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Nuclear heat applications: Design aspects and operating experience

Proceedings of four technical meetings held between December 1995 and April 1998

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FOREWORD

Nuclear energy is playing an important role in electricity generation, producing 17% of the world's electricity. It has proven to be safe, reliable, economic and has only a minimal impact on the environment. Most of the world's energy consumption, however, is in the form of heat. The market potential for nuclear heat was recognized early. Some of the first reactors were used for heat supply, e.g. Calder Hall (United Kingdom), Obninsk (Russian Federation), and Agesta (Sweden). Now, over 60 reactors are supplying heat for district heating, industrial processes and seawater desalination. But the nuclear option could be better deployed if it would provide a larger share of the heat market. Nuclear energy can provide a clean alternative to the burning of fossil fuels for the production of industrial process heat, for district heating and for seawater desalination. In several countries nuclear heat is already being used for these purposes. In particular, seawater desalination using nuclear heat is of increasing interest to some IAEA Member States.

In consideration of the growing experience being accumulated, the IAEA is periodically reviewing the progress and new developments in the field of nuclear heat applications. This publication summarizes the results of the IAEA's activities along this line since the status and international progress made in nuclear heat application and associated reactor development were reviewed and evaluated in November 1995.

The first part of this publication is a presentation and discussion on all relevant aspects of nuclear heat applications. This includes operating experience as well as technology, design and operational aspects of reactors, coupling devices and application facilities for district heating, process heat and seawater desalination.

The second part of this publication includes the papers presented at the following meetings:

- Advisory Group Meeting and Consultancy on Experience with Nuclear Heat Applications: District Heating, Process Heat and Desalination, 13–15 December 1995 and 7–9 February 1996;
- Advisory Group Meeting on Technology, Design and Safety Aspects of Non-electrical Application of Nuclear Energy, 20–24 October 1997;
- Advisory Group Meeting on Operational Modes of Nuclear Desalination Plants, 3–5 November 1997;
- Advisory Group Meeting on Materials and Equipment for the Coupling Interfaces of Nuclear Reactors with Desalination and District Heating Plants, 21–23 April 1998.

The proceedings are structured according to subject areas: (1) design and safety aspects of nuclear heat application, (2) operational and material aspects of nuclear heat application, and (3) operational experience with nuclear heat application.

There are now about 500 reactor-years of operational experience with nuclear district heating, industrial processes and seawater desalination. There appear to be no major technical or safety concerns with nuclear heat application systems. The design precautions to prevent the carry-over of radioactivity into the heating network or into the desalted water have proven to be effective. These findings are important for future applications of nuclear heat for district heating, process heat and seawater desalination. They will provide a reliable basis for a more extensive deployment of nuclear heat, thus leading to better energy supply diversification.

It is expected that the information contained in this publication will be useful for policy planners, engineers and scientists in the areas of heat applications and seawater desalination.

The IAEA officers responsible for the compilation of this publication were T. Konishi and G. Woite of the Division of Nuclear Power.

EDITORIAL NOTE

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Experience with nuclear desalination

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[1]Presented at the Advisory Group Meeting/Consultancy on Experience with Nuclear Heat Applications: District Heating, Process Heat and Desalination, 13–15 December 1995/7–9 February 1996.
[2]Presented at the Advisory Group Meeting on Technology, Design and Safety Aspects of Non-electrical Application of Nuclear Energy, 20–24 October 1997.
[3]Presented at the Advisory Group Meeting on Operational Modes of Nuclear Desalination Plants, 3–5 November 1997.
[4]Presented at the Advisory Group Meeting on Materials and Equipment for the Coupling Interfaces of Nuclear Reactors with Desalination and District Heating Plants, 21–23 April 1998.

Part I

BACKGROUND REPORT

1. INTRODUCTION

The overall world energy consumption is steadily increasing. It is expected to increase by almost 50% by the year 2015. About 33% of it is currently used for electricity generation. 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, and wood. Although some of the first reactors were used for heat supply, e.g. Calder Hall (United Kingdom), Obninsk (Russian Federation), Agesta (Sweden), currently less than 1% of the heat generated in nuclear reactors worldwide is used for desalination, district and process heating. The heat market is an open challenge and there are signs of increasing interest in these applications [1].

Nuclear energy is now used to produce electricity in 30 countries [2]. It provides about 6% of global energy and 17% of global electricity supply. As of Dec.1997 437 nuclear plants, with a total capacity of 352 GW(e), were in operation. A few of these plants are used for cogeneration of hot water and/or steam for district heating, seawater desalination or industrial processes. The total heat capacity of these plants is about 5 GW(th). Significant experience in the cogeneration of electricity and heat has been gained in Bulgaria, Canada, Germany, Hungary, Japan, Kazakhstan, Russian Federation, Slovakia, Switzerland, Ukraine and the United States of America. Experience with dedicated nuclear heating plants has been gained in China and Russia.

Although development of nuclear power reactors has been mainly directed to electricity production, the interest in utilizing nuclear heat for other products still remains and is growing. For heat applications, specific temperature requirements vary greatly. They range from temperatures around 100°C for hot water and steam for district heating and seawater desalination, to up to 1000°C for process heat for the production of hydrogen by water splitting. Although various forms of nuclear heat application are technically feasible and pursued between these temperature ranges, the major interest seems to be directed to the lower end using water cooled reactors [1] and to the higher end using high temperature gas cooled reactors [3].

From the experience of using nuclear heat in both district heating and in industrial processes, the technical aspects can be considered well proven [1]. There are no major technical impediments to the application of nuclear reactors as heat sources for non-electrical products. In principle, any type and size of nuclear reactors could be used for direct heat applications. Interest in using nuclear energy for producing potable water from seawater desalination has been growing among many Member States [4].

Adequate quality and sufficient amount of water is essential for human life. The scarcity of freshwater and especially drinkable water is jeopardizing many regions of the world. By 2025, about two-thirds of world population may be exposed to high or moderate water stress. Top concern is in the Middle East and African region, parts of Latin America and South-East Asia [5]. Seawater desalination offers one of the most promising alternatives for the supply of required potable water. The worldwide cumulative seawater desalination capacity has steadily increased in the past few decades, and this trend is expected to continue into the next century [6].

Combining the use of nuclear energy with the industrial process of supplying potable water by seawater desalination has been considered as far back as in the 1960s. While developments in the nuclear field focused on large power reactors for electricity generation, the interest in seawater desalination was less at that time than in district heating and industrial uses of process heat.

Since IAEA activities concerning nuclear desalination[1] were resumed in 1989 in response to the renewed interest indicated by several Member States, a growing number of countries and international organizations have participated in meetings, and provided information and support.

2. REACTORS AND NUCLEAR HEAT APPLICATION TECHNOLOGIES

2.1. Characteristics of heat applications

For heat applications, specific temperature requirements vary greatly (Fig. 1). They range from low temperatures for applications such as hot water and steam for agro-industry, district heating, and seawater desalination, to up to 1000°C for process heat for the production of hydrogen by water splitting.

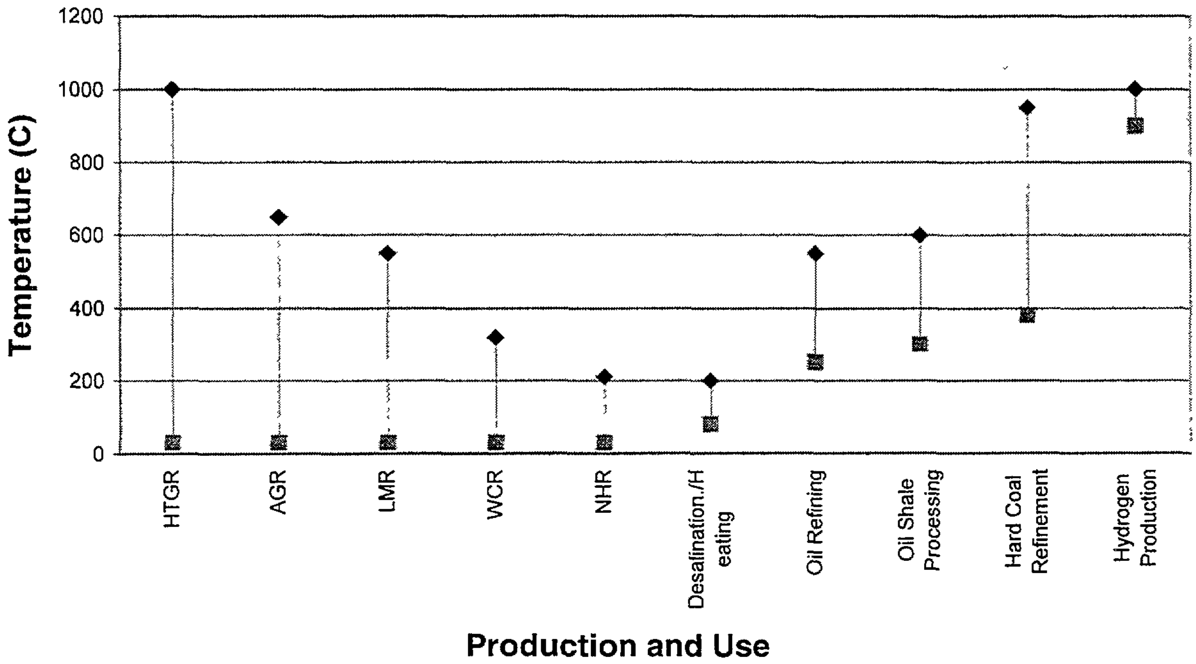

FIG. 1. Temperature ranges in production and use of nuclear heat.

District heating networks generally have installed capacities in the range of 600 to 1200 MW(th) in large cities, decreasing to approximately 10 to 50 MW(th) in towns and small communities. Obviously, a potential market for district heating appears mainly in climatic zones with relatively long and cold winters. In western Europe, for example Finland, Sweden, and Denmark are countries where district heating is widely used. The 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 below what is needed for base load operation of plants.

Within the industrial sector, process heat is used for a very large variety of applications with different heat requirements and with temperature ranges covering a wide spectrum. While in energy intensive industries the energy input represents a considerable fraction of the final product cost, in most other processes it contributes only a few percent. Nevertheless, the supply of energy has an essential

[1] "Nuclear desalination" is taken here to mean the production of potable water from seawater in an integrated complex in which both the nuclear reactor and the desalination system are located on a common site, the relevant facilities and services are shared, and the energy used for the desalination process is produced by the nuclear reactor.

character. Without energy, production would stop. This means that a common feature of most industrial users is the need for assurance of energy supply with a very high degree of reliability and availability, in particular for large industrial installations and energy intensive processes. Contrary to district heating, the load factors of industrial users do not depend on climatic conditions. The demands of large industrial users usually have base load characteristics.

2.2. Reactors for heat application[2]

In nuclear reactors heat is generated by fission of heavy nuclei. For heat applications, there are basically two options: Cogeneration of electricity and heat, and dedicated nuclear heating reactors. Cogeneration has been widely applied and experienced. In the cogeneration mode, electricity will usually constitute the main product. Large size reactors, therefore, have to be integrated into the electrical grid system and optimized for base load 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 will affect the plant optimization criteria. In principle, any amount of heat can be extracted from cogeneration reactors upon demand within the design limitations. However, when the extracted heat is a significant fraction of the reactor power, the impacts of heat and power fluctuations must be carefully analysed. If the heat load is fluctuating daily or seasonally, the electricity production will fluctuate as well. Technically it would raise no significant problem as long as the electricity generation is compatible with the grid load. Technical characteristics of the plant such as anticipated transients may need further evaluation. Heat only reactors have a different approach, although not much experience is available to share. There are some innovative design approaches for dedicated heating reactors.

A variety of reactor designs are being developed for district heating, seawater desalination, or low temperature process heat: dedicated heating reactors (in China), pressurized water reactors (in Argentina, Rep. of Korea and Russia), and Pb–Bi cooled reactors in Russia. Heating reactors in Russia are mostly cogenerating, while China is developing dedicated heat reactors. In Russia further cogenerating plants with improvements based on experience are proposed to replace old plants. China has a demonstration plant project of an NHR-200 which is planned to be built in the North East region of China. Their main technical and safety features are: integrated design, full power natural circulation cooling, dual vessel design, hydraulic driving mechanism of the control rods, primary pressure self-regulation, and low process parameters.

In Russia utilization of lead-bismuth alloy as a coolant for heating plants, either for district heating or for seawater desalination, is being examined. The eutectic lead–bismuth alloy (44.5% Pb–55.5% Bi) has good coolant properties: high boiling temperature and low freezing temperature, low chemical activity, low long lived induced activity, etc. It was reported at an earlier Consultancy at the IAEA (1993) that about 150 reactor-years experience with lead–bismuth cooled reactor systems had been gained. Based on this experience application of lead–bismuth reactor technologies for peaceful uses are being reassessed mainly for small transportable plants, either cogenerating or single purpose for district heating or for seawater desalination. A project is launched on the modular and transportable cogenerating station ANGSTREM.

From about 500 reactor-years of operational experience, no incidents including radioactive contamination have ever been reported for any of the heat supplying reactors. Radioactive contamination of the district heating networks or of the products obtained by the industrial processes has been avoided by adequate design precautions.

The prospects for applying nuclear energy to district and process heating are closely tied to the prospects of deploying SMRs. Design and development status of SMRs as of 1995 are well documented [7]. A recent market assessment for SMRs found that 70 to 80 new units could be installed in about 30

[2] High temperature reactors are described in Section 2.4 together with high temperature heat applications.

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 [1].

2.3. Low temperature heat applications

Low temperature heat applications include seawater desalination, district heating and a large variety of agricultural and industrial processes. Seawater desalination requires temperatures up to about 130°C, district heating up to about 170°C and low temperature industrial processes up to about 250°C. Relevant characteristics of these applications are discussed below.

2.3.1. District heating systems

The technical viability of using nuclear heat for the supply of hot water and steam for district heating has been demonstrated both in dedicated nuclear heating plants and in heat and power cogeneration plants.

Extracted steam from high and/or low pressure turbines is fed to heat exchangers to produce hot water/steam, which is delivered to consumers. Extracted steam from low pressure turbines (LPT) is usually used for the base heat load, while steam from high pressure turbines (HPT) is used, as needed, to meet the peak heat demand.

Depending on the transportation distance and the number of end users, there are usually several pumping stations between the heating source and end users. Heat transport pipelines are installed either above or under ground. They are well insulated in order to minimize the heat loss. Glass wool or rock wool are used often for insulation.

In commercial scale heating networks, the transportation distances are usually less than 10 km, in most cases between 3 km and 6 km. The longest delivery distance known to the IAEA is 24 km in Slovakia. A typical schematic diagram of a nuclear district heating network is shown in Fig. 2.

The design precautions to prevent the transfer of radioactivity into the district heating grid network have proven to be effective in many years of safe and reliable operation. These design features include one or more barriers to radioactive substances, e.g., in the form of a leak-tight intermediate heat transfer loop at a pressure higher than that of the steam extracted from the turbine cycle of the nuclear plant. These loops are continuously monitored, and isolation devices are provided to separate potentially contaminated areas.

District heating systems require a backup heat source when the main heat source is unavailable. Therefore, at least two nuclear power units or a combination of nuclear and fossil fired units are used for district heating grids.

2.3.2. Nuclear desalination systems

Seawater desalination is the processing of seawater, through the separation of dissolved saline components, to obtain fresh water with low salinity, adequate for drinking or industrial use. In general, large scale commercially available desalination processes can be classified into two categories: (a) processes that require mainly heat and some electricity for ancillary equipment (distillation processes, including multi-stage flash (MSF) and multi-effect distillation (MED)), and (b) processes that require only electricity (membrane processes, in particular reverse osmosis (RO)).

Any desalination process requires energy. For standard seawater (35 000 ppm total dissolved solids, 25°C), the theoretical minimum separation work required to produce 1 m^3 of pure water is about 0.73 kW·h. However, the energy consumption of currently available commercial processes is much

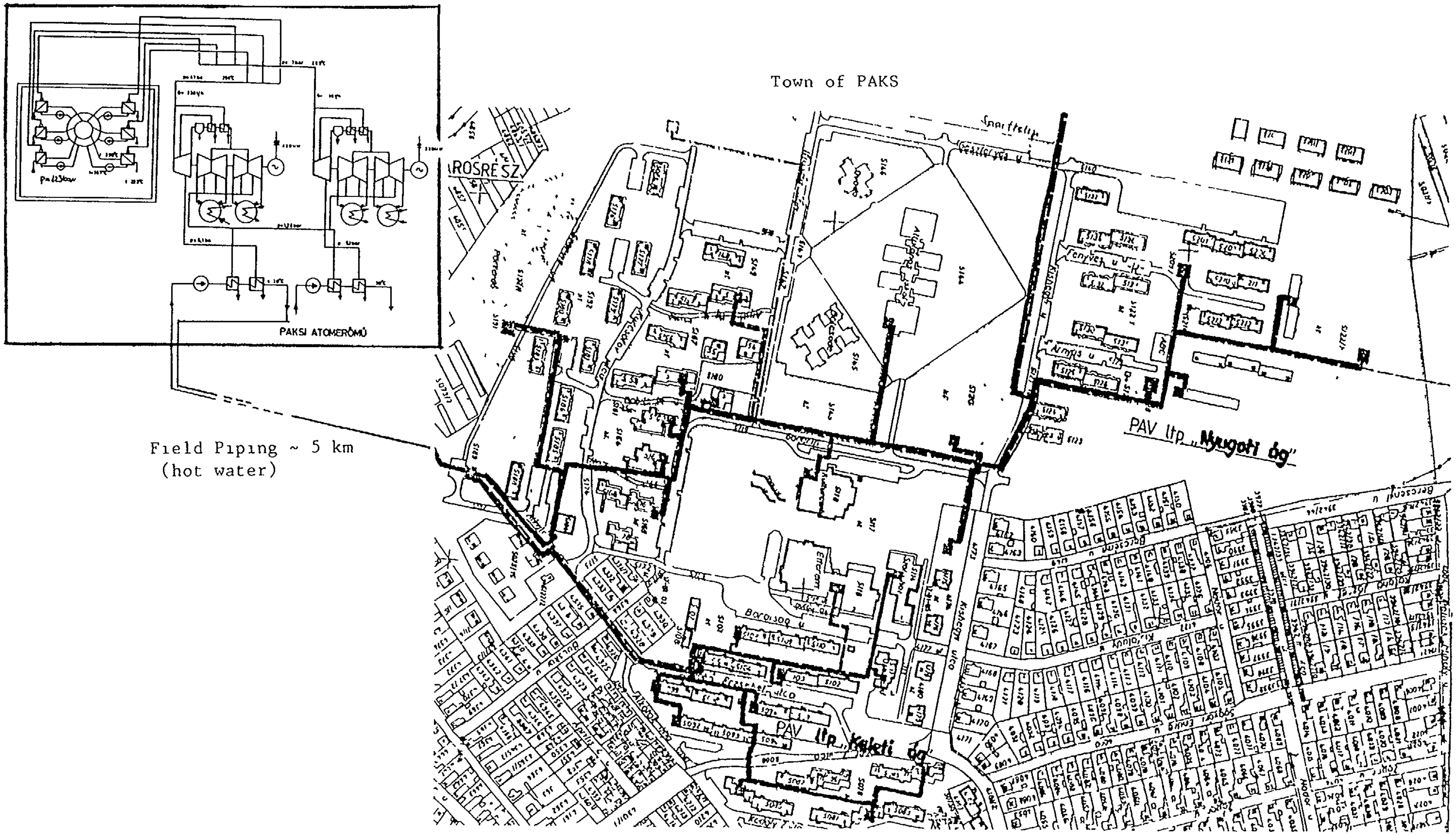

FIG 2 Schematic diagram of a nuclear district heating network in Paks, Hungary

higher due to thermal losses and irreversibility that occur during the separation process. The lowest energy consumption including that for seawater pumps and water pre-treatment is currently obtained with reverse osmosis (RO) plants. It amounts to 4 to 7 kW(e)·h/m^3 of electrical energy depending on product water quality requirement, feed seawater salinity and plant configuration. Among various seawater desalination technologies, only distillation processes and the RO process have achieved commercial large scale application. Further improvements are being developed for membrane processes as well as for hybrid systems comprising mechanically (electrically) and thermally driven processes.

Most relevant and widely applied desalination processes are briefly characterized in the following.

- Distillation processes

In distillation processes, sea water is heated to evaporate pure water that is subsequently condensed. With the exception of mechanical vapour compression, distillation processes are driven by low temperature steam as the heat source, which is usually taken as exhaust/extraction steam from adjacent turbines of power plants. As a result of the high specific heat required to evaporate water, commercial distillation processes are implemented in heat recovery stages placed in series. The larger the number of stages assembled, the better the performance of distillation processes. However, the overall temperature difference between the heat source and the cooling water heat sink as well as economic reasons limits the number of stages. Typical temperature differences for commercial distillation plants are 2–5°C per heat recovery stage. Usually, the thermodynamic efficiency of distillation plants is expressed in kg of water produced per kg of steam used. This ratio is called the gain–output ratio (GOR), which is in the range of 6 to 10 for current commercial multi-stage-flash distillation plants and up to 20 for multiple-effect distillation plants. Because of the low TDS of the product water it needs to be post-treated for drinking use.

– Multi-stage-flash (MSF) distillation

Figure 3 shows the schematic flow diagram of an MSF system [8].

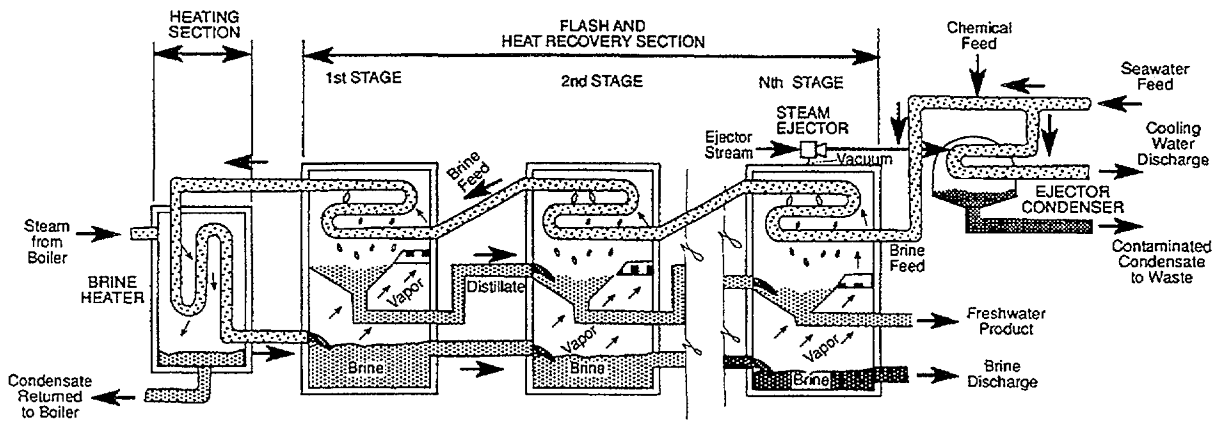

FIG. 3. Diagram of a multi-stage flash (MSF) distillation plant.

Seawater feed passes through tubes in each evaporation stage where it is progressively heated. Final seawater heating occurs in the brine heater by the heat source. Subsequently, the heated brine flows through nozzles into the first stage, which is maintained at a pressure slightly lower than the saturation pressure of the incoming stream. As a result, a small fraction of the brine flashes forming pure steam.

The heat to flash the vapour comes from coolıng of the remaınıng brıne flow, whıch lowers the brıne temperature Subsequently, the produced vapour passes through a mesh demıster ın the upper chamber of the evaporatıon stage where ıt condenses on the outsıde of the condensıng brıne tubes and ıs collected ın a dıstıllate tray The heat transferred by the condensatıon warms the ıncomıng seawater feed as ıt passes through that stage The remaınıng brıne passes successıvely through all the stages at progressıvely lower pressures, where the process ıs repeated MSF plants need pre-treatment of the sea water to avoıd scalıng by addıng acıd or advanced scale ınhıbıtıng chemıcals The vent gases from the deaeratıon together wıth any non-condensable gases released durıng the flashıng process are removed by steam-jet ejectors and dıscharged to the atmosphere

Today, MSF plants have reached a mature and relıable stage of development Unıt sızes up to 60 000 m^3/d have been buılt The thermal heat and electrıcıty consumptıon ıs ın the range of 45 to 120 kW(th) h/m^3 and 3 0 to 6 0 kW(e) h/m^3 respectıvely Usıng polymerıc antı-scalıng addıtıves, the maxımum brıne temperature ıs lımıted to 120°C for MSF-BR (brıne recycle) systems and 135°C for MSF-OT (once-through) systems

– Multıple-effect dıstıllatıon (MED)

Fıgure 4 shows the schematıc flow dıagram of MED process [8]

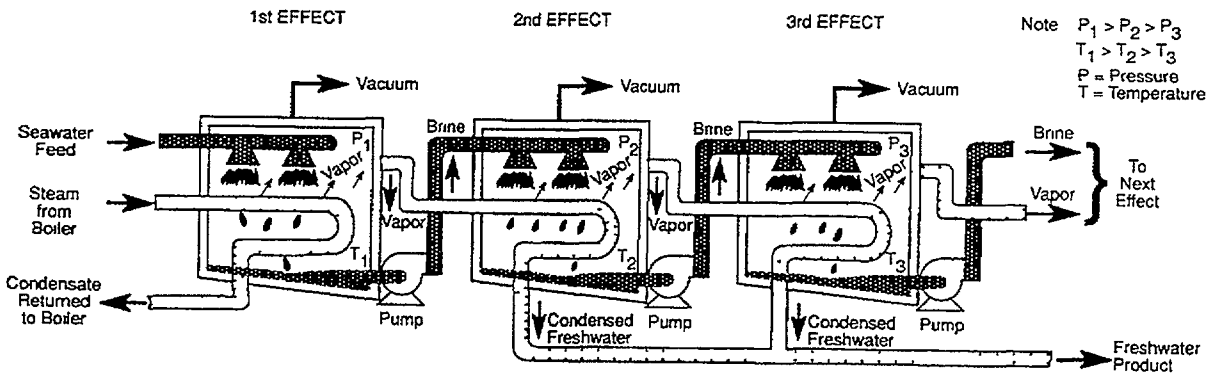

FIG 4 Dıagram of a multıple effect dıstıllatıon (MED) plant

In each effect heat ıs transferred from the condensıng water vapour on one sıde of the tube bundles to the evaporatıng brıne on the other sıde of the tubes Thıs process ıs repeated successıvely ın each of the effects at progressıvely lower pressure and temperature, drıven by the water vapour from the precedıng effect In the last effect at the lowest pressure and temperature the water vapour condenses ın the heat rejectıon heat exchanger, whıch ıs cooled by ıncomıng sea water The condensed dıstıllate ıs collected from each effect Some of the heat ın the dıstıllate may be recovered by flash evaporatıon to a lower pressure (not ıllustrated ın Fıgure 4)

Currently, the most domınant MED processes wıth the hıghest technıcal and economıc potentıal are the low-temperature horızontal-tube multı effect process (LT-HTME) and the vertıcal-tube evaporatıon process (VTE) The maın dıfferences between LT-HTME plants and VTE plants are ın the arrangement of the evaporatıon tubes, the sıde of the tube where the evaporatıon takes place and the evaporatıon tube materıals used In LT-HTME plants, evaporatıon tubes are arranged horızontally and evaporatıon occurs by sprayıng the brıne over the outsıde of the horızontal tubes creatıng a thın fılm from whıch steam evaporates In VTE plants, evaporatıon takes place ınsıde vertıcal tubes Furthermore, ın LT-HTME plants the maxımum brıne temperature ıs lımıted to 70°C, sınce low cost materıals such as alumınıum for heat exchanger and carbon steel as shell materıal are used

MED plants have a much more efficient evaporation heat transfer process than MSF plants. Due to the thin film evaporation of brine on one side of the tubes and the condensation of vapour on the other side, high heat-transfer coefficients are achieved, thus decreasing the specific heat consumption in comparison to MSF plants with identical heat transfer area and the same temperature difference between heat source and cooling water sink. The pre-treatment of sea water for MED plants is similar to that in MSF plants.

In some MED designs, a part of the vapour produced in the last effect is compressed to a higher temperature level so that the energy efficiency of the MED plant can be improved (vapour compression). To compress the vapour, mechanical compressors (isentropic efficiency: about 80%) or steam-jet ejectors (isentropic efficiency: less than 20%) are employed. These designs, however, are usually not applied in cogeneration plants.

The heat and electricity consumption of commercial MED plants is in the range of 30 to 120 kW(th)·h/m^3 and 1.5 to 2.5 kW(e)·h/m^3 respectively depending on the design and the seawater temperature difference.

- Reverse osmosis (RO)

RO can be thought of as a filtration process at the molecular/ionic size level (Figure 5).

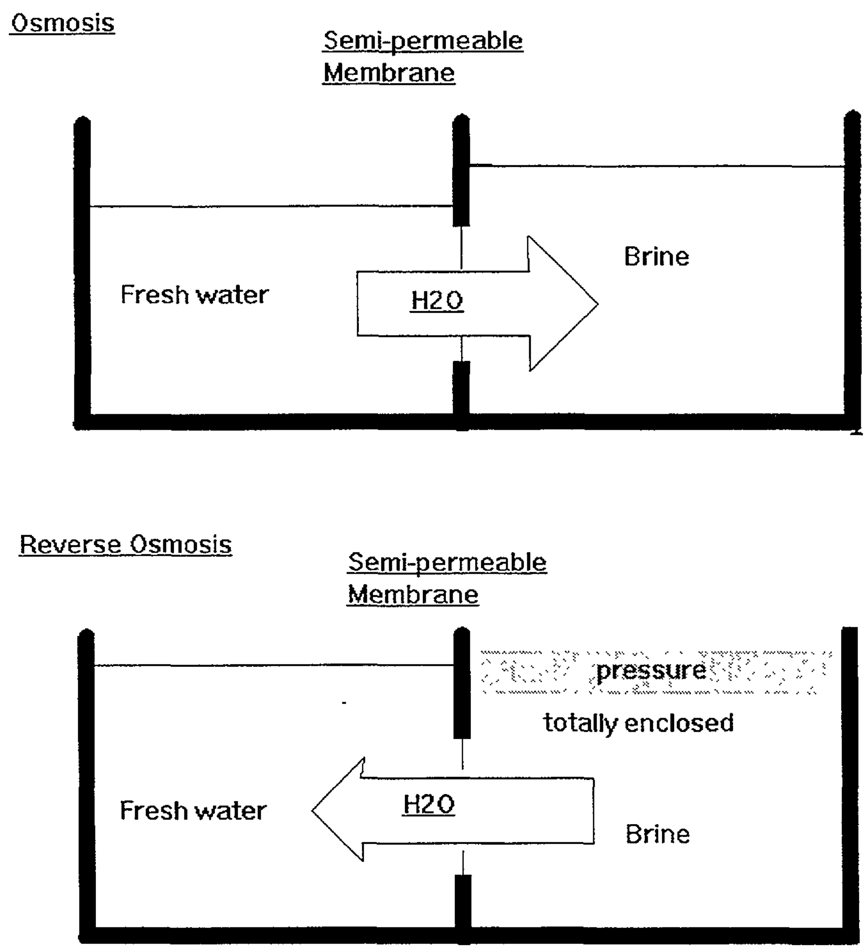

FIG. 5. Principle of reverse osmosis process.

If the solution compartment is now enclosed, and a pressure higher than the natural osmotic pressure of the solution is applied to it, the direction of water flow is reversed. The solution becomes more concentrated, and purified water is obtained on the other side of the membrane, hence the term 'reverse osmosis'.

The rate of flow of purified water depends on various factors such as the chemical properties of the membrane polymer itself, membrane thickness, area, pressure, concentration, pH, and temperature. Under any set of fixed conditions, product flow is proportional to the difference between applied pressure minus the osmotic pressure of the solution, and the permeate pressure.

The saline feed is pumped into a closed vessel where it is pressurized against the membrane. As a portion of the water passes through the membrane, the salt content in the remaining feed water increases. At the same time, a portion of this feed water is discharged without passing through the membrane. In practice, feed has to be compressed up to 70 to 80 bar since the osmotic pressure of the saline solution is about 60 bar, whereas the osmotic pressure of the permeate is negligible. Membrane modules are of tubular type, plate and frame type, spiral-wound type and hollow-fibre type. The most widely used modules today are the spiral-wound and hollow-fibre ones. RO processes need significant pre-conditioning of the feedwater to protect the membranes.

In recent years, seawater RO has become a reliable and commercial process applicable on a large scale. Typical electricity consumption of RO plants is in the range of 4 to 7 $kW(e){\cdot}h/m^3$ depending on the seawater salinity and temperature, recovery ratio, required permeate quality, plant configuration and implementation of energy recovery in the brine blow down. Usually, RO systems can produce fresh water with TDS of about 100 to 200 ppm depending on the system configuration and the feedwater salinity. If lower TDS is required, a two-stage configuration will be needed to get higher separation performance.

- Hybrid desalination systems

Advantages of the high desalting performance of distillation processes and lower energy demand of membrane processes could be combined in hybrid desalination systems for achieving overall improvement in potable water production.

Hybrid desalination systems combine power generation with both mechanically driven (e.g. RO) and thermally driven desalting processes (e.g. MSF or MED). The selection of particular method of coupling depends on the size and the type of the energy source, the desalination process and product water quality requirements.

Hybrid desalination systems have potential advantages of a higher overall availability, low power demand, improved water quality and possible lower running cost as compared to stand-alone RO plants. Appropriate combinations depend on the local conditions and requirements, in particular the required power to water ratio and product water quality. The combination can lead to significant technical and economic advantages of hybrid versus single technology desalination, including:

- Common, small seawater intake
- Optimized feedwater temperature of the RO plant by using cooling water from the heat reject section of the MED or MSF plant
- Improved performance and extended life of membranes
- Single stage RO process with high recovery ratio may be selected
- Blending of product waters and common post treatment
- Low power to water ratio (important when water production is the main requirement)
- Low water production cost.

The variety of possible process combinations and potential advantages of hybrid desalination systems are more fully described in Ref. [9].

2.3.3. Process heat

Nuclear generated heat may be used for industrial processes if:

- the NPP and the industrial facility are located close enough,
- the NPP can reliably supply the required heat amount at the required temperature, and
- the heat cost is competitive with alternative sources.

Most industrial processes require highly reliable heat supply; but some processes (e.g. drying) can also work with interruptible heat supply. Industrial heat consumers can be supplied with steam from a multi-unit or from a single unit nuclear station. In both cases, one or several conventional backup boilers are usually kept on standby.

Several barriers are usually installed between the reactor and the consumers to prevent both the ingress of radioactivity into the heat distribution system and pollution by the consumers of the nuclear heat supply system and/or the turbine cycle.

Temperature requirements range from below 100 to about 1000°C. For some agro-industrial processes, a few degrees above ambient temperature are required which may be supplied as waste heat. Many industrial processes require temperatures of around 100 to over 200°C which may be supplied from the turbine cycle of cogeneration plants. High and medium temperature applications are discussed in the next Section.

2.3.4. Coupling of reactors to heat application systems

Correct function of interface equipment is an important basis for good operating performance. Operating experiences of interface equipment for Nuclear Desalination (ND) and Nuclear District Heating (NDH) are not different from those in commercial thermal plants. Maintaining of interface equipment coupling NHR or NPP with heat extraction with the district heating network and/or desalination, can be done in the same manner like in normal thermal plants. One exceptional precaution is the radioactivity monitoring foreseen/installed in the intermediate and the heat application circuit.

2.3.4.1. Design requirements at the interface

In general, design requirements for the coupling interface are aimed at efficiency and safety, which refer to all engineering designs. Whereas efficiency depends on the expected specific performance of components, subsystems and systems, safety can be specified for the interface as fulfilling the following principles:

- No radioactive ingress to the desalination system.
- No chemical contamination ingress from the desalination to the NSSS.
- Controlled transients in one system and their effects on the other.
- Sufficient means to correct failures in the above three situations, and additional means to correct, or at least limit the damages, if any of the first means fails.

A pre-requisite for the ability to correct failures is to design and keep the monitoring system at best performance to detect necessity for immediate closing, diverting, and/or disconnecting contaminated streams. In addition, mechanical means to implement such handling of the said streams should be ensured: bypass piping, storage tanks, etc. Also means for locating the troubles and detecting the reasons, as well as access for repairs, have to be prepared.

2.3.4.2. Technical aspects on design precautions and analyses of integration at the interface

In most cases at least two mechanical barriers (heat exchangers) exist between the primary loop of nuclear power plants and the heat application system. The heat application system itself has usually, at least one barrier between the heating loop and the product (water or heating media).

Pressure reversal difference across the physical barrier constitutes another safety precaution. Examples are higher pressure in the secondary over the primary, or in the heat application system over the intermediate loop, etc., thus maintaining a barrier against radioactive carryover. Design precautions should be taken to satisfy these conditions at all operating conditions including transient, partial loads, and accidents. In the examples of a nuclear desalination (ND) and a nuclear district heating (NDH) plant using a PWR in Russia and Switzerland, the pressure of the heat application system (steam or hot water) is kept higher than that in the secondary loop. In application of Nuclear Heating Reactors in China, the secondary loop pressure is kept higher than in the primary loop and the heating system.

2.3.4.3. Selection criteria of materials of the interface equipment

(1) Multi-stage flash (MSF)

No difference seems to exist in the required materials of MSF, whether is combined to a single purpose (water production only) or dual purpose (cogenerating) conventional power plant or with a nuclear unit. Also there seems to be no impact on the materials for the nuclear system. The brine heater (BH) which is an inherent component of MSF serves as an efficient "pressure reversal" barrier.

However, in such cases where an additional "pressure reversal" barrier, namely an intermediate loop for heat supply, might be required (e.g. for "heat only" reactors), the materials for such intermediate loops should be selected considering the higher pressures (about 1 bar) and temperatures (10°C–15°C), relative to MSF in normal conditions, because the "cooling medium" is pure steam/pure water rather than sea water (normal or concentrated) in the MSF brine heater.

Also, the relatively high moisture in the steam to the brine heater or the intermediate loop should be considered for materials selection (as well as for the design of the piping to the brine heater or the intermediate loop and its configuration.)

(2) Multi-effect distillation (MED)

Intermediate loop is and should be used as the coupling interface for heat supply to MED in most or all cases. This intermediate loop consists of a steam condenser, which serves as a heat source to either a water recirculation system or a steam supply system. In the case of water recirculation, either concentrated sea water is recirculated (as is the case in several designs) or pure water might be used.

The materials selection should consider the temperature range of 60°C–130°C and pressures ranging between about full vacuum (~30 mmHg abs) for short times — startups, etc. — and 3 bar, for the above mentioned flows and the corresponding atmospheres. In the case of "steam intermediate loop" the design is identical to that of the MSF intermediate loop mentioned above.

(3) Reverse osmosis (RO)

As far as materials are concerned there are no special criteria for their selection for RO connected to a nuclear plant system, whether it is purely electrically driven or pre-heated.

2.3.4.4. Monitoring practice of radioactive contamination in the heat application system

For NDH, there is a strict limit for radioactivity in the heating media, usullay slightly above the natural level. For ND:

- radioactivity levels in the potable water must not exceed the levels specified by WHO standards for drinking water;
- radioactivity levels in the brine and cooling water discharge must not exceed the levels specified in national regulations and international laws, as applicable;
- radioactivity levels in the sea water used as feedwater must not exceed the natural background level.

For both NDH and ND, radioactivity levels in the intermediate loop (between the turbine loop and the heat application loop) must not exceed the natural or preset level.

There are variation of practices to meet this criteria. Intermediate cooling media (water or steam) continuously is monitored to confirm no significant contamination. If any radioactive contamination is detected, the heat application system is isolated. The monitoring device can be also installed in the heat application system. To avoid contamination due to feedwater, the feedwater is monitored, which is practically done in the conventional single purpose nuclear plants as well. "No radioactive contamination" should be confirmed before delivery to end consumers.

2.4. High and medium temperature applications

Applications at the higher temperature end are not well proven and remain in the laboratory or in small scale demonstration. The experience with the early helium cooled high temperature gas-cooled reactors (HTGRs), the Dragon plant in the UK, the AVR in Germany and Peach Bottom in the USA was satisfactory. Problems with the later HTGRs, Fort St. Vrain in the USA and the THTR-300 in Germany, were primarily associated with problems with first-of-a-kind systems and components. There were no safety concern or problems with the basic reactor concept of helium cooling. For their large-scale deployment significant research and development is still required [3].

Extensive programmes are on-going for the application of HTGRs. Both Japan and China currently have test reactors under construction to investigate the high temperature applications of the HTGR. Initial criticality of the Japanese high temperature engineering test reactor (HTTR) is expected in mid-1998. This 30 MW(th) helium cooled reactor will be utilized to establish and upgrade the technology for advanced HTGR development, and to demonstrate the effectiveness of selected high temperature heat utilization systems. The hydrogen production system by steam reforming of natural gas is under design study, and the construction is planned to start in 2002.

The Chinese high temperature reactor (HTR-10) is scheduled to go critical in 1999. This pebble bed reactor of 10 MW(th) will be utilized to test and demonstrate the technology and safety features of the HTGR. It can provide process heat at various temperatures up to 950^0C, which can be used, for example, for heavy oil recovery and crude oil refinement process. Development of the HTGR by INET is being undertaken to evaluate a wide range of applications such as electricity generation, district heat production, combined steam and gas turbine cycle operation, and process heat for coal gasification, steam and methane reforming. The HTR-10 is the first HTGR to be licensed and constructed in China.

The Russian Federation is undertaking a design study on a modular helium cooled reactor wieth core outlet temperatures of 750–950°C. It considers providing heat at various temperature levels from below 450°C to 950°C. Information on nuclear process heat production projects in China, Japan, South Africa and Russia is summarized in Section 5.

3. TECHNICAL AND ECONOMIC ISSUES

The technical issues discussed at the meetings are summarized in this section. For a cohesive presentation, all of the relevant information is grouped into pertinent topics to provide an overview of the collective experience and suggestions for further improvement gained from the operating nuclear heat application systems. The topics are:

3.1. Siting

The siting of reactors intended for nuclear heat application will in many cases be a critical issue since the site must satisfy both the requirements of the nuclear plant and of the heat application.

Siting of nuclear plants has become a major issue, even in the few countries which are proceeding with their nuclear programmes by building new plants. Building additional units at existing nuclear sites has become standard practice lately, and opening up new sites for nuclear plants are rare. An important factor affecting site selection is the NIMBY (not in my back yard) syndrome. It promotes a trend by decision makers to choose remote but accessible locations, in order to avoid potential conflicts and public opposition. Remote siting far from densely populated areas makes it also easier to comply with regulatory requirements, which have become more and more demanding.

Economic factors favour siting as close as possible to load centers. While close siting will be advantageous for electricity generating power plants, but not critical due to the relatively low costs of electricity transmission, it is practically a necessary condition to be fulfilled for reactors designed to supply district heat or industrial process heat since it is very expensive to transport heat. Transport of desalted water costs less than transport of hot water or steam for district heat or industrial process heat; this provides some more freedom for the site selection for a nuclear desalination plant.

Some advanced reactor designs, in particular in the SMR range with improved safety features, could be perceived as acceptable for close siting by the public. They could more easily meet regulatory requirements, could allow siting relatively close to population centers and thus keep heat transmission or water transport costs at reasonable levels.

3.2. Infrastructure and construction

The implementation of a nuclear power project requires the establishment of an adequate infrastructure for construction, regulation and reliable and economical operation. A comprehensive analysis of the need and availability of an adequate infrastructure is essential for an analysis of the technical and economic viability of a nuclear heat applications complex in a region

However, once this is established, the addition of nuclear heat applications does not require a new infrastructure, other than the traditional support facilities used in fossil fuel heating, desalination or electricity generating plants. The existing nuclear heat application systems were installed in regions where a nuclear infrastructure was already established.

The fabrication of seawater desalination equipment does require modern industrial technologies, but they are not more difficult to operate than other traditional industrial facilities.

3.2.1. Construction planning

It is quite normal that nuclear power plants (NPP) are built first and nuclear heat applications systems (NHAS), such as district heating (DH), industrial process heat (IPH), or desalination systems (DS), are added later on. In other cases the heat application systems were completed first. They were initially operated with fossil energy and later with nuclear energy. At some multi-unit stations, nuclear units and heat application systems were constructed in several phases. Some examples of phased start of operation are:

Country	NPP site	Start of operation	Heat application	Start of operation
Bulgaria	Kosloduy	1974–82	DH	1990
Canada	Bruce	1977–87	IPH	1981
Hungary	Paks	1983–87	DH	1978
Japan	Ohi	1979	DS	1973–76
Slovakia	Bohunice	1985	DH	1987
Switzerland	Beznau	169–71	DH	1983–84

In other cases, e.g., in Russia, DH systems were designed and constructed simultaneously with the NPPs. In Canada and China, prototype demonstration NPPs for DH were designed for simultaneous start of operation, but none have been operated on an industrial scale.

Concerning the flexibility of increasing the heat output or water production capacity at a later date, additional heat exchanger(s) would be required in most cases. If spare heat exchangers are installed at the first licensing stage, it would increase the cost of the plant and the product. On the other hand, design modifications at a later stage would require relicensing of the nuclear plant. Thus, heating and distillation desalination systems are not flexible with respect to production capacity.

However, an RO system without pre-heating is flexible and can supply more potable water by simply adding RO trains and electrical power lines. No design modifications of the nuclear power plant are required.

Relevant issues include:

- The basic requirements, such as space for appropriate nuclear heat applications, should be considered and included in the original design of the NPP.
- Multi-purpose plants should be designed as a package to reduce costs, since retrofitting to NPPs is expensive and requires re-licensing.
- It is preferable to stagger the construction schedules of NPPs and NHASs to avoid conflicts. Usually the NPP is built first, but in some arid areas it may be preferable to first construct a DS plant with an auxiliary energy source to provide water for the construction of the NPP.

3.2.2. Service systems

Many aspects, such as regulatory requirements, materials, construction, maintenance, technology and safety culture, are quite different between the NPP and a conventional NHAS. Regulations require nuclear-grade materials and services for the NPP that are much more expensive than non-nuclear industrial materials and services that are sufficient for the NHAS. However, many existing facilities make economical use of common services for both the NPP and the NHAS, for example:

- In Aktau, the NPP is operated jointly with conventional power plants to supply heat. The nuclear desalination complex is used both for production of potable water and feedwater for the nuclear plant.
- In India, there is a common seawater intake facility for the nuclear plant and a multi-stage-flash (MSF) and reverse osmosis (RO) desalination plant.
- At the Paks complex in Hungary, there are two kinds of control rooms for the nuclear power plant and district heating plant. An additional common control and emergency shutdown room is provided for the general overview of all relevant information from the NPP and DH systems.
- Often the NPP and the coupled DH grid have a single common dispatch center for control of the entire heating network. There may, however, be a separate make-up water preparation and chemical control system for the heating grid.

3.2.3. Personnel and facilities

In general, operational staff, facilities and spare parts qualified for use at the NPP can be used for the NHAS — but not the other way around. With proper regulations and operating procedures, the use of common personnel and facilities for fabrication and repair is feasible. This is cost effective and avoids organizational rivalry. With proper planning and training, operating staff and control room facilities can be combined. For example, in Kazakhstan and India, NPP and DS facilities are operated and managed as a single industrial complex.

A relevant and cost-effective guideline is to utilize the existing expertise of the NPP staff and the auxiliary facilities as much as possible also for the other part(s) of the complex.

3.3. Operation

When a nuclear heat application system uses only a relatively small fraction of the nuclear heat from the NPP, there is no need to modify the operating procedures of the NPP. In DH applications, the distribution grid is often the third or fourth heat transport circuit and has relatively little impact on the primary circuits of the NPP, as long as the DH grid is not a major heat sink for the primary circuits. However, routine monitoring of the operational parameters is mandatory, since some of these have to be maintained within regulatory limits to meet the stringent safety requirements of a nuclear power plant environment, for example:

- Limits of radioactivity and chemical substances in the water in NHAS distribution circuits.
- Limits of radioactivity in DS loops and in potable water.
- Inlet and outlet temperatures and pressures in distribution circuits, and
- Pressure and temperature differences in heat exchangers.

- Distillation processes

The distillation processes, such as MSF or MED, can produce potable water with very low salinity regardless of the fluctuation in the quality of the sea water. An increase of salinity or decrease of temperature does not affect the quality of the water produced. Therefore, precise control systems are not needed to keep quality within required standards for potable water for all seasons. The system controllability and maneuverability of distillation systems are thus more advantageous than RO systems with regard to fluctuations of feedwater quality. More detailed information on distillation processes can be found in [10].

- Reverse osmosis (RO)

The operating conditions of an RO system are designed in accordance with the feedwater quality, such as its salinity and temperature. It is not easy to vary the operating conditions manually over a wide range to adjust for fluctuating seawater quality. The tolerable operational range of the membranes is limited, and a fine control system is needed for economic optimization of the operating conditions of RO systems. These can be done usually by a computer aided control system. More detailed information on membrane desalination technologies is contained in Ref. [10].

Relevant issues include:

If a nuclear heating reactor (NHR) is to be installed to provide nuclear heat to an existing DH system, an upgrading of the entire system should be carried out. It could include closure or upgrading of existing boiler plants, reconstruction of heat transport/distribution lines and the addition of inter-links in the networks. A special dispatch center is also needed to provide centralized control and regulation of the heating system. The NHR should operate in a base-load mode and the conventional boilers in a load-following manner.

Desalination plants should be operated at their full design capacity as long as possible, since this is the most economic mode of operation. When the NPP shuts down, it depends on the local situation if a full size backup auxiliary system is required to provide electricity and steam to continue the production of potable water. Economic alternatives include water storage and partial capacity backup systems.

3.4. Design precautions and plant reliability

In addition to the usual safety requirements for a nuclear plant, design precautions to prevent radioactivity carry-over are required for any system that makes direct or indirect use of nuclear technology. Often non-nuclear auxiliary systems at the site of a nuclear power complex must meet the high standards of nuclear components. For DH and DS applications, the economics of the distribution systems dictate siting nuclear plants in the vicinity of heavily populated areas with resultant increase in safety precautions against the release of radioactivity or other accidents. Thanks to the design precautions as well as good maintenance and operation, there have been no accidents at nuclear heat application facilities so far.

3.4.1. Design precautions

In general, the operation of the NPP is not influenced by auxiliary DH systems since the flow to applications such as the DH grid can be promptly isolated from the NPP by gate valves.

There are stringent limits on any ingress of radioactive material from the NPP side to the NHAS side. This requirement is often met by installing an intermediate heat transfer cycle with a higher pressure than both the pressure in the NPP side of the heat exchanger and in the NHAS. The allowable concentrations of radionuclides in the distribution grid and the doses to consumers, workers and the public are thus maintained below specified limits, and the NPP is protected from possible contamination from the NHAS.

In particular, design precautions must be taken to prevent the transfer of radioactivity into the potable water. These design features will include one or more barriers to radioactive substances, usually in the form of a leak-tight intermediate heat transfer loop at a pressure higher than that of the steam extracted from the turbine cycle of the nuclear plant. These loops are continuously monitored, and isolation devices are provided to separate potentially contaminated areas.

When potable water is produced by multiple nuclear desalination units, leakage of radioactivity from one of the units could contaminate all of the water inventory. In such a case, additional design precautions should be implemented so that any accidental radioactivity release in one unit does not contaminate the water produced by the others. Options to minimize the potential impact would be to install multiple heat exchanger in parallel between the nuclear and desalination systems, and/or to install multiple storage tanks for the potable water.

The environmental assessment is specific to each country and is governed by the national regulatory authorities. Public perception of the safety of the technology and environmental concerns can seriously affect the viability of nuclear heat applications.

The experience with the design precautions implemented to safeguard against the potential release of radioactivity have been satisfactory and very encouraging. For example at the Paks complex in Hungary, the ingress of radioactive material into the DH system is prevented by multiple heat exchangers and the steam system of the NPP is monitored continuously. In Japan, operation of the DS plants is to be terminated under any circumstances that could lead to ingress of radioactivity into the DS. Monitoring systems are installed in the secondary loops of the NPP. The systems are monitored continuously and, if any one of the systems would indicate ingress of radioactivity, immediate action is taken to isolate the DS or shut down the NPP when necessary.

Relevant issues include:

- The interface between the NPP and the NHAS should provide at least one additional barrier to prevent radioactive ingress from the NPP side into the DH distribution grid or the DS. This could be an intermediate heat transfer loop at a higher pressure than that of the steam supplied from the NPP.
- For a dedicated nuclear heating reactor where the DH grid is the heat sink, the transient behavior of the distribution grid must be anticipated and analyzed for the design of the NHR.
- Heat extraction from the NHR should be a stable process and potential loss of flow in the distribution grid must be considered in the design and safety analysis of the NHR.
- Inadvertent increase or loss of heat consumption by the grid should be considered as a design basis event due to its potential impact on the operation of the NHR.
- Special stringent requirements should be established for NHRs. A comprehensive study of the impact on the environment, during construction, operation (including possible accidents) and decommissioning should be carried out. The results should be analyzed thoroughly by a special independent review team designated by the regulatory body.
- For DS plants, the impact of natural background radioactivity of sea water should be evaluated.

3.4.2. Plant reliability and maintenance

In Slovakia, the DH systems have operated reliably over the past eight years and heat delivery has increased approximately 2.4 times from 1988 to 1995. The maintenance period is one to two weeks per year. Only basic maintenance of the main control systems of the hot water line in the exchanger station is required. Heat exchangers and water circulation pumps are prepared for repair before the annual maintenance starts.

In Kazakhstan, the nuclear desalination complex at Aktau has been in operation for 25 years. For optimum seawater desalination operation, additional purification of the distillate is necessary to improve water quality. The intermediate sodium circuit at the BN-350 NPP provides reliable separation between the radioactive primary sodium and the steam generator. The steam is produced at 50 bar whereas the pressure in the protective argon gas cavities of the reactor is 1.9 bar on the primary side and 2.5 bar on the secondary side.

In Hungary at the PAKS complex, all of the eight turbines of the NPP are connected to a steam pipe feeding the three heat exchangers for the DH system. This arrangement provides sufficient redundancy for the DH system. The maintenance period is reduced by using a computerized network. A diagnostic system is prepared for DH maintenance. There has been satisfactory experience with nuclear heat applications with only one unscheduled shut down lasting 1 to 2 hours per year. The heat exchangers had to be replaced due to perforation by corrosion. The heat exchangers are monitored by endoscope before the maintenance period. Heat exchangers and hot pipelines are tested.

In Japan, the three distillation plants at Ohi-I and II have performed well. Most of the time a single distillation plant can satisfy the water needed for the two nuclear power reactors. In the distillation plant the heating steam does not directly contact the primary coolant of the nuclear reactor. The heating steam is generated in an independent loop that is connected to the secondary circuit of the NPP. The steam converter acts as a defense barrier. Overhaul and inspection are done once a year according to the supplier's recommended procedures.

In Canada at the Bruce Energy Center, economical nuclear heat has been provided to an adjacent industrial park reliably without interruption from 1981 to 1998.

Relevant issues include:

- It is important to install monitoring and diagnostic systems for the main components of a DH system before it starts commercial service.

- Continuous monitoring of the radioactivity level is essential to take prompt action.
- Inspection levels of radioactivity should be set at a conservative fraction (e.g. approximately one third) of the threshold levels.
- For reliability of supply to a DH system, a fossil-fired boiler with a minimum acceptable capacity should be installed. In Slovakia, a backup system with a capacity of 1/5 of the main heat exchanger is installed at the DH complex.
- A multi-stage flash (MSF) desalination plant can be operated at 65% to 110% load after adjustment of the temperature profile and brine flow quality. However, DS plants should not be operated in the same manner as load following power generation plants.
- A complete maintenance of DH systems should be performed at two year intervals.

3.5. Economics

3.5.1. General

Essential conditions for the viability of nuclear heating systems and nuclear desalination are that the price of nuclear heat or desalted water has to be competitive with alternative supply options, and that the heat or water production must be safe and reliable.

At cogeneration plants, which constitute the vast majority of nuclear heat supplying plants, the main product is electricity. Heat delivery amounts usually to less than 10% of the total thermal power. The cost of the nuclear electricity will thus be decisive for the economic viability of a nuclear project, with heat supply as a byproduct. The energy cost attributable to heat supply is usually calculated from the lost electricity generation and the electricity generation cost (power credit method).

Nuclear heat applications were found economic in a number of cases, but not under all circumstances. Cogeneration has thermodynamic advantages which usually lead to low energy cost, but the heat transport system and other necessary installations may be quite costly. Among other conditions, a large and fairly steady demand for heat or desalted water is essential for economic nuclear heat application.

Essential cost parameters are:

- Construction cost of the nuclear heat supply system, including heat transport to the consumer
- O&M cost attributable to heat supply
- Energy cost attributable to heat supply
- Interest/discount rate
- Heat output capacity
- Load factor
- Plant life.

Cost estimates for the construction of new plants may be derived from bids, recent construction experience or from cost estimates by prospective suppliers. Since only a few NPPs and no nuclear desalination plant were constructed recently, cost estimates for new nuclear desalination plants contain considerable uncertainties. This applies in particular to advanced small and medium nuclear plant (SMR) designs for which no construction and operation experience exists, but which appear to meet the requirements for coupling with desalination plants better than the large evolutionary NPPs deployed in major industrialized countries.

Concerning the load factor, most industrial heat consumers and seawater desalination plants will require rather steady heat supply around the year. All other factors being equal, they will thus have a higher load factor and lower heat cost than DH systems which require seasonal heat supply.

3.5.2. Cost experience

In Canada, several industrial complexes have been established near the Bruce Nuclear Power Development (BNPD) to benefit from the economical source of nuclear heat from the eight-unit nuclear power complex. In addition, steam from the nuclear power stations was used in the nearby heavy water production industrial plant. The nuclear heat cost is significantly lower than heat from natural gas or other fossil fuels.

Nuclear district heating systems are operating under market conditions in Hungary, Slovakia and Switzerland. In Trnava town in Slovakia nuclear heat is cheaper than fossil-based heat and is gaining increased market share. There is potential for the price of nuclear heating to be further reduced in the future. In Hungary, heat from the NPP is sold in bulk to a district heating distribution company (DCTS Ltd). The heat content is measured at the border of the NPP and DCTS pays for the heat content. In Switzerland, the cost of nuclear district heating to the consumer is about 0.07 to 0.095 CHF/kW·h(th). This is higher than the cost of oil-firing, but is accepted by most customers as a contribution to environmental protection. In Russia and in the Ukraine, the cost of nuclear district heating is subsidised, leading to low costs for the consumer.

In spite of the largely positive experience with operating nuclear DH plants, new nuclear DH projects were found uneconomic in several cases (e.g. in the Czech Republic) because of the high construction cost of hot water pipelines from the NPP to the consumers.

3.5.3. Costing approaches

At single purpose plants, the cost of nuclear heat or desalted water can be calculated in a similar way as the cost of electricity, i.e. by dividing the total costs attributable to heat or water production during a defined time period by the heat or water produced during the same period. The time period could be a year or (preferably) the plant life:

$$C_{lev} = \frac{PWC_h}{PWH}$$

C_{lev} levelized cost of nuclear heat or water
PWC_h present worth of lifetime costs attributable to heat production
PWH present worth of nuclear heat or water

Details on present worth calculation and the economic evaluation of nuclear plants are contained in the IAEA Guidebook on the Economic Evaluation of Bids for NPPs [13].

At cogeneration plants, costs have to be allocated to the products, i.e. electricity and heat or desalted water. The overall costs of the plant can be divided into:

- Costs exclusively attributable to electricity generation
- Costs exclusively attributable to heat or water production
- Common costs of electricity generation and heat or water production.

Examples for the first cost category are the turbogenerator and condenser, for the second category the intermediate loop and the balance of the heating or desalination system, and for the third category the nuclear steam supply system. These common costs can be allocated to the products in several ways. A frequently applied method is to calculate the energy cost attributable to heat supply from the lost electricity generation and the electricity generation cost (power credit method). Other approaches include the heat (or water) credit method and the exergetic method [11]. If the cogeneration plant is designed for base heat load and the turbine is not designed to use additional steam at times of low or no heat demand, the cost of nuclear heat can also be calculated from the incremental cost of raising more steam than required for electricity generation. These incremental costs are lower than costs according to the power credit method since the cost of the turbine plant are not included.

More details on the economic evaluation of nuclear electricity and heat cogeneration are contained in [12] and in the IAEA Guidebook on the Economic Evaluation of Bids for NPPs [13].

3.5.4. CDEE spreadsheet

The IAEA has developed a computer code for economic analyses using the power credit method, named the Cogeneration/Desalination Economic Evaluation (CDEE) Spreadsheet [12]. The CDEE methodology is suitable for economic evaluations and screening analyses of various desalination and energy source options. The spreadsheet includes simplified models of several types of nuclear/fossil power plants, nuclear/fossil heat sources, and both distillation and membrane desalination plants. Cost and performance data are incorporated as default values so that the spreadsheet can be quickly adapted to analyze a large variety of options with little new input data required.

The spreadsheet serves three objectives:

(a) Calculation of the levelized cost of electricity and desalted water as a function of quantity, site specific parameters, energy source and desalination technology.
(b) It enables side-by-side comparison of a large number of design alternatives on a consistent basis with common assumptions.
(c) It enables quick identification of the lowest cost options for providing specified quantities of desalted water and/or power at a given location.

However, the spreadsheet is based on simplified models. For planning an actual project, the project costs must be assessed more accurately based on more substantive information including project design and specific vendor data.

Potable water costs from cogeneration plants are calculated in the spreadsheet applying the power credit method. This method is based on the concept that the electricity equivalent of steam and/or electricity provided to the seawater desalination plant could have been sold to the grid, and that this loss in electricity sales revenues should be charged to the water cost. The power credit is calculated by multiplying the reduction in electrical output by the levelized unit electricity generation cost of an equivalent single purpose power plant. Applying the power credit method, the potable water produced is credited with all economic benefits of co-production. This method is often applied in situations with a well established electricity market, and potable water considered as a by-product. Other methods, including the water credit method, the caloric method and the exergetic method, will allocate at least some benefits of co-production to electricity. The most equitable cost allocation method from the thermodynamic viewpoint is the exergetic method described in Ref. [11].

3.5.5. Results of IAEA cost studies

Economic analyses performed for IAEA studies [10, 14] resulted in about the same levelized water production costs for both nuclear and fossil-fuelled plants. The nuclear desalination costs were found in Ref. [10] to range from around US $0.9 to US $1.0/m^3 for large dual purpose plants with cogeneration of electricity and heat delivered to multi-effect distillation (MED) plants. Water costs from multi-stage flash (MSF) plants would be about US $0.3 higher, and from reverse osmosis (RO) plants coupled contiguously with the energy source about US $0.1/m^3 lower than from MED plants. RO plants generally yield lower water production costs than distillation plants. There exists hardly any economy of scale beyond water production capacities of 300 000 m^3/d.

In isolated locations with small electrical grids, stand-alone RO plants powered by diesel generators would yield water costs of about US$ 1.0/m^3. If a distillation process is preferred, water costs would be at least around US $\$1.5/m^3$ for a small MED plant being supplied by a small dedicated heat-only reactor or fossil fuelled plant.

Since these studies were performed, significant progress was made concerning the further development of both RO and MED processes which are expected to lead to significant water cost reductions for both nuclear and fossil fired plants. It remains to be seen to what extent these improvements will materialize at commercial plants.

4. OPERATING EXPERIENCE

Extensive experience with nuclear heat applications has been accumulated, mainly in the low temperature ranges, with district heating, seawater desalination and process heat supply [1]. Tables I, III and V list nuclear plants which provide heat for these purposes. Tables II, IV and VI list nuclear reactors under construction or in an advanced design stage which will supply nuclear heat in the near future.

Experience with nuclear district heating has been gained in Bulgaria, Germany, Hungary, Russia, Slovakia, Switzerland and Ukraine. Application for seawater desalination is another field with some operational experience and good prospects. The Shevchenko complex (now Aktau in Kazakhstan) using BN-350, a liquid-metal-cooled fast reactor, went into operation in 1973, and since then has provided both electricity and heat for the production of potable water. Several nuclear power plants in Japan have been desalting a few thousand m^3/d each for feedwater make-up as well as for household use in the plants. Canadian CANDUs at the Bruce Nuclear Power Development have been providing heat to the heavy water production plants for more than twenty years. The agricultural industries in the adjacent energy center are also provided with heat by these reactors.

There is about 500 reactor-years of operational experience in district heating, industrial process heat and seawater desalination using heat from nuclear power reactors. There appear to be no major technical or safety problems with nuclear heat application systems. The design precautions to prevent the carry-over of radioactivity into district heating systems, potable water or other products have been proven to be effective.

On the other hand, applications at the higher temperature end remain in the laboratory or in small scale demonstration. Although some experience is available from early stage HTGR developments, significant research and development is still required for their large scale deployment. Extensive programmes are ongoing for the application of HTGRs. Both Japan and China currently have test reactors under construction to investigate high temperature applications of the HTGR. Initial criticality of the Japanese High Temperature Engineering Test Reactor is expected in 1998 and of the Chinese High Temperature Reactor (HTR-10) in 1999.

4.1. Nuclear district heating systems

The technical viability of using nuclear heat for the supply of hot water and steam for district heating and other industrial processes has been demonstrated both in dedicated nuclear heating plants and in heat and power cogeneration plants. Nuclear heat application systems have been in service for over 20 years without any serious problems. Information on operating plants and current projects are summarized in Tables I and II.

Dedicated nuclear heating systems were designed and some built and operated in Canada, China and Russia. The plants in Canada and China were for demonstration purposes, whereas the Russian plants are to supply settlements in northern Russia.

TABLE I. OPERATING NUCLEAR HEATING PLANTS

Country	Plant type or name	Location	Application	Phase	Start of operation Reactors / Heat	Power (MW(e) net)	Heat output (MW(th))	°C at Interface (Feed/Return)	Remarks
Bulgaria	Kozloduy 5-6	Kozloduy	E, DH	Commercial	1987-91 1990?	2×953	20	150/70	
China	NHR-5	Beijing	-/DistHeating	Experiment	1989 1989	0	5	90/60	
Hungary	PAKS 2,3,4	Paks	E, DH	Commercial	1983-87	4 × 433	30	130/70	4 × V213 WWER
Russia	Research reactor	Obninsk	DistHeating	Commercial	1954- 1976	0	10-20	130/70	
Russia	EGP-6	Bilibino	E, DH	Commercial	1974-81	4 × 12	133	150/70	
Russia	WWER-1000	Novovoronezh[a]	E, DH	Commercial	1964-	2 × 385 1 × 950	230	130/70	
Russia	WWER-1000	Balakovo	E, DH	Commercial	1986-93	4 × 950	230	130/70	
Russia	WWER-1000	Kalinin	E, DH	Commercial	1985-87	2 × 950	230	130/70	
Russia	WWER-440	Kola	E, DH	Commercial	1973-84	4 × 410	55		
Russia	BN-600	Belojarsk	E, DH	Commercial	1981	600	220	130/70	
Russia	RBMK-1000	St-Petersburg	E, DH	Commercial	1974-81	4 × 1000	ca 170	130/70	
Russia	RBMK-1000	Kursk	E, DH	Commercial	1977-86	4 × 1000	ca 170	130/70	
Slovakia	Bohunice-3,4	Bohunice/Trnava	E, DH	Commercial	1985 1987	2 × 410	240	150/70	2 × V213 WWER
Switzerland	Beznau 1,2	Beznau	E, DH	Commercial	1969-71/1983-84	2 × 360	80	128/50 (water)	
Ukraine	Rovno 1,2	Rovno	E, DH	Commercial	1981-82/ 1982	2 × 370	58	130/70	
Ukraine	Rovno 3	Rovno	E, DH	Commercial	1987 1987	950	233	130/70	
Ukraine	South Ukraine 1-3		E, DH	Commercial	1983-89/ 1976	3 × 950	2 × 151 1 × 232	150/70	

[a]Unit 1 was taken out of operation in 1988, unit 2 in 1990.

TABLE II. NUCLEAR HEATING PLANT PROJECTS

Country	Plant type or site	Location	Application	Phase	Start of operation	Power (MW(e))	Heat output (MW(th))	Temp (°C) at Interface (Feed/Return)	Remarks
Bulgaria	Belene	Belene	E, DH	Design		2x1000	400	150/70	
China	NHR-200	Daqing City	-/DistHeating	In construction	2000	–	200	90/ca 60	
Russia	RUTA	Apatity	-/DistHeating/Air conditioning	Design		–	4 x 55	85/60	
Russia	ATEC-200		E, DH	Design		50–180	70–40	150/70	
Russia	VGM		Process Heat	Design		–	200	900/ca500	
Russia	KLT-40	Floating	E, DH & Desalination	Design		35	110	140/125 (Intermediate circuit)	
Russia	AST-500	Voronez	-/DistHeating	Construction suspended		–	500	150/70	
Russia	AST-500	Tomsk	-/DistHeating	Construction suspended		–	500	150/70	

Atomic Energy of Canada Limited (AECL) constructed and tested a 2 MW(th) SLOWPOKE demonstration reactor in 1980s at the AECL Whiteshell Laboratories in Manitoba. A 5 MW(th) Test Heating Reactor (NHR-5) was commissioned in China and has been in operation since 1989 supplying heat to the INET Center, near Beijing. Russia has operated a 10 MW(th) heating reactor at Obninsk for more than 20 years and has developed the technology of the nuclear district heating reactor AST-500. Construction of AST-500 reactors at the city of Voronez and Tomsk started in 1980s but was suspended in the early 1990s.

Most of the nuclear reactors supplying heat at present are cogeneration plants, in which the main product is electricity and only a small fraction of the reactor power is used for nuclear heat applications. The heat output capacities range from about 20 to 240 MW(th). For thermodynamic reasons, the extraction of low temperature/low pressure steam from the turbine leads to low heat cost to consumers, provided the cost of distribution is not a dominating factor. Nuclear cogeneration plants for electricity and district heating were built and operated in Bulgaria, Germany, Hungary, Russia, Slovakia, Switzerland and Ukraine. Almost 500 reactor-years of quite satisfactory and encouraging operational experience has been accumulated. The plants have operated safely and reliably.

The NPP Kozloduy in Bulgaria has supplied heat to the town of Kozloduy since 1990. The Kozloduy NPP consists of four WWER reactors of 408 MW(e) and two WWER reactors of 953 MW(e). The reactors had some problems of safety and reliability; however, no relevant problems with district heating were reported. The Greifswald NPP in Germany (former GDR) has supplied up to 180 MW(th) for district heating until its decommissioning in 1990.

The Paks Nuclear Power Plant (Hungary) consisting of four units of the Soviet design WWER-440 type V-230 is supplying heat to the town of Paks. The secondary circuit steam leaving the steam generators is monitored by gamma detectors. The water pressure in the heat exchanger is kept higher than the steam pressure to prevent contamination of the hot water system.

The Bohunice Nuclear Power Plant in Slovakia produces electrical energy and low temperature heat for heating and industrial purposes. The plant generates approximately 12 TW·h electrical energy annually. It is shown by operating experience that the heat supply from the nuclear power plant to the town of Trnava is reliable, safe and economically competitive.

The district heat extraction from the Beznau NPP (2 × 360 MW(e) PWR) has been operated reliably and successfully since its commissioning in 19983/84. The peak heat load is about 80 MW(th), leading to about 10 MW(e) loss. The district heating system supplies about 2100 private, industrial and agricultural consumers through 35 km of main piping and 85 km local distribution pipes. Since the consumers are spread over a relatively wide area, the heat cost to them is higher than with individual oil heating, but is accepted as a contribution to environmental protection by most of them.

The most extensive experience with district heat supply from nuclear cogeneration plants has been gained in in the Russian Federation. A research reactor at Obninsk has supplied heat since 1976 and is still in operation. The NPPs of Bilibino, Belojarsk, Balakovo, Kalinin, Kola, Kursk and Sankt Petersburg are supplying heat from steam turbine bleeders through heat exchangers to district heating grids of towns with typically about 50 000 inhabitants, situated between 3 and 15 km from the NPP site. The heat output capacities range from about 50 to 230 MW(th).

In Ukraine, the NPPs Rovno and South Ukraine have supplied heat to district heating grids since 1982 and 1983, respectively. The design characteristics and operating experience are similar to the NPPs and heating grids in the Russian Federation. Figure 6 illustrates the scheme of nuclear heat application from the South Ukraine NPP Unit No. 1.

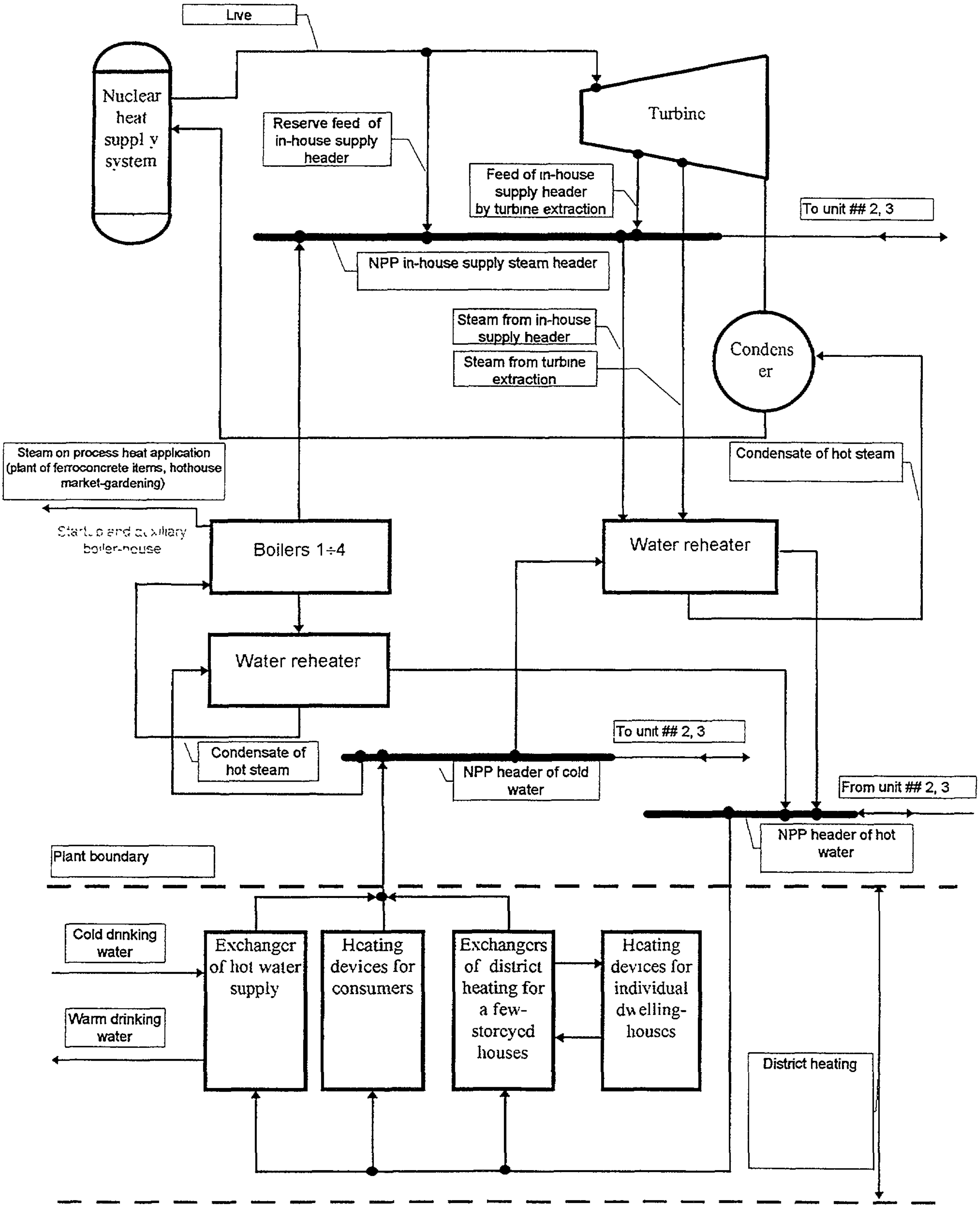

Fig. 6. Scheme of nuclear heat application from the South Ukraine NPP Unit No. 1 (source: V.K. Khokhlov, NNEGC "Energoatom", Kiev, Ukraine).

The design precautions to prevent the transfer of radioactivity into the district heating grid network have proven to be effective in all these countries. These design features include one or more barriers to radioactive substances, e.g., in the form of a leak-tight intermediate heat transfer loop at a pressure higher than that of the steam extracted from the turbine cycle of the nuclear plant. These loops are continuously monitored, and devices are provided to isolate potentially contaminated areas.

District heating systems require a backup heat source when the main heat source is unavailable. Therefore, at least two nuclear power units or a combination of nuclear and fossil-fired units are used for district heating grids.

4.2. Nuclear desalination systems

Integrated nuclear desalination plants have been operated in Japan and Kazakhstan for many years (Table III). Relevant experience has also been gained in Israel and USA. A new project of a nuclear desalination plant has been launched at Kalpakkam, India (Table IV).

In Aktau, Kazakhstan, the liquid metal cooled fast reactor BN-350 has been operating as an energy source for a multi-purpose energy complex since 1973, supplying regional industry and population with electricity, potable water and heat. The complex consists of a nuclear reactor, a gas and/or oil fueled thermal power station, and multi-effect-distillation (MED) and multi-stage-flash (MSF) desalination units. Figure 7 is a flow diagram of Mangyshlak Atomic Energy Complex. The sea water is taken from the Caspian Sea. The nuclear desalination capacity is about 80 000 m^3/d. A part of this capacity has now been decommissioned.

In Japan, all of the nuclear power plants are located at the seaside. Several nuclear power plants of the electric power companies of Kansai, Shikoku and Kyushu have seawater desalination systems using heat and/or electricity from the nuclear plant to produce feedwater for the steam generators and for on-site supply of potable water. MED, MSF and RO desalination processes are used. The individual desalination capacities range from about 1000 to 3000 m^3/d. Figure 8 gives an illustration connecting the NSSS and the desalination facility at the Ohi NPP.

In Israel, the experience gained with the simulation of a nuclear desalination process is also of interest. In Ashdod, Israel, an integrated plant was designed and built to simulate the coupling of a MED plant with a nuclear reactor. A low temperature, horizontal tube, multi-effect (LT-HTME) unit with a production capacity of 17 400 m^3/d was coupled to an old 50 MW(e) oil fired power plant. The steam conditions supplied to the LT-HTME desalination unit were modified to simulate intermediate heat transfer loops (flash loops) from the back-pressure turbine condenser of a nuclear power plant. The Ashdod unit operated continuously for over a year as a demonstration plant and fulfilled its design goals. It was stopped in 1983 because of the high price of oil and the low efficiency of the 50 MW(e) unit [15].

In USA, an integrated RO desalination plant was operated successfully for on-site water supply at the Diablo Canyon NPP since 1985 [4].

For technical and economic reasons, the desalination facilities share the water intake and outfall systems, the main control room and some plant staff with those of the power plant. This practice is similar as at desalination plants coupled to fossil fired cogeneration plants.

The experience gained so far with nuclear desalination is encouraging. Specific features of interest to future operators and designers are summarized below. Most of these features are also relevant for non-nuclear desalination processes.

TABLE III OPERATING NUCLEAR DESALINATION PLANTS[a]

Country	Plant name	Location	Application	Start of operation Reactors / Desal	Power (MW(e))	Water capacity (m^3/day)	Remarks
Japan	Ikata 1,2	Ehime	Electricity/ Desalination	1977–82 1976	566	2 000	PWR/MSF
Japan	Ikata-3	Ehime	Electricity/ Desalination	1994 1993	566	2 000	PWR/RO (2 × 1000 m^3/d)[b]
Japan	Ohi-1,2	Fukui	Electricity/ Desalination	1979 1973–76	2 × 1175	3 900	PWR/MSF (3 × 1300 m^3/d)
Japan	Ohi 3,4	Fukui	Electricity/ Desalination	1991–93 1989	2 × 1180	2 600	PWR/RO(2 × 1300 m^3/d)
Japan	Genkai 4	Fukuoka	Electricity/ Desalination	1997 1988	1180	1 000	PWR/RO
Japan	Genkai-3, 4	Fukuoka	Electricity/ Desalination	1995–97 1992	2 × 1180	1 000	PWR/MED
Japan	Takahama	Fukui	Electricity/ Desalination	1985 1983	870	21 000	PWR/RO
Japan	Kashiwazaki	Niigata	Electricity/ Desalination	1985 (not served)	1100	1000	BWR/MSF[c]
Kazakhstan	BN-350	Aktau	Electricity/ Desalination/ Industrial & district heat	1973 ca 1963	120	120 000	FBR/MED,MSF
USA	Diablo Canyon 1,2	San Louis Obispo	Electricity/ Desalination	1985–86 1985	2 × 1100	2 200	PWR/2 stage RO

[a]All nuclear desalination plants with the exception of Aktau/Kazakstan are for on-site water supply. Some desalination plants were first operated with conventional energy
[b]Brackish water desalted
[c]This desalination facility was not put into service after construction, because other fresh water resources were made available

TABLE IV NUCLEAR DESALINATION PROJECTS

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

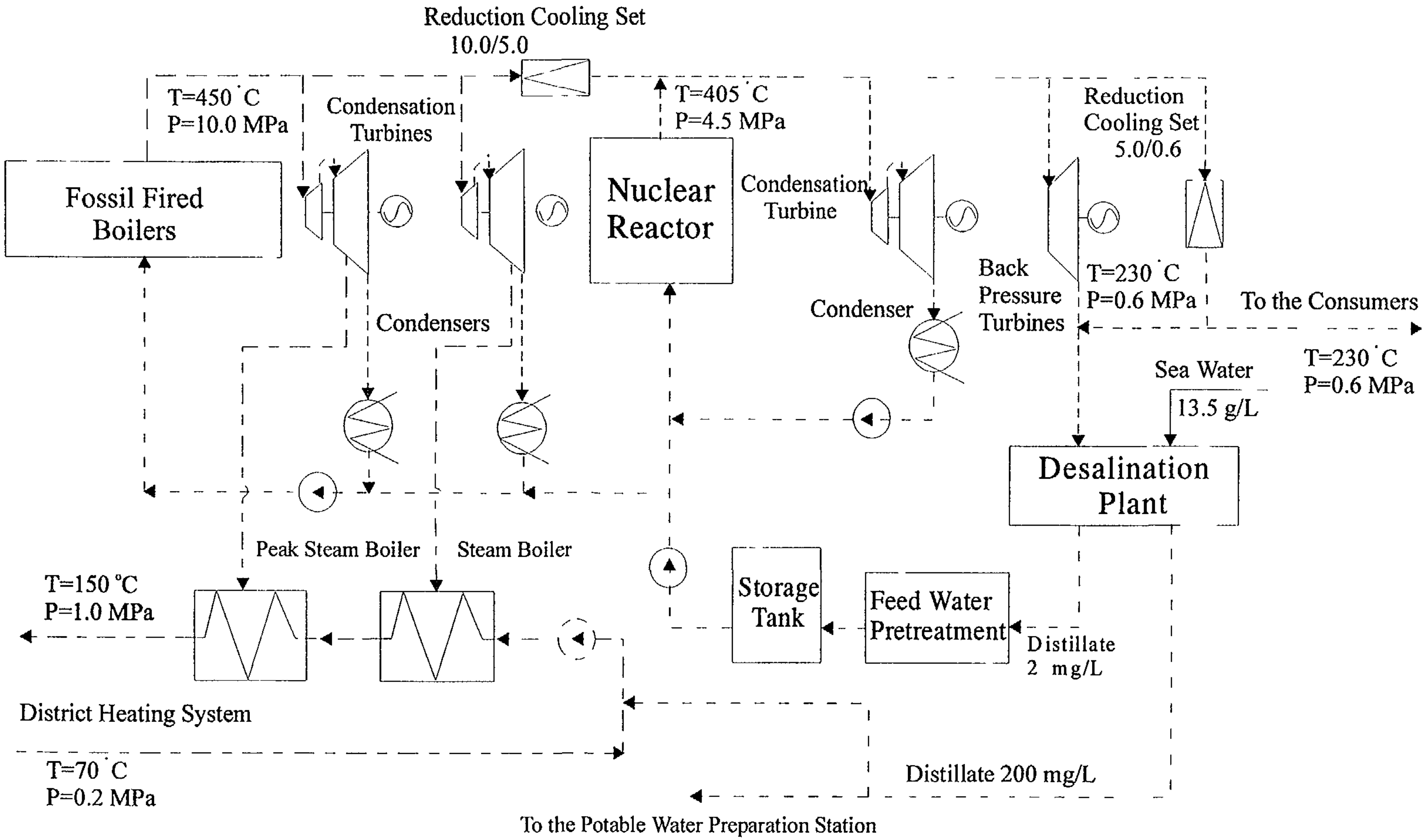

FIG. 7. Principle flow diagram of Mangyshlak Atomic Energy Complex.

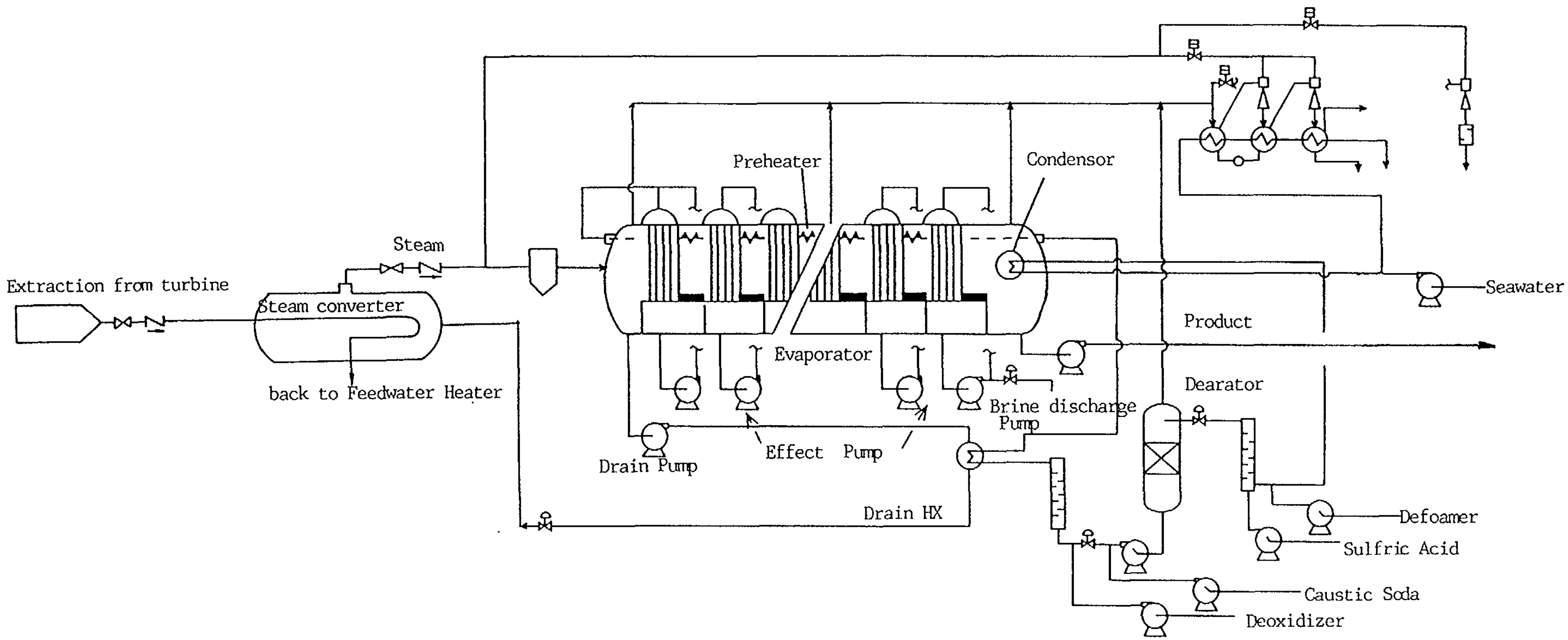

FIG. 8. Schematic illustration of connection between NSSS and desalination facility at Ohi, Japan.

4.3. Nuclear process heat systems

Experience with nuclear process heat systems was gained in Canada, Germany and Switzerland (Table V).

In Canada, steam from the Bruce Nuclear Power Development (BNPD) is supplied to heavy water production plants and to an adjacent industrial park at the Bruce Energy Center (BEC)[3]. BNPD, the world's largest nuclear steam and electricity generating complex, has operated successfully for over 20 years. It includes eight CANDU nuclear reactors with a total output of over 7200 MW(e), the world's largest heavy water plant (HWP), and the Bruce Bulk Steam System (BBSS). The BBSS, capable of producing 5350 MW(th) of medium pressure process heating steam, was built to supply the HWPs from the four 848 MW(e) units of the Bruce A complex at BNPD. Each of the four 2400 MW(th) reactors can supply high pressure steam to a bank of 6 heat exchanger (24 in total) which produce medium pressure steam for the HWP and site services. The normal capacity is approximately 1680 kg/s of medium pressure steam from the reactors with 315 kg/s emergency backup available from oil fired boilers.

In order to meet HWP reliability criteria, the steam system is designed and operated to ensure a maximum steam supply interruption of just a few minutes in winter and 4 hours in summer. One of the 3 oil fired boilers is kept on hot standby and condensate pumps are supplied with uninterruptible power backed by gas turbine standby generators. During its entire 17 years of operation, the HWP has never suffered a loss of emergency steam.

Projected heavy water demand was less than originally forecast giving the BBSS significant long term steam spare capacity. Although the spare capacity is expected to be lower during the next few years because of extensive planned maintenance, the BBSS still provides access to nuclear heat energy sufficient to support substantial industrial development at the adjacent BEC. An industrial distribution supply system feeding the customers was constructed beyond the HWP in the form of a 5 km long, 0.91 m (36 inches) diameter steam line with a 0.46 m (18 inches) condensate return line. There are essentially three barriers between the steam the customer uses and the nuclear plants.

Customers are on an interruptible steam supply system and the operations of the nuclear plants are run independently of the needs and concerns of the customers as they are purchasing interruptible steam. However, should the four-unit Bruce A go down to a one-unit reliability, the oil backup system is brought on line so that the customer's steam supply is not interrupted. The BEC steam prices consist of only a flat rate per thousand pounds steam delivered. The cost is significantly lower than costs of heat energy from burning natural gas, which is the closest competitor.

The six private industries currently established at the park are (1) a plastic film manufacturer, (2) a 30 000 m^2 (7.5 acres) greenhouse, (3) a 12 million liters/year ethanol plant, (4) a 200 000 ton/year alfalfa dehydration, cubing and pelletizing plant, (5) an apple juice concentration plant and (6) an agricultural research facility.

In Germany, the Stade NPP PWR, 1892 MW(th), 640 MW(e) supplies steam for a salt refinery which is located at a distance of 1.5 km since December 1983. The salt refinery requires 45 t/h process steam with 190°C at 1.05 Mpa. This represents a thermal power of about 30 MW and is 1.6% of the thermal output of the NPP. The steam supply from Stade NPP is designed for 60 t/h, of which the remaining 15 t/h are used for space heating at the Schilling oil fired power station nearby, and for an adjacent tank storage facility. Since 1983, the steam supply by NPP Stade had very high time availability, and the operating experience with process steam extraction is very good.

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[3] Although Unit 2 of the Bruce A NPP was laid up in 1995, and the heavy water plant and Units 1,3 and 4 of Bruce A will be laid up in 1998, it is described here, since the experience with BEC is informative. More details on the Bruce Energy Center is contained in the Annex.

In Switzerland, the 970 MW(e) PWR of Gösgen provides process steam for a nearby cardboard factory since 1979. The process steam (1.37 MPa, 220°C) is generated in a tertiary steam cycle by live steam from the PWR. It is then piped over a distance of 1750 m to the cardboard factory. After condensation, it returns as 100°C hot water to the PWR. A maximum process steam extraction of 22.2 kg/s is possible which represents a thermal output of about 54 MW or about 2% of the total thermal power of the PWR.

Experience with nuclear process heat systems at high temperatures are limited, although several projects are currently underway (Table VI).

4.4. Database development for non-electrical applications of nuclear energy

In response to the recommendations made by the Advisory Group Meeting (AGM) in October 1996, the IAEA has initiated preparatory work to establish a database which should contain the experience at all nuclear plants which provide heat for non-electrical products. As most of these nuclear plants are cogenerating, the new database has been designed in such a way that new data tables containing data relevant to heat applications be added to the existing tables in the Power Reactor Information System (PRIS) [2]. The expanded database is being designed to encompass plant design characteristics, performance data, operating and outage statistics and other relevant information on nuclear plants for non-electrical applications. Depending upon the different forms of nuclear heat applications, there are both commonalties and some differences in the detailed design approach.

The data collection of plant characteristics is now in progress. Typical data are seen in Tables VII and VIII[4]. The new database will encompass data at operating plants and those under construction.

5. CURRENT DEVELOPMENT ACTIVITIES IN MEMBER STATES[5]

In the field of nuclear seawater desalination, there are currently relevant ongoing development activities in China, India, the Republic of Korea, Morocco and the Russian Federation. Morocco and China have taken a step through the governmental agreement for a pre-project study on using a small 10 MW(th) heating reactor from China for the production of about 8000 m^3 of potable water per day in Morocco via an MED process. In China, a nuclear desalination plant, based on the 200 MW(th) nuclear heating reactor with a capacity of 150 000 m^3/d is being studied for Dalian City in China. In the Republic of Korea, the design of a 330 MW(th) advanced reactor is in progress as a cogeneration demonstration nuclear plant for seawater desalination. The Republic of Korea offers this project for international cooperation. In India a demonstration project is being launched and civil work at the site started to connect a hybrid desalination system (MSF-RO) to a PHWR at Kalpakkam. India offers this project for international cooperation. In the Russian Federation a small floating nuclear desalination plant is under development using a nuclear reactor originally developed for icebreakers. This project is offered for international cooperation.

In the district heating field, the Russian Federation has accumulated extensive experience. This experience is being reflected in the improved design concept of a local district heating source and the heat supply system. Restarting of the construction work of the site of Voronez and Tomsk is expected, both using AST-500 reactors. Some other cogeneration plants for district heating are also reported to be foreseen for replacing existing plants which are approaching the endl of their design life time in the near future. A project ANGSTREM of a modular, transportable nuclear power-and-heating reactor is launched using a fast nuclear reactor cooled by a lead–bismuth eutectic. The possible evolution of the project may include new applications — seawater desalination or refrigeratory plants. In China a demonstration plant using an NHR-200 is being planned in the North East region of China.

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[4] Upon collection of all data available, prototypes of tables will be documented for dissemination in an appropriate form to all Member States.

[5] Operating plants are not mentioned here.

TABLE V. OPERATING NUCLEAR PROCESS HEAT PRODUCTION PLANTS

Country	Plant name	Location	Application	Start of operation Reactors / Heat	Phase	Power (MW(e))	Heat delivery (MW(th))	°C at Interface (Feed/Return)	Remarks
Canada[a]	Bruce-A	Bruce	Process Heat	1977–87 1981	Commercial	4x848 4x860	5350		D2O production and six industrial heat customers
Germany	Stade	Stade	Electricity/ Process Heat	1983	Commercial	640	30	190/100	Salt refinery
Switzerland	Goesgen	Goesgen	Electricity/ Process Heat	1979 1979	Commercial	970	25	220/100	Cardboard factory

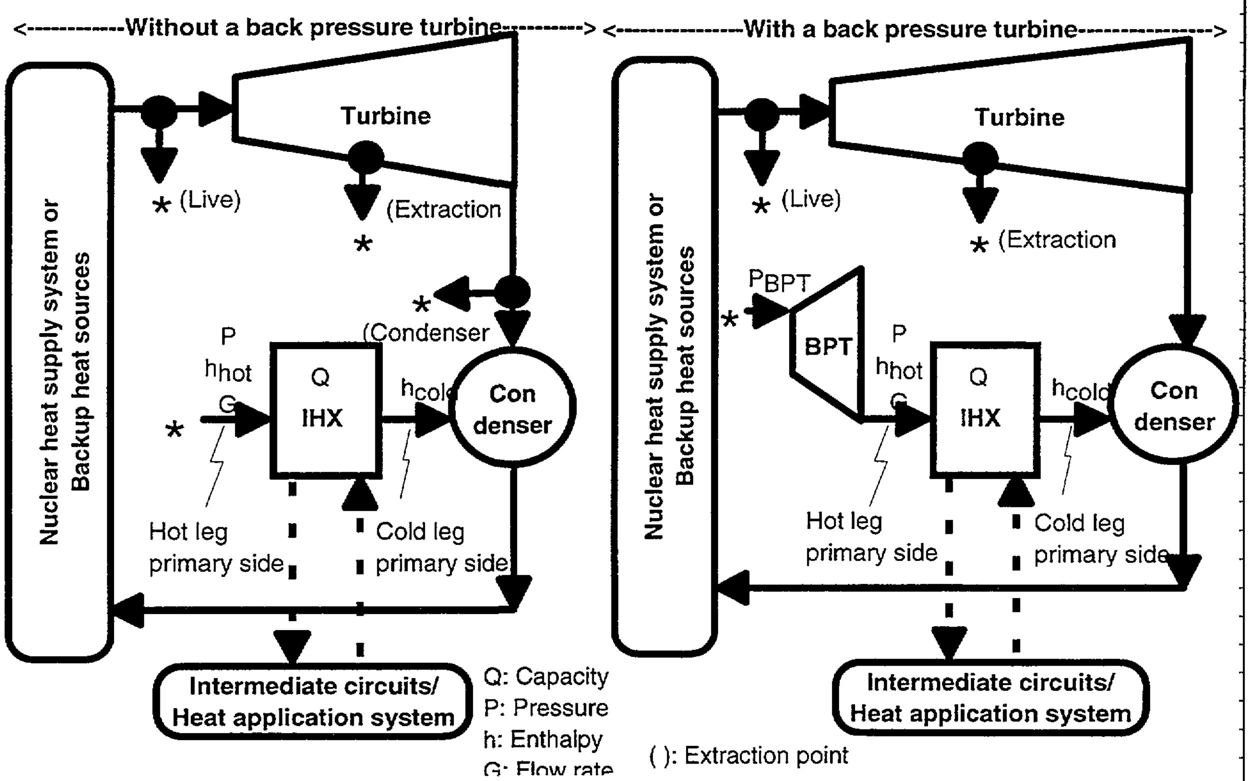
[a]Unit 2 of Bruce A was taken out of service in 1995, units 1, 3 and 4 will be taken out of service in spring 1998.

TABLE VI. NUCLEAR PROCESS HEAT PRODUCTION PROJECTS

Country	Plant name	Location	Application	Start of operation Reactors / Heat	Phase	Power (MW(e))	Heat delivery (MW(th))	°C at interface (Feed/Return)	Remarks
China	HTGR-10	Beijing	Electricity/ Process Heat	Criticality 1999?	Construction	0	10	700–950/250	Experiments for HTR technology development
Japan	HTTR	O-arai	Process Heat	Criticality 1998?	Construction completed	0	30	950/395	Experiments for HTR technology development
Russia	VGM		Process Heat		Design	0		750–950	

TABLE VII. EXAMPLE OF GENERAL DATA FOR NUCLEAR HEAT APPLICATION PLANT

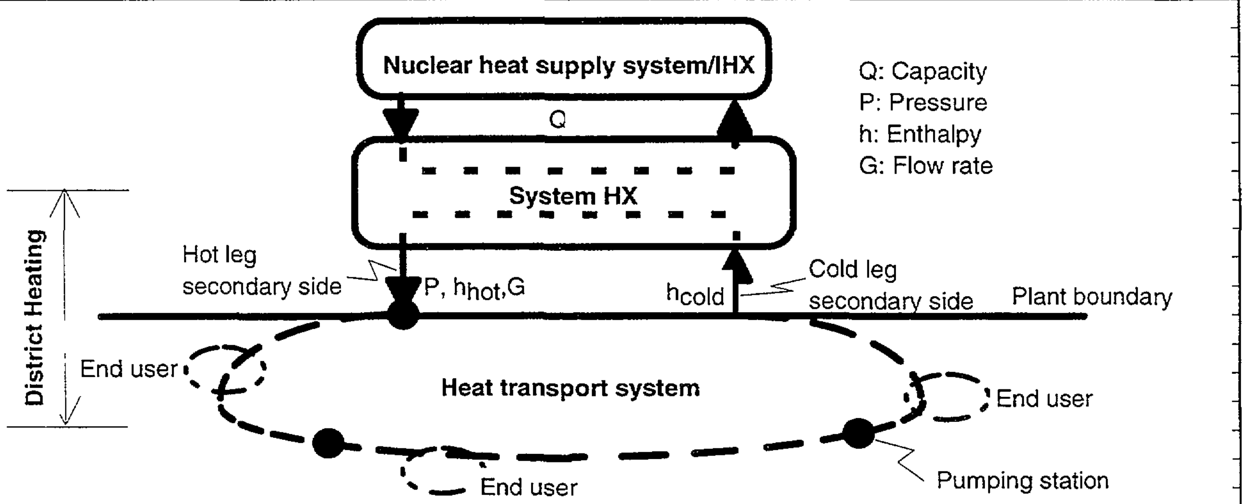
1	**General data**					
			Item	**Unit**	**Design/option**	**Remarks, examples, options**
	1.1	**Heat sources**	Names of connected other nuclear units		PAKS-2/3/4	
	1.2		Backup heat source types and its capacity	–/MWth	-	
	1.3		Heat transport medium		steam	
	1.4		Medium extraction points in the NPP BOP circuit		Extraction (HTP and LPT)	
	1.5		Pressure at BPT inlet: P_{BPT}	Pa	-	
	1.6		Number of intermediate circuits	–	-	
	1.7	**Primary side of IHX**	Capacity	MWth	42	
	1.8		Flow rate: G	kg/s	12.3/7.1	max/nom?
	1.9		Pressure: P	MPa	0.425/0.108	max/nom?
	1.10		Hot leg enthalpy: h_{hot}	kJ/kg	2462.9/2813.2	max/nom?
	1.11		Cold leg enthalpy: h_{cold}	kJ/kg	568.1/426.2	max/nom?



If an intermediate circuit exists, go to "2 Intermediate circuit",					
Otherwise go to		**3A for process heat,**			
		3B for district heat, or			
		3C for desalination.			

TABLE VIII. EXAMPLE OF DESIGN CHARACTERISTICS OF A NUCLEAR DISTRICT HEATING PLANT

			Item	Unit	Design/option	Remarks/examples
	District heating					For each application
3B	3B.1	Heat connection	Service date	yy,mm	83/01/01	
	3B.2		Number of heat connection points	–	1	
	3B.3		Total capacity of heat connections	MWth	42	
	3B.4		Individual capacity of heat connections	MWth	0/42/23	Minimum, maximum, average
	3B.5		End user interface	–	indirect	
	3B.6	System HX	Number of HXs	–	2	
	3B.7		Capacity	MWth	23.3/18.7	
	3B.8		Heat transfer area	m^2	125/200	
	3B.9		Material (tube/plate/shell)	–	steel/steel	
	3B.10	Secondary side	Heat transport medium	–	hot water	
	3B.11		Medium inventory	ton	ca 2000	
	3B.12		Mass flow rate	kg/s	165	
	3B.13		System pressure	MPa	1.4	
	3B.14		Hotleg enthalpy	kJ/kg	547.1	nominal
	3B.15		Cold leg enthalpy	kJ/kg	293.5	nominal
4B	4B.1	Heat transport system	Number of pumping station	–	1	
	4B.2		Longest delivery distance from the plant	km	6	
	4B.3		Nominal pipe diameter at NPP boundary	–	350 mm	
	4B.4		Pipe material	–	steel	stainless steel
	4B.5		Pipe location	–	above	Option: under; above ground
	4B.6		Pipe insulation method	–	glass wool ca 10cm	Glass wool of 1 cm thickness
	4B.7	Monitoring	Radioactivity monitoring method	–	-	
	4B.8		Leakage detection method	–	-	monitoring
	4B.9	Chemistry control	Chemistry control method	–		
	4B.10		pH range	–	8 to 9, 8.2	Minimum, maximum, nominal
	4B.11		Conductivity	μS/cm	-	
	4B.12		Deaeration method	–	manual	



Extensive programmes are ongoing for the application of HTGRs. Both Japan and China currently have test reactors under construction to investigate the high temperature applications of the HTGR. Initial criticality of the Japanese High Temperature Engineering Test Reactor (HTTR) is expected in mid-1998. The Chinese High Temperature Reactor (HTR-10) scheduled to go critical in 1999. The Russian Federation is undertaking design study on a modular helium cooled reactor at the temperature of 750–950°C (Section2.4). Indonesia and South Africa are also interested in utilizing high temperature gas reactors in process heat production. The Indonesian Natuna Project is planning to use an HTGR to produce useful products such as methanol, methane and syngas by CO_2 conversion as well as to desalinate sea water for the complex [4]. The state electric utility of South Africa, Eskom, is finalizing a technical and economic evaluation of a helium cooled pebble bed reactor of German design with a power output of ~228 MW(th) for consideration in increasing the capacity of their electrical system [16].

In order to facilitate these development activities, the IAEA is coordinating, as part of its functions, research programmes, the so-called Co-ordinated Research Projects (CRPs) with participation of interested Member States. In nuclear desalination, a CRP on "Optimization of the Coupling of Nuclear Reactors with Desalination Systems" has started. There are two ongoing CRPs in the field of HTGRs: "Heat Transport and After Heat Removal for GCRs under Accident Conditions", and "Evaluation of HTGR Performance."

6. CONCLUSIONS

The main conclusions related to *nuclear heat application experience* are:

- Positive experience has been gained in the use of nuclear heat for district heating and industrial process heat. The existing systems are economical and socially accepted. This is a good basis for further development.
- The technical and safety related experiences gained from nuclear district heating systems can be directly used for the development of nuclear desalination plants based on distillation processes.
- There are no major technical or safety problems with seawater desalination complexes using heat and/or electricity from nuclear power plants.
- To prevent radioactive contamination, one or more protective barriers are required between the nuclear plant heat transport circuit and the heat application circuit. The heat transport circuits must be monitored continuously and adequate separation devices have to be incorporated in their design.
- If a nuclear infrastructure already exists for the nuclear power plant, the additional infrastructure required for the implementation of cogeneration is practically identical to that required for conventional heat applications.
- If only a small fraction of the reactor power is used for nuclear heat applications, there is very little perturbation on the operation of the nuclear plant. However, if a major fraction of the reactor power is used for nuclear heat application, the effects of all transients anticipated in the application systems have to be considered in the design and safety assessment of the nuclear plant. This may necessitate the installation of sophisticated control systems.
- International information exchange is necessary to further improve the operation, maintenance and the reliability of existing and future nuclear heat application systems.

The following is essential *for future nuclear heat application systems*:

- Obtain public acceptance and establish a nuclear infrastructure.
- Access to technology and the cost of transmission and distribution grids must be considered in evaluations of the viability of nuclear heat applications.
- For a comprehensive comparative assessment of heat supply from a nuclear or a fossil fueled plant, the social costs related to the environmental impacts should be considered, including the emission of NO_x, SO_x and CO_2 from fossil power plants.
- Adequate safety precautions should be adopted for nuclear plants designed for district heating or seawater desalination, since these plants would be built not too far from populated areas to keep the cost of product distribution low.

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ABBREVIATIONS

BEC	Bruce Energy Center
BNPD	Bruce Nuclear Power Development
BOO	build–own–operate
BOT	build–operate–transfer
BWR	boiling water reactor
CANDU	Canada deuterium–uranium (reactor)
CDEE	cogeneration/desalination economic evaluation
CHP	cogeneration of heat and power
DH	district heating
DS	desalination system
ED	electrodialysis
FAO	Food and Agriculture Organization of the United Nations
FBR	fast breeder reactor
FOAK	first of a kind
GCR	gas cooled reactor
GOR	gain–output ratio
HP	high pressure
HR	heating reactor
HTGR	high temperature gas cooled reactor
HTME	horizontal tube multi-effect distillation
HWP	heavy water plant
HWR	heavy water reactor
IACRS	Inter-Agency Committee on Radiation Safety
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
IDA	International Desalination Association
IEA	International Energy Agency (OECD)
IPCC	Intergovernmental Panel on Climate Change
LCD	litres per capita per day
LMR	liquid metal cooled reactor
LT-HTME	low temperature horizontal tube multi-effect distillation
LP	low pressure
LWR	light water reactor
ME	multiple effect evaporation
MED	multi-effect distillation
MSF	multi-stage flash distillation
MVC	mechanical vapour compression
ND	nuclear desalination
NDH	nuclear district heating
NHAS	nuclear heat application system
NHP	nuclear heating plant
NHR	nuclear heating reactor
NPP	nuclear power plant
NPT	Treaty on the Non-proliferation of Nuclear Weapons
NSSS	nuclear steam supply system
NUSS	Nuclear Safety Standards (IAEA)
O&M	operation and maintenance
OECD	Organisation for Economic Co-operation and Development
OIP	Options Identification Programme (for nuclear desalination)
PHWR	pressurized heavy water reactor
ppm	parts per million
PRIS	Power Reactor Information System (IAEA)
PWR	pressurized water reactor

RO	reverse osmosis
TDS	total dissolved solids
TG	turbine generator
VC	vapour compression
VTE	vertical tube evaporator
WHO	World Health Organizatio
WWER	water/water energy reactor (Russian design)

Annex

EXPERIENCE WITH NUCLEAR PROCESS HEAT APPLICATION IN CANADA[1]

1. THE BRUCE ENERGY CENTRE

The Bruce Energy Centre (BEC) is an industrial park serviced with steam from one of the world's largest nuclear energy developments. As a result of co-operation between 3 levels of government, the provincial power utility and private companies, the BEC has grown into an imaginative demonstration of sustainable development. By 1996, 6 industries had established at the BEC.

By using nuclear generated heat, the BEC employs a clean, indigenous, plentiful and low cost energy resource to process and increase the value of local renewable raw materials. Present raw materials are mainly farm crops. It is also expected that by-products of certain BEC industries will serve as inputs for others.

BEC industries areprimarily in the food and special chemicals sectors, using well known processes, e.g. fermentation, distillation, evaporation and dehydration, leveraged by modern biotechnology and a major energy cost advantage. BEC demonstrates on a large scale how integrated energy/industrial systems can achieve economic development without depleting resources or emitting atmospheric pollutants such as CO_2.

The BEC has become a model for sustainable development to secure our common future. While almost all the externally supplied heat for industry in the world today is derived from burning hydrocarbons, producing large amounts of CO_2, nuclear energy is an alternative with the following advantages:

(1) lower long term average and marginal cost,
(2) no emissions of "greenhouse" or "acid" gases, and
(3) very large energy reserves.

The supply of nuclear generated heat to industry is feasible as a co-product from electrical power generation. The cost of steam or hot water pipelines restricts its application to areas within economically determined distances of generating stations. It is generally most applicable, therefore, to new industrial development which can be sited suitably close to existing or planned nuclear power plants.

Looking ahead, there is a potential to site new industrial facilities suitably close to nuclear power plants. In this way, a significant portion of the world's industrial heat energy could eventually be nuclear and this would be an effective way to minimize global emissions of greenhouse and acid gases. The infrastructure created by such developments would integrate well with other proposed clean energy futures, such as thermochemical hydrogen production and alternate uses of the carbon atom.

Ontario Hydro's Bruce Nuclear Power Development, with its bulk steam system, can potentially support the first of such developments.

2. THE BRUCE BULK STEAM SYSTEM (BBSS)

The Bruce Nuclear Power Development (BNPD) is owned and operated by Ontario Hydro, which is the largest corporation in Canada.

BNPD, the world's largest nuclear steam and electricity generating complex, has operated successfully for over 20 years and is vital to Ontario's energy security. It includes 8 CANDU nuclear reactors with a total output of over 7200MW(e), the world's largest heavy water plant, the BBSS, Training Centre, Information Centre and major warehousing, maintenance and other central facilities.

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[1]This information was made available by R. Hart, AECL, Mississauga, Ontario, Canada and compiled by IAEA staff.

The BBSS was built to supply the heavy water plant. Each of four reactors can supply high pressure steam to a bank of 6 heat exchangers (24 in total) which produce medium pressure steam for the heavy water plant and site services. The normal capacity is approximately 1680 kg/s of medium pressure steam from the reactors with 315 kg/s emergency backup available from oil fired boilers.

In order to meet heavy water plant reliability criteria, the steam system is designed and operated to ensure a maximum steam supply interruption of just a few minutes in winter and 4 hours in summer. One of the 3 oil-fired boilers is kept on hot standby and condensate pumps are supplied with uninterruptible power backed by gas turbine standby generators. The heavy water plant has thus never suffered a loss of emergency steam.

Projected heavy water demand is now much less than originally forecast giving the BBSS significant long term spare steam capacity. Although the spare capacity is expected to be lower during the next few years because of extensive planned maintenance, the BBSS still provides access to nuclear heat energy sufficient to support substantial industrial development.

The Bruce Energy Centre is a unique private sector/government/utility collaboration motivated by a group of local citizens intent on sustaining economic growth in their community upon completion of BNPD.

The enormous reject heat from the turbine condensers was identified as a potential resource to initiate an array of new industries in the fields of horticulture and aquaculture. When projections of heavy water demand declined, it was realized that the spare steam capacity represented a high quality energy resource.

An amendment was made to the legislation governing Ontario Hydro enabling the electric utility to sell heat as a primary product, and an environmental assessment was completed to authorize Ontario Hdyro to supply steam to private industry at the BEC. The Government of Ontario provided funding assistance for Ontario Hydro to construct the steam, water and sewer infrastructure for the BEC and these works were completed in the fall of 1989.

Ontario Hydro operates steam, water and sewer services at the BEC on a similar basis as it provides electricity service to the Province of Ontario, i.e. at cost, with a net income requirement only to ensure financial integrity.

BEC Limited has assembled 900 acres of land immediately adjacent to BNPD for the purpose of energy intensive industrial development. 245 acres of this land had been developed with full services to 18 lots, of which 5 have been sold. The 6 private industries currently in business are; (1) a plastic film manufacturer, (2) a 7.5 acres greenhouse, occupying 2 lots, (3) a 12 million litrelyear ethanol plant, (4) a 200,000 tonlyear alfalfa dehydration, cubing and pelletizing plant, (5) an apple juice concentration plant and (6) an agricultural research facility.

3. ECONOMIC ADVANTAGES OF THE BEC

As originally anticipated, the steam costs at the BEC are significantly lower than costs of heat from burning natural gas, which is the closest competitor in Ontario. Nuclear generated steam costs are also more predictable and resistant to inflation because a major part of the cost is already sunk in capital assets.

Most district heating systems sell central steam at a premium above gas in recognition of gains in overall efficiency, reliability, convenience and flexibility and avoidance of boiler capital and operating costs.

In contrast, BEC steam is expected to be always priced competitively with gas and firm prices are provided for 10 years, rolled forward annually. This provides customers with greater foresight on future costs than is typically available feom alternate energy sources.

BEC steam prices consist of only a flat rate per thousand lbs of steam delivered with no complicated extra tariffs or discounts for connection, service, volume, demand level, load factor, season unauthorized overrun, etc., nor any minimum bill. Ontario Hydro only reserves the right to satisfy itself that customers will have sufficient steam demand to justify connection and metering costs.

BEC is accessible to sizeable markets in Southern Ontario and the Northeast U.S. and to the World via Great Lakes shipping routes. Stable electricity, water and sewer rates and relatively low business taxes are further advantages. Finally, the area enjoys favourable socioeconomic conditions.

4. INDUSTRIES OPERATING AT THE BEC

Bruce Tropical Produce Inc.

Bruce Tropical Produce is in the forefront of Canada's horticulture industry, growing vegetables hydroponically, rooted in sterile rockwool through which dissolved nutrients are fed. This system, along with the company's crucial climate control system, is monitored and controlled by a sophisticated Priva micro-computer system developed in the Netherlands.

Employing three dozen core employees, Bruce Tropical Produce is divided into two 3.5 acres, a half-acre propagation area and a technical/packaging/shipping and administration area. The first two growing areas house 40 000 tomato plants each for an impressive total of 80 000 tomato plants on 7 acres.

Bruce Agra Dehy Inc.

Bruce Agra Dehy Inc. is an alfalfa dehydration and cubing plant, processing 90 000 metric tonnes per year, which requires approximately 30 000 acres of locally grown alfalfa. On the world market there is significant opportunity for high quality alfalfa cubes.

Environmental and economic assessment of this venture was based on the concept of capturing and using "waste energy" for production and processing activities.

St. Lawrence Technologies

St. Lawrence Technologies is designed to accommodate companies involved in research and development of technologies for the production of value-added products from agricultural crops and other biomass. St. Lawrence Technologies is the first tenant and is engaged in a wide range of research on products such as: starch, sweeteners, functional fibre food additives, starch adhesives, inulin derivatives, ethanol from grains and cellulosic materials.

A substantial laboratory and pilot plant, located at the Bruce Energy Centre, permit exploration of new opportunities, as well as scale-up of selected processes. The company offers technology, licensing, analytical services, process engineering consulting and contract R&D.

BI-AX International Inc.

Located in the Bruce Energy Centre, BI-AX International was organized in 1986. BL-AX is a processor of biaxially oriented polypropylene film. These products are used for industries that manufacture pressure sensitive tape, package items, such as potato chips, pasta, cookies, coffee; as an inner web in heavier packaging, such as dog food bags and fertilizer bags; and also for graphics laminating, such as presentation folders.

BI-AX International is also working on R&D with other companies for prototype development. This part of the business has seen the launch of many new products.

Commercial Alcohols

Commercial Alcohols Inc., formally Sunroot Energy Ltd, has been operating at the Bruce Energy Centre since September, 1989. The plant produces approximately 16 million litres per year of high grade ethanol for industrial, commercial and fuel uses. In addition to the alcohol, the plant also produces, as a by-product, about 36 000 tonnes per year of wet distillers grains, a high value cattle feed.

Commercial Alcohols uses approximately 42 000 tonnes per year of Ontario-grown corn at this plant, which operates continuously throughout the year.

Bruce Agra Foods Inc.

Bruce Agra Foods is a food processing facility designed to process raw vegetables, fruits and fruit juices into concentrates, sauces and purees to customer specifications. The company produces value-added products from locally grown crops. Their evaporators — fueled by steam energy — can concentrate 84 000 gallons of raw products per day, equating to 10 000 gallons of finished products, depending on the level of concentration required.

Bruce Agra Foods is also equipped with a set of evaporators with a range of 200–600 gallons for small production runs and for experimental runs. A modern laboratory assures a consistently high quality product in bulk tanks and 7300 square feet of warehouse space for packaged product.

Bruce Agra Foods' residual vegetable and fruit pomace is not wasted. In keeping with the Bruce Energy Centre's philosophy of integrated development, the pommace and other by-products are dried for animal feed by the dehydration plant, used as a feedstock for ethanol by the alcohol plant or composted on local agricultural land.

5. INDUSTRIES IN THE DESIGN STAGE

Ethanol Plant

Canadian Agra Ethanol is planning to construct the most technologically advanced anhydrous fuel ethanol production facility from biomass. The yearly capacity of the facility will be 100 million litres of anhydrous ethanol and 75,000 tonnes of high quality distillers dried protein. Locally grown corn and cereals, such as wheat, barley and oats are to be used as the raw feedstock.

The main product, ethanol, is an environmentally clean additive for gasoline. A blend of anhydrous ethanol with gasoline reduces carbon monoxide tailpipe emissions, carbon dioxide emissions, overall hydrocarbon emissions and decreases the ground level ozone. Ethanol fuel is a renewable resource and is a "pollution solution" using agriculture renewable feedstocks.

Canola Crushing Plant

A major canola crushing facility is also being planned by Canadian Agra, using "cold press" technology which avoids the use of chemicals, as in the more conventional press and extraction process. The planned capacity will be to crush 2000 metric tonnes of seed per day or approximately 660 000 metric tonnes per year. This will produce approximately 330 000 tonnes of canola oil and 330 000 tonnes of high protein meal for the livestock and poultry industries.

Canola oil is characterized by a very low level of saturated fatty acids that have been implicated in elevated blood cholesterol levels. Canola oil produced by the cold press method will have the added advantage of being completely chemical free.

Part II

PAPERS PRESENTED AT THE MEETINGS

II.1. DESIGN ASPECTS OF NUCLEAR HEAT APPLICATIONS

Low temperature heat applications — District heating

PROJECT OF DEMONSTRATION NUCLEAR HEATING PLANT USING POOL-TYPE WATER REACTOR

XA9848799

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Abstract

Since 1976 the first NPP equipped with a 10 MWt water-graphite reactor AM has been used for district heating and hot water supply. Development and enhancement of this experience is essentially actual and important scientific and economic task for the SRC RF-IPPE especially after the AM reactor is decommissioned in the nearest future. As one of possible variants of proceeding in this direction, construction of an NPP with a pool-type reactor RUTA is being considered. Development of a series of RUTA reactors of 10 through 50 MWt, and NHPs using a RUTA is being performed aiming at an autonomus nuclear power source for supplying heat to small settlements or cities/districts. A pool-type reactor RUTA possesses self-evident and transparent safety characteristics based on simplicity and reliability of the design, and inherent safety features using laws of nature. The NHP RUTA at the STC RF - IPPE could play a role of demonstration and could provide a more active inculcation of this reactor type to the market. The estimations performed on this basis indicate that if a 30 MWt (~ 25 Gcal/ h) NHP RUTA is coupled with the local heating system, the NHP can secure up to 85% of yearly heat production. Taking into account highly developed site infrastructures, the expenditures for the NHP RUTA project (development and construction) at the SRC RF - IPPE could be approximately estimated as US$ 10 - 15 million.

NUCLEAR HEATING DEMONSTRATION PROJECT

Utilization of heat produced by a nuclear reactor for district heating and hot water supply has been under way in the IPPE since 1976. By that time the turbine at the First NPP had been dismantled and the issue of beneficial utilization of nuclear heat of 10 MWt from the water-graphite reactor AM was raised. A decision was made to supply this heat to the IPPE's consumers, near-by industries and to the old part of the city of Obninsk for district heating.

For this purpose, a Main Water Heater (MWH) was installed in the building of the IPPE's fossil-fired Power and Heat Co-generation Plant (PHCP) next to the first NPP building. The MWH was connected to the main hot water in parallel with the then existed water boiler using the extraction steam from the 6 MWe turbine of the PHCP.

Heat from the reactor to the main hot water is transmitted using a three circuit scheme (Fig.1). The primary coolant of the AM reactor at P=6.0 MPa and t=185°C goes to two Steam Generators (SG) where the steam at P=0.29 MPa and t=175°C is generated. This steam is the heating media for the MHW.

The most concerned issue of nuclear heat utilization - exclusion of the main water radioactive contamination - is solved with reliance on the following design and operational arrangements:

- Pressure channels with annular-type fuel elements are employed in the AM reactor design which excludes the fission product penetration of from the nuclear fuel to the primary coolant.

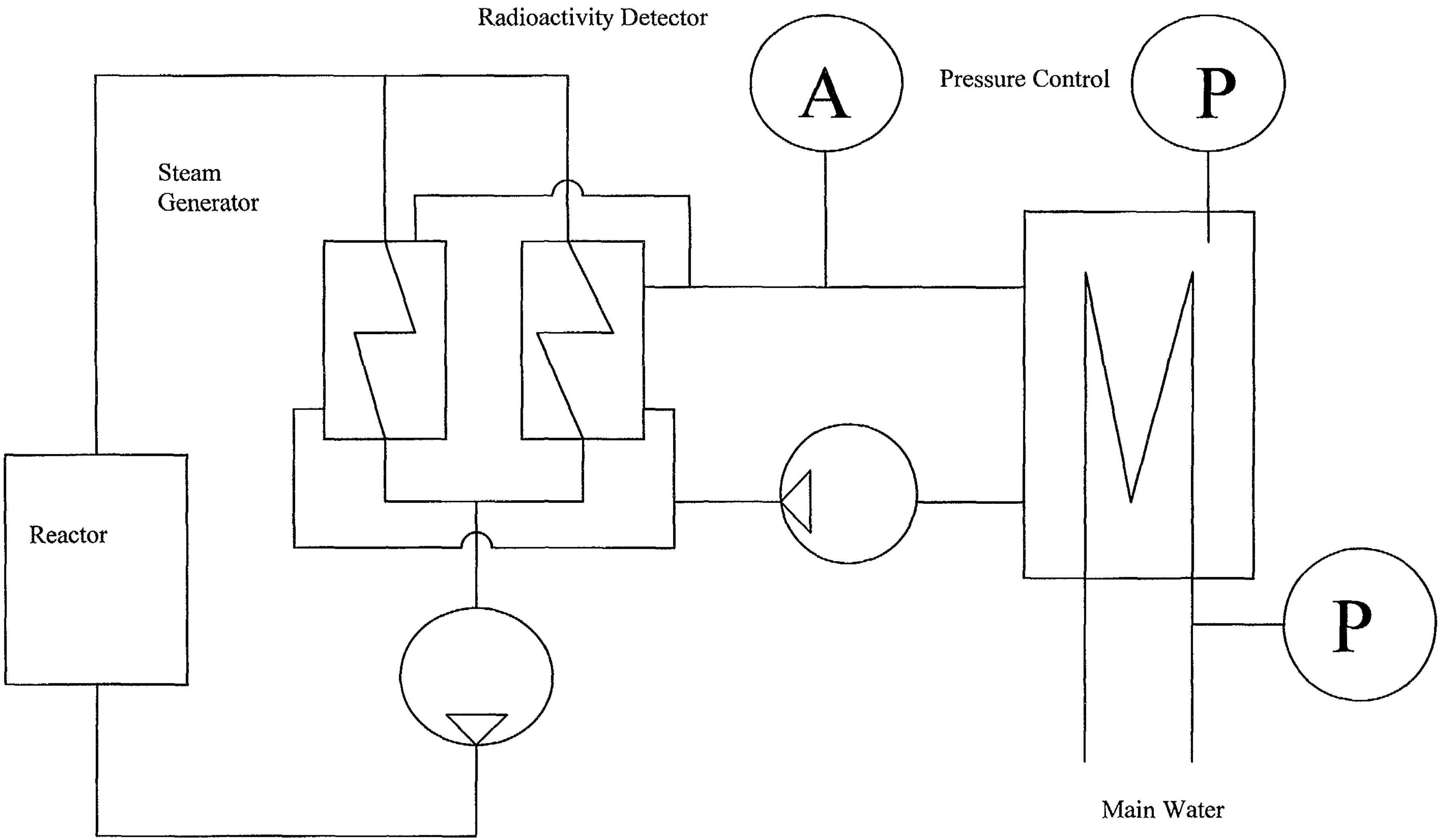

Fig.1 Schematic Diagram of theFirst NPP's Heat Utilization

- Pressure in the intermediate heating steam is kept lower than that of the main water in the MWH. The pressure of the steam from the reactor SG to the MWH its is lowered to ~ 0.1 MPa while main water pressure is maintained in the range of 0.6 - 0.9 MPa.
- The heating steam supply to the MWH is automatically stopped, when the main water pressure is reduced to the level of 0.2 MPa above the intermediate steam pressure in MWH. Simultaneously the valve to the technological condenser is opened in order to dump the intermediate steam.
- The radioactivity in the reactor secondary circuit is continuously monitored Small portion of steam is extracted from the main steam pipeline and condensed in a special heat exchanger. Gamma-activities in the condensate is monitored. Signals from the radioactivity detectors are transmitted to the radiological control panel in the reactor control room. The warning signal is displayed in the control room when the condensate activity exceeds 1.5×10^3 Bq /l (4×10^{-8} Ci/l).

The radioactivity level of scale and corrosion product deposits on the heat exchanger surfaces is measured periodically using the portable radiation control equipment during the maintenance and repair work of MWH. Throughout the operating experience of the AM reactor heat utilization for hot water production, the condensate radioactivity has never exceeded the minimun detectable level which is as high as ~44 Bq/l (1.2×10^{-9} Ci/l) and the deposits activity has been below the background radioactivity level.

Utilization of nuclear heat (steam) from the AM reactor for hot water production was a kind of accidental decision. Design parameters of the reactor were primarily selected as an energy source for a nuclear power plant. Furthermore, the heat transmission scheme is not necessarily optimal for a Nuclear Heating Plant (NHP). Nevertheless, 20 years long successful operation experience in nuclear heating has been accumulated in the SRC RF-IPPE. Development and enhancement of this experience is essentially actual and important scientific and economic task for the SRC RF-IPPE, especially after the AM reactor is decommissioned which is expected in the nearest future. As one of possible variants of proceeding in this direction, construction of an NHP with a pool-type reactor RUTA is being considered.

Development of a series of RUTA reactors of 10 through 50 MWt, and NHPs using a RUTA is being performed aiming at an autonomous nuclear power source for supplying heat to small settlements or city districts. Studies carried out by Joint Stock Company "Malaya Energetica" in recent years have shown that there are a considerable market for small nuclear heating plants in the country. It is particularly so in the Far East and North East regions of Russia [1].

A pool-type reactor RUTA [2] (Fig. 2) possesses self-evident and transparent safety characteristics based on simplicity and reliability of the design, and inherent safety features using laws of nature. From the viewpoint of safety provision, the RUTA reactor has the following important features:

- The primary reactor loop is accommodated inside a single reactor vessel (a water tank or water pool) operating under the atmospheric pressure. This eliminates the possibility of the reactor primary circuit rupture leading to a quick loss of coolant and the reactor core uncovery.
- A large bulk of reactor coolant in the reactor pool enables to provide a large accumulating capacity of the residual heat for a longer time period of up to several days, even in the case of complete malfunctions of reactor normal and emergency heat removal systems.
- Because of low values of reactor process parameters (an atmospheric pressure at the water surface in the reactor pool), no water boiling takes place in the pool.
- Low power density in the reactor core; approximately of 15 kW/ l.
- An integral layout of primary circuit: the primary heat exchangers are accommodated in the reactor pool.
- Natural circulation capability of the reactor primary coolant in all normal and accidental conditions.

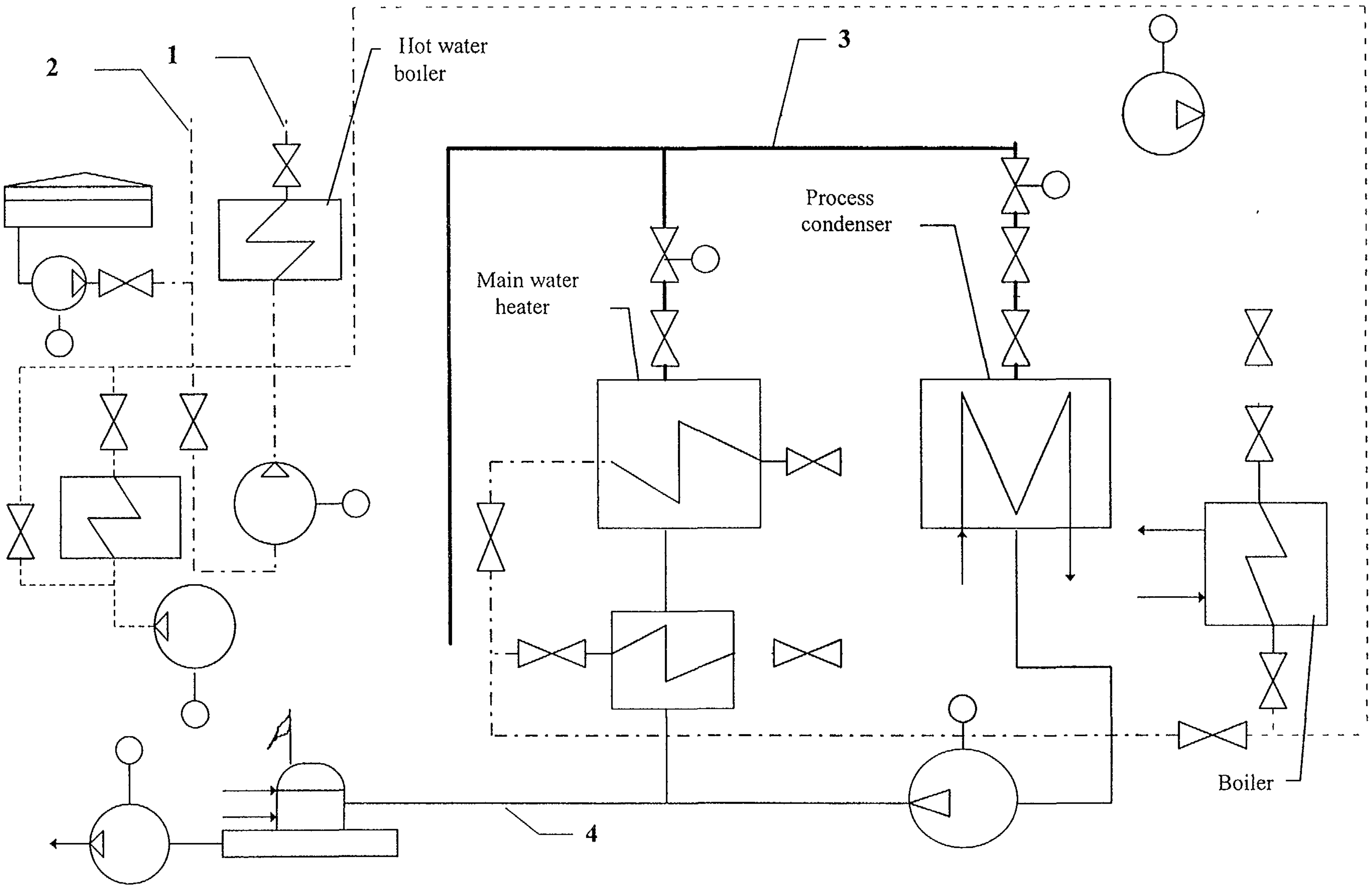

Fig.2 Heat supply system using NPP's steam
1 - Direct main water, 2 - Reverse main water, 3 - First NPP's steam, 4 - Steam condensate

TABLE I. PRINCIPAL DESIGN CHARACTERISTICS OF RUTA NUCLEAR HEATING PLANTS

Reactor thermal capacity, MW	10	20	55
Primary coolant characteristics:			
- coolant	water		
- flow rate, kg/s	115	158	522
- core temperature (inlet/outlet), °C	79/99	70/100	75/99,5
- pressure, MPa			
- core inlet	0.25	0.22	0.25
- core outlet	0.24	0.19	0.23
- primary heat exchanger inlet	0.15	0.12	0.14
Secondary coolant characteristics:			
- coolant	water		
- flow rate, kg/s	105	154	535
- heat exchanger temperature (inlet/outlet), °C	70/93	64/95	66/90
- pressure, MPa	0.39	0.39	0.39
Tertiary coolant characteristics:			
- coolant	water		
- flow rate, kg/s	95.6	167.4	525
- heat exchanger temperature (inlet/outlet), °C	60/85	60/88	60/85
- pressure, MPa	0.6÷1.0	0.6÷1.0	0.6÷1.0
Reactor:			
-water bulk in the pool, m^3	210	265	700
- core dimensions (height/equivalent diameter), m	1/0.95	1/1.23	1.2/2.03
- core power density, MW/m^3	14.1	16.8	18.6
- number of fuel assemblies	37	61	169
- number of fuel pins in the assembly	61/57/54		
- type of fuel elements in the assembly	pin type		
- fuel	UO_2		
- fuel pin outer diameter, mm	13.5		
- enrichment, %	4	4	3,6
- uranium loaded in reactor, kg	1105	1884	5942
- number of control rods	13	19	46
- number of safety control rods	3	6	9
- maximal fuel temperature, °C	600	623	640

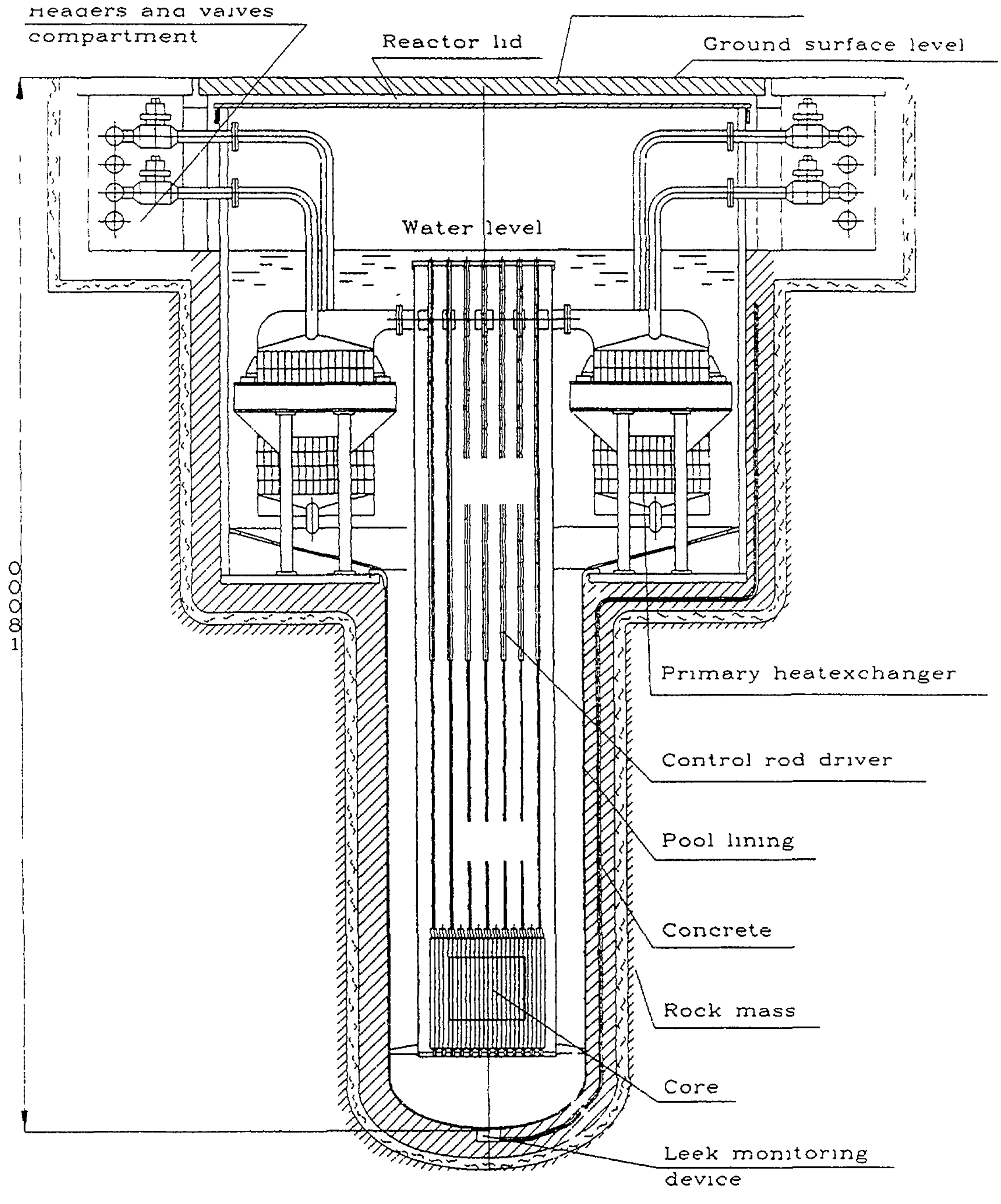

Fig.3 RUTA Energy System

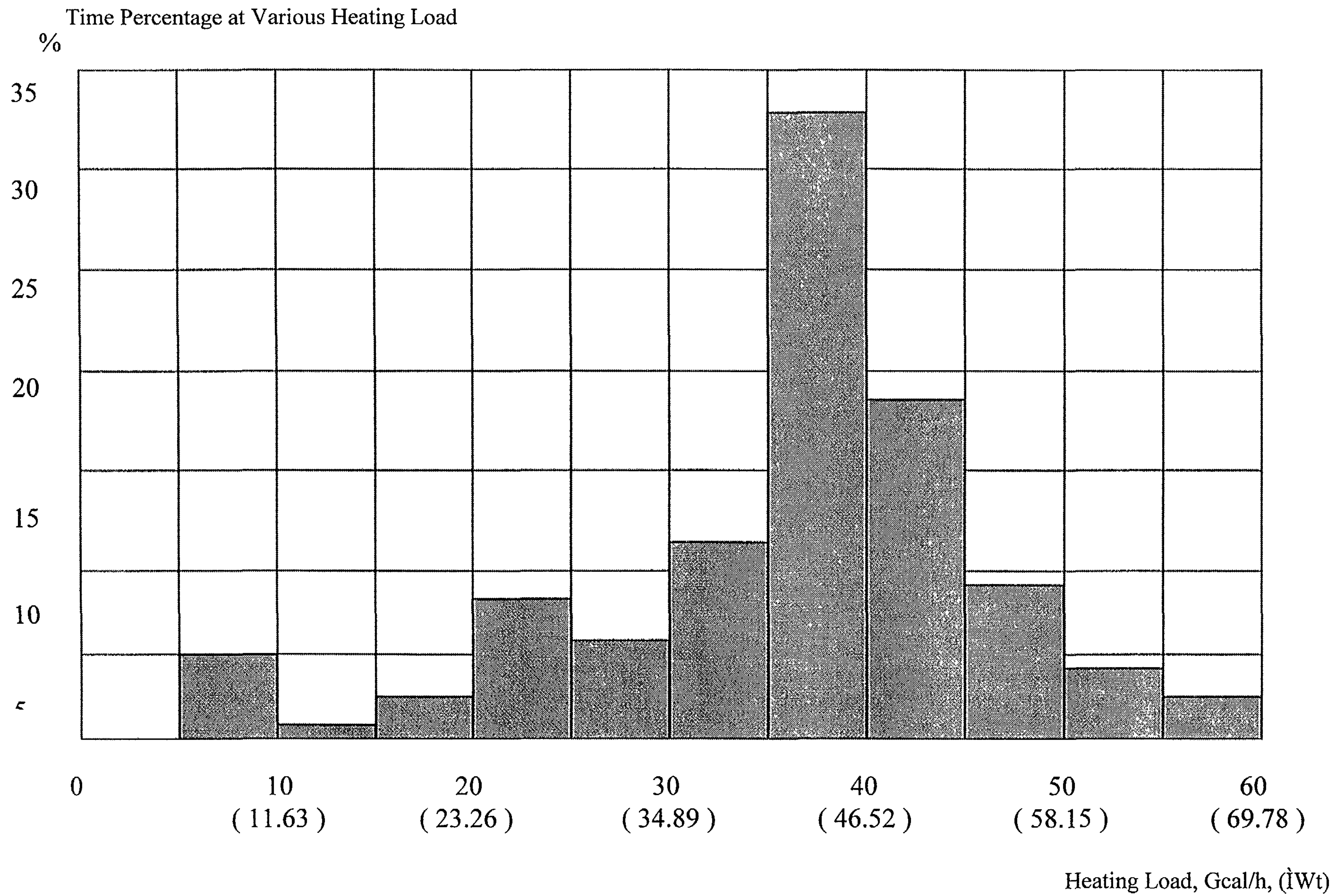

Fig.4 Time Distribution of Heating Load During Heating Period

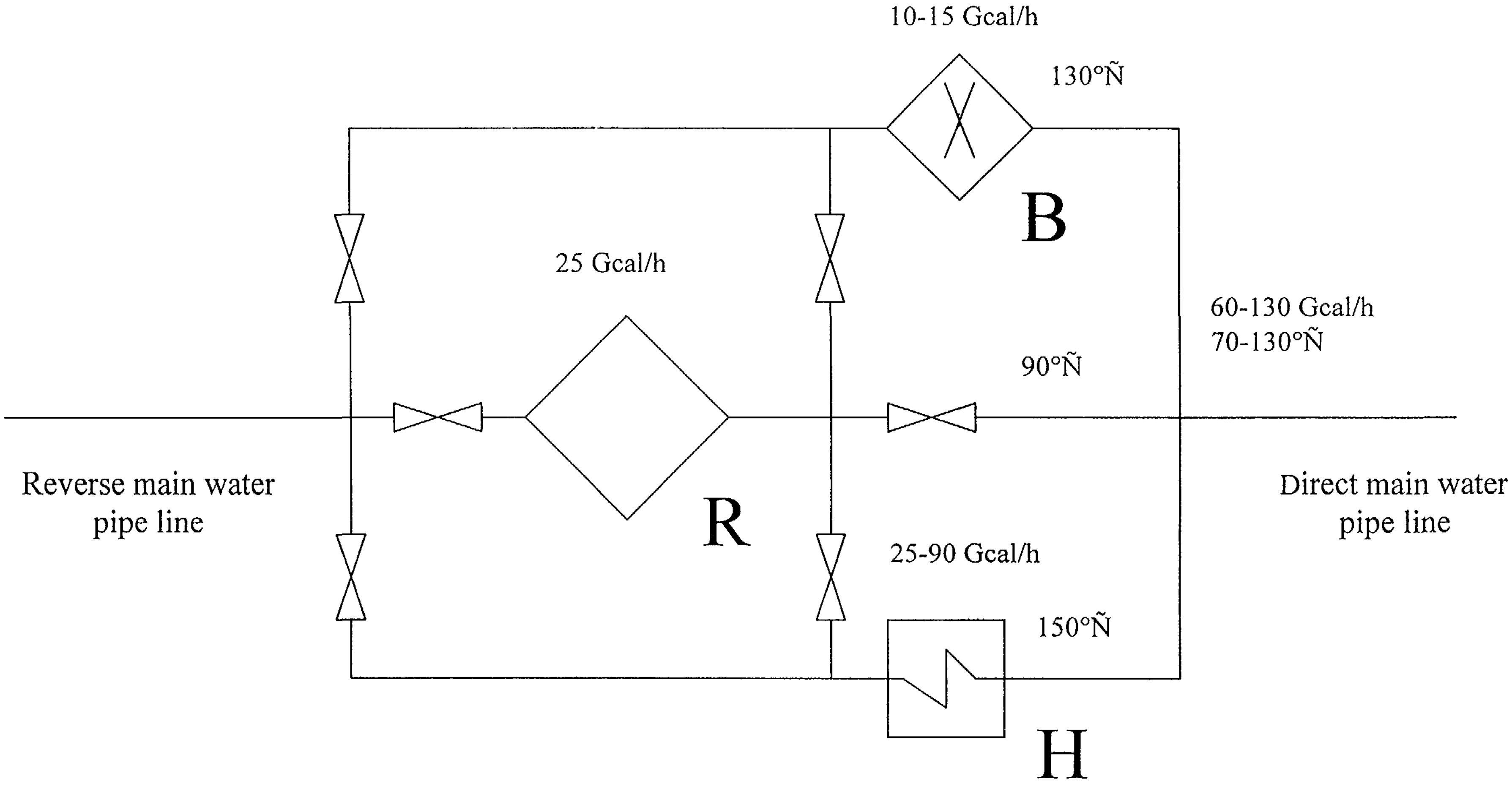

Fig 5 Schematic diagram of coupling of NHP RUTA-30 to IPPE's Heating System

R - NHP RUÒÀ-30, H - Water heater; B - Boilers using steam from turbine

In accordance with the requirements of the Regulatory Documents, the heat from the reactor is transmitted to the main heating circuit through three-stage loops, with its pressure increasing sequentially from the primary to the main circuit.

According to the preliminary safety assessments the radiological impact of the NHP on the inhabitants and the environment does not exceed the level of natural radiation background both in the normal operation conditions and at any realistic accidents.

The principal design characteristics of the RUTA nuclear heating plants are given in Table I.

One of the most important conditions of nuclear project implementation is the positive public opinion (public acceptance). To form it, a convincing demonstration of safety is necessary. A best way to obtain the positive public support would be to construct and successfully operate a prototype or demonstration plant.

A prototype or demonstration plant is to play an important role to confirm the basic design, and operation and economic features of the project. The NHP RUTA at the SRC RF-IPPE could play a role of demonstration and could provide a more active inculcation of this reactor type to the market, i.e. for the commercial use in remote regions of Russia. The design work has shown that, when coupling the 30 MW NHP RUTA with the existing heating system, an effective variant of its use could be found with the reactor operating mainly in the base-load mode.

The heating system in use at the SRC RF-IPPE (Fig.3) is provided with the following heat sources:

- MWH with the design capacity of 20 Gcal/h, using the steam from the First NPP;
- three water heaters with the design capacity of 50 Gcal/h, using the organic fuel; and
- three boilers with the design capacity of 5 Gcal/h, using the steam from the turbine.

Main characteristics of the heat supply system are:

- installed design capacity, Gcal/h 130
- maximun heat load in a typical year, Gcal/h ~ 60
- heat production in a typical year, Gcal/year ~ 200 x 10^3.

Currently water boilers are operated during the period from November till March. The basic fuel for the heating plant is natural gas, and a back-up fuel is heavy liquid fuel. In 1996 the IPPE's expenditures for purchasing these fuels were as high as 10 billion roubles.

Fig.4 shows the distribution of percentage of the time length during which a certain heat load is needed. The estimations performed on this basis indicate that if a 30 MWt (~25 Gcal/h) NHP RUTA is coupled with the local heating system (Fig.5), the NHP can secure up to 85% of yearly heat production. Water boilers (B in Fig.5), and/or water heaters (H in Fig.5) would play a role of back-up peak mode heat sources and be used for further heating of the hot water which is demanded during the cold periods of the year.

Therefore, constructing a demonstration NHP RUTA would result in enhancing reliability and economic viability of heat supply at the SRC RF-IPPE and the surrounding areas of the town. It could also work for implementing a full-scale demonstration of a perspective nuclear technology.

The NHP RUTA project assumes that simple and well developed equipment is to be used which could be manufactured by the Russian contractors. Some of non-standardized equipment could be fabricated at the SRC RF-IPPE's experimental workshops, too. Cooperation with organizations and countries interested in development, design and construction or utilization of such a reactor type is possible and greatly appreciated.

Taking into account highly developed site infrastructures, the expenditures for the NHP RUTA project (development and construction) at the SRC RF-IPPE could be approximately estimated as US$ 10 -15 million. At the same time about US$ 1.3-1.5 million per year would be saved thanks to the reduction of fossil fuel purchase. This covers completely the demonstration plant operation cost.

Authors would like to express deep gratitude to the staff of the SRC RF-IPPE's power shop and staff of the First NPP for co-operation and help in preparing the paper.

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XA9848800

CREATION OF NUCLEAR HEATING PLANTS IN RUSSIA: PRESENT STATUS AND PROSPECTS FOR THE FUTURE

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Abstract

History of heating reactor developments in two sites, Gorky and Voronezh, using AST-500 was reviewed. After interruption of construction for several years, decisions were made to resume the constructions. At Voronezh, based on the environmental assessment and the review of the IAEA OSART mission, the construction was resumed in 1996. In the course of construction resumption, design upgrading has been implemented in the following aspects: reclassification of station-level equipment concerning its importance for safety; control and instrumentation systems retrofitting with reliance on new generation element bases; application of self-actuated safety devices; and implementation of additional instrumentation for extended operating conditions. In Tomsk, Siberia, feasibility study is underway, which aims to replace the currently operating reactors with a twin-unit heating stations with AST-500 in order to provide heat to the district heating grids. In the study the NHP design is being assessed by a joint Russian-American Study Team from evaluation criteria such as design applicability and constructability, maturity of the design, safety aspects, technical uncertainty, available infrastructure, engineering and construction capabilities, site suitability, cost and schedule. Positive possibilities are foreseen to reuse the components previously delivered to the Gorky site, according to the assessments of structures and technological tools necessary for the reerection work were, man-power needed for the equipment dismantling, inspection and reerection and storage conditions. The construction cost is estimated as US$446 per KW(th).

1. INTRODUCTION

1.1. The following conditions are needed for practical application of nuclear heating plants (NHPs):

* heating loads and availability of centralized district heating systems with powerful heat-distributing grids;
* economical efficiency compared to fossil-fueled plants;
* licensability from the nuclear and environmental safety authorities (GAN) and the Ministry of Nature for siting, construction, operation;
* regional and federal authorities approval on the NHP deployment.

In the late 1970s, all the necessary conditions for NHPs were met in two Sites in Russia: (Gorky and Voronezh), where construction of the two pilot nuclear district heating plants (AST-500) started in early 80s. Somewhat later, preparative works began also at the Archangelsk Site. The governmental decision was adopted to construct NHPs at a number of other Sites: Ivanovo, Bryansk, Khabarovsk, etc.

In ex-USSR all NPPs were invested from the State budget and plenty of machine-building plants were involved by the governmental decrees in the production of equipment necessary for the nuclear power generating and heating only plants. Unfortunately, the established nuclear equipment production cooperation collapsed lately. (A new cooperation is being created now, involving some defensive machine-building plants that have to convert their production).

In 1990 decisions were adopted by regional administrations to cease the NHP construction in Gorky and Voronezh. Until that time two sets of AST-500 RP components were delivered to the Gorky Site and one set to Voronezh. By that time construction and erection works of the first unit of the Gorky NHP was 83% complete (in cost term) and more than 30% on the Voronezh Site.

1.2. In 1995 a level of centralized district heating loads was about 1,200 x 10^6 Gcal (in 1990 it was 1,550 x 10^6 Gcal) in the European regions of Russia. This declining trend in the district heating loads is explained by macro-economical difficulties and the significant reduction in the industrial heating loads. However, the district heating loads in the region are now starting to buildup and are expected to rise further in future.

Specialists forecast the following values of household district heating loads in the European regions of Russia (Gcal) in the year 2010.

North-West	140 x 10^6 Gcal
Center	620 x 10^6 Gcal
Middle Volga	210 x 10^6 Gcal
Ural	370 x 10^6 Gcal

The European regions are remote from the fossil fuel production areas and the cost of fossil fuel tends to rise so that it is expected to reach the level of West-European market cost in nearest future. Furthermore, impact of conventional power industries on the environment is most hazardous in large cities. On the other hand, there are many suitable places for the NHP siting in these regions.

In addition, more than 50% of running heat generating plants in the region have exhausted their lifetime or approaching to its lifetime. Therefore, many difficulties arise when it becomes necessary to get licenses every year from the regulatory authorities for continued operation of these fossil-fueled power and heat generating plants. Federal and regional funds available for these plants' modernization are far from actually needed. Private banks and firms are not very interested to invest money in this business. Hence, the following factors are characteristic of the present situation:

* the cost of fossil fuel in many regions of Russia is approaching to the level of the West-European market costs;
* lifetime of many heat generating plants is being exhausted;
* more strict requirements are imposed by the environmental regulations; and
* positive trends in the public attitude toward nuclear energy.

All these factors are objective prerequisites for "renaissance" in the nuclear heating field. The first task on this way is to resume the construction activity on the Site of Voronezh NHP.

2. VORONEZH NHP. PRESENT STATUS

Until the suspension in 1990 of the NHP construction work at the Voronezh Site, one set of AST-500 reactor plant (RP) components were delivered to the site and more than 30% of construction-erection work was completed.

In 1994, under pressure of the acute district heating problem in the city, the regional administration initiated the NHP environmental impact review by the team of a public commission composed of qualified local scientists and engineers. After a comprehensive study and consideration of the design materials, and visiting the NHP Site, a conclusion was drawn up that confirmed the assured safety of the plant for both the local inhabitants and the environment. Also, recommendations were given by the Commission for the administration to make a decision on the resumption of the NHP construction as soon as possible. One of the positive factors was the results of the IAEA OSART mission by the experienced international review team on the Gorky NHP with the same RP AST-500 (1989). Main findings of the review were very positive.

In 1995 the Russian Federation (RF) Ministry of Nature performed the State level environment review. The Committee confirmed the previously made conclusion about a possibility to resume the plant construction. Based on this confirmation the regional and city's administration adopted the decision to proceed with the plant construction activity and to put it into operation.

As a result the construction work on the plant Site was resumed in 1996. The programs have been developed for the NHP design updating in conformity with the requirements of latest norms and rules for nuclear power plants and RP's equipment inspection being currently in force in Russia. Inspections for the plant's equipment available at the Site have been started lately aiming at its preparation for erection.

The design documentation and appropriate licensing materials are under consideration now by the State Regulatory Body (GAN), in order to grant the permission for the station construction resumption. The approval and license may be obtained in the first half of 1998.

Station design updating includes, in particular, the following aspects:

* reclassification of station-level equipment concerning its importance for safety (for some equipment items that are located outside the Reactor Island);
* control and instrumentation systems retrofitting with reliance on new generation element bases;
* application of self-actuated safety devices; and
* implementation of additional instrumentation for extended operating conditions.

Additional validation work should be carried out as well.

The Governmental Decree on the NHP construction resumption and operation has been prepared.

3. PROSPECTS FOR FUTURE. TOMSK NHP

The feasibility study aiming at the NHP deployment at the Tomsk Site in Siberia started in the 1980s. The nuclear heating station was considered as a new energy source to replace the weapon-grade Pu-production nuclear reactors after their lifetime expiration. These reactors are currently generating more than 900 Gcal/h for both the Tomsk and the Seversk district heating grids.

Under the US-RF agreement, Pu-production facilities in Seversk and Zheleznogorsk must be closed by 2000. Then these reactors would be possible to be operated in the "power only" mode (through their conversion) till the end of their lifetime.

The RF Ministry of Atomic Energy (Minatom) has already made the decision to use a twin-unit heating station with AST-500 reactors at the Site of Siberian Chemical Combine in Seversk. This design was evaluated along with other potential types of heat sources by a Joint Russian-American Study Team.

In the feasibility study the NHP design was assessed using nine decision making criteria such as: design applicability and constructability, maturity of the design, safety aspects, technical uncertainty, available infrastructure, engineering and construction capabilities, site suitability, cost and schedule. The Joint Team visit to the site gave an opportunity to review the proposed area for the NHP siting and to better understand geographical relation between the existing facilities and the proposed reactor plant.

Review of the AST-500 design identified no aspects that could raise undue safety concerns. Based on the greater design maturity and minimum research and development work requirements it was concluded that the AST-500 could be implemented with the greatest degree of confidence. This design uses proven technologies, relies on the completed tests of basic components as well as on solid experience of marine NSSSs; also, the AST-500 reactor plant can be operated with bigger margin at low process parameters in the primary circuit (pressure, temperature).

The implementation of the AST-500 for the Seversk Site raises no serious construction issues. The proposed site is located near the city of Seversk and the existing infrastructure (power, transportation network, skilled labor, construction and engineering facilities, etc.) was judged to be sufficient to support the large-scale construction activity for the Project. Considering the near-term district heating demands in Seversk, the AST-500 should be the best choice to locate because of its appropriate design features, lower capital cost and shorter construction period.

The economic analysis of AST-500 was done in parallel by the US specialists using the American methodology (EEDB) and by the Russian specialists using the Russian procedure, respectively. The results of the American estimations were as follows (1994 dollars):

Total Direct Cost	280×10^6
Total Indirect Cost	108×10^6
Contingency	58×10^6
Total Plant Cost (2x500MWth)	446×10^6
Unit Cost, \$/kW(th)	446

Although the methodologies used by the Russian and the American specialists were different, the results turned to be very close (within the range of a few percent).

This year materials have to be prepared for the receipt of the permission for the Seversk NHP siting and construction.

In order to expedite the station construction it was proposed to utilize the components previously delivered to the Gorky NHP. With this aim, the work was carried out on validating the first reactor unit dismantling at the Gorky site followed by their reerection on the Seversk Site. Structures and technological tools necessary for the work were designed, and the man power needed for the equipment dismantling, inspection and reerection was evaluated. Storage conditions for the RP components (delivered to the Gorky NHP in 1984 to 1989) allow good expectation of their satisfactory state.

At present the main task for designers is to obtain licenses for the start of construction work at Seversk. Now the RP designers are working on several engineering modifications for improving the AST-500 annual heat production capability and its economics, such as:

* increase an annual operating time up to 7000 h;
* connection of additional heat loads;
* electricity production during a non-heating period; and
* increase in a unit heat capacity of up to 600 MW.

Pilot AST-500+25%...Ggrid → 600 MW, Tgrid 122/32°C

The 25% increase in the grid water flow allows an increased heat delivery to the grid up to 600 MW.

4. DESIGN ACTIVITY ORGANIZATION

OKBM is the Chief Designer of the reactor plant, the RCC "Kurchatov Institute" is the Scientific Supervisor for the Project. VNIPIET (St.-Petersburg) was the General Designer (Architect-Engineer) of the Gorky pilot station and NIAEP, N.Novgorod, is General Designer of the Voronezh NHP. For the

Tomsk Site VNIPIET is nominated as the General Designer of the NHP and NIAEP as the General designer of the AST-500 Reactor Island.

Joint efforts of the designers will allow a consolidation of best engineering solutions, available experience and lessons that were gained for the period of the pilot NHP_S construction. Selection of the solutions being made with regard to updated safety and economic requirements with reliance on the OKBM experience in construction and operation of different $NSSS_S$.

Our experience confirms the following conditions are necessary for the efficient organization of activities to construct a nuclear power plant:

* Single "master" of the entire Reactor Island (design philosophy, safety concept, economical and technical requirements)
* Safety culture and highest quality in all phases of design life cycle;
* Equal reliability of all components.

The latest design of serial advanced NHPs (AST-500M) includes practically all reactor plant components that were designed by OKBM (main equipment items, safety systems, non-safety systems, control and instrumentation systems, etc.).

5. NHP COMPETITIVENESS ENHANCEMENT

The NHP designers have all grounds to consider the AST-500 as a new-generation reactor plant that meets the internationally recognized safety requirements. It also has quite real potential of economic advantages as compared to fossil-fueled plants, particularly with regard to the new fuel and economic trends in Russia.

Now AST-500 NHP designers are working to achieve better economic characteristics of the plant. The following measures are under consideration:

* increase in annual generating time at rated power up to 7000 hours;
* increase in annual heat production;
* decrease in staff quantity required;
* electricity production during non-heating seasons;
* increase in heat capacity using the identical main equipment; and
* improved reliability and lifetime of auxiliary systems and equipment.

Standard marine NSSS equipment items and technology are the basis for the NHP auxiliary equipment designs (i.e. heat exchangers, valves, filters, etc.). The long-term successful operation of the prototypic equipment has proved its high quality and reliability.

Better reliability and operating characteristics of the auxiliary equipment over its life lead to the less total quantity of systems and equipment, shorter maintenance terms, reduced staff quantity and, as a result, reduction in capital and operation costs. Moreover, it is possible to eliminate some systems and equipment within the reactor island and minimize maintenance work scope taking into account infrastructure available on the Site (as it will be for seen for the Tomsk NHP).

6. CONCLUSION

Objective conditions exist now in Russia which could promote activities in the field of nuclear energy application for district heating. Main goals on this way are as follows: to resume construction work on the Site of Voronezh NHP and to start construction on the Site of Tomsk NHP.

All review missions carried out by various commissions and experts (a pre-OSART mission by IAEA, the Joint Russian-American Study Team, the State Review Committee of the RF (the Ministry of Nature, the State Regulatory Body, etc.) have drawn conclusions about a sound basis of the AST-500 RP designs, well-provenness of technologies, completed testing of all key components, the conformity with requirements of national norms and rules and with IAEA recommendations as well as about the possibility of proceeding with the Voronezh plant construction work and initiating the Tomsk NHP construction.

Experience gained in the design, fabrication and construction of the pilot NHP_S, and conformity with general principles of the internationally recognized safety culture and assured quality identify the basis for the optimization of activity in this field. It is expedient to expand the role and responsibilities of the Chief Designer role in Project management of nuclear facilities. Such approaches allow more efficient work organization in the field and improve the economical competitiveness and safety of the NHPs.

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NUCLEAR POWER PLANT WITH PRESSURE VESSEL BOILING WATER REACTOR VK-300 FOR DISTRICT HEATING AND ELECTRICITY SUPPLY

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Abstract

The viability for Russia of the Boiling Water Reactor (BWR) concept has been shown by a number of feasibility studies fulfilled for perspective sites with increased energy demands. Russia has long (31 year) successful experience in operation of NPPs with the vessel-type boiling reactor VK-50 which is located in the city of Dimitrovgrad. Taking into account the large Russian district heating market, it is expedient to apply this concept (BWR) not only for electricity supply, but also for district heating. This is a way to increase of nuclear power plant competitiveness along with good safety performance. The safety and protection of nuclear heat customer is guaranteed by reliable technical means which are well checked at Russian nuclear sites.

A PRESSURE VESSEL BWR VK-300 FOR DISTRICT HEATING AND ELECTRICITY

Further development of power industry is expedient to be based, to a substantial extent, upon nuclear power provided that nuclear power plants (NPPs) exhibit radically enhanced safety, protection measures against external impacts, public acceptability and economic competitiveness.

The vessel-type boiling reactors with natural circulation coolant refer to one of most suitable solutions for developing, on this basis, NPPs with enhanced safety. The concept of these reactors is based on attaining an extremely simple design, provision of enhanced safety due to inherent properties and passive heat removal from the core in case of accidents and also localization of any accidents without personnel intervention.

Great attention is paid in the world practice to developing new types of reactors which incorporate passive cooling capabilities for new generation reactors for higher safety. Our country has long-year (31 year) successful experience in operation of NPPs with the vessel-type boiling reactor VK-50 which is located in the city of Dimitrovgrad, where NPPs of medium size have been developed.

The most important features of the vessel-type boiling reactors in terms of safety are:

- relatively low steam parameters at the reactor outlet (as a rule, pressure of 7 MPa and temperature of 285 °C);
- reactor neutronics characteristics providing a possibility of inherent safety to be realized mainly due to negative feedbacks between reactor reactivity and its power, fuel temperature and steam quality;
- one circuit flow-chart and reactor coolant natural circulation in all operating modes that allows circulation pumps and steam generators to be dispensed

The design of the vessel-type boiling reactor VK-300 is presently under development. It is based on design specifications evaluated during the VK-50 reactor operation.

The natural circulation loop is of traditional version: it consists of a risign section and a downcomer communicating with each other at the upper and bottom ends. The draft portion is formed as a unit of individual draft tubes; the first steam separation step (gravitational and inertial) is arranged at the unit outlet, and the second one is performed in the intravessel cyclone-type separators fixed on a submerged perforated plate. Under this arrangement, the steam humidity at the reactor outlet do no exeed 0.1 % in mass.

Basic performances of an NPP with the VK-300 reactor and the reactor vessel of VVER-1000 are listed below:

Power, MW:	
thermal	750
electric (as operating in condensation mode)	250
heat generation, Gcal/g	400
Heat removal system:	one-circuit
Coolant	water
Coolant circulation	natural
Reactor outlet steam parameters:	
pressure, MPa	7
temperature, °C	285
flow rate, t/h	1370
Feed water temperature, °C	190
Core overall dimensions, m:	
height	2.4
equivalent diameter	3.16
Fuel charge for uranium, t	31.5
Uranium enrichment, %	4.0
Fuel service life (ef.), day	1870
Average uranium burnup, MW day/kg	43.5.

The design realizes higher design margins for the key reactor characteristics. Among those, the fuel linear power and the temperature during normal operation will remain at the level of up to 300 W/cm and 1300 °C, respectively. They compare with 400-450 W/cm and 1900 °C of moderm PWRs and VVERs.

The reactor core structure selected ensures passive reduction in its reactivity with reduced amount of coolant, for example, due to its boiling or leakage. Along with several channels intended for passive removal of residual heat to enhance the NPP safety, the project envisages:

- installation of a primary protective casing (a safeguard vessel) to localise leaked coolant leak when the reactor vessel and the primary circuit pipelines lose their integrity of leaktightness. This allows the core remaining under water even in the case of the vessel or the pipelines rupture;
- location of the reactor vessel penetraions above the core upper level;
- physical separation of the safety system equipment in order to prevent their simultaneous failures under emergency situations;
- application of fast-acting isolating valves and isolating devices on the primary circuit pipelines which limits, by the protective casing, the radioactive steam ingress into the turbine hall.

In accidents at the NPP, the reactor heat is removed by passive systems as follows:

- in the complete loss of electric power supply, steam is condensed in a submerged heatexchanger-condenser and the condensate gets back to the reactor by gravity;
- in the reactor vessel or primary circuit pipelines rupture within the design limits of the safeguard vessel volume, the steam-water mixture produced is delivered to the pressure-

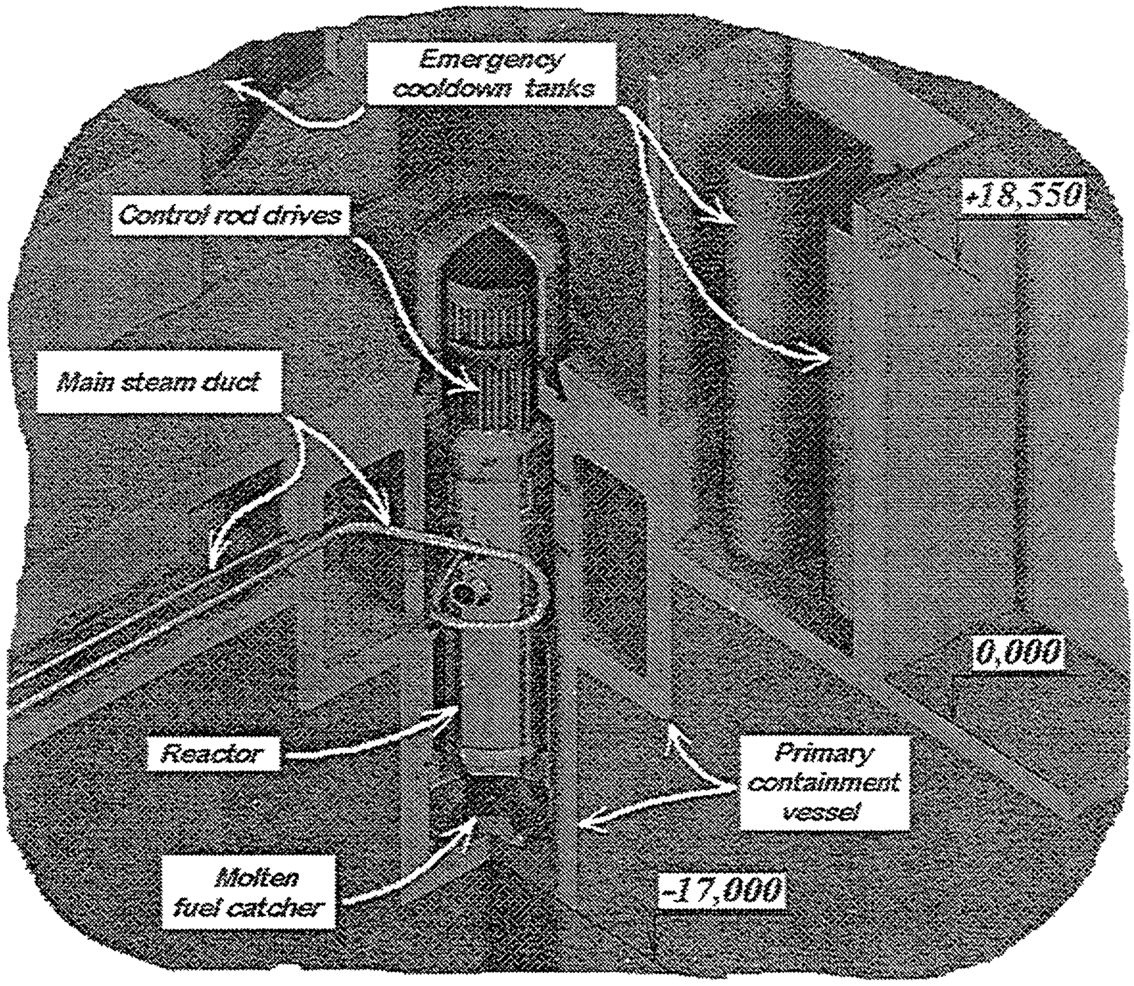

FIG. 1. VK-300 reactor equipment underground layout.

suppression localiser (a heatexchanger of a mixing type), where it is condensed and cooled by pressure suppression. A throttling device is fixed at the protective casing outlet to limit the steam-water mixture release rate and keep the core under the water level. On equalizing of the pressure in the emergency cooling tank and the reactor vessel, water flow to the vessel by gravity.

The simple structure of the reactor and the passive characteristics of safety systems ensure that the probability of the core damage be extremely low (less than $2 \cdot 10^{-7}$ events per a reactor-year).

In order to provide radiation safety, it is envisaged that the nuclear power unit should be equipped with a facility containing the low temperature coal sorbent. This facility plays a role of a protective barrier on the way of radioactive product release to the atmosphere (similar to a steam generator in the two-circuit reactor plants). It exhibits an advantage that gaseous fission products (Kr, Xe) are trapped here before solid fission products (Sr, Cs, Ce) are formed from gaseous ones. The VK-50 reactor operation experience shows that the decontamination factor of such an installation was up to 300 (could be even higher). This allows to make a prognosis that the release of radioactive products into the atmosphere will be at a level of 0.04-0.07 Gcal/MW·day which is comparable with releases from the two-circuit reactor plants.

Safe operation of the NPP with a vessel-type boiling water reactor in an electric load changing mode, which is important for the autonomous power grids, was confirmed by a series of special experiments performed on the VK-50 reactor. The experimental results assert that the reactor is able to successfully follow rapid load changes of up to 25 % nominal power.

The reactor power level is optimized for a combined electric power and heat supply source. The heat for district heating is in the form of extracting steam (or reduced live steam), and directed to the main primary heat exchanger, which is installed in the turbine hall. The primary heat exchanger is connected with a secondary heat exchanger, forming an intermediate loop. The pressure in the intermediate loop is, in principle, higher than the pressure in the primary steam. After the secondary heat exchanger the thermal energy is delivered to the customer. The three-loops system protects the customer in the case of accidental loss of the intermediate loop integrity.

The simple structure of the reactor VK-300 and its safety systems, good neutronics and thermal characteristics ensure its high competitiveness with regard to fossil fuel power sources.

Vessel-type boiling reactors are optimal for underground location due to their inherent features: extremely simple structures, passive cooling capability, small dimensions of the primary protective casing. The main reactor equipment underground layout is presented in Fig.1.

NPPs located underground impart them a number of new characteristics.

1) It can be asserted that NPPs located underground can be protected practically against any external events because of the substantially higher strength properties of massif materials (granite, basalt) compared with concrete (by 4-5 times).
2) Rock massif also serves as a protective casing when the pressure in the reactor plant premises rises due to the "internal" sabotage or "technological" explosions (steam or hydrogen), that may consequently result in a pressure rise of about 1.3-1.5 MPa. Such impacts will inevitably lead to destruction of ground-mounted NPP casings whereas the 50 m thick rock massif is able to withstand these loads.
3) NPPs located underground reduce radionuclide releases into the environment by some orders of magnitude. This can be achieved by disposal pipelines of length substantially larger than it is possible with a ground-mounted NPP. The pipelines are equipped with a system of sluices.
4) There are conditions to release heat directly to the inhousing massif during the reactor emergency cooling.
5) It is obvious that the NPP premises are expedient for use as a radioactive waste storage.
6) Expenditures for underground NPP decommissioning can be considerably lower (as estimated, by 5-10 times compared to ground-mounted NPPs), because the reactor plant premises may serve as their natural storage.
7) Results of numerous studies in seismic stability of underground NPPs and operating experience of underground structures show that within realistic range of contour depth for the reactor plant location, the underground location of reactor compartment leads to reduced intensity of seismic impacts by 1.5-2.0 times compared to those on the ground surface.

According to the international scale of events at NPPs, no accidents above the fourth level (accidents within NPP limits) can take place at the underground NPPs with the design of the underground version of VK-300.

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PROBLEMS OF HEAT SOURCES MODELING ON STAGE OF ISOLATED POWER SYSTEMS EXPANSION PLANNING

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Abstract

It is necessary to use computer codes for evaluation of possible applications and role of nuclear district heating plants in the local self-balancing power and heating systems, which are to be located in the remote isolated and hardly accessible regions in the Far North of Russia. Key factors in determining system configurations and its performances are: (1)interdependency of electricity, heat and fuel supply; (2) long distance between energy consumer centres (from several tens up to some hundred kilometers); and (3) difficulty in export and import of the electricity, especially the fuel in and from neighbouring and remote regions. The problem to challenge is to work out an optimum expansion plan of the local electricity and heat supply system. The ENPEP (ENergy and Power Evaluation Program) software package, which was developed by IAEA together with the USA Argonne National Laboratory, was chosen for this purpose. The Chaun-Bilibino power system (CBPS), an isolated power system in far North-East region of Russia, was selected as the first case of the ENPEP study. ENPEP allows a complex approach in the system expansion optimization planning in the time frame of planning period of up to 30 years. The key ENPEP module, ELECTRIC, considers electricity as the only product. The cogeneration part (heat production) must be considered outside the ELECTRIC model and then the results to be transfered to ELECTRIC. The ENPEP study on the Chaun-Bilibino isolated power system has shown that the modelling of the heat supply sources in ENPEP is not a trivial problem. It is very important and difficult to correctly represent specific features of cogeneration process at the same time.

1. Background

In Russia, from its western border to the Far East regions, a large size Unified Power System (UPS of Russia) is operating. However, UPS of Russia covers only about 40% of the whole territory of the country. This area includes those regions, which are most suitable for the vital activity of population, and therefore are most developed in the aspects of industry, agriculture, demography, etc. The remaining territory of the country is covered by the autonomous power sources and the local isolated power systems (i.e. connected neither to UPS, nor to each other. See Fig. 1). Regions served by the local power systems are primarily those with under-developed or developing economics in the Far North and North-East regions of Russia. The number of autonomous power plants situated in these regions is more than twelve thousand. These plants are mainly small size Diesel power plants (DPP) of only few hundred kW capacity.

The number of power plants, included in the local power systems is about 100. Their power range from 10 to 100 MWe. They produce up to 90% of electricity consumed in the area off the UPS.

If possible application of nuclear power sources in this vast area is considered, it would be natural, first of all, to evaluate the possibility of their inclusion into the local power systems. The power supply objectives should be achieved simultaneously with the heat supply tasks in the area covered by this power system. It is important to note that in the Northern regions the most part of the fuel used for the power engineering needs is consumed for the district heating.

In general, the task of planning of power engineering development is regarded as achieving the required reliability of energy supply with the minimum cost of energy generation, transport and distribution. In order to achieve this goal, it is necessary to consider comprehensively the process of

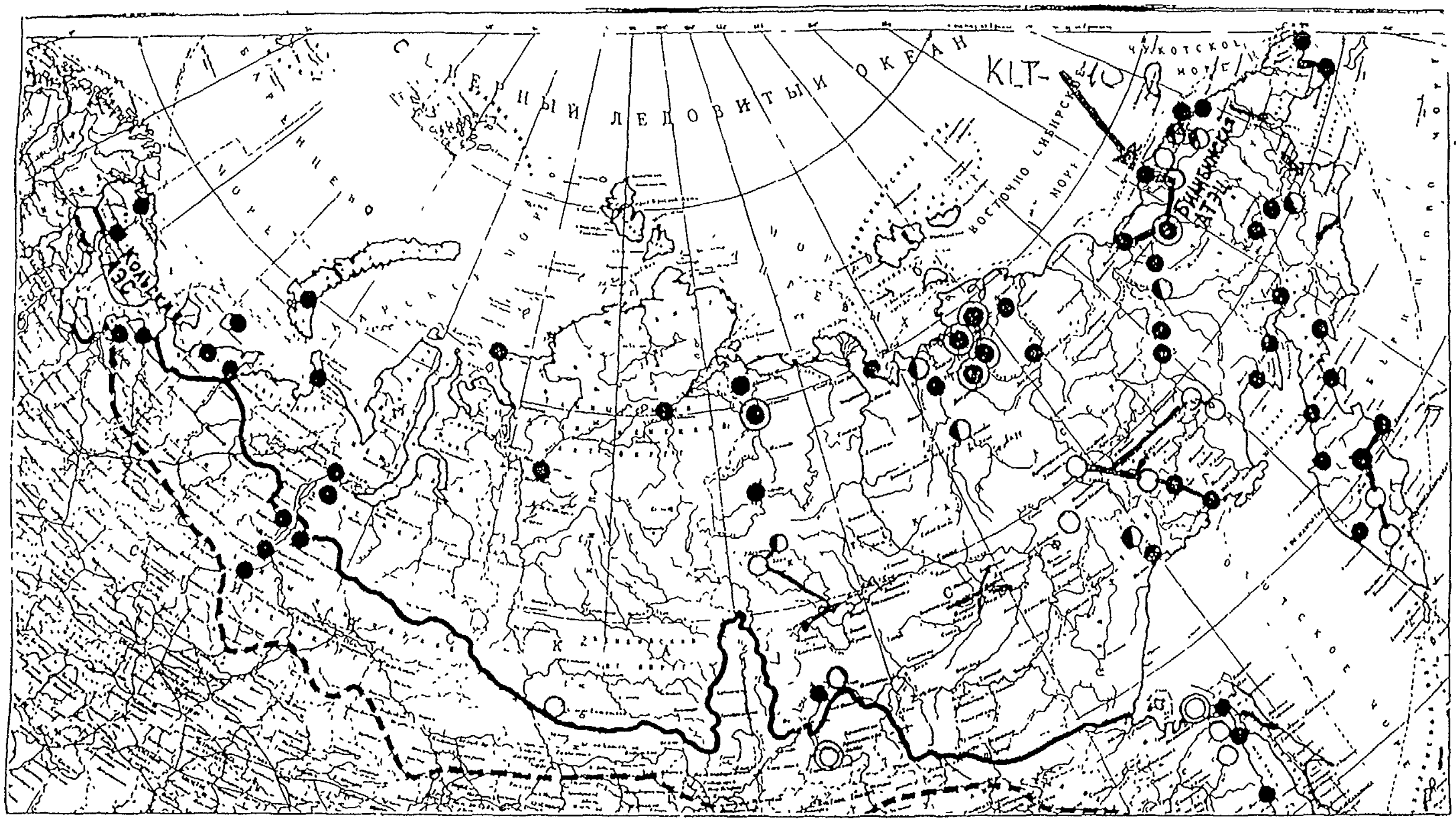

Figure 1. Possible sites of small nuclear power plants construction

- Nuclear cogeneration plants (NCP)
- Single purpose nuclear heating plants (NHP)
- NCP or NHP
- Most probable sites

identification of power system configurations and their performances by means of corresponding choice of respective types, numbers, capacities and operating modes of power units, types of consumed power resources and local, regional and federal level conditions.

It is very difficult or even impossible to satisfactorily realize this approach for the optimized solution without using up-to-date softwares. This is also the case for the consideration of both large size joint power systems and small size local power systems.

2. Choice of tools for study

Development planning of isolated power systems is not a new task, and it has been solved for a long time already. However it was rather difficult to optimize the development plan taking into account the possibilities of joint operation of selected power sources.

This task can be solved using a computer code package ENPEP (ENergy and Power Evaluation Program), which was developed, distributed and supported by the USA Argonne National Laboratory in cooperation with IAEA [1, 2]. The ENPEP software package is one of the integrated tools for power systems expansion planning, consisting of various modules which are shown on Fig. 2. This is a flexible and multifunctional PC-based tool for the optimum power system planning at the federal or regional levels, and for the comprehensive analysis and optimization of the plan.

By now, the ENPEP software package has been used as a tool for the power engineering expansion planning in about 80 countries. It was officially received by Russia in 1992 and adopted by several organizations including IPPE. ENPEP has been used for carrying out several studies.

There are two main models in ENPEP: BALANCE and ELECTRIC. The BALANCE model is used for making forecast studies on the power system and final combined power consumption/production balance under market economy conditions. This model is mainly used for studies and optimization of federal or regional fuel balance.

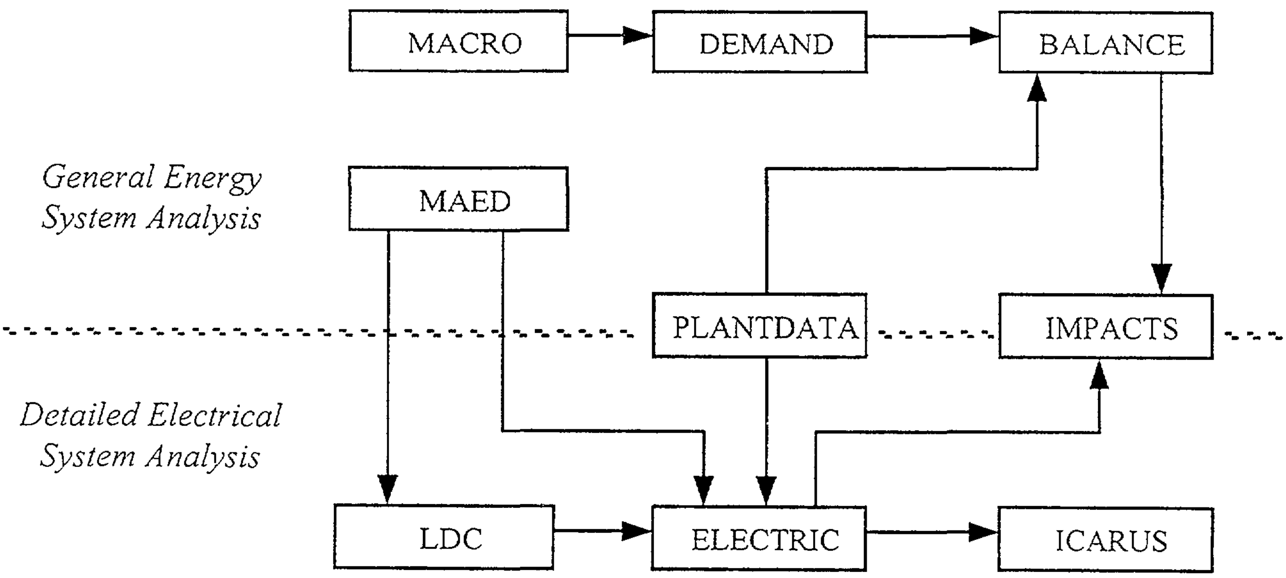

Figure 2. The IAEA ENPEP software package structure

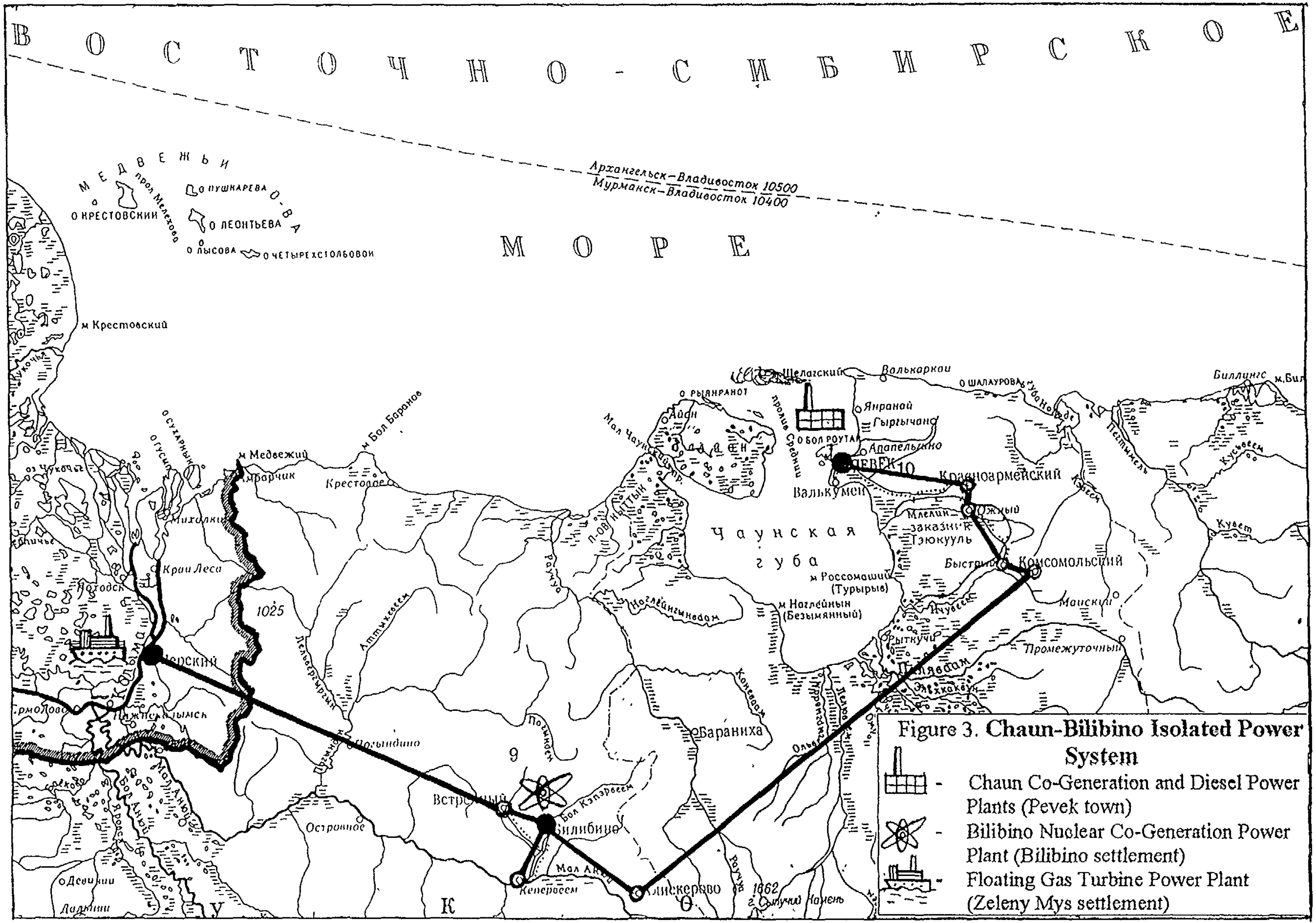

Figure 3. **Chaun-Bilibino Isolated Power System**

- Chaun Co-Generation and Diesel Power Plants (Pevek town)
- Bilibino Nuclear Co-Generation Power Plant (Bilibino settlement)
- Floating Gas Turbine Power Plant (Zeleny Mys settlement)

The ELECTRIC model (PC-based WASP III Plus [3] version) is to be used for working out an optimum plan of electricity generation system expansion (based on minimum of levelized cost criterion), while meeting the demand and reliability requirements.

Service modules of the ENPEP software package can be used for solving specific tasks of planning. For instance, LDC (Load Duration Curve) module is helpful in forecasting load changes in the ENPEP format on the basis of separate data available.

As regards to the task of development planning of isolated systems located in the remote Far North regions, the ELECTRIC model has both advantages and disadvantages. The application of the ELECTRIC model gives the optimum expansion plan of the electricity generating system. This optimum expansion plan would include optimum commissioning schedules of power sources selected from the proposed list of candidates. Minimum levelized costs of the whole power system development, while meeting the demand and reliability requirements for the electricity production system, is the criterion of the optimum plan.

The important advantage of this plan is that it will be worked out taking into account joint operation of the power sources in the system within the evaluated period of planning. This is especially necessary for the power systems which are practically isolated from the other power systems. As a result of such approach, optimized scheduling of power sources into operation is accompanied by the evaluation of energy production, load factor, and fuel demand for each power source in each year of the planning period. The loss of load probability (LOLP) and the system margin are evaluated for the whole power system for each year of the planning period. Thus, the usage of the ELECTRIC model makes it possible not only to work out the optimum plan of the power system development, but also to consider dynamic operation of such system.

However, the ELECTRIC model is capable for simulating a single-product system (electricity production) only, with the concentrated demand at one point, i.e. when the unified power system is considered.

3. Choice of subject of study

The Chaun-Bilibino isolated power system (CBPS) was chosen as a subject of study using the ENPEP software package (Fig. 3). This choice was determined by several considerations: first of all, CBPS is the only multi-unit system in the country when Small Nuclear Power Plants (SNPP), namely Bilibino Nuclear Cogeneration Power Plant (BNCGPP) consisting of four identical power units with 12 MWe are operated as a part of CBPS. The BNCGPP units were commissioned during 1974 - 1976.

TABLE I. Chaun-Bilibino Power System: Existing Power Plants

No	Name	Location	Fuel	Installed Capacity	
				MWe	Gcal/h
1	Bilibino Cogeneration Power Plant	Bilibino	Nuclear	4 x 12	4 x 25
2	Chaun Cogeneration Power Plant	Pevek	Coal	2x4, 1x6, 2x12	~60 (total)
3	Chaun Diesel Power Plant	Pevek	Diesel Fuel	4.5	no
4	“Severnoje Sijanie” Floating Gas Turbine Power Plant	Zeleny Mys (The Kolyma river)	Diesel Fuel	2 x 12	no

CBPS is located in one of the most remote and difficult-to-access regions of north-eastern Russia. Besides BNCGPP, the CBPS includes (see Table I):

- Chaun Cogeneration Power Plant (CGPP) at Pevek, consisting of five power units;
- Chaun Diesel Power Plant (DPP) at Pevek;
- Floating gas turbine power plant "Severnoje Sijanie 01" ("Northern Lights 01"), at Zeleny Mys (Green Cape) settlement on Kolyma river. (See also Table I).

Bilibino NCGPP and Chaun CGPP also supply almost all heat for district heating of the Bilibino settlement (up to 80 Gcal/hour) and the town Pevek (up to 60 Gcal/hours), respectively.

The special urgency of determining the optimum CBPS structure is caused by the fact that by the period from 2000 to 2005 all power sources in the area will exhaust their design life times. It is also important that a floating NCGPP with KLT-40S reactors is considered as one of possible ways of the CBPS expansion. The total operation experience of such a type of reactors installed on the atomic icebreakers and on the lichter-container- carrier "SEVMORPUT" reaches about 150 reactor-years. Now the floating NCGPP design is under development. Its location will be in the vicinity of Pevek. Some design characteristics of NCGPP with reactors KLT-40S and other candidates for inclusion into CBPS are given in Table II.

4. Approaches of Study

4.1. Steps of study

Proceeding from the stated task, the study steps were determined as follows:

- Modeling of the current CBPS status:
  - on the structure and condition of electricity and heat generating units;
  - on the fuel types, their price, amount, means and cost of their transportation to the consumption site;
  - on the amount of produced electricity and heat and its distribution to the consumers.

- Working out of probable scenarios of the demand fluctuations for electricity and heat in the region to be served by CBPS during the evaluated (planning) period of 30 years.

- Development of the data base in the ENPEP format on technical and economic characteristics of the following plants:
  - those included in CBPS at present,
  - candidates for inclusion into CBPS in future.

- Search for CBPS optimal expansion plan during the planning period.

- Sensitivity study of the produced optimum plan with the internal and external factors, which may have the most strong effect on the CBPS structure.

4.2. Modeling of heat supply sources

Some experience on the ELECTRIC module of the ENPEP software package application has already been gained in Russia for the power systems, in which Cogeneration Power Plants (CGPP), producing both heat and electricity, are operated. At the central and north-west Russian power systems which were studied, the shares of electricity generated by CGPPs are 40% and 20% of the total electricity production, respectively[4].

There are several possible approaches to the modeling of power sources generating both heat and electricity, in the calculations using the ELECTRIC module.

The first approach is to set some preliminary schedules of commissioning/decommissioning of CGPPs, thus ruling these CGPPs out of the optimization process. This approach was used in the above mentioned studies [4]. However, it is obvious that this approach is justifiable only when the CGPP share in the total power systems is relatively small.

Another approach consists of two stage iteration. In the first stage iteration, only the electrical part of the power system is optimized. In the second stage, whether the preliminary optimum plan on sizes and locations of the power plants can meet the heat demands for in the region, is evaluated. If the optimum plan does not meet the heat demands, corrections are introduced into the input data, and then the procedure is iterated.

The third approach assumes the heat demand and its production rate to be represented an equivalent amount of electricity.

The main heat consumers in CBPS (Bilibino settlement and Pevek town) are supplied by Bilibino NCGPP and Chaun CGPP. The share of these plants in the total electricity generation in CBPS is nearly 100%. Besides, cogeneration plants only are considered as candidates for entering CBPS (see Table II).

Authors have chosen the third approach to the cogeneration modeling in the ENPEP for solving the task under the CBPS conditions. Thus, in this study the heat supply sources have been modelled by assuming them as an equivalent amount of electricity, i.e. electricity produced from burning the same amount of fuel. The value of installed electric power of CGPP unit is assumed to be equal to that corresponding to the full steam flow rate to the turbine, i.e. the steam extraction from the turbine to the heater of district heating system water is completely stopped. For instance, this value is 16 MWe for BNCGPP, while it is equal to 12 MWe in case of 25 Gcal/hour heat supply rate.

However, it is certain that the representation of all features of heat supply sources in this calculation model is far from being realistic. This is caused by the impracticality of long distance heat transportation. Heat sources are required to be located in the immediate vicinity of the consumers.

TABLE II. Cogeneration Power Plants: Possible Candidates for Expansion of CBPS.

No	Name	Location	Fuel used	Installed Unit Capacity	
				MWe	Gcal/h
1	Bilibino Cogeneration Power Plant Life Time Extension	Bilibino	Nuclear	12	25
2	Bilibino Cogeneration Power Plant - **Second Stage**	Bilibino	Nuclear	40	50
3	Floating Cogeneration Power Plant with KLT-40S reactors	Pevek	Nuclear	2 x 35	2 x 25
4	"Kristall" Floating Cogeneration Power Plant with ABV-6M reactors	Pevek or Zeleny Mys	Nuclear	2 x 12	2 x 25
5	New Chaun Cogeneration Power Plant	Pevek	Coal	25	25

5. Results of studies

The load duration curve (LDC) module of the ENPEP code was used to forecast the CBPS electric load for the period of up to 2025 taking into account the additional load equivalent to the heat production. The obtained LDC were then provided to the ELECTRIC module as Fourier coefficients.

As a result of the ELECTRIC module operation, the CBPS expansion plan was optimized on the levelized cost criterion. The optimal plan included:

- The floating NCGPP with two KLT-40S reactors, the largest of proposed candidates, intended for the base electric load operating mode.
- Two coal fired CGPP units (25 MWe + 25 Gcal/hour), intended for the semi-peak or peak loads.

Sensitivity analyses have shown that the floating NCGPP with KLT-40S can be used as a stable electricity generating source for the base load under any changes of input parameters. Only the types and their numbers of power sources involved for the semi-peak and peak loads can be subject to adjust. These power sources for the "semi-peak and peak mixture" may include the following power units:

- One to three coal fired CGPP units (25 MWe + 25 Gcal/hour).
- A Floating NCGPP "Crystal" (2 x 12 MWe + 2 x 25 Gcal/hour).
- Life time extension of one or more power units of the existing Bilibino NCGPP after their design life time expiration.

Results of sensitivity analysis were not affected by some artificial input modifications assumed by the authors in order to simulate the heat supply features of the CBPS region. The results have shown that the methodological approach applied by the authors for modelling the typical cogeneration systems is not capable of adequately modelling the real CBPS power systems. For instance, none of CBPS expansion plans produced by this method could meet the heat supply requirements in the region.

Thus, the current ENPEP package can not give adequate pictures of cogeneration power plant systems for planning the expansion of isolated power systems on the Far North of Russia. Therefore, the issue of planning such system expansion under the condition of joint operation of different power sources is still open. In the authors' opinion, there are two possible ways in this direction. One is to develop a new tool. The other is to modify the existing modules of the ENPEP package. The experience gained with ENPEP could be used for the code validation.

6. Conclusion

Study on the optimum power system expansion planning for the Chaun-Bilibino isolated power system was performed in the framework of computation capabilities of the ELECTRIC module of the ENPEP software package. This required modeling of generation and consumption of electricity and heat. The common power system model, in which the distribution of power sources and consumers over the territory is not considered, was used for the study. This approach makes it possible to take into account to considerable degree the fuel consumption under cogeneration conditions, while the real controlling features of heat supply sources by the heat consumer locations can not be considered.

Also, it turned out to be impossible to increase the number of heat source candidates of the single purpose heat generating plants, such as nuclear district heating plants or fossil fueled boilers into the power system modelling.

The main conclusion from the above study is that the use of the ENPEP software package (ELECTRIC module) for optimization of electricity and heat supply systems offers rather limited possibilities. Additional means in the calculation model will be required for adequate representation of heat generation plants, and therefore, their role in such power systems.

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XA9848803

THE APATITY NUCLEAR HEATING PLANT PROJECT: MODERN TECHNICAL AND ECONOMIC ISSUES OF NUCLEAR HEAT APPLICATION IN RUSSIA

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Abstract

Traditionally Russia is a country with advanced structure of centralized heat supply. Many thermal plants and heating networks need technical upgrading to improve their technical and economic efficiency. Fossil fueled heating capacities have a negative influence on ecology, which can be seen especially in the northern regions of Russia. Furthermore, fossil fuel prices are rising in Russia.

The above factors tend to intensify the need for alternative heat sources being capable of solving the problem. Nuclear heat sources may be the alternative. In this paper, the main features of a proposed NHP in the Murmansk region are summarized.

1. INTRODUCTION

More than 35% of primary energy resources are consumed for heat supply to towns and villages in Russia. It illustrates the fact that the heating market in Russia is vast, and the demand for thermal energy is vast too. At present this demand is mainly satisfied by using fossil fuel heat sources. Some district heat is supplied by cogeneration NPPs, thus covering a very small fraction of the heat market.

During the last few years the trend of rising fossil fuel prices and transportation costs is clearly seen. These factors lead to increasing thermal energy costs. Heat cannot be transported over a long distance and must be consumed near the place of its generation. The burning of fossil fuel leads to ecological problems and an adverse impact on the health of people. Therefore, in Russia, the population sometimes protest against new fossil-fired power plant construction (e.g. public protest in Moscow against construction of the North power plant in 1993). In the face of public opposition some local governments in Russia adopted ecological laws which obliged the power plant operators to pay compensation for environmental pollution. This is another reason for thermal energy to become more expensive.

The above mentioned obstacles make it imperative to find an alternative to existing energy sources. The nuclear option was considered a reasonable one; therefore, the construction of AST-500 NHPs was started in Nizny Novgorod and Voronez. The State Program of North Regions development was adopted in Russia. This Program considers the nuclear option for improving the North region's heat supply. Russia has a positive experience of nuclear heating. For more than 20 yrs, in the small town Bilibino in Chukotka, the four water-graphite reactors of Bilibino co-generating NPP (4 x [12MW(e)+18.6 MW(th)]) have been supplying the town with ecologically clean electricity and thermal energy.

However, the severe accidents at Three Mile Island and Chernobyl shook the public trust in nuclear energy and in spite of the fact that the designs of AST-500 NHPs were examined by international experts they were not commissioned and the construction suspended.

During the last decade, one can observe a rising tendency of terrorism and local military conflicts in Russia and elsewhere. As a result, the public feel like being hostages of the nuclear establishment. Such a situation raises additional objections against the construction of NPPs in spite of economic and environmental advantages. To meet these concerns, the Russian State Programme on Environmentally clean power foresees the underground location of NPPs. Underground location of nuclear facilities

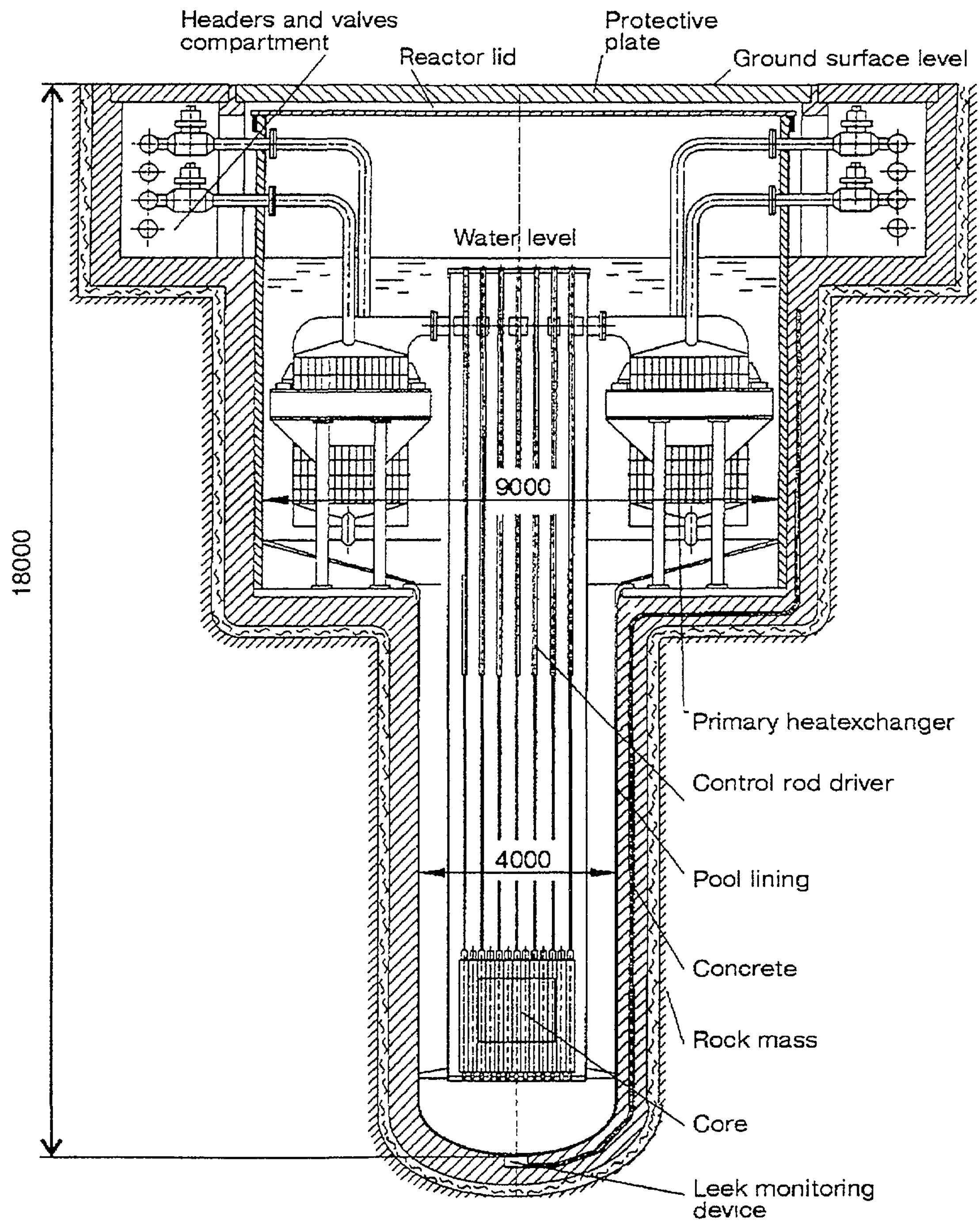

Fig. 1 Reactor RUTA 55

makes the physical defense of the plant very strong and it may withstand internal and external severe impacts.

The Research and Development Institute of Power Engineering (RDIPE) developed the concept of a safe NHP with the following main features:

- The nuclear source must be as simple as possible and supported by proven technologies.
- The nuclear source must be cheap and competitive in comparison with other types of heat sources.
- The nuclear source must posses inherent safety characteristics as far as possible at the present level of nuclear technology development.
- The nuclear source must be protected against diversion and military conflicts.
- For the aim of acceptability, the design of NHP must be well-understood, in other words, it must be well understandable by people without deep technical knowledge.

These considerations led to the idea of a pool-type heating reactor RUTA (Ref.1 and 2). The RUTA reactor design is based on existing pool-type research reactors. It was found that the RUTA reactor is convenient for underground location.

A decision on underground location of a NHP (UNHP) depends considerably on the local conditions of the site, including the following:

- Local fuel price including transportation.
- Availability of heating network at the site.
- Environmental conditions.
- Availability and readiness of local industry.
- Public attitude to nuclear power.

The above mentioned considerations were used in the design of the RUTA UNHP for Apatity, Murmansk region (Kola peninsula).

2. THE APATITY UNDERGROUND NHP RUTA PROJECT

2.1. REACTOR DESIGN

The RUTA.55 reactor unit is a simple nuclear heat source designed to supply 55 MW of thermal energy as water at 85°C. As shown in Fig.1, it is a pool-type reactor designed to operate at atmospheric pressure, thus eliminating the need for a pressure vessel.

The reactor core and primary heat exchangers are in the pool contained inside a steel-lined concrete vault. Pool water serves as the moderator, heat transfer medium and shielding. Primary heat transport from the core is by natural circulation of the pool water through plate-type heat exchangers located in the pool.

The secondary circuit delivers heat to the distribution system by way of the secondary plate-type heat exchangers. The pressure in the secondary circuit is higher than that in primary circuit and the pressure in the distribution system is higher than that in the secondary circuit. Thus, customer protection from a radioactivity leakage is ensured.

Computer controlled absorber rods are used for load following. Periodic adjustment of these absorbers compensate for fuel burnup. All the absorber rods will fall down to the core in case of accidents or when fast reactor shut-down is required. Pool water is continuously pumped through ion exchange columns to maintain water chemistry and to control corrosion.

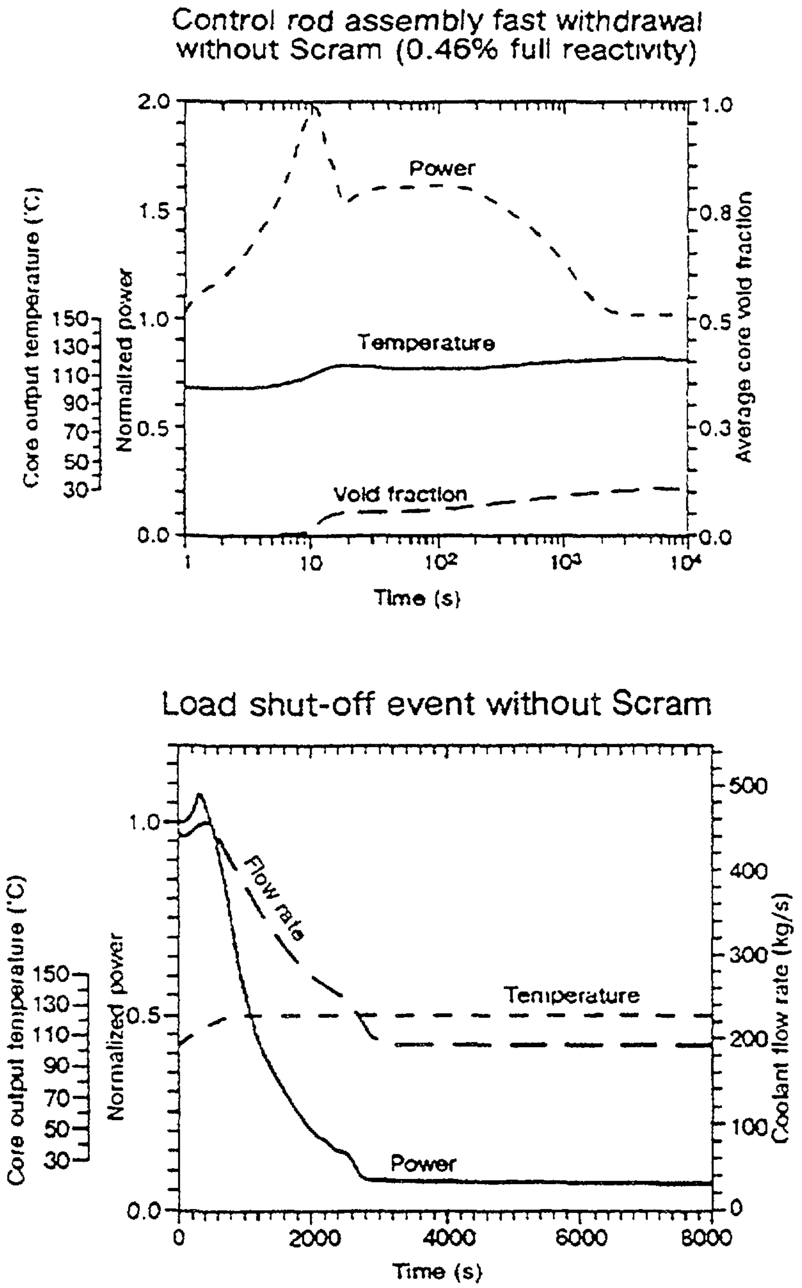

Fig. 2 Transients of RUTA 55

The reactor pool is covered by a lid enclosing a gas space over the pool. The air and water vapor are continuously circulated through a purification system and hydrogen recombiner.

The inherent safety characteristics of the RUTA reactor design include a negative fuel temperature reactivity coefficient and negative coolant temperature and void reactivity coefficients, all of which alleviate power transients following loss-of-regulation. Fig. 2 shows some of the transients. In addition to the inherent safety features:

- Large volume of water in the pool delays the core temperature rise for a long period.
- Natural circulation of water in the pool ensures core cooling under all accidents.
- Atmospheric pressure in the pool makes it impossible for loss-of-primary coolant cased by depressurization.

Major parameters of the RUTA.55 reactor design are presented in Table I.

TABLE I. MAJOR RUTA.55 CHARACTERISTICS

Parameter	Value
Power, MW(th)	55
Coolant pressure, Mpa:	
in primary circuit (above the pool level)	atmosphere
in secondary circuit	0.4
in heating network	0.6-2.0
Coolant temperature, ^{o}C (inlet/outlet):	
in primary circuit	75/100
in secondary circuit	66/90
in heat-supply system	60-85
Number of secondary circuit loops	2
Water circulation in secondary circuit	Forced
Dimensions of the core, m:	
height	1.2
equivalent diameter	2.03
Fuel	UO_2
Enrichment, %	3.6
Fuel burnup, MW day/kg	27.5
Number of fuel assemblies	169
Time interval between partial refuelling, years	3
Fuel lifetime, full power days	2970
Linear heat flux, W/cm	
average	50
Maximum	102

2.2. THE APATITY NHP DESIGN

The town of Apatity is located in the peninsula Kola in northern Russia, close to the Hibiny mountain massif. The Apatity UNHP would consist of 4 RUTA.55 reactor units. As shown in Fig.3, these units are arranged in a horizontal drifting mined into a mountain. This mountain is located in the close vicinity of the town center (less than 4 km.).

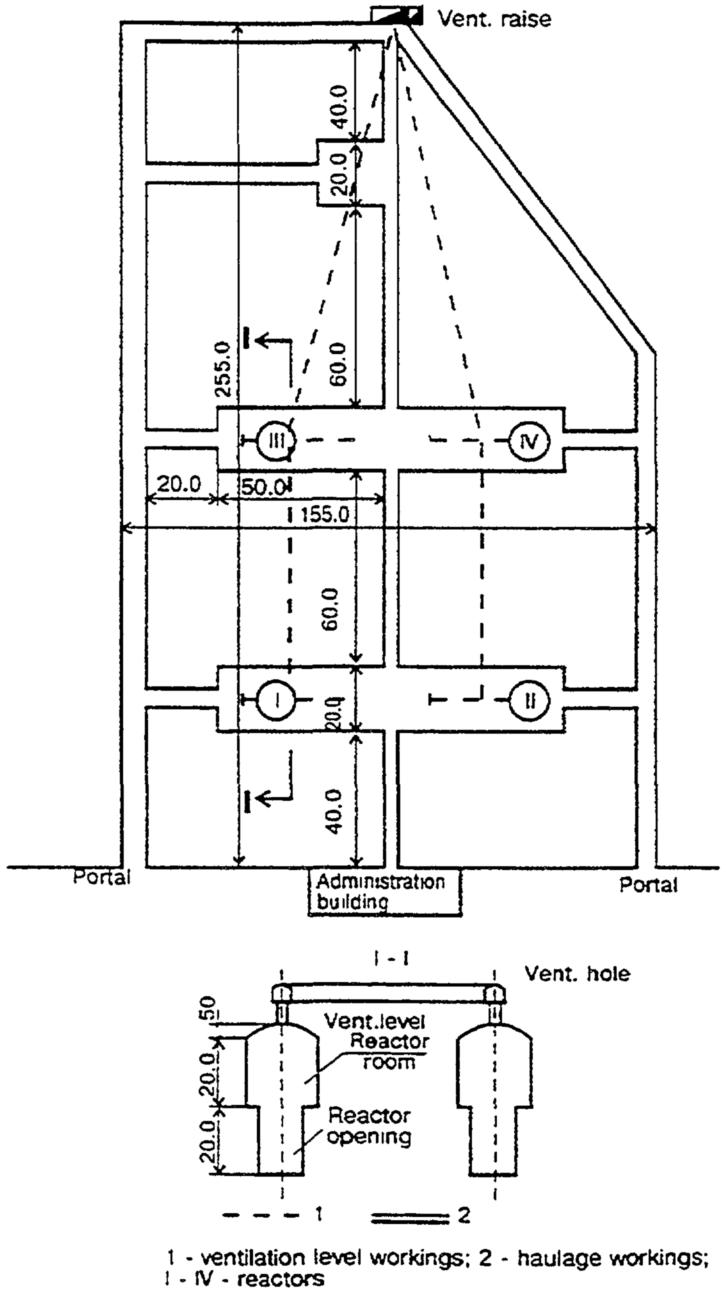

Fig. 3. Principal component plan of RUTA UNTP in Apatity

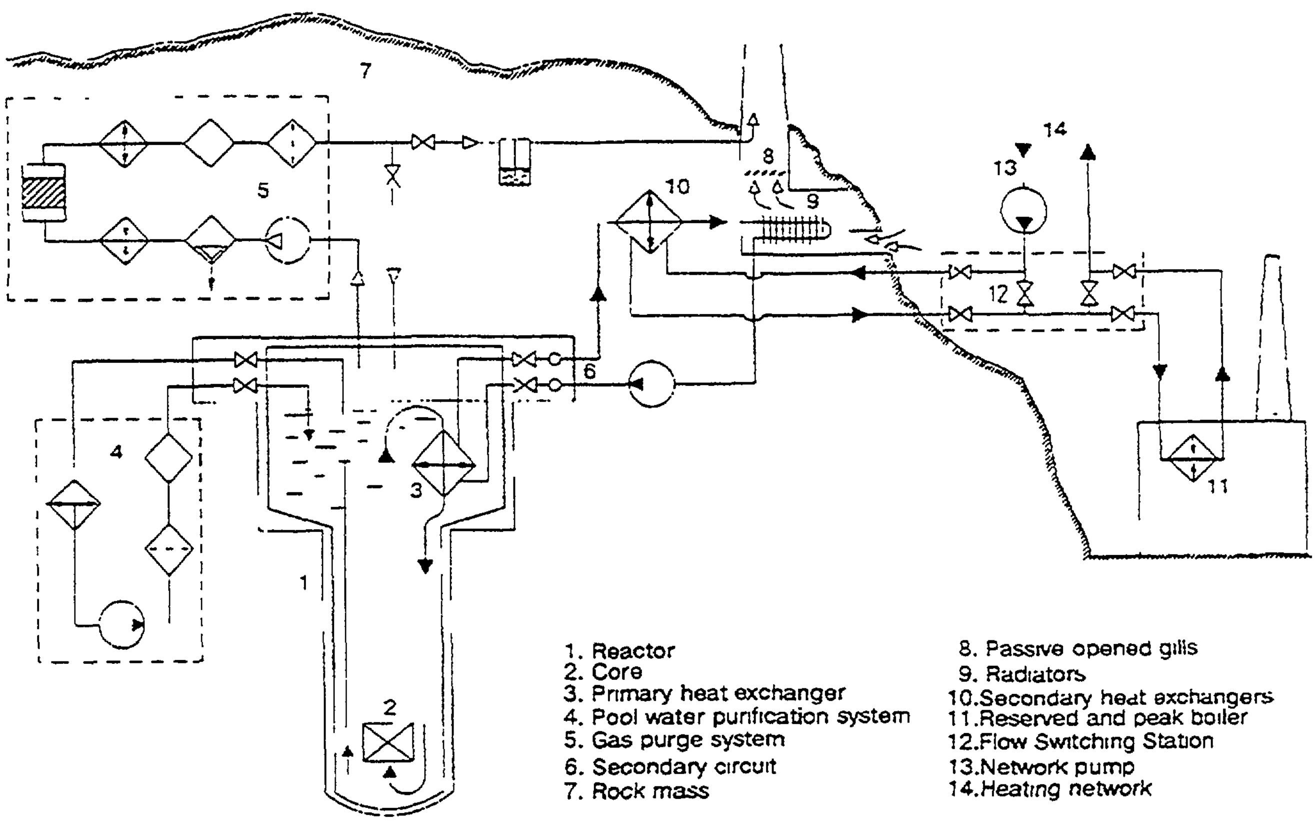

Fig. 4. RUTA NHP flow diagram

The UNHP would supply heat to the existing heating network. The heated water from the UNHP would be pumped to the existing coal-fired power plant for heating up, if necessary, to ensure the required temperature level (Fig.4). The UNHP is designed for a load factor of more than 90% and to supply 75-80% of the annual heating demand of the town.

Two long-term options were analysed:

- Continuation of the existing coal-fired power plant operation (old mode).
- Incorporation of the UNHP RUTA in the local heating network and its operation along with the existing coal-fired power plant (new mode).

The comparisons show that the energy cost in the new mode will be about half that of the old one. As shown in Fig. 5, the thermal energy cost of the UNHP depends on the unit power and if the unit power is more than 20MW(th), the UNHP will be competitive in comparison with alternate heat sources (coal- or gas fired).

The existing Apatity coal-fired power plant with a thermal power 700MW for heat supply (annual electricity output is 500 GW(e)h and heat output 2300 GW(th)h (=2 x10^6 Gkal)) consumes about 10^6 ton of Pechora coal. The coal contains sulphide and has a high ash content. Annually, the power plants emit about 10 000 tons of ash, 31 000 tons of SO_2, 5000 tons of NO_2 and other pollutants. The application of RUTA UNHP would allow to drastically improve the town environment.

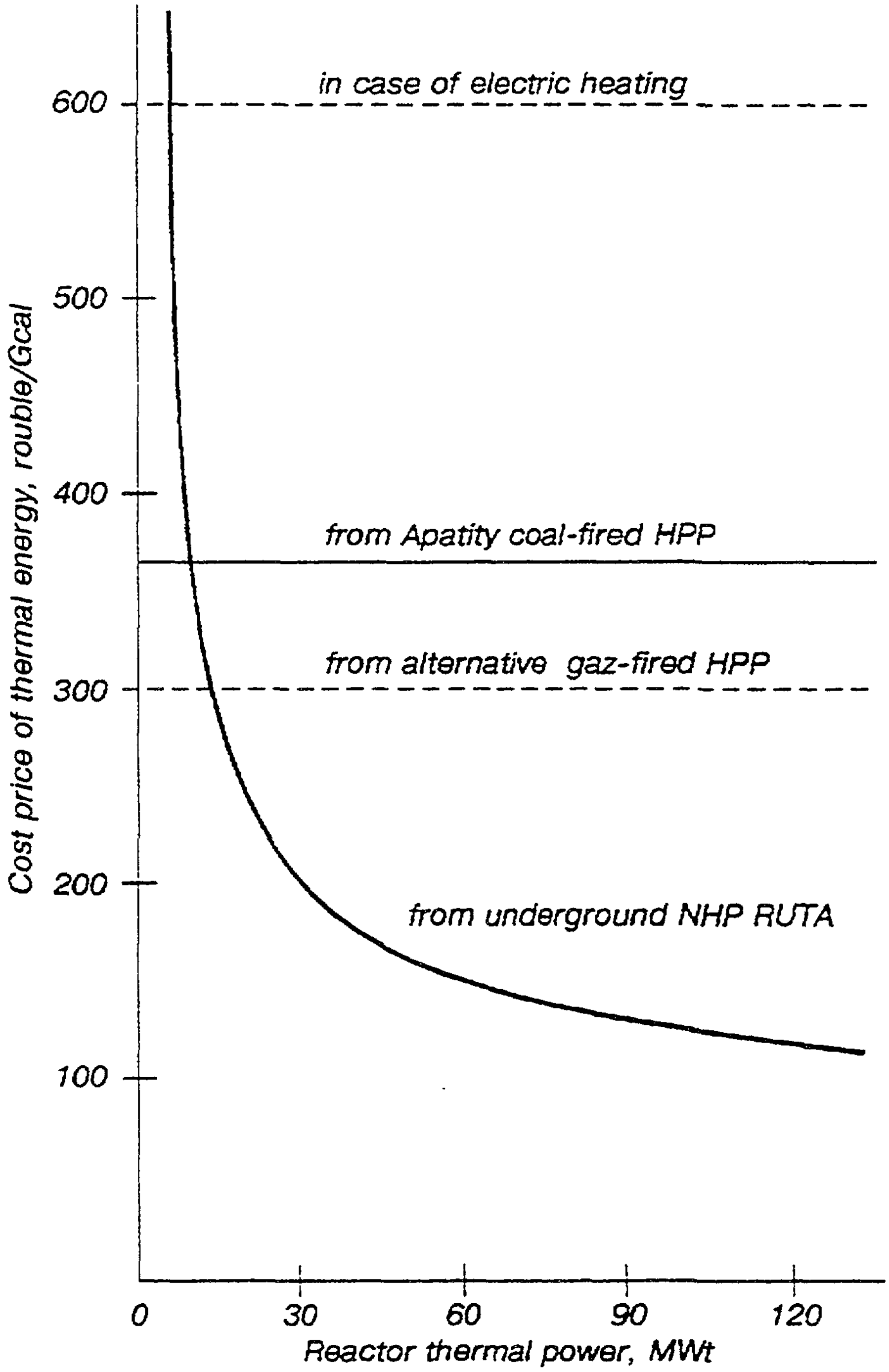

Fig. 5 Cost of heat from different sources (1992)

2.3. PROJECT STATUS

Technical and economical investigation of the project was performed during 1992-1994. In late 1994 a local government adopted the project. In the middle of 1995 the RUTA Joint-stock company was founded in the Apatity town for project implementation.

3. PROSPECTS FOR THE USE

The analysis of annual schedule of heat load shows that there is a considerable reserve of thermal power at the NHP during the summer period. This excess thermal power can be used for centralized production of cold water in an absorption refrigerating machine to be installed at the NHP site.

The refrigerating capacity of a single reactor of 55MW(th) would attain 15MW(th). Cold water at a temperature of $8^{o}C$ would be transported to customers through the standby line of the heating network and used for air conditioning or special room cooling (vegetable storehouses, storage facilities, etc.). The second line of the heating network system would be used for hot water supply.

The reactor can be used as a thermal power source to desalinate sea water by using standard desalinating plants. In this case the desalinating plant capacity will be 815 t/h of fresh water per reactor of 55MW(th).

Electricity production was considered in the RUTA NHP project for regular power supply and supply to the public (not industry) in the heating region. For this purpose, a special ampule (modular channel) is submerged into the reactor pool to produce steam for the turbine. The modular channel is a special pressurized tube-type structure with a nuclear fuel assembly in the bottom and an internal steam generator in the top of the sealed vertical tube, filled with water. The design provides natural water circulation inside the channel and water temperature expansion. The pressure inside the channel is about 9.8 MPa and maximum water temperature is 275 C. Lower side of channels surrounds the core of the reactor. This kind of a structure is the single core from neutron is point of view. Using 78 channels per reactor, it is possible to increase the heating power of the unit up to 65 MWt and electrical power up to 4 MWe. The safety of the reactor will be high in spite of using components with high pressure because the rupture of one channel does not influence the reactor due to the small volume of water in the channel (about 70 liters) and the availability of adequate cooling during the accident.

The project also enables increasing the temperature of NHP water by using thermal transformers. In this case there is no need for peak boiler units, but the heat costs would increase.

4. CONCLUSIONS

- The heating market in Russia is vast, but the nuclear contribution to this market is presently very low.
- Heat is becoming more expensive due to rising fossil fuel price.
- The burning of large quantities of fossil fuels leads to local ecological problems and public protests.
- The nuclear alternative seems to be reasonable, but contrary to the earlier development period, it has become much less popular.
- Underground arrangement of NHPs would allow bringing nuclear heat closer to customers without a significant risk.
- Russian State Programs foresee increasing non-electrical application of nuclear energy.
- Nuclear heat generation plants must be simple, cheap and safe.
- Pool-type reactors have a big potential for district heating, sea water desalination, and air-conditioning.
- The simplicity of pool-type reactors, ease of maintenance and manufacturing make this type of heat source attractive for countries without a developed nuclear industry.

DESIGN PRECAUTIONS FOR COUPLING INTERFACES BETWEEN NUCLEAR HEATING REACTOR AND HEATING GRID OR DESALINATION PLANT

XA9848804

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Abstract

Nuclear heating reactor (NHR) has been developed by INET since the early eighties. To achieve its economic viability and safety goal, the NHR is designed with a number of advanced and innovative features, including integrated arrangement, natural circulation, self-pressurized performance, dynamically hydraulic control rod drive and passive safety systems. As a new promising energy system, the NHR can serve for district heating, air conditioning, sea-water desalination and other industrial processes. For all of these applications, it is vital that the design and performance of the coupling interfaces shall insure protection of user ends against radioactive contamination. Therefore, an intermediate circuit is provided in the NHR as a physical barrier, and the operating pressure in the intermediate circuit is higher than that in the primary system. In addition, the radioactivity in the intermediate circuit is monitored continuously, and there are also other protection measures in the design for isolating the intermediate circuit and the heating grid or desalination plant under some emergency conditions. The excellent performance of the above design precautions for the coupling interfaces has been demonstrated by operational practice from the NHR-5, a 5MW(thermal) experimental NHR, which was put into operation in 1989. This paper presents the main design features of the NHR as well as the special provisions taken in the design for coupling the NHR to the heating grid or desalination plant and some operating experince from the NHR-5.

1. Introduction

Research work on the possible applications of nuclear power for low temperature heating was initiated in the early eighties. During 1983-1984 Institute of Nuclear Energy Technology (INET) of Tsinghua University used its existing pool-type reactor to provide space heat for nearby buildings. Meanwhile, two reactor types, a deep pool type NHR and a vessel type NHR were developed by INET. Based on the specific heating needs in China and a comparison of various NHR design concepts, the vessel type NHR was selected as the main development direction. As a result, construction of a 5MW (thermal) experimental NHR (NHR-5) started in 1986 at INET. The reactor was successfully put into operation for space heating in 1989.

Since 1990, a commercial sized NHR with an output of 200 MWt (NHR-200) has been developed, and it was approved by the central goverment in 1995 that an NHR-200 demonstration plant would be built in Daqing in Northeast China. The NHR can be used in district heating, sea-water desalination, air conditioning and other industrial processes. For all of these applications, there is a risk of radioactive contamination to user ends. Therefore special design considerations should be given to the coupling interfaces between the NHR and the heating grid or desalination plant.

The main purpose of this paper is to present the NHR technical description as well as the design precautions for the coupling interfaces and the operating practice from the NHR-5.

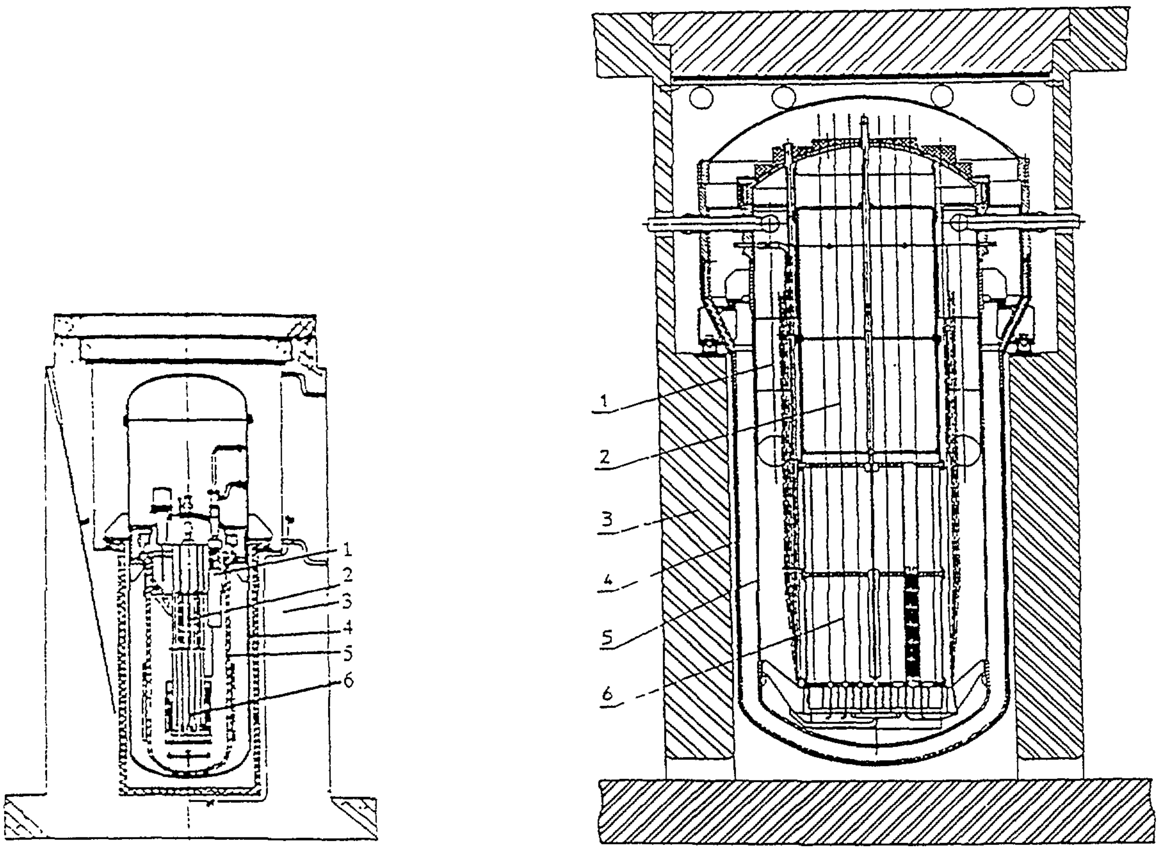
2. Technical Description of the NHR

The NHR has been designed with a number of advanced and innovative features to achieve its safety goal and economic viability. Fig.1 (a) and (b) show the reactor structures of the NHR-5 and the

NHR-200, respectively. Their essential design features are the same. The NHR is a vessel type light water reactor with an integrated arrangement, natural circulation, self-pressurized performance and dual vessel structure. The core is located at the bottom of the reactor pressure vessel (RPV). Primary heat exchangers (PHEs) are arranged on the periphery in the upper part of the RPV. The system pressure is maintained by inert gas and/or steam. A containment vessel fits tightly around the RPV so that the core will not become uncovered under any postulated leakage in the reactor coolant pressure boundary. The reactor coolant circulates due to density differences between "hot" and "cold" regions in the RPV. There is a long riser on the core outlet to increase the natural circulation capacity.

Gadolinium oxide is used as a burnable poison to control the reactivity along with the B_4C control rods. The reactor coolant does not contain boric acid during normal operation. The dynamically hydraulic control rod drive system used in the NHR is designed on the "fail-safe" principle. i.e. control rods will drop into the reactor core automatically upon loss of power supply, depressurization, pipe break and pump shutdown events.

Spent fuel assemblies are stored in racks around the active core. This solution greatly simplifies the refueling equipment.



(a) NHR-5

(b) NHR-200

1. Primary Heat Exchanger
2. Riser
3. Biological Shield
4. Containment
5. Pressure Vessel
6. Core

Fig 1 Cross Section of NHR

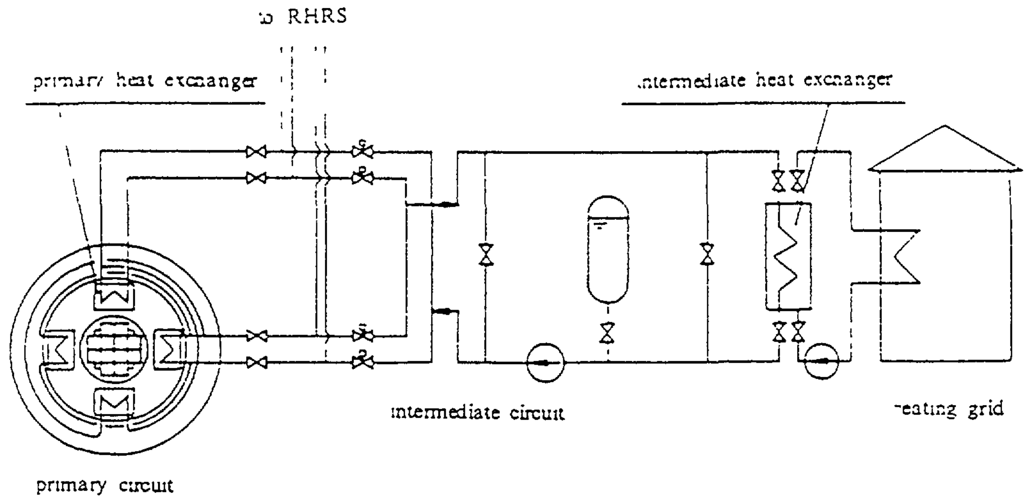


Fig.2 Schematic Diagram of NHR-5 Heat Supply System

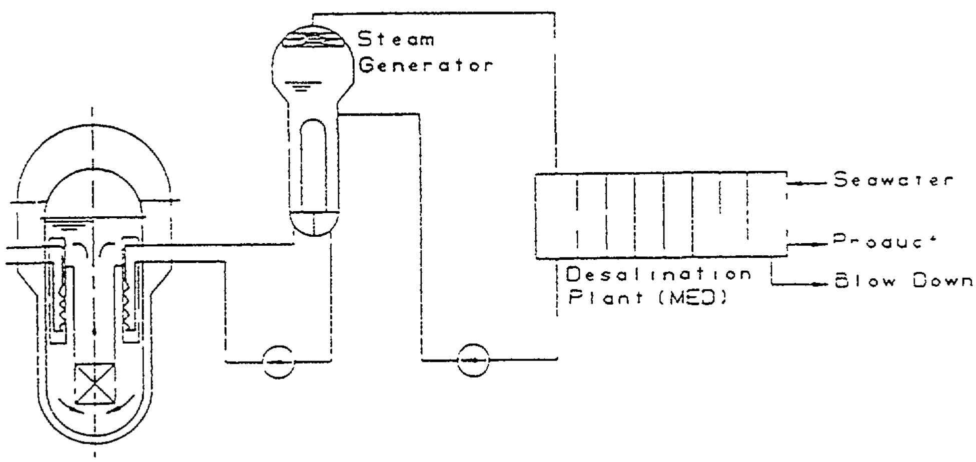


Fig.3 Schematic Diagram of NHR Coupled with H-MED

A simplified schematic diagram of the NHR used for district heating and for the desalination are shown in Fig.2 and Fig.3, respectively. The nuclear heat supply system contains triple loops. The primary coolant absorbs heat from the reactor core, passes through the riser and enters the PHEs where the heat is transferred to the intermediate circuits. Finally, heat is delivered to the heating grid via an intermediate heat exchanger or to the desalination plant via a steam generator. An intermediate circuit is needed in the NHR to insure that the heating grid or the desalination plant is free of radioactivity.

There is no emergency core cooling system in the NHR. The residual heat removal system (RHRS) is the most important safety system for the NHR and is designed as a passive system. The decay heat will be dispersed into the ultimate heat sink by natural circulation. A boric acid injection system, as a secondary reactor shutdown system, will be operated by gravity if an anticipated transient without scram (ATWS) occurs.

The key NHR design data is presented in Table I.

Table I Main Design Data of NHR

Reactor		NHR-5*	NHR-10**	NHR-200*
Thermal power	MW	5	10	200
Primary system pressure	MPa	1.5	2.5	2.5
Core inlet/outlet temperature	°C	146/186	174/210	140/210
Volumetric power density	kW/L	26	23	35.2
Number of fuel assemblies		16	32	96
Number of control rods		13	13	32
Active core height	m	0.69	0.80	1.9
Active core diameter	m	0.57	0.95	1.9
Initial inventory of UO_2	t	0.51	1.4	14.23
Enrichment of initial core	%	3.0	3.0/4.5	1.8/2.4/3.0
Refueling enrichment	%	3.0	3.0/4.5	3.0
Intermediate circuit pressure	MPa	1.7	3.0	3.0
Intermediate circuit temperature	°C	102/142	180/135	95/145
Heat grid temperature (Steam temp.)	°C	90/60	(130)	130/80-90/70***

* NHR for district heating
** NHR for sea-water desalination with H-MED process
*** Temperature in the third and forth loop, respectively.

3. Design Precautions for the Coupling Interfaces

In the present nuclear heat applications, energy is supplied mainly in the form of hot water or low temperature steam. Coupling is accomplished via a heat transmission loop. A major concern for these applications is to prevent radioactivity ingress from the transferring to the heating grid or the product water. To this end, the following design precautions have been taken for the coupling interfaces between the NHR and the heating grid or desalination plant.

(1) An intermediate circuit is provided as a physical barrier in the NHR, so that there are at least two physical barriers generally between the primary system and user ends. As seen in the Fig.2 and Fig.3, the radioactive coolant of the primary system could in principle reach the heating grid or desalination plant only after penetrating the primary heat exchanger and the intermediate heat exchanger or the steam generator in succession. For district heating, there is usually an additional physical barrier provided by the heat exchanger in the local heat distribution station. While in coupling the NHR to a desalination plant with a Multi-Effect-Distillation (MED) process, the first stage of the MED will provide an additional physical barrier to prevent product water from radioactive contamination.

(2) The operating pressure in the intermediate circuit is higher than that in the primary system and the heaitng grid. Therefore, in case of tube failures in the PHEs, the leakage direction is toward the primary side instead of allowing radioactive coolant to leak out. This solution also favors to keep the water quality in the intermediate circuit due to free of contamination from the heating grid.

(3) The pressure and radioactivity of the intermediate circuit are monitored continuously. Either the pressure decreases or the radioactivity increases to a set point, the isolation devices will be triggered to isolate the intermediate circuit. The isolation action can also be done in the heating grid or desalination plant.

The above special design measures for the coupling interfaces will insure protection of the heating grid or the product water against radioactive contamination.

4. Operating Practice of the NHR-5

Since the NHR-5 was put into operation in 1989, a number of experiments have been carried out to demonstrate the operating and safety features of the NHR, including self-regulation and self-stability features, transient behaviour following a loss of main heat sink-ATWS and the heat transfer capability of RHRS with and without interruption of natural circulation in the primary system. Meanwhile, in order to investigate multiple functions for the NHR, experiments were also conducted in the NHR-5 to study concepts such as electricity generation with low pressure steam in a co-generation mode as well as desalination process with high-temperature MED (H-MED) technologies and air conditioning for a large building using the lithium-bromide absorption process.

During reactor operation for space heating and all the above experiments, specific water radioactivity of the intermediate circuit is monitored continuously. However, the specific radioactivity is much lower than the rad-meter sensitivity as no radioactive coolant could penetrate from the primary system. Therefore, to monitor radioactivity in the intermediate circuit and the heating grid can only be performed by reqular sampling analyses. The results of the analyses conducted since 1989 have shown that the specific water radioactivity in the intermediate circuit and the heating grid is as low as the radioactive background level of potable water in the site area, which is about 0.10 Bq/l. Meanwhile, higher standards of water quality in the intermediate circuit can be easily maintained. Analytic radioactivity data in the intermediate circuit are somehow lower than that in the heating grid. Therefore, the operational data and experimental results from the NHR-5 have demonstrated that the overall performance of the NHR is excellent and the design measures adopted for the coupling interfaces function properly.

5. Conclusive Remarks

The NHR, with a number of advanced design and safety features, can serve as a safe, clean and economic energy source for non-electric applications. The special design provisions adopted in the NHR for the coupling interfaces will insure that the heat grid or desalination plant is free of radioactivity. This has been demonstrated by the operating practice of the NHR-5. In particular, to keep the operating pressure in the intermediate circuit higher than that in the primary system and the heating grid is much significant for preventing radioactive coolant from leaking out and for maintaining high standard of water quality in the intermediate circuit.

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II.1. DESIGN ASPECTS OF NUCLEAR HEAT APPLICATIONS

Low temperature heat applications —

Desalination and other applications

NUCLEAR FLOATING POWER DESALINATION COMPLEXES

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Abstract

Russia is a single country in the world which possesses a powerful ice-breaker transport fleet that allows a solution of important social-economic tasks of the country's northern regions by maintaining a year-round navigation along the Arctic sea route. A total operating record of the marine nuclear reactors up until till now exceeds 150 reactor-years, with their main equipment operating life reacning 120 thousand hours. Design and constructional progresses have been made continuously during forty years of nuclear-powered ships construction in Russia. Well proven technology of all components experienced in the marine nuclear reactors give grounds to recommend marine NSSSs of KLT-40 type as energy sources for the heat and power co-generation plants and the sea water desalination complexes, particularly as a floating installation. Co-generation stations are considered for deployment in the extreme Northern Region of Russia. Nuclear floating desalination complexes can be used for drinkable water production in the coastal regions of Northern Africa, the Near East, India etc.

1. Marine nuclear reactors creation and operation experience in Russia

Russia is a single country in the world which possesses a powerful ice-breaker transport fleet that allows a solution of important social-economic tasks of the country's northern regions by maintaining a year-round navigation along the Arctic sea route. The first nuclear ice-breaker "Lenin" was launched in 1956 and commissioned in 1959 as a vessel registered to the Murmansk shipping company.

This year (1997) navigation is the 38th year of navigation in the history of the Russian civil nuclear-powered fleet which consists of seven nuclear ice-breakers and one transport (lighter carrier) ship (Fig.1). The fleet's forefather, the ice-breaker "Lenin," has already retired from operation, while a construction of the latest ice-breaker "50th Anniversary of Victory" is approaching completion now. Main performance indicators of the nuclear-powered ships over the period from 1970 till 1995 are given in Table I.

The total operating record of the marine nuclear reactors under heavy duty conditions (rolling, vibrations, impacts of ice-floes, frequent maneuvering) has exceeded now 150 reactor-years, while that for main equipment items on operating reactors has reached 120 thousand hours. During that period no incidents associated with the chain reaction control violation or the unallowable release of radioactivity have been indicated.

While experience has been accumulated over many years in Russia in the field of marine reactor plant design and construction, cooperation of many participating enterprises has been established. Appropriate infrastructures have been developed to provide maintenance and repair work of nuclear-powered ships at the base of the Murmansk shipping company.

Proven technology and high operational reliability of the ice-breaker reactor plants are achieved mainly by the fact that those plants were developed and constructed for many years by the same enterprises and shipbuilding works that have been constructing nuclear propulsion plants for the Navy

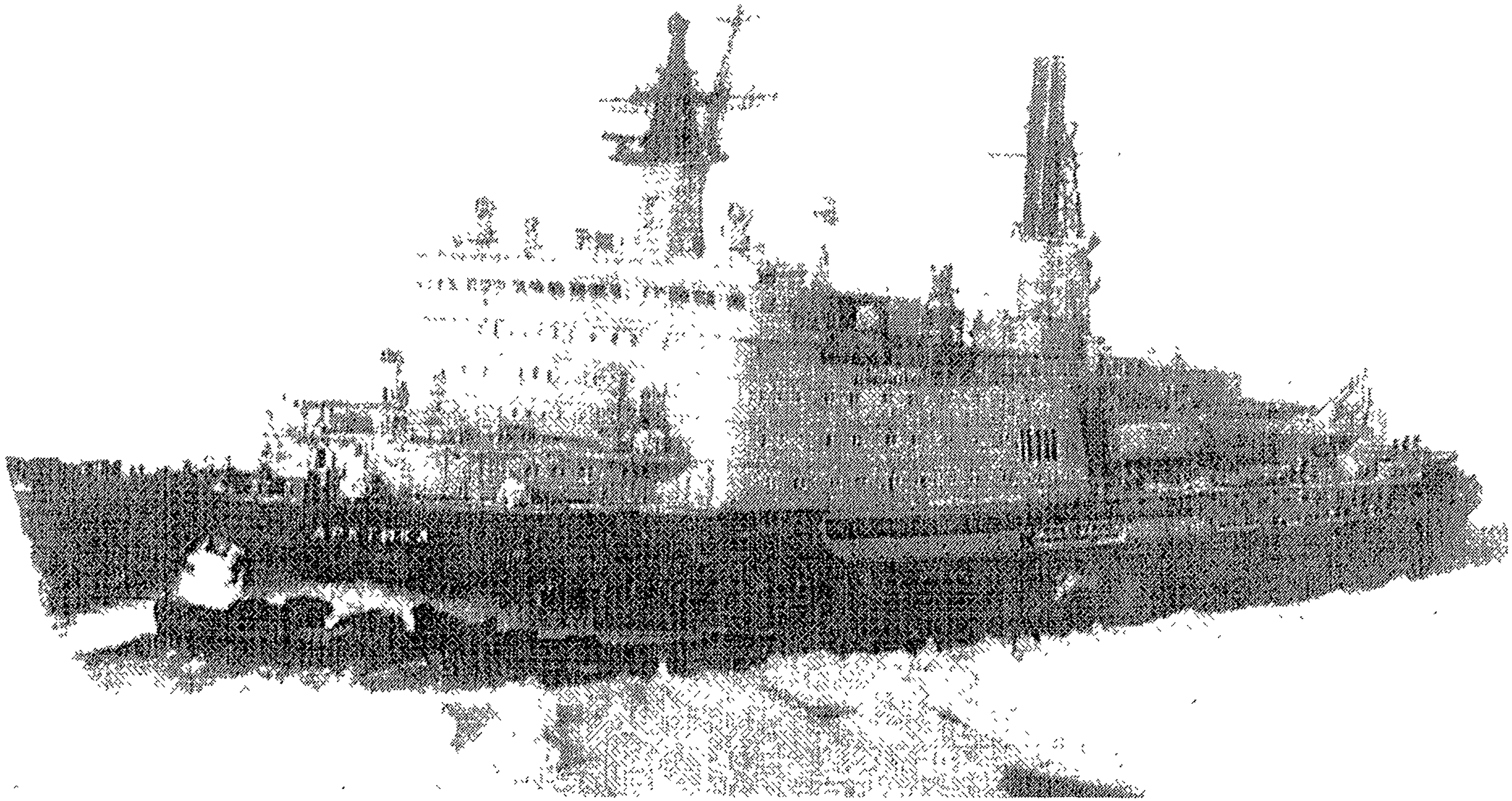


Fig. 1- Nuclear ice-breaker "Arctica" conducting transport ships on the Polar sea route

Table I Nuclear-powered ships performance indicators over the period from 1970 till 1995

Characteristics	Name of ship									
	i.-br. "Lenin"	i.-br. "Arctica"	i.-br. "Sibir"	i.-br. "Russia"	i.-br. "Sov. Souz"	i.-br. "Taymir"	i.-br. "Vaygach"	i.-br. "Yamal"	l.-cr. "Sev-morput"	i.-br. "50th celebration of Victory"
1. Year of commissioning	1970	1974	1977	1985	1989	1989	1990	1992	1988	under construction
2. Averaged duration of operation per year, days	230	225	232	258	232	255	252		291	
3. Total reactor operating record from power start-up, h	106740 106350	**117643*** **118661**	94785 94043	60572 60011	40021 39775	43299	36755	23219 22976	48161	
4. Total energy produced from power start-up, 10^3 MW h	6523 6398	**7756*** **7244**	6095 6933	4294 4472	2243 2668	3453	2754	1329 1321	2756	
5. Distance sailed, miles										
1) total;	654400	744478	740786	391761	221993	247516	126057	143539	198537	
2) incl. through ice	560600	649497	472787	361283	183572	230817	96597	131634	10885	
6. Number of ships conducted	3700	2645	1711	1121	300	627	137	445		

Notes: *** - on 01.01.1997**
i.-br. - ice-breaker
l.-cr. - lighter-carrier

(submarines and surface warships). Most advanced machine - and ship - building technologies were always available there. Therefore the nuclear-powered civil ships were from the very beginning an effective line in the conversion of the defence technologies.

There are various grounds to recommend the marine nuclear reactors as advanced heat sources for floating sea water desalination complexes. They are application of advanced constructional and technological provisions, *continuous* perfection of safety systems to meet updated regulatory requirements for nuclear safety and improvement of equipment items based on operational experience feedbacks.

Drinkable water production became one of the acute problems for many regions of the world, e.g. Northern Africa, Near East, India etc. Also in northern regions of Russia drinkable water has high cost. Extensive work on assessing a potential of economical and efficient utilization of nuclear energy for sea water desalination is currently being conducted under the IAEA umbrella. The former USSR was a leading country in nuclear energy utilization for that purpose (e.g. the nuclear power-desalination complex with the fast reactor BN-350 has been operating since 1973 in Aktau on the eastern coast of the Caspian sea, Kazakhstan). Among a number of different configurations, IAEA's consideration includes a floating power-desalination complex, using a marine nuclear reactor plant of KLT-40 type and the desalination facilities of distillation and reverse-osmosis processes.

Main advantages of nuclear floating desalination complexes are as follows:

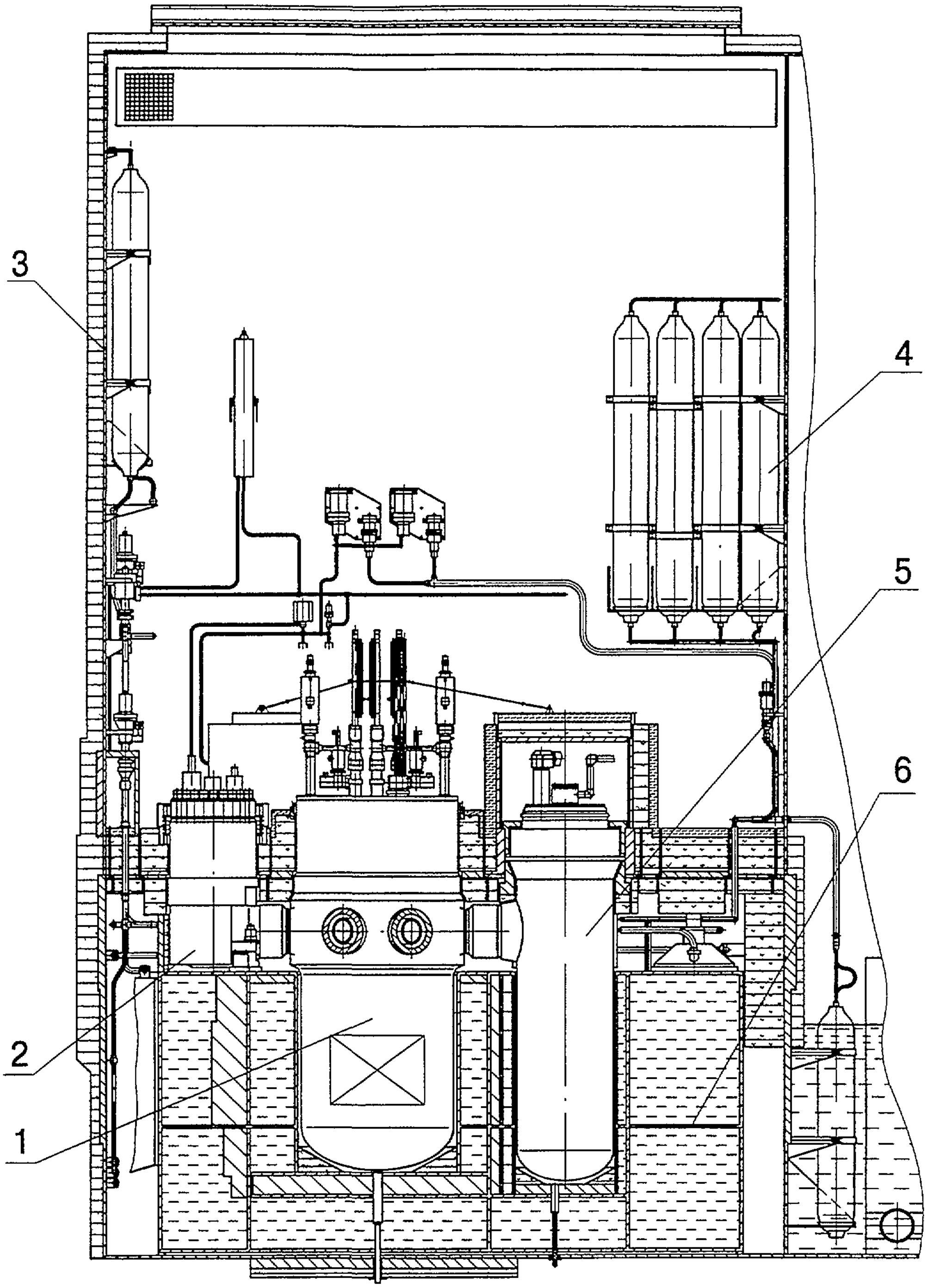
1) high quality of the entire floating power unit fabrications under specialized shipbuilding works conditions followed by delivery to the customer on a turn-key basis;
2) reduction of a station construction period to 4-5 years and of investments as compared with land-base NPPs;
3) a potential for siting at any coastal regions;
4) simplification of antiseismic design measures;
5) cost reduction by serially-produced reactor plants; and
6) simplicity of FNPP decommissioning technology, etc.

Distillation facilities are approximately designed in Russia by the SverdNIIChimMash Institute (Sverdlovsk), while the reverse-osmosis facilities are produced by the Canadian firm "Candesal Inc." A Memorandum of understanding was signed in 1995 and extended in 1997 between "Candesal Inc." and "Russian Ministry of atomic energy (MinAtom)" on the joint design of a desalination complex. At present, potential approaches to an acceptable investment in the project are under consideration, as well as its specific technical aspects.

2. Marine nuclear reactor plant of KLT-40 type

2.1. Technical concept

PWRs which are most widely experienced in the world and their reactor technologies are well proven. KLT-40 is implementing their practice. The plant equipment, i.e., a nuclear reactor, steam generators and reactor coolant pumps are accommodated in a steam-generating unit by short pressure nozzles. The reactor plant is enclosed in a protective shell (containment) designed for a pressure which could emerge as a result of the loss-of-integrity accident of the primary system (Fig.2). The protective shell is housed in a protective enclosure which provides the reactor plant with the protection barrier against the external hazards, like an aircraft crash, a collision with other ship, blast wave impacts, etc.



1- reactor

2- reactor coolant pump

3- protective shell

4- gas cylinders

5- steam generators

6- steam-water shielding tank.

Fig. 2 - KLT-40 nuclear reactor plant

2.2. Safety

The KLT-40-type reactor plant is being designed in conformity with the modern regulatory documentation for nuclear safety (OPB-88, PBYa RU AS-89 etc.). The physical properties of the reactor core (negative reactivity coefficients for power, fuel and coolant temperatures) with a current heat removal capacity maintain the power under self-control without interventions of reactivity control personnel, and the power is self-limited without the aid of the emergency protection system (Vorobiev, 1989).

High heat capacity of the primary coolant and metalwork in combination with the passive residual heat removal system provides an ample time margin for management of hypothetical accidents with complete loss of engineered heat removal means.

Redundant active and passive protective systems are provided for the reactor plant. The inherent self-protection properties of the reactor, and the extensive application of passive and self-actuated safety devices provide the reactor with resistivity against human errors and equipment failures. The reactor plant is equipped with a sophisticated technical diagnostic system.

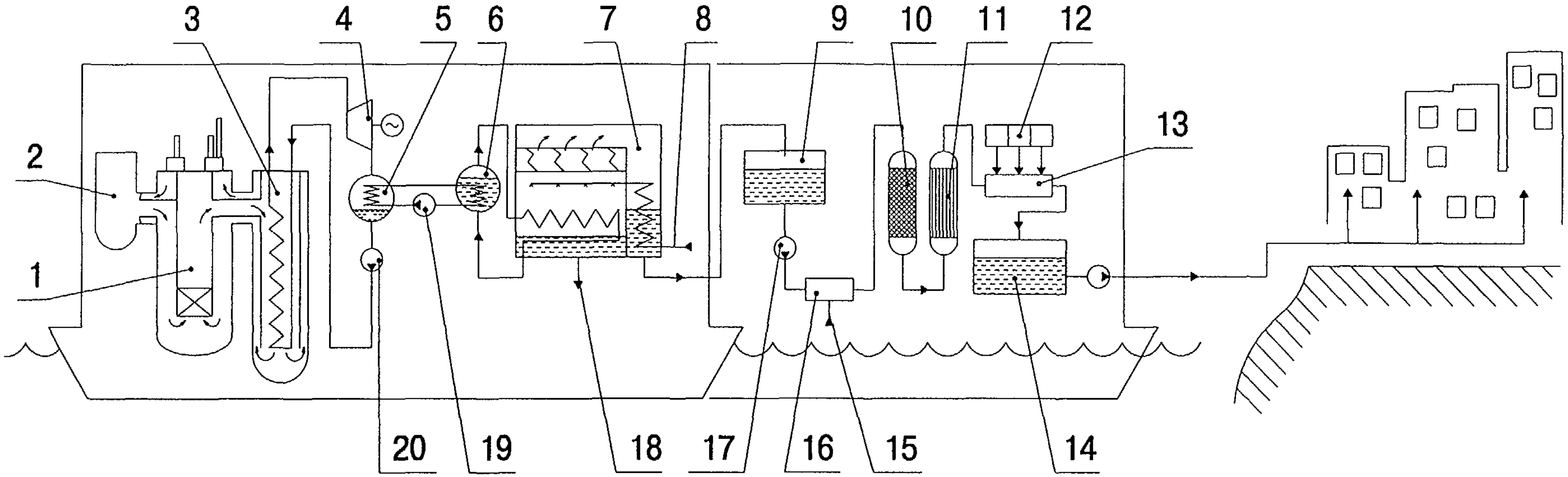
The leak-tight arrangement of the reactor plant prevents releases of noticeable amonunt of radioactive substances during normal operations and accidents. According to the available marine reactor operating experience, small release of radioactivity is conditioned mainly during refueling operations and by air activation under a primary biological shielding.

The exposure dose to an inhabitant at a distance of 1km under normal operation conditions is about 0.01mrem per year. The available systems for radiological safety provision and a multi-barrier system for radioactivity containment limits effectively the radiological consequences of accidents. The population evacuation is not necessary in accidents. Engineering measures in the design, and the enhanced safety corroborated by the multi-year operating experience let the plant to be deployed in the close vicinity of settlements.

Basic design characteristics of the reactor plant are given in Table II.

Table II KLT-40 NSSS basic characteristics

Thermal power, MW (th)	148
Steam flow, t/h	240
Core operating life, h	14600
Refueling interval, yr.	2.5-3
Primary system pressure, MPa	12.7
Core outlet temperature, °C	317
Steam pressure, MPa	3.72
Superheated steam temperature, °C	290
Feed-water temperature, °C	170
Power variation range, %Nnom	10-100
Continuous operation duration, h	8000
Service life, yr.	40



1- reactor

2- primary circuit circulator

3- steam generator

4- turbo-generator

5- condenser

6- steam generator

7- distillation desalination plant

8- sea water inlet

9- distillate intake tank

10- water enrichment facility

11- powdered sorbent filter

12- plant for water fluorination, chlorination and stabilization

13- mixer

14- potable water tank

15- H_2CO_3 solution

16- mixer

17- potable water preparation plant electric pump

18- evaporated sea water brain

19- intermediate circuit electric pump

20- secondary circuit electric pump

Fig. 3- Principal flow diagram of the station with thermal desalination

Table III Basic technico-economic characteristics of floating sea water desalination station

Length of vessel, m	160
Width of vessel (max.), m	44
Draught, m	7
Drinkable water output, m^3/day	80000
Service life, yr.	up to 40
Number of reactors	2
Number of desalination facilities	4
Refueling interval, yr.	2-3
Average load factor	up to 0.85
Staff, persons	60
Term of pilot station creation, yr.	ab. 5
Cost of pilot station creation, million USD	up to 300
Average operation cost per year, million USD	50-60
Cost of 1 m^3 water, USD, not more	2.5
Recoupment period, yr.	8-10

3. Floating water desalination stations

Two options of a floating water desalination station design have been developed: with a distillation desalination facility and with a reverse-osmosis one.

3.1. Floating station for sea water desalination using distillation technology

The station is a special non-self-propelled vessel mounting two nuclear reactor units of the KLT-40-type intended for sea water desalination to be operated under conditions of a protected sea area in combination with a respective land-based support areas at the station destination.

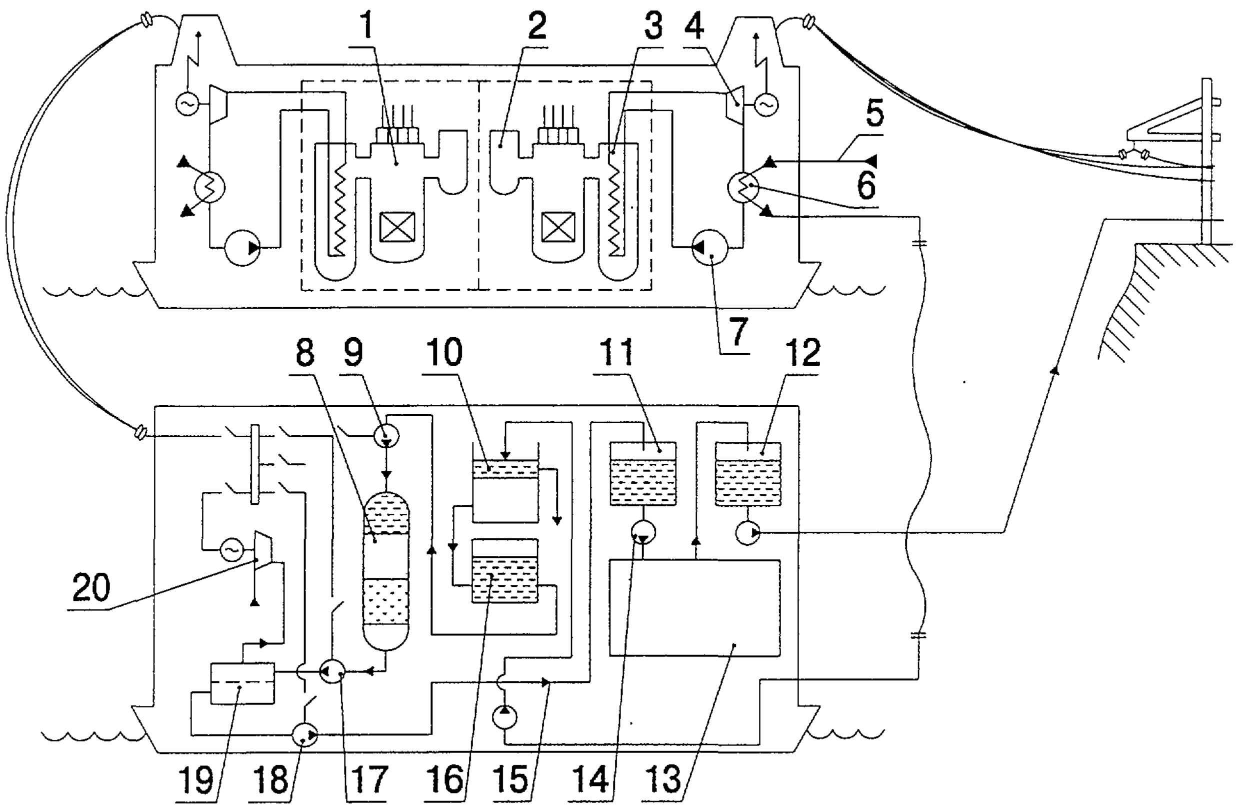
Two reactor units with a rated power of 80MW(th) work on two main turbogenerators with back-pressure turbines (Polunichev, 1995). Waste heat from the turbine condensers is transferred via an intermediate circuit to a twin-unit distillation desalination facility (Fig.3). The desalination unit composes of film evaporators with horizontal tube bundles. Similar evaporators have been successfully operated for many years in the nuclear power-desalination complex at Acktau (Kazakhstan) and at some other sites (Chernozoobov, 1955).

Engineering measures in the design eliminate completely the influence upon the sea and the desalinated water by the reactor unitss. Relatively small quantity of discharged heat during the station operation does not influence the environment.

Basic technico-economic characteristics of the station are given in Table III.

3.2. Floating station for sea water desalination using reverse-osmosis technology

The station is composed of two floating structures: an FNPP with two reactor units of KLT-40 and a vessel for the sea water desalination facility. Part of electricity generated by the FNPP is transmitted to the desalinator vessel to produce drinkable water, and the remainder is directed to the coastal consumers (Fig.4).



1- reactor
2- primary circuit circulating pump
3- steam generator
4- turbo-generator
5- sea water
6- condenser
7- secondary circuit electric pump
8- twin-layer pressure filter
9- booster pump
10- gravity filter
11- filtrate intake tank
12- potable water storage tank
13- potable water preparation unit
14- potable water preparation system electric pum.
15- filtrate
16- clarified water tank
17- high pressure pump
18- fresh water pump
19- reverse osmosis module
20- hydroturbine

Fig. 4 - Principal flow diagram of the complex

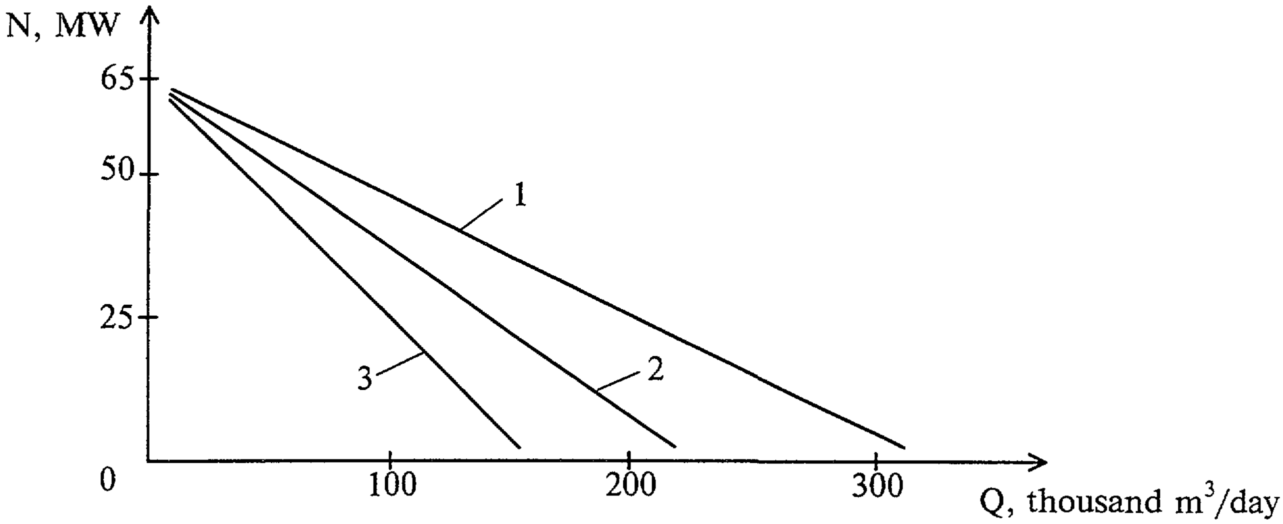


Fig. 5 - Excessive electric power (N) versus desalinated water output (Q) at given thermal power of reactors

1, 2, 3 - specific consumption of power per 1 m^3 of desalinated water - 5, 7, 10 kW.h respectively

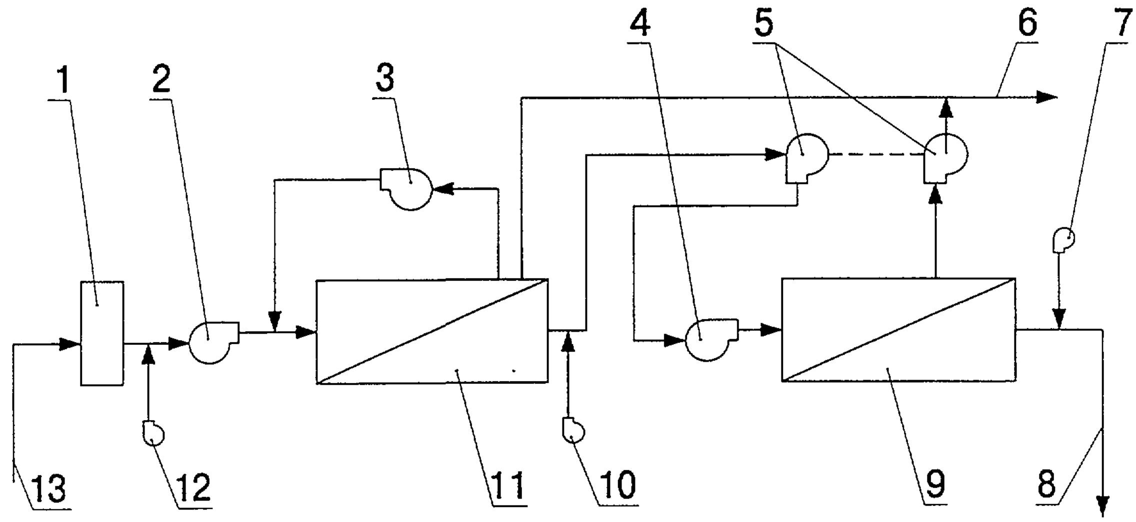
In order to optimize the station for technico-economic performances, different energy allocation can be made between the power and the drinkable water production at the given thermal power of the reactor plants of 2 x 150 MW (th). Possible options of power to drinkable water output relationships are given on Fig.5.

A time span of a pilot station construction is about 5 years, and its investment cost is about 300 million US dollars. One cubic meter of desalinated water will cost about 1 to 1.5 US dollars when the station is operated in a desalination mode.

Membranes which are capable of reducing the salt content from 39-43 g/l (sea water) down to 500 mg/l (distillate) are supposed to be used in the reverse-osmosis desalination facility.

Significant progress in development of reverse-osmosis desalination technologies has been attained by the Canadian firm "Candesal Inc.". The firm envisages implementing the latest sea water desalination technology which reduces significantly both a specific energy consumption for the desalination process and the cost of drinkable water produced (Hamphries, 1995).

The RO system design is based on spiral wound Dow Film Tec high rejection membranes for seawater desalination. Sea water discharged from the reactor's condenser cooling water system at a temperature of 10 ^{o}C above the ambient seawater temperature is used as feedwater for the system. The water first passes through ultrafiltration (UF) pretreatment modules. The filtrate from the UF units is sent to the capacitance tanks, from which suction is taken for the RO modules as feedwater. The feedwater is pumped up to high pressure (1000 psi) and then passes through the RO modules. The permeate from the RO membranes is delivered temporarily into potable water storage tanks, and from there transferred to an off-ship distribution system. Brine concentrate from the RO system is discharged back into the sea. Chemical pretreatment and post-treatment systems for feedwater and product water can be installed as needed (Fig.6).



1- pre-filter
2- medium pressure pump
3- reinculation pump
4- high pressure pump
5- energy recovery system
6- to external consumers
7- chemical additives inlet
8- potable water outlet
9- reverse osmosis membranes
10- anti-fouling additive injection system
11- ultrafiltration membranes
12- prelimiary chemical treatment system
13- sea water inlet

Fig. 6 - Simplified flow diagram of UF/RO desalination system

4. Work organization and financing

Available well proven technologies in the fields of marine nuclear reactors and desalination facilities, their established serial fabrication capabilities, solid experience in their mounting and operation, existing potential of Russian industries and established cooperation of scientific, design and industrial organizations in Russia - all these give the real guarantees of high quality and short term delivery of the station construction (not more 4 to 5 years for a pilot one) and would settle financing issues.

Following options of cooperation in constructing desalination complexes can be possible:

1) Construction of the station by Russian industries with possible participation of foreign and home investors with subsequent repayment of credits during the station recoupment period through sales of electricity or drinkable water produced. If applicable, regional resources (e.g. oil, natural gas, nickel, gold etc.) might be involved in the project investment structure.

   In the present project at Pevek, a pilot FNPP is being financed by the "Rosenergoatom" Concern (a Russian NPPs operator) at the expense of the state budget and loans (about a half of the total investment cost) which may be rendered by home and foreign investors. In this case the station will be a property of Russia.

2) Joint international venture incorporating potential executors and customers (consumers) with shared financing of the project and repayment of investments during the station recoupment period. In this case the floating station would be a property of all participants involved, while Russia will be a designer, vendor and owner of the nuclear reactor plant.

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APPROACH FOR SMART APPLICATION TO DESALINATION AND POWER GENERATION

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Abstract

A 330 MWt integral reactor, SMART, and an integrated nuclear seawater desalination system coupled with SMART are currently under conceptual development at KAERI. The SMART will provide energy to the desalination system either in the form of heat or electricity, or both. The integrated nuclear desalination system aims to produce about 40,000 m^3/day potable water from seawater for demonstration purposes. The remaining energy produced by SMART will be converted into electrical energy. Several important factors are especially considered in the process of SMART and its application system development. The development emphasizes the adoption of technically proven and advanced technology, measures to secure the safety and reliability of the reactor system, consideration of the desalination process for coupling with SMART, a licensing strategy for SMART and the integrated nuclear desalination system, and international cooperation for promoting nuclear desalination with the SMART development program. The current effort to establish the concept of SMART and its application for desalination is being pursued intensively to secure the safety and reliability of SMART, to prove the implemented concepts/ technology considering the coupling with the desalination process, and to formulate an optimum licensing approach. This paper aims to present the technical and strategic approach of SMART and its application system.

Introduction

Electric power generation has become the major area of nuclear energy utilization, and it is expected to remain so in the forseeable future. However, efforts to apply nuclear energy technology to non-electric fields, such as district heating, process heat production, ship propulsion, etc., have been partly successful. Without any promising alternative energy resource, efforts to effectively use nuclear energy will continue for a while. Many countries interested in expanding the areas of nuclear energy utilization have pursued such efforts and have succeeded in realizing such applications.

Korea is one of the countries that has benefited from the use of nuclear energy. Although nuclear energy utilization is currently concentrated on electric power production, its non-electric applications have drawn some interests and various feasibility studies and surveys have been carried out over the past years. One of these studies came up with the promising possibility of developing a relatively small size reactor as the energy source for seawater desalination to solve the potable water shortage problem which is anticipated to occur in the near future.

As a result of these studies, the development of a 330 MWt integral reactor SMART (System-integrated Modular Advanced ReacTor) has been initiated, and it is currently under conceptual development as a national R&D project. A desalination system will be coupled with this reactor to produce potable water from seawater. There are, however, several factors to be carefully considered in developing an integral reactor and an integrated nuclear desalination system. Such factors include the choice of the most promising technical concepts for the reactor, optimization of the coupling of the reactor and desalination system, safety consideration, licensing approach, etc. This paper presents the approaches adopted in the development and design of SMART for the utilization in seawater desalination and power generation.

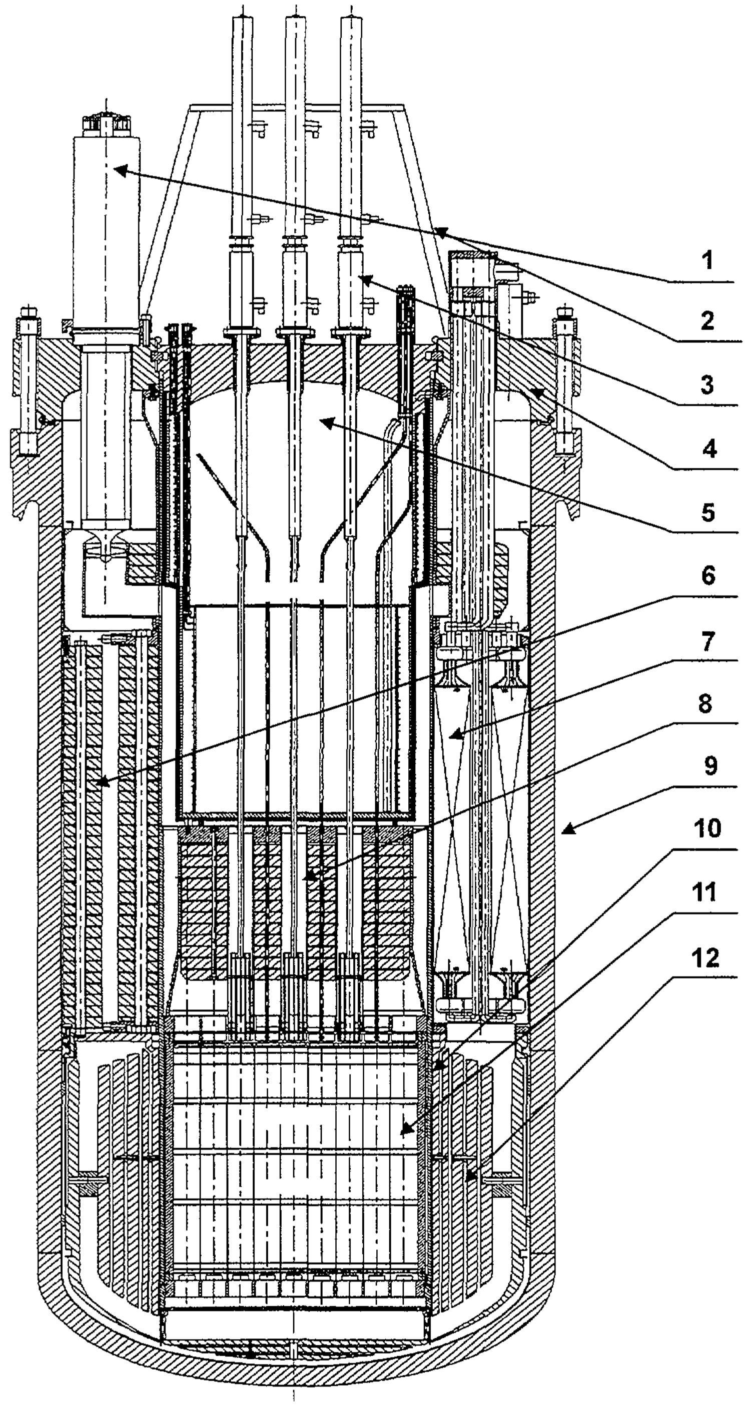

Figure 1. General Configuration of SMART Assembly

1 - MCP (4)	2 - drive support frame	3 - control rod drive (25)	4 - annular cover
5 - pressurizer	6 - displacers	7 - steam generator (12)	8 - shroud tubes
9 - reactor vessel	10 - core support barrel	11 - fuel assembly (57)	12 - side screen

Table I. Conceptual Design Parameters of SMART

Reactor Core		**Reactor Coolant Pump**	
• Nominal thermal power	330 MWt	• Type	Glandless, wetwinding canned motor
• Moderator and coolant	H_2O	• Number of RCPs	4
• Core effective height	2.0 m	• Capacity	1982 m^3/hr
• Core equivalent diameter	1.83 m	• Head	13.5 m
• Core volume	5.27 m^3	• Working medium temperature	310 °C
• Design pressure	17.0 MPa	• Working medium pressure	15 MPa
• Operating pressure	15.0 MPa	**Pressurizer**	
• Average coolant temperature	290 °C	•Type	Self-controlling PZR
• Average power density	62.6 kW/litre	•Operating pressure	15 Mpa
• Core life time	> 3 years	•Operating temperature	120 °C
Fuel		•Control media	H_2O, steam and N_2
• Fuel	5 w/o UO_2	**Residual Heat Removal System**	
• Fuel assembly shape	Square 17x17	•Scheduled reactor shutdown	Turbine bypass
• Number of fuel assemblies	57	•Emergency reactor shutdown	4 train natural circulation
• Average burnup	38.8 MWD/kgU		
• Total uranium loading	13.6 tonne	•Long term cooling	2 train active circulation
Control Rod			
• Number of control groups	6	**Reactor Shutdown System**	
• Absorbing material	$TiDy^{205}$	•Normal shutdown	Control rods
Steam Generator		•Emergency S/D	Control rods/liquid absorber
• Type	Helical once-through	**Emergency Core Cooling System**	
• Number of S/G modules	12	•Not required	
• Steam output	152.4 kg/s		
• Steam pressure	3.0 MPa		
• Superheated steam temperature	274 °C		
• Degree of superheat	40 °C		
• Feedwater temperature	180 °C		

SMART Concepts and its Development Schedule

SMART is an integral reactor with 330 MW thermal power. The internal configuration of the major components in the reactor vessel is similar to that of other integral rectors.

The conceptual development of SMART started in November 1996. This program was established as a national long-term R&D project, its primary purpose being to demonstrate nuclear desalination with relatively small-scale simultaneous electricity generation. The basic design of SMART and its application system includes the establishment of basic models and designs, the development of computer codes used for designs and analysis, the verification of the design feature's performance and reliability throughout the experiments and tests, and the generation of licensing documents.

The general configuration of the reactor assembly is shown in Figure 1, and the conceptual design data of the preliminarily established reactor and its systems are presented in Table I[1].

The safety and reliability of SMART basically relies on passive and inherent safety concepts. These concepts are primarily implemented into the designs of the core and safety systems. The design of SMART with passive and inherent safety characteristics greatly enhances the self-response capability to most reactor transients. Seventy-two (72) hours grace time will provide the operator with sufficient time to respond to the system behavior. One of the important SMART design characteristics is the load follow capability. Transient behavior, such as energy demand reduction by the desalination system, will influence the operating conditions of the secondary system and thus on the primary side. The load follow capability will adequately and safely respond to the change of the energy demand and the consequent system transient behavior.

For severe accident countermeasures, three design concepts are implemented. The reactor safe guard vessel will completely confine any leakage (outflow) of the primary coolant considered in design basis accidents. The external cooling system installed outside the reactor vessel is designed to protect against any leak of the molten core to the containment. The third measure to cope with severe accidents is to install a corium catcher under the bottom of the reactor vessel. These three features will adequately and reliably mitigate the anticipated severe accidents of SMART.

The adoption of proven technologies is one of the basic principles for SMART development. However, for designs using newly developed technologies or new combinations of proven technologies, experiments and tests will be conducted to validate the design features and components for their impact on reactor safety and performance. Experiments currently underway include core flow distribution tests, fuel CHF experiments, the investigation of heat transfer characteristics in the presence of non-condensable gas, and the examination of flow instability in the steam generator. Many other experiments and tests will also be carried out in accordance with the design schedule for SMART. In addition, integral effect tests of the reactor system will be carried out and an approach is currently under investigation. The purpose of these experiments and tests are basically twofold : to provide relevant data for the development and validation of computer codes, and to verify the reliability and performance of the design features. One of many important considerations in reactor design is the manufacturability and stability of the major components, and the ease of component replacements and system surveillance. The designs of major primary components and integrated reactor systems are carefully checked against these important considerations.

A relatively small-size power reactor and its systems (nuclear island) are certain to be less competitive in the power production cost compared to the large-scale power reactor. While SMART is not an exception, the revolutionary and evolutionary technologies to be implemented into SMART, such as passive and inherent safety features, operational flexibility with load follow capability, system simplification and on-shop fabrication of components, will contribute to cost reduction. Furthermore, SMART, as with other small and medium reactors, will have economical advantages by way of multi-unit siting, series production, lower financial risks, and by being a better match to the grid.

The conceptual design of SMART and the coupled desalination system will be completed by March 1999. The following three years are planned for the basic design of SMART. The conceptual design of the integrated nuclear desalination system is to be completed by 2002. Approximately two years of relevant licensing activities will then follow[2].

Coupling of Desalination System with SMART

It is expected that there will not be any technical difficulty in coupling the nuclear reactor system with any of the generally well known desalination processes. However, there are some important factors to be carefully considered, one being the radioactivity carry-over into the product water from the nuclear reactor system, even though the chance of such an occurrence is very slim. When energy in the form of steam is required for the desalination system, one solution to this problem is the adoption of intermediate cooling. Various countermeasures to protect this concern will be thoroughly reviewed and studied and the best option will then be selected for implementation.

A proper method of coping with the feedback effect of the desalination system's transients on the reactor system is another important consideration in the reactor system design. Shutdown or any transient of the desalination system will result in the cutoff or reduction of the energy supply to the desalination system from the reactor system. In the case of nuclear desalination with SMART, this transient feedback can be treated in either of two ways. In the event of the reduction of energy requirements of the desalination system, SMART can quickly respond by re-directing the redundant energy (steam) to the electricity generation system. This energy transfer can be achieved by blocking the steam flow path for desalination. Accordingly, the power generation system will be designed to have the capability to accommodate the full energy produced by SMART.

When the energy produced from the nuclear reactor system is utilized for co-generative purposes, the optimum share of energy by the desalination system and the electricity generation system will be a factor to be considered by the designer. However, the optimum share of the energy by the two systems is strongly dependent on the primary purpose of energy utilization. Thus, this energy share can be dealt with by the proper design of an energy utilization system.

In order to establish the concept of an integrated nuclear desalination system, both the MSF (Multi-Stage Flash distillation) and RO (Reverse Osmosis) processes are being investigated. The MSF requires mainly steam for its operation, whereas RO requires mainly electricity. The current study focuses primarily on the use of the MSF process since the prime objective of the SMART utilization is co-generation. For the target potable water production of 40,000 m^3/day, three desalination cases – that is, one unit, two unit and four unit facilities were evaluated with respect to their major requirements, such as steam consumption, seawater requirements, electricity requirements and required site size. A performance ratio of 8.0 kg dist./2326 kJ was applied to all three cases in the evaluation, and its summary is presented in Table II. As shown in this Table, the results indicate that the single unit facility is marginally more economical than the rest. It should be recognized that the present evaluation is not properly optimized. Once the concept of SMART and its application system, such as the power generation unit and coupling with the desalination unit is established, the optimization for achieving the safety and economy goals with the target production of potable water and electricity will be carried out.

Implementation of the RO process with the pre-heating concept for coupling with SMART is another option under review and investigation. Since the RO process requires energy mainly in the form of electricity, the implementation of RO is relatively simple because SMART is developed as a power reactor. However, further technical investigation and study for the pre-heating concept is needed with respect to establishing the requirements, implementing the concept, and analyzing any impact on the design of SMART and its associated systems. These evaluations will be carried out later.

Table II. Comparison of Desalination Requirements with respect to Unit Size for a 40,000 m^3/day Potable Water Production Target

Item / Units	10,000 m^3/d x 4 units	20,000 m^3/d x 2 units	40,000 m^3/d x 1 unit	remarks
Performance Ratio	8.0 kg dist./2326 kJ	same	same	
Steam Consumption •Sat. low pressure Steam (2.5 ~ 3.0 bar) •Sup. middle pressure Steam (15 ~ 18 bar)	~ 208,000 kg/hr ~ 6,000 kg/hr	~ 208,000 kg/hr ~ 5,000 kg/hr	~ 208,000 kg/hr ~ 4,000 kg/hr	
Required Seawater	max. 20,000 ton/hr	max. 18,500 ton/hr	max. 17,000 ton/hr	
Temperature Rise of Cooling Water at Sea (Summer Design)	max. 8.5 °C	same	same	•Domestic Regulation : max. 9.0 °C
Required Electricity	5 kWh/ton	same	same	
Size of Site (W x L) •Evaporator size per unit •Total plant size	9 x 60 m 100 x 130 m	17 x 65 m 80 x 100 m	25 x 75 m 65 x 110 m	• Storage tank site excluded • For the double decker evaporator, about 35% evaporator size can be reduced
Construction Time	24 ~ 30 months	same	same	

Licensing Approach for SMART and the Nuclear Desalination System

Korea has a well-established licensing and regulatory system for nuclear electricity (power) generation. New policies and technology are continuously developed and implemented in the system by updating existing guidelines, methodology, rules, and practices. However, the present licensing and regulatory systems basically apply to the loop-type nuclear reactor and its power generation system, and no regulatory guide or associated practice exists for an integral reactor like SMART and the nuclear desalination system. This situation can impose practical difficulties for achieving licensing approval on the integral-type reactor and its application system.

In order to resolve the expected licensing problems regarding SMART and the integrated nuclear desalination system, two approaches are being taken simultaneously. In the first approach, the licensing body partakes in the review of the design concepts and features of SMART throughout its development. This is intended to cultivate a better understanding of any licensing issue. In this approach, the licensing body also reviews and investigates any design characteristics of other similar integral reactors under development in other countries. Also, possible licensing issues and unreliable technology recognized by the licensing body will be thoroughly discussed between specialists, including the developer. If any concept or design feature is determined not to be acceptable, the issue will be fed back to the concept for modification. Besides the review of design concepts and features by the licensing body, technology utilized for developing SMART, such as methodology, computer codes, etc., will be provided to the licensing body for review and examination. The licensing policy in Korea plans to introduce a pre-licensing review system and to stimulate the active utilization of this system. This new licensing system is expected to ease the resolution of any licensing issue on SMART and its application system, and also, to shorten the time for licensing.

The second approach is being undertaken by the licensing body itself. The licensing body will begin its own program of establishing new licensing guidelines and an associated system starting in 1998. This program has been already established as one of the national R&D programs on the domestic needs to be prepared for the coming future. The need has been requested in accordance with on-going large-scale reactor development programs, such as the development of a large-scale next generation power reactor, liquid metal reactor, and also of SMART. The purpose of the program with respect to small and medium reactors, including SMART, is to establish detailed safety requirements and licensing guidelines. Several subjects included in this program are : the establishment of safety concepts, search of outstanding safety issues, analysis of regulating requirements, and development of licensing/regulatory guidelines. The outstanding findings obtained from these studies will be furnished as useful information and will serve as good suggestive references for SMART development. This program is thus expected to contribute largely to the efficient development of SMART.

The close cooperation between the licensing body and the reactor developer will contribute to the successful licensing achievement of SMART and the integrated nuclear desalination system.

International Cooperation

As far as safety issues are concerned, the coupling of two well established technology-nuclear energy and desalination technology, will not present any severe technical difficulty. The existing small-scale nuclear desalination facilities in Japan and Kazakistan's long record of nuclear desalination operations are good examples. However, the commercial implementation may face some difficulties not regarding safety issues, but with regards to such issues as financial risk, economic competitiveness, technological reliability in coupling, and licensing concerns. These factors may impose critical adverse impacts on introducing nuclear desalination. The best approach for the resolution of these concerns will be inter-country or inter-organizational cooperation.

The success of international cooperation will depend strongly on the level of common objectives for and mutual benefits achievable from such cooperation. The thorough understanding of a partner's requirements becomes the basis for the cooperation. Typically, the cooperation can be implemented in one of the two ways : joint execution of common interest programs or technology transfer. A combination of the two methods is also possible, and is, in general, the most efficient and practical method for implementating nuclear desalination. This method will produce mutual benefits and will provide the opportunity for technology exchange in nuclear, as well as desalination, technologies. Furthermore, the major concerns mentioned above are expected to be resolved by cooperative efforts.

The development of SMART and its application system can be advocated as a cooperative program by promoting the participation of interested countries or organizations. The cooperation can also be extended to the demonstration of nuclear seawater desalination with SMART. As the first step of cooperation, the direct participation of interested countries or organizations through development phases can be one possible cooperative approach. The areas of participation in the development phases will embody the basic model development of demonstration plants, including SMART and the integrated desalination system, technology development including experiments and tests, and the licensing application of models, technology and associated works. Cooperation for the construction of a demonstration plant can be carried out in the subsequent phase or in a separate cooperative phase. Site selection, detailed design of the demonstration plant with an application of site conditions, manufacturing and installation of components, construction, and test operation of the plant are the work scopes for the cooperation. The cooperation or joint execution of the program during the development phases can be in the form of financial contributions, participation by experienced engineers, provision of proper test facilities, or execution of required tests. For the construction phase, the participants or the cooperative partners can provide a site for the demonstration plant, and participate in the detailed design works and construction. The participant benefits from the cooperation can be identified through discussions and agreement. They may include sharing design data and technical information, royalty pay-back, technology transfer, etc. The type of benefits will be dependent on the cooperation method.

In addition, any technical subject on the development of SMART and its application system can be performed jointly with interested foreign partners through the IAEA's CRP and other programs. KAERI is also willing to participate in any relevant international program to promote the use of nuclear energy for non-electric areas, including nuclear seawater desalination. International cooperation will be a short cut to opening a new era for nuclear energy utilization.

Summary

The non-electric application of nuclear energy and its associated technology has received much attention from worldwide nuclear industries. Some countries possessing the related technology have successfully applied to non-electric fields.

Current Korean efforts in the non-electric application of nuclear energy focus on nuclear seawater desalination. However, all nuclear power reactors currently under operation and construction, and even under design, are large size reactors dedicated solely to electric power generation. Large nuclear reactors of electric power generation are not well suited for the various non-electric applications of nuclear energy, including seawater desalination. In this regard, a national R&D program to develop a relatively small-scale reactor for non-electric applications has been established. Through this program, a 330 MWt integral reactor, SMART, is currently under conceptual development, with the primary objective of utilizing energy in seawater desalination with simultaneous electricity production. SMART adopts highly advanced technology, such as passive and inherent safety features, to enhance its safety, reliability, and performance. All technologies implemented into SMART will be proven through experiments and tests, including an integral performance test.

The integrated nuclear desalination system coupled with SMART is also under investigation, with an emphasis on the key interface conditions that impact the design conditions of SMART. The precursory review of the coupling with the MSF process has been carried out and the coupling of the RO process with a preheating concept will be investigated. MSF requires mainly steam for its operation, whereas RO requires mainly electricity. SMART can provide either steam or electricity or both to the desalination system, with little technical difficulty. In order to facilitate the licensing process and to resolve any licensing issues of SMART and its application system, joint cooperation between the licensing body and its developer is being undertaken. Concurrently, the licensing body will establish related regulatory guidelines and rules by carrying out an independent program.

International cooperation is actively being sought and it is expected to greatly contribute to the early implementation of nuclear desalination. The development of SMART and its application system is open to joint-cooperation with any interested foreign country or organization. The SMART development program will successfully contribute to the establishment of a new era for non-electric application of nuclear energy, especially in the area of nuclear desalination.

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DESALINATION DEMONSTRATION PLANT USING NUCLEAR HEAT

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Abstract

Most of the desalination plants which are operating throughout the world utilize the energy from thermal power station which has the main disadvantage of polluting the environment due to combustion of fossil fuel and with the inevitable rise in prices of fossil fuel, nuclear driven desalination plants will become more economical. So it is proposed to set up nuclear desalination demonstration plant at the location of Madras Atomic Power Station (MAPS), Kalpakkam. The desalination plant will be of a capacity 6300 m^3/day and based on both Multi Stage Flash (MSF) and Sea Water Reverse Osmosis (SWRO) processes. The MSF plant with performance ratio of 9 will produce water (total dissolved solids (TDS-25 ppm) at a rate of 4500 m^3 /day from seawater of 35000 ppm. A part of this water namely 1000 m^3/day will be used as Demineralised (DM) water after passing it through a mixed bed polishing unit. The remaining 3500 m^3/day water will be mixed with 1800 m^3/day water produced from the SWRO plant of TDS of 400 ppm and the same be supplied to industrial/municipal use. The sea water required for MSF & SWRO plants will be drawn from the intake/outfall system of MAPS which will also supply the required electric power pumping. There will be net 4 MW loss of power of MAPS namely 3 MW for MSF and 1 MW for SWRO desalination plants. The salient features of the project as well as the technical details of the both MSF & SWRO processes and its present status are given in this paper. It also contains comparative cost parameters of water produced by both processes.

Introduction

In India nearly three-fourth of the water out of total rain fall of 4000 km^3 runs off to the sea and one-fourth is stored as surface as well as ground water for use. The annual water availability and the consumption can be seen from Table I. Various schemes have been launched by State and Central Government and Municipal Corporations for collecting, storing and distributing rain water specially in water scarce areas for industrial, agricultural and domestic uses. Some of these schemes with capacity & cost are mentioned in Table II. These schemes need large investment and have longer period of construction and cost overruns. Also there is serious environmental concern due to submergence of vast land and displacement of people from the areas. For coastal water scarce region and areas affected by brackishness, desalination plants based on membrane as well as thermal processes can suitably be deployed. Also to augment the water resource, effluent treatment plants using membrane technology have been used worldwide at comparatively low cost. A few desalination plants and effluent treatment plants have already been installed and are operating in India and some plants are under construction for augmenting water as given in Table III. Table IV gives capital & water cost of MSF & Effluent Treatment Plant in India & Table V gives water tariff in Mumbai & Chennai. Mostly these plants are based on membrane processes. One desalination plant based on thermal MED process is operating at Chennai.

MSF desalination plants need steam for heating sea water in brine heater and it is normally drawn from thermal power station. The cost of the steam is generally estimated based on the power loss due to steam supply for desalination plant. A 6300 m^3/d (1.4 MGD) combined MSF and Reverse Osmosis (RO) desalination plant is being set up at Kalpakkam. It would draw steam from Madras Atomic Power Station (MAPS) and use sea water from its outfall system. The reasons for coupling the desalination plant with nuclear power plant are given in the next paragraph.

Table I : Annual water consumption and availability in India (Km^3)

a. Consumption :

Sl. No.	Needs	1985	2000	2025
1	Irrigation	470	630	770
2	Domestic	16.7	24.2	40
3	Industries	10	30	120
4	Power	4.3	5.8	15
5	Miscellaneous	39	60	105
	Total	540	750	1050

b. Water Availability (Km^3)

Total rainfall (Km^3)	Utilizable water (Km^3)	
	Surface water	Ground water
4000	700	350

Table II : Conventional water supply schemes & cost

Sl. No.	Water supply scheme	Capacity (MGD)	US million $
1	Indira Gandhi Canal	--	1000
2	Sardar Sarovar Scheme	500	1715
3	Telegu Ganga Scheme	200	640
4	Krishna Vally Corporation	3000	2040
5	Mumbai Municipal Corporation	100	215
6	RO Sewage Treatment Scheme at Chennai	30	163

Table III : Desalination & RO Plants in India

Sl. No.	Process	Capacity	Location	Supplier	End use
1	RO	225 m^3/h	Madras refinery	Hindustan Dorr Oliver	Test sweage treatment
2	RO	360 m^3/h	Madras refinary	Nuchem weir	-do-
3	RO	360 m^3/h	National Fertilizer, Guna	Ion Exchange	Effluent treatment
4	SWRO	2650 m^3/d	Porbandar	Nu Chem Weir	Sea water desalination
5	SWRO	360 m^3/d	GSFC, Sikka	Thermax	Sea water desalination
6	MED	1650 m^3/d	EID Parry, Chennai	IDE, Israel	Sea water desalination
7	MSF	425 m^3/d	Mumbai	BARC	Sea water desalination
8	128 RO Plants	300 m^3/d	In village of different states	Drinking Water Mission	Drinking
9	SWRO	4500 m^3/d	GEB, Sikka	Ion Exchange	Industrial
10	100 Nos. of RO/NF/UF Plants	10 - 100 m^3/d	Industries	Indian Cos.	Industrial
11	SWRO	4500 m^3/d	Ramanathapuram	BHEL	Drinking
12	MSF-RO	6300 m^3/d	Kalpakkam (under construction)	BARC	Drinking and process water

Table IV: Capital cost and water cost of large MSF desalination and effluent treatment plant in India

Sl. No.	Plant capacity (MGD)	Capital cost million US $	Specific Capital cost million $/MGD
1	1	5.14	5.14
2	5	21.43	4.28
3	20	57.14	2.85
4	30*	163.43	5.43
5	Sea water desalination cost (MSF)		1.48 $ / m^3
6	Sea water desalination cost (RO		1.28 $ / m^3
7	Brcakish water desalination (RO)		0.71 $ / m^3
8	Effluent water tertiary treatment cost		0.57 $ / m^3

*Effluent Treatment Plant

Table V : Water tariffs in Mumbai and Chennai

Mumbai		
Domestic	:	US $ 0.10 / m^3
BARC (R & D)	:	US $ 0.3/ m^3
Industries	:	US $ 1.0/ m^3 + 50 % Sewerage charge
Chennai		
Domestic	:	US $ 0.30 / m^3
Industries	:	US $ 0.80 / m^3

Nuclear Power Plant & Desalination Plant

Nuclear power plants produce large quantities of heat (in the form of steam/hot water) at relatively low cost. Cost of fuel oil on the other hand is high and gaseous emissions from fossil fuel burning pollute the atmosphere. Operational expenses of nuclear power desalination plant are lower than the conventional thermal power desalination plant. High degree of safety & reliability in operation of nuclear power plants have been now achieved. India has nine operating nuclear reactors with a total generating capacity of about 2000 MW. These are mostly pressurized heavy water reactor (PHWR) each of cpacity 220 MW operating in various parts of India near sea shore using natural uranium (UO_2) as fuel and heavy water as moderator and primary coolant. The secondary coolant is water/steam at pressure & temperature of 40 bar and 250^0C respectively. Dual purpose nuclear power & demonstration plants can be installed to produce both electricity and fresh water from sea water using low pressure steam extraction from IP/LP turbine. Also the hot sea water from the reject system can conveniently used as feed to sea water Reverse Osmosis plant.

The Process and Salient Features of the Project

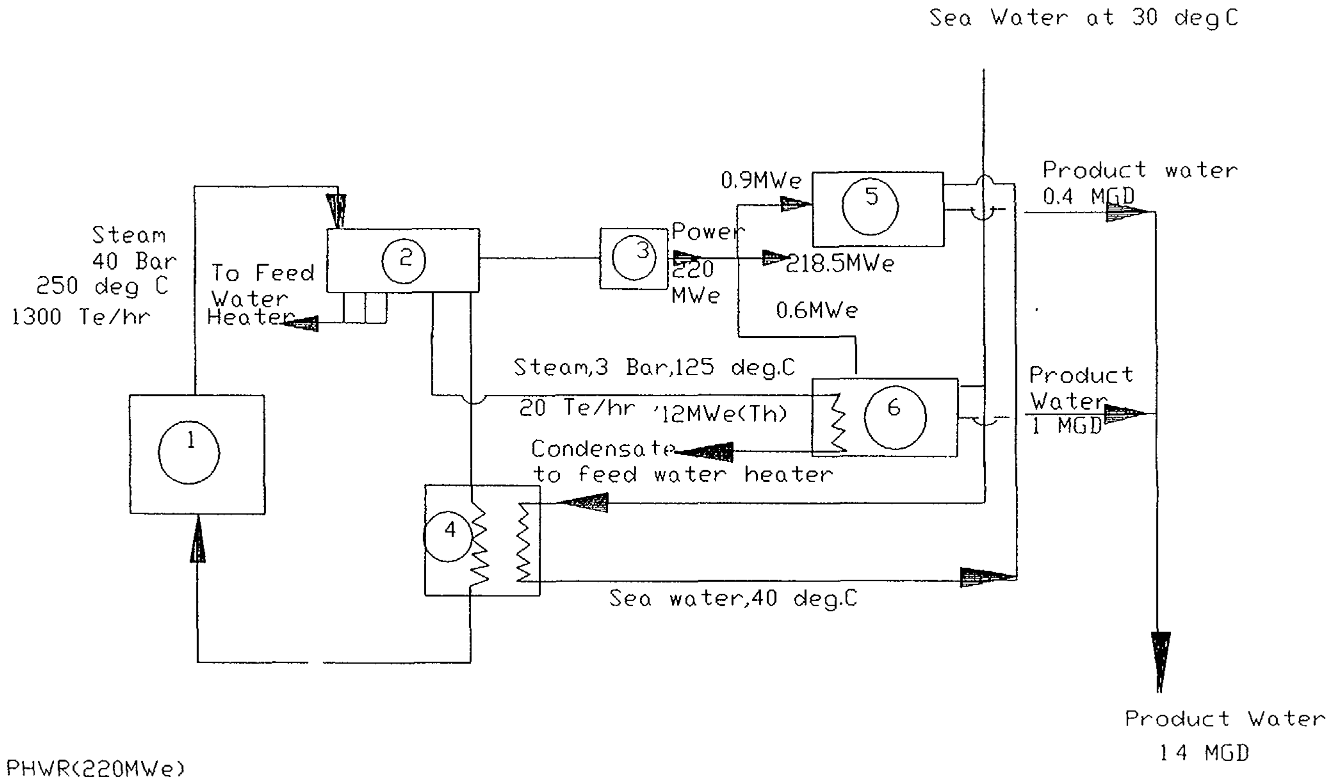
The flowsheet of the 1.4 MGD desalination plant is given in Fig. 1. Steam and the power are drawn from Pressurised Heavy Water Reactor (PHWR) 200 MW at MAPS. Steam at a pressure of 3 bar at 125^0C is drawn from turbine. The product water from MSF plant will be at a rate of 1 MGD. The electrical power loss for MSF plant will be 3.0 MWe including the loss due to withdrawal of steam. The electrical power requirement by SWRO plant will be 1.0 MWe. So the total power loss from MAPS will be 4.0 MWe which will produce 6300 m^3/d of water per day.

MSF Plant

The MSF plant consists of 9 recovery modules each of 4 flash stages and one reject module of 3 stages. Maximum brine temperature is limited to 121°C to prevent calcium sulfate scaling on cupronickel heat transfer tables. Pumps are made of SS 316. The MSF plant will draw sea water at a rate of 1544 m^3/hr from the outfall system and the stream (130 °C, 3 bar pessure) from MAPS at a rate of 20.6 Te/hr and produce product water of 187.5 m^3/hr with less than 25 ppm.

The MSF plant has a performance ratio of 9. The pumping power requirement is 3 kWh/m^3 of water produced. The water produced from MSF plant will be of high quality with TDS of 10 ppm. The part of this water (1000 m^3/day) will be after passing it through a polishing mixed bed unit used as boiler make up water. At present MAPS is producing 940 Te/d of DM water at a cost of $4.2 per m^3. If high purity water from MSF plant is used for making DM water, about 80% of cost can be easily saved.

The remaining 3500 m^3/day of product water from MSF plant will be mixed with 1800 m^3/day product water from the SWRO plant and the mixed stream containing around 250 ppm TDS will be supplied to industrial/municipal use in Kalpakkam. Layout of the desalination plant project is given in Fig. 2. Table VI gives the sea water composition at MAPS, Kalpakkam.



1 PHWR(220MWe)

2 Turbine

3 Generator

4 Condenser

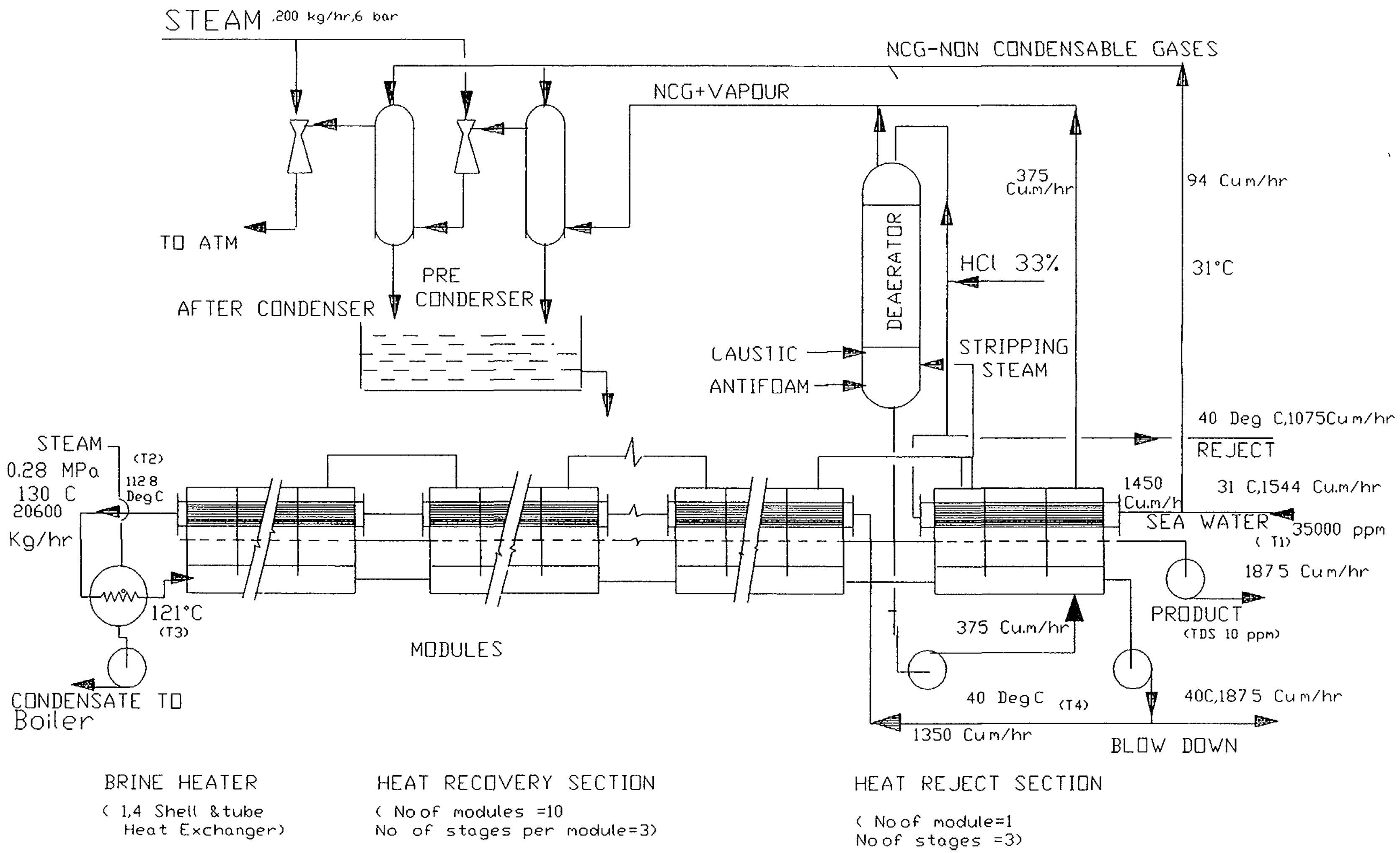
5 SWRO Plant(0.4 MGD)

6 MSF Plant (10 MGD)

Total power loss =0.9+(12/3)+0.6 MWe Electrical

FLOW DIAGRAM OF 14 MGD MSF-SWRO DESALINATION PLANT COUPLED TO 220 MWe COASTAL NUCLEAR POWER STATION

Figure 1



4500 Cu.M/Day MSF Desalination Plant,Kalpakkam

Figure 2

Table VI : Sea water composition at MAPS Kalpakkam

pH	8.1
Total dissilved solids by weight	35600
Total solids (ppm) by wt.	36012
Suspended solids (ppm) (by difference)	410
Total hardness (ppm $CaCO_3$)	6300
Sodium (ppm)	10556
Calcium (ppm)	400
Magnesium (ppm)	1272
Potassium (ppm)	380
Total alkalinity (ppm $CaCO_3$)	138
Chloride (ppm)	18981
Sulphate (ppm)	2650
Fluoride (ppm)	1.3
Iron (pph)	100
Silica (ppm)	0.8

Process

The cold sea water from the outfall system of MAPS is pumped at a rate of 1450 m^3/hr through the tube bundle of the heat reject section (3 heat reject stages). Before it passes through the tube bundle of reject stages, part of the seawater (94 m^3/hr) is used in pre- & inter-condensers. 375 m^3/hr of warm sea water (40^{0}C) from the reject module is subjected to chemical dosing and is sent to the vacuum deaerator. The remaining part of warm sea water (1075 M^3/hr, 40^{0}C) is sent back to the sea (Fig. 3).

The chemical dosing consists of addition of hydrochloric acid to decompose bicarbonates so as to prevent alkaline scale formation on heat transfer surfaces. In vacuum deaerator, the dissolved CO_2 and O_2 are removed to bring it to the level of 1 ppm and 20 ppb respectively. The deaerated feed is then mixed with caustic soda to neutralise excess acid to pH of 6.8 to 7 and a small quantity of antifoaming is injected to avoid foaming during flashing of brine. The deaerated feed is then mixed with recycle brine. It is then passed through the tube bundle of recovery module (9 nos) where it is heated externally by condensation of flashed water vapour. The temperature of recycle brine rised to 112^{0}C which is further heated to 121^{0}C in brine heater. This brine is then gradually passed through all the 36 nos. of stages where it gets flashed & vapours are produced which is then condensed on outside of the tubes and form the product water. Recovery modules are rectangular in shape, long tube design and are arranged in the form of a train. There are total 9 recovery modules and each module has got 4 brine stages. It is made up of carbon steel with sufficient corrosion allowance. The tubes are made of 90/10 cupronickel, 19 mm o.d. and monel demisters are used to separate the brine droplets from the water vapour produced due to flashing. The pumps are made of 316 stainless steel; tubesheets are made of 50 mm thick 90/10 cupronickel.

Noncondensable gases are removed from evaporators by evacuation system. A series of venting is utilized to remove all the gases and to maintain pressure differential in stages. The product water is pumped from last stage and is passed through lime column (calcite bed) before it is distributed. Here it will be mixed with product water from SWRO plant & then will be sent as drinking water.

SWRO Plant

The Sea water Reverse Osmosis (SWRO) plant will receive hot sea water (36-38°C) of 35000 ppm from the condenser outlet of Madras Atomic Power Station (MAPS) at a rate of 215 m^3/hr and it is passed through clarifloceulater, pressure sand filter, activated carbon filter and cartridge filter in order to

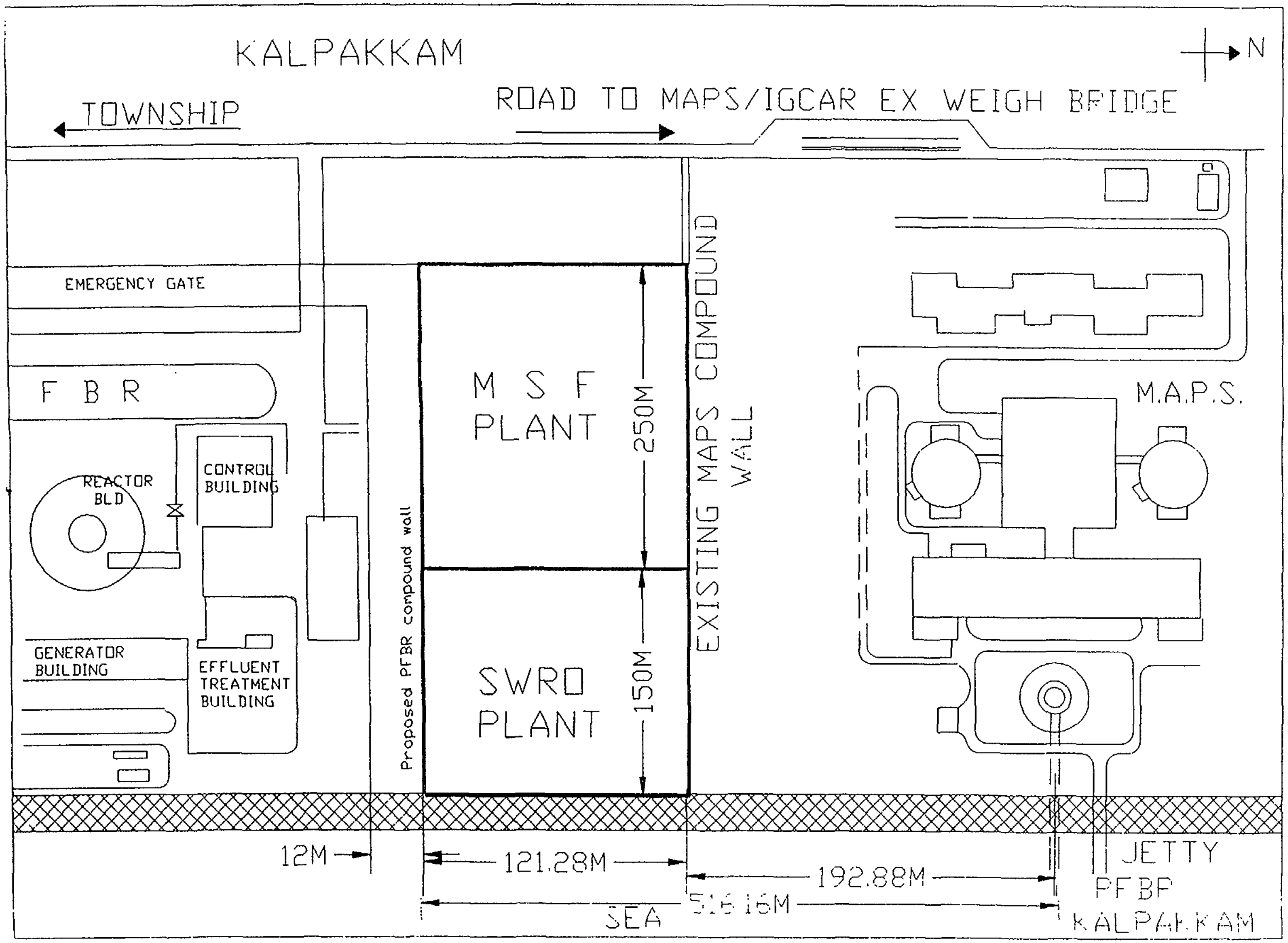

KALPAKKAM
N
TOWNSHIP
ROAD TO MAPS/IGCAR EX WEIGH BRIDGE
EMERGENCY GATE
F B R
REACTOR BLD
CONTROL BUILDING
GENERATOR BUILDING
EFFLUENT TREATMENT BUILDING
Proposed PFBR compound wall
M S F PLANT
250M
SWRO PLANT
150M
EXISTING MAPS COMPOUND WALL
M.A.P.S.
12M
121.28M
192.88M
JETTY
PFBR
KALPAKKAM
SEA

Figure 3

remove suspended and colloidal solids and organics. Since the membrane are polyamides, dechlorination of the feed is carried out by addition of sodium sulfite and to minimize carbonate scaling on the membrane, acid dosing is carried out followed by antiscalant dosing to prevent sulfate scaling. The pretreated seawater is then pumped into two parallel sections through the RO modules at a rate of 110 m^3/hr at a pressure of 55 kg/cm^2 by use of high pressure pumps which is fitted with Energy Recovery Hydraulic Turbocharger (HTC). Use of HTC saves about 30% of energy. Membrane modules are 8040 HSY SWC with TFC spiral wound having solute rejection of 99.6%. There are a total of 26 modules having 156 membrane elements. Each pressure tube has 6 elements and the pressure tube are made of FRP. The product water (75 m^3/hr) has a TDS 50 ppm.

The SWRO plant will receive the hot sea water of 35000 ppm from the condenser outlet of Madras Atomic Power Station (MAPS) and produce potable water of about 500 ppm. Provision is kept to mix the SWRO product water with highly pure MSF water to have drinking water of 200-300 ppm. The SWRO plant will use hot sea water at a temperature of about 36 - 38°C as feed. High temperature feed will increase the membrane flux considerably which will in turn will reduce the membrane cost for a particular plant capacity.

Process Description

The hot chlorinated sea water at a temperature of 36-38^{0}C from the outfall of Madras Atomic Power Station (MAPS) is pumped through the clarifier and pressure sand filter. Large size particles upto 25 micron are removed from sea water at this stage at a rate of 215 m^3/hr. It is then passed through activated carbon filters for removal of organics. It is then passed through 5 micron cartridge filter to ensure the removal of particles below 5 micron in size. Since the membranes are polyamide, dechlorination of sea water is carried out by addition of $NaHSO_3$. To minimize the carbonate scaling, acid dosing is carried out followed by addition of antiscalant or SHMP for removal of sulfate scale.

Table VII : Technical specifications of 4500 M^3/day (1 MGD) MSF plant

(i)	Plant capacity	187.5 m^3/hr
(ii)	Product quality	< 25 ppm of salt
(iii)	Top brine temperature	121^{0}C
(iv)	Blow down temperature	40^{0}C
(v)	Performance ratio	9
(vi)	Steam consumption	20.6 Te/hr
(vii)	Pumping power consumption	600 KWe
(viii)	Power loss to power station due to steam withdrawal for desalination plant	2.4 MW(e)
(ix)	Scale control	Acid treatment
(x)	Flash evaporator	Rectangular, long tube design
(xi) (a) (b) (c) (d) (e)	No. of recovery modules No. of flash stages/module No. of reject module No. of stages Total no. of flash stages	9 4 1 3 39
(x)	Tubes	Cupronickel 90/10
(xi)	Pumps	SS 316 make centrifugal pump

Table VIII : Technical specifications of 1800 M^3/day SWRO plant

(i)	Product out	75 m^3/hr
(ii)	Product quality	500ppm
(iii)	Feed sea water flow	215 m^3/hr
(iv)	Feed sea water TDS	35000 ppm
(v)	Membrane element (a) Type (b) Model (c) Element capacity	 TFC spiral wound 8040 HSY SWC/TFC 2822 SS 22 m^3/day/ element
(vi)	Product recovery	35%
(vii)	Design pressure	55 Kg/cm^2
(viii)	Solute rejection	99.6% at standard sea water test condition (32800 ppm NaCl, pH = 7.5, 25^{0}C)
(ix)	No. of elements required	156 nos
(x)	No. of elements per module	6
(xi)	Total no. of modules	26

Table IX. Flux characteristics of the membrane with temperature

Temperature (^{0}C)	Membrane flux litre/m^2/day (lmd)
30	709
35	815
40	922

This pretreated sea water is then pumped in two parallel sections through the modules at a rate of 110 m^3/hr each at a pressure of 40 bar. Each pump is fitted with Energy Recovery Hydraulic Turbocharger (HTC). Maximum pressure of the feed is 55 kg/cm^2. About 30% of the energy is saved due to the use of HTC. The sea water membrane module is 8040 HSY SWC with TFC spiral wound with solute rejection of 99.6%. There are a total of 26 module having 156 membrane elements. Each pressure tube has 6 elements. Shells are made of FRP. The product water of TDS 450 ppm after degassing to effect CO_2 removal, is dosed with lime or soda ash to adjust pH or mixed with product water from MSF plant whose dissolved salts are as low as 250 ppm.

Table VII and VIII describe the speficiation of the MSF and RO plants. Table IX indicates the performance improvement of RO plants. Table IX indicates the performance improvement of RO plant with rise in feed water temperature.

PROGRAMME AND ACTIVITIES ON NUCLEAR DESALINATION IN MOROCCO

Pre-project study on demonstration plant for seawater desalination using nuclear heating reactor in Morocco

XA9848808

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Abstract

The first part of this paper gives the general information on the pre-project study of a demonstration plant for seawater desalination using a heating reactor being assessed jointly by MOROCCO and CHINA. The progress of the pre-project study is elaborated in the second part.

1. INTRODUCTION

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

To that end, Morocco and China decided, with the assistance of IAEA, to perform jointly a pre-project study concerning an MED (Multi-Effect Distillation) plant coupled to a 10 MW (th) (Mega-Watt thermal) heating reactor. Such a project has been recommended as one of options for demonstration in the IAEA's Options Identification Programme.

2. MAIN CHARACTERISTICS OF PRE-PROJECT-STUDY

In order to perform this project, the following two documents have been established:

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

The agreement was signed by authorities of both sides on the 20th September, 1996 in Rabat, Morocco.

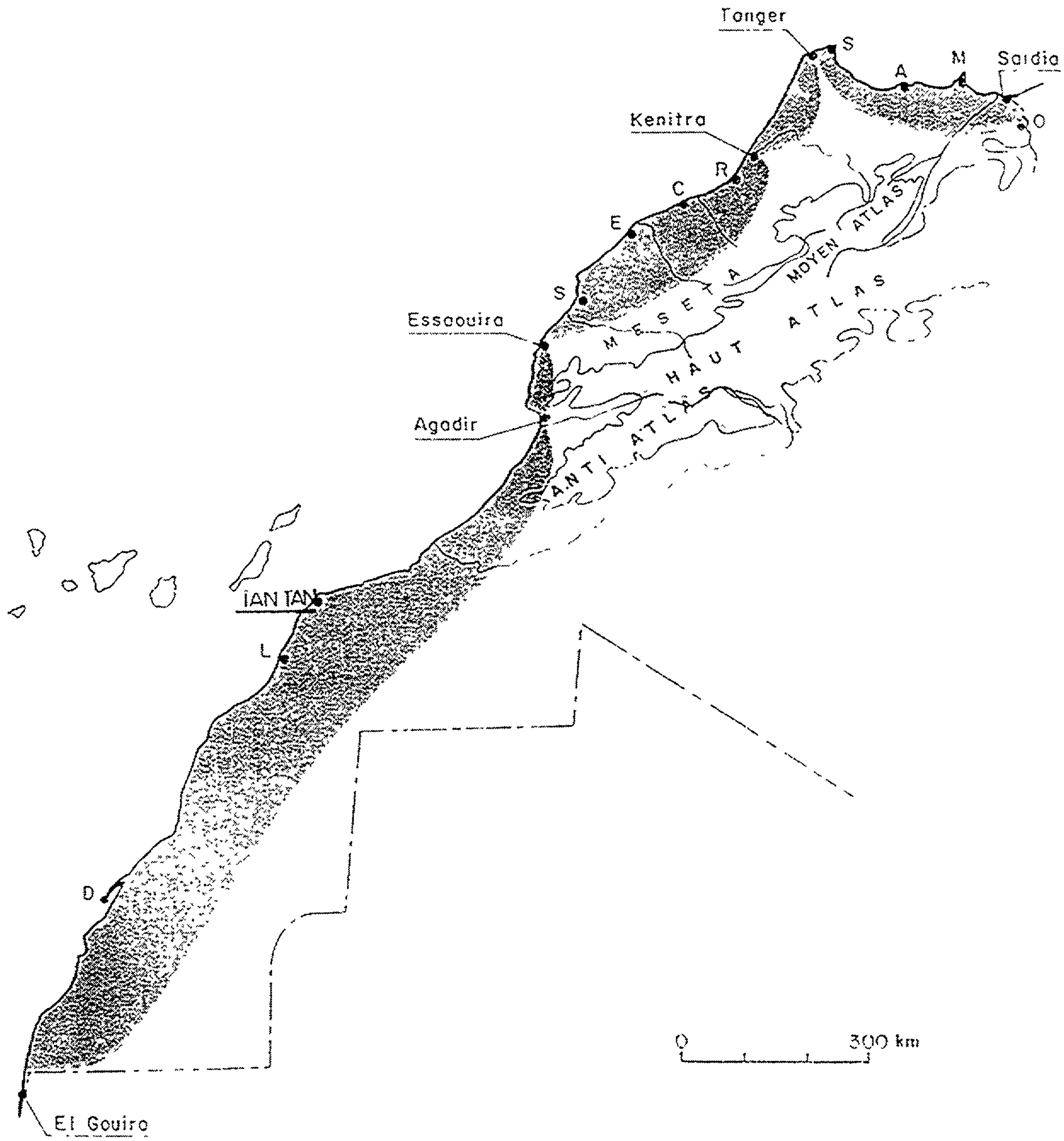
2.1 The objectives of a demonstration plant

The objectives of a demonstration plant are:

- To build up technical confidence in the utilization of a nuclear heating reactor for desalination of seawater; and
- To establish a data base for reliable extrapolation of water production costs to a commercial nuclear desalination plant of the same combination using a 200 MW (th) heating reactor to produce 140 000 m^3/day of desalted water.

2.2 The capacity and the cost for produced potable water

Considering the above-mentioned objectives of demonstration, the capacity and the water production cost have been proposed to meet the following features.


Tanger
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El Gouira
0
300 km

Figure 1

- the production capacity of the demonstration facility will be approximately 8 000 m^3/day,
- the water production cost of the demonstration plant should be evaluated and extrapolated to a commercial scale nuclear desalination plant,
- the water production cost of a commercial nuclear desalination plant should be competitive with fossil options.

2.3 The technical and the economic aspects of demonstration

In the pre-project study, various technical and economic aspects of the propsoed demonstration plant will be evaluated, in particular:

- The coupling scheme of a nuclear heating reactor with an MED desalination plant;
- Safety features of the nuclear desalination system;
- Facility maintainability of the palnt; and
- Economic competitiveness of produced water and providing elements for economic analysis.

3. PERIOD OF THE PRE-PROJECT STUDY

The study was started upon signature of the agreement between Morocco and China for the scheduled period of 18 months starting from the 20th September 1996. The Moroccan side has organized one guiding committee composed of several Moroccan departments. At present, the studies are ongoing and the request of the Technical Assistance of the IAEA has been approved. This IAEA Technical Assistance includes both the experts mission and the scientific visits.

4. THE PROGRESS OF THE PRE-PROJECT STUDY

4.1 Siting

Two candidate sites have been chosen at the TAN-TAN beach county, located in the south of MOROCCO (see Fig. 1). Those coastal sites are located about 330 km south of the city AGADIR. The collection of data and information related to the technical, environmental, and economical aspects of the chosen sites are being studied. During the latest expert mission on siting, held in September 1997, those sites have been visited, and the mission report showed that both sites are suitable for locating the proposed demonstration plant of seawater desalination.

4.2 Reactor study

The MOROCCAN committee has undertaken calculations on the neutron physics and thermal hydraulics related to the 5 MW (th) reactor. As a result of this study, the committe has chosen the 10 MW(th) reactor. The design of the proposed 10 MW (th) heating reactor is based on the 5 MW (th) version which has been in operation in China. The features and design of the proposed reactor are still under going led by the China side.

4.3 Desalination system

The MED (Mulfi-Effect-Distillation) process has been adopted for the demonstration plant by both Moroccan and Chinese sides in consideration of its advantages. The call for cooperation partners for the pre-project study on the desalination process system has been launched. The bids must be evaluated later at an appropriate timing. The study is ongoing.

XA9848809

TECHNICAL AND ECONOMIC EVALUATION OF NUCLEAR SEAWATER DESALINATION SYSTEMS

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Abstract

The IAEA COGENERATION/DESALINATION COST MODEL spreadsheets were used for the economic evaluation of sea water desalination plants coupled with small and medium size nuclear reactors developed in RDIPE. The results of calculations have shown that the cost of potable water is equal to or even below 1$/m^3. This is very close to similar indices of the best fossil driven desalination plants. For remote and difficult-to-access regions, where the transportation share contributes significantly to the product water cost at fossil plants, the nuclear power sources of these reactor types are cost-efficient and can successfully compete with fossil power sources.

1. INTRODUCTION

For a number of years RDIPE has been developing nuclear power plant projects of a small and medium size, such as RUTA-TE [1], UNITHERM [2,3] and NIKA [4]. The said NPPs are intended to solve power supply problems for small settlements located mostly in remote and difficult-to-access regions where the transportation share contributes significantly to the fossil fuel cost. Nuclear power sources of this type are, in this case, cost-efficient and can successfully compete with fossil power sources.

Small and medium size NPPs can be successfully used for solving the problem of fresh water supply by means of sea water desalination. The given work presents feasibility study results aimed at finding an optimum combination of a nuclear power source and desalination plant and also defining the conditions for reasonable, in terms of cost-efficiency, implementation of nuclear sea water desalination projects with application of nuclear power sources developed by RDIPE.

2. SPECIFICATIONS OF NUCLEAR POWER SOURCES

2.1. RUTA-TE reactor plant

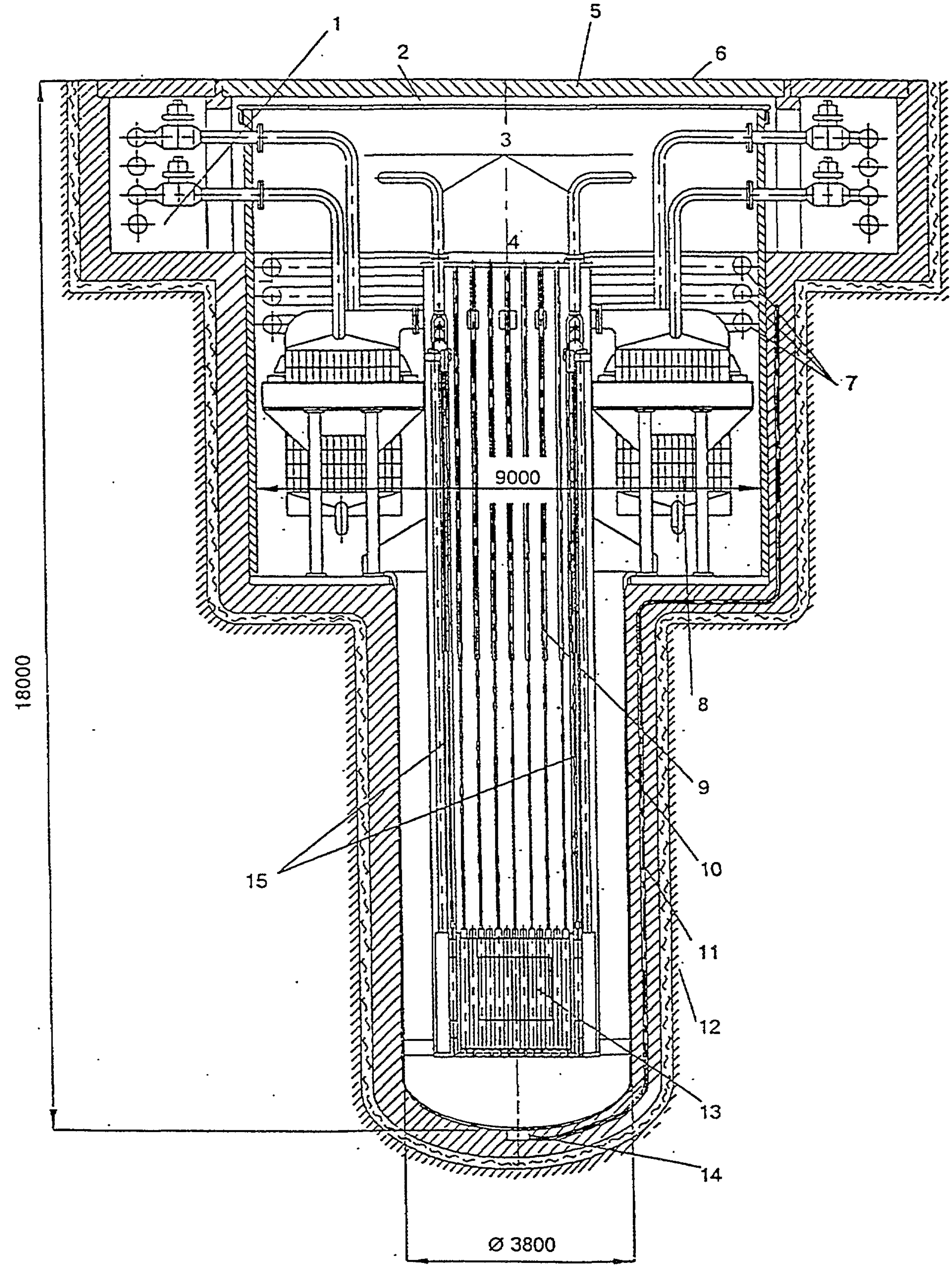
The RUTA-TE reactor is a pool-type thermal reactor whose distinctive features are a high level of safety and reliability being characteristic for pool-type reactors, a relatively low cost of manufacturing and capability to produce some amount of electric power alongside with heat generation. Key specifications of the reactor are given in Table I. Fig. 1 presents the reactor schematic diagram.

2.2. NIKA-type reactors

The NIKA-type reactors under development are integrated PWRs of new generation notable for higher reliability and safety. There are two options for their use: for ground-based and floating NPPs. Table II presents their characteristics. The NIKA-120M NSSS design for a floating NPP is shown in Fig. 3.

2.3. UNITHERM transportable NPP

The UNITHERM reactor represents a small-size PWR whose distinctive features are transportability (delivery to the site as the ready-made blocks), a unique service life (20 years) with a single nuclear fuel loading for remote difficult-to-access places without use of local cooling water



1 - headers and valves compartment, 2 - reactor lid, 3 - feed/steam piping, 4 - water level, 5 - protective plate, 6 - ground surface level, 7 - pressurizers, 8 - primary heat exchanger, 9 - control rod drive, 10 - pool lining, 11 - concrete, 12 - ground, 13 - core, 14 - leakage monitoring device, 15 - modular channels

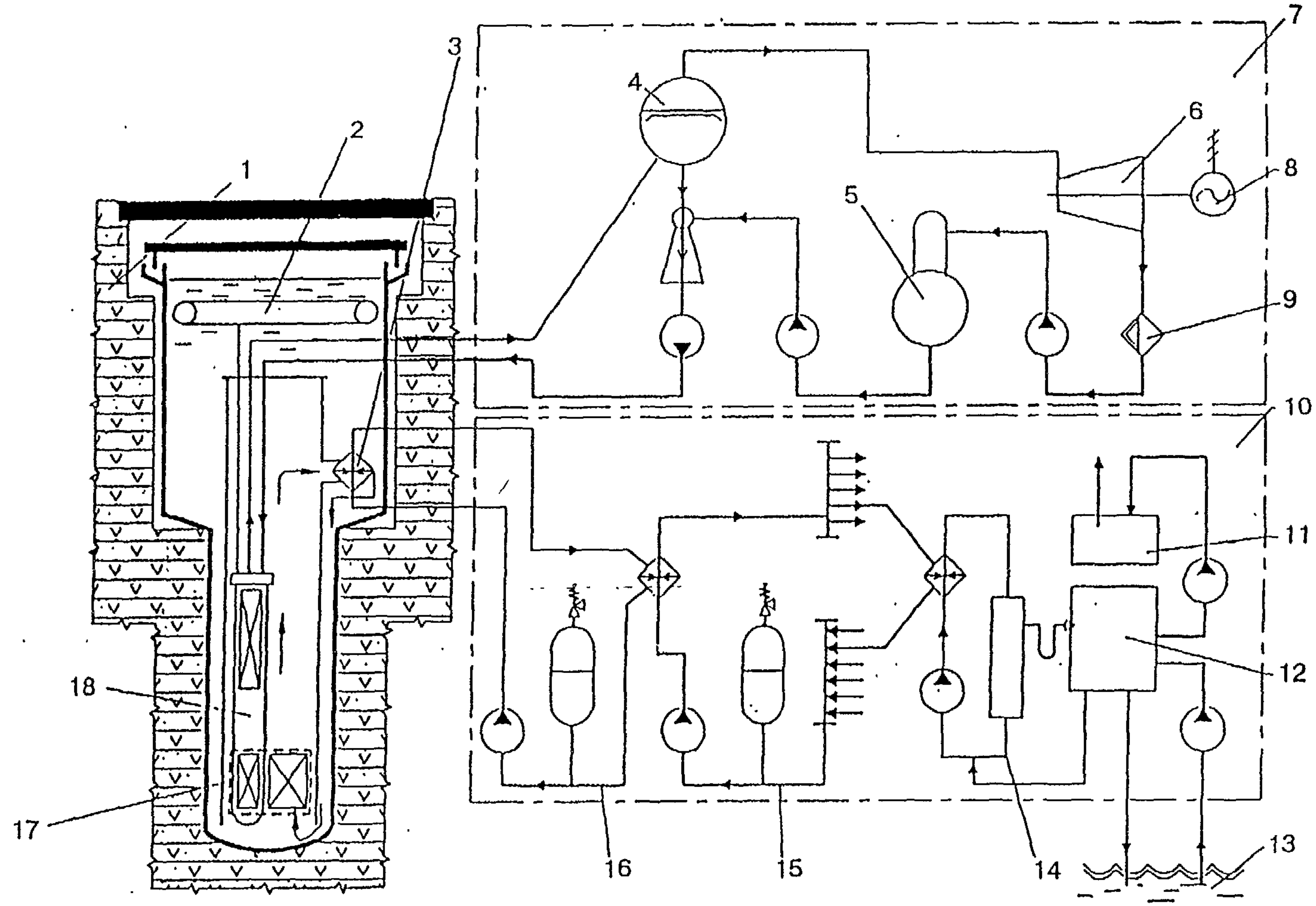
Fig. 1. Reactor RUTA-TE.

Table I Main characteristics of RUTA-TE

Parameter	Value
Total core thermal power, MW(t)	70,4
Net reactor heating capacity, MW(t)	65,5
Net reactor electrical power, MW(e)	3,5
Steam generating capacity, kg/s	6,5
Superheated steam pressure, MPa	3,0
Superheated steam temperature, ^{0}C	263
Feed water temperature, ^{0}C	160
Pool water parameters: ■ inlet/outlet temperature, ^{0}C ■ pressure at the core level, MPa ■ water volume in the pool, m^3	 74/101 0,228 600
Modular channel water parameters(max.): ■ inlet/outlet temperature, ^{0}C ■ pressure at the core level, MPa ■ water volume in the channel, m^3	 274/306 9,8 0,07
Average thermal capacity of the modular channel, MW(t)	192,3
Number of modular channels	78
Number of fuel assemblies in the central part of the core	169
Turbine efficiency, %	26,7
Temperature of reactor plant direct hot water, ^{0}C	85
Temperature of reactor plant return water, ^{0}C	60

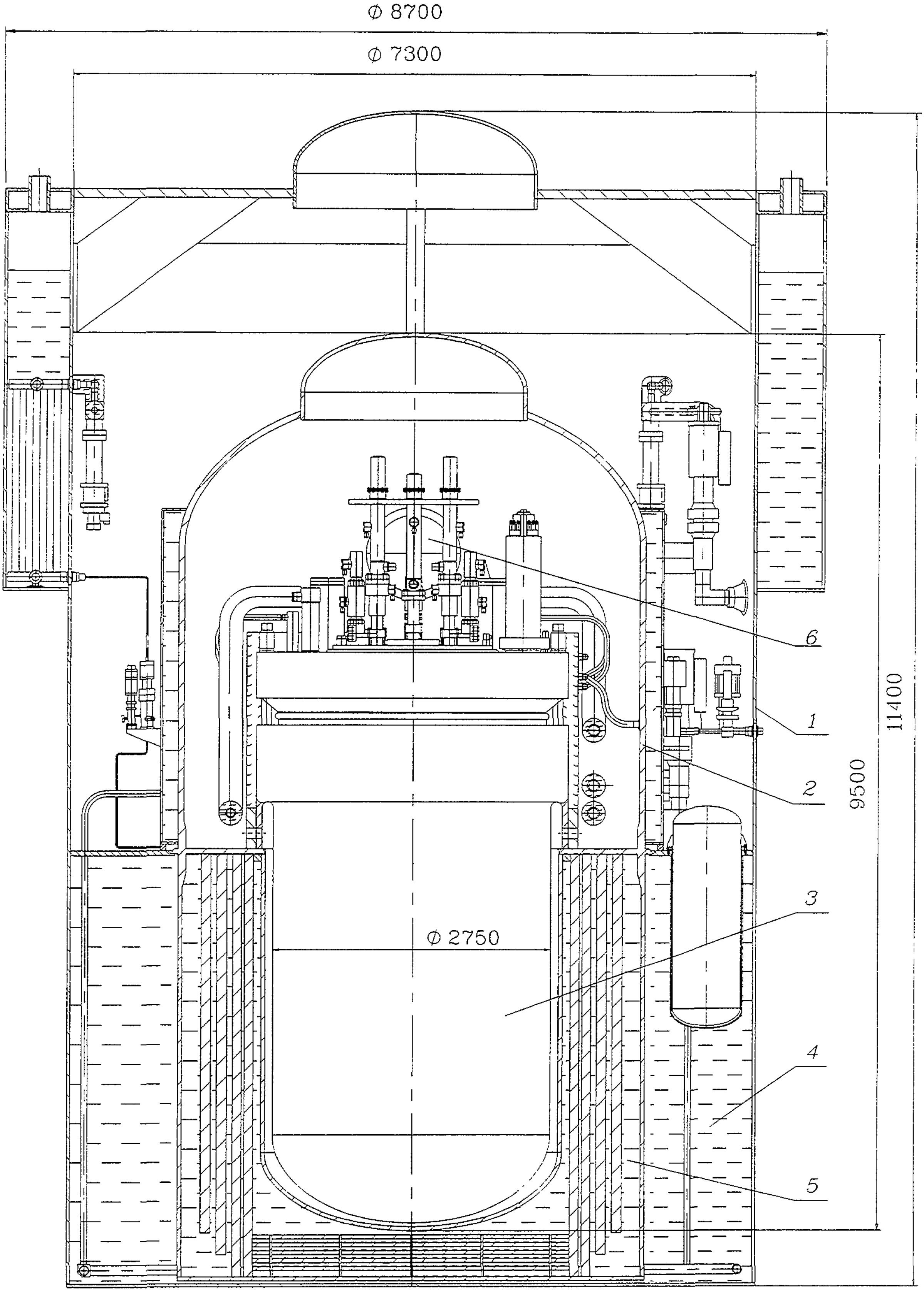
Table II Design characteristics of NSSS of NIKA type and of UNITERM.

No.	Characteristic	NIKA-120M	NIKA-300	UNI-TERM
1	Thermal power of the core, MW(t)	70	300	15.3
2	Steam generating capacity, kg/s	25	138.8	5.6
3	Superheated steam pressure, MPa	3.0	3.0	1.2
4	Superheated steam temperature, at least, °Ñ	274	274	210
5	Feed water temperature, °Ñ	60	180	45
6	Nominal pressure in primary circuit	15	15	16
7	Primary coolant temperature while operating at nominal power, °Ñ: at core inlet at core outlet	 260 300	 270 310	 245 325
8	Operating range of power change, % $N_{nom.}$	20...100	20...100	25...100
9	Effective campaign of core, years	4	4	20
10	Fuel: U^{235} enrichment, % U^{235} load, kg specific power rating, kW/l	 19.7 251 40	 5 637 65	 21 160 15
11	Service life, years	30	60	40



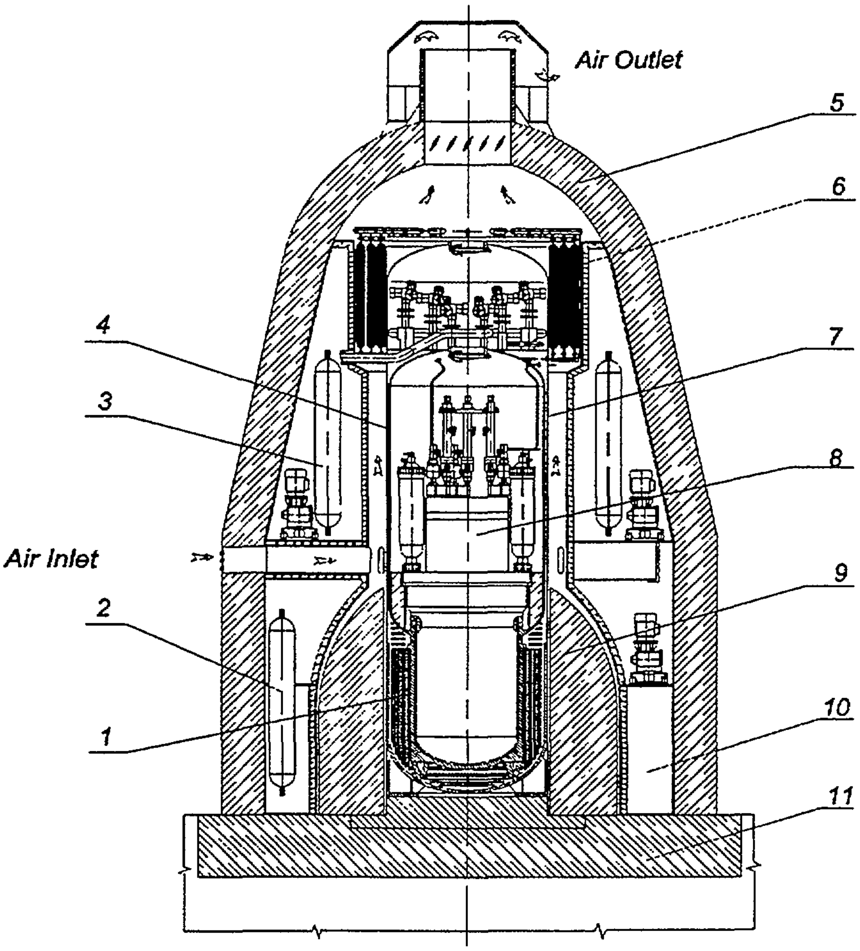
1 - reactor island, 2 - pressurizer, 3 - primary heat exchanger, 4 - separator, 5 - deaerator, 6 - turbine, 7 - turbine island, 8 - generator, 9 - condenser, 10 - desalinating island, 11 - fresh water storage, 12 - water distillation plant, 13 - sea, 14 - vaporization circuit, 15 - third circuit, 16 - secondary circuit, 17 - core, 18 - modular channel

Fig. 2. Co-generating plant flow diagram.



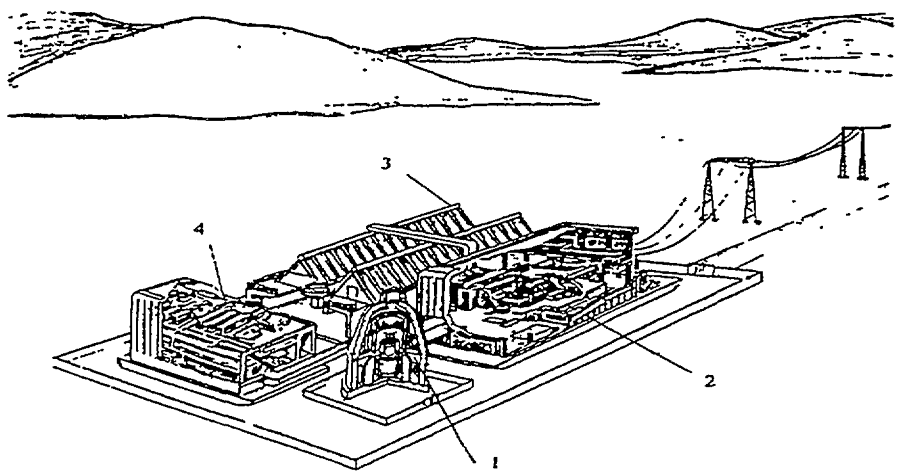
1 Containment
2 Safeguard vessel
3 Reactor
4 External tank
5 Internal tank
6 Entrance hatch

Fig. 3. Longitudinal section.



1 - iron-water shielding tank; 2 - radioactive gases storage cylinders; 3 - liquid absorber supply system; 4 - containment; 5 - shock-proof casing; 6 - cooldown system heat exchanger; 7 - safeguard housing; 8 - steam generating unit; 9 - biological shielding blocks; 10 - liquid and solid wastes storage tanks; 11 - basement

Fig 4. Reactor plant.



1 - reactor plant; 2 - turbine; 3 - air cooler condenser; 4 - main control room

Fig. 5. UNITHERM NPP.

sources, the principle of "green grass" on completion of functioning, and high-level safety. The characteristics are given in Table II. The UNITHERM NSSS design is presented in Fig. 4 and the UNITHERM NPP design is shown in Fig. 5.

3. CALCULATION TECHNIQUE AND INPUT DATA

The economic evaluation of sea water desalination for complexes using nuclear reactors developed by RDIPE were performed by the IAEA COGENERATION/DESALINATION COST MODEL spreadsheets[5]. Basic input data are given in Table III. Some features of calculations are as follows.

An optimum maximum brine temperature at which the cost of fresh water becomes lowest was determined first while calculations for multi-effect distillation (MED) and hybrid water plants were run. The remaining input data for desalination plants were taken from the default values in the spreadsheet programme.

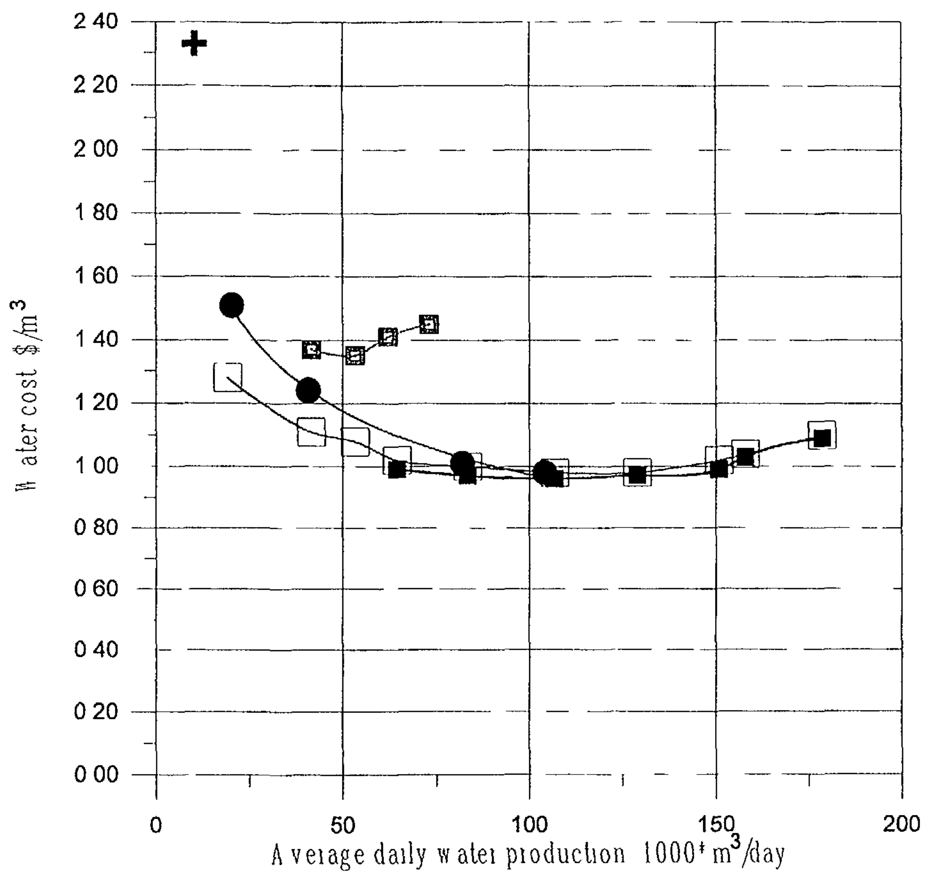
Calculations for fossil driven desalination plants were also performed to compare them with performances of nuclear driven desalination plants. For this, the cost of fossil fuel was assumed to be 15.5 $/BOE.

The given spreadsheets were modified to calculate the RUTA-TE nuclear plant, since they did not envisage a power plant model with an independent heat and power output. The modification consisted in reducing to zero the work lost on turbine shaft due to failure of steam to expand up to the condensation temperature of 37 °C because such losses are not available here.

The UNITHERM NPP secondary circuit consists of two loops. The first loop serves to drive an electric generator of the turbine. The second loop serves for supplying heat as steam or hot water. Since the IAEA spreadsheets do not envisage such a reactor arrangement, it was assumed for the calculations that the whole energy is removed via the first loop to the turbine but the second loop is not actuated. The UNITHERM NPP is expedient for delivery in a two-unit version. The calculations were just performed for this version.

Table III Input data for calculations

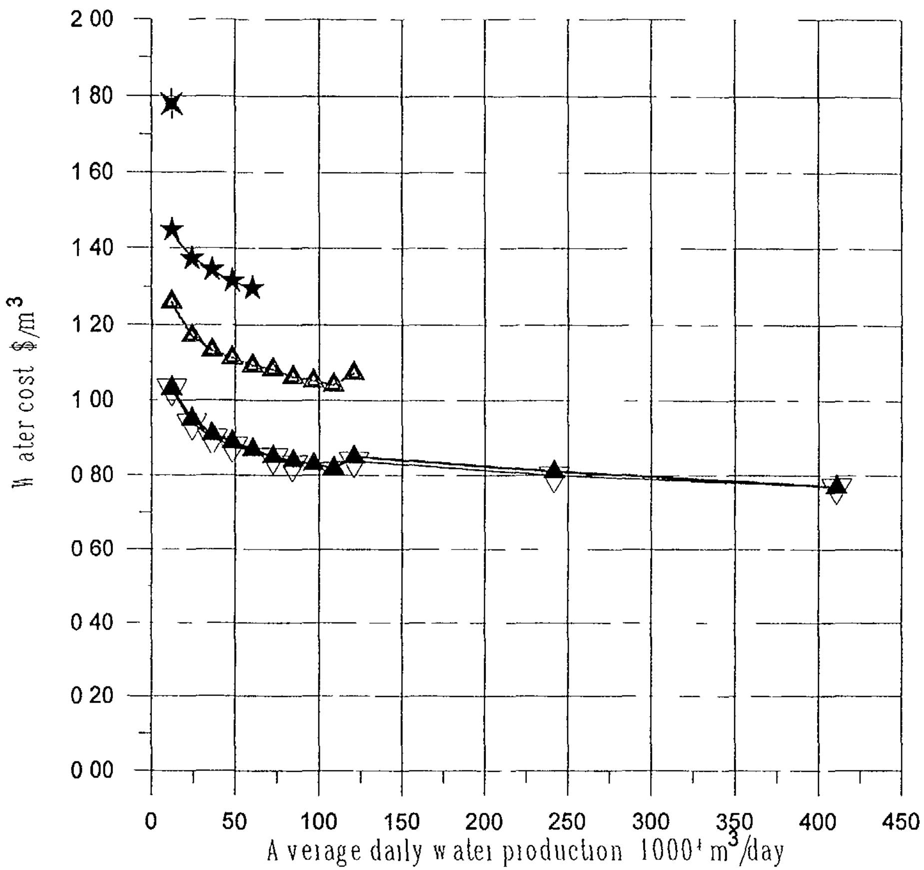
Input data	NIKA-120M (2 units)	NIKA-300	RUTA-TE (2 units)	RUTA-TE (4 units)	UNI-TERM (2 units)
Thermal power of the core, MWt	2*70	300	2*70.4	4*70.4	2*15
Net electrical power, MWe	2*15	90	2*3.5	4*3.5	2*2.07
Specific cost of construction , $/kWe	3500	2100	7300	6800	9000
Specific O&M cost, $/MWeh	6.0	5.0	9.0	9.0	15.0
Specific nuclear fuel cost, $/MWeh	25	17	27	27	21
Operating availability	0.8	0.9	0.95	0.95	0.9
Power plant economic life, a	30	30	60	60	40
Interest rate, %	8	8	8	8	8



Multi-Effect Distillation(MED)
-for NIKA 120M   for NIKA 300   for UNITERM   for RUTA II

FOSSIL at 15.5 $/BOE
-MED/Combine Cycle(CC)

Fig 6a Water cost versus water production for multy-effect distillation(MED) plants



Contiguous Reverse Osmosis(C-RO)
for NIKA 120M   for NIKA-300   for UNITERM   -for RUTA-TF

FOSSIL at 15.5 $/BOF
C-RO/Combine Cycle(CC)

Fig 6b Water cost versus water production for contiguous reverse osmosis(C-RO) water plants

Table IV Results of calculations

Characteristic	NIKA-120M (2 units)	NIKA-300	RUTA-TE (2 units)	RUTA-TE (4 units)	UNI-TERM (2 units)
Levelized power cost, $/kWeh	0.088	0.050	0.151	0.143	0.178
*Average daily fresh water production, 1000*m^3/day*					
Multi-Effect Distillation(MED)	41.6...73.1	64.4...179	40.8	104.1	10.4
Contiguous Reverse Osmosis (CRO)	12.1...121	12.1...411	12.1...24.2	12.1...60.4	12
Hybrid (MED + RO)	60...120	72...409	-	-	-
Fresh water cost, $/m^3					
MED	1.37...1.45	0.98...1.04	1.24	0.98	2.33
CRO	1.26...1.04	1.03...0.77	1.58...1.50	1.45...1.29	1.78
MED+RO	1.35...1.19	1.00...0.89	-	-	-
Total investment cost, M$					
Power Plant	132	252	63	118	45
MED	108...183	154...422	84	233	31
CRO	17...158	17...443	17...34	17...79	17
MED+RO	125...200	154...525	-	-	-

4. CALCULATION RESULTS

Table IV and Fig. 6a,b present some calculation results.

For the RUTA-TE reactor, the cost of water desalinated in the reverse osmosis (RO) plants is higher than the cost of the water produced in RO plants with fossil fuel by about 40 %. This is explained by high specific cost of electricity produced by this reactor. The RUTA-TE reactor is expedient for the application together with the distillation desalination plants. In this case the cost of water is lowest, approximately 1 $/m^3, which is very close to similar indices of fossil fuel plants. In addition, it should be also kept in mind that the quality of distilled water is higher than the quality of water obtained from RO plants (salinity is 25 ppm for MED plants and 300 ppm for RO plants).

For the NIKA-120M reactor the fresh water cost is above those for fossil fuel plants by about 20 to 40 %. But an advantage such as transportability of the reactor by sea (floating design version) may play a major role for choosinga reactor plant for the seawater desalination.

For the NIKA-300 reactor the fresh water cost is very close to the one for the best fossil plants. A power plant based on this reactor can operate in combination with a desalination plant of any type and produce fresh water at a very low cost.

For the UNITHERM NPP the fresh water cost exceeds similar indices for fossil plants. For distillation plants the difference reaches up to 60 % and for RO plants - up to 80 %. However, low capacity desalination plants in combination with the UNITHERM NPP may turn to be cost-efficient in remote difficult-to-access regions.

5. CONCLUSIONS

Desalination plants with the nuclear power plants under consideration have no decisive advantage so far over fossil driven desalination plants in term of desalinated water costs. However, they have merits, such as a substantially longer duration of operation with a single fuel loading and the freedom from environment contamination by the fossil fuel combustion products. Nuclear desalination plants are presently expedient to be applied in remote, difficult-to-access regions whereto the supply of fossil fuel is difficult and expensive, for instance, in Extreme North remote areas and difficult-to-access desert regions. In future, nuclear desalination plants may probably become more cost-efficient than fossil desalination plants as the reserves of fossil fuel decrease and the related fossil fuel becomes expensive.

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APPLYING A SMALL NPP IN THE ARGENTINE MINING INDUSTRY

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XA9848810

Abstract

The CAREM 25 reactor project is a small PWR nuclear power plant of 27 MWe, based on advanced concepts: a self-pressurized integral primary with natural convection of the coolant and a more simple and reliable general design. The CAREM concept has many advantages as a power generator in small electrical grids. Besides, there are some non-electrical applications under consideration, since a co-generation scheme seems very interesting from the economical point of view. In this category two alternatives have been considered: a standard desalination facility and a process plant in the mining industry. In this paper, a conceptual analysis of the second alternative is presented. Mining is a branch of the domestic industry that has shown a remarkable growth in the past three years mainly due to a steady inflow of foreign investments (about two billion dollars for that period). And one of the most attractive markets is in the extraction and manufacturing of nonferrous minerals, coming from deposits in the northwest of Argentina: sodium sulfate, lithium salts, and boron compounds. Nevertheless it faces an unsolved problem in the energy high prices due to the fact that the production sites are located in remote areas where the only achievable energy source is the transportation of fuel oil. In this scenario, a small NPP may be a competitive source of process heat and electricity, with enough autonomy to uncouple fuel requirements from production strategies. The present study analyzes the possible application of the CAREM concept in the non-ferrous mining industry of the Northwest of Argentina, considering a co-generation scheme. The main results of this analysis and the inherent advantages of the approach, show that the alternative may be feasible both from the technical and the economical points of view.

Introduction to this alternative application

In the frame of CAREM project, a dual purpose plant for electricity and process heat is evaluated, pointing to an alternative for the funding of the prototype construction by private investment. The most straightforward dual applications for a small NPP are district heating and seawater desalination, but these seem hardly viable in Argentina where district heating is not being used at all, and fresh-water resources are quite evenly distributed.

As for process heat, the mining industry present blooming, and isolated conditions of some sites **suggests certain commercial viability** for a small NPP. This report assumes the CAREM application for mining / purification process for non-metallic mineral in the western regions of Salta and Catamarca provinces, known as Northwest **Puna**. Among these minerals, the most relevant economically are different compounds of boron and lithium, sodium sulphate, sulphur, and sodium chloride. The energy requirements for such exploitation units are of several tenths of MWth and few MWe , with stringent limitation of distance to the site.

As a first approach, this report will evaluate one of the alternatives of CAREM applications for mining industry: the use of process heat for extraction and purification of sodium sulphate.

Some preliminary advantages

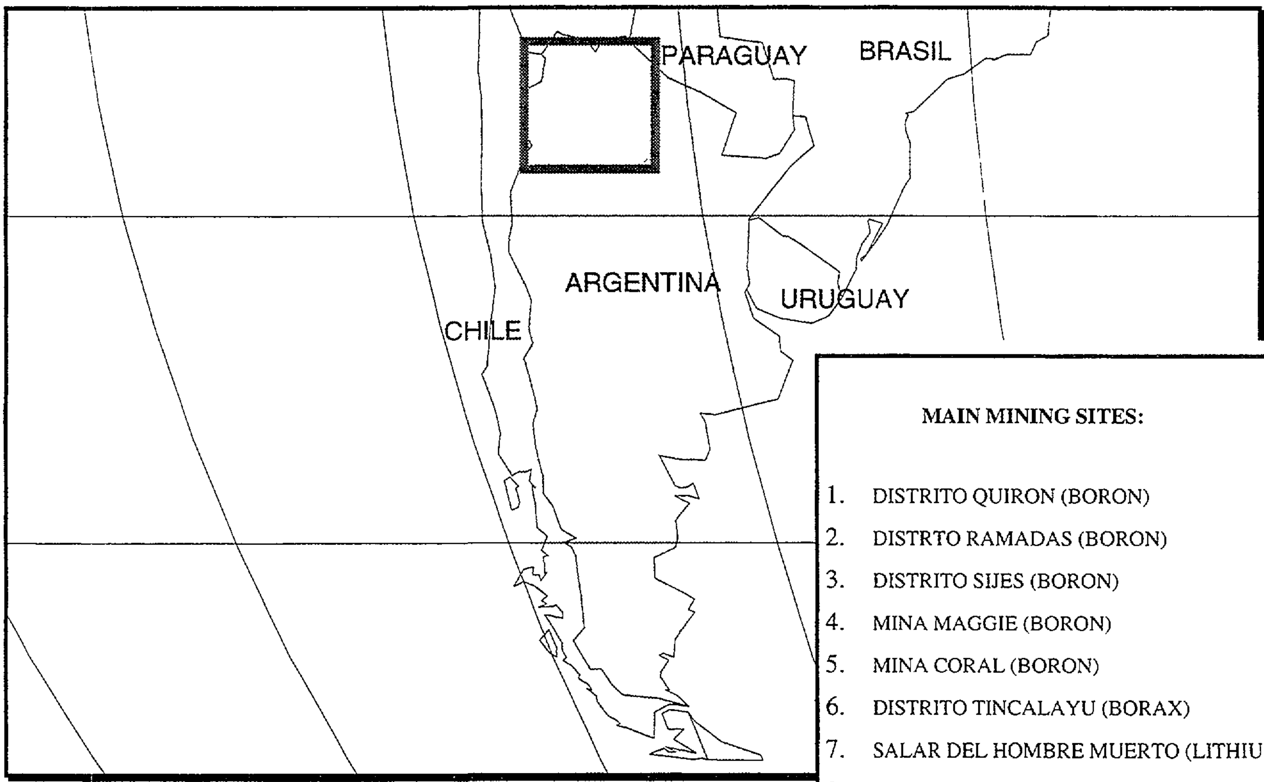
Without going to the detail of an economical evaluation, there are some preliminary features that suggest the existence of comparative advantajes of this alternative application, and encourage a further study:

1. The design concept of CAREM is specifically suited for isolated sites.
2. Placing a NPP or any other important industrial Plant in the Puna implies an important impulse to the regional development.

3. There is presently a mining "boom" in the N-W provinces, and new energy supplies are needed.
4. Traditional energy supplies (extension of natural gas pipelines or of national electricity grid) imply massive investments, while in site generation alternatives imply high transportation costs of fuel.
5. The construction of the prototype as a process heat application enables to simplify the construction, reduces costs and schedules, for the evaporating devices are conventional components much simpler than a turbine thermal cycle.

Argentine mining industry present boom

It must be pointed out that the country's mining potentiality has not been exploited in a way according the known deposits. The contribution to the Argentine GDP of mining production is negligible compared to bordering countries like Chile, or to geographicaly large countries like Canada, Russia, Mexico or U.S.A.



MAIN MINING SITES:

1. DISTRITO QUIRON (BORON)
2. DISTRTO RAMADAS (BORON)
3. DISTRITO SIJES (BORON)
4. MINA MAGGIE (BORON)
5. MINA CORAL (BORON)
6. DISTRITO TINCALAYU (BORAX)
7. SALAR DEL HOMBRE MUERTO (LITHIUM)
8. SALINAS GRANDES (SALT)
9. SALAR DEL RINCON
10. LA COLORADA (COPPER)
11. DIABLILLOS (GOLD, LEAD, ZINC, IRON)
12. INCACHULE (COPPER, GOLD)
13. SAMENTA (COPPER, GOLD)
14. LA CASUALIDAD (SULPHUR)
15. SALINAS GRANDES (BORON)
16. TACA-TACA (COPPER-GOLD)
17. SALAR DE RIO GRANDE (SODIUM SULPHATE)

This production profile is due to social-economical aspects of Argentine history, in which the agricultural exports through the Buenos Aires harbour triggered a development centered in that city. After that, the economical activity grew to certain diversity, but the connecting network left isolated regions, and some industrial activities with a lack of support (among these, mining).

The trend subsisted from the colonial times till this very decade, and was reversed by two main reasons: the settling of foreign mining companies with important investments; and the need to diversify the exportable production (within a global economy model). At present there's a "boom" in the activity in the provinces of Salta and Catamarca. As an example of this, the mining exports of Salta climbed from 10,7 MU$S in 1994 to 36,8 MU$S in the year 1995, mainly by boron minerals. Meanwhile there are prospection activities being performed by 25 companies with a global investment over 30 MU$S for the following three years.

Nevertheless, the most impressive example of the mining boom is given by the auriferous site of Bajo de la Alumbrera, Province of Catamarca, where an Australian based holding (Alumbrera Limited) is investing an amount close to 1000 MU$S.

Finally it must be said that this reactivation is strongly supported by the national and provincial governments that have approved a legal framework that provides benefits for the investments in the N-W region, including tax exemptions on land, labour, financial assets and imports.

Therefore the mining forecasts in the country (and specially in the N-W) are encouraging for local and foreign investors, and new opportunities should arise from the vacancies produced by the Mining Up-Dating Law (Ley de actualización minera).

Available non-metallic minerals in the region

Sulphur

Main deposits are located in the Los Andes department (Salta Province) at a height of 3700 m.o.s.l. near the border with Chile. The provincial company La Casualidad S.A. exploited the site Mina Julia till 1979, and an important infrastructure remains (including a paved road access from the Caipe station of Belgrano railway). The known deposits rise up to 2,200,000 tons, with an average content of 20.4%. The internal market for Sulphur is of around 120,000 tons/year, presently covered by imports in an 80%. A first order estimate of the price of international grade sulphur (minimum fineness 99.5%) is of 120 U$S per metric ton, placed on train wagon in Caipe station. Main use of refined sulphur is for sulphuric acid production, that has an increasing demand.

Boron minerals

From the mineralogical point of view, there are two different areas: Sijes and Tincalayu. In the first the most frequent minerals are hidroboracita, colemanita and ulexita, while in the latter the main ore is of borax (locally known also as Tincal). Both cases are Open Pit exploitations, with the refining process performed in urban sites close to Salta City. Though the borate production figures of the N-W are significant, the relevant potential is in the magnitude of the deposits, for there are only two other countries with comparable reserves: Turkey and USA. Argentine deposits have been estimated in 20,000,000 tons for the global of boron minerals, while the production is close to 70,000 tons/year. Price vary strongly with the kind of borate, but an average value between 500 and 1000 U$S per ton (FOB) may be obtained in exports to Brazil, Australia, Spain, USA and Netherlands. The use of boron includes the industries of enamel, ceramics, glass, detergents (as sodium perborate), agrochemical (as oligoelement), pharmaceutical, cosmetics, etc.

Lithium salts

Due to the increasing demand of metallic lithium the production is drifting from the methods on pegmatite, towards the extraction from the brine of saltpits. Some studies have been made over the possibility of recovering lithium and potassium salts from the N-W saltpits, including prospection on an area of over 100,000 km^2 (samples cover 14 saltpits). The analytical results were encouraging. At present there is an important high technology Plant producing lithium at Salar del Hombre Muerto (placed at the border of Salta and Catamarca provinces), with a staff of 500 skilled workers.

Sodium sulphate

Main deposits are located in the Salar de Río Grande, Los Andes department (Salta Province) that was in commercial production till 1992 by a private company (Altas Cumbres S.A.) with an output of 20,000 tons/year. The historically settled working method was pumping the brine to surface pools, where crystallisation took place under the change in environment conditions (not necessary the most efficient method). The mineral obtained is called "mirabilita", and has a high content of water. The final process of evaporating / drying in that project was performed in Güemes, 50 km from Salta, so the transport cost penalised heavily on the enterprise revenues. The amount of sodium sulphate reservoirs as dissolved in brine is of around 5,000,000 tons. Besides on the bed of the saltpit of Salar de Río Grande there are important mirabilita banks of high content (from .3 to 3 m deep).

There are other production sites in the area of Vega de Arizaro, Laguna de Socompa and Pocitos, where the crystallisation takes place in saltpans by the natural conditions of height weather.
The use of sodium sulphate includes the industries of soap and detergent, paper, textile, and glass. The market for this product is broad, with a world consumption of 4,000,000 tons/year (675,000 of which are for USA, and 150,000 for Mercosur). Though price varies strongly, a first order estimate for international grade sodium sulphate (minimum fineness 99.5%) is of 100 U$S per metric ton, placed on train wagon in Caipe station.

Sodium Chloride

Even though sodium chloride (salt) is generally an unwanted by-product from the purification process of sodium sulphate, it is a product with an assured commercial viability, due to the increasing demand as a raw material and for human consume. The production of Salta Province leads the national market supply with 50,000 tons/year of halite.

Sodium carbonate

Main deposits of sodium carbonate are placed in the surrounding of Cerro Rincón, in the saltpit of Santa María. The low exploitation and fluctuating production rate hinder the possibility of estimating the amount of the reservoir and life expectancy for the site.

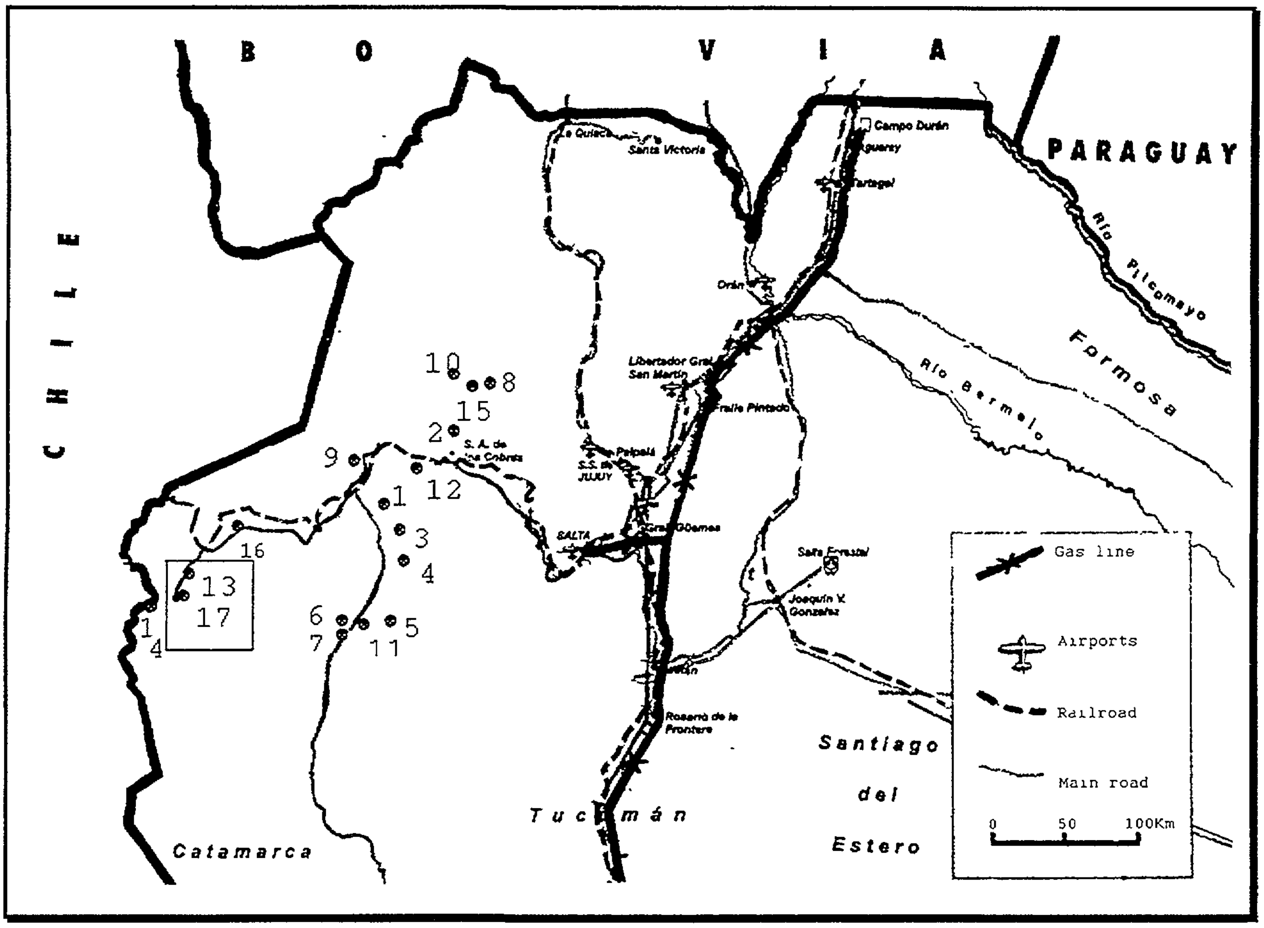


Figure 1. Location of main deposits in Salta Province.

How may a CAREM-NPP fit in the N-W mining?

Assuming the location of a CAREM reactor en the production site (saltpits), i.e. in the N-W Puna at 3700 m.o.s.l. and 500 km from Salta city, two application schemes may be considered.

1. As an industrial Plant. Most of the energy production would be used to obtain and purify (by evaporation) non metallic minerals dissolved in the saltpit water: sodium sulphate, sodium chloride and lithium salts. A minor fraction of the energy would be transformed to electricity for the operating consumption of the NPP, the industrial Plant and auxiliary facilities. A slightly different scheme would include the use of electricity to obtain pure elements by electrochemical methods, as it is shown in Figure 2. In both cases the electricity production would be in a thermal cycle with higher turbine back-pressure, thus with a higher efficiency compared to the standard thermal cycle.

2. As an electricity supplier. In this alternative the reactor would maximise the electricity production, in order to supply the mining industries already located in the Puna region. This is particularly adequate for the boron minerals exploitations, because an energy source near the site, even at a price higher than the national market one, would reduce the overall costs significantly because of the present transport costs. It should be recalled that the process of concentration, refining and delivery preparation for export of the minerals are carried out in the nearbies of Salta city.

It should be pointed out that these alternatives do not exclude each other completely, but simply define the design requirement for the thermal cycle and auxiliary devices for the coupling with the mining industrial Plant.

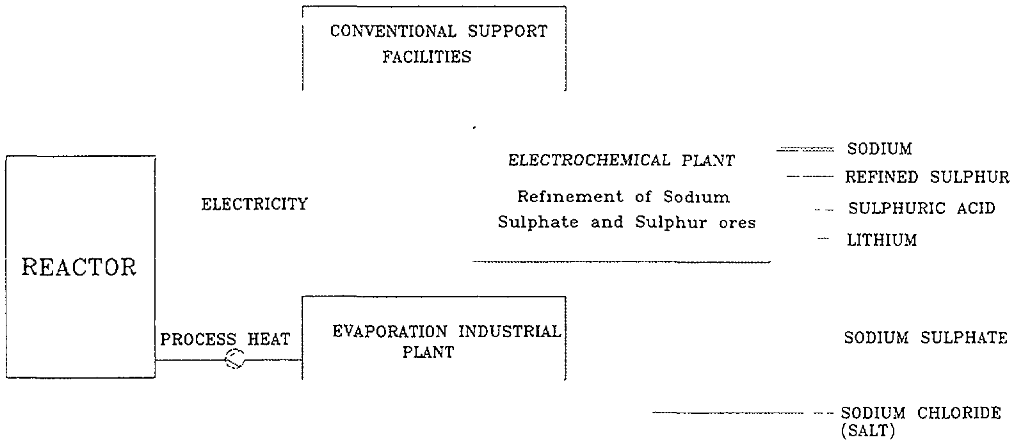


Figure 2. Scheme of the CAREM utilisation in the "Industrial Plant" approach.

First Viability: possible no-go points

A first order approach must first go through all the "no-go" problems this kind of project could meet (patrticularly the non-technical), that include the following points:

1. <u>Siting alternatives.</u> The location at Mina La Casualidad in Salar de Río Grande, site of main deposits of sodium sulphate and sulphur, is adequate for the CAREM application as an industrial plant.

2. <u>Access and communication.</u> Though the site is at present nearly uninhabited, it has an existing infrastructure that enables the integration with regional market: there's a paved road to Caipe Station, from where the Railway connects with Salta, and through Socompa with Antofagasta (Chilean harbour on the Pacific). Figure 3 shows the communication network facing Mercosur market and exports, which are also the access ways for components of the NPP construction.

3. <u>Seismic risk.</u> The area is a plateau surrounded by mountains and extinct volcanoes. From the geomorphologic point of view, the region is formed by extinguished volcanic cones, with slopes (hillsides) of lava and ashes washing. The volcanic bodies are generally of the mixed kind, dated on Plio-pleistocene. The area has a relatively low risk, with no seismic event detected in the last decades.

4. <u>Heat sink.</u> The supply of water as a heat sink is assured by the brook (Vega) La Casualidad, though it is possible to use directly the water from the saltpit as operating heat sink, for evaporation is part of the process.

5. <u>Legal aspects.</u> At present in the Province of Salta there is no legal restriction to the location of nuclear facilities, in fact there are existing facilities under nuclear regulation in the site of uranium mine Mina Don Otto. In the other hand there is legislation promoting mining enterprises and an important government support to energy resources that may enhance the economical performance of mining companies.

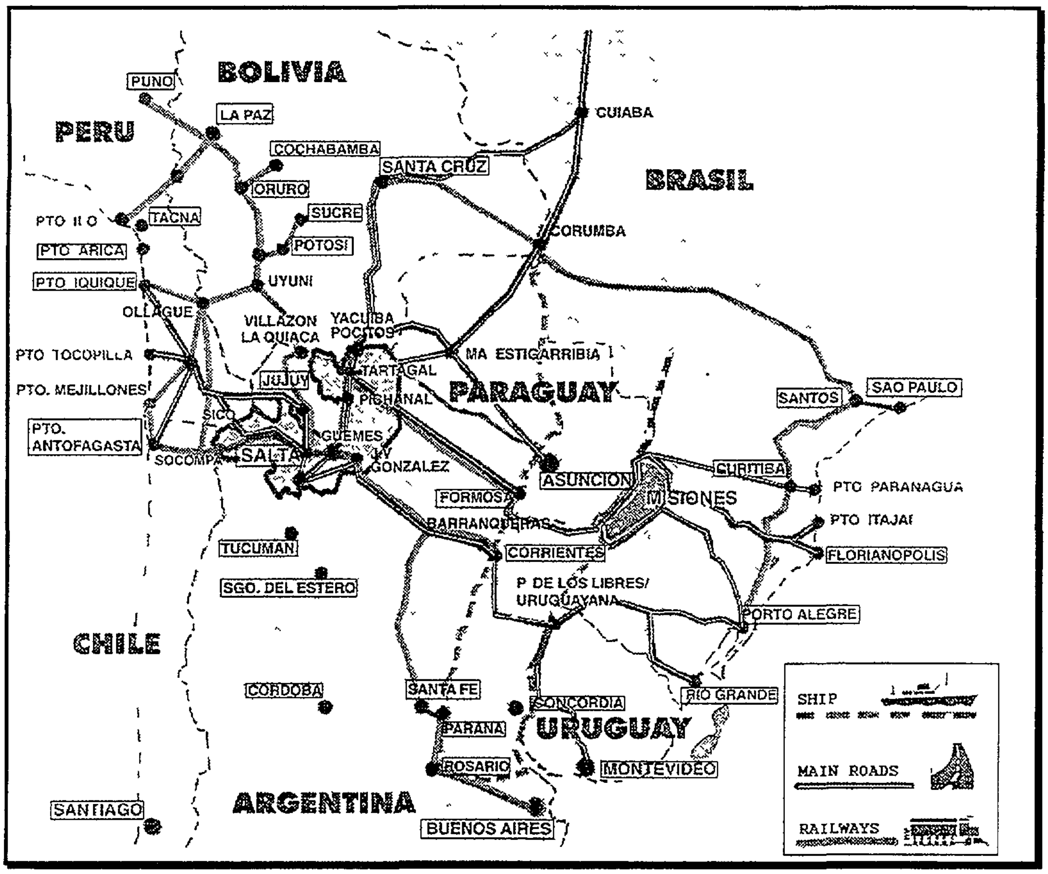


Figure 3. Main access ways of the Puna region towards Mercosur and Harbours (both Oceans).

Analysis case: sodium sulphate production by evaporation

Being this a preliminary study, an example case of a single-product Plant will be taken for the economical evaluation . The sodium sulphate was chosen because of the simplicity of the production process, and **it may not be the most convenient in comercial terms**. The single-product plant assumption is clearly conservative since it is known that a multi-product Plant combining the production of refined sulphur, sulphuric acid, sodium and lithium by electrochemical methods (see Figure 2) would improve the economics allowing a permanent economical optimisation to market fluctuations. Therefore the overall analysis case is strongly conservative.

The sodium sulphate (Na_2SO_4) is of broad industrial use as a pure chemical product also known as “anhydrous”. It is a non-toxic powder that looks like plain salt. In natural deposits the most frequent occurrence is as an hydrate molecule ($Na_2SO_4\,10H_2O$) known as decahydrate, “mirabilite” or Glauber salt. The most simple production method is starting with an hydro-thermal solution at a temperature around 60°C, and then cooling it under 32°C obtaining a precipitation of decahydrate crystals. The following refining to obtain anhydrous may be performed by centrifugation or by a drying furnace (~200°C). Most common deposit are salt pits of the lake-bed type, where an hydro-thermal saturated brine is pumped to man-made pools. Then the purification is achieved either by industrial evaporators or by precipitation in the same pools. The advantage of industrial evaporators is that they allow to obtain directly sodium sulphate without further furnace process. Therefore it is more efficient, but is very dependant on local energy resources (price and availability). A conventional drying device consists of a rotating gas furnace with an inner lining of stainless steel and an helicoidal axe that stirs the mineral. A temperature of 200°C is enough to dissociate the decahydrate molecule and evaporate the residual water.

The outlet is the mineral in irregular grains that must go through a mill to reach commercial requirements on granulometry (powder) before being packed and delivered.

Production scheme

A nuclear reactor may take part in this production process as a heat source for the direct evaporation of the brine containing the sodium sulphate. A basic scheme is presented in Figure 4, and is based in the following items:

1. Tertiary loop. Besides the secondary circuit that would provide the steam generators with the same boundary conditions than in the NPP case (inlet at 200°C, steam at 290°C), there would be a tertiary loop operating at a higher pressure than secondary in single phase. This loop would be the heat source for a set of industrial evaporators, working at an average temperature around 210°C.

2. Evaporating loop. The industrial evaporators are set in parallel and fed with brine at a temperature of ~60°C, coming from a conditioning/filtering stage. The brine pumped form the saltpit has an average base concentration of 3.5%, that may be increased if process pools are used. For the evaluation, the concentration will be taken in the range of 3.5 to 20 %.

3. Electricity generated by the reactor. Only part of the thermal energy available will be transformed to electricity. Therefore, the turbine-group may work with a higher back-pressure and the need of secondary loop cold leg preheating is lower.

4. Cold source for operating conditions. For the nominal operating condition the cold source is the saltpit itself, for all the heat extraction of the secondary loop would be used in the co-generation process. (In fact, the same industrial process provides destilled water.)

5. Cold source for incidental conditions. In the case of loss (trip) of the secondary or tertiary loops, the heat sink for the reactor would be provided by a small dam, with a permanent availability of water. The coolant flow should be assured by level difference between the dam and the reactor. In the case under analysis the dam would be supplied by the brook Vega La Casualidad, and in case of seasonal decreases the supply could be assured by process water production as a by-product.

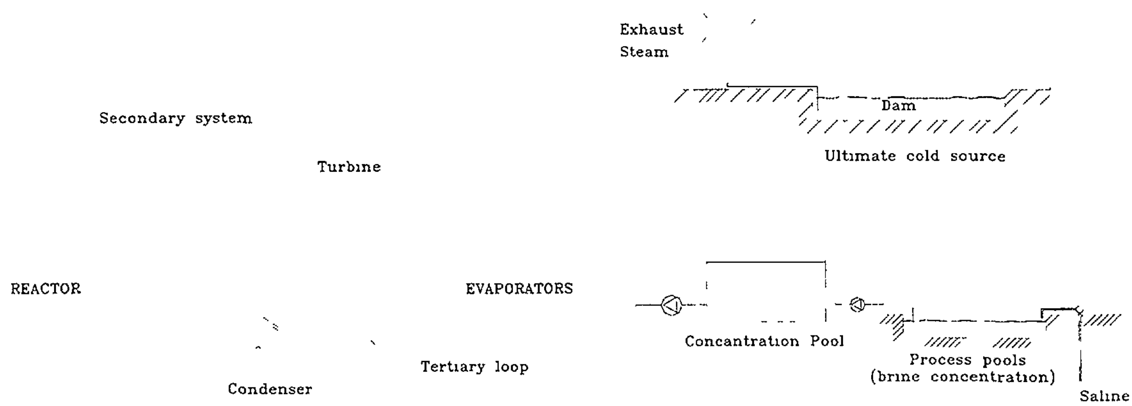


Figure 4. Process loops necessary for the sodium sulphate production by evaporation.

Preliminary economical evaluation

In order to make a preliminary verification of viability, the following energetic and economical balances were performed. Based on the assumed hypothesis and physical properties of the sodium sulphate solutions listed in Table I, the results shown in Tables II and III were obtained for different concentrations in inlet brine.

Table I. Input parameters for balance

Thermal power	80.0	MW
Evaporation heat	2427	KJ/kg
Anhydrous Molecular weight	142.05	
Decahydrate Molecular weight	322.21	
Anhydrous Solute heat at 18°C	280	(cal/g-mol)
Decahydrate Solute heat at 18°C	-18740	(cal/g-mol)
Cost of energy in process	10.50	U$S/MW.h
Overnight cost of energy	7.25	U$S/MW.h
Availability of the Combined Plant	80	%
Price of product	120.0	U$/ton

Table II. Balance with overnight costs

Conc. Na_2SO_4 [% weight]	Solute.: Na_2SO_4 [gr/lt.]	$10H_2O$	Production Na_2SO_4 [ton/year]	Prod. cost [U$S/ton]	Gross gain [U$S/ton]	Gross Revenue [MU$S/year]	Revenue [%/year]
3.5	35.45	44.96	29500.	138.	- 18.	- .531	-.6
5.0	51.35	65.12	42600.	95.	25.	1.065	1.27
8.0	84.40	107.04	70100.	58.	62.	4.346	5.17
10.0	107.38	136.18	89300.	45.	75.	6.700	8.00
20.0	233.40	296.0	194100.	21.	99.	19.200	22.90

Table III. Balance with interest rate of 5%

Conc. Na_2SO_4 [% weight]	Solute.: Na_2SO_4 [gr/lt.]	$10H_2O$	Production Na_2SO_4 [ton/year]	Prod. cost [U$S/ton]	Gross gain [U$S/ton]	Gross Revenue [MU$S/year]
3.5	35.45	44.96	29500.	200.	- 80.	- 2.360
5.0	51.35	65.12	42600.	138.	- 18.	- .770
8.0	84.40	107.04	70100.	84.	36.	2.520
10.0	107.38	136.18	89300.	65.	55.	4.910
20.0	233.40	296.0	194100.	30.	90.	17.400

Comments on the evaluation

Regarding this evaluation, there are some comments to be made:

- The production cost does not include the process pools necessary to increase brine concentration over 3.5%, and the weight of this simplification grows with the concentration required for the evaporators input. Therefore the production costs for concentrations of 10% or more may need further calculation.
- Cost of thermal power of 10.5 U$S/MW.h is composed by
  - * 3.5 U$S/MW.h corresponding to Operation and Maintenance costs
  - * 7.0 U$S/MW.h result from the capital investment of 84 MU$S in an amortisation scheme of 40 years life-time, with an annual interest rate of 5% (viable in the provincial industrial promotion frame).

Conclusions of the analysis

Regarding the particular case evaluation, there are some comments to be made:

- It is necessary to combine the traditional production method of evaporating in pools with the final evaporation and drying in the industrial devices.
- The range of production up to 90,000 tons/year is adequate in terms of the following items:
  - * The fraction of National market covered by imports (96,000 tons/year) .
  - * The Mercosur market.
  - * The size of standard industrial evaporators.
  - * The existing infrastructure for transport (already moving 70,000 tons/year).

Regarding the more general idea of applying a small NPP for the N-W mining activities, the following may be concluded:

- As an **overall result** of the evaluation of the analysis case, it may be said that it is **mildly profitable**, but it should be recalled that several conservative assumptions were made:
  - * The product chosen may not be the most profitable.
  - * The price was of 120 U$S/ton is quite conservative.
  - * A single-product Plant was evaluated, though a multi-product Plant would allow cost optimization and a better economical performance.
- There are more than enough reasons to go further with the study of this alternative application of CAREM design. It seems perfectly viable to find a particular production process (for someone of the available minerals) that would justify a private investment, i.e. a revenue level and payback rate according to international investors requirements.

DESIGN OF A NUCLEAR DESALINATION FACILITY FOR BUSHEHR, IRAN[1]

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XA9848811

Abstract

Three options of coupling schemes were evaluated in order to integrate an MSF desalination plant of 200 000 m3/day with twin PWR units of 3728 MW(th) each for the Halileh Nuclear Power Station in Iran, which were under construction at the time of the investigation: (a) The exhaust steam from a backpressure turbine is fed to the brine heater; (b) The steam extracted downstream of a reheater of the NPP is fed to the brine heater; and (c) Hot water heated by the steam exiting the high pressure turbine of the NPP is fed to the brine heater. Technical and economic advantages and disadvantages of these three options are summarized.

Halileh Nuclear Power station near Bushehr, IRAN

A contract for a dual purpose desalting plant was awarded to the Consortium of Sasakura Engineering Co., Ltd., Mitsubishi Heavy Industries, Ltd. and Sumitomo Shoji Kaisya Ltd., Japan, by Atomic Energy Organization of Iran in 1977. The construction of the two PWR units of 3728 MW(th) and 1293 MW(e) each was started. An MSF desalination plant for 200 000 ton/day (6 x 33 552 t/d) with acid treatment was designed for integration with these PWR units. The design strategy was to keep the influence on the design of the nuclear power station as slight as possible. This was the largest nuclear desalination facility which was designed and contracted at the time.

Coupling of Desalination plant and Nuclear Power Station

The influence of the desalination plant on the design of the nuclear power station must be as slight as possible. The following options were investigated.

Description of the technical options

A) Generation of process steam in a secondary steam generator by means of main steam from the nuclear power plant and expansion of this process steam in a backpressure turbine. The exhaust steam is fed into the brine heater (Figure 1). During the periods of shutdown of the nuclear power station, an oil-fired auxiliary boiler takes over the steam supply of the backpressure turbine and the brine heater. The condensate of the main steam is fed back into the steam generators of the nuclear power station and the condensate of the process steam is returned into the secondary steam generator via a separate feedwater tank.

B) Via an extraction line branching off down stream of the reheater of the saturated steam turbine of the nuclear power station, superheated steam is conducted from the nuclear power station to the seawater desalination plant. In a reducing station with downstream injection-type desuperheater, the steam is throttled down to the saturated steam pressure which is required for the brine heater (Figure 2). By means of a condensate pump, the heating steam condensate is fed back into the nuclear power station, where it is introduced into the hotwell of the condenser after passing through a condensate cooler, through which a partial flow of the turbine condensate is routed parallel to the LP feedwater heaters.

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[1] Although both the NPP and the MSF plant were not completed, the design experience appears valuable and is included here for the benefit of the readers.

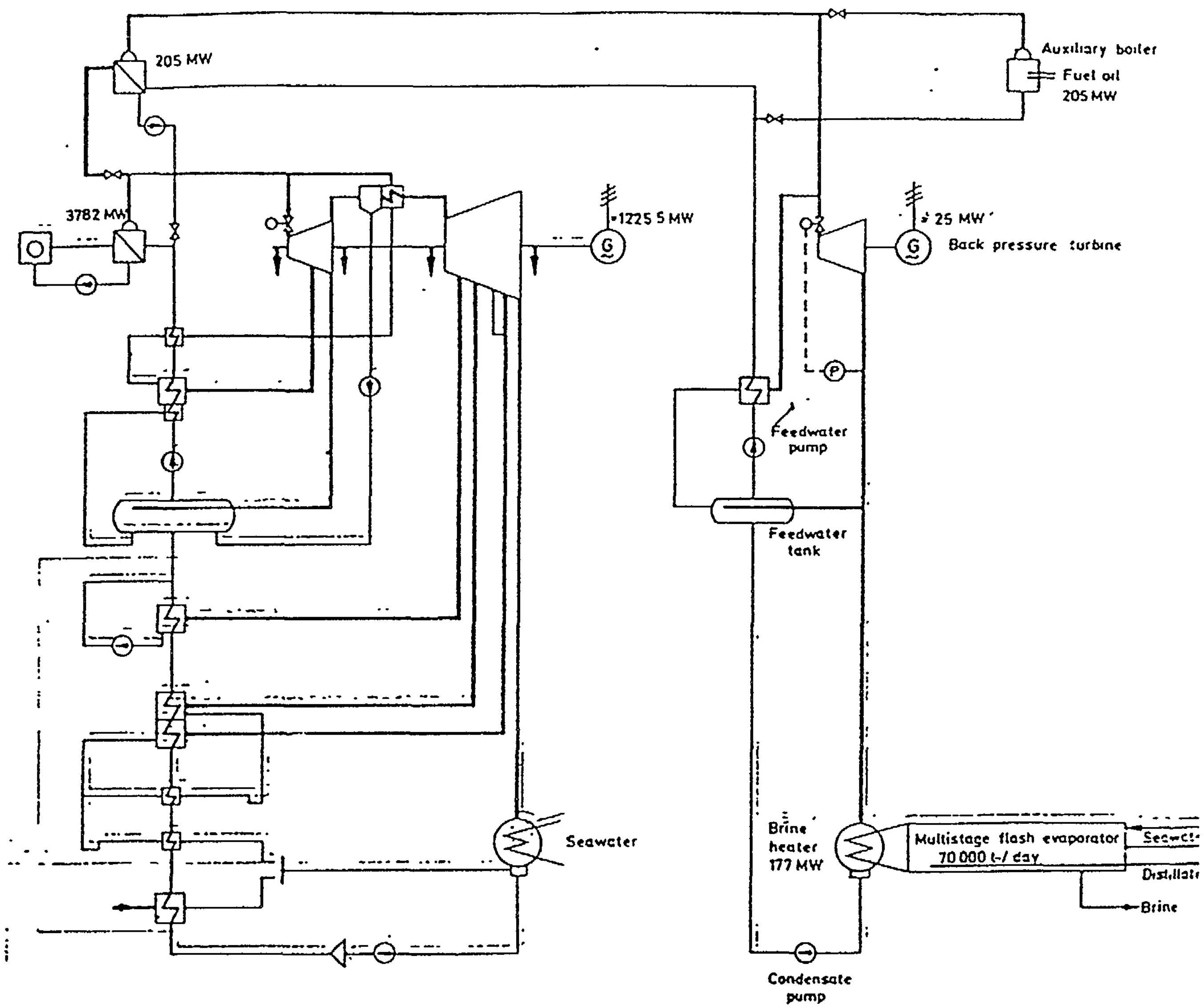

Fig. 1. Combination of a nuclear power plant (PWR) with a desalination plant (MSF) (Option A).

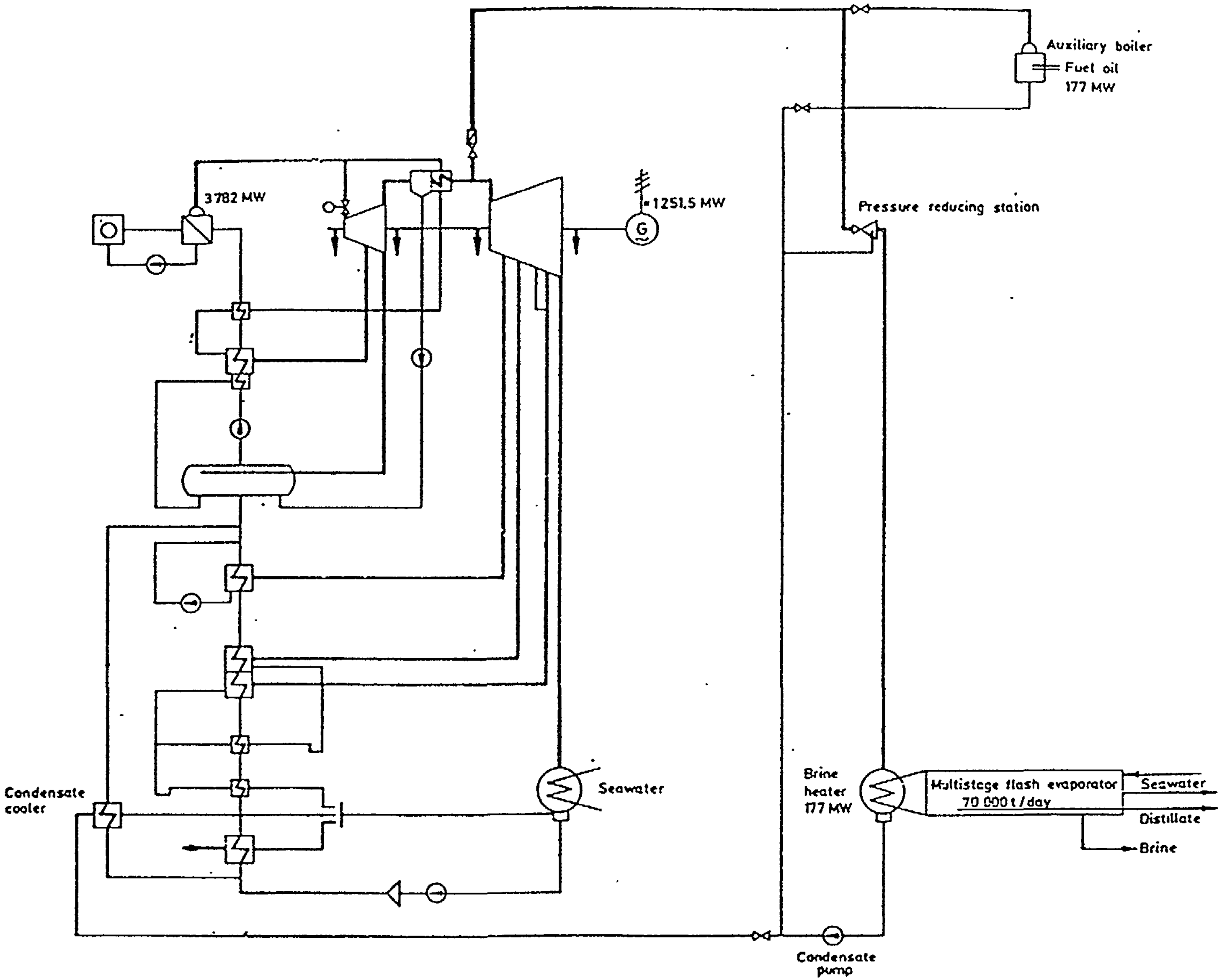

Fig. 2. Combination of a nuclear power plant (PWR) with a desalination plant (MSF) (Option B).

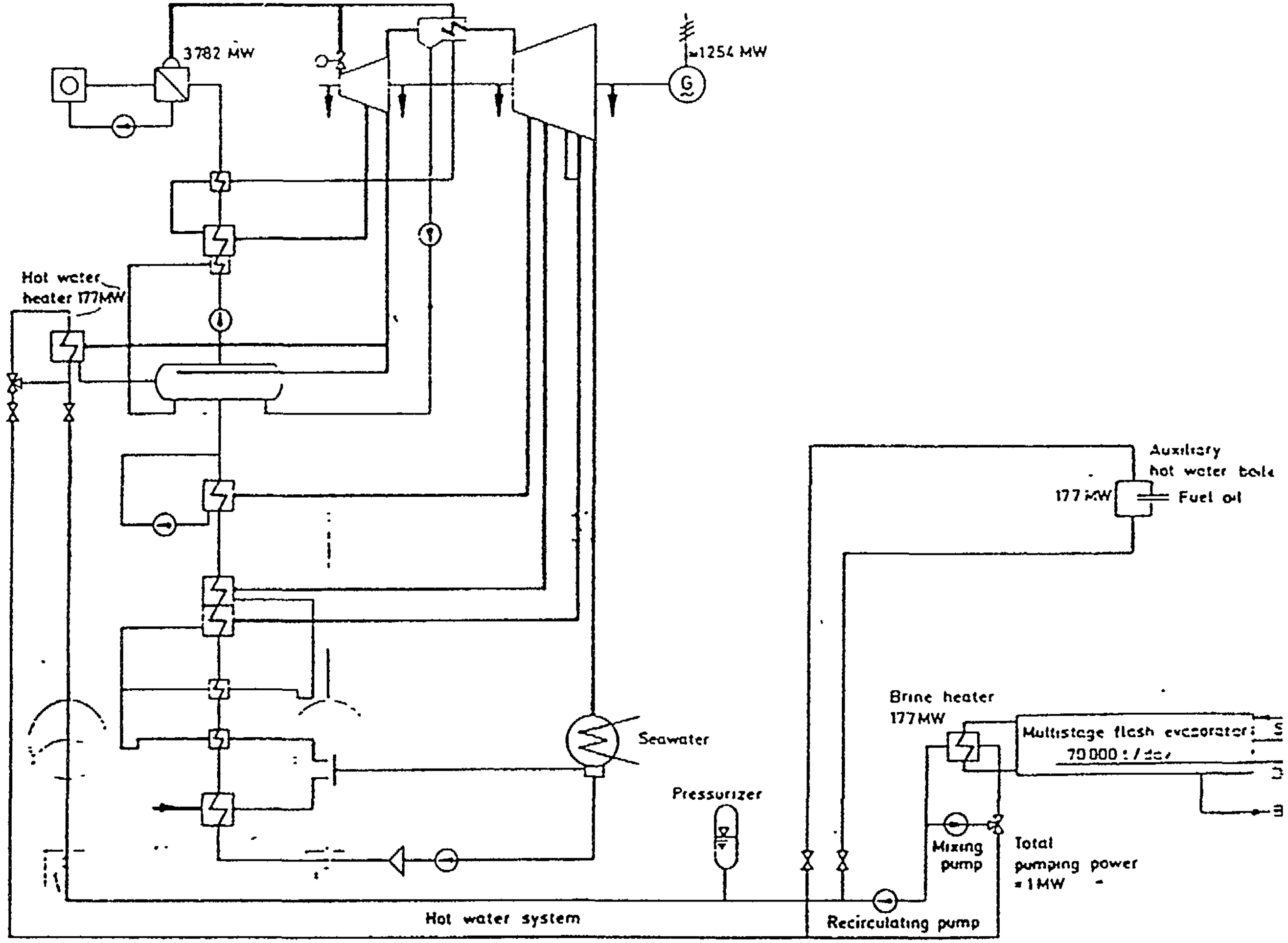

Fig. 3 Combination of a nuclear power plant (PWR) with a desalination plant (MSF) (Option C).

The steam of the nuclear power station will contain practically no radioactivity, because the steam generators will normally work without leaks. On top of these two radioactivity barriers, a pressure gradient from the brine to the process steam ensures that even in case of leakages no radioactivity can enter the brine and the distillate generated from it. An oil-fired auxiliary steam boiler supplies the heating steam for the brine heaters when the nuclear power station is shut down.

C) Heating the brine heaters by means of hot water which is heated in a hot water heater by steam exiting from the HP turbine (Figure 3). The pressure selected for the hot water system is higher than the pressure of the brine in the brine heaters and is also higher than the pressure of the steam in the hot water heater. If any leakages should occur, neither brine nor steam can penetrate into the hot water system and charge it with salt or make it radioactive. An oil-fired auxiliary hot water boiler takes over the heat supply of the seawater desalination plant when the nuclear power station is shut down.

Advantages and Disadvantages of the Solutions

A disadvantage of solution A is that the required large secondary steam generator cannot be accommodated in the turbine house without considerable design changes. This difficulty does not arise in the case of solutions B and C.

The installation, in the case of solution A, of a further small turbine-generator set with approximately 25 MWe also causes a complication as regards operation and maintenance which does not occur with solutions B and C. This complication is not justified for the relatively small amount of additional electric energy generated.

For solution A, steam at 10 bar is extracted when the turbine is at full load, whereas only 3 bar steam is required for the desalination process. The low efficiency of the small 25 MWe turbine generator set compensates fully for the throttling losses of solutions B and C.

The only occasion when the power/process heat combination of solution A is better than that of solution B and C is when the nuclear power station is operated at partial load below 50%. In this case the extraction steam pressure drops so far that it is necessary to use throttled main steam for solutions B and C.

The power supply of the seawater desalination plant is practically always assured by interconnecting to both nuclear power units Iran I and II. Even in the unlikely event of short shutdown period of both reactors, there is still power from the electrical grid available, which is in any case necessary for supply of start-up power to the nuclear power station.

There is also no essential difference between the two solutions as regards the reduction of power generation or the consumption of electric energy. In the case of solution B, the reduction of power generated in the nuclear power station is 2.5 MWe higher, but the hot water circuit in solution C has an additional energy consumption of approximately 1 MWe, which results in a partial compensation of the aforementioned disadvantage of solution B.

The initial expenditure for solution C is only slightly higher than for solution B, because the auxiliary hot water boiler is cheaper than an auxiliary steam boiler and the expenditure for the hot water circuit is thus almost compensated for. The remaining small difference is fully compensated for by the difference in power consumption (1.5 MWe).

In the case of the Halileh plant, the result of all aspects dealt with so far was that solution B (steam) and solution C (hot water) were practically equivalent. It is true that with the present level of technology it also appears perfectly possible to implement solution B, but the additional operating safety of solution C is a relevant point in its favour, particularly since this would be the first time that a combined NP and desalination plant of this type would be constructed.

In addition to this the operation of a hot water system is very simple. It corresponds to the operation of district heating plants which have often been constructed in connection with power stations.

Finally, it may be mentioned that, because of the large heat capacity of the hot water system, the dynamic behaviour of the nuclear power station and of the seawater desalination plant are almost disconnected, with the result that quick changes in one plant do hardly affect the other plant and can easily be controlled.

Because of the advantages described above, the decision was taken in favour of solution C with the hot water system. This solution would be the safest, simplest and most reliable one in operation.

II.1. DESIGN ASPECTS OF NUCLEAR HEAT APPLICATIONS

Low temperature heat applications — Lead–bismuth cooled reactors

XA9848812

THE ANGSTREM PROJECT: PRESENT STATUS AND DEVELOPMENT ACTIVITIES

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Abstract

The project ANGSTREM of a modular-transportable nuclear power-and-heating station with assured safety for remote areas is described. The station is based on fast nuclear reactor cooling by a lead-bismuth eutectic. The possible evolution of the project is mentioned in relation to possible new application - sea water desalination or refrigeratory plants.

DESTINATION

The ANGSTREM Project is a modular-transportable nuclear power-and-heating station with assured safety for remote areas. The cooling of the primary circuit is a lead-bismuth eutectic (Pb is ~44%, Bi is ~56%) and several passive safety systems are included. EDB" Gidropress" is the main designer. IPPE is the scientific adviser of the Project.

The present status of the ANGSTREM Project is presented. The possible evolution of the ANGSTREM Project is mentioned in relation to a new application. It is possible to utilise the thermal power of the station to sea water desalination or to use it in refrigeratory plant.

DESIGN

The project was the first application of the 40 years experience of development and exploitation of lead-bismuth cooled reactors to small power level co-generation plants. It was the first attempt to realize the concept of inherently safe NPPs, in which the laws of nature principally exclude the accidents with severe consequences.

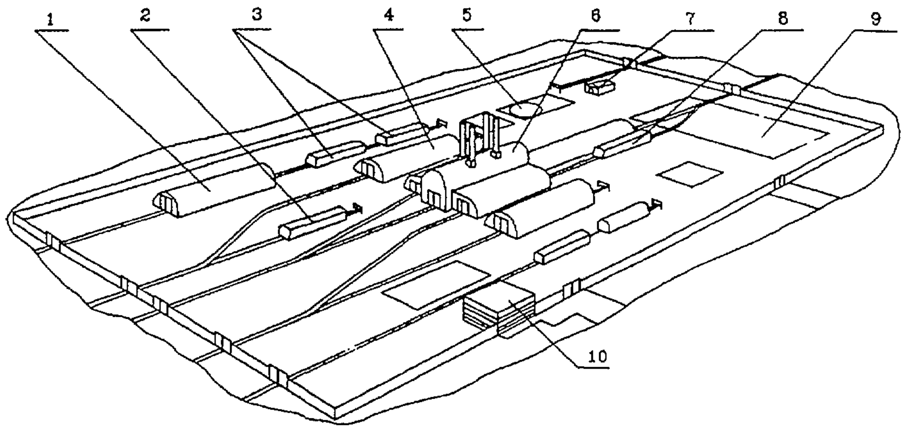
The station consists of:

- a set of buildings on the station site (shown in fig. 1).
- a set of functionally-finished, and completely factory-made modules in these buildings.

The modules can be delivered to the place of operation by all the kinds of transport. The development is in the stage of the detail engineering design. The project is based on the technical decisions which have been proved in practice. There are main equipment prototypes in service and commercial production.

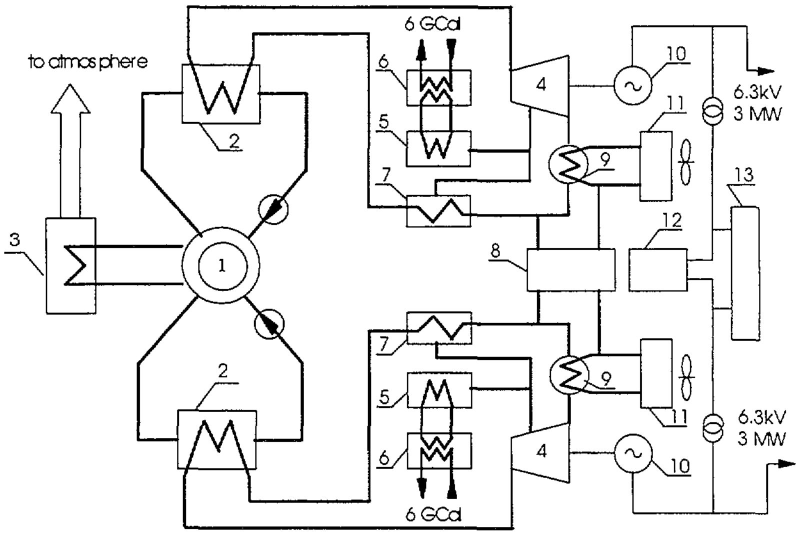
The ANGSTREM can be operated in the territories of increased seismicity, and aridity in climatic conditions from +40°C to -60°C. The reactor plant is a fast neutron reactor cooled by lead-bismuth in two-loops. The station nuclear fuel cycle is in harmony with the fuel cycle procedures of NPPs in current use. The station is equipped with a system of process automatic control and technical diagnostics, which provides safe operation and allows to minimize the operating personnel.

The project of the ANGSTREM station won the competition "Small Nuclear power stations-91" held by the Russian Federation Nuclear Society in the class of similar power level stations.



1 - Central control room compartment
2 - Water treatment system
3 - Cooling system
4 - Electrical switchgear device module
5 - Reservoir for fire-fighting water inventory
6 - Main building
7 - Storehouse for combustible materials
8 - Laboratory-workshop
9 - Supplementory territory
10 - Office building

Fig.1. Station general view.



1 - Reactor
2 - Steam generator
3 - Emergency cooldown passive system
4 - Turbine
5 - Heat exchanger for heat supply system
6 - Boiler
7 - Feedwater heater
8 - Water treatment
9 - Condenser
10 - Electrical generator
11 - Air radiator
12 - Reliable electric power supply system
13 - Ancillary power consumers

Fig.2. Station process diagram.

LEAD-BISMUTH COOLANT FEATURES

Over the long time of investigations, all principal problems related to utilizing lead-bismuth as a coolant in the NPPs were successfully solved: neutronic and thermal physics of the core, heat exchange and hydrodynamic characteristics, corrosion strengths of structural materials and coolant quality, radiation and nuclear safety, study of accident situations, development of fuel and absorbing elements, equipment of the primary circuit, etc. The intensive experimental base was founded which is equipped with unique test facilities.

A lead-bismuth coolant has a number of important safety features:

- Low margin of potential energy excludes the possibility of thermal explosion of the reactor under the internal pressure forces even at very high temperatures. It prevents the loss of coolant due to exclusion of the coolant evaporation by boiling (as it takes place with water coolant) or its blowdown under pressure (as with gas coolant).
- It has high boiling point (1670°C under atmospheric pressure) that eliminates the possibility of DNB occurence, that increase the reliability of the heat removal from the core. It allows to reach high process parameters of the secondary circuit and the high efficiency of the thermodynamic cycle under the low primary pressure.
- Low freezing-point of lead-bismuth (125°C) makes possible to decrease (or exclude by appropriate design) the possibility of the loss of coolant caused by primary leakages.
- Small coolant volume shrinkage on solidification (~1.5 %) and rather high plasticity prevent damages of the primary equipment at the controlled transition of the coolant from liquid to solid states and its cooling to the ambient temperature. This feature can be used for safe handling of the fresh or the spent core, against potential accidents during transportation.
- Low chemical activity eliminates the possibility of fires and explosions in the event of coolant leakages into the reactor room or the liquid metal-water reaction due to the steam generator tube ruptures.
- Low induced long-lived gamma-activity of the coolant and its property to retain iodine and other radionuclides considerably simplify the radiation situation under the primary leak conditions. About in a day after the reactor shutdown the dose rate of gamma-radiation on the outer surface of the primary equipment lowers to the level facilitating to perfom its examination and repair without over-exposure of the personnel.
- Low interaction with neutrons provides the high conversion ratio which allows considerably decrease the burn-up reactivity depletion and extend the core life cycle.
- The reactivity effects of lead-bismuth coolant (void, power, temperature) is negative.

SAFETY

The nuclear power-and-heating station is designed according to the defence-in-depth approach to protect personnel and population. It assumes that the system of barriers is provided on the way of propagation of ionizing radiation and radioactive substances into the environment. Also it includes the technical systems and the organizational measures to ensure the efficiency of these barriers in case of an accident.

The station system of barriers includes:

- fuel matrix;
- fuel rod cladding;
- primary circuit with gas system;
- reactor module hermetic compartment;
- external radiation shield;
- protective guard of the station.

The defence-in-depth approach in the multi-barrier system provides protection of barriers themselves, each of them being protected by the subsequent one. The last barrier - the station protection guard - is designed to withstand the extreme external events:

- earthquakes;
- airplain crashes;
- shock waves.

The nuclear power-and-heating station safety is based on the reactor inherent safety features, utilization of the safety systems and conservative approach in the design and safety analysis. The reliability of heat removal systems should be especially underlined. There are:

- ordinary heat removal systems - secondary circuit, SG and turbogenerator;
- emergency heat removal system - in the loss of feedwater accident, the water inventory in the separators can be utilized for heat removal to the air condenser under natural convection;
- passive heat removal system - always in service.

ECONOMIC ASPECTS

The competitiveness of small scale NPP in comparison with large scale ones can be accomplished, in spite of downsizing, under the following conditions:

- the number of systems should be strongly reduced;
- the specific mass of steel should not be increased; and
- the specific operation and maitainance cost should not become excessive.

These targets are achieved for the ANGSTREM NPP by means of:

- all the NPP modules are functionally-finished, and completely factory-made;
- minimum buildings to be constructed at the NPP site;
- no high and large buildings ligthens the foundation loading and the plant can be built on the seabeach, the soft soil, the seismic area and so on;
- the construction time is reduced to one year;
- partial reloading is excluded to minimize the operation cost; and
- the core lifetime considerably prolonged;
- low decomissioning cost;
- the environment around the plant would not be polluted in any accidents.

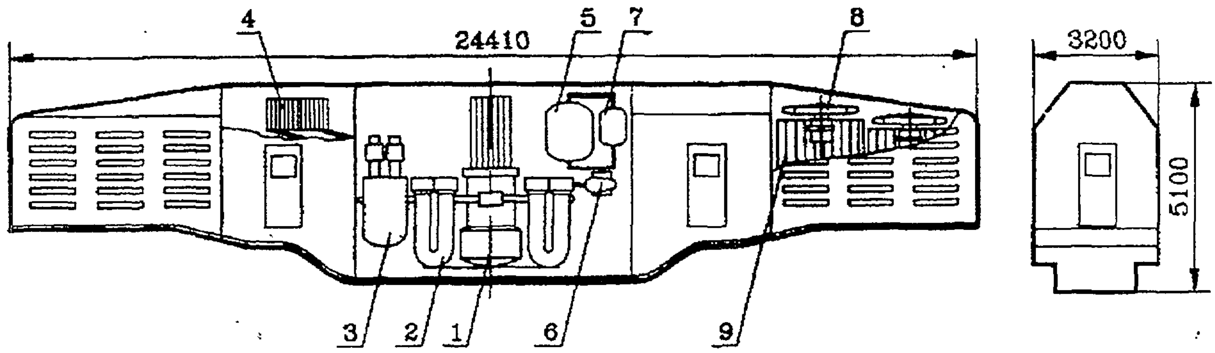
MAIN PERFORMANCES

Name	Value		Remark
Thermal power	30	MWth	
Core inlet temperature	290	^0C	
Core outlet temperature	465	^0C	
Feedwater temperature	190	^0C	
Steam pressure	3 5	MPa	
Steam flow rate	55	t/h	
Steam outlet temperature	435	^0C	
Core size diameter	0 79	m	
fuel length	0 7	m	full length 1 9 m
Fuel cladding material	SS		stainless steel with 13% Cr and 1% Ni
Fuel cladding diameter and thickness	$12 \times 0\ 4$	mm	
Fuel	UO_2		
Maximum fuel burnup	8 6	%	average — 6 %
Mean makeup fuel enrichment	26	%	
Mean volume power density	87	MW/m^3	
Mean linear fuel rod power	17	kW/m	
Electric power, net	6	MWe	
Heat supply power	up to 14	MWth	temperature condition of heat-supply system is 130 / 70°C
Possible desalination capacity	~445	m^3/h	
Possible refrigerating capacity	~10	MWth	production of ~2°C cold water
NPP Lifetime	30	years	
Safe shutdown earthquake	9		magnitude as per scale MSK-64
Load follow rate	up to 1	% per sec	
Core lifetime	70 000	eff. Hour	refuelling in ~ 10-15 years
Number of operating personnel	26	persons	for 5 shifts
Number of station modules	9-12	pieces	depending on water resourses
Mass of modules	60-220	ton	
Frequency of scheduled preventive repairs	8000	h	once a year
Starting-standby source	diesel-generator		500 kWe
Cost for preparation for mastering	75	mln US $	Far North
the commercial station	53	mln US $	Center
Cost of the commercial station	26	mln US $	33 mln US $ with the desalinated water equipment
Annual operational expenses	3-4	mln US $	with account of fuel
Prime cost of energy			
-electrical	0 08	US $ / kWh	
-heat	1 1	US $ / Gcal	
-desalinated water	0 9	US $ / m^3	
Radius of sanitary-protective area	1.0	km	

DESCRIPTION OF THE MODULES

Steam supply system module

The reactor and the heat transport system are mounted on the steam supply system module. The module compartment is hermetic. The module design allows the core to be reloaded in the site or to be transported to the Special Operation Base for refuelling. The core lifetime is about 10-15 years depending on the operation power level. The Module can be exploited alone and the reactor heat can be removed under any conditions.

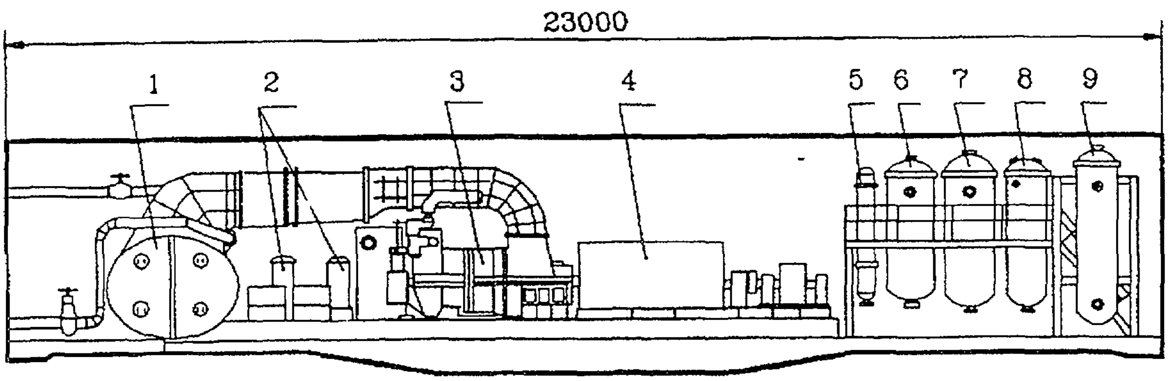


1 - Reactor	4 - Radiator of heat tubes	7 - Cooldown condenser
2 - Steam generator	5 - Separator	8 - Fan
3 - Pumps tank	6 - Controlled circulation pump	9 - Cooling radiator

Fig. 3. Steam supply system.

Turbogenerator module

There are two turbogenerator modules in the ANGSTREM plant. The module includes a turbine, a generator, a condenser and a number of heat exchangers. To increase the district heating (or desalination) capacity up to 20 MWth the backpressure turbine can be used.

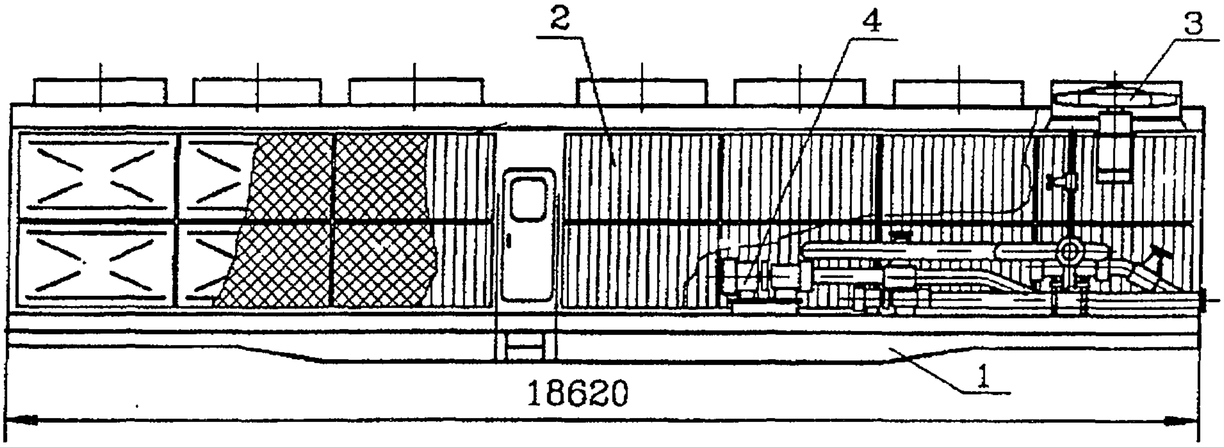


1 - Condenser	4 - Electric generator	7 - High temperature filter
2 - Oil cooler	5 - Low pressure heater	8 - High pressure heater
3 - Turbine	6 - Ion exchanger	9 - Heat exchanger for heat supply system

Fig. 4. Turbogenerator module.

Central control room module

The NPP is controlled from the Central control room module. The module is located in a separate compartment.

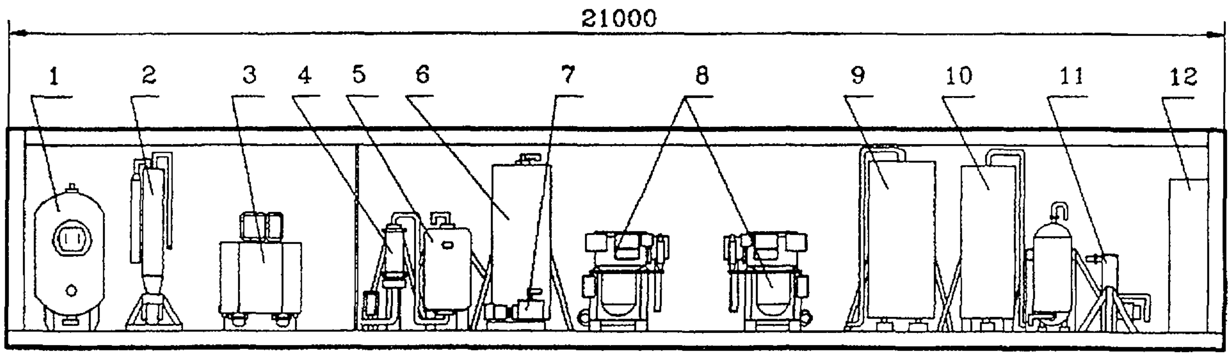


1 - Platform 2 - Air radiator unit 3 - Fan 4 - Electric pump

Fig. 5. Air radiator module.

Air radiator module

There are four air radiator modules in the ANGSTREM plant. The module is designed for the tertiary circuit water cooling. The tertiary circuit water removes the heat of main condensers. When the backpressure turbine is applied, the number of these modules can be reduced to one. Also these modules can be used for remote air-conditioning systems of consumers.

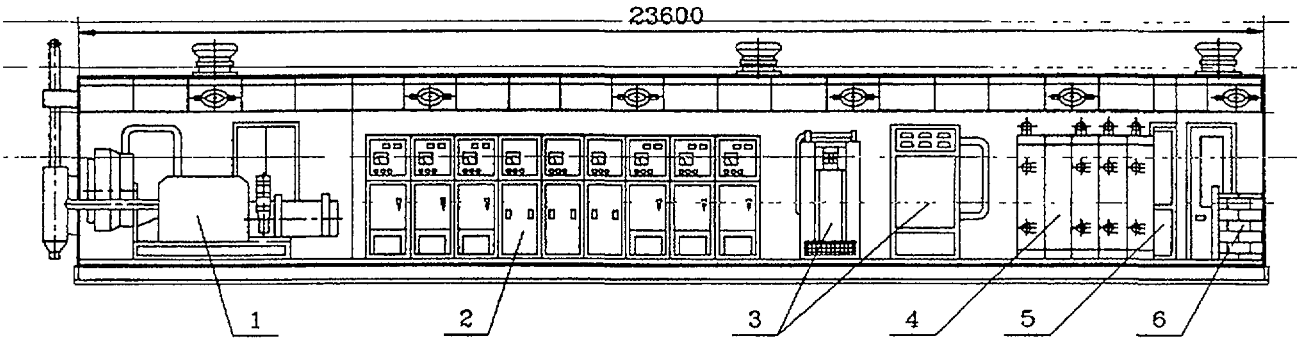


1 - Mechanical filter
2 - Mixing device
3 - Ozonizing device
4 - Electrical coagulator
5 - Clarifying filter
6 - Clarifyed water inventory tank
7 - Pump
8 - Water desalinating distillate plant
9 - Distillate inventory tank
10- Potable inventory tank
11- Water bactericide treatment plant
12- Instrumentation control pannel

Fig. 6. Water treatment module.

Auxiliary modules

The auxiliary modules are: the water treatment module, the electric switchgear device module, and the laboratory-workshop module.



1 - Diesel-generator
2 - Switchgear 6.3 kV
3 - Transformer substation 0.4 kW
4 - Uninterrupted supply set
5 - Reliable power supply switchgear
6 - Battery

Fig. 7. Electric switchgear device module.

CONCLUSION

The ANGSTREM modular-transportable nuclear power-and-heating station is based on experience of the nuclear power reactors for submarines. The main features are: compact reactor layout associated with functionally-finished, completely factory made modules; and the flexible reactor concept for co-generating electricity and heat for district heating or sea water desalination. The economic competitiveness can be made possible by the modular solution and the low sensitivity to downsizing.

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SVBR-75: A REACTOR MODULE FOR RENEWAL OF VVER-440 DECOMMISSIONING REACTORS — SAFETY AND ECONOMIC ASPECTS

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Abstract

IPPE has been developing jointly with EDB "Gidropress" a reactor called SVBR-75 (in English spelling LBFR-75: Lead-Bismuth Fast Reactor), an innovative heavy metal reactor with transparent safety characteristics [1]. The SVBR-75 Reactor Module is designed for the steam production instead of VVER-440 reactors to be decommissioned. At the renewal of the NPP unit the reactor vessel is remained at the same place, and steam generators are replaced by the new modules for SVBR-75.

The main targets are: passively safe behaviour, no pressurisation of the reactor containment (building) under any accident conditions, reduction of plant capital costs and the construction schedule by means of the modular concept and the compact layout. The SVBR-75 design, described in general terms in this paper, is based upon the safety concept proposed by IPPE and on proven technology from both LMR and HMR for submarines.

1. MAIN TARGET OF THE SVBR-75 CONCEPT

1.1. Safety targets

A low frequency of a severe accident, e.g., 10^{-6} 1/ry, is not necessarily a proof of its impossibility to occur, nor evidence that it might happen only once in one million of reactor-years of operation. Besides it is also necessary to take into account wrong aforethought actions of the people. In such a case the probabilistic safety analysis conclusions lose their validity from the ground. Significant efforts have been made during the last years for the deterministic exclusion of severe accidents which can be caused by any external or internal events, as well as the terrorism.

The SVBR-75 concept largely embodies this progress. The main safety targets may be summarized as follows:

— No core melt-down possibility and the negligible release of radioactivity in any accident conditions, by means of the reactor concept itself.
— Prompt and passive reactor shutdown after any abnormal conditions.
— Reactor cooling in natural circulation for unlimited time.
— The overpressurizing and the reactor thermal explosion excluded at a coolant emergency overheating, because there is no pressure increase.

1.2. Economic targets

The economic targets aims at a viable industrial power plant based on the specific overnight-capital cost and the construction time competitive (or less) with those of the light water reactors under development. This is achievable by means of following features:

— Modular reactor concept.
— Integrated components (compact layout)
— No pressurization to be taken into account in the design of the reactor containment.
— Primary system installation after the reactor building completion.

2. BASIC CHARACTERISTICS OF THE SVBR-75 CONCEPT

2.1. Lead-Bismuth coolant features

A lead-bismuth coolant has a number of important safety features [2]:

— It has the high boiling point (1670°C) that eliminates the possibility of over- pressurisation, boiling and thermal explosion of the primary circuit caused by overheating the core, even in the severe, beyond design basis, accident conditions.
— It makes possible to prevent the loss of coolant due to the exclusion of the coolant evaporation. The low lead-bismuth freezing-point (125°C) makes it possible to decrease (or exclude by design) the possibility of the loss of coolant caused by the primary loop leakage.
— Low chemical activity eliminates the possibility of fires and explosions in the event of coolant leakage into the reactor room or the liquid metal-water interaction due to the steam generator tube ruptures.
— The void effect of reactivity by the lead-bismuth coolant is negative.
— The use of the liquid metal with the monoblock reactor design allows passive decay cooling, even when all other heat removal systems are lost, through the reactor vessel cooling by the naturally circulating air or water surrounding the vessel. Excessive core overheating which could damage the core can be avoided.
— The low volume decrease with solidification and the relatively high plasticity of lead-bismuth alloy permit, when necessary, a planned regime of "solidifying-melting" without the deformation and damage to the reactor components.
— The coolant does not boil at the primary circuit rupture and retains the fission products, such as iodine, caesium, etc., which represent a major factor of a radiological danger in short terms after an accident. Also the coolant retains the actinides, which is important to minimize the long term consequences.

2.2. Design

The reactor is in the stage of conceptual design. The project is based on the technical specifications being proved in the past practice. Prototypes of the main equipment already exist. The primary system of the SVBR-75 reactor is of the integrated type (fig. 1). It is the so-called the monoblock reactor type. Steam generator units and primary circuit pumps are housed in the reactor vessel.

SVBR-75 overall design parameters

Name	Value		Remark
Thermal power	250	MWth	
Electric power	75	Mwe	
Core inlet temperature	280	^{0}C	
Core outlet temperature	435	^{0}C	
Operating pressure	0.1	Mpa	the pressure of the primary system gas
Feedwater temperature	226	^{0}C	
Steam pressure	4.6	MPa	
Steam flow rate	458	t/h	
Steam outlet temperature	259	^{0}C	saturated steam

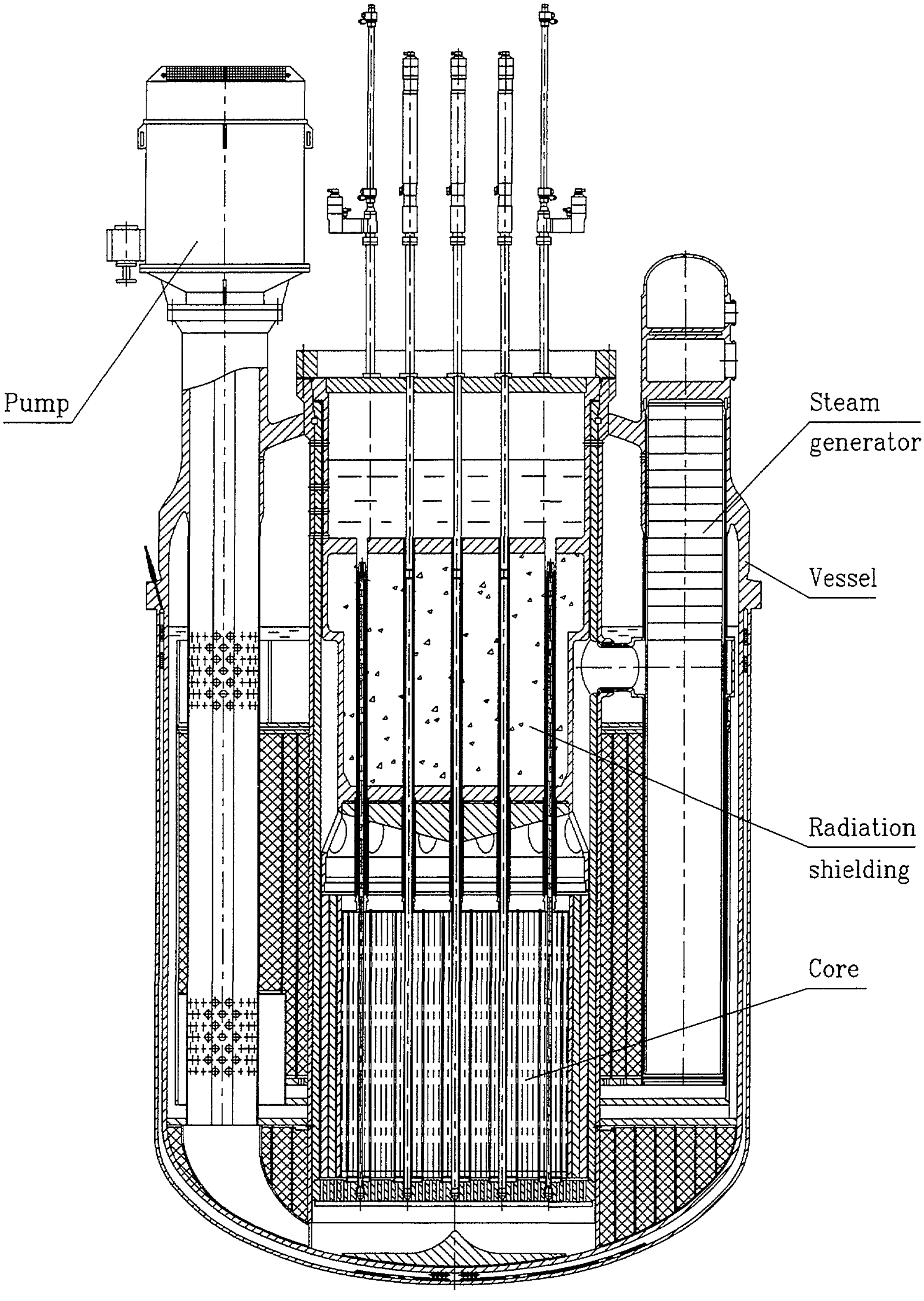


Figure 1. Reactor plant of enhanced safety with liquid–metal coolant.

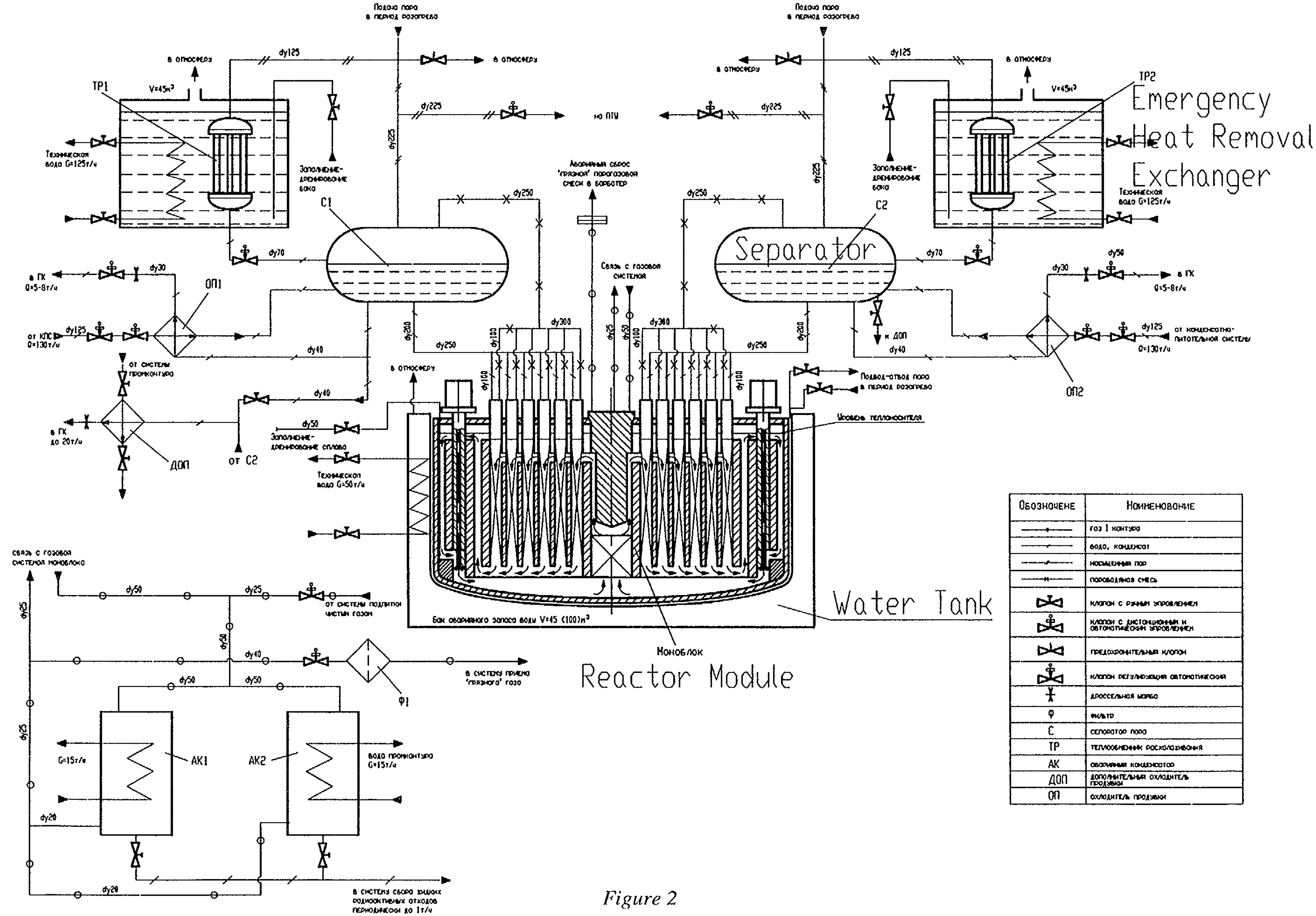

Emergency Heat Removal Exchanger
Separator
Water Tank
Reactor Module
Моноблок
TP1
TP2
C1
C2
ОП1
ОП2
ДОП
от C2
АК1
АК2
Ф1
Обозначене
Ноименование
гоз 1 контура
водо, конденсат
носыщенныя пор
пороводяноя смесь
клопон с ручным упровлением
клопон с дистонционным и овтомотическим упровлением
предохронительныя клопон
клопон регулирующия овтомотическоя
дроссельноя шайбо
Ф
фильтр
С
селоротор поро
ТР
теплообменник росхоложивония
АК
оворийный конденсотор
ДОП
дополнительныя охлодитель продувки
ОП
охлодитель продувки

Figure 2

The reactor module is a completely factory-made module. The modules can be delivered to the operation place by various transport means. The general hydraulic diagram of the SVBR-75 reactor module is shown in fig. 2.

3. MAIN COMPONENTS

Core

The reactor core has the fast neutron spectrum. One core consists of 58 fuel assembles.

Main parameters of core

Name	Value		Remark
Core size: diameter	1.68	m	
fuel length	0.9	m	full length 1.9 m
Fuel rod material	SS		stainless steel with 13% Cr and 1% Ni
Fuel rod diameter and cladding thickness	12×0.4	mm	
Fuel	UO_2		
Max fuel burnup	12	%	average — 8 %
Mean makeup fuel enrichment	18	%	
Mean volume power density	135	MW/m^3	
Mean linear fuel rod power	22	kW/m	
Fuel lifetime	7	year	

Reactor vessel

The reactor vessel is of a cylindrical shape with the semiellitical head and bottom. The material is the austenitic stainless steel. There are an inner vessel and a guard vessel. The guard vessel prevents the loss of coolant accident when the leakage from the inner vessel takes place. The reactor vessel has no penetrations below the coolant level. There are two penetrations for the gas system: the first connecting the primary gas system with the emergency condenser, and the other connecting another with the monitoring gas facility.

Steam generator unit (SGU)

There are 12 steam generator units (SGU) in the reactor module. Each SGU consists of 301 coaxial type tubes, the so-called fild tubes. The SGU produces the saturated steam. The tubeplates are arranged above of the coolant level. When tube ruptures occur, the steam is dumped to the emergency condenser.

Primary circulation pumps

The two primary pumps of the glandless type are fully enclosed in the reactor vessel. The pump motor is located above the reactor head.

Water tank

The reactor module is located in the water tank where the emergency water is reserved. There are heat exchangers in the water tank, too. During beyond design basis accidents the heat from the reactor module is removed from the reactor vessel to the water tank.

4. SAFETY

Concept of safety ensuring [2].

- The reactivity excess (in an operating reactor under working temperature) is less than a beta (delayed neutron fraction), which excludes the reactor runaway on prompt neutrons.
- Even when all steam generator units are failed, the core melt-down (beyond the design basis accident) is prevented owing to the independent emergency cooling system available and a large margin between the anticipated temperature rise and the boiling point of the coolant. This can be possible because of the heat capacity of the reactor internal structures and the coolant itself. The heat can be also removed from the reactor vessel to the water tank, located around the reactor vessel, by means of thermal conductivity and heat radiation.
- The reactor power decreases by the negative reactivity feedback in the case of emergency overheating and the simultaneous failure of the emergency protection systems.
- No materials exist in the core and in the reactor vessel which release hydrogen when irradiated or as a result of chemical reactions with the coolant and overheating.. The coolant reacts very slightly with water and air and no explosions or fires can take place.

The reactor module safety is provided owing to the defence-in-depth approach. There is the system of barriers being implemented in the design to prevent the release of radioactive substances into the environment. These barriers include:

- fuel matrix;
- fuel cladding;
- led-bismuth coolant;
- primary circuit and gas system boundary;
- Guard Vessel of the Reactor Module;
- Reinforced concrete leak-tight Reactor Module compartment.

The defence-in-depth approach in the multi-barrier system provides protection of barriers themselves, each of them being protected by the subsequent one. For example, the lead-bismuth coolant keeps the fuel claddings intact as another barrier as long as the coolant remains in the reactor vessel. The safety systems and inherent reactor safety features provide protection of safety barriers.

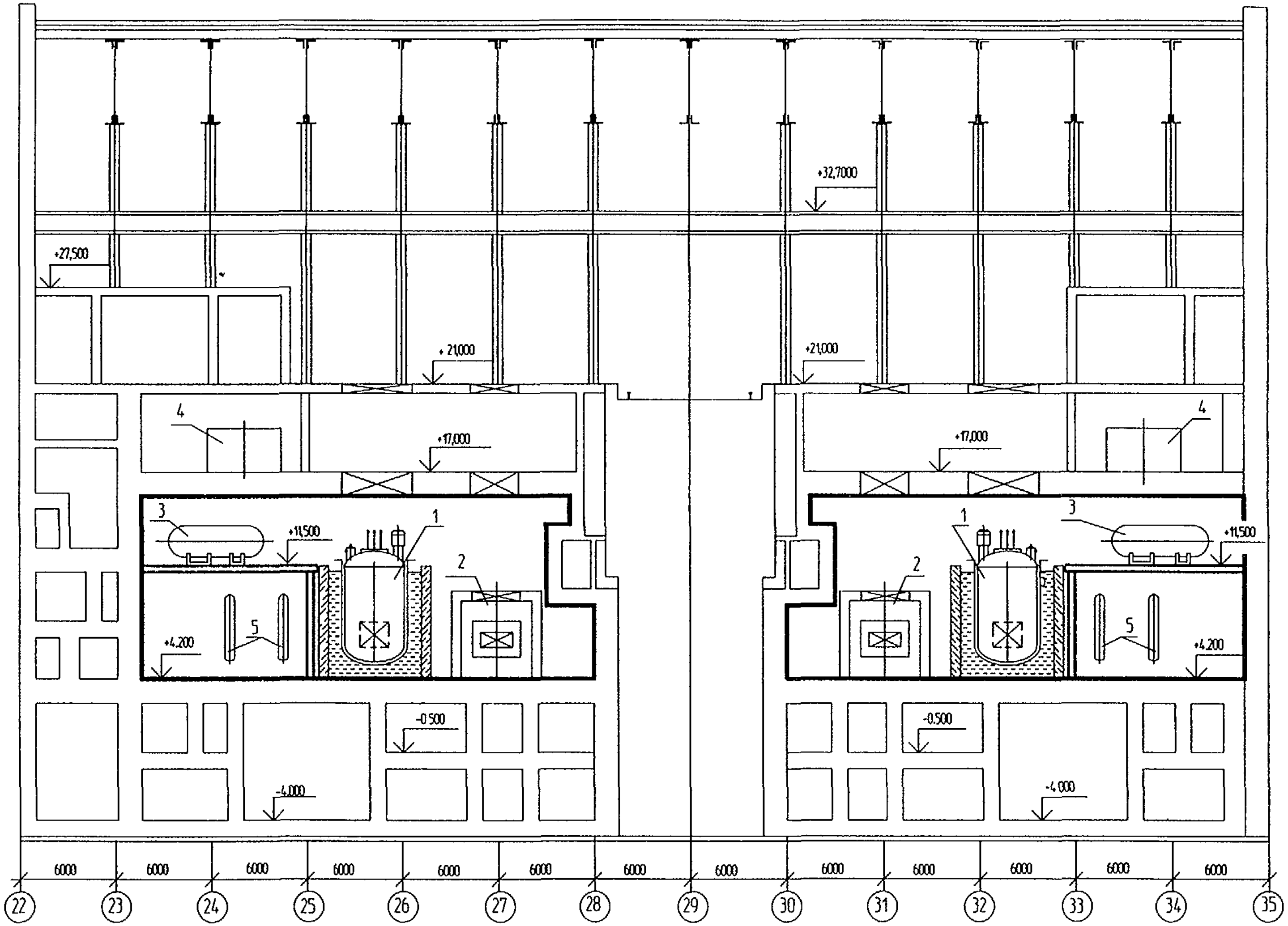
The last barrier - the reinforced concrete leak-tight reactor module compartment - is designed to withstand the extreme external impacts:

- earthquakes;
- aircraft crashed;
- shock waves.

5. MODULAR AND COMPACT PLANT

The present international trend in the nuclear industry is to place focuses on the simplification of the nuclear plant and on the reduction of the construction time. The reduced size of the most attractive modular reactors is dictated by the design target to remove the decay heat directly through the wall of the reactor vessel itself, thereby drastically reducing the number of safety-related systems.

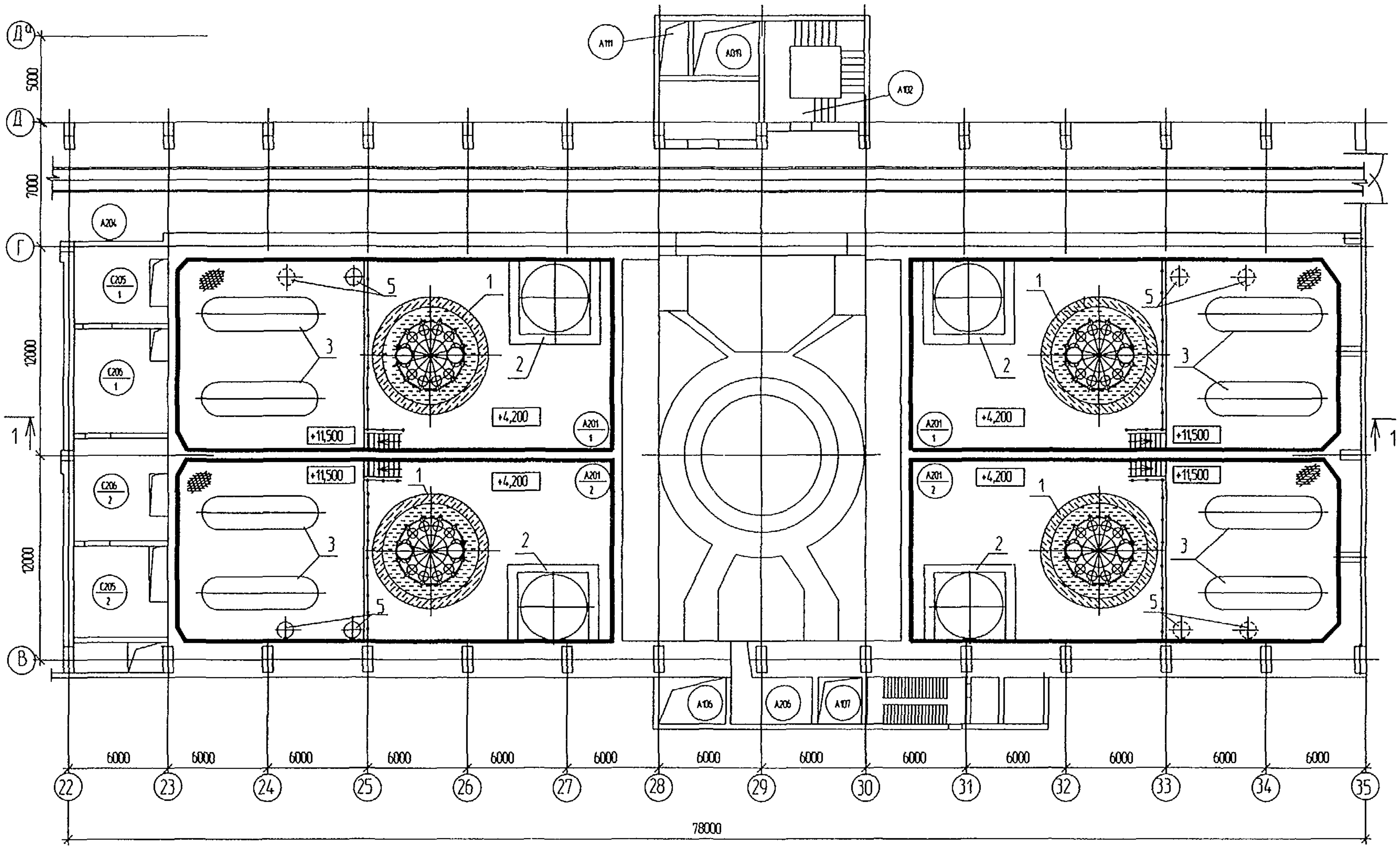
The selected unit power of the SVBR-75 reactor module (250 MWth) is consistent with this design target. The SVBR-75 reactor module is originally intended for use to renew the 2-nd, 3-rd and 4-th Novovoronezh units. Because of the compact overall dimensions of the reactor module, it can be accommodated in the present steam generator buildings. The former steam generators of PWR units will be removed from the buildings during the units decommissioning.



1 - Моноблок АППМ СВБР-75
2 - Хранилище ОВЧ
3 - Сепаратор
4 - Конденсатор аварийного расхолаживания
5 - Аварийный конденсатор

1 - SVBR-75 Reactor Module
2 - Spent Fuel Storehouse
3 - Separator
4 - Emergency Heat Removal Exchanger
5 - Emergency Condenser

Figure 3



Plane on the point 4.200

ПЛАН НА ОТМЕТКЕ 4,200

Figure 4

The unit-2 has been already closed. In the course of 2007-2009 units-2, 3, and 4 with VVER-440 reactors at the Novovoronezh NPP (NVNPP) are scheduled to be decommissioned. In this connection, SSC RF-IPPE, RDB "Gidropress" and "Atomenergoproekt" jointly have carried out an engineering and economical investigation for determining a technical possibility and economical feasibility of the NVNPP 2, 3 and 4 units renewal when their service life comes to an end. Such approach allows to keep the existing electric capacity of the NPP units and to utilize the existing building and equipment of the Novovoronezh NPP.

Figures 3 and 4 show the layout of the reactor modules at the Novovoronezh power plant unit-2 under consideration. Layout studies of the renewed Novovoronezh power plant unit-2 are in progress in "Atomenergoproekt", to optimise component arrangement and reduce the erection time of the reactor modules and of the balance of the renewed plant. The compact reactor layout is made possible by the integrated design of the primary circuit (monoblock design).

The SVBR-75 reactor module could be used as a nuclear co-generation plant for a near-by city where the steam and heat supply is centralized. Because the high and large reactor building is dispensed and the foundation loading is lightened, the plant can be located on the sites similar to the fossil-fuelled power plant like on the seabeach, soft soil, the seismic area, and so on. The environment around the plant would not be polluted in any accidents.

6. ECONOMIC ASPECTS

It is necessary to take into account that the application of small nuclear co-generation plants would be attractive, if they are environmentally-clean, safe and competitive. These are the main problems to be considered.

In the past, the thermal efficiency of electricity production at the nuclear power plants was similar to that of the conventional power plants. Under this circumstance, it was more economical to generate electricity by the large-size nuclear power plants that dominate the nuclear panorama. The thermal efficiency of the modern combined cycle turbo-gas (CCTG) power plants has recently exceeded 50% and in the near future will reach and perhaps exceed 60%, while that of the conventional water cooled nuclear reactors will remain at the about 33% level [3].

This will lead to the following two main consequences:

- if the cost of fossil fuels remain stable, the cost of heat will remain stable, but the cost of electricity will be reduced;
- the ratio of electricity/heat production will increase up to the optimum dictated by the modern fossil-fired co-generating power plants.

Qualitatively it can be affirmed that today an efficient use of energy favours the fossil fuels for electricity production and the nuclear fuel for heat production, because of the lower electric efficiency of the nuclear power plants. Quantitatively, a preliminary economic evaluation carried out comparing a CCTG with 60% efficiency, a co-generating CCTG, conventional boilers and nuclear power plants, has shown that a nuclear power plant could recover part of their lost economic attractiveness only when exploited as a co-generating plant.

A trivial prerequiste for any prospective utility interested in a co-generating nuclear plant is that an adequate co-generating reactor is available. To this end, the reactor designers, in addition to providing the reactor with convincing characteristics of radiological safety, have to overcome the unfavourable scale-demerit of downsizing, because the thermal power needed is mostly in the order of hundred megawatts against the thousands of megawatts available from the today's large nuclear reactors conceived for electricity generation only.

The specific overnight-capital cost of using the SVBR-75 reactor module for the renewal of the Novovoronezh power plant unit-2 does not exceed 560 $/kWe. Moreover, the specific overnight-capital cost for the reactor module itself does not exceed 85 $/kWe. The co-generating nuclear plant with the SVBR-75 reactor module could have the attractiveness.

The smaller reactor can be competitive, in spite of downsizing, provided that:

— the number of systems is greatly reduced;
— the specific mass of steel of the smaller size nuclear plant does not increase;
— the specific operation & maintenance cost does not become excessive.

The co-generating version of the nuclear power plants with the SVBR-75 reactor module is being designed to cope with these requirements. The 1-st unit of such power plant is being planned to be built at the IPPE site.

Furthermore, ongoing studies are evaluating the possibility of reducing operation & maintenance cost, taking advantage of the predicted simple operation of SVBR-75 modules, of the reduced number of systems, and of the modular approach that makes possible to share facilities, such as fuel handling and component handling equipment, between the reactor modules installed in the same reactor building.

Main performances

Name	Value	Unit	Remark
Electric power, net	75	Mwe	
Heat supply power	up to 200	MWth	with decreasing the electric power down to 50 MWe, temperature condition of heat-supply system is 130 / 70°C
Possible desalination capacity	~3700	m^3/h	
Possible refrigerating capacity	~70	MWth	production of ~2°C cold water
Lifetime	35	years	
Safe shutdown earthquake	9		magnitude as per scale MSK-64
Load follow rate	up to 5		% per minute
Core lifetime	50 000	eff. Hour	refuelling in ~ 7 years
Maximum fuel burnup	12		% heavy metals
Frequency of scheduled preventive repairs	8000	h	once a year
Starting-standby source	diesel-generator		500 kWe
Cost of the commercial station	90	mln US $	115 mln US $ with the desalinated water equipment
Annual operational expenses	8	mln US $	with account of fuel
Prime cost of energy:			
-electrical	0.052	US $ / kWh	
-heat	7.7	US $ / Gcal	
-desalinated water	0.8	US $ / m^3	
-			
Radius of sanitary-protective area	0.3	km	

Summary table

Name	Value		Remark
Thermal power	250	MWth	
Core inlet temperature	280	^0C	
Core outlet temperature	435	^0C	
Operating pressure	0 1	Mpa	the pressure of the Primary System Gas
Feedwater temperature	226	^0C	
Steam pressure	4 6	MPa	
Steam flow rate	458	t/h	
Steam outlet temperature	259	^0C	steam saturated
Core size diameter	1 68	m	
fuel length	0 9	m	full length 1 9 m
Fuel rod material	SS		stainless steel with 13% Cr and 1% Ni
Fuel rod diameter	12x0 4	mm	
Fuel	UO_2		
Max fuel burnup	12	%	average — 8 %
Mean makeup fuel enrichment	18	%	
Mean volume power density	135	MW/m^3	
Mean linear fuel rod power	22	kW/m	
Fuel lifetime	7	year	
Electric power, net	75	MWe	
Heat supply power	up to 200	MWth	with decreasing the electric power down to 50 Mwe, temperature condition of heat-supply system is 130 / 70°C
Possible desalination capacity	~3700	m^3/h	
Possible refrigerating capacity	~70	MWth	production of ~2°C cold water
Lifetime	35	years	
Safe shutdown earthquake	9		magnitude as per scale MSK-64
Load follow rate	up to 5		% per minute
Core lifetime	50 000	eff Hour	refuelling in ~ 7 years
Maximum fuel burnup	12		% heavy metals
Frequency of scheduled preventive repairs	8000	h	once a year
Starting-standby source	diesel-generator		500 kWe
Cost of the commercial station	90	mln US $	115 mln US $ with the desalinated water equipment
Annual operational expenses	8	mln US $	with account of fuel
Prime cost of energy			
-electrical	0 052	US $ / kWh	
-heat	7 7	US $ / Gcal	
-desalinated water	0 8	US $ / m^3	
Radius of sanitary-protective area	0 3	km	

7 CONCLUSION

The SVBR-75 reactor module intends the renewal of the Novovoronezh power plant units-2, 3, and 4 It is based on the experience of the nuclear power facilities for submarines Based on the SVBR-75 reactor module design the innovation of the nuclear co-generation power plant is under development

The main features of SVBR-75 are as follows:

- Outstanding passive safe characteristics of the reactor, which include core shutdown and heat removal capabilities in all accident conditions and no release of radioactive substances to the outside of the reactor building.
- Compact reactor arrangement, which was made possible by the modular fabrication and erection, and by the integrated (monoblock) design of the primary circuit.
- Reactor concept flexible for combined generation of heat and electricity, which was made possible by the modularization of the units and the low sensitivity to downsizing.

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PROPERTIES OF LEAD–BISMUTH COOLANT AND PERSPECTIVES OF NON-ELECTRIC APPLICATIONS OF LEAD–BISMUTH REACTOR

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Abstract

Key physical and chemical properties of lead-bismuth eutectic alloy are reviewed. Based on the low chemical activity of the alloy to other work media, a new concept of direct contact heat exchangers is proposed. A series of experiments were performed to validate the concept, using water, model salt solutions of sodium chloride, and oil. Key experimental results are summarized in the report.

PROPERTIES OF LEAD-BISMUTH COOLANT

The eutectic lead-bismuth (44,5%Pb, 55,5%Bi) alloy has a potential to used as a coolant in the primary circuits of special-purpose nuclear power installations. Table I shows some physical and thermal-physical properties characterizing this alloy from the viewpoint of its use as a coolant.

Table I Physical and Thermal-Physical Properties of Eutectic (44.5% weight Pb and 55.5% weight Bi) Lead/Bismuth Alloy

Melting temperature	- 396.6 K
Boiling temperature	- 1943 K
Volumetric expansion coefficient	- 1.19 x 10^{-4} K^{-1}
Volume change at melting	$-\Delta V = \frac{V_l - V_s}{V_s} \cdot 100 - (0 to 1.67)\%$

Dependence of some properties on temperature

Parameter	Temperature, °C					
	130	**200**	**300**	**400**	**500**	**600**
Density, ρ, kg/m³	10570	10486	10364	10242	10120	10000
Heat capacity, c_p, J/kg·K	146	146	146	146	146	146
Kinematic viscosity, 10^8, m³/s	31.4	24.3	18.7	15.7	13.6	12.4
Prandtl number, $Pr \cdot 10^2$	4.45	3.18	2.24	1.72	1.37	1.15
Heat conductivity, λ, W/m·deg.	10.93	11.74	12.67	13.72	14.65	15.81
Thermal conductivity, $a \cdot 10^6$, m²/s	7.1	7.6	8.3	9.1	9.9	10.8

As compared with other liquid metals being used for the same purposes, its high density and the high boiling temperature are noteworthy. The latter property is the obvious advantage of this coolant, because it enhances the reliability and safety of the installations. Specific physical-chemical properties of lead-bismuth melts favour this too. Some properties of primary significance are presented in Table II.

Table II Physical-Chemical Properties of Pb-Bi Melt

Solution of some elements,
where C_s is the saturation concentration (mass %), and T is the temperature (K).

Oxygen: $\lg C_s = 1.2 - \frac{3400}{T}$ (T=673 to 973 K)

Hydrogen: $\lg C_s = -9.65 - \frac{670}{T} + 1.51 \lg P_{H_2}$ (T=398 to 773 K)

Iron: $\lg C_s = 2.01 - \frac{4380}{T}$ (T=823 to 1053 K)

Chromium: $\lg C_s = -0.02 - \frac{2280}{T}$ (T=673 to 1173 K)

Nickel: $\lg C_s = 1.53 - \frac{843}{T}$ (T=673 to 1173 K)

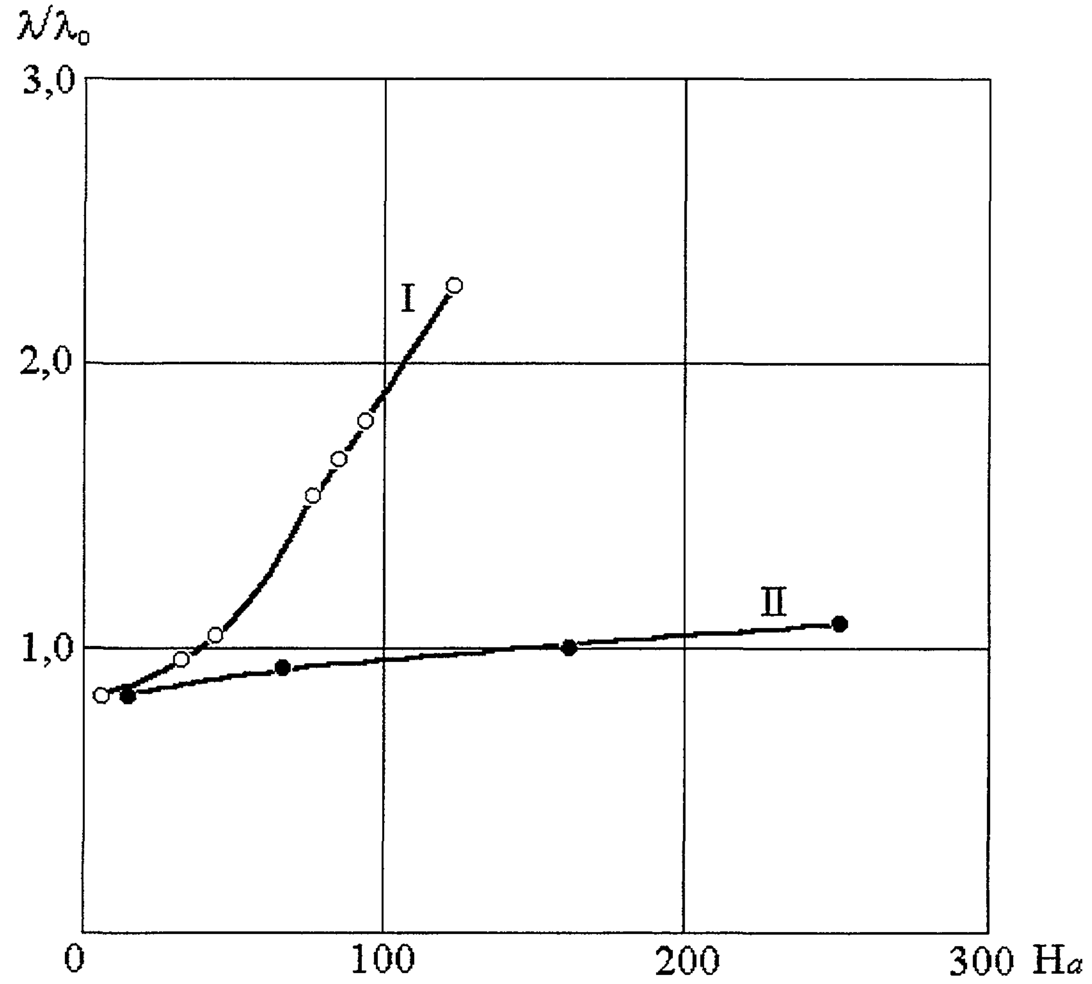
Interaction with some process impurities

Governing chemical reaction	Result of interaction
Water and steam	
$Pb+H_2O \leftrightarrow PbO+H_2$	Llead oxide slags are formed and the hydrogen is released. Accumulation is insignificant.
Air	
$Pb + \frac{1}{2} O_2 \leftrightarrow PbO$	Slags are formed. Amount of accumulation depends upon specific conditions of interaction. With the calm melt surface, the process is damped in time

It is worth noting that the chemical activity of the coolant to water, steam and air, i.e. to substances which can interact with the coolant in some accident conditions of the installation is relatively low. Among the causes of such inertia of lead-bismuth, the following two should be emphasized:

- the first cause is primarily related to the low chemical activity of melt components, ranging in the region close to the noble metals;
- the second cause is related to the ability of this alloy to form on the interface with interacting substance thin layers from the products of this interaction, which prevent further proceeding of the process. In the case of oxidation by air, these layers consist of melt components oxides and in the case of interaction with structural steels of oxides - of components of these steels.

The experimental data on a substantial reduction of hydraulic resistance when the melt flows in the strong magnetic fields illustrate the effect of such layers on coolant properties. Experimental results are presented in Fig.1 in the form of the relative resistance coefficient versus the Ha number, which is proportional to the magnetic field strength. As evident from the figure, the effect of the presence of oxidic layer on the tube surface is remarkable, and it becomes more substantial, as the strength of the magnetic field increases. Thus, unlike other liquid metals, this Pb-Bi coolant is suitable for heat removal in the strong magnetic fields.



$$\lambda_0 = 0{,}0032 + \frac{0{,}221}{Re^{0{,}237}}$$

$$H\grave{a} = \grave{a} B_0 (\sigma/\eta)^{0{,}5}$$

$$Re = 2{,}2x10^5$$

$$T = 300\ ^0C$$

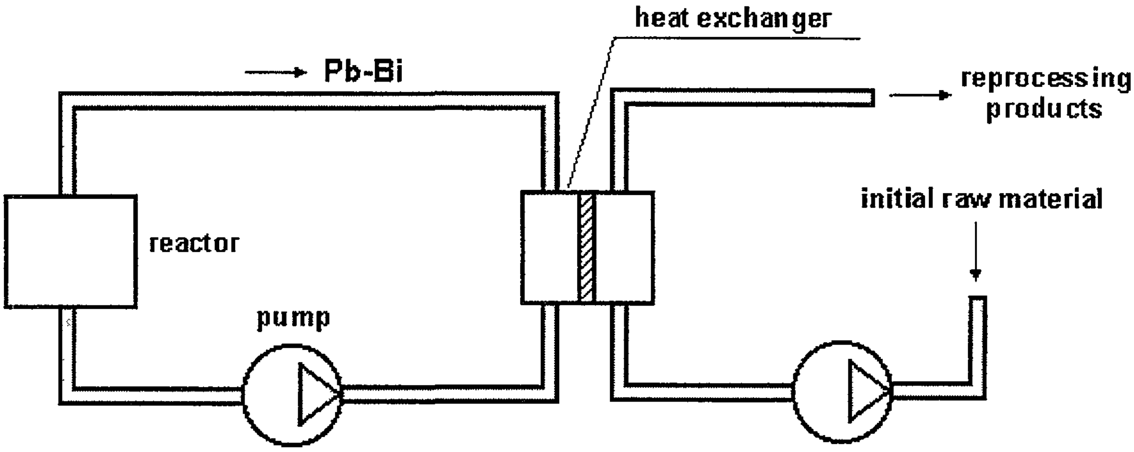
Comparison of Resistance Coefficients for Non-Oxidized (I) and Oxidized (II) Tubes with Pb-Bi Alloy Flow in Cross Sectional Magnetic Field.

FIG. 1. POSSIBILITIES OF USING LEAD-BISMUTH COOLANT FOR HEAT REMOVAL IN STRONG MAGNETIC FIELDS

It should be emphasized that the study of lead-bismuth melt properties and processes carried out on the closed circulation loops made it possible to develop both theoretically and practically a technology and rules for handling this coolant, enabling all questions related to its use in the nuclear power installations to be positively settled.

Regarding the non-electroenergetic use of facilities with the lead-bismuth coolant, Fig.2 shows two absolutely different diagrams, in particular, for the technological processes which require the large amounts of heat supply, for instance, in desalination, oil refining and in some other chemical processes. In the upper part of the figure, a conventional circuit of heat transmission is presented, where heat transfer is implemented in the standard heat exchangers with a partition baffle between the primary coolant and the medium to be reprocessed.

Traditional reprocessing circuit



Non-traditional reprocessing circuit with using direct-contact heat exchangers

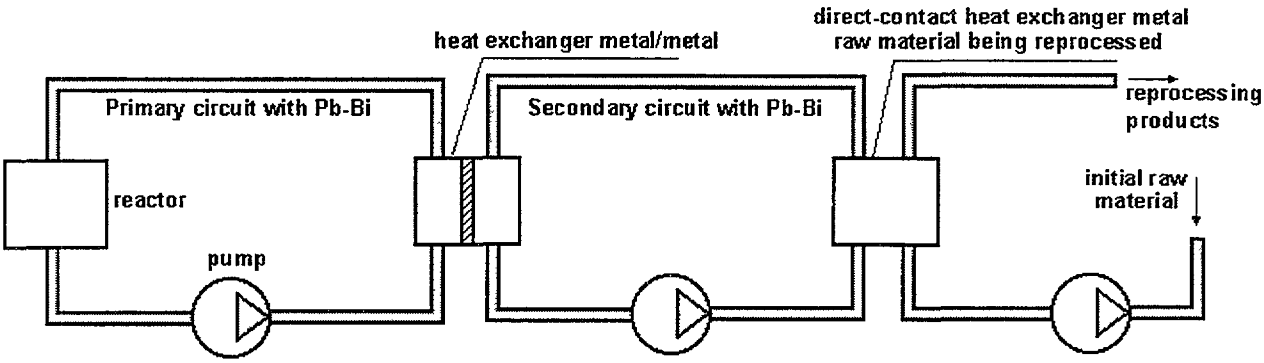


FIG. 2 FLOW DIAGRAM OF POTENTIAL USING OF NUCLEAR POWER INSTALLATION WITH LEAD-BISMUTH COOLANT FOR REPROCESSING VARIOUS ORGANIC AND NON-ORGANIC LIQUID AND GASEOUS MEDIA

The use of lead-bismuth reactor in such a circuit does not pose any additional problems. In principl, it is necessary to solve the problem by manufacturing a reliable tube-in-shell heat exchanger which has no significant problem of corrosion and slagging induced by the coolant and raw materials.

The lower part of the figure shows a new concept of the reprocessing circuit with direct-contact heat exchangers. It has a three-loop scheme where a heat exchanger for direct mixing of raw materials with the secondary circuit coolant is present. The use of such a heat exchanger can be shown to be beneficial in a number of cases. The advantages and disadvantages of such heat exchangers are well known and are summarized below.

Main Advantages and Disadvantages of Direct-Contact Heat Exchangers

Advantages:

- low hydraulic resistances;
- potentially higher specific loadings;
- ability of transferring heat at more lower temperature differences;
- simple design;
- lower cost;
- **free of problems of heat transfer surface corrosion and slagging.**

Disadvantages:

- necessity of maintaining similar pressures at places of mixing;
- **possible mutual contamination of contacting media.**

However, it should be noted that such positive factors as the absence of corrosion and slagging on the heat exchange surface is leveled to some extent by the problem of mutual contamination of contacting media.

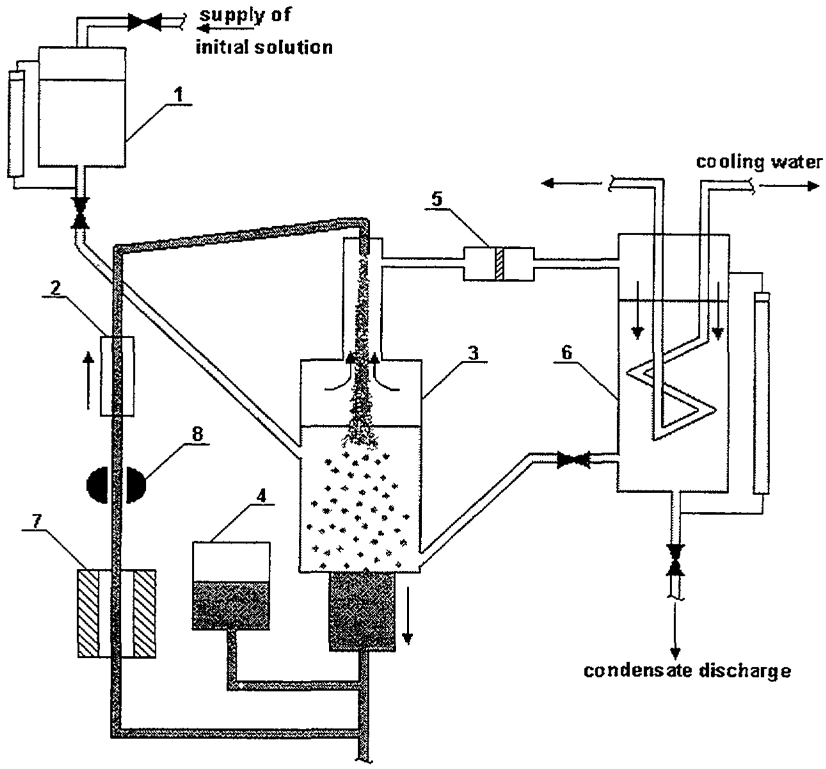
As it was shown above, the properties of the lead-bismuth coolant give good prospects that it will be sufficiently inert not only to water and steam but also to quite a number of other substances including organic ones. To validate this, mock-ups of direct-contact heat exchangers and test facilities were fixed. Some of them are demonstrated in Figs.5[1] and 6. The test facilities simulated different versions of the melt contact with other liquid media under the conditions of both forced and natural circulation of the melt to be investigated. A large number of tests were conducted, where distillate or process water, model salt solutions of sodium chloride with the initial sodium concentration of ~3% to ~30% mass, as well as oil were used as a raw material for reprocessing. The duration of experimental investigations varied on different test facilities from several hours to several hundreds of hours. The results of conducted investigations are shown in Table III, and they can be summerized as follows:

Table III **Main Results of Conducted Experimental Investigations**

Type of reprocessed raw material	Duration of mock-up operation in an experiment, τ, h	Melting temperature, T, ^{O}C	Availability of reprocessing product purification system	Level of melt contamination by impurities, C, % weight	Level of reprocessing liquid product contamination, C, mg/l
Water	τ=20 to 600	500 to 350	not available	C_{O2} ~$2x10^{-6}$	C_{Pb}~0.2 C_{Bi}~0.6
	τ=2 to 10	400÷350	available	C_{O2} ~$2x10^{-6}$	C_{Pb}~0.02 C_{Bi}~0.003
Model salt solutions of sodium chloride, C_{NaCl}=3 to 30 % mass	τ=2 to 10	400 to 300	not available	C_{O2} ~$2x10^{-6}$	C_{pb}~10 to 1 C_{bi}~4 to 0.2
	τ=2 to 10	400 to 300	available	C_{O2} ~$2x10^{-6}$	C_{pb}~0.7 to 0.6 C_{bi}~0.8 to 0.1
Oil	τ=100 to 140	500 to 300	not available	Volumetric contamination is not detected	Light fractions C_{pb}~0.08 to 0.05 C_{bi}~0.02 to 0.01 Heavy fractions C_{pb}~4 to 0. 1 C_{bi}~0.2 to 0.05

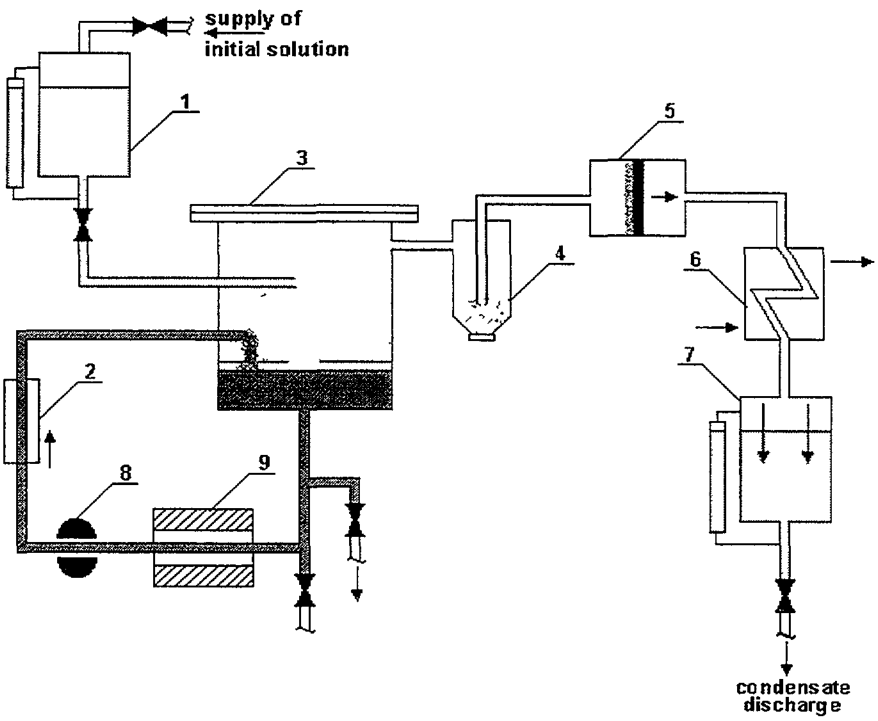
While the lead-bismuth coolant is interacting with the above listed liquid media within a temperature range of ~500^{O}C to 300^{O}C, the coolant contamination by the components of the raw material being reprocessed was not detected during the tests and after their termination, except the oxygen impurity (when interacting with water and water solution). At the same time it was confirmed that the

[1] Because of editing reasons, there are no Fig. 3 and 4.



1. flow rate tank
2. primary heater
3. contact apparatus
4. buffer reservoir
5. filter
6. condenser
7. electromagnetic pump
8. flowmeter

a) Closed-circuit or open-circuit relative to raw material reprocessing



1. flow rate tank
2. primary heater
3. contact apparatus
4. cyclone
5. filter
6. cooler
7. accumulation reservoir
8. flowmeter
9. electromagnetic pump

b) Open-circuit relative to raw material reprocessing

Fig. 5. Schematic Diagram of Direct-Contact Device Mock-Up with Forced Circulation of Coolant

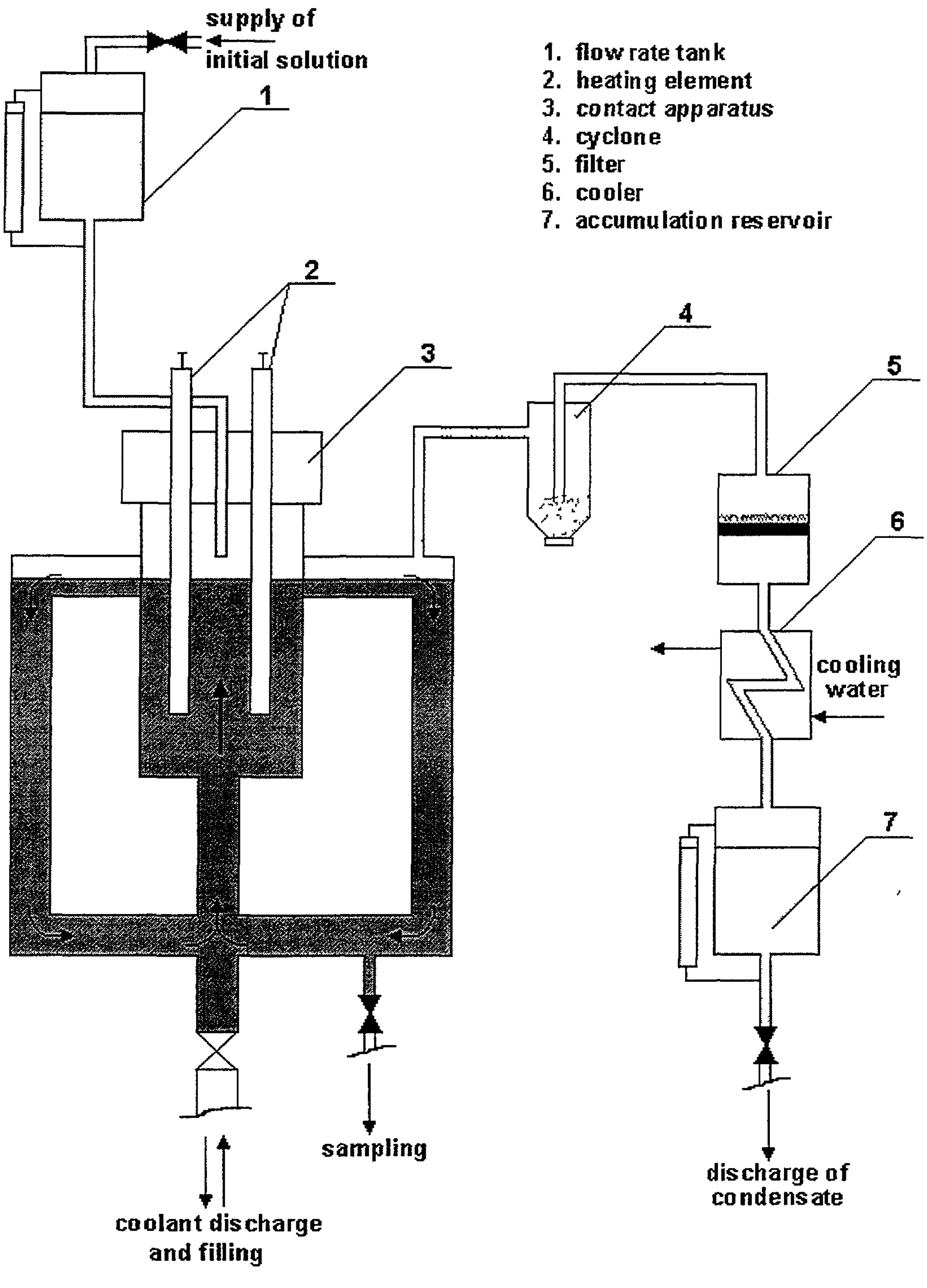


Fig. 6. Schematic Diagram of Direct-Contact Device Mock-Up with Free Circulation of Coolant

initial coolant is purified from electronegative impurities (Fe, Cr, etc.) in a number of cases due to the oxidative refining of the melt in the course of its interaction with water and steam.

With the time, the concentration of the melt components in the liquid raw material reprocessed products was observed to increase. The concentration of lead, a more chemically active substance, was higher than that of bismuth. This difference reached sometimes an order of magnitude and even bigger.

The products contamination rate was evaluated in terms of one year of continuos operation of the facility depending on the type of raw material being reprocessed and the conditions of the direct contact.

Simplest devices such as cyclones, inertial precipitator and metal-ceramic filters for the purification of reprocessed products from impurities, turned out to produce much purer condensed products.

The tests as performed have shown that the problem of the mutual contamination of the coolant and the contacting raw materials tested for reprocessing can be perfectly solved by the existing methods and devices for the liquid and gas purification. The mixer-type heat exchangers with a lead-bismuth alloy can be competitive with conventional tube-in-shell heat exchangers in some cases and can be used in non-electroenergetic technologies for a lead-bismuth coolant.

DEVELOPMENT OF SMALL (10 MW THERMAL) NUCLEAR PLANT WITH LEAD-BISMUTH COOLANT FOR ELECTRICITY AND HEAT CO-GENERATION, PRODUCTION OF FRESH WATER AND HYDROGEN

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Abstract

The paper is presented to evaluate the possibilities of developing a small capacity transportable nuclear power and technology plant (SC TNPTP) with lead-bismuth as a coolant intended for electricity and heat supply, production of fresh water and hydrogen. Its basic distinctive features compared to other various small capacity nuclear power plants (SC NPPs) under development are: simplified design; extended life time; safety and reliability; a pumpless scheme of coolant circulation; utilization of a gas turbine facility with a closed circuit; variable production ratios of the electricity, heat and its products such as fresh water and hydrogen. All these features give grounds for a new generation nuclear power and technology plant (NPTP). TNPTP uses a compact desalinator based on a multi-effect distillation process and/or an electrolizer with metal hydrides for hydrogen accumulation. The plant's capacity of 10 MW(thermal) has been chosen based on the economic optimization of the electricity and thermal heat to be supplied for the production of fresh water and hydrogen (even if the operations are simultaneous, i.e., cogenerating) in the isolated regions far remote from the centralized energy supply sources.

The main purpose of the research

The general purpose of the research was to identify main characteristics of small capacity NPPs (SC NPPs) (10MW thermal), for electricity production and thermal energy supply for heating, desalination of sea water and for hydrogen production. The NPP to be pursued should have superior characteristics compared with other SC NPPs known, in terms of safety, life time, deliverability of the equipment to the construction site with maximum readiness for service, the possibility to produce various products, and to change the ratio between the energy types to be generated depending on the amount of electricity and heat, fresh water or hydrogen demands. In the research the experience of NPPs for ships has been extensively used.

Research substantiation

Under growing shortage and increasing costs of electrical and thermal power supplied from the centralized nets, there are various incentives to develop universal, reliable, safe and economic autonomous sources of electricity and thermal energy to be used for heating, water desalination and hydrogen generation. They are: increasing costs of organic fuels in Russian Federation; increasing number of economic communities independent in their power supply; and the importance of producing effective energy carrier like hydrogen.

The lead-bismuth properties such as no burning, no interactions with water, high boiling temperatures offer a good opportunity of developing a SC lead-bismuth NPP with improved characteristics of safety and economy as compared with SC NPP of other types. A disadvantage of an prototype SC NPP with Pb-Bi named TES-M is that its power is comparatively small (1 MWe per unit). It is necessary to increase its power (at least twice), with no considerable increase of its mass and dimensions, and without impairing favourable other characteristics. This should be the main goal of the project. Research works carried out by specialized institutions (e.g., RPA "Malaya Energetica" - Small Scale Power Engineering - Moscow) made it possible to determine the economically competitive minimum capacity of NPP. It is about 3-4 MWe for the Russian conditions.

The main NPP technical specifications

- A core with thermal (or intermediate) neutron spectrum.
- Application of liquid metal Pb-Bi as a coolant.
- Low core power density.
- Integrated configuration of the primary circuit in one pressure vessel. Thus the circulation contour of the primary coolant becomes simpler, and compact and simple design can be pursued.
- Equipment of the primary circuit (the core absorber rods and gauges, modular air heaters, devices of coolant technology) is of replaceable type.

One of conditions of improved reliability and extended life time of a reactor facility (RF) will be the ellimination of a mechanical pump for circulating the primary circuit coolant. The coolant is circulated by natural convection. The capacity of this facility may be increased up to 10 MWth using additional pumpless means of circulation, e.g., gas injection above the core to intensify the circulation (gas lift method). Furthermore TNPTP includes:

- carnotype gas turbine facility (GTF) with a closed circuit;
- desalinator based on the multi-effect distillation process;
- hydrogen producing and accumulating systems.

The plant can be fabricated in several transportable blocks or as a floating plant. The preliminary main characteristics of the facility proposed are listed in Table I.

Table I Main technical characteristics

No.	Parameter, measurement units	Value
1.	Unit's thermal power, MW	10
2.	Maximum electrical power, MW	2
3.	Maximum thermal power supplied, MW	8
4.	Maximum yield of fresh water, m^3/day	5000
5.	Air temperature at the inlet of the air heater, °C	255
6.	Air temperature at the air heater outlet, °C	500
7.	Outlet air pressure, MPa	0.35
8.	Temperature of coolant at the core inlet, °C	400
9.	Temperature of coolant at the core outlet, °C	520
10.	Core height, mm	~1000
11.	Core diameter, mm	~1000
12.	Integral reactor facility dimensions, D×H, mm	~2500×3000
13.	Weight of a mono-unit, t	~50
14.	Operation time; effective hours	90 000
15.	Uranium-235 load, kg	70
16.	Fuel material	U-Zr alloy
17.	Energy converter type	carnotype GTF with a closed circuit
18.	Fabrication type	Transportable blocs or floating type

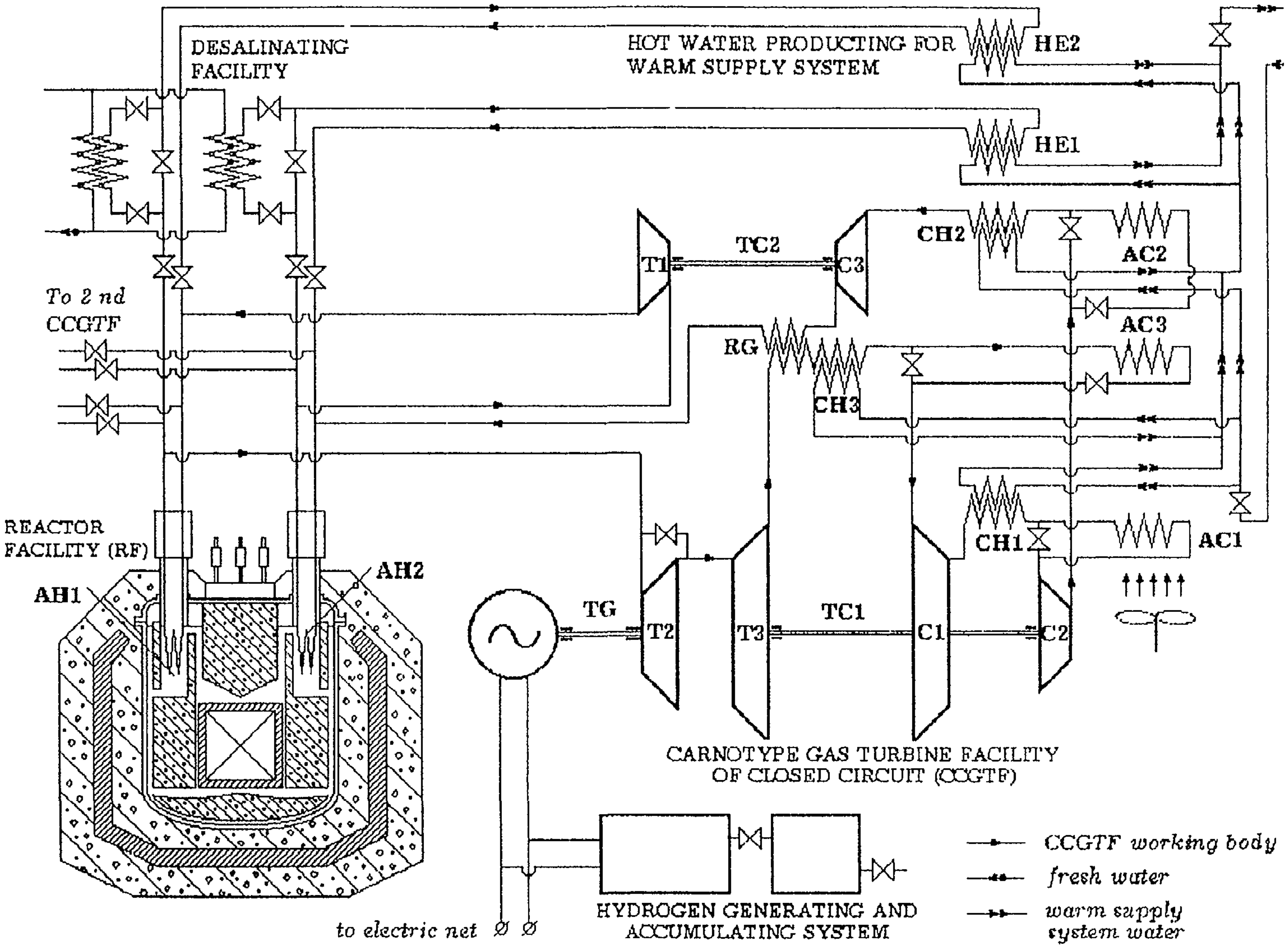

Fig. 1. TNPTP block principle scheme.

The state of the technology offered

Ample experience has been accumulated with all principal elements of the NPP: RF, energy converter, technological systems, desalinating devices and hydrogengeneration facility. Prototype NPPs are available, with pumped coolant circulation and steam turbine installations as energy converters.

The facility proposed differs from the existing ones: it has no mechanical pumps; more compact and safer GTF of a closed type is installed instead of a steam turbine; power density of the core is decreased. SSC RF IPPE and the Central Design Bureacx of Machine Building (CD BM, S.-Petersburg) have prepared a conceptual design of a similar NPP of 1 MWe on the order of Kamchatenergo (without desalinating and hydrogengenerating systems).

Additional means for pumpless circulation of the coolant is introduced in this SC NPP proposed. It provides a considerable increase (at least two-fold) of the prototype NPP power, which may result in a considerable expansion of possible development of the reactor. A characteristic feature of the new NPP is that the open cycle is replaced by a carnotype GTF with closed circuit (CC GTF) improving the plant's competitiveness and safety. Experience necessary for this application is already available from CC GTF, desalination and hydrogen production systems.

Potential advantages of the NPP proposed

The SC NPP proposed is intended for supplying power to isolated regions difficult to access, water desalination and hydrogen generation. It has the following potential advantages as compared to alternative power plants:

- maximum shop fabrication and minimum construction and mounting work at the construction site;
- transportability even in bad transportation conditions;
- comparatively simple design and improved reliability;
- increased safety in transportation, mounting works, operation and dismantling;
- comparatively low cost, due to a small fuel load and simple design;
- ecological compatibility;
- long life time (without reloading);
- minimum services required during operation;
- flexibility of the ratio of energy generated (electric/thermal) depending on required electricity and heat, the output of fresh water and hydrogen.

II.1. DESIGN ASPECTS OF NUCLEAR HEAT APPLICATIONS

High and medium temperature heat applications

OVERVIEW OF HTGR HEAT UTILIZATION SYSTEM DEVELOPMENT AT JAERI

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Abstract

The Japan Atomic Energy Research Institute (JAERI) has conducted research and development of nuclear heat utilization systems of a High Temperature Gas cooled Reactor (HTGR), which are capable to meet a large amount of energy demand without significant CO_2 emission to relax the global warming issue. The High Temperature engineering Test Reactor (HTTR) with thermal output of 30 MW and outlet coolant temperature of 950^{0}C, the first HTGR in Japan, is under construction on the JAERI site, and its first criticality is scheduled for mid-1998. After the reactor performance and safety demonstration tests for several years, a hydrogen production system will be connected to the HTTR. A demonstration program on hydrogen production started in January 1997, in JAERI, as a study consigned by the Science and Technology Agency. A hydrogen production system connected to the HTTR is designed to be able to produce hydrogen by steam reforming of natural gas, using nuclear heat of 10 MW from the HTTR. The safety principle and standard are investigated for the HTTR hydrogen production system. In order to confirm safety, controllability and performance of key components in the HTTR hydrogen production system, an out-of-pile test facility on the scale of approximately 1/30 of the HTTR hydrogen production system is installed. It is equipped with an electric heater as a heat source instead of the HTTR. The out-of-pile test will be performed for four years after 2001. The HTTR hydrogen production system will be demonstratively operated after 2005 at its earliest plan. Other basic studies on the hydrogen production system using thermochemical water splitting, an iodine sulphur (IS) process, and technology of distant heat transport with microencapsulated phase change material have been carried out for more effective and various uses of nuclear heat.

1. Introduction

Consumption of a huge amount of fossil fuels resulted from human activities since the industrial revolution has caused an enhanced global warming. In order to relax the global warming issue, that is, reduction of CO_2 emission, new energy resource/carrier and technology are required to be developed at present. Nuclear energy can satisfy a large amount of energy demands without significant CO_2 emission and its power generation technology by steam turbine is said to be proven. The ratio of electricity to the secondary energy demands, however, is about 21% in 1995 in Japan and the value is estimated to be only 23% in the year 2010. To solve the global warming issue effectively, it is very important to enlarge the nuclear energy utilization in nonelectric field, of which share holds 80% of secondary energy demand, for reducing the usage of fossil fuel. There are various studies of the nuclear energy utilization technology in nonelectric field in the past, but these technology have not yet utilized positively in our society so well as the nuclear power generation.

The present-day Light Water Reactor (LWR) and the developing Fast Breeder Reactor (FBR) provide heat at temperatures of 300 and 500^{0}C, respectively. These are restricted in using heat for nonelectric utilization. About 60% of the total energy demands in Japan are for energy having temperatures below 1000^{0}C. The use of nuclear heat is being considered to meet these demands. The type of reactor which has the capability to generate electricity and high temperature process heat is the High Temperature Gas cooled Reactor (HTGR). The HTGR can generate 800^{0}C to 1000^{0}C heat at reactor outlet because its core is composed of ceramic materials and it is cooled by helium gas. Much of the heat energy now provided by fossil fuels could conveniently be provided by HTGR nuclear heat. In fact, if processes to produce hydrogen gas from water with the HTGR can be realized, a complete clean energy system independent of fossil fuels would be established in the future.

The Japan Atomic Energy Research Institute (JAERI) has been constructing a 30 MW HTGR with reactor outlet coolant temperature of 950°C named HTTR (High Temperature engineering Test Reactor), to develop technology and to demonstrate effectiveness of high temperature nuclear heat utilization. Attainment of the first criticality is scheduled for December 1997, and reactor performance and safety demonstration tests will be carried out for several years [1]. After that, a high temperature nuclear heat utilization system will be connected to the HTTR.

A hydrogen production system by steam reforming of natural gas is selected to be the first heat utilization system of the HTTR since its technology matured in fossil-fired plant enables to connect to the HTTR in the early 2000's and its demonstration test will contribute to all other hydrogen production systems. In a preliminary design conducted from 1990 through 1995, a framework and key design of the HTTR steam reforming system have been developed. The HTTR steam reforming system is designed to be able to produce hydrogen with the production rate of 3800 Nm^3/h, using nuclear heat of 10 MW provided by the HTTR. A demonstration program on hydrogen production was just started on January in 1997, in JAERI, as a study consigned by Science and Technology Agency. Such a heat utilization system is first connected to a nuclear reactor, hence an out-of-pile test, a hydrogen permeation test and a material test of a reforming tube of a steam reformer are carried out prior to the demonstration test of the HTTR steam reforming system. In order to confirm safety, controllability, and performance of key components in the HTTR steam reforming system, a facility for the out-of-pile test is fabricated on the scale of approximately 1/30 of the HTTR steam reforming system. It simulates key components downstream from an intermediate heat exchanger (IHX), being equipped with an electric heater as a heat source instead of the HTTR. The out-of-pile test will be performed for 4 years after 2001. After that the HTTR steam reforming system will be connected to the HTTR and operated demonstratively.

On the other hand, hydrogen production from water is considered as an ideal system for hydrogen production using the HTGR because no CO_2 emission is expected from the system. JAERI has been conducting a basic study on a hydrogen production process by water splitting, a thermochemical iodine sulphur (IS) process as one of the future heat utilization systems of the HTTR following the steam reforming system. Plural chemical reactions are used in the IS process, which works like a chemical engine to produce by absorbing high temperature heat from the HTGR. Stable production of hydrogen and oxygen from water was successfully demonstrated for 24 hours in a closed-cycle operation using a laboratory-scale test apparatus. A study on materials suitable for corrosive process environments is under way for engineering-scale realization. It is also to develop technology for distant heat transport to achieve high performance and enlargement of nuclear heat utilization. Microcapsules of phase change materials have been developed for heat transport.

2. HTTR Hydrogen Production System by Steam Reforming of Natural Gas

2.1 Design of HTTR steam reforming system

The HTTR supplies nuclear heat of 10 MW with 950°C to the IHX in the primary cooling loop, and then the nuclear heat is transferred through the IHX to the secondary helium loop to be utilized for the production of hydrogen. Due to heat loss along the secondary helium gas piping from the IHX to a steam reformer (SR) composed of the HTTR steam reforming system, the secondary helium gas temperature is reduced to 880°C at the SR inlet, whereas helium gas temperature at the IHX outlet is 905°C. Design specifications of the HTTR steam reforming system is shown in Table 1 and an flow scheme of the system in Fig. 1.

The HTTR steam reforming system is designed to utilize the nuclear heat effectively and to achieve hydrogen productivity competitive to that of a fossil-fired plant with operability, controllability and safety acceptable enough to commercialization [2]. Since the steam reforming conditions in the HTTR are higher pressure and lower temperature as shown in Table 2, lower hydrogen productivity is

Table I Design specifications of the HTTR and out-of-pile test systems in steam reforming hydrogen production

items	HTTR system	Out-of-pile test system
Pressure		
Process gas/Secondary helium	4.5/4.1 Mpa	
Temperature inlet at steam reformer		
Process gas/Secondary helium	450/450 °C	
Temperature outlet at steam reformer		
Process gas/Secondary helium	600/600 °C	
Hydrogen production rate	3800 Nm^3/hr	110 Nm^3/hr
Steam-carbon ratio	3.5	3 to 4
Heat source	Reactor	Electric heater

Table II Comparisonof operational conditions and performance of steam conditions

Performance type	Fossil-fired system	HTTR system
Process gas pressure	1 to 3 Mpa depending upon final products	4.5 Mpa at the inlet of steam reformer > Helium pressure of 4.1 Mpa
Maximun process gas temperature	850 to 900 °C	800 °C
Maximum heat flux to catalyst zone	50 to 80 kw/m^2	40 kw/m^2
Thermal energy utilization of steam reformer	80 to 85 %	78 %
CO_2 emission from heat source for heat souce power	3t-CO_2/hr/10MW	0

predicted compared with a fossil-fired plant. The following improvements are, therefore, introduced to a conceptual design of the SR, as shown in Fig. 2, to increase hydrogen productivity:

(i) Increasing heat input to the reforming process gas using a bayonet-type reforming tube to recover heat of the reformed process gas,

(ii) Enhancing the heat transfer between helium and process gases using a fin (or wire net) - type heat transfer promoter on the reforming tube.

These improvements were attained at a high hydrogen productivity, that is, a thermal energy utilization of 78%, which is competitive to that of a fissile-fired plant of 80-85% as shown in Table 2. The improved technologies here are applicable also to other HTGR hydrogen production systems because a heat exchanger type of endothermic chemical reactor is essential technology.

A steam generator (SG) installed at the downstream of the SR in the secondary helium loop provides stable controllability for any disturbance at the SR due to the large capacity of the heat sink. In a thermal transient state such as helium gas temperatures at the SR outlet and the SG inlet increase due to a malfunction in the reforming process gas line, the helium gas temperature at the SG outlet can be kept constant by the saturation temperature of steam. Figure 3 shows analytical gas temperatures in the secondary helium gas loop for the process gas feed rate of 0 to 100% of the rated. The max. temperature difference of helium gas 280^0C at the SR outlet reduces to only 5^0C at the SG outlet in spite of a change in the process gas feed rate. This result suggests that the SG works as a thermal absorber and protects the reactor system from any thermal disturbance caused by the reforming system. Also, the SG is used to avoid a reactor scram due to a malfunction or accident at the reforming system. If the supply of the feed

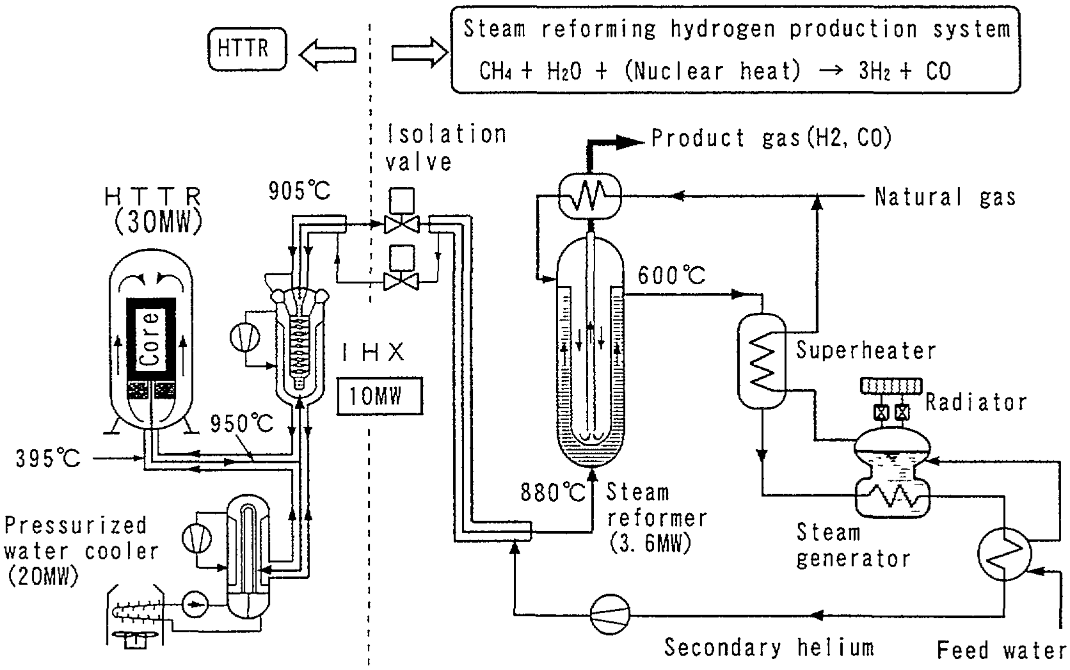

Fig. 1 Flow scheme of the HTTR steam reforming hydrogen production system.

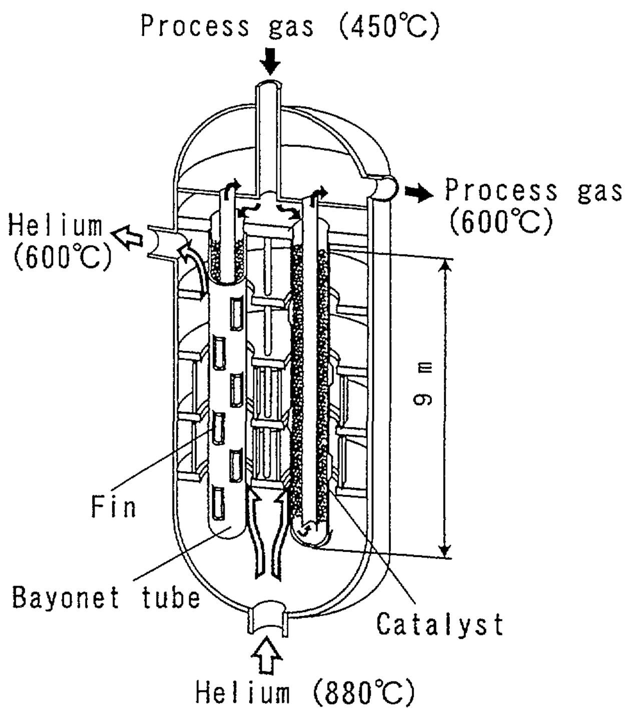

Fig. 2 Schematic view of HTTR steam reformer.

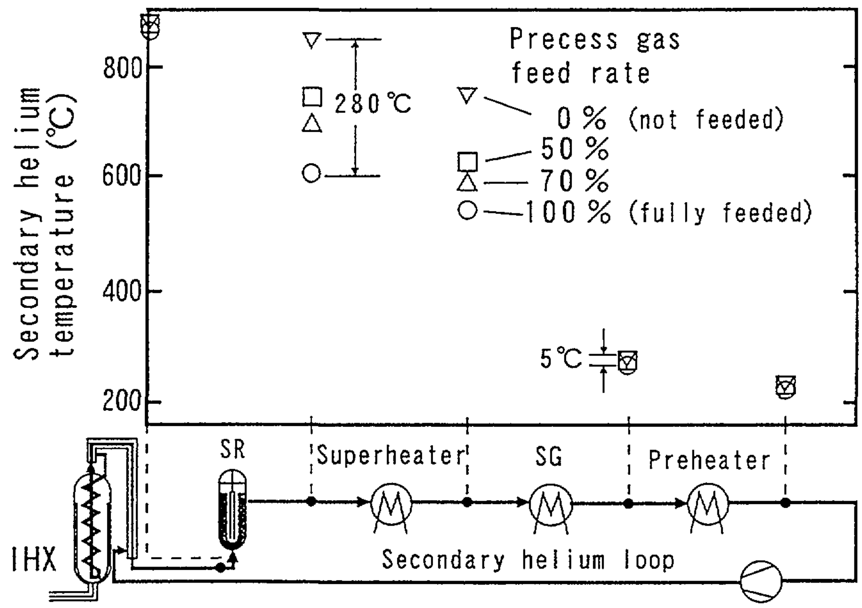

Fig. 3 Change of secondary helium gas temperature at main components for process gas feed rate of 0 to 100%.

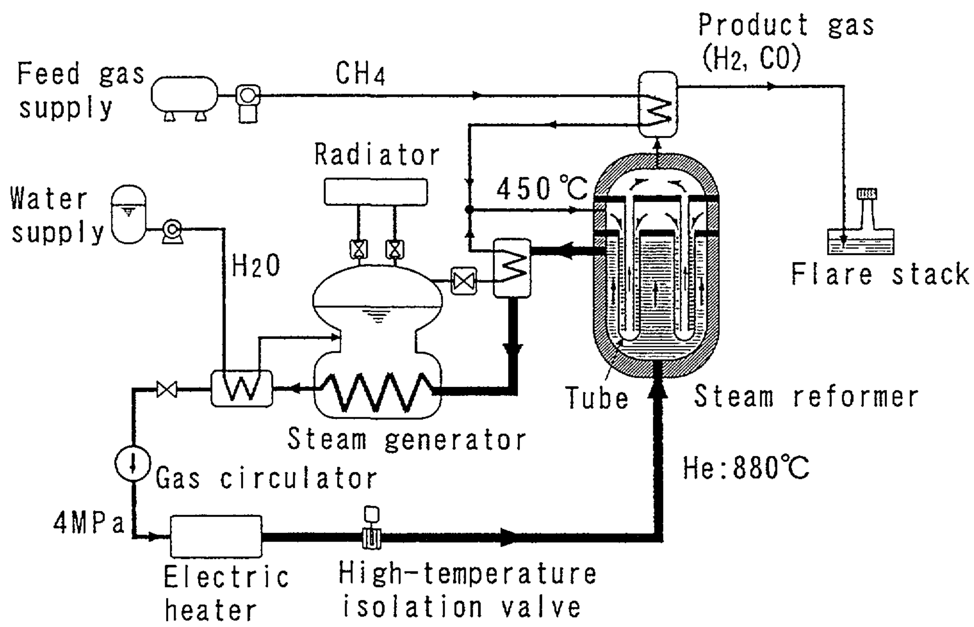

Fig. 4 Flow scheme of out-of-pile test facility.

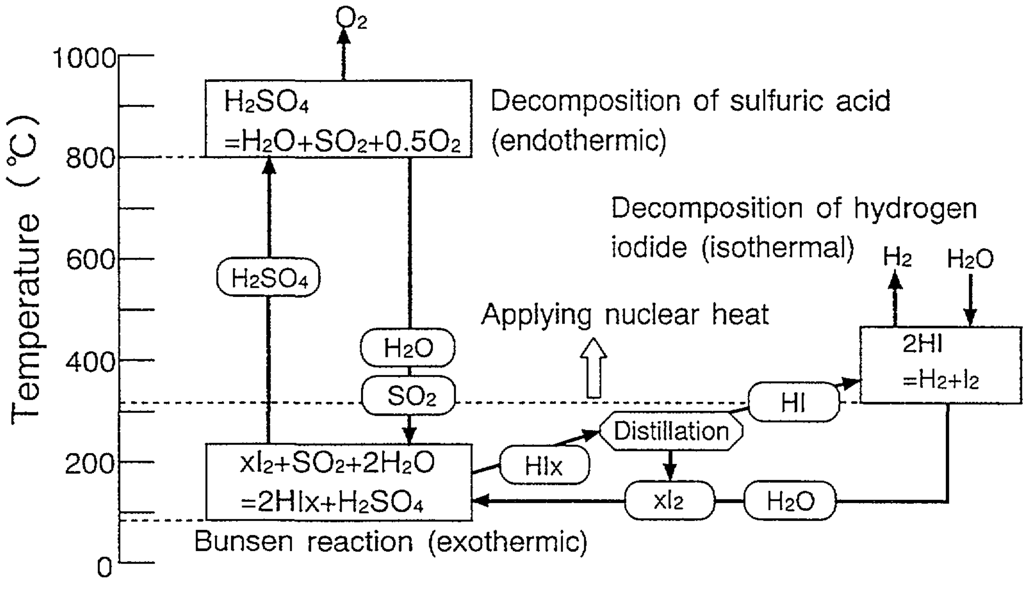

Fig. 5 Reaction scheme of thermochemical IS process.

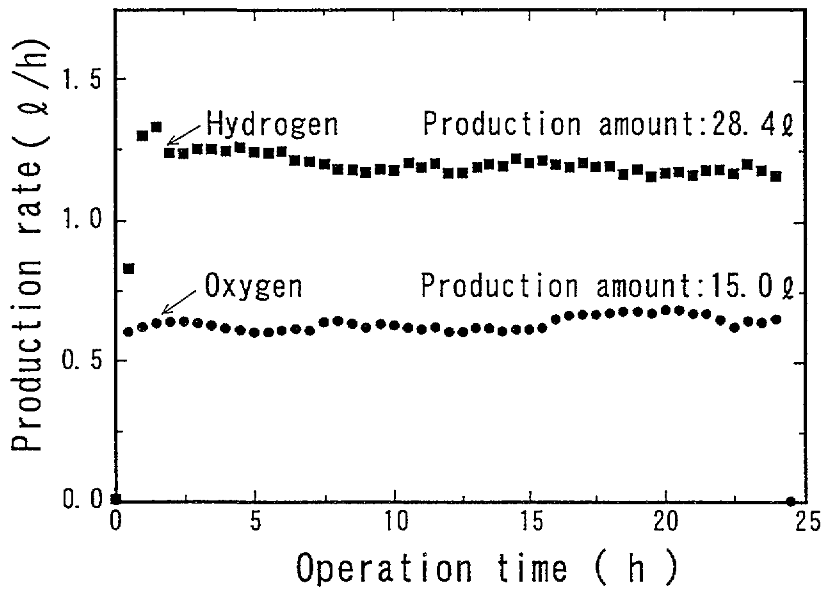

Fig. 6 Results of continuous production of hydrogen and oxygen with laboratory-scale test.

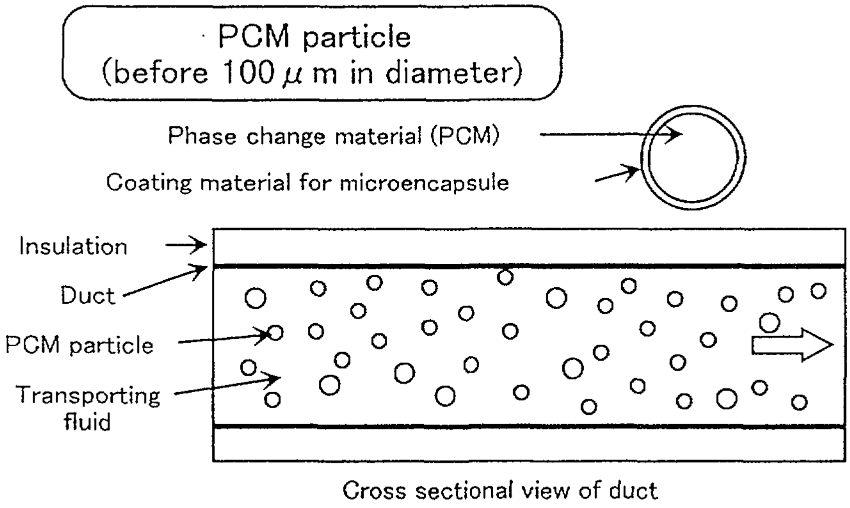

Fig. 7 Concept of slurry heat transport media composed of microencapsulated phase change material.

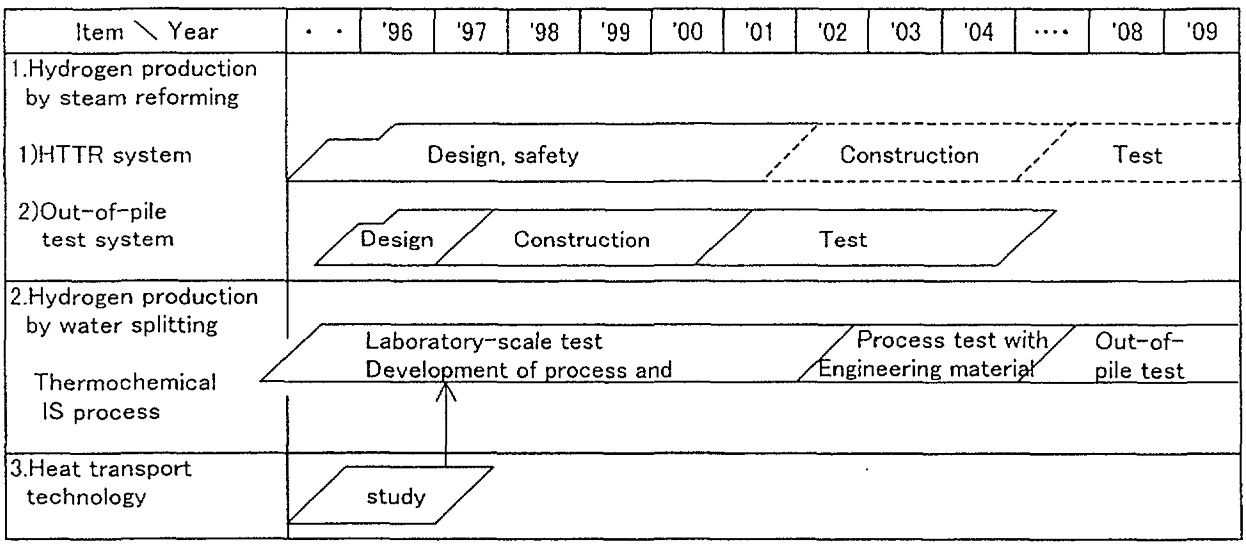

Fig. 8 Development schedule of HTTR utilization system in JAERI.

gas to the SR is stopped, heat of the helium gas can not be removed at the SR because the reforming reaction is curtailed. Therefore, helium gas is passively cooled only by the SG, using the natural convection of steam and condensed water between the SG and a radiator installed above the SG.

Safety barrier (functional or physical) is required to ensure the integrity of the nuclear system against fire/explosion accident caused by combustible gas. Two safety barriers are considered in the HTTR steam reforming system. One is against the accident outside a reactor building (RB), which is a functional barrier of a safety distance to prevent damage of the RB and components related to safety. The other is against the accident inside the RB, which is a physical barrier of the secondary helium gas loop and a combination of isolation valves (IVs) in the secondary helium gas loop and emergency shutoff valves in the process gas feed line to restrict the amount of spreading gas into the RB. In addition to safety consideration at normal operation, tritium transportation is limited as low as reasonably achievable by purification systems of the primary and the secondary helium gas loops.

2.2 Out-of-pile and component tests

Prior to connecting the steam reforming system to the HTTR, the out-of-pile test is required to confirm the safety, controllability and performance of the steam reforming system under simulated operational conditions. The facility of the out-of-pile test has capacity of hydrogen production of 110 Nm_3/h and simulates key components downstream from the IHX [3]. Design specifications of this test facility is also shown in Table 1. The main objectives are as follows:

(i) Design verification of performance of high temperature components, such as the SR, SG and IV,
(ii) Investigation of transient behaviour of steam reforming system,
(iii) Establishment of operation and control technology so as not to give the reactor any significant disturbance at steam reforming system trouble.

Figure 4 shows a schematic flow diagram of the system. An electric heater is used as a heat source instead of nuclear heat to heat helium gas to temperature resulting in 880^{0}C at the SR inlet. The SR has three bayonet-type reforming tubes made of Hastelloy XR, approximately 3500 mm in catalyst layer length, and the heat transfer performance of helium and process gases is investigated to obtain design data for the HTTR. High temperature IVs are installed at high temperature helium gas piping to demonstrate durability and leak tightness. Also, the demonstration of the passive cooling system by the SG is one of the significant objectives of this system. As for passive cooling by the SG, transient behaviour of pressure and temperature of gas and steam, steam production rate and natural convection of steam and condensed water are investigated in detail in passive cooling condition, and the results will be used to improvement of the controllability of the system.

In parallel to the test, the following two tests on key parts are carried out with other small testing apparatuses to obtain detailed data for a safety review and development of analytical codes:

(i) Corrosion and strength reduction of Hastelloy XR reforming tubes in the process gas condition,
(ii) Hydrogen permeation through Hastelloy XR reforming tube walls to evaluate the amount of tritium transported from the HTTR to the steam reforming system, and various kinds of coating at the outer surfaces of the reforming tube will be examined so as to reduce hydrogen permeation for future HTGR heat utilization systems.

The tests results will be used as verification data of design of the HTTR heat utilization system at the safety review and construction.

3. Hydrogen Production by Water Splitting

Hydrogen production using a HTGR offers a quite ambitious concept for future energy systems especially when hydrogen is produced from water alone without fossil fuel. JAERI has been conducting basic studies on a hydrogen production process by water splitting, thermochemical IS process, as one the future heat utilization systems of the HTTR following the steam reforming system. The IS process uses plural chemical reactions and works like a chemical engine to produce hydrogen by absorbing high temperature heat as shown in Fig. 5. The sulphuric acid (H_2SO_4) decomposition reaction requires high temperature heat supplied from the HTGR because it is endothermic. The IS process was first proposed by the General Atomic Co. [4] and has been studied at several research institutions. The thermal efficiency of the process based on the Higher Heating Value (HHV) of hydrogen is theoretically estimated to be 47 to 50% from existing thermodynamic data [5].

Continuous and stoichiometric production of hydrogen and oxygen for 24 hour was successfully achieved with a laboratory-scale apparatus made of glass as shown in Fig. 6. However, the thermal efficiency obtained was much less than the theoretical one. Further studies are needed to improve thermal efficiency. One is a rise of the reaction temperature at a Bunsen reaction step, and the other is highly effective separation of hydrogen and iodine at a HI decomposition step. As for the latter, a hydrogen permselective membrane is under study to improve the separation efficiency. In addition to the above issues, the development of corrosion resistance materials is required for realization of the hydrogen production system. Especially as for H_2SO_4 boiling environment, any material do not exist for commercial material. Material study has been therefore conducted to develop a new material based on an iron-silicon alloy and chemical reactor vessels featuring the high corrosion resistance of silicon based ceramics.

4. Heat Transport Technology

A basic study on new heat transport media with high heat capacity has been carried out to improve transport efficiency of thermal energy from a remote nuclear plant to consumers by means of large latent heat of phase change material (PCM). The PCM is stabilized in shape by means of microencapsulation into tiny particles with diameter of less 100 ƒÊm, coated by a durable and heat-resistant material. The particles are mixed into a slurry with a carrier liquid with low vapour pressure and low viscosity. Figure 7 shows this microencapsulated PCM slurry concept.

For low temperature nuclear heat utilization below about 100^0C, such as district heating and cooling applications, a series of water slurries have been developed using organic PCMs and high polymer coatings. Small size particles have good characteristics to stable particles floating in slurry and their mechanical integrity. But the difference between solidification and melting points, so-called undercooling, increases significantly with decreasing the diameter of particles. This is because smaller particles seem to have few nucleation sites in solidification. Undercooling reduces the utilization temperature. Then undercooling is avoided by a specially selected additive to introduce nucleation sites into each particle [6]. Using the slurry developed, heat transport experiments were carried out for natural convection around a horizontal heated cylinder and for forced convection in a pipe. By an addition of 5 % PCM particles with an average diameters of 8 m, heat transfer coefficient increases by about 35% at relatively small film temperature difference conditions. This tendency is successfully explained by the analysis based on an simple assumption of temperature dependency of specific heat of working fluid. Further heat transport experiments will be to reveal the general characteristics of this newly developed heat transport media in the near future. For higher temperature applications of the HTGR heat utilization, development studies have been started using molten-salts, such as lithium chloride, as PCM, prepared into porous spheres and inorganic coatings such as carbon and SiC, just like coated fuel particles for HTGR.

5. Development Schedule

Attainment of first criticality of the HTTR, being constructed on the JAERI site, is scheduled for December 1997. After the reactor performance and safety demonstration tests will be carried out for several years, the first hydrogen production system by steam reforming of natural gas will be connected to the HTTR. Figure 8 shows the development schedule of the HTGR utilization system in JAERI. The demonstration program on hydrogen production just started on January in 1997, in JAERI, as a study consigned by Science and Technology Agency. The out-of-pile test will be performed for four years after 2001, and the results will reflect on the design and safety review of the HTTR steam reforming system. The HTTR steam reforming system will be constructed from 2002, and the system will be demonstratively operated from 2005 to 2009 at its earliest plan.

On the other hand, the thermochemical IS process is planned as one of the future candidates of the HTTR utilization system following the steam reforming system. Development of the process and material will be carried out by 2002. Continuously a process test using commercially available metal and ceramic materials will be performed during several years. After that, an out-of-pile test is planed to be performed after 2007. Developed heat transport technology will be applied to improve thermal efficiency of the IS process which needs high heat recover and transport.

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NUCLEAR REACTOR DEVELOPMENT IN CHINA FOR NON-ELECTRICAL APPLICATIONS

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Abstract

In parallel to its vigorous program of nuclear power generation, China has attached great importance to the development of nuclear reactors for non-electrical applications. The Institute of Nuclear Energy Technology (INET) in Beijing has been developing technologies of the water-cooled heating reactor and the modular high temperature gas-cooled reactor. In 1989, a 5 MW water cooled test reactor was erected. Currently, an industrial demonstration nuclear heating plant is being projected. Feasibility studies are being made of sea-water desalination using the INET developed nuclear heating reactor as heat source. Also, a 10 MW high temperature gas-cooled test reactor is being constructed at INET in the framework of China's national high-tech program. The paper gives an overview of China's energy market situation. With respect to China's technology development of high temperature gas-cooled reactors and water cooled heating reactors, the paper describes some general requirements on the technical development, reviews the national programs and activities, describes briefly the design and safety features of the reactor concepts, discusses aspects of application potentials.

1. Introduction

China's national economy and energy consumption have experienced rapid growth in the last years. From 1991 to 1995 which is the National Eighth Five-Year Plan period, the average economical growth rate reached 11.8%. In 1994, consumed primary energy was 1227 Million Tons Coal Equivalent (Mtce). Average growth rate of primary energy consumption over 1991-1994 was 5.6%. It is expected that the national economy and energy consumption will continue to grow. China implements the policy of self-reliance for energy production, so that meeting the energy consumption demand by capacity installation and efficiency enhancement remains a challenging task.

Of the 1227 Mtce primary energy consumed in 1994, 31.5% is consumed in the form of electricity, the rest part is consumed for non-electrical applications. To the total of energy consumption in 1994, 75.0% is contributed by coal, 17.4% by oil, 1.9% by natural gas, 5.3% by hydro-power and 0.4% by nuclear power. The energy consumption and production in China is dominated by coal. This fact brings two problems with it: transportation burden and environment pollution. Both are serious problems in China's rapid social and economical development.

Nuclear energy has been recognized in China as a clean, safe and economical energy source. There have been vigorous programs for nuclear power development. Currently, there are three commercial nuclear power reactors in operation with a capacity of 2.1 GW. Eight units of a total capacity of 6.4 GW are under construction or have been planned for construction in the near future. Besides the development of nuclear power generation, great importance is also attached to the development of nuclear energy technology for non-electrical applications. The vessel type water-cooled reactor with low core outlet temperatures (Nuclear Heating Reactors - NHR) has been developed for district heating or other lower temperature applications. Technology development of High Temperature Gas-cooled Reactors (HTGR) is being conducted both for electrical and non-electrical applications. The development work, technical and safety features as well as application potentials of these two reactor types will be presented in the following sections.

2. General requirements on developing heat supplying reactors

Heat is not suitable for being transported over long distances. Nuclear reactors for heat supply purposes should be built near industrial or densely populated areas. Therefore, these reactors should have an extremely high level of safety and should be economically competent over fossil fuels. Following design philosophies are applied to the development of NHR and HTGR in China.

Safety. In the development of heat supplying reactors, nuclear safety is to be improved through the following approaches:

- To improve the reactor's inherent safety. Through appropriate design and proper selection of material's type, quantity and properties, the reactor can be endowed with certain capability of self-protection or will be free of certain potential adversity
- To adopt passive system design. Instead of depending on the operator's action or the effect of external machinery and power, the safety system employs certain natural phenomena such as natural circulation, heat conduction, radiation, internal energy preservation and others to realize its function and thereby to ensure its reliability and effectiveness
- To ensure negative temperature coefficient of reactivity under all conditions by proper nuclear design and to ensure enough negative reactivity to be introduced for reactor shutdown
- To ensure safe removal of decay heat by proper reactor and system design
- To lessen the dependence on the operator's response capacity
- To iron out the influence of internal and external events on the reactor safety
- To further decrease the radiation dose of the operating personnel and the public

Operational reliability. One of the important goals of developing heat supplying reactors is to enhance their operational reliability so as to increase its availability and load factor. In order to accomplish this, the following measures should be taken:

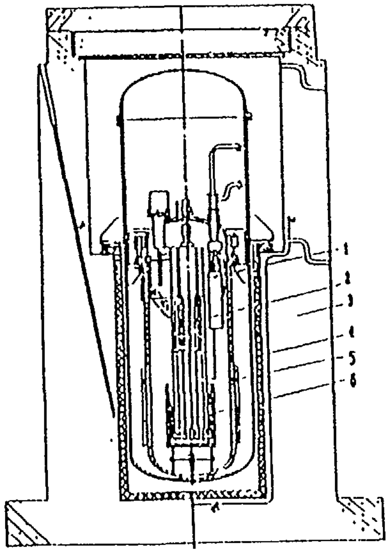
- Simple design
- Accessibility for maintenance and inspection
- Use of proven technology
- Adoption of standard design

Economy. To have a better economy, nuclear reactors for heat supply are required to have the following features achieved by design, construction and operation:

- Longer operational life-span
- Shorter construction period
- Lower operational and maintenance costs
- Lower decommissioning costs
- Smaller investment risk

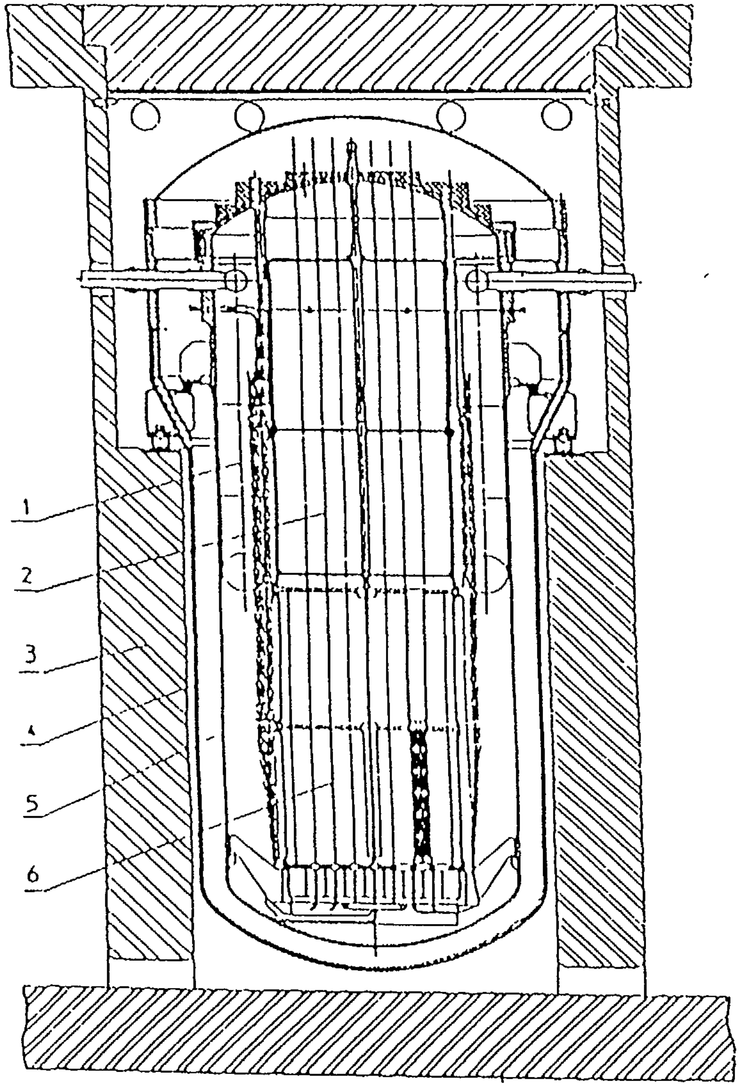
3. Development of NHR in China

The Institute of Nuclear Energy Technology (INET) of Tsinghua University in Beijing has been developing water cooled nuclear heating reactors since the 1980's. In 1983 and 1984, INET conducted successful tests of nuclear district heating using the existing swimming pool type research reactor. In 1984, INET began the project of erecting a 5 MW test NHR on the site of the institute which is about 40 km away to the north of Beijing city. The construction of the 5 MW NHR started in 1986 and was finished in 1989. In November 1989 the test reactor went critical. Since then, the reactor has been successfully operated. For research and development to make more use of NHR, experiments have been conducted on the 5 MW test reactor such as co-generation of electricity and heat, air conditioning and desalination with nuclear energy.



5MW test heating reactor

1. Primary heat exchanger
2. Riser
3. Biological shield
4. Containment
5. Pressure vessel
6. Reactor core



200MW demonstration heating reactor

Fig. 1. The 5 MW test NHR and the 200 MW demonstration NHR.

Table I Key design parameters of the nuclear heating reactors

Reactor		NHR-5	NHR-200
Thermal Power	MW	5	200
Primary pressure	MPa	1.5	2.5
Core inlet / outlet temperature	°C	146 / 186	140 / 210
Height of the active core	m	0.69	1.90
First core UO_2 loading	t	0.508	14.5
Enrichment of the first core loading	%	3.0	1.8/2.4/3.0
Enrichment of the reload fuel	%	3	3
Intermediate circuit pressure	MPa	1.7	3.0
Intermediate circuit temperature	°C	102 / 142	95 / 145
Heating grid temperature	°C	90/60	130/80

Table II Key design parameters of the HTR-10 test reactor

Reactor thermal power	MW	10
Primary helium pressure	MPa	3.0
Reactor core diameter	cm	180
Average core height	cm	197
Average helium temperature at reactor outlet	°C	700
Average helium temperature at reactor inlet	°C	250
Helium mass flow rate at full power	kg/s	4.3
Main steam pressure at steam generator outlet	MPa	4.0
Main steam temperature at steam generator outlet	°C	440
Feed water temperature	°C	104
Main steam flow rate	kg/s	3.47
Number of control rods in side reflector		10
Number of absorber ball units in side reflector		7
Nuclear fuel		UO_2
Heavy metal loading per fuel element	g	5
Enrichment of fresh fuel element	%	17
Number of fuel elements in reactor core		27,000
Fuel loading mode		multi-pass

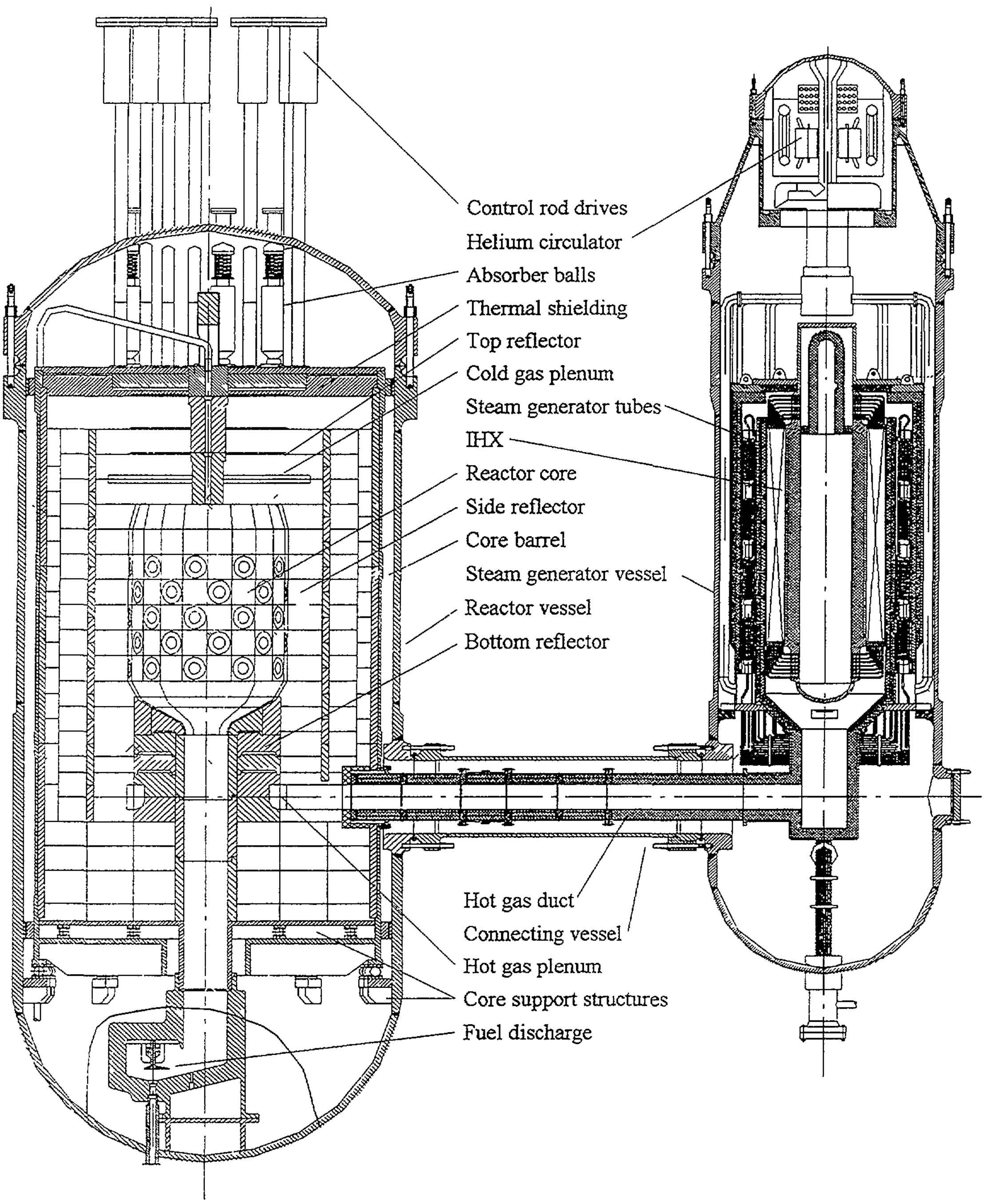


Fig. 2. The HTR-10 reactor and steam generator.

On the basis of the successful 5 MW test reactor, several cities and large enterprises have showed their strong interests in building NHRs. INET is now projecting a 200 MW commercial demonstration nuclear heating reactor which will be built in the northeast part of China.

Technical and safety features of the NHR. Both the 5 MW and the 200 MW heating reactor are of water cooling, vessel type design. Their main technical and safety features are briefly summarized as follows:

- Integrated design. Both the reactor system and the primary heat exchangers are integrated into the pressure vessel. This compact integrated design minimizes the probability of large LOCA accidents.
- Full power natural circulation cooling. At all power levels, the reactor power is designed to be carried out by means of natural circulation, eliminating circulating pumps and ensuring higher system reliability. This is also true for the reactor decay heat removal.
- Duel vessel design. The steel containment vessel is designed closely surrounding the pressure vessel. In case of a very unlikely failure of the pressure vessel, the containment vessel will ensure the immersion of the reactor core without any emergency cooling actions.
- Hydraulic driving mechanism of the control rods. A new driving mechanism of the control rods by hydraulic means has been developed and utilized. This design simplifies the reactor structure design and eliminates the accident of rapid rod ejection.
- Primary pressure self-regulation. With the help of a certain inventory of nitrogen in the primary loop, the primary pressure regulates itself very stable at the designed level.
- Low parameters. The design parameters are chosen which are suitable for district heating purposes and they are much lower than those of large electricity generating reactor. This brings more safety advantages and makes the reactor operation simpler and easier.

Figure 1 shows the overall design of the 5 MW test NHR and the 200 MW demonstration NHR, while the key design parameters of these two reactors are listed in Table I.

4. Development of HTGR in China

Under the support of the Chinese government, INET of Tsinghua University started in the 1970's to develop HTGR technology. A lot of R&D work was performed in the fields of reactor design methodology, system and component technology, coated particle technique, helium technology, graphite materials, treatment of irradiated fuel, etc.

Since 1980's, R&D work in China has been concentrated on the modular reactor concept and process heat applications of HTGR. In 1986, R&D of HTGR was integrated in the National High-Technology Program (NHTP). The future roles of HTGR and the development work were defined. It is projected in the NHTP to establish before the year 2000 a 10 MWth test module reactor (HTR-10) at the site of INET. The HTR-10 is now being constructed. It is expected that commercial HTGR demonstration plants will be projected if the HTR-10 proves to be a success.

Following objectives are expected to be achieved through the establishment of the 10 MW test module reactor:

- Acquiring the key technologies of the modular HTGR with respect to design, construction and operation
- Demonstration of the safety features of modular HTGR
- Demonstration of electricity/heat co-generation
- Demonstration of power generation with gas/steam combined cycle
- R&D of high temperature process heat application, e.g., coal gasification and liquefaction as well as steam reforming
- Providing an irradiation facility for fuel element and material experiments.

Technical and safety features of the HTR-10. Design of the HTR-10 test reactor (see Figure 2) represents the features of modular HTGR design. Reactor core and steam generator are housed in two steel pressure vessels which are arranged in a "side-by-side" way. The two vessels are connected to each other by a connecting vessel in which the hot gas duct is designed. All these steel pressure vessels are in touch with the cold helium of about 250°C coming out from the circulator which sits over the steam generator tubes in the same vessel.

Fuel elements used are the German type spherical fuel elements (6 cm in diameter) with coated particles. The reactor core contains about 27,000 fuel elements forming a pebble bed which is 180 cm in diameter and 197 cm in average height. Spherical fuel elements go through the reactor core in a "multi-pass" pattern. Graphite serves as the main material of core structures which mainly consist of the top, bottom and side reflectors. The ceramic core structures are housed in a metallic core vessel which is supported on the steel pressure vessel. Side reflector is 100 cm thick. In the side reflector, cold helium channels are designed in which helium flows upward after entering the reactor from between the connecting vessel and the hot gas duct. Helium flow reverses at the top of reactor core into the pebble bed, so that a downward flow pattern takes place in it. After being heated in the pebble bed, helium enters into a hot gas chamber in the bottom reflector, and from there it flows with reactor outlet temperature through hot gas duct to the heat exchanging components.

The steam generator is composed of a number of modular helical tubes which are arranged in a circle between two insulation barrels inside the steam generator pressure vessel. The place inside the inner barrel is foreseen for an intermediate heat exchanger which is to be installed in the second phase of the project.

On the wall of the concrete housing, a surface cooling system is designed. This system works on the principle of natural circulation of water and it takes the decay heat via air coolers to the atmosphere. In fact, this surface cooling system is designed to protect the vessel and concrete structures more than the ceramic reactor core from being overheated by decay power.

There are two reactor shutdown systems, one control rod system and one small absorber ball system. They are all designed in the side reflector. Both systems are able to bring the reactor to cold shutdown conditions. Since the reactor has strong negative temperature coefficients and decay heat removal does not require any circulation of the helium coolant, the turn-off of the helium circulator can also shut down the reactor from power operating conditions.

With respect to reactor safety, the 10 MW test reactor is characterized by the following features:

- The overall temperature coefficient of reactivity of the reactor core is always negative. All reactivity transients during power operation can be compensated by the negative temperature coefficient which leads to the self-shutdown of the reactor.
- The decay heat removal requires no circulating coolant systems. It can disperse to the outside of the reactor pressure vessel through heat conduction and radiation within the reactor internals. The dispersed residual heat will be taken away through a surface cooling system on the wall of the concrete cavity which houses the reactor.
- Because of the good capability of coated fuel particles to retain fission products, the primary helium is of low level radioactivity and is allowed to be released unfiltered to the atmosphere at system depressurization, without causing irradiation effect beyond specification. Therefore, a vented confinement instead of a gastight containment is designed for an organized and controlled release of gases in the facility, which serves as the last independent radioactivity barrier following the defense-in-depth principle.

5. Summary and perspective

In China, great importance has been attached to the development of advanced reactor types such as NHR and HTGR. These advanced reactor types are of advantageous safety features and have application potentials in heat supply.

The NHR can be used for district heating and air conditioning in densely populated areas. This can save a great amount of coal consumption and thus ease the associated problems with coal burning. It can also be used to provide low temperature process steam for industrial applications like seawater desalination. With respect to the serious situations of fresh water supply in many coastal areas in the world, there is a good prospect for the NHR to be used for this application.

The HTGR can be used to provide process heat at various temperature levels up to 950°C. With HTGR, process steam can be produced for example for heavy oil recovery and crude oil refinement process. HTGR is the only reactor type which can provide high temperature process heat used for coal liquefaction or gasification. Considering the problems with coal burning and the shortage of liquid fuel supply, the application of HTGR to convert coal into liquid form energy carriers is of special interest in the future energy market.

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UTILIZATION OF HTGR HEAT AND ITS TRANSFER TO INDUSTRIAL FACILITIES

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Abstract

Heat utilization of a modular helium cooled reactor with the temperature of 750-900°C, which can not be attained by other reactor types, and with high level of safety is considered. Requirements of the heat power value and maneuverability of the industrial processes, to which a nuclear reactor is to be integrated, are discussed on an example of a standard oil refinery plant. Heat removal systems from the reactor block to industrial processes are analyzed from a safety, economy and maneuverability point of view. Selection and requirements for the intermediate coolant are also discussed.

1. INTRODUCTION

Oil refinery industry is one of the most energy-consuming sectors in the national economy, spending up to 20% of extracted oil [1]. This fact itself confirms the grounds why using nuclear energy in industry processes is economically expedient in addition to advantages in ecology, transport expenditures etc.

Especially it relates to reactor plants (RP) with high temperature gas cooled reactors (HTGR) having an outlet helium temperature of up to 950°C. In OKB Mechanical Engineering the design of a pilot-industrial modular reactor plant VGM has been developed [2]. Based on this concept industrial RPs are being designed for supplying industrial processes with heat in a wide temperature range (up to 900°C) which can not be provided by other reactor types.

Nuclear power process stations (NPPS) with an HTGR can provide all existing and perspective technologies with heat of required temperature level:

- high temperature processes (up to 950°) for the production of petroleum products and diesel fuel from coal, for the production of hydrogen, ammonia and mineral fertilizers by methane steam conversion, etc.;
- middle temperature processes (up to 600 °C) for mainly the secondary reprocessing of the oil products - reforming , cracking, etc.; and
- low temperature processes (up to 400 °C) for mainly the initial reprocessing of the oil products - hydrocracking, hydrocleaning and centralized district heating, etc.

2. MAIN REQUIREMENTS FOR A NUCLEAR POWER PROCESS STATION SPECIFIED BY THE OIL REFINERY PLANT

Most processes in all the temperature ranges mentioned above are existing at the oil refinery plants (ORPs). In order to specify requirements for the ORP and the heat consumption depending on process temperatures (Table I, Fig. 1), a standard ORP with an output of 12 x 10^6 tons of refining oil per year was used as an example.

The Q-T diagram (Fig.1) shows that, out of all thermal power spent for oil refining (directly in processes) ~240 MW(t) is consumed at temperatures of up to 300 °C, ~500 MW(t) at temperatures of up to 430 °C, and ~125 MW(t) at the temperature range from 430 °C to 830 °C.

Table I Requirements for the ORP and the heat consumption

Parameter	Value
Total power consumption, MW(t), including:	1555
Power direct consumption for processing, MW(t)	865
Process steam generation, MW(t)	475
Electric power consumption, MW(t),	215
(with efficiency~ 30%), MW(e)	64.5
Main operating regime	Operation at the nominal power during at least 8000 hours per year Complete scheduled outage of production is not planned
Minimal power at reduction of load, % of nominal power	~25
Power change rate within the control range, %/ min.	1

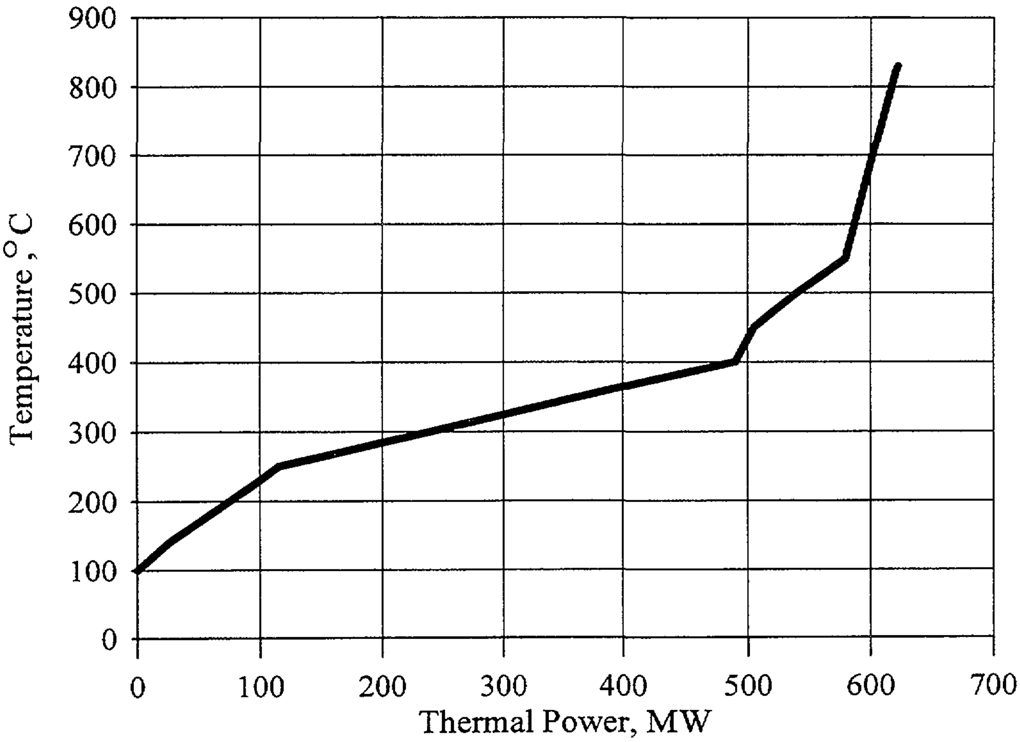


Fig. 1 Typical Oil Refinery Q-T Diagram

A modular HTGR with a pebble bed core that can be refuelled on power is selected as a nuclear heat source for the NPPS. The reactor of this type can provide a high level characteristics of heat supply: high load factor and high availability factor (up to 85 and 90 %, respectively). The reactor has a high level of safety due to inherent safety properties and passive principles if its power is limited to 200-250 MW [3].

The NPPS with such a reactor is possible to be located closer to the ORP and settlements. The power required for the processes can be obtained by an appropriate number of reactor units. Usage of multiple reactor units is better to meet requirements imposed by the oil refinery processes as noted below.

The first requirement is related to the power supply reliability during 8000 hours in consideration of refueling, planned and emergency shutdowns. No shutdowns are required for refueling of the modular reactor with a pebble bed core. At shutdowns and maintenance of one of the reactors, other reactors at the NPPS can supply minimum heat needed.

The second requirement is related to maneuverability over the all range of partial loads. At partial loads the operating temperature level can be maintained steady by shutdown of one or some selected reactor blocks, without changing the power level of other operatng units, than by changing the power level of a single block large NPP. This is possible when the required heat exchange area at partial loads in the intermediate circuit is less than the existing designed one, since multicircuit heat transfer lines connect the reactor and the consumer.

3. SYSTEMS OF HEAT REMOVAL

Capital investments and operating costs of NPPS depend essentially on the number and the length of heat exchange circuits. The number of heat exchange circuits is determined by the following requirements:

- to exclude the possibility of the final product contamination above the allowable levels according to codes and standards; and
- to exclude ingress of the intermediate coolants containing hydrogen or unremovable active and explosive substances to the primary circuit at the loss of integrity of heat exchangers.

Specific activity of a primary coolant in an HTGR is insignificant (~ 2 x 10^3 Bq/cm^3) and depends mainly on gaseous radionuclides (Xe^{133}, Xe^{135}, Kr^{88}). At an emergency outflow of helium from the primary circuit into the central hall of the reactor building the concentration of radionuclides in the ambient air will not exceed the requested level for working rooms. Long-term activity is determined by tritium (H^3) which is practically completely removed by the purification system. Anticipated tritium contamination of the intermediate circuit coolants will not exceed even the requested level for drinking water. Activation of the intermediate coolant by the neutron from the reactor could be reduced to the required values by design measures, such as, usage of step gas ducts limiting direct neutron flux; location of heat exchangers in a shielded concrete compartment.

The intermediate coolant ingress into the primary circuit at the loss of integrity of heat exchangers could be also excluded by design measures, for example, by decreasing the pressure of the intermediate coolant in comparison with the primary one and by releasing the intermediate coolant to the discharge tanks. Discharge tanks are located behind the protection membranes (rupture disks) beneath the gas ducts.

The length of heat exchange circuits is also determined by the following conditions:

- safety of the reactor plant;
- safety of the process (explosion, ejection of toxic substances, etc.); and
- disposition of heat consumers over a plant area.

As noted above according to the radioactivity safety requirements the NPPS with the VGM can be located not far from an ORP. The distance between the NPPS and the ORP will be determined by anticipated impacts on the NPPS by the accidents at the ORP. In addition, considering specific location of the consumer on the ORP area, the length of the heat transfer circuit to the consumer may be much longer than those of the primary and intermediate circuits.

4. HEAT TRANSFER MEDIA

The criteria of heat-transfer coolants selection were:

1) No phase transformation and thermal stability in the whole working temperature ranges including emergency modes;
2) High heat transfer coefficients (minimum heat exchanger areas);
3) Minimum power consumption for coolant circulation;
4) Reliability, simplicity and minimum costs for manufacturing and operation of the circuits; and
5) Safety in emergency modes.

The advantage of using the same coolants in circuits was also taken into account, since it reduces the number of auxiliary systems of heat transfer circuits. For the process circuits with temperatures of up to 550 °C, it is beneficial to utilize accumulated experience in oil refining and petrochemical plants.

With these considerations, following coolants were evaluated:

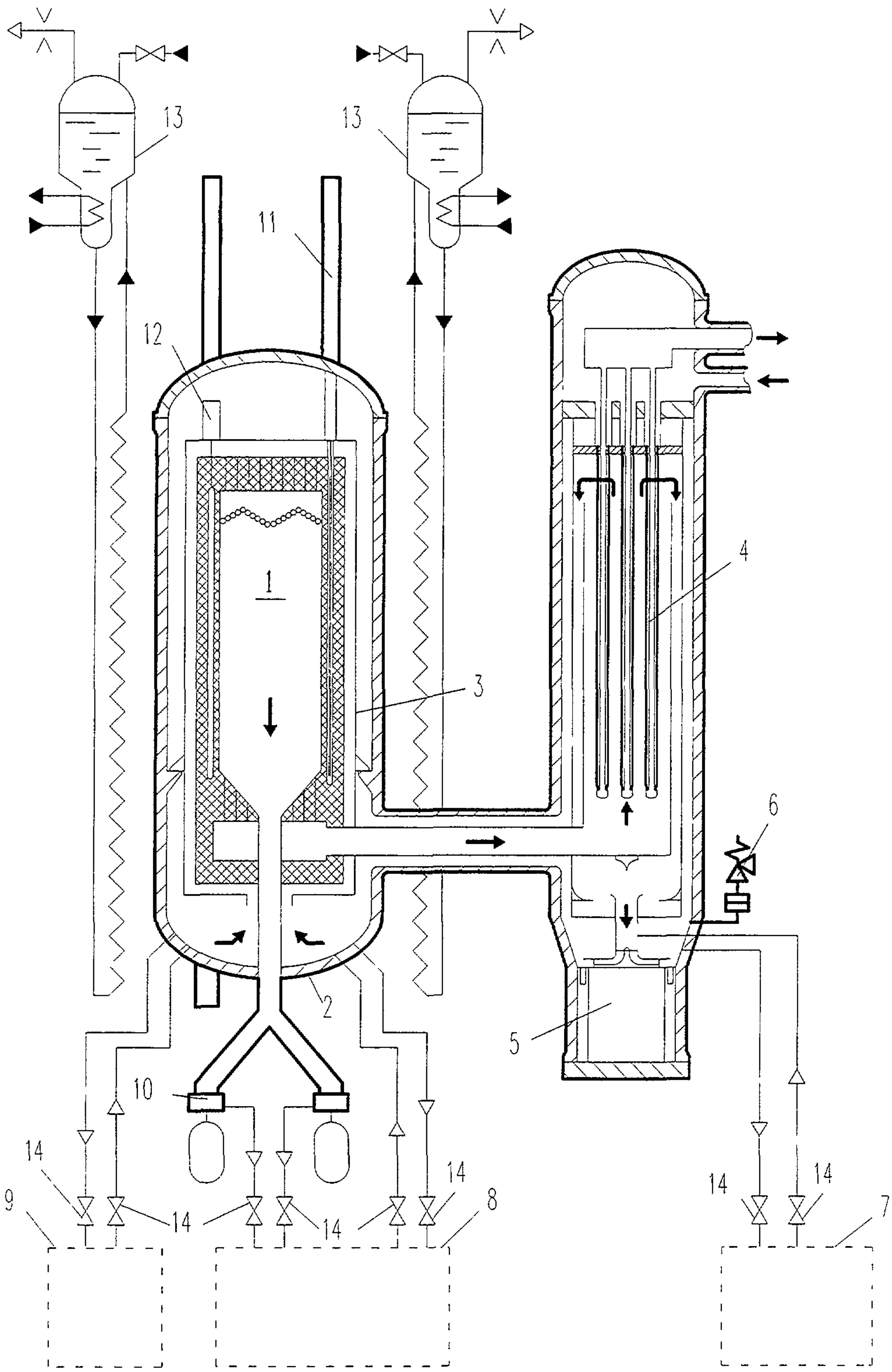
- helium, liquid metals (Na, Na-K, Pb-Bi) for the intermediate circuit with temperatures of up to 900°C;
- liquid metals (Na, Na-K) for the process circuit with temperatures from 550°C to 900°C;
- molten nitrate - nitrite salt on silicon oil for the process circuit with temperatures of up to 550°C.

A helium coolant used in the primary circuit of HTGRs, besides known advantages, has the disadvantages related mainly to the significant consumption of energy for circulation and the increased pressure and thickness of the pressure vessel and the gas ducts. Sodium and sodium-potassium coolants practically meet all requirements. However, they require complicated technologies for application, are rather expensive and have a fire potential at the coolant leak. In addition, the sodium coolant requires its pre-electric heating to its melting temperature (~ 97°C) before putting into service. The Pb-Bi alloy coolant is fire potential free, but has a number of essential shortages: significant consumption of energy for circulation, heavy weight, erosion and vibrating actions on structures, high cost, the necessity of pre-heating. The use of salt coolants in the intermediate circuit limits the temperature potential of transferring heat up to their work temperature of application ~550°C, although the temperature in HTGRs reaches 950°C.

The application of nitrate - nitrite salt as one of the most high temperature coolants has been proven by technological manufacturers [4]. However, its application is limited to the maximum temperature of 550°C, and above 450°C it begins to decompose resulting in the increased melting temperature and its replacement needed. It also requires pre-electric heating to the melting temperature of ~142°C. A silicon oil coolant has the best stability at the temperatures of 500°C -550°C and does not require pre-heating, as compared with salt coolants, as experienced in laboratory tests.

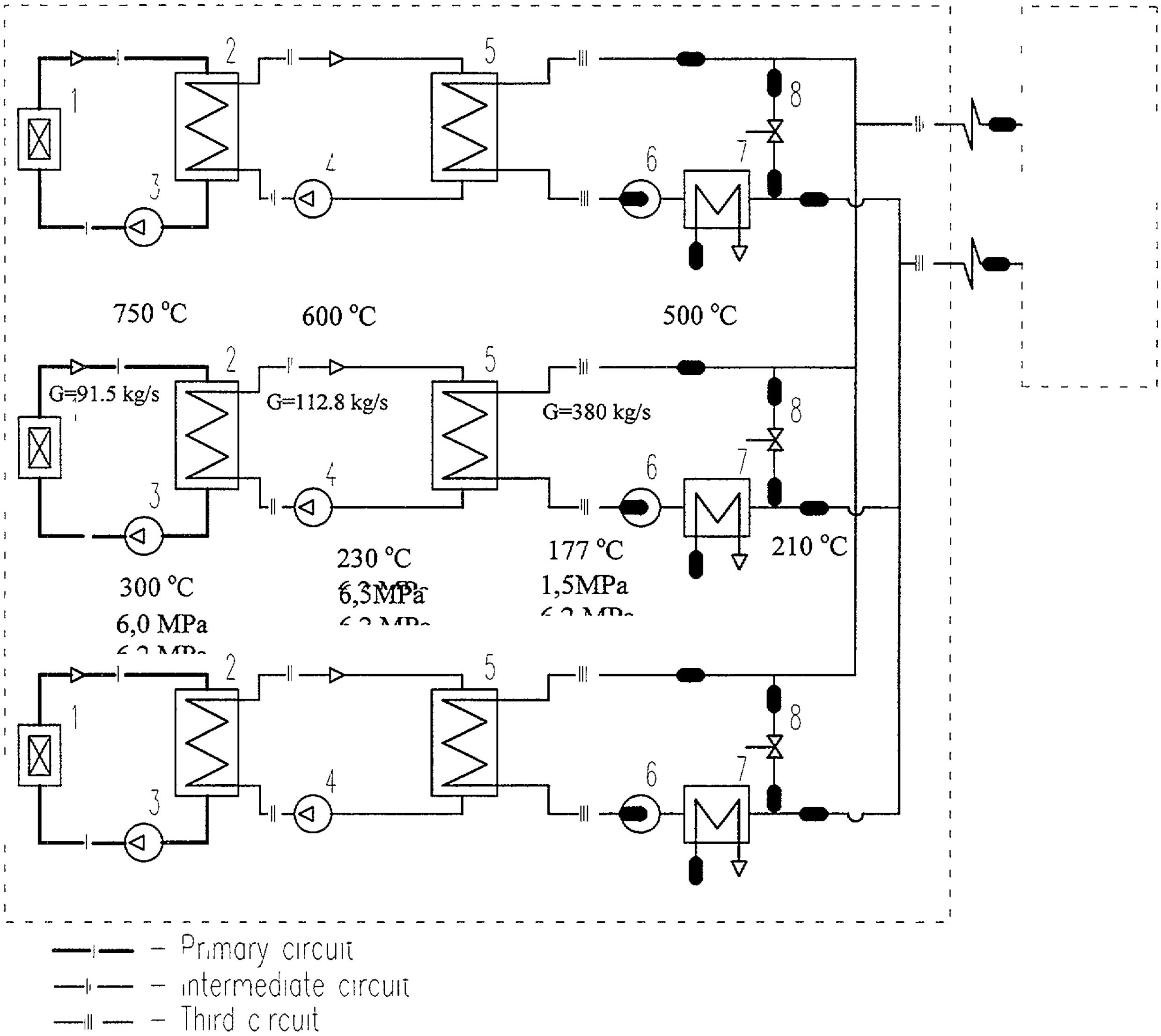
5. THE NUCLEAR POWER PROCESS STATION FOR A STANDARD OIL REFINERY PLANT WITH A MODULAR HTGR

In consideration of a significant fraction (~80 %) of heat need of up to 430 °C, and also in consideration of equipment developed for HTGRs and its readiness of manufacturing, a design of the NPPS for a standard ORP is carried out in OKBM based on the pilot industrial project of the VGM modular reactor. The NPPS consists of three VGM-P reactors of 215 MW power and the outlet helium temperature of 750 °C (Fig. 2). Three circuits for heat transfer to process furnaces (Fig.3) are adopted in the design. The number of the circuits are selected from a point of view of excluding a possibility of ingress of the process coolant into the reactor. Helium is recognized to be expedient as a coolant for the intermediate circuit and nitrate - nitrite salt coolant - for the process circuit. The intermediate circuits are designed to be independent for each reactor module to exclude common mode failures of the NPPS. The process circuit is common for all the ORP. It allows to ensure stable heat supply to the ORP in case of



1 - core; 2 - vessel system; 3 - leak tight core shell; 4 - intermediate heat exchanger;
5 - primary circuit circulator with cut-off valve; 6 - pressurization protection device;
7 - primary coolant purification system; 8 - refueling complex; 9 - small absorber balls system;
10 - fuel element unloading mechanism; 11 - control rod drive; 12 - small absorber balls system drive;
13 - emergency cooldown system; 14 - localizing valves.

Fig. 2 **VGM-P reactor sheet**



1 - reactor; 2 - intermediate heat exchanger; 3 - main gas blower; 4 - intermediate circuit gas blower; 5 - process heat exchanger; 6- technologic circulator; 7 - steam generator; 8 - bypass valve

Fig. 3 **NPPS flow scheme**

planned or emergency shutdowns of one of the reactors. The decrease of VGM-P outlet helium temperature down to 750 °C, in comparison with the base concept of VGM with the outlet helium temperature of 950 °C, allows to prolong the life time of the NPPS to not less than 40 years.

The condition for introducing HTGRs into the industry is its economic competitiveness in comparison with other energy sources. The major economic data of the NPPS with three VGM-P are presented in Table II. (The main economic indexes were taken in 1991 prices and at present are essentially different. Therefore the characteristics in the table should be considered as relative values).

Table II Major economic data of the NPPS with three VGM-P

Characteristic	Value
Thermal power transmitted to the consumer, MW(t)	500
Heat output, GJ/year	12,600 $\times 10^3$
Total investment, million rubles	1657.0
Operation expenses, million rubles/ year	209.1
Manufacturing cost of heat, rubles/GJ	16.6
Saving of natural gas, m^3/ year	340 $\times 10^3$
Saving of oil, tons/ year	430 $\times 10^3$

Calculations showed that, at the taken level of price of mazut (in 1991) of 2500 rubles per ton, the NPPS would reduce expenditures on fuel by more than 1 billion rubles annually.

Another effect of using NPPS for ORP would be the reduction of the adverse impacts on the inhabitants and the environment, which is caused by the pollution of the atmosphere resulting from the burning of the fossil fuel. This effect may amount to some dozens of millions of rubles per year.

6. CONCLUSION

The study of the NPPS project using three VGM-P industrial reactors of 215 MW power with the helium outlet temperature of 750°C showed that the plant can provide up to 80 % of heat consumption in the technological processes of a standard ORP. Mainly unavailability of acceptable process coolants restricts the residual 20 % of heat consumption of the high temperature level above 430°C.

The development of the industrial VGM-P with the helium outlet temperature of 950°C on the basis of the VGM project will make it possible to use scientific and engineering potential accumulated on VGM and its main components supported by research and development works.

The design experience of NPPS with VGM-P showed that one of the key points is a selection of a coolant for the intermediate and especially for the process circuits. Elimination of the intermediate circuit in VGM-P would reduce capital investments and operating expenses. This is the subject of further design developments.

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POWER TECHNOLOGY COMPLEX FOR PRODUCTION OF MOTOR FUEL FROM BROWN COALS WITH POWER SUPPLY FROM NPPs

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Abstract

With the present-day challenge of efficient use of low-grade coals and current restructuring of coal industry in the Russian Federation, it is urgent to organise the motor fuel production by the synthesis from low grade coals and heavy petroleum residues. With this objective in view, the Institute of Physics and Power Engineering of RF Minatom and Combustible Resources Institute of RF Mintopenergo proposed a project of a standard nuclear power technology complex for synthetic liquid fuel (SLF) production using fast neutron reactors for power supply. The proposed project has two main objectives: (1) Engineering and economical optimization of the nuclear power supply for SLF production; and (2) Engineering and economical optimization of the SLF production by hydrogenisation of brown coals and heavy petroleum residues with a complex development of advanced coal chemistry. As a first approach, a scheme is proposed with the use of existing reactor cooling equipment, in particular, steam generators of BN-600, limiting the effect on safety of reactor facility operation at minimum in case of deviations and abnormalities in the operation of technological complex. The possibility to exclude additional requirements to the equipment for nuclear facility cooling was also taken into account. It was proposed to use an intermediate steam-water circuit between the secondary circuit sodium and the coolant to heat the technological equipment. The only change required for the BN-600 equipment will be the replacement of sections of intermediate steam superheaters at the section of main steam superheaters. The economic aspects of synthetic motor fuel production proposed by the joint project depend on the evaluation of integral balances: thermal power engineering, chemical technology, the development of advanced large scale coal chemistry of high profitability; utilisation of ash and precious microelements in waste-free technology; production of valuable isotopes; radical solution of ecological problems.

Nuclear Complex Project for Synthetic Liquid Fuel Production

With the present-day challenge of efficient use of low-grade coals and current restructuring of coal industry in the Russian Federation, it is urgent to organise the motor fuel production by the synthesis from low grade coals and heavy petroleum residues.

With this objective in view, SSC RF – Institute of Physics and Power Engineering (IPPE) of RF Minatom and Combustible Resources Institute of RF Mintopenergo (CRI) proposed a project of a standard nuclear power technology complex for synthetic liquid fuel (SLF) production using fast neutron reactors for power supply.

Fast reactors with liquid metal coolants developed at IPPE have sufficiently high thermal power generation potential requiring the technological processes with low pressure in coolant. More than 20-year experience of successful operation of fast reactors at commercial NPPs has shown that they are highly reliable and safe.

The proposed project of a standard nuclear power technology complex for SLF production has two main objectives:

1) Engineering and economical optimization of the nuclear power supply for SLF production;
2) Engineering and economical optimization of the SLF production by hydrogenisation of brown coals and heavy petroleum residues with a complex development of advanced coal chemistry.

Each of these aspects appears to be a multi-factor task.

The first major aspect is directly associated with the general strategy of nuclear power development in the country, considering its present and perspective structure. The central problem here is the choice of an appropriate type of reactor, nuclear fuel, and the optimal nuclear fuel cycle (NFC). It is coordinated with the strategy of the nuclear power perspective development and the optimization of NFC in Russia (closed NFC with fast reactors involved) [1]. Safety and ecological compatibility are issues of primary importance in the development of the standard nuclear power technology complex for SLF production.

The second major aspect of the project is considered as below.
Essentially, motor fuel synthesis by coal hydrogenization is a process to transform high-molecular substances of coal organic mass (COM) into liquid and gaseous hydrocarbons by pressurized hydrogen. This approach appears to become attractive for the heat power engineering world when the prospective exhaustion of natural gas and oil resources is considered. The scientific fundamentals of this process were developed in the early 20th century by various scientists - V.Ipatyev, N.Zelinski, F.Bergius, F.Fischer, and others.

Technological Background of Synthetic Liquid Fuel Production

In 1930s, industrial enterprises were founded in several countries, in particular, Germany and Great Britain, for the production of petrol, kerosene, diesel fuel, lubrication oil, etc., by coal hydrogenization. In 1950s, coal hydrogenization was developed to the commercial scale in the USSR as well. However, when new oil fields were found in the USSR, Near East countries, etc., the oil prices decreased, and SLF production from coal virtually ceased. In 1970s, the oil price rose sharply, and it became evident that resources would be exhausted in the near future.

It has been recognized world-wide that coal liquefication is a promising method of obtaining liquid motor fuel and diverse products for chemical industry with the raw material resources available for this purpose for a long prospect. To date, in industrialized countries, especially in the USA, facilities and experimental plants for coal liquefication have been operated with their production of more than 21 million tons of SLF a year; 15 basic technologies are supported by intensive R&D work; more than 50 projects of the total annual design output of about 300 million tons of SLF have been launcehed at various stages of development and construction [3].

Brown coals are the best raw material for synthesizing a wide range of motor fuels. Low metamorphization solid combustible minerals of this grade is typified by a small extent of carbonification and a peculiar composition of organic mass components. Brown coals are considered to be low-grade power fuel; however, there are extremely valuable raw materials for coal chemistry. This is confirmed by several examples in Germany in 1930s to early 1940s, where very strong industrial concerns developed on the ground of the domestic brown coal resources before the war. Russia's resources of brown coal are the largest in the world. A giant deposit of high quality brown coal (exceeding trillion of tons), the world's cheapest chemical raw material, serves as a unique base of the Kansky-Achinsky Fuel and Energy Resource Complex (KAFRC), its implementation being given the prime importance in the State Programmes of Siberia development.

Specific features of the composition and structure of these brown coals containing 5-10% ash and up to 40% moisture, their non-transportability and extreme cheap cost of mining (10% of the coal cost in Donbass) give some specific features of operational coal-fueled large thermal electric power plants of the KAFRC (Nazarovski, Berezovski). First of all, this causes an extremely hazardous ecological impacto on that vast region.

The brown coal basin in the Moscow area (Tulaugol) has been exhausted for its long exploitation. Due to a high ash content in that coal (up to 40%), many mines have been closed. However, instead the oil-refineries in the European region of Russia have at their disposal a valuable raw material - heavy petroleum residues which can be reprocessed into standard motor fuel by the hydrogenization method.

Since the laboratory of fuel hydrogenization was established at CRI in 1934, it has carried out a lot of work on this subject. As a result of fundamental research, the scientific basis has been established for solid fuel hydrogenization processes. The theory of hydrogenization transformation of hydrocarbons has been developed. The chemical behaviour and kinetics of catalytic reactions have been identified over a wide range of conditions. New effective catalysts have been created. Principle regularities of hydrogenization reactions of high-molecular compounds have been formulated.

In 1970s, CRI developed technologies for obtaining motor fuel from coals using the hydrogenization method under medium pressure of hydrogen (up to 10 MPa) [4]. The new technology has basic advantages compared to the processes used till then in foreign countries. The main challenge in the technology improvement has been met: decrease of pressure in coal hydrogenization from 30 to 10 MPa and even lower levels. More than 30 inventions have been realized. The pressure decrease made it possible to lower the capital investments approximately by 5 times, using the machine building base in existence now for the manufacturing of the main devices, chemical reactors, and complete lines equipped with high production capacity.

The technology proposed by CRI has an important advantage: being universal, it enables equally efficient synthesis of standard (qualified) motor fuels, using a uniform technological regime and one equipment line, both from the cheapest brown coals and from heavy oil residues (fuel oil, residual tar) which are available in excessive quantities. The CRI technological process is waste-free. Its major advantage is the possibility to obtain the most valuable additional sources of chemical raw materials for chemical industries such as phenols, nitrogen bases and other additional sources of chemical raw materials.

In this aspect, the CRI process for obtaining synthetic motor fuels is fundamental also for the development of advanced large scale coal chemistry. The technology developed provides a rational use of coal ash for the production of building materials; recovering a variety rare, scattered, and precious elements, such as germanium, gallium, yttrium, scandium, mercury, arsenic, natural radioactive elements of the uranium and thorium group, etc., which are geochemically associated in brown coals. In the conventional burning process of coals at thermal power plants, these elements are dispersed in the gas releases, and the ash is concentrated in large masses and pollutes wide areas.

The technology developed by CRI for motor fuel synthesis by hydrogenization of brown coals and heavy petroleum residues has undergone long-term and careful tests under the quasi-industrial conditions at the experimental plant ST-5 of synthetic fuel at Venev, Tula. The new technology effectiveness has been widely recognized by the scientific and expert community.

In 1992, CRI and R&D Institute for Gas and Petroleum Industry (Grozgidroneftekhim) performed a joint feasibility study of constructing an industrial enterprise for the production of motor fuel from the KAFRC brown coals with annual output of 3 million tons. The project was approved by the Mintopenergo. The KAFRC brown coals are the most suitable for the conditions of technology developed by CRI for synthetic liquid fuel producdtion. In the hydrofining of products of the liquefied coal, the ST-5 plant produced components of petrol and diesel fuel which meet the state standards for high-quality oil products.

The project of the nuclear power and technology complex proposed for the production of synthetic motor fuel relies on all achievements in the technology of motor fuels by hydrogenization of brown coals and at the same time is typified by a principle novelty: power supply for all hydrogenization processes by fast reactors. As mentioned above, these technologies of CRI are most efficiently combined with fast neutron reactors with liquid metal coolant, with thermal energy potential reaching 550°C, which is enough to meet the demand of all steps of technological process of SLF production [5-8].

Energy Source for Synthetic Liquid Fuel Production

To obtain one ton of the product (SLF), a thermal power plant has to burn 1.5 – 2 tons of organic fuel, which results in a two-fold increase of expenses for the coal mining and withdrawal of lands from economic use.

In a large-scale production of SLF, ecology-related issues are especially important. Comparative analyses of the general inpacts for public health caused by the operation of nuclear and coal fuel cycle enterprises calculated for equal annual energy yields gives at least a 100-fold advantage to the nuclear fuel cycle (N.Ponomaryov-Stepnoi, "Novy Mir" – New World monthly, 1988, 9, p. 173). The inclusion of ecologically safe nuclear reactors(BOR-60, BN-350, BN-600, BN-800 types) into a commercial complex for the SLF production, electric energy, water steam and gas coolant would allow to reduce the consumption of coal and power source gas by 80%.

The structure of power consumption of the technological complex is characterized as follows: key equipment of the synthetic liquid fuel production line (paste preparation, separation, production of vacuum gas oil, cracking of middle oils, hydrofining, treatment, reforming, distilling) consumes the heating media's heat with temperature potential exceeding 430°C. Its fraction in the general balance is ~30%, if the initial moisture content of the coal amounts to 30%. Heat with lower temperature potential can be used to generate electricity, and to obtain steam that is subsequently delivered for the facility's technologic processes (technological steam), coal preparation and drying of the coals being processed. The required ratio of the high- and low- potential heat in the SLF production scheme can be provided by a fast reactor, which has the cooling metal temperature at the reactor inlet at ~320°C, and at the outlet at ~530°C.

From the standpoint of SLF production, such schemes can provide its maximum possible yield per unit power of reactor. If reactors in operation, e.g., BN-600 are strategically used for the project, the schemes would require the development of new heat exchangers from the secondary circuit sodium to the coolant for the technological equipment (cooling of secondary circuit sodium from ~520°C to ~440°C), and new steam generators for supplying heat to the equipment for technological steam production, coal preparation and drying, and to the electric generation to meet the demand of the technologic facility itself.

As a first approach, a scheme is proposed (Fig.2) with the use of existing reactor cooling equipment, in particular, steam generators of BN-600, limiting the effect on safety of reactor facility operation at minimum in case of deviations and abnormalities in the operation of technological complex. The possibility to exclude additional requirements to the equipment for nuclear facility cooling was also taken into account. It was proposed to use an intermediate steam-water circuit between the secondary circuit sodium and the coolant to heat the technological equipment. As the intermediate heat media, inert gas was used for coal heating and drying, and technical water was used for obtaining technological steam. The closed circuit of the steam generator cooling maintains the water quality required for reliable operations of the steam generator. The water-steam cycle in the intermediate circuit is similar to the tertiary circuit of BN-600. This leads to the situation in which thermal potential of the secondary circuit

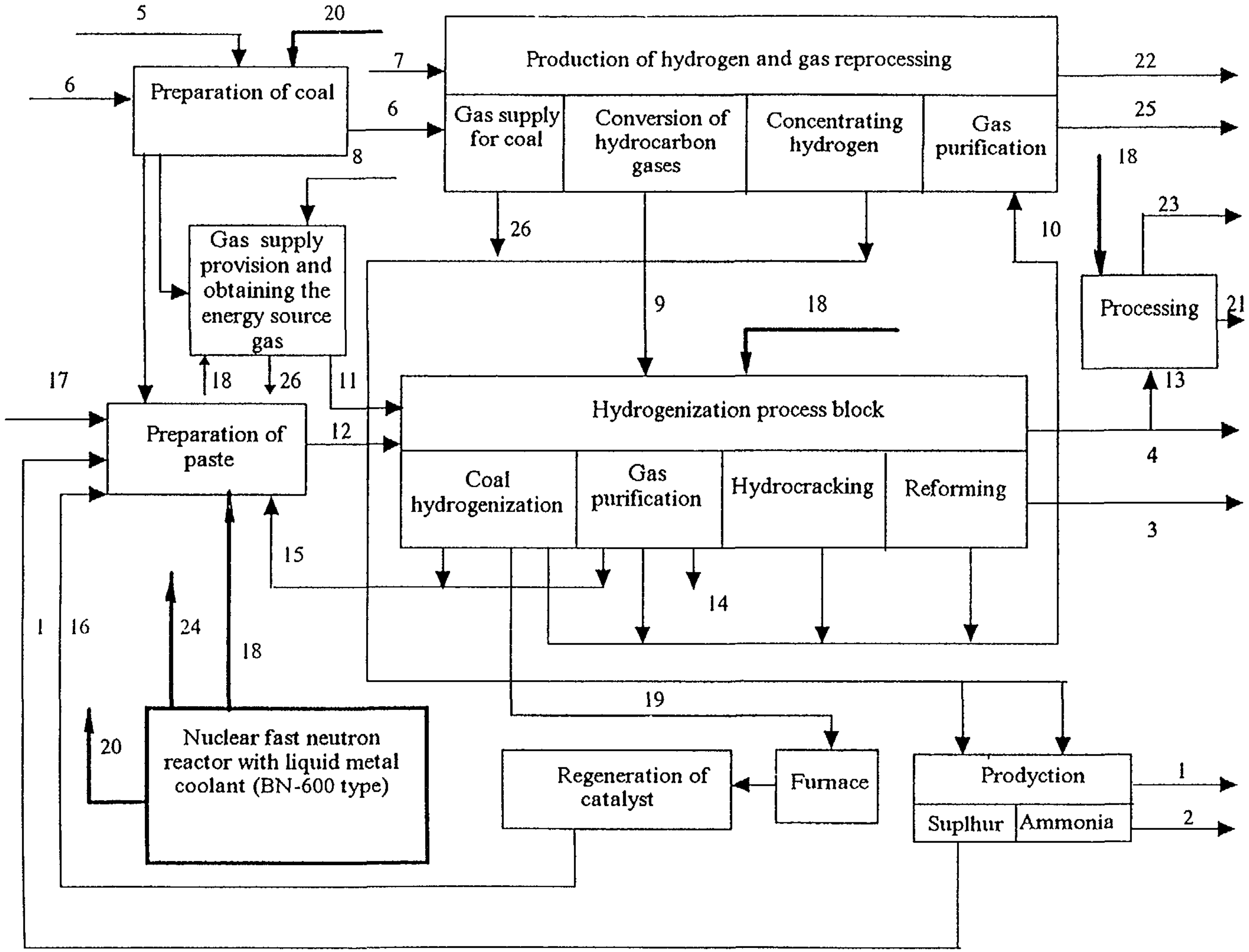

Fig.1
Principle flow chart for obtaining liquid product fromcoal at industrial enterprise with nuclear source of BN-600 type

1 - Sulphur
2 - Ammonia
3 - High octane petrol
4 - Diesel fuel
5 - Reagents
6 - Coal
7 - Oxygen
8 - Air
9 - Hydrogen
10 - Gas
11 - Energy gas
12 - Paste
13 - Low octane petrol
14 - Water
15 - Paste forming source
16 - Return of catalyst
17 - Catalyst
18 - Vapor
19 - Slag
20- Hot inert gas
21 - Automobile petrol
22 - Liquified gases - C_3 - C_4;
23 - Aromatic hydrocarbons - C_6-C_8;
24 - Electric energy
25 - CO_2
26 - Slag

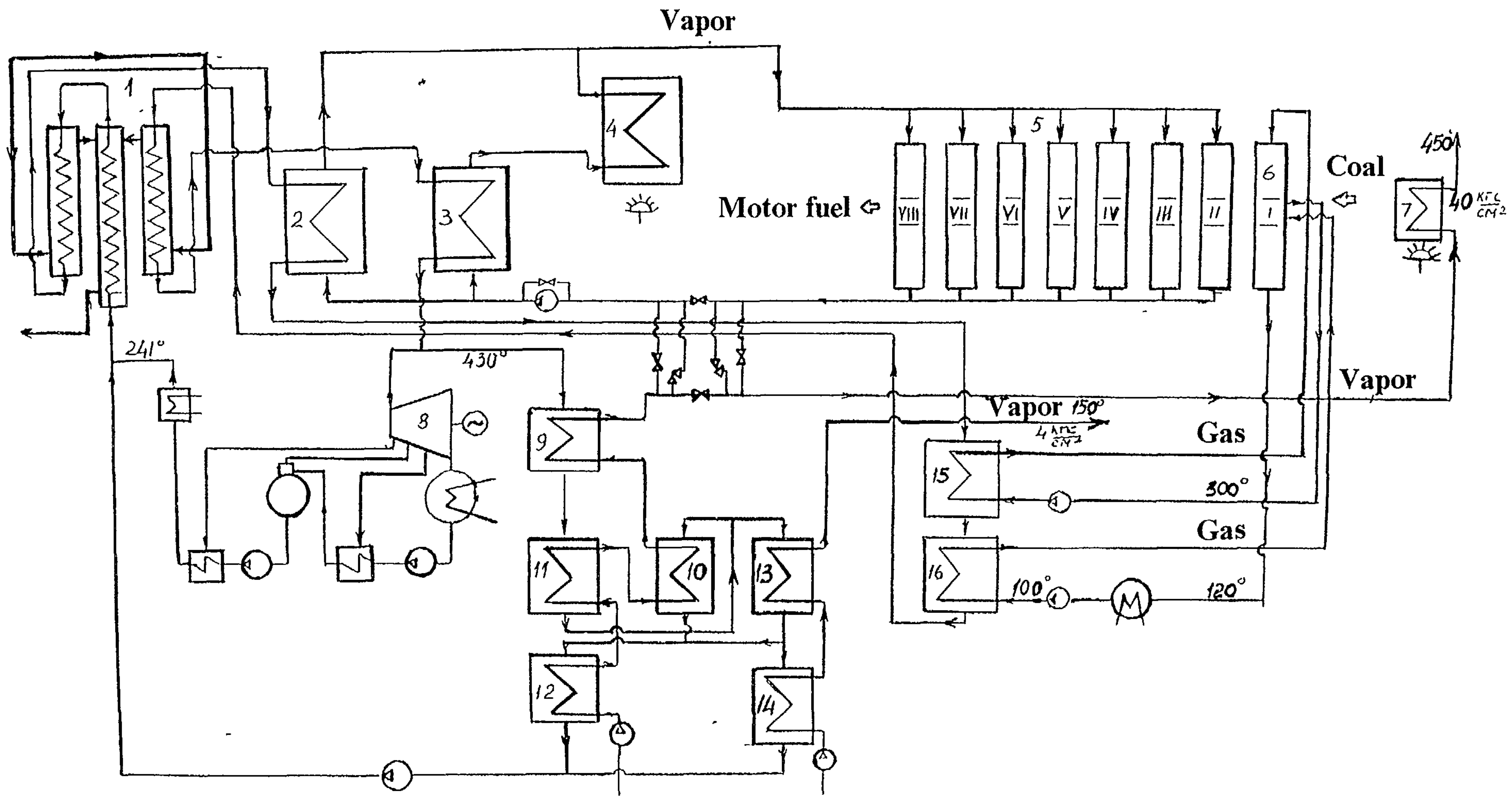
Vapor
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
VIII
VII
VI
V
IV
III
II
I
Motor fuel
Coal
450°
40 кгс/см²
241°
430°
Vapor
Vapor 150°
4 кгс/см²
Gas
300°
Gas
100°
120°

Fig. 2

sodium with temperature above 430°C cannot be used for the power supply of technological equipment only (~60%), since it reduces the SLF production rate per kilowatt of the reactor power. Because the high pressure steam at temperature of 430°C exists in the intermediate circuits in excess of the need for the technological cycle, it can also be used to generate electricity considerably above the need of technological facilities.

Structural scheme of the nuclear power complex based on BN-600 is presented in Figs. 1 and 2.

Designations in Fig. 2 are as follows:
1 – steam generator; 2 – heat exchanger between the steam from the steam generator superheater and the steam for heating technological equipment; 3 – heat exchanger between the steam from the intermediate superheater of steam generator – steam of the heating circuit for technological equipment heating"; 5 – technological equipment (II – paste preparation, III – separation; IV – vacuum gas oil production; V – cracking of middle oils; VI – hydrofining, VII – treatment, reforming; VIII – distillation); 6 – coal preparation, drying of coal; 7 – fire steam superheater of technological steam of high pressure; 8 – turbine generator facility for electric energy production; 9 – intermediate superheater of high pressure technological steam; 10 – evaporator of the high pressure technological steam circuit; 11 – second step heater of water of the high pressure technological steam circuit; 12 – first step heater of water of the high pressure technological steam circuit; 13 – evaporator of the low pressure technological steam circuit; 14 – water heater of the low pressure technological steam circuit; 15 – gas heater of the second step of coal preparation and drying line; 16 – gas heater of the first step coal preparation and drying line.

Principle characteristics of NPP with BN-600 reactor are presented below.

Principle characteristics of NPP with BN-600 reactor

Unit power of facility, MW:

Unit power of reactor, MW: Electric Thermal	 600 1470
Coolant temperature at the outlet of reactor, °C	550
Duration of reactor operation between reloadings, days	150
Parameters of steam generated: Temperature, °C Pressure, MPa	 505 14.2
Productivity of reactor by steam, t/h	1980
Electric efficiency (net) of reactor, %	40
Design-basis operation time, years	40

The only change required for the BN-600 equipment will be the replacement of sections of intermediate steam superheaters at the section of main steam superheaters. The reheaters' sections are not necessary to be replaced, because they are designed for operation below 3 MPa. The changeover to turbines for downgraded parameters will be required (12.0 to 13.0 MPa, 430°C, steam flow rate - ~ 500 t/h).

In the operation of the proposed scheme, the nuclear technology complex based on the power source with BN-600 will produce 8.8 x 10^5 t/year of SLF with the electric power supply of ~300 MW. Another yield of 170-200 MW(electric) can be transmitted to the external power network. Table 1 presents the amounts of commercial products which can be obtained from the scheme shown in Fig.3.

Table 1

Scope of production of commercial products from brown coal of KAFRC (mass. % per dry coal)

Item	
1. Petrol for vehicles	10,64
2. Jet fuel T-8B	3,19
3. Diesel fuel (S<0.05%)	28,88
4. Aromatic hydrocarbons C6 – C8	0,70
5. Methyltretbuthyl ether	2,98
6. Methanol	-
7. Sulphyr	0,07
8. Ammonia	0,40
9. Power fuel (slag of hydrogenization)	6,79
10.Phenols (phenol, crezols, resorcin, etc.)	0,70

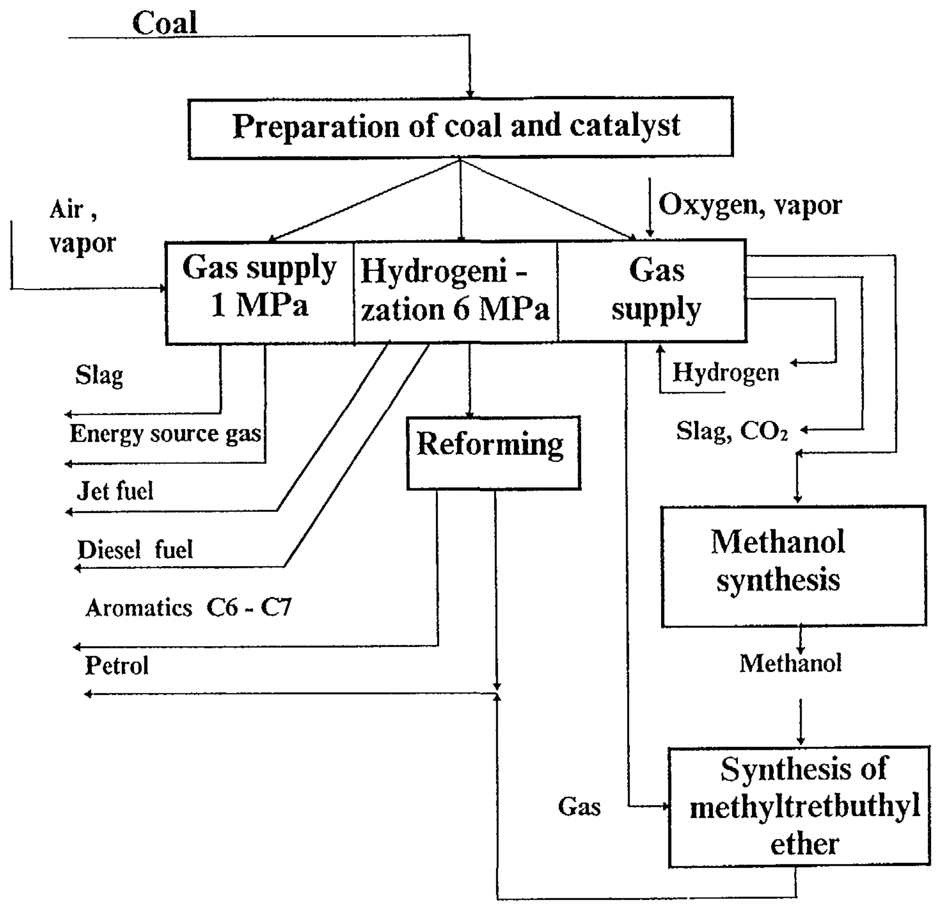

Fig. 3 Flow chart of obtaining fuel-and chemical purpose products from coal

Conclusions

The economic aspects of synthetic motor fuel production proposed by the joint project depend on the evaluation of integral balances: thermal power engineering, chemical technology, the development of advanced large scale coal chemistry of high profitability; utilisation of ash and precious microelements in waste-free technology; production of valuable isotopes; radical solution of ecological problems.

On the eve of the 21st century coal will take undoubtedly its proper place in the solution of power crisis problems in both our country and in the world.

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II.2. OPERATIONAL AND MATERIAL ASPECTS OF NUCLEAR HEAT APPLICATION

Operational modes of nuclear desalination plants

XA9848820

COUPLING OF AST-500 HEATING REACTORS WITH DESALINATION FACILITIES

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Abstract

The general issues regarding NHR and desalination facility joint operation for potable water production are briefly considered. AST-500 reactor plant and DOU GTPA-type evaporating desalination facilities, both relying on proven technology and solid experience of construction and operation, are taken as a basis for the design of a large-output nuclear desalination complex. Its main design characteristics are given. Similarity of NHR operation for a heating grid and a desalination facility in respect of reactor plant operating conditions and power regulation principles is pointed out. The issues of nuclear desalination complexes composition are discussed briefly as well.

1. Introduction

A low-grade nuclear heating reactor is capable to be used effectively as a part of water desalination complexes, taking into account their distinctive features such as:

- high reliability of the heat generation;
- extended refuelling interval (2 years);
- identity of NHR operating conditions both for the heating grid and for a desalination facility;
- enhanced radiological safety, which allows:
    * to construct the nuclear power plant close to a sea shore or any other sources of salted water used for a desalination process;
    * to create an integrated complex incorporating both desalination facilities with desalted water stocks and a nuclear power plant;
    * to deploy such complexes close to fresh water consumers, e.g. industrial-residential centers [1].

2. Desalination Complex Composition

The AST-reactor is designed to produce low parameter heat, which defines the type of an appropriate desalination facility. Direct usage of the low-parameter heat is most effective in evaporating facilities. Long operation experience is available in Russia for the desalination facilities with horizontal film evaporators, in particular more than 20-year experience of plant operation in the industrial nuclear desalination complex with the BN-350 fast reactor (also developed by OKBM) on the Caspian sea shore.

Desalination plants of such type are characterized by the good quality of distilled water produced, low power consumption, long-term (more than 25 years) service life of the basic equipment.

Rate of expansion of the desalination facilities with the horizontal-tube film evaporators for the last 10 years was the highest, according to the IAEA data, compared with the desalination plants of other types, which is explained by their mass production mastering, better economics: relatively low (in comparison with the reverse osmosis facilities) operational cost and lower price of desalinated water, as well as higher quality of potable water (data of IDE Technology LTD., Israel).

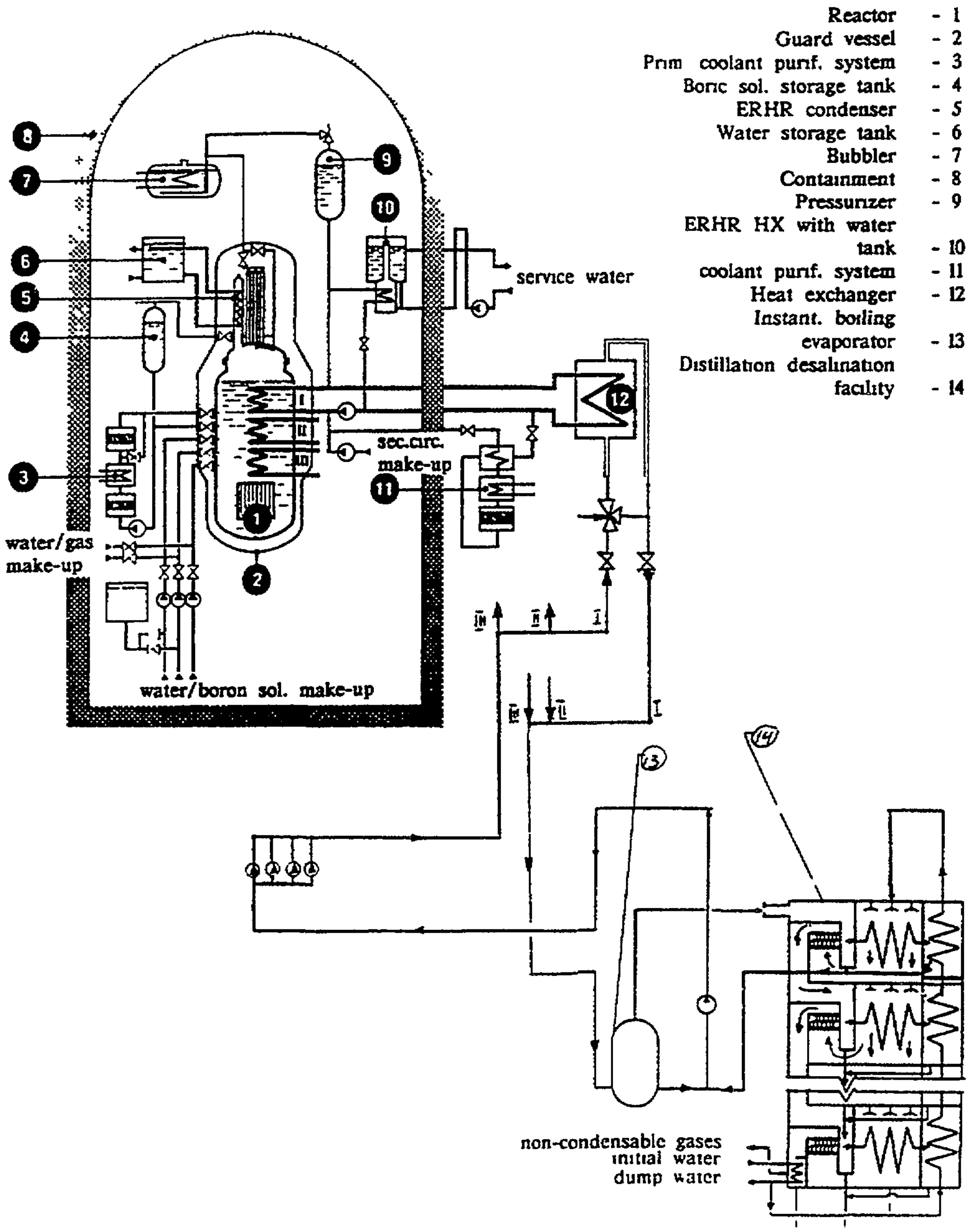

Fig. 1 AST-500 NHR-based Desalination Facility Principal Flow Diagram

Creation of desalination complexes incorporating the AST-500 power units of some 200,000 m^3 per day output and even more in one unit (see Fig.) presumes availability of large consumers of desalinated water in the region, as well as of developed system for water supplying and distribution (pipelines, pump stations, etc.), and of developed power systems. Under these conditions there are grounds to energize the desalination complex from the external power system, while the reactor is operated only for heat supply to desalination facilities. This simplifies the complex operation, and improves the efficiency of the generated heat usage.

However, for regions with weak-developed power grids and high tariffs for electricity, there is a possibility to realize on the basis of AST-500 reactor a desalination complex powered from the own auxiliary turbogenerator. A turbine building, related electrical equipment, etc. should,be additionally included in the plant: in this case the desalination complex output is decreased by appr. 20% due to redistribution of thermal power generated by the reactor between a turbine plant and a desalination facility.

The number of power units in the desalination complex is to be determined by the specified output of desalinated water with account of its maximum consumption and load variation over seasons, and by the availability of other sources of fresh water in the given region. The water output should not be less than a constant portion of the year load curve determined by the residential and industrial water consumption with account of planned outage of one power unit for refuelling. The reactor refuelling should be carried out in the period of minimum load in water consumption, if this factor takes place (e.g. regions with irrigated agriculture).

3. Plant Operation Control Concept

Considering a nuclear desalination plant control principles, its similarity with the NHR operation for the heating grid (load variation over seasons) and for the hot water supply (daytime load variation) should be noted. But in the given case there is wider possibility to flatten the influence of load variations onto the nuclear reactor because of the flexibility to accumulate or consume desalinated water in water tanks following a daytime reduction or rising in water consumption respectively. Besides, there is also a possibility to divide the peaks of industrial and agricultural loads (e.g. fields irrigation during night hours, if possible).

Thanks to the availability of desalinated water stocks, the continuous supply of fresh water to consumers can be secured also in a case of short-term unplanned outage of the desalination unit. Taking account of a heating rate limit specified for the AST-500 reactor start-up mode, the minimum period of interruption in desalinated water production process for this unit amounts to appr. 24 hours.

In general, the choice of the method for heat output regulation to the base-load or load-following principle (or their combination) is dictated by specific conditions and requirements of the User. The actual and prospective loads of industrial, household and agricultural consumers of water are taken into account as well as their variation curves, availability of other sources of fresh water and heat, the range of heating water temperature variation at the desalination facility inlet, etc. The AST-500 reactor ensures the reliable operation of the complex both in base-load and load-following operation modes at the reactor power variation rate limit of 25% N_{nom} per hour.

4. Design Study for AST-500 Integration into Desalination Complex

The design study was carried out by the OKBM together with other design Institutions (SverdNI1ChimMash, NIAEP, Kurchatov Institute, etc.) aiming at the development of a desalination complex which is composed of several (more than two) autonomous units each comprising the AST-500

Table 1 Nuclear desalination plant main technical data

Nuclear reactor plant		
1.	Reactor thermal power, MW	400
2.	Primary circuit pressure, MPa	2
3.	Primary circuit temperature, °C	208/141
4.	Secondary circuit pressure, MPa	1.2
5.	Secondary circuit temperature, °C	160/102
6.	Tertiary circuit pressure, MPa	2
7.	Tertiary circuit temperature, °C	130/98
8.	Auxiliary power (off-site power source), MW	appr. 7
9.	Auxiliary heat consumption, MW	appr. 15
Desalination facility		evaporating type on the basis of DOU GTPA-700 apparatus with horizontal tube- film evaporators
10.	Number of evaporation apparatus	
	total	16
	operating	13
	stand-by	3
11.	Auxiliary power (off-site power source), MW	appr. 12
12.	Nominal output, m^3/day	appr. 220,000 (16,800x 13)
13.	Sea water max. boiling temperature in the first stage of each apparatus, °C	<115
14.	Salt concentration in desalinated water, mg/1	<20

reactor plant and a desalination facility [2]. The number of desalination apparatus in the facility which are working in parallel is defined by taking account of the possibility to disconnect a part of them for preventive maintenance and repair, because the maintenance interval may not coincide with the reactor refuelling one. The level of the redundancy accounts also for the necessity of annual maintenance for desalination apparatus, this work duration (up to 20 days), as well as a probability of additional failure of one of them.

Main technical characteristics of the nuclear desalination unit based on the AST-500 reactor (without a turbogenerator plant) are given in Table 1.

The potential available for the advanced AST-500 M reactor uprating allows to raise additionally the desalination facility output by appr. 15%. The nuclear desalination complex principal flow diagram is given in Fig.1. A three-loop flow diagram being traditional for the AST-500 secondary and tertiary circuits is retained here along with a possibility for one loop disconnection from the consumer.

In the desalination complex, as well as in the AST-500, tertiary circuit loops are joined by "hot' and 'cold' headers. Circulating pumps similar to the grid ones in the AST-500 provide heat transport from the reactor plant to consumers - the desalination apparatus with water recirculation through a closed loop. Heating medium can be supplied to the first stage of desalination apparatus as water or steam produced from this water in an instantaneous boiling facility. Both methods are verified in the operating nuclear desalination complex with the BN-350 nuclear reactor.

At the plant operation under automatic control, the reactor power and parameters of the circuits are established and maintained by the reactor control rods and three-way valves in the loops of the tertiary (heating) circuit according to the control algorithm similar to that adopted for the AST-500. The reactor power variation range (10-100% N_{nom}) overlaps with a margin the range of stable operation of the desalination apparatus (30-100% G_{nom}, where G_{nom} - nominal flow of desalinated water).

When determining a volume of stand-by desalinated water storage tanks it is expedient to take account of the power variation rate admissible for the reactor at planned variations of desalinated water consumption. Transients associated with complete loss of load cause the reactor shutdown by emergency protection signals and its cooling down by the normal or emergency heat removal systems. The reactor operation with incomplete number of heat transport loops without time limitation is permitted in the range of 10-50% $N_{nom,}$ if the design limits and safety conditions specified for the AST-500 are satisfied. The reactor transition to partial operation mode means proportional decrease in desalination facility output by disconnection of several desalination apparatus or by proportional decrease of each apparatus output.

5. Conclusion

The AST-500 NHR has become a reference reactor plant for the whole series of integral enhanced safety reactors which have been developed recently in OKBM. The basic principles and engineering decisions realized in the reactor plant allow to use it effectively as a heat source for an evaporating desalination facility on the basis of well proven apparatus of DOU GTPA-type with horizontal tube-film evaporators.

The rated output of this facility amounts to 220,000 m^3 /day of desalinated water, with the potential for 15% increase in the output. Modes of the AST-500 operation in combination with the desalination facility are similar to those for operation to heating grid.

High level of radiological safety intrinsic for the AST-500 reactor allows to site it close to water sources, to fresh water consumers and in the immediate proximity of desalination facility, thus forming an integrated power-desalination complex.

Russian positive practical experience of in creation and operation of sea water nuclear desalination system based on the BN-350 reactor and on the desalination apparatus of DOU GTPA-type, together with the proven technology of the AST-500 NHR allow to facilitate substantially licensing and creation of the AST-based desalination complexes. Their excellent safety and economic characteristics give grounds to consider them as a rather prospective for the deployment in many regions worldwide suffering from shortage of potable water.

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XA9848821

ANALYSIS OF THE NUCLEAR HEATING REACTOR AND ITS POSSIBLE APPLICATION IN SEAWATER DESALINATION

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Abstract

In order to mitigate the problems of the energy shortage, environmental pollution caused by coal burning and the transport burden in China, the Institute of Nuclear Energy Technology (INET), Tsinghua University, under the support of the state, began the research and development (R&D) of nuclear heating reactor (NHR), which is one of the national key R&D projects in China since the 1980's. Since a 5MW test NHR was completed in November 1989, a lot of experiments have been carried on the NHR-5. The NHR-200 is developed on the experience gained from the design, construction, start-up and operation of the NHR-5. It is designed with a number of advanced and inherent safety features. The main technical and safety features of NHR-200 are: a vessel type light water reactor with the integrated arrangement, full power natural circulation, self-pressurized performance and dual vessel structure. The hydraulic driving system of the control rods is adopted. The design of the NHR-200 insures that the reactor core can be always covered by coolant at any LOCA conditions and the possibility of rods ejection event is excluded by using hydraulic control rods driving system. The excellent performance of the NHR-200 shows that it is suitable to the coupling with a seawater desalination plant from both technical and economic stand. According to the systematic analysis and comparison of economy, technology and safety, the selected coupling design of desalination plant with the NHR-200 are: the steam generator plus multi-effect distillation (MED) process for single water production and the steam generator plus turbine system plus MED process for cogeneration of water-electricity. The economic analysis based on the above mentioned two coupling designs has be conducted. The desalinated water price and its influential factors are determined under present technological circumstances. And some specific proposals of which system to select are given.

1. Introduction

The energy supply has been one of the major problem which strongly impacts the socio-economic development in China. Coal plays an important role in China's primary energy supply systems. According to a study on the energy strategy of China, about 75% of the primary energy consumption is contributed by coal. Enormous consumption of coal leads the very serious problems in environmental pollution, which will become unacceptably serious in some cities if no action is taken immediately. In addition, since the coal production district is highly concentrated in the North and Northwest China, while the main consumption district is located in Northeast, East and Southeast China, the produced coal must be usually transported after a long distance to the end user. Coal transportation takes up more than 40%, 25% and 20% of the railway, highway and waterway freight capacity. It aggravates the serious situation of communications and transportation.

In order to mitigate the problems of the energy shortage, environmental pollution caused by coal burning and the overburdened transportation systems, the Institute of Nuclear Energy Technology (INET), Tsinghua University, under the support of the state, has begun the research and development (R&D) of nuclear heating reactor (NHR), which is one of the national key R&D projects in China, since the 1980's.

The INET conducted successful tests of nuclear district heating using the existing pool type research reactor in 1983 and 1984. The project of a 5MW test NHR (NHR-5) was started in 1984 and begun to construction in 1986 and completed in 1989. In November 1989 the test reactor went the first criticality.

Since then, It has been successfully operated for district heating and a lot of experiments have been carried on the NHR-5. On the basis of the successful of NHR-5, a commercial sized NHR with output of 200MW thermal power (NHR-200) has been developed by INET[1]. Several cities and large enterprises are interested in introducing NHR into their local energy system. The application for the construction permit of the first demonstration NHR-200 was submitted to the National Nuclear Safety Administration in December 1995 with the attached documents and now the preparation of the project is in progress. At the same time, the other three projects are planning. One is to provide process steam for the new developing district in Shanghai, China. The other two are used as energy source to couple with desalination plant in Dalian, China and in Morocco.

2. Main technical and safety features of NHR-200

The NHR-200 is developed on the experience gained from the design, construction, start-up and operation of the NHR-5. It is designed with a number of advanced and inherent safety features, including integrated arrangement, natural circulation, self-pressurized performance, hydraulic control rods driving and passive systems. It could be used in district heating, air conditioning, seawater desalination and other industrial processes. The main design parameters of NHR-200 and NHR-5 are listed in Table I.

Table I Main design parameters of the nuclear heating reactor

		NHR-5	NHR-200
Thermal power	MW	5	200
Primary system pressure	MPa	1.5	2.5
Core inlet/outlet temperature	°C	146/186	140/210
Ave. linear heat rate	kW/m	5.6	7.67
Volumetric power density	kW/L	26	36.23
Number of fuel assemblies		16	96
Number of control rods		13	32
Active core height	m	0.69	1.9
Active core diameter	m	0.57	1.9
Inventory of UO_2	t	0.51	14.87
Enrichment of initial core	%	3	1.8/2.4/3.0
Refueling enrichment	%	3	3
Intermediate circuit temperature	°C	102/142	95/145
Intermediate circuit pressure	MPa	1.7	3.0
Heating grid temperature	°C	90/60	130/80

2.1 Technical description of the NHR-200

The NHR-200 is a vessel type light water reactor. The main technical features can be briefly summarized as follows[2].

a) Integrated design. Both reactor system and the primary circuit, including primary heat exchanges (PHEs), are arranged into the reactor pressure vessel (RPV). The core is located at the bottom of the RPV, and six PHEs are arranged within the annular space between the riser and vessel wall. The riser with height of 5.1m above the core outlet is to enhance the capability of natural circulation. Spent fuel assemblies are stored in the racks around the core. All of the penetrations are on the upper part of the RPV and the biggest diameter of the penetrations of coolant pressure boundary is ϕ50mm. Fig.1 shows the reactor structure.

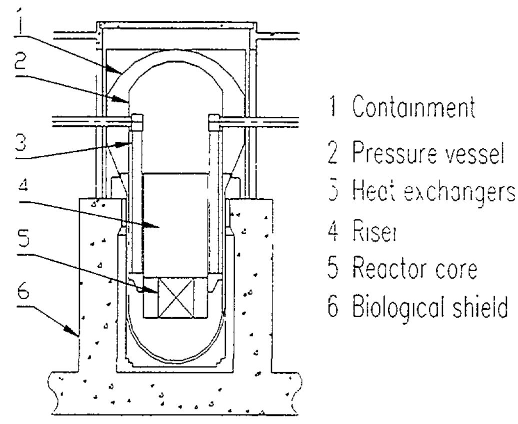

Fig.1 Reactor structure of NHR-200

b) Full power natural circulation cooling. The coolant circulates due to the density difference between hot and cold regions inside the RPV at all power levels so that the primary circulating pumps can be eliminated and the higher system reliability can be ensured. The decay heat is removed also by natural circulation and a special test, which was conducted on NHR-5, indicates that this kind of residual heat removal system is effective even if the natural circulation in the primary circuit is interrupted.

c) Hydraulic driving system of the control rods. This system meets the requirement of "fail-safe" principle i.e. control rods will drop into reactor core automatically by gravity under lose of power supply, depressurization of RPV, pipe break and pump shut down events. This design simplifies the reactor structure and eliminates the accident of rapid rod ejection. There is no boric acid in the coolant during normal operation. Gadolinium oxide as a burnable poison is used to control the reactivity along with the B_4C control rods. In addition, a boric acid injection system as a secondary shutdown system will be operated when anticipated transient without scram occurs.

d) Self-pressurized performance. The primary pressure, composed by the saturate steam pressure corresponding the core outlet temperature and a certain inventory of nitrogen, regulates itself very stable at the designed level.

e) Dual vessel structure. A tight steel containment is equipped around the RPV. The containment vessel will ensure the flooding of the reactor core without any emergency cooling actions in case of a very unlikely failure of the RPV. A secondary concrete containment is adopted to mitigate the mental fear of some resident on the nuclear energy.

2.2 Safety concepts of the NHR-200

The design of the NHR-200 insures that it is operated under the low temperature, low pressure, low power density and low radioactivity content in primary coolant. The main safety features can be briefly summarized as follows[3].

a) A number of passive safety measures. That the core is always covered by coolant is one of the fundamental design criteria for the NHR-200. This result is caused by the follow measures, integrated design to exclude the possibility of large LOCA, on the upper parts of the vessel and small bore of all penetrations limits the loss of coolant, dual vessel design and huge subcooled water inventory. In addition, except full power natural circulation cooling, a passive pattern used in the residual heat remove system, hydraulic driving system for control rods and dual vessel design, all of these above mentioned, low power density and large negative feedback as well as quite large margin of DNBR in the accident condition can maintain the integrity of fuel cladding at any transients and accidents. The results of accident analysis for the NHR-200 indicates that the accident of ϕ50 penetration pipe break outside the steel containment followed by failure of isolation is the most serious LOCA accident. The results of LOCA analysis for the NHR-200 are listed in Table II. The main results of the safety analysis based on any design basis accidents can be summarized as follows. DNBRmin is always greater than safe limit. Peak pressure in primary system is far below its design pressure and the integrity of coolant pressure boundary will be maintained properly. The core will never be uncovered. The maximum fuel enthalpy is much lower than safe limit and the release of radioactivity is much less than the prescribed limit.

b) The effective isolating heating grid or fresh water from radioactivity. The multi-barriers including fuel cladding, reactor pressure boundary, steel containment and secondary containment compose the multi-defenses against the release of radioactive substance. In addition, the nuclear steam supply system (NSSS) is composed by triple loops, i.e. the primary circuit in the reactor pressure vessel, a intermediate circuit and the heating grid or the steam circuit. The working pressure in the intermediate circuit is higher than that in the heating grid or in the steam circuit, so the pollution of radioactivity can be prevented from and the safety of the heating grid or the produced water can be ensured.

Table II Results of LOCA analysis for NHR-200

Events*	Accident circumstances		
	Ultimate pressure in the reactor pressure vessel MPa	Amount of water lost t	Amount of water remained above the core t
1	1.68	23.6	69.3
2	~0.12	45.3	47.6
3	1.04	34.2	58.7
4	0.60	8.25	84.65
5	1.70	11.4	81.5

* Event 1 ϕ50 pipe break inside the steel containment
Event 2 ϕ50 pipe break outside the steel containment followed by failure of isolation
Event 3 Small crack ($\sim 1\,cm^2$) at the bottom of the RPV
Event 4 Safety valve stuck open
Event 5 ATWS initiated by loss of off-site power and safety valve not reclosing

3 Seawater desalination with NHR-200

3.1 Option of desalination process

The excellent performance of the NHR-200 shows that it is suitable to the coupling with a seawater desalination plant. Among the various existing desalination processes worldwide, the following have been selected for the present study as the most interesting for nuclear desalination: reverse osmosis (RO), multi-effect distillation with vapour compression (MED/VC), multi-effect distillation (MED), and multi-stage flash distillation (MSF). And all of these are proven by experience. Among them, MED and MSF may come into consideration to couple with NHR-200. The energy input is mainly in the form of low temperature heat and some electricity for the two distillation processes. But the energy consumption of MSF is grater than MED.

Based on an evaluation of the technological status of the two distillation processes, INET selects MED process for desalination study using NHR-200 or other scale NHR.

In order to match different user's requirements, several scales of the NHR desalination plant are included in the INET's study program, including NHR-200 desalination plant, NHR-10 desalination plant and NHR-5 desalination plant. The pre-feasibility study on the NHR-10 desalination plant in Morocco is in progress.

3.2 Coupling of desalination plant with NHR-200

The main parameters of NHR-200 perfectly match the requirements on the heat source for seawater desalination processes. Based on the study of the technological features of NHR-200 and the MED process, the selecting principles of the coupling plan of desalination plant with NHR-200 are realizable in technology, reliable in safety and optimum in economy. Two kind of interface design between NHR-200 and MED plant was conducted, one is to produce water only and another is to co-generate water and electricity. The choice among these two designs is depended mainly on site features, the design of water production only can gain a larger gain-output ratio (GOR), and the design of water and electricity co-generation can satisfy the requirement for electricity consumption of the NHR-200 and of the MED plant.

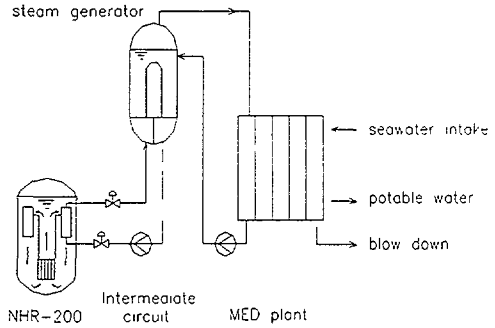

Fig 2 Simplified system diagram of water production only

According to the analysis on the 8 coupling plans[4] for the water production only, the best one is: steam generator + MED plant. The system diagram of this coupling design is shown in Fig.2 and main parameters are listed in Table III. Within this design, the low pressure steam generated in the intermediate circuit steam generator will be directly introduced to the MED plant. A GOR of 20.76 is designed and the daily fresh water production will be 165,052m^3.

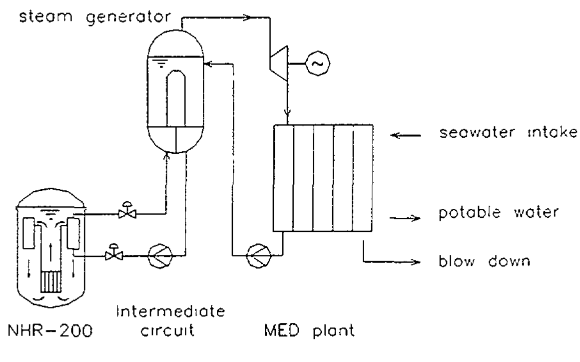

Fig 3 Simplified system diagram of co-generation

The purpose of the heat/electricity co-generation design is to ensure the supply of fresh water and not to intensify the shortage of electricity supply. The basic demand for this purpose is that the generated electricity shall be used for the self-consumption both for the reactor and for the desalination process, and that the output of fresh water shall be grater than 120,000m^3/d. After comparing the other 8 coupling plans for the cogeneration[4], the selected optimum interface is: steam generator + turbine system + MED plant. Fig.3 shows the system diagram of this interface and key design parameters are listed in Table 3. The steam will first be used for electricity generation. And then, the steam extracted from the last stage of the turbine with lower temperature and pressure will go to the MED system. The maximum fresh water output will be 127,583m^3/d with a GOR of 17.89 and the generated electric power will be 14.42MW.

Table III Main parameters of the two coupling designs

Design parameters		Water only	Co-generation
Reactor thermal power	MW	200	200
Pressure in primary loop	MPa	2.5	2.5
Core inlet/outlet temperature	°C	154/213	154/213
Intermediate loop inlet/outlet temperature	°C	135/163	135/163
Ratio of investment and power of the reactor	$/kW	552.0	552.0
Steam generator outlet steam temperature	°C	130	145
Inlet steam temperature of MED process	°C	130	102
Unit capacity of MED plant	m^3/d	48,000	48,000
Number of unit		4	3
Number of effect		30	23
Gain-output ratio (GOR)		20.76	17.89
Efficiency of the turbine generator	%	—	72.7
Electric power	MW	—	14.42
Self-consumption of electricity	MW	—	13.36
Maximum fresh water production	m^3/d	165,052	127,583
Interest rate	%	8	8
Material cost factor*		0.8	0.8
Price of water	$\$/m^3$	1.188	1.206

* The material cost factor is the ratio of the other two ratios, one is the ratio of investment and produced water using low temperature material and high temperature material respectively and another is using high temperature material only under the condition of the same number of effect in MED process.

3.3 Economic analysis

Several advantages can be gained using a NHR for seawater desalination. Compared with a fossil-fueled plant, a NHR desalination plant may not only save large amounts of valuable coal and oil resources and decrease environment pollution but will also be competitive economically under some certain conditions. Due to the inherent and passive safety features of the NHR itself, the NHR desalination plant can be constructed near big cities and industrial consumers. It would lead to a decrease in the cost of the water pipe network.

On the basis of the data provided in IAEA-TECDOC-666, the water prices both of the water production only and water/electricity cogeneration using NHR-200 have been analyzed with the method of output cost per unit and the method of effective credit. Water storage, transport and distribution facilities are not included. The result indicates that the main part of the water price is constituted by the annual fixed capital charge of water plant and heat charge. The water price listed in Table 3 are slightly higher, but the water price may reduced to $0.93\$/m^3$ for water production only and $0.955\$/m^3$ for cogeneration in case of the 5% interest rate. The most sensitive factor, which has influence on the water price, is the load factor of the water plant and the reactor. In addition, the following factors have the more influence on the water price: output of water, investment and service life of water plant, investment and service life of the reactor. The main influence factors on water price are listed in Table 4. The result of sensitivity analysis on the water price indicates that the water price will be dropped in case of the low interest rate and investment of NHR desalination plant as well as high availability of the plant.

Table IV Main influence factors on water price[5]

Sensitive factors	*Change of water price, %	
	Water production only	Co-generation
Interest rate	0.632	0.559
Electricity price	0.109	0.020
Load factor of NHR-200	−0.710	−0.779
Load factor of MED plant	−0.885	−1.035
Service life of NHR-200 (20~30 year)	−0.092	−0.189
Service life of NHR-200 (30~40 year)	−0.035	−0.041
Service life of MED plant (20~30 year)	−0.201	−0.182
Service life of MED plant (30~40 year)	−0.071	−0.057
Investment of NHR-200	0.208	0.287
Investment of MED plant	0.485	0.401
Amount of water production	−0.387	−0.451
Efficiency of turbine generator	—	−0.068

* The percentage of water price change is calculated at the conditions which assume the sensitive factor increase of 1%, others keeping constant

The scale of the NHR-200 desalination plant is more suitable for the demand of potable water about a hundred thousands m^3/d. Under the suitable condition, several NHR-200s could be combined to supply heat and electricity to a large scale seawater desalination plant for cities and industrial districts with large fresh water requirements. The combined NHR-200s desalination plant can not only ensure the continuity of the water production but also improve the economy by sharing of common facilities and service systems including infrastructure, maintenance facilities, reduction of staffs and so on.

4 Conclusion

The safety is a very important factor into the nuclear seawater desalination in the fields of the safety of the reactor itself and the pollution prevention of the produced water. The NHR-200 is designed with a number of advanced and inherent safety features, which have been proved by the successful operation and the results of special experiments carried on the NHR-5. And the technical measures adopted in the NHR-200, such as duel vessel, secondary containment and intermediate circuit, can guarantee the produced water against radioactivity. The NHR-200 can be considered as the new generation nuclear reactor.

The large nuclear power plant may produce cheaper fresh water due to its lower heating price at the present. But the long distance transport would lead to an increase in the cost of water. Due to the excellent performance of the NHR-200, the MED desalination plant coupled with NHR-200 can be located in the vicinity of the consumer, which will bring about reduction of the water price. And the NHR-200 desalination plant or other scale NHR desalination plant may be more suitable for developing countries in respect of its smaller scale, simpler system, easier component manufacture, maintenance and operation.

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TRANSIENT BEHAVIOUR AND COUPLING ASPECTS OF A HYBRID MSF-RO NUCLEAR DESALINATION PLANT

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Abstract

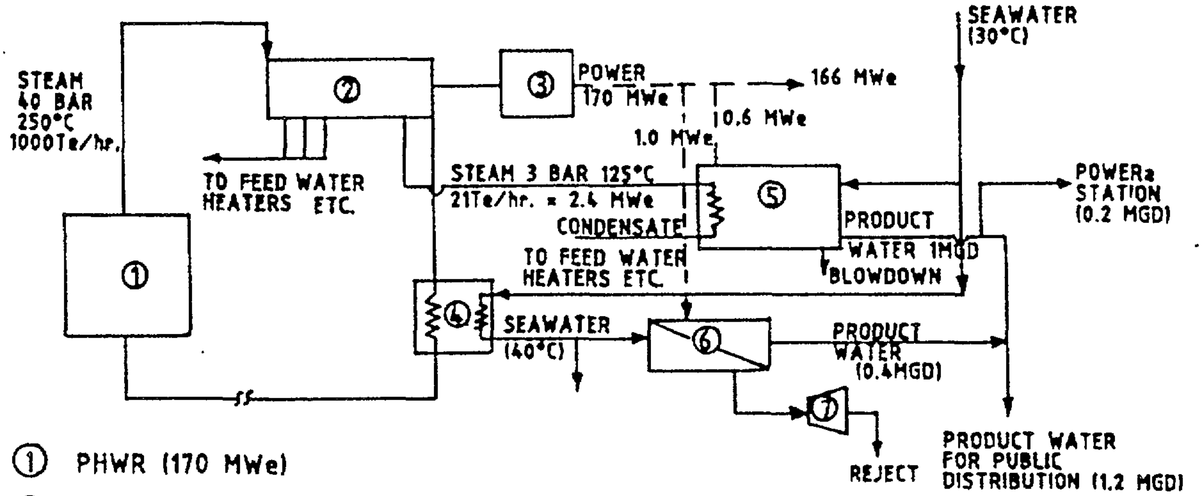
BARC is setting up a 6300 M^3/day (1.4 MGD) hybrid MSF-RO nuclear desalination plant for sea water desalination at Madras Atomic Power Station (MAPS) coupled to a 170 MWe Pressurised Heavy Water Reactor (PHWR). The transient behaviour and coupling aspects of this dual purpose plant has been discussed. A hybrid desalination plant appears to offer high availability factor.

1. INTRODUCTION

Desalination Division at Bhabha Atomic Research Centre (BARC) has been engaged in R&D work on desalination since last several years. It has a fairly advanced ongoing R&D programme on both thermal and membrane desalination. Desalination technologies based on membrane and thermal processes for brackish and seawater desalination have been successfully developed. Multistage Flash (MSF) desalination technology developed by us uses low pressure steam for seawater evaporation. It has been designed keeping in view the higher fuel costs in India. This requires achieving a high gained output ratio (GOR), more number of stages and a higher top brine temperature. It uses less costly material of construction resulting into lower capital investment and water cost compared to international costs. The Reverse Osmosis technology developed at the Centre has been used for providing drinking water in the rural areas of the country. It has also been demonstrated for the effluent treatment and water reuse including production of process water in conjunction with demineralizer.

2. HYBRID DESALINATION

Utilizing design and operation experience of pilot plants, BARC is setting up a 6300 M^3/day (1.4 MGD) hybrid MSF-RO nuclear desalination plant for seawater desalination at Madras Atomic Power Station (MAPS) coupled to a 170 MWe Pressurised Heavy Water Reactor (PHWR) (Fig. 1). The steam required for the 4500 M^3/d MSF is extracted from a tapping point in the low pressure turbine of a power station. A small part of the power generated in the power station is used for operating the 1800 M3/d RO plant and for pumping requirements of MSF plant. The hybrid MSF-RO plant set up at same location aims towards reducing the operation & maintenance (O & M) cost of desalted water by taking the advantage of producing both process and drinking quality water, utilization of water for different applications, having a common sea water intake and outfall systems, common pretreatment to a considerable extent and possibility of using reject streams from one plant to the other. A part of the desalted water (20 ppm TDS) from MSF is used for the makeup requirement. The remaining is blended with the product water of RO plant (500 ppm TDS) and transported for domestic use. The RO plant uses cooling water from the condenser as feed which is at about 6-8^{0}C higher than the ambient temperature. The high temperature operation of RO gives high throughput. It is also possible to use cooling sea water from the reject stage of the MSF plant as feed to RO plant.



① PHWR (170 MWe)

② TURBINE

③ GENERATOR

④ CONDENSER

⑤ MSF PLANT (1.0 MGD)

⑥ SWRO PLANT (0.4 MGD)

⑦ ENERGY RECOVERY DEVICE (ERD)

NOTE:
1. TOTAL ELECTRICAL POWER LOSS INCLUDING POWER LOSS DUE TO THERMAL ENERGY i.e. 3 BAR STEAM DUE TO 1.4 MGD MSF-SWRO WILL BE 4.0 MWe ONLY.
2. SEAWATER REQUIREMENT FOR DESALINATION PLANT IS ABOUT 1 - 2% OF POWER PLANT.

FIG. 1 SCHEMATIC FLOW DIAGRAM OF A 1.4 MGD (6300M^3/DAY) MSF-SWRO HYBRID DESALINATION PLANT COUPLED TO A COASTAL 170 MWe NUCLEAR POWER STATION

TABLE I SALIENT FEATURES OF PHWR

Fuel	Uranium dioxide (UO_2)
Refuelling	Onload
Moderator	
Type	Heavy water (D_2O)
Pressure (MPa)	0.75
Inlet temperature (^{0}C)	44
Outlet temperature (^{0}C)	65
Primary system	
Type	Heavy water (D_2O)
Pressure (MPa)	8.7
Inlet temperature (^{0}C)	249
Outlet temperature (^{0}C)	293
Secondary system	
Coolant	Water/steam
Pressure (MPa)	4.0
Temperature (^{0}C)	250
Heat rejection system	
Coolant	Sea water
Pressure (MPa)	0.00863
Inlet temperature (^{0}C)	30
Outlet temperature (^{0}C)	40

3. COUPLING ASPECTS

When a nuclear reactor is to be used to supply steam for the desalination, the method of coupling has a profound technoeconomic impact. The selection of particular method of coupling depends on the size & type of the nuclear reactor, the characteristics of desalination process and the local factors such as water and power demand.

Where the steam producing capacity of the nuclear reactor is large compared to desalting capacity or when the steam demand is less, a dual purpose plant using extraction steam for desalination and generating power is an ideal choice.

PHWR provides a safer steam generation due to an additional barrier. It uses heavy water (D_2O) as primary coolant and demineralised (DM) water as the secondary coolant. The steam required for MSF is extracted by coupling to the low pressure turbine. The steam is tapped from a suitable point on the low pressure turbine for heating the brine in the MSF plant. The selection of tapping point depends on the maximum brine temperature selected. The arrangement is also made to tap the steam from both the reactor systems for MSF to ensure the continuous operation of the desalination plant. In case of shutdown of a reactor, the steam requirement is met from the other one. Steam line from both the reactors will be connected to a header. Provision is made to regulate the steam supply as well as to isolate the system as and when required. The condensate from the nuclear power station is returned back to MAPS at an appropriate point. A part of the power generated in the nuclear power station is used for operating seawater RO plant by coupling it thermally and electrically with the power plant. The thermal coupling gives higher throughput due to higher feed sea water temperature upto 40^0C.. The remaining power is used for distribution. The pressure of the brine in the brineheater of MSF is maintained higher than the heating system pressure thereby reducing the contamination of brine by radioactive carryover. The quality of condensate is continuously monitored at the outlet from the brineheater. The probability of radioactive contamination of desalted water is extremely low.

The total electrical loss including power loss due to thermal energy i.e. 3 bar steam for 6300 M^3/d hybrid MSF-RO plant is around 4.0 MWe only. Rest of the 166 MWe electrical power is released for distribution. Sea water requirement for desalination is about 1-2% of the power plant.

4. TRANSIENT BEHAVIOUR

A dual purpose nuclear desalination plant consists of three interacting systems - nuclear steam supply system, the turbine generator system and the desalination unit.

The transients are caused due to daily and seasonal load variation. The extent of variation differs from place to place. The potential difficulty in the operation of a dual purpose thermal desalination plant is the dependence of steam flow and thereby desalted water production on the electricity demand. The easiest solution of the problem is achieved by operating the dual purpose plant at full load for most of the time supplying electricity to power grid and water to water grid. In many cases, this solution is feasible, if not, a certain flexibility is required by arranging a steam bypass around the turbine with a pressure regulating valve and a desuperheater. The bypass valve is opened when the turbine steam flow is not sufficient for the stable operation of MSF plant. The MSF does not respond very well to sudden load changes. The main difficulty by sudden load changes in MSF is in the control of brine level in the flash stages by regulating the brine flow and change of area of interstage orifices. There is no difficulty in ensuring the stable operation of desalination plant between 70-110% of full rated capacity with slow change in load. It is difficult to operate MSF plant if there is sudden large reduction of steam flow rate because the brine flow to the flash stages and tubeside brine velocity decreases below the admissible limit. In large MSF plants, this difficulty may be overcome by adopting an operating system in which MSF plant consists of several parallel trains of flash evaporators. In case of sudden large reduction of steam flow rate, some of the trains are switched off to have stable operation of MSF plant.

With the lowering of extraction pressure of the steam at constant steam flow rate, the top brine temperature in MSF comes down giving reduced GOR leading to lower production rate. A steam bypass around the turbine is provided which is opened in order to achieve the top brine temperature. A sea water bypass line around the evaporator is provided to use the brine heater of the MSF plant as a dump condenser for the turbine steam when the MSF plant is temporarily shut down.

In case of a desalination plant using only electricity such as RO, the coupling of the energy source with desalination plant is simple. There is not much of mutual influences due to transients in the nuclear power plant. If the coupling has low availability factor, a RO system coupled to a nuclear power plant has a high availability. On the other hand, a thermal desalination system coupled nuclear power plant gives high overall availability if the coupling has a high availability factor. A hybrid MSF-RO desalination plant coupled to a nuclear power plant appears to give high availability factor in general.

5. CONCLUSION

Transient behaviour and coupling aspects may have some technical fallout but they are like any other technical problems in a plant and solution exists. For the tropical countries as India, the prime need is the power generation at present. Small and medium size desalination plants are required in land and coastal arid zones for process water and safe drinking water requirement unlike Gulf countries where the desalted water demand is high and more important than power requiring high desalting capacity.

II.2. OPERATIONAL AND MATERIAL ASPECTS OF NUCLEAR HEAT APPLICATION

Materials and equipment for the coupling interfaces of nuclear reactors with desalination and district heating plants

EQUIPMENT AND MATERIALS FOR COUPLING INTERFACES OF A NUCLEAR REACTOR WITH DESALINATION AND HEATING PLANTS BASED ON FLOATING NHPS

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Abstract

Intensive design activity is currently underway in Russia on floating nuclear installations, relying on proven marine NSSSs of KLT-40-type, which are capable of generating electricity, producing potable water and heat for industrial and district heating purposes. In particular, design work of the first floating power unit for a pilot nuclear co-generation station, which is due to be situated at the Pevek port area in the Chukotsky national district (extreme north-east of Russia), is approaching completion, and preparatory work is being carried out for fabrication of its most labour-intensive components.

Work is also in progress together with "CANDESAL Inc.(Canada)"on the conceptual design of a floating power-desalination complex. Most suitable options of floating power-desalination complexes are being sought, addressing requirements of potential customers.

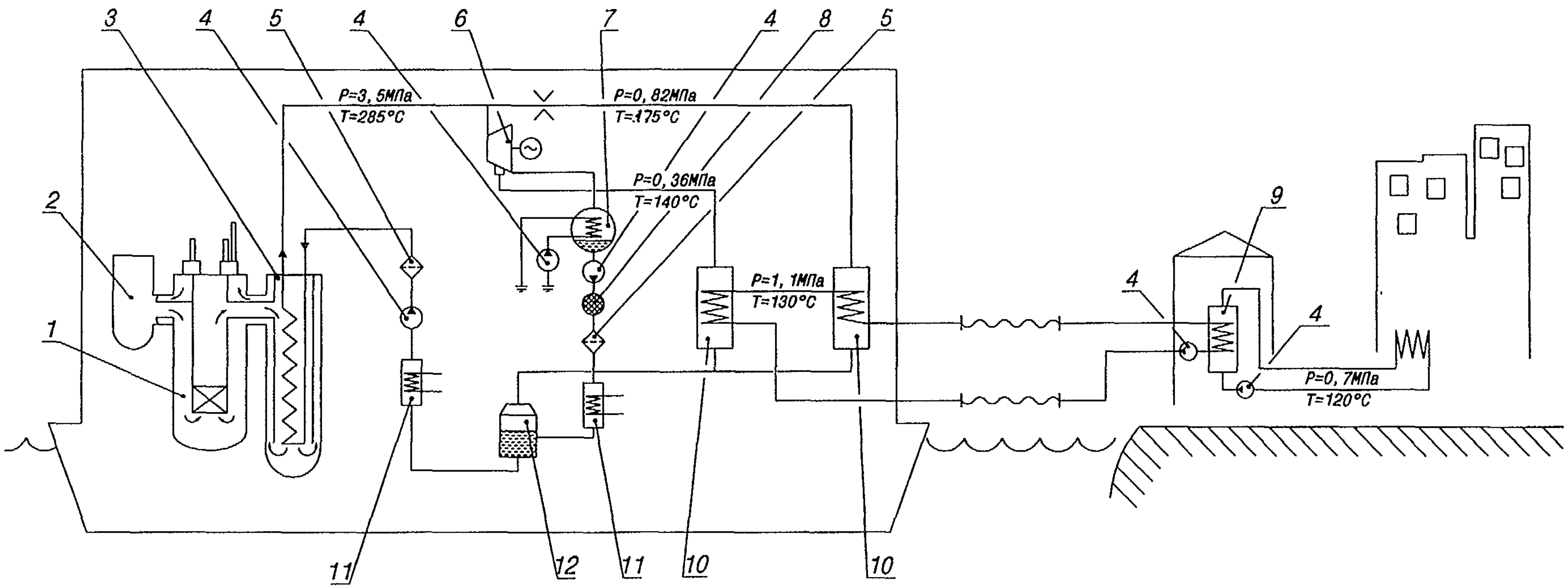
Earlier, at the IAEA technical committee meeting (1993) it was shown that a complex, which combines a highly effective condensation turbine and a modern reverse-osmosis desalination facility, could be considered as most preferable from the view point of efficient utilisation of thermal energy generated by nuclear reactors for co-production of potable water and electricity. The prospective technology for sea water desalination by a reverse-osmosis method is being developed in particular by "CANDESAL Inc." It was also pointed out that another sufficiently efficient installation for potable water and electricity co-production is a dual-purpose complex which integrates both condensation and back-pressure turbines and a distillation desalination facility. Similar flow configurations were adopted for the nuclear desalination complex at Aktau (Kazakhstan) which has been in operation since 1972. "SverdNIIKhimMash" institute (Ekaterinburg) is a Russian leading designer of modern distillation desalination facilities.

This paper presents heat and fluid diagrams of floating complexes, brief description of their key components, used materials, radiological safety provision and instrumentation.

1. NUCLEAR CO-GENERATION STATION WITH FLOATING POWER UNIT FOR REMOTE AND DEVELOPING REGIONS OF RUSSIA

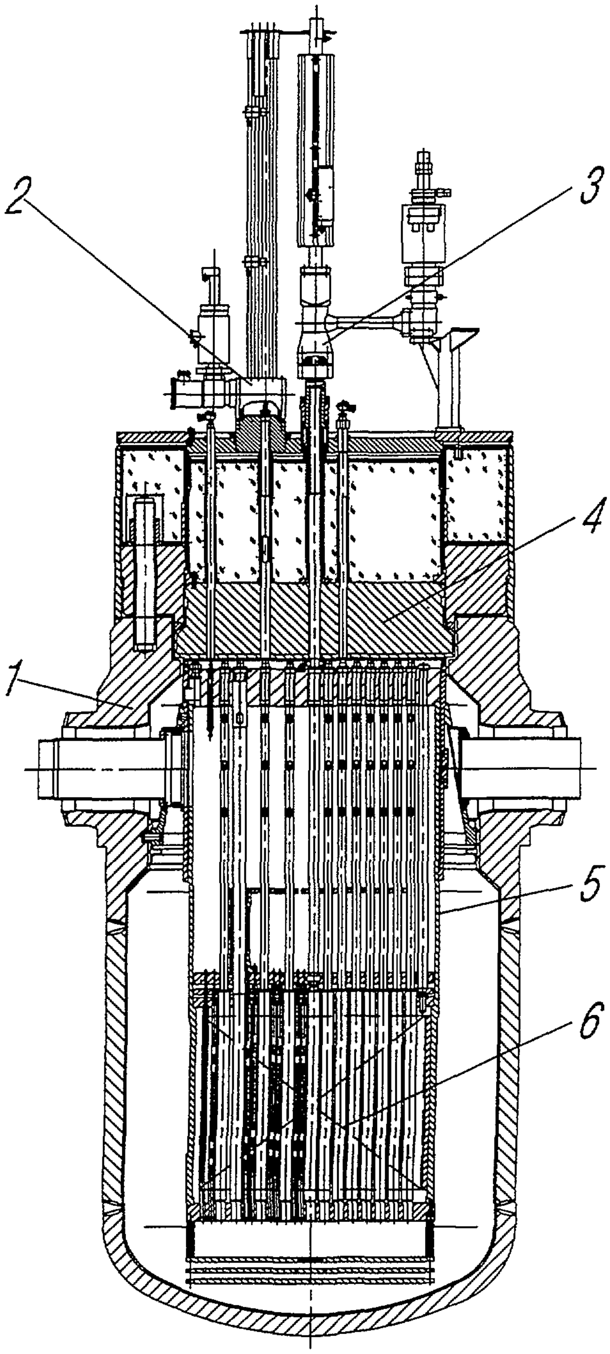
Extreme North and similar remote regions account for more than 50% of Russia's territory. Abundant mineral resources are found out and being exploited there. However, there are no sufficient local fuel and energy resources as a rule at mining and other mineral production areas while delivery of fossil fuel calls for great expenses. So, the use of nuclear heat and power stations (NHPS), especially floating ones, becomes expedient under these specific conditions. At present a design of a floating power unit with a KLT-40C reactor plant is nearing completion in Russia, and preparatory work for production of its equipment with extended fabrication cycle is underway. Area of the port Pevek in the Chukotsky national district is selected as a site for the pilot NHPS [1].

The floating NHPS includes two independent power units, each consisting of a reactor, a steam turbine and electric power plants. Turbogenerators have steam extraction bleed-offs to heat up feedwater



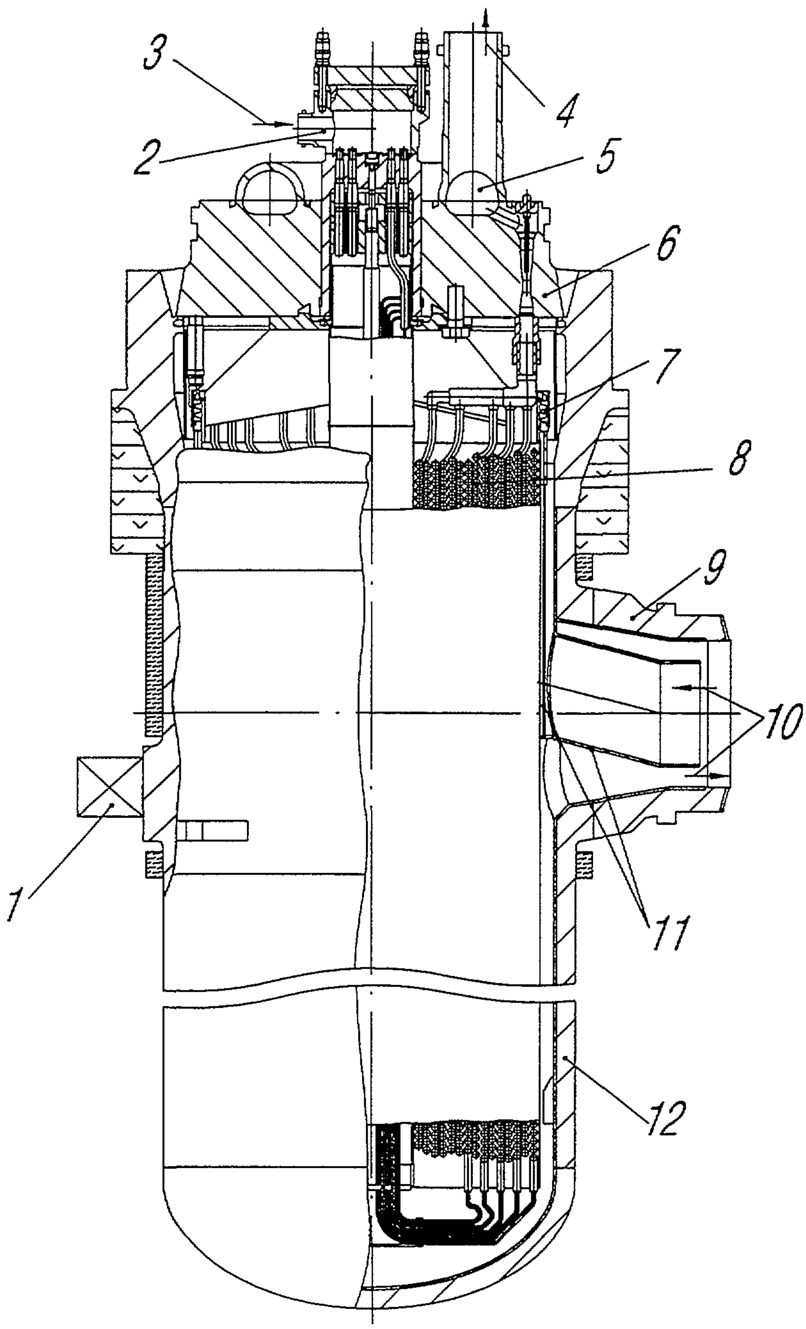
1 - reactor; 2 - reactor coolant pump; 3 - steam generator; 4 - pump; 5 - mechanical filter; 6 - turbo - generator; 7 - condenser; 8 - ion - exchange filter; 9 - district heating system heater; 10 - intermediate circuit heater; 11 - feed water heater; 12 - deaerator

Fig.1. HPNS Principle Flow Diagram



1 - vessel; 2 - EP drive mechanism;
3 - CG drive mechanism;
4 - cover; 5 - removable block; 6 - core

Fig. 2. Reactor



1 - support; 2 - feed water collector;
3 - feed water; 4 - steam; 5 - steam collector;
6 - cover; 7 - bellows - type seal; 8 - tube coil;
9 - "tube - in - tube" nozzle; 10 - primary water;
11 - iln - vessel baffles; 12 - vessel

Fig. 3. Steam Generator

Table I Basic technical characteristics of NHPS floating power unit

Characteristic	Value
1. Number of reactor plants	2
2. Type of reactor plant	KLT-40
3. Thermal power, MW	2x148
4. Turbogenerator plant electric power, MW	2x35
5. Electric output to network, MW	2x30
6. Heat output to heating system, GCal/h	2x25

and intermediate circuit's water, which is then supplied to a coastal heating system. Equipment items used in the NHPSs are based on established fabrication technology and verified by long-term operation experience.

The plant fluid diagram is presented in Fig. 1. Basic technical characteristics of the floating power unit are given in Table I. A PWR-type reactor (Fig. 2) is used. Heat-resistant high strength pearlitic steel with anti-corrosion weld cladding is used as the reactor vessel material. A shielding layer is located between the vessel and the core for limiting the integral neutron fluence ($E > 0.5$ MeV) on the vessel to the value of $2x10^{20}$ cm^{-2}. Leak tightness of the reactor vessel-to-cover joint is ensured by a self-sealed copper wedge gasket.

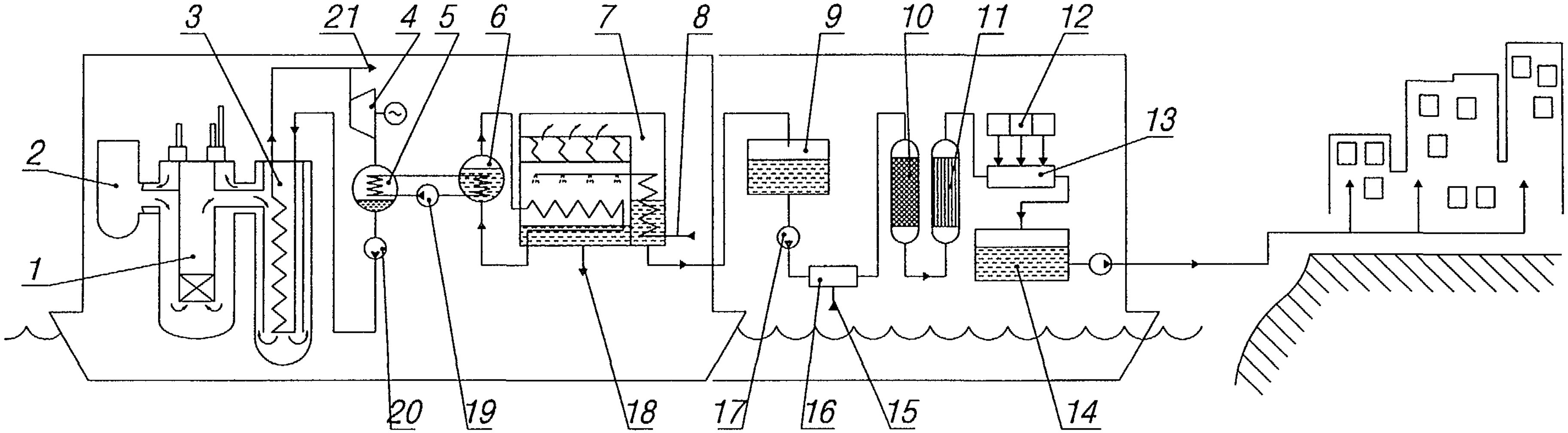
The reactor houses the core and reactivity control members. A radiation resistant material with high neutron-absorption capability is used in the reactivity control members. The core has a heterogeneous channel-type structure consisting of fuel assemblies (FAs) and reactivity control means. FAs include fuel elements and burnable poison rods containing gadolinium. To facilitate the reactor start-up control, some FAs contain a beryllium oxide.

The steam generator is of a once-through type (Fig. 3). Steam is generated inside the tubes, whereas the primary coolant flows outside the tubes. The SG shell is fabricated from low-alloyed steel lined by an anti-corrosion weld cladding. The SG tube system represents a set of cylindrical spatial spiral coils made from titanium alloy and united in sections independent as regards feedwater supply and steam removal. PG-7M and PG-3V titanium alloys are used as structural materials for the tube system. To connect items made from different materials (i.e. titanium alloy and stainless steel), threaded-soldered joints are used.

The steam turbine represents an active, single-cylinder, one-flow condensation-type one with controlled and non-controlled steam bleed-offs for heating feedwater and intermediate circuit's water. A flow part of the turbine consists of 13 stages. A steam bleed-off chamber divides the turbine into high and low pressure parts.

The NHPS's heating system consists of the turbine's steam bleed-offs, and the intermediate circuit with heaters. Part of heaters receive steam from the turbine intermediate bleed-offs, while other peak heaters receive live steam from steam generators. The pressure in the intermediate circuit exceeds that of the heating fluid in the heaters. Therefore, at the loss-of-integrity event in the heaters' tube system, the intermediate circuit's coolant flows back into the steam circuit, thus maintaining a barrier against radioactive carryover.

The district heating grid with heat exchangers and pumps is situated on a coast and connected to the floating NHPS by flexible pipelines. The heaters represent shroud-tube HXs. Carbon and stainless steels are used as structural materials for pipelines and intermediate circuit's heaters. Flexible pipelines are made as multilayer bellows with polymeric coating.



1 - reactor; 2 - reactor coolant pump; 3 - steam generator; 4 - turbo - generator; 5 - condenser; 6 - steam generator; 7 - distillation desalination plant; 8 - sea water inlet; 9 - distillate intake tank; 10 - water enrichment facility; 11 - running water sorbent containing filter; 12 - plant for water fluorine, chlorine treatment and stabilization; 13 - mixer; 14 - potable water tank; 15 - H_2CO_3 solution; 16 - mixer; 17 - potable water preparation plant pump; 18 - evaporated sea water brine; 19 - intermediate circuit electric pump; 20 - secondary circuit electric pump; 21 - to condensation turbine.

Fig.4. Station Principal flow diagram

As operation experience of marine NSSSs shows, reliable operation of equipment significantly depends on quality of coolants circulating in NSSS's circuits. Ingress of sea water into circuits operating with distillate is especially dangerous occurrence. Secondary steam's activity is monitored by gamma-sensors, installed on steam pipelines. Volumetric β-activity of steam is measured in steam pipelines. In addition, volumetric β-activity of steam is measured at ejectors outlet and in the deaerator with sensitivity of $1x10^{-11}$ Ci/litre. Minimum detectable leak into the secondary circuit from the primary circuit is 0.5 kg/h.

2. FLOATING NUCLEAR WATER DESALINATION STATION WITH DISTILLATION DESALINATION PLANT

At present, sea water desalination is one of perspective lines in nuclear energy utilisation. Acute shortage of fresh water in many regions of the world is an incentive factor here.

The station represents a barge on which the KLT-40-type reactor plant, the steam-turbine plant and the desalination facility with the potable water preparation block are mounted (Fig. 4) [2]. Key characteristics of the station are given in Table II.

Table II Key characteristics of nuclear station with distillate desalination plant

Characteristics	Value
1. Thermal power of reactor, MW	2x148
2. Potable water capacity, m^3/day	up to 80,000*
3. Number of desalination facilities, pieces	4
4. Electric power supplied to coastal grid, MW	up to 40

* The capacity is determined by requirements of potential customer

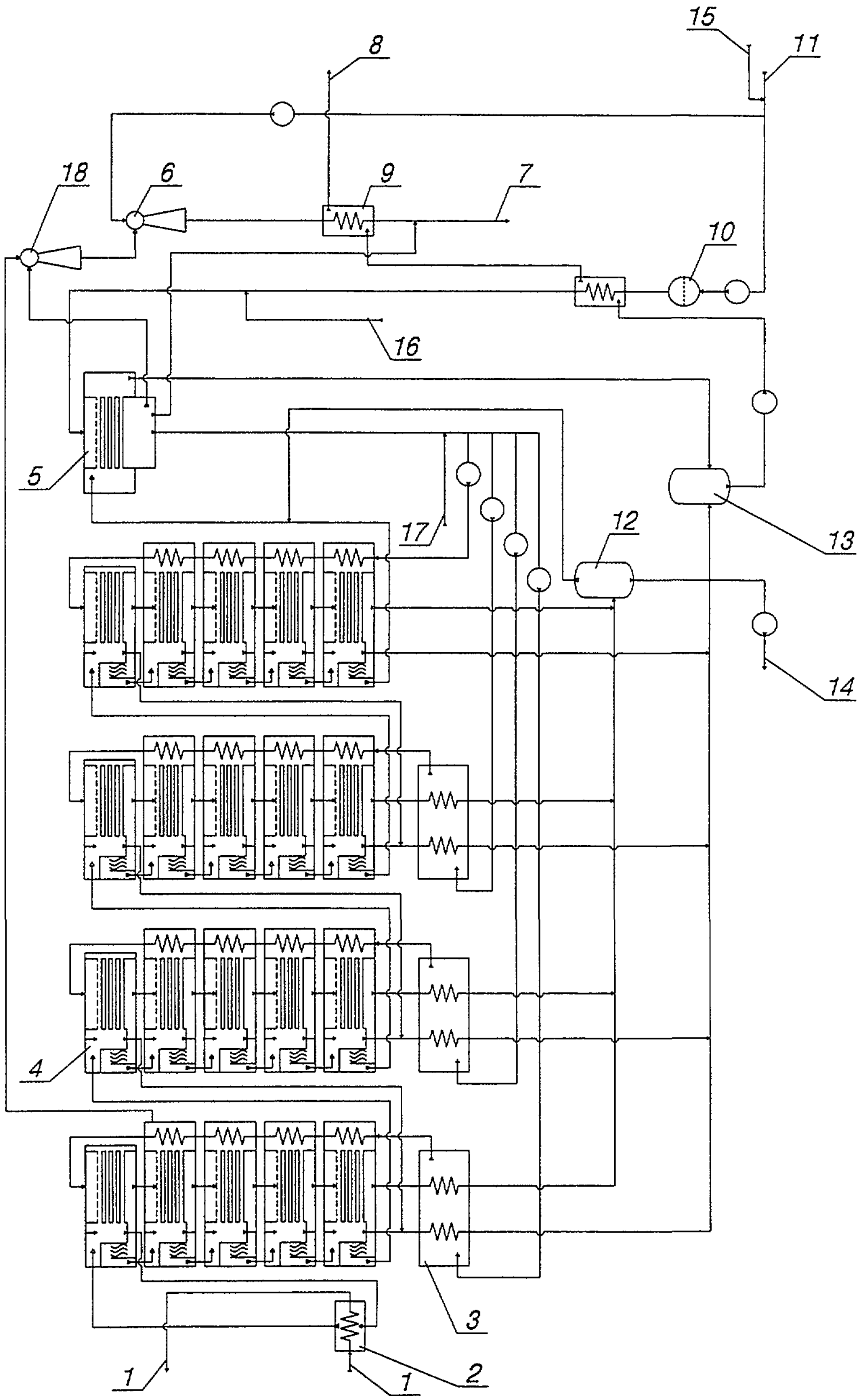
The reactor plant generates superheated steam in steam generators, from which the steam is supplied to a condenser and back-pressure turbines. Heat from the back-pressure turbine's condenser is transferred in a desalination facility's steam generator through an intermediate circuit. Coolant pressure in the intermediate circuit exceeds the pressure in the secondary circuit, thus excluding ingress of radioactive substances into the intermediate circuit and the distillation facility at the loss-of-integrity in the heat exchangers.

A condenser is made of two individual sections separated on cooling water supply, which are joined by a common steam exhaust nozzle and by communicated condensate collectors. The condenser's cooling water path, inlet and outlet nozzles, water chambers, tube sheets and cooling tubes are made from titanium alloy. Its shell is fabricated from a conventional steel, the condensate collector is made from a stainless steel. Steel shell-to-titanium tube sheet joint is sealed by a gasket.

2.1. Distillation desalination facility

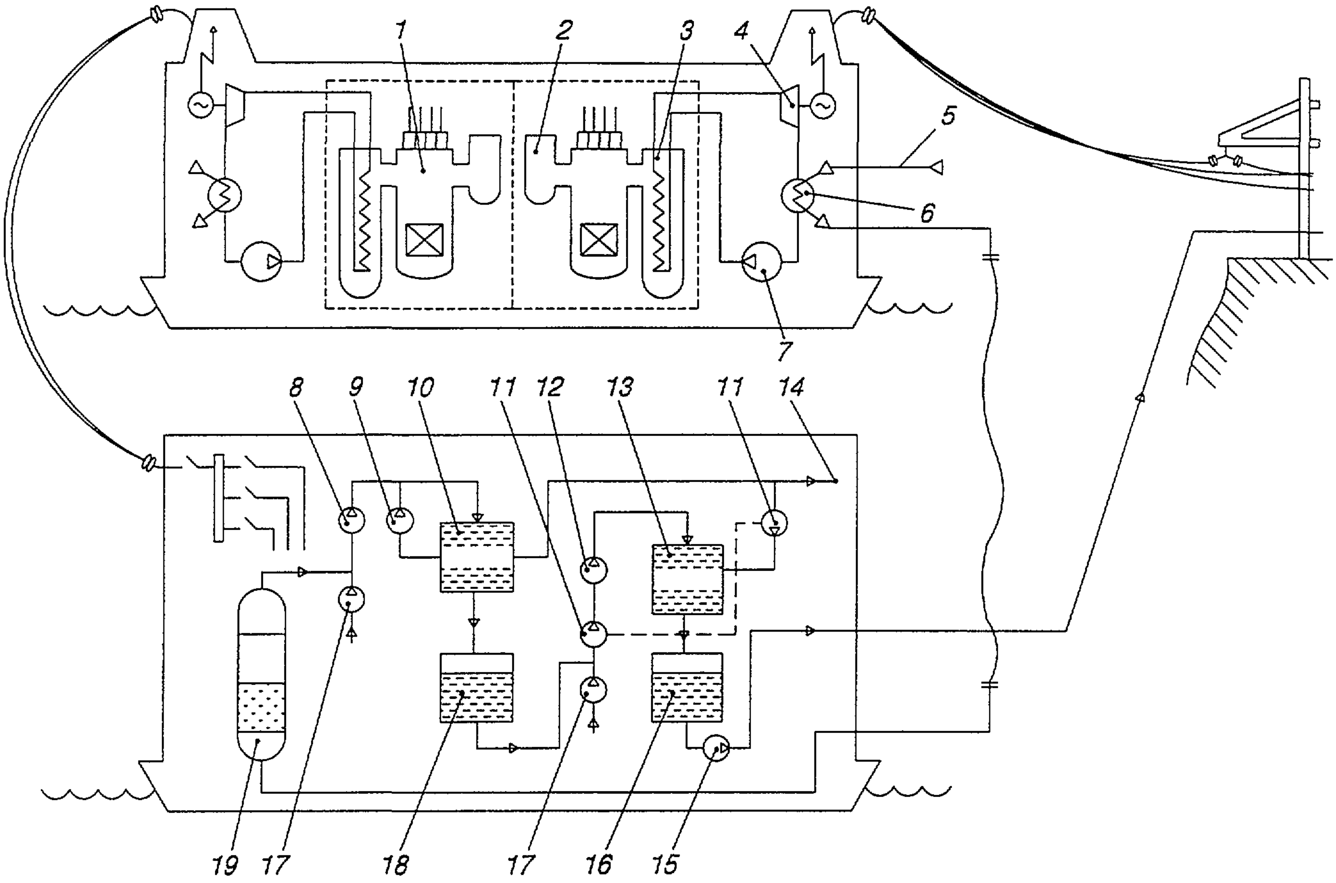
Potable water production by distillation desalination facilities prevails now in its world wide production. Desalination facilities with horizontal-tube film evaporators are considered as the best ones in terms of technical and economic performances and prospects of producing relatively cheaper potable water [3]. An example of these distillation systems is being operated at the nuclear powered desalination complex in Aktau (Kazakhstan).

The flow diagram of a distillate desalination plant is given in Fig. 5. The desalination facility DOU GTP-20000 consists of a steam generator, 20 evaporation stages, three regenerative heat exchangers, a deaerator, a water-jet ejector, a sea water heater, a distillate cooler, a antiscale reagent injection unit, sea water filters, a sea water self evaporator tank, a distillate collector tank, centrifugal pumps for sea water supply, brine and distillate removal pumps.



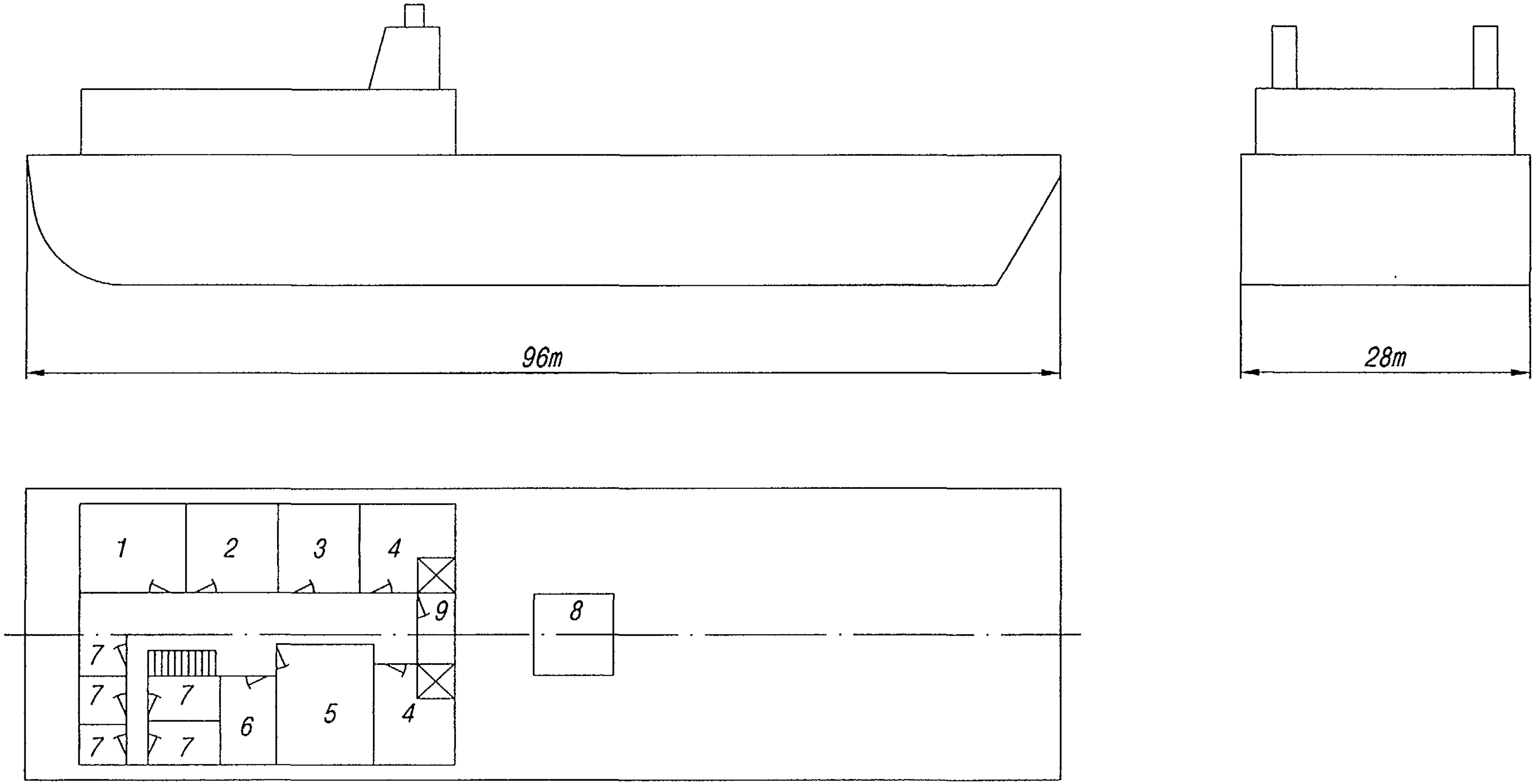
1 - coolant; 2 - steam generator; 3 - preheater; 4 - evaporation stage; 5 - deaerator; 6 - water - jet ejector; 7 - sea water; 8 - distillate to the consumer; 9 - distillate cooler; 10 - filter; 11 - sea water; 12 - sea water concentrate tank; 13 - distillate tank; 14 - sea water with increased salt concentration; 15 - chlorination; 16 - anti - fouling additive inlet; 17 - sodium sulfite inlet; 18 - gas - jet ejector

Fig 5. Desalination plant flow diagram



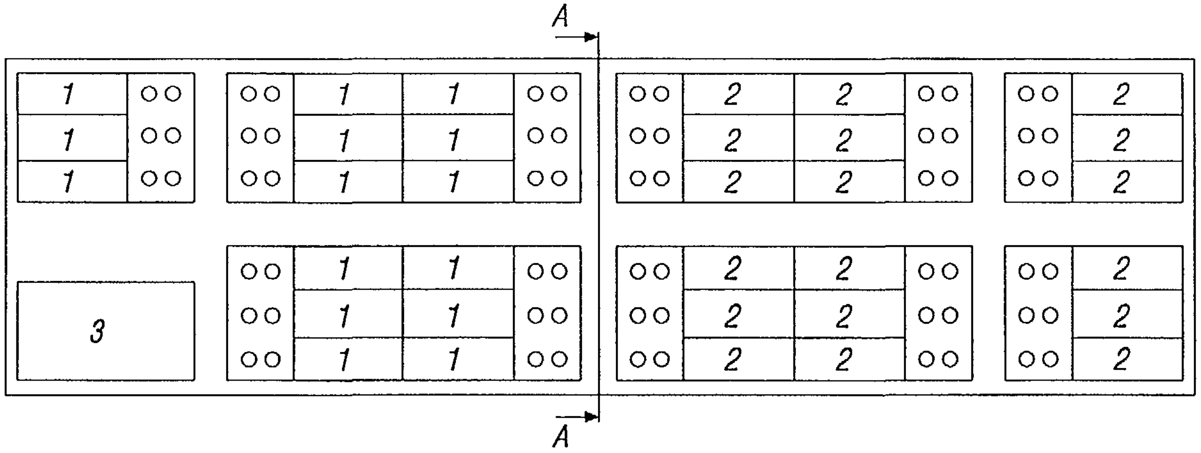
1 - reactor; 2 - primary circuit circulation pump; 3 - steam generator; 4 - turbo - generator; 5 - sea water; 6 - condenser; 7 - secondary circuit electric pump; 8 - medium pressure pump; 9 - recycle pump; 10 - ultra - filtration membranes; 11 - energy recovery system; 12 - high pressure pump; 13 - R.O. membranes; 14 - outfall structure; 15 - potable water pump; 16 - potable water storage tank; 17 - anti - salant injection system; 18 - clarified water tank; 19 - prefilter

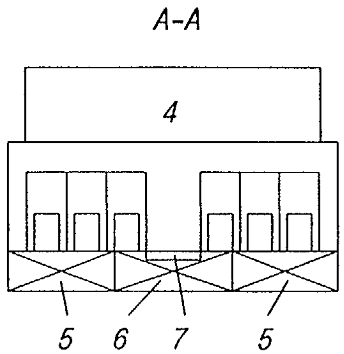
Fig.6 Principle Flow Diagram of the complex



1 - storeroom; 2 - R.O. control room; 3 - galley; 4 - D/G Compartment; 5- recreation lounge; 6 - masters cabin; 7 - crew cabin; 8 - R.O. compartment access hatch; 9 - switchboard

Fig.7. R.O. Desalination Barge





1 - reverse osmosis module (15 off);
2 - ultra - filtration module (18 off);
3 - injection station and service modules;
4 - accommodation area;
5 - ultra - filtration capacitance tanks;
6 - R.O. output tanks;
7 - piping well

Fig.7. R.O. Desalination Barge (cont'd)

The desalination plant's SG is an evaporating-type heat exchanger, in which hot water is driven within the tubes, while the distillate flows in an intertube space. Steam generated in intertube space is liberated from moisture drops in a separation space of the evaporator and directed to a heating chamber of the first horizontal-tube film evaporator. Steam condensate is returned from the horizontal-tube film apparatus into the steam generator.

An evaporation stage includes two main functional units: evaporation and heating sections. The evaporation section is arranged from two horizontal tube bundles, central and two end steam chambers. Horizontal shroud-tube heat exchangers-heaters, in which heat is exchanged between steam which is blown-off from the evaporation section's tube bundles and sea water, are attached to the end steam chambers. Protection of the heat exchange surfaces from scale deposits is provided by antiscale material PAF-13A which is injected into sea water, and by sulfamine acid which partially reduces alkalinity.

The regenerative heat exchanger is an apparatus where heat exchange is effected between distillate, brine and input sea water supplied to an evaporator column. Here two non-mixed flows (i.e. distillate and brine) move in an inter-tube space, while two parallel flows of sea water move within tubes. The deaerator is a heat-and-mass exchange apparatus where sea water is deaerated. The deaerator contains two key parts: a horizontal-tube heat exchanger and a deaeration stage located under the heat exchanger-condenser.

The water-jet ejector is intended to create and keep a vacuum within the plant. It removes non-condensable gases from the deaerator and evaporators. Sea water is used as a working fluid here. The water heater is intended to simultaneously heat up input sea water and to cool distillate produced. The heater represents a shell-tube heat exchanger in which sea water flows within tubes, while a distillate flows in an intertube space. The distillate cooler is intended to cool the distillate produced in the facility. The apparatus is a vertical shell-tube heat exchanger, in which cooled distillate flows in an intertube space upwards. Film of cooling fluid flows within tubes downwards.

The following structural materials are used for above mentioned equipment items: aluminium brass with arsenic for heat-exchange tubes in evaporators, copper-nickel or titanium alloys for tubes in heat exchangers and condensers, stainless steel for equipment items contacting sea water, carbon steel for the rest items.

2.2. Potable water preparation unit

A process diagram of the potable water preparation unit (Fig. 4) is based on a filtration technology, which ensures a production of drinkable water of a calcium group of carbonate grade from distillates. Enrichment of distillates by calcium hydrocarbonate is carried out by preliminary injection and solution of blown carbon dioxide in it and by filtration through a calcium-carbonate bed. Sorption purification of enriched distillates is provided also by filtration through a bed of activated coal. Water conditioning is carried out by fluoridation, desinfection and stabilisation. Quality of the potable water meets the standards of the World Health Organisation.

3. FLOATING DUAL-PURPOSE COMPLEX FOR ELECTRICITY AND POTABLE WATER PRODUCTION USING REVERSE-OSMOSIS TECHNOLOGY

Earlier, at the IAEA technical committee meeting (1993) it was shown that a complex, which combines a condensation turbine and a reverse-osmosis desalination facility, could be considered as most preferable from the view point of efficient utilisation of thermal energy generated by nuclear reactors for co-production of potable water and electricity [4].

A conceptual design of a nuclear floating dual-purpose complex with two NSSSs of KLT-40-type and reverse-osmosis desalinators is being carried out now jointly by institutions of MinAtom of

Russia and "CANDESAL Inc." The nuclear floating complex seems an attractive and flexible solution to attain an optimum relation between both potable water and electricity outputs, thus meeting specific needs of any customers. This relation can vary from maximum production of potable water up to maximum electrical power output.

The nuclear dual-purpose complex consists of two floating structures, i.e. a floating nuclear power plant - FNPP (see Section 1) and a ship for potable water production from sea water by the reverse-osmosis method (Fig. 5) [5]. Electricity generated by the FNPP is partially transmitted to the potable water-production ship and the rest of electricity is delivered to coastal consumers.

The system integrates the proven sea water desalination technology and NPP technology in a single complex, where both installations are coupled by electrical and heat links. Electricity energises a reverse-osmosis process. Thermal link is effected by a waste heat which is released in a process of electricity generation. Waste heat is removed by sea water which cools the condenser and is used as a feed water for the reverse-osmosis system. This flow scheme improves an efficiency of sea water purification process by more that 30%.

Design of the reverse-osmosis system developed by "CANDESAL Inc." is based on utilisation of helically coiled membranes of enhanced permeability produced by DonFilmTech. Sea water from the FNPP turbine's condenser at temperature by 10 °C higher than the ambient sea water temperature is used as a feed water for the reverse-osmosis system. Initially water flows through modules for preliminary treatment by ultrafiltration. Then filtrate is directed to accumulator tanks, from where water is supplied to reverse-osmosis modules. Feed water is pressurised up to 7.0 MPa. Filtrated water is collected for temporal storing in potable water tanks, then it is delivered to an external distribution system. Salty concentrates are discharged into the sea. Preliminary chemical treatment is provided for feed water and subsequent treatment of potable water. Principal lay-out of the desalination complex vessel is depicted in Fig. 7, and its basic characteristics are given in Table III.

Consequently, the floating complex of an FNPP and an RO system is capable of producing up to 100,000 t/day of potable water and up to 40 MW of electricity simultaneously, or up to 60 MWe with idle desalination facility. The design includes collector tanks for both ultra-filtrate and desalted water, which are located at the bottom part of the vessel. Operation premises, cabins for crew and control boards are located on the upper deck.

Table III Basic characteristics of complex for production of electricity and potable water by reverse-osmosis

Characteristics	Value
1. Max. net electrical output, MW	2x30
2. Sea water flow to FNPP condensers, m^3/h	5400
3. Water warm up in FNPP condensers, ° C	10
4. Sea water temperature (design), ° C	18
5. Total amount of solid particles dissolved in sea water, ppm	38,500
6. Water flow to RO system, m^3/day	259,000*
7. Pressure energy recovery in RO system, %	43
8. Potable water flow rate, m^3/day	100,000
9. Potable water facility power load, MW	18-19

*) Defined by sea water flow to FNPP condensers

Proven technology, successful experience of long-term operation of prototypes, possibility of wide variation in relation of electric power, potable water and heat production capacities are the advantages of the floating complexes under consideration.

Multiyear operation experience with similar equipment, structural materials and instrumentation means gives grounds to recommend them as analogues for creation of new prospective floating nuclear stations intended to produce electricity, potable water and heat.

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TECHNICAL ASPECTS OF COUPLING A 6300 m^3/DAY MSF-RO DESALINATION PLANT TO A PHWR NUCLEAR POWER PLANT

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Abstract

Presently, eight pressurised Heavy Water Reactors (PHWRs) each of 235 MWe capacity are operational in India. Four more units of similar capacity are expected to be commissioned soon. Work on two units each of 500 MWe capacity is also initiated. Extensive engineering development work has also been carried out in India, both on the MSF process and the membrane process. Based on the experience obtained from the presently operating 425 m^3/d MSF plant and from the R & D work on the RO process, a 6300 m^3/d MSF-RO plant (4500 m^3/d MSF & 1800 m^3/d RO) has been designed and the work for setting up this plant is undertaken. The steam for the heating duty in the brine heater as well as the steam for the evacuation purpose for the MSF plant is proposed to be obtained from the nuclear plant steam cycle. Sea water feed for the MSF plant as well as for the RO plant will be derived from the sea water discharge system of the nuclear power plant. Provision is made for supply of electrical power also from the power plant. The details of the heating steam supply circuit starting from the steam tapping point on the nuclear plant side to the MSF plant brine heater inlet and the arrangement for the return of condensate to the nuclear plant has been described with component requirement and various technical considerations. All the liquid streams and the steam supplied from the nuclear plant to the desalination plant as well as the product water will be monitored to ensure that there is no radioactive contamination.

1. Introduction

India's nuclear power programme is based mainly on the Pressurised Heavy Water Reactors (PHWRs). At present, the installed capacity is about 1800 MWe. Another 940 MWe is expected to be added in near future with the commissioning of four units each of 235 MWe capacity. The detailed design of 500 MWe unit has been prepared and the work on these units is initiated.

Bhabha Atomic Research Centre (BARC) at Mumbai has been engaged for more than 15 years in the research and development work in the fields of distillation desalination and RO processes. A MSF plant of 425 m^3/d capacity was indigenously designed manufactured and installed at BARC. This plant has been operating for nearly eight years and generated valuable experience for the design and manufacture of commercial MSF plants based on acid pretreatment and long tube evaporator design. Extensive R & D work on the indigenous manufacture of RO membranes has been carried out and a number of small scale RO plants have been installed in India based on work carried out in BARC.

Based on the operational experience and design data from 425 m^3/d MSF plant and the engineering development work on RO, a 6300 m^3/d MSF-RO plant is being set up at Kalpakkam (Tamil Nadu).

MSF plant needs about 21 Te/hr saturated steam at 130°C and 300 kg/hr at 20 kg/cm^2 for the steam jet ejectors of the evacuation system of MSF plant. Both these steam supplies are proposed to be met from the power plant steam cycle of Madras Atomic Power Station (MAPS). Sea water supply for both MSF & RO plants will also come from various sea water outlet streams of the nuclear power plant. Electricity supply for the desalination will also be made from nuclear power plant electrical system.

Use of an oil fired boiler or use of a water ring vacuum pump is an expensive proposal for creating vacuum in the MSF plant. The supply of steam for this purpose also from the nuclear power plant has economic advantage. In a PHWR, large amount of sea water is used for cooling of the

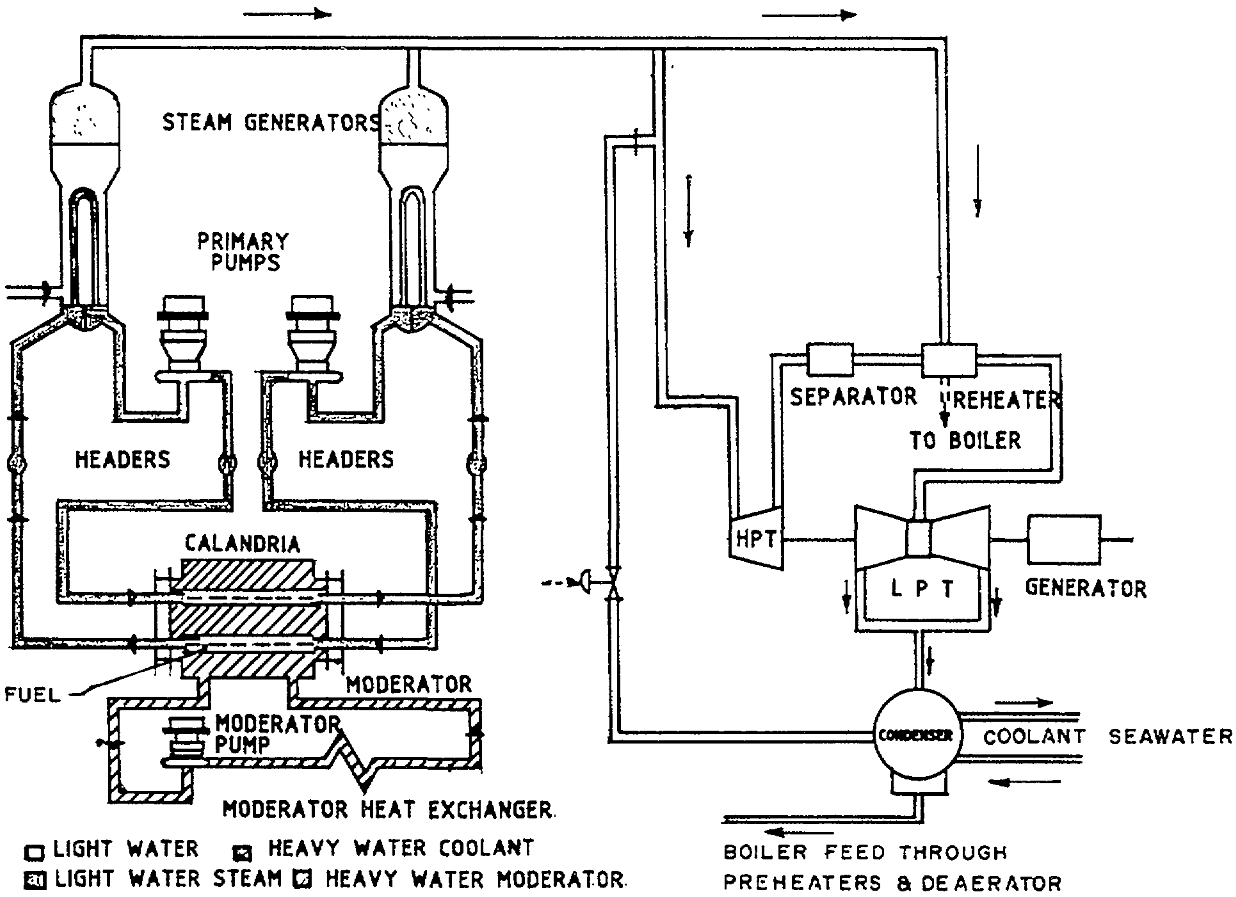
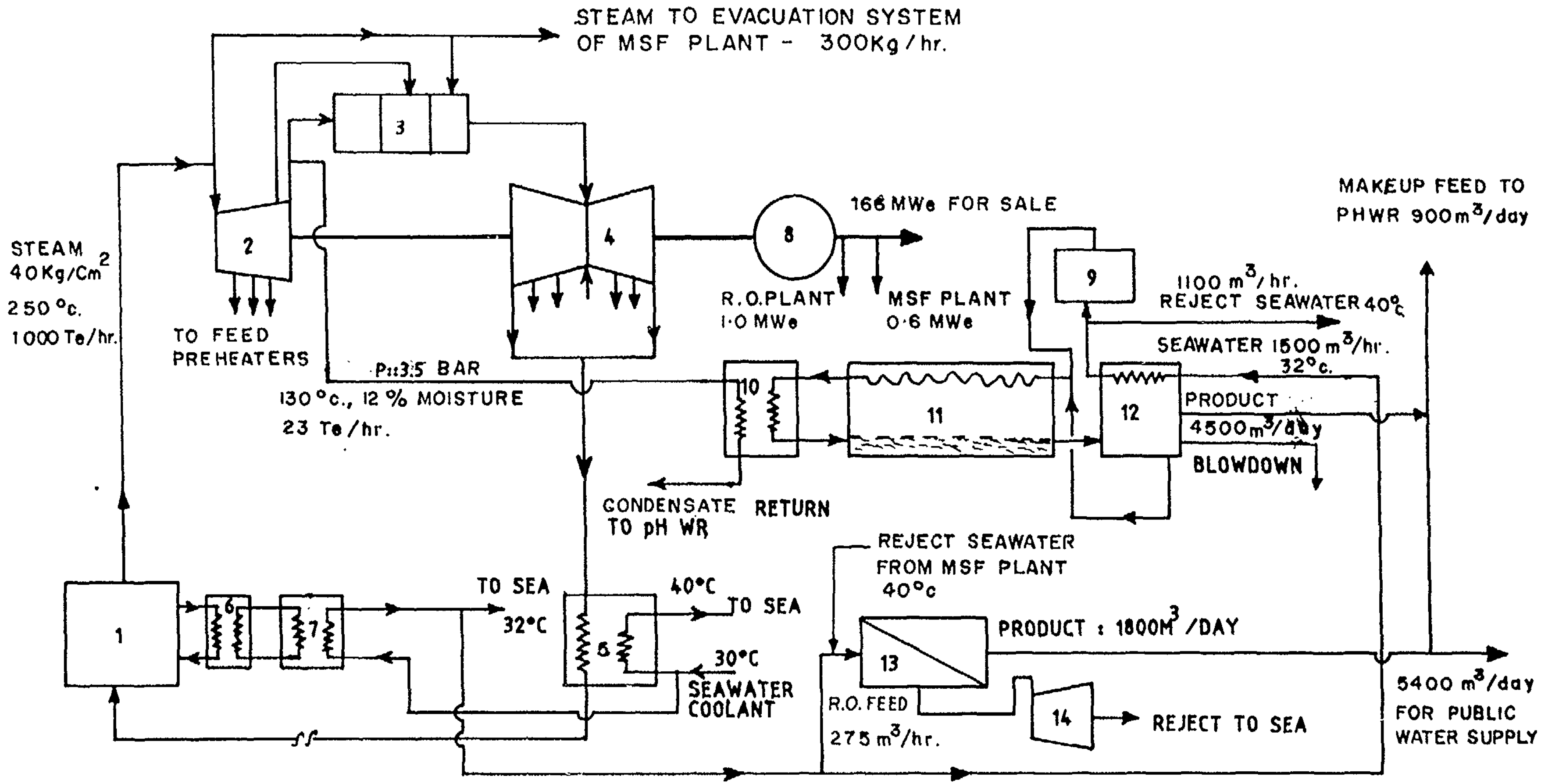


FIG. 1. PHWR SIMPLIFIED FLOW DIAGRAM

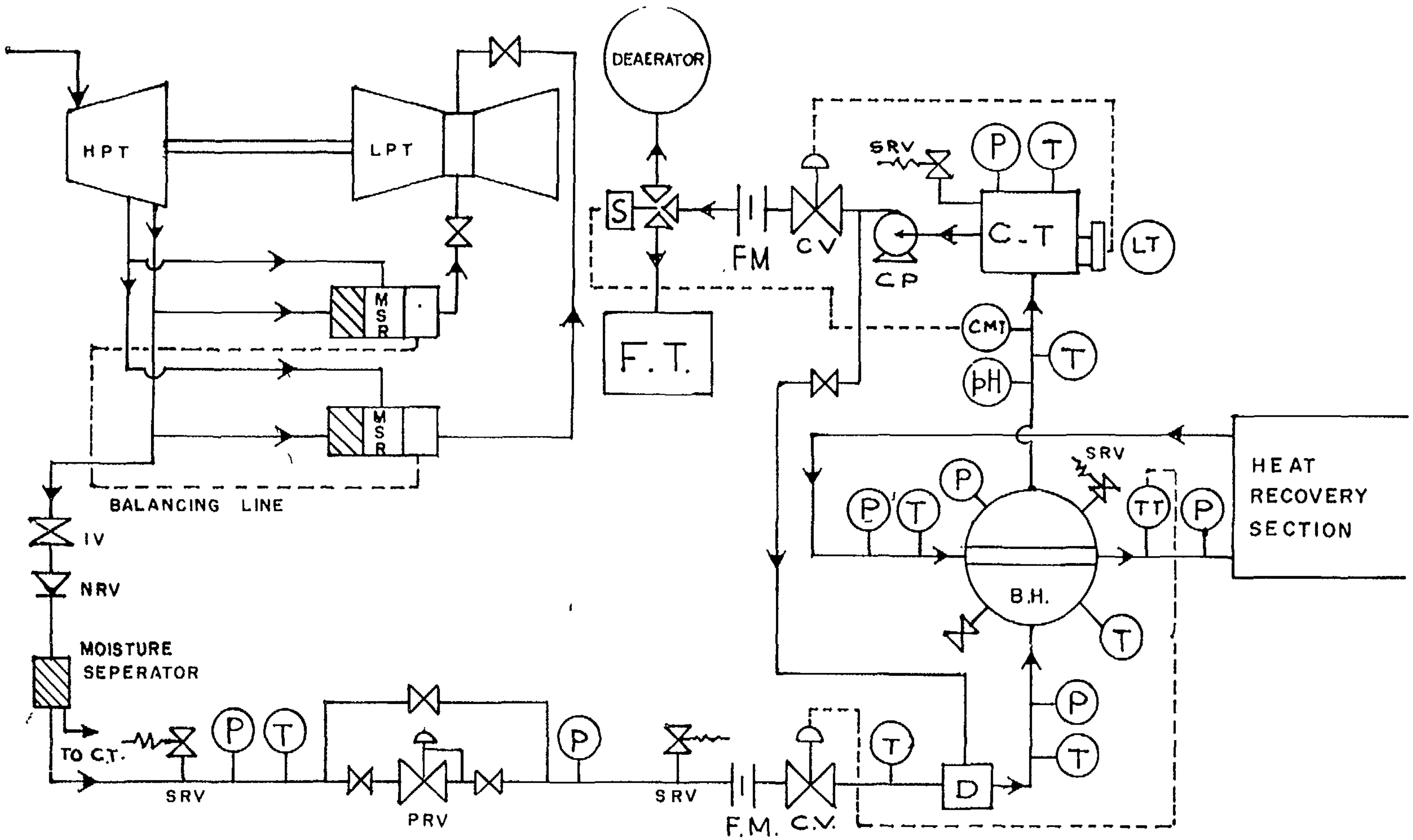
Table I Main Design Parameters of PHWR

1. Reactor Thermal Rating	: 790 MWth
Gross Electrical Output	: 235 MWe
Net Electrical Output	: 220 MWe
2. Moderator	: Heavy Water, p: 7.5 kg/cm^2 T (in) 44°C, T (out) : 65°C
3. Primary Heat Transfer coolant	: D_2O, p: 87 kg/cm^2 T (in) 249°C, T (out) : 293°C
4. Secondary Heat Transfer loop	: Light water/steam Steam p: 40 kg/cm^2, T: 250°C
5. Exhaust steam from H.P. turbine	: p: 6.0 kg/cm^2, moisture : 11.2%
6. Steam at Inlet to L.P. turbine	: p: 5.7 kg/cm^2 T: 233.3°C (superheated)
7. Steam at inlet to condenser	: p: 0.087 kg/cm^2
8. Sea water coolant temperature	: In : 30°C, Out : 40°C



1. PHWR : 170 MWe
2. H.P. TURBINE
3. MOISTURE SEPERATOR/ REHEATER
4. L.P. TURBINE
5. POWER PLANT CONDENSER
6. MODERATOR-DM WATER COOLING LOOP
7. DM WATER-SEAWATER COOLING LOOP
8. GENERATOR
9. MSF PLANT CHEMICAL PRE TREATMENT SECTION.
10. MSF PLANT BRINE HEATER
11. MSF PLANT HEAT RECOVERY SECTION
12. MSF PLANT HEAT REJECT SECTION
13. R.O. PLANT
14. R.O. PLANT ENERGY RECOVERY TURBINE

FIG. 2. 6300 m^3/Day MSF-RO DESALINATION PLANT COUPLED TO 170 MWe PHWR



HPT : HIGH PRESSURE TURBINE.
LPT : LOW PRESSURE TURBINE.
IV : ISOLATION VALVE
FM : FLOW METER
CV : CONTROL VALVE
MSR : MOISTURE SEPERATOR/ REHEATER.

NRV : NON RETURN VALVE
SRV : SAFETY RELIEF VALVE
PRV : PRESSURE REGULATING VALVE
D : DESUPER HEATER
BH : BRINE HEATER
TT : TEMPERATURE TRANSMITTER

CP : CONDENSATE PUMP
pH : pH METER
CMT : CONDUCTIVITY METER TRANSMITTER
CT : CONDENSATE TANK
LT : LEVEL TRANSMITTER
S : 3 WAY SOLENOID VALVE
FT : FLASH TANK

FIG. 3. ARRANGEMENT OF SUPPLY OF HEATING STEAM TO DESALINATION PLANT

moderator through an intermediate DM water heat exchange loop. The outlet sea water at the exit of the moderator cooling circuit is available at 32°C (2-3°C higher than the ambient sea water temperature). This 32°C sea water will form the coolant fed to the heat reject section of the MSF plant. This coolant comes out from the MSF plant at 40°C. A part of this sea water at 40°C is mixed in varying ratio with 32°C sea water from the moderator cooling circuit and form the feed to the RO plant at varying temperature in order to study effect of temperature on the performance of RO membrane modules.

2. PHWR Nuclear Power Plant

Fig. 1 shows a general flow diagram of a power plant based on pressurised heavy water reactors. PHWRs use natural uranium dioxide (UO_2) as fuel and heavy water (D_2O) as moderator as well as the primary coolant. The various process parameters mentioned in PHWRs are given in Table I. The Madras Atomic Power Station is presently working at 2 x 170 MWe capacity due to restriction on flow in the coolant channels. Replacement of the coolant channels is planned and full capacity of one of the power units is expected to be restored to 235 MWe by the time desalination plant becomes operational by the middle of year 2001.

Fig. 2 shows the arrangement of coupling between 6300 m^3/d MSF-RO desalination plant and the Nuclear power plant (2 x 170 MWe Madras Power Station). The heating steam supply to the MSF plant is made from the cross over steam pipeline from HP turbine to the moisture separator/reheaters of the power plant. Since steam is wet (12% moisture)steam requirement is 23 Te/hr for a performance ratio of 9 for MSF plant.

Sea water coolant supply to the heat reject section of MSF plant is proposed to be made from outlet sea water (at 32°C) from moderator - DM water - sea water coolant (process sea water) loop. A part of sea water supply to RO plant is made after mixing outlet sea water (32°C) from process sea water cooling loop with 40°C sea water reject from heat reject section of the MSF plant. By this mixing arrangement, temperature of feed to RO plant can be varied and plant performance with respect to product output and product quality can be studied in the range 32°C to 40°C.

300 kg/hr of motive steam is supplied to the steam jet ejectors (of the evacuation system of MSF plant) from live steam line (40 kg/cm^2, 250°C) after reducing the pressure of steam to 20 kg/cm^2 using a PRV. "An NRV is placed in the vent line from last heat reject stage to the evacuation system so that there is no flow of steam to the flash evaporators. Power requirement for the MSF plant is 0.6 MWe. In addition about 2.4 MWe loss of power occurs in the generator due to extraction of 23 Te/hr of heating steam for the MSF plant. RO plant needs 1.0 MW of electrical power.

Fig. 3 gives the details of the heating supply circuit for the MSF plant starting from the tapping point on the power plant side to the brine heater entry point and the condensate return to the power station. The steam from the nuclear boilers is first expanded in HP turbine. At the exist from the HP turbine, the steam is at a pressure of 3.5 kg/cm^2 g and 12% wet. The outlet steam from the HP turbine is divided into two streams and sent to 2 Nos. of combined moisture separators/reheaters in which it is superheated using steam bled at the intermediate pressure from the HP turbine and then by live steam (40 kg/cm^2, 250°C). Superheated steam then flows to LP turbine.

As shown in Fig. 3, steam is tapped from one of the moisture separator - reheater input steam line. An isolation valve and a non return valve are provided just after the tapping point. NRV is necessary since during start up of power plant, the turbine moisture separator/reheater and condenser come under vacuum. The NRV takes care of any leakages of air which may occur on the desalination plant side. The steam is then taken to a small moisture separator where most of its moisture content is removed and sent to the power plant deaerator. A Wiremesh type moisture separator with SS 304 wiremesh is proposed to be used. Expansion loops are provided between long straight pieces of steam pipe after assessing thermal expansion of the piping. Steam velocity is kept 40 - 50 M/Sec.

After moisture separator a vent valve is provided which will vent the steam to atmosphere for short duration if steam consumption stop abruptly in the desalination plant due to brine recycle pump failure or any abrupt rise (above 121°C) in the temperature of hot brine coming out of the brine heater. This vent line will have an orifice fitted for the venting only fixed quantity of steam roughly equal to the steam consumption in the brine heater in normal operation. The steam further flows to a PRV station. After PRV station a safety relief valve is provided which will open if the steam pressure at the input of the brine heater exceeds the set limit. An arrangement is provided to desuperheat the steam if necessary. The steam then enters the brine heater.

The condensate from the brine heater is collected in a condensate tank and pumped to the deaerator of the power plant. A three way solenoid valve is used in the condensate return line to divert the condensate to an intermediate flash tank if found contaminated with sea water. The contaminated condensate is sent to a mixed bed ion-exchanger for polishing and then return to the nuclear power plant.

Pressure of brine in the brine heater will always be kept 0.25 kg/cm^2 more than the heating steam pressure entering the brine heater to prevent any inleakage of tritium activity in the recycle brine. The recycle will be continuously monitored for α, β and tritium activities.

3. Technical Consideration for the Design of Brine Heater

As the brine heat forms the interface between the heat recovery stages of the MSF plant and the nuclear power plant turbine, design of brine heater should be carefully done to give a useful life of at least 30 years. Selection of materials of construction, design temperatures and pressures, recycle brine concentration factor and pH need special care. As the maximum temperature of brine (at outlet from the brine heater) is 121°C, concentration of the recycle brine should be fixed on the basis that solubility product of Ca^{++} and SO_4^{--} ions is appreciably below to the limiting solubility product. The brine velocity in the tubes should not be kept less than 2 m/sec.

The entire steam supply circuit including brine heater shell (steam side) shall be designed for full vacuum to 5.5 kg/cm^2. g (Maximum possible pressure of steam at the tapping point) and 150°C. The pressure of steam entering the brine heater should not exceed 2.8 kg/cm^2. a and as already mentioned flow of steam will be controlled so that outlet brine temperature does not exceeds 121°C.

An impingement baffle should be provided near the top tube rows in order to prevent direct impingement and erosion by entering steam. Proper arrangement in the tube bundle is necessary to remove non-condensable. A safety relief valve on shell side is to be provided which will open if the steam pressure on the shell side exceeds 3.0 kg/cm^2. Differential expansion between tubes and shell should be carefully assessed.

Materials:

Tubes	:	Cu - N: 90 : 10 (Max. 1.5% Fe) 19 m.m. O.D, 18 BWG
Tubesheets	:	Cu - No. 90 : 10 Thickness to withstand pressure of 7.0 kg/cm^2
Water Boxes	:	Cu - Ni. 90 : 10
Impingement baffle at steam entry	:	SS 316, 5 mm thick

At BARC, a 425 m^3/D MSF plant is being operated for about 8 years. The brine heater is a four pass shell and tube heat exchanger using 346 No. 19 mm OD. 1 mm thick Cu-Ni 90: 10 tubes. So far, no failure of tubes or leakage of brine to the steam condensate has occurred.

Periodically (once in 2 years) tubes are cleaned using high pressure water jet at a pressure of 200 kg/cm^2 and water flow rate of 15 lpm. A small nozzle with 0.5 mm multiple holes and connected to a high pressure water hose is inserted in the tubes. The nozzle moves forward and the scale gets scrapped off by the high pressure water jet. After once cleaning, boroscopic examination of tubes is carried out and if necessary, second cleaning operation is carried out.

The evaluation of corrosion of tubes by sea water was carried out using eddy current technique and practically no damage to the Cu-Ni tubes was observed.

4. Safeguard Against Radioactive Contamination

All the sea water stream of the desalination plant will be continuously monitored to ensure that no objectionable α, β and tritium activities are there in any of the sea water or any other liquid stream coming into the desalination plant

A six hour interim product water storage (2 tanks) is provided. The product water will be continuously monitored both at the inlet and the outlet of the interim storage tanks and the product water will be released for the public consumption only after having certified by the Health Physicist that the radioactivity levels are not exceeding permissible drinking water limits.

SUMMARY OF EXPERIENCE AND PRACTICE IN JAPANESE NUCLEAR DESALINATION PLANTS AT THE INTERFACE BETWEEN NUCLEAR AND DESALINATION SYSTEMS

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Abstract

The widely prevalent large scale desalination of seawater is accomplished by two primary methods: Distillation and reverse osmosis (RO). In any case, an external energy supply source is mandatory for the operation of the desalination plants. Reverse Osmosis is more energy efficient than distillation. The energy input for RO is usually supplied by electric power, whereas thermal energy is extracted from an electric power plant for the distillation processes (dual purpose plant). There are no impediments in using nuclear power plants to supply energy to desalination plants in an integral site. However, it is essential to eliminate the possibility of penetration of radioactive contamination into produced water. Besides, the investigation of possible back-up facilities is detrimental to meet the demand of electric power and water. In accordance with the Japanese regulations, a nuclear power plant cannot be operated if any amount of radioactive contamination resulted from the failure of fuel is detected in the cooling water. In our experience, we have found that no special provisions and no additional selection criteria are needed to install the desalination plants within the nuclear power plants, except for the carbon steel shell utilized for the RO module.

1. Selection criteria of materials of the interface equipment

There is no additional selection criteria of materials of the interface equipment in the distillation system in Japan. However, in most of the cases, higher grade materials are selected for any pressure vessel and equipment of the distillation systems, in accordance with the authorized design guidance for Nuclear Power Plants. Key design specifications of the interface equipment in the distillation system (IKATA Nuclear Power Plant, Shikoku) are compared in Table I, with those of the interface equipment of normal heat exchangers which are used with thermal power plants.

In another case, the carbon steel shell outside the FRP pressure vessel is utilized for the RO module because the design standard for the FRP pressure vessel was not accepted in the design guidance for nuclear power plants.

2. Component structure and design requirements at the interface

A typical sketch of a Low Pressure Steam Generator is shown in Fig. 1. This is a kind of steam converter and a submerged tube type evaporator to obtain clean steam from the waste steam. In some cases, a feed water deaerator is installed on the top of the shell. The capacity and pressure of generating steam is always less than the quantity and pressure of heating steam, as shown in Table I.

However, in order to prevent heating steam leakage into the generated steam side, a Back Pressure Type Liquid/Liquid Heat Exchanger is considered as one of the barriers at the interface. Since radioactive contamination has been very strictly controled in the operation of Nuclear Power Plants, several detectors of radioactive materials are provided at various stages for monitoring the safe operation of the plant.

Table I Overall comparisons of Low Pressure Steam Generators

	Nuclear Power Plant	**Thermal Power Plant**	
Power Station	IKATA-I,II	Tomakomai No.1	Nagoya
(1)Type	U-tube Type Steam Generator (Secondary Loop)	U-tube Type of Steam Generator	U-tube Type of Steam
(2) Capacity (Generating Steam)	20,000 kg/hr	15,000 kg/hr	12,000 kg/hr
(3) Pressure (Generating Steam)	0.7 MPa (7 kg/cm^{2}g)	0.7 MPa (7 kg/cm^{2}g)	1.2 MPa (11.8 kg/cm^{2}g)
(4) Heating Surface Area	60 m^2	110 m^2	62 m^2
(5) Heating Steam : Quantity Pressure	 25,850 kg/hr 2.9 MPa (29 kg/cm^2)	 16,800 kg/hr 1.25 MPa (12.5 kg/cm^2)	 15,574 kg/hr 2.5 MPa (25 kg/cm^2)
(6) Material : Shell Tube Sheet Tube	 Carbon Steel (SB46) Carbon Steel (SB46) Copper Nickel (CNTF-1)	 Carbon Steel (SM41) Carbon Steel (SB42) Carbon Steel (STM42)	 Carbon Steel (SB42B) Forged Steel (SF50) Carbon Steel (STB42)
(7) Purpose of Generated Steam	Distillation System Other Utilities	General Use (no distillation system)	General Use (no distillation system)

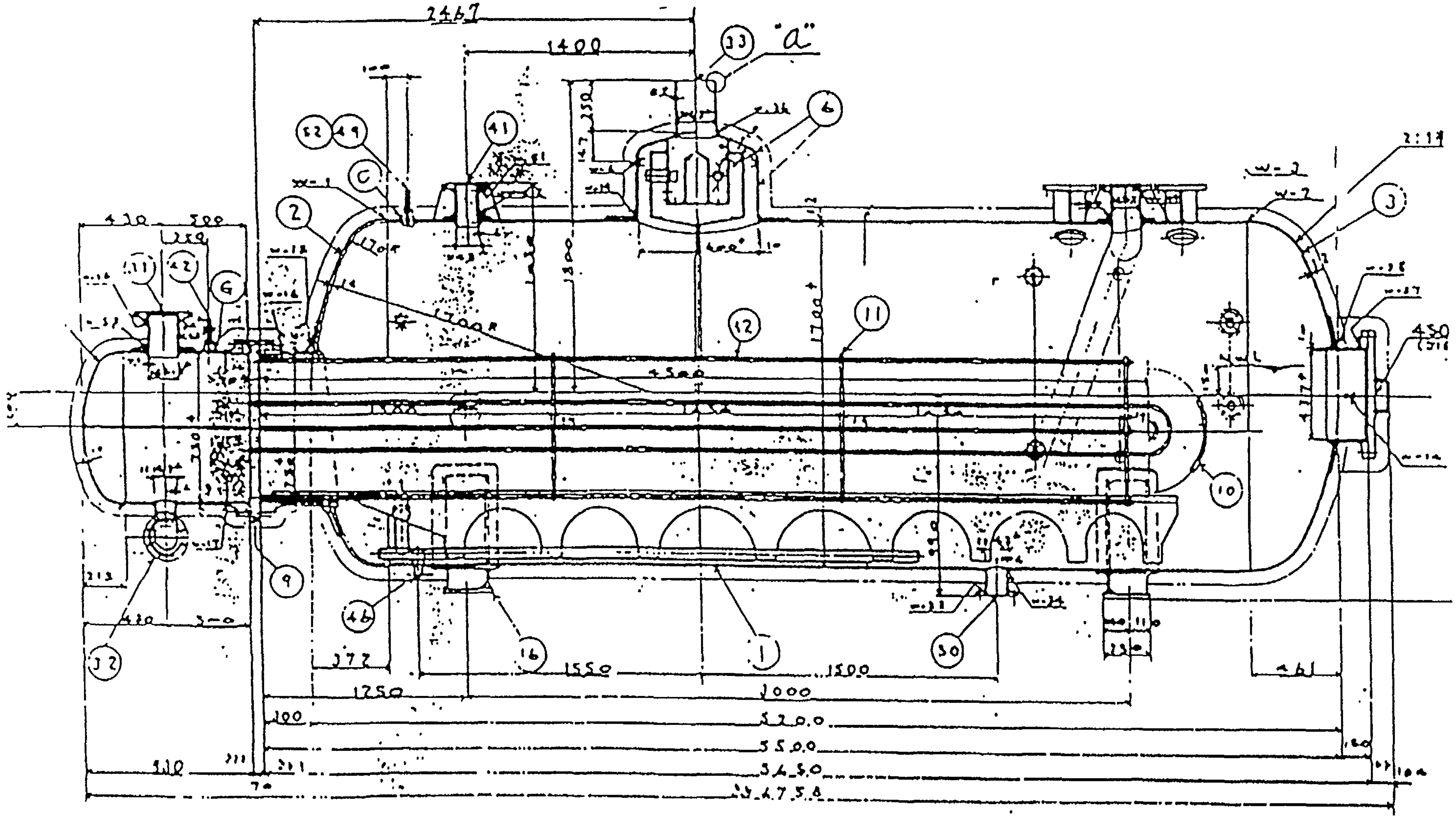
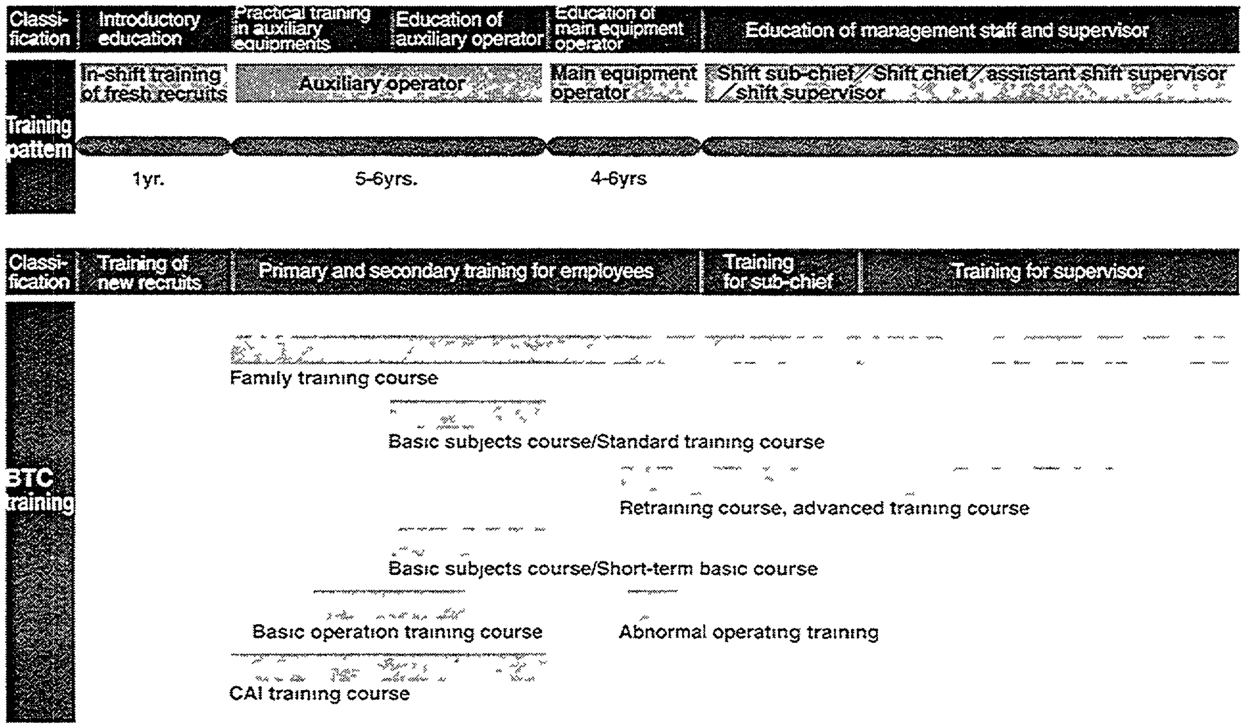


Fig. 1. A typical low pressure steam generator.

Training Pattern of Operators (An Example for a BWR Plant)



Training Pattern of Operators (An Example for a PWR Plant)

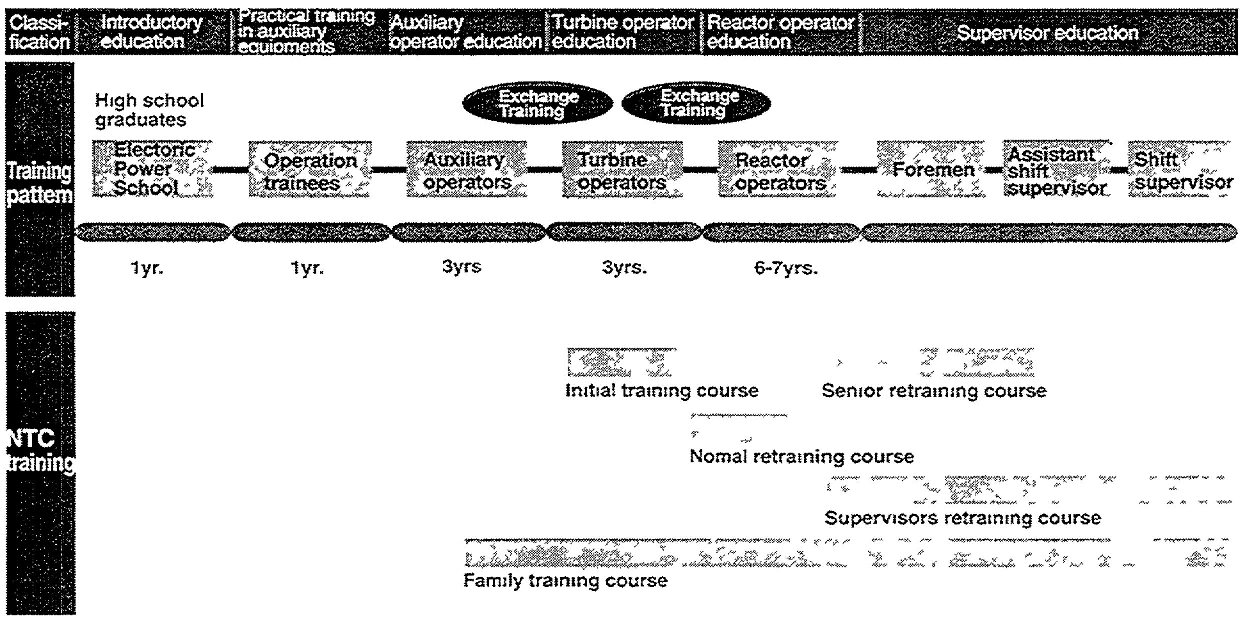
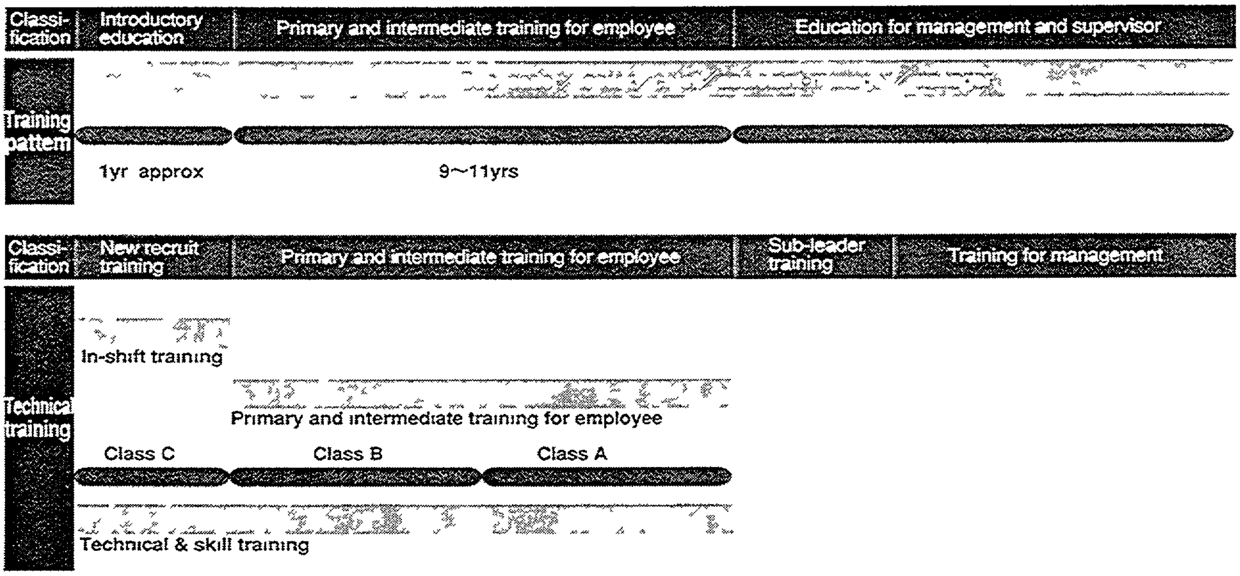


Fig 2(a). Schematic training framework of operators.

Training Pattern of Maintenance Personnel (An Example for a BWR Plant)



Training Pattern of Maintenance Personnel (An Example for a PWR Plant)

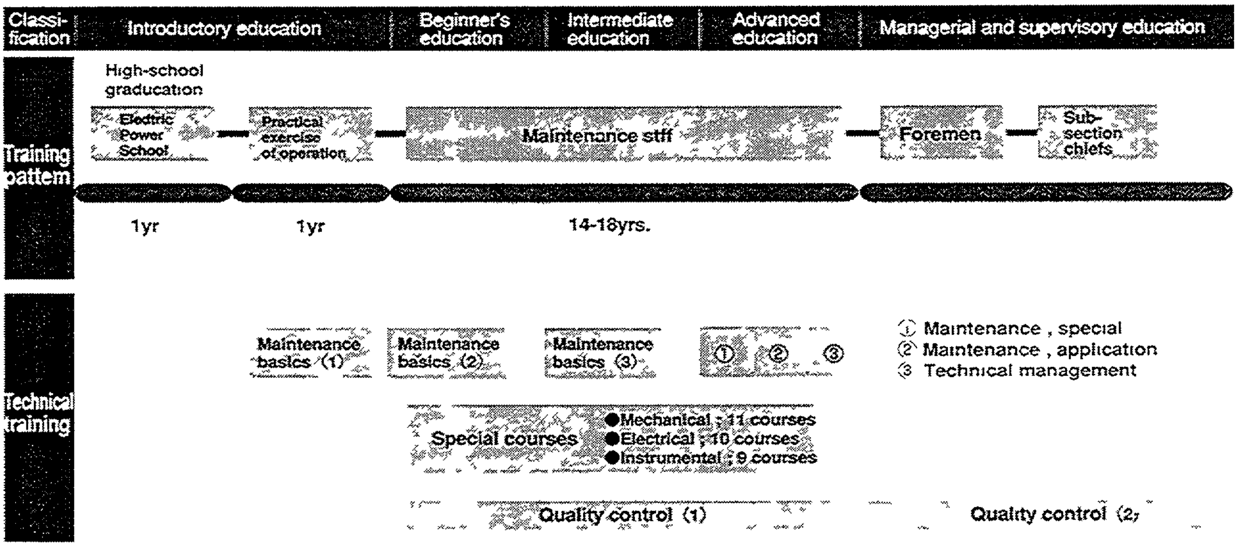


Fig 2(b) Schematic training framework of maintenance personnel

3. Operating experience of the interface equipment, such as performance monitoring, degradation, or regenerating, etc.

The desalination system is licensed to be installed at the nuclear power station after the permission of nuclear installation, in accordance with the Law for the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors. However, its system is operated and maintained under the law of general electrical facility.

4. Maintenance practice of the interface equipment

The scheduled maintenance of the nuclear power plant is carried out once a year or one and a half year. However, maintenance work of the desalination plant does not follow the schedule of the nuclear power plant and is generally carried out independently, because the power station, which has several nuclear power plants, has two desalination systems or two kinds of the desalination systems inside the power station. Maintenance items of the distillation plant at the IKATA nuclear power station are shown in Table II.

5. Technical aspects related to design precaution and analysis of integration of nuclear and heat application systems

5.1 Back-up system of heat energy

Distillation systems can not be operated without steam supply from the adjacent power plant. So, the steam from either of two nuclear power plants or other alternative facility can be always supplied.

5.2 Design requirement for large distillation systems

There is no additional selection criteria of materials of the interface equipment even if the capacity of the distillation plant becomes large. However, when the fraction of heat application for the distillation systems becomes significantly large, an additional system, such as a by-pass loop, may be required to install in the steam side of the heat exchanger at the distillation system, because the heat energy of the steam extracted from the secondary loop is not transferred to the distillation system through the heat exchanger (steam converter) at the shutdown of the distillation system.

In the Japanese application, the fraction of heat application for distillation systems is very small and no special operation or systems are considered. The heat energy generated in the heat exchanger (steam converter) is supplied, in addition to the distillation systems, to the air condition systems inside the building and the auxiliary steam system in the nuclear power plant. The amount of supplied steam from the secondary loop to the heat exchanger (steam converter) is controlled, in accordance with the amount of the total load in those systems.

5.3 Education and training

The education and training of the operators and maintenance personnel for nuclear power plants are conducted by each electric utility. The educational and training for candidate operators are provided from a long-term perspective accoridng to a long-term training programme. Even after assignment to a shift operator, he contiues constant efforts to maintain and improve his skill through field training. Plant -specific simulators are installed at each site to conduct emergency response training (EOP training) and sumulator training for acutual events experienced. Shift operators are provided with such training periodically. Maintenance personnel are educated and trained in class rooms, in daily service and through practical training during scheduled inspections. At the mainitenance training centers of each electric utility, trainees are trained with training machines simulating actual equipment. Fig. 2 shows a schematic training framework of operators and maintenance personnel for BWRs and PWRs in Japan.

Table II Inspection items and its frequency in distillation system

Inspection item	Frequency
(a) Overheat at bearing of pump and motor (b) Seal water and lubricant water (c) Operating condition (inlet or outlet pressure) on pump (d) Ink or recording sheet of instrument	1/1 or 2 days
(e) Measuring instrument of pH (f) Monitoring system of electric conductivity (g) Cleaning of seawater strainer (h) Other strainer (I) Silica gel for dehumidifier in sulfate tank	1/week
(j) Alarm system (k) Air pipe and drain pipe (l) Cleaning of guide pipe for pressure (*)	1/month (*) 4/month
(m) Disassemble of pump (n) Inside inspection of evaporator and other equipment (o) Cleaning of pipe (p) Functioning inspection of instrument system (**) (q) Measurement of insulation (r) Control valve and ground packing (t) Strengthen of wiring terminal for control system (u) Leakage of air pipe	1/year (**) 2/year

6. Monitoring practice of radioactive contamination in the heat application system

All distillation plants (MSF and MED) obtain the heat energy from PWR nuclear power plants. At the side of nuclear power plant, radioactivity in the primary (pressurized water) and the secondary (steam) loops is continuously monitored. If either a fuel failure in the primary loop or radioactive contamination in the secondary loop is detected, the reactor is immediately shutdown, in accordance with the safety regulation. Then, the radioactive contamination in the product water is protected, even if the monitoring systems are not installed both in the steam line at the evaporator and the product water line in the distillation system.

Authors would like to express deep gratitude to the staff of the Shikoku Electric Power Companies for providing the data and relevant information and help in preparing this paper.

II.3. OPERATIONAL EXPERIENCE WITH NUCLEAR HEAT APPLICATION

Experience with nuclear district heating

XA9848826

NUCLEAR SOURCE OF DISTRICT HEATING IN THE NORTH-EAST REGION OF RUSSIA

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Abstract

The operation of the Bilibino Nuclear Co-generation Plant (BNCP) as a local district heating source is reviewed in this paper. Specific features of the BNCP power unit are given with special emphases on the components of the technological scheme, which are involved in the heat production and supply to the consumers. The scheme of steam extraction from the turbine, the flow diagram of steam in the turbine, as well as the three circuit heat removal system are described. The numerical characteristics of the nuclear heat supply system in various operating modes are presented. The real information characterizing current radiological conditions in the vicinity of the heat generation and distribution equipment is also presented in the paper. The BNCP technical and economical characteristics are compared with those of conventional energy sources. Both advantages and some problems revealed during the twenty year experience of the BNCP nuclear heat utilization are generally assessed. Safety and reliability characteristics of the reactor and the heat supply system are also described.

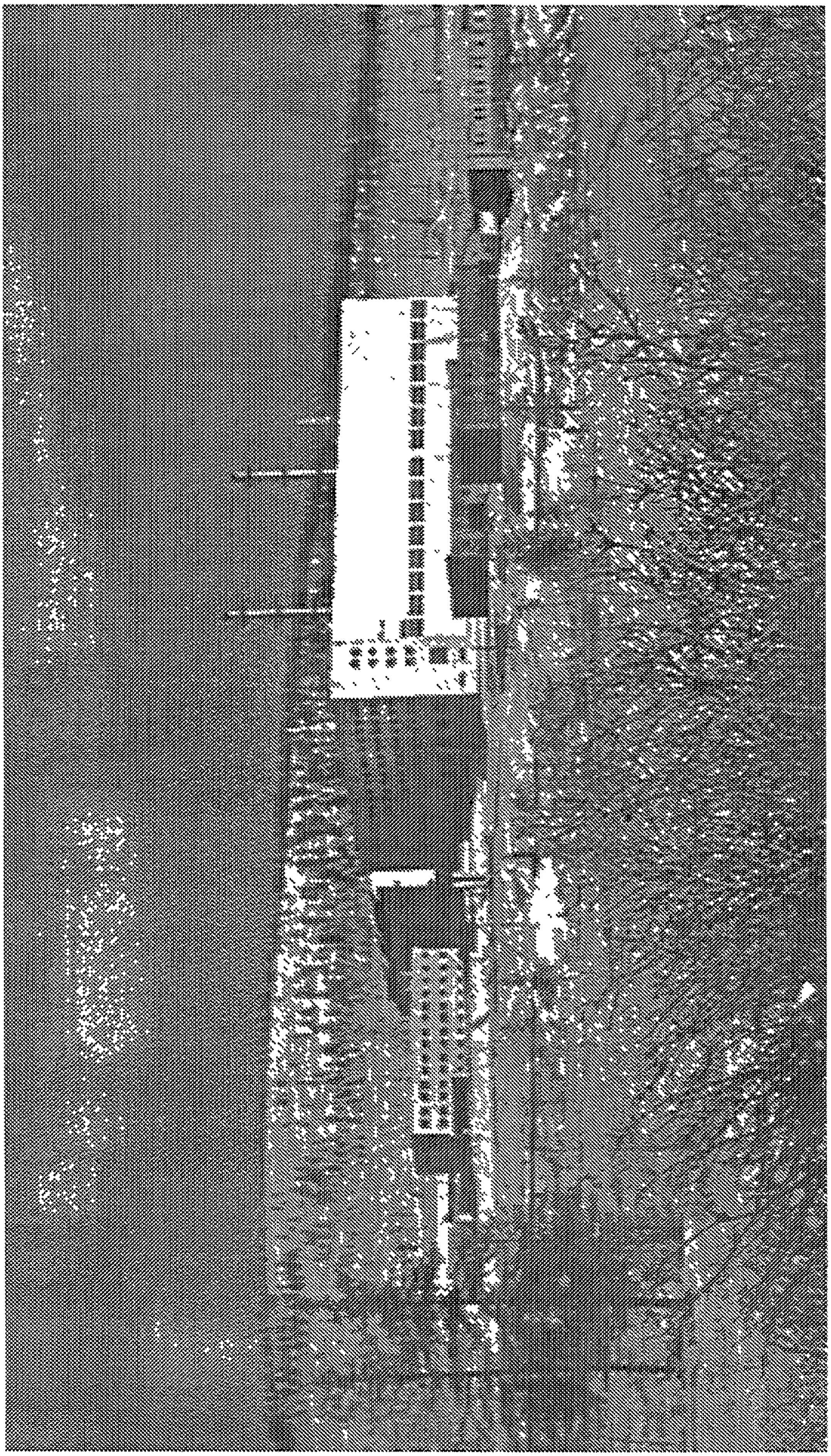
Bilibino nuclear co-generation plant

Along with the development of nuclear plants for the heat supply of large cities in the industrial regions, works are under way in Russia on designing plants for remote and almost inaccessible regions of the country. The experience gained from the Bilibino Nuclear Co-generation Plant design, construction and operation is the first of this kind, as far as the application of nuclear power source under the conditions of Far North-East region of Russia is concerned. The expedience of construction of the nuclear fueled plant in Bilibino, Chukotka Autonomous District was decided to overcome the high cost of fossil fuels, which are mined at the long distance from the site, and therefore the transportation was very complicated and expensive. Simultaneous demand in electricity and heat in the area led to the decision to construct the nuclear power source as a nuclear co-generation plant (NCP).

Specific features of the Bilibino NCP site region, which determined general requirements for the design and construction of the NCP, particularly, for the reactor facility are described in [1]. As regards the heat supply issues, it should be mentioned, that the long winter (at least, eight months) is typical in the Bilibino region, the lowest temperature being -60°C. The heating period is as long as ten months a year (see Fig.4).

The Bilibino NCP consists of four similar type power units. The thermal capacity of each steam generating plant is 62 MW, providing 12 MWe electric power plus heat production equal to 19 MWth (10.5 MWe + 29 MWth, being an alternative option). The first power unit was connected to the grid of Chaun-Bilibino power system in January 1974, while the fourth one came into operation in December, 1976. The simplified flow diagram of the Bilibino NCP power unit is shown in Fig.1.

The water cooled, graphite moderated reactor using tube type fuel elements was specifically designed to serve as a steam generation plant. The natural flow of boiling water is used for heat removal at all power levels of the reactor. In the reactor plant, the dry saturated steam of 96 t/hour is generated at the pressure 6.4 MPa in the single circuit system, the feed water temperature being equal to 104°C. Descriptions of the reactor plant are given in [1].



During the whole period of the BNCP operation (almost 24 years), reactor plants have demonstrated their high reliability. There have been no failures of fuel subassemblies (the main components of the reactor), causing their unloading.

An extraction turbine using the saturated steam was designed and manufactured in Czechia. The steam pressure at the turbine inlet is 5.9 MPa. The turbine consists of high and low pressure cylinders, a moisture separator being placed between these two. The main extraction of steam for the heating purposes is made downstream the high pressure turbine. The extracted steam pressure is controlled within the range 0.4-0.1 MPa. In addition, some portion of the steam extracted under control flows to the deaerator in order to increase water temperature, and to the auxiliary systems. Peak mode extraction (non-controlled extraction) of the steam for the purpose of district heating is made downstream the control stage of the high pressure turbine. Steam pressure of the non-controlled extraction is 1.5 MPa for the rated steam flow in the turbine. Fig. 2 shows the turbine steam flow rate diagram, based on the design data. In case of the turbine trip, live steam is supplied from the reactor to the heaters through the relevant pressure regulators (see Fig.1).

Experience at the Bilibino NCP

Climatic and hydrological conditions of the Bilibino NCP construction site raised the necessity to use the closed service water supply system for the heat removal from condensers, the water being cooled by the atmospheric air in the radiator heat exchangers (Hungarian design and manufacture). Air radiator coolers are used for the first time in the region with low air temperatures during the winter. This system requires a moderate water flow rate for the leaks compensation. It was accepted in the design that the rated temperature would be maintained in the condensers during only 9 months a year, when the daily average air temperature did not exceed 0°C. During the remaining 3 months, the turbine operation with deteriorated vacuum, even at the lowered power level was permitted. In order to eliminate such a drawback as the electric power restriction in the summer period, when the full capacity operation of mining industry plants demands the highest power system load, water-water heat exchangers were installed for the summer only operation in addition to the air radiator coolers. This made it possible to maintain the electric power of the NCP at the rated level in the summertime.

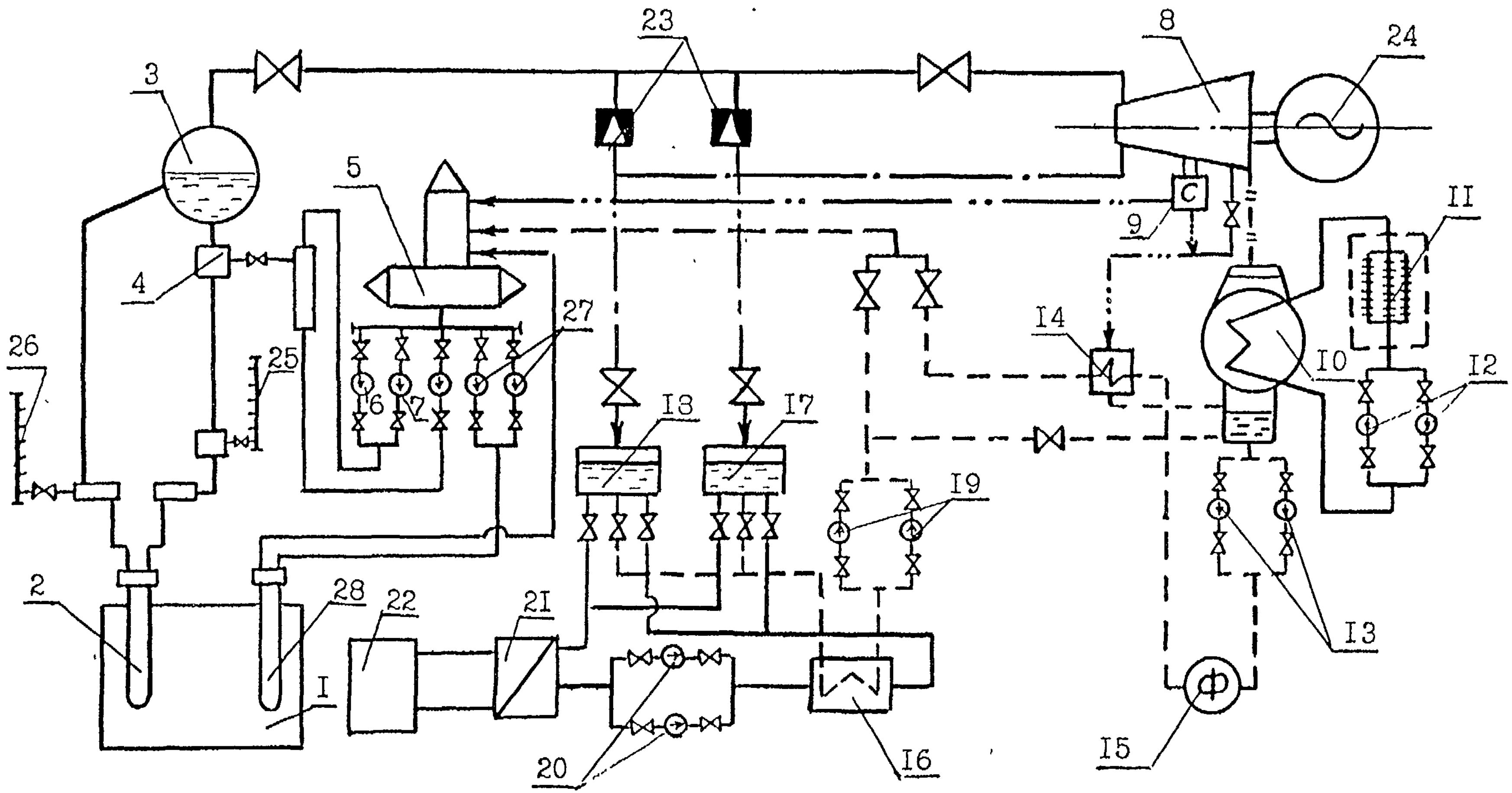
The experience gained on the Bilibino NCP commissioning and operation is reported in [1,2]. Fig.3 shows schematically the heat supply system to the consumers. Heat consumers in Bilibino are:

- heating and ventilation systems of residential and public buildings;
- hot water supply to residential and public buildings;
- heating and ventilation systems of industrial area installations.

Despite the fact that the tube design of reactor fuel elements assures the absence of fission products in the generated steam (except the limited amount of activated corrosion products in the steam), a tertiary circuit scheme has been adopted for the heat supply system.

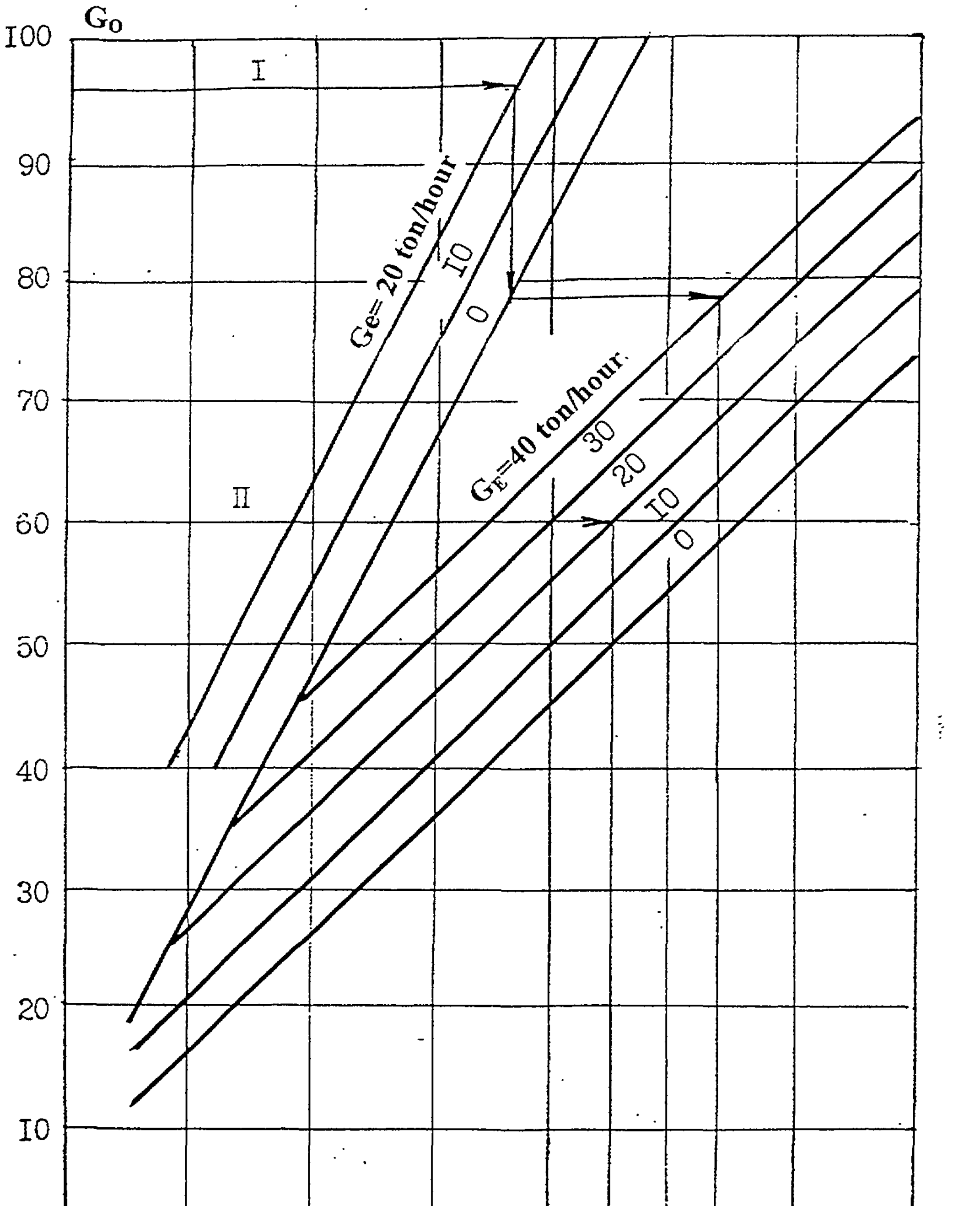
The primary heating medium is the steam extracted from the turbine, as well as its condensate. The highest pressure of the steam in the peak heater does not exceed 1.5 atm. The secondary (intermediate) water is heated at 1.8 MPa in the heat exchangers. The tertiary circuit water, heated by the secondary water in water-water heat exchangers, tramsmits the heat for the operation of heating and ventilation systems of residential and public buildings. Tertiary circuit water pressure does not exceed 0.6 MPa. The hot water supply system assumes heating of potable water in two stage heaters. The intermediate circuit water is used as the the heating medium.

Characteristics of the main operating modes of the heat supply facility of the Bilibino NCP power unit are given in Table 1. The typical chart of the heating load over one year operation is presented in Fig.4.



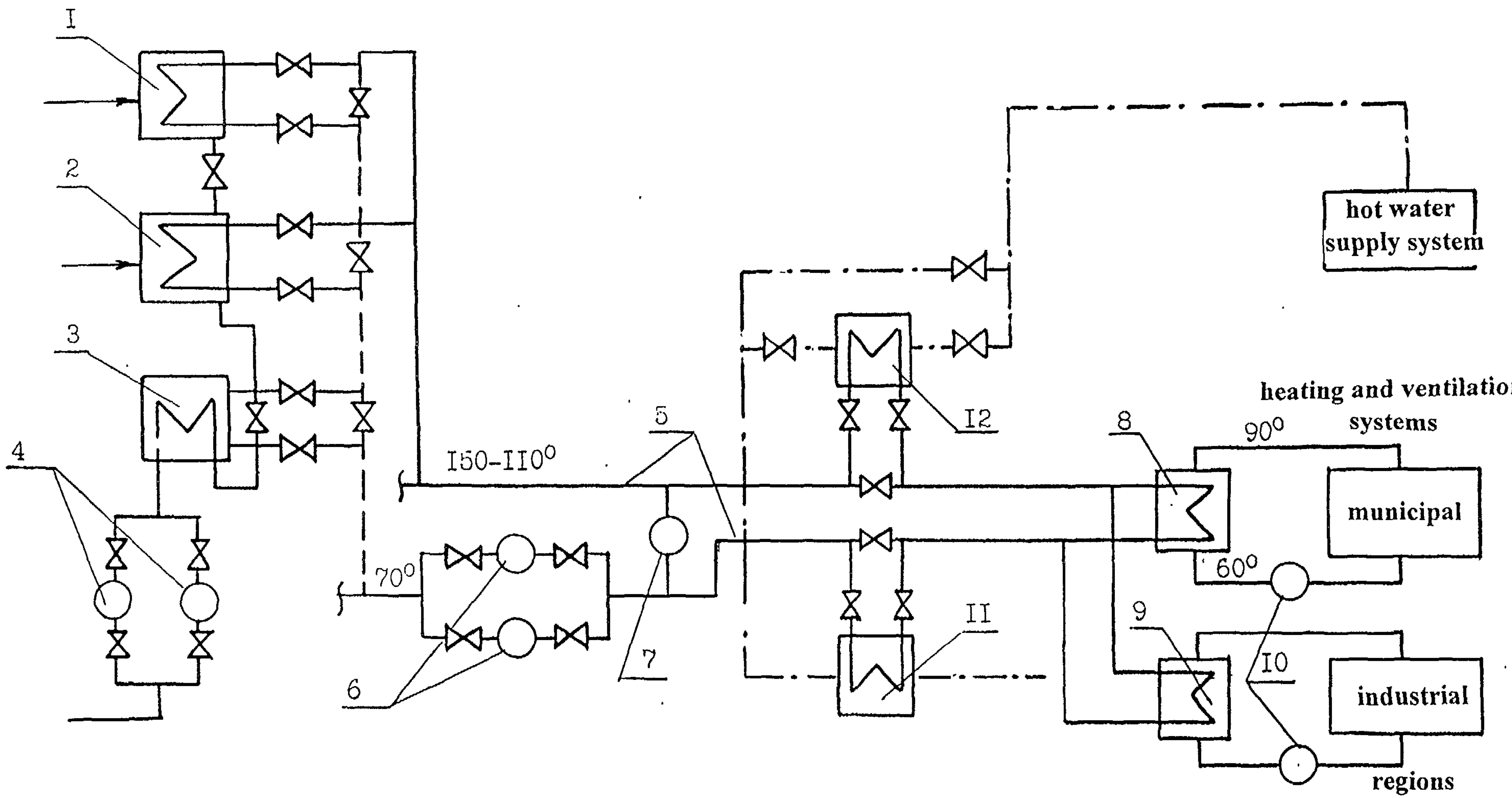
1 - reactor, 2 - fuel subassembly, 3 - steam separator, 4 - mixing device, 5 - deaerator, 6 - feed water pumps, 7 - emergency feed water pump, 8 - turbine, 9 - intermediate separator, 10 - condenser, 11 - air-radiator cooler, 12 - circulating pump, 13 - condensate pumps, 14 - low pressure regeneration heaters, 15 - filter, 16 - additional cooler of heater condensate, 17 - base heater, 18 - peak heater, 19 - heater condensate pump, 20 - intermediate circuit pumps, 21 - water-water heat exchanger, 22 - heat consumers, 23 - pressure regulators, 24 - generator, 25 - ECCS collector, 26 - cross-over collector, 27 - control and scram system cooling circuit pumps, control and scram system channel, 28 - control rod channel.

Fig. 1. Bilibino power unit process flow sheet.



Po = 5.9 MPa, io = 2780 kJ/kg , Pe = 0.25 MPa , tw = 15 °C,
Go = steam flow rate at the turbine inlet,
Ge = peak steam bleed-off, G_E = basic steam bleed-off,
Ge = peak steam bleed-off , Np = electrical power,
I - Go = 96 ton/hour, Ge = 20 ton/hour, Ge = 40 ton/hour
II - Go = 60 ton/hour, Ge = 20 ton/hour.

Fig. 2. Design basic diagram of the steam flow rate through turbine.



1, 2 - peak and basic heat supply exchangers to heat water in the intermediate loop, 3 - condensate cooler, 4 - condensate pumps, 5 - intermadiate loop, 6 - intermadiate loop pumps, 7 - pump of the temperature adjustment, 8,9 - heat exchangers of the heating and ventilation systems of municipial and industrial regions, 10 - pumps of the heating systems, 11, 12 - heat exchangers of 1 and 2 levels of the hot water supply system

Fig. 3. Schematic diagram of the Bilibino heat supply system.

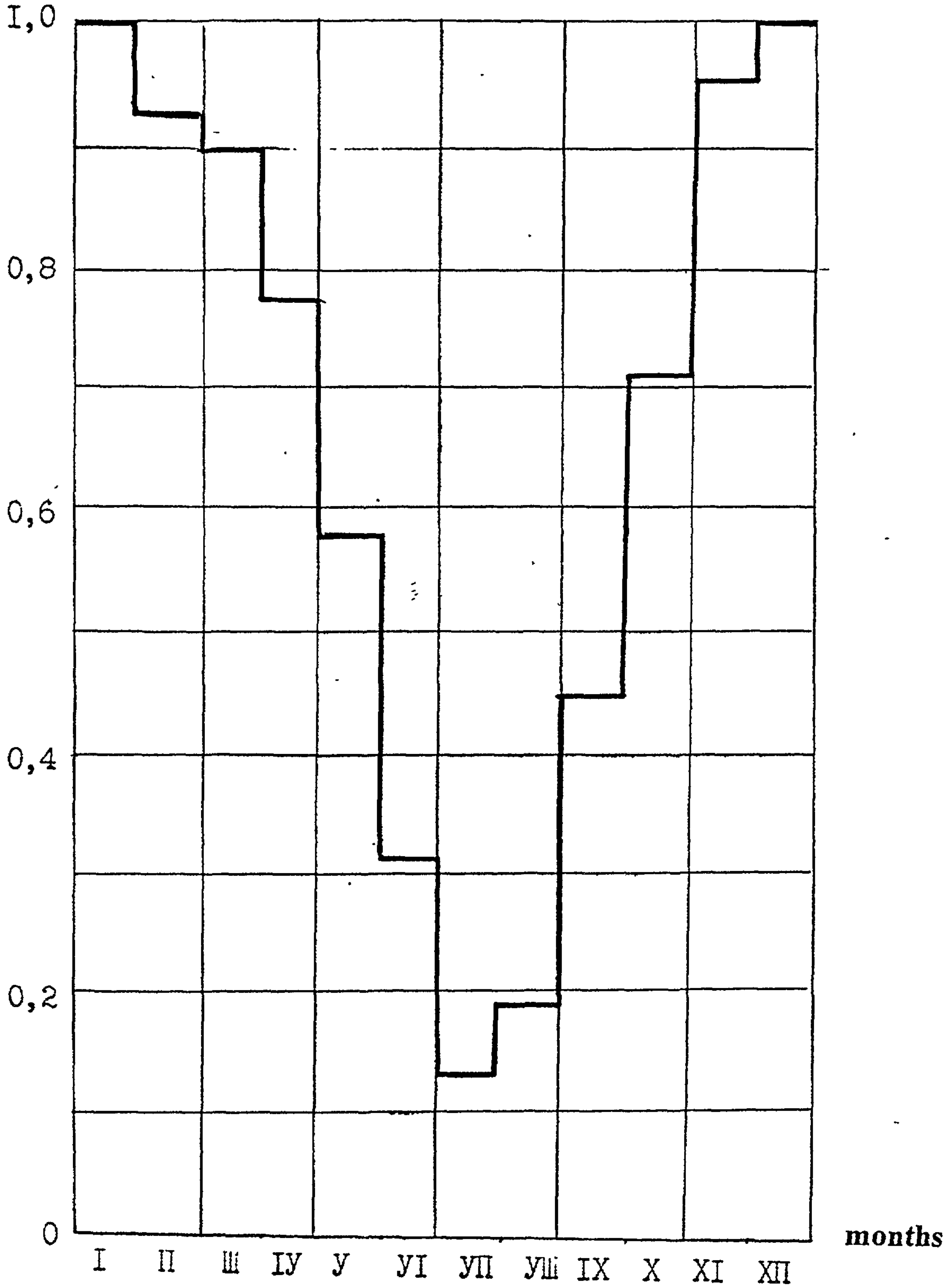


Fig. 4. Annual chart of the heat supply by Bılibino power plant.

Table 1. Operating modes of heat supply facility of Bilibino NCP power unit

No.	Parameters	Unit of measure	Ultimate mode	Rated mode	Medium mode	Turbine stop
1.	Total power	MW	28.9	19.5	13.6	19.5
2.	Peak heater					
2.1.	Hot steam flow rate	ton/hour	25.2	7.95	-	14.0
2.2.	Pressure of extracted steam	MPa	1.48	0.618	-	0.608
3.	Main heater					
3.1.	Hot steam flow rate	ton/hour	23.4	26.9	24.2	15.0
3.2..	Pressure of extracted steam	MPa	0.255	0.393	0.196	
4.	Hot steam condensate flow rate	ton/hour	48.6	34.85	24.2	29.0
5.	Water flow rate in the intermediate circuit	ton/hour	309	207	207	207
6.	Intermediate circuit pressure	MPa	1.81	1.81	1.81	1.81
7.	Intermediate circuit water temperature					
7.1.	Condensate cooler inlet	°C	70	70	52	70
7.2.	Condensate cooler outlet	°C	76	76	58.5	76
7.3.	Main heater outlet	°C	113	133	109	116.8
7.4.	Peak heater outlet	°C	150	150	109	150

The lay-out of the main components of the heating system is as follows: base and peak load heaters are located near the respective turbines in the turbine hall of NCP; the intermediate circuit extends to 3.5 km, namely, from BNCP to the water-water heat exchanger facilities in the town Bilibino; hot water is supplied from these heat exchangers to the consumers through the heating system of the settlement and the hot water supply system. Thus, heat is transported from BNCP by the intermediate circuit to the heat exchanger facilities located in the town. This section of the intermediate circuit is called a heat pipe-line. According to the project, there is the 100% reserve of the heat line: two hot water lines (150°C), and two return lines of cold water (70°C). The capacity of each line corresponds to the rated flow in the intermediate circuit.

There are design measures provided in the heating system of the town to prevent radioactive substance penetration from the primary to the tertiary circuit (heat and hot water supply systems). For this purpose, the increased intermediate (secondary) circuit water pressure is maintained, as compared to that of the primary circuit, i.e. to the steam pressure in the heaters , including the peak load heater. Interlocks are provided as prevention means, in order to isolate automatically the heaters from the intermediate circuit in case of the decrease of the pressure difference between the intermediate circuit and the extracted steam down to 0.2 MPa or lower. In Furthermore, the radioactive substance content in the intermediate circuit water is continuosly monitored. The sensors are attached to the hot leg of the main common pipeline of the intermediate circuit.

Since the intermediate circuit pressure was designed to be higher than that of the heaters, iron oxides can penetrate into the reactor water circuit in case of water leaks from the intermediate circuit caused by failures of the heater tube bundle (the allowable level of the iron oxides in the intermediate circuit is significantly higher than that of the reactor circuit).

The integrity of the tube bundle of heaters is monitored on the basis of intermediate circuit water replenishment and salt content in the condensate of the hot steam. If the salt content in the hot steam condensate reaches a maximum allowable level, the condensate flow is changed from the deaerator to the turbine condenser, thus allowing the hot steam condensate cleaning on the double-function filter (see Fig.1). This is the temporary measure. Repair works on the heater should be carried out during the plant outage.

Special studies on radioactivity have been carried out at the Bilibino NCP. In the course of these studies, isotopic compositions of radioactivity were monitored in the hot steam, water of the intermediate circuit and the heating system of the settlement, and the gamma-radiation was measured on the equipment and piping of the heat supply system and heaters in the living space. The results of these studies along with the data of the external radiation control have shown that the specific radioactivity of the intermediate water is equal to that of the initial water taken from the water storage. Contents of cobalt-60, caesium-137, manganese-54 and zinc-65 were three or two orders lower than those allowable for the potable water. This is also the case for the heating and hot water supply systems.

The results of repeated measurements of gamma-radiation doses on the surface of heat exchangers, radiators and piping of residential and public buildings in Bilibino have shown that these doses do not exceed natural background (10 - 25 μr/hr), thus being in accordance with the results of the observations made during many years on the natural radioactivity background in the BNCP area. Measurements have shown that the NCP based heating system does not cause any increased dangers of radioactive doses to the heat consumers, as compared to those resulting from such system using a fossil fuel fired boiler as the heat source.

The reliability of the Bilibino heat supply system using the Bilibino NCP is achieved by the high reliability characteristics of the reactor facilities used as heat sources (see data given above). These include: for example, three power units supply heat in the winter to the system; 100% heat line is reserved; monitoring devices are provided; and some organizaional measures, such as inspections and preventive maintenance of the equipment, are functioning. More details on the operation of Bilibino NCP based heat supply system are given in [2].

Since the beginning of the Bilibino NCP operation, the plans for electricity and heat production have been successfully fulfilled. The cost of electricity and heat generated by the Bilibino NCP is at least two times lower than that of fossil fuel power plants. Heat supply by the BNCP has saved at least 100 thousand tons of coal a year in the boiler plants of Bilibino.

The Bilibino NCP was designed in the late sixties. Therefore, there are many deviations in its design characteristics from the requirements contained in the current regulatory documents. Reactors of the Bilibino NCP were qualified in Russia as the first generation reactors.

Since the heat sources should not be located too far from the consumers, more strict requirements on safety are imposed to NCP and NHP (Nuclear Heating Plant) designs. After the regulatory documents on safety had been issued, safety analyses of reactor facilities were made [3], resulting in the following conclusions:

First, all reactivity feedbacks determined by the reactor core composition are negative.

Secondly, the reactor facility used on BNCP has a very high level of self-controllability with respect to the accidents with the loss of heat removal, i.e., even in case of the failure of all regular channels of decay heat removal, no fuel element cladding failures due to the unacceptable rise of their temperatures occur, and there is no lack in safety, as far as these accidents are concerned. Reactor facilities of this type (tube fuel element design and graphite moderator) have considerable advantages concerning safety assurance in case of the loss of heat removal system accidents, as compared to the reactors with the pin type fuel elements. This advantage is achieved by the combination of high thermal capacity of graphite moderators and high heat transfer from the fuel elements to the moderator.

Thirdly, design measures are provided to decrease the probability of reactivity related accidents and eliminate its negative consequences if the accident occurs. For the water-graphite reactors under consideration, accidents with several fuel element failures (more than 3 fuel elements) are regarded as the most dangerous. Accidents caused by the reactivity changes would have the most severe consequences: unauthorized insertion of the maximum possible positive reactivity accompanied by the failure of the reactor safety system. In this case, there is a risk of loss of integrity of the reactor shroud. In order to eliminate safety deficiency in case of such reactivity related accident, following measuers are incorporated:

- reliability improvement of control and safety system of the reactor;
- development of a device for the medium discharge from the reactor circuit (apart from the main safety valves);
- development of a device for the medium discharge from the reactor area;
- development of a device for the introduction of liquid absorber into the channels of the contrtol and safety system circuit.

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NUCLEAR HEAT APPLICATIONS IN RUSSIA: EXPERIENCE, STATUS AND PROSPECTS

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Abstract

The extensive experiece gained with nuclear district heating in Russia is described. Most of the WWER reactors in Russia are cogeneration plants. Steam is extracted through LP turbine bleeders and condensed in intermediate heat exchangers to hot water which is then supplied to DH grids. Also some small dedicated nuclear heating plants are operated.

1. INTRODUCTION

Low temperatures during the greater part of the year is characteristic for a majority of regions in Russia. Heating period for town dwellings lasts usually from 7 to 10 months a year. Contemporary dwellings are also provided with centrally-supplied hot potable water for sanitary and hygienic purposes. Moreover, hot water is supplied to public buildings for air conditioning. Average per capita heat consumption in households amounts to about 24 GJ a year and has been increasing over the years.

Fast growth of towns and gradual improvements in the quality of dwellings in the Sixties and Seventies resulted in a dominant share for fossil fuel in the production of heat. By the end of the Eighties it attained 42%, i.e. 1.72 times higher than that for electricity production.

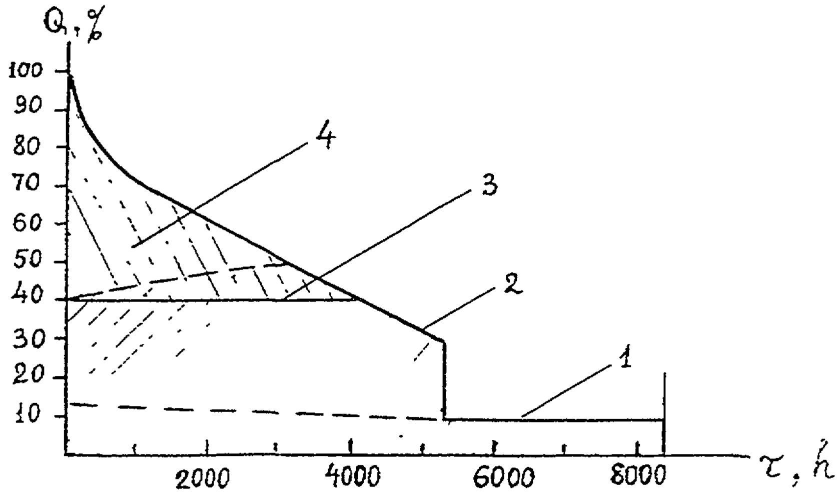
The town-building concept adopted in the country has been based traditionally upon the utilization of centralized heating, i.e. large systems for transport and distribution of heat from co-generation plants and boiler stations to blocks of buildings and to separate multiflat houses. About 70% of total low-grade heat consumed in household sector and in industry (approx. 8.7×10^9 GJ/year) is provided by powerful sources of heat through centralized heating systems, and about half of this value is being produced by co-generation plants [1]. Total heat output capacity of co-generation steam turbines amounted to 175GW (1990). The remaining 30% of heat is generated by decentralized sources, i.e. small boilers and individual fire devices. The share of electric heating is negligible.

A growth in concentration and value of heating demands in towns and availability of hot water transport and distribution, systems create, in principle, favorable prerequisites for utilization of nuclear plants as sources of heat in centralized district heating systems. In regions suffering from shortage of fossil fuel, where expensive fuel is used for heating and/or where environmental conditions do not allow the use of conventional fossil-fuelled boilers, nuclear plants are more attractive.

2. HEAT LOADS IN CENTRALIZED HEATING SYSTEMS

The level of heat demand in a centralized district heating system (CHS) varies depending on ambient air temperature. On the contrary, potable hot water supply does not depend significantly on the season and variations during a day are similar to those in household electrical demand. A typical CHS load diagram is presented in Fig.1.

Heat supply in a CHS is usually provided by a number of heat sources, one of which is a large co-generation power plant or boiler station bearing a base load in the CHS, while rest of the sources are being connected successively to the base as heat demand builds-up, i.e. as ambient air temperature drops. Co-generation power plants are used as a base heat source in CHSs with maximum heat loads exceeding 2500 GJ/h.



1 - hot water demands, 2 - heating demands,
3 - base heat source, 4 - peak heat source

Fig.1 CHS annual heat load variation diagram

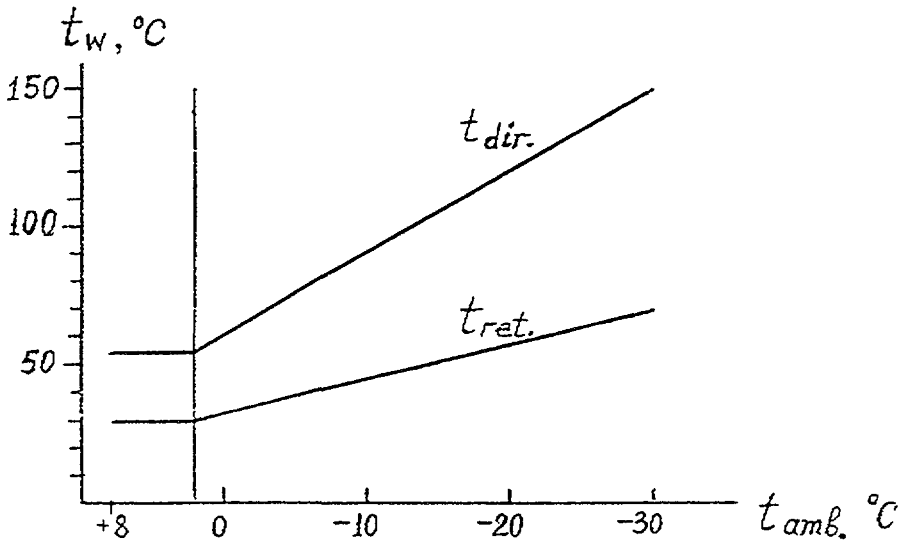
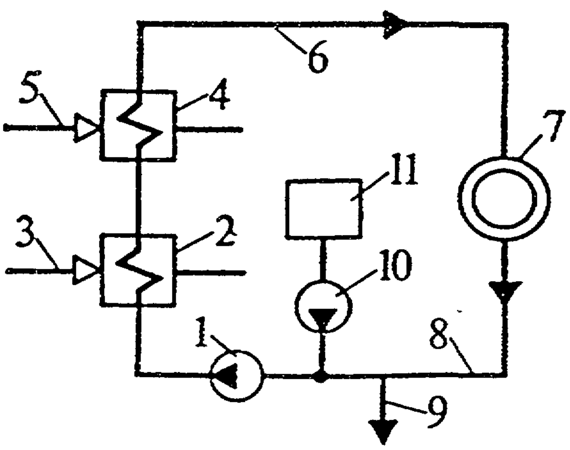


Fig. 2. Heating grid water temperature variation vrs. ambeint air temperature



1 - grid pump, 2 - main grid HX, 3,5 - steam from turbine bleeders, 4 - peak grid HX, 6 - direct water, 7 - heat consumer, 8 - return water, 9 -grid bleed, 10 -grid make-up pump, 11 - make-up water preparation facility.

Fig.3 NPP's heating facility and related CHS flow diagram

Due to the short duration of the period with lowest ambient temperatures a base plant's rated capacity has to be determined not by the maximum heating load in the CHS, but to maximize the plant operation factor,. A base source, therefore, has to supply a heating load which is required by consumers during a major part of the year. Share of the base source in provision of the CHS maximum heat demand is characterized by $_t$-factor, which value should be optimized on the basis of techno-economic analysis during CHS design development. If, for instance, co-generation power plant (CPP) is used in the CHS, it is necessary to take into account in the $_t$-factor optimization, that a reduction in steam flow extracted from a turbine plant when ambient air temperature rises will result in turbine efficiency deterioration. Usually, $_t$-factor adopted is in the range from 0.4 to 0.6.

Maximum temperatures of direct and return water (t_{dir} and t_{ret} respectively) in a full load heating mode of CHS operation are standardized and adopted as follows:

For large towns:	t_{dir}^{max} 150°C	t_{ret}^{max} 70°C
for small heating grids:	t_{dir}^{max} 130°C	t_{ret}^{max} 70°C

Fig.2 shows a typical graph of temperature variation in a heating grid (HG). A HG regulation by changing grid water flowrate ("quantitative" control) is adopted only for a narrow range of positive temperatures of ambient air (+2 -+8 ^{o}C). In the rest air temperature variation range "qualitative" regulation is provided by changing both grid water temperatures and temperature difference between direct and return water. Total gird water flowrate is determined for the maximum heat consumption mode of the CHS operation and remains constant over the whole range of ambient air temperature variation. Reduction in direct water temperature is provided by mixing with colder return water.

3. HEAT SUPPLY FROM CONDENSING NPPS IN THE RUSSIA FEDERATION

Even though historically heat supply was not formulated as an objective for nuclear power industry, first steps in that direction, had been made in the very beginning of the industry development. It is worth reminding that even the first NPP, operating since 1954, served for many years as a heat source for the Obninsk town's CHS. All operating NPPs are equipped with heating facilities including grid water heaters and provide heat for space heating, air conditioning and hot water supply to both the atomic station and related towns. Such towns have a population of 40,000 to 50,000, situated usually 3-15 km away from a NPP site and have heat demands ranging from 100 to 200 MW (360-720 GJ/h) or higher.

Heating facilities of NPPs are being operated from non-regulated steam bleeders of turbine plants and usually have two- or three-stage flow configuration (Fig.3). Heat production characteristics of operating NPPs are given in Table 1. A significant increase in NPP heat supply capacity may be seen as nuclear power units become more powerful.

TABLE I. OPERATING AND MODERNIZED NPPS HEAT OUTPUT PERFORMANCES

Characteristic		Reactor	types	
	VVER-440	BN-600	VVER-1000	RBMK-1000
1. Rated electric power, MW(e)	400	600	1000	100
2. Heat capacity of turbine plant steam bleeders, GJ/h	2 x 100	3 x 240	840	2 x 310
3. Heat capacity of modernized turbine bleeders, GJ/h	840		1900	
4. Steam pressure in turbine bleeders, Mpa				
* upper		0.3	0.6-0.8	
* lower		0.13	0.2-0.3	

NPPs operate under base electrical and heat loads, practically without regulation of energy extracted from steam bleeders. Steam from the turbine is extracted in the range of a turbo-generator power variation from 100 to 60 percent of rated power. Because steam pressure in turbine bleeders reduces proportionally to electric load of a turbo-generator at lower power, steam is supplied to heating facilities from a main steam collector through a pressure-reduction device. To cover the winter maximum demands, a special peak heater may be connected additionally to main heaters.

When NPPs supply heat to large CHSs conventional co-generation power plant plays the role of a load-following source of heat, as well as the town's peak load boilers which come on-line at the lowest ambient temperatures. A co-generation power plant also serves as a back-up source of heat in case the NPP is under a shut-down.

Radiological safety of nuclear heat consumers is provided by continuous monitoring of secondary coolant radioactivity. If radioactivity exceeds the permissible level, fast-acting isolation valves will automatically stop grid water supply to the NPP's heating facility. An additional protective barrier has to be used such as pressure difference between grid water (grid pumps head is 1.6-1.8 MPa) and heating steam in grid heaters. According to regulatory requirements maximum dose to an individual of the wrounding population from NPP-related heating grid must not exceed 0.01 mSv/year [2].

Significant savings of fossil fuel, from 50 to 100 thousand tons oil equivalent per year for one 1000 MW(e) power unit, is the first positive effect of heat extraction from condensing NPPs. Improvement of environmental conditions (less pollution) in NPP-satellite towns and settlements is the second positive factor.

Experience available in operation of relatively small CHSs at the working atomic stations, demonstrates the high reliability and safety of such systems, as well as technical feasibility for creation of more powerful and more effective CHSs connected to NPPs. Design studies of advanced steam turbines have confirmed the possibility to increase significantly heat capacity of non-regulated steam bleeders of NPP turbines - up to 1400 MW (~5000 GJ/h) from one 1000 MW(e) turbine plant [3]. It allows to create large CHSs covering several districts and towns with a big concentration of heat demands. A number of techno-economic studies have been carried out lately for such NPP-related CHSs consolidating heating grids of towns and settlements in a radius of up to 150 km from NPP sites [3,4]. Specific designs of CHSs powered from NPPs which are planned to be constructed at the Kola and Novovoronezh atomic stations have been studied recently [5]. Characteristics of heat loads for these CHSs with different radius of enveloped heat consumers in the vicinity of the NPPs are given below (Table 2).

Fig.4 shows the results of the analysis of showings for CHSs connected to NPPs (both operating and under construction) with VVER-440 and VVER-1000 PWRs, compared to separate production of heat and electricity from conventional plants and boilers (under conditions of central regions of Russia) [3]. One can conclude that utilization of the NPPs for heat supply to CHSs is economical over the entire range of considered heat load variation: the savings vary from 7-25% at moderate heat loads up to 37-48% for the largest CHSs.

Economic impact of such systems is primarily due to significant savings of fossil fuel (up to 5-6 million tons a year), elimination of numerous small low-efficient and polluting heating boilers, transfer of larger boilers to peak or stand-by mode of operation with a low operating factor, and a reduction in economic detriment from polluting releases of conventional energy sources. Economic competitiveness of nuclear-powered CHSs increases substantially as fossil fuel cost rises and NPP investment cost reduces. Thus, a significant rise in economic effect at heat loads below 5000 GJ/h in Fig.4 is explained by transition to larger units of VVER-1000 which have 1.6 times less specific cost than VVER-440 units.

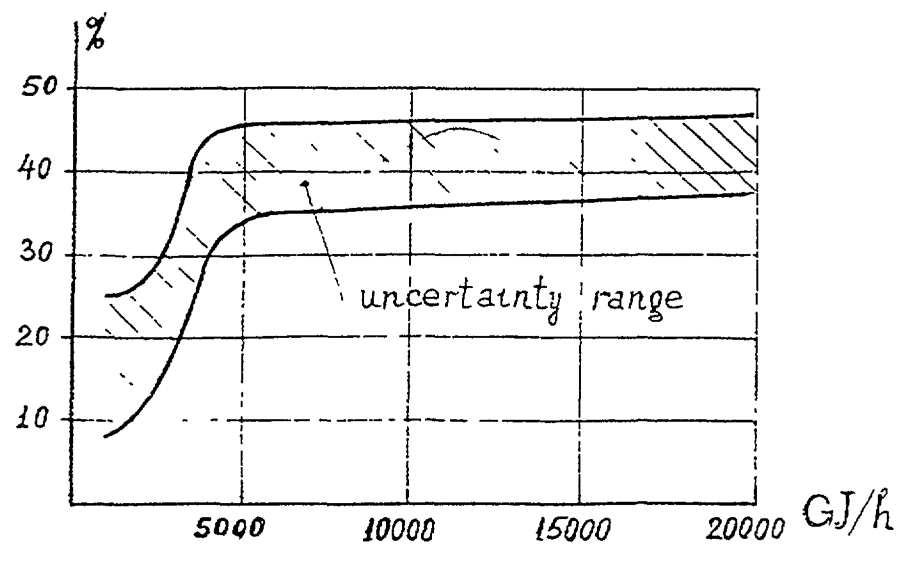


Fig.4. Relative economy of CHS expences vrs. maximum heat demands

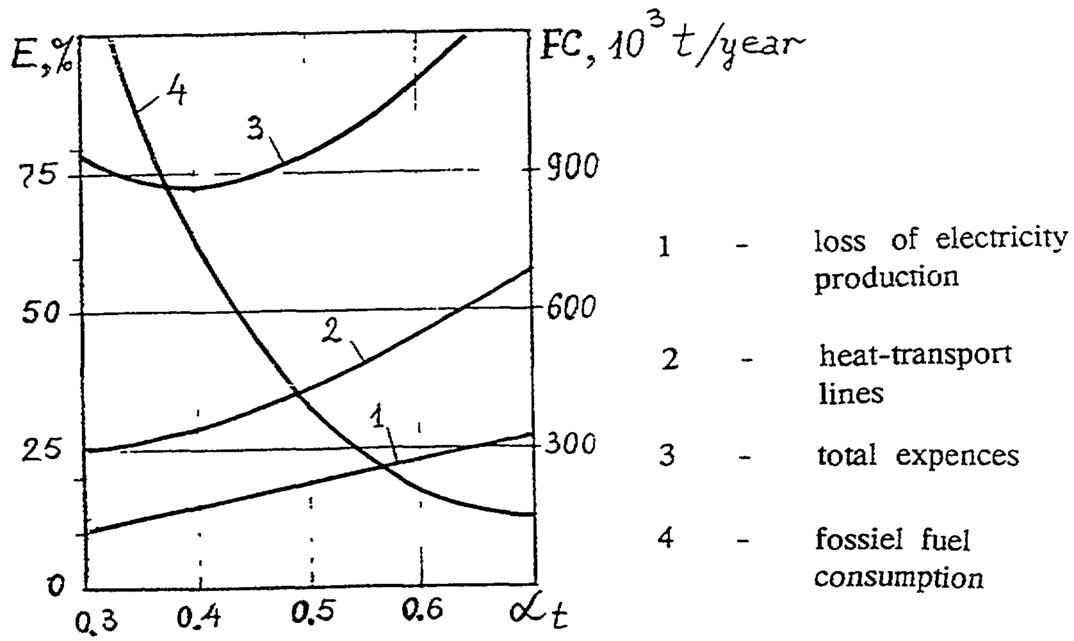


Fig.5 Reduced expences in NPP-based CHS (E) and fossil fuel consumption (FC) vrs. nuclear heat contribution factor α_t

Fig.5 gives an idea about variation of main components of expenses in a CHS with the nuclear heat contribution $_t$-factor [4]. Investigations showed that in large CHSs with heat load more than 5000 MW and large length of heat transport networks, the optimal value of $_t$-factor ranges from 0.3 to 0.5. In smaller CHSs, the optimal $_t$-factor rises to 0.6-0.7.

TABLE II. DESIGN CHARACTERISTICS OF CHSS BASED ON NEW NOVOVORONEZH AND KOLA NPPs (VVER-1000)

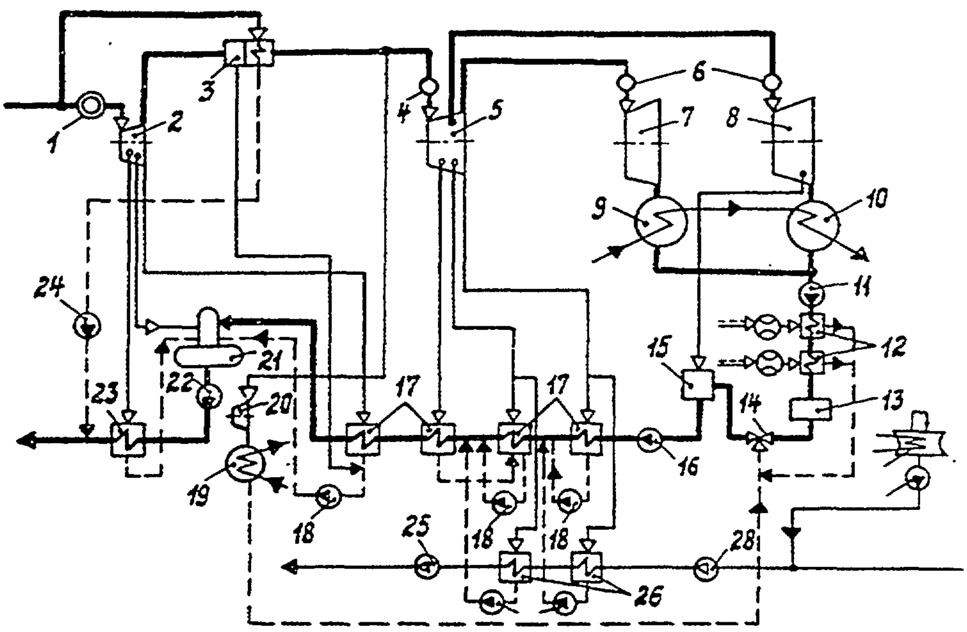
Consumers	Heat load	Distance from	NPPs site, km
	MW	Up to town's boundary	up to most remote consumer
1. CHS from Novovoronezh NPPs			
Total load	6950		
Contibutors:			
- Voronezh city	6310	40.0	55.0
- Novovoronezh town	640	50.0	7.0
2. CHS from Kola NPPs			
Total load	2665		
Contributors:			
- Apatit	2926	51.0	56.5
- Kirovsk	571	64.0	72.0
- Kandalaksha	446	22.5	28.0
- Polar Zori	360	7.0	10.0

4. Heat Supply From Nuclear Co-generation plants (CPPs)

Co-generation of electricity and heat with extraction of significant amount of heat from regulated bleeders of turbine plants is widely used in the power industry of Russia : CPPs give about a third of the total production of heat in the country. This method of reducing losses of heat in a steam cycle permits cutting the expenditure of fuel per kWh. For conditions in Russia, co-generation systems are more economic than separate production of electricity and heat in regions with density of heat demand more than 200 GJ/h per km^2 and maximum heat demand more than 2500 GJ/h [6].

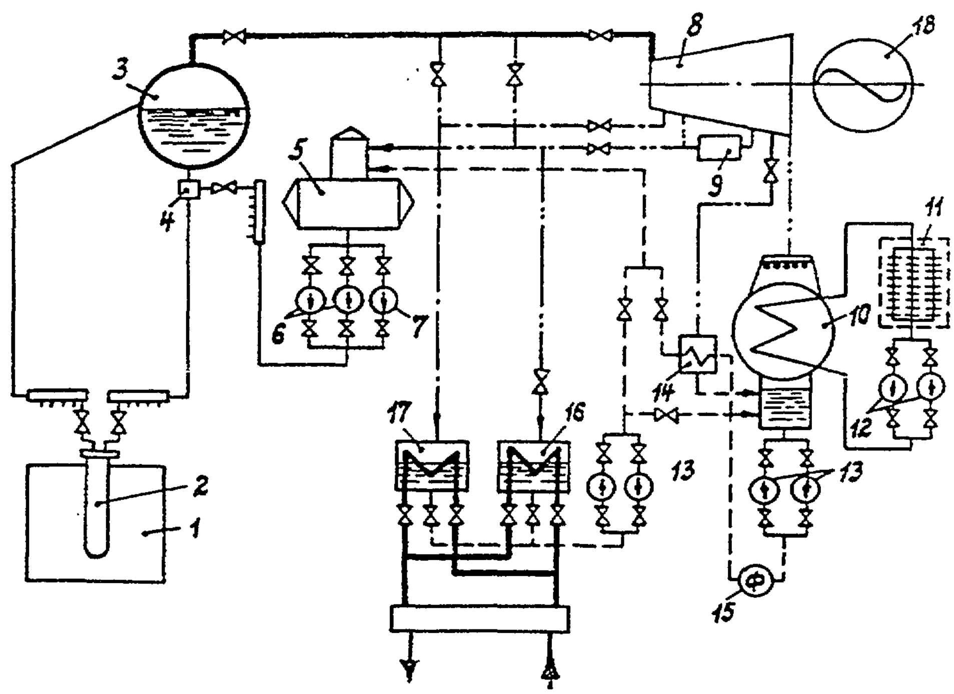
Economic efficiency of co-generation for NPPs, in general, is less than for conventional CPPs, due to lower initial parameters of steam. Large steam supply to a heating facility complicates saturated-steam turbine structure. Moreover, compared with conventional CPPs situated close to a town cost of heat transport and losses in hot water transmission lines are larger for nuclear co-generation power plants (NCPP). Therefore, pressure in steam bleeders of a turbine plant should be higher than increases under production of electricity. Nevertheless, as it was shown in the seventies, utilization of NCPPs might be economically effective in CHSs with large heat loads (more than 4000 GJ/h) at value of $_t$-factor in the range of 0.6-0.7 [6,7].

In the beginning of the eighties in ex-USSR, construction of two large NCPPs with VVER-1000 reactor plants was started near large cities of Minsk and Odessa. For those NCPPs specifically a steam turbine was developed with heat extraction capacity of 1800 GJ/h lower by a factor of 2 than that for bleed turbines of conventional CPPs). The turbine had two regulated steam bleeders and several non-regulated ones, that might be used for additional heating of grid water up to 200 C. In condensing mode of operation the turbine power is 500 MW(e), while in steam-extraction mode at full heat load the power reduces to 450 MW(e). Figu.6 shows the principal flow diagram of the NCPP and its heating facility.



1- stop valve, 2 - medium pressure cylinder, 3- intermediate separator-superheater, 4 - shut-off valve, 5, 7, 8 -low pressure cylinders, 6 - control valves, 9,10 - condensers, 11, 16 - condensate pumps, 12 - ejectors/seals steam coolers, 13 - condensate purification system, 14 - condenser level control valve, 15 - mixing reheater; 17 - low pressure reheater, 18 - drain pumps, 19 - drive turbine condenser, 20 - drive steam turbine, 21 - deaerator, 22 - feedwater pump, 23 - high pressure reheater, 24 - reheater condensate pump, 25 - second stage grid pump, 26 - grid heaters, 27 - grid heaters condensate drain pumps, 28 - first stage grid pump, 29 - grid make - up pump, 30 grid make-up system deaerator.

Fig.6 NCPP's turbine plant principal flow diagram



1 - reactor, 2 - fuel channel, 3 - steam drum, 4 - mixing device, 5 - deaerator, 6 - feed pumps, 7 - emergency feed pump, 8 - turbine, 9 -separator, 10 - condenser, 11 - air coolers, 12 - circulating pumps, 13 - condensate pumps, 14 - low pressure reheater, 15 - ion-exchange filter, 16 - main grid heater, 17 - peak heater, 18 - generator.

Fig. 7 Bilibino NCPP principal flow diagram

Though both projects were canceled after the Chernobyl accident under the pressure of public opinion, the interest and prerequisits for further development of nuclear co-generation are still retained in Russia due to the following reasons:

(i) There is a solid positive experience with operating NCPPs, such as Bilibino, as well as Tomsk and Krasnojarsk NCPPs based on weapon-grade plutonium production reactors.

(ii) Growth in heat load concentration in towns and a significant rise in fuel prices (both fossil and nuclear) enhance the economics of nuclear co-generation.

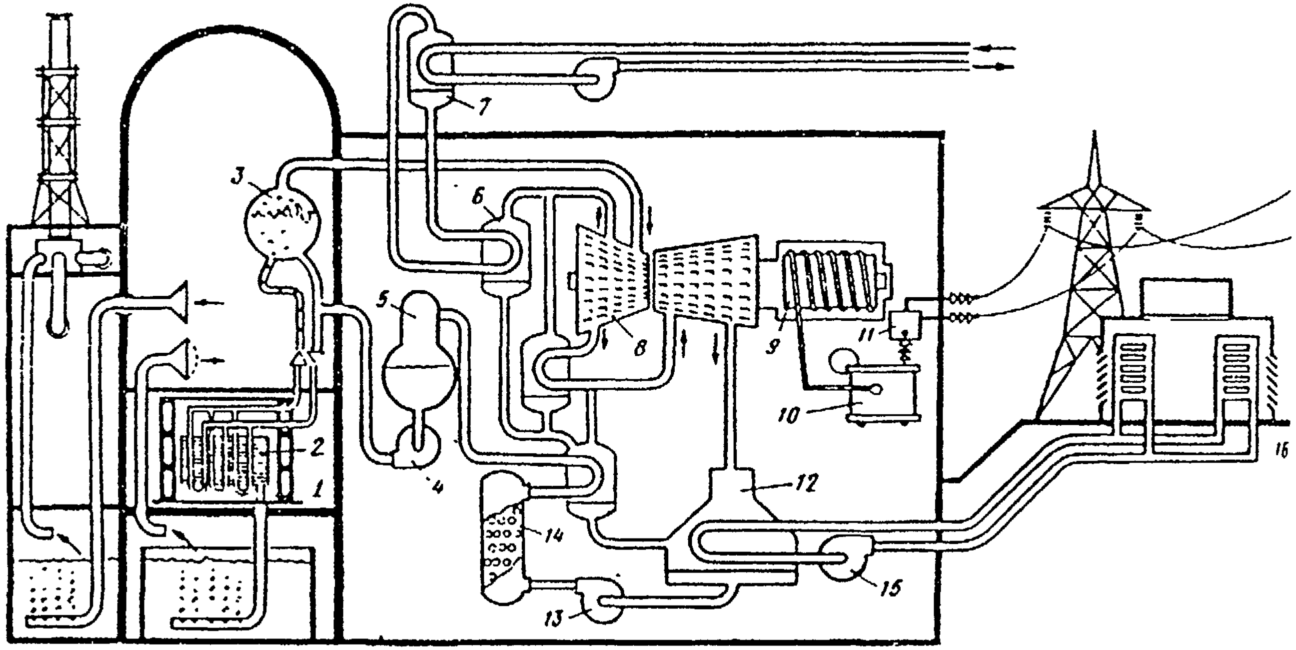
Long-term successful operation of the Bilibino NCPP convincingly demonstrated the economic efficiency and social significance of nuclear heating under specific conditions of the extreme North. Bilibino is the administrative center of important gold-mining region in Chuckot peninsula, far away from the Polar Circle. Severe climatic conditions with extremely low air temperature in winter (down to -60 C) are characteristic for this region, as well as long lasting heating period. Proceeding from the necessity to meet the relatively large demands in heat a NCPP has been adopted as the base energy source for the region. The NCPP comprises of four power units of 12 MW(e) each, that were commissioned in 1974-1976 [8]. At rated electric power, up to 280 GJ/h of heat can be extracted, while maximum heat output (with electric power reduced to 40 MW) is equal to 480 GJ/h (Table 3).

12 MW(e) steam turbines with two regulated and two non-regulated steam bleeders are used in the NCPP. Fig.7 shows the heat transport flow diagram : steam from the turbine bleeders heats water of transmission line (secondary circuit) in the heating facility up to 150 C. This water is supplied to grid HXs located 3.5 km away in a district heating post of the town. Town's heating and potable hot water (65 C) supply grid forms the third heat-transfer circuit. Pressure in heat transmission line is maintained higher (P_{min} = 0.2 MPa) than pressure of heating steam, thus radiological safety of heat consumers is provided. Many years of experience show that the specific radioactivity of grid water is the same as of input water from the local water pond.

TABLE III. BILIBINO NCPP DATA

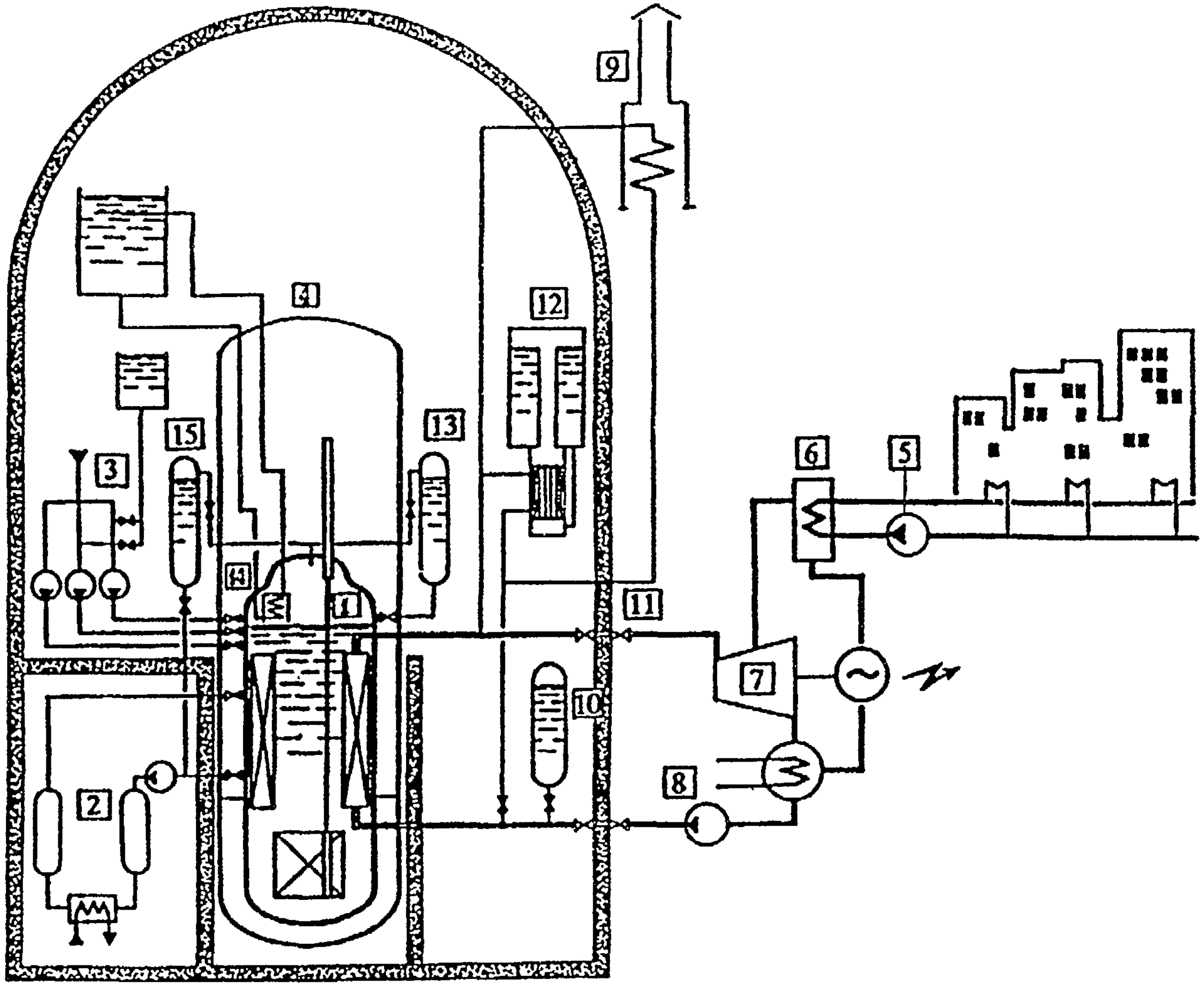
Number of reactor plants	4 (EGP-6)
Reactor type	Channel, uranium-graphite, boiling water, director cycle
Nuclear fuel	UO_2+Mg dispersion, tube configuration
Reactor thermal power MW	62
NCPP rated electric power	48 (4 x 12)
Electric load variation, %	280-480
Reactor coolant parameters	50-100
- pressure, Mpa	6.4
- temperature, °C	275
Transmission line direct	
water temperature, °C	150
- direct	95
- return	70
Steam pressure (max) to water heaters, Mpa	1.5
Heat transmission line pressure, Mpa	1.8
Heating grid pressure (max), Mpa	0.6

The NCPP economic indicators turned out to be much better than those for conventional energy sources operating in the region, e.g. cost of heat is 2-2.5 times lower than that from local heating boilers.



1 - channel-type reactor, 2 - reactor core, 3 - steam drum, 4 - feed pump, 5 - deaerator, 6 - heating facility HX, 7 - grid HX, 8 - turbine, 8 - generator, 10 - transformers, 11 - 110 kV switchgear, 12 - condencer, 13 - condensate pump, 14 - condensate clean-up facility, 15 - raw water pump, 16 - air coolers.

Fig.8. Advanced CNPP with ATU-2 nuclear reactor



1 - reactor unit, 2 - reactor coolant purification facility, 3 - water/boric acid make-up system, 4 - guard vessel. 5 - grid pump, 6 - grid HX, 7 - turbo-generator, 8 - feed pump, 9 - air cooler, 10 - secondary coolant storage tank, 11 - containment, 12, 14 - ERHR trains, 13 - emergency core cooling system, 15 - boric acid injection system

Fig. 9. ATEC - nuclear co-generation plant principal flow diagram

The plant solved the fuel and energy supply problem - it saves some 230,000 tons of fossil fuel a year, thus providing for the economic development of the region and creating qualitatively new conditions of life for the local population (reliable heating and hot water provisions, swimming pool, large greenhouse, improved environmental conditions in the town, etc.).

Taking into account, the high efficiency and positive operational experience of the Bilibino nuclear reactors, the design of advanced nuclear reactor (also of channel type) ATU-2 (40 MW(e), heat capacity of 210 GJ/h) has been developed recently [9]. The new NCPP is designed in compliance with the updated requirements for safety, e.g. an additional heat-transport circuit is introduced between heating steam and a grid HX in the nuclear plant flow diagram (Fig.8). Three such reactors were supposed to substitute the existing nuclear reactors of the Bilibino NCPP which are to be removed from operation in 2001-2004.

A number of medium size power units ATEC-80, -150 and -200 for nuclear co-generation plants has been developed recently on the basis of integral PWR [10]. Proven technical decisions and established technology of marine nuclear reactors are used in the designs, along with engineering features and technology of heat-only reactors created in the eighties (see next chapter). The level of safety attained in the design meets the requirements applied to advanced nuclear reactors and removes the sanitary restrictions on deployment of such reactors in heat-consuming centers. Table 4 gives main design characteristics of ATEC- type NCPP, Fig.9 shows its principal flow diagram.

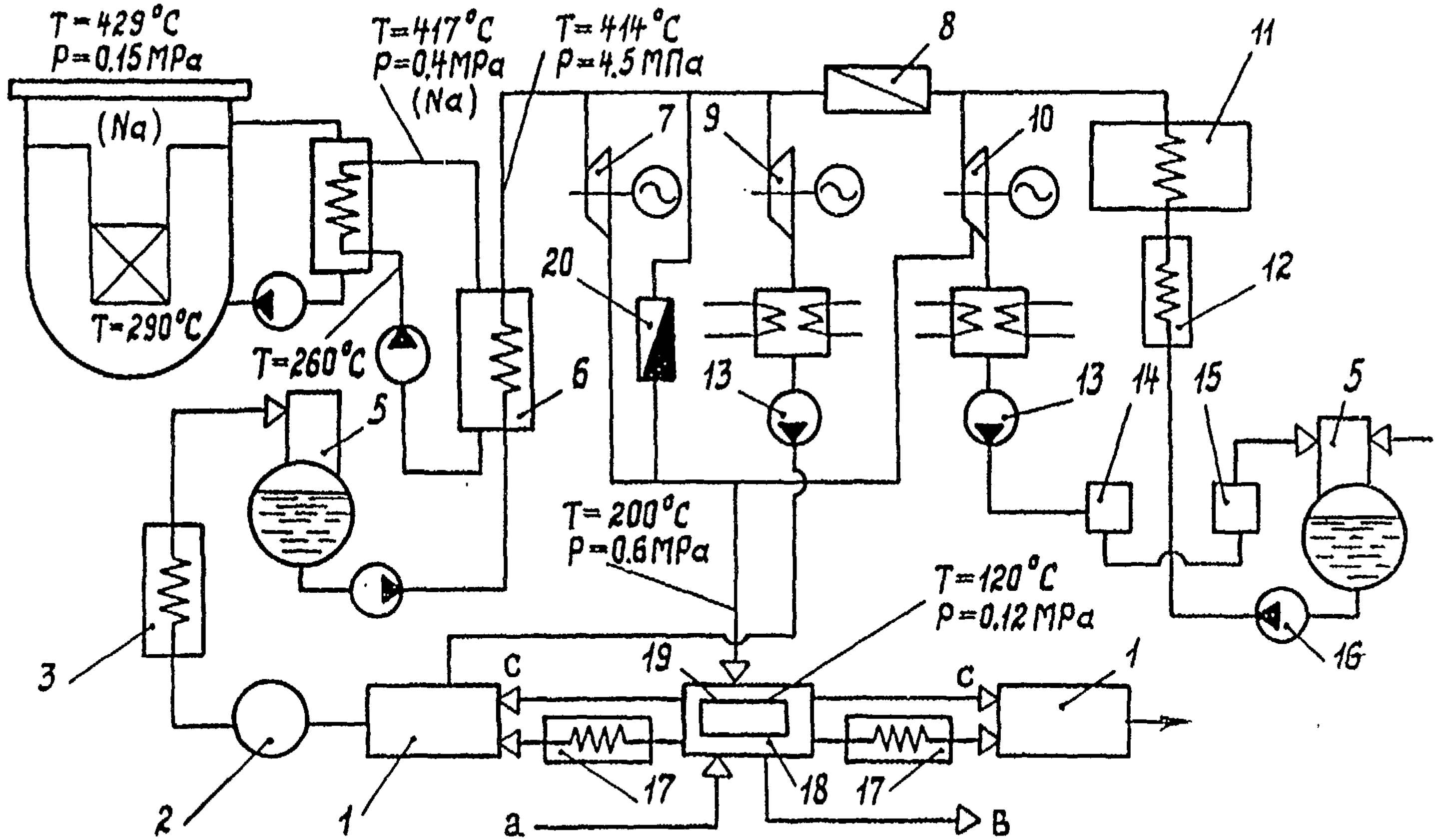
TABLE IV. ATEC-200 DESIGN CHARACTERISTICS

Reactor thermal power, MW	750
Electric power, MW	
- maximum (condensing mode)	250
- heat extraction mode	180-50
Heat output range, GJ/h	1575-2520
Main steam parameters	
- pressure, Mpa	4.4
- temperature, ^oC	290
Grid water temperature, ^oC	
- direct	150
- return	70
Max. distilled water output (*),m^3/day	290,000
Power demand for water desalination	20 MW(e)

(*) This amount of potable water could be produced when there is no heating load.

These power units have been designed as universal power sources capable of producing electricity and heat in different proportions, to desalinate sea water, to provide steam for industry according to specific demands of the consumers.

At present, the project for deployment of the ATEC-200 units on the site of a large nuclear fuel cycle enterprise in Krasnojarsk region is under consideration. Main objective of the project is to substitute a NCPP operating there for a long time on the basis of a plutonium-production reactor, which has to be decommissioned in 2000. The present demands in heat and electricity of both the enterprise and the town (3150 GJ/h) can be provided by two ATEC-200 power units. Underground location of the new units is planned in existing rock caverns, as well as utilization of available hot water transmission line to a satellite town (5 km). Taking account of its deployment on a site of the large production works, it is proposed to use the NCPP for process steam (1.2 MPa) production as well. The design of a boiling water reactor VK-300 is also considered as an alternative.



1 - water preparation facilities, 2 - vacuum deaerators, 3 - desalted water heater , 4 - feed pumps, 5 - 0.7 MPa deaerators, 6 - steam generators, 7 - back-pressure turbines, 8 - 10/4.5 pressure reduction device, 9 - condensing turbine, 10 - conventional CPP's turbines, 11 - CPP's boilers, 12 - high pressure reheaters, 13 - condensate pumps, 14 - condensate purification facility, 15 - low pressure reheaters, 16 - CPP's feed pums, 17 - heating steam condensate coolers, 18 - desalination facility, 19 - desalinator, 20 - 4.5/0.5 pressure reduction device, a - sea water, b - brine removal, c - distillate.

Fig.10. BN-350 nuclear power - desalination complex principal flow diagram

5. PROCESS HEAT SUPPLY FROM NUCLEAR REACTORS AND SEA WATER DESALINATION

In Russia, industry consumes approximately two times more low-and medium-grade heat than the household sector. More than 60% of the heat is provided by low-pressure steam (dominant pressure level from 0.5 to 1.8 MPa). About 40% of heat consumed by industry is used immediately in technological processes, as well as for heating, air conditioning and hot water supply. Large enterprises have usually their own sources of energy, such as CPPs. Hot water supply for small enterprises can be provided from the town's CHS. In the case, when NPP operates as a base source of heat in a CHS, it bears usually an industrial heat load as well.

The first design of a nuclear plant developed specifically for production of process heat (steam) was realized in a NCPP that has been operating in Shevchenko (nowadays Actau, Kazakhstan). This region represents a lifeless stony desert on the eastern shore of the Caspian sea, rich in various minerals (incl. uranium ore) but deprived in potable water sources. From 1972, the first nuclear power-desalination complex in the would has been operating there with a NCPP ussing the fast reactor, BN-350, and a large seawater desalination plant [11]. Table 5 gives main characteristics of the NCPP and desalination facility, Fig.10 shows the complex flow diagram.

TABLE V. BN-350 NUCLEAR REACTOR AND DESALINATION PLANT CHARACTERISTICS

Reactor thermal power, MW	520 (750)
Electric output, MW(e)	125
Primary sodium temperature, ^{o}C	
- outlet	437
- inlet	288
Main steam parameters	
- temperature, ^{o}C	405
- pressure, MPa	4.5
Desalinator type	multi-stage evaporator
Number of desalinators	9
Desalinators capacity, t/day	5 x 15000
	3 x 14400
	1 x 12000
Total distilled water output, t/day	~100.000
Specific production of distilled water, t water/t steam	7.8-8.1

Back-pressure bleed turbine plants (N=37.5 MW(e)) and one condensing turbine are used in the NCPP. Exhaust steam after the back-pressure turbine (0.6 MPa, 200$\pm$ 20 C) is supplied to desalinating facilities which represent multi-stage tube-type evaporators operating on the principle of multiple evaporation of sea water. Distilled water produced in the facilities is used to prepare water of drinkable quality. There is a conventional CPP in the complex, which is used as a back-up source of heat for the desalination facility. During 23 years of operation, the plant has demonstrated the reliable and safe performance of nuclear desalination complex playing the essential role of a water source for the population, industry and agricultural enterprises in the region. Availability of well developed sea water desalination technology proved by long-term operation of the prototypes gives grounds for development and creation of various power-desalination complexes of required capacity.

For production of process heat of medium (up to 500 C) and high (up to 950 C) temperatures a number of high temperature gas-cooled reactors (HTGR) was developed in ex-USSR in the 1980s [12]. Different combinations of those reactors with most energy-intensive industrial processes in chemistry, oil-refinery, petrochemistry, oil production industries etc. were deeply studied [13]. Fig.11 shows spheres of possible heat applications. The following were considered as the most promising ones: production of

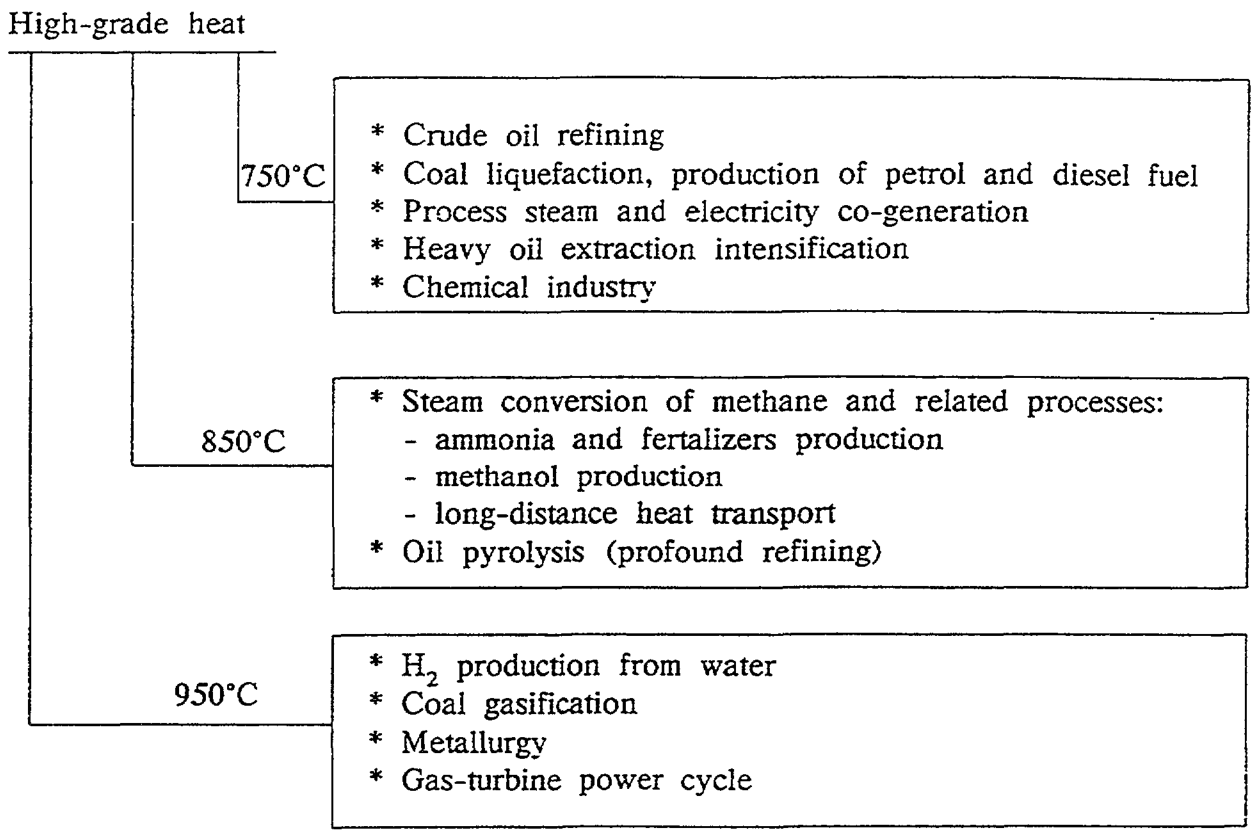


Fig. 11. HTGR process heat applications

ammonia, fertilizers, methanol (synthetic fuel), hydrogen and pyrolysis of oil. An interesting feature of high-temperature heat transmission in chemically bounded form over very long distances (up to 200-300 km) were pointed out. In this case, heat from a HTGR is used initially for steam conversion decomposition) of methane. Mixture of the decomposition products (H_2, CO, CO_2) in "cold" state under excess pressure are pumped through pipes to remote consumers, where a reverse reaction of methane synthesis is performed. The reaction goes on with heat release at temperatures up to 500-600 C. The methane is then returned to the NPP. Advantage of this methodlies in the possibility to cover a large number of dispersed heat consumers by a single large (and consequently economic) nuclear source. Cost of heat transport can be reduced significantly - at least 2.5 times compared with hot water transmission lines.

6. NUCLEAR DISTRICT HEATING STATIONS

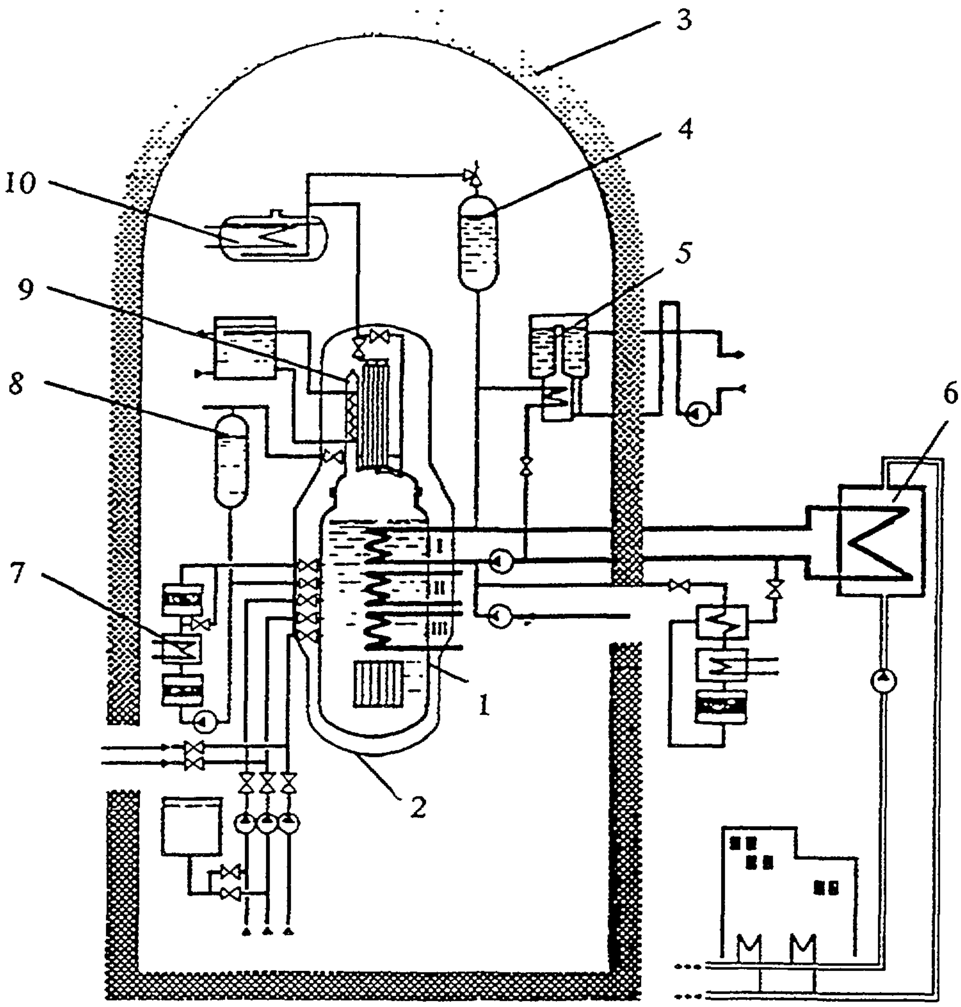
For heating of large towns with significant heat demands and expensive fossil fuel, a heat only low-temperature reactor AST-500 was developed in the late 1970s [14]. From the very beginning, the reactor had been developed as a heat source of enhanced safety to exclude practically any risk to the population from a nuclear heating station (NHS) situated near the center of heat loads. Minimal alienated area and small needs in raw water facilitate NHS siting in the vicinity of towns and in densely populated regions. This simplifies heat transport, while reduced parameters of a reactor coolant and elimination of energy-conversion system give possibility to simplify the structure of a heating reactor and to reduce capital cost of NHS and costs to the CHS.

In the early eighties, construction of two pilot 1000 MW NHSs was started in the cities of Gorky and Voronezh. The NHSs was considered, in that period, as the first stage of an extensive nuclear heating development programme in the country. Significant R&D work was carried out to validate the AST-500 design. Fabrication of the reactor components was established and assembly of the first reactor was almost 90% complete by 1989. Comprehensive safety review of the AST-500 was carried out by an IAEA team in the framework of a Pre-OSART mission, with a positive conclusion in respect of both design decisions and its implementation in Gorky NHS.

According to the design, a twin-unit station in combination with conventional peak sources of heat is capable of providing heat for a CHS with maximum heat demands of up to 2300 MW (8280 GJ/h), which corresponds to a town with 350-400 thousand inhabitants [15]. Radiological safety of consumers is provided by three-circuit heat transport scheme, with a pressure barrier between the second (intermediate) and third (grid) circuits (Fig.12). The NHS enables closing about 300 small heating boiler plants which are sources of significant air pollution. At base load operation for the entire heating period (4500-5500 h/a), the NHS provides about 78% of annual heat consumption of a town, producing more than 16 million GJ of heat, equivalent to burning some 700,000 tons of fossil fuel (oil equivalent). Substitution of such quantity of fuel means that the NHS reduces emission of SOx by 20,000 t and NOx by 2000 t a year. However, in spite of significant anticipated socio-economic benefits and positive conclusions of the IAEA review team on the safety level of the plant, construction of all NHSs in the country has been suspended in the early nineties under the pressure of political factors.

The influence on NHSs economics of such factors as the level of heat loads in a CHS, cost of fossil fuel, operating factor etc. has been investigated lately [16]. As it was shown, NHS efficiency rises with increases in heat loads in a CHS and fossil fuel cost.

It was concluded that the following decisions might be adopted to enhance economic efficiency of NHSs: increase in rated thermal power, connection to a NHS additional heat consumers, production of process steam and electricity, that would enable use of underloaded heat capacity of a NHS in summer, when main (heating) load is not in demand. A study of the potential of NHSs showed that up to 120% increase in rated power can be attained in periods of seasonal rising ambient temperature by lowering the return grid water temperature. There is also a possibility to use a NHS to provide potable hot water demands in summer.



1 - reactor, 2 - guard vessel, 3 - containment, 4 - pressurizer, 5 - RHR HX, 6 - grid HX, 7 - reactor coolant purification system, 8 - boric acid storage tank, 9 - RHR condenser, 10 - bubbler tank

FIG. 12. AST-500 principal flow diagram

To produce electricity a NHS is coupled with a turbine plant operating on saturated steam with pressure of 0.12-0.13 MPa. Maximum electric capacity of one AST-500 unit ranges from 15 to 50 MW(e) depending on the turbine type (condensing or back-pressure one). Selection of a configuration for each design depends on the specific structure and value of heat demands and other particular conditions.

Even though the nuclear heating programme has suffered due to adverse public opinion in the post-Chernobyl period, the prospects for NHSs are still encouraging. State review of ecological safety completed this year for Voronezh NHS, resulted in a favorable conclusion on the possibility to resume construction of the station. Moreover, a special decision of the local authorities has been released removing the hold on construction. On receipt of the official permission from the state regulatory body GAN, the station construction will be resumed in 1996.

Besides, a project for construction of a twin-unit NHS on the site of a Siberian chemical plant is under consideration now. The main objective of the project is to substitute an operating heat source at Tomsk (18 km away, heat output of 2000 GJ/h) consisting of two plutonium-production nuclear reactors which will have to be decommissioned by 2000 in compliance with the Treaty for strategic weapons reduction. To shorten the station construction period it is supposed to use available components of the AST-500 reactor plant erected on the canceled NHS in Nizhny Novgorod (former Gorky).

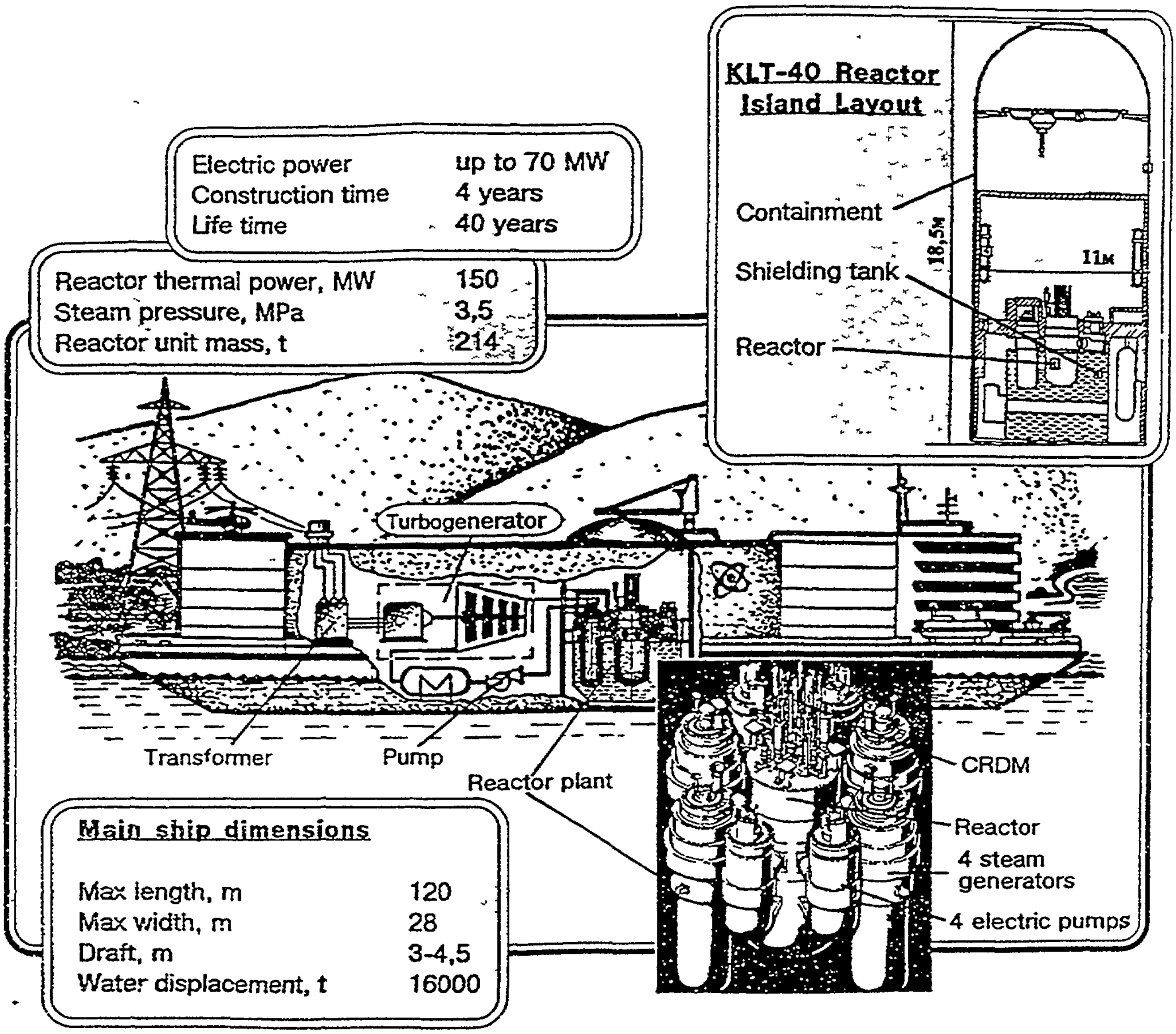
Besides, the AST-500, a number of small heating reactors is under development in Russia with distinctive design approaches and features. One of the designs - "Ruta" represents a number of pool-type nuclear reactors of atmospheric pressure with heat capacity of 10, 20 and 55 MW [17]. A feasibility study is being carried out now for construction of a NHS with these reactors in transpolar town Apatit on Kola peninsula.

7. SMALL NPPS FOR HEAT AND ELECTRICITY PRODUCTION

Activity on small nuclear plants (SNPs) for implementation in remote and difficult-to-access regions to provide electricity and heat supply to isolated consumers has been started in the ex-USSR as long ago as the fifties. Particular interest in this energy source is traditionally connected with the existence in the North and North-East of Russia, vast territories (more than a half of the country's total area) where production and delivery of energy are especially difficult tasks. It is connected with the necessity in large-scale seasonal transportation of fuels over distances of thousands of kilometers under severe natural conditions. Therefore, cost of fuel in some regions of Extreme North is determined almost completely (80-90%) by transport cost, and the cost of electricity and heat turned out to be 10-20 times higher than that in other regions of the country.

Due to the very low density of population and relatively small demands in energy, remoteness of consumers from centralized energy-supply systems, small self-balancing power grids and autonomous energy sources are prevailing there. So, in the northern regions of Russia more than 10,000 small power plants (200 kW capacity in average) mainly of a diesel type are being operated, along with many thousands of heating plants of several GJ/h average capacity. Under such very specific conditions SNPs can be considered as an economically acceptable alternative to traditional sources of energy.

Practical utilization of nuclear fuel for energy production in the Far North was started in Russia by putting into operation the Bilibino NCPP (1974). In the mid eighties a comprehensive study had been carried out by a large group of specialists, concerning prospective demands in electricity and heat for settlements over the northern regions of Russia and energy sources necessary to meet those demands [17]. It was concluded in the study, that for some 90 settlements SNPs could be considered as an acceptable alternative, and co-generation nuclear plants of 6, 12, 25 and 40 MW(e) unit power were selected as the most appropriate sizes for this purpose. The first programme planned for the period until 2000 envisaged construction of SNPs on 33 sites in the North, but it was canceled after Chernobyl accident.



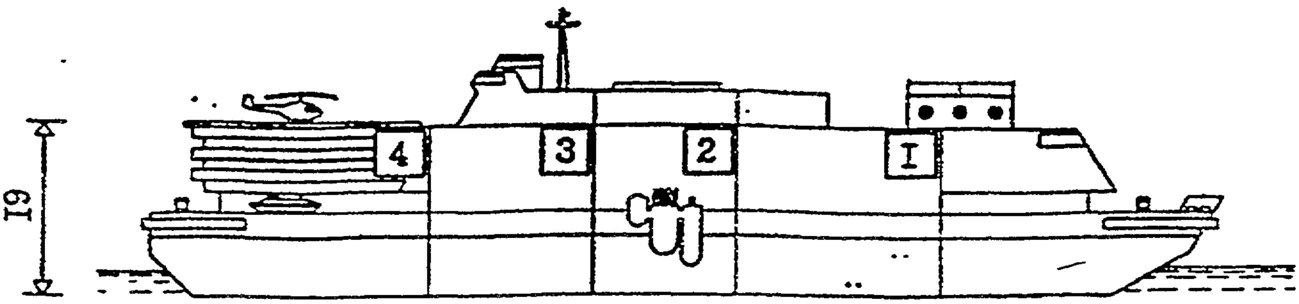


Fig. 13. Twin-KLT-40 floating NCPP

In 1989 a new proposal was validated on construction of SNPs on nine sites situated nearby large gold-mining enterprises in the regions of northern and eastern Siberia [18]. In that period, development of a number of advanced nuclear reactors was started specifically for SNPs to meet the updated safety requirements and to correspond better to the specific conditions of operation in northern regions. A common feature these designs is universality in application options by virtue of the capability to provide energy supply to various consumers with widely different levels and structure of electrical and heat demands. The designs ready for implementation are based on utilization of marine nuclear reactor technology, especially well developed in Russia [19].

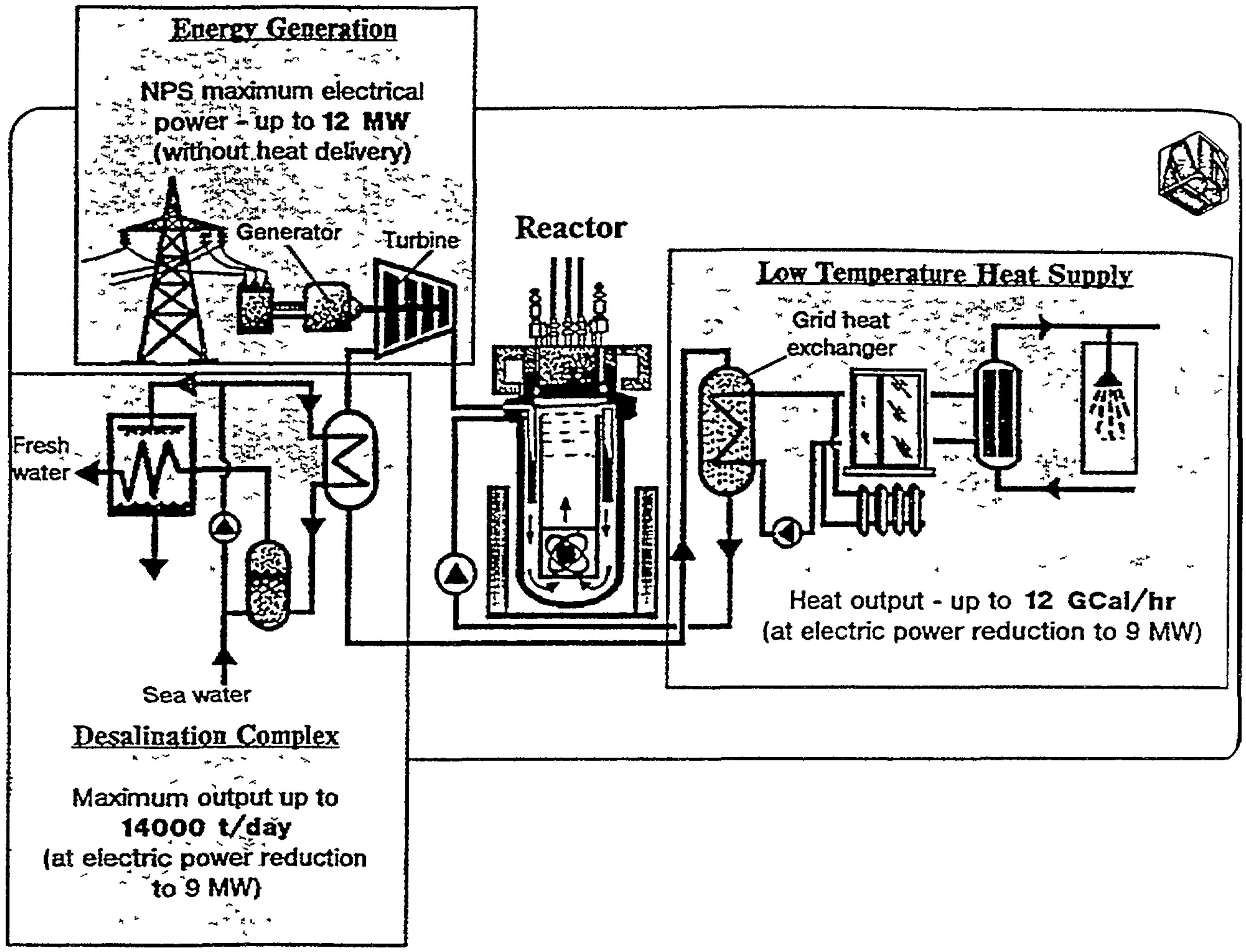
The KLT-40 NPP is based upon serially produced nuclear steam supply systems (NSSS) used in atomic ice-breakers and in ocean-going lighter-carrier "Sevmorput" [20]. Long and highly successful operational experience with this reactor under severe Arctic conditions (total operating record exceeds 140 reactor years), in combination with established technology of nuclear ship construction, operation and maintenance, gave grounds to propose a floating small-power NCPPs with two NSSSs of KLT-40 type.

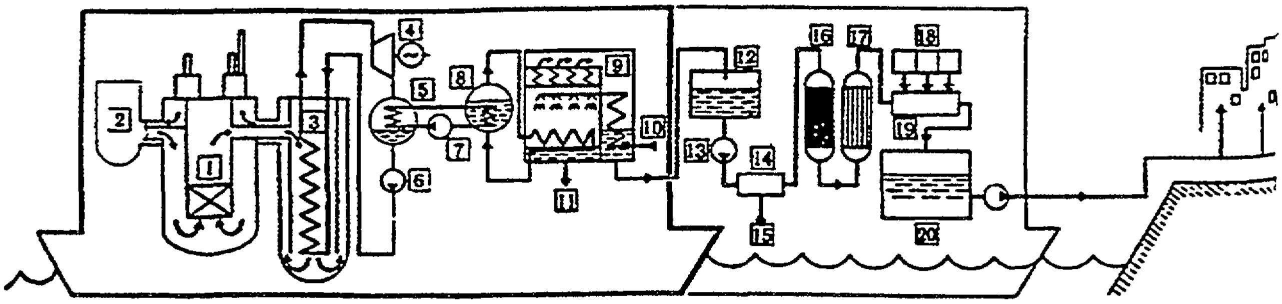
The floating SNP represents a special non-self-propelled ship (Fig.13) which will be built and equipped with NCPP completely at a specialized shipyard, similar to a nuclear icebreaker. After the completion of trial tests and commissioning on a turn-key basis the floating SNP will be towed by water to the region of deployment. There the plant is connected to shore-based transmission lines (electrical and heat-transport) and can be commissioned quickly. The plant is designed for siting in protected water areas (artificial bay or backwater area) of seas or rivers. There is a possibility to change the plant location, if necessary. The entire complex of maintenance work for floating SNPs, including repairs, refueling of the reactors and evacuation of radioactive waste can be fulfilled by available atomic fleet base-ships using established technology. After expiration of the plant lifetime it will be towed to the atomic fleet base in Murmansk to provide all necessary work for decommissioning. These features allow significant reduction on the capital cost of the plant, faster construction and improvement in construction quality, elimination of the need for a large-scale industrial building and creation of local infrastructure under specific conditions of the extreme North. The construction time of the pilot floating SNP is only four years, and subsequently 2 or 3 units could be built in a year. Main characteristics of the SNP with KLT-40 NSSS are given in Table 6, along with data on an other design of a smaller floating plant based on the ABV integral PWR [21]. This design is also ready for construction.

TABLE VI. MAIN CHARACTERISTICS OF FLOATING SNPs

Characteristic	Reactor type	
	KLT-40	ABV
NSSS rated thermal power, MW	160	6
Number of NSSSs	2	2
Max. electric output, MW	2 x 35	2 x 12.5
Electric output, MW	2 x 25	2 x 8
at simultaneous heat output, GJ/h	2 x 400	2 x 85
Operating range of power variation, %	10-1000	10-100
Average load factor, %	up to 85	up to 85
Draught of ship, m	4.5 (3.5)	2.5
Displacement of ship, t	16000	8700
Oprating personnel	55	45

Both the SNPs are designed to operate over a wide range of loads with various ratios of electrical and heat demands. Along with hot water supply for households, a process steam and potable water production can be also provided. At present, feasibility study is under way for building a pilot floating SNP in a large transpolar port Peveck on a coast of the Arctic Ocean.





1 - reactor, 2 - reactor coolant pump, 3, 8 - steam generator, 4 - turbo-generator, 5 - condenser, 6,7 - secondary and intermediate circuit pumps, 9 - distillation desalination facility, 10 - sea water intake, 11 - brane - removal, 12 - distillate storage tank, 13 - potable water preparation system pump, 14 - mixing device, 15 - H_2CO_3 solution, 16, 17 - filters, 18 - water fluorization, cloriding and stabilization facility, 19 - mixer, 20 - potable water storage tanks.

Fig. 14. APVS-80 nuclear floating sea-water desalination plant

On the basis of KLT-40 NSSS, the design of floating sea-water desalination plant (80,000 m^3/day potable water capacity) has also been developed (Fig.14). This design has been thoroughly reviewed in the framework of the special IAEA programme on nuclear desalination and attracted considerable interest. Small reactors of this type allow large-scale potable water production without off-site power consumption and are independent of the availability of indigenous industrial infrastructure.

8. CONCLUSIONS

Extensive experience has been gained in Russia in the field of nuclear heat applications, particularly for district heating and hot water supply to both household and industrial consumers. Notwithstanding serious restrictions on the national nuclear power development caused by macro-economic and social (public opinion) factors, requirements for nuclear heat applications are still existing in the country. Factors favouring nuclear heat applications are a significant rise in fossil fuel prices, fuel shortages in a number of industrial regions and adoption of more stringent requirements for environmental protection.

Investigations performed lately have validated that the following directions in nuclear heat applications have good prospects for the conditions characteristic of Russia:

1) Further expansion of heat supply from NPPs (both operating and under construction) by increase in steam extraction from nonregulated steam bleeders of condensing turbines. At present it is the most proven and studied method of nuclear heat utilization. Its reliability, safety and economic efficiency have been validated by operational experience in Russia.

2) Building of heat only nuclear stations in towns where acute shortage of heat generation capacities exist. Currently, expensive fossil fuel is used and the environment is polluted heavily. The enhanced safety reactor plant AST-500 has been developed to this end. The plant has proven engineering features and established technology of manufacturing and erection. There is a potential for further improvement of the NHSs economics. The AST-500 design has been reviewed by an IAEA team in the framework of a Pre-OSART mission in 1989. Resumption of building on the Voronezh site of two AST-500 units is awaiting clearance. Some other projects are also under consideration.

3) Creation of universal co-generation NPPs for combined production of electricity and hot water (for household and industry) and/or process steam, as well as for potable water production by sea water desalination.

For this aim advanced reactor plants and nuclear power units can be used which are characterized by enhanced safety, improved reliability and better economics.

Economic viability of either of the nuclear heat sources depends greatly both on the structure of energy demands and level of expenditure on fossil fuel for the given region. For isolated and remote regions with expensive fuel and small decentralized energy requirements (e.g. regions of the North of Russia) even small nuclear co-generation plants (up to a few tens of MW(e) could be an economically attractive alternative.

A number of small CNPP designs has been developed ranging from 10 to 250 MW(e) with heat capacity from 80 to 2000 GJ/h. Most of them are based on well-developed and established technology, have operating or manufactured prototypes and are ready for implementation.

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75 MW HEAT EXTRACTION FROM BEZNAU NUCLEAR POWER PLANT (SWITZERLAND)

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Abstract

The district heat extraction system installed and commissioned at the Beznau Nuclear Power Plant 1983 and 1984 is working successfully since the beginning. Together with a six kilometres extension in 1994, the system now consists of a 35 kilometres main network and 85 kilometres of local distribution pipelines. The eight founding communities as well as three networks joined later have been connected. Today around 2160 consumers of the Refuna district heating, small and large private buildings, industrial and agricultural enterprises are supplied with heat from the Beznau plant (1997: 141'000 MWh). The regional district heat supply system has become an integrated part of the regional infrastructure for around 20'000 inhabitants of the lower Aare valley. Nearly 15 years of operational experience are confirming the success of the strict approval conditions for the housing connections. Remarkably deep return flow temperatures in the district heating network were leading to considerable reserves in the transport capacity of the main pipeline system. The impacts of the heat extraction from the Beznau nuclear power plant, in particular its contribution to the protection of the environment by substituting fossil fuels and preventing CO_2-production, have been positive.

1. Swiss Nuclear power plants with Heat Extraction

Five nuclear powers plants (Beznau PWR 1 and 2, Gösgen PWR, Leibstadt BWR and Mühleberg BWR) with a total electrical capacity of 3'077 MW net today are in operation in Switzerland. These stations yearly account for over 40 percent of national electricity supply, only in winter time even for as to 70 percent (Fig. 1).

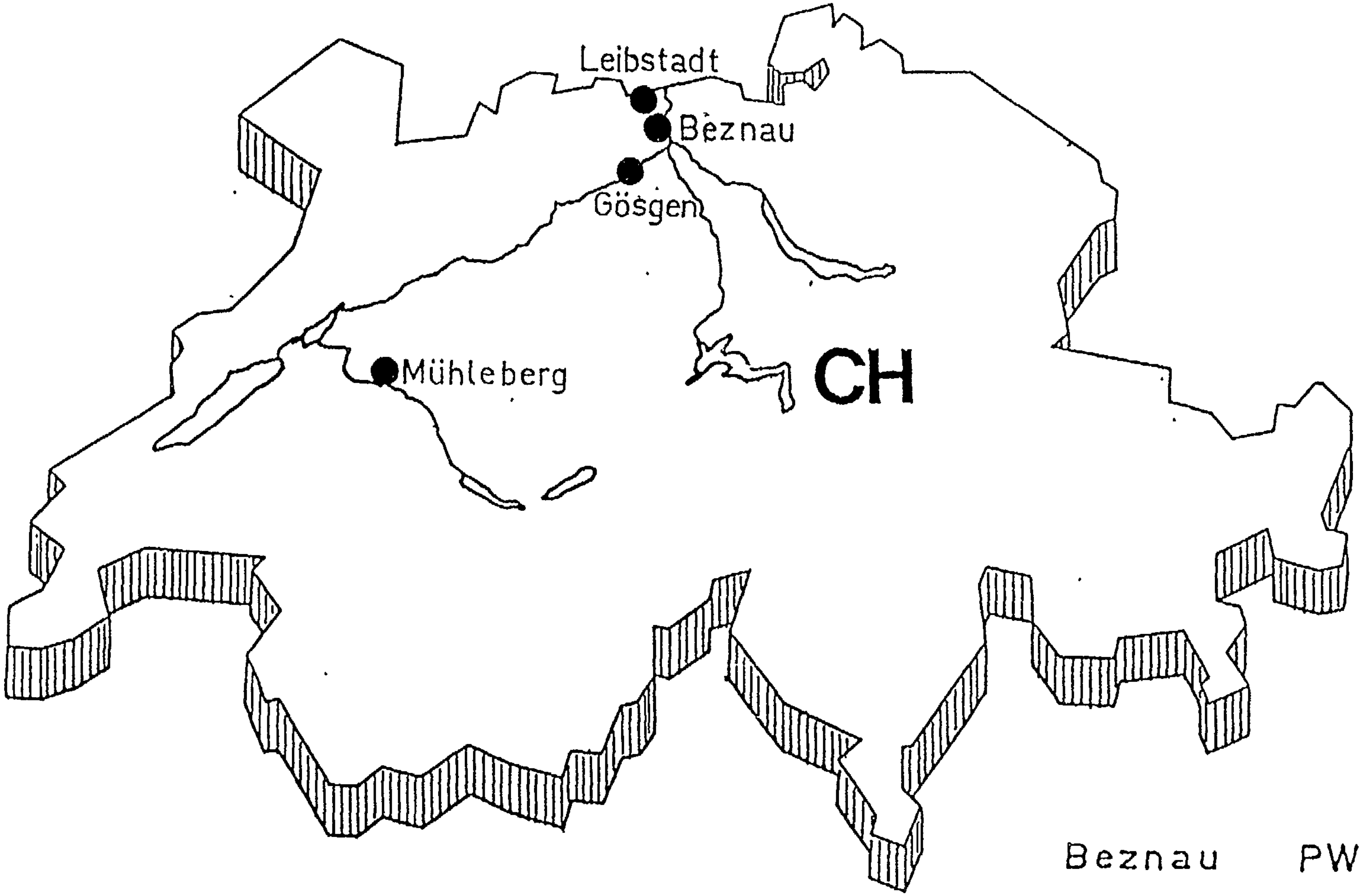
Gösgen, a 970 MWe PWR plant 35 km south-east of Basel, commissioned in 1979, is since that time supplying process steam at 220 °C to a cardboard factory nearby. Nuclear steam supply saves about 15'000 t/year of fuel oil for the factory operators. Gösgen nuclear power plant is situated in the southern foothills of the Swiss Jura, in a bow of the river Aare, 382 m above sea level. 400 kV and 200 kV switchyards are located only 300 m east of the site.

Beznau, a 730 MWe PWR plant, is situated about 35 kilometres north-west of Zurich. It was the second Swiss Nuclear Plant to be adapted to combined heat and power duty, but the first to supply district heating (Fig. 2). The two 1130 MWth reactors at Beznau commenced operation in December 1969 respectively in October 1971. Both are identical Westinghouse PWR's each supplying steam to a pair of originally 182 MWe BBC/ABB turbines. The 1993 steam generator exchange at Beznau-1 and subsequent improvements in turbine efficiency by settling them up with new HP-turbines in 1995 have resulted in better overall performance of both units. After the same activities at Beznau-2 which are planned 1999, the output of both plants together will amount to 730 MW net.

Non-electrical energy production at Beznau and Gösgen continued to function flawlessly. Beznau in 1997 delivered 141 GWh of thermal energy to the Refuna district heating system, while Gösgen supplied 142 GWh of process heat to the nearby Niedergösgen cardboard factory.

2. Refuna District Heating

In April 1981 eight communities, industries, private consumers and national research institutes, all situated in the Canton Argovia, decided to join in a project study of a district heating supply system, using hot water from Beznau nuclear power plant of the north-eastern Swiss Electric Power Company (NOK). Lower heat costs, saving of oil imports and lower environmental impacts were aims, which should favour a nuclear CHP route for the future in Switzerland.



Beznau	PWR	2×350 MW_e
Gösgen	PWR	920 MW_e
Leibstadt	BWR	970 MW_e
Mühleberg	BWR	320 MW_e

Fig. 1 Swiss nuclear power plants

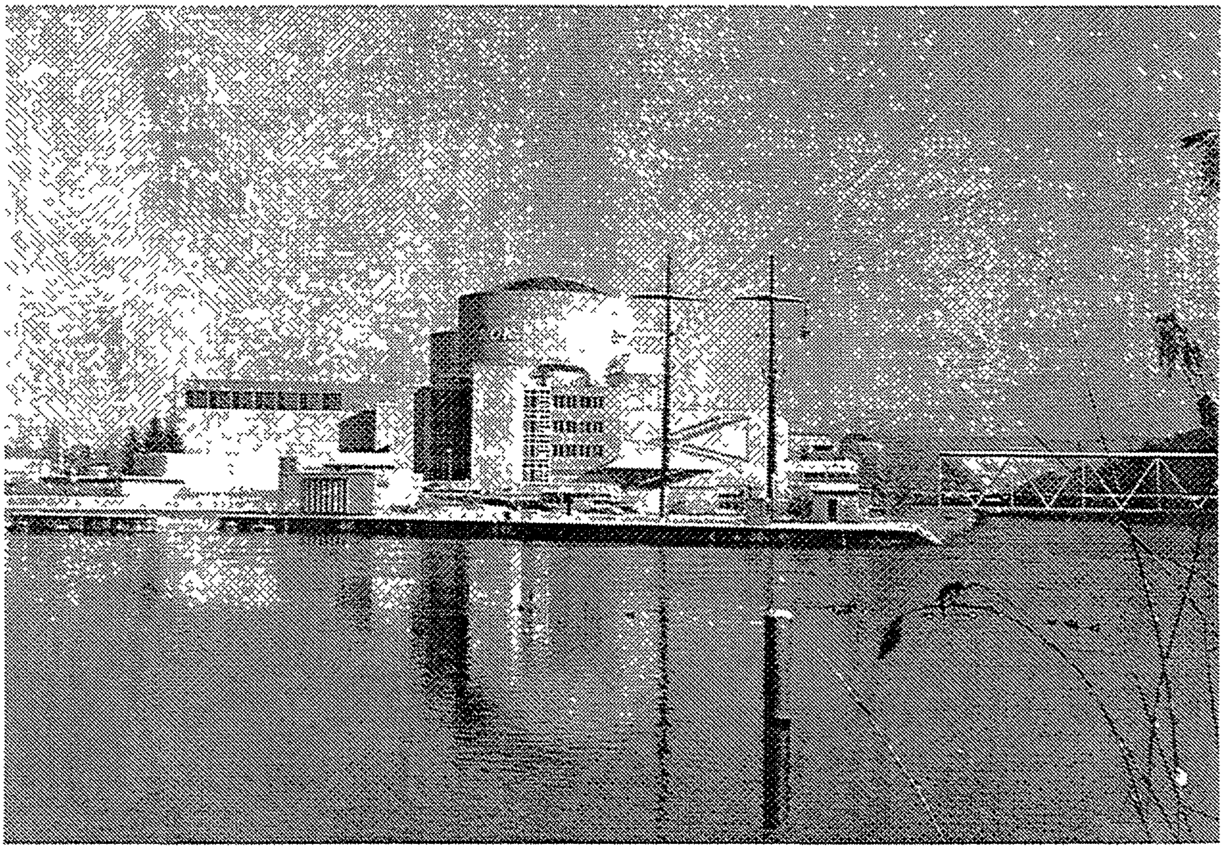


Fig. 2 Beznau PWR Nuclear Power Plant

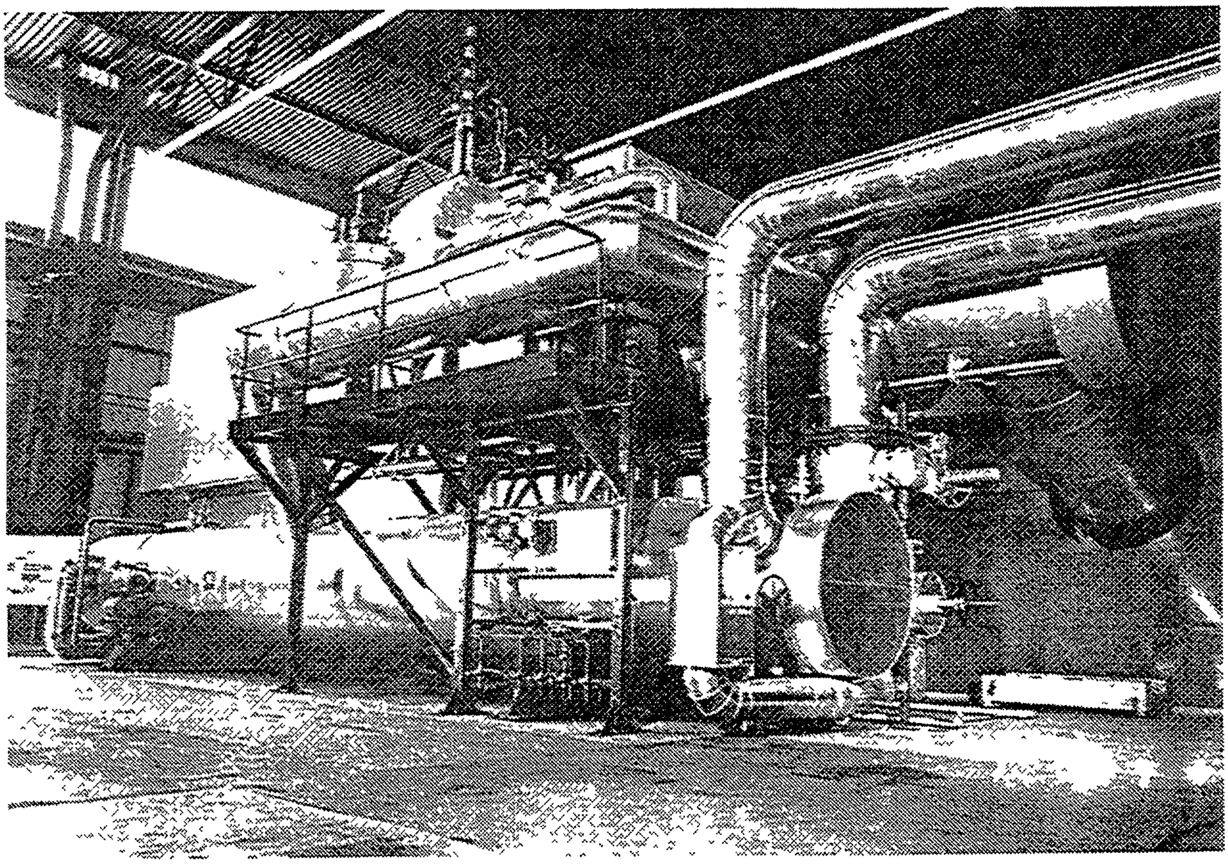
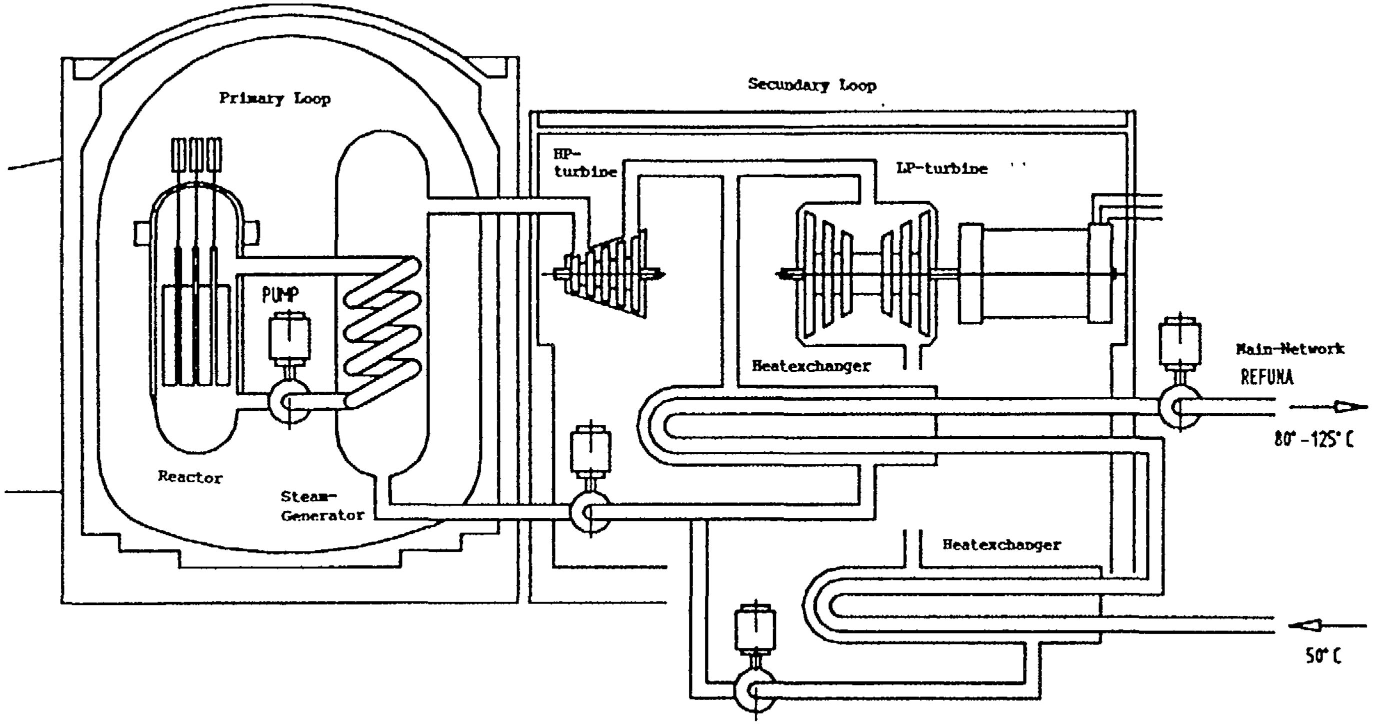


Fig. 3 Large Heat Exchanger of the Beznau Nuclear Plant



Cascaded Heatextraction for REFUNA:
Between the HP- and the LP-turbine, steam of 128 °C is extracted.
Extraction steam at 85 °C from the LP-turbine is taken for preheating.

Fig. 4 Heat extraction at the Beznau NPP (schematic)

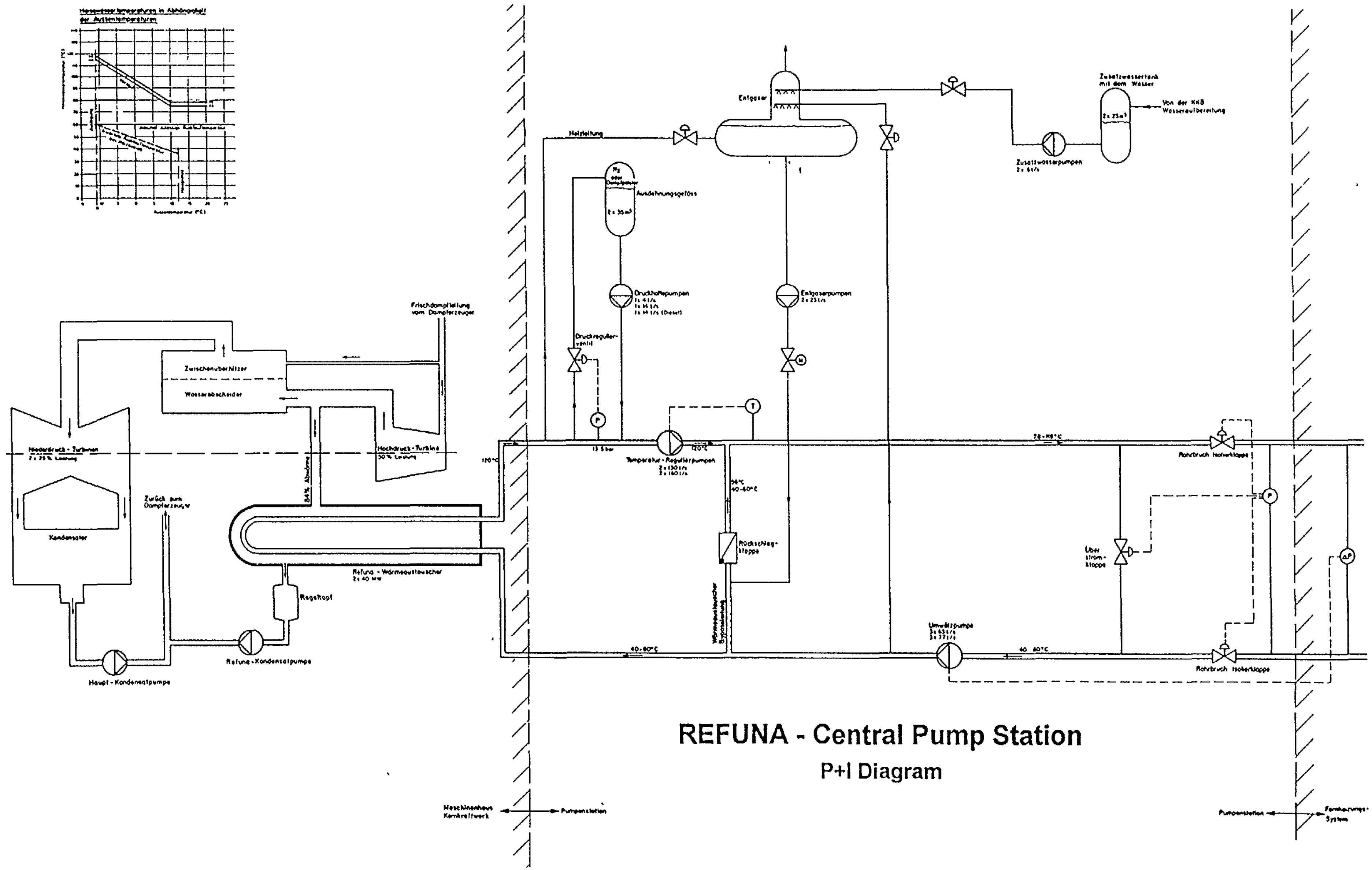


Fig. 5 Safety valves in the Central Pumping station to prevent radioactive carry - over

In 1983, the eight founding communities agreed by public vote to become shareholders of the new company and to build the Refuna district heating system. The Refuna AG was founded with a stock capital of 10 million Swiss francs (CHF), to build and operate the main heat network. 51 percent of the stock capital are owned by the connected communities, 49 percent are distributed among 54 private enterprises and large heat consumers. Each community has to build and maintain its own connecting pipes and hot water networks from the main pipes to the individual consumers. In 1985 it was agreed to extend the Refuna system to feed additional local networks in three new communities. Some 2'100 heat consumers should be connected in the future [1].

The district heating system includes the following components:

- Heat extraction from the Beznau Nuclear Power Plant (Beznau-1 and Beznau-2)
- Central pumping station with dispatching center
- Main pipeline system and local networks
- Oil fired package boilers for emergency back up

3. Heat Extraction from Beznau-1 and Beznau-2

Heating steam is taken from one turbine of each pair. The two heat exchangers No. 1 and No. 2 are placed in parallel with the turbosets (Fig. 3). They obtain the steam required for heating from cold re-heat crossover pipe between the high pressure - and the low pressure turbine at about 2,2/2,8 bar and 122 °C/128 °C (depending on the actual heat extraction). The heat exchanger - only one heat extraction normally is operating - delivers heating water at 16 bar to the central pumping station (Fig. 4).

Thus heat is available from two nuclear power plants, heat supply is therefore guaranteed during the annual inspections of Beznau-1 and Beznau-2, which were for a long time carried out one after another in summer each year. Beginning 1994 maintenance and refueling time was extended to 18 months cycle. If no heat can be supplied from both power plants - such situations were given only once a year in the past and were reaching seldom more than some hours - emergency boilers will feed the district-heating network. Oil fired back-up boilers are installed in the northern network with a total capacity of 28 MWth. Other boilers with a total heat capacity of 22 MWth supply the south-eastern heating system. On the very southern end of the network, additional boilers with a total of 16 MWth are available for emergency back up.

A central pumping station, built nearby to the nuclear power plant, drives the northern and the southern part of the heating system [2]. The supply temperature into the district-heating network ranges from 125 °C in winter down to 80 °C in summer. The actual return temperature in winter 1997 is below 50 °C. Totally nine booster pumping stations are installed, for the heat consumers in the communities are located up to 12 kilometres distant from Beznau NPP. These pumping stations have to guarantee a minimized pressure at the top levels of the pipeline system, which has to cap a eight-difference of nearby 140 meters between the central pumping station and the most distant consumers.

With respect to the nuclear safety, pressure in the district heating system (16 bar) is strictly kept higher than pressure in the crossover pipes of the turbine (2.2/2.8 bar). This is to avoid radioactive carry - over into the distribution system through the heat exchanger due to leakage. Safety valves in the main pipelines are installed on two points, determined by the concept of pressure- barriers. One group is situated inside the central pumping station (Fig. 5). In case of an abruptly loss-of - circulating water in the Refuna system, accompanied by significantly loss of pressure, they are closing rapidly at 6 bars. The second group of safety valves installed in the pilots of the piping bridge outside of the Beznau island, are closing next, ensuring the DH-system in the communities against loss of pressure. These valves are self-actuating. With respect to the pressure difference and the supplementary installations to ensure the pressure-difference at each time no further monitoring is carried out.

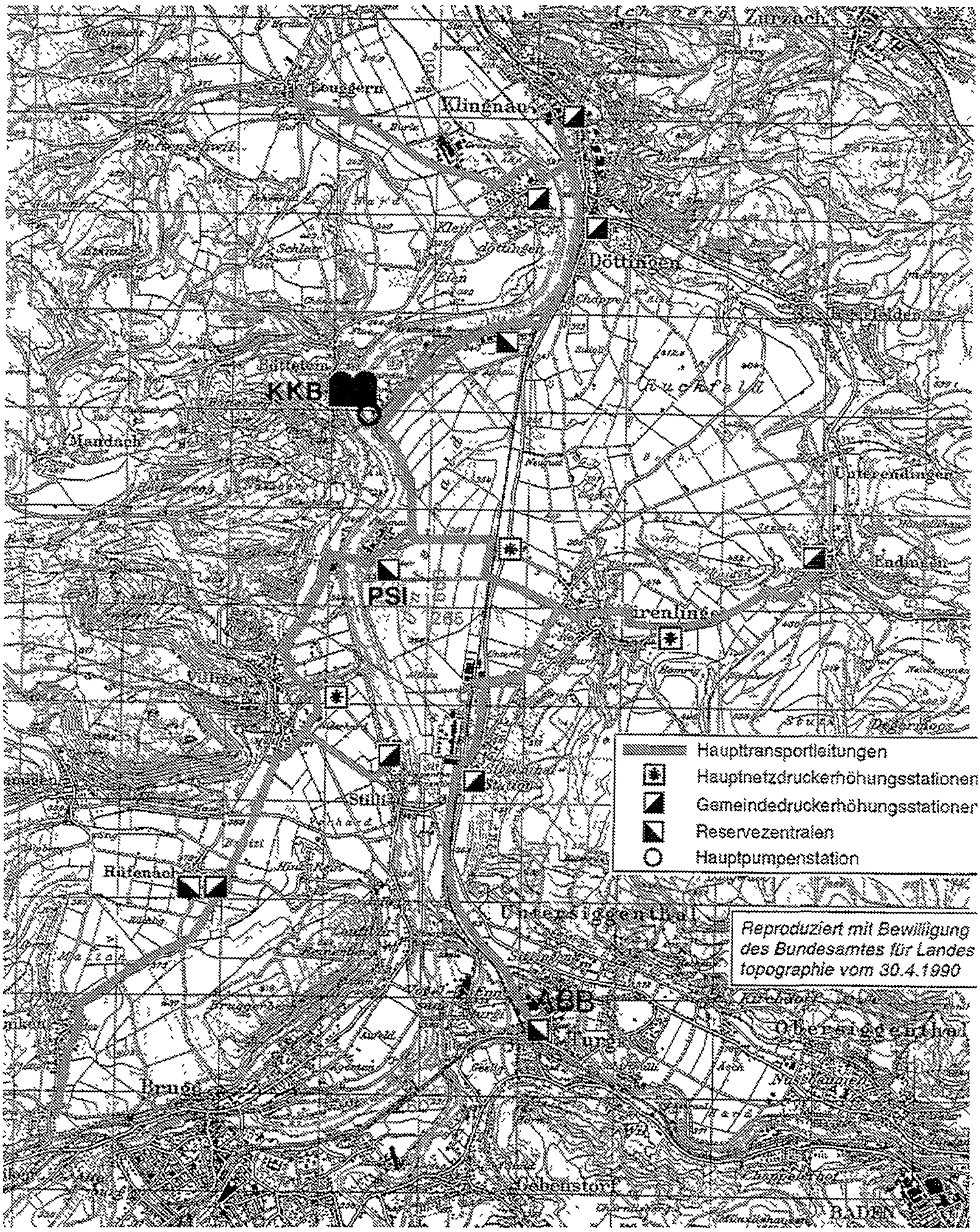


Fig. 6 Main Pipeline System of the District Heating

Contracted Heat Load in the District Heating System

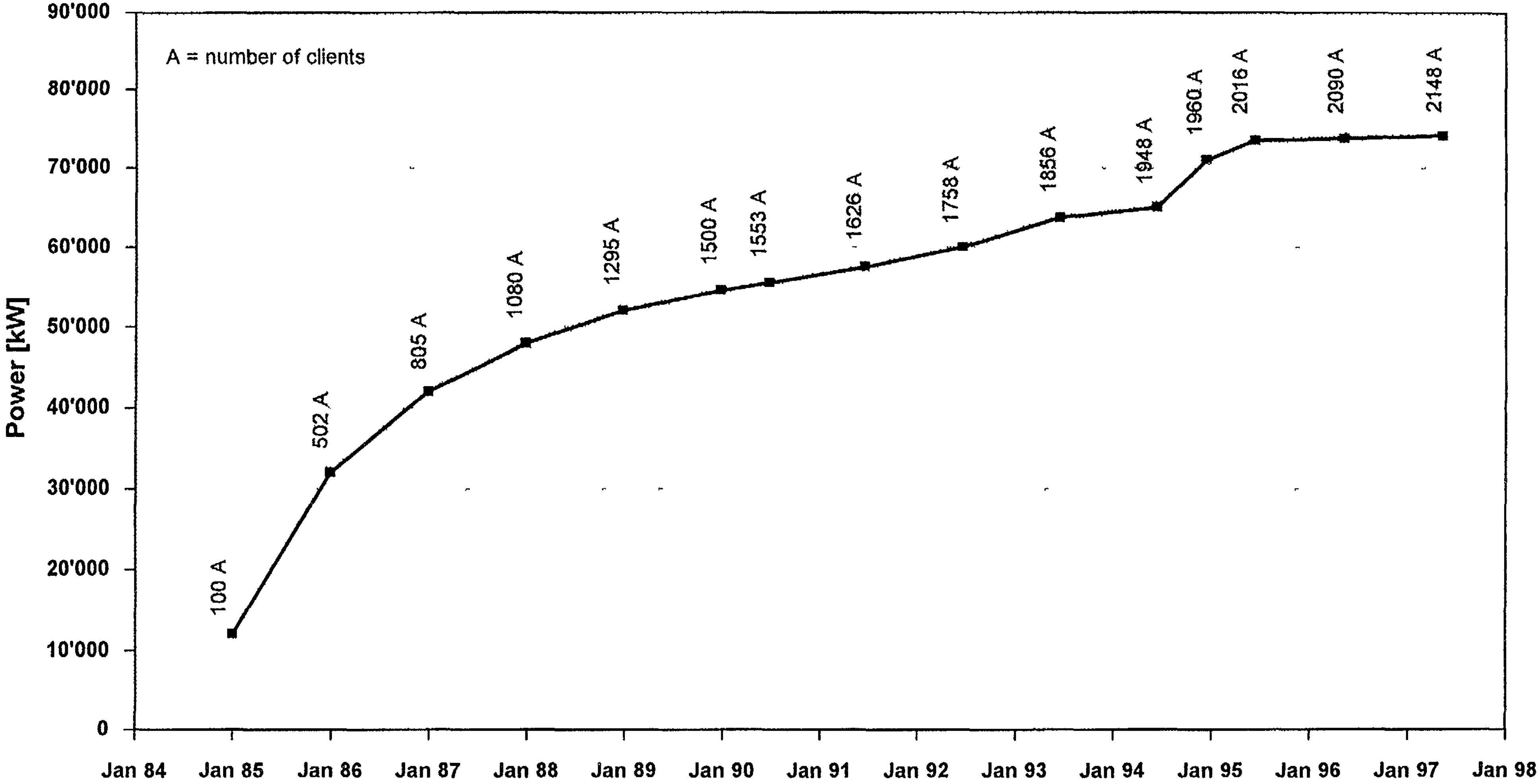


Fig. 7 Housing connections and contracted Heat Load

4. Heat Transport System

There are two main heat transmission pipelines coming from the Beznau nuclear power plant. One runs north supplying Döttingen, Kleindöttingen, Klingnau and Leuggern, the other runs south-eastwards to Würenlingen, Station Siggenthal and Endingen with a southern branch to Villigen, Stilli, Rüfenach and Riniken. The total length of the main pipelines is amounting 35 kilometres (Fig. 6).

Heat consumers normally are connected indirectly to the heat distribution network, using exchangers to transmit the heat to building heating and domestic hot water heating loops. Only one factory with a newly installed 16-bars heating system is fed directly without heat exchangers. All pipelines of the main system and of the local networks have a special built- in safety system, which detects damages and humidity within the insulating material immediately. Local networks together with the house service connections today amount to a total length of about 85 kilometres.

To minimize the heat losses everywhere high-effectively insulated pipes are installed. Heat losses of the main transmission system actually amount to 6 percent. Depending on the community structure, heat losses in local networks range from 6 to 12 percent. Total heat losses of the district heating network amount to 15 percent in full operation.

5. Housing Connections

The first phase of Refuna began in 1983 by supplying heat from Beznau to two research laboratories at Villigen and Würenlingen, which are situated 1.8 kilometres from the power plant. For this service one of the Beznau-2 turbines was converted in summer 1983. A 40 MWth (max 52 MWth) heat exchanger and auxiliaries were installed in the turbine hall during the scheduled downtime for maintenance and refueling [3].

Construction of the district heating network and construction of a second heat exchanger in Beznau-1 were starting in 1984. During the winter 1984/85, after a very short time for construction and testing of the pumping stations and pipe systems, six communities surrounding the Beznau power plant received district heating. 100 pilot-consumers were connected with 15 kilometres of main pipelines and local networks.

During the following heating period 1985/86 approximately 400 consumers were connected to the local distribution system, 1986/87 the figure was amounting to 800. At the end of 1997 more than 2'100 housing connections are in operation. The contracted heat load at an ambient temperature of -11°C is 74'500 kW(th) (Fig. 7).

6. Double-Stage Heat Extraction

With more than 2'100 buildings connected, peak heat demand will reach 80'000 kW in the next years. To be ready to meet this demand, which will be significantly higher than proposed in the original system concept, a third heat exchanger has been taken in service at the Beznau-1 nuclear power plant in November 1990 [4].

Heating the Refuna water using a double-stage extraction (first stage to 85 °C using steam of a new low pressure extraction and second stage to 125 °C, as mentioned in the beginning) was increasing the total efficiency of the heat recovery by almost 20 percent. The total heat extraction capacity was increasing by 30 percent. Peak load of 60 MWth is guaranteed using only the Beznau-1 heating extraction.

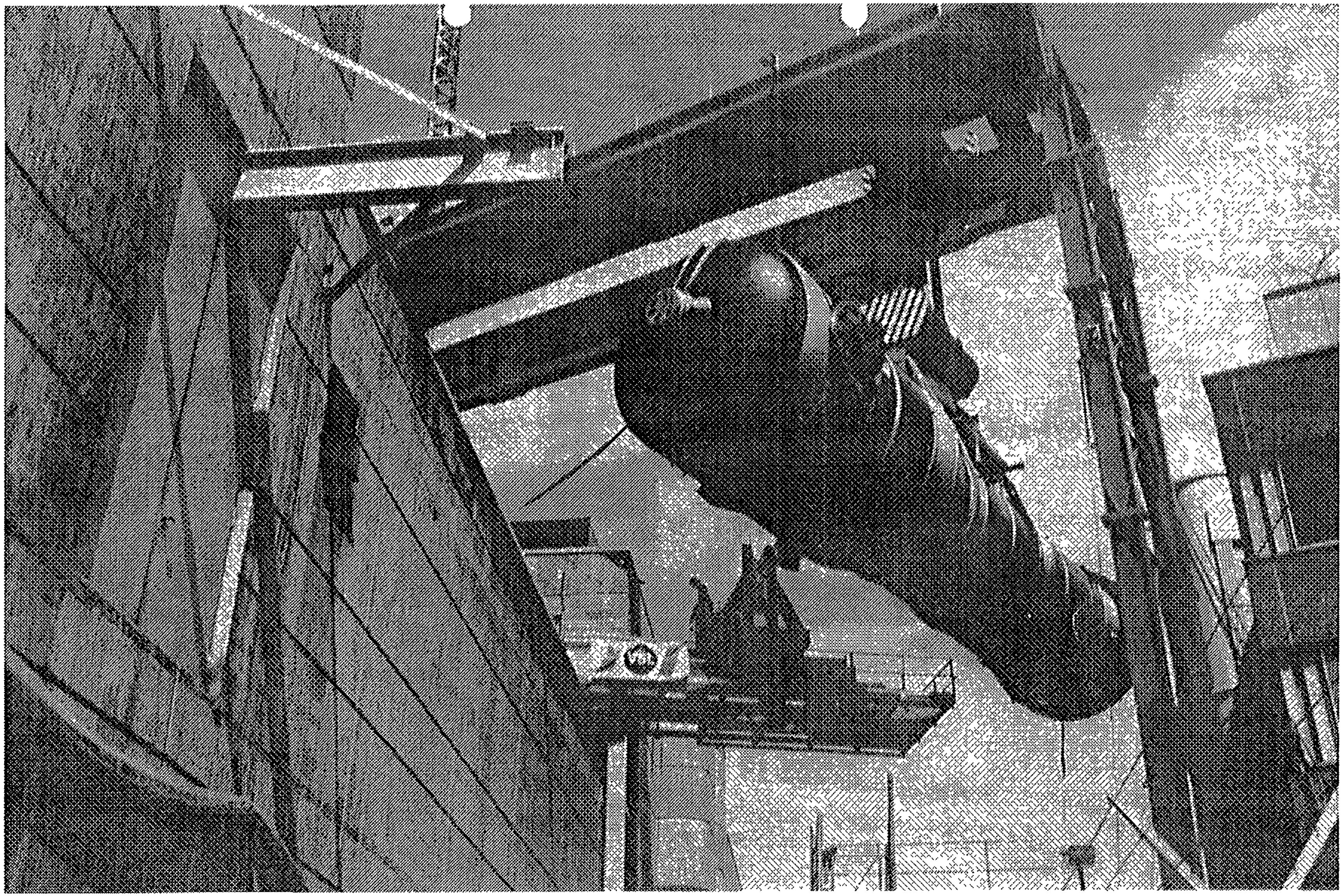


Fig. 8 Exchange of Steam-generators Beznau-1 (1993)

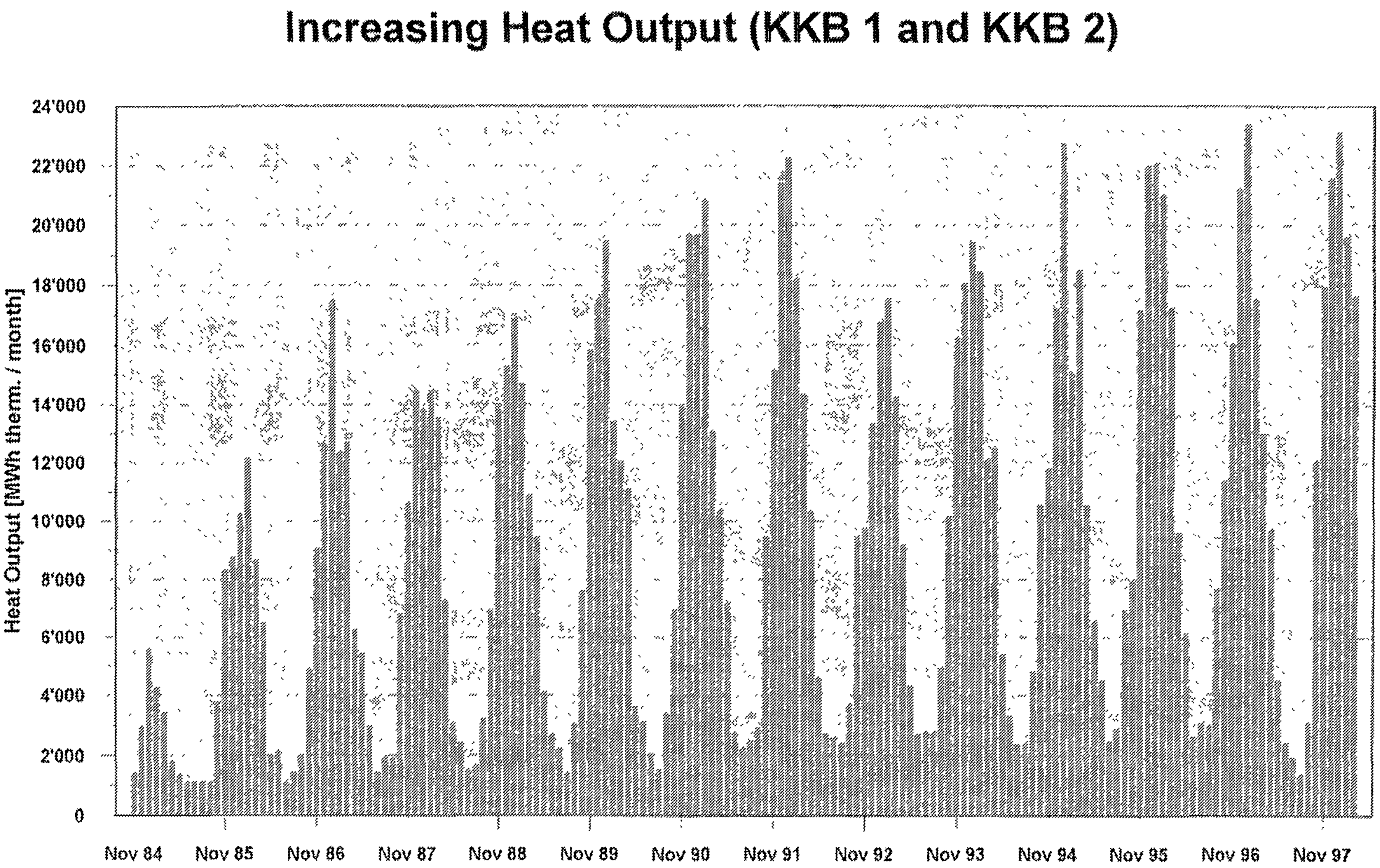


Fig. 9 Heat output from Beznau NPP 1984-1998

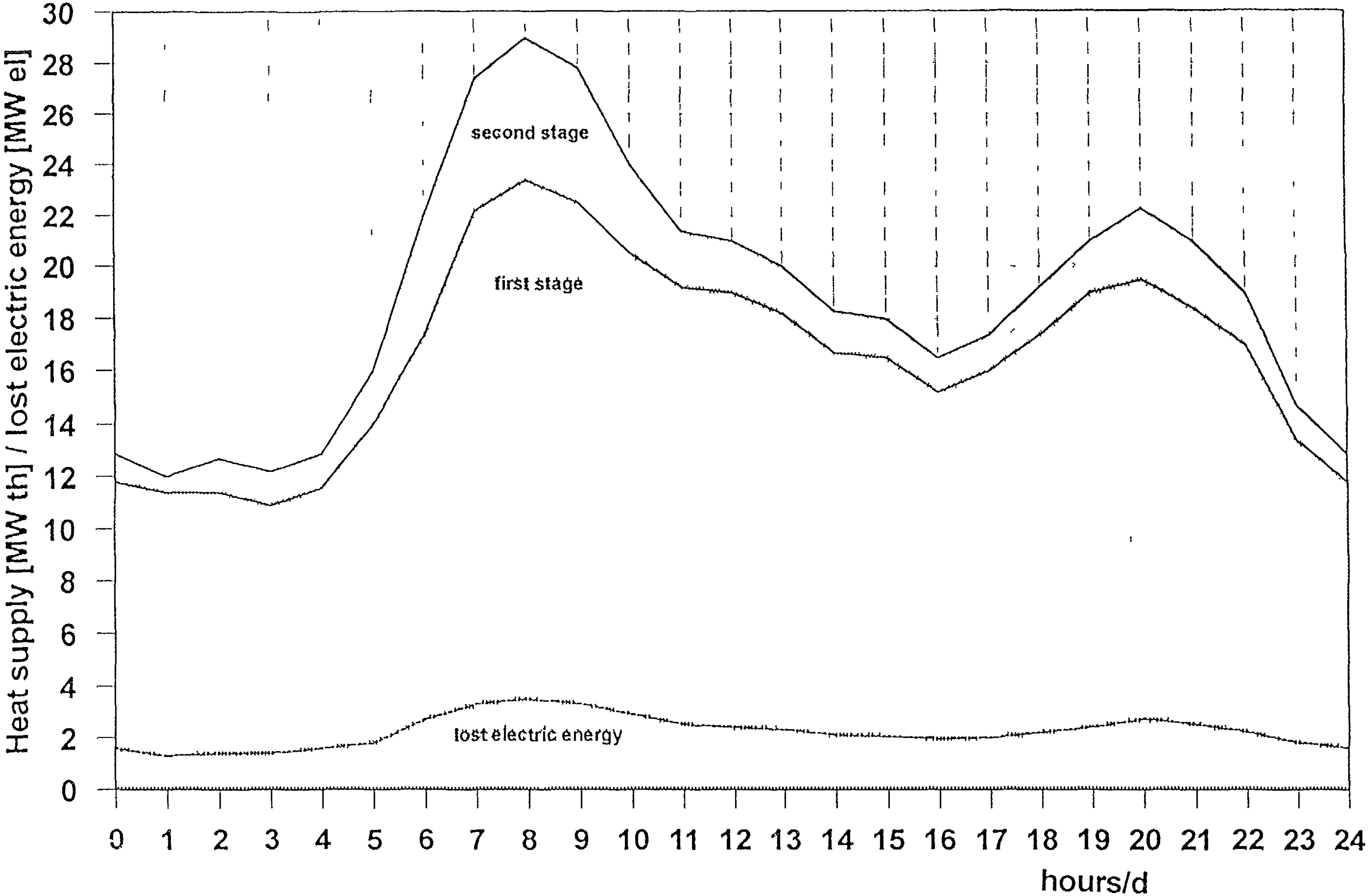


Fig. 10 Heat Supply on March 5, 1991. The Contribution of the first stage (1) of Heat Extraction is reaching 60 percent

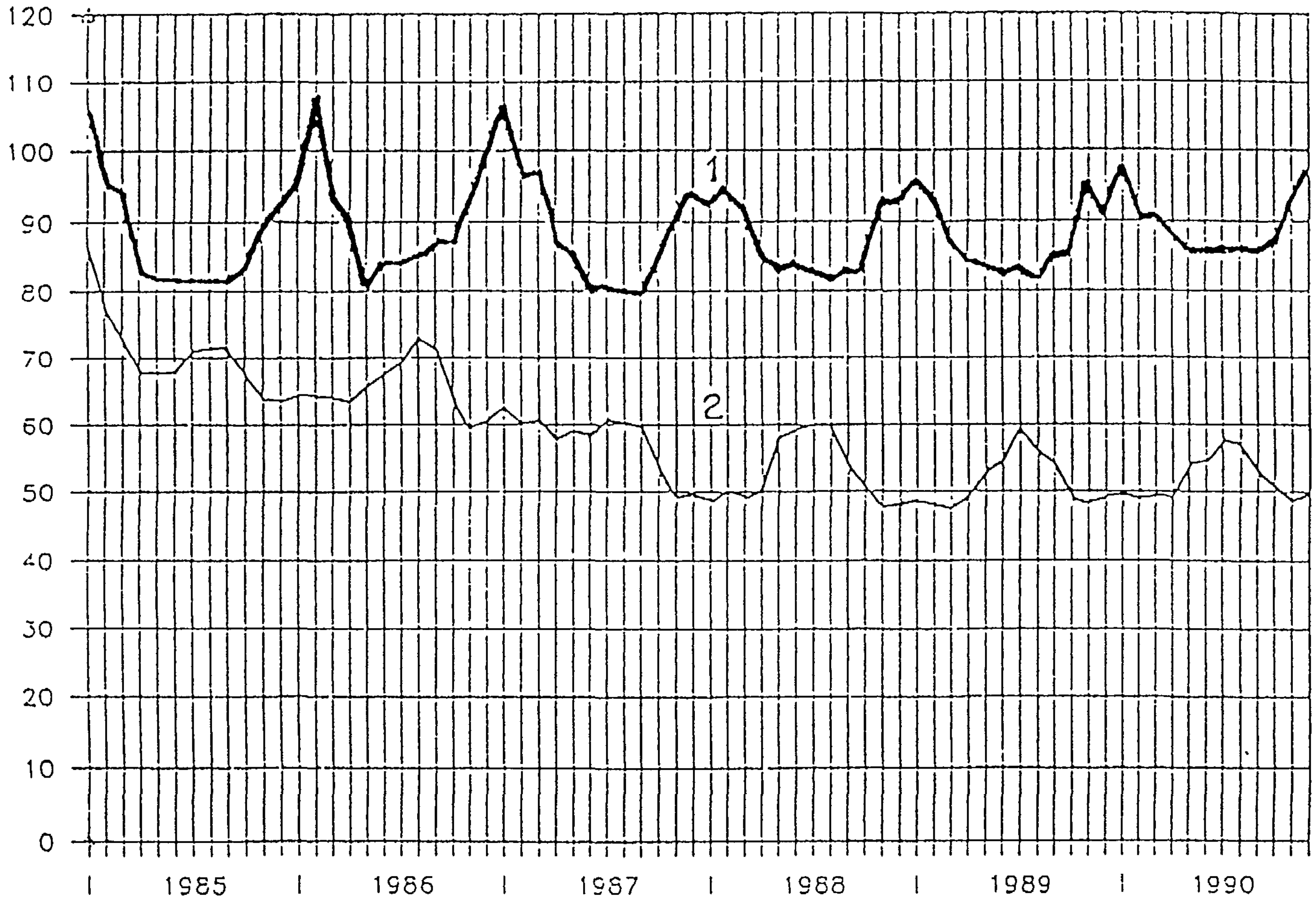


Fig. 11 Decreasing tendency of the return flow (2) temperatures in the Refuna network

Since the excellent exchange of the two existing steam generators in the Beznau-1 nuclear power plant in 1993, the specific loss of electric power by the heat extraction for district heating purposes was reduced. Better conditions (higher temperature, higher pressure) of the live steam to the turbines was leading to a heat extraction factor below f=0.12 following commissioning the new steam generators. 88 percent of the heat load drawn by the heating network is now recovered heat, which would be normally discharged by the condenser cooling water into the river Aare. The benefit of the new installation - higher efficiency and lower heat costs - is given by the Beznau owner NOK to the Refuna district heating system. Refuna itself has to pay back the investment of 3.5 Mill CHF within 15 years (Fig. 8).

7. Operating experiences during 15 years

Growing heat delivery in spite of warm winters

In parallel with the year by year growing number of housing connections, heat supply from the Beznau nuclear power plant up to now is increasing without interruption. From January 1996, to December 1996 heat delivery reached 141'100 MWh, according to an outdoor temperature (average) of 8.5 °C. These figures are to compare with 132'567 MWh in 1995 (9.0 °C) and 128'536 MWh (9.2 °C) in 1997. The monthly course of development of heat consumption from the beginning of official heat supply in October 1984 demonstrates a good correspondence between the increasing heat load contracted and the heat supply (Fig. 9).

The peak load in 1995, 1996 and 1997 were registered each times at 53 MW in January, following an outdoor temperature between -8.9 °C (1995) and -12.6 °C (1997). This is to compare with the contracted heat load of 74.5 MW. In these years we had very warm winter periods. The highest heat delivery within one month was registered in January 1997, amounting 1'140 MWh.

About 15 % of heat losses are expected normally together in the main and the local DH networks. Significantly "reduced" losses in 1992/93 and 1996/97 were caused by measuring failures in the power plant.

Extended reserves in heat capacity

Seven years of operating experiences with the new double-stage heat extraction are confirming the high value of the decision to extend the heat exchange: Not only in springtime, summer and autumn, but also under the actually moving climatic conditions with increasing temperatures and warming winter periods the effectiveness of the measure is obvious. The difference of loss of electric power production between the one-stage heat extraction at Beznau-2 and the double-stage heat extraction at Beznau-1 is amounting to more than 20 percent (Fig. 10).

Using optimized housing stations, connecting best insulated new houses and together with additional activities, it was possible to reduce the return flow temperature of the district heating water continuously (Fig. 11). Corresponding with this fact the pumped mass flow in the heating water circuit and therewith the specific costs for pumping energy could be reduced each year. Following consequently extended temperature differences between out flow and return flow the district heating network was getting to a comfortable situation with regard to load reserves in the main pipelines. So it is possible to extend the district heating system year by year.

The operational experience to date is confirming the success of the strict approval conditions for the housing stations connections: very deep return temperatures in the district heating network are leading the main pipeline system. These reserves of capacity now can be used, when specific extensions of the DH-System are offering new dimensions of power supply in the connected areas. In the sense of a comprehensive exploitation of the heat sources available in the region, the waste heat from a large wood processing plant is now fed into the Refuna network.

Enlarged capacity of emergency boilers

Connected with the growing heating load in the last years the capacity of emergency back up boilers had to be extended. Today 80 percent of the peak load (at -11°C) are secured by boilers. Installation of a further boiler is foreseen.

Leakage in the district heating network

The water volume of the whole district heating system is amounting 2'500 m3. Normally the water losses vary between 1.0 and 1.5 m3 a day. In the case of higher losses - figures of 20 to 25 m3/d were measured - a wide searching campaign has to be started, to determine the location of the leakage. Various methods are being used, e.g. "Thermo-Vision".

Failures in measuring heat delivery

The first measurement failure in winter 1992/93 was amounting about 20'000 MWh heating energy. From November 1992 to February 1993 measurement installations in the Beznau NPP were influenced by polluted circulating water (Fig. 12). After that event shorter cleaning intervals on the inductive measuring instruments were determined. On the other hand such cleaning processes can only be executed in periods of a plant shut down for maintenance or refueling. A second failure happened in winter 1996/97, when 11'000 MWh of heat delivered were missing in the bills.

Now the installation of a second measuring system (ultra sonic) is in consideration to have a better and quicker information in case of polluted measurement installations.

Functional tests on prevention of radioactive carry-over

Safety tests of the valves installed in the central pumping station and in the bridgehead of the piping bridge, some hundred meters from the pumping station, will be carried out each month.

8. Costs and economics of the Refuna district heating

The Refuna regional heat supply company, founded in 1983, today is owned by 62 stockholders. 51 percent of the stock capital of today 22.9 million Swiss francs is owned by the connected communities, 49 percent is distributed among 54 private enterprises and large heat subscribers.

Investment costs for the complete district heating network with main pipelines and local networks amount to CHF 100 million. CHF 40 million fall to the share of the main system and CHF 60 million to the local distribution networks, including housing connections and heat measurement installations. Additional costs of about CHF 10 million for the two large heat exchangers at Beznau nuclear power plant including auxiliaries and the pipelines on the Beznau island (about 200 meters) were paid by the electric utility NOK in 1983. Building costs of about CHF 3,5 million for the new low-pressure heat exchanger in Beznau 1 as discussed before and CHF 3.0 million for the central pumping station were paid by NOK 1997 supporting the financial balance of Refuna.

NOK undertakes conversion at the power plant and delivers heat into the hot water transmission system. Primary energy from the nuclear plant to Refuna during the last years was supplied for CHF 0.015 per kWh thermal energy. This figure depends actually on the price for the lost electric energy and on the ratio of loss of electric energy to extracted heat energy, which, following the installation of the new heat exchanger in 1990 and with the new steam generators in 1993, now is below f=0.12 on average.

All heat extraction services in the Beznau power plant and the service of the central pumping station are included in the price for the heat delivered by the owner NOK. Only 3 employees are working for the Refuna district heating main system and for three small local networks. In eight local networks,

Problems measuring Heat Output

January 1997

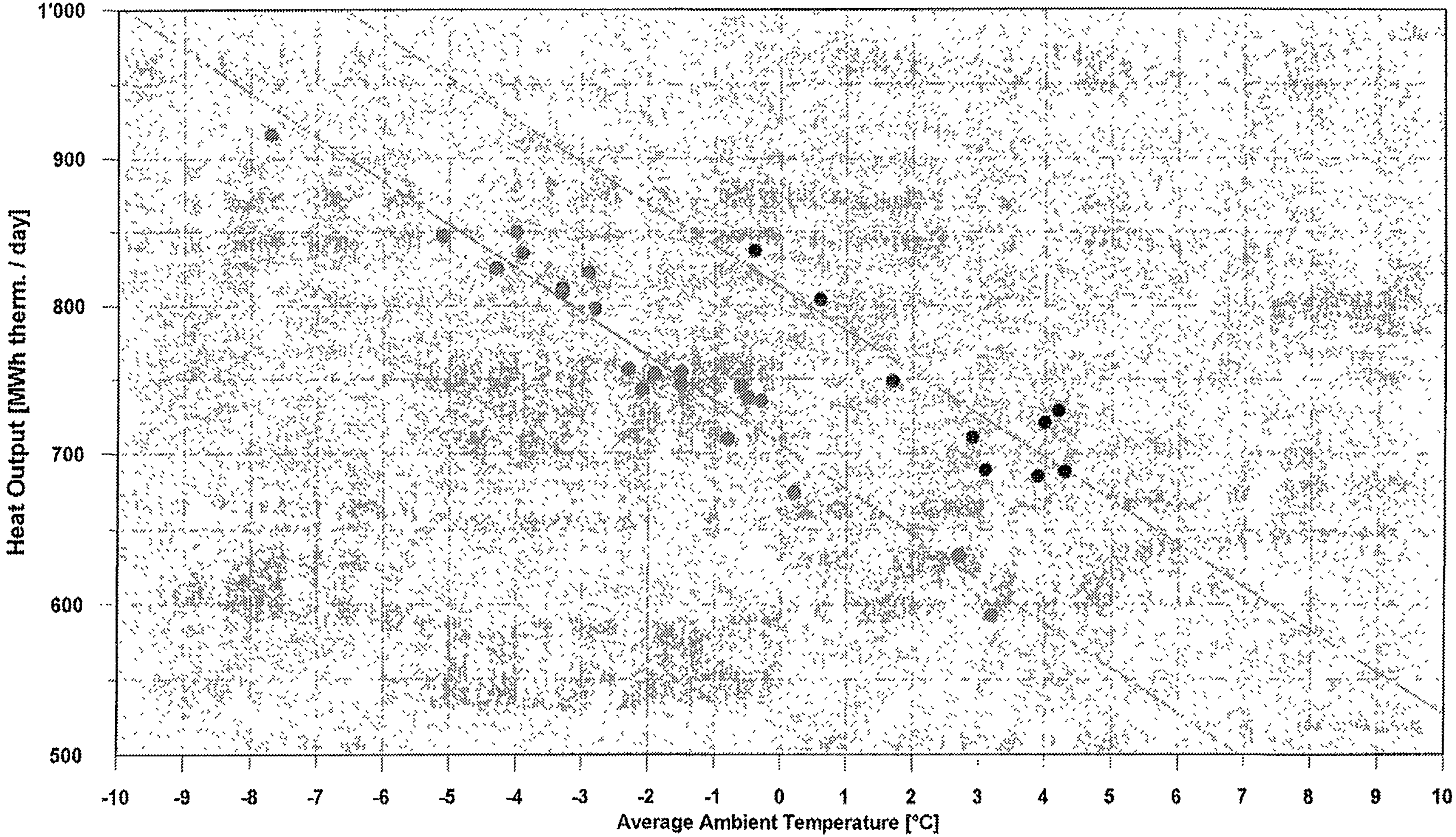


Fig. 12 Influence of magnetite on the measuring of heat delivery from the NPP

which are owned by the communities, the local energy commission is taking care of correct function. In the case of extended problems in the district heating system a contract for engineering and emergency assistance by NOK is supporting Refuna.

The total costs of heat production in the district heating system can be roughly by detailed into capital costs (60 percent), primary energy (20 percent) and operation, maintenance etc. (20 percent). Although primary energy is fairly cheap, the high costs of the distribution network in a predominantly rural district required careful optimisation of financing modes for the district heating system. About 40 percent of the capital are paid by the shareholders, the balance is long-term loan capital from banks and insurance companies.

The heat consumer pays a fixed, once only charge for connection to the local network and pays for the conversion of his existing house heating system - if it is an existing building. The fixed charge depends on the peak heat load of the consumer's house contracted. The tariff for heat delivery is a dual rate consisting of a basic price and a heat price. The costs of district heat for the consumers are varying between the single local networks and are also depending on the full-load hours of the contracted thermal power (water flow). In average, total of both components today range from CHF 0.07 to CHF 0.095 per kWh(th). With these figures the costs of district heat in the Refuna area today are significantly higher than oil-firing, which is still the main heat source in Switzerland. Oil heating actually costs about CHF 0.045 to 0.060 per kWh(th), including maintenance and periodical testing of the oil tanks.

All figures presented show very clearly that Refuna is not a typical district heating system with a big market of large heat consumers near the heat source. On the contrary, distances between heat production in the power plant and the communities are great and consumers are spread over a wide area. 75 percent of the consumers live in one - or two family houses.

9. Contribution to the CO_2 - Balance

High acceptance of nuclear district heating in the Refuna-area is at first based on very good operating experience over many years with Beznau nuclear power plant. In 28 years (Beznau-1) respectively 26 (Beznau-2) years of operation both plants have run with average load factors of more than 85 percent. Successful heat supply from the beginning of district heating 1984 up to now is an important influence, which has to be seen. Information about the actual heat load is presented to the employees of the Beznau NPP on-line in the entry, together with the total electric output of the plant.

Every year, depending on the climatic conditions, up to 20'000 t less oil will be fired once the schemes are completed and when the supply networks of the eleven communities connected are in full operation. [5]

Calculation of the reduction of CO_2-emission due to the operation of the heat extraction plant is showing a yearly saving of 50'000 tons of CO_2. In addition savings of 100 tons of SO_2 and 50 tons of NOx are to be considered. In addition there are some 100'000 MW(th)h less waste heat in the river cooling water. [6]

The impacts of the heat extraction from the Beznau nuclear power plant Beznau, in particular its contribution to the protection of the environment by substituting fossil fuels and preventing CO_2-production, are positive. Based on the successful heat extraction from Beznau nuclear power plant, district heat supply systems in Switzerland will be favoured. With respect to its positive environmental aspects the heat extraction at Beznau for the regional REFUNA pilot scheme serves the aims of the World Power Conference 1989 at Montreal and other International Conferences.

District heat supply from Beznau nuclear power plant is representing a remarkable example of efficient energy use - unfortunately, because oil is so cheep, today out of competitiveness. Although up to now only one large consumer, a steel factory, closed their 1 MW connection of district heating, when the factory was closed in 1996 and each year the number of consumers is growing. Expecting a peak load of 80 MWth at end of the century loss of electric power output at Beznau nuclear power plant will amount to 10 MWe. Loss of electric energy generation at that time will reach about 25 GWh per year. In 2005 the participating communities are expecting the final peak load with about 91'000 kW.

It is obvious that the small contribution of the Refuna DH to the world-wide CO_2 balance is negligible. However it demonstrates a possibility how to reduce CO_2 significantly in a number of countries by heat extraction from NPPs.

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XA9848829

HEAT DELIVERY FROM BOHUNICE NPP, SLOVAKIA

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Abstract

. Experience with nuclear district heating in the Slovak Republic is reported. The heating system of the town of Trnava is supplied by the Bohunice NPP and conventional sources. Construction of the hot water heating system from the Bohunice NPP began in 1983. Commercial operation began on 10 December 1987. Heat delivery has gradually increased from 478 TJ in 1988, to 1,104 TJ in 1995. The heat cost is low, resulting in an increasing number of consumers

1. COMBINED ELECTRICITY AND HEAT GENERATION

The perspective of energy balance led Slovakia towards heat supply from large energy facilities through regional heating systems with heat transport over relatively long distances. On the basis of governmental decisions, the conversion of existing single purpose power plants to combined electricity and heat generation is taking place and the nuclear power plants in Slovakia will deliver heat energy, in addition to electrical energy. In the Slovak Republic there is only one NPP in operation.. This NPP of Bohunice has four units, out of which units 3 and 4 deliver heat to the centralized heat supply system of the town of Trnava.

In the vicinity of the Bohunice NPP, there is a heat exchanger station with an installed power capacity of 240 MW(th). Heat exchange is achieved through four basic heaters, each supplied with LP steam from one of the turbo-generators of units 3 and 4. In the basic heaters, circulating water is heated from 70 to 130°C by steam from the fifth bleeding stage of the turbo-generator. Behind each pair of basic heaters there is a top heater which heats the circulating water up to 150°C. The circulation of the water is ensured by three pumps, each with a flow rate of 1,200 t/h. The flow can be regulated from 600 to 1,200 t/h by means of hydrodynamic coupling which permits regulation of revolutions from 600 to 1,450 r.p.m. (revolutions per minute).

Construction of the hot water heating system began in 1983. Commercial operation began on 10 December 1987. Heat delivery has gradually increased from 478 TJ in 1988, to 1,104 TJ in 1995. This quantitative increase in delivery is closely connected with the quality of the heat delivered, which resulted in an increasing number of consumers and a steady growth of heat supply.

2. THE MOST CHARACTERISTIC QUALITATIVE PARAMETERS

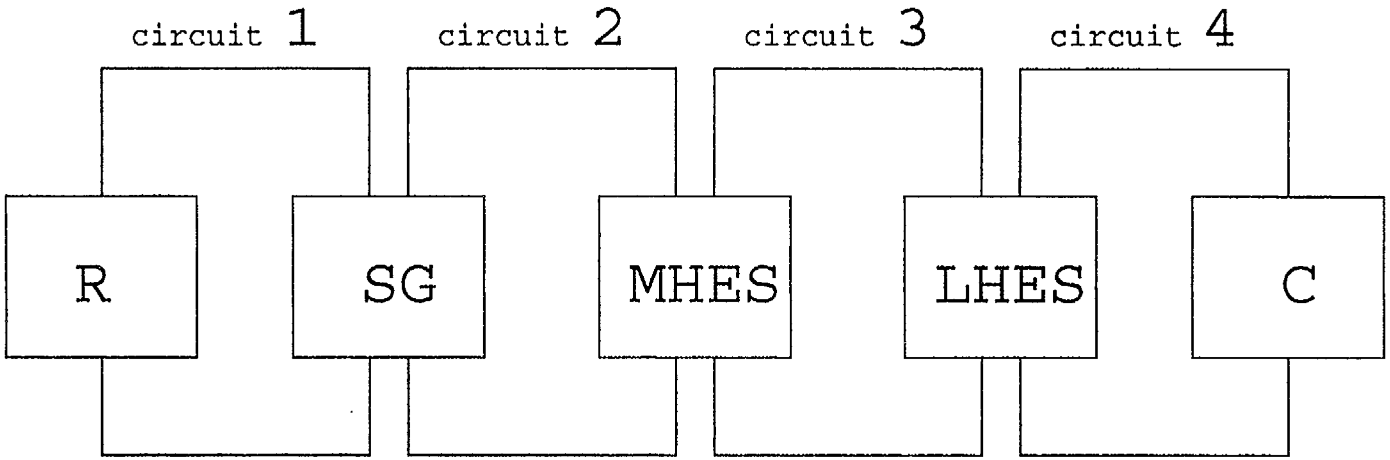
2.1. ACTIVITY (RADIOLOGICAL SITUATION)

The principal scheme of circuits for heat delivery is shown in Fig. 1. To avoid release of radioactivity from circuits 1 and 2 into the circulating water of circuits 3 and 4, several technical measures were taken. In addition to these measures, regular inspection and limiting levels were established for radiological control of working fluids.

For the secondary circuit, the inspection levels were based on the level of values for drinking water. The total activity in the secondary circuit has not yet reached the inspection level of 1 Bq/l.

In the third circuit, total activity of 1 Bq/l is the limiting value and for ^3H (tritium) the limiting value specified is lower than the level in the make-up river water.

Neither the limiting values in the third circuit, nor the inspection values in the second circuit, have yet been reached.



R - reactor

SG - steam generator

MHES - main heat exchanger station

LHES - local heat exchanger station

C - consumer

Fig. 1 Scheme of heat circuits in the NPP and district heating system

Source	Share [%]
Fossil source	100
Nuclear source	61
Fuel component of fossil source	27
Fuel component of nuclear source	4

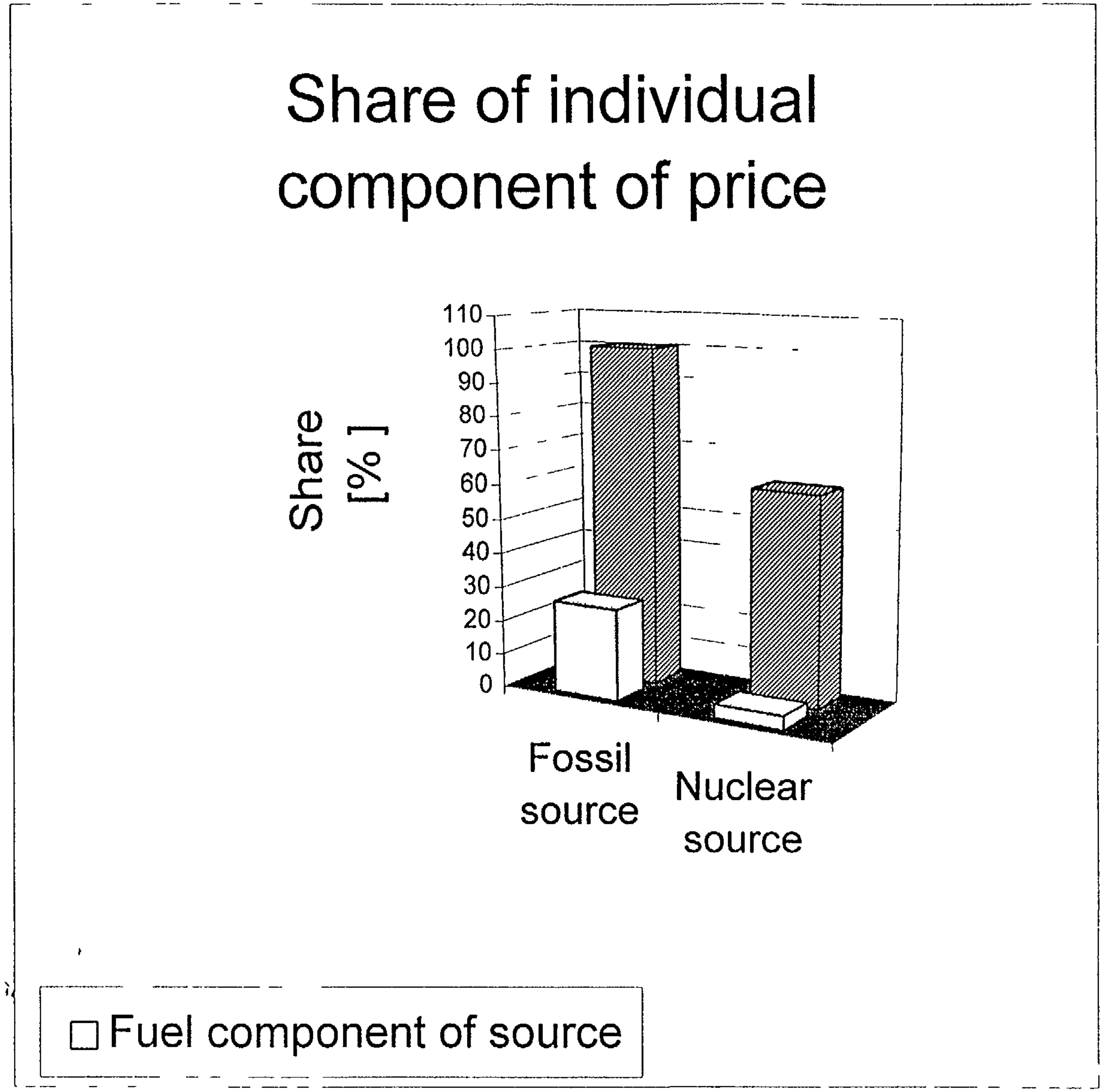


Fig. 2 Relative heat prices

Year	Heat of delivery [GJ]
1987	59 738
1988	478 782
1989	607 786
1990	775 835
1991	850 837
1992	903 172
1993	1 012 399
1994	935 264
1995	1 104 401
Total	**6 728 214**

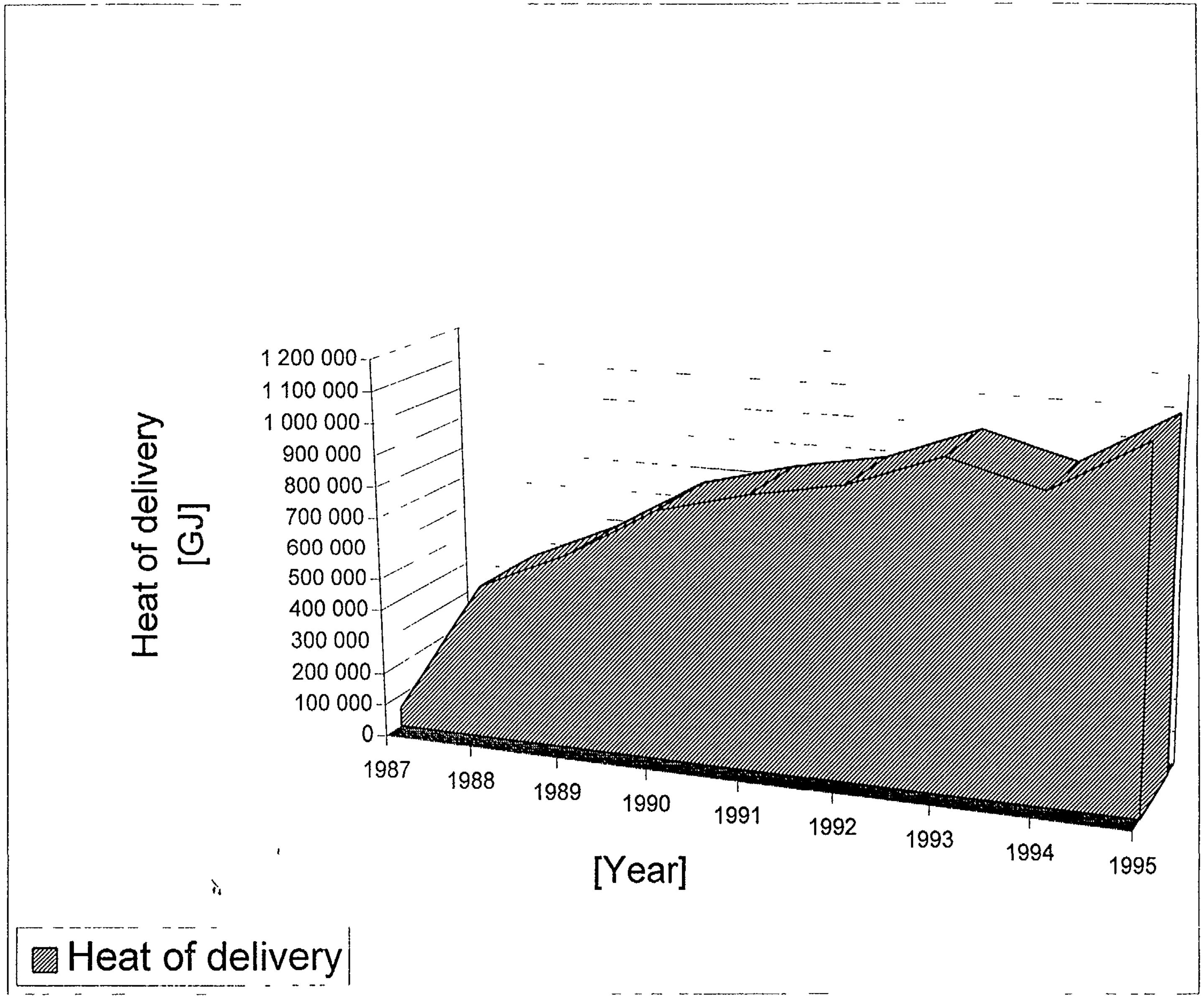


Fig. 3 Heat delivery from the Bohunice NPP

Year	Load Steam [MWt]	Load Hot water [MWt]
1993	126 0	120 0
1994	120 7	125 3
1995	97 4	150 7
1996	93 4	154 3
1997	89 4	157 8
1998	85 4	161 3
1999	81.4	164 9
2000	77 4	168 4
2001	51 4	194 4
2002	45 5	200 2
2003	39 7	206 1
2004	33 8	212 0
2005	30 1	215 7

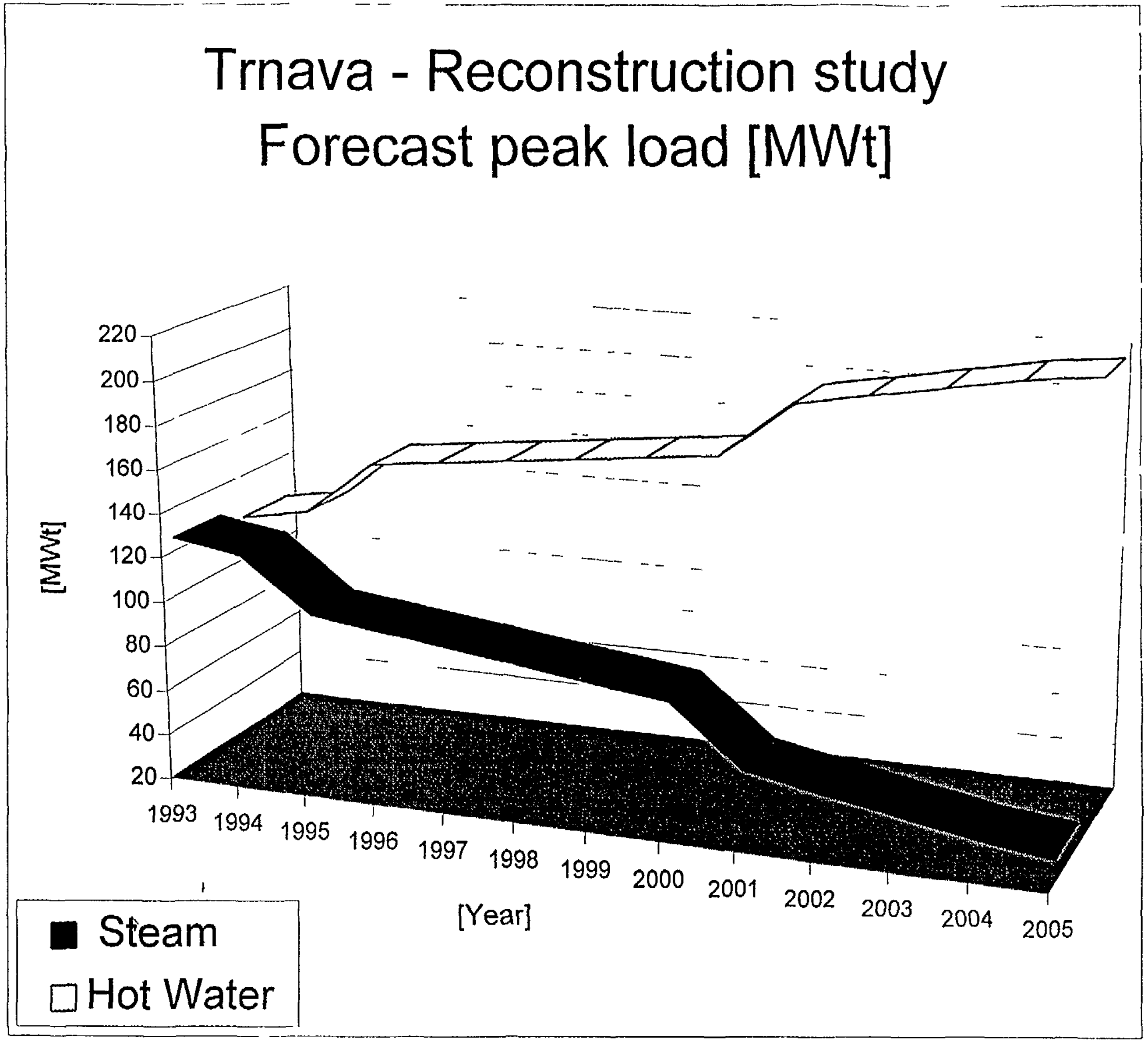


Fig. 4 Forecast of peak heat load

Year	Share [%]
1993	51
1994	49
1995	39
1996	38
1997	36
1998	35
1999	33
2000	31
2001	21
2002	19
2003	16
2004	14
2005	12

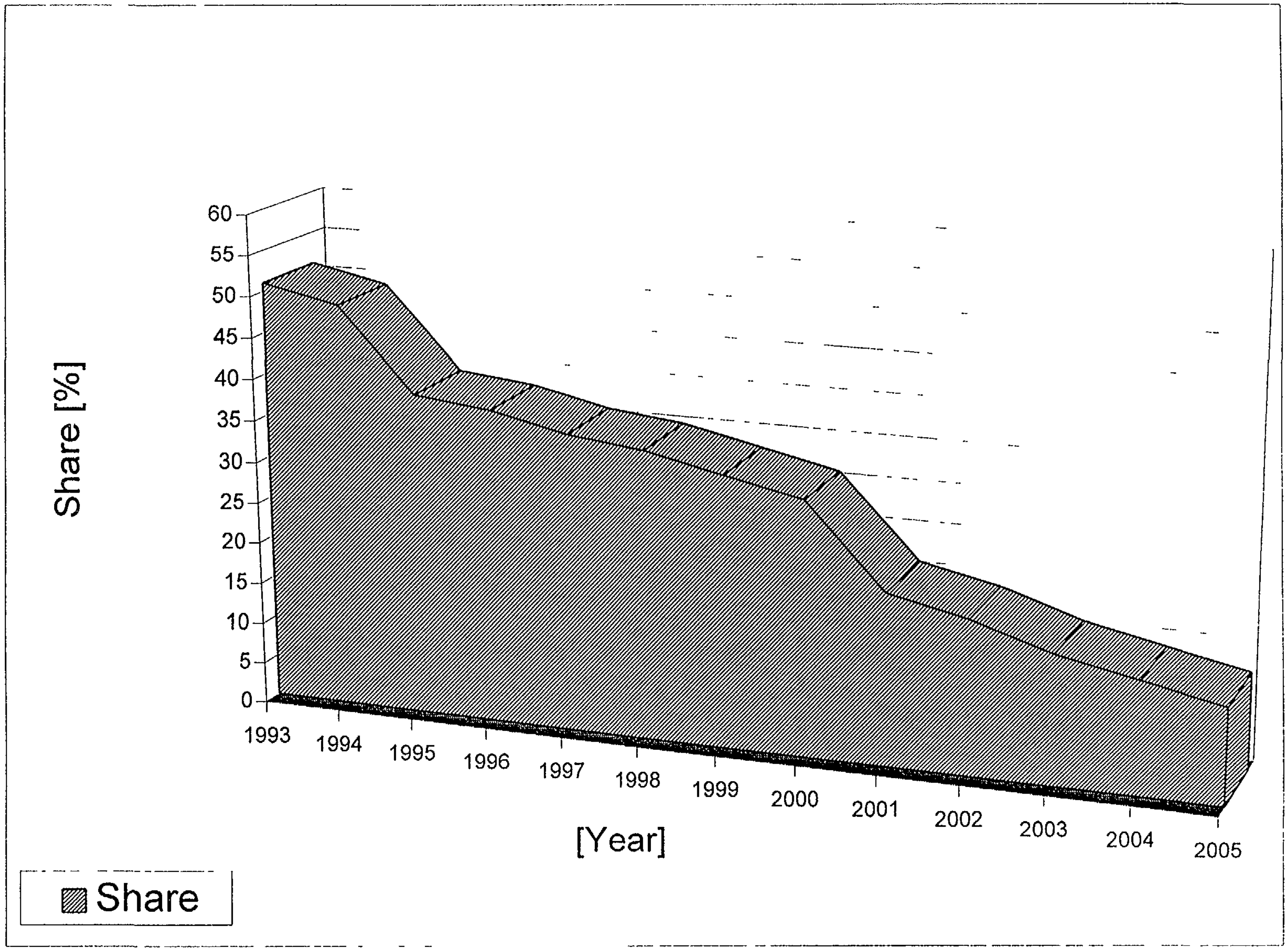


Fig. 5 Forecast of conventional share of heat load

2.2. ECONOMY

Heat supply to Trnava is ensured both from nuclear and conventional sources. In 1995, this relation was approximately 60% nuclear and 40% conventional. In Figure 2, the relative heat prices are shown.

An interesting point is not only the relative cost of heat nuclear heat shown as a percentage of the cost of heat from a conventional source, but also the low nuclear fuel cost. This leads to a lower average heat price as the number of consumers increases and the nuclear heat consumption increases. The specific fixed costs, which constitute the main part of the price, are gradually decreasing.

3. PROJECTIONS OF HEAT CONSUMPTION IN THE TOWN OF TRNAVA

Heat is delivered from conventional sources (steam system) and from nuclear sources (hot water system). The estimated development of energy consumption for heating purposes from different sources is shown in Figs. 3-5.

The data indicate a falling share of heat delivery from conventional sources, from the present 39% to 12% in the year 2005, and a correspondingly increasing share of nuclear heat.

EXPERIMENTS ON SAFETY FEATURES AND COMPREHENSIVE APPLICATIONS OF THE 5 MW NUCLEAR HEATING REACTOR

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XA9848830

Abstract

The 5MW Nuclear Heating Reactor (NHR-5), was developed by the Institute of Nuclear Energy Technology (INET), Tsinghua University. The NHR-5 is a vessel type light water reactor with integrated arrangement, self-pressurized and natural circulation. The construction of the NHR-5 began in March 1986, its initial criticality was on November 3, 1989. and full power operation on December 16, 1989. Until now, the NHR-5 has been successfully operated for four winter seasons. Subsequently it is being utilized for various experimental studies. During the last few years, a number of experiments on safety features and a variety of applications have been carried out and important results were obtained. The experiments have demonstrated that the NHR is suitable for district heating, refrigeration, air conditioning and seawater desalination. The NHR as an advanced nuclear energy system with inherent and passive safety features will play an important role in solving the energy supply and environmental problems of China in the future.

1. BRIEF DESCRIPTION OF NHR-5

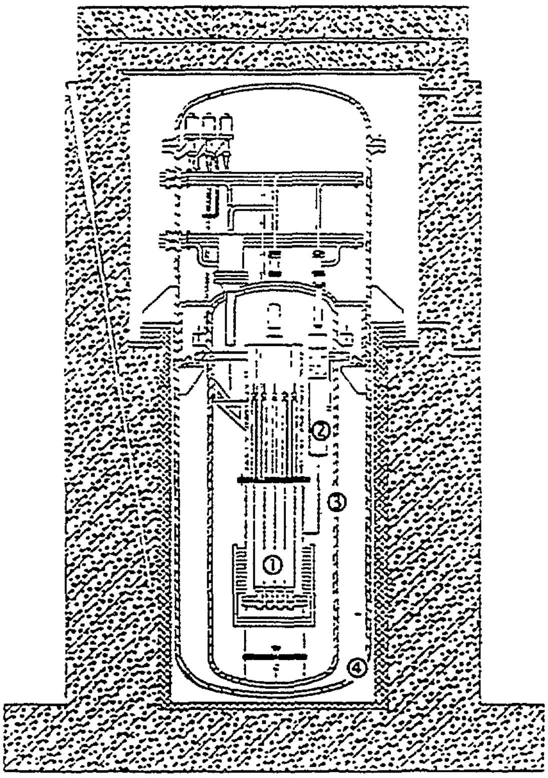
1.1. REACTOR DESIGN

The reactor core and main components of primary circuit are housed within a reactor pressure vessel (RPV, Fig. 1). The reactor core is located at the bottom of a hanging barrel, underneath the hanging barrel there is a secondary support. There is a long riser above the core outlet to enhance the natural circulation capability. The reactor core is cooled by natural circulation and heat is transferred to the intermediate circuit via primary heat exchangers. There are four primary heat exchangers in the downcomer between the riser and the vessel wall. Outside the RPV is a metallic containment vessel. The design pressure is 1.5MPa.

The gap between the RPV and the containment is very small. The containment is called "tight-containment". The positioning of all RPV penetrations is at a height of 3m above the core outlet and there are no large diameter penetrations of the RPV. Fig. 1 shows the structure of the NHR-5. A volume above the coolant level inside the RPV functions as a self-pressurizied volume of about 1m^3. The pressure inside the RPV depends on the initial partial pressure of nitrogen and the saturated vapor pressure corresponds to the core outlet temperature. Due to the nitrogen partial pressure, the coolant can be kept subcooled at the core outlet. This is a pressurized water operation mode.

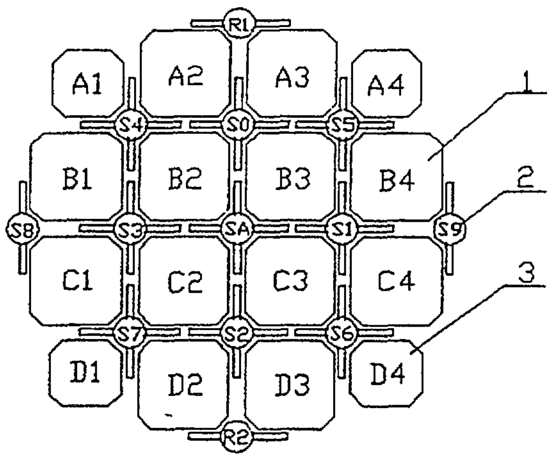
The core cross section of the NHR-5 is shown in Fig 2. In the core there are 12 fuel assemblies with 96 fuel rods and 4 assemblies with 35 fuel rods. The fuel rod with Zircaloy-4 cladding has an active length of 690mm and a diameter of 10mm. The nuclear fuel is uranium dioxide with an enrichment of 3%. The total amount of UO_2 loaded in the core is 0.508 tons.

The reactivity is controlled by a combination of fuel rods containing fixed burnable poison of 1.5% Gd_2O_3, movable absorption rods (boron carbide) and negative reactivity coefficient. In the core there are 13 control rods which are driven by a hydraulic driving system. The control rods can be dropped into the core by gravity in case of reactor shutdowns. A standby boron injection shutdown system is initiated by pumps or pressurized nitrogen during an anticipated transient without scram (ATWS).



① Core ③ Reactor pressure vessel
② Primary heat exchanger ④ Containment

Fig. 1 The NHR-5 integrated structure



1. Assembly with 96 fuel elements
2. Control rod
3. Assembly with 35 fuel elements

Fig. 2 The cross section of NHR-5

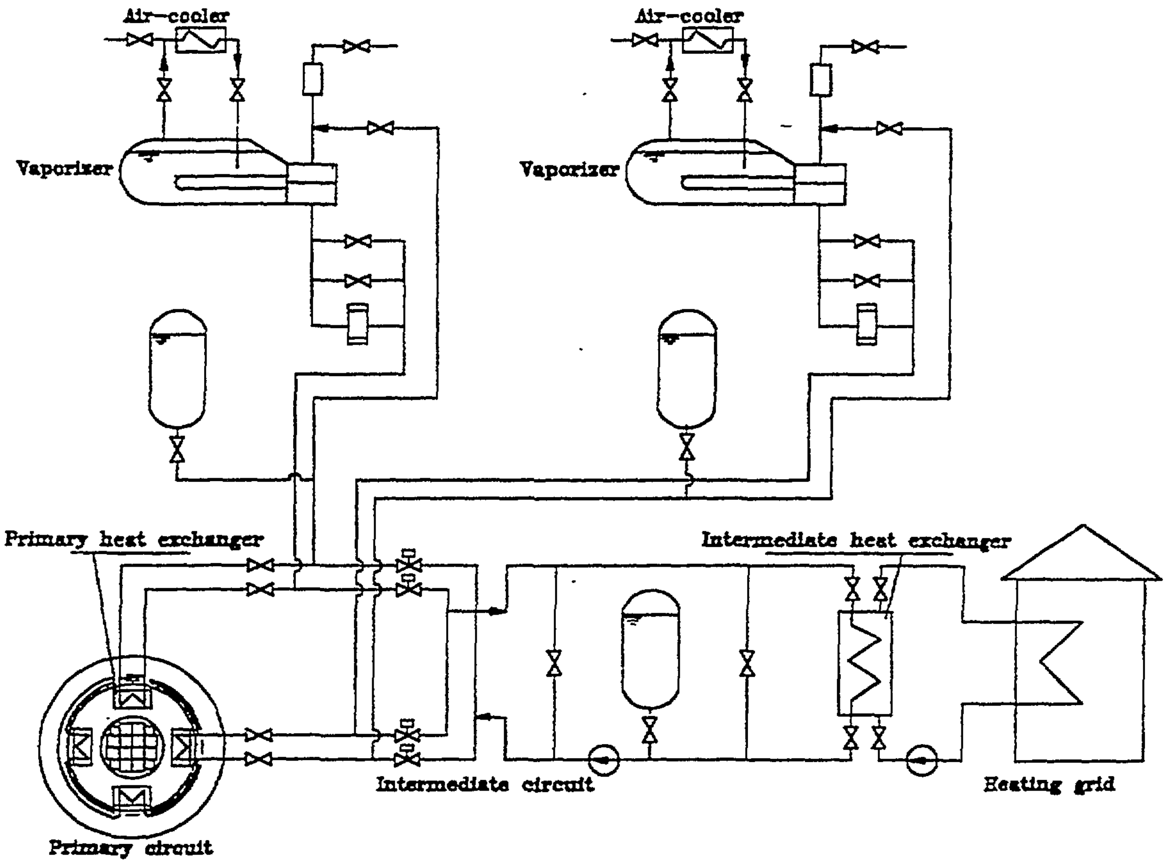


Fig. 3 Main heat transfer system and residual heat removal system

1.2. MAIN HEAT TRANSFER SYSTEM

The main heat transfer system is composed of three circuits, i.e. the primary circuit, the intermediate circuit and heat grid. The intermediate circuit connects with the primary circuit and the heat grid via the four primary heat exchangers and two intermediate heat exchangers, respectively. Four primary heat exchangers are divided into two groups parallelly, which are connected to a single loop through isolating valves. Heat generated in the core is transferred to the heat grid via the intermediate circuit. The main heat transfer system is shown in Fig. 3.

1.3. RESIDUAL HEAT REMOVAL SYSTEM

The residual heat removal system (RHRS) of the NHR-5 consists of two independent trains connected to two primary heat exchangers. In each train there are three natural circulation loops. Figure 3 shows a schematic diagram of the RHRS. After reactor shutdown decay heat is transferred to the intermediate circuit via the primary heat exchangers. The heat is carried to a vaporizer located at a high elevation in the reactor hall. This is the first natural circulation loop. The second natural circulation loop consists of the vaporizer and air cooler which is a process of vaporization and condensation. Finally, the decay heat is discharged to the atmosphere via the air cooler on the floor of the building by natural convection of air.

2. FEATURES OF DESIGN AND SAFETY

(1) Integral reactor arrangement and natural circulation

Due to an integral reactor configuration and natural circulation, the large bore coolant loops and big pumps outside the RPV are eliminated. All piping connections to the RPV are of small-bore and are located in the upper part of the RPV. The possibility of a large-break loss of coolant accident (LOCA) is thus eliminated, and the severity of small-break leaks is also reduced. Thus, the NHR-5 is not sensitive to a loss of coolant accident (LOCA) in the primary circuit.

TABLE I. MAIN OPERATING PARAMETERS OF NHR-5

Reactor core		
Reactor thermal power	MW(th)	5
Outlet temperature	°C	186
Inlet temperature	°C	151
Pressure in RPV	Mpa	1.37
Intermediate circuit		
Primary heat exchanger		
Outlet temperature	°C	144
Inlet temperature	°C	100
Intermediate heat exchanger		
Outlet temperature	°C	80
Inlet temperature	°C	144
Pressure	Mpa	1.70
Heating grid		
Heating water temperature	°C	84
Return water temperature	°C	56

(2) Large inventory of primary coolant

The specific coolant inventory in the RPV is larger than that in a commercial pressurized water reactor (PWR). The ratio of water inventory to power is 2t/MW (th) in NHR-5 and only about 0.2t/MW (th) in PWR. A large coolant inventory not only mitigates the possible accident consequences of a LOCA but also results in a longer grace period for operator actions.

(3) Tight-containment

The containment is very close to the RPV. The advantage of the design is that even in the case of an RPV failure at the worst location the core will continue to be covered with water.

(4) RPV

Owing to the integral design, the RPV has a large diameter for accommodating primary heat exchangers. The fast neutron dose in the wall of the RPV is considerably reduced so that the problem of radiation-induced embrittlement is avoided.

(5) Self-pressurization

The self-pressurized system is a passive system to maintain the pressure in the RPV, which has adequate capacity to meet any operational conditions and keep subcooling at the core outlet. The ratio of self-pressurized volume to power output in NHR-5 is also larger than that in the PWR ($0.2m^3$/MW (th) for the NHR-5 and about $0.01m^3$/MW (th) for a PWR). This makes the reactor operation easy and simple.

(6) Residual heat removal

Reactor decay heat is removed by a passive system. The principle of natural circulation by convection is used to transfer heat from the core to the ultimate heat sink (atmosphere) via primary heat exchanger, vaporizer and air-cooler in the RHRS. The RHRS can be operated under all conditions including the interruption of primary natural circulation caused by a LOCA.

(7) Installation of triple loops

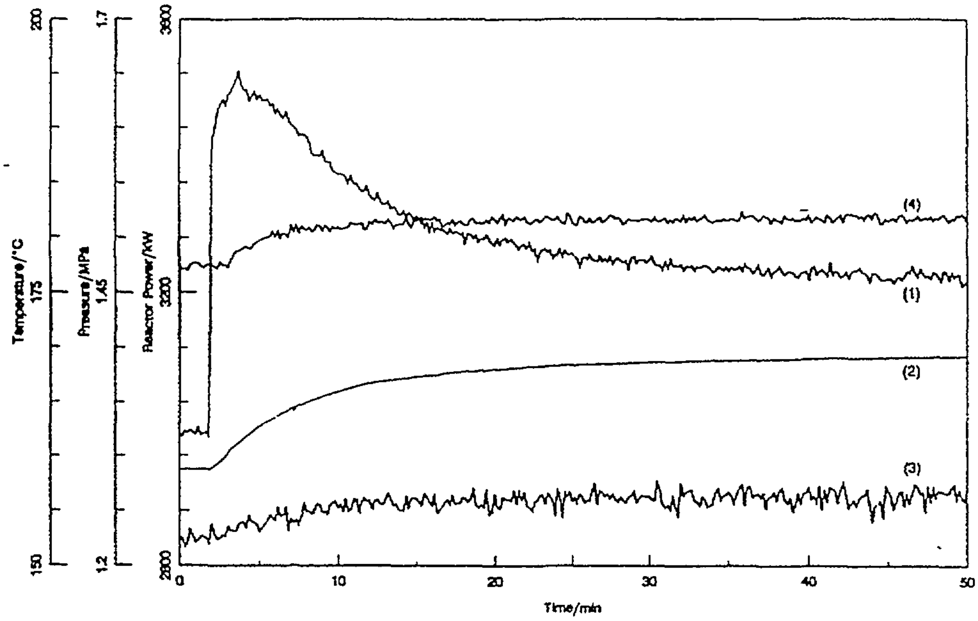
There is an intermediate loop between the primary circuit and the heating gird. The operating pressure in the intermediate circuit of the NHR-5 is always maintained higher than that in the primary circuit under all operating conditions, to keep the heating grid free of radioactivity.

For NHR, the pressure barrier can be installed at the primary or at the intermediate heat exchanger (HX). If the pressure barrier is installed at the primary HX, the pressure in the intermediate circuit should be higher than that of the primary circuit; if the pressure barrier is installed at the intermediate HX, the pressure in the heating grid should be higher than that of the intermediate circuit. Examples for the first and second approach are the NHR-200 and AST-500 design, respectively.

(8) Control rod driving system

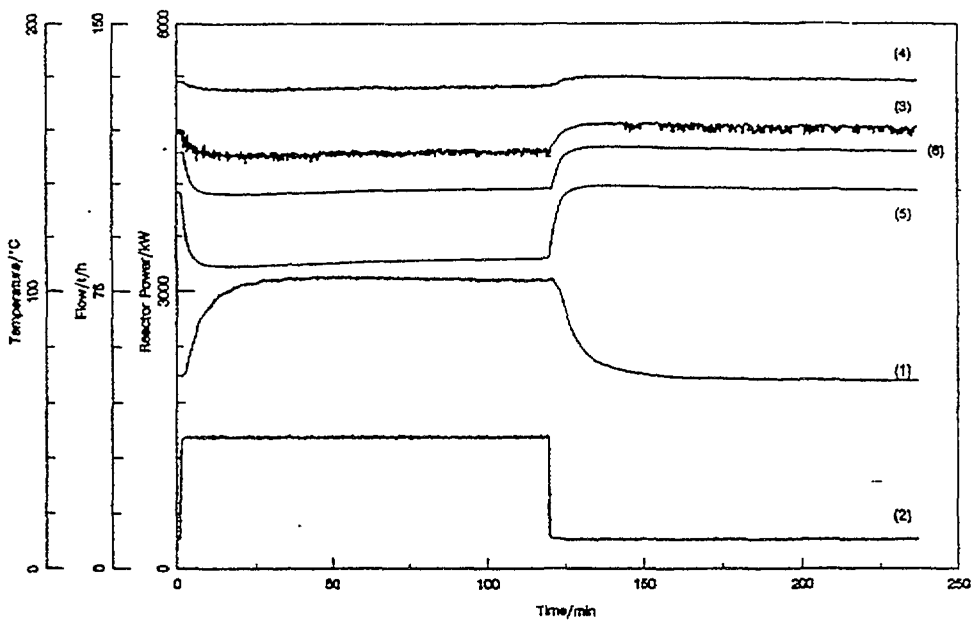
The control rod driving system uses a hydraulic driving mechanism. The control rods can be dropped into the reactor core by gravity under system failures, such as loss of power supply, depressurization piping break and pump shutdown as well as loss of pressure in the RPV.

Design features of the hydraulic driving mechanism prevent the control rods from being ejected in any condition.



1. Reactor power 2. Pressure in RPV
3. Core inlet temperature 4. Core outlet temperature

Fig. 4 The transient performance after inserting a reactivity of 2mk



1. Reactor power 2. Flow rate through intermediate heat exchanger
3. Core inlet temperature 4. Core outlet temperature
5. inlet temperature in 2nd side of primary heat exchanger
6. outlet temperature in 2nd side of primary heat exchanger

Fig. 5 The feature of self-regulation at NHR-5

(9) Simplified systems

Simplification of systems is an important feature of the NHR-5. Due to the design features mentioned above, the need for a separate emergency core cooling system, emergency feedwater system and emergency power source for active equipments such as large pumps etc., is eliminated.

(10) Strong negative reactivity coefficient

The NHR-5 is designed with strong negative reactivity coefficients. This makes a major contribution to safety and ease of operation. Various reactivity coefficients are listed in Table II.

TABLE II. VARIOUS REACTIVITY COEFFICIENTS

Fuel temperature coefficient	$-1.62 \times 10^{-5}/ ^{o}C$
Isothermic temperature coefficient	$-1.62 \times 10^{-4}/ ^{o}C$
Void coefficient	-2.20×10^{-3}/% void
(measured value at a temperature of 172 ^{o}C)	

3. VERIFICATION OF SAFETY FEATURES OF NHR-5

The NHR-5 is a new generation reactor with a higher degree of inherent safety characteristics, which have been proven by a number of experiments. In these experiments operator intervention in deliberately with held to demonstrate the level of safety achieved by the design.

3.1. TRANSIENT EXPERIMENTS

(1) Self-stability feature

The self-stability experiment was performed at a power of 3 MW (th) with a step increase in reactivity of 2mk. Figure 4 indicates the variation of reactor parameters. At the beginning of the transient the reactor power increased rapidly due to the extra reactivity and reached a maximum relative value of 1.18 in 100 seconds. Then the reactor power began to decrease due to the feedback of negative reactivity coefficient and came to a new stable power level at relative value of 1.08 in 30 minutes. The core inlet and outlet temperatures experienced an increment of 3.8^{o}C and 4.2^{o}C respectively. The reactor pressure increased by 0.102 MPa.

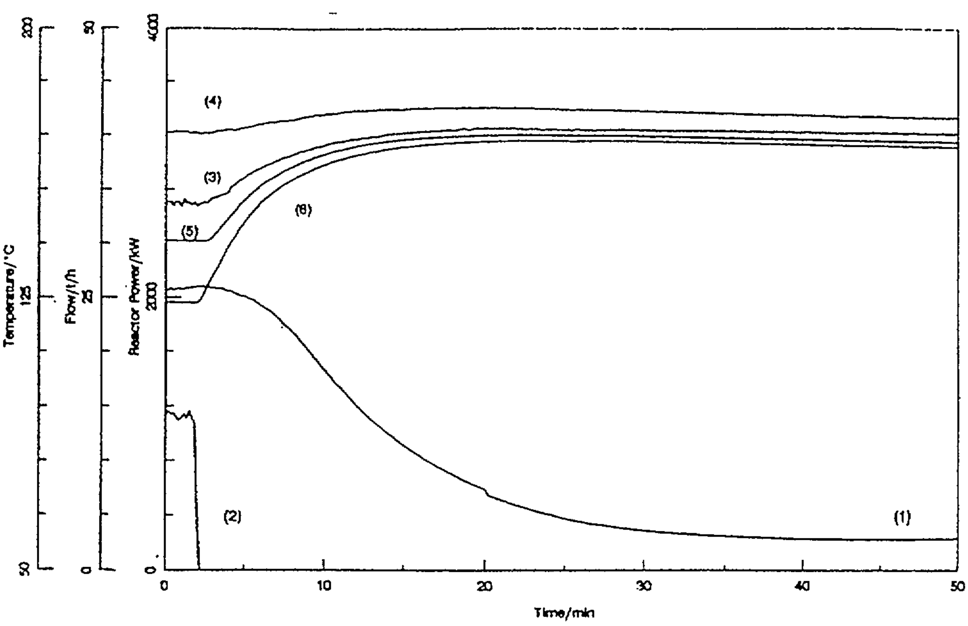
(2) Self-regulation feature

The self-regulation experiment was carried out at power level of 2 MW (th). The experiment is to investigate the ability of the reactor to automatically follow the changes in heating load. The heating load is varied by means of changing the flow rate through the intermediate heat exchanger.

The flow rate through the intermediate heat exchanger was varied from 8t/h to 35 t/h, then back to 8 t/h. This value corresponds to a heating load change from 2 MW (th) to 3 MW (th) and then back to 2 MW (th). Figure 5 shows the behavior of NHR-5 following the heating load change. The reactor power began to change by the self-regulating mechanism automatically after 90 seconds and reached a new power level to match the heating load within 30 minutes. The moderator temperature coefficient plays the main role in this process. The experimental results show that the NHR-5 has a very strong self-regulation ability to follow a load change without any operator action.

(3) Simulation Experiment for ATWS

In order to study the self-protection ability of the NHR-5 under the ATWS condition, an experiment has been carried out which simulated the ATWS, i.e. a loss of the main heat sink followed by the failure of reactor shutdown.



1. Reactor power 2. Flow rate through intermediate heat exchanger
3. Core inlet temperature 4. Core outlet temperature
5. Outlet temperature in 2nd side of primary heat exchanger
6. Inlet temperature in 2nd side of primary heat exchanger

Fig. 6 The transient of loss of main heat sink without scram

In this experiment, the intermediate heat exchanger was isolated at a reactor power of 2MWth, and no control rod was inserted into the core. Figure 6 shows the power variation and the changes in temperature and pressure in the primary circuit. The power decreased gradually and reached a stable value of about 0.2MWth by feedback of negative coefficients in about 30 minutes. The inlet and outlet temperatures of the reactor core rose by 20.4°C and 4.7°C, respectively. The temperature variation is not serious. The primary system pressure rose by 0.23MPa. The results of the experiment demonstrate that the NHR-5 has excellent passive safety features.

3.2. RESIDUAL HEAT REMOVAL UNDER THE INTERRUPTION OF NATURAL CIRCULATION IN THE PRIMARY CIRCUIT

When a loss of coolant accident (LOCA) occurs in the primary circuit, the water level inside the RPV will decrease. As a result of the water level decrease, the natural circulation of the primary circuit might be interrupted. In this case the residual heat of the reactor core will be transported by vapor condensed at the uncovered tubes of the primary heat exchangers.

In order to demonstrate the capability of residual heat removal under LOCA conditions, a special experiment was done at the NHR-5. After reactor shutdown the water in the reactor vessel was discharged by opening a valve to the blowdown tank.

The water discharge rate was $1.6m^3/h$ and an amount of $2.4m^3$ water was drained off. The water level in the reactor vessel decreased below the entry point to the primary heat exchanger and natural circulation was interrupted. In this case, the residual heat removal mainly occurred by condensation of the vapor. Due to the discharge of $2.4m^3$ water, the partial pressure of nitrogen reduced from 0.29MPa to 0.022MPa, so that the water subcooling of the reactor outlet temperature decreased form 12°C and 2°C

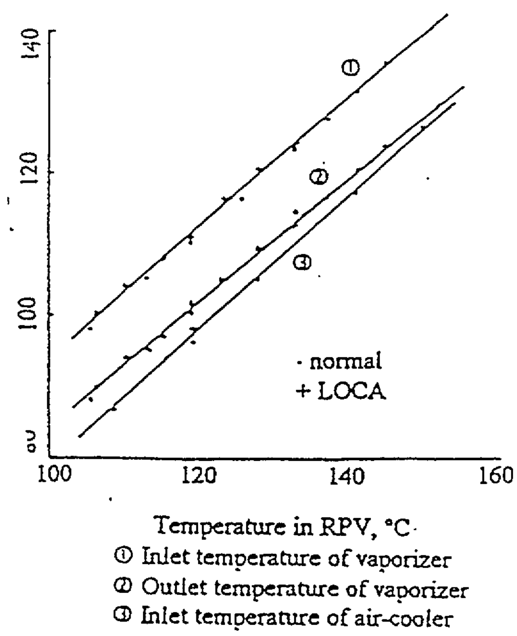


Fig. 7 The comparison of normal and abnormal operation condition of NHR-5

The reduction of subcooling enhanced the vaporization-condensation process. Figure 7 shows the comparison of the residual heat removal capabilities between LOCA condition and the normal operation condition. In Figure 7 it can be seen that the decay heat removal capacity under both LOCA and normal operation conditions are almost the same. Decay heat can be reliably removed by means of vapor condensation on the primary heat exchanger under LOCA conditions.

For nuclear heat only reactors (e.g. AST-500, NHR-5 and NHR-200) where the heating grid works as an ultimate heat sink and substantially influences the thermal conditions in the reactor, the intermediate circuit and heating grid transients and possible malfunctions and failures must be evaluated at the safety analysis. Such events as loss of flow in one of the two loops (e.g. valve closure or pump shutdown), inadvertent increase in heat comsumption (e.g. flow rate increase) are considered as design basis events.

At the NHR-5, the experiment on loss of heat sink without scram was conducted with the intermediate heat exchanger (grid HX) insulated on the side of the intermediate circuit. The experimental result indicated that a loss of flow in the intermediate circuit can cause the event of loss of the ultimate heat sink. As for the flow rate increasing in one of those two loops, the influence on the reactor operation is not serious (see the experiment of self-regulation at NHR-5).

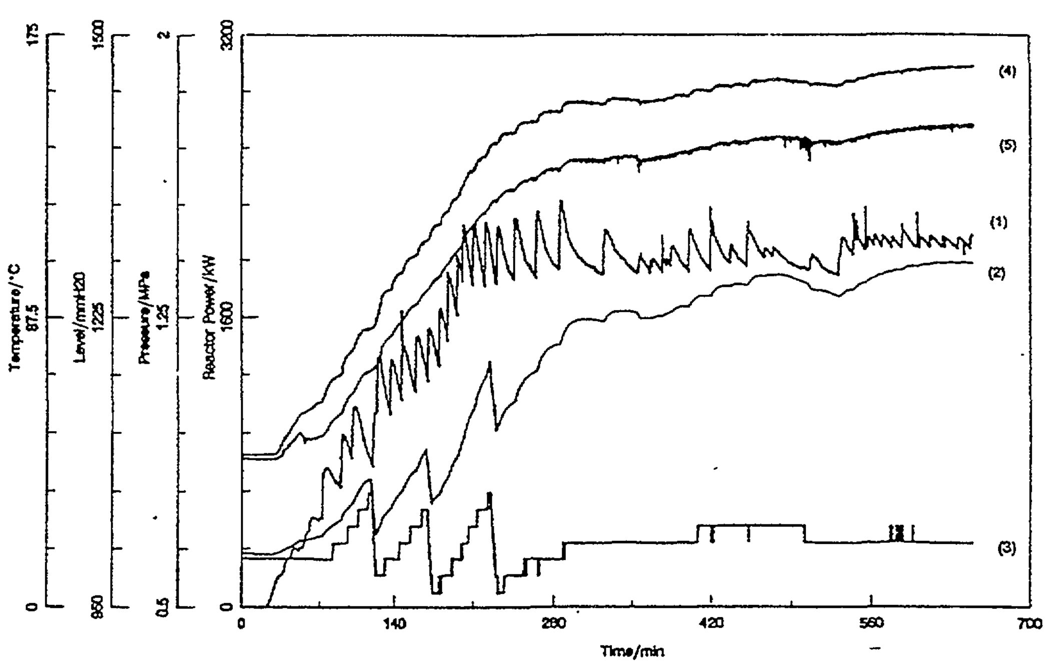
4. OPERATIONAL EXPERIENCE OF NHR-5

The NHR-5 is the first integral type heating reactor put in to operation in the world. Therefore it is highly valuable to understand the operational experience and the overall performance of the NHR-5.

4.1. REACTOR OPERATION

(1) Start up and operation

It is easy and simple to start up and operate the NHR-5. Start up of the NHR-5 is a process of taking the reactor from cold condition to the expected operational condition by means of its own nuclear heating. During the start up process attention should be paid to the following: need to set up initial partial



1. Reactor power 2. Pressure in RPV 3. Coolant level
4. Core outlet temperature 5. Core inlet temperature

Fig. 8 The start up process of NHR-5

pressure of nitrogen in the RPV; need to limit the rising temperature rate to less than 50°C/h in the primary circuit and the need to keep the coolant level within a certain range in the RPV. Figure 8 shows the start up process with full external heating load.

Adequate information is provided by the computer system on line for operators to know the reactor operating status. Due to inherent and passive features of the NHR-5, the operators would not be in a tense state of mind during the shift.

In the NHR-5 there is no power regulating system. The reactor power can follow the changes in external load by its self-regulating ability.

(2) High operation availability

Simplification of systems may be the key to obtain high operational availability. The NHR-5 as a heating reactor is only operated in winter. The operational availability is evaluated by a ratio of actual operated days to planned operation days. From December 1989 to March 1993, the NHR-5 has been operated for more than 9330 hours. The average availability for heating was more than 90%. During the period, there were four unexpected outages caused by failure of electricity supply and auxiliary systems.

4.2. PERFORMANCE OF THE MAIN SYSTEMS

(1) Control rod driving system

The hydraulic driving system of control rods was satisfactory for starting up, regulating reactor power and reactor shutdown during the past operation period. The full travel time for dropping the control rods into the core was maintained at a value less than 2 seconds.

Owing to the use of a temperature compensation device in the hydraulic driving system, there is no need to adjust the flow rate of the system at high temperatures.

The ultrasonic position indicators were satisfactory for indicating the position of control rods under the pressurized water operation mode also. The ultrasonic indicator system might not work under two phase flow conditions.

(2) Main heat transfer system

The primary circuit operated very well. Natural circulation could be established quickly at a very low reactor power. An unsymmetrical operation experiment of primary heat exchangers was conducted. The results show that the core inlet temperatures would have a large difference which may affect the power distribution in the core. There was no variation of temperature distribution in the RPV.

The intermediate circuit has the functions of isolating, heat transfer and detecting its own leakage. Operational experience indicates that the isolating function can be maintained under all conditions; leakage with more than 2 l/h can be found by the level variation in the pressurized tank in the loop; the normal operation mode of the circuit is to keep the flow rate through the primary heat exchangers constant and only adjust the flow rate through the intermediate heat exchangers. The advantage is that the average temperature in the circuit can be increased especially when the reactor is operated at partial power.

(3) Residual heat removal system

The operational experience of passive residual heat removal system is satisfactory. The three natural circulation loops can be reliably established and can be put in operation under any conditions including an interruption of natural circulation in the primary circuit.

To prevent the air-coolers from freezing it is adequate to have a small flow pass through the vaporizer from the intermediate circuit during normal operation. Experience indicates that the RHRS can also be operated at a Lower than 100^{o}C. That means the reactor can be cooled down to the cold shutdown state by the RHRS alone.

4.3. WATER CHEMISTRY OF NHR-5

The water chemistry of NHR-5 is different from that of PWR and BWR. It has a neutral pH without containing boric solution. There is no need to add hydrogen in primary coolant as oxygen is removed by the chemical additive (C_2H_4).

The results of monitoring and analysis show that the dissolved oxygen can be maintained at a level of 40g/kg and pH value of 6-7. Table III listed the results of analysis and the specifications of primary coolant chemistry.

TABLE III. SPECIFICATION AND MONITORED RESULTS FOR PRIMARY COOLANT

Item		Specification	Results of analysis
Dissolved oxygen	µg/kg	<50	30-40
PH (25°C)	---	6-10	6-7
Fluorine (F)	µg/kg	<100	<50
Chlorine (Cl)	µg/kg	<100	<50
Chromium (Cr)	µg/kg	<10	<0.1
Iron (Fe)	µg/kg	<10	<0.05
Sodium (Na)	µg/kg	<5	<5
Copper (Cu)	µg/kg	-	<0.2
Nitrate (NO_3)	µg/kg	-	<5
Nitrite (NO_2)	µg/kg	-	<5
Total solids	mg/kg	<1	<0.5-1

The nitrate and nitrite are less than 5μg/kg in the coolant at all operating conditions. This concentration is too low to cause metal structure corrosion . So the use of nitrogen as covered gas is feasible for NHR-5.

In order to effectively decrease the dissolved oxygen level in the primary coolant, some modifications are needed in the future. The first is to remove oxygen from the makeup water, the second is to add an additive into the primary circuit continuously and the third is to exhaust the air from the nitrogen supply lines, especially after the nitrogen cylinder is replaced.

4.4. RADIATION PROTECTION AND ENVIRONMENTAL IMPACTS

(1) Specific radioactivity of water in the three loops

During operation the water radioactivity level in the primary circuit, intermediate circuit and heating grid have been regularly monitored. The radioactive background of potable water in this area is about 0.10 Bq/l. The radioactivity level in the water of the intermediate circuit and the heating grid are as low as that of potable water. In the primary circuit the specific radioactivity of coolant is in the range of 2.5×10^2 to 2.7×10^3 Bq/l. The nuclide analysis showed that there were no fission products in the coolant. The intermediate circuit provides a perfect solution to keep the heating grid free of radioactivity.

(2) Dose levels

The distribution of dose levels in the NHR-5 is reasonable. A large part of the building has very low dose level, very near the background level. A higher dose level is found outside the biological shielding where the regenerator of the primary purification system is located. A local shielding with lead has to be added to reduce the dose level.

(3) Effluent

During normal operation, radioactivity of the gaseous effluents is radioactive at the same level as that of the background. The nuclide analysis indicated that there were no nuclides in the effluents. The nuclides in the effluents are natural ^{40}K and Radon daughters.

The amount of waste water produced from operation and maintenance is about $10.2 m^3$ in four years.

(4) Collective dose

The collective dose for all operators in each heating period is also very low as listed in Table IV. The data demonstrate that the radiation protection design of the NHR-5 was satisfactory.

TABLE IV. COLLECTIVE DOSE FOR ALL OPERATORS IN EACH HEATING PERIOD

Period	Collective dose (mSv-man)
1989.11-1990.3	2.4
1990.11-1991.2	3.2
1991.11-1992.3	11.4

Other parameters regularly monitored onsite and offsite include dose levels and cross - radioactivity levels in aerosol, service drain, and other samples.

All the measured data indicate that the NHR-5 operation does not cause any change in the background radioactivity level in this area.

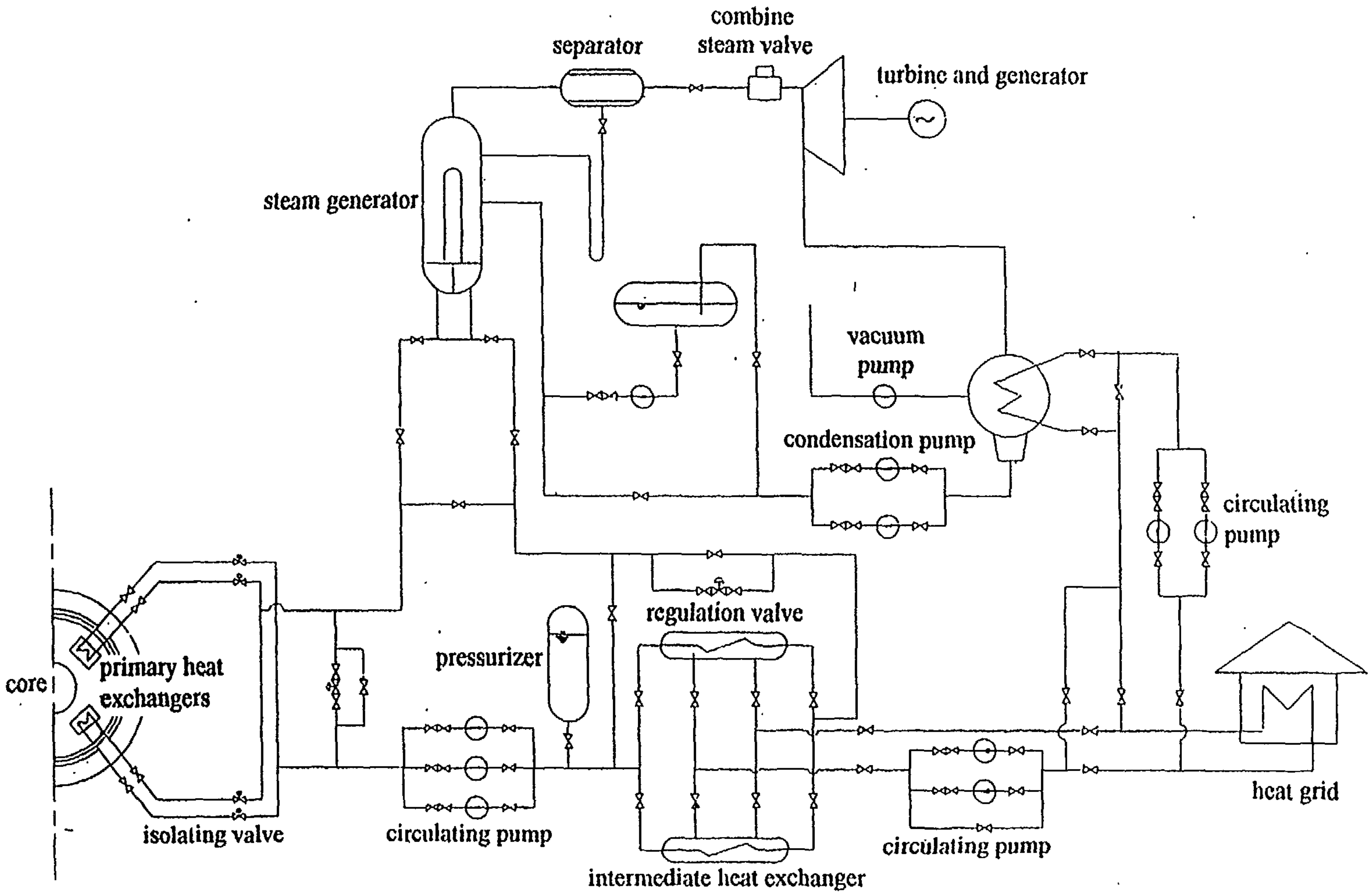


Fig. 9 Steam-power conversion system of NHR-5

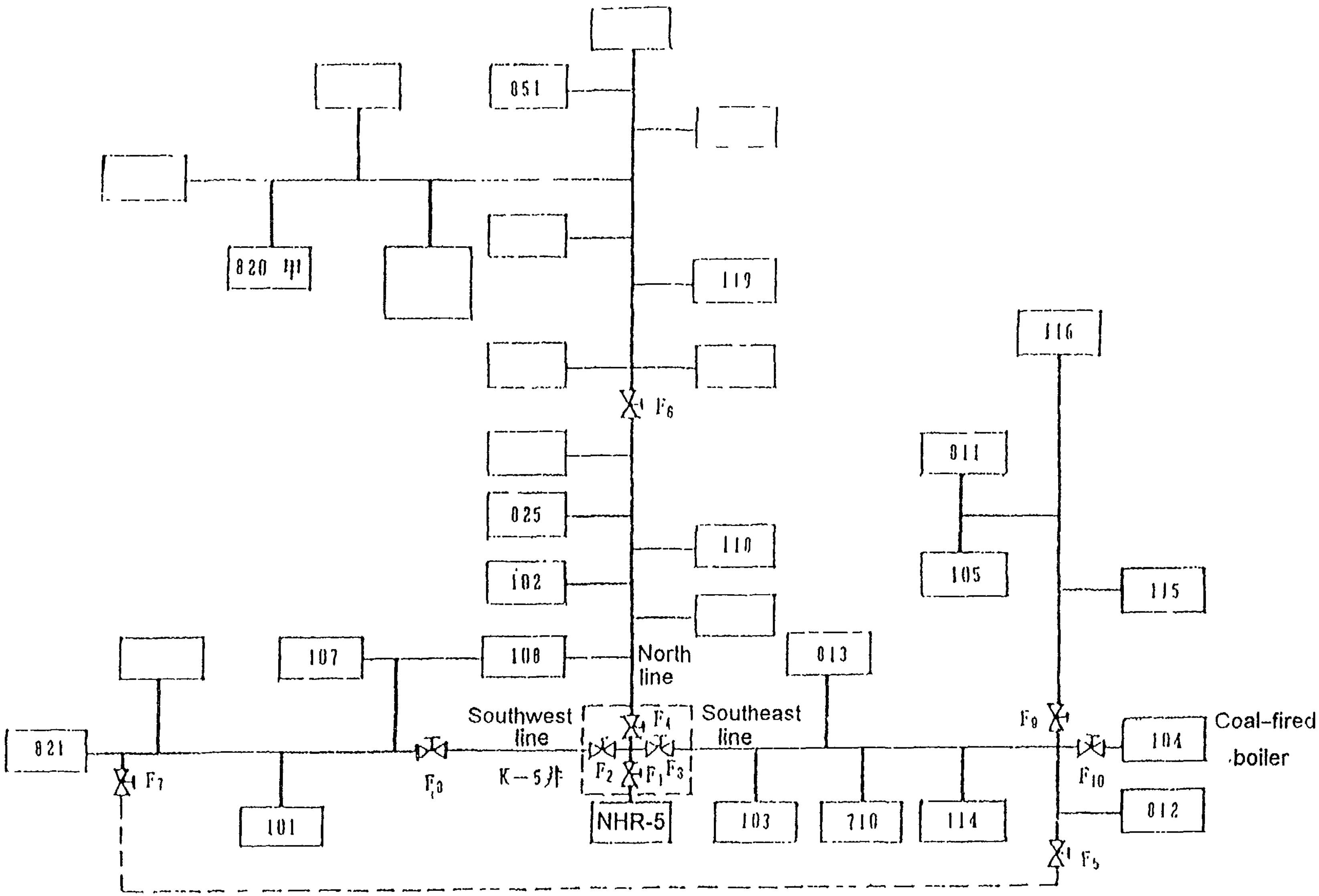


Fig 10 Diagram of heating grid of the NHR-5

5. EXPERIMENTS FOR VARIED APPLICATIONS OF NHR-5

A nuclear heating reactor can supply hot water and low temperature steam for district heating, air conditioning, seawater desalination and other industrial processes. A number of experiments on various applications of the nuclear heating reactor have been carried out over the past years at the NHR-5. All of these experiments indicated that applications of nuclear heat would have very good prospects in future.

The development of nuclear heat applications is dependent on the safety and economics of the NHR energy system.

5.1. COGENERATION OF HEAT AND ELECTRICITY

The experimental cogeneration of heat and electricity was performed at the NHR-5.

The experimental system given in Figure 9 includes four loops. A steam generator (SG) is installed in the intermediate circuit, the steam loop is for electricity generation and the circulating water loop is for heating.

Low pressure steam from the SG was first used for electricity generation and then exhaust steam was extracted from the last turbine stage for heating. The electricity generation efficiency was 7.5% in this experiment. Operating parameters are listed in Table V.

This experiment shows it is feasible for the nuclear heating reactor to cogenerate electricity and district heating or desalination as well as supply industrial process steam (0.8 MPa), as far as the design parameters of NHR can meet the requirements.

TABLE V. OPERATING PARAMETERS OF COGENERATION-HEAT AND ELECTRICITY OF NHR-5

Reactor thermal power	MW	2.0
Steam generator power	MW	1.8
Electricity output	MW	0.15
Primary circuit		
Pressure (g)	MPa	1.47
Core inlet temperature	°C	173
Core outlet temperature	°C	192
Intermediate circuit		
Pressure (g)	MPa	1.7
Inlet temperature of primary heat exchanger	°C	147
Outlet temperature of primary heat exchanger	°C	167
Flow rate	t/h	124
Steam circuit		
Outlet pressure of SG	MPa	0.43
Inlet pressure of turbine	MPa	0.38
Back pressure of turbine	MPa	0.025
Temperature of condensate	°C	65
Heating grid		
Temperature of supply water	°C	55
Temperature of return water	°C	45

The INET located in the north of Beijing is 40 km away from Tsinghua University. The heating grid in INET consists of three parts-the southwest, southeast and north line (Fig. 10). The heating area of all buildings in INET was about 40,000 m^3 in 1989. The heating for all buildings was provided by a

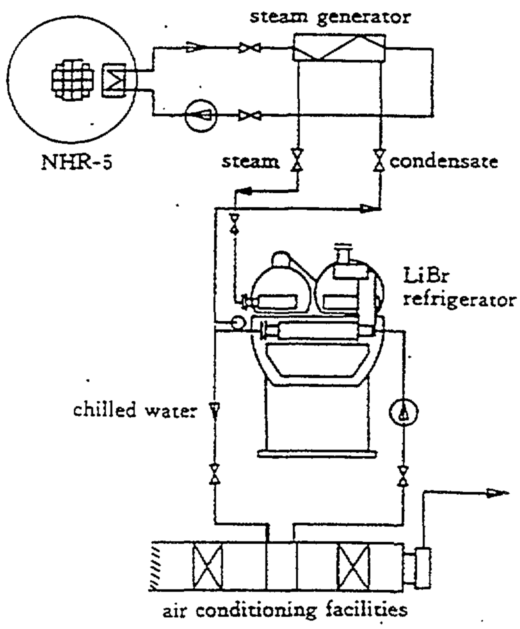


Fig. 11 Schematic diagram of an NHR air-conditioning system

TABLE VI. MAIN PARAMETERS OF NUCLEAR HEAT REFRIGERATION SYSTEM

Reactor thermal power	MW	0.6
Primary circuit		
Pressure	MPa	1.2
Core inlet temperature	^{o}C	160
Core outlet temprature	^{o}C	169
Intermediate circuit		
Pressure	MPa	1.7
Inlet temperature of primary heat exchanger	^{o}C	155
Outlet temperature of primary heat exchanger	^{o}C	159
Steam circuit		
Outlet pressure of SG	MPa	0.55
Steam flow rate	kg/ h	400
Steam temperature	^{o}C	154
Refrigerator		
Steam inlet pressure	MPa	0.40
Steam inlet temperature	^{o}C	143
Power output	kW	232.6
Chilled water circuit		
Outlet temperature of chilled water	^{o}C	7
Temperature of return chilled water	^{o}C	12
Refrigeration coefficient	-	0.9

coal-fired boiler before the year 1989, and from 1989 to 1993 the heating was provided by the NHR-5. In order to connect with the NHR-5, four valves (F1-F4) were added in the heating grid, other parts of the heating grid were not changed.

5.2. REFRIGERATION

Using a nuclear heating reactor as the energy source for air conditioning is an important aspect of its application. In this way, it may be used not only for heating in the winter but also for air conditioning and industrial refrigeration in the summer. According to this idea the nuclear heating reactors may be constructed in cities such as Shanghai. Nanjing and Wuhan, where district heating in winter and air conditioning in summer are needed.

In order to demonstrate the feasibility of nuclear heat refrigeration for air conditioning an experiment was performed at the NHR-5.

The experimental system is shown in Figure 11. A double-effect LiBr absorption refrigerator with a capacity of 0.84 MJ/h was used in this expermental system for producing chilled water. The air-conditioned area was 2500m^2. The main operating parameters are listed in Table VI.

The experimental results show that use of nuclear heat for refrigeration is feasible.

5.3. NUCLEAR DESALINATION

In China most coastal cities have suffered from fresh water shortage for a long time. Seawater desalination using nuclear energy would be one of the ways to solve the problem of fresh water shortage. Work on a nuclear desalination feasibility study using NHR-200 and experimental investigations have been conducted for the past few years in INET. The status of experimental investigations is presented below.

An experimental desalination system using MED process with 5 effects is planned to be built in INET. In fact the experimental system is a part of a desalination system with 21 effects, which would be able to simulate operation of the five initial effects and any successive four effects. If the experimental system simulates operation of any successive four effects, the first effect will serve as "re-boiler" to supply vapor with the required parameters to the following four effects. The experimental desalination system will be coupled with the NHR-5. The MED system with 5 effects is the fourth circuit. The design parameters of the 21 effect desalination system, are listed in Table VII.

TABLE VII. DESIGN PARAMETERS OF SEAWATER DESALINATION SYSTEM

Steam temperature at inlet of the first effect	^oC	120
Steam flow rate	kg/h	310
Temperature of seawater at inlet of the first effect	^oC	119
Seawater flow rate	kg/h	10
Steam temperature in last effect	^oC	39
Seawater temperature in last effect	^oC	36
Raw seawater temperature	^oC	25
Gain-output ratio (GOR)	-	17.2
Fresh water production	t/d	120

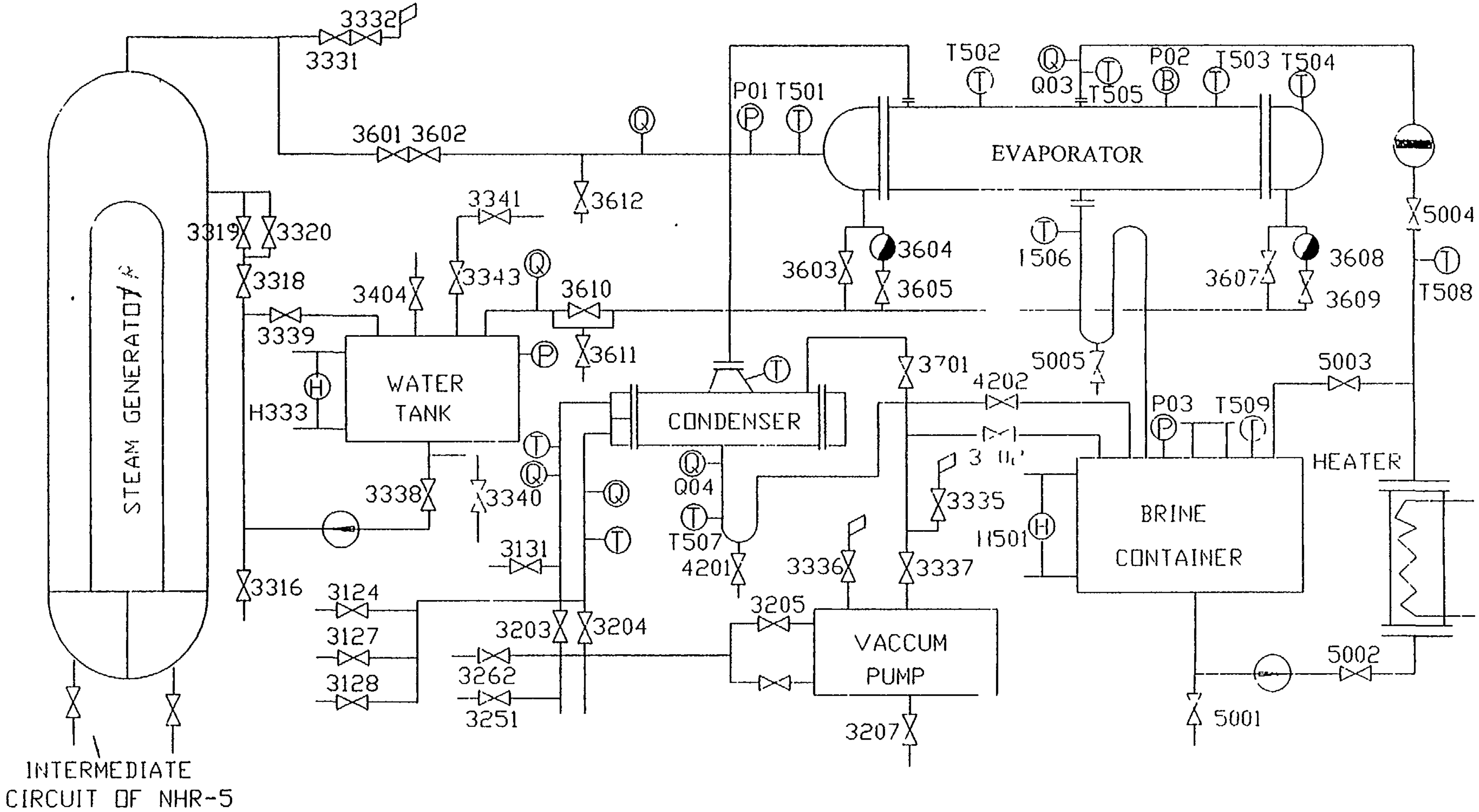


Fig. 12 Flow chart of sea water desalination

The main objectives of building the 5 effect desalination system are as follows:

- To accumulate design and operating experience.
- To investigate the interactions between the NHR-5 and the water plant at different coditions.
- To explore the safety aspects and the way to increase the efficiency of water production.

As a first step, an experimental system with one MED effect has been built. The system shown in Figure 12 is composed of four parts: a vaporizer of the MED, a condenser, a seawater container with heaters and vaccum pump.

The heating steam from the SG in the intermediate circuit of the NHR-5 enters the tube side of vaporizer to heat the sea water on the shell side. The vapor distilled on the shell side of the vaporizer is condensed to fresh water in the condenser and the rest goes back to the sea water container. The performance test of vaporizer and re-boiler have been carried out using this system.

The vaporizer and re-boiler would be used in the seawater desalination system with 5 effects of MED process. The experimental results indicate that the vaporizer design is suitable, and the reboiler can provide vapor with temperatures of 85, 71 and 54^{o}C for simulating experiments of seawater desalination.

6. SUMMARY AND CONCLUSIONS

The operational experience and the results of a number of experiments on safety features have shown that the NHR-5 does achieve an excellent performance. The experiments on comprehensive application have demonstrated that the NHR is suitable for disrict heating, refrigeration, air conditioning and seawater desalination. The NHR as an advanced nuclear energy system with inherent and passive safety features will play an important role in solving the energy supply and environmental problems of China in the future.

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II.3. OPERATIONAL EXPERIENCE WITH NUCLEAR HEAT APPLICATION

Experience with nuclear desalination

EXPERIENCE IN THE APPLICATION OF NUCLEAR ENERGY FOR DESALINATION AND INDUSTRIAL USE IN KAZAKHSTAN

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Abstract

Key design features of the Aktau complex in Kazakhstan with a 1000 MWth fast breeder nuclear reactor are outlined. The experience gained over 20 years of operation and maintenance is briefed. The water costs, the impact on the environment and the water and steam quality have confirmed the efficiency and the reliability of nuclear energy application for seawater desalination and industrial use.

1. INTRODUCTION

Recently the interest in using nuclear energy for non-electrical applications has increased. In line with the ongoing development of nuclear engineering and technology, more reliable and safe reactors are being designed. The public opinion is changing in many countries in favour of development of nuclear energy.

It is the time to evaluate existing experience gained in various countries. Mangyshlak Atomic Energy Complex has been in service since 1973, providing the whole region with electricity, heat and potable water. This paper outlines some representative features of this experience.

2. DESCRIPTION

2.1 Layout

The atomic energy complex is located on the Mangyshlak peninsula, on the East Coast of the Caspian Sea. It is located in an arid zone, with a small amount of rainfall and a limited stock of groundwater which has a salinity of 3.5-5.8 g/litre. The complex was constructed on a platform located 12 km from the city, next to the developed industrial enterprises, and a potable water preparation station was built at a place closer to the town.

Relevant objectives for constructing the complex at the place were:

- To minimize heat losses during the transmission of steam to the industrial enterprises;
- To minimize the water transport costs;

Prevention of the inhabitants exposure to radiation beyond the specified limits for accidents had been also taken into consideration.

The Mangyshlak Atomic Energy Complex consists of:

- A liquid metal fast breeder reactor BN-350;
- A gas-oil fuelled thermal power station;
- A potable water preparation station;
- Ten multi-effect distillation (MED) desalination units, with a production capacity of 8 000 to 14 500 m3/day each;
- A feed water preparation plant for the steam generators.

2.2 Nuclear Reactor

The BN-350 is a loop type fast breeder reactor cooled with liquid sodium. Its main design features and operational history are described in [1]. The reactor has six primary loops located in individual airtight rooms and six secondary loops located in two rooms at opposite ends of the reactor vessel [1]. Each loop has a pump and an intermediate Na-Na heat exchanger. The thermal section of the reactor consists of six steam generators with natural circulation. The reactor core is surrounded by a blanket of depleted uranium. Iron ore concentrate, graphite, steel and concrete are used for the biological shielding. The negative power and temperature coefficients of the core provide self-stability of the reactor. Heat release in the core is stable and the excess reactivity is low. Operation of the reactor is simple. The low pressure of the sodium coolant and the absence of noticeable corrosion ensure the leak tightness of sodium piping and components.

The reactor was designed to produce 1000 MW thermal power, 150 MW of electrical energy and 120 000 ton of water per day but has never been operated at its designed capacity. Operation of the reactor started on 29 November 1972 and since then the reactor has been supplying heat for desalination, industrial use and for electricity production. Tables I and II present the main technical parameters and operating record that were achieved in practice. The maximum output of 750 MWth was achieved in 1984. The restriction of thermal flow in the steam generators was a main obstacle for the operation of the reactor at its rated power. It is possible to increase the power up to 800 MWth. But according to the latest safety requirements, expensive modernization of the safety systems would be necessary for the power increase.

The reactor BN-350 was designed to operate for 20 years. This period ended in May 1993. Thorough inspections led to the authorities to conclude the possibility of continued operation of the reactor with safety and reliability up to 2003.

Table I. MAIN TECHNICAL PARAMETERS OF BN-350

Reactor thermal power	750 MW (th)
Inlet/outlet sodium temperature of the primary loops	288/437°C
Inlet/outlet sodium temperature of the secondary loops	260/420°C
Steam outlet temperature	405°C
Steam pressure	4.5 MPa
Steam flow	1070 t/h
Loop number	5

Table II. PRINCIPAL RESULTS OF THE REACTOR BN-350 OPERATION FROM 1973 TO1995

Power level operation	159 921 hours
Average power	592 MW (th)
Average load factor	0.85
Refueling number	56
Number of unplanned power decrease	62
Fuel burnup	11.8 %

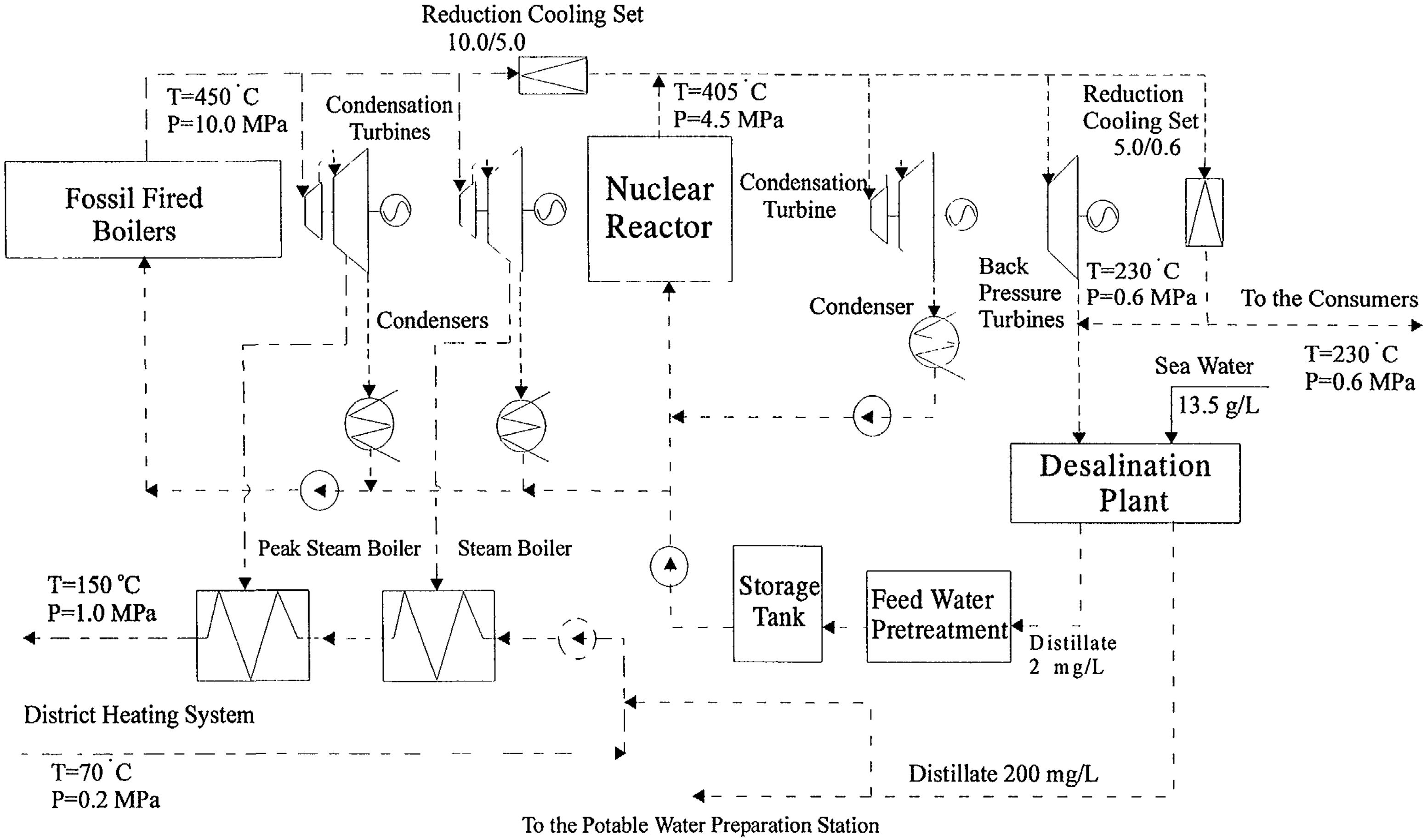

Fig. 1. Principle Flow Diagram of Mangyshlak Atomic Energy Complex

2.3 Heat Application Scheme

The complex includes a thermal power station, a nuclear reactor, a condenser and three back pressure turbogenerators (Fig. 1). The exhaust steam (0.6 MPa) from the back pressure turbines is used as the heat source in the heating chambers of the first stage evaporators of the desalination plant. If more steam from the reactor is available than required for the desalination, it is used to supply heat energy to the industrial enterprises and settlements located within 7 km from the complex.

There are also some steam boilers for district heating and for hot water supply. The boilers are usually heated by extraction steam from the turbogenerators of the thermal power station. A transit pipeline exists between the thermal power station and the nuclear power station, which allows to use the 0.6 MPa steam from the back pressure turbines of the reactor for supplying heat to the district heating system.

Part of the distillates from the desalination plant is fed to the district heating system to keep the normal pressure, since hot water in the district heating system is partially consumed by inhabitants.

Pressure reducing and cooling devices are installed in order to provide redundant high grade steam of the reactor to the desalination units during the periods of low electrical power demand, or in case of turbine failures. Another set of devices can provide steam to the desalination plant from the fossil fired boilers of the thermal power station when the reactor is shut down or when steam from the nuclear reactor is not sufficient for required water production. To meet daily peak demands of water, reservoir tanks are installed for the distillates and for the feedwater of the steam generators and boilers.

The turbogenerators are in the turbine hall of the thermal power station, separate from the reactor building. Maintenance is performed by the thermal power station personnel.

There is an independent source of electric power and water which can start up the complex, even when the complex is disconnected from the regional electric grid.

3. OPERATION AND MAINTENANCE

3.1. Reliability of the operation

The reactor has been in operation since 1973, with a cumulative operating time of more than 150 000 hours. During this period, there were no significant sodium leaks in the primary and secondary loops. Abnormal operation of the sodium pumps was not observed. Cavitation damage in the driving wheels of the pumps was insignificant. In the first period of reactor operation, depressurization of the heat exchanger tubes of the steam generators was repeatedly experienced due to manufacturing defects of the tubes. When the steam generators and the water preparation system were reconstructed, one loop of the thermal section was disconnected and another loop is maintained as a reserve. Because of this reserve loop, failure of the heat exchanger tubes no longer result in a decrease in the operating power of the reactor. After a modification of the feedwater preparing system, operation of this section has become more stable (Table III). Since the nuclear reactor came into service there have been no serious problems with the reactor equipment.

Twice a year the nuclear reactor is shut down for twenty days for refuelling and scheduled maintenance. During the periods the heat for the desalination plant is supplied by the thermal power station. Such switching between heat sources for the desalination plant has been carried out regularly for more than 24 years, and no problems have arisen.

Table III. FEEDWATER CHARACTERISTICS

Conductivity, (μS/cm)	< 40.0
pH	9.1 ± 0.2
Iron, (μg/l)	< 10.0
Copper, (μg/l)	< 5.0
Total Hardness, (μg/l)	< 33.2

3.2. Safety Aspects

The anticipated events of the abnormal operation of the fast breeder reactor BN-350 include 40 initiating events. In some of them, one loop of the thermal section should be disconnected. It calls for the decreasing of the reactor power by approximately 10-12%. Sixty four such events have been experienced during the 24 years of reactor operation.

When necessary, the high-speed reactor shut down system is activated. Since 1973 there were 77 such cases when the emergency response system functioned. During the last six years, such events were not observed.

Water and steam leaks in the steam generator result in sodium-water reactions, generating hydrogen and emitting energy. But even a complete destruction of heat exchanger tubes does not cause a pressure pulsation of above 0.5 MPa. When it happens, the pressure of the water circuit is decreased quickly, and the defective loop is disconnected from the steam and water collectors. The ingress of the reaction products to the thermal section steam is avoided. These products are completely contained in the second contour and transferred to a separate tank.

There are four variants of probable failures considered for BN-350.

- Sodium leaks in the primary loop;
- Gas system tightness failure;
- Loss of sodium circulation in one of the core subassemblies;
- Damage of the core subassembly during refuelling.

None of these failures would lead to the contamination of the steam delivered to the consumers. This is supported by relevant features of the fast breeder reactor with liquid metal cooling:

- Low pressure in the reactor vessel
- Three circuits system of reactor cooling, where the first one has the lowest pressure, and the last one the highest

Owing to FBR features fast destruction of the core can be avoided. Even for such heavy failures the steam circuit of the reactor is not exposed to radioactive contamination.

Three independent water sources are connected to the deaerated water circuit and to the emergency feed water tanks: Storage of feed water for the reactor steam generators; water tanks on the thermal power station, and storage of distillate of the desalination plant. They can provide the necessary heat sink to the reactor during emergency cooling. It provides high reliability and safety of the emergency cooling of the reactor.

3.3. Emergency response systems

The emergency protection systems of the nuclear desalination plant are the same as those established for other reactors and thermal power stations. The activation of local protection systems does not have much influence on operation of the nuclear desalination plant. However the situation changes if the high-speed reactor shut down system is activated. In this case only one steam generator remains in the operating mode for reactor cooling. Steam production decreases from 700 t/h to zero within a few minutes, therefore it is necessary to increase the steam supply from the fossil fired boilers in order to keep the desalination system in service. Since it is impossible to offset such a deficit in a short time, some heat and electric power consumers have to be temporarily disconnected.

3.4. Product water quality

There are two distillate lines from the Desalination Plant: for drinking water quality for potable water production (TDS up to 200 mg/L) and for high quality water (TDS= 2 to 10 mg/L) for preparing feedwater and technological need of industrial use. The characteristics of the product water are given in Tables IV. The quality of the product water meets the WHO requirements, and does not depend on the heat source used, i.e. nuclear or conventional. The amount of radioactive substances in the product potable water is determined by the contents in the mineral water which is added to the distillate to obtain the drinking water quality. The analysis of tritium in the nuclear desalination plant streams has shown that it is at the background level, close to that of the sea or the ground water. The tritium concentration in the steam and water of the thermal sections does not exceed 1.6×10^{-10} Ci/kg, though the allowable level for the drinking water is 0.81×10^{-6} Ci/kg.

Still measures are taken to reduce the tritium concentration in the distillate for preparing potable water. The distillate streams for preparing potable water are isolated from the high quality water line where condensates of the steam from the reactor steam generators and fuel fired boilers.

Table IV. WATER PRODUCT CHARACTERISTICS

Characteristics	WHO guideline values	Distillate «G»	Distillate «A»
Total dissolved solids (mg/L)	<1000	1.96	198.6
Temperature (^{o}C)	NG	45	28
Color (TCU)	15	-	-
Turbidity (FTU)	5	-	-
Conductivity (μS/cm)	NG	4.05	326.7
pH	6.5-8.5	8.46	8.07
Total Hardness (mg/L) ($CaCO_3$)	500	0.78	66.0
Chloride (mg/L)	250	0.48	55.6
Sulfate (mg/L)	400	0.31	33.2
Calcium (mg/L)	N.G.	0.08	7.6
Magnesium (mg/L)	N.G.	0.09	8.2
Sodium (mg/L)	200	0.18	48.5
Aluminum (mg/L)	0.3	-	-
Copper (mg/L)	1.0	0.013	0.06
Iron (mg/L)	0.3	0.033	0.09
Zinc (mg/L)	5.0	-	-
Fluoride (mg/L)	1.5	-	-
Nitrate (mg/L)	10.0	-	0.27
α-activity (Bq/L)	0.1	-	-
β-activity (Bq/L)	1.0	-	-

NG - no guideline value set

The feedwater for the steam generators is partly taken from the distillate, too. The distillate is already of high quality and requires only some minor final processing for that purpose. As compared with conventional technologies, the costs of feedwater preparation for the distillate are several times lower. Regeneration of the ion exchange filters is done twice a year.

3.5. Materials and corrosion

The operating behaviour of reactor equipment made of special steel has been generally satisfactory. Cu-Ni and stainless steel tubes were used for the tube bundles of the steam generators. Repeated damage of Cu-Ni steam generator tubes have been caused by their poor durability at elevated temperatures.

Because of the high quality of the feedwater, scale depositions were not observed in the heat exchanger tubes of the reactor's steam generators or in the fossil-fired boilers.

4. EVALUATION OF PRODUCTION COSTS

The costs of water, electric power and heat depend on the cost of fuel. As the prices of gas and reactor fuel rose between 1994 and 1996, the production costs have risen accordingly. Analysis of the average data has shown that these costs increased by 10-15% when the reactor was shut down for refuelling or repair works (Table V).

TABLE V. COMPARISON OF PRODUCTION COSTS

	Reactor operation (Normal)	Gas/oil plant operation (Reactor shutdown)
Distillate (US$/t)	0.956	1.08
Electric power (US$/kW(e)*h)	0.016	0.018
Thermal Energy (US$/GJ)	1.09	1.22

5. RADIOLOGICAL IMPACTS ON THE ENVIRONMENT

A big advantage of BN-350 is its minimum radiological impact on the environment. The average emission of radioactive gases is, including argon, xenon and krypton, 0-15 Ci/d (as compared with the allowable level of 500 Ci/d. These gaseous effluents have a short period of half-lives and are not harmful to the inhabitants. Operational experience and analysis of the design basis and beyond design basis accidents of the reactor have shown that the radiological consequences at all normal and abnormal operating conditions do not have any effect on the quality of the product water and steam.

The chemical contents of the brine and the cooling water discharged into the sea are also within the allowable limits. Overall emission of the radioactive substances to the sea does not exceed 1 Ci per year, that is below 3% of the allowable level. For many years, the artificial shallow lake, in which the disposal water is aerated and cleaned, has served as a place for the wintering birds and fishing.

6. CONCLUSION

The scheme applied at MAEC in Aktau has shown high reliability and flexibility of control owing to the combined use of a nuclear reactor and fossil fuel fired boilers.

No adverse effect on the environment and no contamination of the steam and water was experienced. The radiological characteristics of the product water is not different from those of the water obtained from the traditional desalination plants, and meet the WHO standards.

Over 20 years experience of safe and successful operation of MAEC in Aktau has confirmed the high reliability and efficiency of application of nuclear energy for desalination and industrial heat.

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