PANEL DISCUSSION — INCENTIVES FOR INTRODUCTION OF SMALL AND MEDIUM REACTORS IN DEVELOPING COUNTRIES

(Session 8)

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INCENTIVES FOR INTRODUCTION OF SMRs IN DEVELOPING COUNTRIES

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Abstract

Small and medium size reactor development has many incentives, some are economic and others are safety related. Another incentive to SMR development has been its suitability for implementation of new design approaches. Small reactors have a potential for substantial simplification of the design, facilitating more economical operation and maintenance.

A most serious constraint for nuclear power introduction in developing countries is the difficulty in obtaining financing. The high investment costs may even exceed the overall credit limits of lending institutions for an individual developing country. Two other factors are longer construction time and potential uncertainties in schedule and costs due to regulatory and public intervention and policy changes.

On the basis of this orientation, a feasibility study and siting of the first Nuclear Power Plant of SIDI BOULBRA in Morocco had been undertaken between 1984 and 1996 in the framework of the agreement concluded between ONE and the French company SOFRATOME under the support of IAEA. This study has shown that the nuclear power option could be a viable solution. The chosen site of SIDI BOULBRA was recognized by the Agency as a qualified site. It has now been decided to update the above mentioned study in order to introduce a Small and Medium Reactor (300-600 MW) which is suitable for the national electric grid and which will contribute to the energy demand in the horizon of 2015.

1. INTRODUCTION

The population growth and the development of new regions of the world are inevitably leading to increase energy needs, and particularly an increase in electricity consumption.

The world electricity market responds to general economic conditions, but also to more specialized political and strategic preoccupations. Besides hydropower, the only renewable energy source that can be mobilized in large quantities, this market is dominated by four main players : oil, coal, natural gas, and nuclear energy.

In the past few years, the very low levels of fossil fuel prices, the opening-up and internationalization of the markets, the appearance of new technologies, and preoccupations concerning environmental protection have significantly modified the competitive positions of these main energy technologies.

Thus, for each of them, to face up to the developments that will continue, new solutions have been developed to satisfy the energy demand.

To meet smaller needs, for less robust electric power grids and, especially countries which have noprimary energy sources, SMR's are well suited.

These new reactors should be safer, with higher performance, should show greater economy, and should respond precisely to the needs for future energy units, in the medium-power range.

As concerns nuclear energy, public opinion is also very sensitive to the still incomplete solution of the problem of the disposal of high-level radioactive wastes. New reactors design must at least minimize the quantities of such wastes produced.

2. THE SMR MARKET

The current growth of population and energy demand is dominated by developing countries. There are many places and applications where this increased demand will be best met by power plants in the SMR range, due to a small grid system or for application in a remote area or for a special purpose.

A half of the world primary energy consumption amounts is used as hot water, steam and heat. Only a few nuclear power plants are being used for heat applications (district heating, heat for indstrial processes, and seawater desalination). The need for potable water in some parts of the world is large, vital for sustaining development, and ever increasing. Clearly nuclear heat and power production could play a major and important role.

Nuclear power at present is used mainly for electrical power generation which only forms 30% of the energy market. There have been many studies on the use of SMRs for heat applications rather than electrical generation and some of these studies have shown the SMR option to be viable both technically and economically. Future expansion of nuclear application, beside addressing large power generation demand, may also come from more spread energy market involving smaller units for process heat applications and small scale power generation in remote areas.

3. INCENTIVES FOR DEVELOPMENT OF SMR'S

3.1. Stimulus to deployment of SMR's

Small and medium size reactor development has many incentives, some are economic and others are safety related. The motivation for these developments has included the need to enhance public acceptance of nuclear power. The simplification of designs should improve the transparency of their reactor safety. Another incentive for SMR development has been its suitability for the implementation of new design approaches. New innovative and evolutionary designs have been implemented in the SMR range. A passive safety approach has so far been the technology of small and medium reactors. SMRs have particular characteristics which can enable them to be economically viable in spite of losing the advantage of the economics of scale.

The incentives for the development of SMR's can be summarized as follows :

- Simpler design,

An SMR can be modularised more easily and constructed in a shorter time than larger plants, thus reducing construction costs (including interest during construction) and generating earlier revenues.

- Increased safety margins leading to a longer grace period,
- Lower severe core melt frequency and minimum accident consequences,
- Better match to small grid requirements,

SMR's can provide a better match to small grids or to a slow growth of energy demand. Taking into consideration the potable water demand and the corresponding energy requirement, a SMR would be a suitable candidate for a developing country starting its nuclear programme.

- Better use of nuclear industry infrastructure and manpower skills in countries with small nuclear programmes.
- SMR's could open up energy markets,
- SMR's can be used for process heat, desalination, district heating as well as power generation.
- Lower financial risk due to lower financing requirements per unit, shorter and better predictable construction schedule.

3.2. Objectives and requirements for SMR's

Development of SMR's could take place in a programme under the following general objectives which are applicable to reactors of any size but the particular aspects of SMRs help in meeting them.

- The size of reactor is appropriate to a geographical location, distribution network or application like cogeneration,
- SMRs are appropriate for remote regions with limited load. They are appropriate for utilities with small grid systems and for some dedicated applications such as desalination, district heating or process heat possibly in a cogeneration mode,
- It should be economic within the constraints of the other objectives.

SMR's are designed to reduce costs and modularisation allows a greater element of factory construction and assembly and is generally less expensive than work on site. It leads to shorter construction times and saving in interest during construction. The reduced capital requirements compared with large plants may well be attractive to some purchasers.

- It must be demonstrably safe and licensable.

3.3. Technology and Safety

Small commercial reactors are designed for different operating conditions, and hence involve somewhat different technologies than for large power plants. A number of small reactor concepts exists, but proven technologies available on a commercial basis are not evident. Specific technical characteristics needed for some small reactor applications could include a high core operating temperature or a reduction or elimination of high pressure systems. Further, a focus, on improved energy conversion efficiency for reactors of small size, is required.

Inherent safety features as well as the use of passive systems, for example natural heat dissipation and natural circulation for core cooling, should as far as possible be emphasized in the design. Complexity, cost and the need for manual intervention would be lowered by the absence of large numbers of active systems.

Small reactors have a potential for substantial simplification of the design, facilitating more economical operation and maintenance. A simple design allows for a corresponding simplification of the control procedures and significantly reduces the probability of operator errors.

Low stored energy is a characteristic of small reactors which permits use of passive safety systems for safety functions such as decay heat removal. This in turn eliminates the redundancy needs of active components. This is believed to improve safety, while simultaneously reducing plant costs (for components and building volume). The simplicity reached by these features will increase plan availability, lower the size of the operating crew, and reduce the probability of serious accidents.

The safety aspect has been and will remain essential. The consequences of a core melt accident being unacceptable, all measures must be taken to reduce the probability of such an accident to extremely low levels. More generally, nuclear safety based on regulatory measures must prevent incidents or accidents and ensure protection, under all circumstances, of the public, the environment, and the nuclear power plant staff.

4. ECONOMICS AND FINANCING

A most serious constraint for nuclear power introduction in developing countries is the difficulty in obtaining financing.

There are three main characteristics of nuclear power projects which make financing difficult:

- the high investment costs which may even exceed the overall credit limits of lending institutions for an individual developing country ,
- the longer construction time than for fossil fuelled power plants,
- the potential uncertainties in schedule and costs due to regulatory and public intervention, unforeseen construction problems and policy changes.

Several trends, developed worldwide, favour SMR's. Among these is the global trend to electricity supply deregulation. This has resulted in many large industries constructing power plants to serve their in-house power requirements, many of which provide both electricity and process heat. The surplus power is sold to external users. A further result of deregulation is a proliferation of power generation facilities operated by Independent Power Producers (IPP's), these units are currently dominated by combined cycle plants. IPP's place a strong emphasis on low total plant cost and fast cost recovery, and take advantage of the minimum electricity transmission costs often associated with small power plants.

SMR's must be competitive with alternate nuclear and non-nuclear energy sources in all important areas and gain public acceptance in order to be commercially viable. Hence, on a specific output basis, SMR's must meet or exceed the capabilities of large modern water cooled reactors, for example, their capital cost per MW output, their operation and maintenance costs per MW output, and the risk posed to the staff and the public on a per MW basis must not exceed those of large nuclear power plants. In addition, they must be economically competitive with both fossil and renewable energy producers in a large number of applications.

Innovative financing models like BOT (build-operate-transfer) or BOO (build-own-operate) are now being used in some cases for thermal power plants. A few attempts have been made to use them for nuclear plants but they have not been successful due to their slow cost recovery and to the problem of civil liability in case of nuclear accident.

5. MOROCCAN NATIONAL ELECTRIC POWER DEMAND

5.1. The National Office of Electricity : O. N. E

The "Office National de l'Electricité" (O.N.E) is the national utility in charge of electricity generation, transmission and distribution in Morocco.

5.2. Electricity Generation : Present and Future

By the end of 2000, the installed capacity in Morocco has reached a total of 4445 MW (13942 GWh of energy demand). This capacity is composed of :

- 5 thermal power plants (coal/oil) 2545 MW
- 24 hydroelectric plants 1175 MW
- 8 gas turbines power plants 615 MW
- diesel power plants 56 MW
- Wind power 54 MW

About 90% of fossil energy sources are imported.

Table 1 shows different sources of Electric Power generation current 1998, 1999 and 2000.

TABLE I. SATISFACTION OF ELECTRIC ENERGY DEMAND

Source of Energy Generation	1998	1999	2000
1 - ONE Production	GWh		
- Hydropower	1759 (14,1%)	817,2 (6,2%)	704,7 (5%)
- Thermal power	4998 (40,1%)	5650,5 (42,6%)	4333,8 (31,1%)
- Wind power	-	-	1,7
- Other National Producers	42 (0,3%)	27,2 (0,2%)	39,5 (0,3%)
- Importation from Algéria	11 (0,1%)	33,8 (0,2%)	94,9 (0,7%)
- Importation from Spain	705 (5,7%)	1811,9 (13,7%)	2267,8 (16,3%)
- Independant Power Production (IPP) - JLEC	4938 (39,6%)	4923,7 (37,1%)	6441,8 (46,2%)
- Wind power (IPP)	-	-	57,7 (0,4%)
Total Net Energy Demand	12453	13264,3	13941,9

Fig.1 shows the contribution of different power sources in the electrical energy demand current 1999 and the projected status for 2010.

Figure 2 shows the evolution over the next 15 years of the national energy demand. The growth rate during the next decade is around 6% per year.

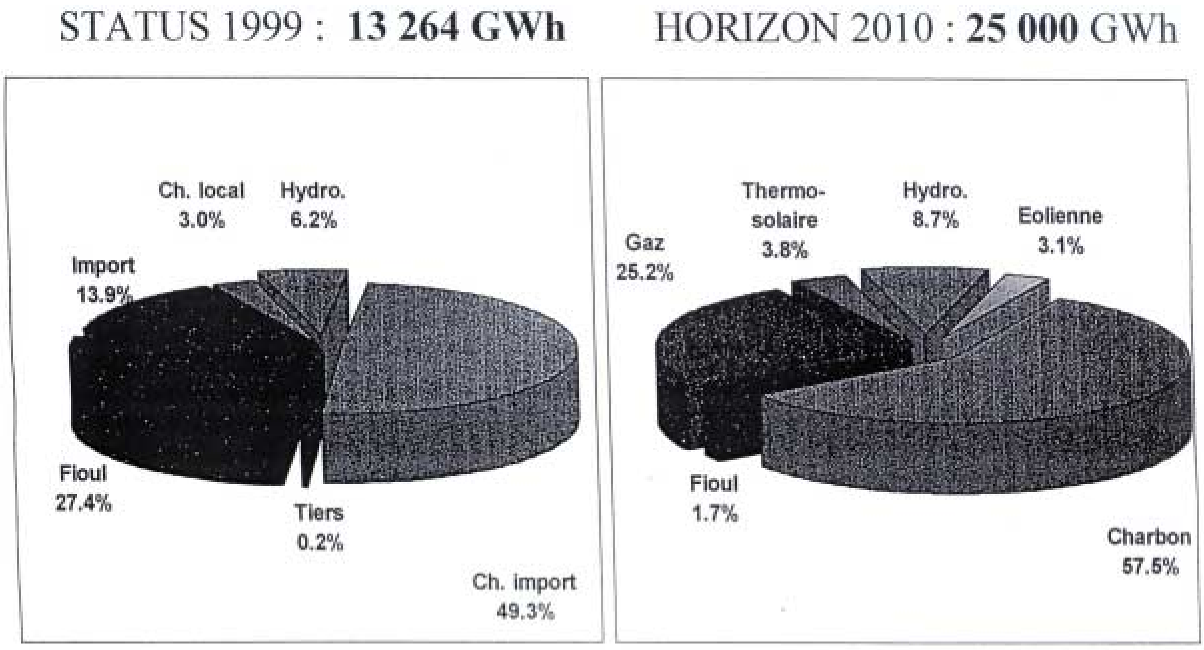
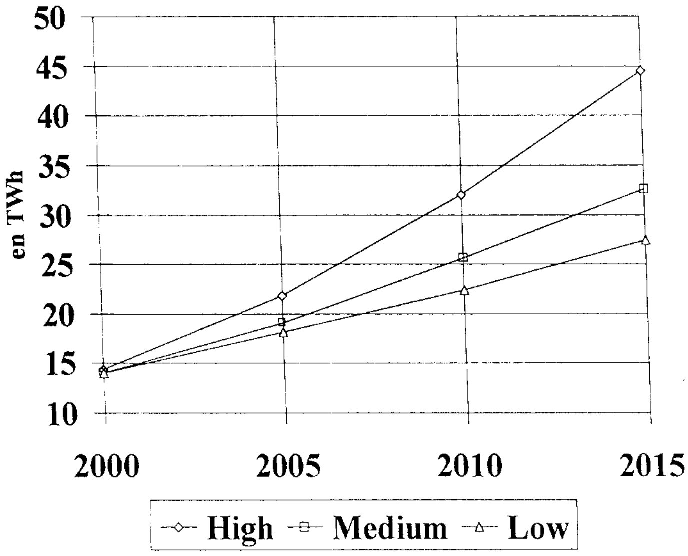


FIG. 1. Satisfaction of energy demand per source (medium scenario).



Average Growth Rates

	2000-2005	2005-2010	2010-2015
High	8 .5%	8 %	7 %
Medium	6 %	6 %	5 %
Low	5 %	5 %	4 %

FIG. 2. Evolution of the national energy demand.

Figure 3 shows the evolution of the installed, available and demand power projected during the next 10 years. At 2010, the total installed power will reach about 7000 MW.

As the ratio of the power of a single production unit to the total installed should be around 10%, a single Nuclear Power Plant (N.P.P) of 600 MW or two N.P.P units of 300 MW each could be inserted in the grid by 2010-2015.

5.3. Nuclear Power Plant Project of SIDI BOULBRA

In the framework of diversification of energy resources policy and in compliance with governemental orientations, the ONE is considering nuclear power option as an alternative to meet the needs of power generation.

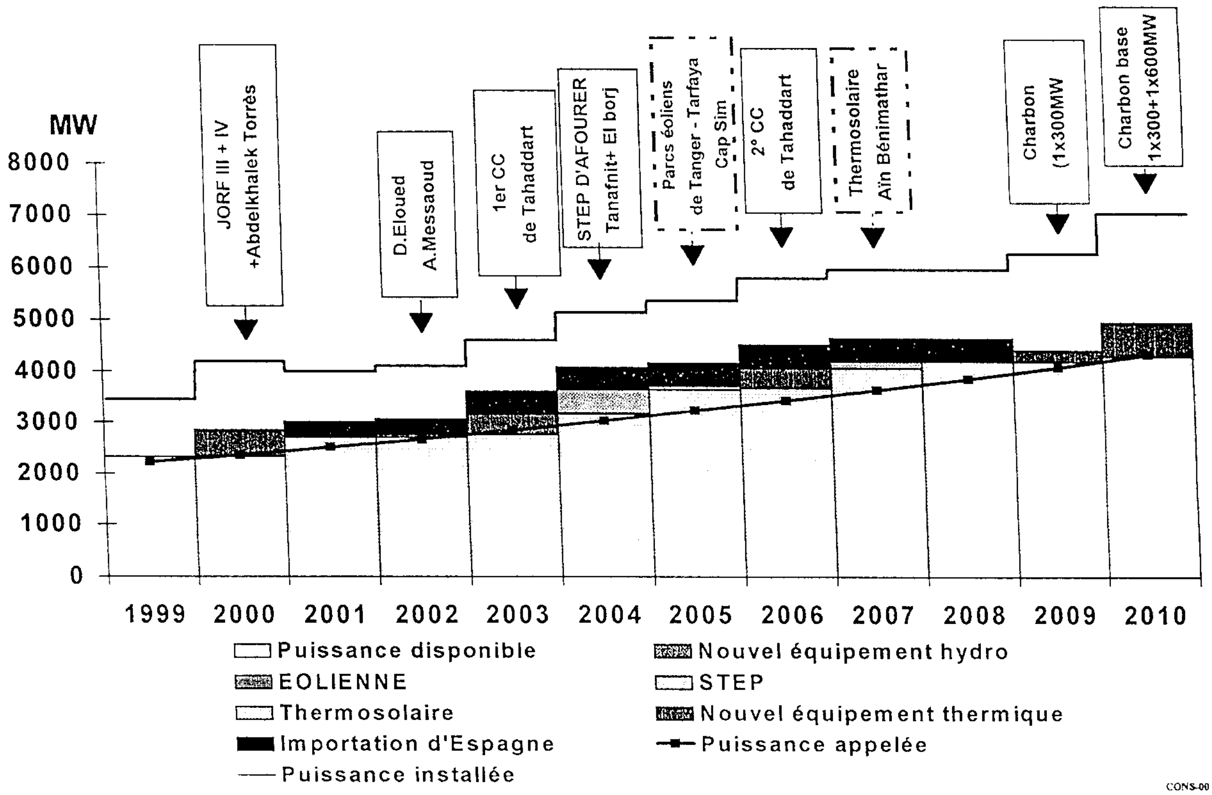


FIG. 3. Evolution of the installed, available and demand power (average scenario).

On the basis of this orientation, a feasibility study and siting of the first Nuclear Power Plant of SIDI BOULBRA in Morocco has been undertaken between 1984 and 1996 in the framework of the agreement concluded between ONE and the french company SOFRATOME under the support of IAEA. The conclusion of this study has shown that the nuclear power option could be a viable solution.

The chosen site of SIDI BOULBRA was recognized by the Agency as a qualified site able to receive nuclear power plant which can be operated in the required safety conditions.

Nevertheless, the feasibility studies based on large sizes reactors (900 MW) have led to the conclusion that the earliest date which can allow the introduction of the first unit is by 2015 and that's mainly for reasons related to physical insertion in the national grid.

The actual prospect leads to forecast a growth of foreign energy supply sources marked by shutting down of the unique national coal mine and the trend towards a saturation of hydropower potential equipment as well as the increasing of environmental pollution by fossil fuel plants.

For those reasons and due to the present state of reactor technology, it has been decided to update the above mentioned study in order to introduce the Small and Medium Reactor's (300-600 MW) which could be safely inserted in the national grid and will contribute to satisfy the energy demand in the horizon of 2015. Besides the technical aspect, other items should be reviewed such as the economic competitiveness, the financial issues, etc…

CONCLUSION

Small an medium sized reactors have good potential for future deployment specially in developing countries both for power generation and process heat application.

Small and medium reactor systems provide an attractive option for a wide range of applications worldwide. The design approach and design characteristics of the SMR's with regard to size, economics and safety appear to provide favourable conditions. Specific requirements on these topics will provide a common ground for the suppliers and interested users to further the discussion on specific design requirements such as performance, operability, maintainability and reliability. For successful deployment, the overall cost must be competitive with other alternatives, taken into consideration the main objectives.

An important aspect to the introduction of nuclear power in a developing country is a well planned and executed programme on the development of the required infrastructure according to the objectives of the programme.

The government commitment must be seen as stable and continuing for the long-term, since only a long-term nuclear power programme, involving several plants over a period of time, can justify the considerable effort and costs which will have to be devoted to national infrastructure development. Decisions on nuclear power projects can be made only in the context of a long-term economic electricity generation expansion plan, which will entrail long-term policy decisions for financing, conditions for international supplies of technology and nuclear fuel cycle services, and waste management and disposal.

ROMANIAN EXPERIENCE, POLICY AND CAPABILITY IN THE NUCLEAR POWER SECTOR BASED ON CANDU 6 REACTOR TYPE

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Abstract

Some 20 years ago, a joint Romanian - Canadian team prepared a feasibility study that led to the decision that the CANDU 6 nuclear power plant would be the basis for the Romanian nuclear program. The first nuclear site for five 700 MWe CANDU 6 units was developed at Cernavoda, about 160 km east of Bucharest. Cernavoda Unit 1 started its commercial operation on December 2, 1996. Romania has developed an industrial support structure within the electric utility consisting of nuclear fuel and heavy water facilities, a research & development institute, and a design engineering center. Cernavoda Unit 2 is expected to be commissioned in 2005.

1. INTRODUCTION

Situated in the East European region, Romania is a country of 23 millions inhabitants and an area of 238,391 square kilometers. Romania has 8 cities over 300,000 inhabitants, the biggest being Bucharest-the Capital, having 2.5 million inhabitants. The Gross Domestic Product (GDP) is 1,640 USD/capita and the actual energy consumption about 2400 kWh/capita.

Romania started its nuclear power program some 20 years ago. In 1977 the Romanian and Canadian Governments formally agreed to cooperate in the field of peaceful use of atomic energy. A joint team prepared a feasibility study that led to the decision that the CANDU 6 nuclear power plant, design of Atomic Energy of Canada Limited (AECL) would be the basic plant upon which Romania was to build its nuclear program. A tremendous effort was spent to build a national infrastructure to maximize Romanian participation in a large National Nuclear Power Program, which at that time envisaged the construction of more than twelve (12) CANDU 6 and three (3) VVER-1000 units.

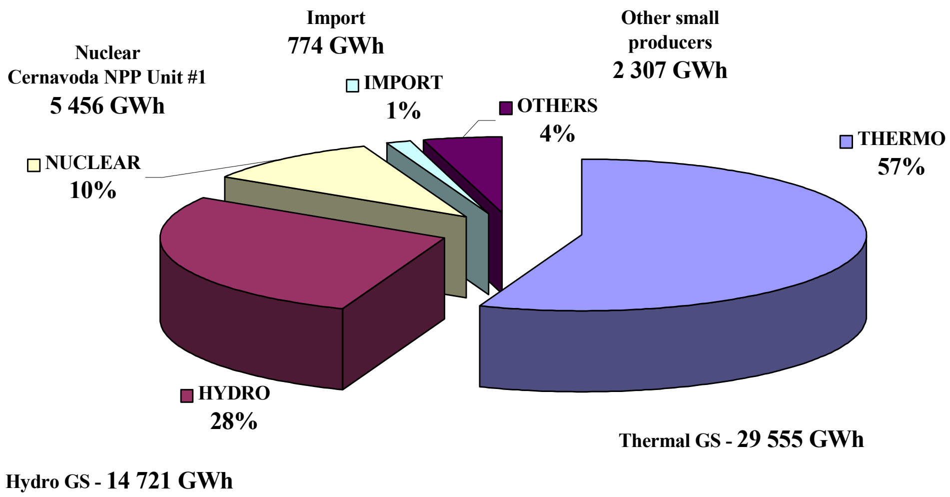
This first nuclear site, for five CANDU 6 units of 700 MWe capacity each, was developed at Cernavoda, in the south-east area of Romania (Dobrogea region) on the right side of the Danube River, about 160 km east of Bucharest. Construction work for Unit 1 started in 1980, under Romanian management, and in 1991 45% completed. After 1990, the program was reconsidered and following the recommendations of the International Atomic Energy Agency - IAEA, in 1990, the Romanian utility RENEL signed a Project Management Contract (PMC) with the AECL - ANSALDO Consortium (AAC), for the completion and commissioning of Unit 1. Cernavoda Unit 1 started its commercial operation on December 2, 1996 and until December 2000 it produced over 22.75 million MWh of electricity. In the process, it attained a capacity factor of 86.83%, which is very good as per international standards. The unit is professionally managed by Romanian specialists and has earned praise from foreign experts.

Within RENEL an industrial support structure was developed for the Cernavoda Project, represented by the Nuclear Fuel Plant in Pitesti, and the Heavy Water Plant, located in the southwest of Romania, near Drobeta-Turnu Severin. The "brain" support for the Romanian Nuclear Program was provided by the Nuclear Research Institute - ICN for specific Research and Development (R&D) activities and by the Center for Nuclear Projects Engineering and

Technologies - CITON for design-engineering activities. Romania also implemented a dedicated nuclear infrastructure, beginning with an educational system to industry and research-engineering capabilities. Specialized industries such as uranium mining, milling and concentrating in Compania Nationala a Uraniului were also developed.

FIG.1 POWER GENERATION IN ROMANIA IN *2000*

TOTAL - 52 813 GWh



The Unit 1 CANDU station at Cernavoda is as safe as any other in the world, a statement that is supported by the following arguments:

- the experience of advanced countries such as Canada, Italy and the United States is implemented in the Cernavoda Project, Canadian and Italian specialists were deeply involved in the construction, commissioning and initial operation of the station;
- Romanian staff, very well trained in Canada, with hundreds of thousands of training hours in the Canadian sister nuclear power plant, Point Lepreau, are successfully managing the business process;
- the international nuclear community support is very strong, with significant results in process improvement and in implementation of the “safety culture” among the Romanian specialists and Romanian authorities; PRE-OSART missions from IAEA, or training programs, seminars, co-operation with IAEA, OECD-NEA, WANO or COG are supplementary warranties to the safe operation of the Cernavoda NPP.

All this argument places Romania in an excellent position to optimize its power resources and nuclear infrastructure through the development of its nuclear sector based on the CANDU 6 reactor type. The convenience of the completion of Cernavoda Unit 2 Project was definitely demonstrated by the "Least cost power and heat generation capacity development study", prepared by SEP (Holland), Tractebel (Belgium) and EDF (France) under PHARE Energy Program Management Unit. This is evidenced and considered by Romania’s Strategy for Energy Sector - Middle Term (2000-2005) and Forecast (2010), which is a component of the National Development Strategy. Cernavoda Unit 2 NPP is considered by the new Government, elected in December last year, “the most efficient new capacity to be commissioned until 2005” and that the horizontal infrastructure is available (nuclear fuel plant, heavy water, R&D). Recent developments of oil price on the international market are strengthening Unit #2's position as priority number one for the Romanian Power Sector and open the door for Unit #3 completion.

The nuclear option represents also a good opportunity for Romania to reduce polluting emissions, even the country is not under the pressure of Kyoto Protocol, having a credit of 8% for CO_2 emissions.

The Romanian public attitude towards nuclear power has remained very positive and the Romanian experts, including those from non-governmental organizations, take advantage of any opportunity to explain the benefits of the nuclear energy and to discuss the key elements of awareness, such as radiation or waste management and disposal.

For further development and good operation of the nuclear sector, Romania receives also an important support from international community as IAEA, WANO, COG, EURATOM and other organizations.

COMMENTS ON THE INTRODUCTION OF SMRs IN DEVELOPING COUNTRIES (THE ASIA PACIFIC REGION)

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Abstract

Small sized reactors have been constructed for R-D or for specific purposes since many years. The practical experience in the design and exploitation of SMRs is now encouraging their new peaceful applications, such as electric generation, sea water desalination, thermal supply etc. The demands for these applications would be diverse in different socio-economic and geological areas.

SMRs fit well to the moderate economic scale of a developing country and are flexible to adjust to the electrical grid. The scale of SMRs easily facilitates the financial arrangement, construction and operation for a NPP project. The technological reliability of SMRs must be also very helpful in gaining public acceptance. While the economical benefit and safety issues of the SMRs still remain under consideration, the low capital cost of SMRs and the enhanced nuclear safety would make them more attractive and promising in the near future for many developing countries.

1. NUCLEAR POWER PROGRAM IN ASIAN PACIFIC (AP) DEVELOPING COUNTRIES

While North America and Europe are slowing down in nuclear power generation, Northeast Asian countries such as Japan and the Republic of Korea (ROK), in opposite, are steadily developing their nuclear program. Although China and India, the two giant developing countries, are widening the construction of nuclear power plants (NPP), most other AP developing countries are still moderate in nuclear power introduction. However, they are participating actively in a regional cooperation agreement (RCA) for peaceful application of radiation and isotopes. Among ASEAN countries, Philippines has approved an NPP; Indonesia and Thailand are studying for the introduction of NPPs. Viet Nam is planning a nuclear power program.

We are focusing below on the status of the nuclear power programs of such developing countries, with emphasis on the possibility of application of small and medium sized reactors (SMR).

2. NUCLEAR POWER AS A CLEAN AND STABLE ENERGY OPTION

The lack of natural resources is a common reason for a strong interest of the Northeast Asian countries in the nuclear power program. The goal of their efforts is to gain independence from any economic or energetic crisis. The ways they have chosen are more or less different. By advancing LWR technology, Japan keeps on recycling spent fuel to reduce radioactive waste. After introducing both light and heavy water technologies, ROK is focusing on the Korean standard PWR and accepting disposal storage without reprocessing. Nowadays, nuclear energy shares as much as 35% and 43% of total electric generation of Japan and ROK, respectively [1]. After 30-40 years of experience, these countries are now keeping a stable tendency of nuclear power development, despite a slight increase in the anti-nuclear opposition, which sometimes causes some difficulties in the new site selection.

Other developing countries in Asia, including China, are considering nuclear power as an alternative for industrialization in order to meet high growth of electricity demand without green house effect. There are 3 reactors in operation and 8 others under construction now in China. This country is pursuing a very fast growing nuclear program and expects to have 16-18 reactors with total capacity of 16GWe in about 20 years, which would be the best way to reduce very large portion (82%) of its fossil electricity generation [2].

In ASEAN there is no operating NPP at the moment. Philippines was the first to build a NNP with 2 PWR units. However, this project has been abandoned due to political reasons and the uncompleted blocks are to be reconstructed into a non-nuclear thermal station. Even the country is still interested in nuclear power, it will take time to change the situation.

Of all the energy consumption in Indonesia for 1997, oil contributed 42%, followed by gas 34%. A feasibility study for nuclear power was completed in 1996, and a site has been selected in Muria Peninsula in Central Java for a set of NPPs, which would have a total capacity of 7 GWe and would be implemented as a BOO project [2,3]. Due to the recent financial crisis, this program has been delayed.

In Thailand, 36% of the electricity consumption was based on coal and 28% based on gas for the year 1998. This country had studied a nuclear power program more than 20 years ago [4]. However it was stopped, probably due to the same regional financial crisis. Indonesia as well as Thailand would go back to their nuclear power program in the near future, after their economy is stabilized. The generation mix of AP countries is reviewed in the table 1.

TABLE I. GENERATION MIX IN AP COUNTRIES FOR 1999

Country	Fossil,%	Hydro,%	Nuclear,%	Others,%	Total, Bill.kWh
Japan	59.3	8.8	31.5	0.4	972.7
ROK	63.0	2.5	34.5	0.0	212.0
China	82.0	16.9	1.1	0.0	1036.4
India	82.2	15.3	2.5	0.05	425.7
Indonesia	84.0	11.0	0.0	5.0	60.0*
Philippines	65.5	19.0	0.0	15.5	58.0**
Thailand	69.7	15.8	0.0	15.5	92.1**
Viet Nam	41.3	58.7	0.0	0.0	23.7

* *as for 1997 and ** for 1998* (see [3]).

TABLE II. THE GENERATION MIX FOR THE BASE LOAD AND THE HIGH LOAD AROUND THE YEARS 2016-20

Type of generation	Installed capacity(MW)		Share (%)	
	Base load	High load	Base load	High load
Hydro	12890	12890	39.0	33.7
Gas	10785	10785	32.6	28.3
Coal	4000	6300	12.1	16.5
Imported	4000	4000	12.1	10.5
Nuclear	1200	4000	3.6	10.5
Geothermal	200	200	0.6	0.5
Total	**33075**	**38176**	100	100

In Viet Nam, the share of different generation sources is 58.7, 22.7 and 18.6%, respectively, for hydro, coal and gas/diesel [5] for the year 1999. Hydro power, as well as oil and gas are limited, and coal generation is no more welcome. Viet Nam now is studying the possibility to import energy or to go nuclear. The research result of INST (VAEC, MOSTE) and Institute of Energy (MOI) shows that after 2015, depending on how fast the GDP growth rate is, (6.5% and 8.2%), Viet Nam has to import electricity and it will need from 1,200 to 4,000 MWe nuclear power in addition to other sources. The forecast for 2016-2020 of the energy consumption in Viet Nam [5] for a base load (for the GDP rate of 6.5%) and for a high load (the GDP rate of 7.0%) is shown in table 2.

The limit of conventional resources as hydro, oil and gas is a common problem of AP developing countries. Moreover, the global problem of the green house effect rejects the coal and oil electricity. Renewable energy resources such as solar, wind, geo-thermal and bio-gas… are under research. They all are expensive and cannot bring about any significant capacity. For the upcoming 50 years the only proven alternative to meet the demand of the mankind would be nuclear energy. Certainly, radioactive waste management remains the most sensitive problem, which has to be solved in the near future. Public acceptance of NPPs depends also on the reliability and safety of the nuclear technology, which has been improved significantly after the Chernobyl accident.

3. OPPORTUNITY OF SMR APPLICATION

Small sized reactors used to be constructed for research and development purposes or specific applications. Examples of the first peaceful applications are Russian atomic icebreaker constructed in 1959 with a power of 32 MW and the thermal stations with reactors type ATU-15 (15MW) serving for Arctic area during 60's. The good experiences on SMRs design in developed countries such as Russia and USA are encouraging their new applications.

TABLE III. ADVANTAGE AND DISADVANTAGE OF THE SMRS

Advantages of SMRs	Disadvantages of SMRs
-Lower absolute capital cost, with smaller financial burden. -Distribution of economic risk through several smaller plants. -Better controlled construction schedule due to the less on-site work and smaller size of components. -Earlier introduction of nuclear power will give environmental protection vs. fossil fired units. -Lower absolute heat rejection permits better adaptation to cooling capacity and extends the number and location of possible sites. -Better fit to smaller and weaker grids and lower requirements on grid. -Fit to low load growth rate situations. -Better past performance records than for larger plants. -High degree of shop fabrication and potential for series production. -Earlier introduction of nuclear power with potential for longer term technology transfer if the introduction does not come too early.	-Larger units have lower specific capital cost per kW(e) and better economic viability. -In many cases non-standard design, with provenness, licensing, and commercial availability questions. -Break in normal technology development for industrialized countries that are used to larger units. -More limited possibilities for domestic participation due to trends for shop prefabrication but the smaller size of components vs. larger nuclear power plants can bring an increase in domestic participation. -Domestic participation targets and seismic design requirements can work against construction time. -Essentially the same infrastructure requirements as for big plants.

In developing countries the demands of SMRs are raised for electrical consumption, water desalination, thermal supply, etc. In different areas the demand would be specifically emphasized. In Africa and Arabia water desalination would be of the high priority; however, some other countries including European ones are interested more in introduction of thermal stations. In Asia or Latin America electrical supply would be a more important issue. As the economy is growing up in Asia, the other needs such as the water desalination and the thermal consumption will also be considered.

SMRs would fit to any economical growth scale of developing countries. They are especially convenient for isolated areas. The safety and reliability of this technology makes it easier to convince the public. This will also facilitate the construction, operation and management of a project, even when countries have some lack of technical manpower.

During the 80's IAEA has initiated different projects for SMR feasibility [6-8]. Results of these studies for the advantages and disadvantages of SMRs are shown in table 3.

However, since the TMI and Chernobyl accidents, the reputation of nuclear energy has suffered seriously, that the interest in SMRs has been interrupted for some time.

The problems may be raised on economical aspects of the SMRs in comparison with the nominal ones. Safety as a common concern of the public still remains a subject for discussion. Those problems may be solved by standardization and serial fabrication, which would facilitate the installation works on-site and reduce the construction duration.

In the new era of industrialization under the environmental sustainable concept, when NPPs may have a new opportunity, it would be foreseen that SMRs are very attractive and promising for developing countries in the new century. Medium sized reactors with capacity from 600 to 900 MWe are indeed large enough for the electricity consumption of many developing countries and they are certainly accepted more easily than larger sized reactors. Such medium sized reactors have got very rich experiences of successful exploitation in nuclear developed countries.

Smaller sized reactors have poorer experiences of world NPP exploitation. The exception is the Russian VVER-440 MWe, supplied during 70's and 80's for East European countries: Bulgaria, Czech, Finland, former Eastern Germany, Hungary, Slovakia, Slovenia, and Ucraina and Russia itself which had quite stable operation [1]. The post-Chernobyl philosophy demanded higher safety requirements for the reactors of this generation. However, the experience of VVER-440 serves a good example for the SMR application. In 90's Canada planned to design CANDU-3 with a similar capacity at 450 MWe [9].

Nowadays India is the only country developing NPPs with serial small sized HWRs of 220 MWe. There are 14 reactors in operation and 2 others under construction in this country, which would contribute to the consideration of the small sized reactor application.

Japan is now initiating renewal of the study on SMRs with advanced performances and competitive cost benefit [10]. This new trend may attract other countries in this region including developing ASEAN, potentially looking for the nuclear power in the new century.

CONCLUSION

For developing countries, medium sized reactors are probably the most promising for introduction. Moreover, medium capacity is often large enough for many countries with limited grids. Small sized reactors will be also very attractive for the beginning of the nuclear program in a developing country. Small reactors would be useful not only for developing countries, but also for developed ones looking for different specific applications. Small sized electricity stations will also fit to isolated areas, such as archipelagos and mountains. Although small sized reactors can not replace medium or large sized reactors with proven cost benefit, some specific advantages of the SMRs and the improvement of their performances in near future would make them more attractive and will contribute to the peaceful use of nuclear energy for the mankind.

ACKNOWLEDGEMENT

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PANEL DISCUSSION
(Summarized by Debu Majumdar)

This panel on incentives for introduction of small and medium reactors in developing countries generated a lot of questions. The purpose for this session was to hear directly from the developing countries about their interests and problems in introducing these reactors. What incentives are needed? In the keynote speech, Ms. Jan Murray of the World Energy Council said the following: “virtually all of the 21st century’s population growth will occur in developing countries. If nuclear energy cannot play a role there, it is destined to be a sideshow.” So it is important to make sure that the developing countries have nuclear power. That is where the needs and the demands are.

A starting question was why there is no order for a small and medium reactor even though these are being advertised as economic, safe, proliferation-resistant, etc. It was said that a suitable small reactor was not available and that unless the reactor is being used by the developer country, no developing country would buy it. For developing countries proven technology is very important. It is stressed that the two partners in the game - the supplier and the user - should try to be together: Most of the suppliers from the advanced countries, having strongly interconnected grids, are going for economics of scale and need larger unit sizes of the order of 900 MW(e) and higher. But for several countries, as for example in Africa, 1500 MW(e) is more than the total installed capacity in the country. So it would not be possible to use a plant of this size in Africa. The smallest available in the market is around 300 MW(e) size. It seems the suppliers are waiting for the market and the users are waiting for a reference plant with a good record. It is a circle that has to be broken, if the situation were to be improved.

A couple of panellists mentioned about financing and infrastructure requirements. The question debated was if the developing countries must build up their infrastructure first before introducing an SMR, or, a small- or medium-size reactor should be developed that would lead to easy financing and facilitate gradual strengthening of available infrastructure. From the Bulgarian experience, it is important that a country should start development of appropriate infrastructure years before starting a nuclear project, such as education, operator training, training in maintenance procedures, for safety principles and so on because after that it is too late.

Many reactor concepts were presented at the Seminar, but very few have been built and it was said that the reactors would be ready in the next 10 or 20 years. However, according to Pakistan the developing countries have problems right now, not after 10 years or 20 years.

In Japan there are three 300 megawatt light-water reactors. These were constructed at the beginning. They also built several 600 megawatt light-water reactors, which are still operating. So there is another option: to take these same designs to the developing countries but to construct them in indigenous way - to use the developing country’s technology as much as possible. That will lead to cost reduction. This is the way China has taken their nuclear program; they constructed one 300 MW(e) unit but they ordered fabrication of the reactor pressure vessel from Japan, which is an important component. The 300 or 600 MW(e) reactor design is already there, so the cost of making a detailed design is already covered. And even for these reactors one can introduce new technology, for example I&C can be digital. So an idea is to take the existing design of 300 or 600 megawatt light-water reactors already operating in another country and use the developing countries own technologies as much as possible. Then one can build up from here.

Egypt tried three times to produce nuclear power. The first attempt was in the 1960's and at that time they were discussing co-generation plant - 150 megawatt electricity and 20,000 cubic metre per day desalted water; the second attempt was in 1974 and the last one was in 1984 and in all these attempts they were quite flexible and they did not try to impose too much requirements of their own on existing designs. All these attempts had failed basically for external reasons and sometimes for internal reasons as well. Thus for developing countries to be able to achieve their programmes, there is a need to have an international political environment that supports their efforts. For example, proliferation resistant design is receiving emphasis now, but there is no precedent that a nuclear power reactor has been diverted to military purposes. So it is necessary to create an international environment, which will help nuclear energy to proceed in developing countries.

Development of a new type of reactor that is attractive to both the developing and the industrialized countries simply takes time, and the first step is to accumulate the requirements for these reactors. There is one solution that may help developing countries immediately without the need for the very sophisticated infrastructure; this is the concept of a black box reactor. They may be transported to the country, which needs energy and operated there and the electricity can be sold to the country. So there are immediate solutions but it is not clear if developing countries want this kind of technology. Until now they always wanted technology transfer included and this simply takes a longer time.

It is believed that the incentive for developing countries is the same incentive that the developed countries had in taking on nuclear power. They wanted cost-effective power and they wanted reliable supply. That's the reason North America, including Canada and the US, is going back to nuclear power - because of reliability of cost-effective electricity supply. The reason nobody is buying these new reactors or any reactors is because they are too expensive. So economics is the key and if the economics is not right, then nuclear energy is not going to be successful.

Russia has many projects of small reactors in different stages of development. Some are practically ready for construction, or limited efforts are required to finalise and licence the design. The small barge-mounted reactor is planned to be implemented within the current decade in Russia and is promising for application in developing countries. Russia also has several existing and new proposed reactors in the medium-size range. Russian WWER-440 has been used for a long period in Russia and Eastern Europe. This is a proven but rather old design and its further application in developing countries seems to be questionable. New proposal in this range is represented by evolutionary design of WWER-640. Russia is going to build this reactor in the nearest future. Right now it is still not a proven design and this is a problem for construction. However, joint efforts by suppliers and users for demonstration of a medium-sized reactor concept in a developing country would be a good idea.

Pakistan stressed that co-operation between developing countries themselves is important. Pakistan and Peoples Republic of China have a very good example of that. The PRC developed their own version of Westinghouse PWR (300 MW(e) Qinshan NPP) and found a ready customer in Pakistan. It was made possible because the exporting country had first built a demonstration reactor of its own and had successfully operated it for some years. The nuclear power plant they built in Pakistan (at Chashma) has been very successful. It is an example of cooperation between the developing countries themselves for the development of SMRs that suit their requirements.

France emphasized that there is a need for a strong safety authority in the developing countries and it is not a specific case for the SMRs but it is needed for all the nuclear installations. The nuclear safety authority should be competent and the staff of the safety authority should be stable and independent in order to assure fulfilment of its responsibilities.

India suggested that one has to take a long term view for economic viability instead of a short term view due to the scarcity of capital because power reactors have been operating economically in many countries for the last 50 years.

Finally, all participants and the panellists were thanked for their lively and stimulating comments and discussion. This panel was designed to address the incentives and issues of deploying SMRs in developing countries. The incentives are as follows: SMRs are more suitable for smaller distribution systems, easier to finance, and require lower capital costs. They also employ inherent safety designs, passive safety features, and design simplicity. With regard to the issues and challenges, the following items are raised. Economic competitiveness and proven technologies are requirements from developing countries. The serious constraints for introduction of nuclear power in developing countries are: financing, investment risk due to potential uncertainties in licensing, and public acceptance.

APPLICATIONS AND EXPERIENCE

(Session 9)

Chairpersons

Y. Oka
Japan

M. Sultan
Egypt

OVERVIEW OF SMR APPLICATIONS

J. KUPITZ
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Abstract

Energy demand will continue to grow worldwide, with a faster growth in developing countries; consequently a wide range of options needs to be maintained for energy production and utilization in order to achieve environmental and economic objectives. This paper describes the various possibilities of non-electric applications of nuclear energy

The temperature requirements for these applications vary greatly from low temperature heat for district heating and desalination to high temperature process heat for coal gasification and hydrogen production. Processes requiring temperatures of up to 300°C can be supplied by water-cooled reactors while breeders may be applied to processes requiring up to 540°C. The high temperature gas cooled reactor can provide process heat temperatures of 950°C. Thus, nuclear energy has the potential to provide heat for many of the world's industrial heat application processes. Small and medium sized reactors are more suited for this purpose because of their reduced output capacity and grid size requirement. The attractiveness of using nuclear energy for non-electric applications, compared to fossil energy is the long-term stability of nuclear fuel prices in contrast to the rising prices of fossil fuels, increase of energy independence, decrease of the environmental impact and contribution to national technology and manpower development.

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. The prospects for the future will depend on where and how the demand characteristics of the heat market can be matched by what nuclear reactors are able to offer.

1. INTRODUCTION

In October 1956, the first nuclear power reactor at Calder Hall, United Kingdom, came into commercial operation. It provided electricity to the grid and heat to a neighboring fuel reprocessing plant. After more than 40 years, the four 50 MW(e) units of the Calder Hall station are still in operation. In Sweden, the Agesta reactor provided hot water for district heating to a suburb of Stockholm for a decade, starting in 1963. These examples show that the application of nuclear reactors for providing heat to industrial processes as well as for urban district heating started at a very early date, practically at the same time when nuclear power reactors were first applied to electricity generation.

Since these early days of nuclear power development, the direct use of heat generated in reactors has been expanding. Countries such as Bulgaria, Canada, China, the Czech Republic, Germany, Hungary, India, Japan, Kazakstan, Russia, Slovakia, Sweden, Switzerland and Ukraine, have found it convenient to apply nuclear heat for district heating or for industrial processes, or for both, in addition to electricity generation.Though less than 1% of the heat generated in nuclear reactors worldwide is at present used for district and process heating, there are signs of increasing interest in these applications.

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.

2. THE ENERGY MARKET

About one third of the total energy consumption in the world 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 strong increase in the generation of nuclear electricity is expected during the next couple of decades. The real need for energy is in the developing countries, which in many cases have limited grid capacity. SMRs are well suited for these grids as for provision of electricity to remotely located areas without interconnected grids. SMRs are also well suited for industrialized countries with low growth rate projection.

In the transport sector, practically no application of nuclear energy is foreseen, except indirectly through the increased use of electricity, e.g. transport by rail.

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.

3. CHARACTERISTICS OF THE HEAT MARKET

Transport of heat is difficult and expensive. The need of 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 networks generally have installed capacities in the range of 600 to 1200 MW(th) in large cities, decreasing to approximately 10-50 MW(th) in towns and small communities. Exceptionally, capacities of 3000-4000 MW(th) can be found. Obviously, 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, and this approach is also applied in Austria, Belgium, Germany, France, Switzerland, Norway and the Netherlands, though to a much lesser degree. 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 back-up 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 100-150^{o}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 fueled by gas, oil, coal or wood.

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 per cent. Nevertheless, the supply of energy has an essential character. Without energy, production would stop. This means that a common feature of practically all industrial users is the need for 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 then 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 1-300 MW(th) range, 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 demand of large industrial users has usually base load characteristics.

The temperature requirements depend on the type of industry, covering a wide range up to around 1500^{o}C. The upper range above 1000^{o}C is dominated by the iron/steel industry. The lower range up to about 200-300^{o}C includes industries such as seawater desalination, pulp and paper or textiles. Chemical industry, oil refining, oil shale and sand processing, coal gasification, are examples of industries with temperature requirements of up to the 500-600 ^{o}C level. Non ferrous metals, refinement of coal and lignite, hydrogen production by water splitting, etc. require temperatures between 600 and 1000^{o}C.

All industrial users who require heat, also consume electricity. The proportions vary according to the type of process, where either heat or electricity might have a predominant role. The demand for electricity can either be supplied from an electrical grid, or by a dedicated electricity generating plant. Co-generating electricity and heat is an attractive option. It increases overall energy efficiency and provides corresponding economic benefits. 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, and which in turn would serve as a back-up for assurance of electricity supply. Such arrangements are often found to be desirable.

4. CHARACTERISTICS OF NUCLEAR HEAT SOURCES

From the technical point of view, nuclear reactors, as has been mentioned before, are basically heat generating devices. 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 measures, such as intermediate heat exchanger 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, up to about 300^{o}C are obtained in light and heavy water reactors, up to 540 ^{o}C in liquid metal cooled fast reactors, up to 650^{o}C in advanced gas-cooled reactors, and up to about 1000^{o}C in high temperature gas-cooled reactors.

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. Whatever heat is not needed to supply the heat demand, can be used for electricity generation, which 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 70-80 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 close to 100% levels required by most large heat users. Consequently, as for fossil-fueled heat sources, redundancy is needed. Multiple unit co-generation power plants, modular designs, or back-up heat sources are suitable solutions.

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 served admits 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 have to be also 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. 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 power reactors (SMRs). Current design SMRs are not scaled down versions of large commercial reactors, and they are intended to be economically competitive.

Siting of nuclear plants has become a major issue, even in those countries which are proceeding with their nuclear programmes by initiating new projects. Building additional units at existing nuclear sites has been standard practice lately, and opening up new sites for nuclear plants are a rare occurrence. Economics promote siting as close as possible to load centers even for electricity generating power plants, and for co-generation or heat-only reactors it is practically a necessary condition to be fulfilled. 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, which are getting more and more demanding. Advanced reactor designs, in particular in the SMR range with improved safety features, could be perceived as acceptable for close siting by the public, 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.

5. PROSPECTS OF 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. The prospects for the future 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.

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 byproduct. 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 are given.

For small co-generation reactors corresponding to power ranges of up to 300 MW(e) and up to 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, but in addition, 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. 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) was initiated in Russia in 1983/85, but later interrupted. There are several designs being pursued, and the feasibility for construction of a 200 MW(th) unit in China is currently reviewed. 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 costs are very high 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.

Intermediate temperature process heat can be provided directly or in a cogeneration mode. Typical loads involve supply of process steam for large industrial operations such as heavy oil recovery, refineries and chemical processing plants as well as process industries such as textiles, building materials etc. The higher heat rejection temperatures, relative to district heating, for these applications cause a larger reduction in the efficiency of electricity production, but the load can be supplied in a cogeneration mode by water reactors as well as gas and liquid metal cooled reactors.

Process heat for high temperature applications can be provided in a dedicated heat only mode or by parallel supply of electricity and process heat. Typical high temperature process heat applications include the following:

- Steam reforming of methane for hydrogen and methanol production
- CO_2 reforming of methane for hydrogen
- Thermochemical water splitting for hydrogen production
- Coal conversion to liquid and gaseous fuels.

These processes require higher temperatures available from gas cooled reactors, and in some cases from liquid metal cooled reactors.

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, which by definition constitute the district heating market. 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 or even eliminate 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 in operation already supply process heat to industrial users. The largest project is in Canada (Bruce, heavy water production and other industrial/agricultural users). 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. It would, however, involve additional costs, which would have to be justified by a cost/benefit analysis. 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 an existing and interested industrial user, or even better, proceeding with 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, offers the best chances for overall optimization, and hence, has the most promising prospects.

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. These reactors still require some development/demonstration efforts in order to achieve commercial maturity. Should they achieve economic competitiveness as expected, their prospects seem to be very 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, but this is something yet to be proven in practice. 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 of applying nuclear energy for district and process heating are closely tied to the prospects of deployment of SMRs. As a result of a recently performed market assessment for SMRs, 60 to 100 new units are expected to be implemented 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 very well supply heat in addition to electric energy, while a few are expected to be heat-only reactors.

CANADIAN EXPERIENCE IN DEPLOYING SMRs AND PLANS FOR FUTURE DEPLOYMENT

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Abstract

Climate change, air quality and conventional energy costs are emerging as increasingly important issues. The application of nuclear technology to meet energy requirements is an important potential option to address these issues, particularly for countries where nuclear power plants have not yet been introduced. Since SMRs are far easier to introduce into new national nuclear programs, it is instructive to consider the characteristics that are important for successful implementation. Canada has introduced the CANDU 6 SMR into a number of different countries and operating conditions.

In addition to Canada, CANDU 6 SMRs are operating or are under construction in Argentina, Romania, the Republic of Korea, and China. As part of these projects, Canada has assisted with all aspects of the nuclear program, including project financing, regulatory & licensing affairs, establishment of a project and operating infrastructure, localization and technology transfer, cooperative R&D programs, and operations support.

While investments in nuclear power are excellent long-term choices, some countries need financing assistance and financial guarantees to launch nuclear projects. AECL's approach is to draw on our project experience, network of partners, and our relationship with the Canadian government to assist with this aspect of a project. For example, for the Qinshan Project in China, AECL involved Bechtel (USA) and Hitachi (Japan) in the project team.

For the future, nuclear power designs must respond to demanding requirements. To meet the emerging market requirements related to cost, schedule and project risk, an evolutionary Next Generation CANDU SMR product is being developed. This product builds on the strengths of the current SMR CANDU 6.

For the Next Generation CANDU system, AECL has adopted the evolutionary approach, accommodating significant changes to design to significantly improve economics, performance and safety margins, while retaining the essential characteristics.

Based on these principles, a conceptual design program for the next generation SMR CANDU was initiated at the beginning of year 2000.

1. INTRODUCTION

The expansion of nuclear power will lead to large improvements to both the local and world-wide environment, with the elimination of airborne pollutants and greenhouse gas emissions. However, the adoption of large nuclear plants is often impracticable for countries that are initiating nuclear programs. In virtually all cases, even developed countries started with SMRs before evolving to larger systems. There-fore, the SMR has a unique role to play in securing the future role of nuclear power. In addition, technology transfer in the form of the supporting technology and science, operational experience, regulatory experience, and basic infrastructure must also be addressed for implementation of a successful program. Canada has a large domestic nuclear program, with 22 reactors, which consists of 10 SMRs and 12 larger reactors. Two of the SMRs are CANDU 6 reactors. The CANDU 6 is a 700 MWe class nuclear power plant designed and developed by Atomic Energy of Canada, Limited (AECL) for domestic use and for export to other countries. This design has successfully met criteria for operation and redundant safety features required by Canada and by the International Atomic Energy Agency (IAEA).

In addition to Canada, CANDU 6 SMRs are operating or are under construction in Argentina, Romania, the Republic of Korea, and China. As part of these projects, Canada has assisted with all aspects of the nuclear program, including project financing, regulatory & licensing affairs, establishment of a project and operating infra-structure, localization and technology transfer, cooperative R&D programs, and operations support.

In this paper, we summarize Canadian experience with SMR deployment and plans for future deployment.

2. CANDU 6 CHRONOLOGY

The origins of the CANDU 6 design go back to the 1970's, when an updated, single -unit adaptation of the 8-unit Canadian Pickering station was developed. The first four CANDU 6 units all entered service in 1983-84: Point Lepreau and Gentilly 2 in Canada, Embalse in Argentina, and Wolsong-1 in Republic of Korea.

Subsequently, further CANDU 6 units have entered service in Romania (Cernavoda-1; 1996) and Republic of Korea (Wolsong 2,3, 4; 1997-99). Additional units are presently under construction in China (Qinshan Phase 3, Units 1 and 2) and Romania (Cernavoda-2).

Thus, CANDU 6 SMRs have been successfully introduced in a number of different countries, operating settings and regulatory jurisdictions. In addition, CANDU 6 units have achieved excellent operating performance including situations where the unit represents the only nuclear asset of the owner-operator, i.e., where there is less supporting infrastructure. As discussed below, this success arises from both the CANDU 6 design and from the network of operating support available to CANDU owners.

3. CANDU 6 DESIGN SUMMARY

The CANDU 6 is a pressure tube type nuclear power plant, which uses heavy water as a coolant and moderator. Figure 1 shows a 3D CADDS model of the plant.

The basic modular unit of the CANDU reactor is the fuel channel. Each fuel channel contains 12 short fuel bundles and is positioned horizontally in a large tank, called a calandria. The calandria is filled with low pressure and temperature heavy water to moderate the neutrons. The power output from the reactor depends on the number of fuel channels. For example, the 700 MWe class CANDU 6 reactor contains 380 fuel channels, while a 1000 MWe class reactor contains 480.

This configuration results in "high neutron economy", such that more of the neutrons produced are avail-able for sustaining fission. A major reason for this is that heavy water has a low cross-section for neutron absorption compared to light water. This means that CANDU reactors are highly efficient burners of fissile material and converters of fertile material. All CANDU reactors operating today are fuelled with natural uranium, although the high neutron economy means that several other fuels can also be used.

Another key feature of CANDU reactors is on-power refuelling. A fuelling machine pushes fuel in from one side of the fuel channel, and a similar machine accepts the spent fuel bundles that are ejected from the other end, while the reactor is operating at full power.

The fuel design is relatively simple – a 0.1 m diameter, 0.5 m long fuel bundle that can be easily manufactured and handled. The bundle consists of a number of zircaloy-sheathed elements (for example, 28 or 37 in the fuel in common use today) arranged in concentric rings. The fuel material itself consists of high-density UO_2 pellets.

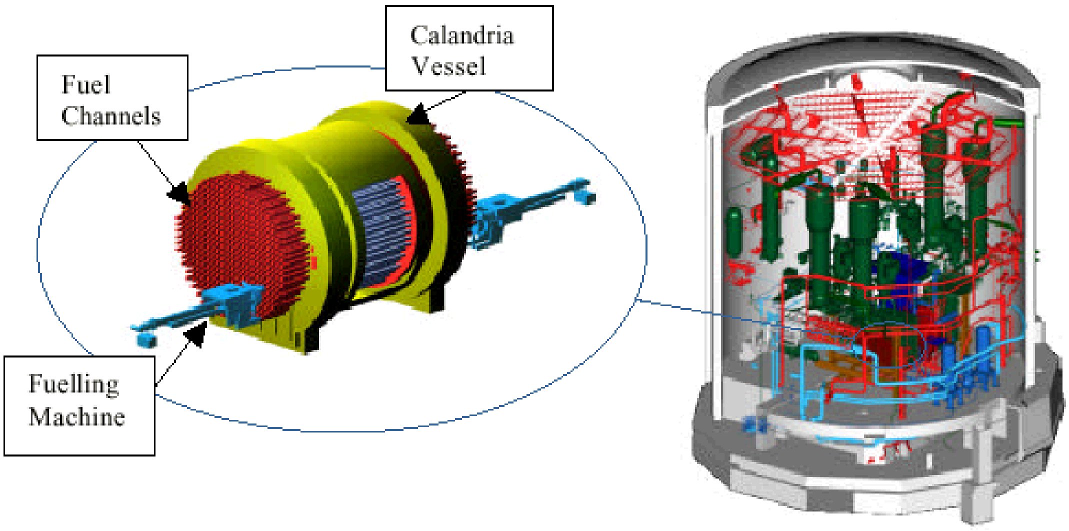

FIG. 1: The 700 MWe CANDU 6 SMR

It is the combination of high neutron efficiency, on-power fuelling, and simple bundle design that gives CANDU reactors the flexibility to use several different fuels and fuel cycles. The exploitation of this flexibility results in fuel cycles that optimize the use of uranium resources and that secures long-term fuel supply even if uranium resources become scarce or expensive. A prime example of this is the ability to burn used LWR fuel.

The reactor also includes a number of robust safety features. For example, reactivity devices are located in the low pressure and temperature moderator, and are not subject to high temperatures or pressures.

Accidental reactivity fluctuations have reduced consequences, since excess reactivity available from the fuel is small, and the relatively long lifetime of prompt neutrons in the reactor precludes rapid changes in power levels. Two independent and diverse reactor shutdown systems ensure the reactor can be shut down under any conditions. And the presence of the large moderator tank ensures that heat can be passively removed from the core in the event of loss of both the primary coolant and emergency cooling systems.

The design meets current and evolving licensing requirements in Canada, the current and evolving IAEA safety standards and guides, and includes a comprehensive design definition adaptable to host country licensing requirements without fundamental changes.

4. REGULATORY SUPPORT AND LICENSING BASIS

CANDU 6 units have been licensed for construction and/or operation both in Canada and in four other countries. The licensing basis has thus been demonstrated to meet both Canadian licensing standards and those elsewhere, including, for example, standards in Republic of Korea based on U.S. licensing experience. AECL has built up a very broad and up-to-date experience in supporting local licensing activities. Also, the Canadian regulator, the Canadian Nuclear Safety Commission (CNSC), has a long-standing commitment to assist with the development of licensing agencies world-wide. This has been and continues to be applied in licensing support to CANDU 6 host countries. For example, the CNSC is currently providing formal classroom and on-the-job training in Canadian licensing practice to regulators from China for the Qinshan project. A companion paper and the panel discussions at this conference will describe this Canadian sup-port to client countries.

5. PROJECT FINANCING & PROJECT TEAM

While investments in nuclear power are excellent long-term choices, some countries need financing assistance and financial guarantees to launch nuclear projects. AECL's approach to this is to draw on our project experience, network of partners, and our relationship with the Canadian government to assist with this aspect of a project. For example, for the Qinshan Project in China, AECL involved Bechtel (USA) and Hitachi (Japan) in the project team. Each of the team members accessed export loans from their governments to secure the overall project financing.

For all projects, once financing is in place, the project team is strengthened by involving local construction companies and the local utility. For Qinshan, as for previous projects, the project team members are working with local companies to transfer their expertise and in so doing to transfer technology to the client country.

6. CANDU 6 PROJECT EXPERIENCE & TECHNOLOGY TRANSFER

The CANDU 6 reactor design has proven to be adaptable to construction in varied sites and locations, and with a range of project models, from component contracts to turnkey. Individual projects have readily included adaptation of supply for significant localization of both nuclear and non-nuclear components.

For example, all exported CANDU 6 units have resulted in local manufacturing of reactor fuel, due to the relatively simple CANDU fuel design.

A large degree of localization and technology transfer has been accomplished in the course of building a series of units, by tailoring project equipment supply, construction, commissioning and operation responsibilities to host country priorities and capabilities. For example, in the Republic of Korea, local content increased from 25% for Wolsong Unit 1, to more than 75% for Wolsong Unit 4. This was coordinated with increasing technology transfer to host organizations, providing both "know-how" and "know-why" to enable engineering, equipment manufacturing, and licensing to be localized to the fullest extent.

While the initial CANDU 6 design was established in the early 1980s, both the design itself, and project delivery methods have continued to evolve. As a result, the most recently committed CANDU 6 project, at Qinshan in China represents the state -of-the-art in nuclear construction. The Qinshan project is de-signed fully via a series of CADDS-3D models for process, civil, and I & C design. The models are integrated with databases for engineering documentation, procurement documentation and on-site material management. As a result, dramatic improvements have been made in reduction of rework and material waste, elimination of equipment interferences, and efficiency of installation.

In addition, the Qinshan project includes extensive use of open-top construction techniques, based on application of a Very Heavy Lift (VHL) crane. All major vessels and a number of piping modules are in-stalled in the reactor and turbine building using the VHL crane. Currently, for example, the reactor building dousing modules are being prefabricated and will be ready for installation before the Unit 1 reactor building dome is built.

Many of the electronic engineering tools were pioneered for the Wolsong 2,3,4 construction project. As a result, the Wolsong 3 project achieved an unequalled start-to-finish total project speed – 69 months from the project Contract Effective Date (CED) to Commercial Operation. This 69-month duration included 14 months for initial engineering and construction license approval; 41 months for main construction; and 14 months for main commissioning. Current studies for future CANDU 6 projects have demonstrated a total project schedule of 66 months is achievable – unmatched for currently available nuclear plants.

7. OPERATING RECORD & OPERATIONS SUPPORT

In the last few years, increasing worldwide emphasis has been placed on improving production from nuclear power plants (NPPs). In a number of countries, concerted efforts have been made to improve NPP capacity factors, and also to ensure that other production and safety performance indicators are improved.

This is more challenging for countries that may only operate one or two nuclear plants, as is the case for some CANDU 6 SMRs. Therefore, for CANDU owners worldwide, alternative support is provided very effectively by the CANDU Owner's Group Inc., an operator funded cooperative organization, which pro-vides powerful information exchange, logistics assistance and jointly sponsored operating support pro-grams.

In addition, AECL, as a fully comprehensive nuclear services organization, makes a priority of maintaining support to CANDU 6 owners through internal design feedback and plant life management programs, as well as by the provision of engineering and on-site products and services.

The results of this cooperation among CANDU 6 owners are evident in the good operating record of CANDU 6 units. CANDU 6 capacity factors have been consistently excellent for both the original and more recent units, covering more than 80 reactor years of operation during a period of more than 17 years.

The average capacity factor for CANDU 6 units averaged over the last 12 years is approximately 85%. This is an excellent record considering that the CANDU 6 units are operated by five different utilities in four countries under very different conditions. This demonstrates that the CANDU 6 SMR design enables reliable operation for utilities with a wide range of experience in operating nuclear plants.

This average capacity factor has remained high, as the years of unit operation have increased. The record shows that plant aging does not have a large effect on CANDU 6 capacity factor. In addition, new CANDU 6 units coming on line over the last four years are performing equally well, further demonstrating that the design robustness is a lasting feature. For example, in the first nine months of year 2000,Wolsong units 3 and 4 both operated at 100% capacity factor. This shows that after a rapid, on-time construction project, the CANDU 6 SMR has proven its ability to quickly settle in to reliable operation.

CANDU 6 plants have achieved high operating performance in other key respects also. For example, low-level waste volumes are typically about 1/3 of those arising from comparable plants. Emissions from CANDU 6 plants have met traditional Canadian targets (< 1% of regulatory derived limits) with ample margins. And unplanned shutdown frequency has been consistently less than 1/year.

8. PLANT LIFE MANAGEMENT

The first four CANDU 6 units were based on a conservative 30-year design life. After 17 years of operation, studies have confirmed that this includes a great deal of margin, and that life extension to 50 years of operation is practical. Economic evaluations of a mid-life refurbishment outage to life-extend the two Canadian CANDU 6 SMRs (Gentillly-2 and Point Lepreau) to 50 years life are well under way, and similar studies are planned for Embalse and Wolsong 1.

The most recent CANDU 6 units, at Qinshan, have a design life of 40 years, and similar life extension studies would be advantageous in the future as these units reach their mid-life.

In the lead-up to these Plant Life Extension studies, AECL has also worked cooperatively with the Canadian CANDU 6 utilities on Plant Life Management programs, both to upgrade equipment monitoring and condition assessment, and to streamline the effectiveness of equipment maintenance. These joint studies have been very beneficial. By applying the techniques of Reliability-Centred-Maintenance (RCM) to plant equipment, the CANDU 6 utilities have been able to find opportunities to improve equipment operating reliability, while at the same time reducing maintenance costs.

9. EVOLVING DESIGN

As noted above, the CANDU 6 design has not remained static. In particular, the Qinshan project represents the current nuclear state-of-the-art. The Qinshan control rooms have been adapted from the original CANDU 6 control room design to include an advanced human-machine interface via AECL's Plant Display System software (see Figure 2). This includes detailed intelligent listing of messages and alarms for more reliable operator action. Future CANDU 6 SMRs will couple this interface to Distributed Control System modules for simpler construction and more reliable and easier-to-maintain operation.

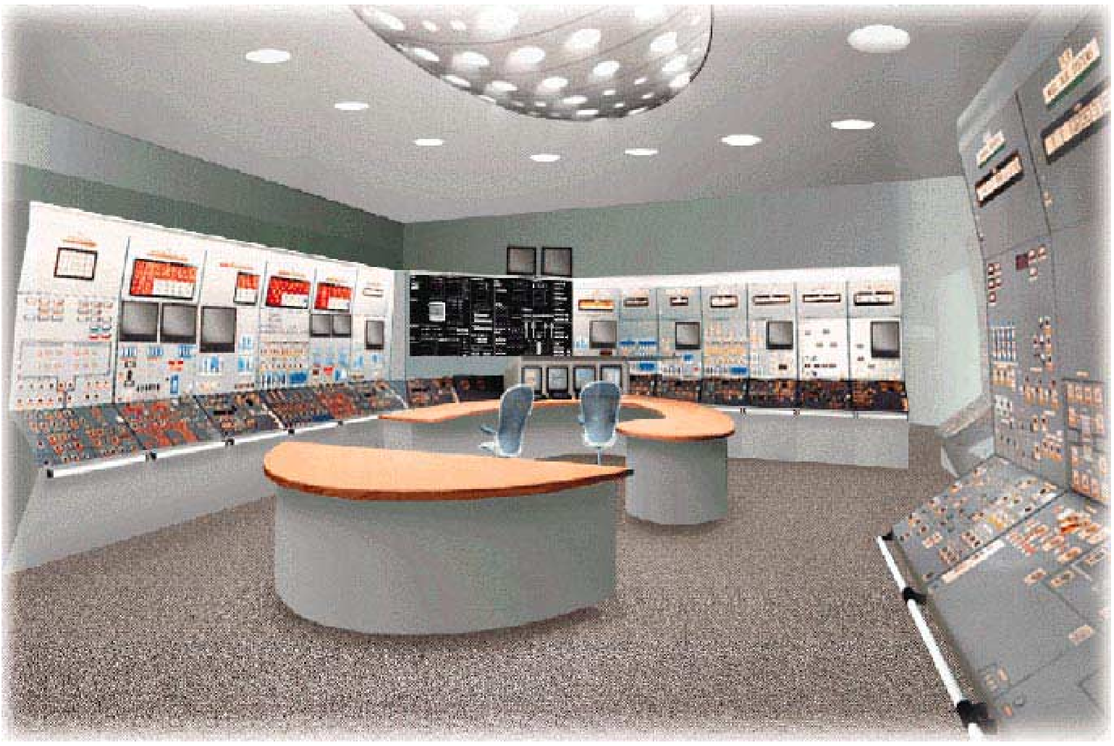

FIG. 2. CANDU 6 Control Centre.

AECL's CANDU 6 development program maintains the CANDU 6 SMR as a state-of-the-art system, and incorporates new technology from the development programs as part of a continuous improvements process.

Recent studies have, for example, evaluated plant adaptation for synergy with desalination facilities, design changes to reduce routine plant emissions, and improvements to spent fuel storage and handling.

10. CONCEPTUAL DESIGN: THE NEXT GENERATION SMR CANDU

Nuclear power plant designs for the future must respond to very demanding requirements from the emerging energy marketplace. This means additional substantial development.

For the CANDU system, AECL has adopted the evolutionary approach, accommodating significant changes to design while retaining the essential characteristics:

- Modular horizontal fuel channel core
- Available Simple, economical fuel bundle design
- On-power fuelling
- Separate cool, low-pressure moderator with back-up heat sink capability
- Low neutron absorption for good fuel utilization

Based on these principles, a conceptual design program for the next generation CANDU was initiated at the beginning of year 2000. A design configuration to extend the medium-sized CANDU 6 concept has been established. The configuration has the following technical characteristics:

- Nominal output of approximately 600 MW (e) net, dependent on site conditions and customer needs
- Light-water coolant at hot leg pressure of approximately 13 MPa.
- Steam outlet pressure to turbine at 7MPa, allowing turbine cycle efficiency of 35% or more.
- Traditional fuel channel configuration with 12 CANFLEX fuel bundles per channel
- Reactor core configuration consisting of up to 256 fuel channel locations
- Average channel power increased from 5.3 MW (CANDU 6) to 6.8 MW, while peak channel power increases by less than 10% due to an inherently even core flux shape (reflector-driven thermal flux), while peak fuel element ratings are approximately 20% lower than CANDU 6 due to the CANFLEX fuel design.
- Fuel enrichment nominally 1.6%, with burn-up of 20,000 MW/tU.
- Reactor coolant system in a single -loop, figure of eight configuration with two steam generators.
- Double-ended refuelling employing two on- line fuelling machines and 2-bundle shifts refuel-ling.

Figure 3 shows the conceptual design of the NG CANDU reactor. There are obvious similarities to the CANDU 6 SMR shown in Figure 1. While following the evolutionary approach, the Next Generation medium-size CANDU nuclear power plant represents a substantial step in the evolution of the CANDU system. A development program has been planned to generate the detailed, project-ready design and component qualification for this plant, for first-project implementation in the period 2005-2010. To bring the Next Generation CANDU to market readiness in this time-frame requires a strong emphasis on con-current development activities:

- Development and qualification of enabling technologies
- Plant conceptual and detailed design
- Development of improved project delivery
- Development of safety, licensing and environmental case

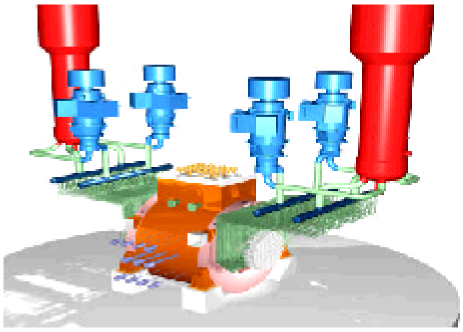

FIG. 3. Next Generation CANDU Reactor

AECL is proceeding with each of these activities in parallel, within a single coordinated program. The Next Generation CANDU program m represents a vital step in the continuing process of development of the CANDU system.

11. NUCLEAR POWER PLANT ADOPTION IN DEVELOPING ECONOMIES

Since SMRs are far easier to introduce into developing economies, it is instructive to consider the characteristics that are important for successful implementation. The CANDU 6 SMR has been successful for the following reasons:

- **Size** : At 700 MW(e) nominal output, the CANDU 6 represents a mid-size power plant suitable for addition to moderate size electricity grids. CANDU 6's load following responsiveness is also an ad-vantage in enhancing grid reliability.
- **Proveness**: The CANDU 6 SMR has a proven track record for both construction and operation. A strong design and operational feedback system is maintained to ensure that new projects take advantage of previous experience, and the current design is constantly being upgraded with state-of-the-art technology.
- **Low Ris**k: The most recent CANDU 6 projects at Wolsong were completed exactly on time against tight schedule requirements. Operational performance of CANDU 6s has remained remarkably consistent over different plant ages and different sites and jurisdictions.
- **Flexibilit**y: CANDU 6 units are readily sited, as shown by the different project examples. The cur-rent plans have been licensed in several different jurisdictions, thus building and demonstrating a comprehensive licensing track record.
- **Project and Operational Suppor**t: Canadian support to offshore CANDU 6 reactors has been essential for achieving successful build projects, along with extensive localization and technology transfer.

Also, the experience with operating CANDU 6s shows that a utility can be highly successful even if it only operates a single nuclear unit, due in part to the strength of operational support from the CANDU Owners' Group and from the cohesive CANDU industry.

To meet the emerging market requirements related to cost, schedule and project risk an evolutionary Next Generation Next Generation CANDU product is being developed. This product builds on the strengths of the current SMR CANDU 6.

12. SUMMARY

The CANDU 6 SMR has been implemented successfully in both large and small countries, and under a variety of different conditions and infrastructure. CANDU design evolution, leading to a next generation CANDU design, will add to the competitiveness of future SMR's. This experience has shown that SMRs can be successfully introduced into developing countries by ensuring all aspects of a project are meticulously addressed, including financing, regulation, construction, commissioning, and operation. In addition, the importance of establishing long-term technical cooperation with countries adopting SMRs is critical to the ongoing success of an emerging nuclear program. If all these elements are present, then SMRs can be easily introduced into a country's energy infrastructure.

PROSPECTS OF SMALL LIGHT-WATER REACTOR UTILIZATION

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Abstract

OKB Mechanical Engineering (OKBM) has accumulated a many years of experience of designing world famous low power PWRs, including the KLT-40 type propulsion reactors and has gained from the positive experience of their operation and technical perfection.

With reliance on this experience, a whole range of prospective small power reactor plants with KLT-40S-type reactors has been developed, including ABV, ATEC co-generation power plants, SAKHA etc., which may be recommended for the immediate future as small power nuclear sources of energy for regions with a decentralized power supply, where the utilization of traditional Diesel generators in entails, in the majority of cases, considerable difficulties associated with the need for stable provision of large quantities of liquid fuel and lubricants.

The report contains information on the peculiarities of small power RPs utilization. We also consider the major characteristics of nuclear power plants based on the advanced reactors of the above mentioned types, which meet all modern safety requirements. We demonstrate the possibility of the use of small power RPs in complexes for electric power, heat and fresh water co-generation.

1. INTRODUCTION

The growing deficit of electrical energy, heat and potable water is currently characteristic of many regions, primarily those with decentralized provision of power demands. This prevents effective economic development of the regions and consequently impedes the improvement of social life conditions for the local population. It is in particular characteristic of vast territories of the Russian Federation's sub-Arctic regions, especially of North-East and Far East regions.

The utilization of traditional Diesel generators in the majority of cases entails considerable difficulties associated with needs in stable provision of great flows of liquid fuel and lubricants, that eventually leads to a drastic rise in the cost of energy produced there. All these afford grounds for consideration of nuclear power sources to be a more optimal solution of the decentralized electric and heat provision problem under certain conditions.

The largest international experience has been accumulated at present in the construction and operation of LWR-type reactors. Because of this reason, first small size NPPs with acceptable effectiveness will evidently use the same reactor type. However, the following considerations should be taken into account.

In designing a small NPP it is necessary on the one hand to rely to a full extent upon the available considerable experience in construction and operation of large size NPPs. On the other hand, the design features of large NPPs could not be directly used for small NPPs since they would not allow acceptable economic characteristics to be obtained. Consequently, ahe search for original design and structural solutions is needed. Since a unit power to a great extent determines admittance of those or other solutions, there are grounds for attaining positive results there.

The requirements for a small size NPP are determined in general by specific conditions of its utilization and operation.

Practical attainment of optimal values of all the necessary parameters will certainly require an adequate period of time, especially since non-optimal individual characteristics of pilot plants might be in a number of cases taken in favour of proven technical features for the sake of reliable demonstration of the whole plant's positive performance within an acceptable period of time.

OKB Mechanical Engineering (OKBM) has accumulated a multi-year experience of designing small power PWRs, including KLT-40 type propulsion reactors; a positive experience of their operation and technical perfection has been gained [1].

The nuclear reactor plants have gone through a long process of evolutionary development from their utilization in nuclear icebreakers to application for nuclear floating power plants and integrated power & water desalination systems [2].

With reliance on the experience with KLT-40S-type reactors, a whole range of prospective small power reactor plants has been developed, including ABV, ATEC co-generation power plants, SAKHA etc., which may be recommended for the nearest future as small power nuclear sources of energy.

2. SMALL REACTORS DESIGN AND APPLICATION FEATURES

Addressing the particular purpose of small reactor plants (RP), i.e. provision of outlying regions (with decentralized energy sources) demands in electrical energy, heat and potable water, very specific requirements are imposed on them, associated primarily with reactor plant units delivery to a construction site, in-service maintenance and decommissioning.

The following are attributed to such specific requirements:

- modular configuration of a reactor plant with ability to be transported as large prefabricated units by any type of transport to a construction site, aiming at minimization of construction/erection work scope at a site;
- ability to be located and operated on ships and floating vessels, if necessary;
- minimal demands for experienced maintenance personnel, possibility to provide telemetering control of a reactor plant operation from a remote control center and possibility to be periodically serviced by a watch method;
- high safety level, eliminating environment contamination and need for population evacuation in the event of any accidents;
- economic effectiveness compared to fossil plants used in the same region;
- simplification of and reduction in reactor plant systems;
- extension of equipment and systems service life up to that of the reactor plant itself;
- reactor power self-control capability under load variation conditions;
- economically attractive "price" of small NPP safety, that is considered as a key factor for successful development of such power sources.

The solution of the problem supposes a formulation of validated probabilistic safety criteria for small NPPs deployed in remote regions with a low density of population, as well as development of Federal (and international) standards for such NPPs. Also that the risk concept should be used along with the deterministic design approach. Small NPPs and their components should be designed with reliance upon the modern probabilistic safety analysis techniques, based on available operating experience of similar small NPPs, with further reduction in probabilistic targets compared to that for operating analogs.

Deployment of the reactor plants on floating vessels, e.g. on non-self-propelled barges, seems to be the most attractive for coastal regions. The main advantages of floating NPP are as follows:

- manufacturing of power units at a shipyard in relatively short terms with high industrial quality, followed by NPP handing over to a customer on the “turn-key” basis;
- possibility of NPP to be deployed at various coastal regions;
- convenience of NPP maintenance by a special tender-ship at a floating plant mooring place;
- 1.4 to 1.6 times reduction is construction costs compared to land-based NPPs;
- NPP decommissioning by its vessel tugging to a place of dismantling facility location, etc.

3. .SMALL POWER REACTOR PLANT DESIGNS WITH LWR

Considerable experience has been gained in OKBM in the design, construction and operation of small size nuclear power plants with LWR, ranging from 1 MWe to 200 MWe. Some designs within the indicated power range, that were developed in the Russian Federation are summarized in Table below:

TABLE I. SMALL POWER REACTOR PLANT DESIGNS

Main characteristics	Name of design				
	KLT-40	ABV	ATEC-80	ATEC-200	SAKHA-92
Thermal capacity, MWth	up to 160	up to 60	250	540	7
Reactor plant configuration	Modular	Integrated	Integrated	Integrated	Integrated
Primary coolant circulation	Forced	natural	natural	natural	natural

The reactor plant designs to different extents meet the basic requirements for small size NPPs. A succession of the designs described below reflects their evolutionary development towards meeting the requirements more fully.

3.1. KLT-40S reactor plant

The KLT-40S reactor plant is based on the experience of design and long successful operation under severe navigation conditions of Russian nuclear-powered ships with similar reactor plants. Also, international experience and trends in NPP safety enhancement have been taken into account in the design.

The major technical characteristics of the KLT-40S reactor plant are as follows:

- Reactor plant configuration – modularized with PWR-type reactor.
- Thermal capacity – up to 160 MW
- Primary coolant circulation – natural up to 10% N_{nom} (shutdown cooling), forced
- Steam temperature – up to 300 ^{o}C
- Steam pressure – up to 4 MPa.
- Operating expperience of key equipment in similar reactor plants amounts to 150,000 hours.

The reactor plant's primary components, viz.: reactor, four steam generators and four reactor coolant pumps are incorporated by short load-bearing coaxial nozzles in a single compact steam generating unit. The unit is located inside a metal-water shielding tank, which is housed within a protective shell (Fig.1).

The prototype KLT-40 reactor plant is included in the "Small and Medium Power reactors Status Report" issued by IAEA (1994). The plant was a winner of the small power reactor designs competition held by the Russian Nuclear Society in 1995. ABV and KLT-40 reactor plants were at the first place among all other designs of reactor plants of the same power level.

The commercial status of the fabrication/construction process and multi-year positive experience of KLT-40-type plants operation in nuclear-powered ships (operated by the Murmansk shipping company) open wide prospects for the Russian Federation in creation of small floating nuclear power plants with this type reactor plant [2].

The work status attained by efforts of a number of involved engineering and industrial enterprises collaborated with Rosenergoatom Concern permitted to start with practical realization of the first floating power unit project for a pilot heat & power co-generation plant. It is planned to be deployed at Peveck port area (the Chuckchee Peninsula) and in Severodvinsk.

In conformity with "Programme for nuclear power development in the Russian Federation at 1998 to 2005 and until 2010" the floating co-generation plant will be built in Peveck between 2006 and 2010.

The major technical characteristics of the floating power unit are as follows:

- number of reactor plants	2
- electric power, MWe	70
- heat output, Gcal/h	up to 50 (at reduced electric output)
- total plant service life, a	40

The floating power unit overall dimensions are:

length, m	140
width, m	30
draught, m	5.6
displacement, t	21000

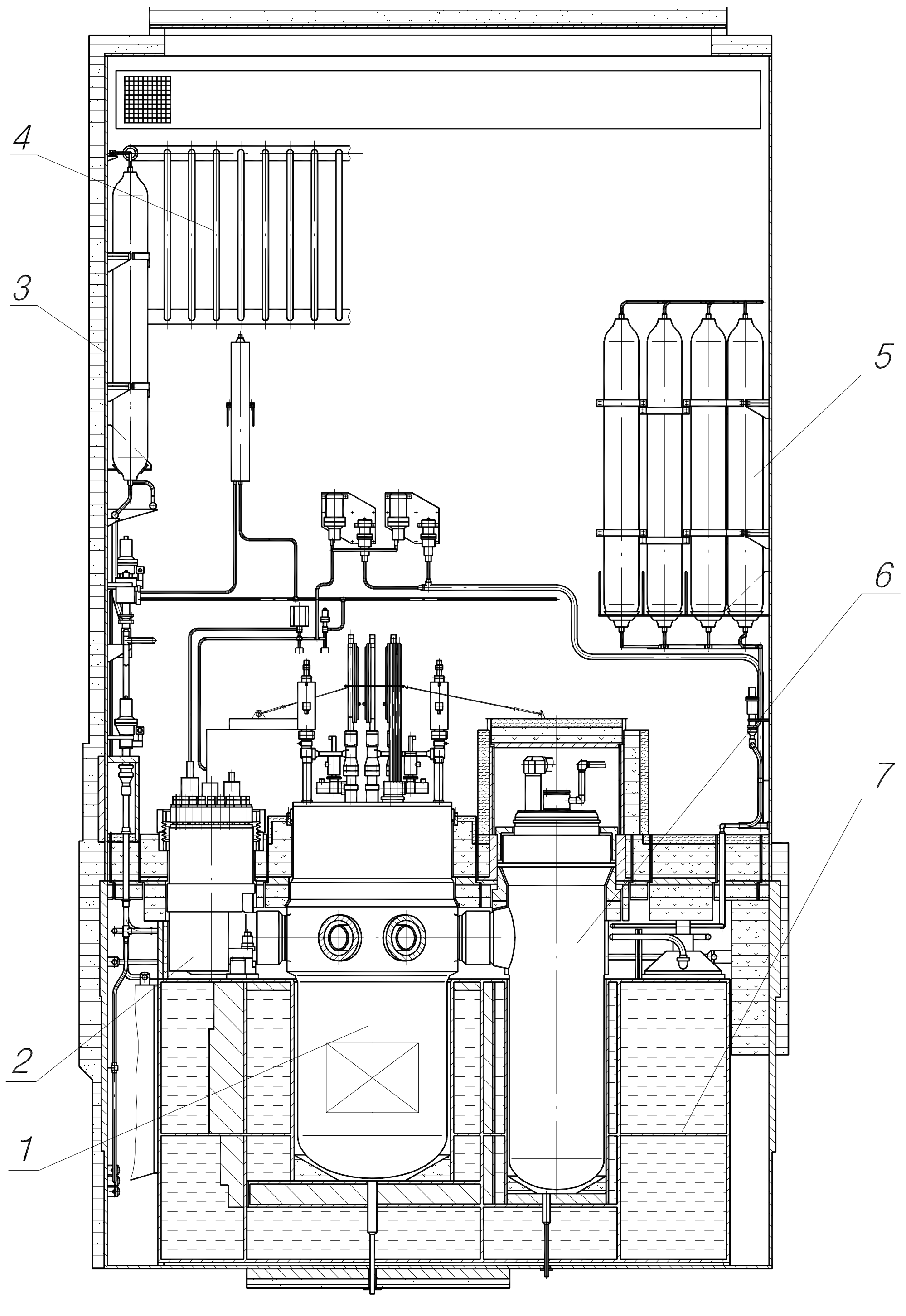

Fig. 1 – KLT-40S reactor plant

1 – reactor
2 – primary circuit circulation pump;
3 – protective shell;
4 – protective shell pressure suppression emergency condensation system;
5 – high pressure gas cylinders;
6 – steam generator;
7 – metal-water shielding tank

The integral reactor plant designs are described below: The integral configuration has certain advantages compared with a modular non-integral one in terms of safety, manufacturing quality, construction time, decommissioning etc. That was a ground for development of new generation power units with integral LWRs. A range of designs has been developed, starting from ABV-type plants and finishing with ATEC-type co-generation plants. The integral reactor plants have either an outside gas pressurization system (as in ABV-type reactors) or integrated steam-gas (ATEC-type reactors) pressurization system. Safety of the plants is basically ensured by passive means.

3.2. ABV reactor plant

Main technical characteristics of the power unit with ABV reactor plant are as follows [3]:

Reactor thermal capacity, MW – 38

Primary coolant circulation – natural

Steam flow, kg/s – 14.7

Steam temperature, ^{o}C – 290

Steam pressure, MPa – 3.14

The ABV reactor plant has the following design features (Fig. 2):

- it can be used in land-based and floating NPPs;

- it uses an integral reactor with natural circulation of coolant in the primary circuit.

The reactor core and steam generators are located inside RPV. The gas pressurization system is located outside RPV.

A detailed design of the ABV-reactor plant has been accomplished and agreed with the RF regulatory authority.

The compact configuration of the ABV-reactor plant permits its optimal layout in a floating NPP vessel.

The major technical characteristics of the floating power unit are the following:

–	number of reactor plants	2
–	electric power, MW	12
–	heat output, Gcal/h	up to 24 (at reduced electric output)
–	total service life, a.	50

overall dimensions:

–	length, m	78
–	width, m	22
–	draught, m	2.5

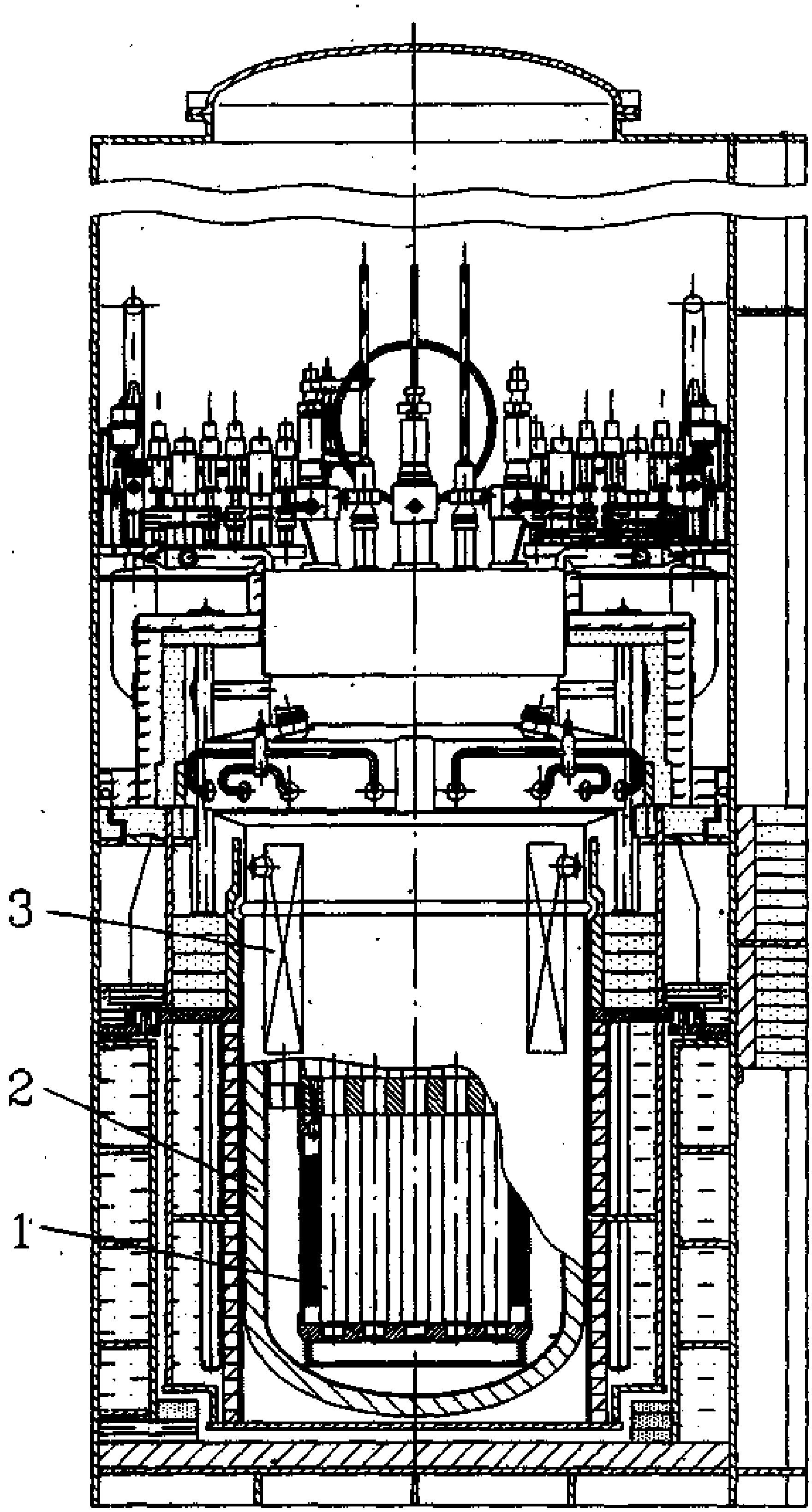

FIG. 2. ABV reactor plant

1 – active core;
2 – reactor pressure vessel;
3 – steam generator

Design study of a land-based NPP with RP housed inside an industrially-produced container (8.5 m diameter, 12 m length and 600 t mass) was carried out. The site area for twin-reactor plant is 7 ha.

3.3. ATEC-80 reactor plant

Major characteristics of ATEC-80 reactor plant are the following [3]:

Thermal capacity, MW	250
Primary circuit parameters:	
– - nominal pressure, MPa	15.7
– - core outlet temperature, °C	340
– - core inlet temperature, °C	265
Secondary circuit parameters:	
– - steam temperature, °C	290
– - steam pressure, MPa	4.4
– - feedwater temperature, °C	195.

The conceptual design of the reactor plant is currently being developed.

The following basic technical features are used (Fig. 3):

- integral configuration of the reactor with active core, steam generators and steam-gas pressurization systems located inside RPV;
- guard (safety) vessel around the RPV;
- natural circulation of coolant in all operating modes;
- passive safety systems;
- two independent reactivity control systems.

The nuclear co-generation plant with ATEC-80 reactor has the following characteristics:

– number of reactor plants	2
– electric power, MW	170
– heat output, Gcal/h	112
– total plant service life, a	60

A whole range of ATEC-type reactor plants within 50 MWe to 230 MWe has been developed.

3.4. SAKHA-92 reactor unit [3]

The design of an ultra-small (7 MW capacity) reactor plant has been made with the aim to create autonomous unattended sources of energy with long (up to 25 a) service life without refueling [6].

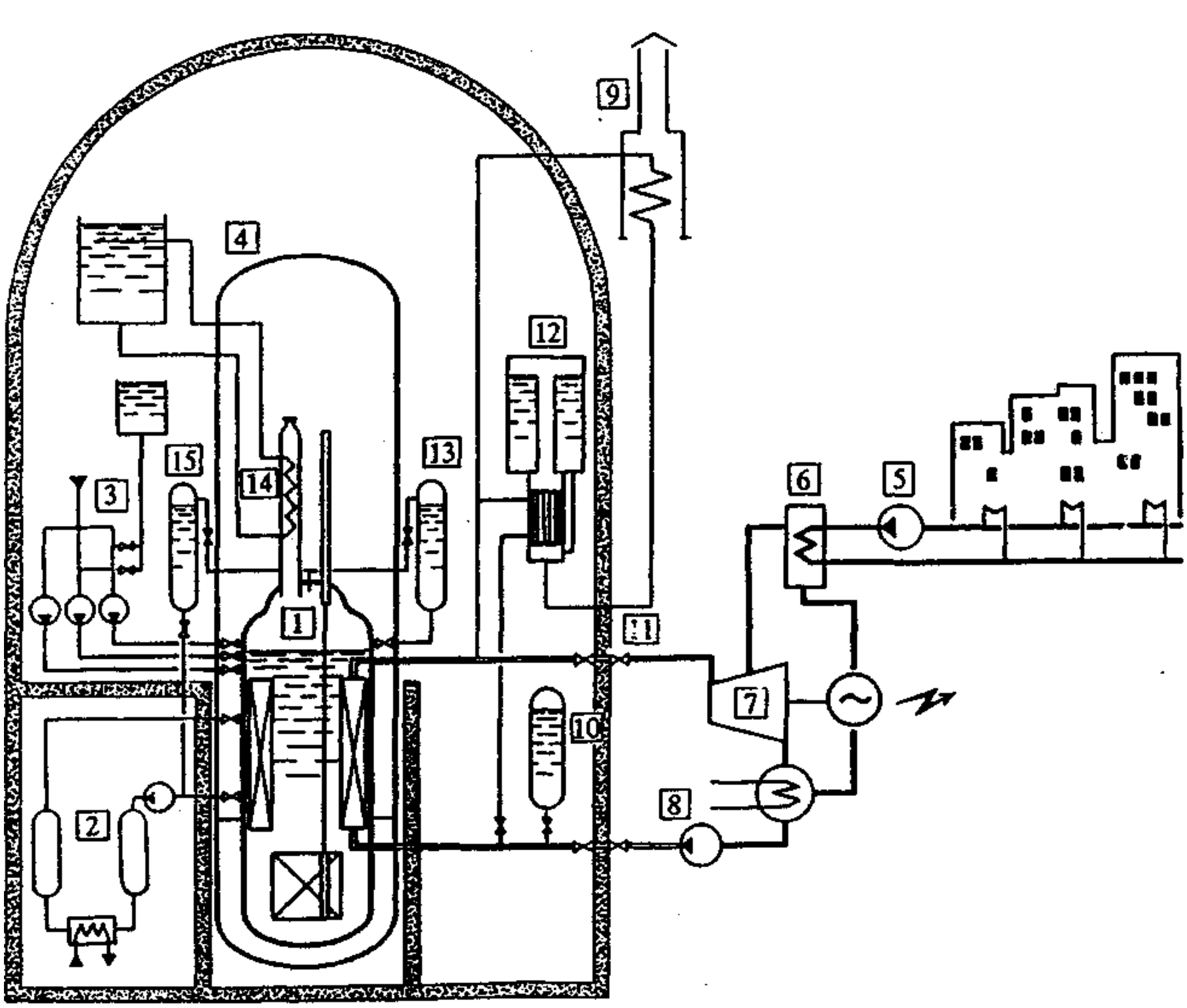

FIG. 3 ATEC (co-generator) reactor plant

1 – Reactor
2 – Purification system
3 – Makcup and boron injection system
4 – GV
5 – Grid circuit pump
6 – Grid HX
7 – Turbogenerator
8 – Fide pump
9 – Air heat exchanger
10 – Water storage tank
11 – Containment
12 – ERHRS heat changer
13 – Emergency injection system
14 – Channel of emergency residual heat removal system
15 – Boron injection system

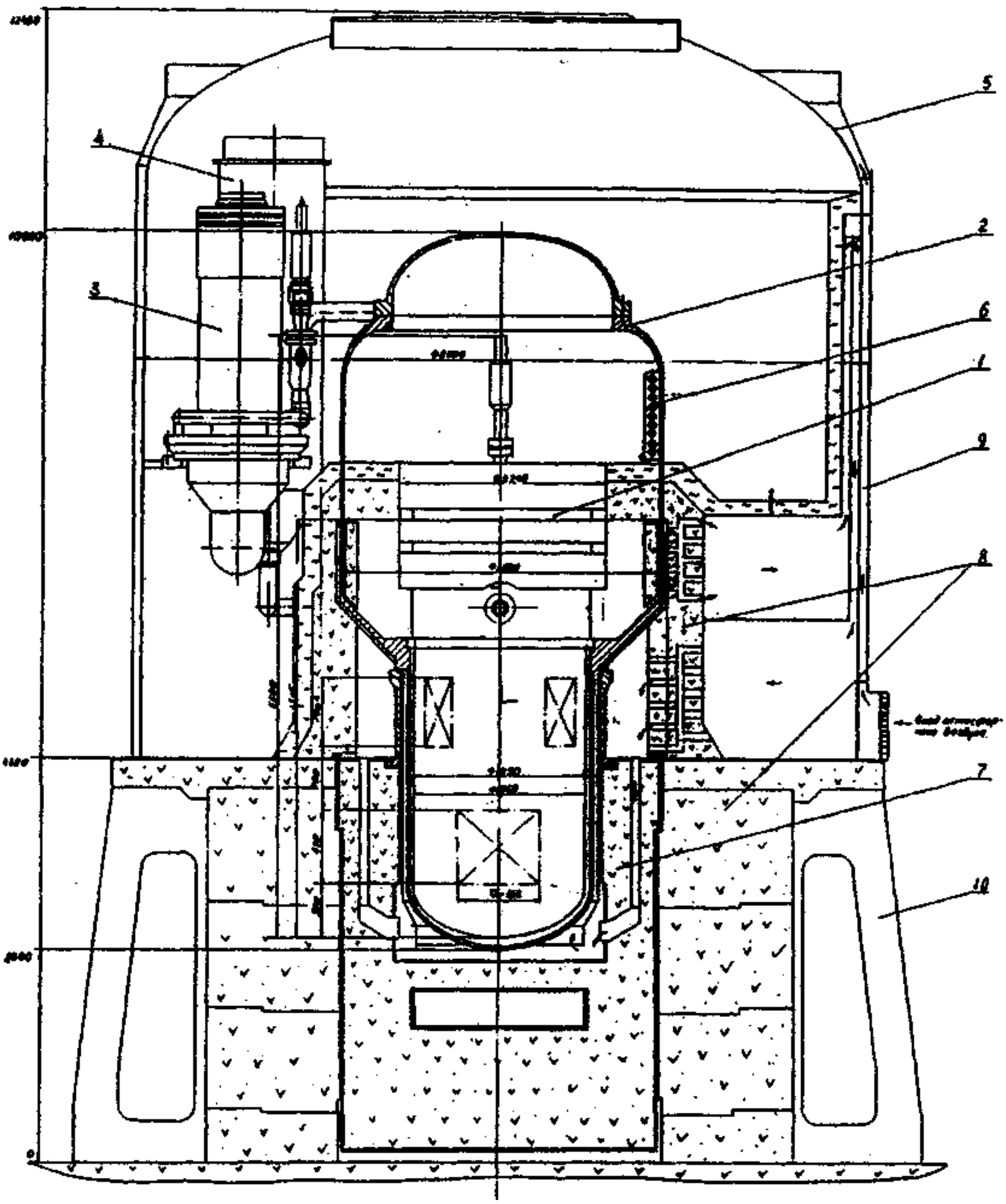

FIG. 4 "SAKHA" reactor plant

1 – steam generating module;
2 – guard (safety) vessel;
3 – turbogenerator;
4 – condenser;
5 – hermetic vessel;
6 – heat exchanger;
7, 8 – biological shielding;
9 – air cooler;
10 – support structures

The major technical characteristics of SAKHA-92 power unit are the following [3]:

– Reactor thermal capacity, MW	7
– Primary coolant circulation	natural
– Steam flow, kg/s	2.6
– Steam temperature, ^{o}C	300-335
– Steam pressure, MPa	3.2

– A two-circuit configuration is used.

Integral PWR-type reactor, where an active core, steam generators, control rods, steam-gas pressurizer are located inside a single pressure vessel (Fig. 4) is used.

Implementation of conversion technologies (e.g. leak-tight asynchronous turbogenerator, leak-tight secondary circuit including pumps) allows a large number of systems to be dispensed with, and a significant reduction in the number of personnel, so periodic servicing of the plant becomes possible.

The leak-tight design of an asynchronous generator predetermines elimination of an oil systems, that enhances the NPP fire safety, simplifies design of its condensate-feedwater system.

The conceptual Design of the SAKHA-92 reactor plant has been developed.

NPP with one SAKHA-92 reactor plant is capable of generating up to 1 MW electric power and up to 2.7 Gcal/h heat. Service life of the plant is 25 a. Shipping mass of the module is 80 to 90 tons.

CONCLUSION

1. Small reactor plants, which have already been widely used for propulsion of ice-breakers and freight ships providing economic development of the Russian Federation's Extreme North and Far East regions, have a real prospect of further utilization.

2. Successful operating experience of the propulsion reactor plants permit them to be recommended as energy sources for heat&power co-generation plants and power-sea water desalination systems.

3. Co-generation small nuclear plants are currently being developed in the Russian Federation. They use a floating power unit based on the KLT-40S reactor plant. The plants will be deployed at port Peveck in Chuckot national district and in Severodvinsk (Arkhangelsk region).

4. Based on the ABV and KLT-40-type reactor plants a number of conceptual designs of floating sea water-desalination systems for co-generation of electricity, heat and potable water has been developed. Distillation desalination facilities developed by the leading Russian enterprise "Sverdchimmash" [4] and reverse-osmosis ones developed by "Candesal Inc." (Canada) [5] are used in the designs.

5. The small NPP designs considered here, which are based upon PWR-type reactors, apparently do not fully meet the requirements for small NPPs suitable for a wide application.
6. Further perfection and development of small NPPs should be expected in the following main directions: extension of core generating potential for the entire plant service life; simplification and reduction in the number of plant systems, optimization of the plant construction and operation costs.

7. There are no grounds to consider LWR as alone base for future development of small nuclear power plants. Other reactor types might be used when they will attain their technological maturity. In particular, noticeable advantages could have helium–cooled high temperature reactors.

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BILIBINO NPP: OPERATION EXPERIENCE AND DESIGN LIFETIME EXTENSION

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Abstract

The Bilibino NPP (BiNPP) has been operated since early 1974 near the town of Bilibino in Chukotka. A high effectiveness of the nuclear energy sources use under the rigorous conditions of the Russian Federation's North-East region has been demonstrated. BiNPP was designed as nuclear co-generation plant. Some specific features of the area where the BiNPP is sited have necessitated several original engineering solutions in the reactor plant development and in the design of the NPP. Their correctness has been confirmed by the operating experience. BiNPP consists of four power units of the same type. The BiNPP's installed capacity amounts to 48 MW, with simultaneous heat production of 78 MW. In the period when the Russian economy was stable (up to 1991), the plant capacity factor amounted to 85%, with that of operating availability of 90-92%. The BiNPP's economic parameters were considerably superior vs. those of local organic fuel fired power sources. This advantage appears to increase to date due to significant rise of prices for organic fuel brought to the area. The analysis of accidents with normal operation of safety systems and with their failures has revealed features of rather high inherent self-protection of the reactor. The reactors' failure-free operation ensures the high reliability of BiNPP as power source under extreme conditions of the Far North-East. The BiNPP 1-st unit's design lifetime of 30 years is to end in January of 2004, and that of the 4-th unit – in December 2006. The question of extending the operation lifetime of Bilibino NPP beyond the design limit was raised because the project of the BiNPP-2 (second construction stage) developed in 1992 proved not to be feasible due to the high construction cost.

1. INTRODUCTION

Bilibino NPP, "firstling" of national small power engineering, has been operated since early 1974 at Bilibino settlement in Chukotka (in latitude 69° North). The BiNPP was designed as nuclear co-generating plant. The major advantage of nuclear power sources as applicable to some outlying areas includes lower transportation cost compared to the transportation costs of organic fuel (in some cases over 100 times in terms of cost of energy equivalent). This circumstance is a major economic pre-requisite for possible expedience of constructing the nuclear power sources in outlying and difficult-to-access areas.

2. SPECIFIC FEATURES OF BiNPP CONSTRUCTION SITE

The area of Bilibino town is typified by a severe climate and complicated geological and hydro-geological conditions: long winter (of 8 months a year) with temperatures reaching -60°C; permafrost ground, mountain rocks penetrated with ice galls to large depths; lack of water in the region, with all rivers and natural lakes freezing in winter; large distance of thousands km from industrial regions, the nearest ports of the Northern water-way being at 300 km and 550 km.

The Chaun-Bilibino power supply system (ChBPS) the BiNPP had been built for has extended electrical grid (~800 km) crossing the mountain tundra, which is a pre-condition for higher accident occurrence (especially in summer time, when grounds are de-frosting). Grid capacity is small, so the BiNPP's gross plant output amounts to ~2/3 of total capacity. The ChBPS is also typified by a considerably irregular profile of the power load (e.g., four maximums-minimums in summer period with minimum-to-maximum ratio of 0.6).

3. ENGINEERING SOLUTIONS IN THE BINPP PROJECT

Due to the specific features of the region listed above, very original problems were resolved in the development of the BiNPP project.

The channel-type water-graphite reactor plant was developed based on the tubular-type fuel elements (TTFE) generating saturated steam over the direct steam cycle with natural circulation of boiling water at all power levels, having the necessary level of reliability in load-following operation mode (the nuclear power units must be involved in covering of the varying part of the power system's operation schedule).

The closed water cooling engineering system was developed for the first time for the polar area conditions. It is based on dry cooling towers, which require very small water flows (in terms of circuit make-up compensating for leaks therefrom).

The siting of a large industrial building on loose rocks under permafrost conditions was provided.

4. GENERAL CHARACTERISTICS OF BiNPP

The BiNPP's installed electrical power is 48 MW, with simultaneous heat production of 78 MW. The maximum heat production of 116 MW in terms of possible steam extraction from the turbine and heat-exchanging equipment capacity is available, with NPP's electrical power decreasing to ~40 MW. BiNPP includes 4 power units of the same type. The first unit was commissioned in January 1974, and the fourth one – in December 1976. Each power unit includes the following equipment: nuclear steam supply system of 62 MWt rated power generating 95 t/h of 6.37 MPa steam at feedwater temperature of 104°C; the turbine plant with steam extraction for district heating operated on saturated steam at a pressure of 5.88 MPa with an intermediate moisture separation; an electric generator of 12 MW power, transformer, scheme of power supply to the electrical grid of ChBPS; the heat exchangers and equipment for heat transfer to a district heat supply system; service water supply system; auxiliaries of the reactor and turbine plant [1]. Figure 1 shows the power unit's flow diagram**.**

5. REACTOR PLANT

The BiNPP Nuclear Steam Supply System is a Channel-type water-graphite reactor plant EGP-6 generating saturated steam in the direct steam cycle [2]. The technical specifications of the reactor EGP-6 are shown in Table I (the performance parameters are given above).

Six fuel elements form a fuel assembly (FA). The fuel elements are of the tubular-type (TTFE), with dispersion fuel composition, which has a rigid connection with the internal pressure tube and outer cladding (shroud).

TTFEs, inferior as they are vs. pin-type ones in terms of economic parameters, have several advantages, some of them being of principal importance for NPPs in outlying areas:

a. TTFEs with dispersion fuel composition and steel claddings are efficient for systematic load-following operation of reactor during the whole operating cycl
b. TTFEs ensure the water-graphite reactor inherent safety features under LOCA.

TABLE I. PHYSICAL AND ENGINEERING SPECIFICATIONS OF EGP-6 REACTOR

	Parameter	Units	Value
	Height of the core	M	3
	Diameter of the core	m	4.1
	Square lattice pitch	mm	200
	Number of fuel assemblies (FA)	item	273
	Number of CPS channels,	item	60
	including those of safety control	item	8
	automatic power control	item	4
	Uranium loading	t	7.2
	Fuel enrichment	%	3 and 3.6
	FA cycle duration accounting partial reloading	eff. days	1150
	Burn-up (maximum)	kg/t	20
	FA external diameter	mm	88
	FA total length	mm	7700
	FA downcomer pipe (0X18H10T steel) dimension	mm	25x1
	Number of fuel elements in FA	item	6
	Fuel element:		
	internal tube (0X18H10T steel) dimension	mm	12x0.6
	outer shroud (0X18H10T steel) dimension	mm	22x0.3
	fuel	-	UO_2 particles
	matrix material	-	magnesium
	The gas filling the graphite pile	-	nitrogen
	FA maximum power	kW	360
	FA average power	kW	227
	Maximum heat flow	kW/m^2	800
	Maximum fuel temperature	^oC	380
	Maximum temperature of the graphite moderator	^oC	550
	Water flow rate through FA	kg/h	2000-2500
	FA inlet water temperature	^oC	250
	FA outlet temperature	^oC	280
	Steam fraction at FA outlet (maximum/average)	%	30/16

The primary coolant circulation loop is arranged as follows: water is supplied from the distributing collector to the FA inlet connector; then it follows downcomer (FA central tube) and enters the bottom plenum of FA. It is dispatched therefrom to the riser formed by internal pressure tubes of six tubular fuel elements.

In the TTFE, the coolant passes consecutively through the inlet orifice, the spiral expansion joint made from pipe of ∅9.4 × 0.6, the fuel element itself of 3 m length, the outlet part of TTFE made of pipe of ∅12 × 0.6, length of 840 mm. The steam-water mixture from the fuel elements enters the upper plenum of the FA, which is of length of 3 m. From the upper plenum of FA, the steam-water mixture is supplied into the outlet header.

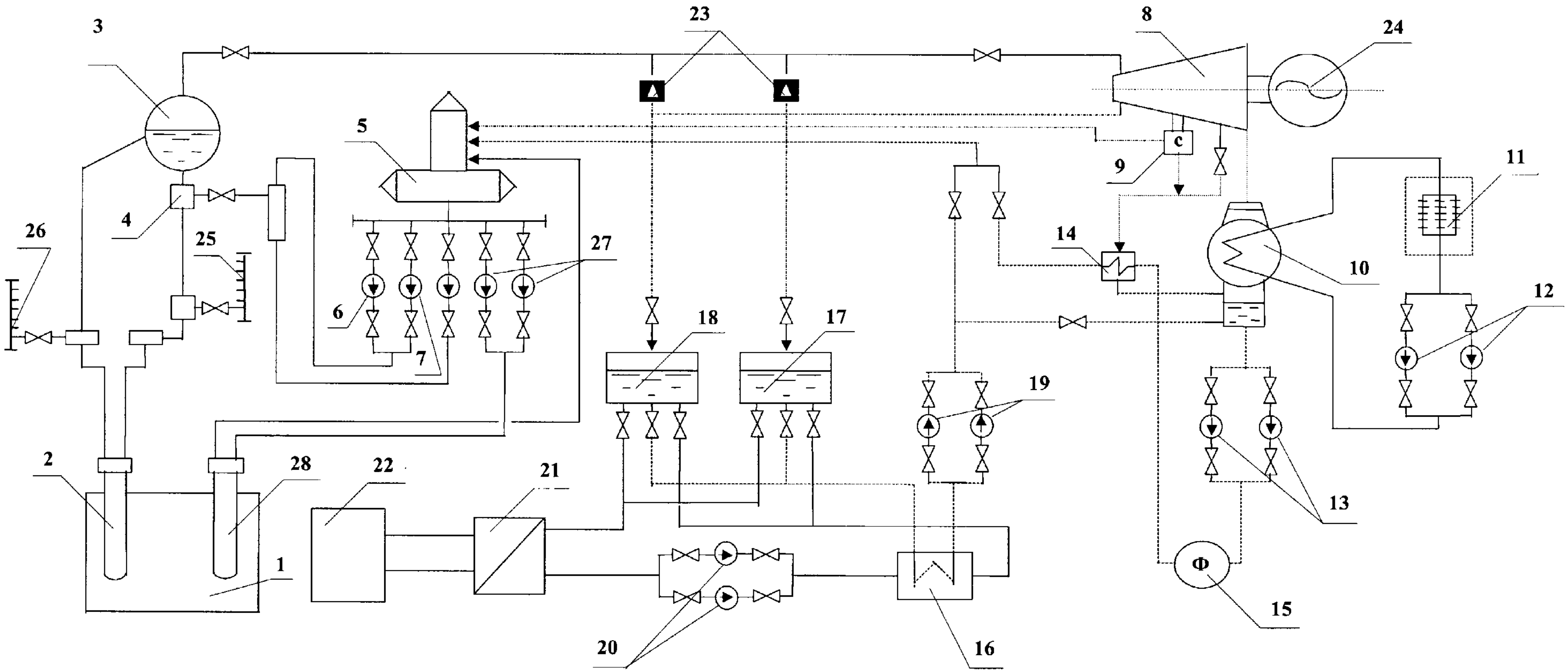

FIG. 1. Principal flow diagram of the Bilibino NPP power unit.

1 - reactor, 2 - fuel assembly, 3 – drum-separator, 4 – mixing device, 5 – deaerator, 6 – feedwater pumps, 7 – emergency feed pump, 8 – turbine, 9 – intermediate separator, 10 – condenser, 11 – air-radiator coolers, 12 – condenser cooling water pumps, 13 – condensate pumps, 14– low pressure regenerative heater, 15 – filter of combined action, 16 – aftercooler of heaters condensate, 17, 18 – main and peak heaters, 19 – heaters condensate pumps, 20 – intermediate circuit pumps, 21 – water-to-water heat exchanger, 22 – heat consumers, 23 – pressure-reducing devices, 24 – electric generator, 25 – Emergency Core Cooling System (ECCS) collector, 26 – bypass collector, 27 – cooling pumps of Control and Protection System (CPS) circuit, 28 – CPS channel.

The steel shielding plug for protection against gamma-radiation in the upward direction is placed in the upper plenum of the FA.

Cylindrical graphite sleeves are put on the central tube of the FA and fuel elements in the core and reflector area.

In order to improve nitrogen circulation over the external surface of the FA structural elements, a gas passage is formed therein, which ensures necessary movement of gas over the FA. The gas passage of the FA includes the supplying nozzle in the upper part, the annular gap between the upper plenum tube and outer shroud, the expansion joint cavity, and openings in the bottom FA shroud. Along each of six openings for passing fuel elements trough the graphite sleeves there are vertical chases throughout ensuring the flow of nitrogen along the fuel elements.

Heat removal from the FA is performed with natural circulation of boiling water for all levels of the reactor power, up to the rated one. The facility's natural convection loop consists of six independent loops connected to the drum-type steam separator. Each of the loops (Fig.2) includes the downcomer with the unit for mixing feed water and that separated in the drum, the horizontal dispatching group collector, which is a continuation of the downcomer pipe ($\varnothing 219 \times 12$), a group of FAs, horizontal outlet group collector connected to the riser ($\varnothing 219 \times 12$), along which the steam-and water mixture is transported to the drum-separator. Saturated steam is dispatched therefrom to the turbine.

The reactor FAs are subdivided into six groups according to the loops they belong to. Two central groups (by their arrangement in the core) include 65 FAs, two middle ones – 104 FAs, and two peripheral ones – 104 FAs. Despite the relatively high power density in the fuel elements (800 kW/m2), adequate cooling by natural circulation of boiling water is ensured by the design and geometric parameters of the natural convection loop as well as optimal choice of the core height (3 m) and primary pressure.

The channels of the Control and Protection System (CPS) are installed into cells similar to those for FAs. A CPS channel consists of a set of annular graphite sleeves, put on four parallel cooling pipes $\varnothing 9.4 \times 0.6$. A control rod is placed into the central cylindrical passage of the sleeves, which is connected with the pile via gaps between the sleeves. The water cooling pipes in the upper part of the channel are connected to the upper chamber consisting of two plenums, the inlet and outlet ones, and in the bottom part - with the bottom plenum. Cooling water flows downwards along two pipes of the channel, and upwards - along the other two pipes.

The circuit for heat removal from the CPS channels is connected to the de-aerator, which allows the utilization of the heat produced by radiation capture in the CPS (Fig.1). The temperature of water at the inlet to the CPS channels is 104°C, which excludes steam condensation on CPS channel pipes and on the circuit pipelines, thus eliminating the hazard of corrosion-active medium formation.

In order to improve the gas atmosphere in the reactor, graphite with a low content of chloride impurities is used (impurities being removed by a special technology), and nitrogen is pumped through cells of all FA (there is a gas circulation circuit in the reactor).

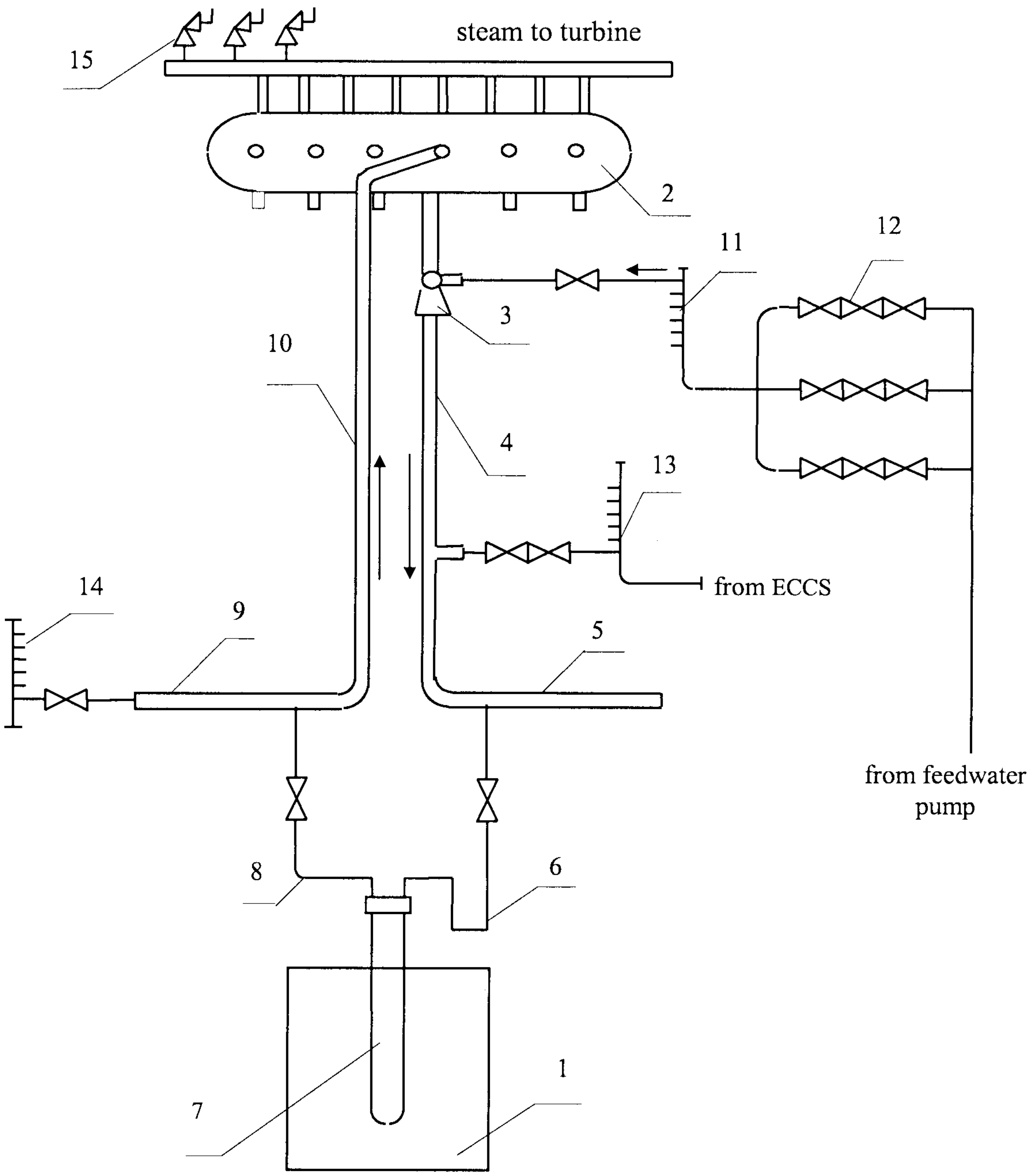

FIG. 2. EGP-6 natural convection loop diagram.

1 – reactor; 2 – drum-separator 3 – jet mixer; 4 – group downcomer; 5 – group inlet header; 6 – feed pipe; 7 – fuel assembly; 8 – discharge line; 9 – group outlet header; 10 – riser; 11 – feedwater header; 12 – feed unit; 13 – ECCS header; 14 - bypass header; 15 – Main Check Valve (MCV) system.

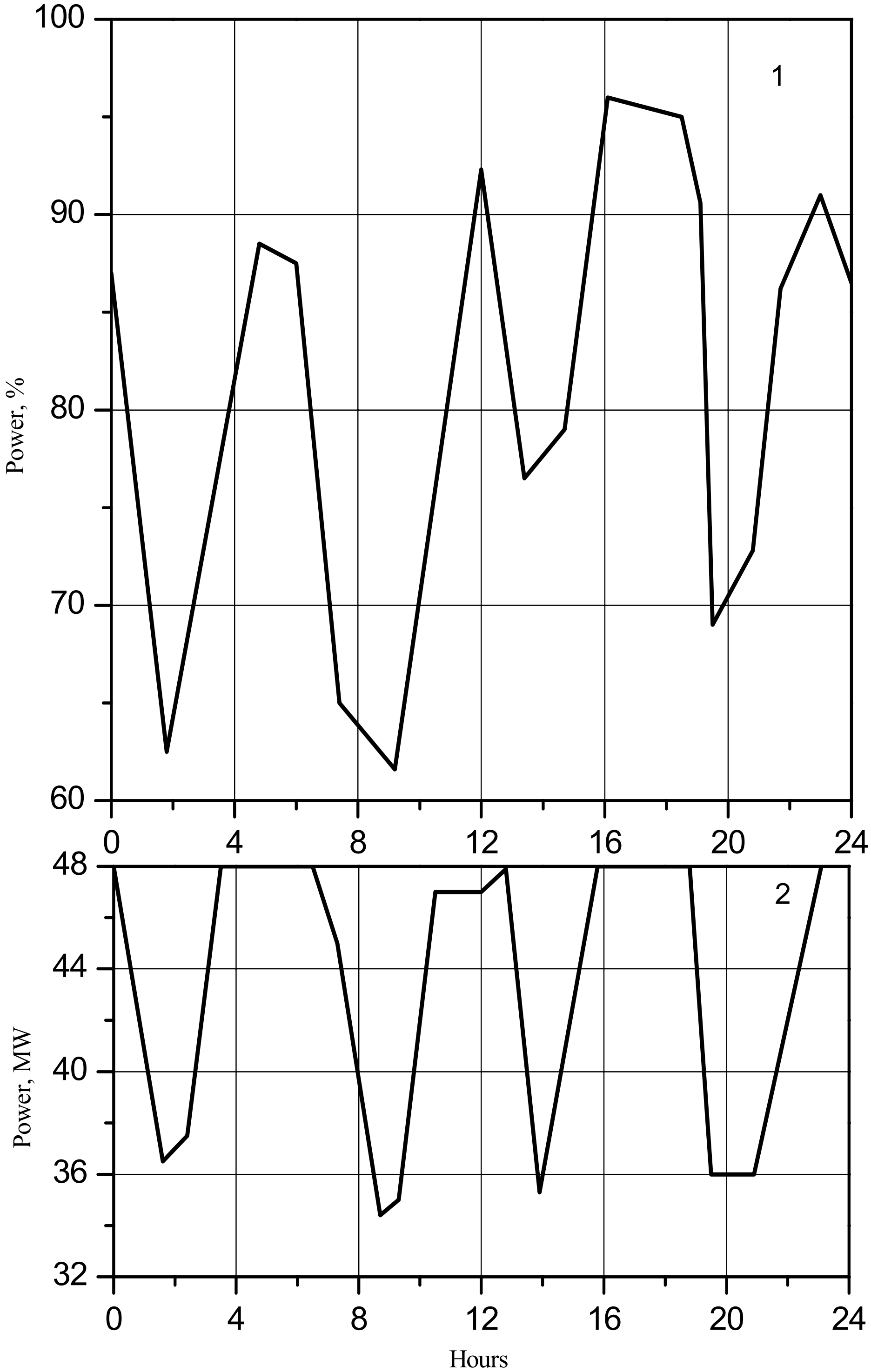

Fig. 3. Typical daily electric load profile of Chaun-Bilibino electric grid (1) and Bilibino co-generation plant (2) in summer time

6. BiNPP POWER UNITS OPERATION IN LOAD FOLLOWING MODE

Working within the relatively small ChBPS the BiNPP has to participate in covering the alternating part of the ChBPS operating schedule. An example of typical electric load profile of ChBPS in summer time and the BiNPP load following operation is illustrated in Fig.3.

The operator varies the power unit's output by the ChBPS according to the grid dispatcher order or according to the predetermined load profile. The power of units selected for load following operation is varied in the range of 50-100% (in electrical output) 3-4 times a day - according to the number of maximums in the ChBPS loading schedule [3].

7. SAFETY OF BiNPP

The analysis of transients at the reactor facility in normal operation of safety systems and in their failure has shown the reactor plant to have rather high inherent self-protection features. This is due to several factors: negative thermohydraulic and neutronic feed-backs; natural circulation of coolant; tubular-type fuel elements having thermal contact with graphite moderator, which heat accumulation capacity is much higher than that of fuel; the operating temperature of graphite moderator and fuel elements is considerably lower vs the temperature level at which a large-scale fuel elements cladding damage is possible; relatively small size of the core (relatively large heat sink into the adjacent space in accidents) [4].

8. ECONOMIC EFFICIENCY OF BINPP

In the period when the Russian economy was stable (up to 1991), the plant capacity factor amounted to 85%, with that of operating availability of 90-92%. Cost of power was 1.3–1.5 times as low as that of fossil-fired power plants in the most favorable locations in terms of organic fuel costs for the given region – at the Arctic Ocean coast. The cost of heat produced by the BiNPP for district heating was 2-2.5 times lower than that of boiler houses operated in the town of Bilibino, and consuming the organic fuel supplied from other regions. Nowadays, this difference appears to be even higher in favor of the BiNPP because of a sharp increase of prices for organic fuel.

The great economic significance of BiNPP construction is that there is a large-scale power source created for the Chukotka region. Its operation does not require the involvement of large transportation mean (dozens of tankers or coal carriers, hundreds of vehicles with appropriate number of personnel to be hired). The quantity of thermal energy that can be produced by the BiNPP reactors during a year corresponds to burning 210-230 thousand tonnes of TCE of organic fuel. The transportation of an equivalent quantity of nuclear fuel to the BiNPP (FA together with transportation containers) ranges to about 50 tones a year and is supplied by air.

9. EXTENSION OF BINPP OPERATION TIME BEYOND THE DESIGNED PERIOD

9.1. General issues

The problem of the service life extension for the nuclear unit or for the NPP beyond the time stipulated by the plant Project appears to be urgent all over the world. The extension has been shown to give a considerable economic effect. This is due to the fact that, first, the construction cost of a NPP is very high, and second, NPP decommissioning requires large expenses (compared to fossil-fired thermal power plants) as well.

At the beginning of NPP development and designing (60-s and early 70-s), when long-term operating experience with power reactor was lacking, the design lifetime of NPPs was scheduled with a certain underestimation “to be sure”. In particular, that was the case with the 30-year operation time: it was determined as analogous with organic fuel power plants. Now, for many costly elements of NPPs it is clear that they can be in use for significantly longer periods without additional upgrading and repair. Actually, their contribution into the NPP costs is the largest. It is obvious with the BiNPP example: the costs of reactor equipment together with mounting thereof amounted to only 11% of the whole plant’s cost. It is clear motivation for assessing possibility for nuclear islend equipment lifetime extension.

In the process of the development and implementation of the project of extending the BiNPP power units’ operation life for 10-15 years the following aspects is to be taken into account.

9.2.Socio-economic expedience of extending the BiNPP operation time

Socio-economic issues constitute a special aspect in decision-making process on extending the BiNPP operation time. BiNPP is a part of ChBPS network being total ~800 km length. The BiNPP located in the bottom (Southern) part of the V-shaped electrical network is the main element of ChBPS forming the entire system. If the BiNPP was to be decommissioned without a substitution with a new power source in Bilibino, it would mean in fact the liquidation of the ChBPS, because in this case the electrical network operation would appear extremely complicated. The system of power supply would be broken over a large area of the Chukotka Autonomous county. The town of Bilibino would face the problems in heat supply as well. The probability of emergencies in power and heat supply for this area would become very high. The threat of a considerable aggravation of the situation in the region as a result of BiNPP shutdown is a decisive factor for substantiating the extension of BiNPP operation beyond the design lifetime.

9.3. Assessment and substantiation of BiNPP safety in terms of extending its operation time

According to the commissioning dates the Bilibino reactor plant (EGP-6) belongs to the first generation reactors. The BiNPP design was finished before basic regulatory documents for NPP safety were issued (the BiNPP’s 1 unit reactor first criticality and publication of the first regulatory requirements on NPP safety took place simultaneously in 1973). For this reason, the BiNPP has a considerable number of deviations from some requirements of the current regulatory documents. Some essential ones have been resolved in the past fifteen years. Due to the above-said, the practical assessment of EGP-6 facilities appears to acquire a decisive significance in the consideration of extending the operation time.

Primarily, the works for BiNPP safety substantiation are aimed at the development in-depth safety assessment report (IDSAR) for BiNPP. The structure of IDSAR is recommended by the Russian Radiation and Nuclear safety authority – Gosatomnadzor (GAN), it includes the Summary volume and four principal supplements. The Summary volume contains the review and evaluation of all factors determining the current level of the power unit safety, with upgrading this level by means of implementation of the measures planned. The Summary volume includes the brief description of the methodology of the analysis accomplished. The information in this volume is represented in an abridged form with references to corresponding supplements, in the scope sufficient for independent expert evaluations.

Detailed safety assessment for a current status of the NPP is to be developed in the following supplements:

Supplement 1 – “NPP safety assessment report”

Supplement 2 – “Materials for extended substantiation of safety”

Supplement 3 – “Probabilistic safety analysis”

Supplement 4 - “Analysis of beyond the design-basis and severe accidents”

9.4. Analysis of engineering capabilities of the technological equipment and structures of BiNPP in terms of extending its operation time

The top level document for the BiNPP life extension is “General program for complex review of BiNPP for the extension of its operation time” prepared by the operating organization – concern “Rosenergoatom”. The objective of complex review of BiNPP is obtaining technologically substantiated information:

- on the status and state of main buildings, structures, and structural units of BiNPP
- on the status and state of the balance-of-plant equipment and elements, including their lifetime characteristics
- on the status and state of the nuclear island equipment and elements and residual operating resource in the operation mode and conditions accepted by the time of review
- on actual status of the system of BiNPP operation management and control and their accordance with regulatory standards and requirements now in force, which regulate safe implementation of work in the nuclear power industry.

The review of any facility commences with studies of critical elements (conventionally irreplaceable or difficult to replace objects). This category at BiNPP includes:

- concerning the balance-of-plant equipment and systems:
  - main building of the NPP, including the basement slab;
  - the water reservoir dam.
- concerning the reactor unit equipment and systems:
  - graphite pile of the reactor;
  - metallic structures of the reactor (the bottom slab, reactor vessel, biological shielding tank);
  - drum-separator.

In order to assess the state of the elements listed, special investigation programs or additional techniques of investigation are to be developed obligatorily and approved by the utility. This work is charged to the BiNPP staff, and specialized entities and organizations having the necessary engineering tools and proficiency personnel licensed by the regulatory authorities.

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IAEA-SR-218/60

THE PRESENT ROLE AND THE FUTURE OF SMALL AND MEDIUM SIZED REACTORS IN THE BULGARIAN ELECTRICITY GENERATION INDUSTRY

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Abstract

At the moment, six PWR units WWER type are in operation in the Bulgarian electric grid, located in Kozloduy NPP. All of these units operate in the basic load diagram. There are four reactors of 440 MWel and 2 reactors of 1000 MWel. During the last ten years the total demand of power in the national grid varies between 3 500 MW during the summer and 7 500 MW in the winter. For certain periods the capacity even of a single reactor WWER-440 has exceeded more than 10% the total system demand. This fact has imposed some restrictions on reactor operation, but the reactors nevertheless operate in the load diagram basis. More often these restrictions are in form of operation at reduced power. For WWER-440 reactors this means operation at 50-55% (with one turbine). This fact leads to some negative consequences as for example unbalanced fuel burning and increased core peaking factors.

Another sensitive problem for the plant is the case when due to technical specification operational limits and conditions violation one unit has to be shutdown and put in a cooldown condition for a period of time for different reasons (forced outage, maintenance,etc.). In some cases, this is not desirable for the grid dispatchers due to lack of redundant generating capacities. As far as Kozloduy NPP is concerned, since April 2000 it has been separated from the National Electric Company (NEC) as a stand-alone electricity generating company, it suffers some penalties applied by NEC, based on the contract for purchase of the electricity between NEC and the plant.

The scheduled modernizations of the units will lead to some extensions of the outages, which will force NEC to put into operation less profitable power plants in the country.

In order to cope with the problems of the grid frequency regulations and the backup of the power generation capacity of the biggest unit in the national grid, which is WWER-1000 MW, a pumping accumulation hydro-station with capacity 840 MW in turbine mode was constructed and put in operation in Bulgaria in 1998.

In spite of above-mentioned, Kozloduy NPP staff operate the plant units with definitive priority of safety.

This year 2001, the State Agency for Energy and Energy Resources of Bulgaria together with National Electric Company elaborated an update of the National plan for development of electricity generation capacities in the grid. According to this Plan, a new nuclear generation capacity with 2 units has to be constructed and put in operation on the site of Belene between 2008-2015. The preliminary feasibility study of this project, performed by the national architect-engineering company Energoproekt shows that the optimal unit capacity for this project is between 600-900 MWel. The preliminary engineering activity of the project is expected to start effectively by the end of 2001.

1. THE PRESENT ROLE OF WWER-440 UNITS IN THE BULGARIAN ENERGY SYSTEM

Four units equipped with the Russian designed WWER-440 reactors (B-230 type) are in successful operation in Kozloduy NPP since 1974, when unit 1 was commissioned. Units 2, 3 and 4 were put into operation respectively in 1976,1982 and 1984.

Safety

All deviations from the normal operation of Kozloduy NPP are recorded, analyzed and reported to the Inspectorate of Safe Use of Atomic Energy in compliance with the Bulgarian legislation and the internal regulatory requirements. The number of recorded events in 2000 and their levels, evaluated under the IAEA INES-scale, are an indicator for the plant safety during the period.

Distribution of the events in 2000 by scale levels and by units are reported below:

Unit/ INES level events	**I**	**II**	**III**	**IV**	**V**	**VI**	**Total**
"1"	0	0	1	2	0	0	3
"0"	4	11	10	12	10	8	56
Total number of events	8	14	16	20	12	12	84

Fifty-six events were assessed as "0" level, below the INES-scale, since they were related to safety but did not result in its impairment.

Three events in 2000 were assessed as level "1" and they brought about partial deterioration of the defence-in-depth. The total number of safety related events in 2000 under the INES-scale were 59, which represents 86.82% of the total number of reported events. The total number of reactor SCRAMS in Kozloduy NPP in 2000 is better than the average performance for this type of reactors and is a characteristic of their acceptable operational reliability.

There were no events in 2000 leading to environmental or plant site radiation contamination.

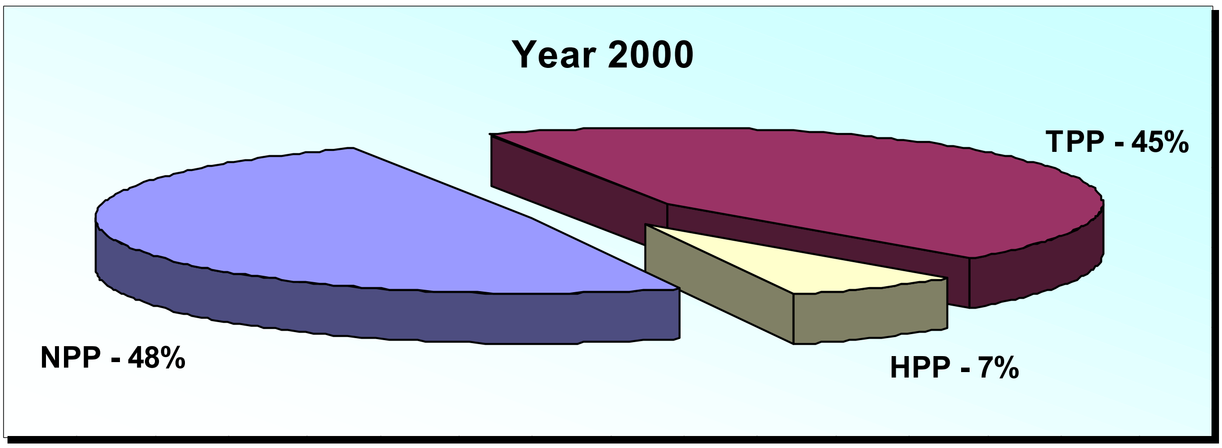
The relative share of safety related events compared to the total number is nearly constant during the years and is about 60%.

In 2000 a considerable reduction of the number of events is registered caused by decrease of the human errors. This fact gives evidence that the safety culture in Kozloduy NPP has been improved significantly compared to 1998 and 1999. All these data reveal the full transparency of Kozloduy NPP operation and safety reporting.

Electricity Generation

The share of the electricity generated in the National energy system of Bulgaria for year 2000 is as follows:

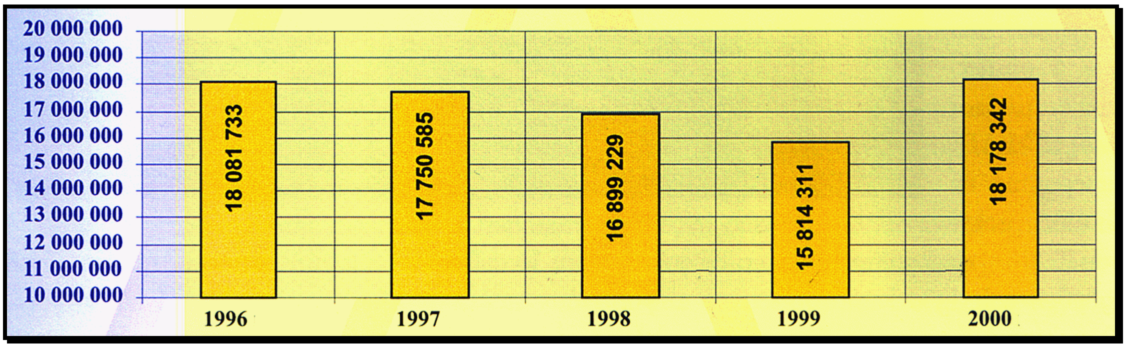
- Kozloduy NPP – 18 160 MWh
- Thermal Power Plants – 16 168 MWh
- Hydropower Plants – 2 450 MWh



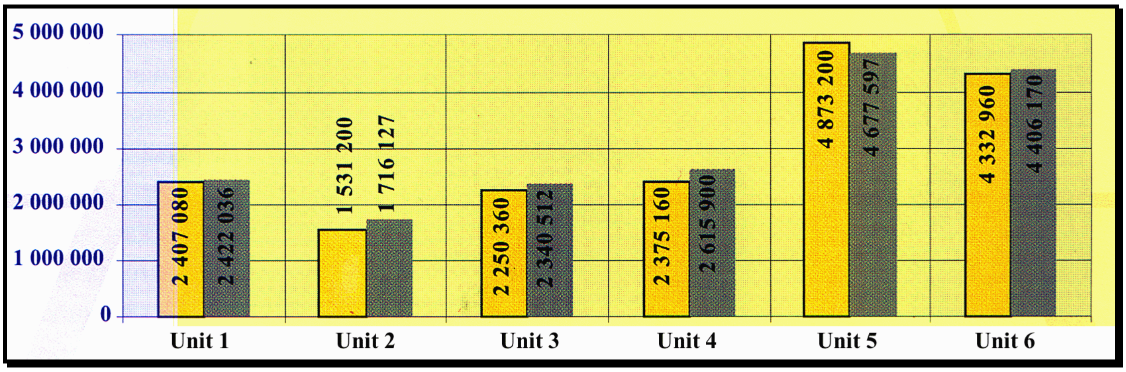
After downsizing of the National Electric Company of Bulgaria (NEK) in 2000 when Kozloduy NPP was established as a stand alone public limited company, it signed a contract for sale of electricity and provision of standby backup capacities to the "Single Buyer". The period of the contract was yearly based – for the period May-December 2000. For 2000 the contract was fulfilled at 103% in terms of net sale of active electricity. In terms of provision of the backup generating capacities the obligations were fulfilled 96.92% for the duration of the contract.

In 2000 Kozloduy NPP hit a record in the electricity generation, producing of 18 178 342 MWh in full compliance with the safety requirements, against the planned 17 769 960 MWh. Compared to 1999 it gives an increase of the annual generation with about 19 %.

Kozloduy NPP electricity generation in the last years [in MWh]:

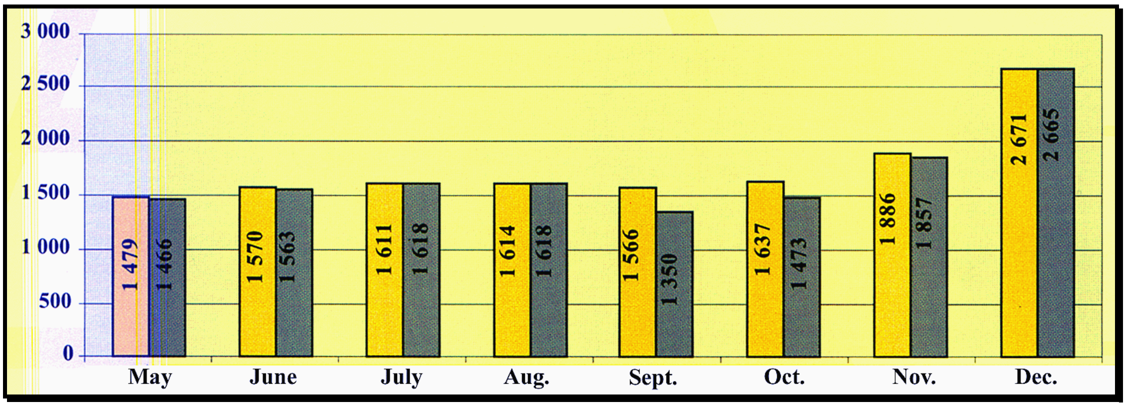


Electricity Generation by units [in MWh]:



The generation keeps its seasonal character – it begins with a drop already in the middle of March, the tendency continues by the end of April and May and then fixes a stable summer load of about 1400 MWel. The main indicator for the effective use of the installed capacity throughout the year is the average power output of the plant. This factor for 2000 is 2 069 MW and represents 4,9% increase of the installed capacities, compared to 1 805 MW for 1999. The average annual workload for the time in operation is 2 986 MW.

Planned and actual availability for the period May–December 2000 in GWh



2. GENERAL CONSIDERATIONS FOR CONSTRUCTION OF NEW NUCLEAR POWER PLANT

The forecasts for energy sector development based on the forecasts for general economic development of the Republic of Bulgaria shows necessity new nuclear electric power generating capacities to be implemented as a replacement capacity of the planned decommissioning of units 1 to 4 of Kozloduy NPP. Nevertheless there are plans for decreasing of energy intensity of the industrial production by improvements of the energy efficiency , for larger utilization of renewable energy sources in the energy sector and natural gas for the household consumption it will be a precondition for reliable electric power supply and stable development of our country.

The emerging European trends for the development of the energy sector (moratorium over the development of the nuclear energy in Germany, eventual worsening of the Russian capabilities for electricity export due to the amortization of the existing capacities within the next 10 years) as well as the international and European trends of the development of the markets of primary energy sources, could lead to a certain, even temporary stagnation of the presently developing European energy market that will have double impact on the energy sector in Bulgaria.

The restrictions of green gas emissions combined with the decommissioning of nuclear capacities and possible increasing of the oil and natural gas prices could impose difficulties on construction of new conventional power plants within pan-European extent that could lead to a temporary market shortage of electric power after the year 2008.

The above circumstance as well as the relative isolation from the large electric power transmission capacities in the region in the center of which Bulgaria is located calls in question the capabilities for electricity export in case our country has sufficient installed capacities available. In case of electric power shortage of our country, it will be difficult to cover it by import from neighbour energy systems even through UCTE. If our energy system would not have technical capabilities and capacities available to ensure the possible electric power export a favourable conjuncture moment for the Bulgarian power sector will be missed.

The accession of the Republic of Bulgaria to different Conventions for environmental protection also determines the construction of new nuclear electric power generating capacity as a most realistic opportunity to meet the future electric power needs simultaneously fulfilling the obligations undertaken under these conventions.

There are some preliminary information that the public opinion in the country in general accepts the necessity for development of the nuclear power sector. The expected positive social effects both in regional and national extent could only enforce this public believe.

During the consideration of the issue the expected significant positive social and economic effect resulting from construction of new NPP in our country not only in regional but also in national scale shall not be neglected. It is expected for a long-term period at least 15-20 thousand new working places to be opened as well as number of companies and industrial enterprises to be involved. It is doubtless the impact on the general social and economic development of the Republic of Bulgaria will be beneficial.

3. SITE SELECTION FOR THE CONSTRUCTION

For the moment there are only 2 sites on the territory of country which are unconditionally proved as suitable for NPP construction – Kozloduy and Belene sites.

The possibilities for construction of new nuclear power capacities on these detailed investigated sites are considered. The analysis of the condition of both sites and of the additional factors shows that the Belene site is more suitable for construction of new nuclear capacity.

For both sites within the period 1990-1997 under the management of the International Atomic energy agency based on program "*Studies and activities for improvement of reliability of the sites of Belene NPP and Kozloduy NPP*" developed for this purpose entirely based on the international approved document of IAEA N 50-C-S "Safety of Nuclear Power Plants – selection of NPP site" additional studies were made aiming to clarify their suitability for NPP construction.

The new data about Kozloduy site only confirm once again the already known ones and they clarify some of the quantitative features of certain parameters. It is possible construction of new capacities next to the existing ones to be done.

The results of the new course of studies of Belene site made mainly by BAS, including results of the needed additional studies of the design seismic characteristics of the site, were considered, discussed and accepted during the last two missions of IAEA (10-14 October 1994 and 4-7 June 1995). The seismic characteristics together with the entire data base of engineering geology, tectonic and seismic of the site and region were subject of discussions during the last IAEA mission held on 7 till 11 July 1997 in Sofia.

The main conclusion of all held missions and analyses is that from the point of view of seismic tectonic and seismic risk there are no conditions impeding the Belene site to be used for construction of nuclear power station. Due to the pressure of some non-governmental organizations in the country more additional seismic studies were ordered aiming data to be proved unconditionally. They were made till the middle of year 1999 by *Bulgarian Academy of Sciences -Geological Institute*. All conclusions were confirmed in uncontestable way. The main conclusion of these additional studies of Belene Site [BAS, Geological institute, "Additional Geological studies in Belene NPP region], accepted on 5 August 1999 by Expert Council of Geological Institute of BAS and on 13 August 1999 by the Technical Council of NEK-EAD is as follows: "*….under the Belene site and next to it (within the local 5 km zone) there are no traces of action of active fractures during the Quaternary (or at least during the last several hundred thousand years) including the area alongside the Danube river. On this* *point of view the site meets the IAEA requirements (items 601 and 604, page 27 of Safety Series 50-SG-S1, Rev.1, 1999) and it is suitable for construction of nuclear power station."* Based on the meteorological monitoring held during the last recent years it was found out that the meteorological conditions are favorable for construction of nuclear facility.

From the point of view of the electric power grid stability the Belene site is preferable because the commissioning of new nuclear facility will increase the reliability of the grid system and will decrease the electric power transmission losses.

Regarding all rest factors – demography, extreme impacts etc. – no results impeding the opportunity for construction of new nuclear capacity were found out. By the construction of the replacement nuclear capacity on the Kozloduy site the investment concentration at one site will occur, the infrastructure will be overloaded and the last, but not least, the risks will be increased that makes this site not suitable for new construction within a medium term prospect. Taking into consideration the constructed large infrastructure, readiness to start the construction works, the possibility to use the constructed facility that would allow to accelerate and perform cheaper the construction works as well as the positive social effect on the region the Belene site is preferable for construction of new replacement nuclear capacity.

4. NUCLEAR INSTALLATION TYPE

While considering the possibilities for construction of new types of nuclear power stations several main factors are taken into account:

- safety level;
- possibility for eventual partial use of the equipment supplied to Belene NPP, unit 1 and the facilities constructed there;
- single unit capacity suitable for our electric power system;
- availability of project ready for commercial offer;
- gained experience and possibilities for most full value use of the existing domestic scientific and engineering potential.
- Nevertheless the fact that presently new projects are under development that are in different stage of readiness, the above considerations led to the narrowing of the range of considered possibility as follows:

Option 1 – Construction of 2 new units with 600 MWel unit capacity under the AP-600 project of WESTINGHOUSE;

Option 2 – Construction of 2 units with about 640 MWel unit capacity under the B-407 project developed by ATOMENERGOPROEKT – St. Petersburg.

Option 3 – Construction of 2 units with about 900 MWel unit under the European project with possibilities for partial utilization of the engineering infrastructure already available on site;

Option 4 – Completion of the “frozen” construction of Unit 1 of Belene NPP as a modernized option of WWER-1000 unit similar to Temelin NPP in Czech Republic (provisional indication B-230M);

The main considerations for the above selection are:

- in our country quite valuable and high quality experience was gained in operation of PWR reactors (WWER in Russian);
- PWR reactors are presently broadly used worldwide, they are relatively easy for management, the applied technologies and their safety are proven by the decades of operation of the plenty of NPP facilities with this type of reactors.
- Possible partial utilization of the supplied equipment and constructed infrastructure on the Belene site;
- On the point of view of the stability of the national electric grid the most suitable are nuclear installations with single unit capacity in the range of 500–600 MWel. Single unit capacity of 1000 MWel will have forced reduced annual utilization (by about 15%). The unit capacity higher than 1000 MWel is unacceptable, nevertheless its performance characteristics could be the best;
- The necessity to construct capacity of new generation under a new project having obviously higher safety and reliability indicators as well as considerably improved technical economic characteristics.
- In the 2 tables below a comparison of the features of 4 discussed options for construction of the new nuclear installation is presented.

TABLE I. MAIN SAFETY CHARACTERISTICS

Indicator	Dimension	Option			
		1	2	3	4
Core damage frequency	1/r.year	$1,6.10^{-6}$	$1,8.10^{-5}$	$1,6.10^{-6}$	$\leq 3,10^{-6}$
Probability for radioactivity release outside the containment	1/r.year	$\leq 3.10^{-7}$	$<1.10^{-7}$	$<1.10^{-7}$	$<1.10^{-7}$
Dimensions of the sanitary protection area in case of DBA	Km	<1	<1	<1	<1
Emergency Planning Area	Km	No data	No data	<5	<7.5

TABLE II. MAIN TECHNICAL PARAMETERS OF THE OPTIONS:

Indicator	Dimension	Option			
		1	2	3	4
Construction period including licensing and site preparation	Year	≈6	<9	≈6	<8
Operation Life Time	Year	50	50	45	35
Gross installed power capacity	MWel	610	645	900	1000
Annual design utilization	eff.hrs	>7000	>7000	6500	6500

It has been found that, options 3 and 4 have certain advantages because they provide a possibility to utilize a part of the investments made in the past and frozen for during 10 years already.

On the other hand, some specific difficulties are expected (in **Option 4**) related to the upgrading of the project reflecting the modern safety requirements. This project is expected to be relatively risky due to some difficulties associated with the licensing and the completion schedule . The large single unit capacity of this project and its expected low utilization due to grid requirements are an important disadvantage.

As it could be seen from the data in the above tables the new designs of the reactor installation with medium capacity (Options 1 and 2) have certain advantages regarding requirements to the grid stability also from the safety point of view (especially comparing with Option 4). Their compliance with the safety standards presently in force is unconditional and internationally accepted. The main disadvantages are comparatively higher investment costs, but especially important disadvantage is the fact that both types of reactor installations do not have a prototype implemented anywhere in the world. No doubt it will lead to long and more difficult licensing of these projects. In this regard the option 3 is more advantageous because these reactor installations are of serial type and a lot of them are in operation in Europe. They are offered on the market in "turn key" basis. The total investment costs for such a project amount to about USD 1250 millions .

Analyzing the advantages and disadvantages of the considered options and mostly taking into account the risk factor, a conclusion could be made that for the present conditions in Bulgaria a selection of new generation reactor facility with medium capacity is most expedient, typical representatives of which are option 1,2 and 3.

However it also has to be considered that the available information on new designs of nuclear installations and their technical and performance indicators is limited and this is a reason at this stage not to be possible to review full range of probable nuclear installations and to make a concrete selection based on precise technical and financial comparisons. The needed additional information could be obtained only after a Bid launching and preliminary commercial discussions.

For the entire clarification of all issues it is necessary the following principle subsequence of actions to be implemented:

- Principle decision to be made for construction of the new replacement capacity according to the Energy and Energy Efficiency Act provisions while simultaneously informing the public on the motivation and scheduled activities;
- Actions to be undertaken to look for possible investors by an international tender procedure. Such investors are expected to be foreign companies in joint ventures with Bulgarian ones like nuclear operators, utilities, grid operators, etc.;
- Clarification of the conditions for financing and crediting;
- Negotiations with potential suppliers of equipment;
- Announcement of open international tender for selection of an architect-engineering company and effective start of the project.

SMALL SIZED REACTOR FOR LASER RADIATION

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Abstract

Substantiation of the possibility of the creation of an autonomous nuclear power plant generating laser radiation is given in this paper. The work of the power plant is based on the use of the small sized reactor, generating electric energy, with non-self maintained discharge lasers built-in reactor core or reflector, having high flux density of a thermal neutron $\Phi_{Th} \geq 1,0.10^{13}$ $cm^{-2}.s^{-1}$ in the channels, where the lasers are located.

A thermionic fast reactor-converter with beryllium reflector of the space nuclear-energetic installation was chosen for consideration by us, because the neutron radiation and electric energy necessary for operating of non-self maintained discharge lasers should be produced in one nuclear reactor.

To prove the possibility of operation of the non-self maintained discharge lasers in the reactor and estimate parameters of the laser systems in the reactor, we used the results of experimental and computing researches of the neutron characteristics of non-self maintained discharge lasers built-in the beryllium reflector. These experiments were carried out by us on the critical assembly PhS-1, simulating that thermionic fast reactor-converter. Moreover, we used the research results of the in-reactor diagnostics of the nuclear-excited plasma of the laser gas mixtures, data of experimental characteristics of non-self maintained discharge, threshold and output data of lasers of different waves lengths, carried out by us earlier on the Kazakhstan Research Nuclear Reactor WWR-K.

Thus the possibility of achieving an autonomous, compact nuclear power plant generating not only electrical energy, but also laser radiation concerning the large capacities of infra-red and ultra-violet range of waves lengths in stationary and pulse modes is shown.

1. INTRODUCTION

Substantiation of possibility of the creation of an autonomous nuclear power plant, generating laser radiation, is given in this paper. The work of the power plant is based on the use of small sized reactor, generating electric energy, with high specific energy issues, with non-self maintained discharge lasers, built-in reactor core or reflector.

The application of a nuclear reactor allows the excitation of lasers of large working volumes and at large pressures. The reactor installation with non-self maintained discharge lasers will allow to obtain a stationary laser generation or laser generation with high frequency of impulse replication and correspondingly to provide a high average level of power of laser radiation, as against impulse reactors with a direct nuclear pumped lasers. Moreover, non-self maintained discharge in reactor will allow to pump a row of lasers in the visible and ultra-violet (UV) ranges (for example, ultra-violet excimer laser on XeF *), that is not accessible to direct nuclear pumped lasers in impulse nuclear reactors because of the longer duration of pumping impulse.

It is important, that the neutron irradiation and electrical energy necessary for operating of non-self maintained lasers should be produced in the one nuclear reactor. Therefore, a thermionic fast reactor-converter with a beryllium reflector in a space nuclear-energetic installation [1] was chosen by us for consideration.

2. NEUTRONS CHARACTERISTICS OF THE THERMIONIC REACTOR-CONVERTER CORE WITH NON-SELF MAINTAINED DISCHARGE LASERS BUILT-IN REFLECTOR.

The Physico-Energetical Institute (Obninsk, Russian Federation) provided us a possibility to carry out research on the critical assembly PhS-1 [2], which simulated **a** thermionic fast reactor-converter (TRC) with beryllium reflector.(See Fig.1). The study of the neutron characteristics of the thermionic reactor- converter was carried out on the critical assembly PhS-1, which included the physical models of the lasers built-in the beryllium reflector (see Fig.1).

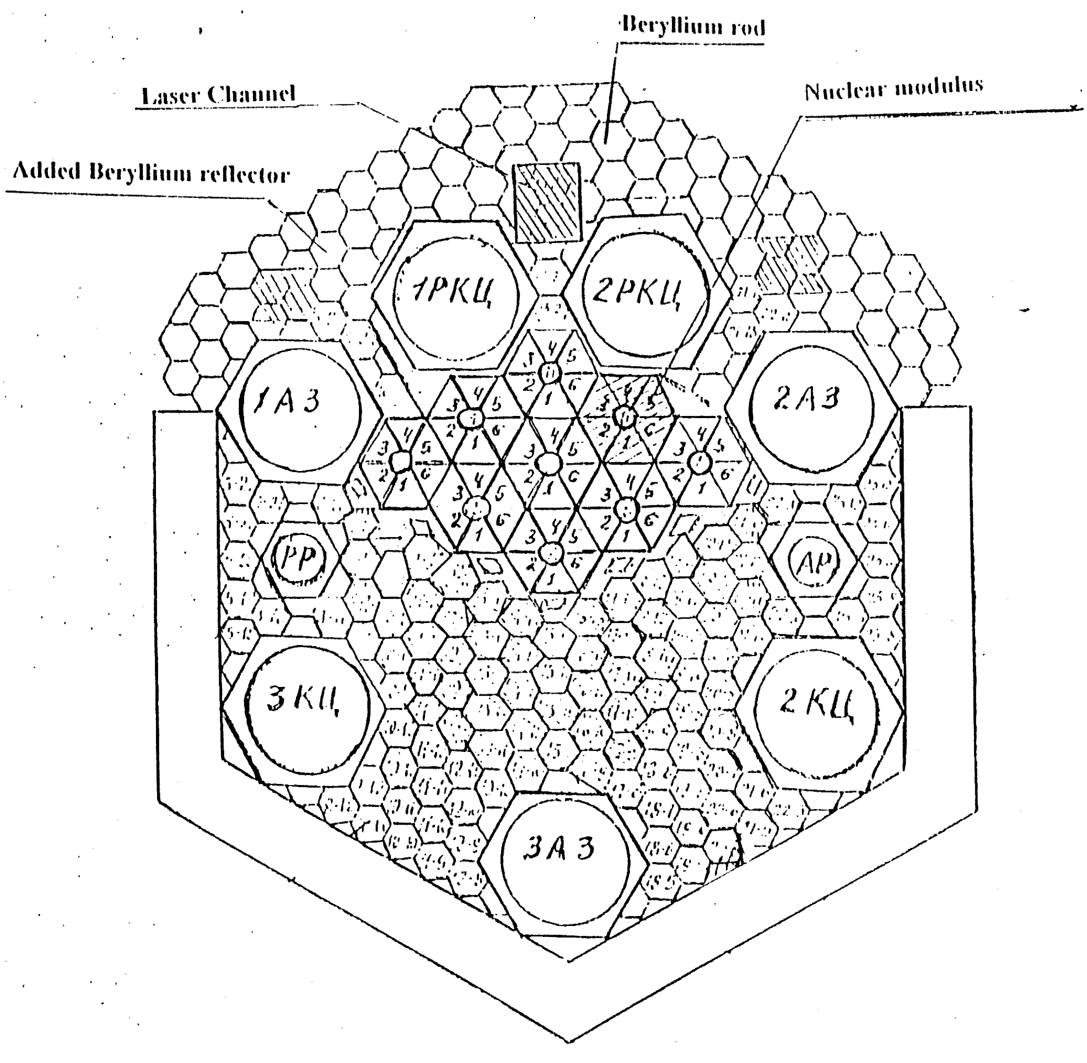

FIG. 1. Plan of the critical assembly PhS-1, simulating thermionic reactor-converter.

The beryllium reflector thickness of the critical assembly was increased by an additional thickness 10 cm (total - 25 cm) to create space for laser model channels disposal and to increase the reactivity stock to compensate negative reactivity connected with 12 lasers models channels disposal. It needed also to increase flux density of thermal neutrons in the place of location of channels with laser models.

TABLE I. NEUTRON CHARACTERISTICS OF THE CHANNEL WITH LASER CHAMBER PLACEDIN THE EXTENDED BERYLLIUM REFLECTOR OF THE THERMOIONIC REACTOR-CONVERTER (TRC)

Lasers place in critical assembly	Measurement place	Experiments				
		Detector	Φ_T For W_{Th} = 1 wt For W_{Th} = 7 Mw	Cadmium ratio	Φ_T^{Eff} For W_{Th} = 1 wt For W_{Th} = 7 Mw	Nuclear reaction rate $^3He(n,p)^3H$ $cm^{-3}.c^{-1}$
Be - reflector	Inside laser Scale-model	Au	$(1{,}02 \pm 0{,}17).10^6$ $(6{,}8 \pm 1{,}1).10^{12}$	$0{,}20 \pm 0{,}02$		
		Counter $^3He(n, p)^3H$	$(1{,}1 \pm 0{,}2).10^6$ $(7{,}1 \pm 1{,}2).10^{12}$	$5{,}6 \pm 0{,}8$	$(1{,}2 \pm 0{,}2).10^6$ $(8{,}4 \pm 1{,}4).10^{12}$	$(1{,}8 \pm 0{,}3).10^5$ $(1{,}2 \pm 0{,}2).10^{12}$
Calculations of Monte-Carlo method						
Be - reflector	Inside laser Scale-model		$(2{,}34 \pm 0{,}05).10^6$ $(1{,}6 \pm 0{,}03)\ 10^{13}$	$5{,}3 \pm 0{,}3$	$(2{,}78 \pm 0{,}05).\ 10^6$ $(1{,}9 \pm 0{,}1).\ 10^{13}$	$(3{,}7 \pm 0{,}05).10^5$ $(2{,}5 \pm 0{,}04).10^{12}$

The neutron characteristics measured by experiments and calculated by a Monte-Carlo method are shown in Table I. Absolute and effective values of flux density of thermal neutrons, specific rates of reactions ^{3}He (n, p) ^{3}H, normalized to the power of the critical assembly of 1W and on thermal power of the thermionic reactor-convertor ~ 7 MW are given in this Table. Given in the Table the effective flux density of thermal neutrons is such one at which reaction rate $^3He(n, p)^3H$ is equal to the rate of this reaction on a complete spectrum of neutrons $\{\Phi_{Th}^{eff} = (1+1/R_{Cd}^{He-3})\Phi_{Th}\}$.

The experimental values of neutron flux density are lower than the calculated ones, obtained by a Monte-Carlo method, because additional Be- reflector was grown up only in solid angle 120° and had void content.

The values obtained for the effective flux density of thermal neutrons for laser channels located in the beryllium reflector with enlarged width were $\Phi_{Th}^{eff} > 1{,}0.10^{13}$ $cm^{-2}.s^{-1}$. These values are quite sufficient for effective operation of the lasers considered, as will be proved hereinafter.

3. CHARACTERISTICS OF THE LASERS, EXCITED BY NON-SELF MAINTAINED DISCHARGE IN REACTOR RADIATION.

To estimate parameters of laser systems in the thermionic reactor-converter, we had used the results of the following research carried out by us earlier on the Kazakhstan Research Nuclear Reactor WWR-K: diagnostics of nuclear-excited plasma of the lasers' gas mixtures [3,4,5];

experimental measurements of characteristics of non-self maintained discharge [4,6] and threshold and output result of generations of different lasers [6,7,8]. Moreover, we had used references data of high efficiency and large-power lasers with ionization by electron beam.

The electron density in a plasma of the gas mixtures of the CO_2- and CO- lasers ($CO_2:N_2:{}^3He=1:4:5$, $CO:N_2:{}^3He=1:6:7$), obtained experimentally by an electrical probe method and by calculation, versus gas pressure and flux density of thermal neutrons are shown in Fig.2 [3,4]. Ionization of the gas medium was carried out by products of nuclear reaction ^3He (n, p) ^3H. According to the references data, the influence of a space charge on the electrical current of the discharge in the plasma becomes non-essential and the volt-ampere characteristics of the discharge achieves linear Om function at an electron density $N_e=10^{10}$-10^{11} cm^{-3} [17]. In our case, the indicated electron density was achieved at thermal neutron flux density $\Phi_{Th} \geq 5.10^{12}$ $см^{-2}.s^{-1}$.

The volt-ampere characteristics of non-self maintained discharge of the CO- laser are shown in Fig.3 [4]. Due to Figure the volt-ampere characteristics of the discharge has linear Om function at thermal neutron flux density $\Phi_T \geq 5.10^{12}$ $см^{-2}.s^{-1}$.

Figures 2 and 3 indicate the realization of high current discharge, excited by electroionization method [18] at thermal neutron flux density $\Phi_T \geq 5.10^{12}$ $cm^{-2}.s^{-1}$. Hereinafter, the threshold of the generation of the electro-ionization infra-red CO- and CO_2 – lasers was achieved experimentally in the core of the Kazakhstan research nuclear reactor WWR-K at thermal neutron flux density $\Phi_T \geq 5.10^{12}$ $см^{-2}.s^{-1}$ [6,7,8].

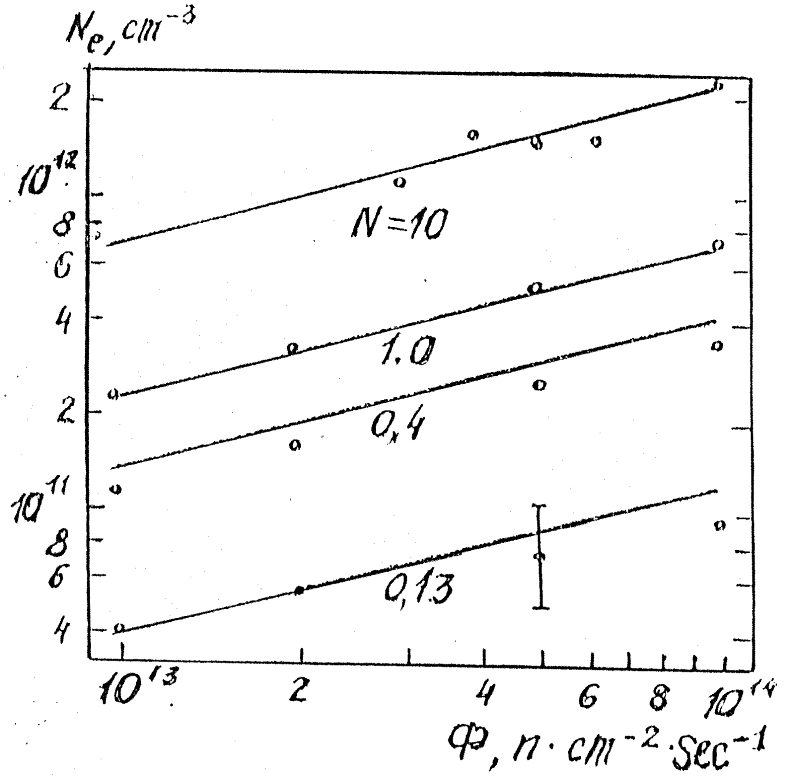

FIG. 2. Electron density versus thermal neutron flux density and total gas pressure in $CO_2 : N_2 : {}^3He = 1 : 4 : 5$ and $CO : N_2 : {}^3He = 1 : 6 : 7$ gas mixtures.

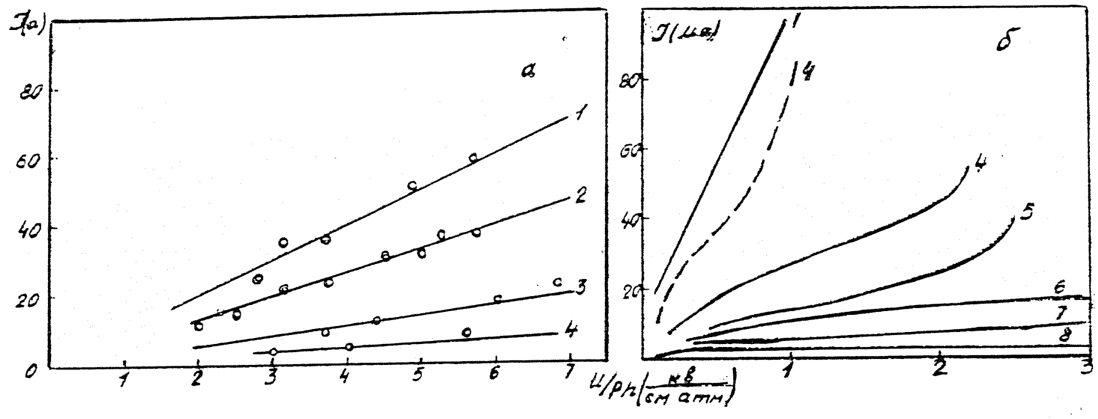

FIG. 3. Current of discharge of CO–laser versus parameter U/ph (kv/cm.atm) and thermal neutron flux density: a - impulse discharge; b –stationary discharge; 10^{14} n/cm^2.c -(1) , 5.10^{13} -(2), 10^{13} -(3), 5.10^{12} -(4), 2.10^{12} -(5), 10^{12} -(6), 5.10^{11} -(7), 10^{11}-(8).

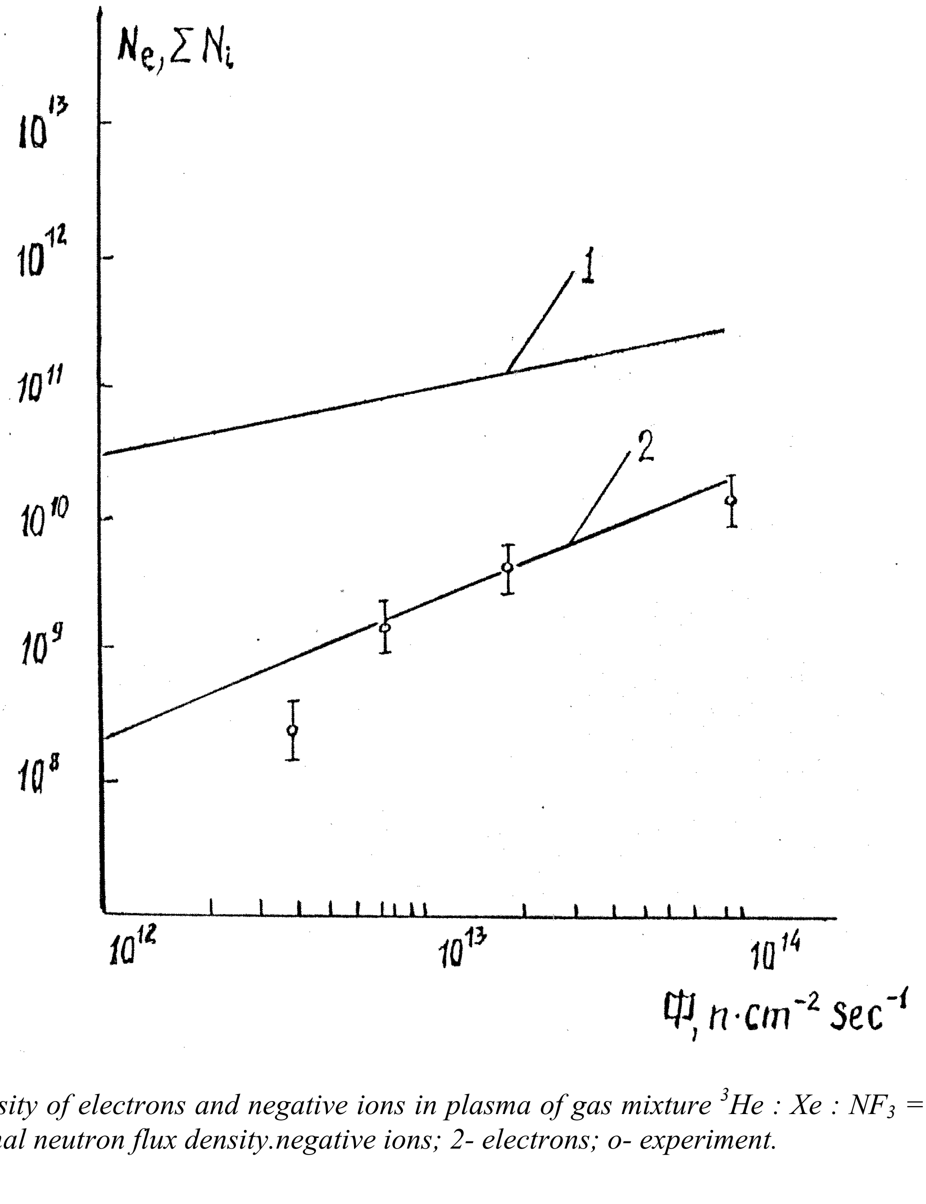

FIG. 4. Density of electrons and negative ions in plasma of gas mixture $^3He : Xe : NF_3$ = 760 : 3 : 2 versus thermal neutron flux density.negative ions; 2- electrons; o- experiment.

The investigation results of the gas mixtures plasma of ultra-violet XeF* excimer laser ($^3He:Xe:NF_3$=350:1,5:1) are shown in Fig.4 [5]. As can be seen, the minimum of concentration of negative ions $N_i = 10^{10}$ cm^{-3} and electrons $N_e = 10^8$ cm^{-3} in the plasma necessary for operation of electrical discharge excimer laser with ionization by electronic beam, are reached at flux densities of thermal neutrons $\Phi_T \geq 10^{12}$ $cm^{-2}.s^{-1}$. It shows the possibility for operation of electro-discharge excimer laser with nuclear ionization in a steady-state nuclear reactor at flux density of thermal neutrons $\Phi_T \geq 10^{12}$ $cm^{-2}.s^{-1}$. This threshold of generation $\Phi_T \approx 10^{12}$ $cm^{-2}.s^{-1}$ was confirmed by us in generation experiments of the ultra-violet eximer XeF*- laser into WWR-R reactor.

4. APPROXIMATE EVALUATIONS OF THE LASER SYSTEMS' PARAMETERS IN THE THERMIONIC REACTOR-CONVERTER

The important result of our investigation is the good correlation of results of in-reactor plasma diagnostics and discharge, threshold and output characteristics of created CO- and CO_2- lasers excited by non-self maintained discharge in the WWR-K reactor with analogous parameters of lasers with ionization by e-beam in the similar conditions. It allows hereinafter to use some research data of high efficiency and large-power lasers with ionization by electron beam for estimation.

TABLE II. APPROXIMATE ASSESSMENT PARAMETER OF CO_2 - ,CO - LASER SYSTEMS, WHEN THEY PLACED IN THE BE – REFLECTOR OF THE THERMIONIC REACTOR-CONVERTER (TRC)

Parameter	Lasers in Be –reflector $W_{EL.}$ = 500кW (for load)
$V_{DISCH.}$ (laser) (l)	1 laser - 0,92 l 3 lasers - 2,76 l. 12 lasers - 11 l.
$V_{ACT.}$ (laser) (l)	1 laser - 0,92 l 3 lasers - 2,76 l. 12 lasers - 11 l.
Φ_T^{Eff} ($cm^{-2}.c^{-1}$) for P (^3He) = 0,5 atm)	(2.5 ± 0,4) 10^{13}
$\Phi_{T.}^{EFF.}$ ($cm^{-2}.c^{-1}$) for (P (^3He) = 1 atm)	(1,9 ± 0,4) 10^{13}
Efficient CO_2 - laser Efficient CO - laser	15 ± 5 40 ± 10
Longitudinal circulation	
$W_{DISCH.}^{CO_2,CO}$ (кW)	1 laser. - 15 12 lasers. – 180
$w_{CP.DIS.P.}^{CO_2,CO}$ (Wt. cm^{-3})	16
$W_{RAD.}^{CO_2}$ (кW)	1 laser - (2,0 ± 0,7) 12 lasers - (21 ± 7)
$W_{RAD.}^{CO}$ (кW)	1 laser - (5,0 ± 1,6) 12 lasers - (60 ± 15)
Lateral circulation	
$W_{DISCH.}^{CO_2,CO}$ (кW)	3 lasers - 500
$w_{CP.DIS.P..}^{CO_2,CO}$ (W.cm^{-3})	~ 180
$W_{RAD.}^{CO_2}$ (кW)	3 lasers - (70 ± 25)
$W_{RAD.}^{CO}$ (кW)	3 lasers - (160 ± 40)

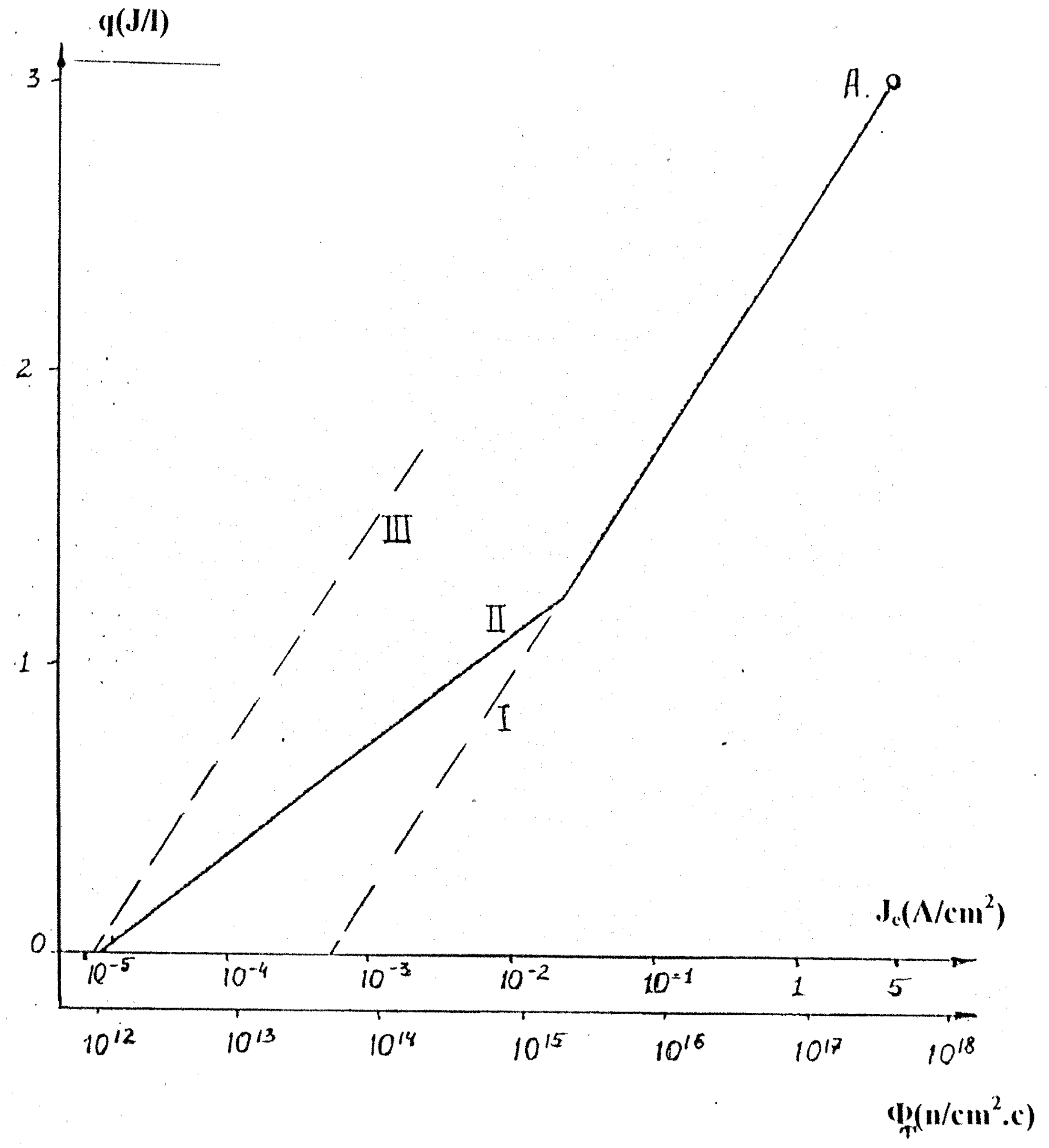

FIG. 5. Specific radiation energy versus flux density of thermal neutron or current density of electron beam.

The analysis of experimental results of plasma diagnostics, discharge, threshold and output characteristics of the CO_2- and CO lasers, and also those of similar lasers with ionization by an electronic beam [9,10,11], has allowed for CO- laser for T_G = 100 K to accept values of efficiency η=(40±10) % and for CO_2- laser for T_G = 300 K – η = (15±5) %.

While evaluating power of CO and CO_2- lasers it needs to take into account the main limitation due to the extreme energy deposition on one gram of pumped gas of laser, which is determined by discharge stability. This value of extreme specific energy deposition equals to~300J.g^{-1}.

The laser generation threshold at atmospheric pressure can not be reached due to a limitation of specific energy deposition. To decrease the generation threshold it is necessary to decrease the gas mixture pressure to P = 0,2 atm. The powers of laser radiation of CO_2- and CO- lasers were evaluated accounting for efficiency and received discharge power value.

TABLE III. APPROXIMATE ASSESSMENT PARAMETER OF EXIMER XEF* - LASER SYSTEM WHEN THEY PLACED IN THE BE – REFLECTOR OF THE THERMOIONIC REACTOR-CONVERTER (TRC).

Parameter	Eximer XeF*- Lasers in the Be - Reflector $W_{EL.}$ = 500 **кw** (for load)
$q_{RAD.}^{EXP.}$ (j.l^{-1})	0,4 ± 0,3
$Q_{RAD.}^{EXP.}$ (j)	1 laser - (0,3 ± 0,24) 3 lasers - (0,9 ± 0,72) 12 lasers - (4 ± 3)
Longitudinal circulation $W_{RAD.}^{EXP.}$ (кw)	1 laser - (0,10 ± 0,08) 12 lasers - (1,2 ± 0,9)
Lateral circulation $W_{RAD.}^{EXP.}$ (кw)	1 laser - (1,5 ± 1,2) 3 lasers - (4,5±3,6)

The results of the approximate assessment parameter of CO_2 - , CO - lasers systems, placed in the Be – reflector are shown in Table II.

As seen, in the case of longitudinal moving of gas mixture through lasers with rate v = 200 m.s^{-1} only part of electrical power of installation is used for discharge of lasers, because of the indicated limitation of specific energy deposition ~ 300 J.g^{-1}. If the amount of lasers in the beryllium reflector is reduced from 12 to 3, with created transverse pumping at the same speed v = 200 m.s^{-1}, then all the electrical power of installation 500 kW can be realized for discharge of lasers and transformed at corresponding efficiency into coherent radiation CO_2- and CO - lasers.

As seen in Table II, summary power of laser radiation will be (160 ± 40) kW for 3 CO-lasers, located in beryllium reflector with transverse pumping at electric power ~ 500 kW and efficiency η=(40 ± 10) %. It will be (70 ± 25) kW for CO_2 - lasers, respectively. The ratio of the active area volume to the discharge area volume of the lasers equals to π/4. This is taken into account in the estimation.

For all indicated cases the radial durability of optical materials of lasers is provided.

Approximate evaluations of the laser system in the thermionic reactor -converter, based on excimer lasers, are carried out.

Approximate dependence of specific energy output of the excimer laser on XeF* versus flux density of thermal neutrons (or ionization rate) is shown on Fig. 5. It is received on the basis of analysis of parameters of nuclear induced plasma and threshold data of excimer laser generation, together the references for excimer lasers with ionization by accelerated electrons [12,13,14,15,16]. As seen, the specific energy output of excimer lasers, located in the beryllium reflector, equals to 0,4 J.l^{-1} for thermal neutron flux density $\Phi_T^{eff.}$ = 2,0.10^{13} cm^{-2} s^{-1}.

Basic limitations are imposed on the frequency of repetition of impulses. Proceeding from conditions of complete change of gas in working volume between sequential impulses at gas rate v = 200 m.s^{-1}, the frequency of repetition of impulses for longitudinal gas pumping f = 300 Hz, for transverse pumping f = 5 kHz is obtained.

The results of the approximate assessment parameter of the XeF^* eximer lasers systems, when they placed in the Be – reflector are shown in Table III.

An irradiation power 4,5 ± 3,6 kW can be obtained for 3 excimer lasers, located in the beryllium reflector with increased thickness for transverse pumping.

Thus opportunity to realise and independent, compact nuclear power plant generating not only electrical energy, but also lasers radiation concerning the large capacities of an infra-red and ultra-violet range of lengths of waves in stationary and pulse modes is shown.

Note we do not pay attention to many technical problems which will appear during installation and operation of lasers in the thermionic reactor-convertor.

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CEA PROGRAM ON FUTURE GENERATION LIGHT WATER MODULAR REACTORS AND GAS COOLED REACTORS

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Abstract

The CEA programme on "Future Generation Reactors and Fuel Cycles" aims at studying and developing the mean and long term most promising options for nuclear reactors, fuels and reprocessing. These options should contribute to make the nuclear energy a major source of the sustainable development. The program also aims at maintaining at the highest level of competency the technologies with which the CEA will be able to bring to national achievements or international projects in the next decades, projects whose specifications and calendar are today unknown.

These studies on the "Future Generation Reactors and Fuel Cycles" constitute a field privileged for international collaboration.

The corresponding researches are structured in four main axes: Innovations for LWR; Systems of 4th generation; Sodium-cooled reactors; Systems which are the object for survey or exploratory studies. Studies on future nuclear gas technologies are mainly covered by the 4th generation programme (Gen IV).

Within this context, the goals pursued, in particular the minimization of the production of long lived waste and the saving of resources (i.e.: the optimised utilisation of fissile and fertile nuclear fuels), could justify an evolution towards hard neutron spectra and high temperatures, to cover applications other than the electricity production, e.g.: hydrogen production, desalination, cogeneration.

The main R&D axis for these long-term objectives currently the area of Gas Cooled Reactors (GCR).

The corresponding program is structured through eight R&D projects details of which are presented within the paper.

1. INTRODUCTION

The CEA has launched a very ambitious program on Future Reactors and Fuel Cycles aimed at studying and developing the mean and long term most promising options to make nuclear energy a major component of future sustainable development.

This program, open to a very large international cooperation, covers all aspects of systems for nuclear reactors, fuels and reprocessing.

The four main axes of research are:

- Innovations for LWRs,
- Fourth generations systems
- Sodium cooled reactors
- Exploratory studies and survey on other concepts.

2. OBJECTIVES FOR FUTURE REACTORS AND CYCLES

In relation to the objectives for future reactors and their associated fuel cycles, there is a consensus at the international level:

- Economic competitiveness
  - Investment, operation, fuel cycle
- Enhanced safety
  - No off-site effects in case of severe accidents
  - Enhanced resistance of core technology to severe accident damages
  - Towards a strategy of core melting exclusion
- Minimization of long lived radioactive waste production
  - A request from the French government
  - Subsequent need for spent fuel processing and actinides recycling (Plutonium and Minor actinides)
- Efficient use of available fissile and fertile nuclear fuel resources (Pu, U_{dep}, Th...)
- Increased resistance to proliferation risks
- Compliance to fuel cycle integration with the nuclear production plant on the same site
- Potential for other applications than electricity generation

These objectives and in particular the minimization of the production of long lived waste and the saving of resources (i.e.: the optimised utilisation of fissile and fertile materials), could justify an evolution towards hard neutron spectra and high temperatures, to cover applications other than the electricity production, e.g.: hydrogen production, desalination, cogeneration.

3. THE PROJECTS

The above mentioned main areas are structured through thirteen projects managed in parallel:

1. General studies
   - Economic studies
   - Reactor park scenarios
   - Public acceptance
   - Non proliferation
2. Thermal GCR[1] concepts
   - GT-MHR
   - Potential for Pu management
3. HEGCR[2] – Systems studies and core design
4. HEGCR – Fuel and fuel cycles studies
5. HEGCR – Materials resistant to high temperatures
6. HEGCR – Technology of high temperature helium circuits/components

[1] GCR: Gas Cooled Reactor
[2] HEGCR: High Efficiency GCR (High temperature, hard spectrum,…).

7. HEGCR – Potential for hydrogen production and cogeneration
8. PWR – Pu management and other advanced fuels
9. PWR – Advanced designs
10. Sodium Cooled Fast Reactors
11. New processes for spent fuel reprocessing.
    - advanced hydrometallurgical
    - pyrochemical
12. Other advanced nuclear systems
    - Liquid metal cooled reactors (Lead alloys)
    - Supercritical water reactors
    - Molten salt reactors
13. Ad hoc studies
    - Water desalination
    - Space applications.

It is interesting to stress that the aptitude of the technologies under consideration, to also satisfy the required characteristics looked for the systems dedicated to transmutation (critical and subcritical systems studied in addition), led to a strong synergy between the two fields of research, being able to go until carrying out a significant part of the studies in a joint approach (e.g. on core, systems, materials, technology).

3.1. General studies project

Technical-economic studies, like the analysis of nuclear scenarios, will come to support the evaluations of new options and will contribute to orientate the corresponding research.

The safety approach definition and implementation will drive the design efforts to achieve the requirements of improved safety.

Standard assessment methodologies will be defined to guarantee the global coherency of the "Future systems" activity.

Public acceptability and non proliferation concerns will also be tackled within the frame of this project.

3.2. GCR projects

3.2.1 Thermal GCR concepts

In the frame of the "Thermal GCR concepts" project, two main objectives are pursued:

a. To define the role and place of these concepts within the frame of nuclear fleet renewal, both at national and international level.

b. To implement technical support for the international GT-MHR. The PBMR (Pebble Bed Modular Reactor) is also considered among the concepts of interest.

The first of these objectives covers simultaneously strategic studies – in which the HTR must be included (e.g. for the Pu inventory management) – and concepts assessment to check the neutronic, thermodynamic and safety performances.

The answer to the second objective results in the setting up of a program of R&D aimed at (re)acquiring mastery of HTR technology, while concentrating more particularly on the technological issues which concern: the fuel, materials and the technology of the components.

The project content is divided in three main domains: Computer codes development and qualification; Concept evaluations; Technology development.

3.2.2 HEGCR – Systems studies and core design

In the frame of the "HEGCR – Systems studies and core design" project, the main studies are focused on:

- The definition/verification of the feasibility of cores with hardened spectrum with the objective of iso-generation (eventually breeding).
- The definition of one or several reactor images with the architecture of systems (functional, auxiliary and safety related).
- The definition of the operational and abnormal boundary conditions (neutronic characteristics, temperatures, pressures, etc.) for the design of reactor components (fuel, materials,..), i.e. operational and safety concept assessment.

As complementary objectives the project aims:

- The define the R&D needs to look further into the evaluations or to validate the concepts;
- The development of qualified tools (codes for design, dimensioning, operation, safety) necessary to achieve preliminary design studies (APD);
- The implementation of the experimental tools to analyse ignored physical phenomena, to constitute data bases, to examine, in connection with project " HEGCR: Technology of the helium circuits at high temperature ", the technological feasibility of concepts, to take part in the assessment of the needed R&D (duration, cost).
- Possibly the supply of a motivation report for an experimental facility to demonstrate relevant technologies (around 2015).

3.2.3 HEGCR – Fuel and fuel cycles studies

Among the objectives for future generation reactors and cycles, those which relate to fuel and core are the following: increased safety; the minimization of the production of waste and the capacity to burn the waste produced by the current generations of reactors; increased flexibility; increased resistance to proliferation.

Moreover, the fuel and the associated cycle will certainly have repercussions on economic competitiveness.

Lastly, in response to the above objectives, the multi-recycling of the nuclear material - U, Pu - and of minor actinides (AM) constitutes an essential element of the cycle strategy.

The challenges for the fuel are the following:

- to establish a very good behaviour at high temperatures to make easier the management of abnormal situations,
- to demonstrate a strong in-situ fission product containment capacity;
- to demonstrate an ability to reprocess by current processes or an advanced version or by a new process;
- establish the ability for re-use of the material which results from reprocessing.

These specifications are conditioned by very ambitious technological projections and underlie an original approach for safety (e.g. with the implementation of a Core melt exclusion strategy – CMES). Particle fuels or the advanced microstructures show a great potential to achieve such specifications.

The fuels developments and those of the associated reprocessing process appear rather specific and there is no guarantee that an evolutionary approach will allow the pursued goals to be met. This is due to the technological ruptures necessary to pass from HTR Triso particles to a fuel compatible with a hardened spectrum and able to sustain a higher fast fluence, one order of magnitude ($\sim 10^{27}$ n/m²) greater than the current one. This fuel, which must reconcile a sufficient concentration of heavy nuclei inside the cores and preserve a good integrity (tightness in particular) at strong fast fluence, represents a technological challenge and largely calls upon innovations.

The R&D domains for fuel and its cycle (the actions relative to this last aspect are also treated by the project " New processes of reprocessing for burned fuel”) cover at the same time the experimental field and that of simulation. They are as follows:

- Analysis of the potential of various technologies for fuel
- Fuel behaviour under reactor conditions and the mastering of its technology
- The ability to reprocess HTR particles;
- The basic study for HEGCR materials
- Inert fuel matrix
- Implementation of modern materials;
- Thermodynamic data;
- Modelling and simulation;
- Specific instrumental developments.

The first selection is expected to occur around 2010.

3.2.4 HEGCR – Materials resistant to high temperatures

Among the objectives retained for future systems, those which relate more directly to materials capable of withstanding high temperatures are as follows: economic competitiveness; increased safety; the minimization of the production of radioactive waste with long life; potentialities for applications other than the electricity production (e.g. hydrogen production, desalination…). To these main guidelines, the needs related to minimization for waste coming from structural materials shall be added.

The study of high temperature materials and the technology of the helium circuits, clearly show a common synergetic base for the developments necessary to the high-efficiency systems generation (i.e.: thermal HTR and HEGCR). The effort of upstream research and modelling necessary to drive the innovative effort towards structural and functional materials also appears as a common base.

Moreover, in the field of the HEGCR, the developments relating to materials appear to be able to benefit from significant dualities with conventional industry. The acquired experience in the R&D programs on high temperature structural materials for the nuclear power and non nuclear industrial field, makes possible to gather in the table below the material classes likely to resist the various operating conditions from the components of HTR and HEGCR.

TABLE I. MATERIALS FOR COMPONENTS OF HTR AND HEGCR

Reactor component	HEGCR (Direct cycleC) (*)	HTR (*)	HEGCR (Indirect Cycle) (**)
Vessel	AFMA 9 - 12% Cr		
Primary circuit	• Ni based with structural hardening • 32 Ni–25Cr–20 Fe–12,5W–0,05C (german HTR) • Ni-23Cr-18W-0,2C (Japanese HTTR) • Thermal barriers		• AMFA 9 12% Cr • Serie 300
Fuel element	• AFM ODS • Refractory metals and alloys • Ceramics	- Particles	• AAA from LMR Na • AFMA 9-12%Cr
Functional materials	• Refractory metals and alloys • Ceramics	- Graphite	• AFMA9-12 Cr • ceramics • refractory alloys
Heat exchangers	- primary circuit materials		- AFMA à 9-12%Cr

(*) Tmax (He) = 850°C; (**)Tmax (CO_2) = 650°C.

AFMA – Advanced Ferritic Martensitic Steel; AAA – Advanced Austenitic Steel; AFMODS Ferritic Martensitic Steel Oxide Dispersed

These analyses, on the potentialities of various alloys, should be complemented by basic research studies.

The preliminary selection is expected to occur around 2006 on:

- a reference set of refractory materials (ceramics and metals)
- advanced ferritic and martensitic steels (Phenix, Joyo, Bor-60)
- advanced austenitic steels and oxide dispersed steels (Phenix)
- in service behavior of steels and superalloys

3.2.5 HEGCR – Technology of high temperature helium circuits / components

Among the objectives retained for future systems those which more directly relate to technologies of the helium loops are the economic competitiveness and the increased safety. To these objectives, needs related to minimization of waste coming from structural materials should be added.

As for the materials, the technology of the helium circuits clearly show a common synergetic base for the developments necessary to the high-efficiency systems (i.e.: HTR and HEGCR). The effort of upstream research and modelling necessary to drive the innovative studies also appears as a common base. Moreover, as for the materials, in the field of the HEGCR, the developments relating to helium loops technology show significant dualities with conventional industry.

The definition of the objectives of a gas technology program is established starting from the review of the needs identified for a project such as the GT-MHR with thermal spectrum and direct cycle and extrapolated to hardened or fast spectrum. The following needs are identified:

- Purification and control of the coolant quality and inventory, and monitoring of the interaction with the materials;
- Thermal and thermo-aerolic of the core, the circuits and the heat exchangers;
- Dynamics of the circuits and the structures; acoustics of the cavities;
- In-service monitoring;
- Waste processing and dismantling
- Technology of specific components.

To achieve this Helium systems technology program, the following facilities are tentatively planned:

- Small experimental loops (tightness, oxidation, tribology...)
- Multipurpose 1 MW test loop (impurities, thermal exchanges...) (>2005)
- Multichannel 15 MW test loop (thermal exchanges, components...) (>2010)
- Test loop for dynamic purification

3.2.6 HEGCR – Potential for hydrogen production and cogeneration

Starting from current research in the field of fuel cells and hydrogen storage systems, the CEA intends to implement a large R&D programme on hydrogen also covering the aspects of production (new CO2 free processes), transport and related safety requirements. Five research lines have been identified to investigate specific questions associated with this type of production:

a. Modelling and performance assessment of promising processes,

b. Technical and economic assessment,

c. Control of the technology of processes through the modelling of the process engineering,

d. Proposal of a safety approach applicable to the production on the nuclear site,

e. Preliminary design of production units based on various processes and comparative evaluation of their performances and cost.

New processes may be put into two categories depending on the current state of technical advances:

- H2 production processes for which power prototypes already exist:
  - Thermochemical cycles.
  - High temperature electrolysis,
- H2 production processes for which development is required:
  - Gas decomposition through gliding arcs,
  - Water decomposition in plasma phase.

This endeavour is intended to reinforce the contribution of the CEA to the national and European research effort on non-fossil energy sources, and to open new opportunities of international collaborations and networking.

3.3. New processes for spent fuel reprocessing

It is a question of adapting technologies of the cycle (backend) to the evolution of fuels and strategies, to open the possibility of recycling plutonium and minor actinides while preserving and reinforcing the economic competitiveness of the nuclear option in the medium and long term.

The current context has led CEA to seek solutions which:

- Minimize discharges to the environment (gaseous and liquid waste);
- Take into account, in their design, the management of induced secondary waste (processing and conditioning);
- Make easier the integration of the operations of reprocessing and manufacture of the fuel (concept "all in one").

Two reprocessing families are to be analysed: advanced hydrometallurgical and pyrochemical.

3.4. PWR – Advanced designs

In the frame of the above mentioned "PWR – Advanced designs" project, CEA is investigating two main axes:

1. Small and medium size PWRs
   The work is focused on safety, economic and non proliferation aspects and the use of such reactors for special applications like co-generation. The simplification in the design by using modular concepts is being studied and also the safety approach in a strategy to exclude core melt.
   Other points under consideration are the use of passive safety systems particularly in the case of compact designs where the "threshold effects" must be identified for simplification.
   The economic evaluations of investment and electricity production costs must complete the analysis of such concepts.

2. The examination of the potential of PWRs to cope with optimisation of natural fissile resources and waste management.
   In this case, double phase cooled cores and tight lattices are under examination for selection of the most promising options to be calculated in detail in a second phase of the project.

Within the project N° 13, "Ad hoc studies: desalination and spatial", the objective is the identification of new applications for nuclear energy and the support for the corresponding projects Two areas are identified:

- the first relates to the analysis and the assessment of the economic feasibility of the sea water desalination with energy provided by nuclear reactors;
- the second aims at the implementation of activities in support to the design of a nuclear engine for space applications; after a first bibliographical step, the programme content will be defined as a function of the interest that could again raise on behalf of the CNES, ESA or NASA.

4. MAIN STEPS AND TIME SCHEDULE FOR THE DIFFERENT PROGRAMMES

4.1. HTGR concept

2003 – Pooling of the experience acquired in Europe on HTR through the 5^{th} FP - INNOHTR program;
2003 – First reprocessing tests on particles fuel (e.g. TRISO);
2005 – Manufacturing quality assessment of TRISO HTR particles;
2006 – First irradiation tests of fuel particles manufactured by CEA.

4.2. 4^{th} Generation reactors and cycles

2001 – Report on motivations and justification for the reference option;
2001 – Requirements on core performances;
2001 – Requirements on materials;
2004 –Technical report on retained operational options for fuel, core, system and cycle, as well as for the safety related options

2001 – 2005 – Technology of the helium loops: achievement and assessment of experiments on small test facility (<< 1 MW)
2006 – Assessment of the tests on advanced steels irradiated in Phenix (AFMA, ODS…)
2001 – 2006 – Refractory materials resisting the irradiation damages (coatings, matrices): test launching and assessment, and launching of the irradiations (< 2015)
2001– 2006 – Fuel manufacturing: test launching and assessment, and launching of the irradiations (< 2015)
2010 – Assessment of research on fuel
2010 – Technology of the helium loops: assessment of the experiments on 1MW loop

4.3. New reprocessing processes

2002 – Options choice for pyrochemical processes adapted to the considered fuels;
2004 – Options choice for the advanced hydrometallurgical processes;
2005 – Assessment of the first active tests on the selected pyrochemical processes;
2010– Final assessment on experimental laboratory test on the hydrometallurgical and pyrochemical processes.

CONSTRUCTION AND MANUFACTURING EXPERIENCE

(Session 10)

Chairperson

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Romania

MEDIUM REACTOR CONCEPT AND CONSTRUCTION/MANUFACTURE TECHNOLOGIES IN HITACHI

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Abstract

It is expected that the demand for medium and small nuclear power plants will increase in the areas where the electric supply network is weak in the near future. Actual plant construction and operational experience is very important due to earlier introduction of nuclear power plants in these areas. Hitachi is developing a medium-sized BWR, designated HABWR by downsizing ABWR that is the latest type of BWR in the world. The systems and equipment in HABWR are simplified and rationalized to overcome the scale demerit while keeping the basic design of the ABWR. Therefore, the construction/manufacture technologies and experience will be almost completely applicable to the HABWR. This paper describes the plant concept of HABWR and the latest construction technologies and experience in Hitachi.

1. 1. INTRODUCTION

Hitachi is developing two types of medium BWR concept which are the 600MWe forced circulation type BWR denoted HABWR and the natural circulation type BWR denoted HSBWR, respectively.

HABWR is downsized from the latest large-sized BWR called by the ABWR (1356MWe) in Ref [1]. HABWR is rationalized by the use of proven simplified technologies and it is unnecessary to do any T&Ds, for example component reduction of large capacity valves and the simplification of the turbine island 52” turbine developed for the 1350MWe ABWR, necessary to overcome the scale demerit. However, the HABWR has kept the characteristics of the ABWR, i.e. the reactor internal re-circulation pumps, the reinforced concrete containment vessel and so on. Therefore, all construction and manufacturing technologies of the ABWR can be completely applied to the HABWR.

HSBWR is a simplified BWR (300MWe to 600MWe) with natural-circulation core cooling, passive ECCS of accumulated tanks, passive containment cooling system and so on in Refs [2]-[7]. The confirmatory tests of these passive systems have been completed. The construction and manufacture of HSBWR have also been investigated to apply the same technologies as ABWR. Moreover, the simplification in HSBWR will contribute to easy construction and manufacture and to facilitate decommissioning.

The construction and manufacturing technologies should be improved reflecting field experience and it is very reliable to apply the latest technologies developed and adopted in the ABWR to the medium reactors. Hitachi had already applied the 3D-CAD system to the plant layout and composite design and adopted the large-scale module construction method for the ABWR construction in Ref [3]. These will be applied to the above medium BWRs more easily. These technologies contributed to the short length of the construction period and to the reliability of manufacture. Hitachi is newly developing the visualization system for construction engineering helpful to easily understand the situation and environment around the installation area, to easily plan a work sequence and confirm the planned schedule. This will be applied to the next ABWR construction.

2. DESIGN FEATURES OF THE HABWR

HABWR is downsized from the latest large-sized BWR called by the ABWR (1356MWe).

The HABWR is characterized by the following.

1. High reliability, Safety, and operability equivalent to the ABWR
2. Use of construction/manufacturing technologies same as ABWR
3. System simplification and rationalization by adopting the large capacity equipment developed for ABWR
4. Use of proven technologies that do not require additional T & D s

TABLE I. MAJOR SPECIFICATIONS OF HABWR AND ABWR

	HABWR	ABWR
Rated Electric Power	600 MWe	1356 MWe
Rated Thermal Power	1800 MWt	3926 MWt
Main Steam Line	700A × 2	700A × 4
Number of SRV	7	18
Feedwater Line	400A× 2	550A× 2
Reactor recirculation pump	RIP×5 to 6	RIP×10
ECCS	Active 3-div.	Active 3-div.
Primary Containment Vessel	RCCV	RCCV
Type of Turbine	TCDF-52"	TC6F-52"

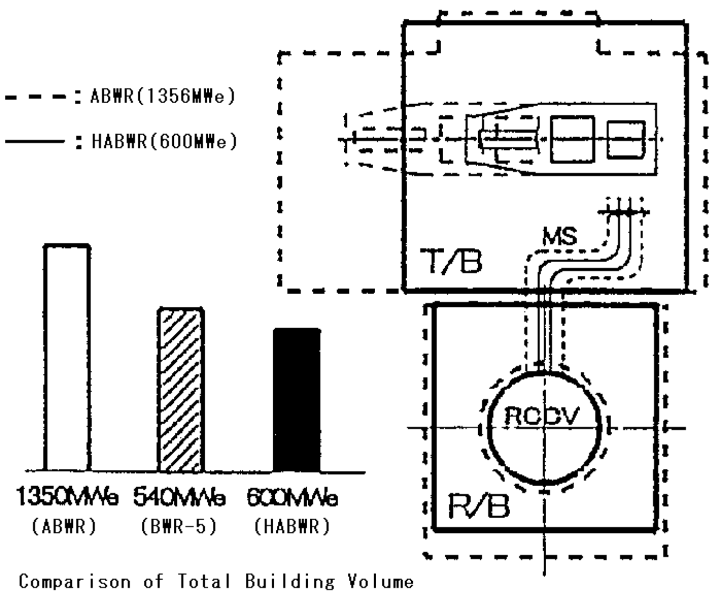

FIG.1. The comparison of Total Building Volume among ABWR, BWR-5 and HABWR.

Table 1 shows the plant specification in comparison with the ABWR. The HABWR is rationalized by the use of proven simplified technologies for which it is unnecessary to do any T&Ds, for example component reduction by large capacity valves and the simplification of the turbine island by 52” turbine developed for the 1350MWe ABWR, necessary to overcome the scale demerit. As the results of the simplification and rationalization, the total volume of reactor and turbine buildings is reduced to about 88% of a conventional BWR-5 (540MWe) and to about 63% of a standard ABWR (1356MWe) as shown in Fig.1.

3. IMPROVEMENTS OF CONSTRUCTION AND MANUFACTURING TECHNOLOGY

Nuclear power plants aim at providing a cheap and steady electric power supply, and electric power companies and plant suppliers are working on rationalization of construction technology and management. With respect to the construction aspect, large-scale modules, expansion of concurrent work between civil and mechanical and various automatic machines have been adopted in order to improve work efficiency, reduce the amount of fieldwork, and level fieldwork. A construction supporting system like scheduling system, which use the design data of 3D-CAD(three-dimensional CAD) system, is developed and applied as a construction planning tool.

In the construction management aspect, fine management is executed by developing and operating the construction supporting system which enables a work order and progress management based on construction planning. The essential information is uniformly managed by using the network, and communication between the office and the job site can be done promptly.

In this paper, the actual result of construction engineering and large-scale modularization method using 3D-CAD and integrated information system is described in the details.

3.1. Construction Engineering

Hitachi has developed and applied Hitachi Integrated Information System, which is consistent among design, production and construction. This system has design information and schedule information made electronic as a basic database, and is characterized by visualization planning and project management functions based on that. The content of the construction planning system is described as follows.

Construction planning of nuclear power plant can be classified into schedule plan, installation plan and temporary facility plan. The construction plan starts at 5 or 6 years before the start of construction and engineers who have abundant experience of construction execute it from a basic plan to a detailed plan coordinating with customers, civil companies and other manufacturers. However, a lot of manpower is required and it is necessary to repeat many adjustments to optimize the planning.

In addition, skillful engineers are decreasing recently and reproduction of the site image is requested at planning. Moreover, the schedule plan is required to high accuracy and is adjusted with the engineering schedule and the manufacturing schedule at the early stage to meet the requested arrival date of the drawing and commodities. Under these backgrounds, the planning support system effectively uses 3 D-CAD data and has been developed applying the computer graphic technology. This system is composed of the following three systems.

1. Schedule planning system
2. Modularization planning system
3. Early carry-in, temporary stage and scaffolding planning system

This section introduces the outline of these three systems as follows.

3.1.1 Schedule planning system

This system supports making the sub-master schedule and the area schedule planned in the office. This system has a relation with the 3D-CAD system and gets the necessary data from it. Moreover, this system is a consistent schedule system from sub-master schedule to 3 weeks schedule cooperating with the schedule support system, which are the subsystem of Hitachi Construction Supporting System described later.

1. Sub-master Schedule System

The sub-master submits documents for the customer and work on it begins at about 1 or 2 years before the start of construction. The purpose of this schedule is to confirm the whole schedule of each construction and testing and to clarify the interfaces between other companies including the customer. The main features of this system are as follows.

(1) Making the schedule semi-automatically based on equipment data and architectural schedule data.

(2) Cooperation with the downstream schedule

(3) Making the construction simulation based on the schedule data (refer to Fig 2)

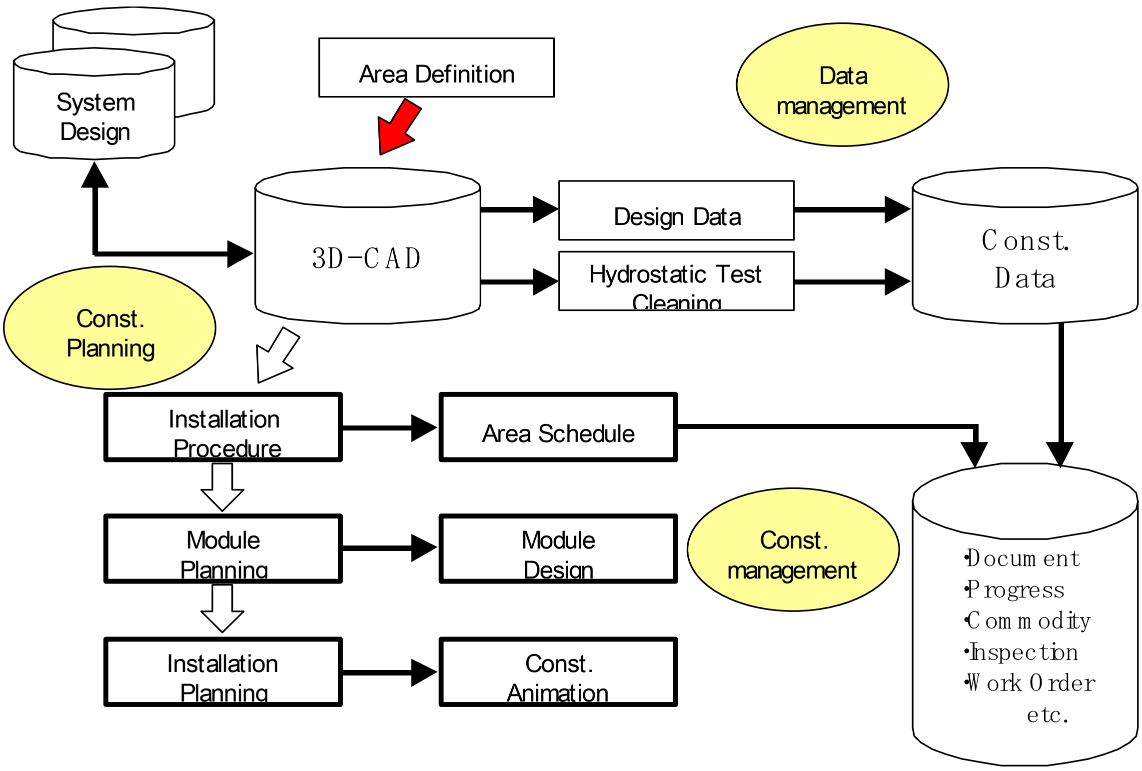

Fig2: Overview of Integrated Information System.

2. Area Schedule System

Area means a divided zone for construction use and there are about one hundred areas in each building. The area schedule is made before the construction office is established and is the basic schedule for the execution of all field works.

This schedule is made by skilful engineers, who confirm the layout of an area of 3D-CAD before and spent a lot of time in examination of the construction procedure. This system solved a past problem and has the following features and functions.

(1) Integration and division of 3D-CAD data according to schedule activity

(2) Making schedule data in the CAD screen (work procedure and work duration)

(3) Interface with other schedule systems

(4) Making animations based on the schedule

(5) Input and accumulation of scheduling know-how and construction notes, etc.

These functions allow the work procedure to be defined and construct the schedule by directly looking at the 3D-CAD layout and it is possible to certainly verify and make an accurate plan by construction animation. Moreover, it is possible to inform of a certain plan intention and content for field workers by leaving the procedure decision reason and note.

In addition, the scheduling engineers in charge of each construction gather in front of 3D-CAD and adjust the construction sequence before. In this system, the each scheduling engineer can plan the area schedule concurrently because they can examine and adjust own construction sequence on their PC referring visual procedure of another construction work.

3.1.2 Modularization planning system

One of the technologies to reduce manpower at the site is modularization which prefabricates equipment, piping and structures together at the workshop.

In order to promote modularization, it is necessary to study the ease of the design, the purchase date of commodities, how to fabricate and install the module from the early stage of the engineering schedule.

Fig.3 shows an example of the studies by 3D-CAD for the drywell module inside the RCCV (Reinforced Concrete Containment Vessel). In this way, using 3D-CAD enables speedy engineering, concurrence of design, fabrication and manufacturing, and visualization of feasibility studies for various kinds of modularization.

3.1.3 Early carry-in, temporary storage and scaffolding planning system

The number of carrying objects before slab work has increased and the temporary storage plan has become important. In the previous construction plant, all temporary storage planning drawing were made taking a lot of manpower by CAD output chart or hand. This system aims at saving labor of making the planning drawings, coordination with civil company and improvement of work efficiency. The main function is described below.

FIG. 3. Sample of T/B Construction Simulation.

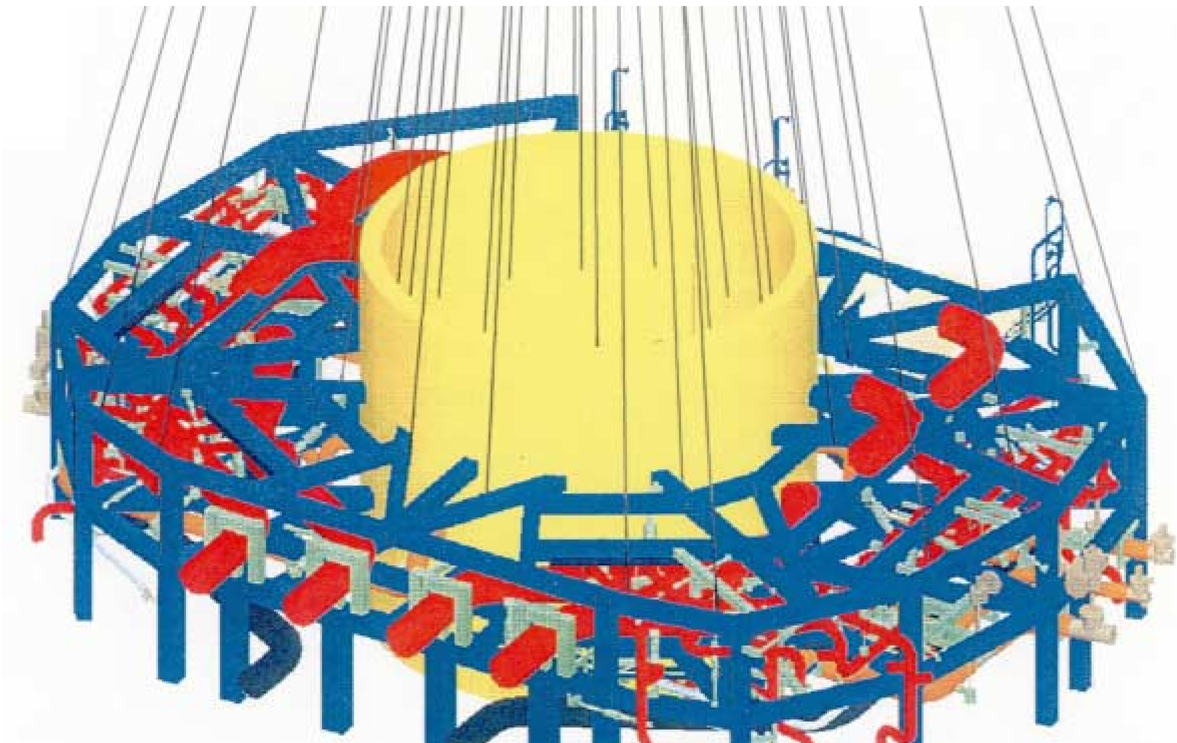

FIG. 4. Modularization planning system.

(1) Temporary storage function
Temporary storage position is decided by using the function of movement, rotation and position specification, etc. considering the following installation sequence.

(2) Scaffolding planning function
Scaffolding planning is executed by the function of making, the movement and the copy, etc. considering share of the scaffolding and minimization of re-making the scaffolding.

(3) Drawing making function
This system has function of automatically sizing line and making the commodity table for making a temporary storage planning drawings and a temporary scaffolding drawings.

The worker carries in the pipes, sets the scaffolding based on this drawing and is using the drawing as a work installation directly. Fig 4 is shown as for the example of this system.

3.2. Modularization method

Hitachi adopted the modularization method in 1979. However, the crane capacity of those days was small (130 ton), and the weight of a module was also as small as 10 to 60 ton. In 1985, an 850-ton mobile heavy lift crane (crawler crane) was put into use. Based on the better lifting devices available, the amount of factory work can be increased. Here, the 3D-CAD system plays an important role in such a planning as this large-scale modularization. To avoid transportation limitations, the fabrication area on the site is enlarged. Thus, adoption of the large-scale modularization method which applied 3D- CAD and the heavy lift crawler crane are contributing to laborsaving and construction schedule shortening.

The cross section of the Kashiwazaki Kariwa nuclear plant No.7 reactor building is shown in Fig. 5. This ABWR plant was constructed from 1993 to 1997. The inside of a building is divided into RCCV area and the general area made of reinforced concrete for every room. The actual construction method in each area is described below.

(1) RCCV area

This area was roughly divided into five regions. Plant supplier's products and the civil materials, which were contained in each region, were combined into the composite modules whose weights were from 290 tonnes to 650 tonnes. Then, reduction of the on-site work volume and construction schedule shortening were realized by assembling those modules like blocks using the heavy lift crawler crane.

(2) General area

Conventionally, the installation work of the bulk commodities (piping, support, cable tray and HVAC duct) amounts to a very large amount of work and forms the critical paths. As the installation work progressed to the upper floor, many workers were needed for a shorter installation period, and the peak of the number of workers had occurred in the last construction stage.

To advance these works, the bulk commodity modularization method was employed. Rather than fabricating bulk commodities at the construction field as was done initially, these products are fabricated from 10 tonnes to 120 tonnes in the factory or an on-site workshop, and transported by cranes.

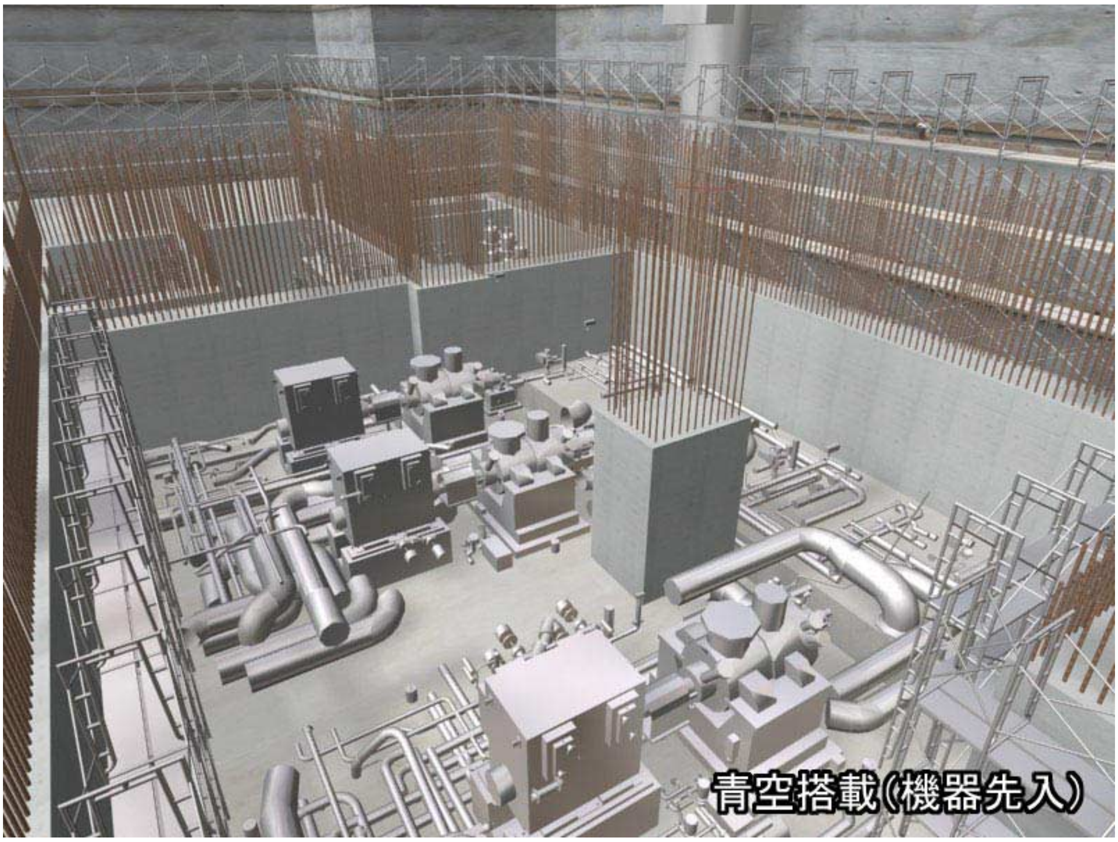

FIG. 5. Sample of scaffolding planning.

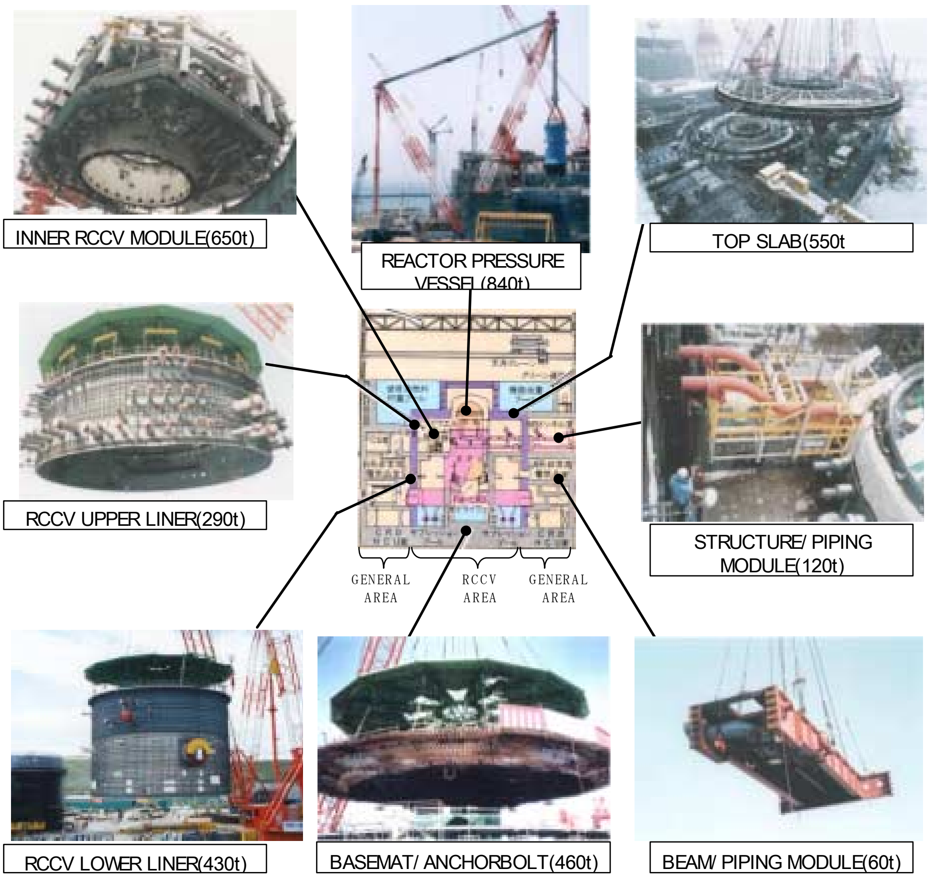

FIG. 6. Photos during construction.

This method has been used very much and reduced an on-site work volume, peak of workers and a construction schedule shortening to a large extent.

CONCLUSION

In Japan, Hitachi has several planning construction plants of ABWR. The construction/manufacturing technologies are still being advanced for next plant constructions. These technologies can almost completely be applied to the HABWR of a medium-sized ABWR due to adopting the same systems and equipment as the ABWR.

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DEVELOPMENT & APPLICATION OF AN ADVANCED CONSTRUCTION MANAGEMENT SYSTEM

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Abstract

In a gigantic and complex project such as the construction of a nuclear power plant, certain types of management methodologies and tools are prerequisite in order to ensure the satisfaction of all the project stakeholders in terms of time, cost, quality, and others. The advanced construction management system applied to Wolsong Nuclear Power Plant Project by Daewoo E&C Co., Ltd. has proven to be the major contributor to its successful completion. Based on this success we are very sure that we can serve our potential clients for the existing and prospective projects including SMR projects with our capability in combination with a more advanced construction management system.

1. CHARACTERISTICS OF NUCLEAR POWER PLANT CONSTRUCTION

1.1. Multiple Interfaces & Conflicts

Construction of the Wolsong nuclear power plant was composed of approximately 118,000 activities, of which roles and responsibilities were assigned to civil, mechanical, piping, electrical, and instrumentation departments in accordance with technical specialities. All the activities have interactivity dependencies to be closely monitored, controlled and coordinated, especially in the space-limited areas such as reactor building. In addition, departmental staff and engineers have their own priorities to address. Therefore, frequent interfaces and conflicts are likely to occur during the various phases of the project.

In order to prevent inappropriate and time-consuming interfaces and conflicts, the considerations listed below should be required in the project planning phase and fully understood by those involved for project execution.

- Proper scope definition
- Clear roles and responsibilities assigned to each department
- Understanding of priorities for project success
- Share of information through systematical communication management

1.2. Quality Oriented Project

Societal and environmental impact caused by severe or tiny accidents in nuclear power plants is beyond our imagination under the certain circumstances. Therefore, more intensified quality operations through quality planning, quality assurance, and quality control are required to ensure that the project will meet its needs and expectations in terms of reliability, stability, safety, government regulation and public acceptance.

1.3. Enormous and Various Resources

More than 120,000 kinds of materials and manpower of 3.6 million man-days were necessary for the Wolsong nuclear power plant.

What is to be procured, when it should be available and how it is delivered to the site are the main concerns. Moreover, due to the huge amount of materials and the traceability requirement on raw materials that is one of the nuclear specific features, the special controlling method is inevitably needed.

The availability of appropriate manpower is one of the important factors because the construction of a nuclear power plant strictly requires experts and workers with the specific skills and workmanship. Sometimes it is very hard to make the local competent workers or engineers available for the specific works within the performing organization. In that case, acquisition of manpower from the sources outside the performing organization will be more effective.

1.4. Strict Time and Cost Management Required

Construction of a nuclear power plant generally needs a relatively long period of time. During this period, the key milestones and specifically obligated activity sequences should be strictly achieved. This requirement demands effective and efficient tools to be implemented and controlled.

In addition, the intensified and strict cost management due to the enormous resources such as human, equipment, supplies and services is essential for the success of the project. Further considerations such as inflation forecast, financial cost, spending plan, return on investment, and payback period should be required in order to respond appropriately to the long period of construction.

2. ADVANCED CONSTRUCTION MANAGEMENT SYSTEM APPLIED TO WOLSONG PROJECT

2.1. Project Summary

The Wolsong Nuclear Power Plant Unit 3 & 4 can be summarized as below;

- Client: Korea Electric Power Corporation
- Location: Wolsong, Southern part of Korea
- Reactor Type: Pressurized Heavy Water Reactor (CANDU Type)
- Nuclear Fuel: Natural Uranium (about 0.7% of UO_2)
- Coolant and Moderator: Heavy Water (D_2O)
- Capacity: 700MW $\times$ 2 Units
- Construction Period (from the First Concrete Date to Commercial Operation Date)
  - Unit 3: March 1994 ~ June 1998 (51M)
  - Unit 4: July 1994 ~ June 1999 (59M)

Daewoo E&C Co., Ltd. performed the construction of the two units simultaneously as a general contractor covering the entire work scope including civil, mechanical, piping, electrical and instrumentation works for NSSS (Nuclear Steam Supply System), BNSP (Balance of Nuclear Steam Plant), BOP (Balance of Plant), Ancillary Facilities, D_2O Upgrading System, and Sea Water Intake & Outlet Structures, etc.

2.2. CM System Modules

The following construction management system modules were developed based on the previously accumulated experience from various types of projects that Daewoo E&C Co., Ltd. had undertaken.

2.2.1 Daewoo Planning & Scheduling Control (DPSC)

The scope of work described in the contract provisions was distributed to each department having the unique specialty in terms of technology and management. Each discipline produced the Work Breakdown Structure through scope definition processes followed by activity lists, activity duration estimates and network diagrams. All this information and data were combined and analyzed by using the expert judgement in parallel with the program called Daewoo Planning & Scheduling Control System. The outputs of reviews were feedback to each discipline and the final schedule was issued under the understanding of all the involved project team members with the approval of the client.

Once the final plan and schedule was established, progress monitoring, site detailed schedule control, and project analyses were performed on a regular basis. Furthermore, interfaces between various departmental activities were handled through schedule control process on the spot and recorded in DPSC system for the future uses and performance improvements.

2.2.2 Daewoo Material Tracking System (DMTS)

Materials were divided into two broad categories according to the sources of supply, namely owner supplied materials and contractor furnished materials. However, for the purpose of ease of control, Daewoo used one management system for both of them.

When the materials arrived at the site, the material controller performed the receiving inspection including documents reviews and physical surveillance and finally issued the acceptance and maintained the records in Daewoo Material Tracking System.

In addition, the system covered the information on material requisition, purchase order, manufacturing, expedition, and source inspection. By doing so, project staff who were in charge of material control for each discipline could access the system from their offices and find the status of materials. This system was the most effective tool in order to ensure the traceability requirement, which is one of the distinctive characteristics in nuclear industry.

2.2.3 Daewoo Cost Management System (DWCMS)

To reflect the characteristics of nuclear power plant as much as possible into the existing cost management system previously developed for other common projects, extensive and comprehensive resource planning, cost estimating and cost budgeting activities were performed in advance. Especially, advanced computerized tools and techniques such as spreadsheets and programming software turned out to be very useful and efficient in terms of handling and incorporating the enormous number of activities and resources in a consistent and coherent manner.

The allocated resources for a specific activity were controlled and managed in accordance with the appropriate measures instructed by the Daewoo Cost Management System. The causes, necessary actions and results regarding cost change were incorporated into the system whenever the changes occurred. With this system, it is possible to assess the performance analysis such as earned value analysis and predict the estimated cost at completion, spending plan, cash flow, and return on investment, etc. In addition, information and data stored in the system can be referred to as one of the reliable sources for future similar projects.

2.2.4 Daewoo Drawing & Document Control (DDDC)

Prior to the development of the Daewoo Drawing & Document Control System, several factors were considered. They were project organization and stakeholder relationship, departmental and individual roles, responsibilities and reporting relationship, information retrieval and distribution, etc. Based on the comprehensive understanding, Daewoo Drawing & Document Control System covered the collection and filing structure which details the methods of gathering and storing various types of information including procedures mandating the collection, distribution, revisions and corrections to previously distributed material.

All the information with the details on time of distribution, personnel to whom the information was provided, description of the information, the filed location of the information, revision status were recorded and controlled in a database format in order to ensure easy access and traceability.

2.2.5 Quality & Safety Control

Quality management is mainly composed of quality planning, quality assurance and quality control. In the case of the Wolsong nuclear power plant, Daewoo modified its own quality system in order to accommodate the specific inputs for the construction of the nuclear power plant. Quality management was performed by the quality assurance department and quality control department. The quality assurance department was responsible for the quality planning and quality assurance by using such tools as system or process flowchart and quality audit. The quality control department was responsible for the quality control by using such tools as inspection, control charts, statistical sampling, attribute sampling, flowchart, etc.

The safety control department was organized solely for safety concerns with the functions of training, monitoring, reporting to meet the regulatory guidelines. The records were filed and stored in a systematical manner.

2.3. One Hundred Critical Item Monitoring System

One hundred critical items were selected and completion of the selected individual item in a timely manner was prerequisite to meet the imposed target dates. Intensive daily monitoring and controlling was given to those items in order to check the progress and pending issues. The priority for resources such as labor, equipment, material was granted to those items. Interfaces and pending obstacles were resolved through regularly held status review meetings with the relevant departmental staffs and stakeholders including the client. In addition, all the related information and data produced were maintained and recorded for future use.

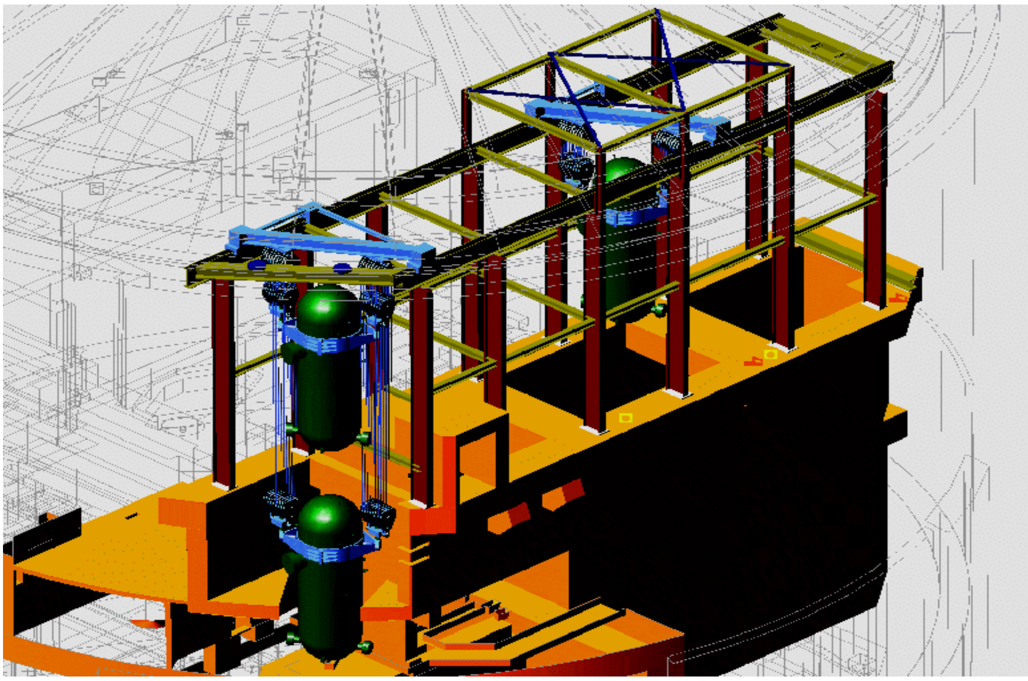

FIG. 1. An example of 3D CADD application.

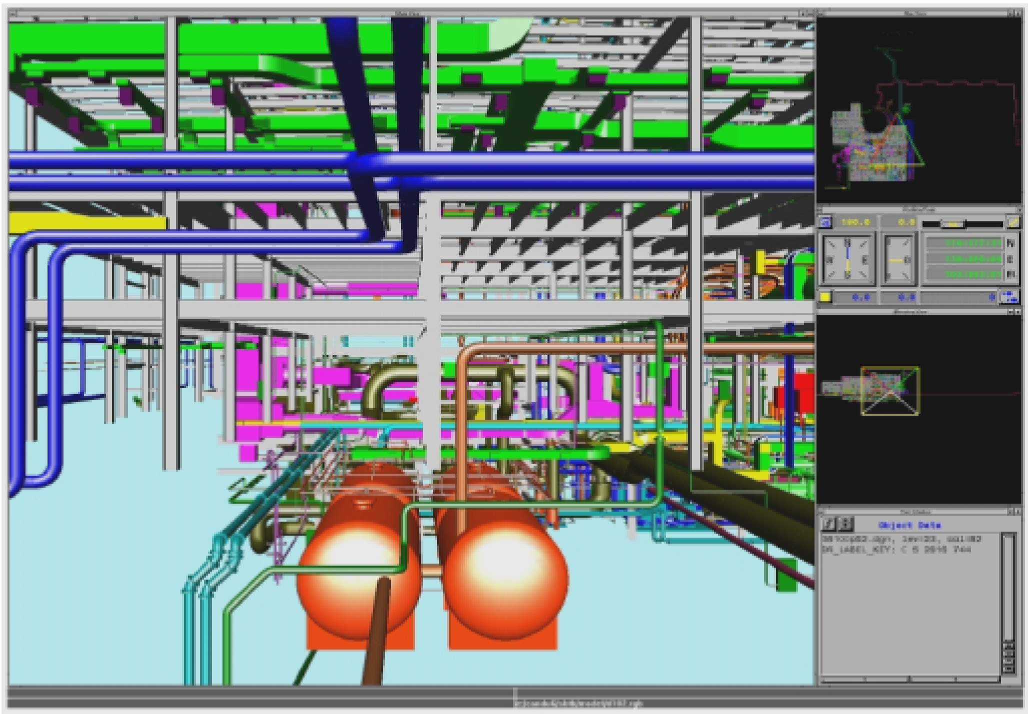

FIG. 2. Second example of 3D CADD application.

2.4. 3-Dimensional CADD Model

3D CADD Modeling is one of the newly developed and most advanced technologies for the manufacturing and construction industries around the world. Its effectiveness for the successful fulfillment of the complicated projects has been already proven and turned out to accomplish cost savings and shortening of the planned project schedule.

Daewoo developed the 3D CADD Model for Wolsong nuclear power plant in collaboration with AECL (Atomic Energy of Canada Limited) which had established the RDB (Reference DataBase) for several years. More than 20 engineers and CAD Designers participated in the development for 4 years. Total investment cost ended up to approximately 5 million US Dollars. With 3D CADD Modeling, the following were possible;

- Visualization and animation of multiple construction scenarios for the project in order to evaluate conflicts, risks and critical path events.
- Detecting and resolving conflicts and interference between equipment and structures within the plant.
- Development of detailed drawings (Isometric drawings).
- Easy understanding of systems and structures.
- Quick and efficient communication between the design and construction site.

The applicable scope of 3D CADD Modeling for the projects is extensive, for example, but not limited to, civil, architectural, mechanical, piping, HVAC, C&I, commissioning, O&M services, etc. Examples of 3D CADD application are shown on Figure 1 & 2.

3. 3. MAJOR ACHIEVEMENTS

3.1. The shortest Construction Period

As the result of applying the construction management system modules based on the full understanding of nuclear specific characteristics, Daewoo E&C Co., Ltd. accomplished the completion of Wolsong unit 3 in the shortest construction period (51months) from first concrete to commercial operation date in the world among the similar reactor types.

3.2. Conspicuous Records

The outstanding accomplishments which had major effects on the shortest construction period with assured quality are illustrated as below;

- The shortest construction period of the reactor building perimeter wall for Unit 4 (Verticality: 34mm, Torsional Alignment: 16mm)
- The shortest installation period (64 days for “C” face, 51 days for “A” face) of lower feeder in the world for Unit 4
- The shortest test period (255 hours) and the lowest leakage rate (0.223%/day) during SIT (Structural Integration Test) & ILRT (Integrated Leakage Rate Test) for Unit 3

3.3. Advancement to Overseas Nuclear Projects

With the recognition of successful performance at home and abroad, Daewoo E&C Co., Ltd. participated in the following nuclear power plant projects and had valuable opportunities to provide its clients with more advanced and proven technologies and management skills;

- Technical Assistance Services and Material Supply for Qinshan CANDU Project, China
- Technical Assistance Services for Civil Works for Reactor Building & Control Building for Lungmen NPP Project, Taiwan
- Technical Assistance Services for Mechanical Equipment & Piping System Installation Works of Nuclear Island for Lungmen NPP Project, Taiwan
- Design of Temporary Weather Cover for Lungmen NPP Project, Taiwan
- Acquisition of ASME NA & NPT Certificates of Authorization for Lungmen NPP under the joint name of DAEWOO and a Taiwanese company
- Joint Constructability Study with AECL for CANDU 9
- Member of consortium for the construction of KEDO project in North Korea

4. INTEGRATION OF ADVANCED CONSTRUCTION MANAGEMENT SYSTEM

The next step is to improve the efficiency and effectiveness of advanced construction management system modules shall be the integration and upgrading of them by making it possible that all the data and information are shared by end-users at real time through LAN (Local Access Network) and WWW (World Wide Web) environments.

It shall provide the project staff and stakeholders with the current status of project and progress measurement in terms of cost, time and quality as well as ease of communications.

5. CONTRIBUTION TO SMR (SMALL & MEDIUM SIZED REACTOR)

With the development and application of computerized CM system modules, the application of 3D CADD system to construction management and the enhancement of quality for nuclear power plants through advanced construction management, Daewoo E&C Co., Ltd. is actively taking part in SMART (System Integrated Modular Advance Reactor) project led by KAERI (Korea Atomic Energy Research Institute).

Daewoo are always ready to contribute its know-how in advanced CM system and technical assistance services for successful completion of SMR in future.

PARTICIPATION OF SMART DEVELOPMENT BASED ON THE CONSTRUCTION EXPERIENCES FROM THE NUCLEAR POWER PLANT IN THE REPUBLIC OF KOREA

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Abstract

World wide increase of awareness in “Quality of Life” by the implementation of industrialization has created the mass consumption of energy in which has caused environmental problems such as the global warming effect from the greenhouse gases and the lack of water supply that could cause various regional conflicts unless a valid solution can be adopted. Despite some opposition from environmentalists, nuclear energy is being considered a valid solution faced with the uncertainty of these environmental problems.

In compliance with these issues and to develop a proper solution, development of the 330MWt unit SMART reactor has been considered to produce electricity and desalinization in cooperation with the Government. Therefore, HDEC (Hyundai Engineering & Construction Co., Ltd.) has been participating in the strategic planning of SMART based on the over 30 years work experience of nuclear power plant construction. HDEC first started in the ‘70s by participating as a subcontractor of a foreign company, continuously invested in developing technologies, which has resulted in the ‘90’s in self-reliance in the area of technology of construction. Finally, HDEC has reached a position to transfer its technology and experience of construction to other countries. In addition, HDEC could shorten the construction period (from 1st concrete placement to the completion) from 71 months to the current, 58 months total of 13 months in 1,000MW unit in Yonggwang unit 5 NPP. It is expected that the schedule could be reduced a further 2 months in the 1,000MW Yonggwang unit 5 NPP, making a total saving of 15 months.

HDEC expects that the Total Construction Management System could further reduce the construction period in a Korea Standard Nuclear Power Plant (KSNP$^+$) & Advanced Power Reactor 1400(APR 1400), which might be ordered in the future, with the accumulated experience and feedback to the new NPP project.

The following will explain the process of HDEC’s construction technology accumulation ever since they participated in the first nuclear power plant construction, and how it made a stride towards economic prosperity and an effective construction management. Additionally, it will explain how that construction technology became to be used in SMART’s technology development and overall construction processes.

1. INTRODUCTION

HDEC’s (Hyundai Engineering & Construction Co., Ltd.) nuclear business is divided into the NPP construction part and the others. Nuclear Power Plant construction part began with the Kori PWR, Republic of Korea, in 1970s, and as is shown in Figure 1, 10 of the 16 units that are being operated have been built by HDEC, and the additional 4 units are currently being built. The other parts are consists of HANARO Research Reactor which ranked top 10 in the world in these areas as well as Irradiated Material Examination Facility (IMEF), Spent Fuel Dry Storage Canister for CANDU, and Steam Generator Replacement for Kori NPP #1.

Fig. 2 indicates the location of NPP projects in operation, and currently being built NPPs.

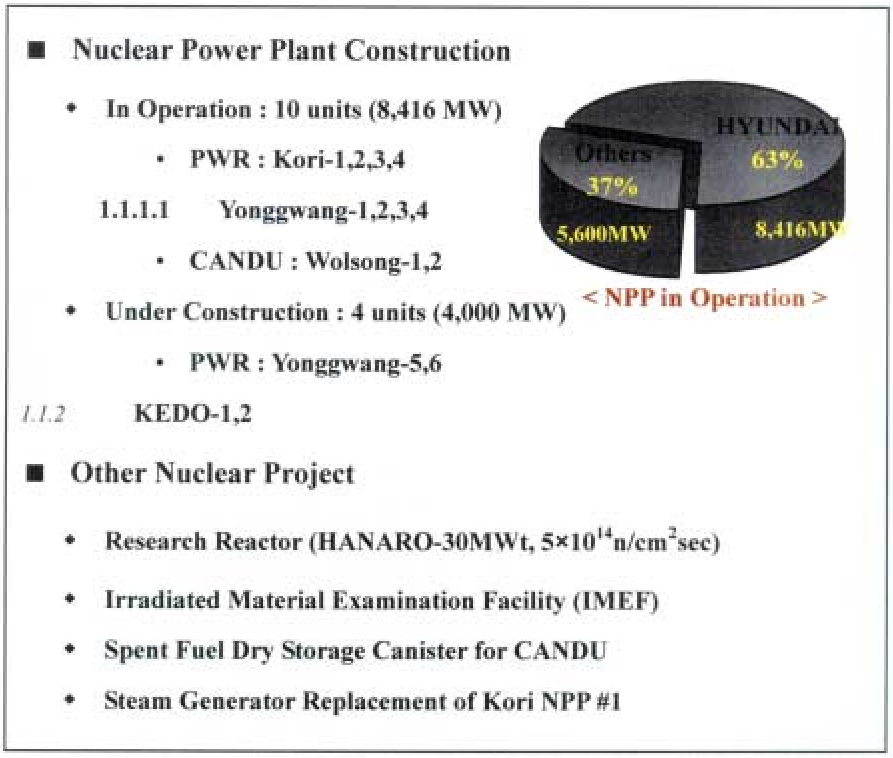

FIG. 1 Nuclear Power Plant Construction in the Republic of Korea.

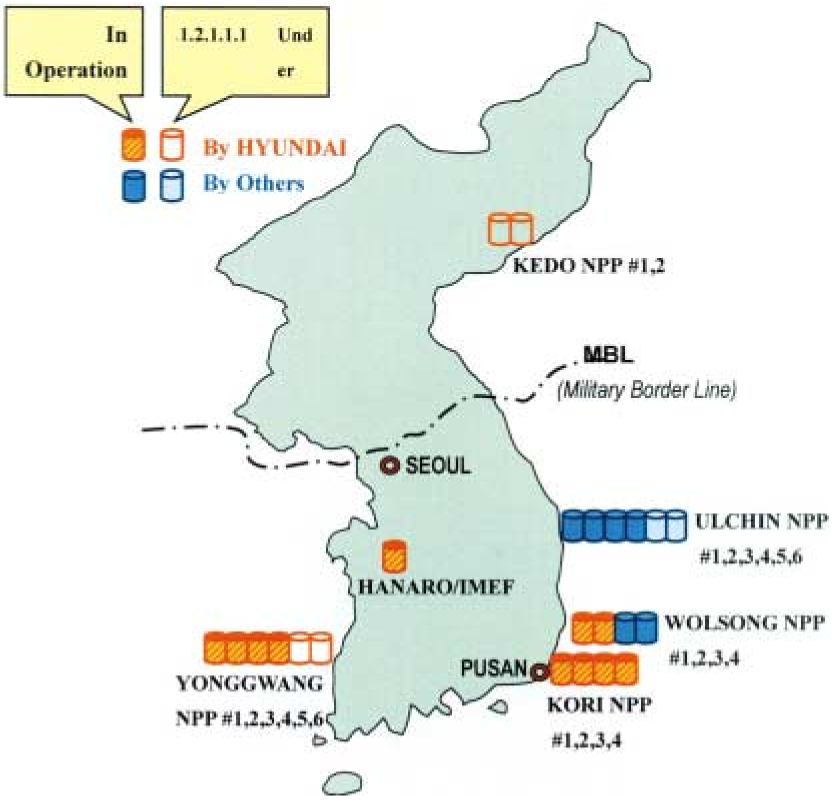

FIG. 2. Map of NPP projects in the Republic of Korea.

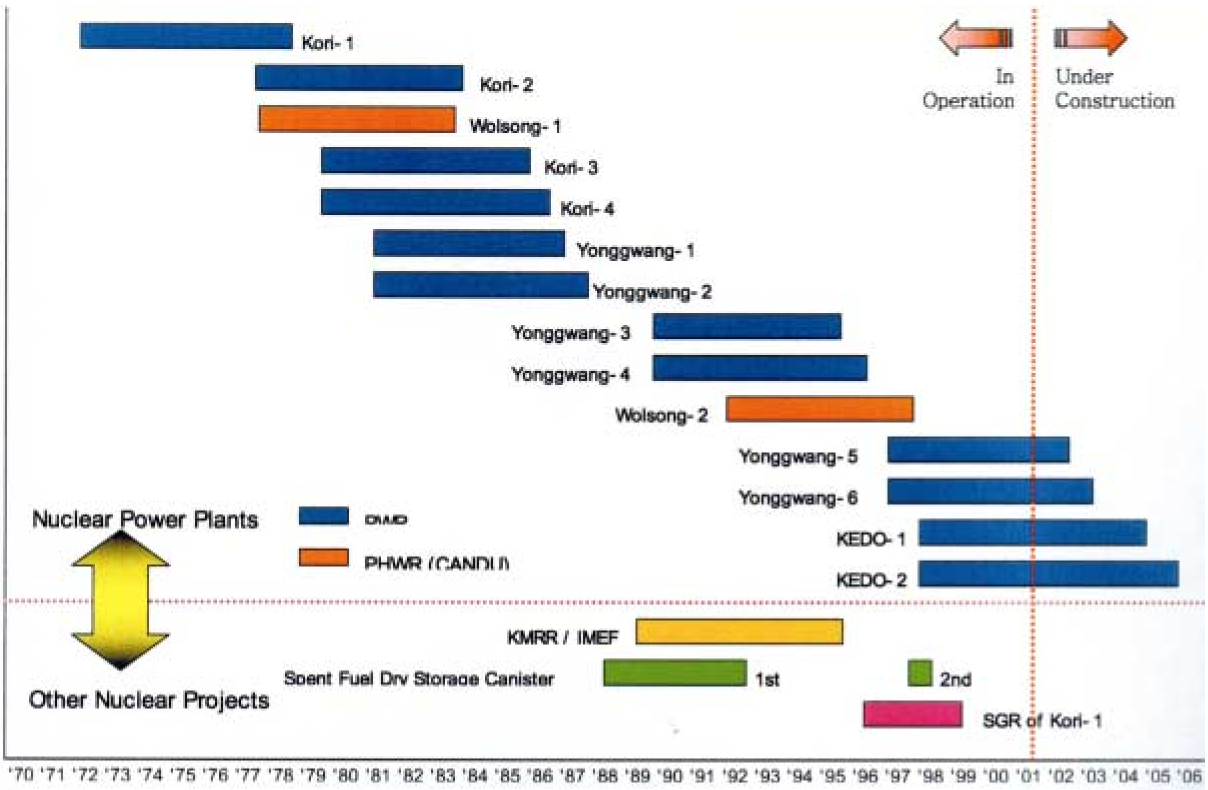

FIG. 3. Chronicle of NPP related projects in the Republic of Korea.

Figure 3 indicates the chronicle of NPP & other nuclear related projects carried out since the early 1970s. Over 30 years of continuous construction experience has resulted in the reduction of construction periods and increase of safety as well as higher economic value in nuclear power plant construction.

Accumulated experience through the nuclear power plant construction process from self-reliance of the construction technology to the current Younggwang-5, 6 as well as 400 other domestic and overseas projects' compound construction management system have been applied. These construction technologies are being utilized in the design and construction of SMART.

2. COMPLETION PROCESS OF TECHNOLOGY SELF-RELIANCE

HDEC's nuclear power plant construction businesses coincide with the history of the Republic of Korea's nuclear power plant construction. As shown in figure 4, technology reliance on the foreign company was the case in 1970s, and a technology transfer from the foreign company was conducted during the 1980s which resulted in becoming the main contractor in construction technology, and finally, enabling to transfer it's management skills to other companies.

In compliance with the Republic of Korea's nuclear power plant planning of self-reliance, KSNP adopted HDEC's Younggwang-3, 4 as a fundamental model, and ever since, it has been continuously carried out in fields of constructability and reducing of construction period in other projects of KSNP^{+} & APR 1400, which those issued later.

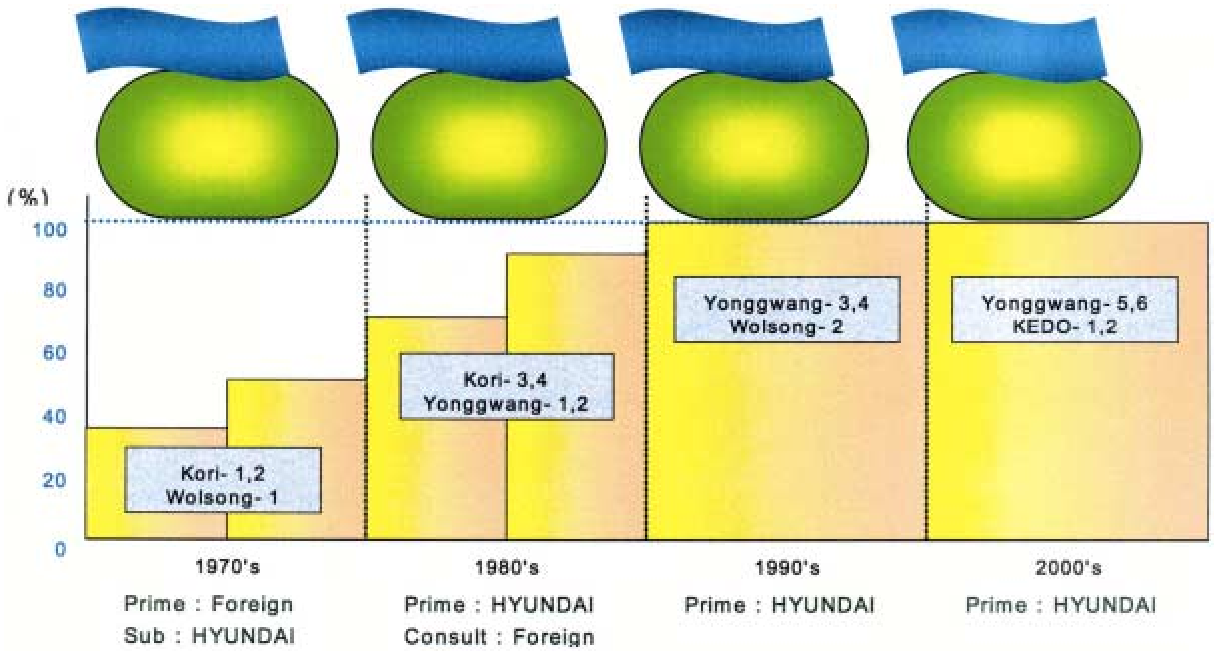

FIG. 4. Change of technology reliance in NPP construction.

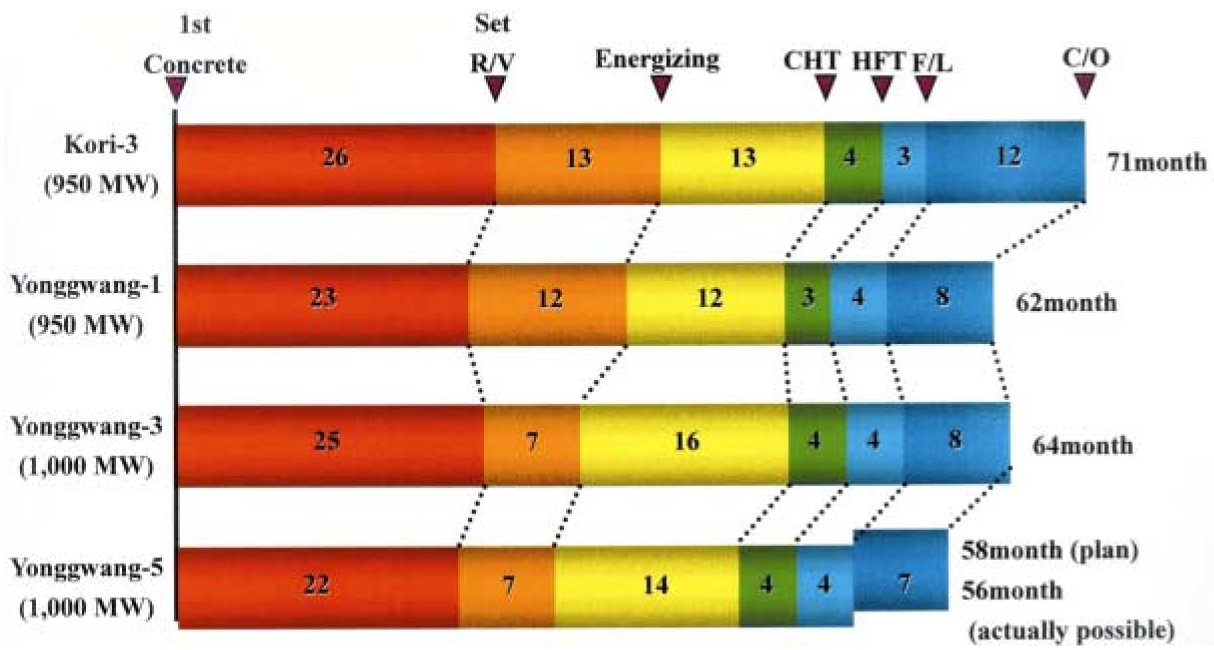

FIG. 5. Decrease in construction period at different projects.

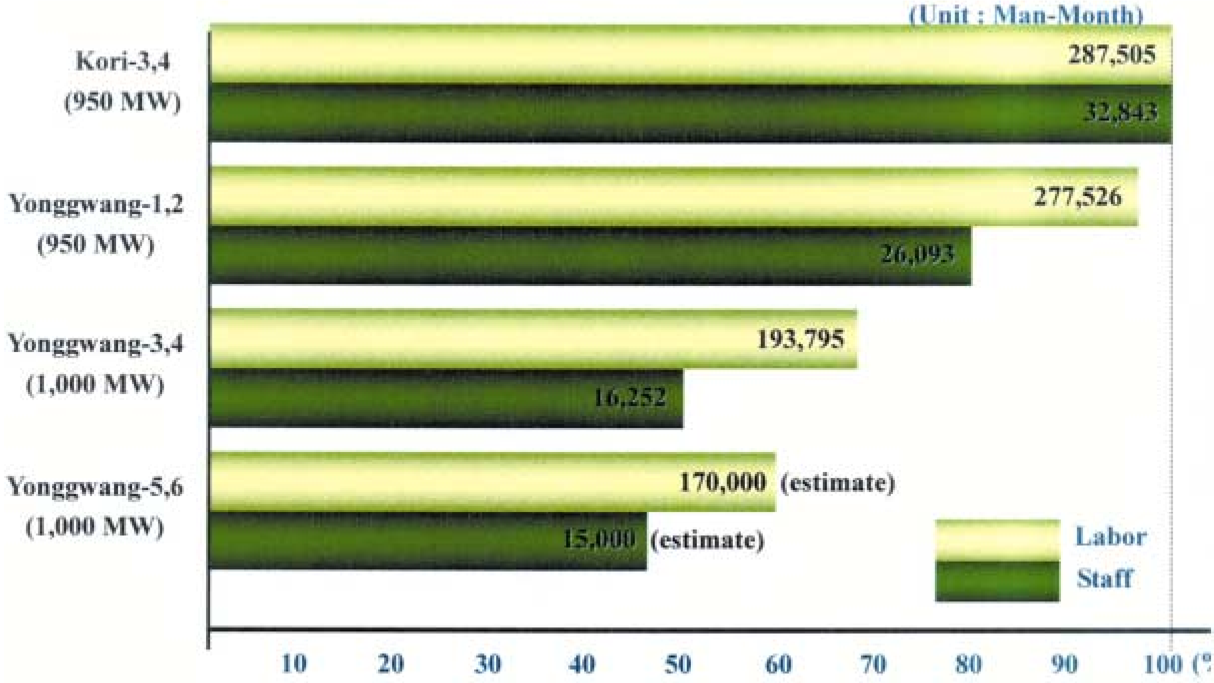

FIG. 6. Ratio of Non-Manual Staff in Kori-3&4 and Yonggwang (from 1 to 6) projects.

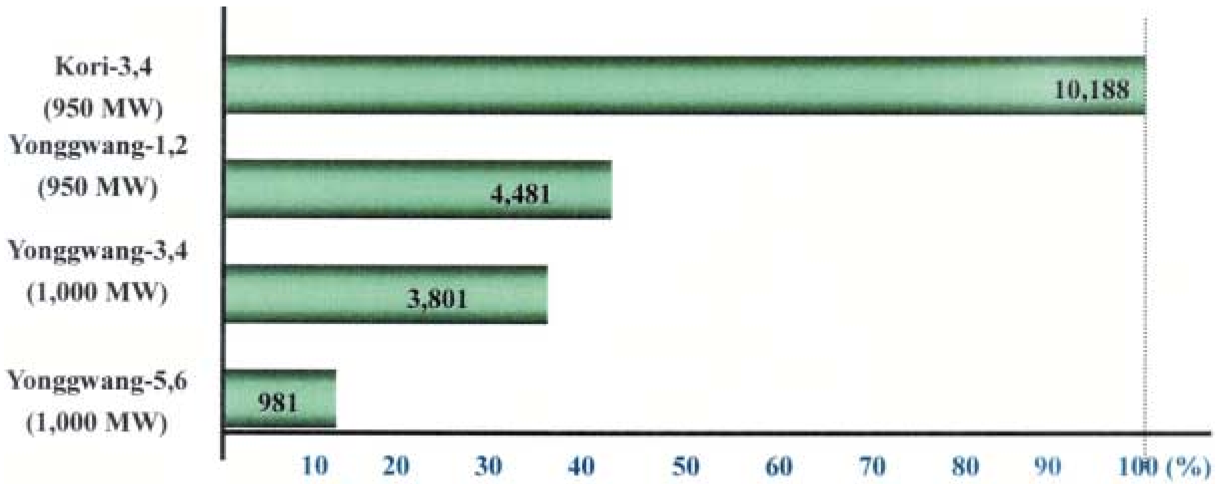

FIG. 7. Number of NCR issued at different projects.

3. INCREASE OF ECONOMIC VALUE THROUGH CONTINUOUS CONSTRUCTION EXPERIENCES

1. Decrease in Construction Period

As shown in figure 5, Kori-unit 3&4, which were completed during the mid 1980s, the construction period was 71 months after the 1st concrete placement however, based on that experience, Yonggwang-1 took 62 months, and Yonggwang-3 took 64 months. At the same time, current Yonggwang-5 is expected to be completed in 2002, which adds up to 58 months of construction period, and one month has been shortened in current position than expectation schedule and finally 56 months' construction period is expected by reducing 2 months.

This 56-month is same as the New Kori unit 1 construction period and is expect 15 months reducing of construction period compare to Kori unit 3.

KSNP^{+}'s New Kori-1 is expected to shorten even further construction period that will result in the increase of significant economic values and is helpful in view of financing face.

2. Decrease in Labor Worker Supply

Figure 6 indicates the ratio of Non-Manual Staff in Kori-3&4 to Yonggwang-5&6 to be from 32,843 MM (Man-Month) to 15,000 MM (presumption) which results in 54% decrease rate. In case of labor workers, the change occurs from 287,505 MM to 170,000 MM (presumption) which results in decrease rate of 40%.

3. Occurrence of NCR (Non-Conformance Report)

Figure 7 indicates a 63% decrease from 10,188 to 3,801 when comparing Kori-3&4 to Yonggwang-3&4 in number of NCR occurrence. At the same time, based on December 2000, there were 981 NCR issued which shows 90% decrease in Yonggwang-5&6.

This statistics indicate that strong efforts put into reducing the delay of construction period and an entrance of unnecessary material & labor force into the site.

4. EFFORTS TO IMPROVE CONSTRUCTION TECHNOLOGY

HDEC's main headquarter and the construction sites are being connected with their own internet line as well as the business owners and subcontractors in order to send necessary information to each other through the connected network. Especially, each department within each construction site is being connected with LAN to decrease the amount of paper work and to save time. HDEC's reference library contains more than 500 construction sites' including all of the nuclear power plants' information to be available by each project base, and can be provided to any team in need.

At the same time, in order to continuously develop new technologies and to save construction period, HICT has been built to support various technical difficulties, which can occur in construction sites.

Following this process, continuous data is being updated through problem solving, past experience, new technologies, and new construction techniques which is being transferred from the beginning of similar natural & complexity project to the actual design, which in turn, can prevent the waste of workers & construction materials.

These efforts contribute to early detection of problem-prevention measures, and helps to select proper construction materials and techniques, in order to maximize the final outcome, as well as to improve the speed and accuracy of information being used, which in turn, will create the overall economic value & production improvement. HDEC is already participating in SMART as well as other proposed project like KSNP^{+} & APR 1400 in the review of economic feasibility & construction methods.

5. INTRODUCTION OF HYUNDAI INSTITUTE OF CONSTRUCTION TECHNOLOGY (HICT)

HICT was developed in October 1989, at Mabuk-ri, in dedication to the research and development of applied and practical technology. Current human resources at HICT include 25 Doctorates and 52 Master's Degree recipients. As an important part of practical research, HICT is equipped with various well-advanced and high-tech laboratory facilities. Main research activities in structural engineering laboratory covers structural system and safety issues in large-span structures, aerospace industry, and nuclear power plant.

The research project involving technical application of composite panel floor on nuclear power plants (project granted from KEPCO, Korea Electric Power Corporation) demonstrated HICT's international competitiveness on cost savings and construction methodology by reducing labor cost, construction period as well as the overall cost. Other research projects consist of structural analysis of wall liners in containment, interpretation of the secondary building's inner-wall seismic measures, development of an expert system for crack control of concrete structure, practical development of high performance concrete for dome structure, and the investigation of cracks in the tendon Gallery and upper shell.

6. NUCLEAR POWER PLANT CONSTRUCTION TECHNOLOGY ON SMR

In order to improve economic feasibility in high quality electrical power plant, the final outcome should be built the way it's designer has intended originally.

The construction technique requires a proper combination of design, construction period, & construction method along with the appropriate management system to result in an optimum condition. In here, technical categories, like improving work efficiency, decreasing unnecessary work load on site, carrying out a design which included a consideration for construction & economic efficiency from the very beginning to the end, should require accumulated field experience. At the same time, in the case of designing necessary materials for construction, an appropriate size of modularizing and automotive facility for each construction stage should be carried out in the construction on site.

1. Construction Considered Design

Design considerations requires product size, weight, transportation, & storage for the construction site's location and schedule period. Especially, the intended construction region's weather, customs, industrial water, and environmental location should be considered to plan an overall work stage, and it is one of the most essential elements.

2. Economic Values Considered Design and Construction

For the items that require an on-site production & installation, convenient access to buy from a local market on each material should be possible, at the same time, the site area, infrastructure, and labor availability should also be considered.

3. Construction Cost Calculation

The optimum solution should be reflected in the design to calculate construction survey, period, method, & the process for the given project area, site layout plan, building volume & material quantity survey.

4) Maintenance and Repairing Management for Facility

Facility consideration in design is necessary to keep good conditions and repairing the facilities in nuclear operation and to carry out a given task even though it may no be provided on a design drawing.

CONCLUSION

Having carried out various nuclear plant construction projects for over 30 years and with the accumulated experiences and technical data, HDEC is participating in the design phase of SMART with the emphasis on construction and economic feasibility as well as a design consideration for construction, on-site characteristics for construction method. At the same time, the building's arrangement plan, on-site layout plan, labor workers & materials' resource plans are also being considered for safety & high quality performance, in order to achieve an economically feasible construction business.

NUCLEAR DESALINATION

(Session 11)

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IAEA-SR-218/83

PERSPECTIVES OF NUCLEAR DESALINATION OF SEAWATER BY SMALL AND MEDIUM REACTORS (SMRs)

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Abstract

Interest in nuclear desalination has been growing in many Member States over the past decade. Following the review of the state-of-the-art technologies of nuclear and seawater desalination using experience from plants in operation, and taking into account feasibility studies under site-specific conditions, several nuclear desalination demonstration programmes are already under way or being planned by IAEA Member States. The energy required for desalination can be provided by nuclear reactors in the form of low-grade heat and/or electricity.

In this paper, various factors, which support the attractiveness of nuclear desalination by small and medium reactors (SMRs), are identified and discussed (e.g. growing concerns about the environmental effects of burning fossil fuels; recognition of the benefits of diversification of energy sources; expected spin-off effects in industrial development and the development of new advanced reactor concepts in the small- and medium-power range). It is further illustrated that many nuclear reactor types can provide the energy requirements for various desalination processes. Operating experience with nuclear desalination, which has been gained by a liquid-metal cooled fast reactor BN-350 in Kazakhstan and several Pressurized Water Reactor (PWR) units in Japan, is discussed. Other reactor types, which are also being evaluated for application (i.e. integral type PWRs, nuclear heating reactors (HTGRs) and a Boiling Water Reactor (BWR)), are also discussed. Economic analyses of nuclear desalination using specialized computer software were carried out in order to demonstrate the competitiveness of this technology relative to fossil fuel-powered desalination operations. Guidelines are also given for the preparation of site-specific user requirements documents.

1. THE NEED FOR WATER

Freshwater is a fundamental necessity for all life on earth; yet it is a limited resource. Indeed, only about 2.5% of the world's water is freshwater, and of that two-thirds are locked up in ice-caps and glaciers. Of the remaining amount, some 20% is in areas too remote for human access, and of the rest about three-quarters falls at the wrong time and place — in monsoons and floods — and is not captured for use by people. Accessible freshwater for human use thus is less than 0.08% of the total water on the planet.

The limited, but easily accessible freshwater resources in rivers, lakes and shallow groundwater aquifers are dwindling owing to over-exploitation or water quality degradation resulting from human activities. An estimated 1.1 billion people lack safe water. The resulting human toll is roughly 3.3 billion cases of illness and 2 million deaths per year [1]. According to forecasts, if nothing is done today, a large part of the world, with about two thirds of humanity, will face shortages of clean freshwater by 2025. By that time, the world will also need better water management to grow food for an additional 2 billion inhabitants [2].

As these problems become acute, countries and international organizations need to work together to pool expertise and resources, as demonstrated at the second World Water Forum in The Hague, in March 2000, where more than one hundred government ministers, and representatives of UN system and other inter-governmental agencies and non-governmental organizations gathered together to discuss the problem and the possible solutions [3].

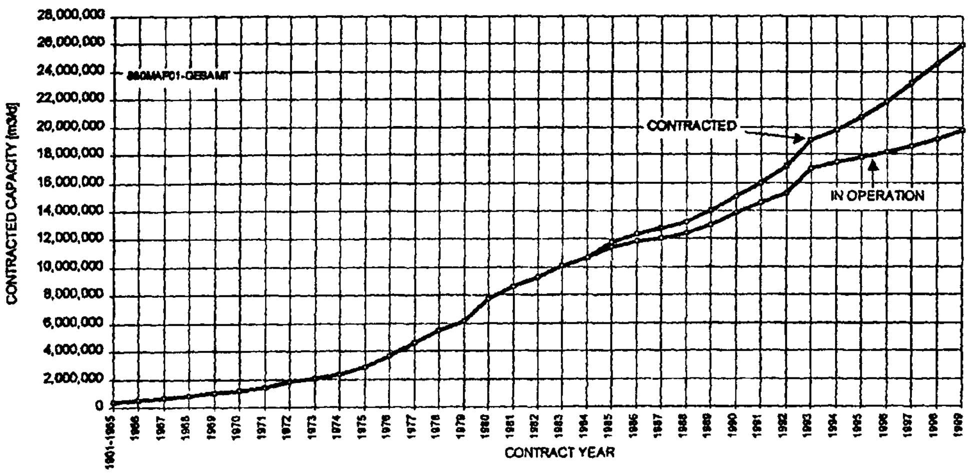

Fig. 1.1. Cumulative worldwide desalination capacity (installed and in operation). Source: ref. [4].

It is essential that all available and appropriate technologies, including nuclear and related technologies, be used for the sustainable development and management of freshwater resources. One particular approach is the desalination of seawater. Figure 1.1 shows that there has been a steady increase in water production capacity from desalination plants over the past few decades [4]. The energy for these plants is generally supplied in the form of either steam or electricity. Conventional fossil fuels have normally been utilized as the primary sources but their intensive use raises increasing environmental concerns.

The prospects of using nuclear energy for sea water desalination on a large scale remain very attractive since desalination is an energy intensive process that can utilize the heat from a nuclear reactor and/or the electricity produced by such plants. Many years of successful operation of nuclear desalination[1] plants in Kazakhstan and Japan have proved the technical feasibility, compliance with safety requirements and the high degree of reliability of co-generation nuclear reactors.

2. CONTRIBUTION OF NUCLEAR DESALINATION

Over the past 50 years, nuclear power has become an important part of the energy mix in many countries. At the end of 1999, some 434 nuclear power reactors in over 30 countries were producing about 16% of the world electricity supply. Sixteen countries have relied on nuclear power for 25% or more of their electricity needs.

Integrated nuclear desalination plants have been in operation for over two decades in countries such as Japan and Kazakhstan. Indeed, nuclear reactor operating experience for desalination exceeds 150 reactor-years as of 2000 [5] with a "clean" safety record. The IAEA's "Options Identification Programme for Demonstration of Nuclear Desalination [6]", completed in 1996,

[1] *Nuclear desalination* is defined as the production of potable water from seawater in a facility in which a nuclear reactor is used as the source of energy (electrical and/or thermal) for the desalination process. The facility may be dedicated solely to the production of potable water, or may be used for the generation of electricity and the production of potable water, in which case only a portion of the total energy output of the reactor is used for water production. In either case, the notion of nuclear desalination is taken to mean an integrated facility in which both the reactor and the desalination system are located on a common site and energy is produced on-site for use in the desalination system. It also involves at least some degree of common or shared facilities, services, staff, operating strategies, outage planning, and possibly control facilities and seawater intake and outfall structures.

and the international symposium on "Nuclear Desalination of Seawater [7]" in 1997 gave momentum to many Member States to take a step forward in evaluating, planning or initiating nuclear desalination projects.

The IAEA provides a framework for facilitating such activities in Member States through information exchange and provision of technical assistance. IAEA support activities include: co-ordination of research projects between interested institutes; development and application of an economic evaluation computer program; co-ordination of joint development of integrated nuclear desalination plants by technology holders and potential end-users and provision of information exchange forum through meetings and publications [8].

One of the most important motivations for large-scale implementation of nuclear desalination is its economic competitiveness with alternative energy supply options, especially in developing countries. Operating experience in Kazakhstan using an LMR and in Japan with large PWRs demonstrated the technical viability of nuclear desalination, as was mentioned above. Design efforts of advanced SMRs with higher safety and low cost could potentially establish nuclear desalination as a feasible and competitive technology for water desalination in developing countries.

Although significant experience with nuclear desalination has been accumulated and its technical feasibility has been shown, additional requirements have to be met under specific conditions in order to introduce nuclear desalination on a broader scale [8]. Technical issues will include, among other things, economic competitiveness (both electricity and water for co-generating plants), to meet more stringent safety requirements (nuclear reactors themselves and nuclear-desalination integrated complexes), and performance improvements of the integrated systems.

Many Member States are currently taking a step forward in evaluating, planning or initiating nuclear desalination projects. In Egypt, for example, a feasibility study for an electricity and desalination plant at El-Dabaa has been underway for quite some time now. In India, a hybrid (i.e. combined distillation and RO) desalination unit is being connected to the existing heavy water reactor units at Kalpakkam. Civil work is underway with commissioning expected in 2002. In the Republic of Korea, a nuclear desalination unit is currently under design using the co-generating reactor SMART. Morocco has completed a feasibility study in 1998 for a desalination plant based on a heating reactor. In Russia, efforts continue in developing a floating power unit based on a KLT-40 nuclear reactor for multipurpose use, including desalination. The first such nuclear desalination plant is foreseen in the Russian Arctic Sea coast area (Severodvinsk or Pevec).

3. OVERVIEW OF PRACTICAL EXPERIENCE

As was previously mentioned, some of the nuclear desalination plants have been in operation in Japan and Kazakhstan. The first Japanese nuclear power and seawater desalination plant was commissioned in 1978 at the Ohi Nuclear Power Station. All of the Japanese nuclear desalination plants have capacities in the range of 1000 to 3900 m^3/d (Table 3.1). The water produced is primarily used for in-plant use for steam cycle make-up water and other general household uses.

An RO facility set up at the Karachi Nuclear Power Plant (KANUPP) in Pakistan has been in service since the early 1970s. It provides freshwater for the nuclear and constitutes, in case of emergency, an independent source of feed-water for the steam generator in the event of a nuclear plant emergency.

TABLE 3.1. EXPERIENCE IN NUCLEAR DESALINATION PLANTS

Plant name	Reactor Type[2]	Gross Power (MWe)	Desalination process[3]	Water Capacity (m^3/day)
Ikata-1,2 (Ehime, Japan)	PWR	566	MSF	2000
Ikata-3 (Ehime, Japan)	PWR	890	RO[4]	2000
Ohi-1,2 (Fukui, Japan)	PWR	2x1175	MSF	3900
Ohi-3,4 (Fukui, Japan)	PWR	2x1180	RO	2600
Genkai-4 (Fukuoka, Japan)	PWR	1180	RO	1000
Genkai-3,4 (Fukuoka, Japan)	PWR	2x1180	MED	1000
Takahama-3,4 (Fukui, Japan)	PWR	2x870	MED	1000
Kashiwazaki Kariwa-1 (Niigata, Japan)	BWR	1100	MSF[5]	1000
KANUPP (Karachi, Pakistan)	PHWR	137	RO	454
BN-350 (Aktau, Kazakhstan)	LMFR (closed in 1999)	150	MSF&MED	80 000

In Aktau, Kazakhstan, a sodium-cooled fast reactor BN-350 provided electricity and nuclear heat to the Aktau power and desalination plant complex located in an arid region on the East Coast of the Caspian Sea. The heat energy needed for the desalination process was supplied in the form of steam by both a BN-350 reactor and a fossil power plant. There were two distillate lines from the desalination plant: drinking water and industrial water for boiler feed and other industrial uses. The desalination plant could continue its service even during nuclear reactor shutdown periods by switching the heat source for the desalination plant to the thermal power station. Failures experienced in the desalination equipment included corrosion and erosion of pipes, shell and tubes of evaporators' and preheaters' heat exchangers, and circulating pump blades [9]. The overall desalination system availability over the whole service period was reported to be about 85%.

4. TECHNICAL ASPECTS OF NUCLEAR DESALINATION PLANTS

4.1. Specific design considerations for combined plants

The design approaches of a nuclear desalination plant are based on those of a nuclear reactor and some additional aspects, which are unique to such a plant. Some of these additional aspects are highlighted below.

(a) Safety

The safety of a nuclear desalination plant depends mainly on the safety of the nuclear reactor and the interface between the nuclear desalination systems. Adequate measures must be introduced to ensure no detectable radioactivity release to the product water. In the case of a PHWR, the tritium level in the heating steam and product water must be regularly checked.

[2] PWR: pressurized water reactor; PHWR: pressurized heavy water reactor; LMFR: liquid metal fast reactor.
[3] MED: multi-effect distillation process, MSF: multi-stage flashing process, RO: reverse osmosis process.
[4] Brackish water desalted.
[5] Desalination unit was idle after test completion, since other freshwater resources were made available.

(b) Plant design life

The main components of the nuclear plant are designed for more than 40 years, whereas desalination plants are usually designed for an economic life of around 25 years or less. Therefore, it is essential that the design and layout of the nuclear desalination plant should accommodate the possibility of replacement or expansion of the desalination section with minimum interruption of electricity generation and water production.

(c) Operational flexibility

The water to power ratio in a co-generating station changes with daily and seasonal variation. As electricity cannot be stored, only the steam flow rate in the turbine may be adjusted to meet varying power demands. The design of a co-generation plant must provide a minimum degree of flexibility to avoid the breakdown of production units when either the turbine generator or the desalination plant production is reduced or shut down. The two major balance-of-plant (BOP) systems (i.e. the turbine and the desalination plant) should meet the requirements of smooth and convenient operation during transients, such as start up, load changes and shut down of one BOP system during operation or outage of the other, and steady operation with either one of the two major BOP systems at partial load.

(d) Design limitations

The seawater intake and outfall system and the environmental limitations with respect to temperature and salinity of seawater discharge from a nuclear power plant influence the coupling of a desalination plant with a nuclear reactor. The temperature and pressure of steam or hot water produced in a heating reactor also have an effect on the selection of the desalination system and its specifications including the coupling arrangement.

(e) Construction schedule

The construction period for the nuclear plant is much longer (at least 5-6 years) than for desalination systems (usually 1½-3 years). If the user wants to commission the two units simultaneously for minimizing financing expenses and for optimizing construction time, the desalination system construction should be scheduled accordingly. If desalted water is needed for construction, one option is to start an RO system using power from the electrical grid.

4.2. Design concepts being evaluated

The amount of energy (heat or electricity) needed for desalination is not big and the low grade steam can be used for the distillation processes. This leads to a good potential to use nuclear reactors in the small and medium size ranges (SMRs). In principle any nuclear reactors can provide energy, as required by desalination processes, and could be chosen as an energy source for desalination. The safety, regulatory and environmental concerns in nuclear desalination are those related directly to nuclear power plants, with due consideration given to the coupling process. Existing international safety standards and guides seem to be appropriate covering desalination plants [10].

Several design concepts are being pursued in IAEA Member States. Coupling combinations currently under evaluation are summarized in Table 4.1 [11]. The work under progress demonstrates the high degree of flexibility in the technical coupling solutions. All major reactor lines and the three most important desalination technologies can be used in various coupling combinations.

TABLE 4.1. DESIGN CONCEPTS PURSUED IN IAEA MEMBER STATES (2001)

Reactor Type	Desalination Process	Member States
PWR	MED, RO	Argentina, Japan, Republic of Korea, Russian Federation
PHWR	MSF/RO	India
	RO with preheating effects	Canada
BWR	RO	Japan
LMFR	MED, RO	Russian Federation, Japan
NHR	MED	Morocco, China
HTGR	MED, MSF	South Africa, France, The Netherlands, Japan, Russian Federation

Each of these design approaches is characterized by specific technical solutions for the coupling between the nuclear part and the desalination part. These solutions and design concepts are described below in more detail.

4.2.1 Application of a pressurized water reactor (PWR)

At existing nuclear desalination plants with PWRs in Japan, steam extracted from the secondary circuit or the turbine is provided to the brine heater via a steam converter, which distributes steam to various auxiliary systems. This arrangement secures a multiple barrier between the primary coolant and the brine heater. Similar concepts are followed in other design approaches. In some cases condenser cooling water is used as preheated feed water for the desalination process for improving its performance.

4.2.2 Application of a boiling water reactor (BWR)

At Kashiwazaki-Kariwa, Japan, an additional heat exchanger was installed between the primary cooling loop and the brine heater for the MSF process. A new approach was recently proposed, in which RO systems are connected mechanically to turbine-driven pumps in order to receive pressure for the membrane system. With this approach, intermediate loops or heat exchangers for isolating the coolant and seawater are not required.

4.2.3 Application of a liquid metal reactor (LMR)

At Aktau in Kazakhstan, steam discharged from a back-pressure turbine located in the tertiary system of the sodium-cooled BN-350 was fed to the brine heater of the desalination system. Recently a simpler connection is being evaluated by Russia using a Pb-Bi cooled reactor. Part of discharged steam from a high pressure turbine is provided to the brine heater. The Japanese approach is to combine the reactor electrically with an RO process.

4.2.4 Application of a pressurized heavy water reactor (PHWR)

Part of steam from the secondary loop is extracted and fed to the brine heater for the MSF process via an intermediate heat exchanger in the Indian design. Waste heat from the condenser is used to preheat the feed water for the RO process in the Canadian approach.

4.2.5 Application of heating reactors

The interface using Heating Reactors is a simple steam generator, through which heat is provided to the brine heater. Specific consideration needed for the application of heating reactors is that the desalination system is the main heat sink of the reactor. All transient variations from either side (heat source or heat sink) must be carefully evaluated to avoid any adverse effect on the other side.

4.2.6 Application of high temperature gas cooled reactors (HTGRs)

Heat discharged from the primary helium coolant can be efficiently used for desalination via the intercooler/precooler and the intermediate water loop with minimal penalty to the main heat cycle. This potential application is being evaluated using Pebble Bed Modular Reactor (PBMR) in South Africa and Gas Turbine-Modular Helium-cooled Reactor (GT-MHR) in France, Russia and Japan.

5. IAEA SUPPORT ACTIVITIES

The programme of the IAEA encompasses the broad areas of nuclear technology, safety and safeguards, together with outreach activities which seek to provide information to decision makers, civil society and the public. However, the discussion below will be confined to the nuclear technology area.

International co-operation has played an essential role in the global development and deployment of nuclear power plants. In the current situation, when the resources assigned to innovative R&D are limited and the efforts invested are dispersed among many countries and projects, international co-operation could once again be of vital importance. For more than forty years the IAEA has fostered international communication and co-operation in nuclear technology development, through information exchange (see, for example, Refs. [5–25]), the management of national, regional or interregional technical co-operation projects and the organization of co-ordinated research. It is well positioned to serve as a catalyst in the continuing development of nuclear power technologies for developing countries.

In the past two decades, considerable effort has been directed towards defining the needs of developing countries, assessing existing technologies relative to these needs, and exploring nuclear plant concepts that have the potential to address these needs.

The IAEA has also been actively assisting its Member States in the area of nuclear desalination and has provided resources and means to help in the preparation and conduct of national programmes. A number of technical co-operation projects have assessed the feasibility of particular projects, and documentation on economic and technical aspects of nuclear desalination has been made available in the form of publications [6,17–21]. The "Guidebook on the Introduction of Nuclear Desalination" will help all interested Member States to prepare for and implement nuclear desalination projects [8].

Since 1997, the IAEA has provided the only comprehensive and regular worldwide forum for the exchange of information on nuclear desalination technologies and programmes through the International Nuclear Desalination Advisory Group (INDAG), and many countries have used this forum to share information or to co-ordinate activities.

A co-ordinated research project (CRP) on "Optimization of the Coupling of Nuclear Reactors and Desalination Systems" was initiated in 1998 with participation of research institutes from nine Member States. The main objectives of the CRP include a review of suitable reactor designs for coupling with desalination systems, optimization of coupling technology, possible performance improvements, and advanced technologies of desalination systems for nuclear desalination. The CRP is aimed at identifying optimum coupling conditions (configuration, process parameters, etc.) for a wide variety of nuclear system and desalination processes. A number of technical co-operation projects between technology developers and end-users (e.g. the 1999 project entitled "Integrated Nuclear Power and Desalination System Design") have assessed the feasibility of particular projects.

A nuclear desalination plant is, in most cases, a co-generation plant, producing both electricity and desalted water. Several cost allocation methods suitable for the evaluation of co-generation plants have been developed. The IAEA offers the computer package DEEP (Desalination Economic Evaluation Program [23]), for the economic comparison of different energy-desalination plant options based on the "Power Credit" method. Results of a recent study which utilized DEEP are in the following section.

6. ECONOMIC PERSPECTIVES

The IAEA has recently employed DEEP in performing a series of detailed economic calculations of desalination using a range of fossil and nuclear energy sources (e.g. pulverized coal-fuelled electric power plant, pressurized light water nuclear reactors, high temperature nuclear reactors, and others), which were coupled to selected desalination technologies (i.e. Multi-Flash Distillation (MSF), Multi-Effect Distillation (MED), and Reverse Osmosis (RO)) [24]. The study employed specific seawater and economic conditions, which are prevalent in various regions of the world. Conditions included a seawater salt concentration of the range 38,000 – 45,000 ppm total dissolved solids (TDS), desired water production capacity of the range 60,000 – 480,000 m^3/d, energy source power levels of 100 – 900 MW_e, and discount interest rates of 5 – 10%. Furthermore, two main economic scenarios were considered: one favouring nuclear energy sources and the other favouring fossil energy sources.

It was found that desalination costs ranged from 0.40$/m^3$ to about 1.90$/m^3$ depending upon the water plant type and size, energy source, specific region and economic scenarios. The differences between water production costs of the RO and MED processes were found to be typically smaller than the differences introduced by changes in discount rate over a wide range of power sources and regional conditions. In addition, calculated desalination cost results for the scenario favouring nuclear energy sources suggested that the nuclear option was particularly advantageous with both RO and MED. On the other hand, costs from nuclear and fossil options were found to be comparable for the scenario favouring the fossil option. It should also be noted that water production costs by MSF were systematically higher than production costs using MED or RO, independent of the energy source or region considered. Moreover, RO water desalination costs were found to be lower than for MED and MSF when the less stringent WHO (World Health Organization) water quality standards were used.

Water production costs with small reactors dedicated exclusively to heat production were systematically higher compared to larger dual-purpose nuclear reactors. When considering the MED process, for example, water production costs from the heat-only reactor were about 30-40% higher than those from the dual-purpose reactor, mainly because energy costs were approximately 2-fold higher. Results also suggested that the competitiveness of the nuclear option could become questionable if the capital costs of nuclear power plants are increased by 15-20% (assuming fossil fuel cost to be 25$/boe (or lower). In addition, existing nuclear power plants appeared not be competitive for discount rates above 11%; however, a limiting value did not appear to exist for discount rate with innovative nuclear reactors.

It was encouraging that results obtained from calculations and analyses made independently by five countries in the context of specific national programmes yielded trends, which are in line with the above findings.

Additional analysis of results obtained indicate that the competitiveness of nuclear desalination relative to conventional desalination would be significantly increased if capital cost could be reduced, as currently envisaged for innovative reactors under development [24].

7. GUIDANCE FOR PREPARING USER REQUIREMENTS DOCUMENTS (URD)

When nuclear desalination is considered as an option to contribute to the fresh water issue in developing countries, the policy planners must establish certain user requirements. The Agency documented a guidance in order to assist professionals involved in energy planning and in building and operating Small and Medium Reactors (SMRs) in developing countries, who will determine and specify requirements for SMRs to be used for electricity generation and other applications e.g. nuclear desalination [25]. The scope of the guidance covers the entire plant up to the interface where the product (i.e. electricity, heat, and desalted water) is delivered to the distribution grid. Essential key points are summarized below.

Rationale for nuclear desalination programme

The rationale for choosing seawater desalination relative to other potable-water source options should be established. Steps aimed at the management of potable water consumption through the reduction of water use and the development of existing water resources should be taken before deciding on seawater desalination (whatever energy source is assumed). In addition, rationale for choosing nuclear energy in comparison with other energy options to power the desalination plants should be made clear.

Nuclear desalination projects have to be carried out in the context and within the framework of the national water plan, taking into account the national industrial development plan and the national energy plan. Information should be collected regarding national water standards (domestic, agricultural and industrial), national water resources and the degree of their utilization, national water treatment and distribution networks and their performances, water demand projections and forecasts, and existing and planned water infrastructure.

Site-imposed requirements

Important information to be specified is site conditions and site infrastructures related to the design of the desalination plant. Site-specific conditions include: feed water quality (e.g. total dissolved solids (TDS), electrical conductivity, chemical oxygen demand (COD), seawater turbidity etc.); sea bottom quality (e.g. appearance, pH, Water content, granular earth composition, COD, etc.); oceanographic data (e.g. daily, monthly and seasonal seawater temperature, tide tables for the proposed site, etc.); topographical data of the area where the nuclear desalination plant is being sited and the product water will be distributed.

Site-specific infrastructure information includes: the institution (authority) in charge of the production and distribution of potable water; definition of the quality and the quantity of the potable water to be supplied; and storage facilities and the distribution system.

Financing requirements

A clear definition of the future owner-operator(s) of the nuclear desalination plant at an early stage is helpful for financing arrangements, separate or unified, for the nuclear and the desalination plants. The user should specify the different financing options to the vendor (as practiced in the case of a nuclear power plant), who has to respond by giving the preferred financing option such as the Build-Own-Operate, Build-Operate-Transfer or other alternative approaches.

Special national requirements

There are other important requirements to be established by the planners prior to the project implementation. These include: manpower development, infrastructure and national participation, technology transfer, licensing support, contractual options and responsibilities, extended guarantees and warranties, special materials and spare parts supply, technical support and long-term partnerships.

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A DISTINCTIVE COUPLING CONCEPT OF SMALL NATURAL CIRCULATION BWR AND RO DESALINATION SYSTEM

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Abstract

Although small sized nuclear plants are generally disadvantaged in economics, a coupling concept consisting of a small natural circulation boiling water reactor (BWR) and a reverse osmosis (RO) seawater desalination system potentially overcomes it without sacrificing the level of safety. Small natural circulation BWRs are known with their simple designs in reactor, safety system and balance of plant (BOP). The reactor core is cooled by natural circulation of water without use of pumps. The reactor directly supplies steam without use of steam generators. The safety system can be rationalized for this type of reactor, which accommodates a large coolant inventory above the core in the reactor pressure vessel (RPV) to enhance natural circulation of the coolant. The large coolant inventory results in a large safety margin and consequently mitigates the design requirements on the safety systems. In the BOP, steam generated in the reactor works at the steam turbines to produce electricity. Some steam is used in turbine-driven pumps (TD-pumps) which compress the saline feed up to 7 to 8 MPa for the RO seawater desalination process. The use of TD-pumps as the interface between the BWR and the desalination process physically eliminates the possibility of radioactive contamination of potable water from the BWR steam. Moreover, it facilitates the use of motor driven pumps as backup instead of backup boilers or heat exchangers. Besides, all the technologies used in this concept are proven and existing so that neither large R&D nor new investment in manufacturing facilities is necessary. From these features, this simple coupling concept promises advantages not only in economics but also in safety for the areas that suffer shortage of potable water.

1. INTRODUCTION

Interest in nuclear energy application to seawater desalination has been increasing in many regions that suffer a shortage of potable water. For those regions, seawater desalination offers one of the most promising alternatives for the supply of potable water. Although seawater desalination has been mainly realized by the use of fossil energy, the issue of the greenhouse effect has become crucial as well as the issue of energy security. Under these circumstances, interest in nuclear application to seawater desalination has been increasing in many regions [1].

Such interest has been also increasing among the nuclear suppliers who look for new markets for nuclear installations. Although nuclear installations have been directed mainly towards electricity production, increase of electricity demand has recently flattened out in the developed countries so that the nuclear market has become stagnant. Nuclear installations for non-electric purposes or for co-generation purposes offer one of the most promising alternatives to revive the nuclear market. Under these circumstances, interest in nuclear installations for non-electric purpose including seawater desalination has been increasing among nuclear suppliers.

One of the major items to be solved for such nuclear applications to seawater desalination is economy without sacrificing the level of safety mainly because of the following four reasons:

- In general, capacity required for such seawater desalination plants is relatively small compared with nuclear power plants because water transportation to a wide area is economically less favorable than electricity transportation. This often results in disadvantages in economics according to economy of scale.
- Nuclear desalination needs special measures such as extra barriers (extra heat exchangers) to practically eliminate the possibility of radioactive contamination into potable water produced in the system.
- Backup systems are also needed to achieve high reliability as potable water supply system.
- Additionally, the level of safety for any nuclear plants shall not be sacrificed for economy.

As a potential solution to this item, a coupling concept mainly consisting of a small natural circulation BWR and an RO seawater desalination system through TD-pumps as interface (TTBWR+RO) is proposed in this paper. Both the BWR and the RO system are known for their simplicity in design, which improves economics as well as reliability. The use of TD-pumps, which are often used in nuclear power plants, enhances economics as well as safety because it can eliminate the use of an extra heat exchanger as an interface between the nuclear system and the desalination system. Additionally, all these technologies are well proven and existing so that neither large R&D nor new investment in manufacturing facilities is necessary. From these features, the coupling concept consisting of a small natural circulation BWR and an RO desalination system with the use of TD-pumps potentially overcome the economical disadvantages of small sized nuclear plants.

2. CONCEPTUAL DESIGN

Design study was performed under the principle of the maximum utilization of proven and existing technologies to show the technical feasibility of this concept focusing on reactor, safety systems and BOP. This principle contributes to improve not only the economics/reliability but also licensability where the same type of nuclear plants has already received the license.

2.1. Reactor design

A standard BWR design was further simplified to fit for a co-generation plant producing electricity and potable water under the design principle. Even the smallest size of standard BWRs is, however, too large (1600 MWth) to fit for the co-generation purpose [2]. Because the design principle is maximum utilization of proven technologies, the power density of the core was decreased instead of changing the core and/or fuel designs.

This decrease in power density results in a simplification of the coolant circulation system of the BWR because natural circulation is high enough for the core with such low power density. In general, BWRs have a high natural circulation capability in the RPV because of large difference in the coolant density between inside and outside of the core shroud. According to the core-flow-map of a BWR loading 368 fuel assemblies, up to 33 % (2.1 t/s) of the rated core flow is achievable with 37 % (19 kW/*l*) of the rated power on the natural circulation curve as shown in FIG. 1. The external recirculation loops including recirculation pumps are unnecessary from the BWR circulation system with the low power density core.

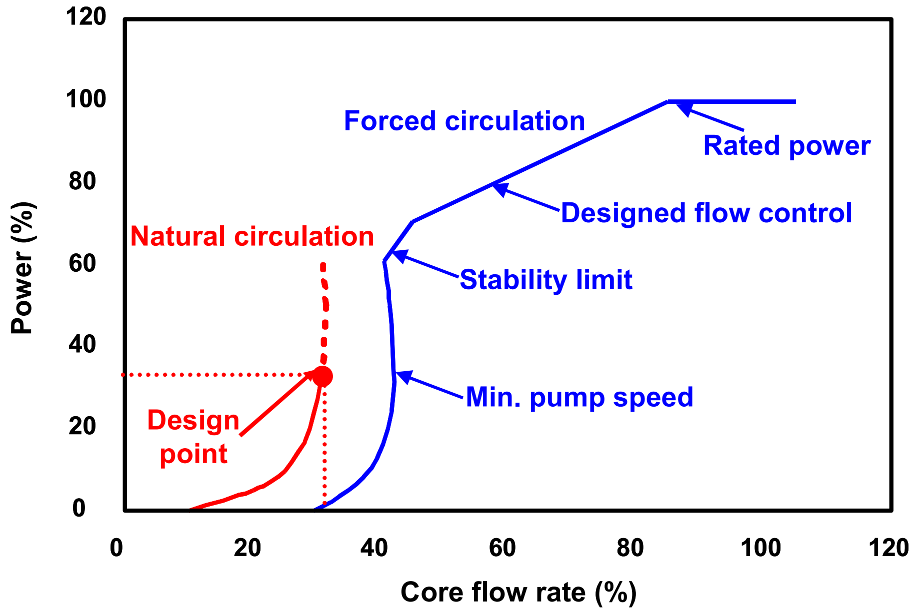


FIG. 1. Core flow map of a BWR including 368 fuel assemblies

The low power density core results in improvement not only in design simplification but also in availability of the plant. The low power density lengthens refueling intervals and consequently enhances availability of the plant. For example, 48 effective-full-power-month (EFPM) cycle length is achievable with the standard 45 GWd/t of BWR fuels. Theoretically, the availability exceeds 95 % with the low power density core.

The major characteristics of the reactor are summarized in Table I together with a cross-sectional view of the reactor with the core configuration.

2.2. Safety system design

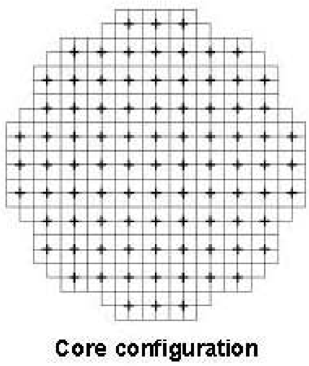
The safety systems of the most recent BWRs were further sophisticated to fit with this small natural circulation BWR with low power density [3, 4]. The emergency core cooling system (ECCS) configuration was rationalized taking advantage of the relatively small power of the core. The emergency power sources were diversified into two types: diesel generator (DG) and gas turbine generator (GTG), owing to the relatively small capacity required for them. The passive containment cooling system (PCCS) was adopted for overpressure protection of the primary containment vessel (PCV) in case of a severe accident. A configuration of the rationalized safety system is depicted in FIG. 2.

High-pressure injections were eliminated from the most recent ECCS design owing to the following two features: large coolant inventory above the core and small nozzle sizes on the RPV. The large coolant inventory resulted from the small thermal power density of the core increases the time for the water level in the RPV to reach the top of the fuel in case of loss of coolant accident (LOCA) for example.

The small nozzle sizes on the RPV resulted from the required capacity of piping for this small natural circulation BWR and give a smaller discharge flow rate in case of LOCA. This feature also increases the time for the water level to reach the top of the fuel in case of LOCA. These two features enhance the safety margin to eliminate rapid injection (high-pressure injection) of the coolant in the event of accidents. Therefore, the ECCS consists of only low-pressure flooders (LPFLs) together with a reactor core isolation cooling system (RCIC) and two automatic depressurization systems (ADSs) as shown in FIG. 2.

TABLE I. MAJOR CHARACTERISTICS OF TTBWR+RO NUCLEAR BOILER

Thermal output		589 MWth
Reactor pressure		7 MPa
Steam temperature		286 °C
Core flow rate		2.1×10^3 kg/s
Steam flow rate		0.3×10^3 kg/s
Core	Power density	19 kW/*l*
	Cycle length	48 EFPM
	Active fuel length	3.7 m
	Equivalent diameter	3.3 m
	No. of assemblies	368
	No. of control rods	89
Internals	No. of separators	85
	No. of dryer units	4
RPV	Height	21 m
	Inner diameter	4.7 m



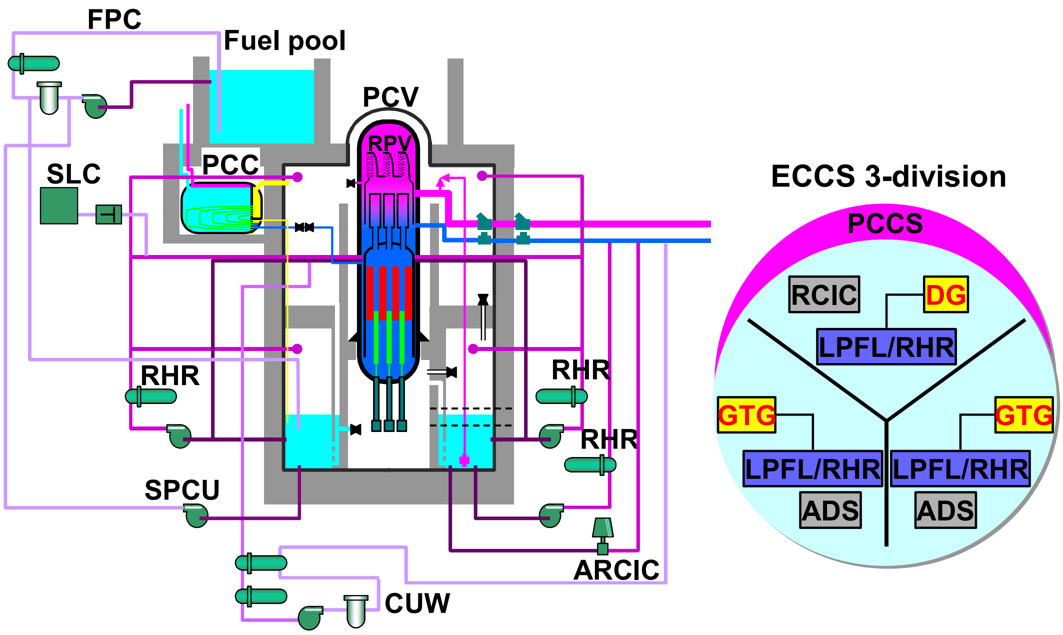


FIG. 2. TTBWR+RO safety system and ECCS division

The DG and two GTGs are adopted to improve the reliability of emergency power sources as well as economics. While DGs are often used as emergency power source in nuclear power plants, GTGs are hardly used due to their longer start-up time than that for DGs. This natural circulation BWR, however, features a large safety margin in response time so that it facilitates the adoption of GTGs. Because the required capacity for these emergency power sources is in the range used for the transportation industry including air and ship industry, less expensive DGs and GTGs with high reliability are obtainable from the market. Furthermore, the use of DG and GTG enhances the diversification of emergency power sources.

PCCS is adopted for overpressure protection of the PCV in case of a severe accident when no ECCS works. The PCCS is composed of passive components such as shell-and-tube heat exchangers (passive containment cooler: PCC) located above the core without active components such as pumps. The PCC works as follows. In the event of a severe accident, steam generated in the RPV by decay heat flows out into the PCV. Some steam in the PCV is absorbed in the tube side of the PCC and condensed by heat exchange with water in the shell side. The condensed water returns to the RPV from the PCC by gravity. Steam generated by heat exchange in the shell side flows out to the atmosphere [4]. Through these functions, the PCCS protects the PCV in case of a severe accident.

2.3. BOP design

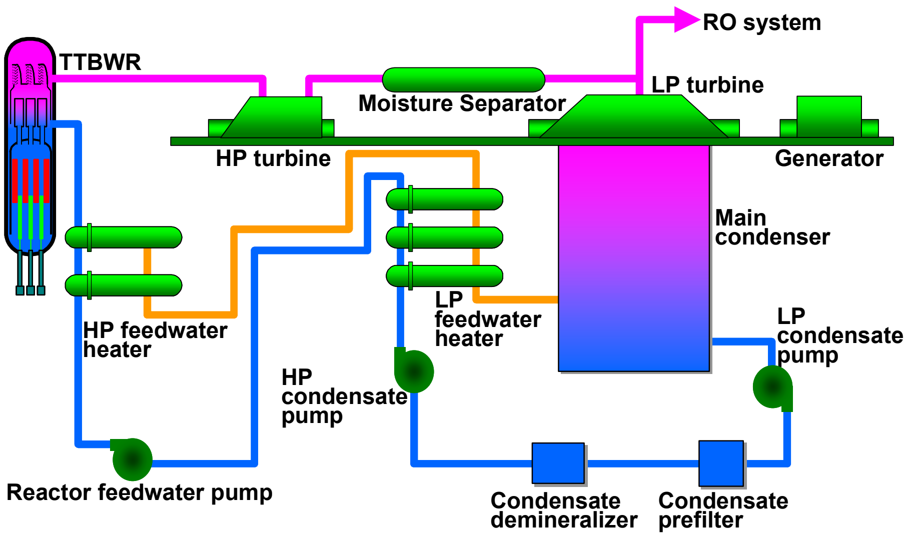
The BOP of this coupling concept mainly consists of a turbine system generating 182 MWe and a seawater desalination system producing about 100×10^3 m^3/d of potable water as a reference design (FIG. 3). The whole system was designed based on existing technologies.

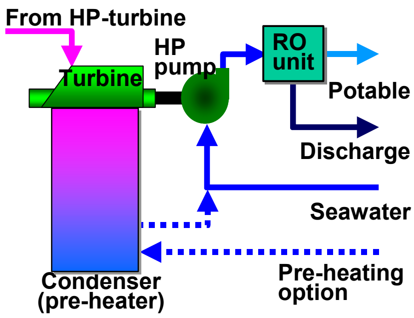
The turbine system employs the regenerative steam cycle consisting of two stages of high-pressure feedwater heating and three stages of low-pressure feedwater heating. Steam generated in the reactor works in the high and low-pressure turbines producing electricity of 182 MW. Steam condenses at the condenser then the condensed water is heated by the feedwater heaters to improve the thermal efficiency before it reaches the reactor.

Some part of steam (30 kg/s) bled after the high-pressure turbines works in the turbine side of the two TD-pumps. The TD-pumps compress seawater up to the osmotic pressure (about 7 MPa) required for the RO process. The RO units produces about 100×10^3 m^3/d of potable water assuming the energy recovery ratio of 40 % [1].

Instead of backup boilers, which are often accommodated in the distillation seawater desalination systems, a motor-driven pump (MD-pump) is equipped in this RO system for backup. Because the MD-pump is powered by the external sources, backup boilers together with the associated systems are unnecessary.

This RO system including TD-pumps as the interface has advantages in efficiency, in economics, and in safety over the conventional distillation system for seawater desalination. This RO system produces about 100×10^3 m^3/d of potable water while the distillation system produces up to 80×10^3 m^3/d if the same amount of steam is used for distillation process [1].





RO system

General	
Rated thermal power	589 MW
Rated electric power	182 MW
Water production rate	102×10^{3} m^{3}/d
Steam cycle	
Turbine type	TC4F23"
Stages of feedwater heating	HP×2, LP×3
Desalination	
Process	RO
HP pumps	TD×2 (50 %×2)
Capacity	17 MW/unit
Backup	MD×1 (50 %)
Pre-heating by condenser	Option

FIG.3. BOP of TTBWR+RO

Only a MD-pump is added for backup of the RO seawater desalination while backup boilers together with the associated systems including fuel tanks are necessary for the distillation system. Because the possibility of radioactive contamination of seawater from BWR steam is physically eliminated, no extra barrier is necessary for this RO system. While the distillation system need extra barrier (extra heat exchanger) to lower the possibility because the only thin wall in the heat exchangers separates BWR steam and seawater. From these features, this RO system is more favorable as nuclear seawater desalination system than the distillation system.

3. PERFORMANCE EVALUATION

The advantages of this coupling concept were evaluated focusing on its ECCS performance, its core damage frequency (CDF), and its brief economics. ECCS performance was evaluated through design-base-accident (DBA) analyses. CDF was evaluated with probabilistic safety analysis (PSA). Economics was estimated based on cost data for advanced BWRs (ABWRs).

3.1. ECCS performance

LPFL functions were evaluated through DBA analyses because the ECCS consists of only LPFLs together with a RCIC and ADSs (FIG. 2). Among DBAs, LPFL line small break LOCA (LPFL SBLOCA) was chosen to be analyzed in this paper because the accident is generally most severe for this type of BWR. Accident analyses were carried out with the conventional safety evaluation code, which is often used for existing plants' analyses.

The assumptions used in the analysis were based on those used for the existing plants' analyses. The major assumptions are as follows. The most severe single failure, that is one train of the LPFL is out of order, was assumed for LPFL SBLOCA. Safety systems are triggered by the water level signals which have been set as listed in TABLE II. The rated capacity of single LPFL is 150 t/h (15 % of ABWR LPFL capacity).

TABLE II. WATER LEVEL SET POINTS USED FOR SAFETY SYSTEM'S TRIGGER

Water level set point (outside shroud)	Function
Level 8 (high)	RCIC, LPFL injection stop Main turbine trip
Level 3	Reactor SCRAM
Level 2	RCIC (make-up function) start-up
Level 1.5	MSIVs closure RCIC (ECCS function) start-up GTG loaded LPFL start-up
Level 1	DG loaded LPFL start-up ADS activation

The analytical result shows that water level in the shroud has been above the top of the fuel even in case of LPFL SBLOCA (FIG. 4). Right after LPFL break, reactor pressure drops because of scram triggered by the level 3 signal (FIG. 5). When the water level reaches at the level 1 set point, ADS is activated to depressurize the RPV. Because reactor coolant discharges through ADS valves, water level in the shroud begins to decrease after flushing according to depressurization. With some delay due to DG start-up process, one train of LPFL starts injection assuming that a line break on one LPFL and that "single failure" on another LPFL. This single LPFL injection recovers the water level in the shroud before it reaches the top of the fuel in case of LPFL SBLOCA.

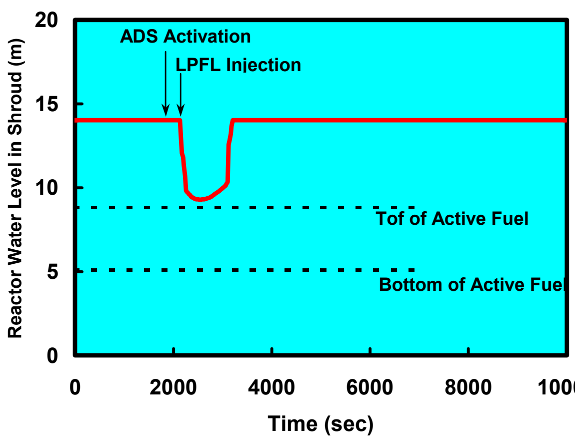


FIG.4. Water level response after LPFL SBLOCA

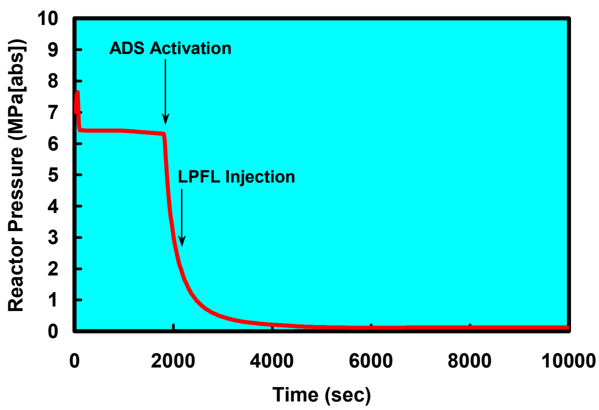


FIG.5. RPV pressure response after LPFL SBLOCA

The rationalized ECCS equipped on the small natural circulation BWR has a sufficient capability to maintain the reactor of safe condition. This capability was confirmed by LPFL SBLOCA analysis with severe assumptions. Other accident analyses including large break LOCA and loss of feedwater also support the capability of the rationalized ECCS.

3.2. Core damage frequency

To compare the safety level of this small natural circulation BWR with that of ABWR, the core damage frequency (CDF) was evaluated through probabilistic safety assessment (PSA). ABWR's total CDF had been evaluated at far below 10^{-6} per reactor year, which is one of the best in the operating nuclear reactors in the world [5].

The total CDF mainly consists of generating frequencies of the following six core damage sequences using frequency of initiating events and unavailability of the safety systems/functions. Frequency of initiating events as well as unavailability of the safety systems/functions were basically derived from data used for ABWR PSA. Those which are not available for ABWR PSA were set by engineering judgment for TTBWR PSA.

TQUX: Failure of high-pressure core cooling and depressurization

TQUV: Failure of low-pressure and high-pressure core cooling

TB: Station blackout

TW: Loss of decay heat removal

TC: Anticipated transient without scram

LOCA: Failure of all ECCS following a LOCA

The total CDF of TTBWR is as low as ABWR's although there are small differences in initiating frequency of each core damage sequence between TTBWR and ABWR (FIG. 6). The initiating frequency of TQUX, TB and TW sequence for TTBWR is slightly better than that for ABWR, respectively. This improvement results mainly from ADS function for all events, the diversified emergency power sources and the PCCS make-up capability on those sequences. The initiating frequency of TQUV sequence as well as LOCA sequence for TTBWR is slightly worse than that for ABWR. This difference is derived from rationalization of high-pressure ECCS for TTBWR, which results in reduction in number of make-up trains to the RPV and consequently raises the unavailability of make-up function on the TQUV sequence as well as LOCA sequence. Although the initiating frequency of the TC sequence for TTBWR is also worse than that for ABWR, it is so low that it yields negligible influence on the total CDF. Counting these positive and negative influences, the total CDF for TTBWR is assessed at the same level of the total CDF for ABWR.

3.3. Economics overview

To evaluate the energy generation cost (EGC) of TTBWR+RO, a parametric survey was carried out mainly counting a co-generation effect, an availability effect, a scaling effect and a learning effect on the ABWR cost data [6]. The EGC was defined as the total cost per the total product consisting of electricity and potable water value in this paper. The total cost consists of construction cost, operation and maintenance cost, and fuel cycle cost.

The co-generation effect was applied to count the value added by seawater desalination. The potable water value is represented by the equivalent value of electricity in kWh. For example, value (price) of 1 m^3 of potable water is equivalent to about 10 kWh of electricity in a city of Japan where water shortage hardly occurs. The water value is higher in the areas that suffer water shortage. Potable water value was changed as a parameter in the EGC evaluation.

The availability effect was applied to count the higher availability of TTBWR+RO because of the longer refueling intervals than that of ABWR. With the standard 45 GWd/t of BWR fuels, the availability theoretically exceeds 95 % resulting in cost improvement by over 10 %.

The scaling approach was applied to estimate the construction cost of TTBWR+RO from that of ABWR considering the difference in capacity/volume of major components such as nuclear boiler, ECCS/safeguards, turbine equipment, electrical equipment and structures. The cost of each component was calculated assuming it is generally proportional to the component capacity/volume powered to the 0.7. Small sized nuclear plants generally encompass a disadvantage in economics due to this scaling effect.

The learning effect was applied to count what is known with standardization or factory fabrication effect. Because the power of TTBWR+RO is smaller by about 85 % than that of ABWR, 5-6 times larger number of plants is projected to be implemented to meet the demand. This facilitates standardization or factory fabrication to improve the construction cost.

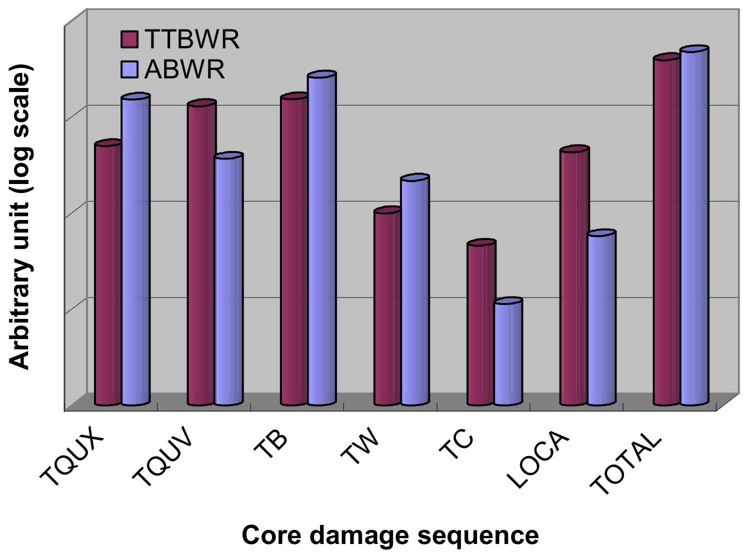


FIG. 6. Comparison of CDF between TTWR+RO and ABWR

The parametric survey shows the range of water value (more than 50 kWh/m^3) where the EGC of TTBWR+RO is competitive with that of ABWR (FIG. 7). The EGC decreases with increase of water value mainly according to the co-generation effect, which counts the additional product value by seawater desalination. If the water value is more than about 50 kWh/m^3, the EGC falls below the EGC of ABWR (100 %/kWh).

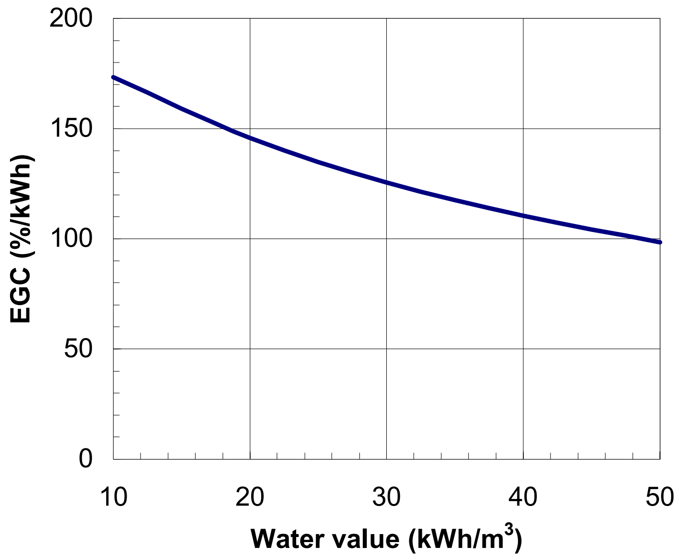


FIG. 7. Energy generation cost of TTBWR+RO (ABWR EGC = 100 %/kWh)

The EGC of TTBWR+RO is competitive in the areas where potable water value is relatively high compared with electricity value because the co-generation effect together with the availability effect and the learning effect overcomes the scaling effect.

CONCLUSION

The coupling concept consisting of a small natural circulation BWR and an RO seawater desalination system was proposed because interest in nuclear energy application to seawater desalination has been increasing both from the many regions who suffer scarcity of freshwater and from nuclear suppliers who are looking for new nuclear market.

The design study showed the technical feasibility of this concept focusing on reactor, safety systems and BOP under the principle of the maximum utilization of proven and existing technologies. The low power density core of this natural circulation BWR facilitates long operating cycle resulting in a high availability. The safety system is rationalized due to the large coolant inventory in the RPV. The co-generation system consisting of electricity generation and RO seawater desalination system is simple and reliable thanks to the use of TD-pumps as interface between the BWR and the RO system.

The advantages of this coupling concept were confirmed focusing on its ECCS performances, its CDF, and its economical competitiveness. The rationalized ECCS has a sufficient capability to maintain the reactor of safe condition. The total CDF is assessed at the same level of the total CDF for ABWR, which is one of the best among the operating nuclear plants. The energy generation cost of this concept is competitive in the areas where potable water value is relatively high compared with electricity value because the co-generation effect together with the availability effect and the learning effect overcomes the scaling effect.

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OPTIMIZATION OF NUCLEAR POWER USAGE: CO-PRODUCTION OF ELECTRICITY & WATER WITH THE GT-MHR

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Abstract

The new generation of high temperature reactors produces electric power through a direct gas turbine cycle. This design gives a high electric efficiency of 47% but also releases low temperature heat at the cold sink which is usable by a desalination plant without affecting the electricity production. In this configuration, 75 to 80% of the nuclear heat is used and soft water production is inexpensive because of the use of free rejected heat.

1. INTRODUCTION

Nuclear power has mostly been used for electricity production and contributes about 17% to the world electricity needs. Light water reactors have taken by far the largest share of this production. However, the enticement to save natural resources of all kinds and to minimize wastes is demanding an improvement in energy efficiency. The trend is clear in fossil fueled plants with natural gas combined cycles reaching over 50% efficiency. In the nuclear field, the only proven concept that can pursue such a trend is the high temperature reactor, as all others are limited in thermodynamic efficiency by their coolant. High temperature reactors connected to a steam cycle have proved efficiencies up to 40%. However, for them to achieve their full potential with the high temperature helium coolant, their coupling to a direct cycle gas turbine or a combined cycle allows them to reach about 47% efficiency in the first case or slightly over 50% efficiency in the second case with today's technology.

Nowadays, the limiting factor to these values is material related, therefore, as material physical characteristics improve, these types of plants will be able to follow the upward efficiency trend. The direct Brayton cycle is the short term favored cycle over the combined cycle because the complications and costs of the combined cycle cannot be justified for just a few percentage points of improved efficiency.

While electricity is a recognized commodity linked to modern society anywhere in the world, fresh water for human usage or agricultural irrigation is so short that it is the main obstacle to development in many areas of the world. Hopefully there is a lot of water in oceans but its desalination, while well proven technically, is often too expensive because of the large required energy.

The direct gas turbine Brayton cycle implemented in the GT-MHR (Gas Turbine- Modular Helium Reactor) turns out to be an ideal source of heat to supply a MED desalination unit because in the present case this heat is virtually free bringing down the cost of supplied fresh water tremendously.

Wherever such a combination of electricity and water supply is needed, the GT-MHR brings a very competitive solution to both supplies, while at the same time making a very efficient use of the nuclear heat since in this case 75 to 80% of the heat is used.

I – Brief description of the GT-MHR

The GT-MHR is being developed in an international collaboration framework and has been described in session G of this conference. Here we only summarize the most salient characteristics. The GT-MHR is an advanced generation of High Temperature gas cooled Reactor which fully takes advantage, by design, of the exceptional properties of the coated fuel particle to ensure a high degree of safety while taking advantage of modern gas turbine technology to reach high efficiency. A prototype unit should start tests in 2010 and commercial units could be available around 2013.

HTR fuel is based on the coated fuel particle principle (see fig. 1) which has been proven to be very sturdy and contains fission products within its cladding as long as its temperature stays bellow 2000°C approximately. In order to keep some margin it is customary to design modern HTR's with a goal to keep the fuel below 1600°C during the most challenging transients.

The GT-MHR is designed to meet this requirement in any circumstance thanks to the modular design, the large thermal inertia of the core and the capacity of the steel pressure vessel to thermally radiate heat out, including the residual heat after a loss of coolant accident. In these circumstances, no core melt nor significant release ever occur.

The core, with its annular design and enrichment zoning, has been designed to meet the previous safety requirement for an optimal power level in order to be as competitive as possible. This optimization led to the following characteristics (table 1) and a 600 MW thermal core. The principle of the Brayton cycle to which this nuclear unit is coupled is shown on fig (2) while its lay out is illustrated in fig (3).

TABLE I. THE OPTIMIZED CHARACTERISATION

• Reactor Power MWth	600
• Core Inlet/Outlet Temperatures, °C	491/850
• Core Inlet/Outlet Pressures, MPa	7.07/7.02
• Helium Mass Flow Rate, Kg/s	320
• Turbine Inlet/Outlet Temperatures,°C	848/511
• Turbine Inlet/Outlet Pressures, MPa	7.01/2.64
• Recuperator Hot Side Inlet/Outlet Temps °C	511/125
• Recuperator Cold Side Inlet/Outlet Temps °C	105/491
• Net Electrical Output, MWe	284
• Net Plant Efficiency, %	47

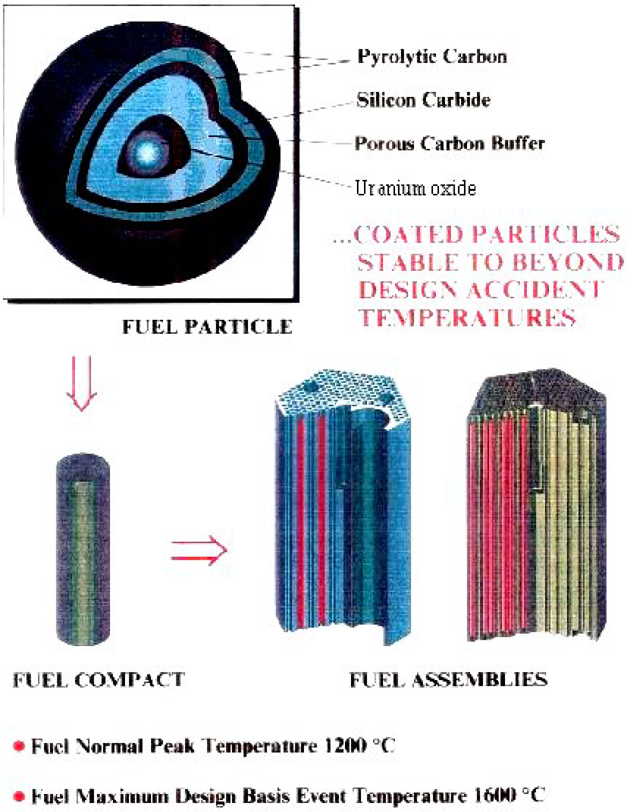

FIG.1. HTR fuel.

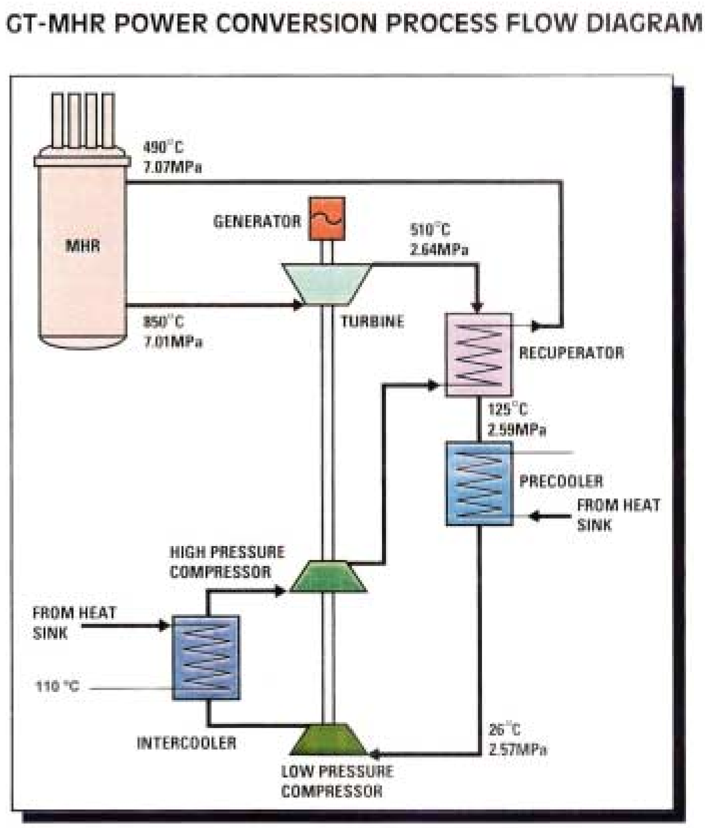

FIG. 2. The Brayton cycle.

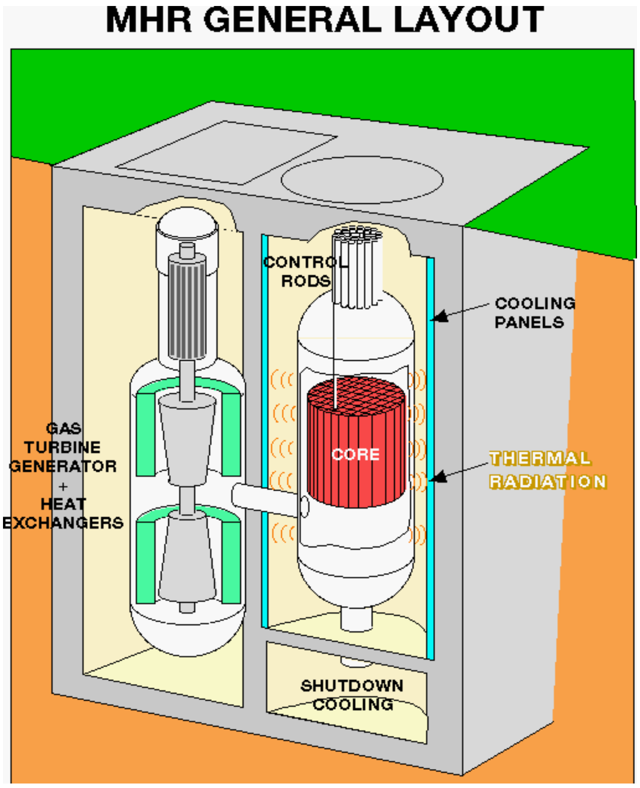

Fig 3. The layout of the GT-MHR.

We wish to stress a specific feature of this thermodynamic cycle. Heat is rejected at the cold heat sink at the precooler and intercooler at a temperature of about 110°C. This temperature is sufficiently high with respect to air temperature anywhere in the world to allow use of dry air cooling heat sink thereby permitting the plant localization anywhere irrespective of river or sea availability which is an interesting feature as potential sites for power units along rivers are becoming scarce.

But, and this is where this feature can be put to efficient use with desalination, heat removed from the precooler and intercooler can be put to good use in supplying heat to a (MED:Multiple Effect Desalination) desalination unit.

Indeed, in contrast to all other nuclear desalination schemes where heat supplied to a desalination unit is taken away from the steam cycle and therefore induces a decrease in electricity production, the GT-MHR electricity production is not affected at all by the use of the cold sink heat. Therefore its highly efficient electricity production remains unchanged but its released low temperature heat is put to good use virtually free. This is actually the only nuclear/desalination coupling where both end products add up to the nuclear heat use for an effective usage of 75 to 80% depending on sea water temperature.

II – Coupling with a desalination MED plant

The MED desalination process has been chosen here because it is the most efficient user of thermal heat for water desalination and because it uses low temperature heat. Indeed, in order to control corrosion and crud accumulation in the desalination plant, it is advised not to exceed 70°C into the MED plant. Practically we prefer to stay away from the limit and assume a head inlet temperature Tp into the MED plant of 65°C.

According to figure 4, in these conditions, the total recoverable energy for the MED plant from the precooler and intercooler rejected heat is :

$$F = \frac{T8 - Tp + T2 - Tp}{T8 - T1 + T2 - T3} = 60\%$$

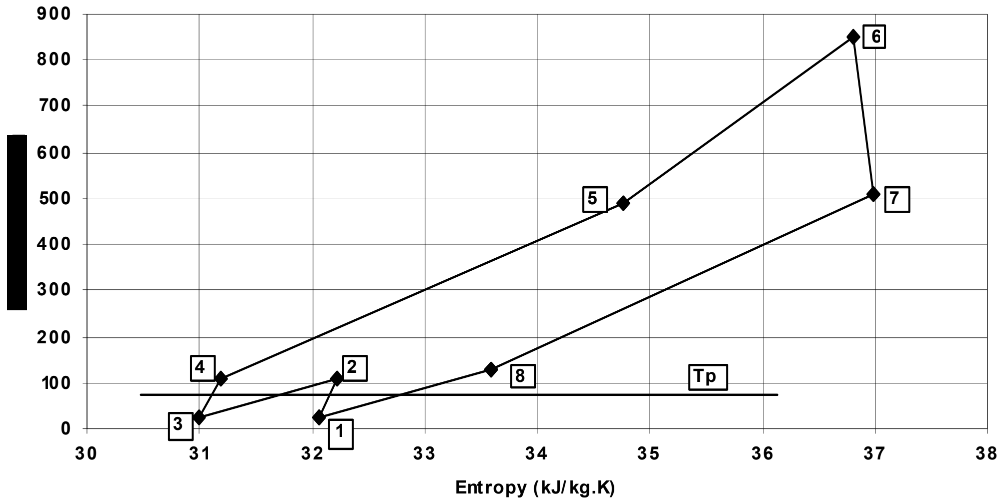

FIG. 4. TS diagram of GT-MHR cycle - Ratio of recoverable thermal power.

Therefore about 180 MW out of the rejected 300 MW at the heat sink are used for desalination.

Depending on the need for water the desalination plant can be designed as a single set of effects, typically 5 which would produce about 25 000 m^3/day of fresh water or as several lines in parallel which would optimize water production, typically 3 lines for a production of 42 000 m^3/day. Of course the desalination plant is quite larger in the second case because more effect stages are used to improve production. The following is a typical cost evaluation of the desalination unit (25 000 m^3/day).

These costs have been provided by SIDEM (Société Internationale pour le dessalement de l'Eau de Mer) a major world desalination unit supplier.

The same calculation carried out for a 42 000 m^3/day unit leads to the same cost of produced water.

Of course these are typical costs used to illustrate the large cost advantage of the coupling thanks to the free rejected heat of the GT-MHR. It basically shows that the price of water production is about half the cost of water typically produced by MED in general because heat accounts for about half of the production cost.

Two examples of coupling between the heat rejection from the GT-MHR cycle and the desalination unit are shown on figures 5a and 5b.

• In order to avoid fouling and scaling inlet MED temperature is assumed to be 63 °C

• Sea water temperature is typical Gulf temperature 35 °C
(These assumptions penalize performances)

Calculation assumptions :

.1.1.1.1.1 MED plant delivered ready to run: 20 M$

Availability:	95%
Annual water production:	7 880 000 m3
Plant life:	20 years
Interest on money:	5%
Amortizing rate:	8%

Amortization

$$\frac{20\,000\,000 \times 8\%}{7\,880\,000} = 20.3 \text{ US c/m3}$$

Energy

Thermal (free)	=	0
Electricity (2 kWh/m3@5US c/kWh)	=	10 US c/m3

Chemical products

(antiscaling,antifoam)	=	3 US c/m3

Maintenance

25% of investment

$$\frac{20\,000\,000 \times 2{,}5\%}{7\,880\,000} = 6.43 \text{ US c/m3}$$

Personnel

8 operators + 2 supervisors (300 000 $)

300 000/7 880 000	=	3.8 US c/m3
Total	=	43.5 US c/m3

Figure 5a shows a process heat exchanger supplying hot water to the desalination line. The water crosses the line in separate evaporators to recover a maximum of thermal power. A second heat exchanger rejects the unused thermal power ; it is able to reject the full power of the GT-MHR when the desalination unit is not working.

Figure 5b shows another way to feed the desalination unit using a flash vessel to produce the first steam supplied to the head effect.

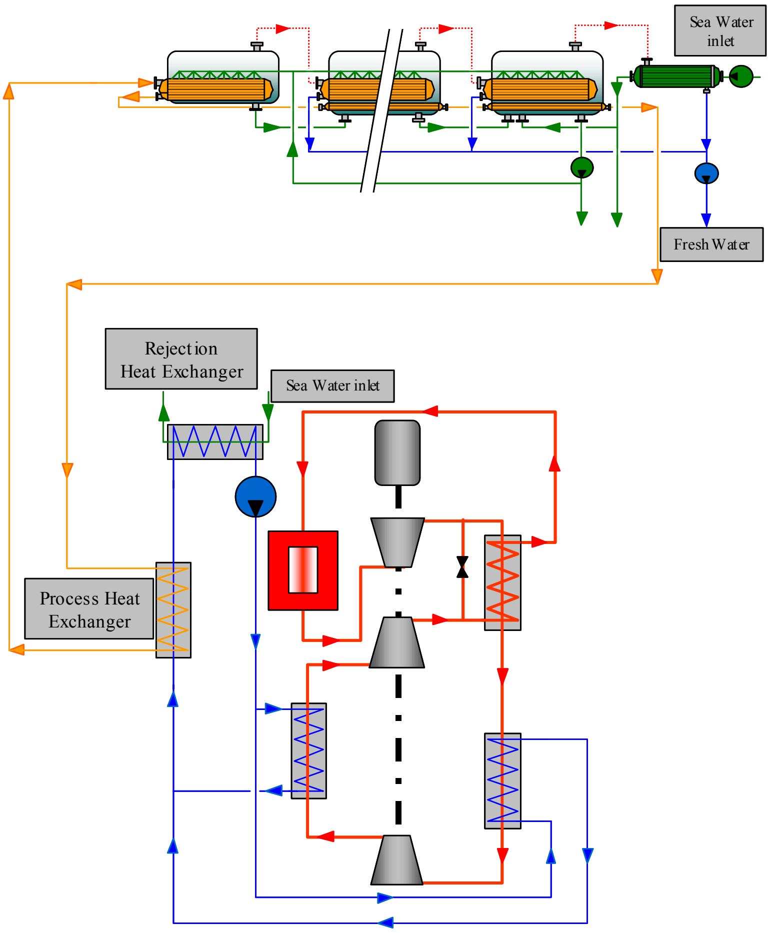

FIG. 5a. GT-MHR Distribution Water Production Arrangement (42 000 m^3/day)

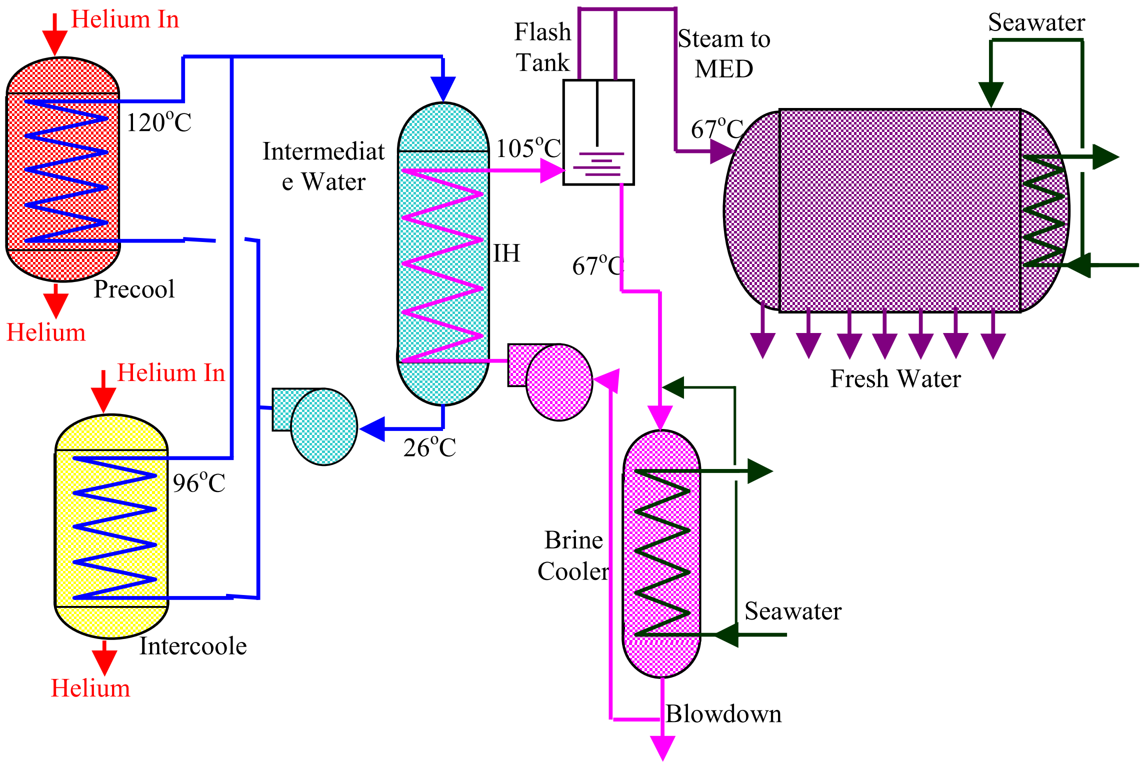

FIG. 5b. GT-MHR Distribution Water Production Arrangement (25 000 m^3/day)

It should be pointed out that water produced by MED has a very low level of solids content, typically less than 10 ppm.

If larger amounts of water were needed and more electricity as well, several sets of GT-MHR - MED units would be needed. However if more water were needed in proportion of the electricity, electricity could be used to power a reverse osmosis plant producing more water. The resulting water from reverse osmosis contains more dissolved solids and may not be suitable for some uses. In this case proper blending between water produced by the MED and reverse osmosis units can provide the right water quality.

CONCLUSION

The GT-MHR is an advanced generation nuclear reactor with very high safety and high efficiency producing 280 MWe sufficient for a population of 300 000 about. Where water is scarce it can also provide up to 42 000 m^3/day of fresh water at a very competitive price when coupled to an MED desalination unit thanks to use of the free rejected heat of its gas turbine thermodynamic cycle. This would typically represent more than 50% of that population needs. If more water were needed by that population, an additional reverse osmosis plant could supplement water production at the expense of a lower net electricity production and water blending can optimize water quality to its end use.

The coupling of the GT-MHR to the MED plant leads to a total useful conversion of about 80% of the nuclear heat. Such efficient use of nuclear energy is exceptional, contributes to minimization of wastes and to the overall welfare of areas where electricity and water are needed.

RUTA POOL-TYPE REACTOR FOR HEAT SUPPLY AND THE POSSIBILITY FOR ITS APPLICATION AREA EXPANSION

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Abstract

RUTA, a reactor facility with a pool-type reactor, has been designed for heat supply of residential districts. A relatively low potential of the heat generated by the reactor requires a special approach to building up heat supply systems with RUTA facilities. The application of the RUTA facility as a heat source for seawater thermal distillation has been considered. It is possible to use the reactor for neutron therapy. The reactor optimization provides for the improvement of the facility's consumer qualities.

1. 1. INTRODUCTION

RUTA is a single-purpose reactor facility for production of thermal energy [1,2]. The prime objectives of the RUTA project development are to:

- develop an alternative high-reliability heat source for the heat supply market (or for the seawater desalination market);
- reduce dramatically the fossil fuel consumption for heat supply;
- reduce the thermal energy cost in the heat supply market;
- improve in general the culture of heat production and quality of heat supply;
- improve the ecology in the heat consumption locations.

Variants of the RUTA reactor with a thermal power of 10 to 70 MW have been developed at the conceptual level. The facility's economic efficiency improves significantly as the reactor power is increased. Reactors with the thermal power of over 50 MW should be considered as most promising. RUTA-55 reactor with a thermal power of 55 MW is presented in the report.

2. DESIGN AND BASIC TECHNICAL CHARACTERISTICS OF THE REACTOR

RUTA is a pool-type reactor with a light-water coolant. Pool-type nuclear reactors are the most safe of modern nuclear installations. The existing experience of designing and operating pool-type research units successfully run for a long time in many cities in Russia and in the world was generalized and used in the development of the RUTA power reactor facility's design. Low water coolant parameters (the atmospheric pressure in the pool) and the simplicity of the design make the reactor and all of the facility's components highly reliable. The basic technical characteristics of the reactor are presented in Table 1.

TABLE I. BASIC TECHNICAL CHARACTERISTICS OF THE RUTA REACTOR

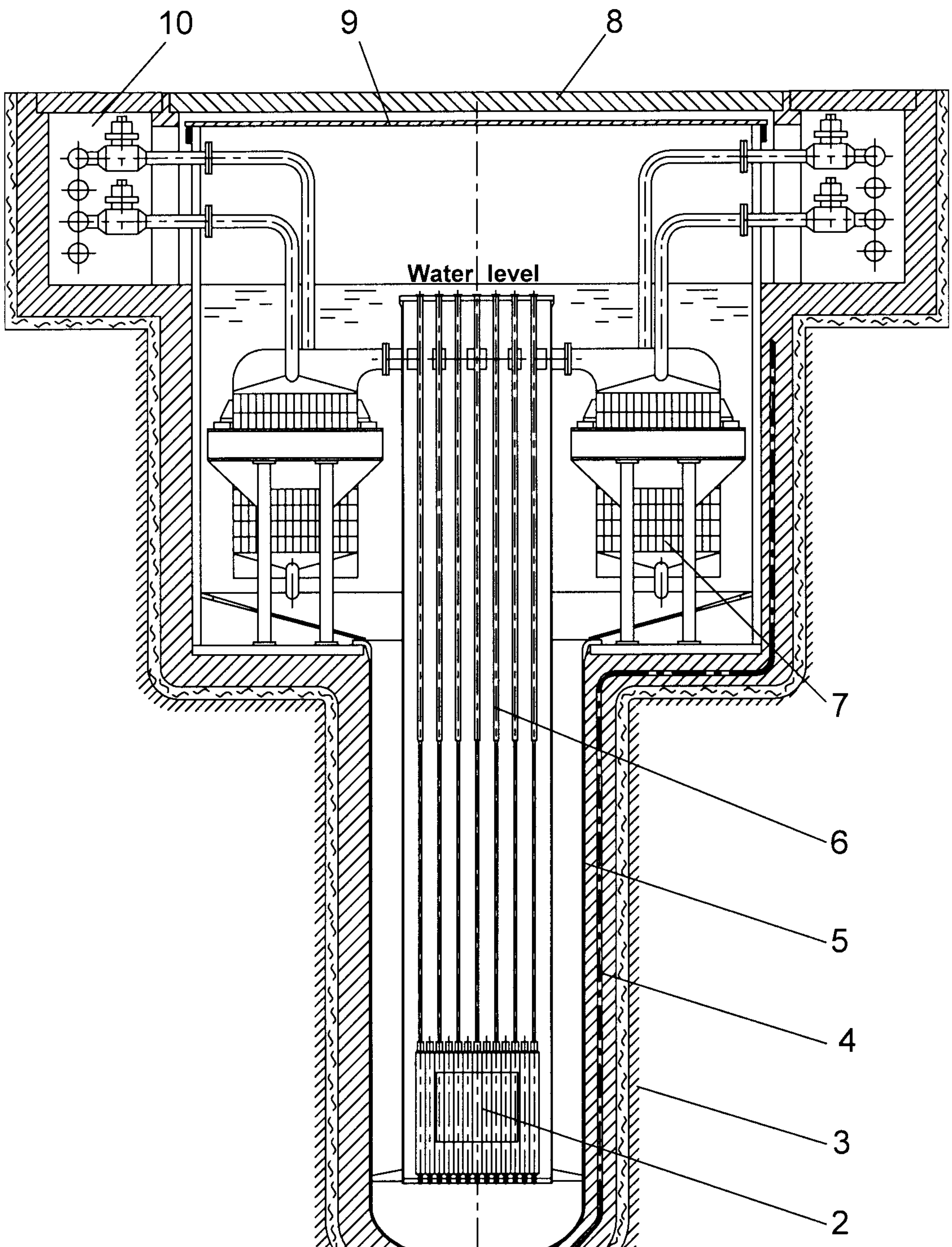
Characteristic	Value
Reactor thermal power, MW	55*
Parameters of the reactor facility coolant**:** Pressure, MPa primary circuit secondary circuit tertiary circuit Temperature, °C Primary circuit Secondary circuit Tertiary circuit	 0.25/0.14 0.4 0.6 –1.0 75/100 66/90 60/85*
Core dimensions (height/diameter), m	(1.2÷1.5)/(1.42÷2.03)
Fuel	UO_2
Fuel enrichment by U_{235}, %	3.6 – 6.0
Fuel life, eff. days	2300 – 2970
Core power density, MW/m^3	14 – 25

* Application of forced circulation in the RF primary circuit during high-power reactor operation mode provides for a power increase of up to 70 MW and ensures the tertiary coolant temperature of t = 70/90°C.

The structural diagram of the reactor is presented in Figure 1. The reactor is located in a concrete tank vessel with an internal bimetallic lining. The concrete vessel also serves as a radiation shield. Concrete plates are installed over the reactor to protect the equipment against external impacts. The reactor tank is filled with distilled water at the atmospheric pressure in the above-water air space. The primary circuit components with an integral in-tank arrangement operate at the hydrostatic pressure. The primary circuit's hydraulic train is arranged inside the reactor's water volume by a system of the internals and consists of the core, the upcomer section, primary heat exchangers, the reactor pool and the downcomer section. The core is build up of hexagonal fuel assemblies with fuel rods. The fuel is uranium dioxide. A core partial refueling is envisaged with the FA partial replacement once in three years. The overall nuclear fuel life is 6-9 years.

The RUTA reactor facility has three circuits. The coolant circulation in the primary circuit is natural and a power boost through forced circulation is possible when pumps are installed.

Heat is transferred from the primary circuit to the secondary one and from the secondary circuit to the tertiary one via heat exchanging surfaces. The secondary circuit is intermediate and intended for the heat transfer from the primary to the tertiary (grid) circuit. The secondary circuit is also used for the reactor cooldown (in normal and emergency modes). The secondary circuit consists of two autonomous loops. The tertiary circuit is the consumer circuit - it is a heating network for the nuclear heating plant, and for the nuclear desalination complex it is a system of steam generating loops for heating the desalination modules.



7.
8.
9.
10.

1- Leak monitoring device
2- Core
3- Ground
4- Concrete
5- Pool lining
6- Control rod drive
7- Primary heat-exchanger
8- Protective plate
9- Reactor lid
10- Headers and valves compartment

FIG. 1. Reactor RUTA-55.

The RUTA reactor facility is characterized by high reliability, maximum safety and ecological friendliness due to which the heat source can be brought closer to the consumer and transportation heat losses can be reduced. The relative pressures in the reactor facility circuits prevents radioactive products from entering the consumer circuit. The RUTA facility's enhanced safety is determined by the following basic intrinsic properties:

- no overpressure in the reactor tank;
- high heat-accumulating capacity of the water in the reactor tank;
- integral in-tank arrangement of the primary circuit;
- coolant natural circulation in the primary circuit;
- low nuclear fuel rating (14-18 kW/l);
- developed power self-regulation properties at the expense of negative reactivity feedbacks.

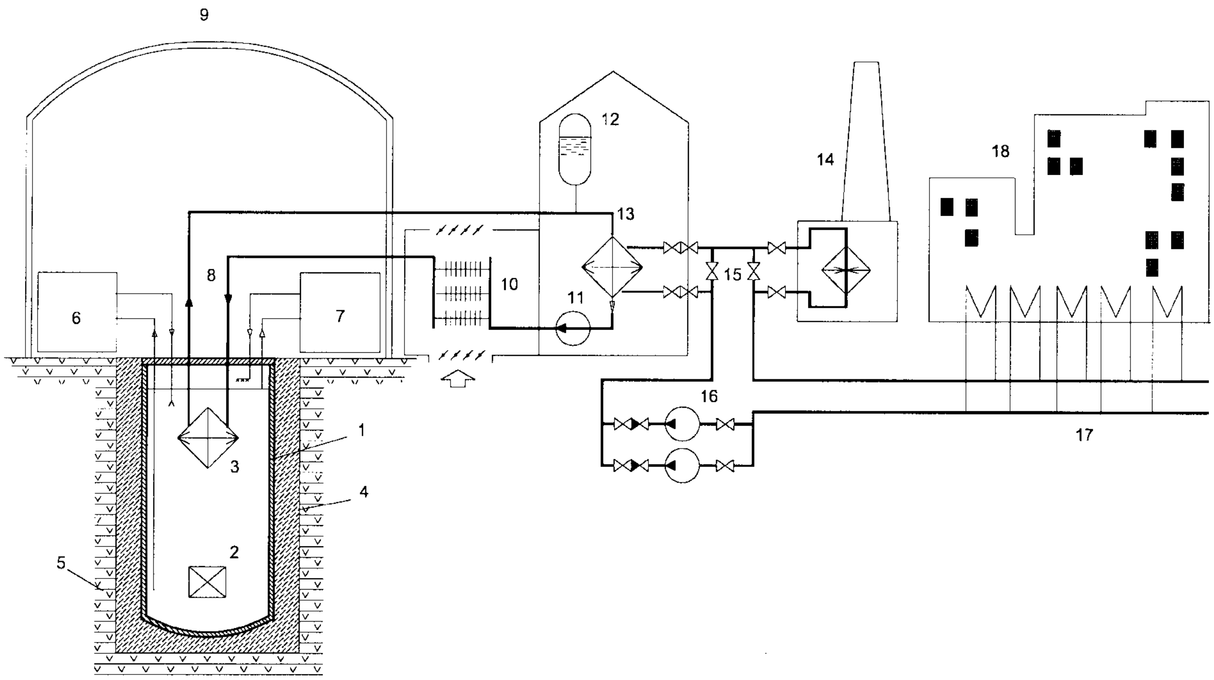
The combination of these factors makes it possible to exclude deterministically the probability of an accident with a melt or severe damage of the core in the RUTA reactor. A high level of the reactor self-protection at any design basis and beyond design basis failures of the reactor components and systems, the control system included, creates pre-conditions for setting up an all-automated reactor facility processes control system (with no operating personnel interference). The control concept [3] suggests the availability of the smallest full-time number of personnel (not more than 30) and short-term engagement of specialized teams for planned preventive maintenance and repair activities. All refueling and fuel handling operations (both with fresh and spent fuel) are done by special temporarily engaged personnel of the RF Ministry for Atomic Energy.

3. PECULIARITIES OF USING THE RUTA REACTOR FACILITY IN DISTRICT HEATING SYSTEMS

The peculiarity of the RUTA pool-type reactor is atmospheric pressure in the pool's air space. The inevitable, natural consequence of the overpressure absence in the reactor vessel is a low potential of the heat generated by the reactor. As a maximum, the temperature of the grid water supplied from the RUTA reactor facility may reach ~90 ^{o}C at a reasonable pool depth (of up to ~20 m) and acceptable values of the heat exchanging surfaces of the primary and grid heat exchangers. The direct grid water temperature is 85 ^{o}C for the considered RUTA-55 reactor with natural in-pool circulation as the return water temperature is 60 ^{o}C. With regard for the low grid water temperature values, the following approach to the arrangement of heat supply systems with RUTA reactors has been taken.

Most preferable is the joint use of a RUTA NHP and a peak non-nuclear heat source based on an fossil fuel application. Figure 2 presents a simplified diagram of the RUTA NHP and peak source joint use. The peak boiler and co-generation boilers are connected to the NHP grid circuit and used in the coldest period of the year.

Figure 3 shows a standard load regulation chart for a heating network with the parameters of 130/70 ^{o}C depending on the outdoor air temperature (weather conditions). The grid circuit water is heated in the RF intermediate heat exchangers up to 85 ^{o}C and further (if required) goes to the peak heaters where it is brought to the required temperature. In this case, the reactor, as it operates at the steady-state (maximum) power level, covers the base portion of the annual heat load chart. The peak portion of the load chart is covered at the expense of additional grid water heating in the peak boilers. For heating networks with more or less standard temperature regulation charts, the optimal NHP installed power is 30-50% of the maximum heating system load (Figure 4). If this approach is implemented, the share of the heat generated by the RUTA NHP will be 70-80% of the total annual heat consumption and the peak heat sources will account for 30-20% respectively. The costs of purchasing fossil fuels will decrease dramatically and the dependence on the probable irregularity (unreliability) of their supplies will be reduced.

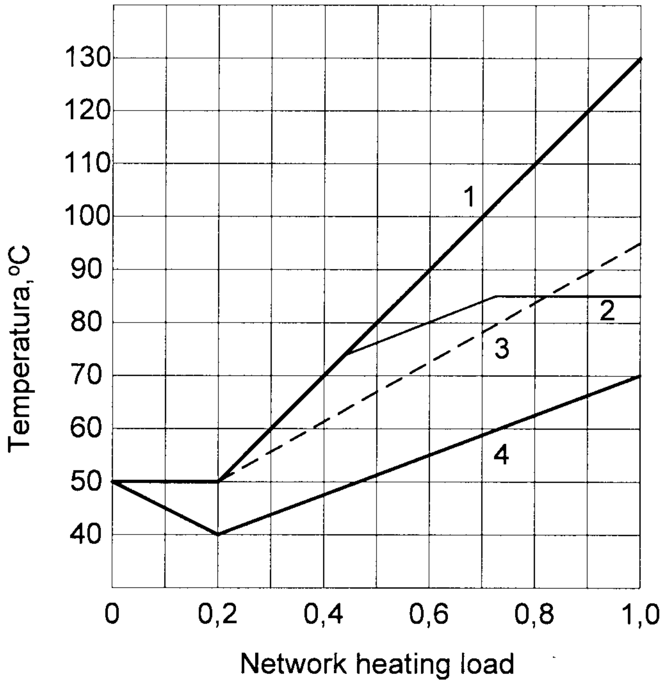


1- Reactor
2- Core
3- Primary heat-exchanger
4- Concrete
5- Ground
6- Pool water purification system
7- Gas ventilation purge system
8- Secondary circuit
9- Containment
10- Decay heat removal system
11- Secondary circuit pump
12- Pressuriser
13- Secondary heat exchanger
14- Reserved and peak boiler
15- Flow switching station
16- Heating network pumps
17- Heating network
18- Dwellings

FIG. 2. RUTA Nuclear Heating Plant.

The NHP may satisfy in full the heat demands at a small power and low loading parameters of the heating network with the use of fossil fuels being excluded. However, though a larger RF contribution to the total heat production ensures an additional fossil fuel economy, it simultaneously, as the annual thermal load duration chart shows (see Figure 4), causes a lower NHP capacity factor and may result in a lower economic performance of the heat supply system.

The results of special feasibility studies, including recent (2000) ones, have proved the economic efficiency of the RUTA NHP use in Russian regions with expensive fossil fuels and long heating seasons in the country's northern, north-west and eastern territories (including Urals, Altay, Amur Region and Khabarovsk Territory).



1 – direct water temperature in the network supply line;
2 – direct water temperature downstream of NHP heat exchanger;
3 – temperature at the heating devices inlet of a consumer;
4 – network return water temperature

FIG. 3. Temperature chart of qualitative regulation of heat loading in 130/70°C.

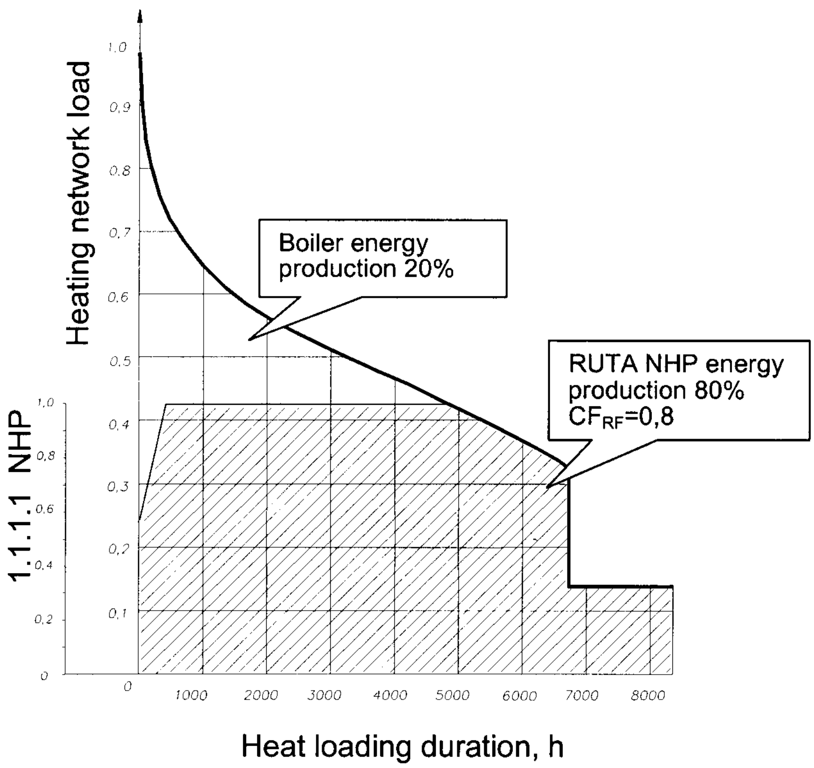


FIG. 4. Contribution of RUTA reactor facility to meeting the network heating load.

4. NUCLEAR DESALINATION COMPLEX WITH THE RUTA REACTOR

Apart from the use in heat supply systems, another potential application of the RUTA reactor – for heat supply of desalination facilities – looks promising. Desalination processes based on the use of thermal distillation (MED and MSF technologies) require a considerable thermal energy consumption at a relatively small electric power consumption. Proposals on using the RUTA-55 reactor facility as a heat source for distillation facilities have been developed within the framework of the IAEA Coordinated Research Program on Nuclear Desalination in 1999-2000.

Seawater in the RUTA nuclear desalination complex (NDC) is desalinated in multi-effect distillation facilities with horizontal-tube film units (MED technology) developed and manufactured in Russia. These are modern highly efficient distillers meeting the international development standard of desalination technologies. A low temperature potential of the heat generated by the reactor has caused the necessity of special technical decisions to be taken to ensure as high distillate production capacity of the complex as possible.

The interface of the reactor and desalination facilities (Figure 5) includes steam generation equipment. The steam generation circuit is designed as a loop of the reactor facility tertiary circuit. The number of such loops connected in parallel to the common header depends on the number of desalination modules in the NDC. Each MED-unit is equipped with a pre-connected multi-stage self-evaporator in which the tertiary circuit water evaporates in part to be supplied for the heating of the MED-unit head effects. The tertiary circuit water losses are made up for with distillate.

The daily production capacity of the desalination complex including one RUTA-55 reactor with a thermal power of 55 MW and four desalination modules, is 20000 m^3/day. Such an amount of water is enough for the water supply of a small city with a population of around 100,000 with regard for the infrastructure and local technologies.

The optimal configuration of the complex includes a 2- or 3-unit power source (depending on demand) and a desalination system with a 5-10% margin in terms of the installed capacity. Economic estimates show that NDCs with the RUTA-55 reactor are capable of competing similar desalination facilities based on fossil fuels use in regions with expensive imported fuels.

The cost of the distillate for the RUTA NDC may be reduced significantly as the result of optimizing the reactor parameters. The arrangement in the in-pile circulation circuit (primary circuit) of an additional driving head using pumps allows the reactor power to be increased by 30-35% in the facility and raise the supplied heat potential in the same reactor dimensions and without radical changes to the design. An increase of the coolant temperature in the heat consumer circuit from 85/60^{o}C to 90/70^{o}C provides for an additional increase in the capacity of desalination facilities by 12-15%. As the result, the NDC total distillate production capacity increase of 45-50% is achieved.

To evaluate the economic effect of the proposed decisions, the cost of the distillate for NDCs with the optimized RUTA reactor with a thermal power of 70 MW and the respective set of desalination modules have been calculated. The economic calculations have shown that the distillate cost is reduced by 12-14%. Besides, capital investments may be reduced significantly due to the NDC optimization as the following example shows.

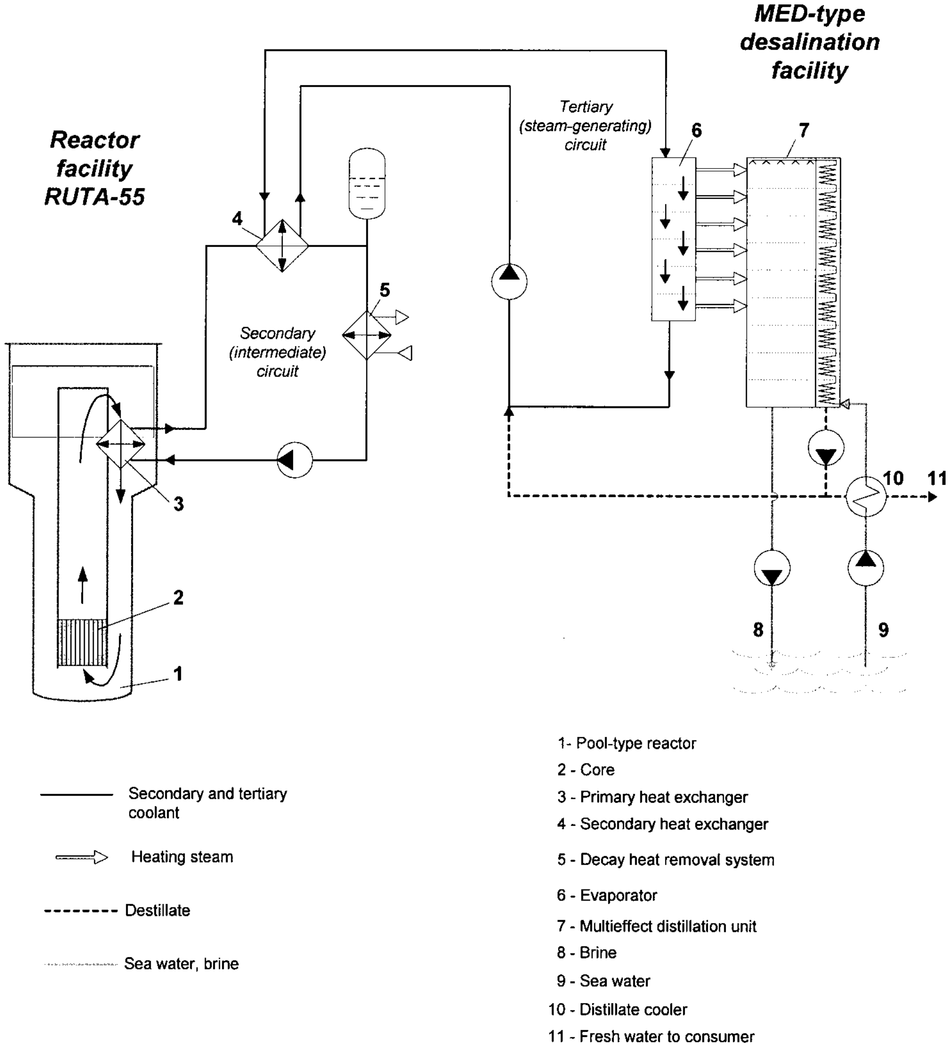


FIG. 5. RUTA Nuclear Desalination Complex.

Let's assume that the considered region's fresh water demand is 60000 m^3/day. Such a distillate amount may be produced by an NDC with three RUTA-55 reactors (not optimized) or with two optimized RUTA-70 reactors. The capital costs for the construction of an energy source with two optimized reactors are some 30% lower than for the variant with two RUTA-55 reactors. Total capital investments in NDCs are reduced by more than 20%. Therefore, the above estimates confirm the economic efficiency of the proposed decisions.

5. OPTIMIZATION OF THE RUTA REACTOR PARAMETERS

The economic performance of the RUTA facility may be improved as the results of:

- increasing the RF unit power;
- improving the consumer properties through raising the heat temperature potential;
- reducing the nuclear fuel costs.

One of the possible ways to solve this task is connected with the application of natural circulation in the primary circuit or, to be more exact, with the arrangement of an additional effective head in the reactor's natural circulation circuit using pumps.

The idea of boosting the reactor power using pumps does not run counter to the concept of a higher safety and reliability of the RUTA reactors as during the reactor operation in a broad power range from the minimum level to 70%N_{nom}, and during the reactor cooldown, the core is cooled in the natural circulation mode which ensures a high safety. The power increase from 70%N_{nom} to 100%N_{nom} is attained at the arrangement of an additional head by circulation pumps. The primary circuit uses axial pumps installed in the in-pile natural circulation bypass downstream of the primary heat exchanger outlet. The coolant is delivered to the circulation circuit's downcomer section inlet from the pump pressure side. The circulation modes (natural/forced) are switched over in the primary circuit at the actuation of the pumps automatically by means of the check valve passive closing/opening.

The grid water temperature for the optimized variant is 90/70°C. The return water temperature equal to 70°C matches the conditions of the reactor operation as part of the NHP as it matches the temperature assumed in the standard regulation charts of low-temperature heating systems. If a reactor is designed specially for NDCs, it may turn out reasonable to raise the temperature of the RF circuit's cold legs still higher (up to 75-80°C in the return line from the consumer).

The use of forced circulation in the primary circuit enables to accept a more compact variant of the core for the RUTA reactor by making it higher and its diameter smaller (see Table 1) which ensures optimization of its fuel performance.

6. USE OF THE RUTA REACTOR AS A NEUTRON SOURCE

In parallel with the use of the RUTA reactor for its intended application as a heat source in district heating systems and seawater desalination, its functions may be expanded. The reactor design opens up rather broad opportunities for it being used as a neutron source [4] for generation of beams for neutron therapy (a flux of fast or epithermal neutrons on the patient seat of $\sim 10^9$ n/cm$^2\cdot$s).

The realization of these capabilities makes the project of an NDC with the RUTA reactor more commercially and socially attractive.

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A PROPOSAL FOR REUSABLE TYPE SMALL PWR FOR SEAWATER DESALINATION

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Abstract

Demand for energy and water has been increasing, especially in developing countries where populations are growing at a high annual rate. If such demand is met by fossil-fuel energy production, the impact on the environment, such as global warming, cannot be disregarded. Accordingly, we have devised a reusable type small PWR exclusively for seawater desalination with mid-range pressure and temperature specifications. A distinct feature of this small reactor is that spent fuel is removed together with the reactor vessel and refueling is carried out at an exclusive fuel exchange base that is independently installed in a different place. This feature is aimed at reducing construction cost and increasing safeguards against nuclear proliferation, as the reactor vessel lid is not opened at the plant installation site. Also, since there has been a worldwide trend to employ reverse osmosis method of seawater desalination system, we studied the reverse osmosis method directly using low-pressure steam generated from a small PWR.

1. INTRODUCTION

Demand for energy and water has been increasing year by year, especially in developing countries where populations are growing at a high annual rate. If such demand is met by fossil-fuel energy production, the impact on the environment, such as global warming, cannot be disregarded. On the other hand, since developing countries are behind in their preparedness of social capital infrastructure such as power transmission grids, etc. decentralized small reactors are considered to be more suitable for introduction than the large reactors that are common in developed countries. Therefore, we have devised a reusable type small PWR for exclusive use in seawater desalination that provides mid-range pressure and temperature service, but does not require on-site refueling, thus taking into consideration proliferation safeguards.

2. FEATURES OF REUSABLE TYPE SMALL PWR

2.1. Core design

For the core concept of a reusable type small PWR, the design of the marine reactor of Japan's experimental nuclear ship "Mutsu" (thermal power: 36MWt) was modified for land use [1], and a mid-range temperature and pressure service reactor for seawater desalination was configured based on this land use reactor.

For the fuel assembly, the length of the 17×17 type fuel assembly used in current PWRs was shortened to suit the core thermal power, in order to save the cost for development of new fuel rods and to limit the increase in fabrication cost. The shape of the core was determined so that the ratio between the core equivalent diameter and the core effective length might be approximately 1. As a result, the core equivalent diameter became 1.11m and the core effective length 1m, with 21 fuel assemblies containing gadolinium. In order to prolong the service life of the core as much as possible, the fuel enrichment was determined to be within 5wt% of the limiting value by improving the operating rate. As a result, the core lifetime became approximately 5.8EFPY, and the burn-up approximately 28GWd/t when the fuel is taken out.

As for the core reactivity control method, the control of reactivity in a current PWR is carried out by control rod and boron concentration adjustment. In this case, a method to control it only by control rod adjustment was selected, because stability can be maintained by increasing the control rod worth. The specifications and cross-section of the core are shown in Table 1 and Fig. 1.

TABLE I: SPECIFICATIONS OF THE CORE

Items	Unit	Numerical value	Remarks
Core configuration			
Core equivalent diameter	m	1.11	
Core effective length	m	1.0	
Fuel assembly			
Number of assemblies	-	21	
Total number of fuel rods	-	5,333	
Fuel rod			
Fuel Assembly Type	-	17 x 17	most popular in the world
Fuel pitch	cm	1.26	the same as PWR 17 x 17 fuel
Fuel cladding outer diameter	cm	0.95	ditto
Fuel cladding inner diameter	cm	0.84	ditto
Fuel cladding material	-	Zircaloy-4	ditto
Pellet diameter	cm	0.82	ditto
U-235 enrichment	Wt%	below 5.0	
Control rod			
Absorber material	-	B_4C	B^{10} of 90wt% concentration
Control rod outer diameter	cm	2.0	
Number of control rods	Rods	64	
Core characteristics			
Thermal power	MWt	36	
Average linear power	kW/m	6.1	
Core power density	W/cc	36	
Fuel charge amount	t	2.74	
Average burn-up rate	GWd/t	Approx. 28	
Core service life	EFPY	Approx. 5.8	

Remarks: Water-soluble boron (Chemical shim) is not used for reactivity control.

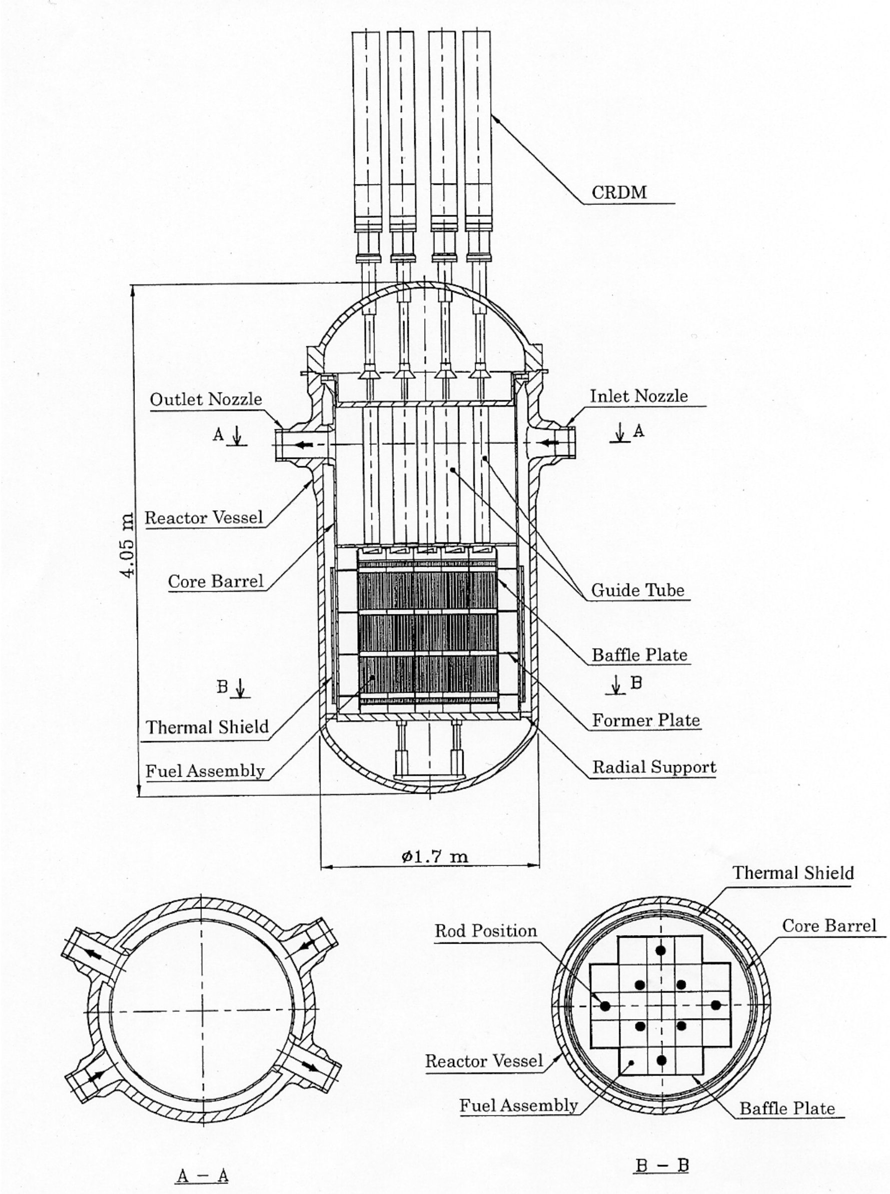

FIG. 1: Cross Section of Reactor Vessel.

2.2. Reusable type reactor vessel

To transport the reusable type small PWR to the installation site together with its reactor vessel, the reactor vessel can be separated from the reactor coolant loop during refueling and must be configured so that it is replaceable. In addition, the size and weight must be reduced as much as possible. In view of the above, the reactor vessel became 8t, the core internals 12t, the reactor coolant 6t, and thus the total weight 26t.

In order to separate the reactor vessel, it was decided that the reactor vessel and the reactor coolant pipe were to be flange-connected with spool piece. (As an alternative, the vessel can be also isolated with isolation valves.)

The concept envisions repeated use of the reactor vessel up to its service life, by going and returning between the plant site and the fuel exchange base. Nevertheless it takes a long cooling period for decay heat removal by natural heat dissipation after reactor shutdown, but it becomes possible to reduce this period by adopting transfer means with a forced cooling device.

2.3. System design

The system design of the reusable type small PWR was aimed at reducing to a minimum the element to be newly developed and fully utilizing current technologies that had already been proven. The features of the system are as follows:

1. Simplification/realization of small capacity of equipment

   By adopting the following methods, the simplification and realization of small capacity of equipment were aimed at:

   - The operation conditions of reactor coolant system have been changed to mid-range temperature and pressure service, that is, from 320°C and 16.0MPa to 240°C and 4.4MPa.
   - Adoption of canned motor type reactor coolant pump
   - Elimination of boron regeneration and treatment system and anion demineralizer accompanied with not using water-soluble boron (chemical shim) for reactivity control.
   - Liquid waste is only stored on-site, and treated in a batch off-site after accumulation in the tank.
   - Component cooling water system and service water systems are divided into safety system and normal operating system.
   - Basic configuration of other safety systems is the same as that of a current PWR.

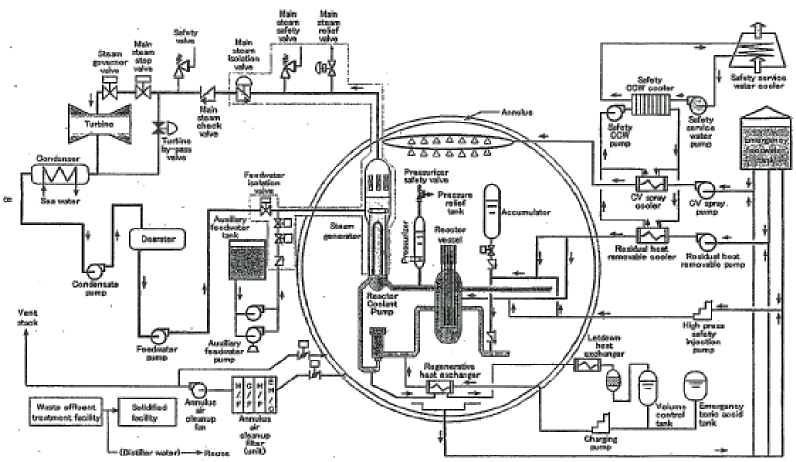

FIG. 2: Reusable Type Small PWR – Conceptual Diagram of Reactor System.

Installation of intermediate heat exchanger (option)

Steam generated in the steam generator is directly introduced to the turbine-driven high-pressure pump for the seawater desalination plant. If this reactor is used as a district heating reactor, it can be applied without changing the design parameters of the reactor. By installing an intermediate loop between the reactor coolant system (primary system) and the heat utilization system, radioactive materials are prevented from mixing into the utilization system.

The system concept of the reactor system is shown in Fig. 2.

2.4. Layout

Two reactor vessels are located in a containment vessel so as to allow the continuous operation of the seawater desalination system without being stopped, even when one reactor vessel is transferred to the outside. (Refueling outage one of two reactors can be performed at any time.)

In the reactor building, there are arranged the reactor containment vessel in the center where primary cooling system is installed, as well as reactor vessel, etc. the safety systems, and volume control system, etc. relating to them. In the auxiliary building are mainly arranged the electrical and instrumentation equipment such as central control room, electrical equipment, instrumentation equipment, etc. Both the reactor building and the auxiliary building are of anti-seismic Class A.

Fig. 3 shows the buildings conceptual layouts.

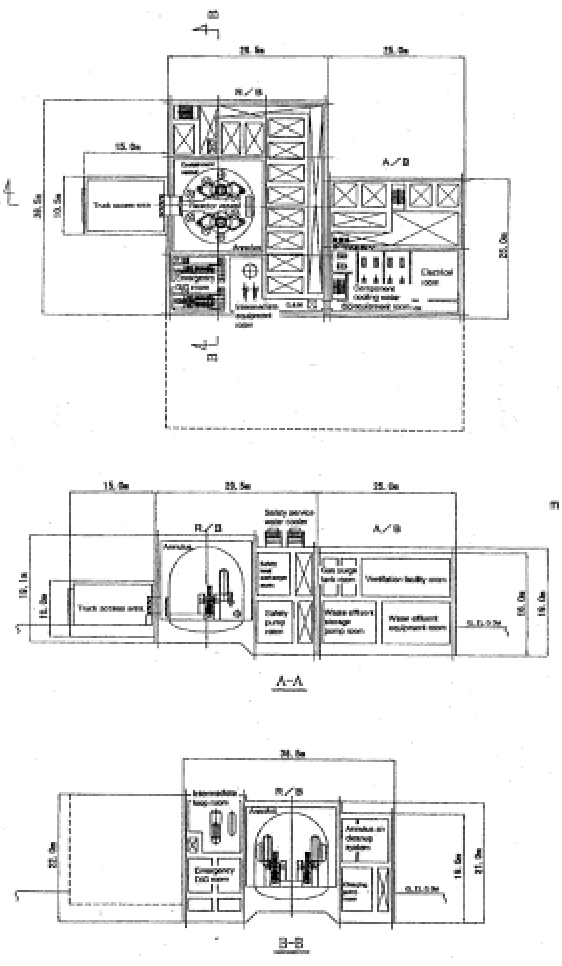

FIG. 3: Reusable Type Small PWR – Plant Layout.

2.5. Refueling

The fuel and the reactor vessel of the reusable type small PWR are transferred to the exclusive fuel exchange base, being laid down horizontally within a transfer cask after a long cycle operation (approximately 6 years). After refueling and periodic inspections are carried out, they will be returned to the site for a long period operation. Their reuse is enabled with the so-called cassette method. The concept of transfer cask is shown in Fig. 4.

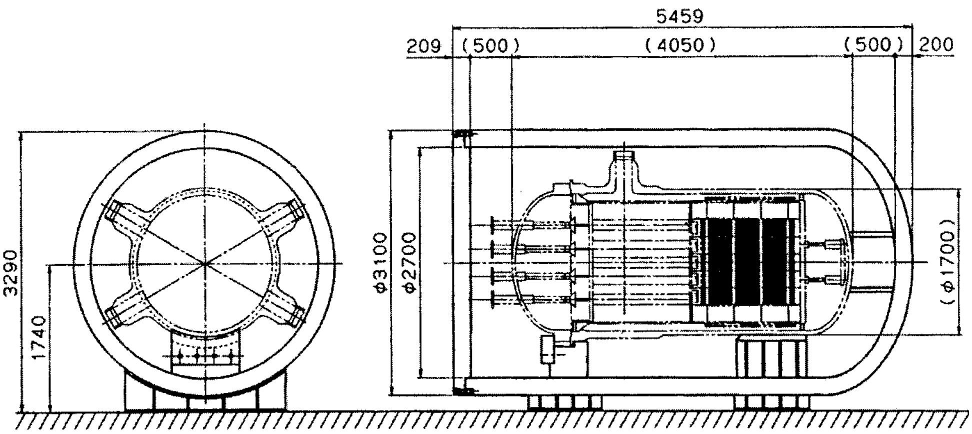

FIG. 4: Conceptual Diagram of Transfer Cask.

Centralization of fuel handling facilities in the fuel exchange base contributes to reduction of plant construction costs, as fuel handling facilities become unnecessary at the plant installation site. Furthermore, the reactor vessel is re-used until its service life expires.

The diagram of the removal procedure is shown in Fig. 5. After the reactor vessel is separated from the reactor coolant pipes, it is horizontally transferred by a crane, laid down horizontally on a support frame and fixed to the frame. Next, the reactor vessel is taken out to the truck access area outside of the containment vessel by means of an air carrier, and is contained along with the support frame, in the transfer cask carried on a trailer.

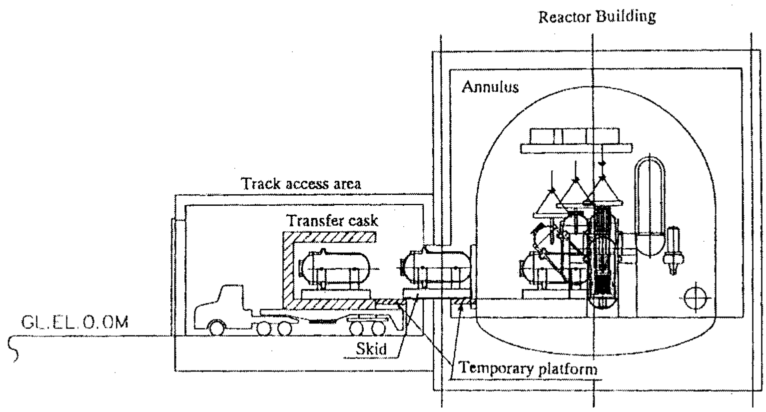

Fig. 5: Reusable Type Small PWR – Diagram of Procedure to take out Reactor Vessel.

3. COUPLING SMALL REACTOR WITH SEAWATER DESALINATION PLANT

A seawater desalination system using a reverse osmosis membrane (RO) is more economical than that using the distillation process (MED or MSF), and there is a worldwide trend toward adoption of RO. [2]

For the sake of direct use of the heat generated from the small PWR without the heat being converted into electricity, a method has been selected where the high-pressure pump is directly driven with the steam. [3] Because the steam turbine driven pump is used as auxiliary feed water pump of PWR and almost the same specifications can be used, there are few development elements. Fresh water can be produced when seawater is pressurized by the steam turbine driven high-pressure pump and passes through the reverse osmosis membrane (RO). Since speed reduction gears are not used in between, fresh water can be produced efficiently. The concentrated seawater (brine) that does not pass through the reverse osmosis membrane also can be used as an on-site electric power source for pumping power by extracting it as electricity a power recovery turbine. Figure 6 shows the conceptual diagram of the nuclear seawater desalination system where the reusable type small PWRs are coupled with the RO system using a turbine driven high-pressure pump. From this system, fresh water amounting to approximately 40,000m^3/day per reactor can be produced. The heating reactor of mid-range temperature and pressure service can provide almost the same performance as that of a power reactor of high temperature and pressure service, considering that water amounting to approximately 4,000m^3/day per 1MWe is produced by the RO method using an electric motor driven high-pressure pump.

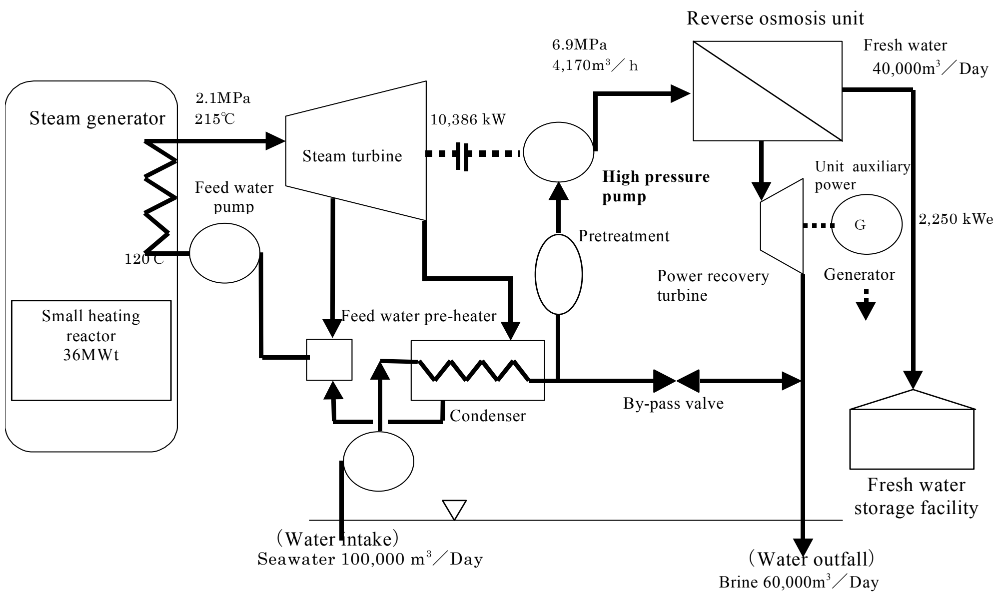

FIG. 6: Coupling Scheme of SMR with Reverse Osmosis Membrane System (RO)

One disadvantage of the RO method is that it is not suitable for intermittent operation, because backwash is required, since the seawater inside the system corrodes when the operation is shut down for periodic inspection, etc. In order to maintain the operation as continuously as possible, a facility to store excess fresh water is required during periods of low demand, and a huge space for storage is necessary. To solve this problem, one idea is that a water storage bag installed in sea is available utilizing the difference in specific gravities between seawater and fresh water, as shown in Fig. 7.

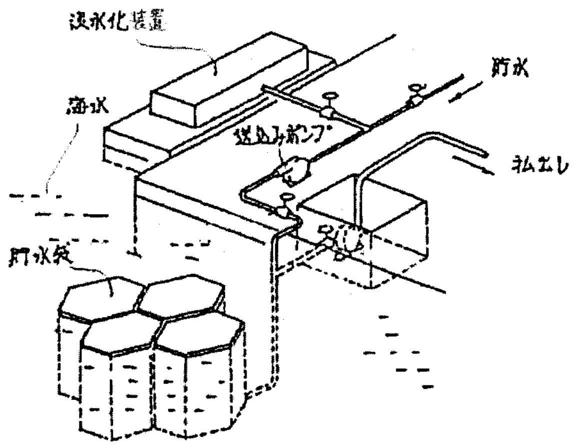

FIG. 7: Water Storage Facility

On the other hand, it is supposed that some developing countries have already mastered the distillation process (MED or MSF) using fossil-fuel energy and that there is also a need to continuously use the distillation process, whose operation is easier. A system to connect the distillation process is also taken into consideration as an option. When the distillation process is adopted, it is possible to use the heat from the steam generator via a steam drum for evaporation. By making the system, including the steam drum, an intermediate loop between the small PWR and the seawater desalination system, contamination by radioactive materials in the heat utilization system can be prevented. Figure 8 shows, as an example, a conceptual diagram of a nuclear seawater desalination system coupled to the reusable type small PWR and the multi-effect distillation process (MED).

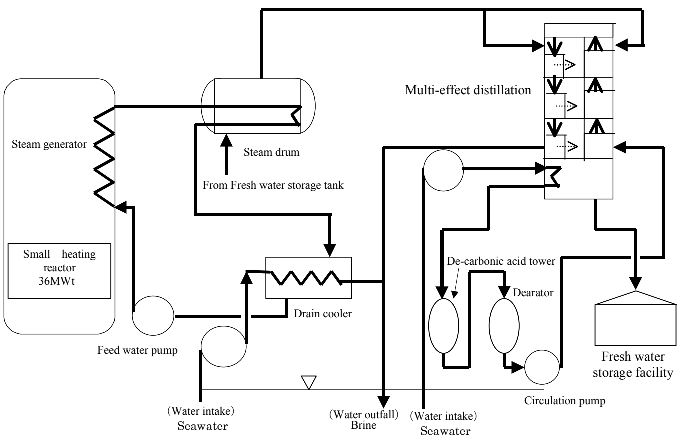

Fig. 8: Coupling Scheme of SMR with Distillation Process (MED)

4. ISSUES TO BE CONSIDERED IN THE FUTURE

4.1. Licensing

In order to improve the facility efficiency of the reusable type small PWR, it is necessary to standardize the reactor vessel . and prepare a qualification regime to transport reactor vessel together with new and spent fuels in addition to enable free unrestrained shuttle transportation, between not only the specific site and the fuel exchange base but also the fuel exchange base and another site.

4.2. Transport of reactor vessel

There is no experience in removing and re-installing the fuel assembly together with the reactor vessel. The outline of the procedure for removing or re-installing has been studied. Since the control rod driving mechanism must be separated and laid down when the reactor vessel is transported, sufficient review and verification tests are necessary to ensure structural integrity during separation, re-attachment and operation when the mechanism is operated again after transportation with the rod laid down.

4.3. Economy

We started our study based on the core of "Mutsu" of 36MWt. From the viewpoint of the enhanced economy through merits of scale, we need to investigate how much the thermal power can be enlarged with the concept of transferring the reactor unchanged. However, the investigation must also consider the assumed demand for fresh water, as the standardization of thermal power is required.

4.4. Others

Since the fuel exchange base for this reactor must be located off-site, the installation site must be carefully selected and where issues clarified the business proprietor. Also, waste treatment and disposal facilities are necessary at off-site.

CONCLUSIONS

We have proposed a reusable type small PWR with mid-range temperature and pressure specifications for exclusive use of seawater desalination. For this reactor, we aimed at enhancement of the economy and the nuclear proliferation safeguards by removing the spent fuel together with the reactor vessel, and adopting a concept of refueling at an exclusive fuel exchange base installed independently. Thus we have demonstrated the prospect of feasibility. The reverse osmosis method was studied for the seawater desalination equipment by mainly using low-pressure steam generated from this reactor. It is also technically possible to combine this reactor with the distillation process (MED or MSF). It is supposed that some developing countries have already mastered the distillation process using fossil-fuel energy, and there also exists a need to continuously use the distillation process, whose operation is easier.

Although the prospect of feasibility of the reactor seawater desalination equipment has been shown, our research will continue in the future, as various detailed studies are required, such as a method to remove the reactor vessel, etc. in the event of the actual application of this proposal.

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JOINT RUSSIAN–CANADIAN PROJECT FOR A FLOATING NUCLEAR DESALINATION COMPLEX

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Abstract

The Russian Federation has established the requirement for floating nuclear power plants to supply electrical power and process steam to areas on the Russian northern and eastern littoral. Recognizing that there also existed a substantial global need for increased potable water production, engineers from OKBM approached CANDESAL Technologies in the mid 1990's to investigate the potential for a joint project to meet these identified and established requirements. The paper describes some of the customer oriented aspects as they could apply at the national, regional, municipal or private sector level in the customer country, and site specific needs relative to the system and system product, and goes on to describe the concept of operation of the Floating Nuclear Desalination System (FNDS) and the proposed interaction and partnership with the customer for the reception and distribution of the electrical power, potable water and, potentially, process steam. The technical aspects of the Floating Power Unit (FPU), and the Floating Desalination Unit (FDU) are described which cover not solely technical issues, but also the licensing and safety issues including the employment of preheat in the system and system optimization for seasonal, site specific conditions and considerations.

1. INTRODUCTION

1.1. Small Nuclear Power Plants

In 1994 the Nuclear Society of the Russian Federation organized a competition of small nuclear power stations. Reactors falling into three power level categories were considered: below 10 MW_t, between 10 and 50 MW_t, and above 50 MW_t. More than 20 different projects were submitted to the competition.

The KLT-40 project was the winner in the third category. This reactor has been designed, built and operated for nuclear icebreakers, and has been endorsed by Russian State and international experts. The recommended project was to design a Floating Energy Block (FPU) based on the KLT-40 design, for application with particular customers and locations where appropriate.

Market research for a small size nuclear heat power plant (SS NHPP) took place at the same time as the competition. Feasibility study reports were carried out for several parts of the country, including the Chukotsk region, the Far East and Habarovsk areas, and the distant North and Northeast territories. More than 250 locations were considered, especially in northern Russia, and after careful analysis 26 locations were selected as potential sites for SS NHPP.

Preliminary analysis of the efficiency of these small atomic power thermo-electric stations has shown that they are competitive with respect to more traditional energy sources in distant regions of the Russian Federation, even if these regions have conventional energy resources.

1.2. Advanced Reverse Osmosis Desalination Systems

The focus of CANDESAL's early design concept development work was placed first on the determination of an appropriate seawater desalination technology for coupling to a nuclear reactor. With no prior commitment to any particular technology, all options were open for consideration. The only prerequisite was that commercially available, well-proven desalination technologies be considered. Upon investigation, it was determined that the reverse osmosis method of desalination was the most efficient and cost-effective way of creating potable water when coupled with an electrical generating system.

As CANDESAL did more investigation into the RO system to be coupled with a nuclear reactor, a whole new approach to reverse osmosis desalination was developed in the process. That advanced technology is being applied in the joint Russian Federation-Canada project for a floating nuclear desalination complex. The special features of the CANDESAL advanced RO seawater desalination system, and in particular it's coupling to the KLT-40C reactor, are described below. Briefly, however, these include the use of waste heat from the energy generating system to improve the efficiency of the RO process, the use of ultrafiltration as an advanced RO system feedwater pretreatment, advanced energy recovery technology, and non-traditional, highly optimized design and operating parameters.

The result of the CANDESAL approach to RO system design is expected to result in reductions in the cost of potable water production on the order of 15% relative to systems designed and operated in a more traditional manner.

1.3. Russian Federation-Canada Joint Project Background

It became clear while under development that the KLT-40 reactor has applications outside of the Russian Federation and could be used as a source of electricity and energy for desalination systems in regions with shortages of potable water. The engineers of OKBM and JSC "Malaya Energetica" (Russian Federation) and CANDESAL Technologies (Canada) have joined efforts to develop a floating energy block that will be coupled with a floating desalination system. After appropriate discussion of the issues surrounding a nuclear desalination project, a Memorandum of Understanding between MINATOM of the Russian Federation and the Canadian company CANDESAL was signed in 1995 and extended in 1998. A formal agreement to build a floating nuclear desalination system was reached in June 2000.

2. DESCRIPTION OF THE RUSSIAN SS NHPP PROJECT

2.1. The Small Size Nuclear Heat Power Plant

Russian specialists have completed the project design concept for a Small Size Nuclear Heat Power Plant (SS NHPP) to provide distant regions of the country with heat and electrical energy. The station consists of a floating power unit (FPU), a berthing facility and a coastal infrastructure.

The berthing facility provides the fixtures and mounting structures for mooring the FPU at a coastal site. The berth construction provides for transport of heat and electricity to the coastal infrastructure. The design also accommodates direct mooring of support and provisioning ships to the FPU.

The coastal infrastructure consists of special equipment, buildings and facilities designed for the transport of electric energy and heat to customers. The main hull and superstructure are constructed from steel with an enhanced resistance to low temperatures.

2.2. The Floating Energy Block

The floating energy block is built on a flat deck barge with a rectangular hull outline and multi-deck superstructure. The basic dimensions of the FPU are: length – 140 m; width – 30m; keel draught – 5.6 m; and a displacement tonnage – 21,000 t.

The FPU is designed to work within the framework of the SS NHPP. This block is a defining component of the SS NHPP, which generates electricity and heat and supplies them to customers through the coastal infrastructure. The all-welded hull of the FPU is ice resistant and has special equipment for towing and mounting. The submerged part of the hull is protected from corrosion by electro-chemical coating and an ice-resistant varnish-and-paint coating.

The reactor compartment and spent fuel bay are protected from external events (collisions, shoal landing) by an anti-impact design. The FPU construction contains storage for spent fuel elements and liquid and solid radioactive wastes. It includes a reactor refueling complex, but operates without the use of special refueling technology between regularly scheduled maintenance. The KLT-40 reactor is designed with safety systems (such as a protective vessel, shielding, and autonomous ventilation system), which completely prevent any radiation release to the environment.

The FPU contains two main functional modules: a living module, located in the stern section of the barge, and an energy module, located in the central and forward sections. FPU operation in general is provided by regular vessel systems. Both functional modules are supported by an automatic control system based on contemporary microprocessor technology.

Typical FPU Service Life

1.	Design life of FPU (years) The service life of the main equipment is either equal to this life or corresponds to the period between regularly scheduled maintenance.	35-40
2.	Design interval between plant maintenance outages (years)	10-12
3.	Expected period of continuous operation (hours)	8000

The energy module is designed to generate electricity and heat energy. This module contains two reactor installations, two steam turbines, and an electric energy generating system. A block principle (one reactor, one turbine, one generator) is used in the equipment configuration.

2.3. KLT-40C Reactor Description

The reactor installation for the FPU is uses the KLT-40C reactor, which is based on icebreaker reactor technology that has been proven by over 40 years of successful, accident free operation.

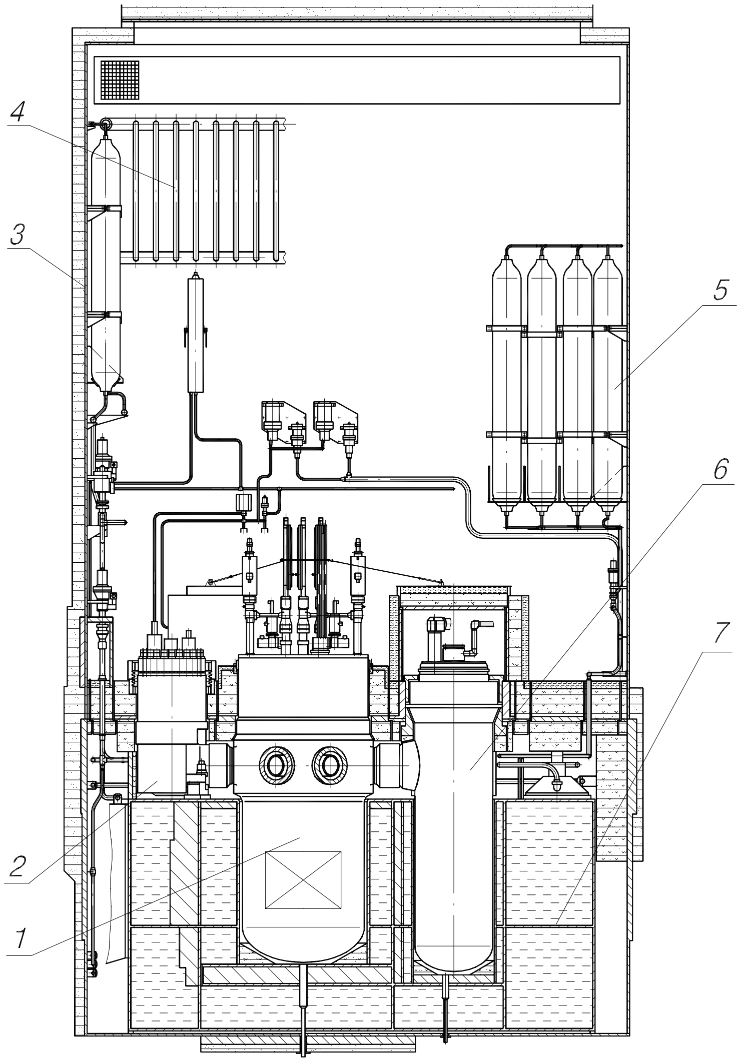

FIG. 1 – KLT-40C Reactor.

The KLT-40C reactor installation contains the reactor, four steam generators, and four pumps within the primary circuit. These are joined by all joined by steam lines into the steam generating block. This block is located within a metal-water shield tank, which is surrounded by a protective containment barrier (Figure 1).

Technical improvements have been implemented in the KLT-40C reactor installation relative to the early prototypes in order to further increase system safety. These include, in particular:

- A two-channel passive residual heat dissipation system**,** which during accidents with full depressurization of the FPU provides a safe stability operating condition for twenty four hours;
- Pressure reduction in the containment, that would operate under any extreme design basis accident, with bubbling and condensing passive subsystems;
- An active emergency core cooling system for loss of coolant accidents from the primary circuit: this system contains two loops, each with a water tank and two pumps, and keeps the core in a safe condition in accordance with single failure criteria. Each loop also

contains a passive subsystem. Facilities for condensed steam collection and return to the reactor are also included in order to decrease the water consumption and consequently decrease the release of radioactive wastes;

- A passive system of external cooling of the reactor core is supplied for protection of the reactor core from melting under "beyond design basis" accidents leading to fuel dry-out and severe core damage;
- Self-activated devices are included as protective measures for "beyond design basis" accidents, associated with failure of safety systems in the reactor safety and protection system.

1 – reactor 2 – primary circulating pump 3 – protective shell 4 – condensation system of reducing pressure in the containment 5 – high pressure gases cylinders 6 – steam generator
7 – metal-water shielding tank

General Technical Characteristics Of The Reactor Installation

Thermal power, MWt	150
Steam production, t/h	240
Primary circuit pressure, MPa	12.7
Steam pressure behind PG, MPa	3.8
Temperature of superheated steam, C	290
Temperature of feedwater	170
Design limits of power variation, % N nominal	10-100
Refueling interval, years	2.5-3

2.4. Energy Generating Systems Description

Steam Turbine Installation

There are two TK-35/38-3.4 type steam turbines in the FPU. The turbines are designed for production of heat and for process steam for the generator as a source of electric energy. Fresh steam consumption by the turbine is equal to 220 t/hour at 285 C. There are three steam outputs from the turbine. The first and the third outputs are not controlled and they are designed for feed water heating. The second output is a controlled one, and steam from this output is used for feed water heating and heating of water in the intermediate circuit. The range of regulation of heat energy supply for the intermediate circuit is equal to 0-100% if the load on the generator is not less than 30% of nominal. This latter restriction is associated with cooling of last turbine blades. Independent control of heat and electric energy is provided under the load range of 30-100% of nominal. The turbine may also be used with intermediate circuit heaters out of service.

Another turbine heat extraction scheme provides additional heat output through intermediate circuit heaters from steam taken from the system prior to the turbine. Electric power generation decreases under this process. This application is necessary during winter periods to compensate for peak heat loads. Heat supply from the turbine occurs through pressurized water in the intermediate circuit that provides an additional barrier to release of radiation materials.

General Technical Characteristics Of The TK-35/38-3.4 Turbine

1	Nominal electric power, MW	35
2	Nominal heat power, GCal/hour	25
3	Maximum electrical power (without heat power generation), MW	38.5
4	Steam pressure under control bleeding, MPa	0.357
5	Steam temperature under control bleeding, °C	139
6	Nominal water heating in the capacitor, °C	13.4
7	Method of heat power transfer from FPU	Industrial circuit
8	Heat carrier of the intermediate circuit	water
9	Water pressure in the intermediate circuit, MPa	~1.6
10	Water consumption in the intermediate circuit, cub. m/hour	420
11	Nominal temperature of water in the intermediate circuit, output/input °C	130/70

* Characteristics for design temperature 10 °C of cooling water

The period of FPU autonomy with respect to oil reserves for stem generator installations is equal to one year

Electric Energy Supply System

The system of heat production and electric energy supply to the energy system

Electrical energy production is carried out by two TAG 8123 synchronous generators with a nominal power of 35 MW each (voltage – 10.5 kV, frequency - 50 Hz), coupled with steam turbines and with indirect air-cooling system. There are two main distribution devices for electrical energy from the generators; energy panel outlets and electric power distribution less than 10.5 kV. The energy panel outlets are designed for energy transmission via flexible hose-type cable from the FPU to coastal receiving devices.

The system of electric energy supply to meet the needs of the FPU

The system is designed for electrical energy supply to customers under normal operating conditions for the FPU, and for transfer or maintenance of reactor installations into a safe state during normal and accident regimes. This system contains subsystems of normal and accident electric supply.

The normal supply subsystem contains eight transformers, four main distribution panels, four 230 volt panels, a reserve diesel-generator with its four panels, and eight 400/230 volt reduction transformers. Electric power is supplied from the turbine generator in the normal operating regime or from the reserve diesel generators, 800 kW each, when the main electrical energy supply is not functional.

The emergency energy supply subsystem serves for energy supply for the reactor installation safety systems if the main and reserve generators fail under an accident. This subsystem contains four diesel generators, each producing 200 MWe, panels for reactor systems, and 400/230 volt reduction transformers.

The period of FPU autonomy with respect to diesel fuel reserves for accident and reserve generators corresponds to 30 days.

3. DESCRIPTION OF THE CANDESAL DESALINATION SYSTEM

3.1. The CANDESAL Design Approach

The CANDESAL Reverse Osmosis Desalination System incorporates several key features in its design approach. One of these is the use of the condenser cooling system discharge stream, carrying away waste heat from the energy generating system, as preheated feedwater to the RO system. CANDESAL is the originator of the concept of preheated feedwater, and has carried out extensive studies of the performance improvements available through its use. The use of KLT-40C reactor plant condenser cooling water as preheated feed stream for the desalination plant allows for substantial gains in fresh water production efficiency, resulting in reduced plant capital cost as well as reduced energy consumption per unit of water produced. Additional benefits can also be obtained through use of the process heat generated by the FPU to generate additional feedwater preheat, when the process heat is not required for district heating.

Ultrafiltration (UF) pre-treatment is used to provide high quality feedwater 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 a reduced requirement for membrane maintenance and replacement.

Sophisticated analysis techniques drawn from reactor design experience are used in the CANDESAL desalination/cogeneration system design. Drawing on the combined expertise of desalination system and nuclear power plant designers, the design is numerically modeled to allow design optimization and integrated system performance analyses. This comprehensive design optimization allows further performance enhancements and reduced water production costs, which are site specific and optimize the inherent advantages of the site, which vary depending on geographic location and quality of available water.

Maximum use is made of energy recovery techniques. Much of the electrical energy consumed in RO desalination is used to pressurize the RO feed stream to the high operating pressures required for optimum performance. Since there is relatively little pressure drop through the RO membranes, a significant portion of this energy can be recovered, thereby reducing energy consumption and hence energy costs and water production costs. Many gains have been made in recent years in energy recovery and the CANDESAL system uses the latest energy recovery technology, which gives significant improvement in performance.

3.2. The Floating Desalination System

The floating desalination system (FDS) is a self contained, barge mounted system that is designed to complement the FPU. Under conditions when the FPU is not available, it can also be used in conjunction with shore based fossil fuelled power plants or as a stand-alone desalination plant to provide operating reliability and security of water supply.

Several configuration options are being addressed in the design of the desalination barge. While it is established that the barge will not be self-propelled the onboard facilities are still under active consideration:

Option 1.

Option 2 would be a fully self-contained barge, with onboard accommodation for the plant operators and maintainers complete with galley and recreational facilities.

This option would require only the supply of the electrical power for the desalination plant and the heated condenser cooling water, along with any process heat available to the system.

Option 2.

Option 2 would be a fully self-contained barge from the standpoint of operation of the RO system. However, it would not include onboard accommodation or facilities.

This option would utilize the accommodation facilities of the FPU for the operators and maintainers of the desalination barge and, in situations where the barge was utilized independently of the power barge accommodation facilities would be erected ashore.

Both options would locate on the barge the desalination modules, clean in place system, auxiliary equipment rooms including air-conditioning, post treatment facility, the control and data-logging rooms, switchboard and cyclo-converter room, emergency generator, workshops and stores for spare and repair parts

3.3. Reverse Osmosis Module Description

The RO system consists of a set of modules operating in parallel to provide redundancy and reliability to the system. The RO modules are each sized for a daily production of 12,000 m^3/day of permeate, with ten modules fitted on the barge. Each module will comprise an ultrafiltration section, a feed treatment section, an RO section comprising pumps and RO pressure vessels, and energy recovery turbines. The pressure vessels will be connected to appropriate sub-headers that can be isolated for maintenance and the replacement of membranes. Pump characteristics will be fine-tuned to accommodate the prevailing conditions by the use of variable frequency drives to the electric motors. The modules will be baseplate mounted and installed as complete units into the barge structure Central headers will supply heated feed water to each of the modules and central headers will also carry away the concentrate and the permeate, the permeate header passing through the post treatment room. Because of the size of the plant modules and the fitting of them into a barge, consideration has been given to the layout of the pressure vessels and their attendant machinery from the point of view of space and accessibility for maintenance. Current indications are that the final module configuration may have the machinery mounted on a deck above the UF and RO pressure vessels, with the modules laid out in athwartships pairs on either side of a fore and aft layout of main headers (a 5 × 2 layout of the modules).

The design basis for the modules is for seawater conditions of 37,000 ppm and 25°C, resulting in feedwater to the RO system at about 35-40°C.

Common systems supplying all modules in the barge would include:

- Feedwater preheating from the FPU
- Common feed header
- Common permeate header

- Common concentrate header
- Clean In Place System
- Post treatment System

An example of a typical module is illustrated schematically in Figure 2 below.

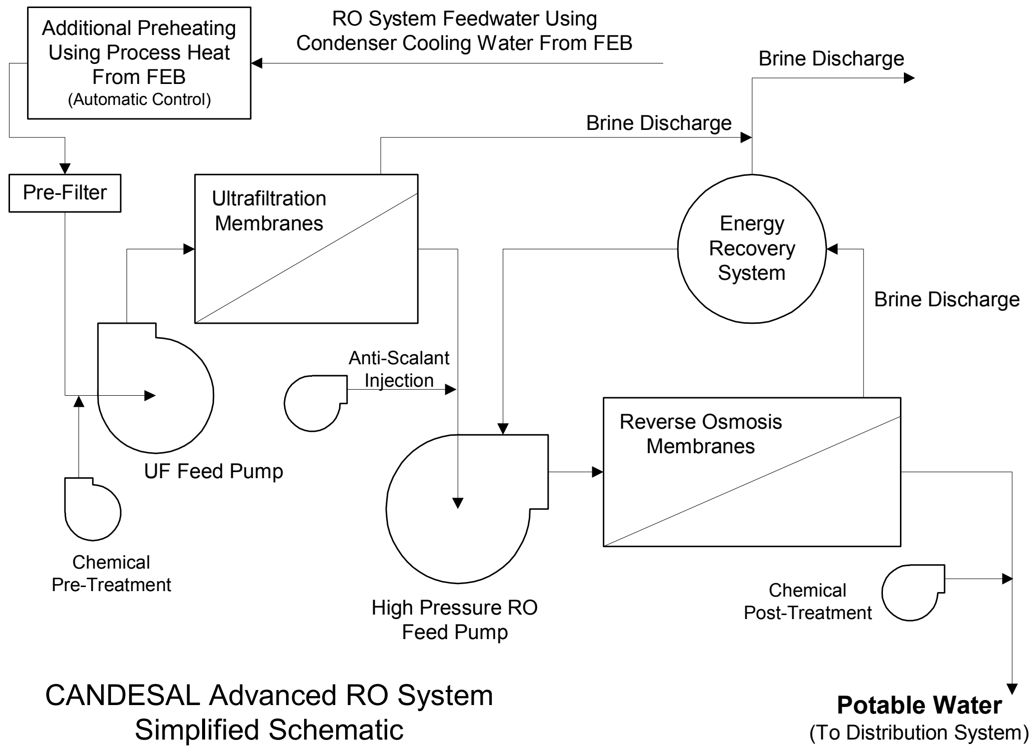

FIG. 2. Simplified Schematic Of Typical CANDESAL Advanced RO System Module.

3.4. Coupling To The Small Size Nuclear Heat Power Plant

Coupling of the floating energy block and the floating desalination system includes both direct and indirect coupling considerations.

Direct coupling includes the supply of electrical power from the FPU, heated seawater from the FPU resulting from condenser cooling, and heated water from the process steam stream generated in the FPU. The indirect coupling results from considerations of economic optimization with respect to the sale of electricity and process steam from the FPU to the FDS, as well as the potential provision of accommodation on the FPU.

The amount of process steam available for the heating of feed water is another area for study in establishing the coupling between the two barges. In the northern regions of the Russian Federation, where some (or possibly all) of the process heat is required for district heating, there will be little or no process heat available to the desalination system. In countries where process heat is not required for district heating, all of this heat is available for use by the desalination system. The most economical and energy efficient method for utilizing this heat is a site specific design consideration that is subject to design optimization.

3.5. The Economic Benefits of Cogeneration

Two of the most critical issues facing nuclear desalination as a commercially viable technology are energy utilization and the cost of water production. It was recognized early in the design process that improvements in the efficiency of energy utilization could be achieved by taking advantage of waste heat normally discharged from the reactor through the condenser cooling system. Hence, a strong emphasis has been placed on the integration of the energy and water production systems into a single, optimized design for the cogeneration of both water and electricity. Such a system provides the capability for economical supply of potable water and electricity without substantial infrastructure investment. This approach to the integration of the RO seawater desalination system with KLT-40C reactor 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. In essence, the reactor operates without "knowing" that there is a desalination plant associated with it. Transients in the desalination plant do not have a feedback effect on reactor operation. 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 also be undesirable to have reactor shutdowns, whether unanticipated or for planned maintenance, that would require shutdown of the water production plant.

As a result of the coupling of the FPU and FDS into an integrated facility, the economic advantage of using preheated feedwater can be realized. Figure 3 illustrates the relative increase in potable water production as a result of increasing RO system feedwater above ambient seawater temperature. As can be seen in this figure, the increased production capacity, achieved with the same feedwater flow and no additional pumping power required, is substantial.

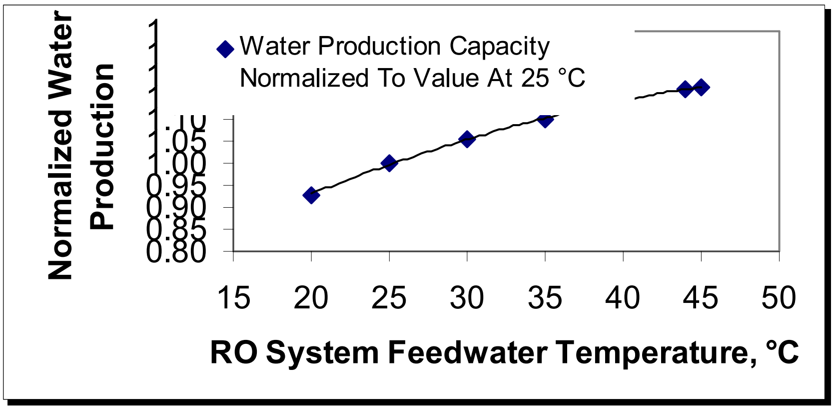

FIG. 3. Normalized Water Production As A Function Of RO Feedwater Temperature.

An economic analysis of this effect has been carried out using the Desalination Economic Evaluation Program (DEEP) developed by the International Atomic Energy Agency. In its present form, DEEP does not currently accommodate the effects of preheated feedwater, and modifications have been made to the RO performance algorithms in DEEP to properly account for the increased production capacity illustrated in Figure 3. Water production costs have been calculated for a contiguous RO plant (C-RO), in which the RO system feedwater temperature was at ambient temperature, and for two cases of preheated RO (PH-RO) to illustrate the effect of increased feedwater temperature. For purposes of comparison, water costs were also calculated for a stand-alone RO plant (SA-RO), in which a more traditional system design was evaluated.

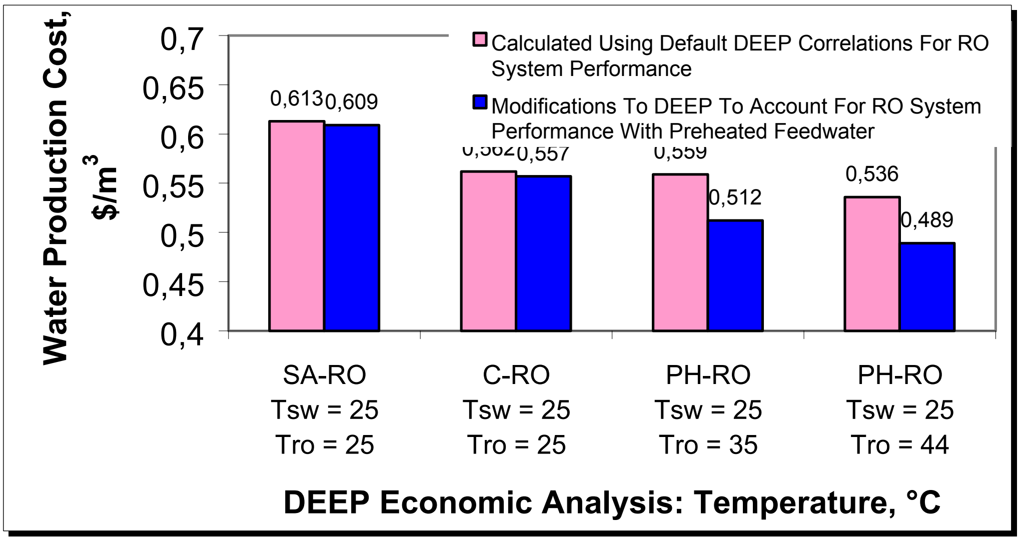

FIG. 4. DEEP Economic Analyses Of Water Production Costs.

4. SAFETY CONSIDERATIONS

The two key safety issues that must be taken into consideration in assessing nuclear desalination are the potential for radioactive carry-over into the product water and the potential effect of transients in the desalination unit on the safety of the reactor.

With respect to transients, as noted above, the very "loose" coupling associated with this design concept virtually eliminates any real concern. Nevertheless, it is recognised that transient effects must be considered in the design of the integrated system.

The public dose consequences from normal operation of a combined reactor and desalination facility have been conservatively assessed in a study carried out by Atomic Energy of Canada Limited. It should be noted that all operating pressures in the desalination plant and the coupling to the reactor system are such that any leakage in any of these systems would result in leakage into the reactor plant, not into the desalination system. It was assumed for the purposes of this analysis that these normal pressure barriers had failed. The radiological consequences were found to be insignificant compared with background doses and are a small fraction of public dose limits. Since an RO desalination facility has a high rejection rate for removal of elements with large molecular weights, there is likely to be a very high level of radionuclide removal by the plant, even in the unlikely event of an accident.

Ecological Safety

The release of radioactive materials from the SS NHPP under normal operating and design basis accidents does not contribute measurably to the natural radiation background and hence does not have an adverse effect on people and or the environment. An annual total output of radioactive gases to the atmosphere does not exceed 10 Ci, while the population exposure dose does not exceed 0.01 mrem/year. In addition, the SS NHPP with KLT-40C, as compared to fossil fuel stations,

- Saves 300 000 t/year of fuel;
- Keeps 400 million cubic meters of air a year from being used (in fossil fuel combustion);

- Does not pollute the atmosphere with harmful substances, such as sulfur anhydride, nitrogen and vanadium oxides, etc.

Thus, an absence of measurable output of harmful substances, under normal operating and expected safety levels (no people evacuation under any accidents) provides an opportunity to locate SS NHPP in the vicinity of living areas. SS NHPP produces definite ecological advantages in comparison with fossil fuel power sources.

5. CURRENT STATUS OF THE RUSSIAN FEDERATION-CANADA JOINT PROJECT

5.1. Project Overview

As noted above, a formal agreement to build a floating nuclear desalination system was reached in June 2000. The June Meeting took place in the framework of the 4th Intergovernmental Economic Commission meeting between the Russian Federation and Canada. A formal Declaration of Cooperation in the Field of Nuclear Desalination between the Russian State Concern "Rosenergoatom", the Russian public joint-stock company Malaya Energetica, and the Canadian company CANDESAL Technologies was signed at this meeting. In addition, a Letter of Intent between Malaya Energetica and CANDESAL setting out the basis for proceeding with the project was signed. These documents underlined the intentions for long term cooperation for the purpose of developing a floating nuclear desalination complex, where the Russian floating energy block is used as a power source for the Canadian desalination system.

Following the meeting in June, a Working Meeting was held in November 2000. A wide spectrum of technical, economic, marketing and other issues was discussed during this Working Meeting. In addition, it was agreed to create joint executive bodies, a Coordinating Committee and a senior Working Group, to address policy and economic issues and to facilitate project execution.

5.2. Current Status Of The Floating Energy Block

The technical design of the FPU with KLT-40C is completed and prepared for consideration by Safety &Technology Council of MINATOM. The first issue of the Safety Report, which is necessary to initiate construction of the first FPU, is issued. The procedures for State environmental assessment are developed. The ship assembly plant for the first FPU is identified, and work for preparation of industrial facilities is now underway

At present there are three sites identified for construction of small atomic power thermo-nuclear stations. Severodvinsk, of the Archangelsk region, has been selected as the location for the first SS NHPP. Marketing in the Russian Federation and abroad is developing with the intent of constructing of a set of these stations.

5.3. Current Status Of The Floating Desalination System

Design of the basic RO system module is underway, and optimization studies have been carried out to evaluate the best configuration for such a module. Preliminary modifications have been made to DEEP to allow economic analysis of RO systems with preheated feedwater and to properly represent the performance characteristics of an optimized RO system design. The modified code has been used to evaluate the effect of using process heat in the coupling scheme. Additional work is required, however, to make more substantial changes to DEEP to fully represent the FPU and FDS designs.

An experimental facility has been constructed and commissioned in Canada to validate the non-traditional approach to design of RO systems developed by CANDESAL. Functional testing of the facility has been completed, and the initial data obtained has shown extremely good agreement with expectations. Even though continued testing is planned, the data obtained to date has provided a high level of assurance that there is no technological risk in this unique approach to desalination system design and operation.

5.4. Coupling Schemes

A variety of coupling schemes are under consideration in order to obtain the optimum output of both electricity and potable water from the floating nuclear desalination complex. Since both condenser cooling water and, in some cases, process heat are available from the FPU there are a number of ways in which the FPU and the FDS can be coupled to take advantage of these two sources of heat energy. Figure 5 illustrates and example of two of the coupling schemes currently being evaluated.

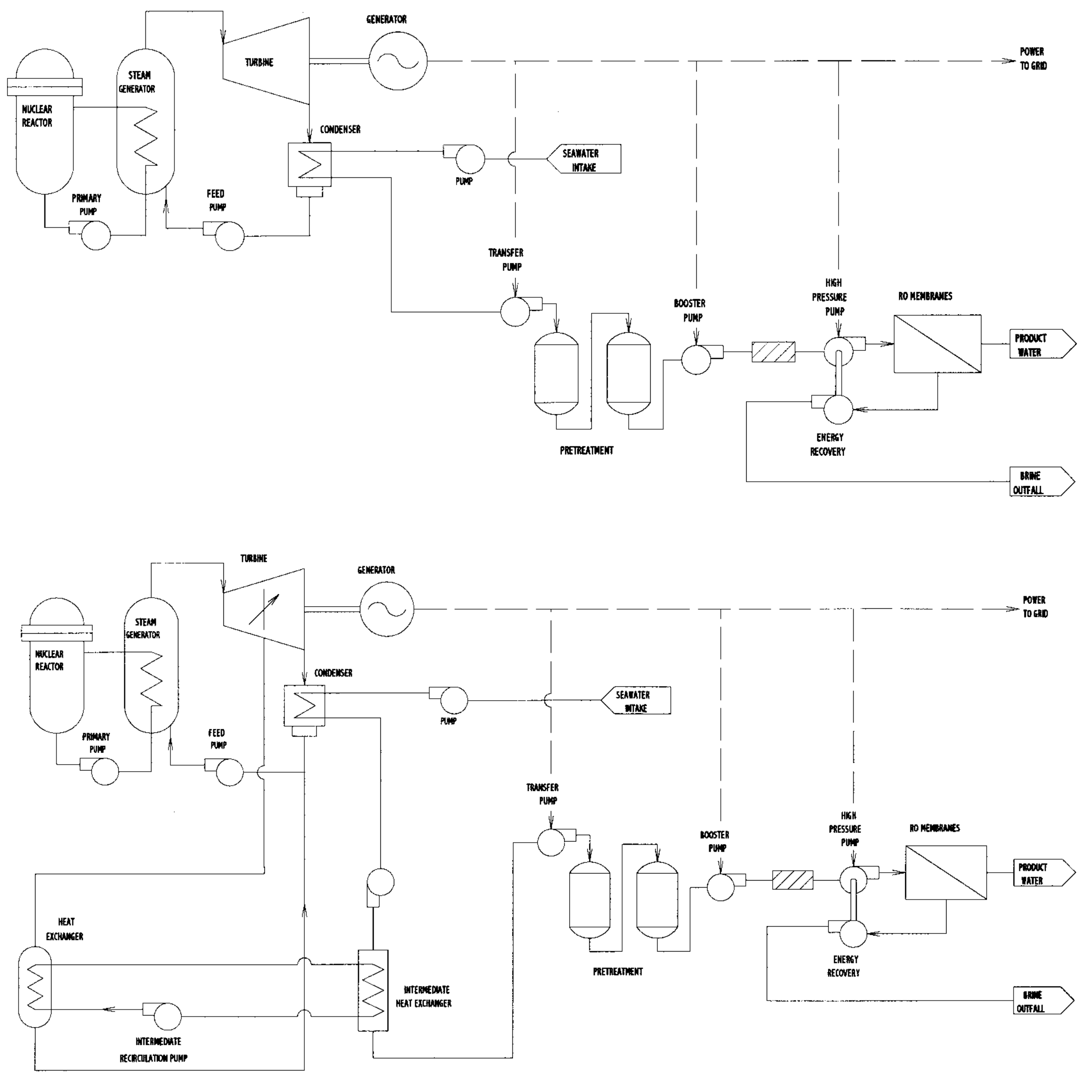

Figure 5. Example Coupling Schemes.

CONCLUSIONS

The work completed to date, as well as the results of the Working Meeting, has shown that the joint Russian Federation-Canada project for development of a floating nuclear desalination complex is a realistic task, and is occupying increasing levels of effort by specialists of both countries. Apart from this, the joint project requires support at the national level in both countries, as well as support from international organizations. Successful realization of the joint project offers the opportunity to resolve problems of potable water deficit and stable energy supply within a relatively short time frame, and with reasonable expanse, in regions where there is a shortage of potable water and decentralized energy supply.

In addition to providing a solution to very serious problems in the International energy and potable water markets, this unique joint project for development of a nuclear desalinating complex, with the FPU as a power source, offers investors a leading role in the supply of small energy systems and optimized, economical RO systems.