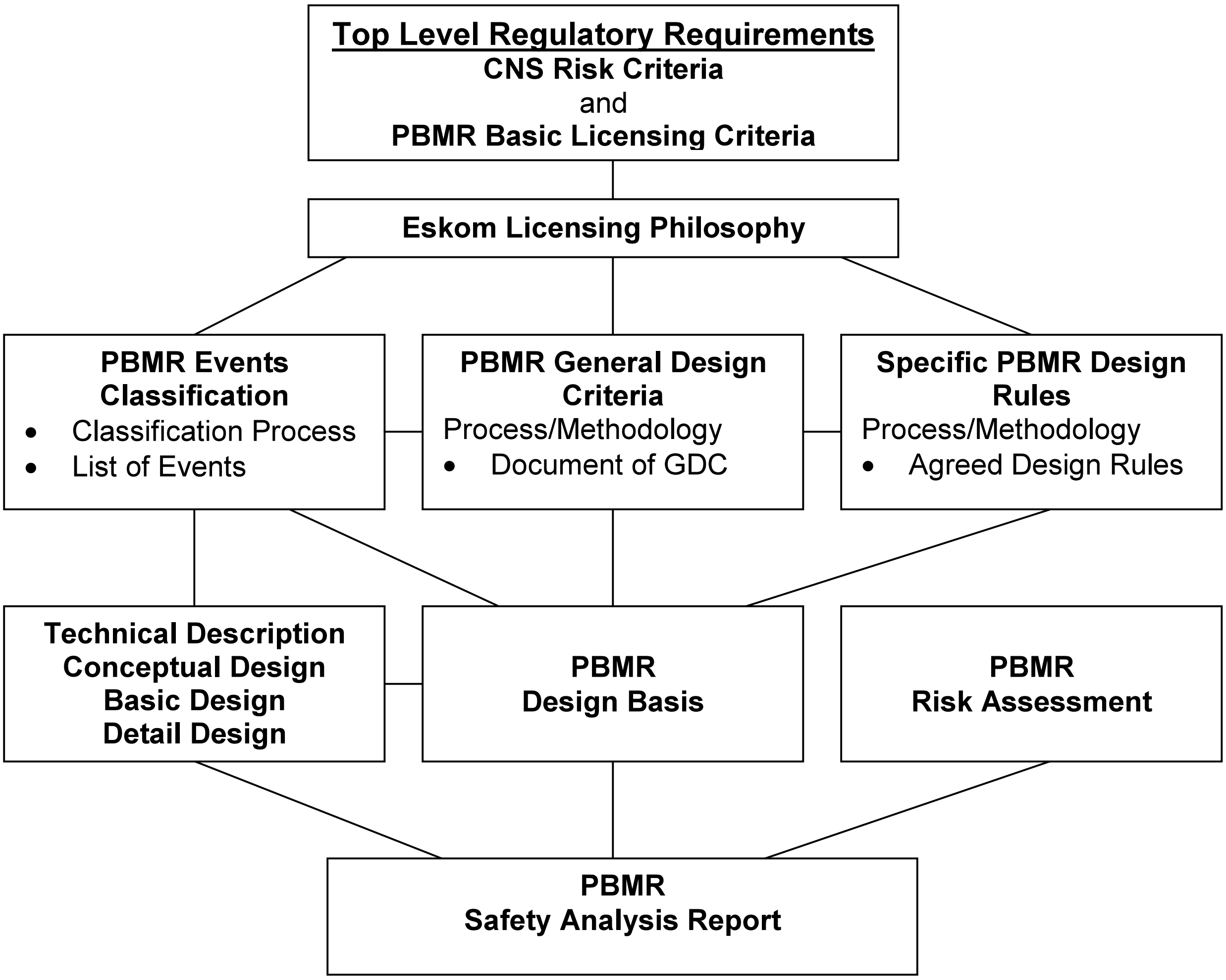

FIG. 1. Documentation forming the PBMR licensing basis.

CONCLUSION

In spite of some inevitable minor delays in the PBMR project development, significant progress has been achieved. Expected completion of the first stage of the licensing process is the end of 1999. The issue of a licensibility statement defining acceptability of the safety bases for the proposed PBMR is expected at the beginning of 2000. This target date is optimistic although not unattainable and is very much dependant on the availability of documentation and information required during the licensing process.

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