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Status of design concepts of nuclear desalination plants

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FOREWORD

Interest in using nuclear energy for producing potable water has been growing worldwide in the past decade. This has been motivated by a variety of factors, including economic competitiveness of nuclear energy, the growing need for worldwide energy supply diversification, the need to conserve limited supplies of fossil fuels, protecting the environment from greenhouse gas emissions, and potentially advantageous spin-off effects of nuclear technology for industrial development.

Various studies, and at least one demonstration project, have been considered by Member States with the aim of assessing the feasibility of using nuclear energy for desalination applications under specific conditions. In order to facilitate information exchange on the subject area, the IAEA has been active for a number of years in compiling related technical publications. In 1999, an interregional technical co-operation project on Integrated Nuclear Power and desalination System Design was launched to facilitate international collaboration for the joint development by technology holders and potential end users of an integrated nuclear desalination system.

This publication presents material on the current status of nuclear desalination activities and preliminary design concepts of nuclear desalination plants, as made available to the IAEA by various Member States. It is aimed at planners, designers and potential end-users in those Member States interested in further assessment of nuclear desalination.

Interested readers are also referred to two related and recent IAEA publications, which contain useful information in this area: Introduction of Nuclear Desalination: A Guidebook, Technical Report Series No. 400 (2000) and Safety Aspects of Nuclear Plants Coupled with Seawater Desalination Units, IAEA-TECDOC-1235 (2001).

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CHAPTER 1. INTRODUCTION

1.1. Background

The global need and demand for fresh water is ever increasing. As illustrated in Table 1.1 below, The World Water Vision [1] presents two diverging projections toward the year 2025 for use of renewable freshwater resources for business as usual scenario. The projections by Shiklomanov (1999) are based on the assumption that current trends can be extrapolated that reservoirs be constructed as in the past, and that the world's irrigated area will expand by 30% from 1995 to 2025. The projections by Alcamo and others (1999) assumed a limited expansion of irrigated area, which, combined with rapidly increasing water use efficiency, leads to reduced agricultural use but a rapid increase in municipal and industrial use linked to rising income and population.

TABLE 1.1. TWO DIVERGING PROJECTIONS FOR USE OF RENEWABLE WATER RESOURCES FOR BUSINESS AS USUAL [1]

			Expanding Irrigation	Stable Irrigation
Year	1950	1995	2025[1]	2025[2]
Use				
Agriculture: Withdrawal:	1100	2500	3200	2300
Consumption	700	1750	2250	1700
Industry: Withdrawal:	200	750	1200	900
Consumption	20	80	170	120
Municipalities: Withdrawal:	90	350	600	900
Consumption	15	50	75	100
Reservoirs (evaporation)	10	200	270	200*
Total: Withdrawal:	1400	3800	5200	4300
Consumption	750	2100	2800	2100

Note: All numbers are rounded in cubic kilometres.

* *Not calculated, but since the business as usual scenario developed by the World Water Vision assumes that relatively few additional reservoirs will be built, the value for 1995 is used to obtain comparable total use figures.*

Of the total average global amount of renewable water resources of about 42 000 km^3, only 10% is withdrawn and 5% is consumed. The problem lies in the uneven distribution of water resources both geographically and seasonally. Fresh water is an essential element for human existence. It is also vital for sustained industrial, and agricultural development. Fresh water demand for the various uses ranges in amount and quality. In terms of a typical indicator of water quality, the concentration of total dissolved solids (TDS) in potable water should be in the range of 500–1000 ppm. Typical water quality standards are given in Table 1.2. Water for irrigation can range from about 500 to 2000 ppm, water for livestock can contain up to about 3000 ppm of TDS: [2], while water quality for industrial uses varies depending upon the type of industry. In many cases, high water quality is required with very low TDS. TDS for the make-up water for the steam cycle at power plants, for example, is required to be as low as about 2 ppm.

[1] Shiklomanov projection.
[2] Alcamo projection.

TABLE 1.2. STANDARDS FOR DRINKING WATER

		WHO	EEC 1980			USA	JAPAN	GERMANY	CANADA
		1993	80/778/E WG			12/89			
Unit	Constituent	GV	RV	MALC	NMC	MCL	MALC	MALC	MACC
Important to aesthetic quality									
^{o}C	Temperature	NGV		25				25	15
-	pH	<8	6.5–8.5			6.5–8.5	6.8–8.6	6.5–9.5	6.5–8.5
	Taste and odour	NGV							
TCU	Colour	15	1	20			5		15
NTU	Turbidity	5	0.4	4			2	1.5	5
	Detergent	NGV		0.2					
mg/l	Oxygen dissolved	NGV			>75% sat				
μS/cm	Conductivity		400					2000	
mg/l	TDS	1000		1500		500	500		500
mg/l	Total hardness	NGV			60		300		
mg/l	Alkalinity				30				
mg/l	Calcium		100					400	
mg/l	Magnesium		30	50				50	
mg/l	Strontium								
mg/l	Sodium	200	20	(150)				150	
mg/l	Potassium		10	12				12	
mg/l	Ammonium	1.5	0.05	0.5				0.05	
mg/l	Phosphate		0.4	5				5	
mg/l	Iron	0.3	0.05	0.2		0.3	0.3	0.2	0.3
mg/l	Manganese	0.1	0.02	0.05		0.05	0.3	0.05	0.05
mg/l	Zinc	3	0.1			5	1	5	5
mg/l	Copper	1	0.1			1	1	3	1
mg/l	Aluminium	0.2	0.05	0.2				0.2	
mg/l	Chloride	250	25	(200)		250	200	250	250
mg/l	Sulphate	250	25	250		250		240	500
mg/l	Hydrogen carbonate								
Important to health									
mg/l	Antimony	0.005		0.01				0.01	
mg/l	Arsenic	0.01		0.05		0.05	0.05	0.01	0.05
mg/l	Barium	0.7	0.1			1		1	1
mg/l	Beryllium								

mg/l	Boron	0.3	1				1	5
mg/l	Cadmium	0.003		0.005	0.01	0.01	0.005	0.005
mg/l	Chromium	0.05		0.05	0.05	0.05	0.05	0.05
mg/l	Cyanide	0.07		0.05		NTD	0.05	0.2
mg/l	Fluoride	1.5		1.5	2	0.8	1.5	1.5
mg/l	Lead	0.01		0.05	0.05		0.04	0.05
mg/l	Manganese	0.5						
mg/l	Mercury	0.001		0.001	0.002	NTD	0.001	0.001
mg/l	Nickel	0.02		0.05			0.05	
mg/l	Nitrate	50	25	50	10	>10	50	10
mg/l	Nitrite	3		0.1		>	0.1	1
mg/l	Kjeldahl-Nitrogen			1			1	
mg/l	Oxidability $KMnO_4$		2	5		10	5	
mg/l	Selenium	0.01		0.01	0.01		0.01	0.01
mg/l	Silver	NGV		0.01	0.05		0.01	0.05
Bq/l	Gross α	0.1						
Bq/l	Gross β	1.0						

NMC = necessary minimum concentration
RV = recommended value
NGV = no guideline value set
MACC = maximum acceptable concentration

MCL = maximum containment level
GV = guideline value
MALC = maximum allowable concentration
NTD = not to be detect

If the steps to manage freshwater resources through the reduction of water use and the development of existing resources cannot meet freshwater demand, seawater desalination is the best way to meet increasing demands of freshwater especially along coastal areas been widely deployed in the past several decades in many arid and semi-arid zones. Seawater desalination can produce freshwater with necessary quality by choosing an appropriate desalination process and post-treatment methods of the product water.

According to the market survey performed by the World Resources Institute on the future growth of seawater desalination, the worldwide demand for desalination is expected to double approximately every 10 years in the foreseeable future. Most of the demand would arise in the Arabian Gulf and North African regions, but this is likely to expand to other areas.

The prospects of using nuclear energy for seawater desalination on a large scale remain very attractive since desalination is an energy intensive process that can utilize the heat from a nuclear reactor and/or the electricity produced by such plants. Many years of successful operation of a nuclear power plant in Kazakhstan have proved the technical feasibility, compliance with safety requirements and reliability of co-generation nuclear reactors. Also, a few small-scale nuclear desalination plants have been successfully operated in Japan. Large-scale commercial deployment of nuclear desalination[3] will mainly depend on its economic competitiveness with alternate energy supply options. For example, economic studies by the IAEA have shown that the nuclear desalination option can offer potable water at a cost that is competitive with fossil fuelled plants in the North African coast and other locations with similar conditions. It is expected that most future desalination plants will be built in three distinct sizes: small (capacity of less than 10 000 m^3/d), medium (50 000–100 000 m^3/d) and large (greater than 200 000 m^3/d). Owing to the relatively high cost of water transport, it is doubtful whether plants larger than 500 000 m^3/d would be economic, except under unique circumstances.

However, additional effort is required to take advantage of the nuclear option for future production of fresh water. More research effort is directed toward reduction of both nuclear and desalination costs. International and regional co-operation is employed to promote desalination R&D and to assist future owners of desalination plants with their technology selection, installation and management. Several nuclear desalination programmes are under way or being planned in several Member States. For the effective and successful progress of these nuclear desalination programmes, it is especially important to share the knowledge and experience gained among interested IAEA Member States. In order to facilitate these on-going demonstration programmes in Member States, relevant information has been collected and disseminated at various IAEA technical meetings and shared by many Member States, which are interested in nuclear desalination.

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

1.2. Objectives and Scope

The purpose of this publication is to provide an overview of various nuclear desalination plant design concepts, which are being proposed, evaluated, or constructed in the Member States with the aim of demonstrating the feasibility of using nuclear energy for desalination applications under specific conditions. Future potential applications of a variety of nuclear reactor designs in nuclear desalination are being proposed for examination. These include: high- temperature gas reactors (HTGRs), liquid metal cooled reactors (LMRs) such as lead-bismuth cooled or sodium cooled reactors, and other innovative reactor design concepts.

Chapter 2 gives an overview of operating experience of nuclear desalination and design approaches based on lessons learned and/or experienced, which are informative to further application of nuclear energy for seawater desalination. It also summarizes the status of major ongoing and up-to-date activities in Member States focusing on demonstration of nuclear desalination. The information has been updated in most activities since it was reported at various IAEA meetings.

Chapters 3 and 4 contain technical information of various nuclear desalination plant design concepts, which are currently evaluated in Member States. Chapter 3 gives an overview of the co-generation design concepts, while Chapter 4 includes an overview of dedicated heat design concepts available to the IAEA.

CHAPTER 2. OVERVIEW OF EXPERIENCE AND ONGOING ACTIVITIES

2.1. Experience

Integrated nuclear desalination plants have been operated in Kazakhstan and Japan for many years. Pakistan has also recently reported its operating experience. The overall operating experience exceeds 150 reactor years as of 2000 [3]. Relevant experience has also been accumulated in district heating and process heat production for industrial use. No nuclear related safety problems have been reported for any of these nuclear energy applications. Experience has shown that nuclear desalination is technically feasible.

TABLE 2.1. OPERATING EXPERIENCE OF NUCLEAR DESALINATION PLANTS

Plant name	Location	Gross Power (MW(e))	Water Capacity (m^3/day)	Energy/Desalination plant type
Shevchenko	Aktau, Kazakhstan	150	80 000	LMFBR/MSF&MED BN-350 closed in 1999
Ikata-1,2	Ehime, Japan	566	2000	PWR/MSF
Ikata-3	Ehime, Japan	890	2000	PWR/RO[4]
Ohi-1,2	Fukui, Japan	2 × 1175	3900	PWR/MSF
Ohi-3,4	Fukui, Japan	2 × 1180	2600	PWR/RO
Genkai-4	Fukuoka, Japan	1180	1000	PWR/RO
Genkai-3,4	Fukuoka, Japan	2 × 1180	1000	PWR/MED
Takahama-3,4	Fukui, Japan	2 × 870	1000	PWR/RO
KANUPP	Karachi, Pakistan	137	454	PHWR/RO

This section provides an overview of nuclear desalination plants operating experience, and lessons learned from IAEA support activities, and major ongoing national programmes on nuclear desalination in Member States. Some additional relevant information e.g. data, flow diagram, etc. are given in the Appendix.

2.1.1. Nuclear desalination system at Aktau, Kazakhstan

A sodium-cooled fast reactor BN-350 provided electricity and nuclear heat to the Aktau power and desalination plant complex (located in an arid zone in the Mangyshlak peninsula, on the East Coast of the Caspian Sea, since 1973 until its shut down in 1999). The complex was constructed at a location situated 12 km from the city, next to several developed industrial enterprises. The complex consisted of the BN-350 and a fossil power plant, which together provided steam to a condensing turbine and three backpressure turbines. The exhaust steam (0.6 MPa) from the backpressure turbines was used as the heat source for the first stage evaporator of an MED desalination plant (See *Flow diagram* in the Appendix). If more steam from the power plant was available than that which was required for desalination, it was used to supply heat energy to the nearby industrial enterprises and settlements.

During the service period of the BN-350 reactor, there were no reports of sodium leakages in the primary and secondary loops, or abnormal operation of the sodium pumps and cavitations in the driving wheels of the pumps were insignificant. Failures in the desalination equipment included corrosion of the pipes, shell parts and heat exchange tubes of the evaporators and preheaters, and erosion and corrosion of the circulating pump blades.

[4] Brackish water desalted.

However, these did not influence the operational reliability of the complex as a whole. This was guaranteed by the selective shutdown of one of the evaporator units, which did not cause a significant decline in the amount of product water [4].

The nuclear reactor was shut down twice a year for a period of twenty days to enable refuelling and scheduled maintenance. During these periods heat for the desalination plant was supplied by the thermal power station. The overall desalination system availability over the whole service period (1973–1997) was reported at about 85%.

There were two distillate lines from the desalination plant: drinking quality water (TDS up to 200 mg/l) and high purity water (TDS = 2 to 10 mg/l) for boiler feed water and other industrial uses (see diagram in the Appendix). The product water quality was independent of the utilized heat source (i.e. nuclear or fossil).

Both reactor operating experience and the analysis of design specifications and potential accidents have shown that the radiological consequences of all abnormal operating conditions did not, and could not have any effect on the desalinated water quality.

2.1.2. Nuclear desalination plants in Japan

There are 53 operational nuclear power plants in Japan as of December 2000, which are operated by 10 different electric power companies. These are all located in coastal areas in order to use seawater as an ultimate heat sink. Some of these plants are equipped with seawater desalination plants in order to provide high quality make-up water for the boiler feed water as well as for other uses after an appropriate water post treatment.

The first Japanese nuclear power and seawater desalination plant was commissioned in 1978 at the Ohi Nuclear Power Station. The plant consists of an 1175 MW(e) PWR coupled to an MSF distillation plant with a capacity of 1300 m^3/d. As of 2000, nine additional nuclear seawater desalination plants were installed. Eight of these plants are currently in operation. The desalination plant capacities are in the range of 1000 to 3900 m^3/d. The average salinity of seawater in-take is 35 000 mg/l and the average feed temperature is 17ºC. Selected highlights from the operating experience of these plants include:

- Successful operation with no evidence of any anomalies to date.
- No occurrence of leakage of radioactive substances into the product water.

The desalination plants have become vital and effective facilities for supplying high quality make-up and potable water for the nuclear power stations. Despite low capacities of the desalination plants, operating data obtained to date is fully applicable to the expected operations of a larger scale nuclear desalination plant. The data highly supports the use of nuclear power for seawater desalination worldwide. The seawater desalination plant designs for nuclear plant coupling- are identical to those of fossil plant coupling, with exception of desalination plants using RO technology, which in the case of a nuclear desalination plant, the plastic casings of RO membranes are covered with carbon steel.

2.1.3. Nuclear desalination system in Pakistan

A 2 × 227 m^3/day (2 × 50 000 gpd) reverse osmosis seawater desalination plant has been set up at KANUPP to meet the normal operational requirements of the plant in addition to providing an independent source of emergency feed water to the steam generator. Technical specifications for the plant are summarised in Table 2.2 The RO plant shares many facilities with the main plant including product water storage facility, water distribution system, as well as manpower for maintenance operation and chemical control of the plant. The plant has recorded satisfactory performance since it became operational. The raw water supply was initially planned to be from two deep wells, however additional two wells are being dug due to the rocky terrain at the KANUPP site. The salinity of the raw feed water is in the range of 21 000–22 000 ppm of TDS. The quality of treated water from this plant meets WHO standards for potable water-(<1000 ppm TDS, see Table 2.2).

TABLE 2.2. TECHNICAL SPECIFICATIONS FOR 2 × 227 m^3/DAY (2 × 50 000 IGPD) RO DESALINATION PLANT FOR SEA WATER AT KANUPP

No. & Capacity of RO Modules	2 × 227 m^3/day (2 × 50 000 Igpd) Skid Mounted
Supplier	AQAUA Clean Technology, Australia
Permeators	Polyamide type, Model TFC-2832 Magnum
Size & No. Of Membranes/P. Vessel	8” diameter × 60” length, 4 Membranes
Membranes per module	16
Make	Fluid Systems (USA)
Filter Feed Pumps	2 × 100%, each 22 m^3/h capacity
Double Stage Filters	2 × 100%
Cartridge Filters	2 × 100%, 5 micron
High Pressure Pumps	4 × 100%, each of 22 m^3/h capacity at 37 & 65 bars (2 active & 2 standby)
Turbo Recovery System	2 × 100%
TDS Product Water at 35 °C	400 ppm
pH value Range	6.5–9.00
Overall recovery ratio	45 percent
Power Consumption with ERT	About 5 kWh/m^3
Measured Seawater Temperature	68–86°F (20–30°C)

2.2. Design approaches

For wider deployment of nuclear desalination, additional requirements have to be met under specific conditions [5]. Technical issues include, meeting more stringent safety requirements specifically for nuclear-desalination integrated plants and improvement on performance of the integrated systems. Another important factor for consideration in wider deployment of nuclear desalination is economic competitiveness compared with other options (e.g. fossil fuel powered co-generation plants).

This section summarizes some key approaches recommended in nuclear desalination complex design. Successful approaches based on operating experience of nuclear desalination plants coupled with practices and experiences at nuclear power plants, as well as conventional fossil-fuelled desalination facilities. (See an IAEA publication on guidance for developing countries [6].

2.2.1. Safety

The safety of a nuclear desalination plant depends mainly on the safety of the nuclear reactor and the interface between the nuclear plant and the desalination system. It must be ensured that any load variation of steam consumption in the desalination plant would cause no anomalies in the nuclear plant. There should be suitable provision for monitoring the radioactivity level in the isolation loop and desalination system. In the case of a PHWR, the tritium level in the heating steam and product water must be regularly checked. Adequate safety measures must be introduced to ensure no detectable radioactivity release to the product water. The risk should also be assessed for accidental radioactivity carry over. An agreement of all relevant parties on safety and quality standards and clear regulations are very important for nuclear desalination applications [7].

The basic requirement for preventing radioactive contamination of the desalination plant and/or the atmosphere is of utmost importance in thermal coupling. At least two mechanical barriers and pressure reversal between the reactor primary coolant and brine must be incorporated. In the case of a pressurised water reactor, the steam generator is the first barrier. The second barrier could be the condenser of a backpressure turbine. In the case of heat generation reactors, careful attention must be given to providing sufficient barriers to prevent radioactive contamination. The most suitable heat generation reactors for desalination coupling are those with a closed primary cooling circuit such as a low temperature PWR or PHWR. In this arrangement, heat is supplied through an interface with a steam generator or primary heat exchanger. This provides a barrier between the reactor coolant and the steam (or hot water) for desalination. Direct supply of steam from the reactor core to the desalination plant is not suitable for desalination without an intermediate barrier.

2.2.2. Design life

The design life of the coupling system should be as long as the design life of the individual processes, i.e. the nuclear and desalination systems. The main components of the nuclear plant are designed for more than 40 years of operation. On the other hand desalination plants are usually designed for an economic life time of around 25 years, although some MSF desalination plants have been operating for as long as 30 years. MED vessels and piping are similar to those of MSF. MED systems with aluminium tubes must be usually retubed at about 15 years interval. Titanium, cupro-nickel and stainless steel alloy tubes may last for 25–30 years. In an RO plant, membranes and filters are components with shorter lives and must be replaced at relatively short intervals. It is therefore essential that, the design and layout of the nuclear desalination plant should accommodate the possibility of replacement or expansion of the desalination equipment with minimum interruption of electricity generation and water production.

2.2.3. Operational flexibility

The water to power ratio in a co-generating station changes with daily and seasonal variation. As electricity cannot be stored, the steam flow rate in the turbine may be adjusted to meet power demand. A certain degree of flexibility is required in the plant to match local conditions. The design of a co-generation plant must provide a minimum degree of flexibility to avoid the breakdown of production units when either the turbine generator is shut down or the desalination plant production is reduced or altogether stopped. This can be accomplished by using a backup condenser and/or a backup heat source.

The coupling of an RO or MVC desalination plant to a power plant has a higher degree of flexibility because the main interface is the power coupling, on the other hand the coupling of an MSF or MED plant to a power plant introduces close interaction between the operations of both plants. The use of electricity from the grid for RO or MVC plants can allow them to operate as stand alone plants, thereby improving their operational flexibility in case of outage of the heat source from the nuclear plant.

The transient behaviour, including operational flexibility to meet varying power and water demands and methods of maintaining steam supply to MSF and MED plants at low electrical load, must be analysed in detail for nuclear co-generation plants. The transients are normally caused by load variations. The extent of variations differs from location to location. Power demand variation is a critical factor in countries where the summer power demand is largely due to air-conditioning and the load in winter may be only 30% or less of that during summer.

The potential difficulty in the operation of a co-generation plant with thermal desalination is the dependence of steam flow on electricity demand. A thermal desalination plant does not respond very well to sudden load changes. There is usually no difficulty in ensuring the stable operation of a thermal desalination plant between 70–110% of full rated capacity with slow changes in load. A sudden large reduction in steam flow rate to the desalination plant may lead to difficulty in operation because the brine flow may decrease below the permissible limit. This difficulty may be partially overcome by adopting a system in which the desalination evaporators are connected in parallel and taking some of the units out of service to ensure stable operation. This takes care of transient behaviour due to daily and seasonal load variation.

2.2.4. Reliability/availability

The reliability requirements of the desalination systems must be taken into account while designing a steam supply system using nuclear energy. These requirements must be addressed in the design phase of the integrated nuclear desalination system. These requirements may be met by different measures.

A nuclear co-generation desalination plant consists of three interacting systems: nuclear steam supply system (NSSS), the turbine generator system and the desalination unit. If the coupling has a high availability factor, the thermal desalination system coupled to the turbine generator may give a high overall availability factor. A thermal desalination plant requires a back up heat source for its operation during the reactor shut down. If the desalination plant is shut down, provision should be made for power production increase or a back up condenser for discharge of the steam from the power plant.

A hybrid thermal desalination and RO plant coupled to the NSSS appears to have a high availability factor. During the shut down of the reactor, the thermal desalination plant must also be closed due to non-availability of steam. The RO plant, however, can continue to operate using power from the grid

2.2.5. Design limitations

A nuclear power reactor as such can accommodate almost any desalination plant size. The seawater intake and outfall system and the environmental limitations with respect to temperature and salinity of seawater discharge influence the coupling of a desalination plant with a nuclear power reactor. The temperature and pressure of steam or hot water produced in a heat only reactor also have an effect on the type of desalination system to be used and its specifications including the coupling arrangement.

2.2.6. Economics

Economic competitiveness of the nuclear desalination option is one of the most important factors in deciding whether to employ such an option. A comprehensive economic investigation using the IAEA's Desalination Economic Evaluation Program (DEEP) generally shows that nuclear seawater desalination yields water costs in the same range as fossil fuelled options hence both can provide viable options depending upon site specific conditions [8]. The target cost of the product water from nuclear desalination plants is not easily derived, since it depends on energy cost (thermal and/or electricity) and many other local conditions. Nevertheless reasonable product water will be in the range of 0.7–0.9 US$/$m^3$ as recently experienced in the Arabian Peninsula [9].

2.3. IAEA support programmes

The use of nuclear energy to produce potable water by seawater desalination has been considered as far back as in the 1960s. There was great optimism at that time regarding the use of nuclear energy for seawater desalination as well as other forms of heat applications such as district heating. Individual countries, organizations and nuclear industries under took several studies.

Since the renewal of the IAEA's activities concerning nuclear desalination following the 1989 General Conference, the IAEA took steps to update its review of available information on desalination technologies and the coupling of nuclear reactors with desalination plants. [10] With the participation of a growing number of countries and international organizations, the IAEA has also assessed the economic viability of seawater desalination by comparing the use of nuclear energy with fossil fuels [11]. The study encompassed a broad range of both nuclear and fossil plant sizes and technologies, in combination with various desalination processes.

In 1997, the IAEA organized an international Symposium on Desalination of Seawater with Nuclear Energy in order to provide a forum for reviewing the latest technological experiences, designs and developments, and future prospects of nuclear seawater desalination. An overview of activities on desalination was given by participants from selected organizations- This include: a review of experiences from existing nuclear desalination plants and relevant conventional desalination facilities; national and bilateral activities including research, design and development aspects of nuclear seawater desalination, as well as forecasts and challenges lying ahead. [12].

For the purpose of economic evaluation and analysis of various desalination and energy source options, IAEA has also developed a Desalination Economic Evaluation Program (DEEP) [16] which is based on — spreadsheet methodology. The spreadsheet serves three important goals:

(a) side-by-side comparison of a large number of design alternatives;
(b) quick identification of the lowest cost options at a given location; and
(c) an approximate cost of desalinated water and power.

The DEEP software package has been disseminated to many Member States and organizations for specific applications. This is to enable the distribution of costs to the two products in a co-generation plant (i.e. power and water) DEEP uses the "power credit method", i.e. the loss of electricity generation is charged to the water costs.

The energy and water problem is not easily solved by individual countries, but requires regional or even global approaches. Developing countries are, on the one hand, suffering mostly from energy and/or water shortages and on the other hand face difficulties solving these problems on their own. Therefore there exist a need for technical partners for international information exchange and co-operation. The next step for proceeding with a nuclear desalination demonstration programme would be for one or more Member States to initiate the project-related preparatory steps towards the demonstration of an international collaborative framework. Several countries are performing technology development programmes on nuclear desalination and/or conducting feasibility studies on the possible introduction of nuclear desalination. Many activities at the IAEA are focusing on supporting such programmes in Member States some of which are described below.

2.3.1. Co-ordinated research project (CRP*)*

A Co-ordinated Research Project (CRP)[5] on "Optimization of the Coupling of Nuclear Reactors and Desalination Systems" was initiated in 1998 with participating institutes from nine countries[6] in order to share the relevant information, optimize the resources, and integrate related R&D efforts. The CRP covers a review of reactor designs suitable for coupling with desalination systems, the optimization of this coupling, possible performance improvements and advanced technologies of desalination systems for nuclear desalination.

Three Research Co-ordination Meetings (RCM), were held in November 1998, February 2000 and October 2001. Highlights of the work over this period include:

- Identification of optimum steam extraction conditions from NSSS to the desalination system and plant safety analyses;
- Development of an analytical model of thermally coupled configurations, which simulates protective measures for ensuring no radioactive materials carry-over into the product water;
- Planning of experimental verification of performance improvements of the Reverse Osmosis system with preheated feedwater using nuclear heat;
- Performance prediction and system design of the MSF-RO hybrid system to be coupled to existing nuclear power units.

The CRP is scheduled to be completed in 2003 and the results will be documented and shared by interested Member States for facilitating improved design of nuclear desalination plants.

2.3.2. Development of an economic evaluation tool and its application

As was previously mentioned the IAEA has developed a computer program, DEEP (Desalination Economic Evaluation Program), based on spreadsheet methodology. The spreadsheet includes simplified models of several types of nuclear/fossil power plants, and nuclear/fossil heat sources, as well as thermal and membrane desalination processes. Current cost and performance data has also been incorporated so that the spreadsheet can be quickly adapted to analyse a large variety of options with very little required new input data.

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[5] A new CRP on "Economic Research on, and Assessment of, Selected Nuclear Desalination Projects and Case Studies" has been initiated, with the participation of thirteen research institutes. Three CRPs will facilitate coordination of current and planned national studies on seawater desalination in Member States.

[6] Argentina, Canada, China, Egypt, India, Indonesia, Rep. of Korea, Russian Federation and Tunisia. Canada completed its planned mission in 2001.Two other institutes from Morocco and Libya joined the CRP in 2001.

DEEP was applied in a study for three broad regions with different seawater conditions and two economic scenarios [8]. For an economic assessment of nuclear seawater desalination versus fossil options, calculations were performed for a simplified cost assessment of nuclear seawater desalination in comparison with fossil options using DEEP Calculations of the base power cost (i.e. overnight cost) of the power unit, the distillation system water cost and the RO system water cost were performed by summing the annual capital fuel (or energy) and O&M costs and dividing the total sum by the annual product output. The electricity cost is calculated as "lifetime levelized electricity cost" (i.e. dividing the discounted sum of all expenditures over the whole plant life associated with the generation of water by the discounted value of the product). Calculations were carried out for three representative regions: Southern Europe (South of France, Italy, Greece, Turkey and Spain), South East Asia, the Red Sea and the North African region, and the Arabian Gulf region (based on different average seawater salinity and temperature). The results generally show that nuclear seawater desalination yields water costs in the same range as fossil options hence both can be seen to be competitive with each other.

2.3.3. Technical assistance through Technical Co-operation programme (TCP)

The IAEA has over the past three decades assisted developing Member States in capacity building, and the establishment of infrastructures in the area of nuclear science and technology. The Technical Co-operation Programme is an established mechanism to provide technical assistance to such Member States for their specific needs. Thus the TCP could provide assistance to member states, which are considering a demonstration programme of nuclear desalination. In this respect, the TCP launched in 1999 an interregional project on "Integrated Nuclear Power and Desalination System Design (INT/4/134). The project has been designed to provide a forum for technology suppliers and prospective end users to jointly develop of integrated nuclear desalination concepts with the, aim of demonstrating the viability of nuclear desalination for specific potential sites. In the first phase of this interregional project (1999/2000), participants were briefed by the IAEA on the following related issues:

- State-of-the-art technologies (nuclear and seawater desalination);
- Status of planned and on-going nuclear desalination demonstration activities in some Member States and their readiness to welcome international participation in the projects;
- Availability of technologies for transfer;
- Willingness of technology providers to assist others; and
- Possible mechanisms for international co-operation-and collaboration.

The IAEA is continuing its efforts to evaluate the needs of "technology seekers" and opportunities for "technology providers" Based on the recommendations of INDAG, a questionnaire was sent in June 1999 to interested Member States requesting them to specify their needs and requirements for participating and benefiting from the project. The IAEA then consulted with Member States to identify the needs of individual "technology seekers" having concrete government endorsements. In the process the IAEA received specific TC requests from Tunisia, Indonesia, Pakistan and Iran by the end of the year 2000.[7]

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[7] The National Nuclear Energy Agency (BATAN) of Indonesia and the Korea Atomic Energy Research Institute

2.4. Relevant Activities in Member States

The IAEA's Options Identification Programme for Demonstration of Nuclear Desalination [39] completed in 1996-and the international symposium on "Nuclear Desalination of Seawater [12]" in 1997 gave momentum to many Member States in taking a step forward in evaluating, planning and/or initiating nuclear desalination projects, as discussed in the following subsections.

2.4.1. Argentina

A small integrated PWR, CAREM 25, has been under development since the early 1990s in order to employ nuclear energy in power output ranges much lower than those currently deployed, but large and at the same competitive enough as energy source for seawater desalination. The CAREM 25 is proposed for nuclear desalination demonstration, and the expertise accumulated in the project, and related R&D will be open to international collaboration under the IAEA's INT/4/134. The construction site of the CAREM 25 reactor was identified in 2000 and construction is expected to commerce soon.

In view of the economic competitiveness of nuclear power for production of both electricity and water, the competitiveness of CAREM 25 was analysed using an economic evaluation and optimization program based on integral PWR designs. The CAREM 25 design was determined to meets the economic criteria for a nuclear power source for desalination in the given power range.

The CAREM project is being developed jointly by the Comisión Nacional de Energía Atómica (CNEA) and INVAP S.E. Various experimental facilities were used to verify its design, which included a test rig for verifying the dynamic response of the primary circuit (including pressure control through feedback on power); a prototype control rod drive mechanisms; and the RA-8 experimental facility to measure the neutronic parameters of the CAREM core. Also as part of the research programmes for the project, INVAP S.E. is developing an analytical model of coupling various configurations of nuclear desalination plants. The model is to contribute to the safety design of the integrated system and to ensure that there is no carryover of radioactive materials to the product water. The research programme is one element of the IAEA's CRP on "Optimization of the Coupling of Nuclear Reactors and Desalination Systems" and exchange of technical information with other participants of the CRP.

2.4.2. Canada

Activities in Canada on nuclear desalination comprise nuclear reactor development by AECL and desalination technology development by CANDESAL Technologies. From its successful experience of more than 50 years with CANDU, AECL supplies complete nuclear

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(KAERI) agreed on cooperative arrangement in October 2001 for a two-year preliminary economic feasibility assessment of nuclear desalination.

Another collaborative work has been initiated between the French Commissariat a l'Energie Atomique (CEA) on one hand and the National Nuclear Research Centre (CNSTN), The National Water Distribution Company (SONEDE) and the Tunisian Electricity and Gas Company (STEG) representing Tunisia. The Pakistan Atomic Energy Commission (PAEC) has solicited in 2001 technical assistance through the Agency for installing a nuclear desalination facility at its Karachi Nuclear Power Plant site.

generating stations in all aspects of nuclear technology including its application to non-electrical products. CANDESAL has been developing a unique approach to the application of reverse osmosis technology and design methodology to improve energy efficiency and effectively reduce the life cycle production cost of potable water. Work is currently underway to utilize the moderator cooling system, a feature unique to the Pressurized Heavy Water Reactor, as a source of additional waste heat to further improve the effectiveness and efficiency of the coupling between the reactor and the desalination system.

An experimental programme has been initiated to investigate and demonstrate the validity of the CANDESAL design methodology. The programme is designed to investigate the effects of high temperature and high-pressure operation of spiral wound membranes using UF treated seawater in a specialized test rig in parallel with the development of system design optimization algorithms. With these data the programme will design and build a 350 m^3/day demonstration facility that will incorporate preheated feed water, high feed pressures, ultra-filtration pre-treatment, energy recovery and site-specific optimization.

CANDESAL is also collaborating with the Russian Federation on the application of its technologies to a floating nuclear desalination system, consisting of a barge-mounted Russian reactor and a barge-mounted Canadian desalination unit.

2.4.3. China

Based on research work on the possible application of nuclear energy for low temperature heating initiated in the early 80s, a 5 MW(th) experimental Nuclear Heating Reactor (NHR-5) came into operation for space heating in 1989. A large-scale NHR with an output of 200 MW(th) (NHR-200) has been developed since 1990. The NHR can be used in district heating, seawater desalination, air conditioning and other industrial processes.

An NHR-200 demonstration plant was once planned in the city of Daqing on the northeast coast in 1995, but it was changed to the new site of Shenyang City for institutional reasons. The study on the new site is on going. A smaller NHR-10 with an output of 10 MW(th) has been evaluated as a prospective heat source for a demonstration plant in Morocco in order to produce 8000 m^3/d of potable water using an MED process (see Section 2.4.9).

2.4.4. Egypt

Egypt has assessed the introduction of nuclear power and has approved the El-Dabaa site as the location for the first plant. Egypt like many other countries is also experiencing serious fresh water shortages and has participated in an earlier regional project for evaluating the feasibility of nuclear seawater desalination (RAF/4/014) the country is now studying the feasibility of a nuclear desalination plant under specific site conditions at El-Dabaa.

The Feasibility Study of nuclear power generation and desalination at the El Dabaa site is underway by the NPPA with the IAEA's assistance under a TCP (EGY/4/040).[8] The test facility at El Dabaa will also be used to study the effect of pre-heating feed water, which can have a favourable effect on plant productivity.

[8] The study was completed at the end of 2001.

2.4.5. France and European Union

France has initiated nuclear desalination feasibility and economic studies as part of CEA's own innovation programme and as part of a proposed joint European Study (The EURODESAL PROJECT). Shortages of water supply for drinking and irrigation purposes will be a major problem in the years ahead, specifically in the southern regions of the European Union, namely, south of France, Italy, Greece, Portugal and Spain. It is for this reason that the partners of the Michael Angelo Initiative Concerted Action (MICA) are proposing the EURODESAL project. The combined efforts of the 14 industrial and R&D organisations, which constitute MICA, represent a very considerable pool of knowledge and experience in advanced technologies, which could provide sustainable and economic solutions to the water and energy needs of the region.

The basic goals of the project are to propose:

– Appropriate nuclear reactor systems (incorporating improved safety features and proliferation resistant technologies), as well as clean combustion and renewable energy systems for desalination;
– Assessment and evaluation criteria of such systems (with other energy sources) based on sustainability, sound economics and safety.
– Efficient, innovative and economic coupling schemes using innovative membrane and\ or distillation processes.

2.4.6. India

Based on the earlier experience in desalination pilot plants (MSF and RO), the Bhabha Atomic Research Centre (BARC) has undertaken the establishment of a demonstration scale hybrid desalination plant to be coupled to two PHWR units (170 MW(e) each) at the Madras Atomic Power Station, Kalpakkam, in south-eastern India. The desalination plant consists of a 4500 m^3/d MSF plant and an 1800 m^3/d RO plant. The product water will be provided to the nuclear power station and the local inhabitants for drinking. The nuclear desalination demonstration plant (NDDP) was licensed in 1999.

The tenders for the major equipment of this plant are released and are under various stages of procurement or fabrication. Tenders for seawater intake/outfall and steam supply are in preparation. The civil and electrical work was started in 1999 and completion was scheduled for 2001. Useful design data is expected from this plant on the coupling of SMR based on a PHWR with a hybrid desalination plant. India will share the O&M experience of NDDP to Member States when the plant is commissioned.[9]

2.4.7. Indonesia

A preliminary economic study is in progress to consider a nuclear desalination plant as an alternative to fossil-fuelled desalination plants for the Madura Island. The purpose of the plant is to provide the Madura Island with sufficient power so that it would be less dependent on the Java-Bali–Madura interconnected grid, as well as on outside supply of potable water for public uses. The plant will also support the expansion of the tourism industry and general industrial development. Further additional investigation is being done on the utilization of the waste brine produced by the desalination plant as a concentrated seawater feed for traditional salt production.

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[9] Commissioning of RO section due in 2002. Commissioning of MSF section is due 2003.

The Ministry of Research and Technology has decided to support the initiation of the "integrated featured research" which started in 2001. The study will be performed by BATAN as the lead organization. It will focus on two goals: a short or medium-term solution and a long-term solution for the water and energy problems at hand. Indonesia is collaborating with the Republic of Korea for the study under the umbrella of IAEA's TC project INT/4/134. This activity represents a part (i.e. nuclear alternative) of the integrated feature research, which is aimed at assessing the economics of the electricity generation and desalination plant operations before introducing nuclear desalination to Madura Island.

In 1997 Canada (AECL and CANDESAL) carried out a joint study with Indonesia to evaluate the site-specific technical performance and economic characteristics of a CANDU 6 reactor coupled with a reverse osmosis desalination system based on the CANDESAL concept using RO feed water preheat and system optimisation. The Muria site was used for the study. The study showed that even under the severe seawater conditions at the Muria site (high temperature and salinity), the CANDU-CANDESAL system design could economically meet all water production and quality requirements. The cost of potable water produced by the system was found to be about 15% lower than that from a system designed without taking advantage of RO preheat and design optimisation.

The present policy of the Indonesian government gives low priority to the introduction of nuclear power in Indonesia. The government has accepted, however, the IAEA's proposal of performing a national study called "Comparative (comprehensive) assessment of different energy resources in Indonesia" (TC-INS/2001/008). The purpose of this two-year study, which started in 2001, is to "support planning and decision making process in the energy and electricity sector in Indonesia.

2.4.8. Republic of Korea

Well-established nuclear energy technology can be readily extended to several applications. In this regard, Korea initiated, specific programmes on nuclear desalination in 1997, which will result in the start of construction of mainly small and advanced nuclear reactors, coupled to a desalination plant. The scope of the programme includes:

– Reactor and fuel design, and associated technology development;
– Design verification)e.g. experiments, tests);
– Power plant design;
– Component design-and manufacturing technology development; and
– Desalination system design.

The Korea Atomic Energy Research Institute (KAERI) as the lead organization with governmental support and participation is carrying out the work by various industries. Following the conceptual design, which was completed in 1999, concrete design of the integrated nuclear desalination system has commenced and is. Expected to be completed by March 2002. Plans for the next phase of the programme for the verification of system and technology are currently under preparation and will be finalized in the first half of 2001.

The central part of the integrated system is a 330 MW(th) SMART (System-integrated Modular Advanced Reactor) for dual-purpose application. The integrated nuclear desalination plant with SMART aims to produce both electricity and water. The capacity of the desalination system under design is 40 000 m^3/day using the MED process, but it can be adjusted for various demands. From the very beginning, the programme has been open to involvement or co-operation from any interested countries and/or overseas organizations. For this purpose, the SMART project has been integrated in the IAEA's interregional technical co-operation project on "Integrated nuclear and desalination system design" which started in 1999. KAERI is also participating in the IAEA's CRP on "Optimisation of the coupling of nuclear reactors and desalination systems" based on activity in the SMART project.

2.4.9. Morocco

Morocco was one of the participating Member States of the IAEA's feasibility study on the use of nuclear energy for seawater desalination in the North African Region. [13]. In that study, Morocco identified possible sites for nuclear desalination plants for further study in Morocco. In 1997 and 1998, Morocco participated in a technical co-operation project -with China under the umbrella of the IAEA to carry out a pre-feasibility study on a nuclear desalination demonstration plant with a 10 MW(th) Chinese Nuclear Heating Reactor (NHR-10) to be built in Tan-Tan, Morocco. The plant was designed to supply heat for the production capacity of 8000 m^3/d of potable water using a coupled MED plant. The production capacity of the demonstration plant was chosen so that it will augment the current water supply in Tan-Tan and provide sufficient water for its growing population which is expected to reach 70 000 inhabitants by the year 2010. The project will also produce a database for reliable extrapolation of water production costs for a future commercial nuclear desalination plant that would produce 160 000 m^3/d of potable water using a 200 MW(th) NHR.

2.4.10. Pakistan

Pakistan has been interested in nuclear technology and its application to seawater desalination in the coastal areas near Karachi. Pre-feasibility and design studies were carried out for desalination projects involving solar, nuclear and diesel power as energy sources. Since the arid zones along the coast are sparsely populated, (with the exception of Karachi), the water and power requirements are essentially small, which can be adequately met by means of conventional power plants or even an off-shore floating barge-mounted plant.

Recent work by the Pakistan Atomic Energy Commission (PAEC) on nuclear desalination has been undertaken to evaluate several options: (1) connecting a 2 × 227 m^3/day RO seawater desalination plant to the existing 137 MW(e) nuclear power plant at Paradise Point, the Karachi Nuclear Power Plant (KANUPP); (2) planning a 4550 m^3/day demonstration MSF plant coupled to KANUPP; and (3) design study of a large dual-purpose nuclear desalination plant using a 300 MW(e) or 600 MW(e) NPP.[10]

2.4.11. Russian Federation

The Russian Federation has a long history of developing and utilizing nuclear icebreaker transport fleets for its northern regions. Using its experience, the Russian Federation has been developing the concept of a Floating Nuclear Power Unit (FNPU) and its application for

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[10] The RO plant in Option (1) is operable since early 2000. Option (2) is being examined under the umbrella of the IAEA's Interregional Technical Co-operation programme (INT/4/134).

desalination. The following advantages of floating nuclear desalination complexes are envisaged:

– high quality of the entire floating power unit fabrication under shipbuilding work conditions followed by delivery to the customer possibly on a turn-key basis;
– short construction period of the station (4–5 years) and reduced investments as compared with land-based NPPs;
– possibility of siting at any coastal region;
– simplification of anti-seismic design features;
– cost reduction by serially-produced reactor plants; and
– simplified decommissioning of the station.

The design activities for a floating co-generation plant based on FNPU with KLT-40C reactors started in the mid-1990s at OKB Mechanical Engineering, with participation of other relevant organizations. The final design and licensing activities are in progress and construction permit was expected in 2001. Application of NFPU as an energy source for seawater desalination is also under consideration. Conceptual design of the coupling of a NFPU with MED facilities was prepared and further development is currently being planned. Coupling of a NFPU with a reverse osmosis process is also being investigated through a co-operation project on development of nuclear floating desalination plant using a Russian NFPU and a Canadian barge mounted RO desalination facility. The Russian atomic authority, Minatom, has solidified their commitment to the joint development project with the Canadian CANDESAL Company. Six possible sites have been identified in the country, so that seven NFPUs can be most effectively deployed with maximum efficiency (See Figure 2.1.). Various coupling schemes for several other Russian small reactors (RUTA, NIKA) are also being investigated in the framework of the IAEA's CRP on "Optimization of the coupling of nuclear reactor and desalination system".

2.4.12. South Africa

The South African Power Utility, Eskom, in conjunction with local and overseas partners, is leading the development of a High Temperature Gas Cooled Reactor (HTGR) of ±265 MWt capacity designed primarily for electric power production (100 to 110 MW(e)), but from which the waste heat could readily be utilized for either reverse osmosis feed water preheating as the source of thermal energy for vacuum evaporative desalination. This reactor is known as the Pebble Bed Modular Reactor (PBMR).

Development work on the PBMR Programme has now been underway in South Africa for several years. Initially plans included the construction of only Demonstration Module, with five or ten-module plants anticipated to be constructed subsequently in either South Africa or elsewhere. At the present time, the South African work is at the detailed feasibility study phase, wherein all designs are being finalized to the extent that capital cost of the plant can be estimated to an accuracy of ± 5%. In parallel with this work, the statutory Environmental Impact Study of the project is also in progress (with the tentative site adjacent to that of Eskom's existing Koeberg Nuclear Power Station on the Atlantic Ocean a few kilometres north-west of Cape Town). In addition the nuclear licensing process via the South African National Nuclear Regulator is currently underway, in order to obtain their necessary Construct License.

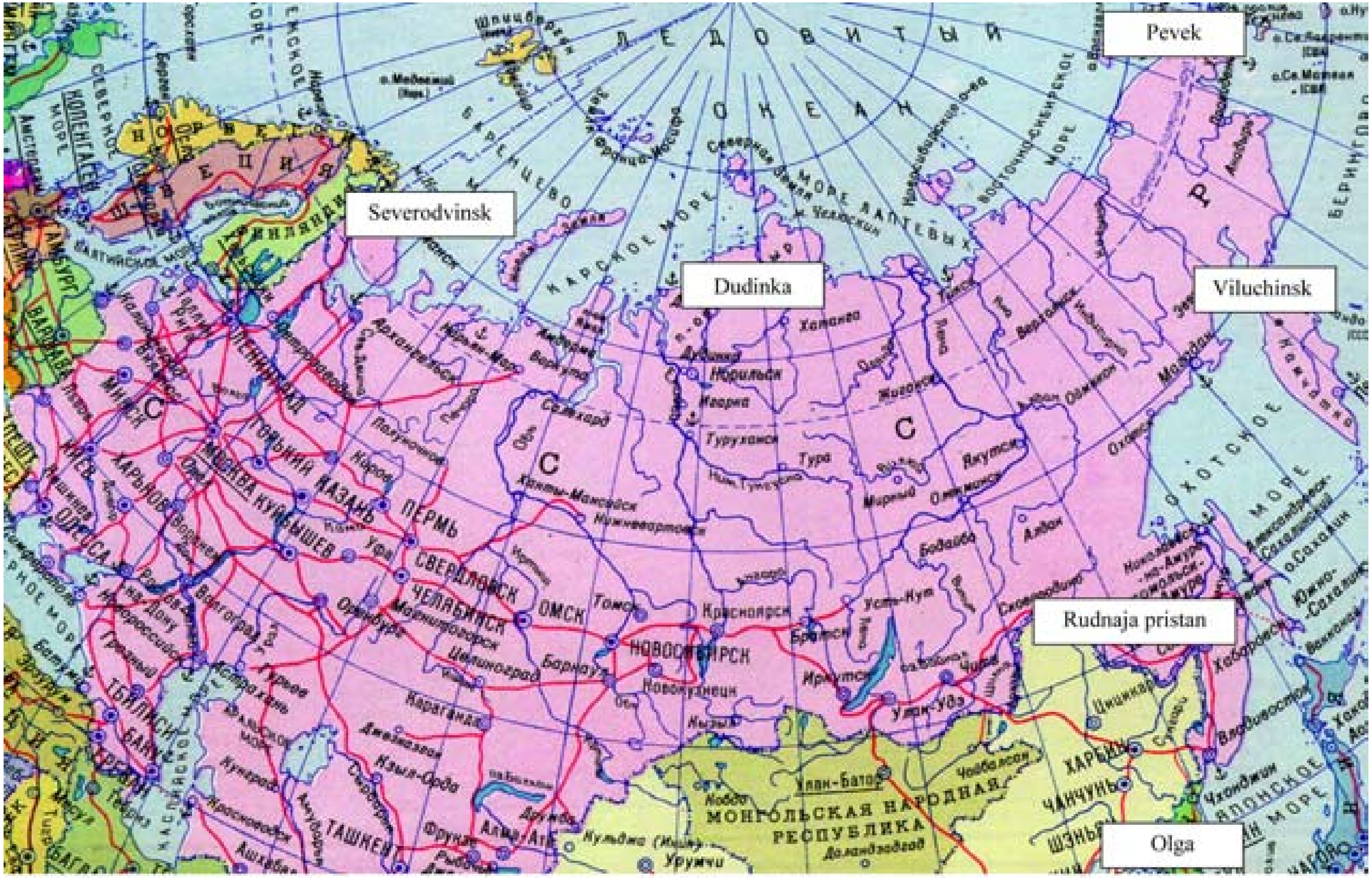


FIG. 2.1. Supposed areas of first FNCGP siting.

It is expected that all necessary approvals for the project will be obtained in early 2002. Construction is scheduled to commence in the first quarter of 2002 and run through mid 2004. Cold and hot non-nuclear testing of this first Demonstration Module (with reactor being simulated by a ± 20 MW helium electric heater for hot commissioning) are expected to be complete by mid 2005, at which time nuclear fuel will be loaded. Again, for this first plant, some 15 months of nuclear hot commissioning and testing is planned, resulting in turnover for commercial operation in mid to late 2006. Without any sacrifice to either electrical output or to the overall thermodynamic efficiency of the electricity production process, the design of the plant could facilitate the coupling of any desalination process at some later date, although currently this first Demonstration Module does not provide for this. For specific information regarding the potential coupling of PBMR to large-scale desalination processes please refer to Section 4.1 of this publication.

2.4.13. Tunisia

Tunisia was also one of the participating countries in the IAEA's feasibility study on the use of nuclear energy for seawater desalination in the North African Region (RAF/4/013). Tunisia is already experiencing a deficit of 50 000 m^3/day of potable water, which is alleviated by brackish water desalination using the RO process. This deficit will reach about 100 000 m^3/day by 2010.

Future studies are being carried out to select a suitable desalination process for bridging the deficit gap including those using nuclear energy. Two sites, Skirat and Zarat, located in the Southeast area of the country, were identified for specific studies. Recently Tunisia organized a dedicated project team with participation of three relevant organizations (Nuclear Research and Technology, Electricity, and Water) to carry out a feasibility study of a nuclear cogeneration plant for electricity and water in the country.

CHAPTER 3. CO-GENERATION PLANT CONCEPTS UNDER EVALUATION

There are various design concepts currently under evaluation for the purpose of demonstrating technical and economical viability of nuclear desalination under country specific conditions. Technology holders are proposing some design concepts for consideration by potential end users, while some others are being developed by end-users themselves. Information on most of these design concepts are made available to other Member States through the IAEA mechanisms, such as, technical meetings, co-ordinated research projects, and technical co-operation projects. The concepts can be divided into two main categories:

- Co-generation types where heat for desalination purposes is associated with electricity production; and
- Dedicated heat types of plants where no electricity production is sought.

This chapter gives an overview of the co-generation design concepts, while the next includes an overview of the dedicated-heat design concepts, available to the IAEA.

3.1. A small integrated PWR CAREM with RO and MED (Argentina)

3.1.1. Background

CAREM-25 is an advanced, but simple project for a small nuclear power plant. Its conception is based on new design solutions, after having accumulated significant worldwide experience in safe operation of light water reactors. This joint project involves CNEA (Comisión Nacional de Energía Atómica) and INVAP SE. An alternative design called CAREM-D has also been developed for the co-generation of electricity and potable water (modules of 10 000 m^3/day). This design involves the electrical and thermal coupling of desalination technology. It also involves design optimisation in order to allow modular flexibility of capacity and to maximise plant availability.

The size of the CAREM plant makes it especially adequate for nuclear desalination of seawater. The CAREM is an indirect cycle plant with some distinctive features, which greatly simplify reactor design and contribute to a higher level of safety. These features include:

- Integrated primary cooling system;
- Primary cooling by natural circulation;
- A self-pressurised reactor;
- Safety systems relying on passive features; and
- A unique coupling system to minimise the risk of water cross contamination.

3.1.2. Design description

Reactor

The CAREM nuclear power plant (NPP) has an integrated reactor. The entire high-energy primary system-core, steam generators, primary coolant and steam dome is contained inside a single pressure vessel. The flow rate in the reactor primary systems is maintained by natural circulation. The driving force obtained by the differences in the density along the circuit are balanced by friction and form losses, producing a flow rate in the core that allows for sufficient thermal margin to critical phenomena. The coolant acts also as a moderator.

Self-pressurisation of the primary system in the steam dome is the result of the liquid-vapour equilibrium: the core outlet bulk temperature corresponds to saturation temperature at primary pressure. Heaters and sprinklers that are typical of conventional PWR's are eliminated. Twelve identical 'Mini-helical' vertical steam generators, of the "once-through" type are used to transfer heat from the primary to the secondary circuit, producing dry steam at 47 bar, with 30°C of superheating. (Figure 3.1 shows the reactor main components.) The location of the steam generator above the core induces natural circulation in the primary system.

The secondary system circulates upwards within the tubes, while the primary system does so in counter-current flow (downward circulation). An external shell surrounding the outer coil layer, with an adequate seal guarantees that the entire stream of the primary system flows through the SGs. As another safety feature, steam generators are designed to withstand the pressure from the primary system up to the steam outlet and water inlet valves in case of loss of secondary pressure.

The CAREM plant has a standard steam cycle with a simple design. In accordance with the behaviour of once-through boilers, steam is superheated under all plant conditions and no super-heater is needed. Likewise, no blow-down is needed in the steam generators; this reduces waste generation. A single turbine is used, and the exhaust steam at low pressure is condensed in a water-cooled surface condenser. The condensate is then pumped and delivered to the full stream polishing system in order to maintain ultra-pure water conditions.

High purity water exiting the polishing system is sent to the low-pressure pre-heater using turbine extraction as a heating medium. The warm water is delivered to the water accumulator in order to perform degassing operations with additional heat using extraction steam. Water is then pumped to the high-pressure pre-heaters (two in tandem using extraction steam) and sent to the steam generators as feedwater, closing the circuit. The CAREM secondary circuit is not a safety-graded system, i.e., the nuclear safety of the plant does not rely on the functioning of the steam circuit.

Safety systems

The main criteria used in the design of Safety Systems are simplicity, reliability, redundancy and passivity. Special emphasis has been placed on minimising dependency on active components and operators' actions. The following is a list of these systems (see also Figure 3.2):

(1) First shutdown system (FSS): Consist- of Ag-In-Cd alloy rods.

(2) Second shutdown system: It is a gravity driven injection system of borated water at high pressure.

(3) Residual heat removal system: This reduces the pressure on the primary system and removes the decay heat in case of a lost of heat sink.

(4) Emergency injection system: This system prevents the core exposure in case of Lost of Coolant Accidents (LOCA).

(5) Containment system: This is a pressure-suppression type with two major compartments, (a dry well and wet well).

(6). Pressure relief system: This is aimed at protecting the integrity of the reactor's pressure vessel against over pressure in the event of an imbalance between the core power generated and the power removed by the systems.

Plant response to accidents

(1) **Blackout:** It is one of the events with a major contribution to core meltdown probability in a conventional light water reactor. In the CAREM NPP, the feedback coefficients will produce the self-shutdown of the nuclear reaction. The extinction and cooling of the core and the decay heat removal are guaranteed without electricity by the passive features of the safety systems. Loss of power produces the interruption of the feed water to the hydraulically driven CRDs, and thus produces the insertion of the absorbing elements into the core. The residual heat removal system removes the decay heat.

(2) **Loss of coolant accident (LOCA):** Since only small LOCAs are possible, and due to the large water inventory in the RPV, there is a long time span between the initiation of the LOCA and core exposure in comparison with conventional PWRs. The largest break allows some minutes of depressurisation before triggering the emergency injection system with the RPV at 15 bar and the core fully covered.

(3) **Main steam pipe break:** It produces a transient that can be easily handled by the safety systems due to the small water inventory of the steam generators in the secondary side and the large water inventory of the primary system.

NPP–Desalination Plant coupling

Reverse Osmosis plant coupling

For the co-generation option, there are few small changes in the CAREM plant Balance of Plant (BOP) design. The main change is in the condenser design. In order to optimise the thermal coupling, the cooling-water outlet temperature is taken to 43.8 C, while the turbine back-pressure is taken to 0.1238 bar. In a first approach, this change facilitates the pre-heated feed of for two 10 000 m^3/day RO modules, with a direct extension to a third possible module. In addition, the BOP for co-generation has additional piping for the seawater cooling flow, which bypass the condenser. It allows the seawater intake to be used by the sea water Reverse Osmosis (SWRO) plant when the condenser is down for maintenance or repairs. This BOP change reduces the mechanical power delivered by turbine by a small amount, while enhancing preheating for a single module. With a second module in operation, there is a clear benefit from the thermal coupling. This benefit is amplified with a third module.

The above coupling scheme may not be the most efficient option for a single module. In this case the overall plant efficiency is operated to be slightly better than a CAREM-25 coupled to a stand-alone SWRO plant. However this coupling scheme fields the maximum flexibility in terms of both capacity increase (by number of modules) and performance variation in the desalination plant without a significant impact on the nuclear plant.

The electrical output of the CAREM-25 allows the addition of several modules, (up to 10 modules, depending on the membrane technology chosen for the upgrade). These additional modules would require a new (most likely separate) seawater intake.

The co-generation plant layout places the desalination plant near the outlet channel, and foresees the connection of both plants by thermally isolated piping.

The make-up built houses, with two make-up tanks, an isolating system, and the membranes building, are designed and placed so that their walls may be shared by another building (in between) for additional RO modules (see Figure 3.4). This co-generation option was developed within the CAREM project at the conceptual stage, and implies a few changes in the plant BOP design, mainly in the condenser, to facilitate RO coupling.

Multi-Effect Distillation plant coupling

An analysis of coupling a CAREM plant with a thermal desalination system was performed within the framework of supporting IAEA's activities on nuclear desalination. This was due to general -interest shown by Member States in coupling of nuclear power plants with MED systems. Within the Co-ordinated Research Project, "Optimisation of the coupling of Nuclear and Desalination systems", the CAREM project contributed a modelling tool for the simulation of contamination migration through a coupling system (upon failure). In addition, a conceptual safety analysis of a nuclear desalination plant (NDP) with thermal coupling was elaborated and is presented in the following section. A generic MED Plant, most likely Low Temperature MED, consists of a variable number of evaporators or effects, through which steam from the turbine and seawater couple. This coupling is either in parallel flows (co-current) or in opposite direction flows (counter-current). A simplified scheme of the NDP is shown in Figure 3.5

Conceptual safety analysis of the NDP

In the nuclear industry "defence-in-depth" is singled out amongst the fundamental principles since it underlies the safety technology of nuclear power. The concept is centred on several levels of protection including successive barriers providing a graded (envelope) protection against a variety of transients. These transients include those resulting from equipment failure and human error, or from internal or external events that may lead eventually to accidental conditions. The graded (envelope) protection should prevent the release of radioactive material to the environment. The implementation of the defence in depth concept is mainly carried out through deterministic analysis (which may be supplemented with probabilistic studies) and application of sound engineering practices based on research and operational experience. The application of the concept of defence in depth to the design process consists of the consideration of a series of multiple and successive levels of protection aimed at ensuring appropriate protection in the event that the previous one fails. Now that the main safety concepts are presented and the NDP coupling has been described, a conceptual safety analysis can be performed these including:

– Identification of the defence-in-depth barriers in the NDP;
– Construction of the postulated initiating event list;
– Definition of critical group;
– Acceptability of the design against the criterion curve.

An examination of, the *defence-in-depth barriers* and a look at the NDP as a whole, five main barriers can be identified to avoid contamination of the fresh water product, these are:

– Fuel matrix (pellet)
– Fuel rod cladding
– Steam generators tube walls
– Heat exchanger tube walls
– DP Isolation System (between the DP and distribution-piping grid).

This means that, in case there is a release of fission products due to the catastrophic damage of the fuel matrix and fuel element cladding, contamination could reach the fresh water product only by a chain of failures. The steam generators would need to break, the turbine trip would have to fail, the heat exchanger tube walls would have to break simultaneously, and then the Isolation System would have to fail to perform its function.

This is a chain of independent events that would need to take place simultaneously in order to allow a situation that, if not managed properly, would produce consequences to the consumers. The probability of this "chain of occurrences" may be estimated, as well as the effective dose it may produce. In the case that the results become unacceptable, according to the criterion curve [13], the Isolation System design would be improved in order to assure the performance of the isolation function, in the required time and with the required reliability.

Interaction between plants

Effects of DP events on the NPP: From the Deterministic Analysis it may be recalled that there is no event initiated at the DP level that could seriously affect the NPP [13].

Effects of NPP events on the DP: Concerning the effects of the NPP on the DP, there are events initiated in the NPP that could lead to the shut down of the DP. Most of them would be related to operational transients.

Special requirements

As a result of the nuclear power plant and desalination plant coupling it is of great importance to set the main concepts to perform a safety analysis within the framework of safety culture and defence in depth widely spread on the NPP's design.

It is necessary that the desalination plant cause no perturbation to the nuclear power plant while perturbations to the desalination plant due to the nuclear power plant must be analysed.

The most commonly accepted techniques used for the deterministic safety analysis of nuclear power plants are adequate for nuclear desalination plants.

3.2. A small advanced integral PWR SMART with MED (Republic of Korea)

3.2.1. Background

The use of well-established and advanced nuclear energy technology for seawater desalination is recognized as providing dual benefits namely the promotion of nuclear energy utilization and security of freshwater resources. A national R&D project has been conducted for developing a small and advanced nuclear reactor, SMART, and an integrated nuclear desalination plant with the SMART for the demonstration of nuclear seawater desalination.

Data Sheet

TABLE 3.1. GENERAL INFORMATION

Design Name	CAREM D	
Plant production	Electricity–potable water co-generation	
Reactor Type	Integrated PWR	
Desalination technology	Pre-heated RO	
Gross Thermal Power	100	MWth
Max. Electrical Power Output	27	MW(e)
Max. Water Output per Module	10 000	m^3/day
Max. # of Pre-heated Modules	3	
Core and Reactivity Control		
Initial Enrichment of Fuel	3.4	%
Refuel Cycle	390 full power days	
Clad Material	Zircaloy-4	
Control Rod neutron absorber	Ag-In-Cd	
Additional Shut-down system	Boron Injection	
Burnable poison	Gd_2O_3-UO_2	
Reactor Cooling System		
Cooling Mode	Natural Circulation	
Coolant Inventory	39	m^3
Coolant mass flow through core	410	Kg/sec
Operating Coolant Pressure	12.25	MPa
Core Inlet/Outlet Temperature	284/326	°C
Reactor Pressure Vessel		
Overall Length/Vessel Diameter	11/3.2	m
Vessel Material	SA508 Grade 3 Class 1	
Lining Material	SS-304L	
Design pressure	14.5	MPa
Gross Weight (without internals)	130	Ton
Steam Generator		
Number	12	
Type	Once through	
Configuration	Integrated–mini helical	
Tubes Material	Inconel 690 (SB 163 N06690)	
Shell Material	SS-304 L	
Feed Water Pressure	4.7	MPa
Feed Water Temperature	200	°C
Steam Pressure	4.7	MPa
Min. Steam Temperature	290	°C
Containment		
Type	Pressure Suppression	
Design Pressure	0.5	MPa
Design Temperature	175	°C

TABLE 3.2. SAFETY SYSTEMS

First shutdown system

Absorbing material	Ag-In-Cd
Shutdown function driven by	Gravity
Number of elements of the Fast Extinction System	6
Number of elements of the Adjust and Control System	19

Second shutdown system

Neutron Absorber Material	Borated solution
Operation Mode	Gravity driven discharge
Redundancy	Tanks 2 × 100%
	Valves: 4 × 100%

Residual Heat Removal System-Emergency Condenser

Operation Mode	Steam Condensation
Redundancy	Condenser 2 × 100%
	Valves: 4 × 100%
Autonomy	> 48 hours

Emergency Injection System

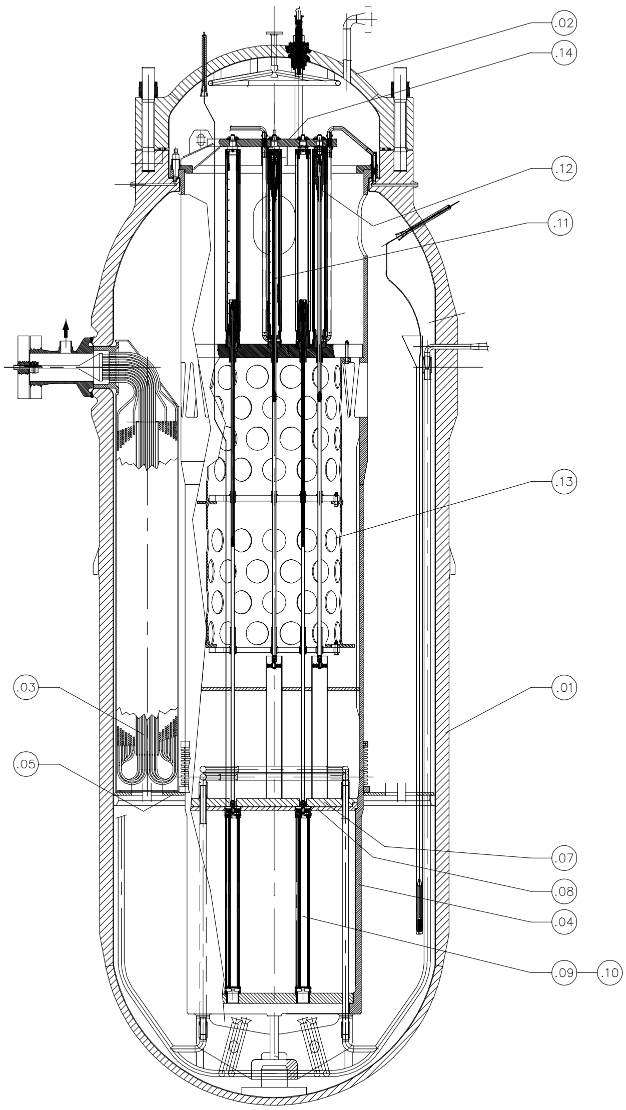
Pressure of Injection	1.5	MPa
Operation Mode	Pressurized Tanks	
Redundancy	Tanks 2 × 100%	
	Valves: 4 × 100%	
Autonomy	> 48 hours	

Safety Relief System

Pressure Set Point	14.0	MPa.
Redundancy	3 × 100%	

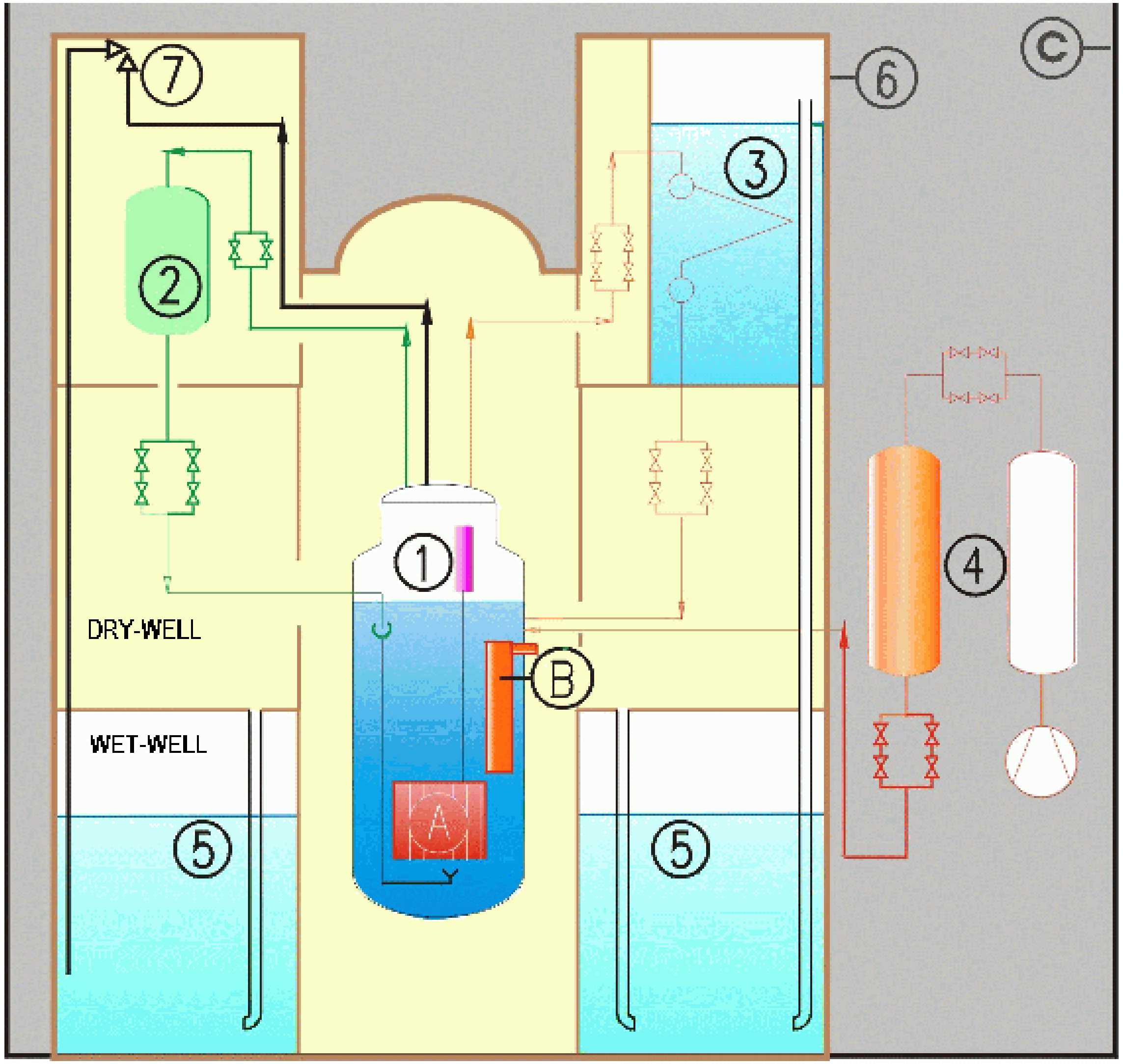
Turbine System

Type	Condensing	
Stages	1	
Speed	3000	Rpm
Steam Pressure	4.7	MPa
Steam Temperature (30°C superheated)	290	°C
Steam Flow rate	175.32	Ton/h



Reactor and Pressure Vessel
.01 Reactor Pressure Vessel
.02 Cover head
.03 Steam generator
.04 Barrel
.05 Flow separator device
.06 Core lower grid
.07 Core upper guide structure
.08 Core
.09 Absorbing element
.10 Fuel element
.11 Adj. And ctrl. system CRD
.12 Fast extinction system CRD
.13 CRD Rods guide structure
.14 CRD feeder structure

FIG. 3.1. Main reactor components.



1 First shut-down system
2 Second shut-down system
3 Residual heat removal system
4 Safety injection system
5 Pressure suppression pool
6 Containment
7 Pressure relief system
References
A CORE
B STEAM GENERATOR
C NUCLEAR BUILDING (Secondary containment)

FIG. 3.2. CAREM safety systems.

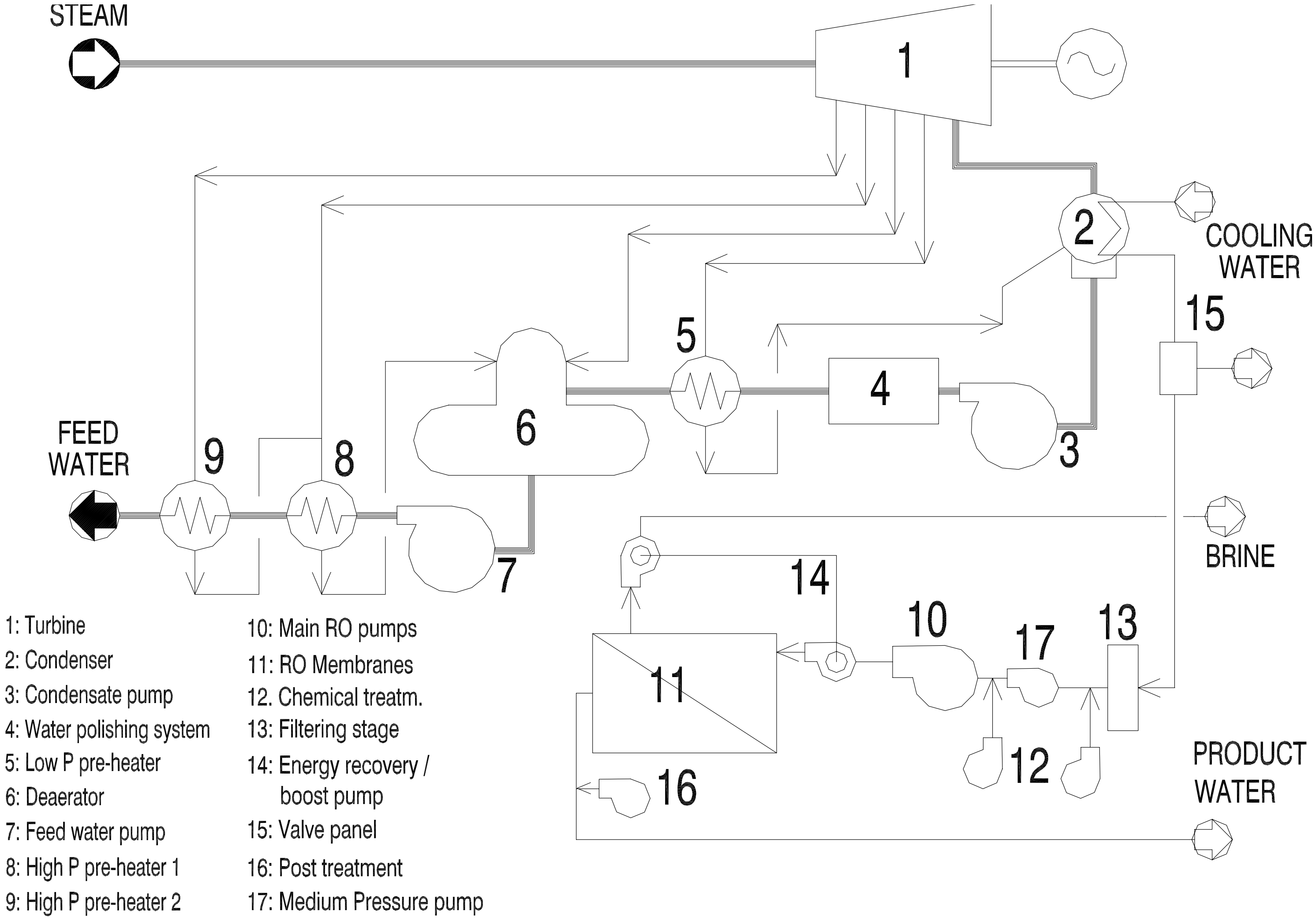


FIG. 3.3. CAREM D, balance of plant and preheated SWRO desalination system.

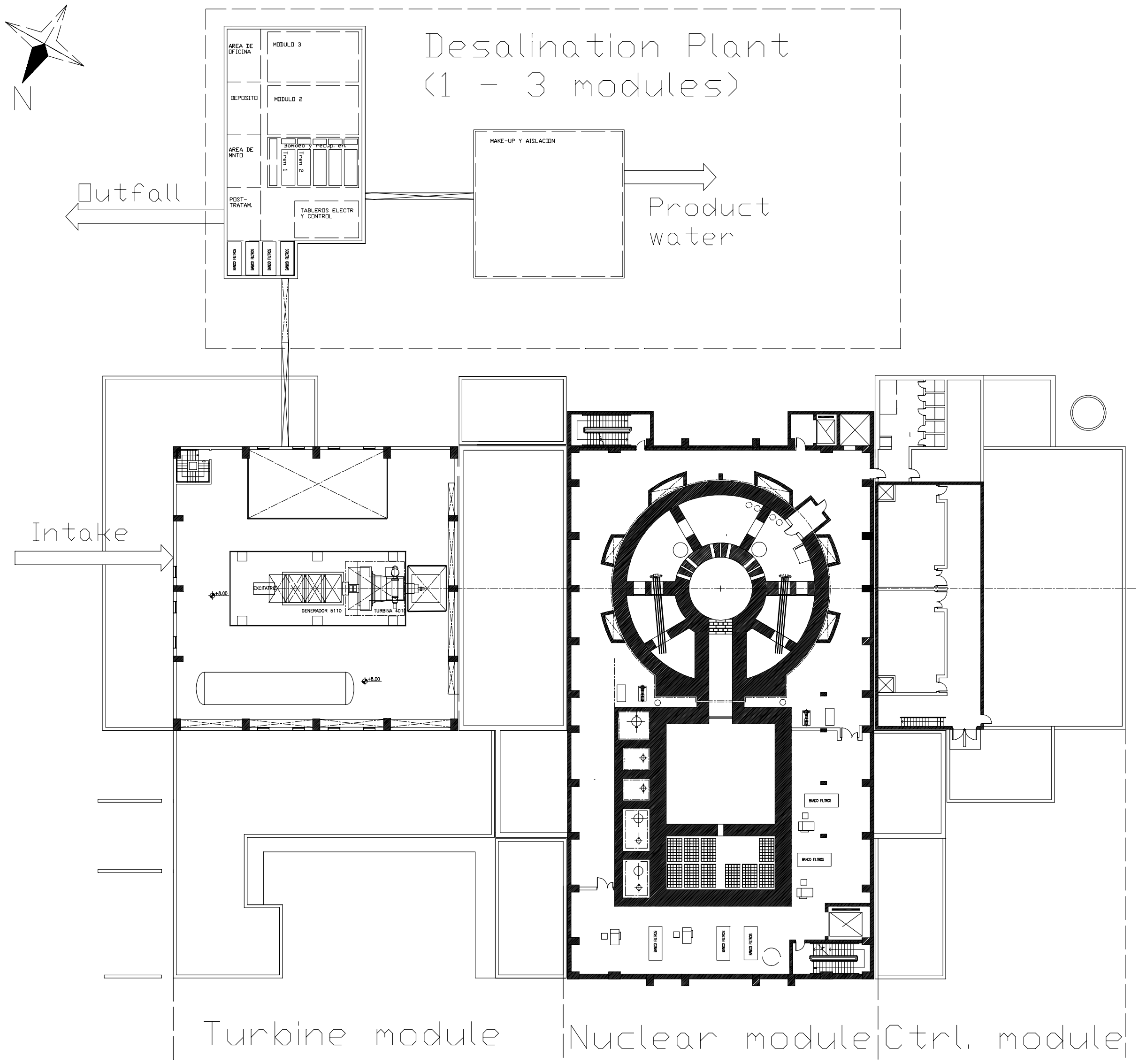


FIG. 3.4. CAREM-D general layout in a top view, including NPP and SWRO.

3.2.2. Design description

SMART Design Concepts

The SMART is an integral type power reactor with a rated thermal power of 330 MWt. It is different from the loop-type reactors due to the arrangement of its primary components. All major primary components, such as core, steam generators, pressurizer, control element drive mechanisms, and main coolant pumps, are installed in a single pressure vessel, as shown in Figure 3.6.

The integrated arrangement of these components enables the elimination of large pipe connections between the components of the primary reactor coolant systems, and thus fundamentally eliminates the possibility of large break loss of coolant accidents. This integral arrangement, in turn, becomes a contributing factor to the safety enhancement of the SMART.

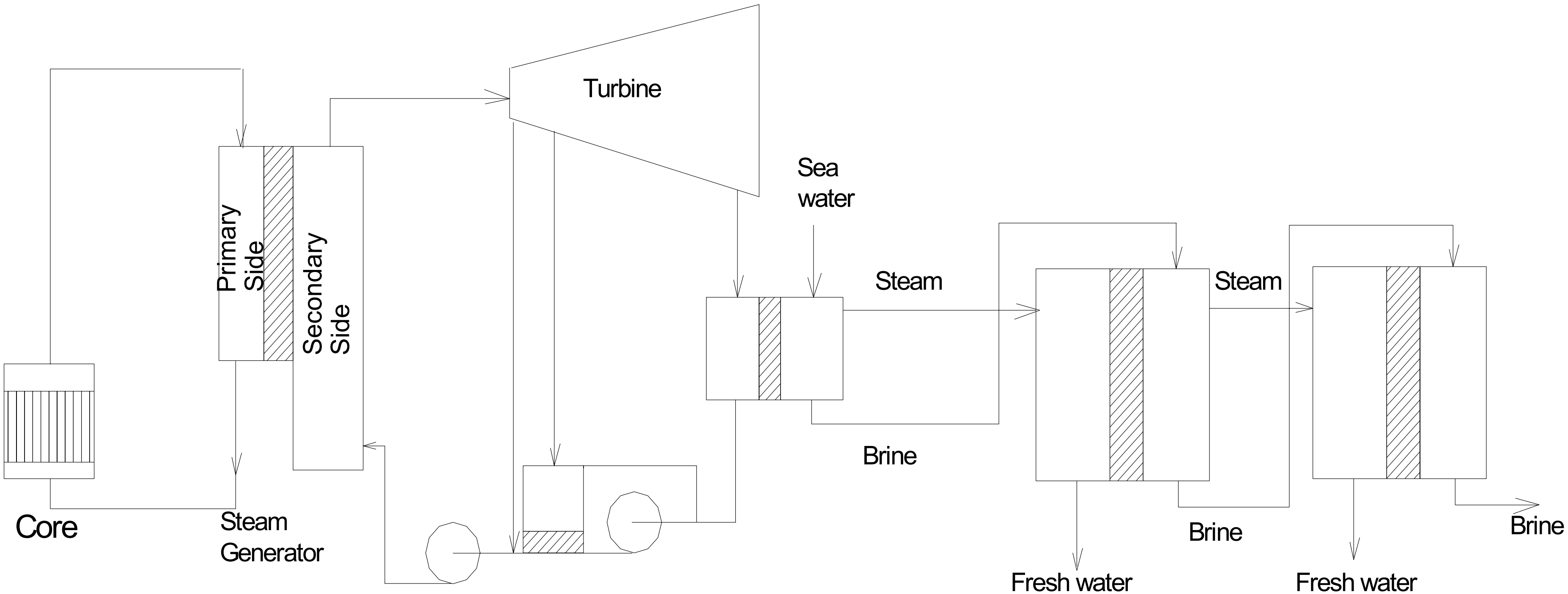


FIG. 3.5. Scheme of a generic coupling of CAREM Plant with a MED desalination system.

These innovative and advanced features are adopted in the SMART design to enhance its safety, reliability, performance, and operability. Most of these technologies and design features implemented in the SMART are those that have been well proven through the operation of commercial power reactors, and new features will be proven through various tests.

Reactor and major components

Fuel and reactor core: The SMART core consists of 57 fuel assemblies with a design based on a well-proven low-enriched 17 × 17 UO_2 fuel assembly. The assembly is designed to accommodate power ramps during load following. Soluble boron-free operation is one of the evolving design characteristics of the core along with the low core power density design.

Axially zoned solid burnable absorbers and Control Element Drive Mechanism (CEDM) compensate the reactivity change due to the fuel burn-up during normal operation with very fine-step manoeuvring capability. A single or modified single batch reload design is adopted to provide longer than a three-year refuelling cycle.

Enhanced safety is achieved due to the strong negative moderator temperature coefficient and sufficient thermal margin. The on-line core monitoring and protection systems are provided to assess the real core operating conditions and thus to provide the plant operator with adequate alarms for the proper responses.

Steam generator (SG): Twelve identical SG cassettes are located on the annulus formed by the reactor pressure vessel (RPV) and the core support barrel. Each SG cassette is of once through design with helically coiled tubes wound around the inner shell. The primary reactor coolant flows downward in the shell side of the SG tubes, while the secondary feedwater flows upward in the tube side.

The secondary feedwater exits the SG in a superheated steam condition. For performance and safety, each SG cassette consists of six independent modules, and six modules from three adjacent SGs are then grouped into one nozzle. Three nozzles eventually compose one section. This concept of SG grouping minimizes the asymmetric impact of a SG section isolation of the reactor system.

Pressurizer (PZR): An in-vessel self-pressurizing concept is adopted for the PZR of the SMART. The PZR is located in the upper space of the reactor assembly and is filled with water and nitrogen gas. The concept of the self-pressurizing design eliminates the active mechanisms such as spray and heater.

By keeping the average primary coolant temperature constant with respect to power change, the large pressure variation due to power change during normal operation can be reduced. To achieve self-pressurizing, a PZR cooler for maintaining a low PZR temperature, and a wet thermal insulator for reducing heat transfer from the primary coolant are installed.

Control element drive mechanism (CEDM): Fine control of core reactivity during normal operation is a design requirement for the SMART CEDM. This is due to soluble boron-free operation. To meet this requirement of fine reactivity control, the CEDM is designed for very fine power manoeuvring capability using linear step motor. Forty-nine (49) CEDMS are installed in the SMART.

Main coolant pump (MCP): The MCP is a canned motor pump that does not require any pump seal. This characteristic eliminates a small break loss of coolant accident associated with a pump seal failure in the case of a station black out.

The SMART has four MCPs installed vertically on the RPV annular cover. Each MCP is an integral unit consisting of a canned asynchronous 3-phase motor and an axial flow single-stage pump. A common shaft rotating on three radial and one axial thrust bearings connects the motor and pump. Table 1 summarizes the major design parameters of the SMART system.

Safety systems

Besides the inherent safety characteristics of the SMART, further safety enhancement is accomplished with highly reliable engineered safety systems. These systems are designed to function passively. The following is a summary of major safety systems adopted in the SMART design.

Reactor shutdown system (RSS):

The shutdown of the reactor can be achieved by one of two independent systems. The primary shutdown system is the control rods with Ag-In-Cd absorbing material. In the case of the failure of the primary shutdown system, the emergency boron injection system is provided as an active backup. One of the two trains is sufficient to bring the reactor to sub-critical condition.

Passive residual heat removal system (PRHRS):

The PRHRS removes the core decay heat by natural circulation in emergency situations. The system consists of four independent trains with a 50% capacity for each train in core decay heat removal, and the operation of any two trains is sufficient to remove the decay heat. The system is designed to be capable of decay heat removal for 72 hours without any corrective action by operators for the “design base accidents”.

Reactor Over-Pressure Protection System (ROPS): The function of the ROPS is to reduce reactor pressure during the postulated “beyond design base accidents”. The system consists of two parallel trains that are connected between the PZR and the internal shielding tank through a single pipeline. When the primary system pressure increase over the set point value, pilot operated safety relief valves (POSRV) on both trains are opened to discharge the steam into the internal shielding tank.

Containment Over-Pressure Protection System (COPS): The containment is a steel structure in a concrete building. During any accident causing the temperature and thus the pressure to rise in the containment, the cooling is accomplished in a passive manner. The heat is removed through the steel structure itself, and through the PRHRS cool-down tank installed in the containment.

TABLE 3.3. KEY DESIGN PARAMETERS OF SMART SYSTEM

General Information	
Reactor name/type	SMART/Integral PWR
Thermal power (MWt)	330
Design life time (year)	60
Max. Electric power (MW(e))	100
Fuel and Reactor Core	
Fuel type	UO_2 Square FA
Enrichment (w/o)	4.95
Active fuel length (m)	2.0
No. Of fuel assemblies	57
Core power density (w/cc)	62.6
Refuelling cycle (year)	>3
Reactivity Control	
No. Of control element banks	49
No. Of control banks/material	49/Ag-In-Cd
Burnable poison material	Al_2O_3-B_4C, GD_20_3-UO_2
Reactor Pressure Vessel	
Overall length (m)	9.8
Outer diameter (m)	3.96
Average vessel thickness (mm)	19.8
Vessel material	SA508, CL-3
Reactor Coolant System	
Design pressure (Mpa)	17
Operating pressure (MPa)	15
Core inlet temperature (ºC)	270
Core outlet temperature (ºC)	310
Steam Generator	
Type	Once-through with helically coiled tubes
No. Of steam generators	12
Design temperature (ºC)	350
Design Pressure (Mpa)	17
Main Coolant Pump	
Type	Canned motor pump
No. Of MCP	4
Flow rate (m^3/h)	2006
Water head (m)	17.5
Control Element Drive Mechanism	
Type	Linear pulse motor driven
No. Of CEDM	49
Step length per pulse (mm)	4
Make-up System	
No. Of trains	2
Operating mode	Active
Secondary System	
Feedwater pressure (MPa)	5.2
Feedwater temperature (ºC)	180
Steam pressure (MPa)	3.0
Steam temperature (ºC)	274
Degree of superheating (ºC)	40

SMART integrated nuclear desalination plant

Major desalination processes that are widely utilized are the distillation process and Reverse Osmosis (RO). Distillation processes such as Multi-Stage Flash (MSF) and Multi-Effect Distillation (MED) require heat energy, while RO requires mainly electricity. The prime interest in the application of the SMART to desalination is the utilization of steam rather than electricity.

In this regard only the distillation process was taken into consideration for the SMART integrated nuclear desalination system. The desalination system aim is to produce 40 000 m^3/day of desalted water. This amount of product water is assessed to be sufficient for a population of about 100 000.

Since the objective of the SMART system is to economically produce both water and electricity, the prime factor for the selection of the distillation process is based on minimizing energy consumption. For the economic choice of a distillation process and for its thermal coupling with the SMART system, a preliminary sensitivity analysis was carried out for both MSF and MED with regard to the method of energy extraction from the turbine system.

Three methods of steam extraction were considered in the analyses, which were prime steam, turbine extraction and backpressure turbine. Based on the results of the sensitivity analysis, the MED with steam extraction was selected for coupling in the SMART integrated nuclear desalination system. Based on this concept of coupling, the desalination system was composed of four units, each having a water production capacity of 10 000 m^3/day. Figure 1 shows the configuration of the coupling scheme for the SMART integrated nuclear desalination plant.

Safety analysis of SMART and the integrated nuclear desalination system

The two major safety aspects considered in the design of the coupled system are:

Protection of the product water from possible contamination by radioactive materials, and protection of the SMART reactor from potential disturbances of the desalination system.

Regarding the first aspect, two protection mechanisms are provided. One of the mechanisms consists of two barriers namely the steam generator and brine heater (steam transformer) along with the pressure reversal between the energy supply and the desalination system.

The other mechanism adopted in the system is the continuous radioactivity monitoring system installed in the line of the water production system to check for any symptoms of radioactivity carry-over. Due to the thermal coupling between the nuclear and the desalination system, any transient of the desalination system can directly impact reactor safety.

A slow transient, such as a gradual reduction in the energy demand of the desalination system, can be easily accommodated by the SMART system through either the load following capability or the cutback of energy supply to the desalination system. Thus, only fast transients induced by the desalination system become important events to be considered for reactor safety. For the current desalination system, the following three events were identified as the major potential disturbances induced by the desalination system that may affect the safety of the SMART:

– turbine trip due to desalination system disturbances;
– excess load due to increased steam flow to the desalination system;
– loss of load due to desalination system shutdown.

The safety of the SMART conceptual design was evaluated against the limiting design base events (DBEs). A safety analysis methodology was developed, including a set of DBEs based on the ANSI/ANS–51.l-1983 (1988). A best estimate system analysis code, MARS/SMR, which is under development, was used to evaluate the safety. MARS/SMR is based on the best estimate three-dimensional system analysis code, MARS, and has integral reactor specific thermal hydraulic models, systems and components.

Based on the design, a set of limiting safety related DBEs was determined for the safety evaluation of the SMART system. The limiting DBEs include steam line break, turbine trip, feed water line break, and total loss of flow, SG tube rupture, and Small Break Loss of Coolant Accident (SBLOCA). In the safety analyses, conservative assumptions including initial and boundary conditions were employed to evaluate the safety envelope.

The results of the safety analyses showed that the peak reactor coolant system (RCS) pressure, specified acceptable fuel design limit (SAFDL) on minimum departure from nucleate boiling ratio (DNBR), and the minimum collapsed core level are within the design limits of 110% design pressure, 1.30 and no core uncovery, respectively.

It was found that feed water line break, steam line break, and the SBLOCA gave the maximum peak pressure, minimum DNBR and the minimum collapsed core level. It was also confirmed that the SMART design has a sufficient safety margin.

TABLE 3.4. LEVELIZED WATER PRODUCTION COST WITH RESPECT TO THE MAXIMUM BRINE TEMPERATURE

MBT (ºC)	GOR	Base Unit Cost ($/(m^3/d))	Water Production Cost ($/m^3)	Net Salable Electricity (MW(e))
40	6.2	669	0.92	90.1
45	7.8	713	0.86	90.5
50	9.4	758	0.83	90.9
55	10.7	794	0.83	90.9
60	12.1	833	0.83	90.8
65	13.3	866	0.83	90.7
70	14.5	900	0.84	90.5
75	15.6	930	0.84	90.4
80	16.6	958	0.85	90.3
85	17.5	983	0.86	90.2
90	18.3	1,005	0.87	90.0
95	19.0	1,025	0.88	89.9
100	19.7	1,044	0.90	89.7

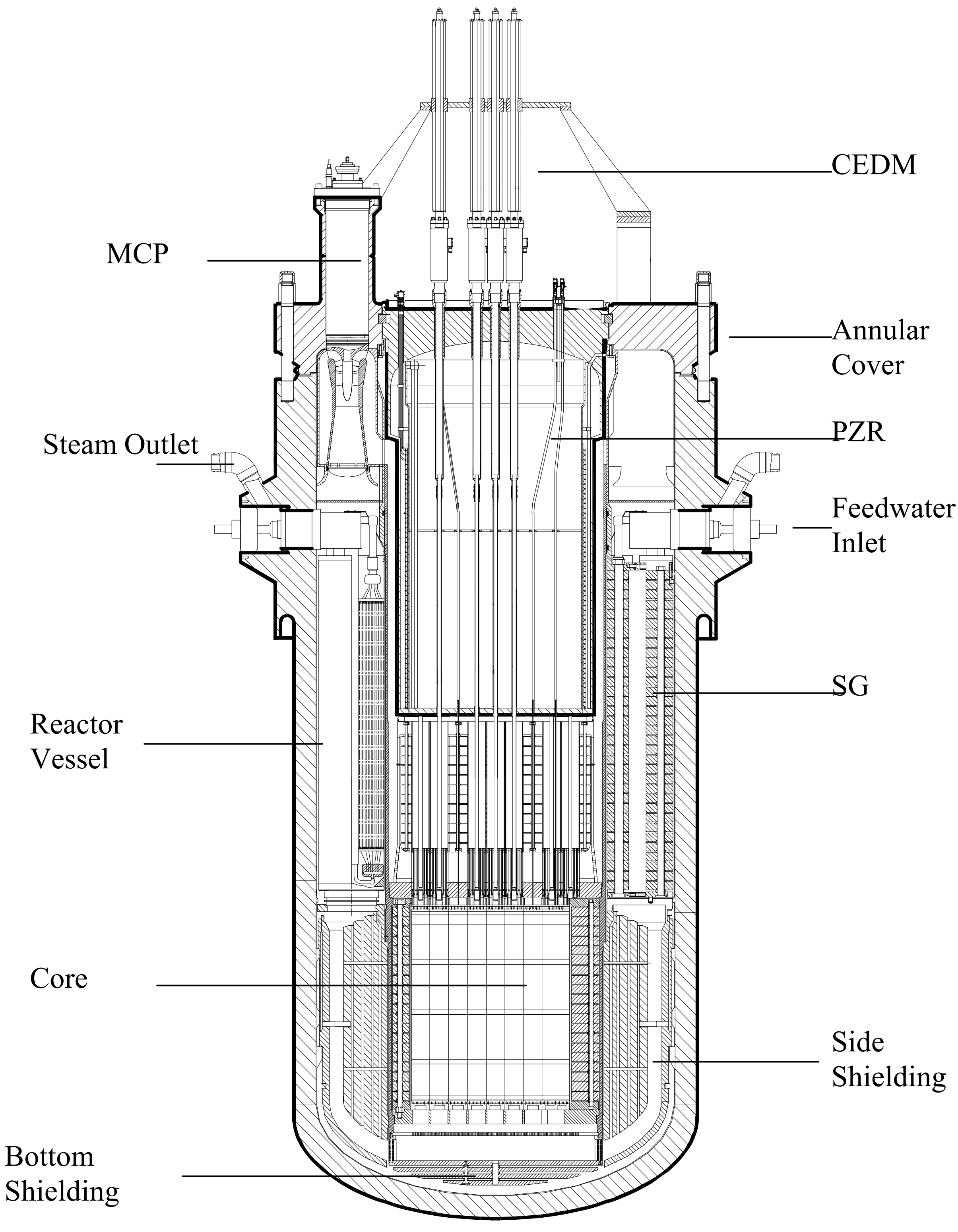


FIG. 3.6. SMART Reactor Vessel Assembly.

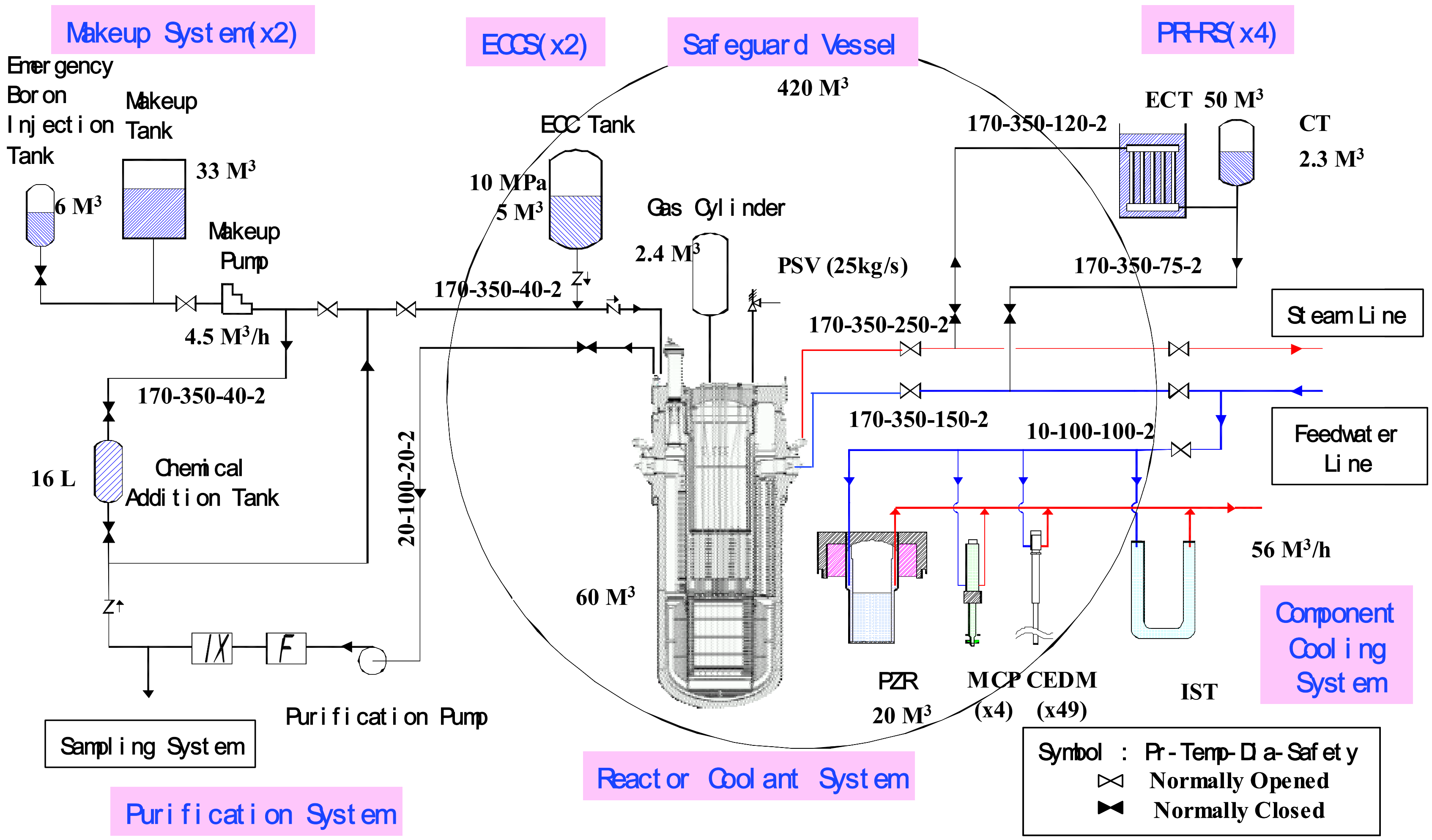
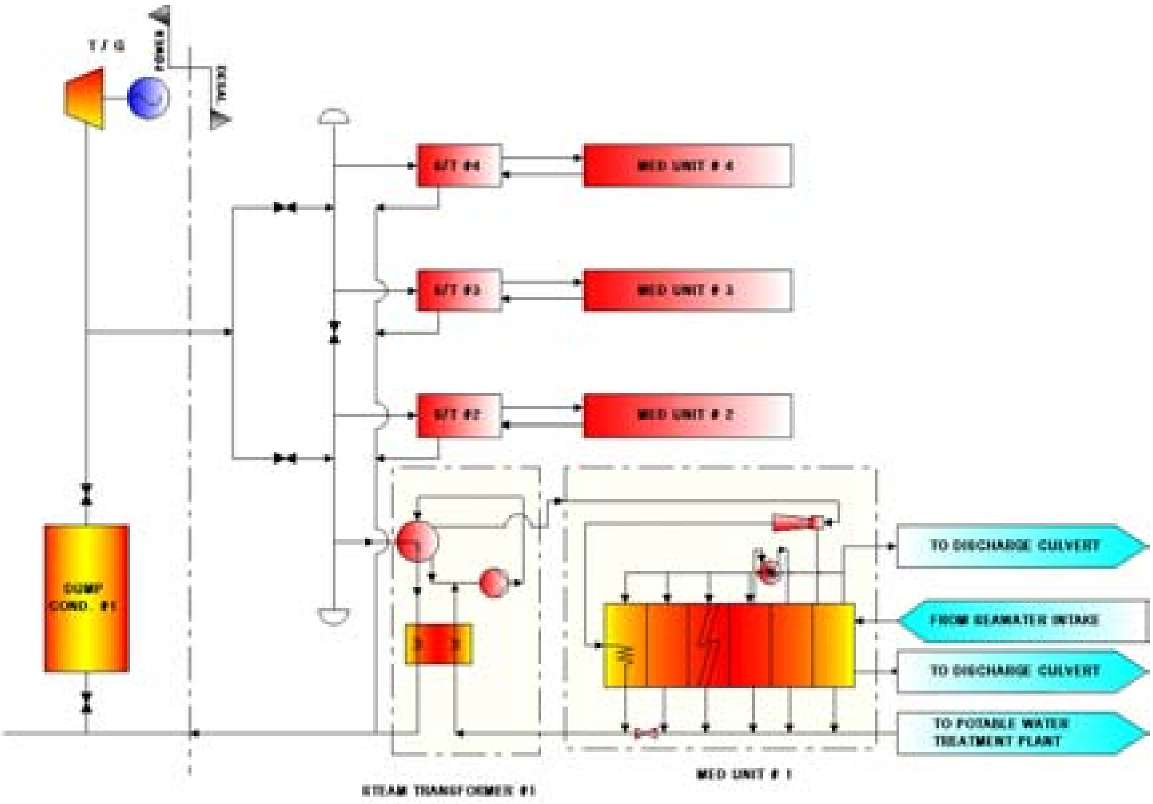
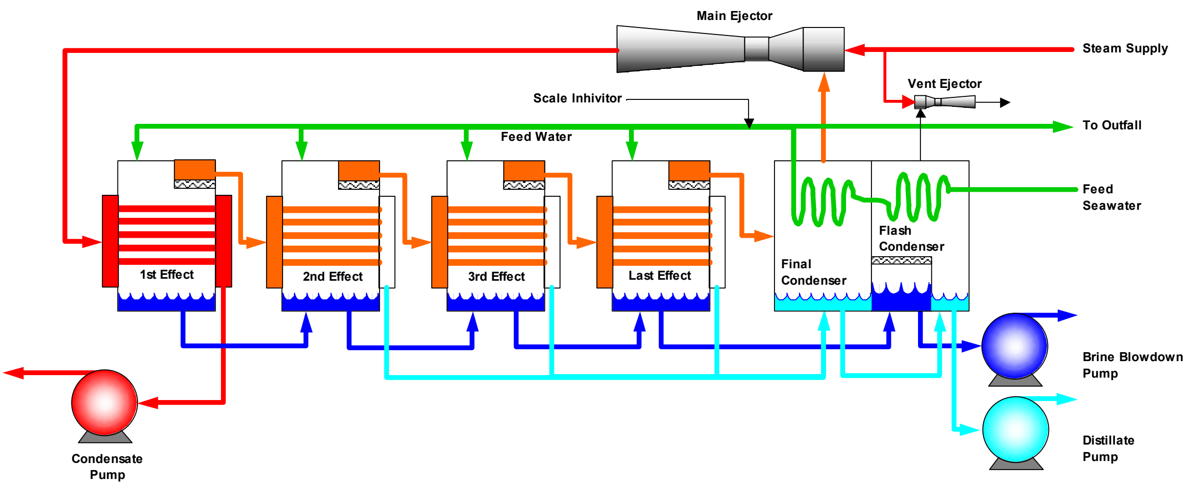


Fig 3.7. Schematic diagram of the SMART NSSS configuration.



Schematic Diagram of Desalination Plant



Schematic Diagram of MED-TVC

FIG. 3.8. Schematic diagram of the integrated nuclear desalination system of SMART.

Thermal-hydraulic interactions exist between the SMART system and the desalination system due to the thermal coupling of both systems. The effect of these interactions on the safety of the SMART was also evaluated. A turbine trip may be caused by the potential desalination system disturbances. Upon the reactor trip, the PRHRS comes into operation automatically and then removes the core decay heat enough to keep the RCS pressure and DNBR below the design limits.

The increased steam flow to the desalination system causes the overcooling of the SMART primary RCS, leading to the excess load event due to the large negative moderator temperature coefficient and causing the high power reactor trip signal or low secondary steam pressure trip signal, depending on the causes of the increase of the steam. The results of this event are to be bounded by Main Steam Line Break (MSLB) accident.

A sudden stop of the steam flow to the desalination system due to unexpected desalination system shutdown causes the secondary system pressure to increase and thus causes the reactor trip by the high secondary steam pressure trip signal. The system behaviour by this event was bounded by the total loss of load by the secondary system or turbine trip that were well controlled by the proper system responses. These potential events have been confirmed to be bounded by the design bases accidents of the SMART.

3.2.3. Economic perspectives

For the SMART integrated nuclear desalination plant, two types of preliminary economic evaluation were carried out to assess the cost of water production and electricity generation. The IAEA Desalination Economic Evaluation Program (DEEP) was used for the evaluation. Table 3.4 shows the levelized water production cost of the target water product of 40 000 m^3/day, as a function of the maximum brine temperature (MBT) for the fixed over-night construction cost of the SMART system.

The lowest water production cost of 0.83 \$/m^3 was obtained at an MBT of 65ºC and at the Gain Output Ratio (GOR) of about 13. The plant also generates electricity of about 90 MW(e). The amount of electricity generation remains nearly the same over the entire MBT range.

The results indicates that the optimum value of MBT with respect to economic use of energy lies in the range of 60–70ºC for the current coupling arrangement. From this result, it was also found that approximately 10% of the energy produced by the SMART is consumed for the target water production of 40 000 m^3/day.

3.3. PHWR and PWR with MSF (Pakistan)

3.3.1. Background

Like other Asian countries Pakistan, is likely to face a severe shortfall of water in the coming decades. Agriculture, population growth and urban expansion has an increasing demand on the nation’s dwindling water resources:

- Agriculture is an important sector of the Pakistani economy and has contributed about 25% of its GDP over the last decade. The climate in Pakistan, being arid to semi-arid, has had its agriculture almost entirely dependent on irrigation. The bulk of Pakistan’s agricultural production (about 78%) comes from irrigated land. Water availability rather than the land resource is going to be the main limiting factor for the future food and fibre needs of the country.
- The population density in Pakistan increased three times between 1961- 1998 period, making Pakistan the seventh most populous country in the world.
- The urban water demand is also rising sharply in Pakistan. Karachi is a typical example. From a tiny fishing village in the 1860’s, it is now grown into one of the largest and busiest metropolitan area of the country and this has put an enormous strain on its water supply system.

Pakistan though endowed with substantial water reserves, has not been able to take full advantage from this invaluable resource. According to some estimates, a substantial proportion of Pakistan's water resources go to waste due to percolation, evaporation, faulty irrigation methods and discharge to the sea.

On average about 169 billion cubic meters (Bm^3) of river water flow into Pakistan per annum, the annual flow variation is in the range of about 124 -230 Bm^3 and more than 80% of this flow occurs during summer season. The irrigated plains of the Indus basin are underlain by an extensive ground water aquifer of varying water quality. The ground water resources are extensively exploited with the help of public and private tube-wells for irrigation and drainage.

The Indus Basin Irrigation System is one of the largest man-made systems in the world. It is comprised of the Indus river and its major tributaries, three major reservoirs of about 18.5 Bm^3 of conservation storage, 23 barrages, head-works, siphons, 12 inter-river links and 48 canal commands. The total length of the canals is about 57 200 km, with watercourses, field channels and field ditches running another 1.6 million-km. This system supplies water to about 17 million hectares.

The present per capita availability of water in Pakistan has fallen to about 20 percent of the value at the time of Independence in 1947 and may well reach near "water stress" levels in the next 20 years. The near-term national water requirements and availability are summarized in Table 3.5.

The use of nuclear energy for power generation and seawater desalination seems quite in order. There was a definite need to set up a nuclear desalination demonstration plant around KANUPP, for which a detailed engineering study is being carried out. Parallel to this activity, is the planning of a large-sized dual purpose NPP to meet the growing needs of power and water in the Karachi metropolitan region by PAEC

The work on the nuclear desalination demonstration plant and the large sized plant has not progressed as fast as expected due to national economic difficulties. Our efforts for obtaining inputs on the latest available technologies (of Japanese, Korean & Russian origin) through IAEA's INT/4/134 Project also did not yield the desired response.

TABLE 3.5. NATIONAL WATER REQUIREMENT & AVAILABILITY

	Year	1992–93	1997–98	2002–03	2010–11
1.	Population, million	115	130	158	167
2.	Water Requirement, Bm^3			189 (153.1)	237 (192)
3.	Availability at Farm Gate, Bm^3	155 (125.12)	165 (133.28)	166 (134.48)	170 (137.6)
	Cubic meter/Capita	1348	1269	1051	1018
	Shortfall, Bm^3			23 (18.6)	67 (54.4)

Values in parenthesis are in Million Acre feet

Source: For Water Data, 9th Five Year Plan Draft Document

An opinion is now being expressed against linking a 1 MGD MSF plant to KANUPP, whose life is undergoing extension from 2002 to 2012 AD. This period of 10 years may not be sufficient to justify a large thermal-based desalination process (unless further extension of the life of KANUPP was intended), instead, a smaller facility for R&D may be preferable.

3.3.2. Design study for KANUPP

The Karachi Nuclear Power Plant (KANUPP) is located at Paradise Point on the seashore, 15 miles west of the city of Karachi (see coastal map of Karachi, IAEA 1997). The 137 MW(e) (gross) and 125 MW(e) (net) plant of the horizontal CANDU-PHW type was constructed by Canadian General Electric (C.G.E).

The power station has 6 steam generators where heavy water coolant in the primary loop exchanges heat to produce steam in a light water secondary circuit. Saturated steam at 559 psa or 38.5 bar (kept constant irrespective of load) measured at the throttle valve is supplied to a tandem compound turbine with one high pressure and two double flow low pressure cylinders. Feed water is regeneratively heated in 5 stages and returned to the steam generators.

The choice of the dual-purpose scheme is fairly restricted in the case of KANUPP, as it has been basically designed only as a power plant, however, with the small water to power ratio envisaged, an extraction-condensing scheme (pass-out turbine) would be both practical and economical, as shown in Figure 3.9.

The steam take-off for the brine heater would be at the crossover point between the moisture separator and the low-pressure cylinders. Scale control techniques employed in the 1960s restricted the extraction steam for the brine heater to a maximum pressure of about 35 psia (2.4 bar). The steam conditions at the crossover point vary with power output in the following manner:

Load	Steam Conditions
100%	2.2 bar (32.0 psia), 0.96% wet,
75%	1.7 bar (25.0 psia),
50%	1.3 bar (18.3 psia), 1.8% wet and
25%	0.7 bar (10.0 psia).

Steam conditions corresponding to 50% of the load were taken as the basis for the design of the MSF plant. An automatic throttling device would be used to maintain a constant steam inlet pressure to the brine heater.

Energy requirement of the MSF process consists of thermal energy for the brine heater and power needed for pumping. The thermal energy consumption for the 4550 m^3/day (1 MGD) plant attached to KANUPP has been evaluated in terms of its power raising value, while the pumping power requirement has been calculated as a function of the performance ratio. The total requirement has been expressed in terms of equivalent MW(e) power, Table 3.6.

TABLE 3.6. POWER CONSUMPTION (MWE) AND NORMALISED VALUES OF UNIT WATER COST FOR THE 4550 M^3/DAY (1 MGD) DESALTING PLANT

Gained Output Ratio	Performance Ratio		Power Equiv. to Thermal Energy	Pumping Power	Total Power	Normalised Unit Water Cost
	Lb/1000Btu	kg/kWthh				
4	3.82	5.91	4.60	0.48	5.08	1.0442
5	4.76	7.37	3.68	0.49	4.17	1.0088
6	5.72	8.85	3.08	0.53	3.61	1.0088
7	6.68	10.34	2.64	0.56	3.20	1.0000
8	7.65	11.84	2.32	0.59	2.91	1.0354
9	8.55	13.23	2.07	0.62	2.69	1.1150
10	9.51	14.72	1.75	0.65	2.40	1.1593

TABLE 3.7. OPTIMISATION FOR NUMBER OF STAGES FOR THE 4550 M^3/DAY (1MGD) DESALTING PLANT ATTACHED TO KANNUP

Cases	I	II	III	IV	V
No. Of Recovery & Rejection Stages	17, 3	19, 3	21, 3	22, 4	24, 4
Avg. LMTD, Rec. Stages, °C	6	6.22	6.44	6.5	6.56
Flash Temp. Drop per stage, °C	2.92	2.66	2.43	2.24	2.08
Circulation Ratio	11.35	11.35	11.35	11.35	11.35
Heat Transfer Surface, m^2.	7836	7471	7297	7321	7279
Tube Length (Stage Width), m.	6.1	5.2	4.6	4.3	4.0
MS Width, m.					
a)	0.55 (1–17)	0.55 (1–18)	0.55 (1–20)	0.55 (1–21)	0.55 (1–23)
b)	0.58 to 0.82 (18–20)	0.58 to 0.85 (19–22)	0.61 to 0.91 (21–24)	0.58 to 0.91 (22–26)	0.58 to 0.91 (24–28)
Brine Level, m.	0.37	0.43	0.46	0.46	0.46
MS Height above brine level, m.	1.83	1.91	1.98	1.98	1.98
Tube Plate Height, m.	1.83	1.83	1.83	1.83	1.83
Stage Height, m.	3.05	3.05	3.05	3.05	3.05
Stage Length, m.					
a)	1.04 (1–17)	1.04 (1–18)	1.04 (1–20)	1.04 (1–20)	1.04 (1–23)
b)	1.04 to 1.30 (18–20)	1.07 to 1.65 (19–22)	1.04 to 1.65 (21–24)	1.07 to 1.55 (22–26)	1.10 to 1.40 (24–28)
Total Steel Area required m^2.	1198	1274	1276	1291	1341
Total Normalised Cost of Recovery & Rejection stages:	1.0000	0.9878	0.9737	0.9761	0.9864

* Includes cost of steel, heat transfer surface and moisture separators.

This table also shows the normalised values of the unit water cost, leading to an optimum gained output ratio of 6.5 (or performance ratio of 9.6-kg condensate/kWthh or 6.2-lb distillate/1000 Btu).

As an innate advantage of the MSF process, it is possible to decrease the cost of heat transfer surface for a given performance ratio by increasing the number of stages. The optimum number of stages is a function of the relative cost of heat transfer surface and the cost of the chambers comprising the plant. The total cost of heat recovery and rejection sections has been determined for various numbers of stages, optimising the stage geometry to give minimum steel requirements in each case. Results are shown in Table 3.7.

The optimum number of stages was found to be 25, with a performance ratio of 9.3 kg/kWthh (6 lb/1000 Btu). Details of the chamber geometry with a cross-tube design were worked out for the final design as in the optimisation studies. Some of the main specifications of the optimised desalting plant are summarised in Table 3.8.

TABLE 3.8. MAIN SPECIFICATIONS OF THE 4550 M^3/DAY (1 MGD) OPTIMISED DESALTING PLANT ATTACHED TO KANUPP

Desalting Plant Type	MSF-BR, Cross Tube Design
Fresh Water Output	4550 m^3/day (One million imperial gallons per day)
Product Quality	Less than 50 PPM total dissolved solids (TDS)
Raw Sea-water Condition	38 500 PPM
Recorded Sea Water Temperature	18–33 ºC (65 ºF–92 ºF)
Design Sea Water Temperature	32.2 ºC (90 ºF)
Energy Source	1.3 bar (18.3 psia) steam, 1.8 per cent wet (106.3 ºC or 223.3 ºF)
Brine outlet Temperature	98.9 ºC (210 ºF)
Total Flash Temperature Drop	40.6 ºC (105 ºF)
Performance Ratio, R	9.3 kg/kWthh (6 lb. of distillate per 1000 Btu of heat input)
Optimum No. of Stages	25 (4 rejection stage)
Scale Control Method	Acid Pre-treatment and pH Control

3.3.3. Design study for large dual-purpose nuclear desalination plant

The PWR cycle for the study assumed the initial condition of steam to the turbine inlet as 700 psia (48.3 bar), 0.25% wet. Input data for steam cycle calculations were also listed in the study and used in obtaining the saturated steam cycle expansion lines. The dual-purpose scheme consisted of a back pressure (BP) turbine generating electricity corresponding to the rated capacity of the water plant; there is a steam by-pass around this turbine to compensate for the variations in LP steam flow to the evaporator and fluctuations in power demand from the BP turbine, besides allowing for the operation of evaporator when BP turbine is out of order, and a condensing turbine (optional) to provide for the power requirement over and above the back-pressure turbine capacity (Figure 3.10.).

TABLE 3.9. RESULTS OF COST OPTIMISATION STUDY FOR NUCLEAR DESALINATION PLANT 400 MW(e) NET, 100 MGD)

Sr. No.	Top Brine Temperature ºC	Performance Ratio, R kg/kWthh	Reactor Output MWth	Turbine output (MW(e))			Figure of merit
				BP	Condensing	Total	
1	99	11.0	2210	490.4	-	490.4	1.0000
2	93	11.5	2122	490.3	-	490.3	1.0006
3.	104	10.4	2298	490.7	-	490.7	1.0036
4.	91	11.8	2088	490.2	-	490.2	1.0039
5.	110	9.9	2396	491.2	-	491.2	1.0113
6.	116	9.3	2515	491.8	-	491.8	1.0245
7.	116	14.0	2202	328.0	169.7	497.7	1.0283
8.	110	14.6	2135	334.6	163.6	498.2	1.0292
9.	121	13.5	2265	323.0	175.0	498.0	1.0295
10.	104	15.9	2056	323.0	176.5	499.5	1.0404
11.	99	16.4	2007	328.5	171.4	499.9	1.0525

A computer programme was developed to find the optimum conditions for the given water and power outputs, 454 600 m^3/day (100 MGD) and 400 MW(e) net respectively. The method, in brief, involved finding, by the process of iteration, the performance ratio for the backpressure only case and then varying it on either side so that steam by-pass or condensing turbine could also be included. Knowing the reactor thermal rating, the gross power outputs from T/G islands and auxiliary power consumption, the programme then used the parametric cost equations to find out the capital and total annual costs for each performance ratio. A search was then performed to find out the performance ratio yielding the lowest annual cost for a particular top brine temperature, which later was then varied in the range of 90–125ºC (195 to 250 ºF). Finally, by comparing these optima for various temperatures, the most optimum temperature corresponding to the minimum annual cost could easily be ascertained.

As was expected, the minimum point for a given back-pressure was generally in the BP turbine-only region with only one exception namely the 121ºC (250ºF) brine temperature case, which had a minimum in the BP condensing zone of operation. For the above stated water/power ratio, the use of the highest brine temperature 121ºC (250ºF) did not yield the lowest annual cost, which occurred at 99ºC (210ºF) with optimum R of 11 kg/kWthh (7.1 lb distillate/1000 Btu).

The trend for lower optimum R and corresponding brine temperature continued for larger electrical capacity. Thus for 500 MW(e) net and 454,600 m^3/day (100 MGD) outputs, the most optimum point would occur at about 91ºC (195ºF) (R = 9.8 kg per kWthh or

6.3 lb/1000 Btu) for BP turbine only case. In this case as well, local minima would occur in the BP condensing mode for respective temperatures of 116ºC (240ºF) and 121ºC (250ºF).

When the T/G size was reduced to 300 MW(e) net, with the same water output, the optima for all temperatures occurred in the BP only case and the most optimum point was at 110ºC (230ºF) (with R = 12.5 kg per kWthh or 8.10 lb/1000 Btu). Increasing the water/power ratio raised not only the optimum performance ratio but also the brine temperature at which it occurred.

Table 3.9 gives the results of a computer optimisation study for various top brine temperatures, involving BP turbine only cycle (cases 1–6) and combined BP-condensing turbine cycle (cases 7–11). Obviously the annual costs would be slightly higher for near optimum cases as compared to the most optimum BP-turbine only case.

A typical near optimum case corresponding to the top brine temperature of 116ºC (240ºF) (R = 9 lb/1000 Btu or 13.9 kg/kWthh) was subsequently used for the detailed design calculations for 4546 m^3/day (1 MGD) and 45 460 m^3/day (10 MGD) desalting units of the cross-tube, MSF-BR type. Total number of stages for the optimum design was 36 and 38, for the two sizes respectively, each having 4 rejection stages.

A number of supplementary slides (Figure 3.9–3.18) are also presented herewith to further illustrate the results of the study.

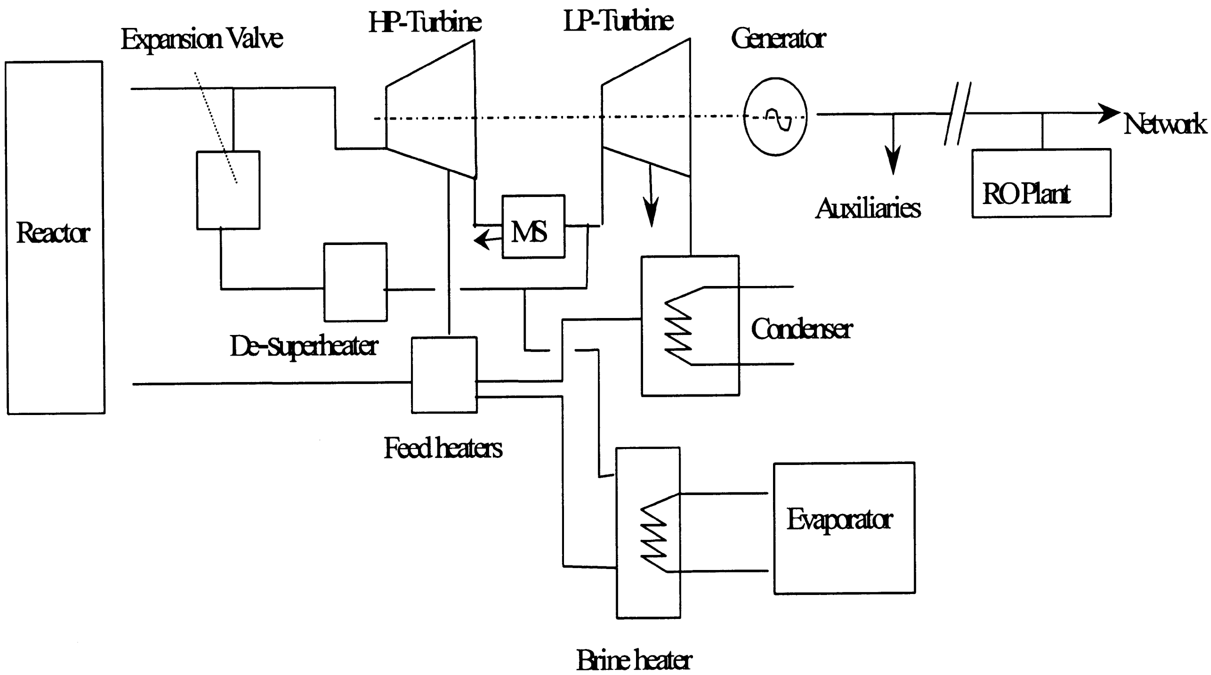


FIG 3.9. Nuclear desalination scheme with extraction condensing cycle and RO plant.

Text cont. on page 54.

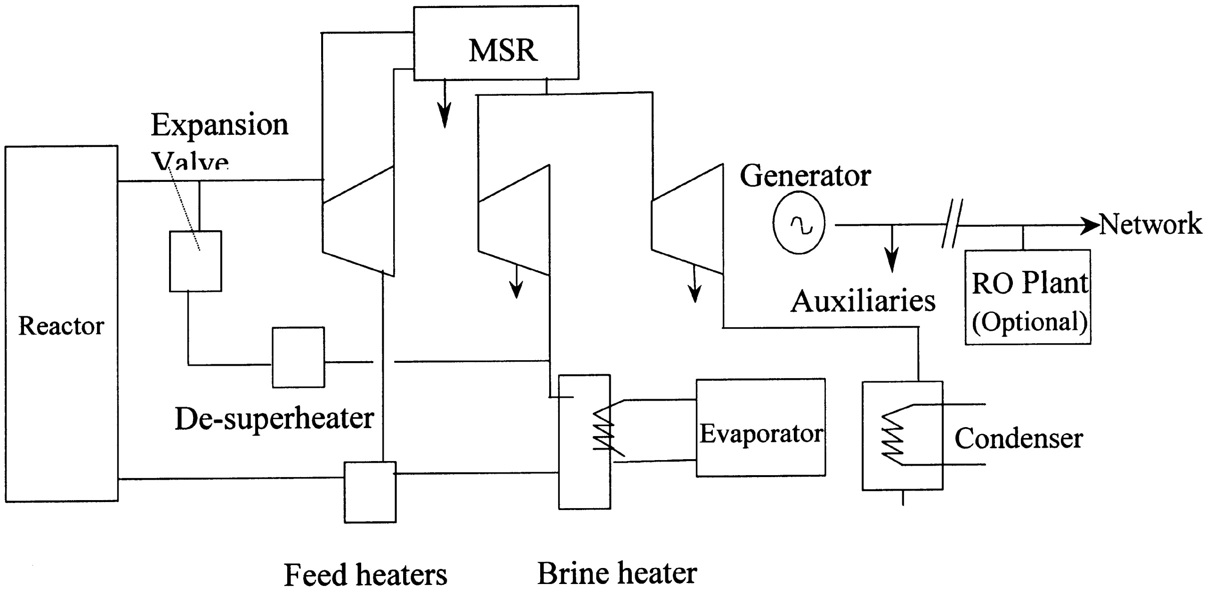


FIG. 3.10. Nuclear desalination scheme with BP-condensing cycle.

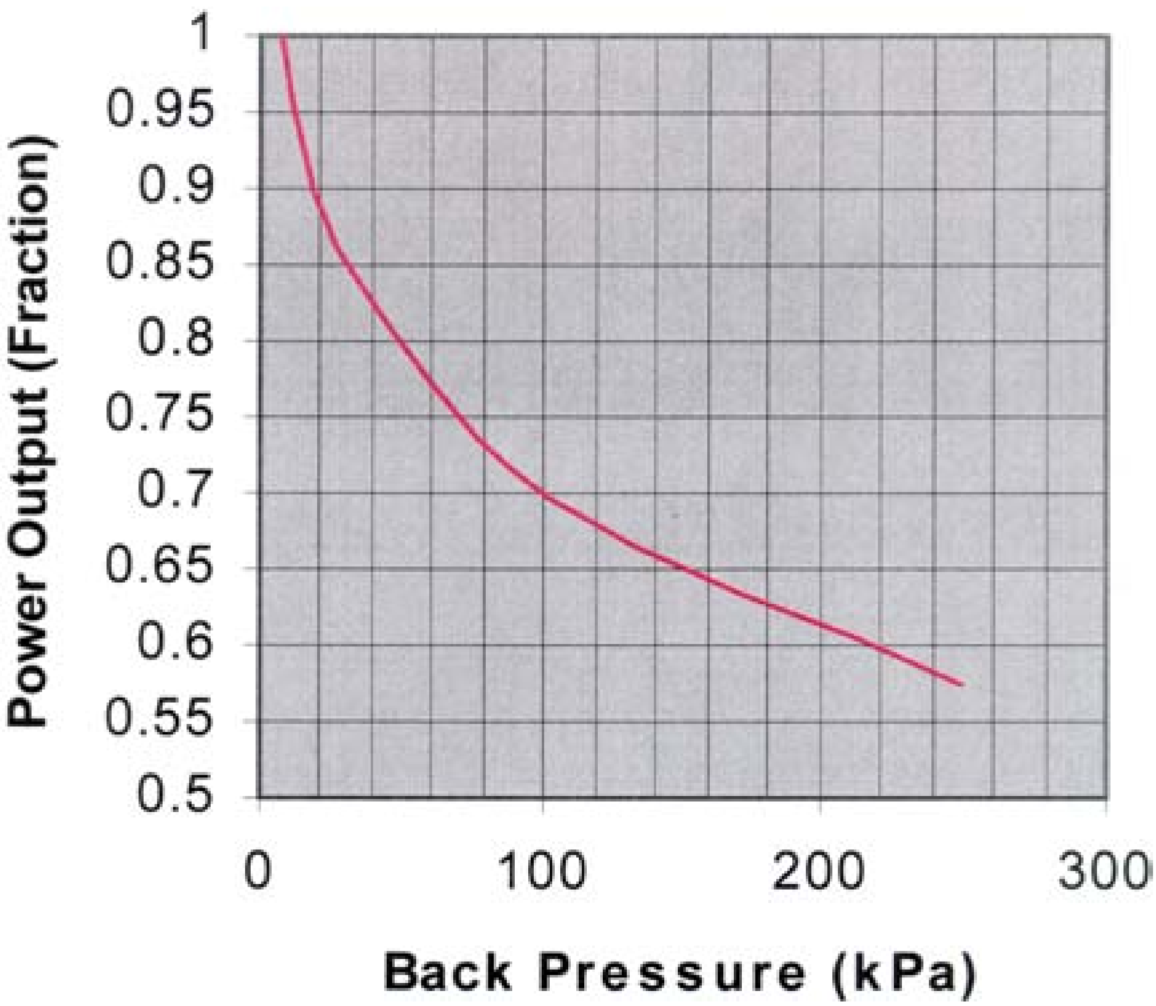


FIG. 3.11. PWR electric power outputs for different back pressures (main steam at 4.826 MPa or 700 psia, 0.25% wet).

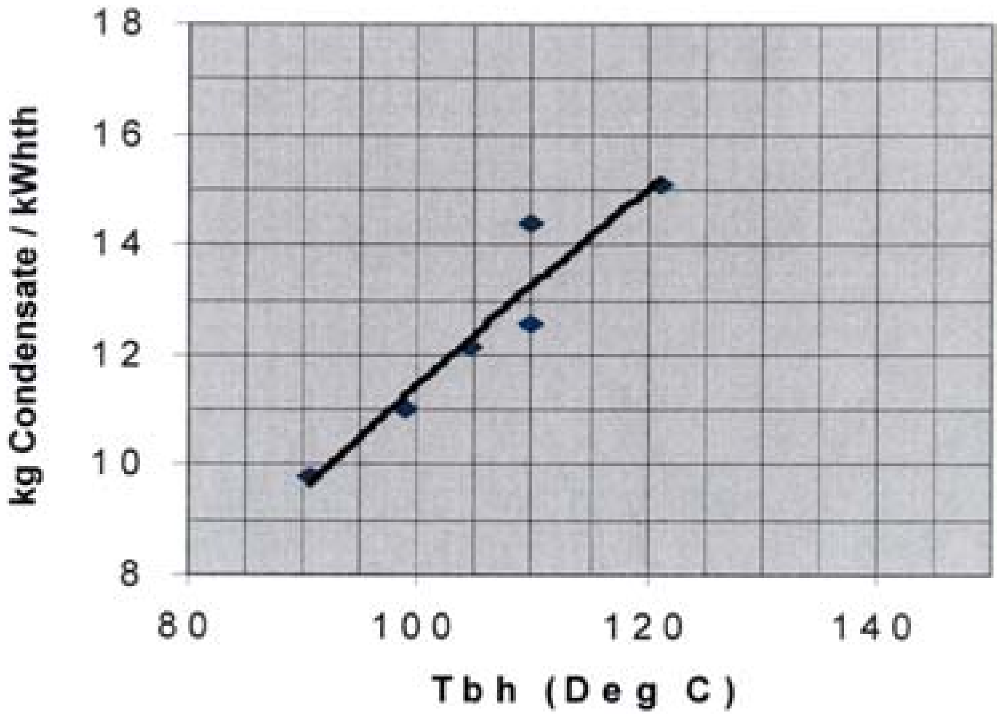


FIG. 3.12. Optimum performance ratios versus brine heater outlet temperatures.

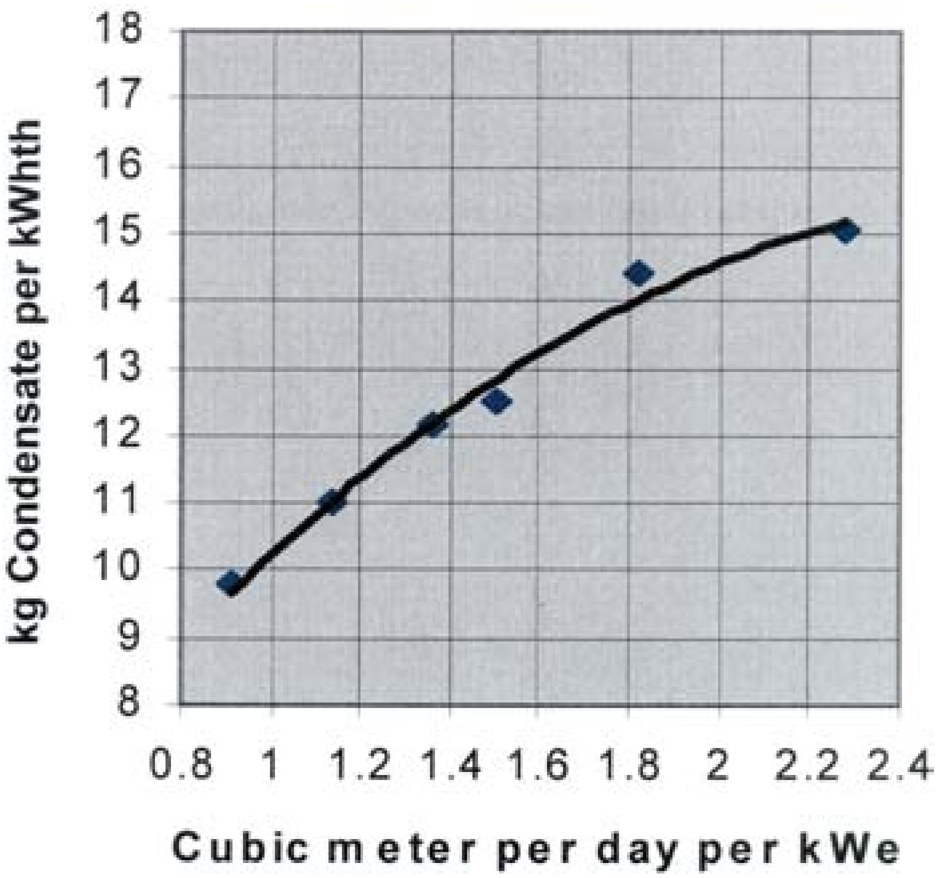


FIG. 3.13. Optimum performance ratios versus water to power ratios.

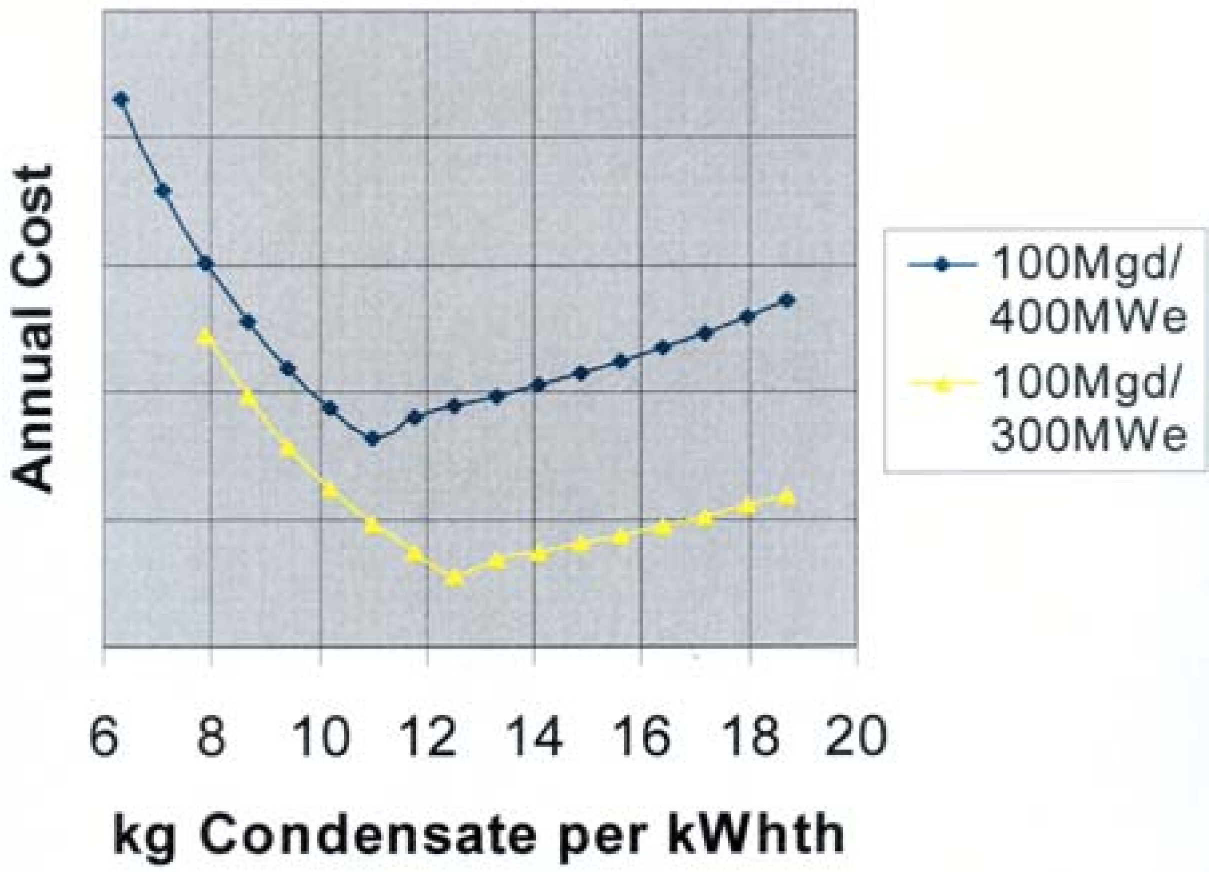


FIG. 3.14. Optimization study for BP, BP + bypass and BP + condensing modes [upper: 90.6℃ (195℉) and lower: 110℃ (230℉)].

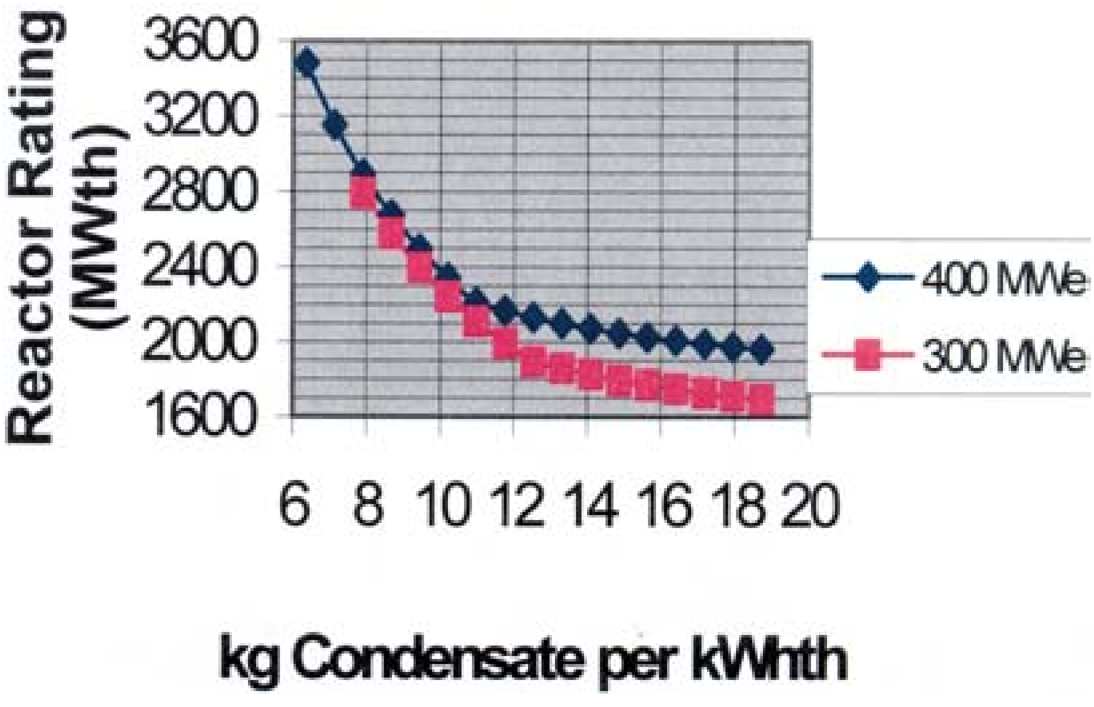


FIG. 3.15. Reactor ratings versus performance ratios.

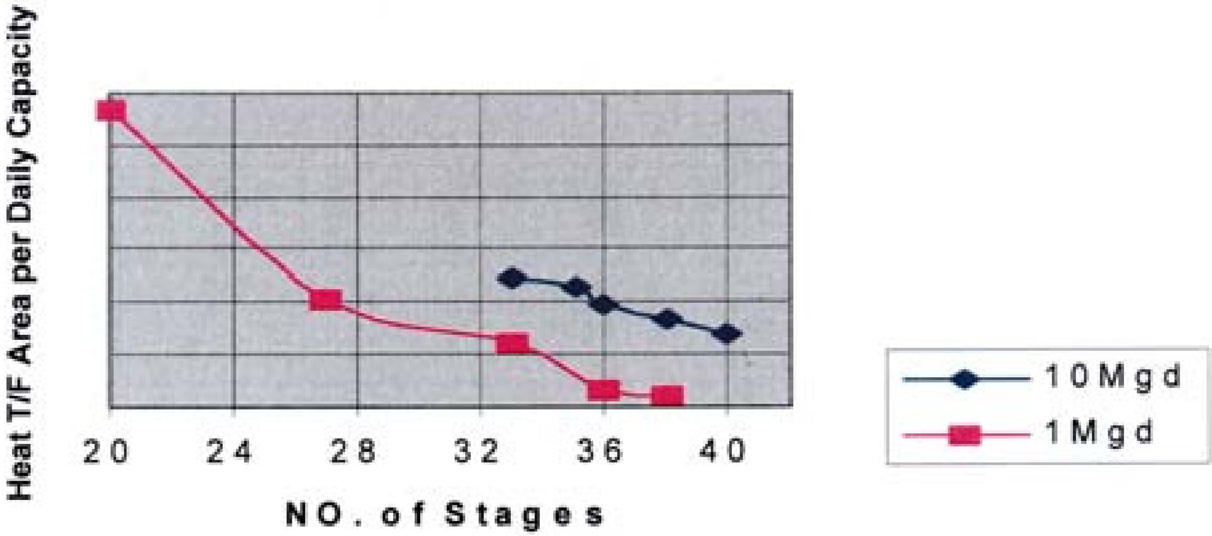


FIG. 3.16. Variation of specific heat transfer area with no. of stages for 1 Mgd and 10 Mgd (4550 m^3/day) unit sizes.

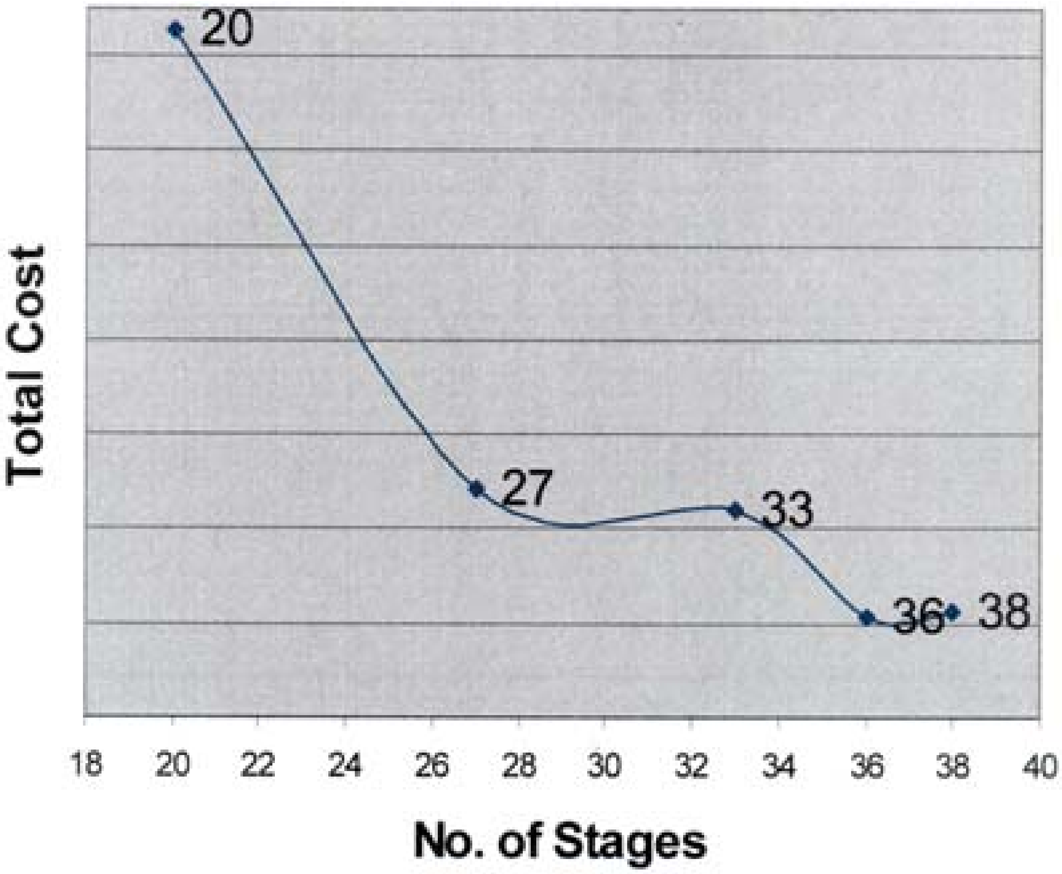


FIG. 3.17. Cost optimization for 1 Mgd (4550 m^3/day) unit.

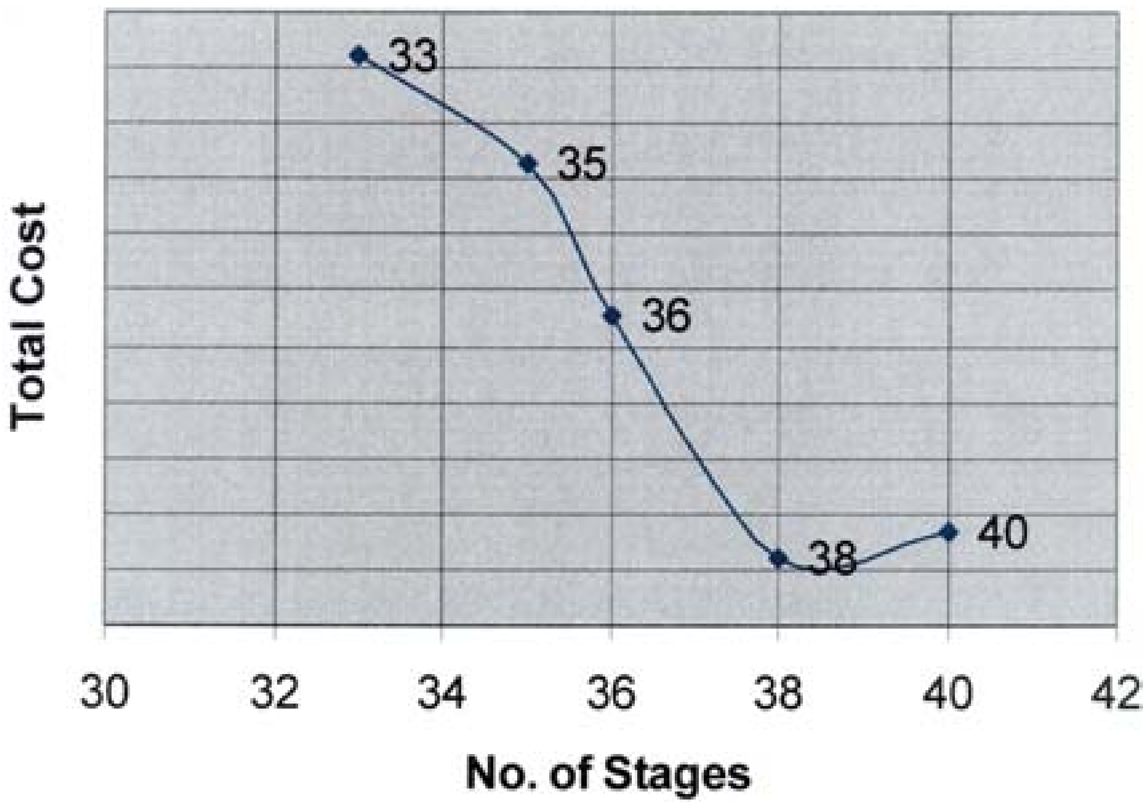


FIG 3.18. Cost optimisation for 10 Mgd (45 500 m^3/day) unit.

3.4. PHWR with MSF/RO (India)

3.4.1. Background

Bhabha Atomic Research Centre (BARC) is setting up a 6300 m^3/d combined MSF-RO nuclear desalination demonstration plant (NDDP) at the existing PHWR at the power station (Madras Atomic Power Station), Kalpakkam. The preliminary design and detailed design of this plant has been completed — and tenders with complete specifications for various equipments for the MSF and RO plants have been prepared and released. These are under various stages of fabrication and procurement. Civil works consisting of administrative building, RO plant, MSF plant, electrical sub-station, stores, etc., are in progress. The detailed specifications for the steam line, seawater intake and out-fall are under preparation for tendering the work.

3.4.2. Design description of the demonstration plant

Figure 3.19 and Figure 3.20 show the general layout of the demonstration plant and the schematic of the hybrid plant including the coupling with the power station (Madras Atomic Power Station).

Nuclear heat source

Two units of PHWRs have been in operation at Kalpakkam (Unit-1 since 1984 and Unit-2 since 1986). Both units produce gross outputs of 170 MW(e) each. The primary coolant (D_2O) in pressure tubes has a temperature of 293ºC at 87 bars at the core outlet. The secondary coolant (H_2O) transports heat to the turbine, from which steam at a pressure of 3 bars at 125ºC is extracted to the desalination system.

Desalination plant

Due to non-availability of raw seawater from MAPS, the seawater intake for the NDDP is considered from the process seawater out fall. The temperature is normally 3–5ºC higher than the ambient seawater temperature. As no significant modifications are possible in an existing reactor, it has been planned that the steam of around 3.5 bar pressure will be tapped from the manholes in the cold reheat lines after HP turbine exhaust from both of the nuclear reactors. The moisture content will be removed through a moisture separator and steam will be sent to an intermediate isolation heat exchanger (IHX) to produce process steam (using DM water) for the brine heater of the MSF plant. The condensate from the IHX will be returned back to the MAPS. The feed to the RO plant is taken from the MSF plant cooling water reject and is mixed with ambient temperature feed seawater resulting in a RO plant feed water temperature of 36–38ºC.

The water produced from the MSF plant will be of high quality with TDS of 10 ppm. Part of this water (1000 m^3/day) will be used as boiler make-up water for the nuclear power reactors. The remaining 3500 m^3/day of product water from the MSF plant will be mixed with 1800 m^3/day of product water from the SWRO plant and the mixed stream containing around 250 ppm TDS will be supplied to industrial/municipal use in Kalpakkam.

Desalination processes

Hybrid desalination system

The combined MSF-RO plant is envisaged to have a number of advantages viz.

(i) the RO plant will continue operation in order to provide the minimum quantity of water essential for drinking purposes during the shut down of the power station.
(ii) the return cooling seawater from the reject stages of the MSF plant is to be utilised (after blending with raw seawater) as feed for the RO plant for its enhanced throughput.
(iii) a part of high purity product water from MSF plant will be used for the make-up process water requirement (after necessary polishing operations) for the power station.
(iv) blending of the product water from both RO and MSF plants would provide requisite quality drinking water.

The 6300 m^3/d combined MSF-RO nuclear desalination project is located in between the existing 170 MW(e) PHWR station and proposed 500 MW(e) FBR at Kalpakkam, Chennai. The construction site for the project is shown in Figure 3.19. The process flow sheet of this plant showing the coupling of NDDP with NPP is shown in Figure 3.20.

The steam at around 3.5 bar pressure will be tapped from the manholes in the cold reheat lines after HP turbine exhaust from both nuclear reactors. The moisture content will be removed through a moisture separator and steam will be sent to an intermediate isolation heat exchanger (IHX) to produce process steam (using DM water) for the brine heater of the MSF plant. The condensate from the isolation heat exchanger is returned back to the power station de-aerator section, while the condensate from the brine heater is sent back to the isolation heat exchanger. Adequate provisions for monitoring and control have been made for isolation of the steam supply in case of shut down of the reactor and/or the desalination plant. A separate steam source directly from the reactor is to be utilised for ejectors of the MSF plant (after passing through another isolation heat exchanger).

For feed seawater to the project, two alternatives have been envisaged out of which one would be selected soon after the ongoing detailed techno-economic studies are completed. One source is the return process seawater, which will be delivered through a concrete conduit under gravity to the sump of the desalination plant from where it will be pumped to the MSF plant. Alternately, the possibility of an exclusive intake for supplying raw seawater is also being investigated.

The return stream of cooling seawater from the reject stages of the MSF plant will be blended with raw seawater to bring down the temperature to 36–38ºC before it is sent to the pre-treatment section of RO module.

The pre-treatment scheme for the MSF plant involves acidification, vacuum de-aeration for control of O_2/CO_2 concentration, pH control by alkali neutralisation followed by antifoam dosing. The pre-treatment for the RO plant is presently conventional using chlorination, clarifloculation, sand filtration, acid dosing, anti-scalant dosing, dechlorination and 5μ cartridge filtration. At a later date partial adoption of ultra-filtration is also is envisaged.

MSF plant

The MSF desalination plant has been designed to produce 4500 m^3/day of desalted water of a very pure quality from seawater. The plant is based on a long tube design concept, a re-circulation type with an acid dosing system, a top brine temperature around 120ºC and a Gain Output Ratio (GOR) of 9. As per design, seawater enters the heat reject section at 30ºC and comes out at 40ºC. As BARC is coupling the desalination plant to an existing PWHR, the seawater intake for NDDP is available from the process seawater outfall which is normally 3–5ºC higher than the ambient seawater temperature. The effect of the seawater temperature and heat reject section condenser coolant temperature rise on the production of the MSF plant is given in Table 3.10.

The thermal coupling of the MSF and PHWR plants consists of an isolation loop between the steam of the nuclear power reactor (PHWR) and the brine of the MSF desalination plant. The isolation loop has been provided for eliminating any possibility of radioactive contaminants penetrating the desalination plant or the atmosphere. In the case of MSF desalination, the brine heater also serves as an additional barrier.

The brine in the brine heater is maintained at higher pressure than the heating fluid for pressure reversal so that direction of the leakage, if it occurs, will be from the desalination system and not into it. The isolation loop consists of a closed loop between the nuclear steam and the MSF desalination plant. In the isolation loop, the steam is condensed, transferring its heat to another heat transfer medium, which is used to heat the brine. The heat transfer medium in this case is boiling water, generating steam in the shell side at around 3 bar.

The nuclear steam at around 3.5 bar is used in the tube side of the system. The isolation loop provides enhanced safety at the cost of electrical power loss due to extraction of steam at 3.5 bar rather than 3.0 bar for the desalination plant.

TABLE 3.10. EFFECT OF SEAWATER TEMPERATURE AND HEAT REJECT CONDENSER COOLANT TEMPERATURE RISE ON DESALTED WATER PRODUCTION

Sr. No.	Condenser coolant temperature rise (ºC)	Coolant sea water temperature (ºC)		Flashing brine temperature entering heat reject section (ºC)	Seawater coolant flowrate (m^3/h)	Production rate (m^3/h)
		In				
1.	10	30	40	47.8	1160.9	187.88
2.	10	32	42	49.8	1160.9	183.76
3.	10	34	44	51.8	1160.9	179.63
4.	10	36	46	53.8	1160.9	175.47
5.	9	30	39	46.8	1289.91	189.93
6.	9	32	41	48.8	1289.91	185.85
7.	9	34	43	50.8	1289.91	181.69
8.	9	36	45	52.8	1289.91	177.56
9.	8	30	38	45.8	1451.15	191.97
10.	8	32	40	47.8	1451.15	187.87
11.	8	34	42	49.8	1451.15	183.77
12.	8	36	44	51.8	1451.15	179.63
13.	7	30	37	44.8	1658.46	194.0
14.	7	32	39	46.8	1658.46	190.0
15.	7	34	41	48.8	1658.46	185.85
16.	7	36	43	50.8	1658.46	171.71

The cold seawater from the outfall system of MAPS is pumped at a rate of 1450 m^3/day through the tube bundle of the heat reject section (3 heat reject stages). Before it passes through the tube bundle of reject stages, a part of it (94 m^3/h) is used in pre- and inter condensers. The remaining part of warm seawater (1075 m^3/h, 40ºC) from the reject module is sent back to the sea and only 375 m^3/h of warm seawater (40ºC) is subjected to chemical dosing and is sent to the vacuum de-aerator.

The chemical dosing consists of addition of hydrochloric acid to decompose bicarbonates so as to prevent alkaline scale formation on heat transfer surfaces. In the 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 de-aerated 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 the flashing of brine. The de-aerated feed is then mixed with recycled brine. It is then passed through the tube bundle of the recovery module (9 nos) where it is heated externally by condensation of flashed water vapour. The temperature of recycled brine is raised to 112ºC, which is then further heated to 121ºC in the brine heater. This brine is then gradually passed through all 36 stages where it gets flashed and vapour is produced, which is then condensed on the outside of the tubes and form the product water. Recovery modules are rectangular in shape, with a long tube design and are arranged in the form of a train. There are a total of 9 recovery modules, each with 4 brine stages, made up of carbon steel with a 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; tube sheets are made of 50 mm thick 90/10 cupronickel.

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

RO plant

The SWRO plant will receive hot seawater of 35000 ppm from the MSF plant cooling water reject and produce potable water of about 500 ppm. The SWRO product water will be mixed with highly pure MSF water to make drinking water of 200–300 ppm. The SWRO plant will use hot seawater at a temperature of about 36–38ºC as feed. High temperature feed will increase the membrane flux considerably, which will in turn reduce the membrane cost for a particular plant capacity.

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

This pre-treated seawater is then pumped in two parallel sections through the modules at a rate of 110 m^3/h each at a pressure of 40 bar. Each pump is fitted with an 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 seawater membrane is 8040 HSY SWC or equivalent with TFC spiral wound membranes with solute rejection of 99.6%. There are a total of 26 modules with a total of 156 membrane elements. Each pressure tube has 6 elements. The 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.

Table 3.11 and Table 3.12 describe the specification of the MSF and RO plants [3.4.1].

3.4.3. Project Schedule and Major Milestones

Project Schedule

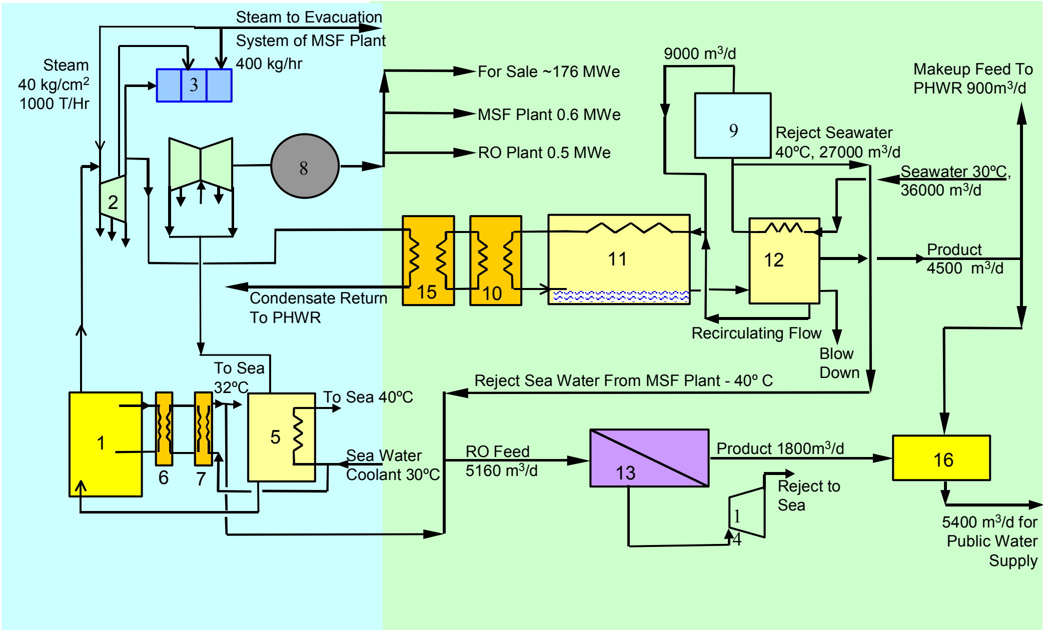
- 1997 — Project is approved by the government of India.
- 1998 — The preliminary and detailed designs were completed and technical specifications of major equipment were prepared. Tenders for a few items were released. During this year the PSAR was prepared, reviewed and approved by the Safety Committee.

– 1999 — The civil work started and most of the buildings housing the MSF and RO plants as well as administration are nearing completion. Tenders for all major equipment were released.
– 2000 — The detailed specifications for the steam supply line and FSAR have been prepared and the tenders released.
– 2002 — Commissioning of RO section is expected.
– 2003 — Commissioning of MSF section is expected.

TABLE 3.11. TECHNICAL SPECIFICATIONS OF 4500 m^3/DAY (1 MGD) MSF PLANT

(i)	Plant capacity	187.5 m^3/h
(ii)	Product quality	< 25 ppm of salt
(iii)	Top brine temperature	121^0C
(iv)	Blow down temperature	40^0C
(v)	Performance ratio	9
(vi)	Steam consumption	20.6 Te/h
(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)	No. of recovery modules	9
(b)	No. of flash stages/module	4
(c)	No. of reject module	1
(d)	No. of stages	3
(e)	Total no. of flash stages	39
(x)	Tubes	Cupronickel 90/10
(xi)	Pumps	SS 316 make centrifugal pump

FIG 3.19. Construction site for Kalpakkam nuclear desalination project.



1. PHWR 170 MWe
2. HP Turbine
3. Moisture Separator / Reheater
4. LP Turbine
5. Power Plant Condenser
6. Moderator - DM Water Cooling Loop
7. DM Water - Sea Water Cooling Loop
8. Generator
9. MSF Plant Chemical Pretreatment Section
10. MSF Plant Brine Heater
11. MSF Plant Heat Recovery Section
12. MSF Plant Heat Reject Section
13. RO Plant
14. RO Plant Energy Recovery Turbine
15. Intermediate Heat Exchanger
16. Product Storage Tank

6,300 m^3/d MSF- RO DESALINATION PLANT COUPLED TO 170 MWe PHWR

FIG. 3.20.

TABLE 3.12. TECHNICAL SPECIFICATIONS OF 1800 M^3/DAY SWRO PLANT

(i)	Product out	75 m^3/h
(ii)	Product quality	500 ppm
(iii)	Feed sea water flow	215 m^3/h
(iv)	Feed sea water TDS	35000 ppm
(v)	Membrane element	
	(a) Type	TFC spiral wound
		8040 HSY SWC/TFC 2822 SS
	(b) Model	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

3.5. PHWR CANDU with preheat RO (Canada)

3.5.1. Background

In late 1993 Canada began active participation in the IAEA's potable water program. Due to pressing global need for a large scale, efficient energy water production facility, CANDESAL (a private Canadian business enterprise) began working with AECL (Atomic Energy of Canada Limited, a corporation overseen by the federal government) to develop an approach to the application of the Canada Deuterium Uranium (CANDU) reactor for seawater desalination.

CANDU 6 (700 MW(e) class) reactors have consistently ranked amongst the top 10 in the world for lifetime performance. Canada has operated Units successfully, as well as the Republic of Korea, and Argentina for more than 17 years. CANDU 6 units have the highest lifetime capacity factor within their class. The design of the CANDU makes it a safe and natural choice for coupling with potable water production as its inherent safety assures the quality of the product water for public consumption or industrial or agricultural use. Evolutionary improvements continue to be incorporated in the CANDU 6 design, taking advantage of CANDU operating experience, AECL research and development, and technical advances worldwide in order to further enhance safety, reliability and economics.

The focus of CANDESAL's early design concept development work was placed first on the determination of an appropriate seawater desalination technology for coupling to the CANDU. With no prior commitment to any particular technology, all options were open for consideration. The only prerequisite was that commercially available, well-proven desalination technologies be considered. Initial investigations indicated that with this prerequisite as a constraint, only two technologies warranted further consideration. These were multi-effect distillation and reverse osmosis. Accordingly, more detailed preliminary studies were carried out to evaluate the use of process steam from the nuclear steam supply system to provide the thermal energy for MED and the use of electrical energy generated by the nuclear plant to provide the pumping power for reverse osmosis. It was found that in order to match the required thermal conditions for MED, changes were required to the balance of plant design, which was considered expensive to implement and this led to a reduction in electrical generating efficiency. Moreover, the loss in electrical generating capacity was such that the combined water and electrical production capacity was not as great as that which could be achieved using RO combined with the standard CANDU design.

CANDESAL conducted more investigation into the use of RO system coupled with a CANDUa new approach to reverse osmosis desalination was developed in the process. The work expanded beyond the scope of nuclear desalination and now the CANDESAL system is a viable option for other desalination applications as well. The special features of the CANDESAL system, and in particular it's coupling to the CANDU reactor, are outlined below.

3.5.2. Design Description

The CANDU 6 reactor system's basic design features

The CANDU 6, described in more detail in [17], incorporates all of the basic and well-proven features, which are the hallmark of CANDU. These include:

- A reactor consisting of small diameter horizontal pressure tubes housed in a low pressure, low temperature moderator-filled calandria (tank)
- Heavy water (D_2O) for moderator and reactor coolant
- The standard CANDU 37-element CANDU fuel bundle, and the ability to operate on natural uranium or other low fissile content fuel
- On-power refuelling, to eliminate the need for refuelling outages
- Two diverse, passive, fast-acting and fully capable shutdown systems which are independent of each other, and of the reactor regulating system
- Automated digital control of all key Nuclear Steam Plant and Balance of Plant functions
- The total absence of all chemicals in the reactor coolant (Heat Transport System) for reactivity control.

Safety is assured in CANDU 6 through a defence in depth approach that builds on diversity and redundancy, and which takes advantage of the unique CANDU pressure tube reactor concept. Passive systems are used whenever they are shown to be reliable and economic; these systems are complimented by engineered systems. The consistent application of human factor principles, and detailed attention to all aspects of plant designs also contributed to CANDU 6 safety.

CANDU design practice places emphasis on the performance of the special safety systems. The design incorporates four special safety systems. These consist of the two passive, diverse, dedicated reactor shutdown systems; the emergency core cooling system; and the containment system. Each is separated from and — independent of — normal operating plant systems, including other safety systems. The initiation and operation of all special safety systems, if required, is fully automatic, based on diverse and redundant measurements. For example, two independent and diverse reactor trip (shutdown) signals are provided for each of the shutdown systems for every design basis accident requiring reactor shutdown.

The special safety systems themselves are:

- Independent of each other and of the normal control and process systems;
- Separated physically from each other, and from the control/process systems, so that common cause events cannot affect more than one safety system;
- Redundant, at both the system and active component level, so that isolated failures, either of active components or of an entire system, cannot disable the safety function;
- Testable during service, to meet a reliability target of 999 times out of 1000 tries;
- Diverse in design and operation, so that a generic fault in design, maintenance or operation cannot affect more than one safety system.

Other features include a number of passive safety features as well as a range of engineered systems, which contribute to CANDU 6 safety.

The CANDESAL reverse osmosis desalination system — key features

Preheated feed water is one of the key features of the CANDESAL design. Preheated feed water concept owes its origin to CANDESAL, and it's now a standard feature in many desalination systems being built today. The use of reactor plant condenser cooling water as a preheated feed stream for the desalination plant allows for substantial gains in fresh water production efficiency, resulting in reduced plant capital cost as well as reduced energy consumption per unit of water produced. In addition to the condenser cooling water, the unique design of the CANDU 6 allows for the use of waste heat from its moderator cooling water system giving an additional temperature rise to the RO system feed water. The benefits of this additional feed water preheat will be examined in a later section.

Ultrafiltration (UF) pre-treatment is used to provide high quality feed water to the RO process. This serves to protect the RO membranes and enhance their performance, thereby reducing the total number of RO membranes required and increasing their lifetime. The result is a reduction in capital cost of plant and subsequent reduction in the requirement for membrane maintenance and replacement.

Sophisticated analysis techniques drawn from reactor design experience are used in the CANDESAL desalination and cogeneration systems design. Drawing on the combined expertise of desalination system and nuclear power plant designers, the design is numerically modelled to allow design optimization and integrated system performance analyses. This comprehensive design optimization allows further performance enhancements and reduced water production costs, which are site specific and optimize the inherent advantages of the site, which vary depending on geographic location and quality of available water. Maximum use is made of energy recovery techniques. Much of the electrical energy consumed in RO desalination is used to pressurize the RO feed stream to the high operating pressures required for optimum performance. Since there is relatively little pressure drop through the RO membranes, a significant portion of this energy can be recovered, thereby reducing energy consumption and hence energy costs and water production costs. Many gains have been made in recent years in energy recovery and the CANDESAL system uses the latest energy recovery technology, which gives significant improvement in performance. This design concept was first presented at an IAEA Advisory Group Meeting on "Coupling Aspects of Nuclear Reactors with Seawater Desalination Processes" held in Vienna in September 1993 [18].

Design features of the integrated CANDU-CANDESAL plant

Introduction

Two critical issues facing nuclear desalination as a commercially viable technology are energy utilization and the cost of water production. It was recognized that improvements in the efficiency of energy utilization could be achieved by taking advantage of waste heat normally discharged from the reactor through the condenser cooling system. Use of the condenser cooling water as well as the moderator cooling water as preheated feed water to the RO system improves the efficiency of the RO process, and therefore the economics of water production. A strong emphasis has been placed on the integration of the energy and water production systems into a single, optimized design for the cogeneration of both water and electricity.

This approach to the integration of seawater desalination systems with nuclear reactors has the advantage of maximizing the benefits of system integration while at the same time minimizing the impact of physical interaction between the two systems. In essence, the reactor operates without "knowing" that there is a desalination plant associated with it. Transients in the desalination plant do not have a feedback effect on reactor operation. This is extremely important, since there must be a high degree of assurance that unanticipated operating transients in the desalination unit do not have an adverse impact on either reactor safety or operational reliability. Conversely, it would also be undesirable to have reactor shutdowns, whether unanticipated or for planned maintenance, that would require shutdown of the water production plant.

As the CANDESAL nuclear desalination and cogeneration system design concept has evolved, it has developed in a direction which allows use of a standardized off-the-shelf CANDU reactor without modification, while at the same time accruing significant benefits from the systems integration due to improved performance characteristics and energy utilization. A schematic of the RO desalination system is shown in Figure 1, showing its feed stream drawn from the reactor's condenser discharge stream.

The benefits of moderator cooling for additional feedwater preheat

A feature of the CANDU 6, which makes it ideally suited to reverse osmosis desalination, is the moderator cooling system. The waste heat from the moderator cooling system can be used to preheat the feed water of the RO plant in addition to the waste heat from the condenser cooling water. Other reactors, which do not offer this source of waste heat, are restricted to the availability of waste heat from the condenser cooling water — Analysis — carried out by CANDESAL shows that this additional source of waste heat can be used for further increase in temperature rise of the feed water stream by as much as 9ºC under the design conditions for this specific case. This translates into significant performance improvement, as discussed below.

Tritium Release Considerations

Any coupling of a nuclear power plant with a desalination plant producing potable water requires special examination of the safety issues involved. AECL carried out an evaluation of a combined CANDU-CANDESAL Nuclear Desalination Facility; the results are presented in Reference [19]. The conclusion of that work is that there is no additional exposure to the public when desalination is combined with nuclear power.

Benefits Of Cogeneration

There are benefits arising from the use of the standardized CANDU reactor design. Firstly there is the use of waste heat from the nuclear power generation process and the ability to optimize the overall system design, while other benefits — result from an innovative cogeneration systems design. A fully integrated cogeneration design based on co-located nuclear energy and desalination systems allows for shared land acquisitions and commonality of many on-site facilities including water intake and outfall structures, staff, maintenance as well as administrative facilities. These all have clear economic benefits. Fresh water and electrical transportation costs may also be reduced through the use of common rights-of-way to bring these two resources to their markets. By designing the power plant and desalination facility to operate independently even though they are thermally coupled, show that the CANDESAL system allows for flexibility of phased increases in the size of the desalination plant with no collateral requirement to modify the power plant.

Additionally, coupling the reactor with the desalination system in this manner provides the flexibility of varying water production without adversely impacting the operation of the power plant. The CANDU nuclear power plant can be operated at maximum electrical production efficiency, while the desalination plant is operated so that fresh water production meets or exceeds requirements under various operating conditions, including annual variations in site specific feed water conditions and daily variations in demand. During periods where the power plant is off-line and the feed water preheat is unavailable to the desalination plant, the desalination process can still continue, at a reduced efficiency. Through this combination of design and performance optimization, the unique electrical and thermal coupling of the energy source and desalination system, shows significant improvements in water production efficiency and reductions in desalination plant capital cost. The result is, a reduction in levelized water production costs. Although the costs for any given facility are highly specific to the site, seawater conditions, and other design requirements, detailed cost assessment models nevertheless indicate that savings typically on the order of 10–15% in plant capital cost and 10–20% in water production costs are achievable [20].

3.5.3. Economic perspectives

DEEP Calculations — modifications required

The IAEA's programme for economic evaluation of nuclear desalination plants, Desalination Economic Evaluation Programme (DEEP) [21] does calculations for stand-alone and contiguous RO plants that are co-located with a reactor system. Although the stand-alone plants do not really constitute nuclear desalination as they connect directly to the grid for their power source, they do provide a basis for comparison to illustrate the benefits of co-locating the systems. DEEP accommodates contiguous designs, and the calculations of RO performance characteristics are based on correlations derived from the performance of stand-alone plants and therefore do not properly represent the effects of using preheated feed water in the RO system. In addition, the correlations represent "typical" performance characteristics derived from the operation of a wide variety of systems. In order to carry out economic analysis that properly accounts for specific membrane performance characteristics operating in an optimised system design, modifications to the DEEP code are required.

To represent the performance and economic characteristics of the CANDU-CANDESAL system, operating as an integrated nuclear desalination system and taking maximum advantage of waste heat from the condenser and moderator cooling systems, modifications to DEEP were made to bring the spreadsheet calculations in line with design code projections for the system. These changes included changes to the energy and water plant input data that are not normally allowed by DEEP, as well as modifications to the basic calculational modules themselves.

Changes to power plant input

Rather than use the default data supplied by DEEP for the calculation of the performance of the nuclear power plant, actual CANDU 6 performance and economic data was entered (see Table 3.13). The Table lists only those data that differed from the DEEP default. The changes made were not to the evaluation programme itself, but rather to the input data used as a basis for the DEEP economic analysis.

TABLE 3.13. INPUT DATA TO POWER PLANT SECTION IN DEEP

Data Definition	DEEP default data	CANDU 6 data
Additional site related construction cost	214 $/kW	168 $/kW
Construction lead time	60 m	52 m
Operating availability	0.801	0.856
Planned outage rate	0.100	0.038
Plant economic life	30 years	40 years
Reference energy plant net output	450 MW	668 MW
Reference net thermal efficiency	29.7%	30.9%
Specific construction cost	2,140 $/kW	1,677 $/kW
Specific decommissioning cost	1.00 $/MW(e).h	0.72 $/MW(e).h
Specific nuclear fuel cost	4.35 $/MW(e).h	2.49 $/MW(e).h
Specific O&M cost	11.00 $/MW(e).h	5.82 $/MW(e).h

Changes to reverse osmosis desalination plant input

For the purpose of economic evaluation of the CANDU-CANDESAL system, DEEP was modified based on performance characteristics from actual design code analysis results for a specific case. This method allows the calculations to properly represent the effect of an increased RO feed water temperature and the performance improvements in water production and economics that go along with it. Changes to RO system performance correlations to represent increased water production as a function of increasing feed water temperature, and cost contributions based on water production rate were adjusted accordingly.

RO system input data was entered into the reverse osmosis section based on analysis of a specific case with 25 °C seawater having a TDS of 38 500 ppm and a feed flow suitable for producing about 100 000 m^3/d of potable water at ambient seawater temperature. Results of that analysis are presented below.

TABLE 3.14. INPUT TO REVERSE OSMOSIS SECTION IN DEEP

Data Definition	DEEP default data	CANDESAL analysis
Average annual cooling water temperature	21°C	25°C seawater temperature
Contiguous RO design cooling water temperature	31°C	25, 35 and 44°C
Desalination plant optional unit size specification	24 000 m^3/d	12 000 m^3/d
High head pump pressure	Calculated by DEEP	69 bar
Recovery ratio	Calculated by DEEP	0.388, 0.427 and 0.453 based on RO system calculations
Required water plant capacity at site	User input, m^3/d	100 000 m^3/d
Stand-alone RO design cooling water temperature	21°C	25°C

Performance and economic improvement with preheated feedwater

Using the CANDESAL system, significant reduction in water costs was achieved through the optimisation of the systems design. This took advantage of the rise in temperature from the total waste heat available from the CANDU reactor, including the water quality range as given in the WHO standards, and maximum membrane flux as allowed by the manufacturers, with other features of optimization when designing a CANDESAL system.

The main performance characteristics are shown in Figures 3.22–3.25, which show the installed daily water production capacity, the relative water production, the RO system recovery ratio and the product water quality (ppm) as a function of temperature. The economic analysis was carried out using both the standard version of DEEP and the CANDESAL modified version that accounts for the proper treatment of RO systems with feed water preheating and with performance characteristics

DEEP calculations were done using the same energy plant data for each case, and two sets of calculations were done for comparison. The first used DEEP as it is currently programmed with the default RO performance calculations. While Cases were ran for a stand-alone plant and a contiguous plant, which is co-located with the reactor but not coupled with it, so that the feed temperature to the RO plant was still at seawater temperature. Calculations were then run for two contiguous plants assuming that the feed water was drawn from the condenser cooling discharge at 35°C and 44°C. All of the other conditions remained constant and unmodified in DEEP. The energy plant characteristics were the same for all of the cases, based on the data for the CANDU reactor, as listed in Table 1. The second set of economic analyses evaluated the same four cases (stand-alone, contiguous at ambient seawater temperature, feed water preheat at 35°C and feed water preheat at 44°C) using the modified version of DEEP, which was changed to properly represent the effects of RO preheat. The results of both sets of economic analyses are shown in Figure 3.26.

Analysis results from the first 2 cases provided an informal code validation of the DEEP modifications. Significant changes in the results between the default DEEP code and the modified code would not be expected because in both the stand-alone case and the contiguous case, which receives its feed water from the intake canal for the power plant at ambient seawater temperature, the RO system would be operated at seawater temperature. Indeed, no significant changes resulted when the modified code was for those cases using ambient seawater temperature the modified code and the standard DEEP code produce very similar results. This validates the changes made to the code and demonstrates that errors were not inadvertently introduced through modifying the code to accommodate the effects of preheated feed water.

The second two cases, for RO operating temperatures of 35°C and 44° (leaving all other parameters the same) show that there are some significant changes when proper accommodation has been made for preheated feed water through modification of the DEEP code. The calculations show significant economic improvement as the feed water temperature rises. In addition, the difference between preheat using just the condenser cooling water, and that using the additional waste heat available from the CANDU's moderator cooling system (calculated to be an additional 9°C for this site specific case) results in even further savings in water production costs. Interestingly, the default DEEP cases do not follow the same pattern, which suggests that there may be faults with RO performance correlations beyond just their lack of ability to model preheat conditions.

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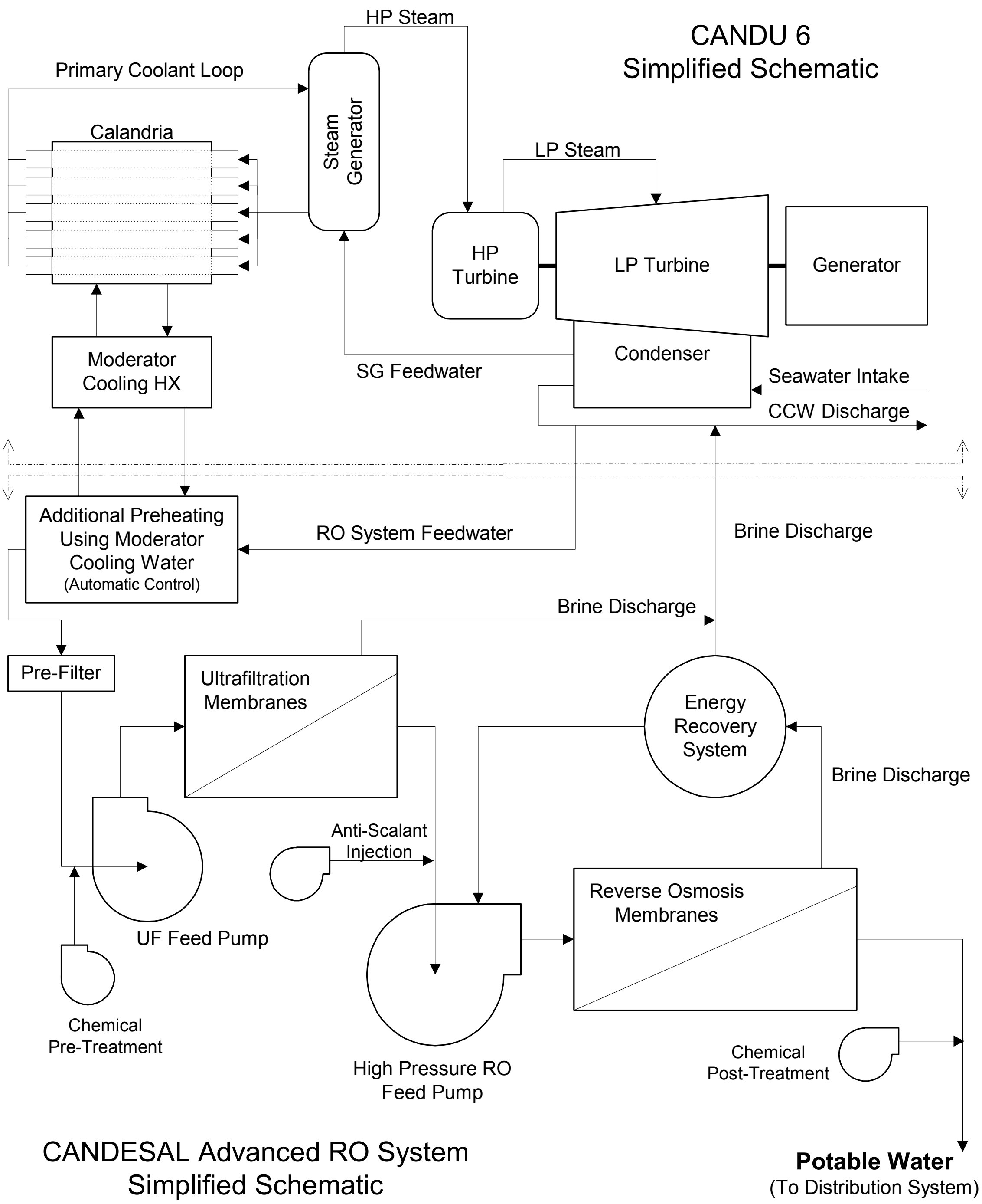


FIG 3.21. Simplified schematic of CANDU-6 CANDESAL nuclear desalination system.

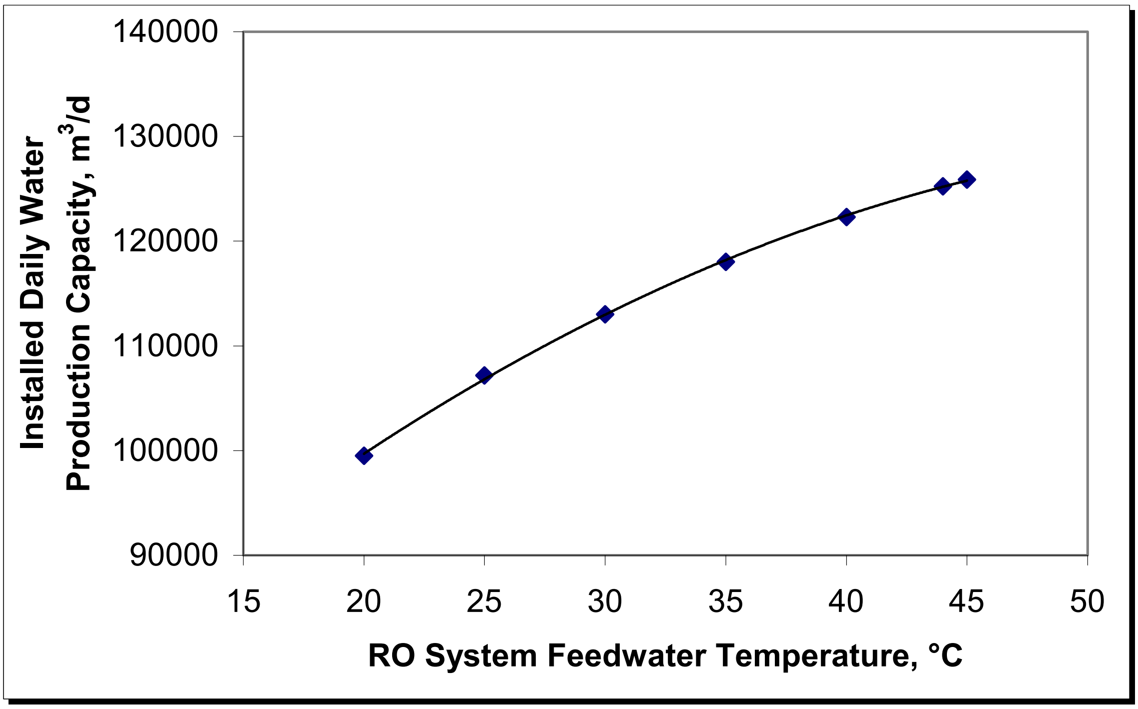


FIG.3.22. Installed daily water production capacity as a function feedwater temperature.

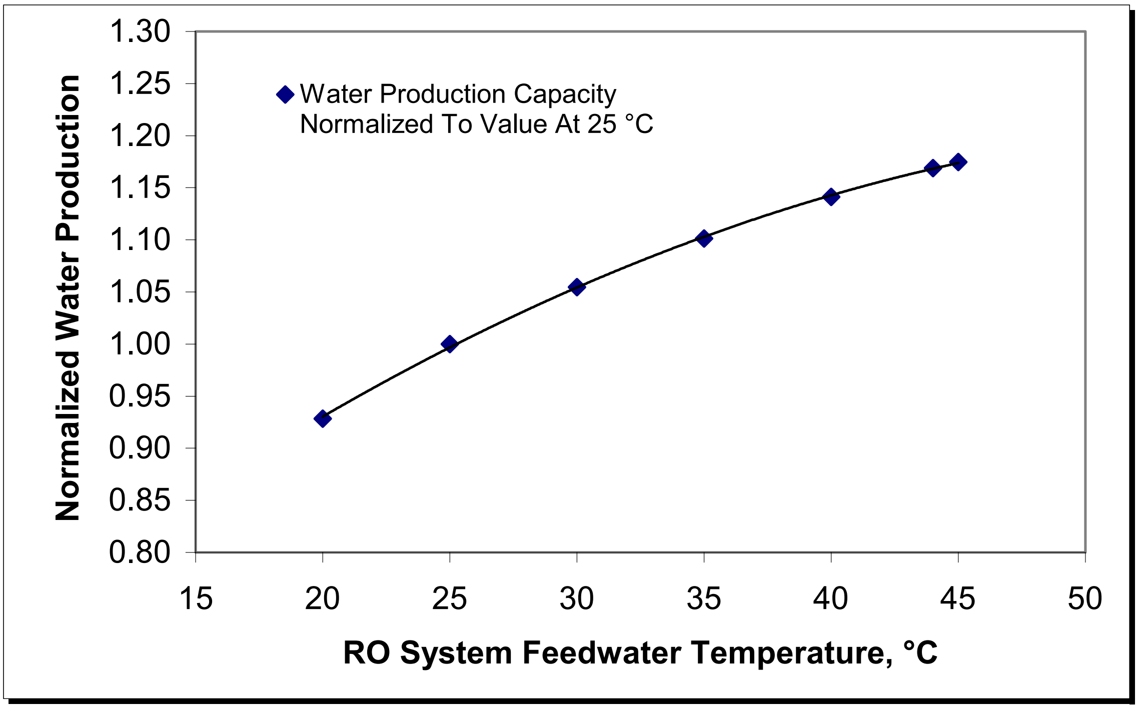


FIG. 3.23. Normalized water production as a function of RO feedwater temperature.

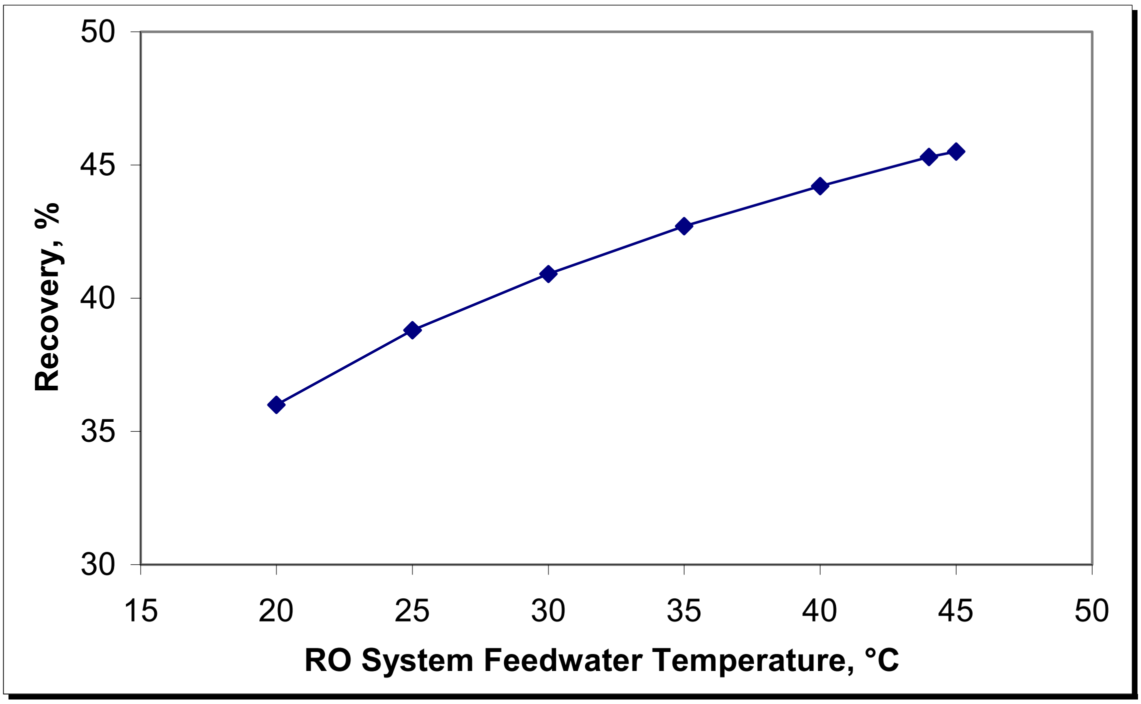


FIG. 3.24. RO System recovery as a function of RO feedwater temperature.

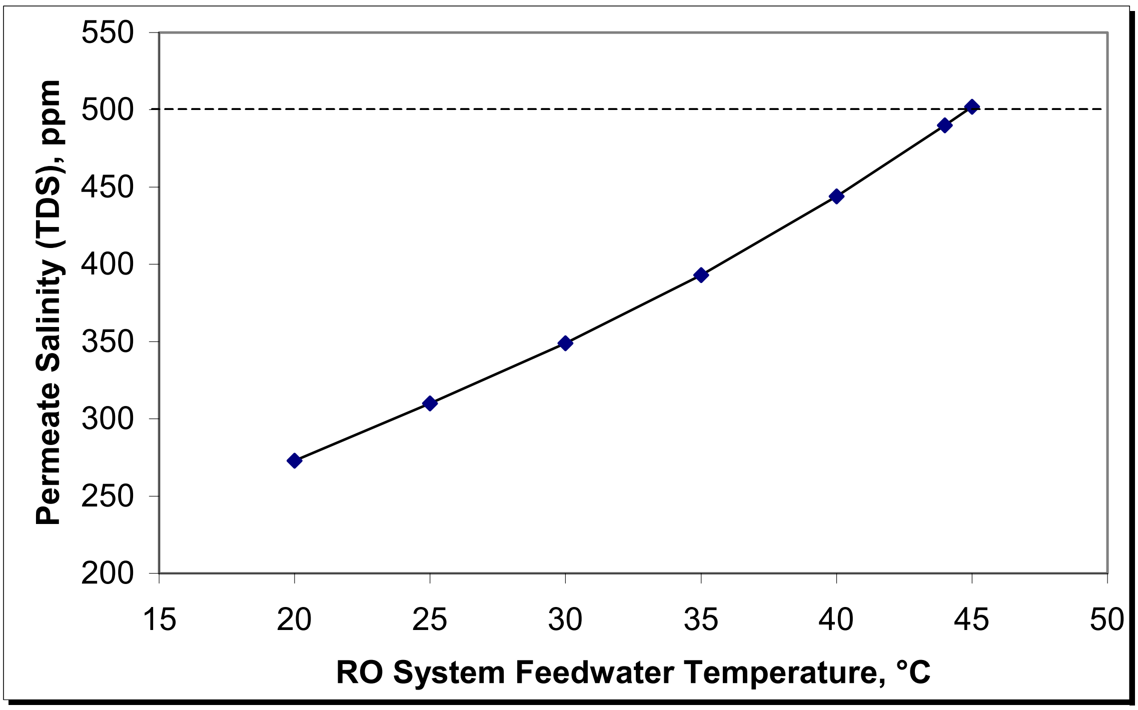


FIG. 3.25. Product water quality as a function of feedwater temperature.

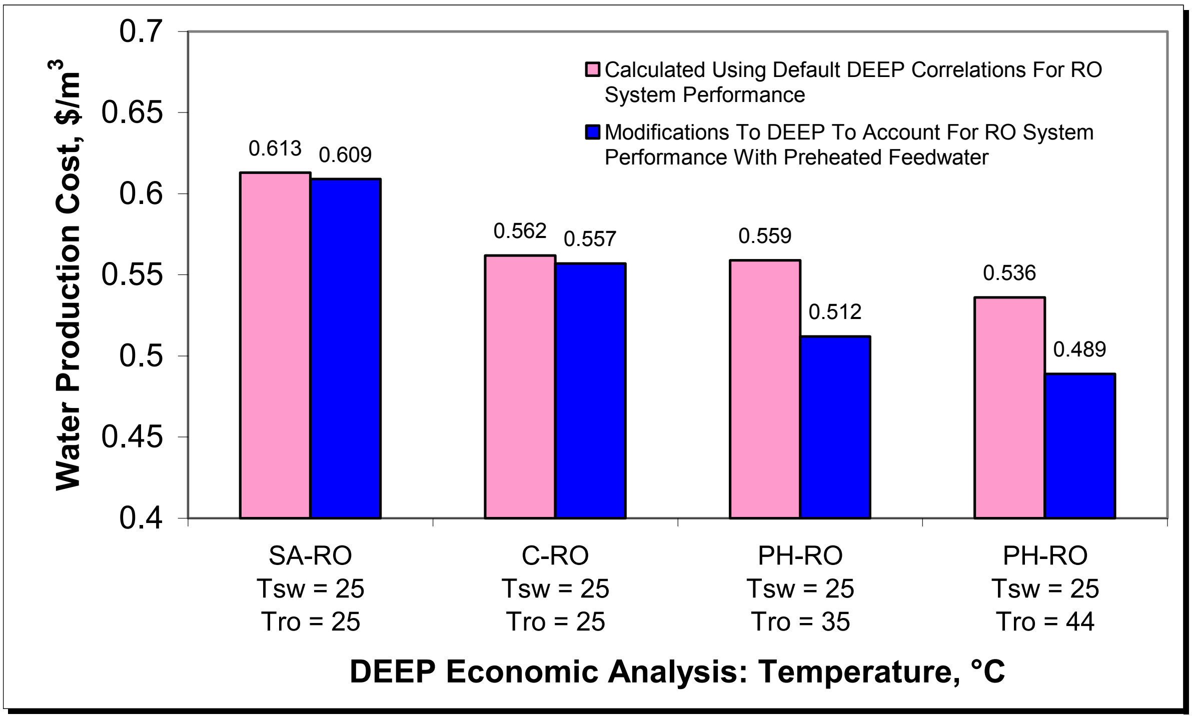


FIG. 3.26. Water production cost results using DEEP.

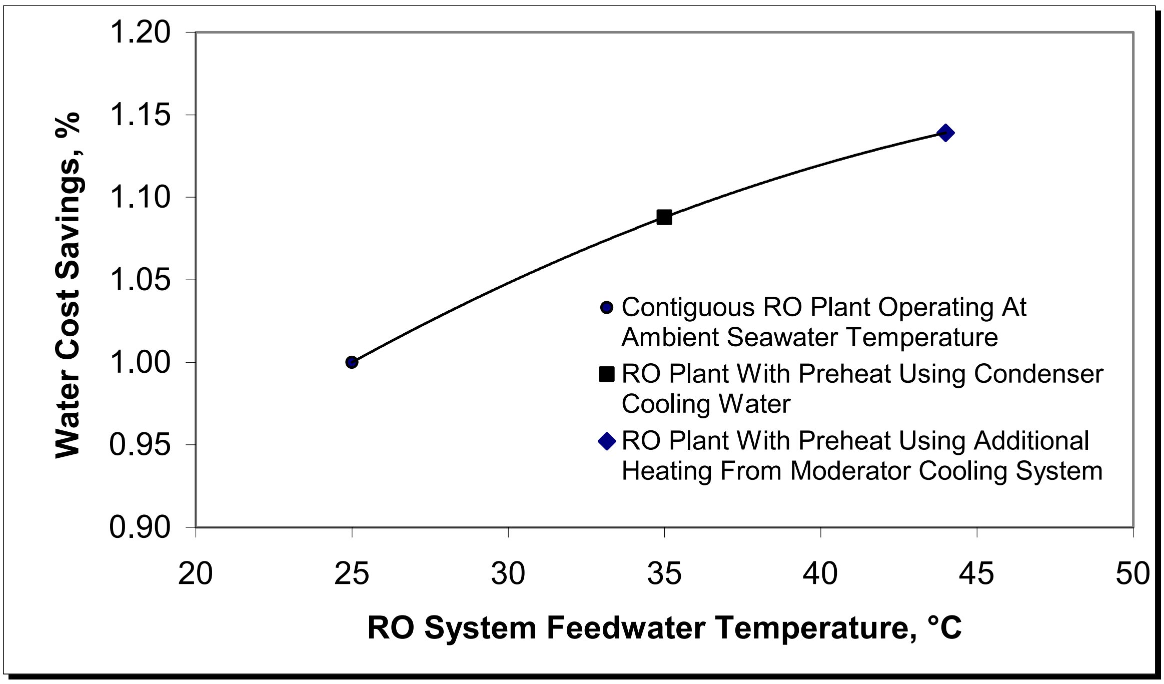


FIG. 3.27. Water cost savings calculated using the modified DEEP version.

This work has shown that for this specific set of seawater conditions, coupling an RO system to a reactor using just waste heat from the condenser cooling system would result in water production costs on the order of US$ 0.51/m^3, which is 15% less than the cost of water produced by a traditional stand-alone RO plant operating at ambient seawater conditions. A plant designed according to the CANDESAL approach and coupled with a CANDU 6 reactor, where there is the added benefit of using waste heat available from the moderator cooling system, could be expect to yield water costs on the order of US$ 0.49/m^3 an additional 5% savings in water cost.

Figure 3.26 provides a graphical illustration of the cost savings that can be achieved for an RO system co-located with and coupled to a reactor plant as its energy source. The stand-alone case is not included in this figure, as it represents an independently operated plant and includes the additional costs of seawater intake and outfall structures. Again, it can be seen from this figure that the ability to take advantage of additional RO system feedwater preheat as a result of using waste heat from the CANDU 6 moderator cooling system provides a significant additional savings in the cost of water production.

3.6. Barge-mounted PWR KLT-40C with RO and MED (Russian Federation)

3.6.1. Background

The Russian Federation's activities for the development of a nuclear desalination complex are currently focused on the application of a Nuclear Floating Power Unit (NFPU). This is being developed for a floating nuclear electricity and heat co-generation plant (FNCGP), on the basis of two KLT-40C reactors. The NFPU (Figure 3.28) is equipped with two 148 MW KLT-40C reactors, two KLT-40C reactors and two turbine generators form two separate power units. Each of these carries an installed capacity of 35 MW(e). Turbine steam extraction enables heating water in the intermediate circuit of the district heating system, each power unit producing 25 Gcal/hour of heat. Characteristics of the NFPU are presented in Table 3.15.

TABLE 3.15. BASIC CHARACTERISTICS OF NUCLEAR FLOATING POWER UNIT (NFPU)

Number of reactors	2
Total thermal power of reactors, MWt	2x148
Steam production, tons/hour	2x240
Steam pressure at SG outlet, MPa	3.8
Steam temperature at SG outlet, ^{0}C	290
Feedwater temperature, ^{0}C	170
Gross electric power, MW(e)	2x35
Net electric power, MW(e)	2x32.5
Thermal power for heat application system, Gcal/hour	2x25

3.6.2. KLT-40C, the energy source

Two KLT-40C units are mounted as the energy source for the NFPU nuclear desalination complex. The KLT-40C is a modified version of the well-proven KLT-40 reactor used for nuclear icebreakers. Figure 3.29 illustrates the basic flow diagram of the KLT-40C nuclear steam supply system. Table 2 gives the NSSS basic design characteristics of the KLT-40C.

TABLE 3.16. BASIC NSSS DESIGN CHARACTERISTICS OF KLT-40

Thermal power, MW(th)	148
Steam flow, t/h	240
Core operating life, h	14600
Refuelling 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

3.6.3. Desalination processes for the nuclear desalination complex

Two design options have been developed for the purpose of a floating water desalination station: one is with a distillation desalination facility and the other with a reverse osmosis system [22]

Floating station for seawater desalination using distillation technology

Two reactor units with a rated power of 80MW(th) run on two main turbo generators with backpressure turbines. Waste heat from the turbine condensers is transferred via an intermediate circuit to a twin-unit distillation desalination facility (Figure 3.30). The desalination unit is composed of film evaporators with horizontal tube bundles. Similar evaporators have been successfully operated for many years in the nuclear power-desalination complex at Aktau (Kazakhstan) and at some other sites.

Engineering measures in the design completely eliminates the environmental impact on sea and desalinated water by the reactor units. A relatively small quantity of heat is discharged during station operation, but does not have a significant impact on the environment. Basic technical and economic characteristics of the station are given in Table 3.17.

TABLE 3.17. BASIC TECHNICAL AND ECONOMIC CHARACTERISTICS OF THE FLOATING SEAWATER DESALINATION STATION

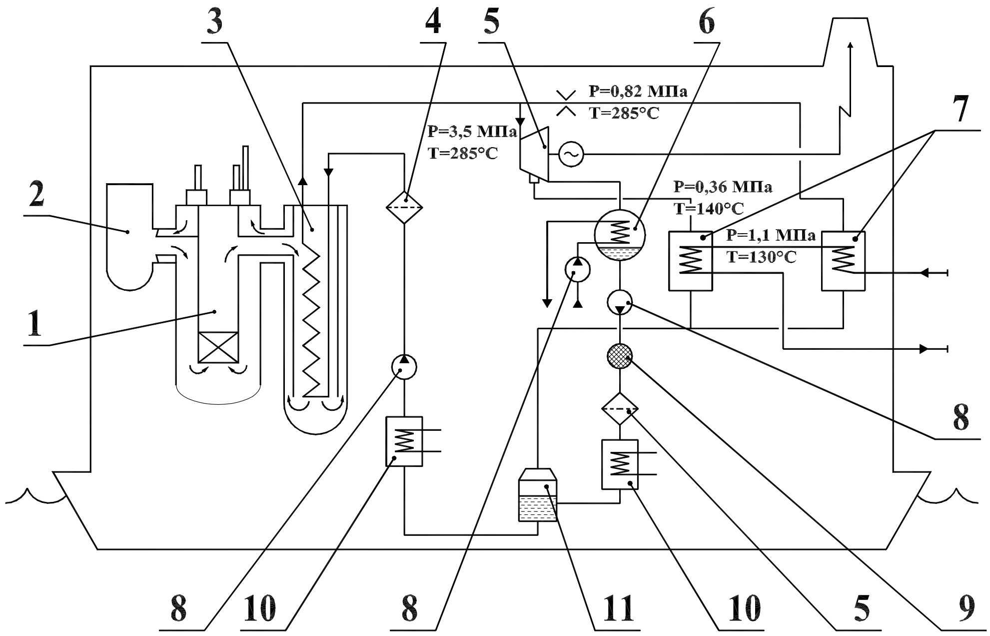
Length of vessel, m	160
Width of vessel (max.), m	44
Draught, m	7
Drinkable water output, m^3/day	80000
Service life, yr.	up to 40
Number of reactors	2
Number of desalination facilities	4
Refuelling interval, yr.	2–3
Average load factor	up to 0.85
Staff, persons	60
Term of pilot station creation, yr.	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
Payback period, yr.	8–10

Floating station for seawater desalination using reverse osmosis technology

The station is composed of two floating structures: an FNPP with two KLT-40C reactor units and a vessel for the seawater desalination facility. Part of the electricity generated by the FNPP is transmitted to the desalination vessel to produce potable water, and the rest is channelled to consumers in the coastal area (Figure 3.31).

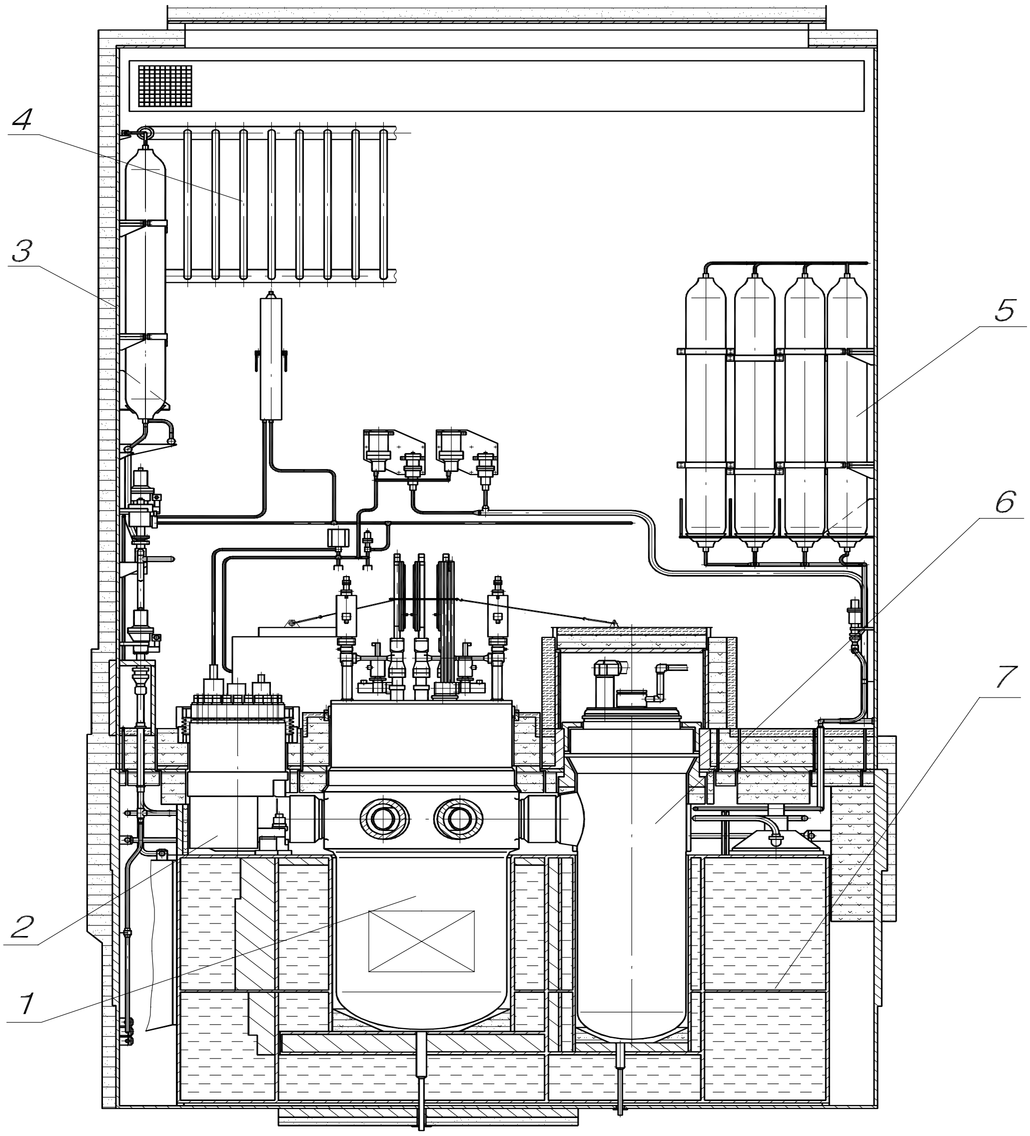
In order to optimise the station's technical and economic performance, different energy allocation can be made between the power and the potable water production at the given thermal power of the reactor plants of 2 × 150 MW(th). Possible options of power to potable water output relationships are given in Figure 3.32.

The construction time of a pilot station is about 5 years, and its investment cost is about 300 million US dollars. One cubic meter of desalinated water is expected to cost about 1 to 1.5 US dollars when the station is operated in a desalination mode. Membranes that are capable of reducing the salt content from 39–43 g/l (seawater) down to 500 mg/l (distillate) would be used in the reverse osmosis desalination facility.



1 – reactor;
2 – primary circuit circulation pump;
3 – steam generator;
4 – mechanical filter;
5 – turbo-generator;
6 – condenser;
7 – intermediate circuit heater;
8 – pump;
9 – ion-exchange filter;
10 – feed water heater;
11 — deaerator

FIG. 3.28. Schematic flow diagram of nuclear floating power unit (NFPU).



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

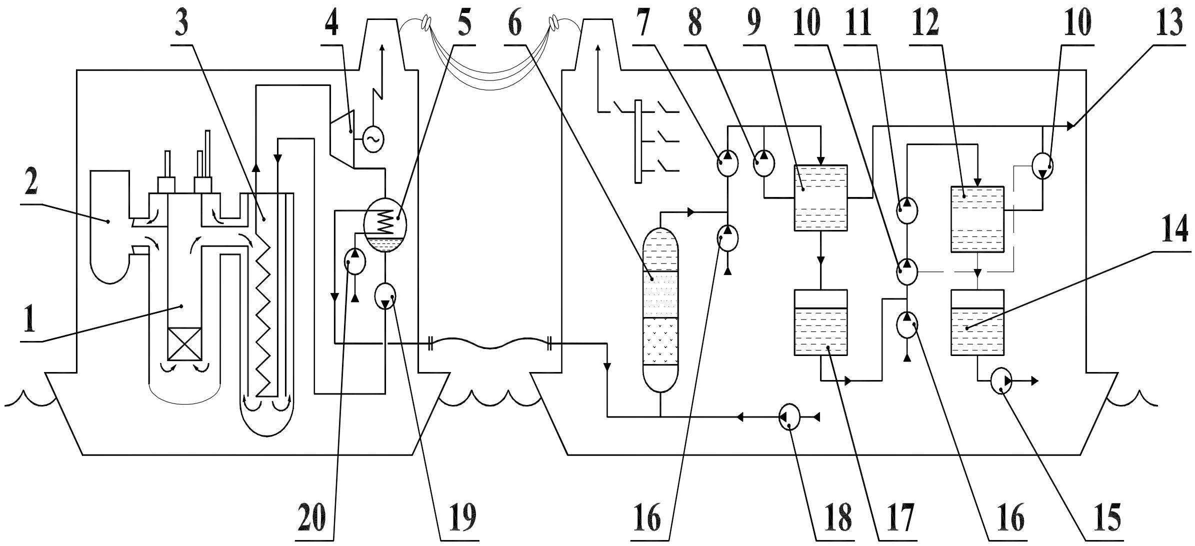
FIG. 3.29. KLT-40S reactor plant.

3.7. A small integrated PWR NIKA-70 with MED and RO (Russian Federation)

3.7.1. Background

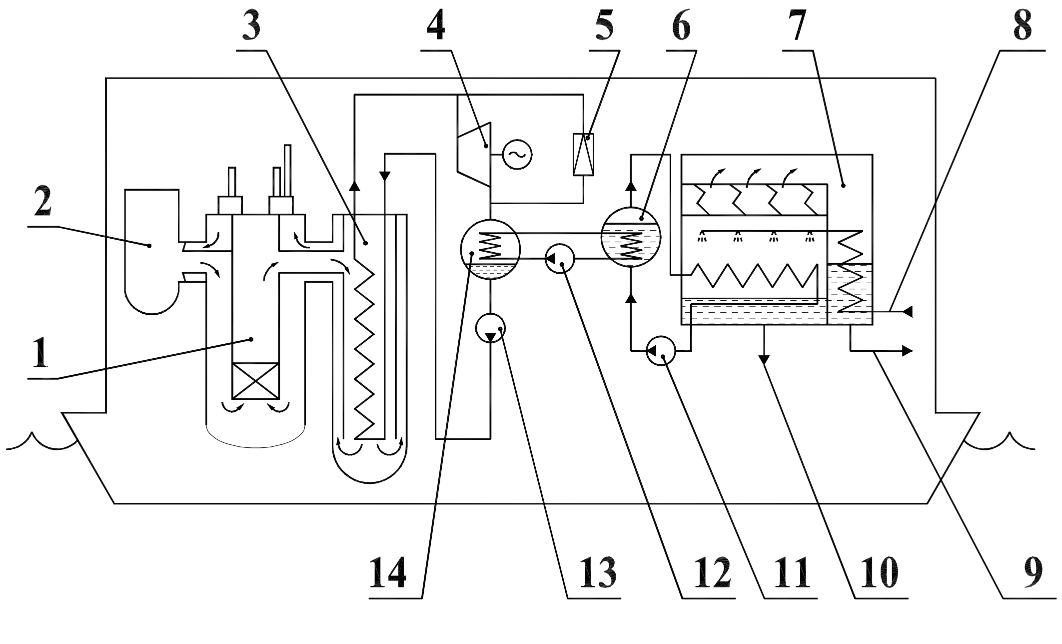
The Research and Development Institute of Power Engineering (RDIPE) has in recent years been actively involved in designing the NIKA nuclear facilities with advanced integral PWR reactors with enhanced safety capabilities. These efforts rely on long-term experience of RDIPE in designing the mobile nuclear facilities.

The NSSS NIKA-70 will be used as a part of the co-generation floating nuclear plant. Specific design features of this facility permit it to carry out all manufacturing activities and testing on the manufacturer's site and deliver a completely finished NSSS to its designated location.



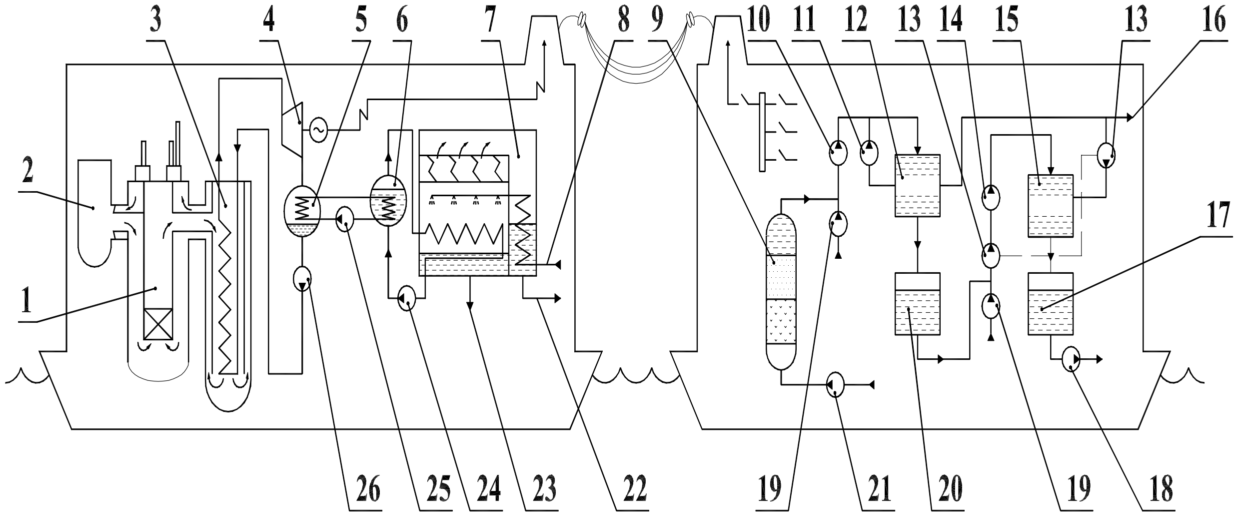
1 – reactor; 2 – primary circuit circulation pump; 3 – steam generator; 4 – turbo-generator; 5 – condenser; 6 – prefilter; 7 – medium pressure pump; 8 – recycle pump; 9 – ultra-filtration membranes; 10 – energy recovery system; 11 – high pressure pump; 12 – R.O. membranes; 13 – outfall structure; 14 – potable water storage tank; 15 – potable water pump; 16 – anti-salant injection system; 17 – clarified water tank; 18 – pump; 19 – secondary circuit electric pump; 20 – pump

FIG. 3.30. Principle flow diagram of the system with RO desalination.



1 – reactor; 2 – primary circuit circulation pump; 3 – steam generator; 4 – turbo-generator; 5 – reduction cooling set; 6 – steam generator; 7 – distillation desalination plant; 8 – sea water inlet; 9 – sea water; 10 – evaporated sea water brain; 11 – pump; 12 – intermediate circuit electric pump; 13 – secondary circuit electric pump; 14 – condenser

FIG. 3.31. Principal flow diagram of the system with thermal desalination.



1 – reactor; 2 – primary circuit circulation pump; 3 – steam generator; 4 – turbo-generator;
5 – condenser; 6 – steam generator; 7 – distillation desalination plant; 8 – sea water inlet;
9 – prefilter; 10 – medium pressure pump; 11 – recycle pump; 12 – ultra-filtration membranes;
13 – energy recovery system; 14 – high pressure pump; 15 – R.O. membranes;
16 – outfall structure; 17 – potable water storage tank; 18 – potable water pump;
19 – anti-salant injection system; 20 – clarified water tank; 21 – pump; 22 – sea water outlet;
23 – evaporated sea water brain; 24 – pump; 25 – intermediate circuit pump;
26 – secondary circuit pump

FIG. 3.32. Principle flow diagram of the system with combined desalinators.

Its particularly small mass and dimensions, and the insignificant weight of the barge carrying the CNPP (estimated as ~2.6 m) allows for transportation — by waterways to distant regions. It therefore seems appropriate to use NIKA-70 for seawater desalination taking into account its high performance characteristics and high level of safety. Calculations also demonstrated that in terms of fresh water cost such nuclear reactors can be competitive with fossil options.

3.7.2. Design description

Reactor Design

The NSSS NIKA-70 [23] is based on a water-water integral reactor (see Figure 3.33). All components of the primary circuit (i.e., the core and control rods, steam generator, main coolant pumps, pressurizer) are located in a single cylindrical vessel. Such reactor design offers the following advantages:

- Almost all primary circuit pipelines can be excluded from NSSS (because of its integral arrangement) and, hence, the probability of a leakage can be substantially reduced;
- High flow rate of natural circulation of primary coolant can be provided; and
- Water inventory above the core can be increased and, therefore, in case of a leakage in the primary circuit boundary there will be no possibility of water supply to the reactor, and core-cooling conditions can be significantly improved.

NSSS has been designed to make use of the materials, parameters and media characteristics that are broadly applied both in the Russian Federation and elsewhere. In combination with its operationally proven elements of primary components (the core, steam generator, coolant pumps, absorber rods, etc.) an approach that enables it to take advantage of the R&D data on thermal hydraulics, properties of the structural materials, water chemistry and others, has resulted in limiting, the needed scope of R&D activities for the pilot NSSS — and only a minimum amount of efforts will then be required for the creation of the pilot plant.

Reactor core heterogeneous- channel type with a single-pass coolant flow from bottom to top inside and outside fuel assemblies (FA). FA is composed of fuel rods with a square cross-section. At the corner there are fins that are spiral with respect to the longitudinal axis of the fuel rod. Fuel composition is uranium-zirconium alloy with ~20% enrichment by U^{235}. Fuel cladding is made of zirconium alloy. Burnable rods placed in FAs and absorber rods moving outside fuel assemblies are used to compensate for reactivity change in the core.

Primary coolant circulation — provided by two **main coolant pumps (MCP) with asynchronous motors —** are installed on the reactors cover. As an additional benefit, a simple configuration and short length of the primary circuit path permits a high flow rate of natural circulation to be sustained in the reactor, and a capability for the NSSS operation at power not lower than 25% of nominal when MCPs are stopped.

In-vessel once-through **steam generator** is designed as surface-type helical heat exchanger with tubing made of titanium alloy. The heat exchange surface of the steam generator is divided into 16 cylindrical cassettes that are placed in the reactor annulus formed by a cylindrical part of the reactor vessel and core barrel. From the steam and feed water sides the SG cassettes are connected via pipelines to form four independent sections that can be isolated by valves outside the reactor vessel.

Control element drive mechanisms (CEDM) of the reactor control and protection system are meant for motion of control clusters in the core and their holdings at required position. CEDM incorporates a rotary step motor used for motion of control rods under all normal and emergency modes of NSSS operation. The step motor is backed up with a spring-type actuator that inserts the rods in the core in case of loss of power for the step motor or control system under any position of the reactor, including its capsizing. Implementation of this engineering solution is especially important keeping in mind that the reactor is to be mounted on a ship.

Unlike the known designs of integral reactors under development in many countries where either steam or steam-gas pressurizers are applied, the integral reactor of the NSSS NIKA-70 uses **a gas pressurizer**. Selection of this solution was motivated by several reasons, firstly, the intention was to simplify and, consequently, enhance the safety of primary circuit pressure compensation system by eliminating — heaters and — sprinkler systems. Secondly, this approach is based on a 40-years experience in the design and operation of ship-mounted NSSSs with gas pressurizers in the primary circuit. It should be pointed out, however, that in the previous cases the pressurizers were placed outside the reactor vessel.

The NSSS equipment layout depicted in Fig 3.34. A regular cylindrical shape of the reactor vessel enables the most optimal application in terms of efficiency and mass-dimensional characteristics iron-water biological shielding, which consists of two concentric circular tanks. The design of the biological shielding allows — no possibility of reactor vessel melting-through in case of a beyond design-basis accident (postulated accident) resulting in core dryout.

All primary circuit equipment does not require maintenance during power plant operation and is located in a leak-proof strong safeguard vessel, which confines radio nuclide releases from the primary circuit during all DBAs.

The remaining NSSS equipment is located in a strong, leak-tight container, which serves as an additional protective barrier in the way of radionuclides propagation into the environment.

The NSSS thermo hydraulic configuration is extremely simple as compared with those of the world's operating reactor plants of loop or modular design. All safety systems are of passive type and are designed for at least 72-hour operation without operator's intervention.

The specific design features of NSSS NIKA-70 enable a rather efficient solution to the urgent problems of power plant decommissioning and utilization after the expiration of its service life. When the core is unloaded, the reactor and internal biological shielding tank shall be removed by crane from the power plant vessel and transferred to a storage facility. The remaining NSSS structures are of low activity and after a certain cooling time can be handled in the usual way. Table 3.18 summarizes the major design parameters of NSSS NIKA-70.

TABLE 3.18. DESIGN CHARACTERISTICS OF NIKA-70 REACTOR

No.	Characteristic	Unit	Value
1	Thermal power of the core	MW(th)	70
2	Total electric power	MW(e)	15
3	Nominal steam generating capacity	kg/s	25
4	Superheated steam pressure	MPa	3.0
5	Superheated steam temperature, min	°C	274
6	Feed water temperature	°C	60
7	Nominal pressure in primary circuit	MPa	15.0
8	Primary coolant temperature at full power:		
	at core inlet	°C	260
	at core outlet		300
9	Operating range of power variations	% N_{nom}	20 ÷ 100
10	Effective campaign of the core	hour	30000
11	Core–water-water type:		
	equivalent diameter	mm	1500
	height	mm	1200
	Fuel:		
	U^{235} enrichment	%	19.7
	U^{235} load	kg	250
	specific power density	kW/l	40
12	Service life	years	30

NSSS Safety Concept

Essential to a high safety level of the NSSS is implementation of the following solutions:

Use of an integral water-cooled water-moderated reactor with inherent self-protection and the following unique features:

- Negative coefficients of reactivity throughout the operating range of parameters;
- High rate of natural circulation of the coolant which affords effective cooling and heat removal from the core during design-basis and beyond design-basis accidents;
- High heat storage capacity of metal structures and a great mass of coolant in the reactor, which result in a relatively slow progression of transients during accidents with upset heat removal from the core.

Defense-in-depth provided as a system of barriers to off-site releases of ionizing radiation and radioactive uranium fission products, and implementation of a package of engineering and organizational measures to protect these barriers against internal and external impacts. The system of safety barriers includes:

- Fuel matrix;
- Fuel cladding;
- Leak-tight primary circuit;
- Safeguard vessel;
- Isolating valves of the secondary circuit;
- Containment.

Use of passive systems and safety features whose operation is based on natural processes with no need for external power supply.
Such systems and facilities include:

- CPS drives the design assures insertion of control rods into the core by gravity and drop springs;
- Passive systems for emergency residual heat removal;
- A safeguard vessel which ensures core coverage with coolant and heat removal under all accidents, and guarantees radioactivity confinement in case of a leak in the primary circuit;
- A containment which limits radioactive releases from an open safeguard vessel and under beyond design-basis accidents;
- Iron and water biological shielding, which apart from their direct functions, serves as bubbler tanks with cooling water and provided heat removal from the reactor vessel to avoid its melt through under a postulated beyond design-basis accident with core dryout.

Safety systems reliability

High reliability of the safety systems is provided owing to the following philosophy:

- The systems are passive, i.e. they do not need special actuators to initiate them;
- The safety systems and features are diverse, i.e. they are based on different principles of system operation (for example, electromechanical CPS drives and liquid poison injection system are used for emergency shutdown);

- The safety systems are redundant (for instance, the redundancy of the reactor shutdown system is 2 × 100% , of ECCS – 4 × 50%, etc.)
- Systems and equipment are subjected to periodic in-service inspection or continuous monitoring.

Protection against human errors

The design safety philosophy pays more attention to prevention, or mitigation of the consequences of human errors and deliberate actions meant to render the nuclear plant inoperative.

These measures include:

- Minimum scope of on-load maintenance and repair of major systems and equipment;
- Design solutions and organizational measures intended to prevent an unauthorized access to NSSS systems (all vital systems are housed in a safeguard vessel or containment);
- Use of systems satisfying as far as possible the fail-safe principle (failure of a system component triggers the safety function in the system or system fails in a safe state);
- Passive safety systems and features are used so that they do not have to be actuated with special means (a safeguard vessel, a containment) or they can be brought into action in a passive way (emergency cool down systems, ECCS, system for reducing overpressure in the reactor, safeguard vessel and containment);
- Reliable control systems are used which minimize or disable erroneous operator's actions, with personnel having no access to interlocks and set points;
- Operator support systems are provided which rapidly assess the plant state and suggest optimum control actions;
- Special hardware is used for training of the operating and maintenance personnel and maintaining their skills and knowledge; in particular, a simulator is used to drill operating personnel in various situations, including emergencies.

Choice of desalination technology

For the production of desalted water NIKA-70 can be coupled with all types of modern desalination plants, i.e. with reverse osmosis (RO) plants, with distillation plants and their various combinations.

Nuclear desalination combined with an RO plant has the following advantages:

- Maximum level of safety from the view point of preventing radioactive contamination of fresh water, since these two plants would be connected only by electrical cables;
- Connection of NSSS and the desalination plant via electrical cables only will facilitate the transient conditions for the nuclear plant in case of disconnection of the desalination plant.
- As was precisely established, fresh water cost will be minimum if the reverse osmosis technology is used.

When choosing the membrane type for the desalination plant, it should be remembered that the Russian Federation does not have developed capabilities for manufacturing membranes for water desalination and the appropriate equipment would be purchased from foreign manufacturers.

The disadvantages of membrane desalination are as follows:

– The need for significant pre-treatment of water so as to protect membranes from bacteria, free chlorine and oxygen;
– Low resistance of membranes to possible operational departures of the desalination plant resulting in failure of membrane and their costly replacement;
– A limited service life of membrane elements (require replacement over several years);
– Not a very high degree of water desalination, though it meets the international requirements of the World Health Organization (WHO) for the quality of drinking water.

For the above reasons it is recommended that reverse osmosis facilities be used to produce the cheapest desalted water (though not of premium quality).

As opposed to reverse osmosis plants, distillation plants have some advantages (they are capable of producing desalinated water of higher quality and have higher reliability and longer life cycle); therefore it seems reasonable to consider the option of coupling a NIKA-70 with a distillation plants. The most acceptable distillation technology for coupling with NIKA-70 is multi-stage distillation. Multi stage flash water plants also can couple with NIKA-70 but preliminary economic estimations have shown that the cost of desalted water is too high so this type of desalination plant was not considered later on.

It seems reasonable to use distillation desalination plants with a horizontal-tube film MED apparatus developed by SverdlNIIKhimmash, Ekaterinburg, Russian Federation. Those MED water plants are up-to-date facilities characterized by high cost efficiency and productivity of desalination process, quality and stability of produced distillate, low consumption of energy resources, low need in metal and occupied area, enhanced reliability and flexibility, simplicity of control, maintenance and repair. As an additional benefit, the cost of these plants is not very high and rather competitive in the international market. The following options of these plants are offered: 240 600, 1200, 2400, 3600, 16 800 and 20 000 m^3/day [25].

Combined (hybrid) distillation plants consisting of RO and distillation plants can be of special interest for consumers. In this case one can obtains very pure fresh water from the distillation plant, as well as fresh water from the membrane water plant with a higher level of salt but at a lower cost. The consumer has a choice for the optimal ratio of the distillation and RO product water.

Depending on site conditions and customer’s requirements, desalination plants can be placed on a single barge with the reactor, on a separate barge or on the shore.

Coupling Between Reactor and Desalination Systems

The task of coupling nuclear power and water plants is very important especially when coupling with distillation plants. Coupling designs should exclude, firstly, any possibility of radioactive contamination of the desalted water, secondly, the possibility of penetration of salt water into the turbine circuit and, thirdly, it should not be too expensive.

The following variants of coupling NIKA-70 with desalination plants have been considered:

- Through an extra intermediate circuit with hot water throttling in the first stage of the distillation plant;
- With a distillation plant using an extra isolated intermediate circuit;
- With a preheat reverse osmosis plant;
- With a hybrid desalination complex including distillation and reverse osmosis plants.

In addition, a reverse osmosis plant could be coupled without preheat, but is not being considered here because it is straightforward.

A variation of coupling through an additional isolated water circuit at a higher pressure, which has been proposed in [26], seems to be best from the viewpoint of the radioactive contamination protection of the desalinated water and the turbine plant loop salinization. However, using an additional isolated water circuit would result in significantly higher desalted water costs due to generated power loss resulting from a higher temperature in the turbine condenser.

A variation of coupling through an extra intermediate circuit with hot water throttling in the first stage of the distillation plant proposed by the IAEA [27], in our opinion, is the best way to meet the requirements of both economic effectiveness of the desalination process and radioactive contamination protection.

This variant has the following advantages:

- A low temperature drop between the secondary circuit of the reactor plant and the distillation plant, which saves the electrical power produced by the turbine generator;
- A higher pressure in the circuit of the desalination plant prior to the throttling make a reliable barrier against radioactive leakage in the event of a leak.

A disadvantage of the above variant is that seawater could enter the secondary circuit of the reactor plant in case there is a leakage in the condenser. However, the experience of operating nuclear power plants with sea-water-cooled turbine condensers has shown that this disadvantage was not a determinant. There are various ways of reducing the risk of secondary circuit seawater contamination to zero, etc., namely:

- A high reliability of the design,
- Sectionalization of the condenser,
- Control of secondary circuit purity.

This variant is chosen as the most optimal for coupling the NIKA-70 reactor plant and distillation plant.

Figure 3.35 shows the optimal scheme of coupling between the NIKA-70 reactor plant and multi-effect distillation plant. The distillation complex consists of a power and heat source, i.e. NIKA-70, and three MED facilities having a total installed production capacity of fresh water production of 3 × 12 000 m^3/day.

The coupling variants for a preheat reverse osmosis plant and a hybrid plant shown in Figure 3.36 and Figure 3.37 seem reasonable to be used as proposed in the IAEA publication [27]. The possible design characteristics of nuclear desalination plant with NIKA-70 reactor are listed in Table 3.19. Those characteristics were obtained by calculations using the IAEA-developed software DEEP [27].

One of the challenges of designing a nuclear desalination complex coupled with a distillation desalination plant is choosing the optimum turbine steam bleed (extraction). An examination of ways of steam extraction from the turbine plant (from a back-pressure turbine, live steam extraction at inlet to low pressure stages, using steam bleeder at low-pressure stages, using a back-pressure turbine and a low-pressure condensing turbine in parallel) shows that the most appropriate for the NIKA-70 reactor was using steam for desalination from an interim steam bleeder at low-pressure stages. This meets the requirements for coupling the reactor plant and distillation plant at best — with respect to the regulation of desalted water and electric power production. The turbine used in this configuration allows for regulating the steam extraction for distillation purposes under the parameters as follows:

– Pressure 0.082 MPa,
– Temperature 94°C,
– Flow rate from 0 up to 80 tons/hour.

Such controlled extraction of steam permits the customer to respond flexibly to changes in consumption of power and desalted water.

TABLE 3.19. DESIGN CHARACTERISTICS OF A NUCLEAR DESALINATION PLANTS WITH NIKA-70 REACTOR

Parameter	Distillation (optimal water productivity)	Reverse Osmosis	Preheat Reverse Osmosis	Hybrid Desalination Plant (50% MED+50% RO)
Installed Water Production Capacity, m^3/day	36 000	71 000	80 000	72 000
Electric Power to the Grid, MW(e)	7.7	0	0	0.8

The temperature of steam extraction from the turbine and, accordingly, the maximum brine temperature T_{mb} have a great influence on the performance of the nuclear desalination complex with the distillation plant. This temperature should be the highest possible maximum for desalted water production. But from view point of minimal losses of electrical power, this temperature should be as low as possible. There exist optimal value of this temperature by cost of desalted water. Calculations have showed that the optimal maximum brine temperature lies in the range of T_{mb} = 86–90 °C.

It is evident that for a given nuclear desalination complex electrical power production and desalted water capacity are connected with each other. It would be interesting to analyze the interference between the electrical power and fresh water productivity. Figure 6 shows the dependence of the net saleable power versus the average daily water production for various kinds of desalination plants coupled with NIKA-70. Using this diagram the customer can make a choice between required electrical power and fresh water production. It can be seen, for example, that for a given power of NPP the maximum installed water productivity makes up to 48 700 m^3/day for distillation plant with the residual salt content of 10 ppm + 3.3 MW(e) to the grid, up to 71 000 m^3/day with the residual salt content of 320 ppm — for the membrane plant without pre-heat, 80 000 m^3/day with the residual salt content of 320 ppm — for the membrane plant with pre-heat, and 72 000 m^3/day with the residual salt content of 166 ppm + 0.8 MW(e) — for hybrid plant (50% distillation + 50% RO with pre-heat).

Maximum water capacity is limited for the distillation plant by the maximum seawater heating temperature of 125^{o}C connected with corrosion conditions and for membrane water plants by available electrical power.

TABLE 3.20. INPUT DATA FOR FEASIBILITY EVALUATION OF NUCLEAR SEAWATER DESALINATION

Parameter	Value
Energy Plant	
Total Thermal Power, MW (th)	70
Net Electrical Power, MW(e)	14
Steam Temperature at Turbine Outlet, ^{0}C	60
Specific Construction Cost (including additional remote site construction cost +10% by default), $/Kw (e)	4125
Construction Time, months	36
Specific O&M Cost, $/MW(e)h	10
Specific Nuclear Fuel Cost, $/MW(e)h	20
Specific Decommissioning Cost, $/MW(e)h	1
Operating Availability	0.8
Purchased Electricity Cost, $/kW(e)h	0.09
Oil Price, $/boe	20
Power Plant Life, years	30
Discount/Interest Rate,%	8
Desalination plants	
Distillation plant base unit cost, $/($m^3$/day)	900
Membrane plant base unit, $/($m^3$/day)	800
Average Annual Cooling Sea Water Temperature, ^{0}C	25
Seawater Total Dissolved Solids, ppm	41 000

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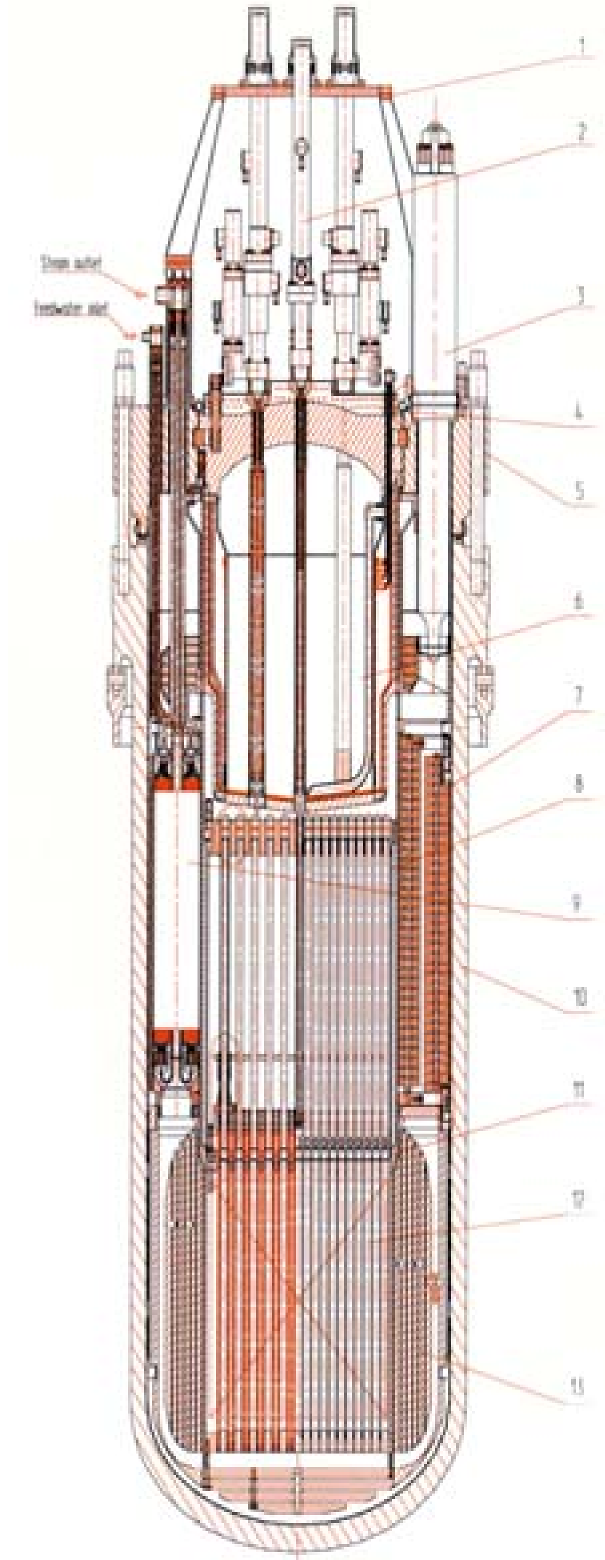
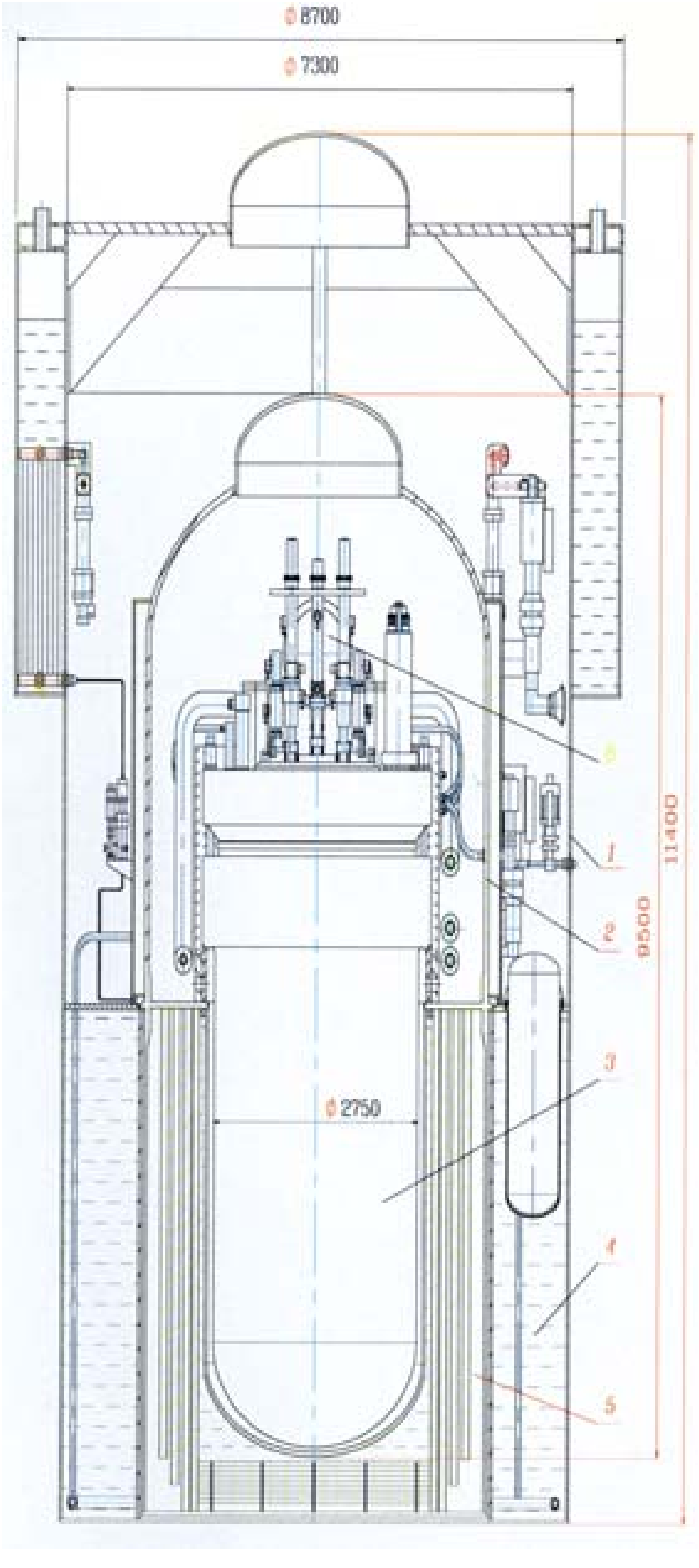


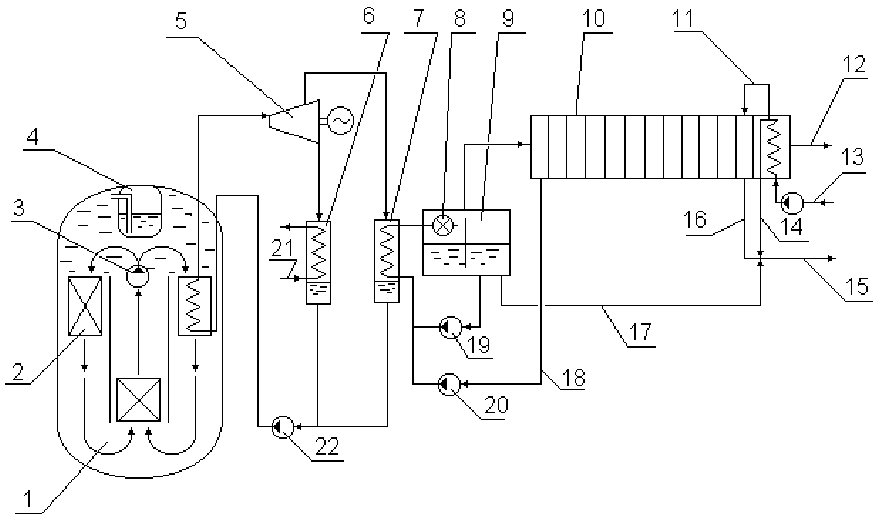
1 - drive fastening frame; 2 - shim rod group drive; 3 - MCP; 4 - thermal insulation; 5 - annular cover; 6 - pressurizer; 7 - displacers; 8 - metalwork with control rod clusters; 9 - SG; 10 - vessel; 11 - core barrel; 12 - fuel assembly; 13 - side shield.

Fig 3.33. General view of the reactor NIKA 70.



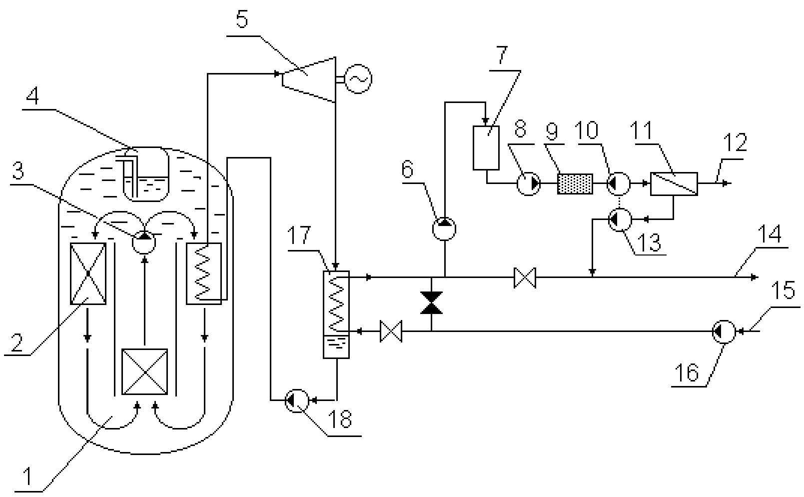
1 - containment; 2 - safeguard vessel; 3 - reactor; 4 - biological shielding external tank; 5 - biological shielding internal tank ; 6 - entrance hatch

FIG. 3.34 Reactor structure of the NIKA-70.



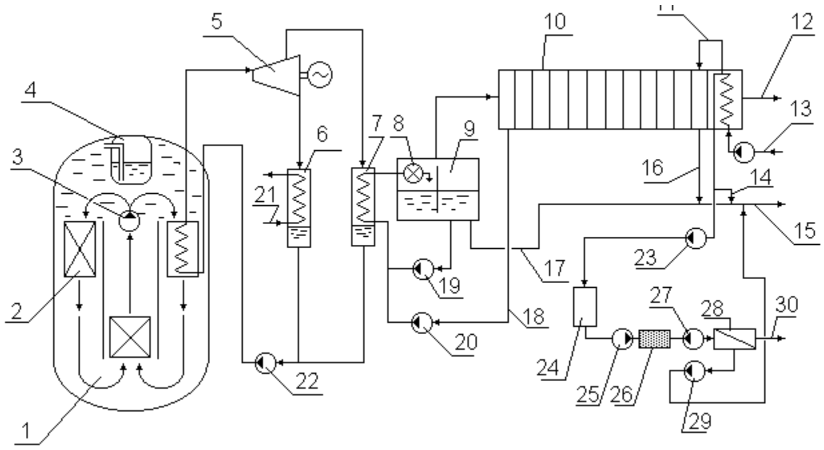
1 – nuclear reactor; 2 – steam generator; 3 – primary pump; 4 – pressurizer; 5 – turbogenerator; 6 – turbine condenser; 7 – condenser-heat exchanger of distillation plant; 8 – throttle; 9 – flash tank; 10 – multi effect distillation plant; 11 – feed makeup; 12 – product water; 13 – seawater intake; 14 – reject cooling water; 15 – brine outfall; 16 – brine discharge; 17 – flash tank blowdown; 18 – preheated water makeup; 19 – intermediate recirculation pump; 20 – makeup pump; 21 – cooling seawater; 22 – feed pump

FIG. 3.35. Schematic diagram of NIKA-70 coupled with MED.



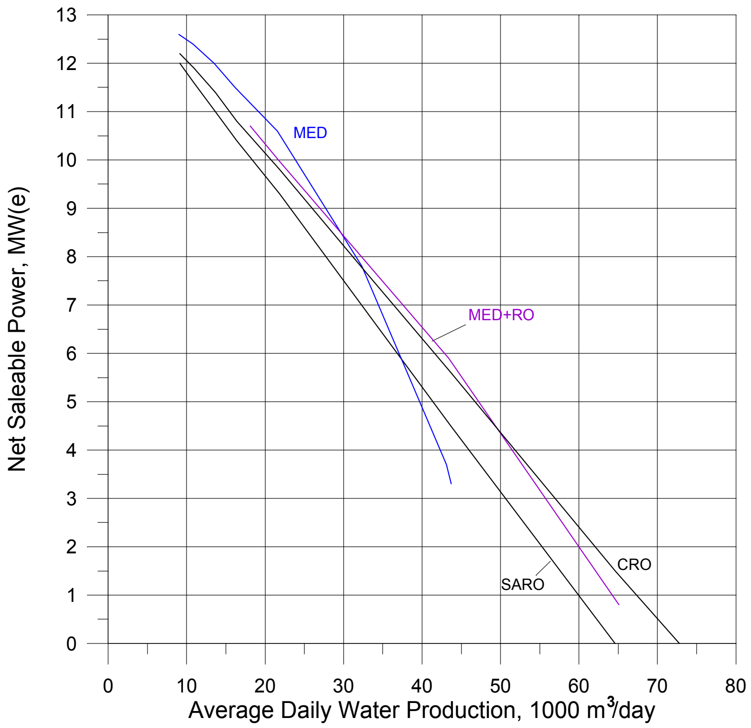
1 – nuclear reactor; 2 – steam generator; 3 – primary pump; 4 – pressurizer; 5 – turbogenerator; 6 – transfer pump; 7 – pre-treatment; 8 – booster pump; 9 – ultra filtration membranes; 10 – high pressure pump; 11 – RO membranes; 12 – product water; 13 – energy recovery; 14 – seawater outfall; 15 – seawater intake; 16 – seawater pump; 17 – turbine condenser; 18 – feed pump

FIG. 3.36. Schematic diagram of NIKA-70 coupled with CRO.



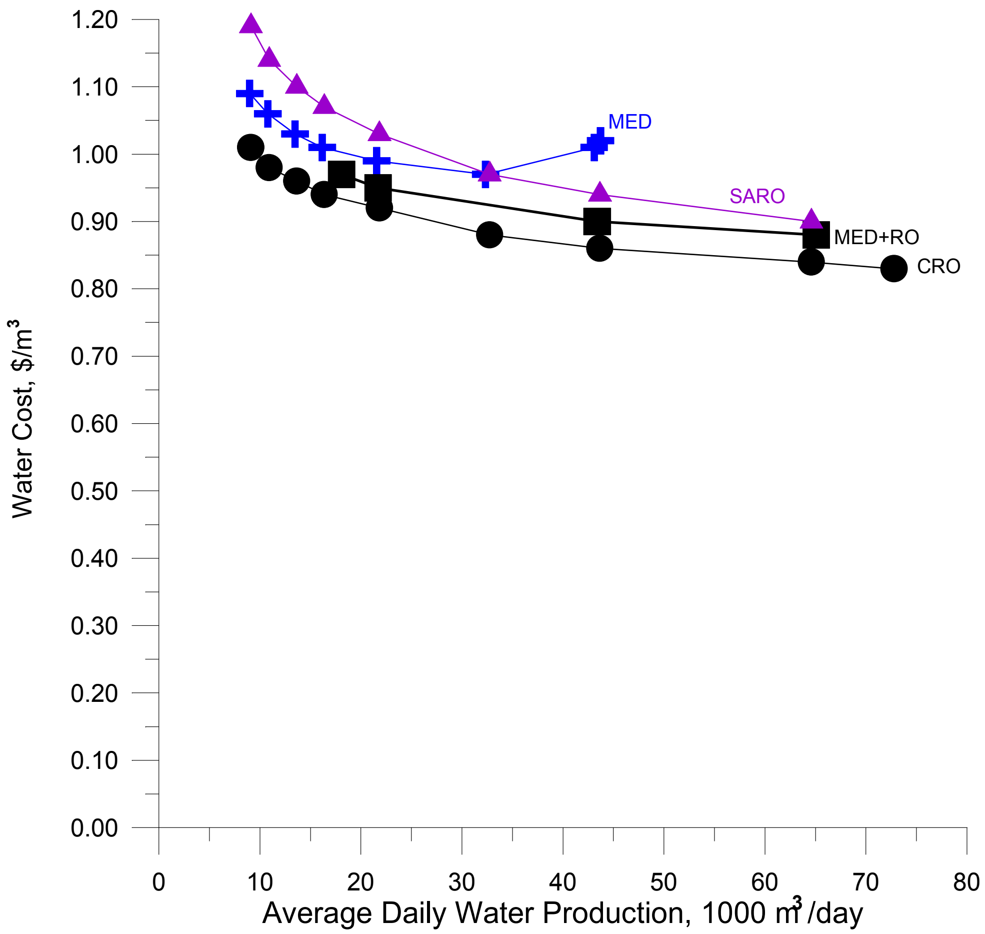
1 – nuclear reactor; 2 – steam generator; 3 – primary pump; 4 – pressurizer; 5 – turbo generator; 6 – turbine condenser; 7 – condenser-heat exchanger of distillation plant; 8 – throttle; 9 – flash tank; 10 – multi effect distillation plant; 11 – feed makeup; 12 – MED product water; 13 – seawater intake; 14 – reject cooling water; 15 – brine outfall; 16 – brine discharge; 17 – flash tank blowdown; 18 – preheated water makeup; 19 – intermediate recirculation pump; 20 – makeup pump; 21 – cooling seawater; 22 – feed pump; 23 – transfer pump; 24 – pre-treatment; 25 – booster pump; 26 – ultra filtration membranes; 27 – high pressure pump; 28 – RO membranes; 29 – energy recovery; 30 – RO product water

FIG 3.37. Schematic diagram of NIKA-70 coupled with hybrid MED and RO.



MED – Multy-Effect Distillation, SARO – Stand-Alone Reverse Osmosis, CRO – Contiguous Reverse Osmosis, MED+RO - Multy-Effect Distillation (50%) + Reverse Osmosis (50%).

FIG. 3.38. Net saleable power vs. average daily water production for various desalination plants.



MED – Multi-Effect Distillation, SARO – Stand-Alone Reverse Osmosis, CRO – Contiguous Reverse Osmosis, MED+RO - Multi-Effect Distillation (50%) + Reverse Osmosis (50%)

Fig. 3.39. Water cost vs. average daily water production for desalination water plants of various types.

3.7.3. Economic perspectives

An economic evaluation of seawater desalination for complexes using NIKA-70 reactors was performed with the IAEA software package, DEEP [4], for the economic comparison of seawater desalination plants. Basic input data for this evaluation are given in Table 3. The remaining input data were taken from spreadsheets by default but with specifics of North African Region.

Water cost versus average daily water production for desalination water plants of various types is shown in Figure 3.39.

Water costs are 0.99–1.09 $/m^3$ for a distillation plant, 0.83–1.01 $/m^3$ for a membrane plant with pre-heat and 0.88–0.97 $/m^3$ for a hybrid plant.

For a distillation plant the average daily water production optimised by cost equals 32 324 m^3/d of fresh water and 7.7 MW(e) of net saleable power at a water cost of 0.99 $/m^3$.

Membrane plants with pre-heat have an advantage in terms of water cost. However, one should remember that water produced by a distillation plant has a higher quality in our calculations; product water salinity is 10 ppm for distillation plants, 320 ppm for membrane plants, and 166 ppm for hybrid plants. The customer can select from different desalination technologies or their combinations to meet his own requirements. The hybrid plant seems most attractive because it allows production of both very pure freshwater by the MED process and fresh water with increased salinity in the reverse osmosis process at a rather low cost.

3.8. A small natural circulation BWR with RO (Japan)

3.8.1. Background

Water-cooled reactors have been widely used in nuclear systems and have good operating performance as a commercial energy supply systems. Boiling water reactors (BWRs) in particular are known for their simple direct cycle configuration, in which steam generated in the reactor directly flows and expands in the steam turbine without large steam generators between the reactor and the turbine.

Although distillation processes including multi-stage flash (MSF) and multi-effect distillation (MED), which are widely used for seawater desalination, the reverse osmosis (RO) process has become especially reliable and economical in recent years. In the RO process, energy is consumed to compress the saline feed up to 7 or 8 MPa in order to overcome the osmotic pressure of the saline solution of about 6 Mpa [27]. Steam turbine driven pumps (TD pumps), are widely used in BWR feed water systems and can be applied to the RO process because their discharge pressure exceeds 7 MPa. From these considerations, coupling of BWR and RO process with TD pumps seems promising in spite of the fact that only a few designs have been proposed. The objective of this section is to introduce design of coupling BWR and RO plants for the purpose of seawater desalination.

3.8.2. Design description

Design policy

Maximum utilization of proven technologies is of paramount importance in the design of BWR+RO plant. This utilization contributes to improving the economics of the nuclear system because neither large R&D nor new investment in manufacturing facilities is necessary. Moreover, it provides an advantage in licensability in case where similar types of nuclear plants have already obtained the necessary licence.

A standard BWR plant design is simplified and rationalized to be suited for seawater desalination using the RO process. Major design targets and design bases used in this section as summarized below. These can always be tailored to specific user requirements.

Nuclear boiler

From the standard available BWR designs, namely, BWR/4, BWR/5 and ABWR, the BWR/4 was chosen owing to its small dimensions compared with others and its suitability for co-generation purposes. The BWR/4 remains however, too large in terms of power generation capability (1600 MWth), design principle is maximum utilization of proven technologies, power density of the core was decreased instead of changing the core and fuel configuration.

In general, BWRs have a high natural circulation capability in the reactors pressure vessel (RPV) because of boiling in the core shroud. According to the core-flow-map of BWR/4 as shown in Figure 1, up to 37% of the rated power is attainable with 33% of the rated core flow on the natural circulation curve. BWR/4 with the low power density of 19 kW/l becomes a natural circulation BWR and no recirculation pump is necessary.

TABLE 3.21. MAJOR DESIGN TARGETS AND BASES OF BWR+RO

General	Purpose	Co-generation: electricity & seawater desalination
	Thermal power	600 MW class
	Electric power	100–200 MW class
	Water production rate	80–120 × 10^3 m^3/d
Nuclear boiler	Core	Conventional BWR core with low power density
	Fuel	Conventional BWR fuels
	Core cooling	Natural circulation
	Internals	Conventional BWR internals
Safety system	ECCS	Rationalized ABWR ECCS (active)
	RHR	Rationalized ABWR RHR (active)
	For severe accident	Containment protection system (passive)
Balance of plant	Turbine system	Conventional steam turbine system
	Desalination process	RO
	RO pump type	Turbine-driven pump
	Pre-heating	No (option)

Such natural circulation BWR (TTBWR) with low power density has some economic advantages compared with forced circulation BWRs with equivalent power output. Elimination of recirculation pumps results in reducing not only in construction cost but also in reducing maintenance cost of the recirculation system. The low power density reduces refuelling needs and consequently enhances availability of the plant. For example, 48 effective full power month (EFPM) cycle length is achievable with the standard 45 GWd/t BWR fuels. Major characteristics of the nuclear boiler are summarized in Figure 3.41.

Safety system

TTBWR+RO is a natural circulation BWR with no external recirculation loop, which is present in ABWRs. Therefore, safety systems including primary containment vessel (PCV), used in the ABWR, are suitable for TTBWR+RO. The ABWR safety systems are very reliable and have been well proven to show that core damage frequency (CDF) is evaluated far below 10^{-6} for Kashiwazaki-Kariwa units 6 and 7 in Japan [28].

Based on the ABWR safety systems [29], TTBWR+RO safety systems were rationalized. Because the coolant inventory above the core in the RPV is relatively large against the power and because the size of nozzle on the RPV is small, high pressure ECCS is unnecessary and only three low pressures ECCS; namely low pressure flooder (LPFL), are equipped in TTBWR+RO. This is another advantage derived from low power density of the core.

To achieve higher level of safety, a couple of features are added in the TTBWR+RO. One of them is the use of gas turbine generators (GTGs) as an emergency power source. Together with the conventional diesel generator (DG), the use of GTG enhances the diversity of emergency power sources. GTGs are widely used in many industries and the reliability of GTGs is as high as that of DGs. The other feature involves design for preventing a severe accident. For overpressure protection of the PCV, passive containment cooling system (PCCS) is included. The PCCS is composed of three independent trains in which shell-and-tube type heat exchangers condense steam in the PCV and water condensed in the PCC flows back to the RPV by gravity. The safety systems used in TTBWR+RO are depicted in Figure 3.42 with a schematic division of the ECCS.

CDF of TTBWR+RO was preliminarily evaluated based on methods and data used for CDF evaluation of ABWR. Figure 3.43 shows that total CDF of TTBWR+RO is as small as that of ABWR because the configuration of the safety system used for TTBWR+RO is basically identical with that of ABWR. There are, however, small differences in particular CDF in case of a loss-of-coolant-accident (LOCA). This difference is derived from rationalization of high-pressure ECCS, which results in reduction in number of make-up trains to the RPV and consequently worsens the probability of a make-up failure in case of a LOCA. But the probability is still very low.

Balance of plant

The possibility of radioactive contamination of seawater from the BWR coolant is physically eliminated by the use of TD pumps for the RO system. Since no heat transfer is necessary between BWR coolant and seawater, no intermediate loop or heat exchanger between the coolant and seawater is used unlike the coupling of a nuclear reactor and seawater distillation system.

A diagram of TTBWR+RO BOP is shown in Figure 3.44 with its major characteristics as a co-generation plant. Steam generated in the TTBWR flows into the turbines and condenses at the condenser. Total electric power is about 182 MW.

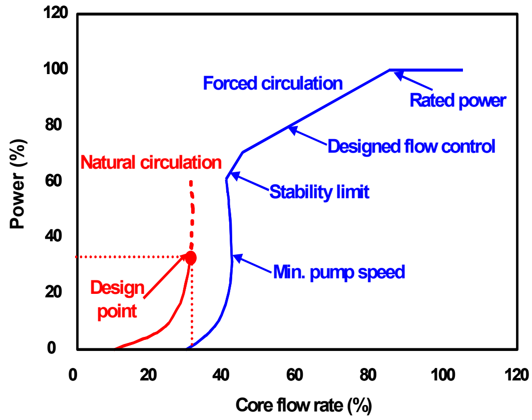
A part of the steam (about 106 t/h) is bled-off before the LP turbines and diverted to two TD pumps. The RO unit with the TD pumps produces potable water at a rate of 102×10^3 m^3/d. If the same amount of steam is used in an MSF process with huge heat exchangers, the water production rate will be up to 80×10^3 m^3/d [27].

Instead of back-up boilers, one MD pump is used in the RO system as a back-up pump, and could be used as the potable water demand rises. Pre-heating of seawater could be accomplished by feeding the condenser cooling water to the RO plant if required.

3.8.3. Economic perspective

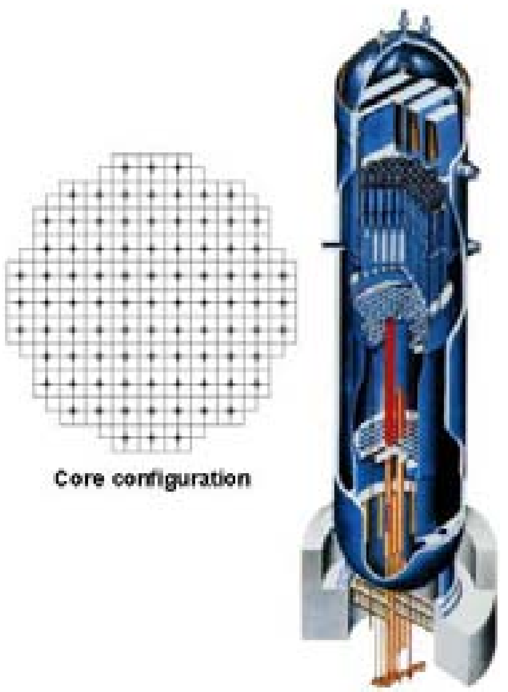
The preliminary study showed that TTBWR+RO is technically feasible. While TTBWR+RO generates electricity of 182 MW with conventional designs in the nuclear boiler and the steam cycle, it produces fresh water of about 102×10^3 m^3/d through an RO process. CDF of TTBWR+RO is as small as that of ABWR thanks to the reliable and well-proven safety systems. The possibility of radioactive contamination of seawater from the BWR coolant is physically eliminated owing to the use of TD pumps.

Further study is under way including a comprehensive economic evaluation. Although small sized nuclear plants generally have a disadvantage in economics, TTBWR+RO potentially overcomes it. TTBWR+RO only uses proven technologies and neither large R&D nor new investment in manufacturing facilities are necessary. Its natural circulation core with low power density results in a reduction not only in construction cost but also in maintenance cost of the recirculation system. The low power density lengthens refuelling intervals and consequently enhances availability of the plant. The rationalized safety system reduces the construction cost and facilitates the use of equipment of non-nuclear standard owing to their small sizes. Besides, the adoption of an RO process improves economics due to its high efficiency and no need of back-up boilers.



(100% of power: 1593 MWth, 100% of core flow rate: 6.4×10^3 kg/s)

FIG. 3.40. Core flow map of BWR/4.



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

FIG. 3.41. Major characteristics of "TTBWR+RO" nuclear boiler.

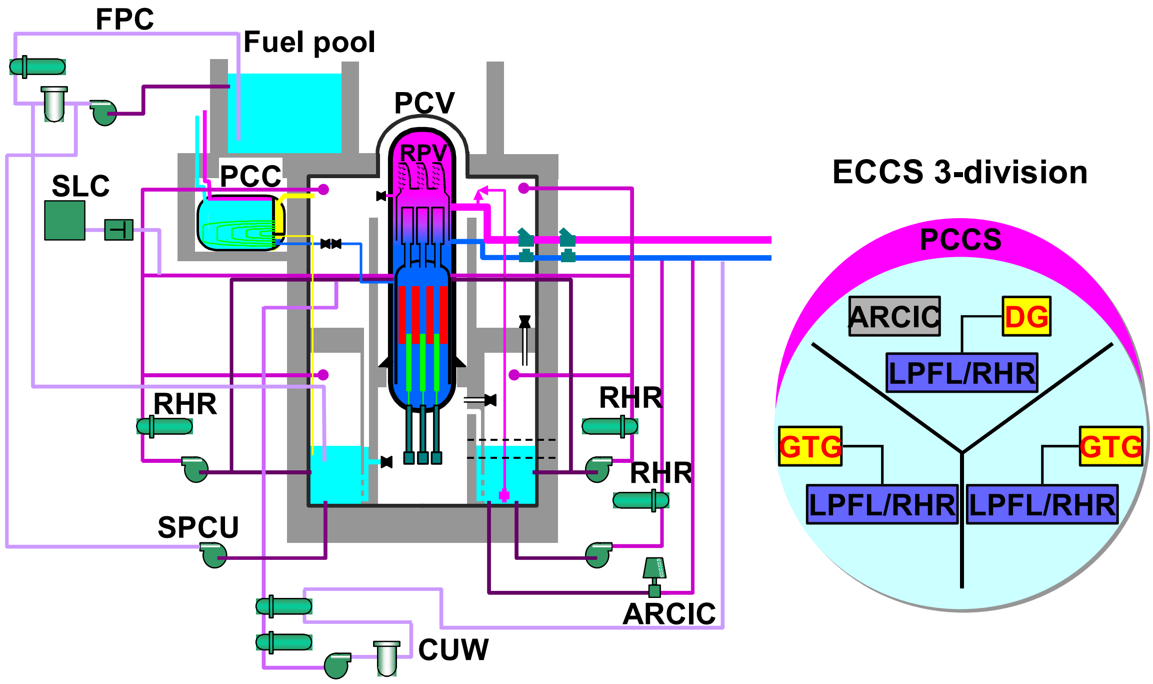


FIG. 3.42. TTBWR+RO safety system and ECCS division.

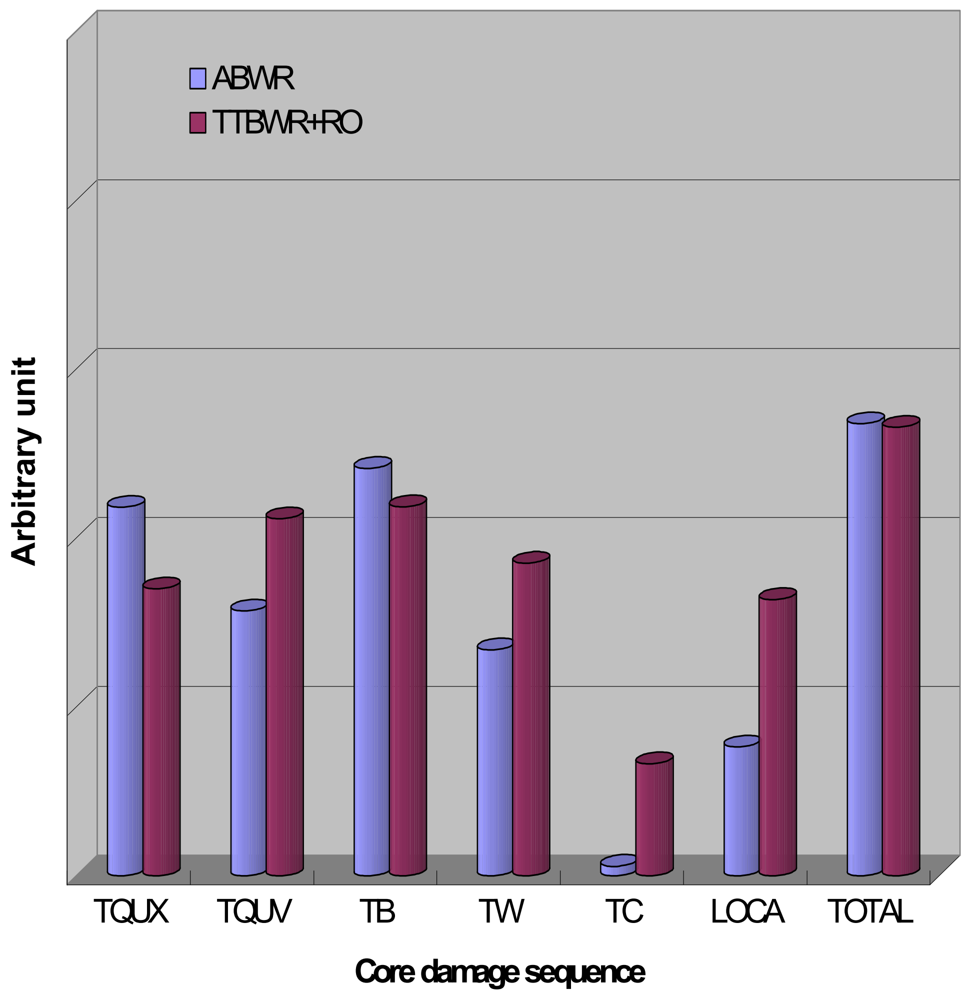
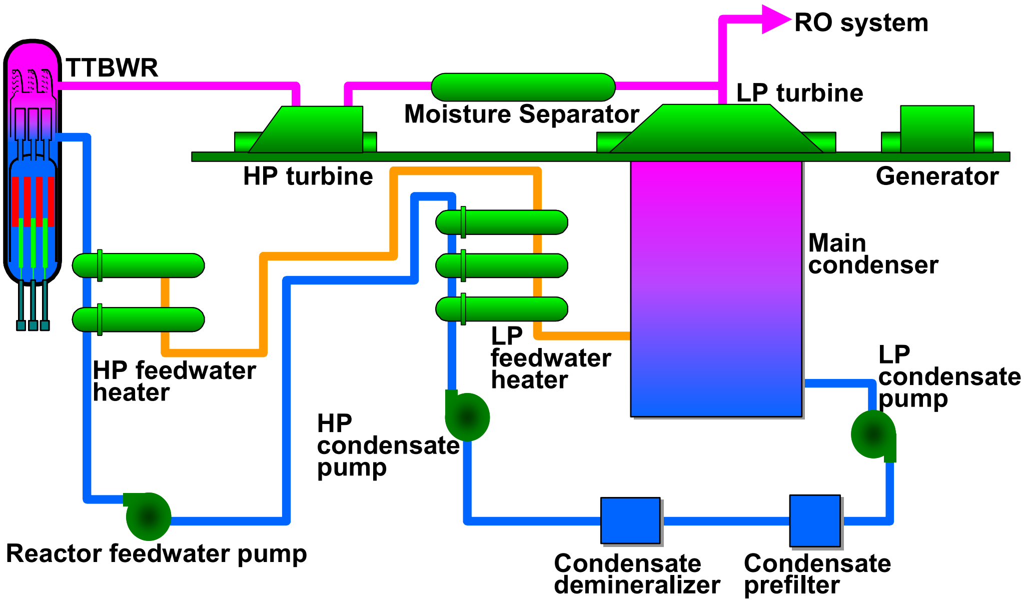


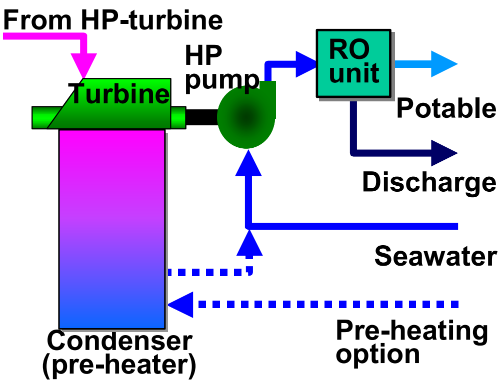
FIG. 3.43. Comparison of CDF between ABWR and TTWR+RO.

3.9. Helium-cooled PBMR with RO and MED (South Africa)

3.9.1. Background

The South African PBMR is currently being designed for the sole purpose of electric power production (100–110 MW(e) per module). Construction of the first such plant called a Demonstration Module, is expected to commence in early 2002 and commissioning is expected sometimes during 2005–2006. Modifications to the design would not be required in order to enable its waste heat stream (seawater at 40°C) to be used as feedstock for a reverse osmosis desalination process. Minor design modifications, with no impact on performance, would be required to increase the temperature of this waste heat stream required to serve as feed stock for a vacuum and evaporative desalination process. Hence the future application of PBMR to a cogeneration form of desalination is seen as an obvious future development. The fact that the primary heat stream from PBMR is helium gas at 900°C, economically precludes, however, the use of this type of plant for desalination in anything but a cogeneration mode.





RO system

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

FIG. 3.44. BOP of TTBWR+RO.

3.9.2. Design description

Referring to Figure 3.45 (a simplified schematic of the basic PBMR thermodynamic cycle) it can be seen that the PBMR overall thermal process disposes — almost all of its waste heat via two large helium gas to buffered demineralized water heat exchangers; the Pre-Cooler and the Inter-Cooler. The cooling water to these heat exchangers is in turn cooled by a common demineralized water to seawater heat exchanger, with a common demineralized water circulating pump supplying the Pre-Cooler and Inter-Cooler in parallel (with approximately equal flows to each). Under full load conditions the helium and water temperatures flows to and from these heat exchangers as shown on the diagram.

Several additional important comments should be made:

- The helium flows and temperatures shown are fixed by the Brayton cycle. The cold helium temperatures should not be allowed to increase except as matter of absolute necessity, i.e. in the event of a very high heat sink (seawater ambient) temperature.
- All water flows and temperatures shown are as per the current PBMR Demonstration Module design, but could be readily customized for any specific application. The seawater ambient temperature of 18°C represents the normal maximum for the proposed site of the Demonstration Module; on the South Atlantic Ocean a few kilometers north of Cape Town.
- A helium gas to seawater thermodynamic heat balance on the temperatures and flows — shows a discrepancy; i.e. less heat in than out-due to the fact that a number of other relatively minor heat sources (generator cooling and other minor cooling applications), are cooled by the buffered demineralized water loop, are not shown.

Potential flexibility in the configuration of PBMR's cooling system

Under full load conditions PBMR obviously has a fixed amount of heat, which must be dissipated. This heat removal process is clearly a function of coolant flow versus coolant temperature rise. From the flow and temperature figures previously noted, this may be approximately* represented by:

* Approximately, since as water temperatures increase so will radiation losses.

$$Q = 32.56 \div (T_2 - T_1) \textit{ or } T_2 = T_1 + (32.56 \div Q)$$

where: Q = Sea Water Flow (m^3/sec)
T_1 = Seawater Supply (ambient) Temperature (°C)
T_2 = Seawater Discharge Temperature.

Clearly there are finite limits to the temperature to which the respective cooling water loops may be heated. The buffered demineralized water coolant loop operates only marginally above atmospheric pressure in the Pre-Cooler and Inter-Cooler. This is to ensure that any possible leakage is always helium outward rather than water inward. It is essential to ensure that no localized boiling is allowed to take place within these heat exchangers-though no absolute limit has been calculated. The assumption should be that, the maximum possible seawater discharge temperature is in the order of 80^o C.

Coupling of PBMR to Desalination

Coupling of PBMR to desalination can be accomplished in a number of ways, as follows:

Using the Electric Power Generated by PBMR to Power a Reverse Osmosis Desalination (No use of PBMR's Waste Heat): Although this cannot be categorized as Nuclear Desalination, it is mentioned only for the purpose of emphasizing that by — direct electrical coupling of a power generation plant to a large power user (without going through a transformation/transmission/distribution process), the large power user (the RO Plant) should generally be able to enjoy substantially lower power tariffs than would otherwise be the case.

Using PBMR's cooling water discharge stream as the feed stock for an RO Desalination process and powering the RO process with electric power from PBMR: As was noted in above, via customized design of PBMR's helium to demineralized water and demineralized water to seawater heat exchangers, the seawater stream can be heated to any reasonable desired temperature to serve as preheated RO feed stock. The only physical coupling between the two plants is a pipe connection between PBMR's water discharge and the RO plant's water intake. This pipe connection would incorporate a "dump valve" to enable PBMR to continue normal operation in the event that the RO plant was non-operational.

Using PBMR's cooling water discharge stream as the feed stock for a Vacuum Evaporative Desalination process and powering the desalination process with electric power from PBMR:

The coupling of the two plants for this purpose would be exactly as described above for RO plant coupling. The only difference is in the design of PBMR's were there are various heat exchangers, so as to optimize the feed stock temperature to the desalination process utilized.

3.9.3. Economic perspectives

In the above, two potential practicable means of coupling PBMR with desalination were discussed: RO using PBMR's waste heat to preheat the feed stock and Evaporative Desalination using PBMR's waste heat as the source of thermal energy for the evaporative process. Both of these would utilize some small portion of PBMR's electrical output to serve their electric power needs (with RO option the more economic one).

RO with feed preheating:

If it is assumed that the optimum feed stock temperature for RO is 35°C and that the seawater ambient temperature for a hypothetical location is 20°C, and it is further assumed that all of one PBMR Module's cooling water discharge is diverted as RO feed stock, this feed stock flow would amount to approximately 2.17 m^3/sec. Assuming 40% recovery of desalinated water, this would amount to a product flow of some 75 000 m^3/day. If one further assumes an electric power consumption of 4 kilowatt-hours per cubic meter of product water, and the cost of power at 2.5 US cents per kilowatt hour the total energy cost would be US$ 7500 per day or 10 US cents per cubic meter of product. All of these projections are regarded as "order of magnitude" approximation.

The above would, in reality, require no heat exchanger or other modifications within PBMR as currently designed. The desired feed stock temperature could be obtained by dilution of PBMR's cooling water discharge, as presently designed, with additional cold seawater as required.

Evaporative Desalination Using PBMR's Waste Heat Stream as Feed Stock and Thermal Energy Source: Many evaporative technologies exist, with vastly different performance parameters. If the same raw water temperature of 20°C is assumed, and a seawater discharge temperature from PBMR of 80°C is assumed, then a feed stock flow of approximately 0.5 m^3/sec could be achieved. Depending on the process selected, a recovery rate of 25% treated water is possible, and this would amount to 10 800 m^3/day of product water. If it is further assumed that electric power is consumed at the rate of 1.2 kilowatt-hours per cubic meter of product, water and cost of power at US 2.5 cents per kilowatt hour, the total energy cost would be $US 325 per day or 3 US cents per cubic meter of product water. This amounts to roughly 14% of the product flow of the RO option, at approximately 30% of the energy cost per cubic meter of product using the RO option.

Again, these figures are "order of magnitude" approximations only, and do not take into account, for instance, the write-off of the capital cost of modifications necessary to PBMR's heat exchangers to enable the feed stock to be heated to 80°C. Also excluded are costs of write-off of any other capital expenditures as well as maintenance and operating costs (aside from energy costs).

The purpose of the above simplest analysis is to illustrate the possibilities that exist for integrating PBMR with large-scale desalination processes.

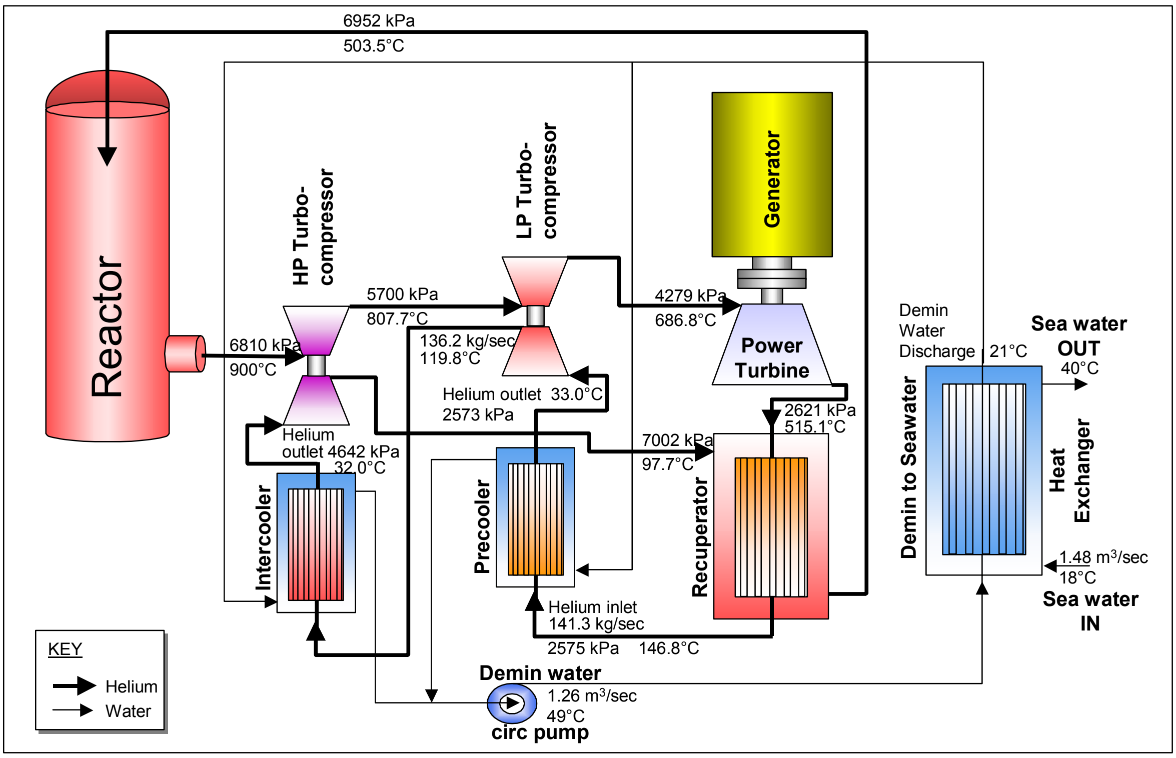


FIG. 3.45. PBMR thermal cycle (Koeberg siting).

3.10. Helium-cooled GT-MHR with MED

3.10.1. Background

The GT-MHR (Gas Turbine Modular Helium Reactor), is an advanced high temperature gas cooled reactor which is jointly being developed by a consortium including Minatom of Russia, General Atomics, Framatome and Fuji Electric with the goal of burning weapons grade plutonium. It can, however, operate on uranium fuel and be competitive as a stand-alone electricity producer. By design, it releases waste about 100°C; and the recovery of “free” heat for desalination lowers the price of the product water by a factor of 2, making the combination of the GT-MHR and an MED unit a very attractive set economical option (see Figure 3.46 for a simplified illustration).

3.10.2. Design Description

Nuclear Reactor

The Nuclear Reactor has a 600 MWth core with micro particle fuel included into prismatic fuel elements. This type of core has been successfully employed in the Fort Saint Vrain plant in the US. In the modular design, the safety of the concept is simplified by use of natural phenomena such as thermal radiation, which in any event maintains the fuel temperature below the temperature that leads to silicon carbide cladding damage. This ensures that the nuclear material is confined within the fuel all the time. (See main characteristics in Table 3.22.)

With helium as a coolant, that core is coupled directly to a gas turbine in a Brayton cycle (see Figure 3.47 for the principle diagram.). Helium at 850° C is expanded in a turbine that drives two compressors and an alternator yielding a net electricity production of 285 MW(e) for an efficiency of 47.5 %.

A special feature of the Brayton cycle, optimised for our operating conditions is the release of heat at the cold source via the precooler and intercooler at more than 100°C. Normally, this heat is released only through a cooling tower or to the river, but with proper adaptation it can be converted to useful heat to be used, for example to heat the feedwater of an MED desalination unit.

Desalination and coupling

As illustrated in Figure 3.48[11], an intermediate loop transfers heat from the precooler and intercooler to the MED unit transforming the sensible helium heat into water latent heat at around 67 to 73° C. Indeed, the MED unit should have an upper operating temperature in that range in order to avoid scaling problems. The MED unit is not described here, as it is a usual one unit, with specific design changes.

3.10.3. Economic perspectives - Example of water cost estimate

The following is an example of water test analysis that has been worked out with the help of SIDEM, “Société Internationale pour le Dessalement de l’Eau de Mer”, the main French Company in the field of desalination. Here are some remarks that are in order:

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[11] Please note that a heat exchanger at the outlet of the intermediate water circuit was not added in Figure 3.48 due to an aleardy crowded figure. The figure provides only an overall symbloic representation of the plant.

- the assumption of a seawater temperature of 35°C in the Gulf area is penalizing performance, and a more typical temperature of 25°C would significantly increase performance (by about 25%);
- the price of electricity, the second largest contributor in the price of water, has been taken at 5 US cents/kW·h. If the electric plant company owns the desalination plant, a price close, to 2.5 to 3 US cents would be more appropriate.

This is a single case example in order to show what kind of performance can be expected by coupling a GT-MHR and an MED plant. In this case we obtain very high quality water (~ 10 ppm TDS) that can be mixed with other water for use in agriculture, for example, thus decreasing the water price even further.

3.11. Shore-based lead-bismuth cooled SVBR-75 with RO and MED (Russian Federation)

3.11.1. Background

For several decades the lead-bismuth cooled nuclear submarines (NS) were designed and constructed in the Russian Federation under IPPE scientific supervision. This innovative nuclear power technology, which has no analog elsewhere in the world, has been demonstrated in industrial application. Land-base nuclear power plants can now utilize this technology. Given the amount of experience gained through prior research, the natural properties of the coolant, and the reactor's physical features, it is possible to design reactor installations (RI), which meet the strictest safety standards. These RIs do not require many safety systems, or accident localizing systems, thus lowering the nuclear power plant's (NPP) construction costs and simplifying its operation. On the basis of this technology the nuclear desalination power complex (NDPC) with RI SVBR-75 is proposed for countries which require additional potable water supply.

NDPC includes nuclear, electric power and desalination installations and satisfies the stringent safety and environmental requirements. Closeness of the scale factor of RI SVBR-75 to that of NS's RIs [31, 32] (see Figure 3.49) makes it possible to use practically developed technical solutions from that system and reduce the scope of R&D.

The design and operation experience of NS's RIs was used in designing RI SVBR-75. The total operation time for these reactors (along with the ground reactors-prototypes) is 80 reactor-years. When the RI was designed, all of the prior accidents that had occurred were taken into account, and design faults of the RI were eliminated [33].

3.11.2. Design description

SVBR-75 Reactor

Safety Concept

The proposed RI with a fast reactor cooled by LBC satisfies the highest safety requirements due to the inherent safety properties of this reactor [34]:

- The high boiling point and latent evaporation heat practically eliminate the possibility of primary circuit over pressurization or reactor thermal explosion during any conceivable accident as the pressure does not increase.
- Impossibility of coolant's boiling during any accident enhances reliability of heat removal from the core and safety due to lack of the heat removal crisis phenomenon.

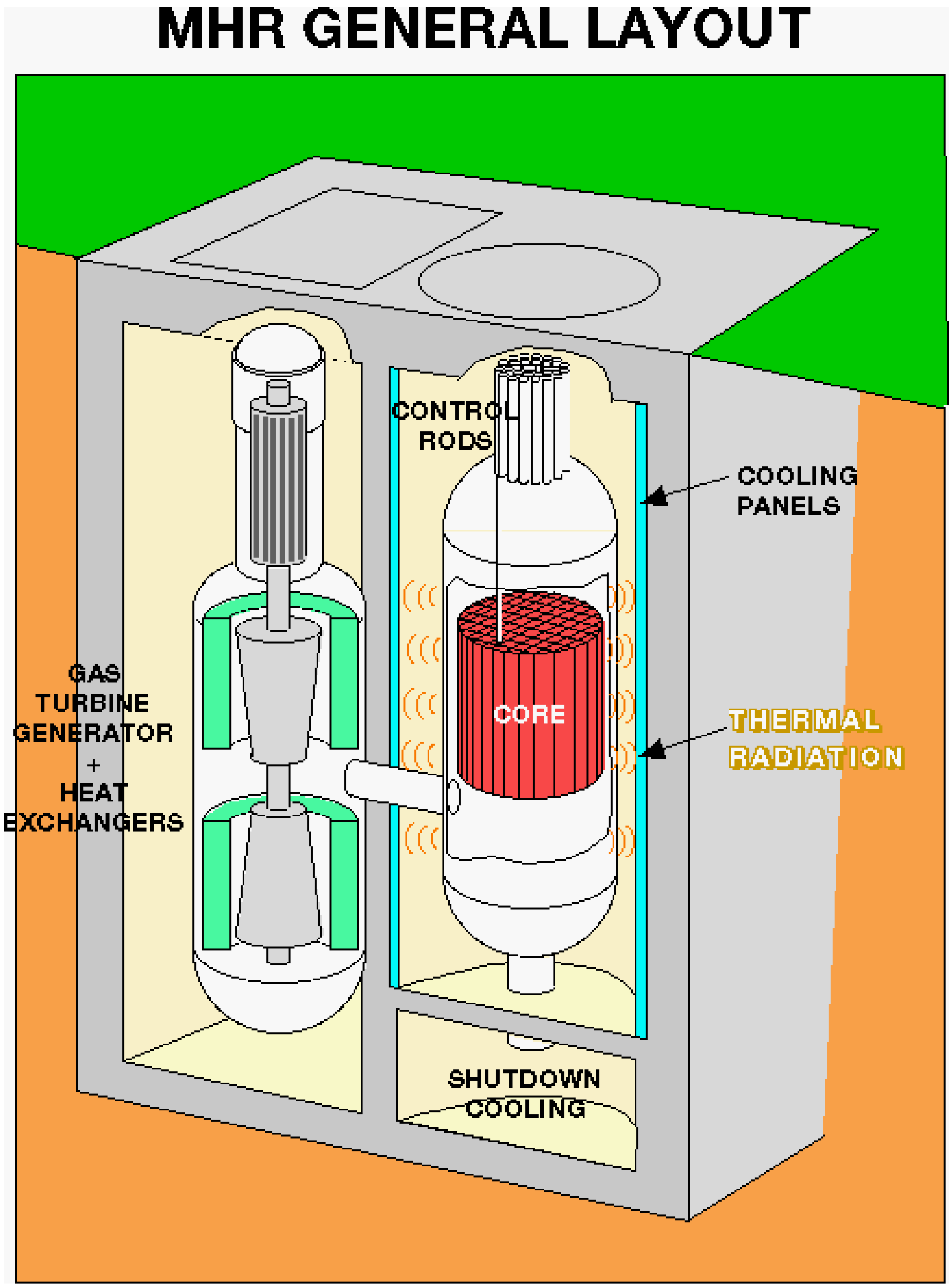


FIG. 3.46. GT-MHR power conversion process flow diagram.

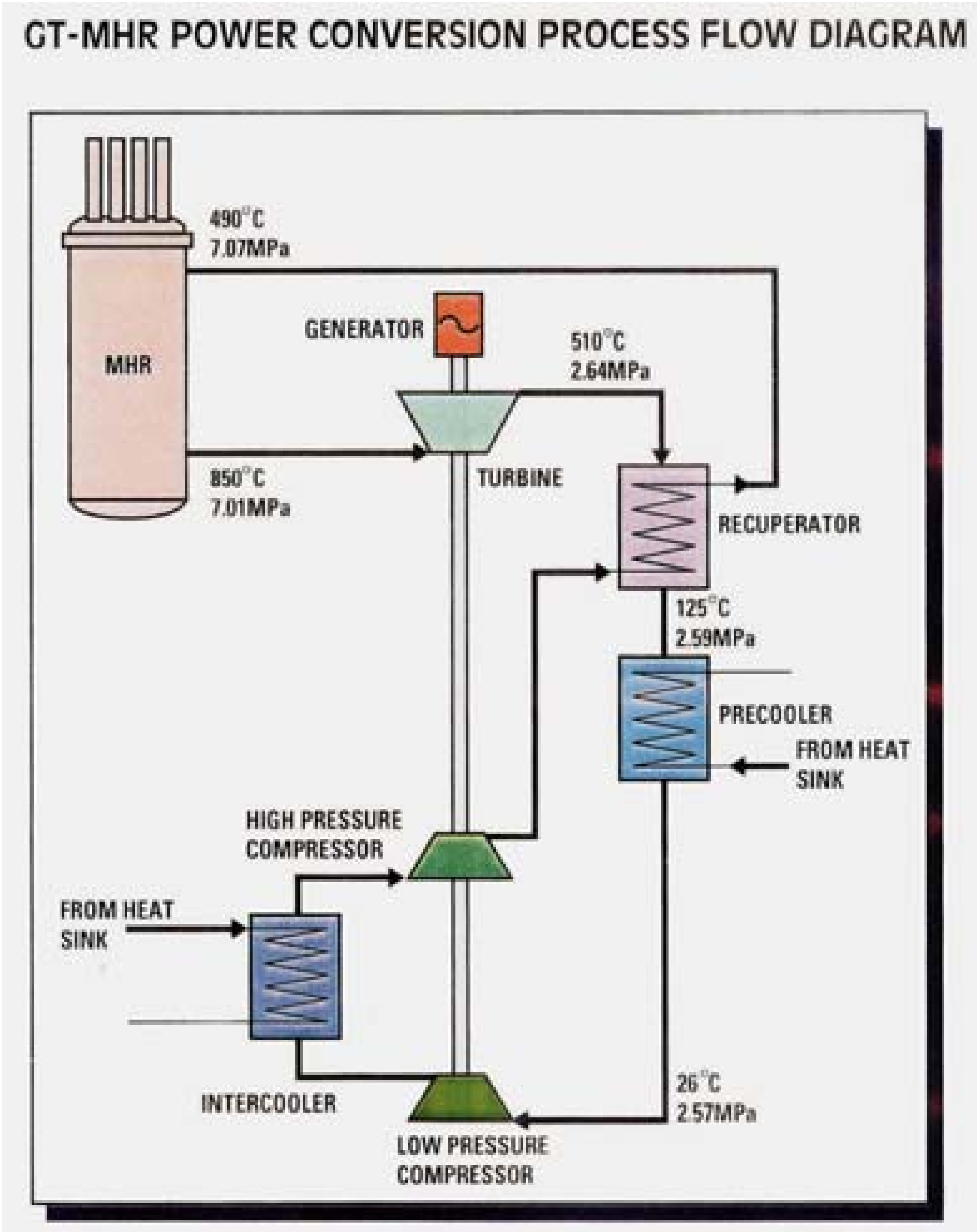
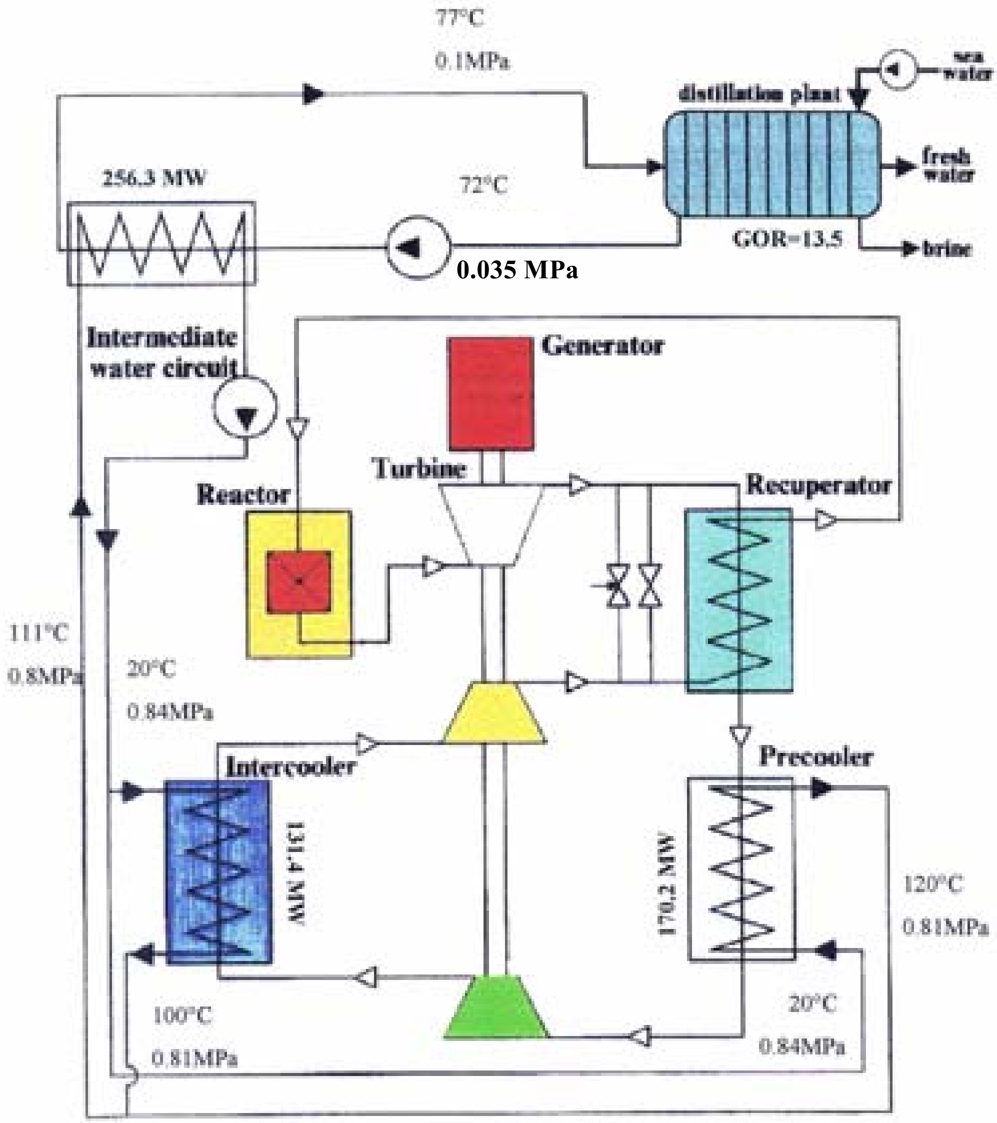


FIG. 3.47 GT-MHR power conversion process flow diagram.



1 - Principle diagram of the GT-MHR plant coupled with a MED desalination plant

FIG. 3.48. Principal diagram of the GT-MHR plant coupled with a MED desalination plant

For the proposed integral (monoblock) design of the RI, a loss of coolant, with interruption of coolant circulation through the core, caused by leakage from the reactor main vessel (postulated accident) is eliminated by introduction of a safe-guard vessel with a small free gap between the main reactor vessel and the safe-guard vessel. In case of leakage from the primary circuit gas system, coolant loss is eliminated by impossibility of the coolant's boiling off.

- In the case of failure of all cool-down emergency systems (a postulated accident), elimination of core melting caused by heat decay effect and keeping the vessel of proposed reactors intact are ensured completely by passive way with a large margin to boiling due to heat accumulation in in-reactor structures and coolant with short-time increase of its temperature. In this case, heat is removed through the reactor vessel to the water storage tank around the reactor vessel.
- In the case of emergency overheating and simultaneous postulated failure of emergency protection system (EPS), the reactivity negative feedbacks ensure decreasing the reactor power down to a level that does not cause the core damage.
- Within the core and RI there are no materials that release hydrogen as a result of thermal and radiation effects and chemical reactions with coolant. The coolant itself reacts with water and air only very slightly, and the coolant's contact with them would only be caused by the circuit's leakage. Therefore, the likelihood of chemical explosions and fires as internal events is virtually eliminated.
- Elimination of water or steam penetration into the core caused by SG leak and consequent over pressurization of the monoblock vessel, which was designed to withstand maximum possible pressure under this condition, are ensured by the coolant's circulation scheme. This scheme provides that steam bubbles and water drops are thrown out on the free coolant level by up going coolant flow. Thereby, steam effective separation occurs in the gas space of the primary circuit above the coolant's level, whence steam goes to the system of emergency condensers. In the event of postulated failure of emergency condensers, steam goes to the bubbler through the bursting membranes.
- The operating experience of LBC cooled RIs at the NSs has demonstrated the possibility of RI safe operation for a certain time period, under conditions of a small SG leak that does not cause any significant deviations from design technical parameters. Due to this fact, necessary repair work would not be urgently required; it can be carried out at a convenient time.
- Coolant's chemical inertness, the impossibility of its boiling in case of primary circuit leak, its property to retain iodine (the radionuclide that, as a rule, represent the major factor of radiation risk just after an accident), as well as the other fission products (inert gases are an exception) and actinides, sharply reduce the scale of radiation consequences of such an accident.
- The dry dock building, in which the RI is installed, is an additional safety system barrier against external events. Low margins of potential energy in the primary circuit restrict the scale of RI destruction to only external impact forces.

RI Safety Systems

(a) Emergency Protection System

The system consists of the two subsystems for bringing the reactor to the sub critical state:

– The emergency protection rods subsystem. Their operating mechanisms drop the rods in case of de-energizing the electromagnetic clutches in response to EP signals. EP rods are designed with fusible seals that allow for dropping EP rods into the core when LBC temperature exceeds safe operation level in case of mechanical damage of the operation mechanisms;
– The subsystem of the operating group of rods for reactivity compensation. Their operating mechanisms have springs that provide for drop of the rods in case of de-energizing the electromagnetic clutches in response to the EP signal.

(b) Autonomous Cooling System (ACS)

The autonomous cooling system (ACS) is a protection safety system and is designed for cooling the reactor core at any time in life and from any power level for any accident considered to be within the design basis. The ACS does not use the turbine installations of the complex.

Each ACS channel provides reactor cooling without exceeding the fuel elements' damage limits that are established for accidents that are considered as within the design basis.

When operating at power levels within the normal operational limits, the ACS is in a waiting state.

If the SG pressure exceeds the given limit (see Figure 3.52), the once-through drain valves at the condensate overflow to the SG separators are opened. The secondary circuit steam is transferred to the heat exchanger. Distillate of the cooling circuit of the RI equipment circulates constantly through the heat exchanger. Steam is condensed in the ACS heat exchanger and the condensate drains back to the SG.

(b) System of Passive Heat Removal

The system of passive heat removal (SPHR) is a technical means designed for overcoming the postulated accidents considered as beyond the design basis. It includes coincidence of any combinations of such postulated events as failure of all secondary circuit equipment with simultaneous failure of the primary circuit circulation pumps, failure of the reactor's EP system, leakage of the main reactor vessel, total blacking out the complex. In these cases, SPHR ensures passive heat removal from the core through the reactor vessel.

The SPHR design (see Figure 3.53) includes the water pool, in which the monoblock is installed, heat exchanger installed in the pool, air cooler that is installed on the external side of the floating unit vessel and the water natural circulation circuit which transfers the heat removed from the monoblock vessel to the air inside the protection dock. During normal operation, the SPHR is in readiness and removes the heat that flows from the monoblock vessel.

In case of an accident considered to be beyond the design basis and related to failure of normal operating systems and protection systems, the LBC temperature of the primary circuit begins to increase. Under the LBC natural circulation the reactor power begins to reduce due to the negative temperature reactivity effects. In the course of heating the monoblock vessel, heat losses increase, the water temperature in the SPHR pool increases. The power removed by the SPHR increases correspondingly. Step-by-step, the system achieves the state of heat equilibrium when heating is stopped and the total reactor power is removed through the SPHR. In this state the RI can exist for a long period of time.

As computations reveal, this regime is followed by temporary core heating, increasing the coolant temperature and the temperature of the monoblock vessel. However, core element damage does not occur.

The SPHR and the monoblock's safeguard vessel eliminate the coolant's losses which are prohibitive for the core cooling conditions and are caused by the postulated accident considered to be beyond the design basis and in which the basic vessel of the monoblock loses its tightness. In this case, self-localization of the leak within the safeguard vessel occurs. And then, when the leaking coolant comes in contract with the safeguard vessel's walls cooled by water (LBC solidifying temperature is ~125°C), it "freezes".

The volume of the gap between the monoblock's strong vessel and safeguard vessel determines the maximum possible coolant loss that might be caused by the strong vessel leakage. In the case of coolant loss, its level in the monoblock does not drop below the limit value that maintains the conditions for LBC natural circulation in the monoblock.

(c) SG Leak Localizing System

This is the localizing safety system. It is designed for removing steam out of the monoblock's gas space in case of an accident involving an SG intercircuit leak. The system is designed for guillotine rupture of one SG tube. At the same time, this system is a technical means for overcoming the accident considered to be beyond the design basis and including the postulated guillotine rupture of several (more than one) SG tubes.

In case of an accident, the emergency condensers condense the steam phase of the steam-gas mixture in the amount that corresponds to the guillotine rupture of one SG tube. In this case, the gas system pressure does not exceed 0.5 MPa. When the emergency condensers condensate collectors are filled, the condensate is transferred to a special condensate tank – see Figure 3.54.

In case of the beyond design accident such as the rupture of more than two SG tubes, the pressure in the system and in the monoblock increases. When the pressure reaches 1 MPa, the membrane-safety mechanism operates. Through this mechanism the steam-gas mixture is transferred to the bubbler, in which the steam phase of the blown off mixture is condensed and non-condensed gases are dumped through the filters to the reactor compartment.

(d) Localizing System

The RI localizing system is a multi-barrier shield against proliferation of radioactive products into the environment. It includes:

- Fuel matrix of fuel elements;
- Cladding of fuel elements;
- Coolant;
- The walls of the coolant's primary circuit, which include the main vessel of the monoblock, safe guard vessel of the monoblock, vessels of equipment elements and the gas system pipelines;

– Gas-tight vessel of the floating unit;
– Protecting shell (containment) of the dry dock.

Concept of fissile materials nonproliferation

The proposed concept of the NDPC incorporates the following measures against fissile materials proliferation:

– Fuel enrichment of the RI does not exceed the values recommended by IAEA;
– No manipulations with fuel (both fresh and spent), no equipment for such works is provided at the NDPC site. After 10 years of operation the floating unit with the RI is towed for repair and refuelling to the Supplier's plant;
– Organizing and technical measures on RI physical protection should be provided at the NDPC site.

Electric power installation concept

As an NDPC component, it is the intention to use a steam turbine with industrial steam bleeding for an MED desalination installation. In doing this, the interdependency between produced electric energy and steam will be flexibly controlled from nil to the nominal power of each of these components. which will not depend on each other. It is further proposed to use the heat output of the turbine condenser for heating the feed seawater, which can then be fed into an RO desalination facility.

Coupling system concept

Low pressure in the SVBR-75 primary circuit allows for the simplification the coupling system and the use of water and steam produced by the turbine circuit for direct desalted seawater heating directly.

Regulatory rules and Russian standards for nuclear district heating installations (which can be applied to nuclear desalination) require four safety barriers between the primary reactor circuit and the water supply grid. Both hermetic walls between circuits and pressure drop between circuits may be considered as safety barriers.

Increasing feed seawater pressure for the RO system to values higher than the steam pressure in the turbine condenser, and increasing seawater pressure applied to the MED system, to values higher than the pressure of turbine steam bleeding for its heating, result in several possible safety barriers:

– SG tube wall between the reactor coolant circuit and turbine circuit;
– pressure drop on the SG tube wall (pressure of turbine steam and water is higher than pressure in the reactor circuit);
– wall of the heat exchanger for seawater heating between turbine circuit and MED circuit, and turbine condenser wall between turbine circuit and RO circuit;
– pressure drop on the walls of these heat exchangers (pressure of heating water and steam of turbine circuit is lower, than pressure of heated seawater in MED and RO circuits).

These features of SVBR-75 allow for simplifying the coupling system significantly and providing the radiation safety required.

NDPC concept

- All installations in the complex (nuclear, desalination, and electric power generation) are designed with a capability of flexible changing connections with each other and can operate autonomously or in the complex.
- In order to improve the reliability and provide continuous supply of electricity and fresh water during the periods of scheduled reconditioning works, alternative heat sources (reserve boiler-room and electric power grid) will be included into the NDPC.
- In order to provide a continuous supply of potable water in case of emergency, reservoirs for storing 2 operating days' worth of water are included in the design. It is possible to supply MED installations with steam from the reserve boiler-room; RO installations may be supplied by electricity from the grid.
- Stationary desalination and electric power generating units are located on shore. When constructing these units, local resources, industry, and personnel are maximized to reduce the cost and construction term. These NDPC parts are property of country-user.
- Shore-mounted NDPC design (in comparison with a barge-mounted one) simplifies reactor module protection against external impacts (like gales or tsunamis) and possible acts of terrorism (fighting scuba divers). Also it simplifies NDPC maintenance (excluding expanses to provide barge float age).
- Desalination and electricity generating units do not significantly affect the safety systems and do not impact RI safety.
- Reactor units are delivered as "Build-Own-Lease". This means that the supplier leases the reactor unit for the time period determined by the reactor core lifetime duration (~10 years). Such core lifetime will make it possible to keep stable costs for the NDPC products (potable water and electricity).
- Reactor is installed in the gas-tight and durable compartment of the floating unit with 1500–2000 tons of displacement. This unit includes all the systems necessary for safe operation of the reactor under design scenarios and passive safety systems for overcoming possible accidental situations. The compartment is separated from the environment; there is no discharge of contaminants from the reactor system.
- After being manufactured at the plant, the floating unit with the freshly refuelled reactor is towed to the nuclear desalination power complex and installed by sluicing in the closed dry dock, which protects the reactor unit against objects falling on it, as well as other design external events.
- On-shore operation with radioactive materials including reactor refuelling is not performed. For that reason, highly qualified maintenance personnel are required and the risk of plutonium proliferation is reduced.
- In the event of an accident, radioactive products are kept in the RI compartment and on-site radioactive contamination does not occur; no decontamination operations are required.
- At the end of its lifetime, the reactor unit is sluiced to the cooling compartment protected against external effects. It stays there for about a year until the level of residual heat release decreases and LBC solidifies (melting point ~125°C). Another reactor replaces it. After cooling, the supplier transports the unit with solidified coolant to the plant-manufacturer for refuelling, necessary repair works, and renewing the expired equipment.

– When withdrawing from operation, the reactor unit will be towed to the - manufacturing country after the necessary cooling time. After this removal no radioactive waste will be left on the NDPC site.
– The proposed NDPC concept makes it possible to decrease the investment risk to a level typical of non-nuclear projects.

NDPC general site layout is shown in Figure 3.50. A simplified flow diagram is presented in Figure 3.51.

Operational Concept

It is presumed that all control operations from the moment of first start-up to the final shutdown of the reactor will be automated to the maximum extent possible. When the NDPC control system is designed, the experience of designing high-automated NSs of the “Alpha” class, current achievements on control systems, reactor diagnosing systems, and RI inherent safety properties will be used. The number of the NDPC personnel will be ~70.

Three operators should control all NDPC installations from one hall and the shift head should coordinate their operations. The operators control actions should be minimal; operators should monitor the status of the installations and operatively change their power according to the system requirements. For example, start-up and shutdown of the RI should be performed automatically by pressing the button on the RI control desk.

Scheduled repair works at the NDPC are limited to its non-nuclear part. Equipment of the basic RI systems should be highly reliable, and not need replacement and maintenance during operation. It is only allowed to replace the separate elements, which do not need for carrying out radiation-hazardous work and which do not affect the RI safety.

Step-by-step, control of the complex is transferred to the User-Country’s- personnel. It is presumed that at the initial stages Supplier-Country operators will control the RI. User-Country’s trained personnel will work as the fieldworkers with the Supplier Country operators. Furthermore, after receiving a license for controlling the RI by Organization-User (OU), control will be transferred to the OU of the User-Country.

The RI Supplier is responsible for carrying out annual repair and maintenance, and elimination of accidents consequences or serious faults of the equipment. For that purpose, the Supplier has a mobile team of specialists.

The inherent safety properties ensure overcoming an accident event without drawing on operating personnel. Emergency situations are recognized automatically as parameters exceed their operational limits. In case of an emergency situation at the RI, the operator informs the RI Supplier representatives about the event. They make a decision about the possibility to continue RI operation and organize the necessary repair/reconditioning works.

TABLE 3.22. BASIC PARAMETERS OF THE NDPC

Thermal Power, MWt	250
Design electric power, MWt-e	75
Thermal efficiency	0.30
Electric power under nominal power of MED, MWt-e	50
RO electric consumption, MWt-e	12.5
Auxiliary, MWt-e	2.5
Net electric power, MWt-e	35
Construction terms, month	36
RI load factor	0.9
RI lifetime, year	60
RI operation term before reloading, year	10
Potable water output, m^3/day	80 000 (50% RO + 50% MED)
Fuel cost Capital outlays (only for RI)	Are taken into account only at rent
RI decommissioning expenses	Not required (see operation model)
Approximate rent cost for reactor module, USA $ million/year	6.5
RI approximate specific operation and service cost, $/MWt*hour	8.1
Approximate cost of electricity (including RI rent, turbine-generator set amortization cost, cost of RI and turbine-generator set operation and service), $/MWt*hour	25
Approximate contribution of RI to the cost of fresh water (RI rent, electric power according prime cost for RO — ~6 кWt/m^3 – turbine-generator set amortization, RI and turbine-generator set operation and service), $/$m^3$	0.24

3.11.3. Economic perspectives

– It is expected that all control operations from initial commissioning of the proposed NDPC concept will decrease the investment risk to a level typical of non-nuclear projects.
– Stationary desalination and electric power generating units are located on shore. When constructing these units, local resources, industry, and personnel are maximized to reduce the cost and time of construction. The designer and equipment supplier for non-nuclear part of NDPC is chosen by the user country for the specific site.
– Desalination and electricity generating units do not significantly affect the safety of the systems and do not impact RI safety.
– Reactor units are delivered as “Build-Own-Lease”. This means that the supplier leases the reactor unit for the time period determined by the reactor core lifetime duration (~10 years). Such core lifetime will make it possible to keep stable costs for the NDPC products (potable water and electricity).

From start-up to the final shutdown of the reactor operations will be automated as much as possible. All of the following elements will influence NPDC control system design: the experience of designing highly automated NSs of the “Alpha” class, current achievements on control systems, reactor diagnosing systems, and RI inherent safety properties will be used. The number of the NDPC personnel will be approximately 70.

All NDPC installations should be controlled from one hall by three operators and the shift head, who should coordinate their operations. The operators control actions should be minimal; operators should monitor the status of the installations and operatively change their power according to the system requirements. For example, start up and shutdown of the RI should be performed automatically by pressing the button on the RI control desk.

Scheduled repair works at the NDPC are limited to its non-nuclear part. Equipment of the basic RI systems should be highly reliable, and not need replacement and maintenance during operation. It is only allowed to replace the separate elements, which do not need for carrying out radiation-hazardous work and which do not affect the RI safety. Such approach allows minimizing RI operation expenses.

The RI Supplier is responsible for carrying out annual repair and maintenance, and elimination of accidents consequences or serious faults of the equipment. For that purpose, the Supplier has a mobile team of specialists. Economical risks due unplanned downtime are insuring and this insures fee is included to operation expenses.

Since the desalination installations do not significantly affect RI safety, their construction may be performed with maximal use of local industry and manpower to reduce the construction cost. Auxiliary boiler-house and power grid availability make it possible to commission desalination installations after construction and prior to RI readiness. This will enhance reliability of the complex in case of reactor unit failure. Depending on the available energy resources, the RO installation will be able to operate using electric power supply from the external power grid or the MED will be able to operate using steam supply from the back-up boiler-house in the event the reactor is off-line (Figure 3.55). To ensure water supply in abnormal situations, the NDPC is equipped with reservoirs to store desalted water supply equal to a two-day required capacity.

3.12. An advanced small sodium-cooled reactor (4S) for nuclear desalination

3.12.1. Background

"4S (Super Safe, Small and Simple)" is a small sodium-cooled fast reactor, which concentrates intensive efforts on meeting energy requirements in region where technical and engineering infrastructures are limited. To meet this objective, "4S" is designed on the principles of simple operation, simplified maintenance including refuelling, increased safety, and improved economic and proliferation resistant features. These technical features of "4S" are most effectively applied to a co-located co-production plant for electricity and water in coastal arid zones. "4S" can supply electric power to an RO for seawater desalination by an RO system and also for transportation of water to consumers.

3.12.2. Design description

"4S" is designed to have a long life core with a small diameter surrounded by an annular reflector to control the reactivity depletion due to burning and enhance the core safety. Its lifetime is set at ten years [34, 35], to eliminate the need for refuelling. A co-production "4S" plant can continuously produce fresh water for more than 10 years without nuclear refuelling. It has also high resistance to nuclear proliferation since there is no need to access to nuclear fuel.

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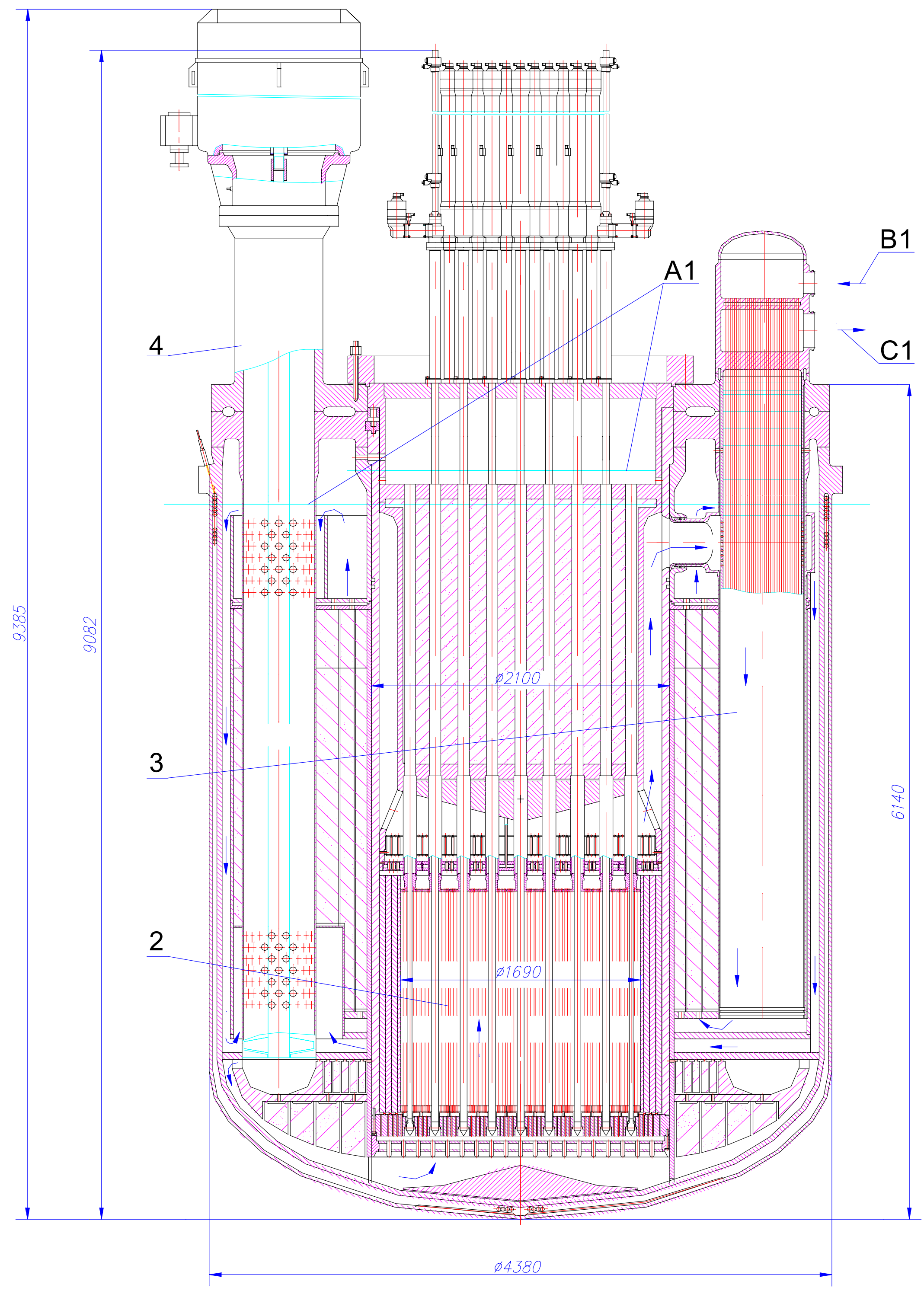


2- reactor;
3- SG module;
4 - main circulation pump;

A1- coolant level;
B1- boiler water inlet;
C1- steam outlet.

FIG.3.49. Reactor general view.

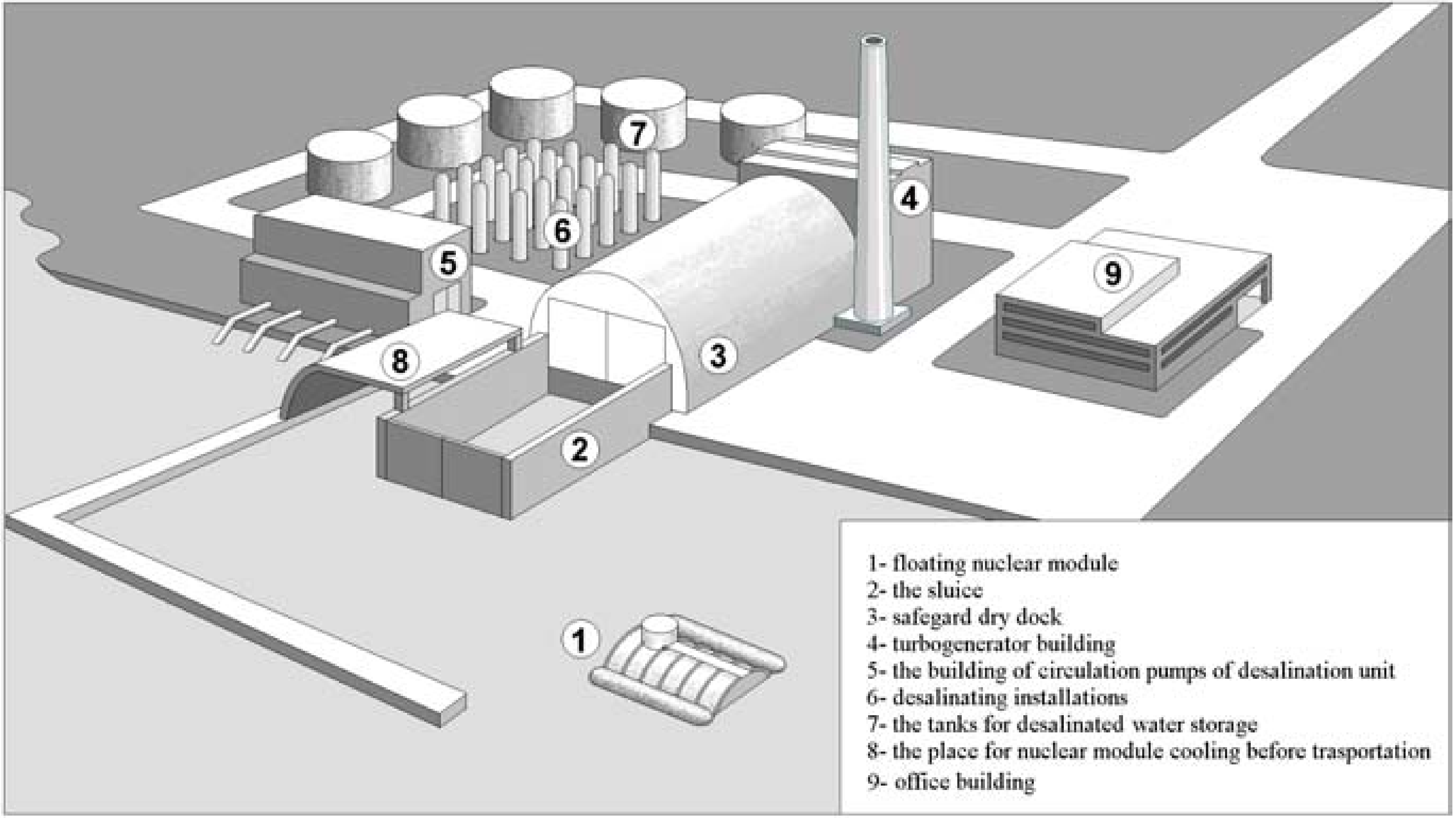


FIG. 3.50. Site of nuclear desalination power complex. General view.

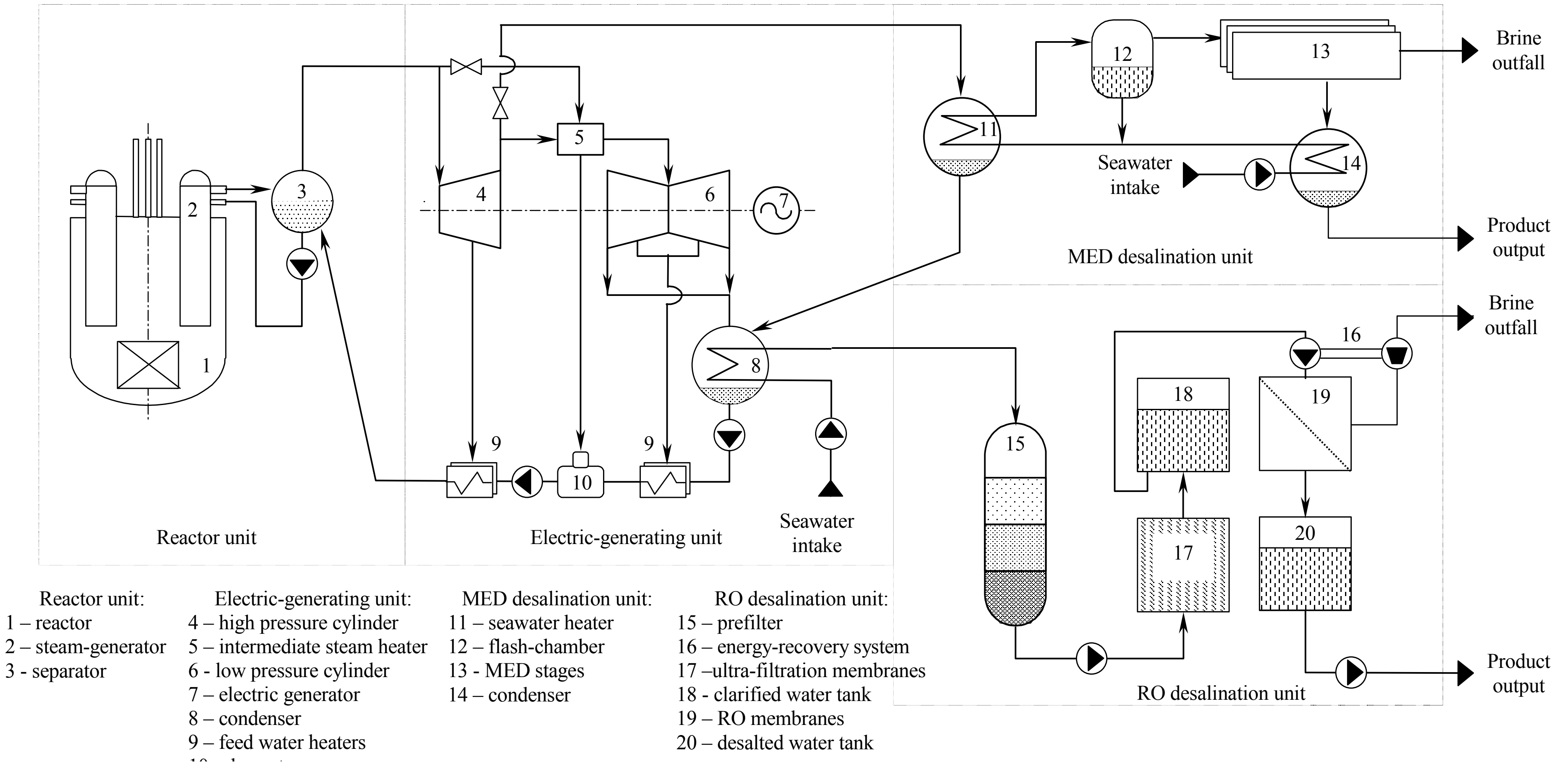


FIG. 3.51. Nuclear desalination power complex. Simplified flow diagram.

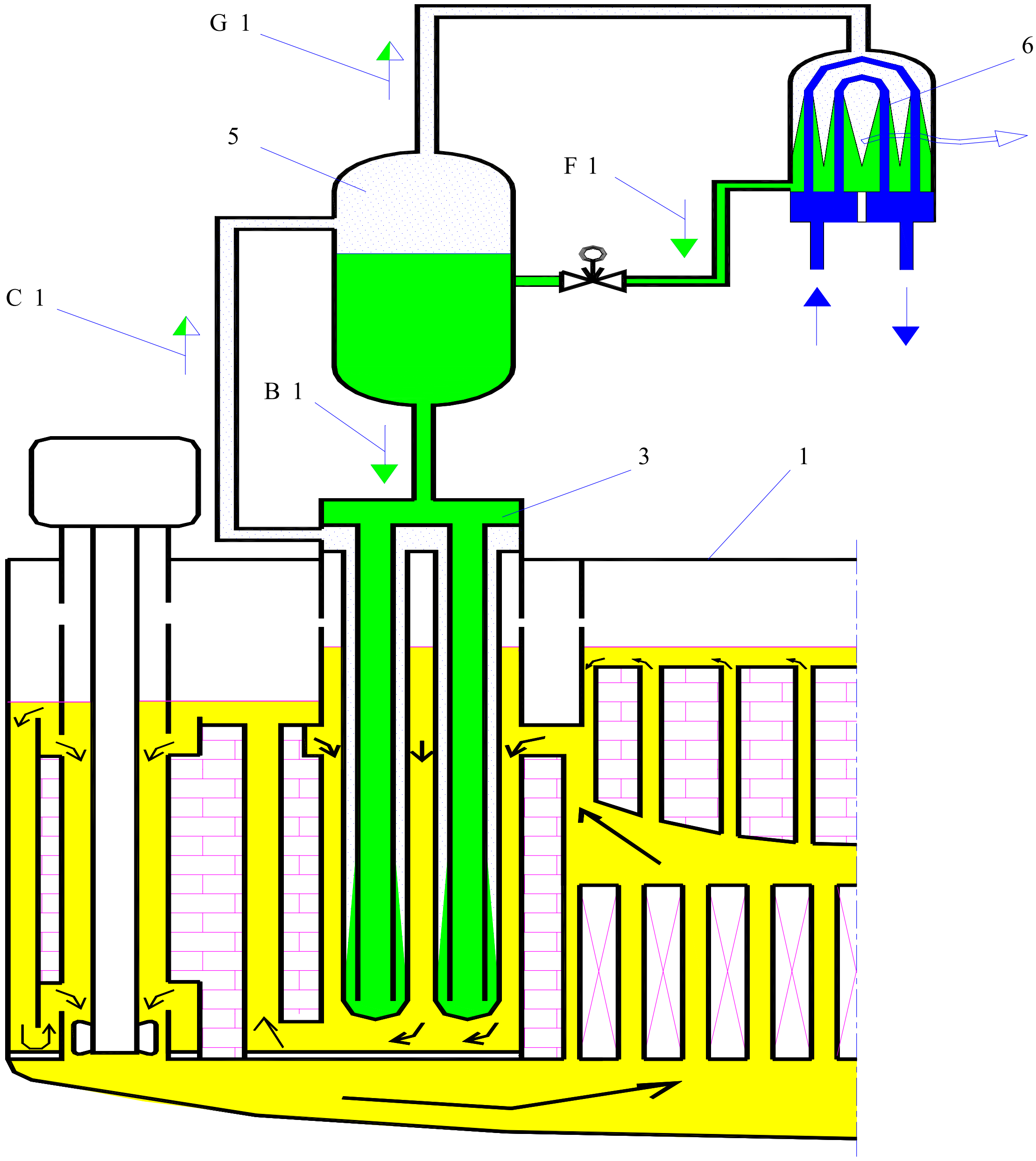
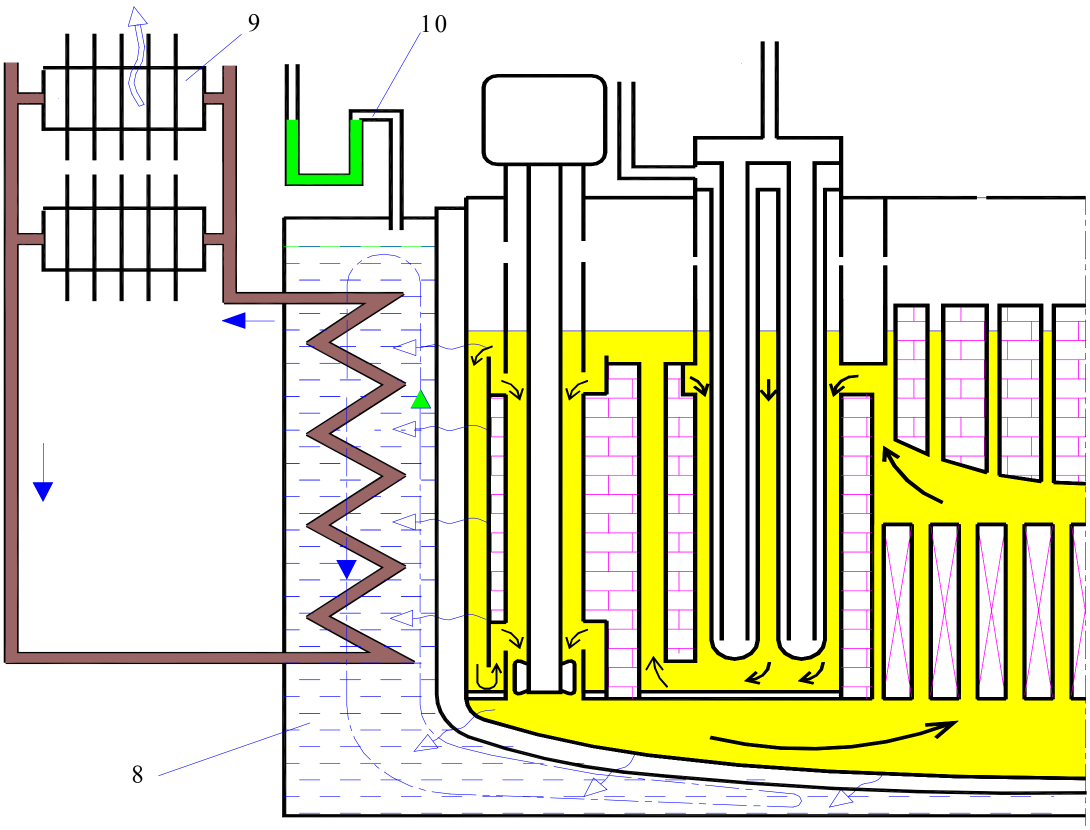


FIG. 3.52. Autonomous cooling system (ACS).

1 – monoblock;	*G1 – saturated steam;*
3 – SG module;	*C1 – steam-water mixture;*
5 – separator;	*B1 – boiler water;*
6 – emergency condenser;	*F1 – condensate draining .*



8 – Passive heat removal system (PHRS); 9 – air cooler; 10 – hydroseal.

FIG. 3.53. Passive heat removal system.

Regarding the desalination system, reverse osmosis (RO) system has been selected in view of its high efficiency as well as its recent technological development and the economical advantages compared with distillation systems. Another reason of selecting an RO system is that electric energy is necessary to transport the product water to consumers.

By co-locating with “4S”, RO units can be directly powered. Auxiliary systems like feed water intake or brine discharge, workshops, control and monitoring panels can be accommodated together for better economics. Where seawater temperature is relatively low, utilization of nuclear heat from the tertiary circuit will be also feasible for preheating feed water to improve the performance.

(a) Nuclear reactor

Major parameters of the reference design are shown in Table 3.23. The reactor building, as shown in Figure 3.56 is an embedded structure with seismic class A. It contains the reactor, secondary systems, a steam generator, a coast down control system, a power switchboard and the refueling pits. The plan of the building measures 26 m × 16 m, requiring only a small ground base.

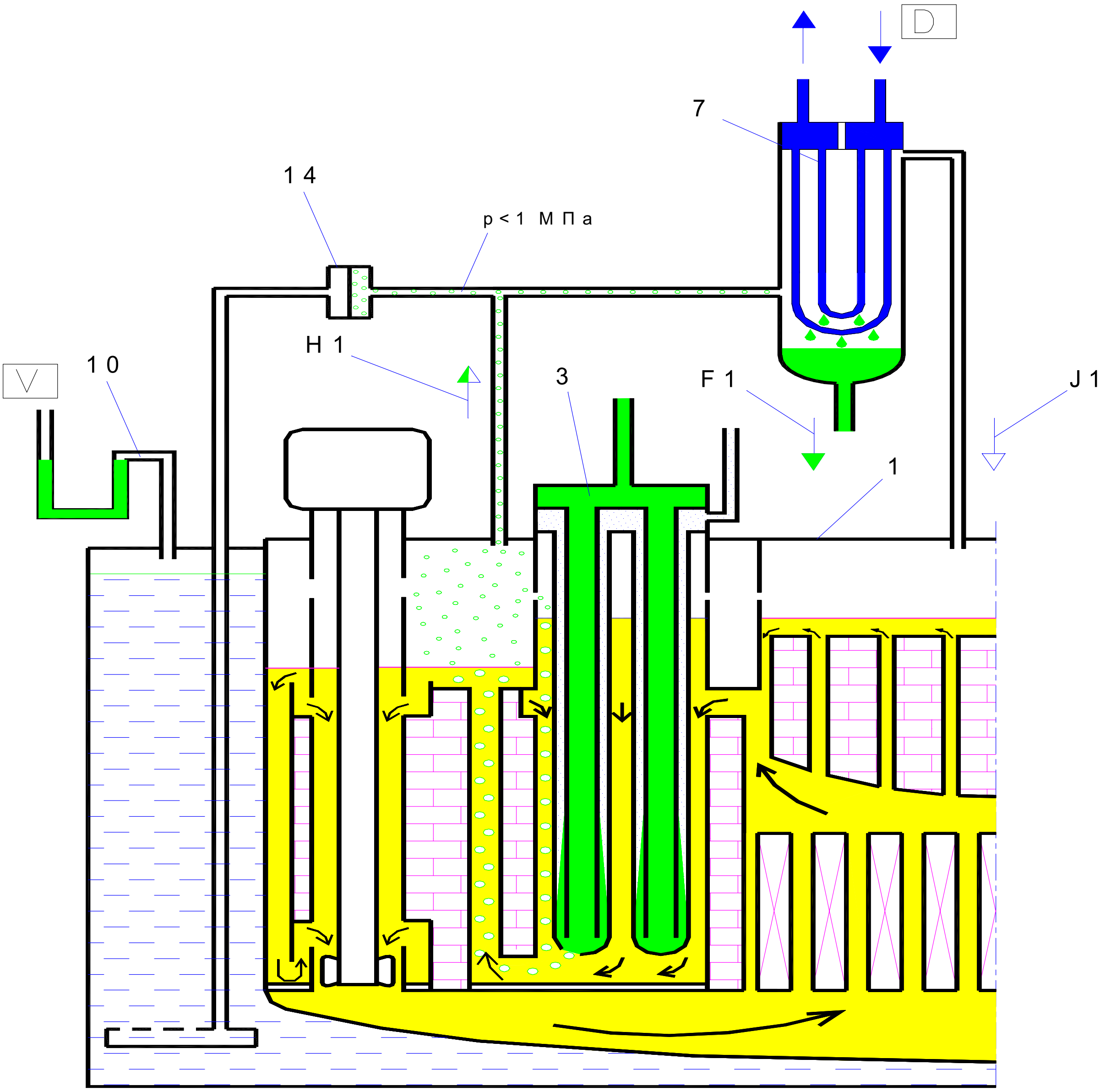


FIG. 3.54. SG intercircuit leak localizing system.

Small leak. Gas pressure P < 1 MPa.

1 – monoblock

3 – SG module

7 – emergency condenser

10 – hydro-seal

14 – membrane device

H1 – steam-water mixture

F1 – condensate draining

J1 – cover gas return

D – cooling system (intermediate circuit)

V – filter-ventilation system

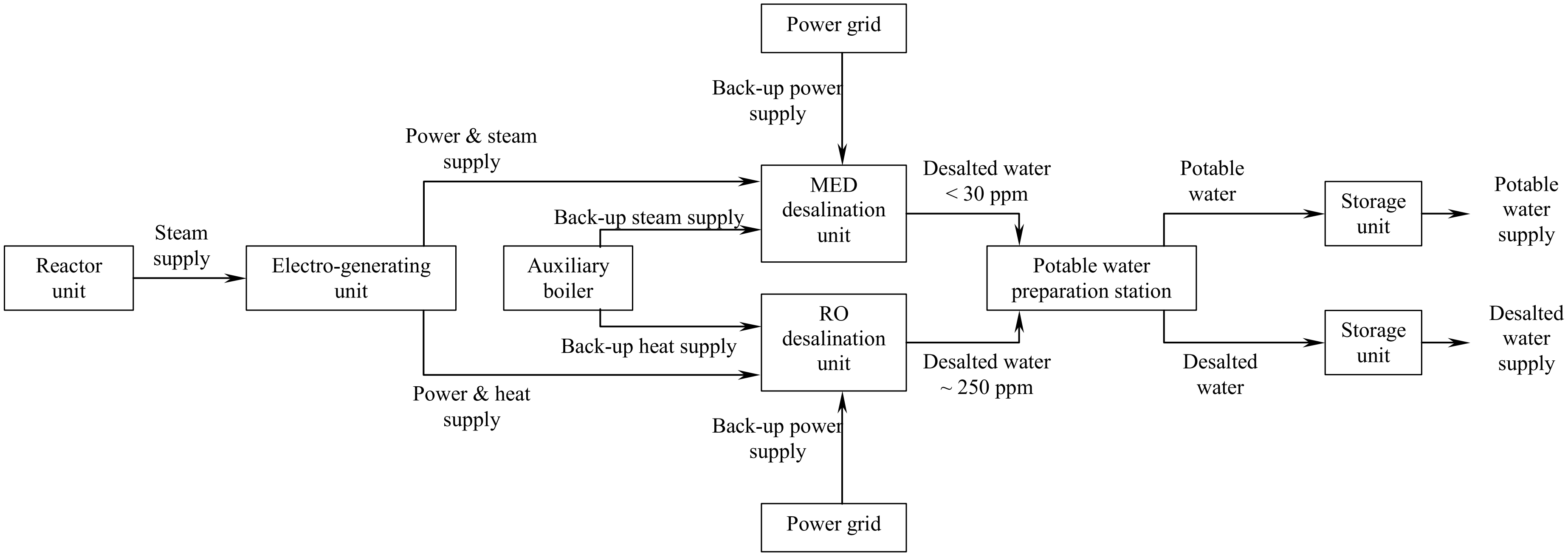


FIG.3.55. Main and backup power and heat supply of desalination plant.

The reactor system is shown in Fig. 3.57. Primary coolant flows out of the core, raises the hot pool and descends in the intermediate heat exchanger through which the heat is transferred to the secondary sodium. It is pressurized by the primary electromagnetic pump at the bottom of the intermediate heat exchanger and flows down along the inner hole of the in-vessel shielding structure. Then it turns at the bottom of the reactor vessel and returns to the core.

TABLE 3.23. MAJOR PARAMETERS OF REFERENCE DESIGN

Item	Specification
Core Power	
Thermal Output	125 MW(th)
Electric Output	50 MW(e)
Core Inlet/Outlet Temp.	355/510ºC
Reactor Assembly	
Diameter length	2.5 m 23 m
Thickness Material	20 mm
Reactor Vessel	304 SS
Core Barrel	Mod.9Cr–1Mo
Reflector Guide	Mod.9Cr–1Mo
Others	304 SS
Fuel	
Composition	U-Pu-Zr
Pu Enrichment(Ave.)	19.5%
Pu Fissile Weight	1.3 ton
Core	
Breeding Ratio	0.7
No. of Sub. Assembly	18
Reflector	
Material	Mod.9Cr–1Mo
Thickness	15 cm
Length	1.5 m

"4S" employs a burn-up control system with an annular reflector in place of the control rods and control rod driving mechanisms. This eliminates the need of frequent maintenance services. No replacement of the reflector is required for the entire plant life. The core geometry with a reflector control system has been chosen to meet requirements for negative void reactivity and no refuelling for ten years.

The reflector is driven hydraulically at the start up and the shutdown. At the start-up the reflector is driven upward at a rate of 1 mm/sec by the hydraulic pump. The reflector is fixed by the hydraulic and moves up for the burn-up control at a constant speed of 1 mm/day by a motor, which is designed so that the reflector is positioned by integration of generated power frequency.

For shutdown of the reactor, a scram valve is opened to let the reflector descend at a rate of 10 cm/sec down to one meter for the sub-critical cold shutdown state. Longer life core [36]

In order to enhance its applicability in developing countries, "4S" has a long core life with a single batch fuelling. The reference design (Table 3.23) has a ten-year life core. A longer core life can be achieved by introducing at the core center a burnable poison assembly, which contains a mixture of Gd (poison) and ZrH (moderator to soften the spectrum). This reduces the reactivity depletion of the core and extends its life to 30 years. The longer life core enhances proliferation resistance with no need of refueling or processing of plutonium.

The driving speed of the reflector is programmed to compensate the balance of reactivity.

(b) Desalination system

Three cases to produce drinking water are designed:

– a two stage RO system to produce drinking water meeting the EC standard (below about 200 ppm of TDS);
– one stage RO system to produce drinking water meeting the WHO standard (below 500 ppm of TDS); and
– an advanced two stage RO system to obtain high recovery ratio (60%).

(b.1) Two stage RO system for the EC water quality standard

Existing RO membranes have been developed to achieve the water quality meeting the WHO standard (TDS less than 500 ppm and chloride ion less than 200 ppm). A two-stage RO system is required to meet the EC standard (TDS less than about 200 ppm, Cl less that 25 ppm).

The cellulose acetate (CA) membrane was selected as the membrane of the first stage due to the easy maintenance. However, as the membrane of the second stage, TFC (thin film composite) membrane was selected to reduce the energy consumption because TFC can be operated at lower pressure than CA and also has been well experienced in the industry water purification field.

The seasonal changes of the temperature and the salt density of seawater affect the performance of the membrane. In order to absorb its seasonal changes, the pressure of the pump is controlled. The energy recovery system with Pelton wheel type is selected to obtain higher energy recovery.

Filtered water through the dual media filter, which is mixed with the second stage RO brine as shown in Figure 3.58 is supplied to the guard filter, then fed to the first stage RO feed pump. The feed pump increases the pressure up to 7 MPa so as to produce desalinated water at a rate of 45% of feed water. The water through the first stage RO train, which is stored in the intermediate tank, is fed to the second stage RO train through the second RO feed pump where the pressure increases up to 1.6 MPa.

Easy operation and maintenance can be obtained by reduced number of equipment and a simple process flow. Fully automatic operation is applied by the computer control to minimize the number of operators.

The energy consumption of this process is about 5 kW(e)h/m^3. Additional electric power of 1 kWh/m^3 is needed for transportation of the product water to the urban area. Then the total required electric energy is about 6 kW(e)h/m^3. 46 MW(e) from 4S is available for seawater desalination. 168 000 m^3/day of the product water can be obtained with the 7 trains of the two stage RO systems (the capacity of each train is 24 000 m^3/day). An average capacity factor of 0.9 was assumed for the whole system. Overall process flow sheet is shown in Figure 3.58.

The first stage produces water of about 450 ppm of TDS, which is further purified to 125 ppm of TDS in the second stage.

(b.2) RO system with advanced technology [38]

The above-mentioned process is a design to meet the EC water quality standard. The following process is a typical design to meet the WHO water quality standard using an advanced RO technology, which is a brine conversion two stage seawater desalination system (BCS).

Based on the following assumptions, energy consumption of each system (existing RO technology and BCS) are compared, as shown in Table 3.24.

(1) Product water quantity:	50 000 m^3/day, BCS-10 000m^3/day × 5 trains, One stage-5000 m3/day × 10 trains.
(2) Membrane element:	SU-820 for Conventional and BCS of 1st stage, SU-820BCM for BCS of 2nd stage.
(3) Temperature:	20 °C.
(4) Feed pressure:	Module outlet osmosis pressure + 2 MPa (maximum osmosis pressure).
(5) Energy recovery system:	Reverse pump for the first stage, Turbo charger for BCS.

TABLE 3.24. COMPARISON OF ENERGY CONSUMPTION

		With energy recovery device					
		Recovery ratio (%)	Feed pressure (MPa)	RO energy consumption (kW(e)h/m^3)	Pretreatment energy consumption (kW(e)h/m^3)	Total energy consumption (kW(e)h/m^3)	Reduced with conventional (%)
One stage	Conventional	40	6.4	3.99	0.54	4.53	0
Two stages	BCS	60	5.9/9.0	3.5	0.31	3.81	–15.9

Assuming the same electric power consumption for desalination and transportation and the average capacity factor as above, BCS can produce 210 000 m^3/day of water of about 350 ppm of TDS meeting the WHO standard, while a conventional system of one RO stage can produce 180 000 m^3/day. The BCS has a 13% of energy reduction compared with the present technology. Each flow sheet is shown in Figures 3.59 and 3.60, respectively.

(b.3) Prospect of RO re-heating

If the seawater temperature is relatively low, e.g., below 30ºC, and there is low cost waste heat, RO with pre-heat may become advantageous. Heat from the tertiary circuit of “4S” could be used in this option.

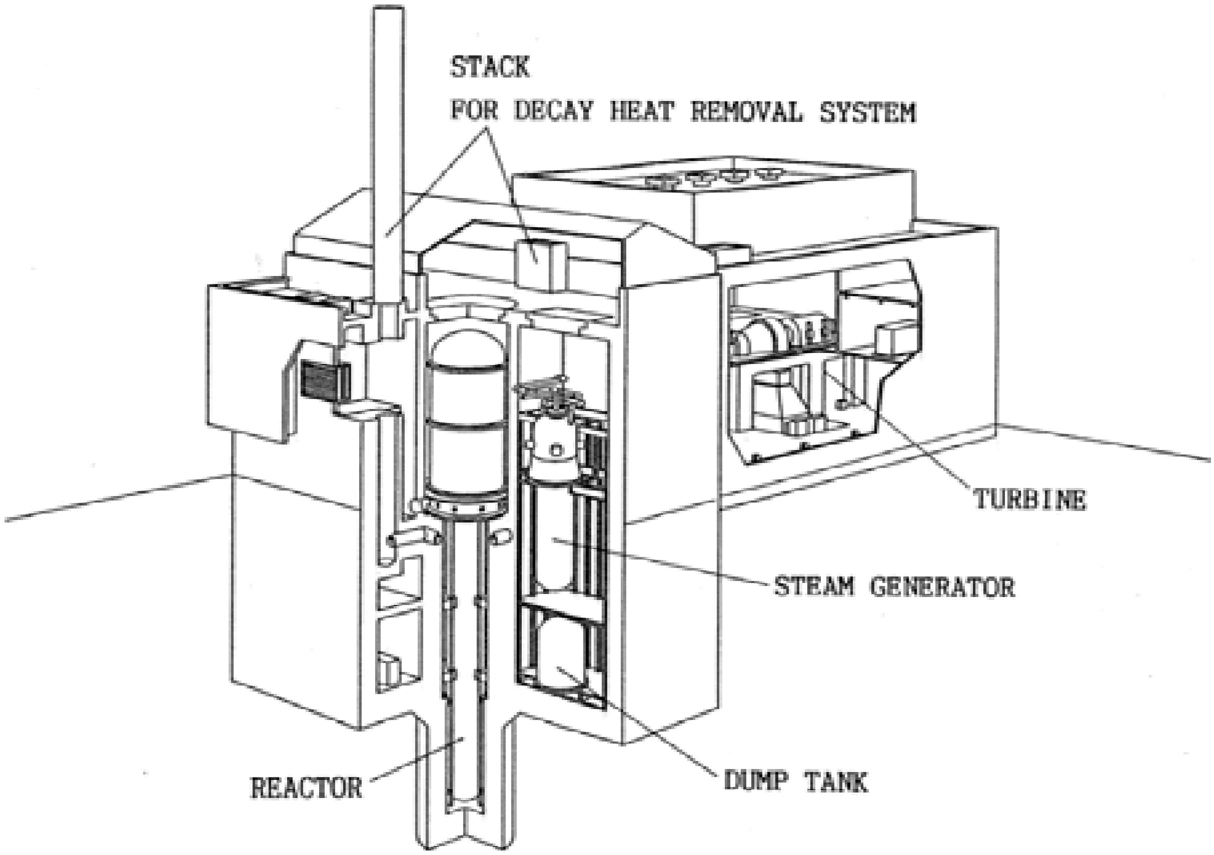


FIG. 3.56. Reactor concept.

(c) Safety

– The reactivity insertion is designed to be below 1 $ so that the core will not experience a prompt power transient. The inherent safety characteristic of the metal fuel also stabilizes the core. All other nuclear specific design basis events have been confirmed safe reactor shutdown. This includes the abnormal ascent of the reflector (reactivity insertion), and the PRACS failure (loss of heat sink).

– The reactor can be scrammed by automatic descent of the reflector. Even if the reflector does not descend, the metal fuel switches over to safe state of extremely small output with its inherent safety characteristics.

– "4S" is capable of load following operation in the range of 20–110% of rated output. Therefore, "4S" can be best used as a co-production plant for electricity and RO desalination. The electric power to the RO system should be in the range of above-mentioned power range. .

– In the option of "RO with preheat" the heat is taken from the tertiary circuit. As well experienced at a nuclear desalination complex at Aktau, Kazakhstan using the BN–350, there will be at least three physical barriers (IHX, SG, Brine Heater) between the radioactive primary coolant and the brine. This excludes a possibility of carry-over of radioactive substances into the product water.

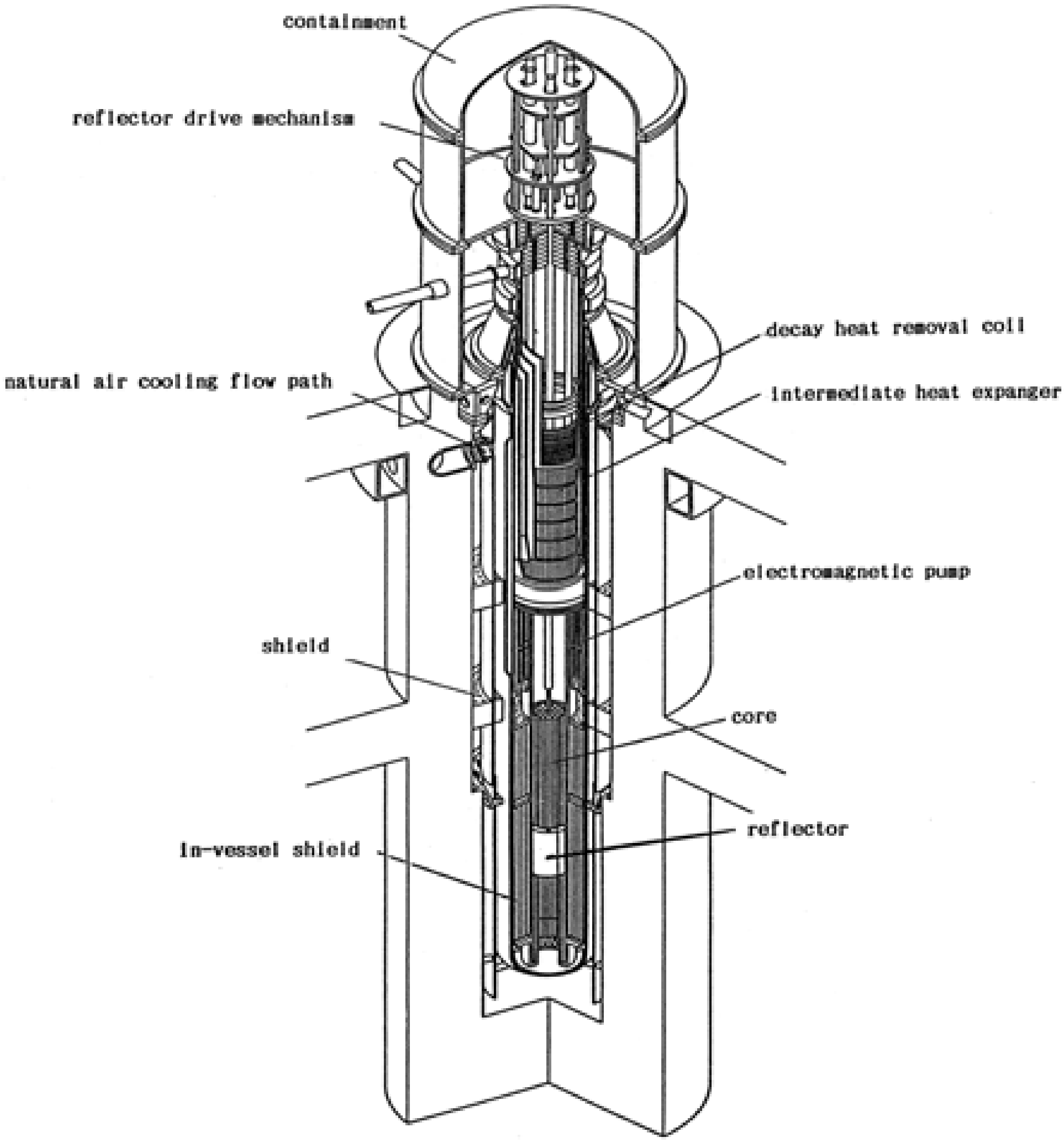


FIG. 3.57. Reactor assembly.

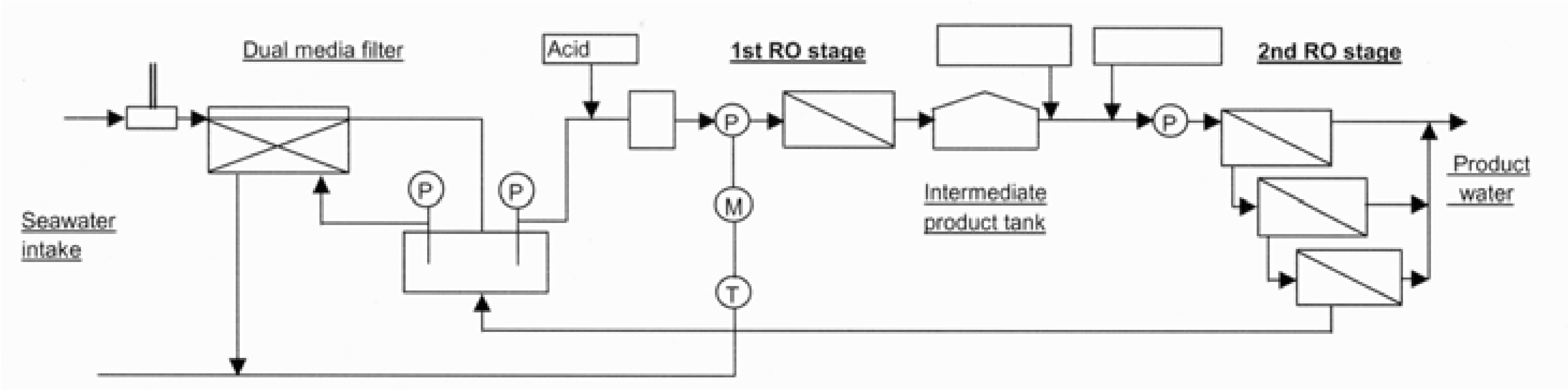


Fig 3.58. Process flow sheet of RO system with two stages.

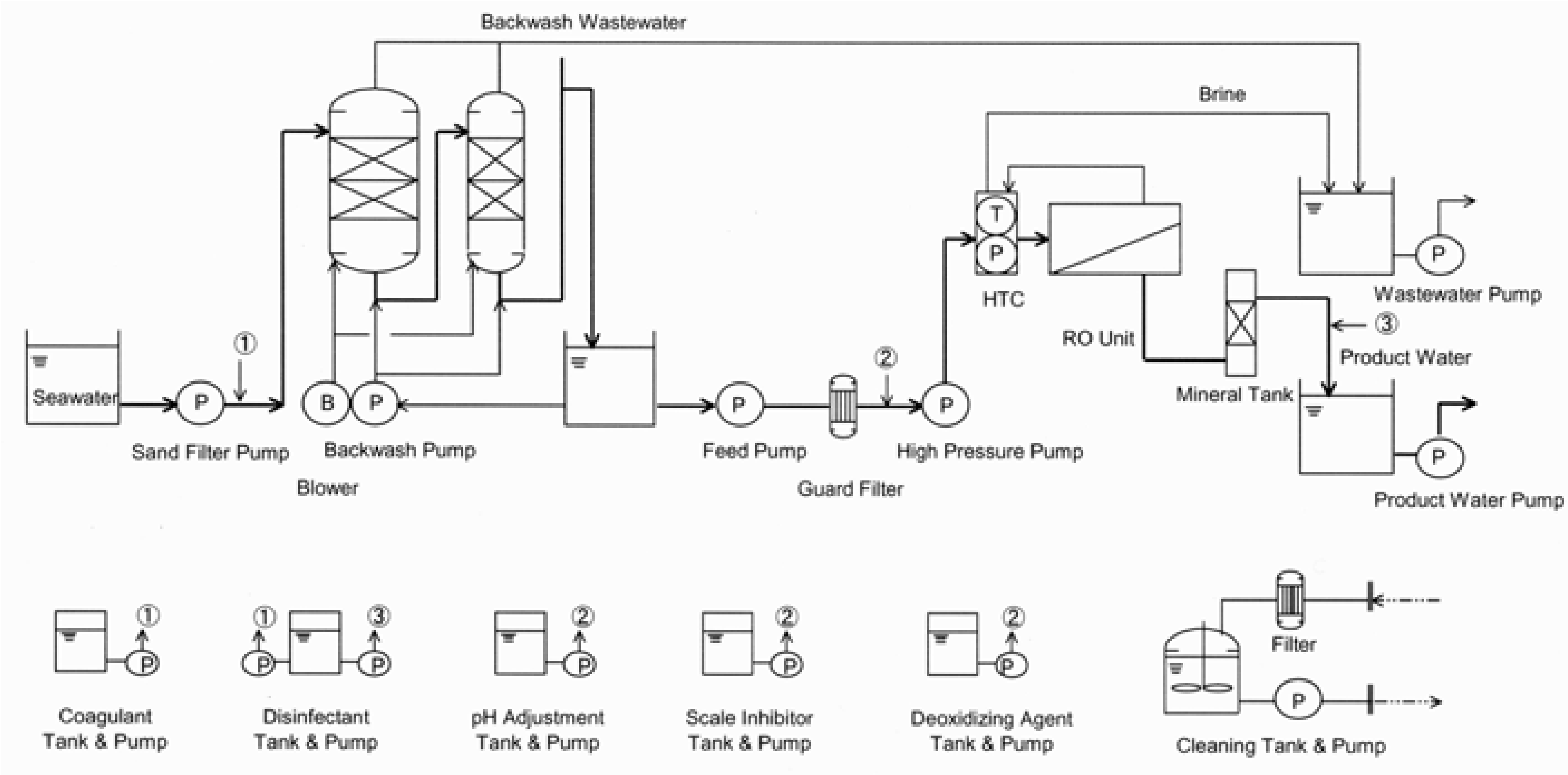


FIG. 3.59. Process flow sheet of existing RO system (recovery ratio of 40%).

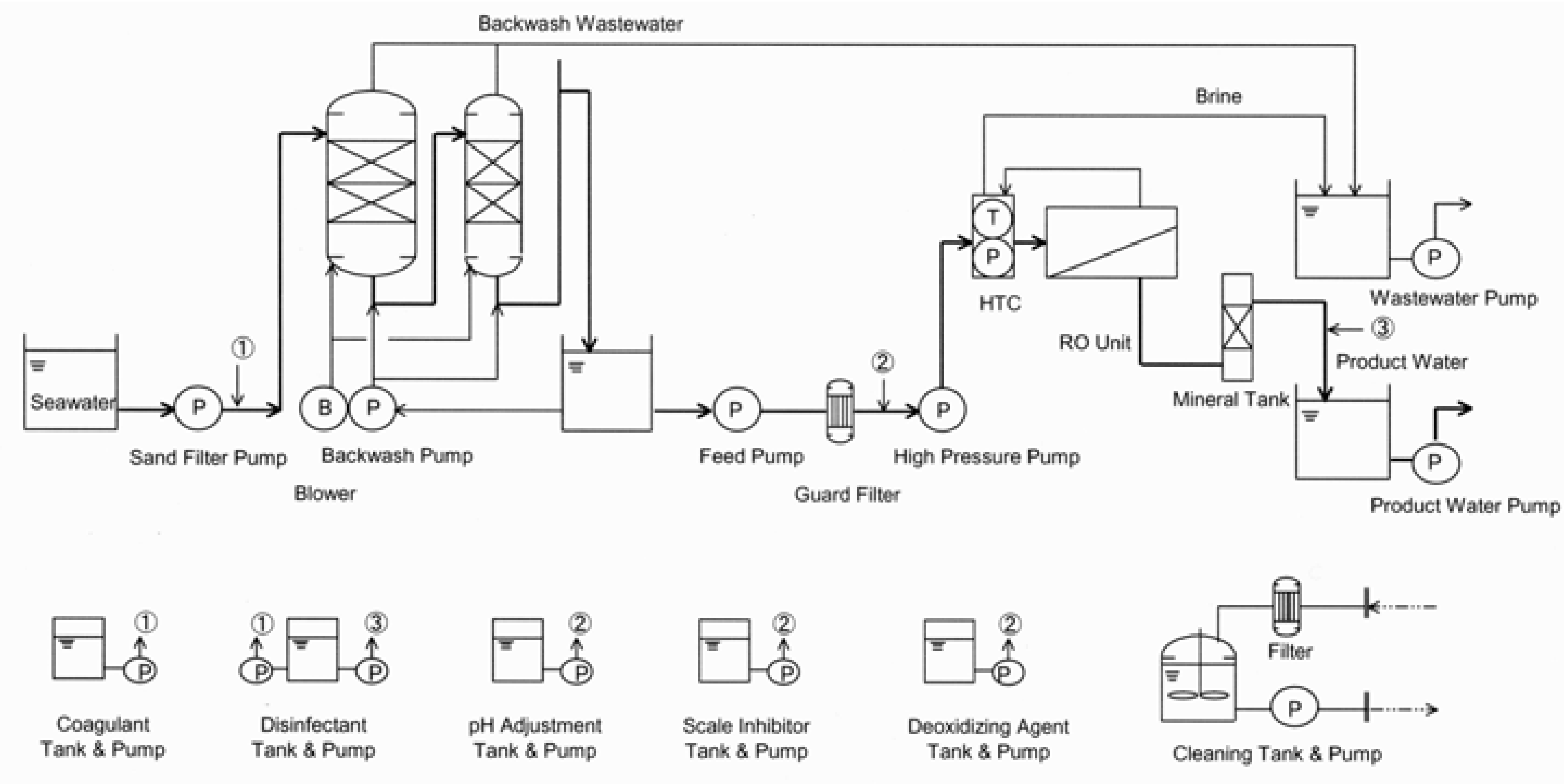


FIG. 3.60. Process flow sheet of BCS (recovery ratio of 60%).

(d) Economy

The water cost is generally obtained by the following equation:

$$\text{Water cost} = \frac{\text{capital cost + fuel cost + operation maintenance cost}}{\text{total produced water}} \quad (\$/m^3)$$

Here, interests, depreciation, fixed property tax are included in the capital cost. The capital cost of the reactor and desalination systems depends on the interest, term and way of the depreciation and other factors. The construction cost of the reactor and desalination system affects the absolute value of the capital cost because multiplying construction cost by the rate of capital cost represents the capital cost.

Nuclear power supply system cost

This small reactor concept has a big potential of design standardization, series production and shop fabrication of NSSS equipment, which are necessary to obtain the low construction cost compared with the large one. The "4S" capability of easy manoeuvrability and maintenance can greatly reduce the O&M cost of the energy source leading to lower levelized electricity cost. This reduced energy cost is another potential element to lower the water cost.

If "4S" is on a series production line of a 10 units/year annual rate over 10 years, its construction cost could be comparable to that of a large reactor (1000 MW(e), provided that the electric power of 4S is 50 MW(e) and the lifetime of the fuel is 10 years.

Desalination system cost

The capital cost of the desalination system was estimated for the first system with a two stage RO system, which produces 168 000 m^3/day of water.

The unit cost of 1600 $\$/m^3$/day for an RO membrane plant (1992$), was used for the estimation of the capital cost. As a result the capital cost of the desalination system is USM$270.

Water cost

The following assumptions were used to estimate the water cost at the plant boundary:

– All electrical output of "4S" is used for desalting water and its transportation;
– Refuelling interval is 10 years;
– Interest rate is 5% and;
– The lifetime of reactor and desalination systems is 30 years.

Under the above assumptions, the water cost is estimated to be about 1.2 $\$/m^3$, taking account of other costs such as fuel cost, operation and maintenance cost.

If the following technology developments are incorporated in updating cost estimation, lower water cost will be obtained compared with 1.2 $\$/m^3$:

– Advanced "4S" plant design with 30 years of the fuel lifetime.
– Low cost of RO membranes and/or advanced two stage RO systems.

CHAPTER 4. DEDICATED-HEAT PLANT CONCEPTS UNDER EVALUATION

4.1. Heating reactor NHR-200 with MED (China)

4.1.1. Background

The NHR-series including (NHR-200, NHR-10 and NHR-5) with thermal power of 200 MW, 10 MW and 5 MW respectively, was developed by the Institute of Nuclear Energy Technology (INET), Tsinghua University, China. It is specially designed to provide heat to seawater desalination, district heating, refrigeration, and similar applications.

The Nuclear Heating Reactor (NHR) has been designed with a number of advanced and innovative features to achieve high degree of safety and economic viability. Inherent safety features of the NHR include integrated arrangement of the primary circuit, natural circulation of primary coolant and residual heat removal, self-pressurized performance, hydraulic control rod drive and passive systems. The test reactor NHR-5 became operational in 1989, with a number of experiments — carried out to demonstrate the operating and safety features of the NHR. These include self-regulation and self-stability features, transient behaviour following a loss of main sink-ATWS and the heat transfer capability of RHRS with or without interruption of natural circulation in the primary system. Test operation of heat and electricity co-generation was also performed on NHR-5. NHR-200 design and NHR-10 design were based on the experience gained from the design, construction, start-up, operation and maintenance of NHR-5.

4.1.2. Design description

Technical description of the NHR

The structures of the NHR-5 and the NHR-200 reactors are depicted in Figure 4.1 (a) and 4.1 (b) respectively. The key design data of the NHR is presented in Table 4.1. 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 are arranged on the periphery in the upper part of the RPV. The system pressure is maintained by inert gas and steam. A containment vessel fits tightly around the RPV, so that the core will not become uncovered under any postulated leakage at 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.

Pursuant to the special safety requirement for nuclear reactors to be located near the consumers and directly connected with desalination plants, the NHR is designed with inherent and passive safety features different from those of general nuclear power plants, which strongly depend on engineering safety facilities.

Basic objectives and features of the NHR-200

The basic design objectives of HR-200 are as follows:

(a) *Safety*

– Reactor core is cooled with natural circulation.
– Reactor core is prevented from being uncovered under any accidents.
– Integral primary circuit arrangement, self-pressurization. Dual pressure vessel structure and all the penetrations located on the upper part of the reactor pressure vessel (RPV).
– Use of simple, dedicated, independent, passive safety systems that require no operator action for accident mitigation, and maintain core cooling without AC power.
– Predicted core damage frequency $<10^{-8}$/year and a significant release frequency $<10^{-9}$/year.
– Reliable reactivity control and shutdown system.
– Low operating parameters and large safety margin.

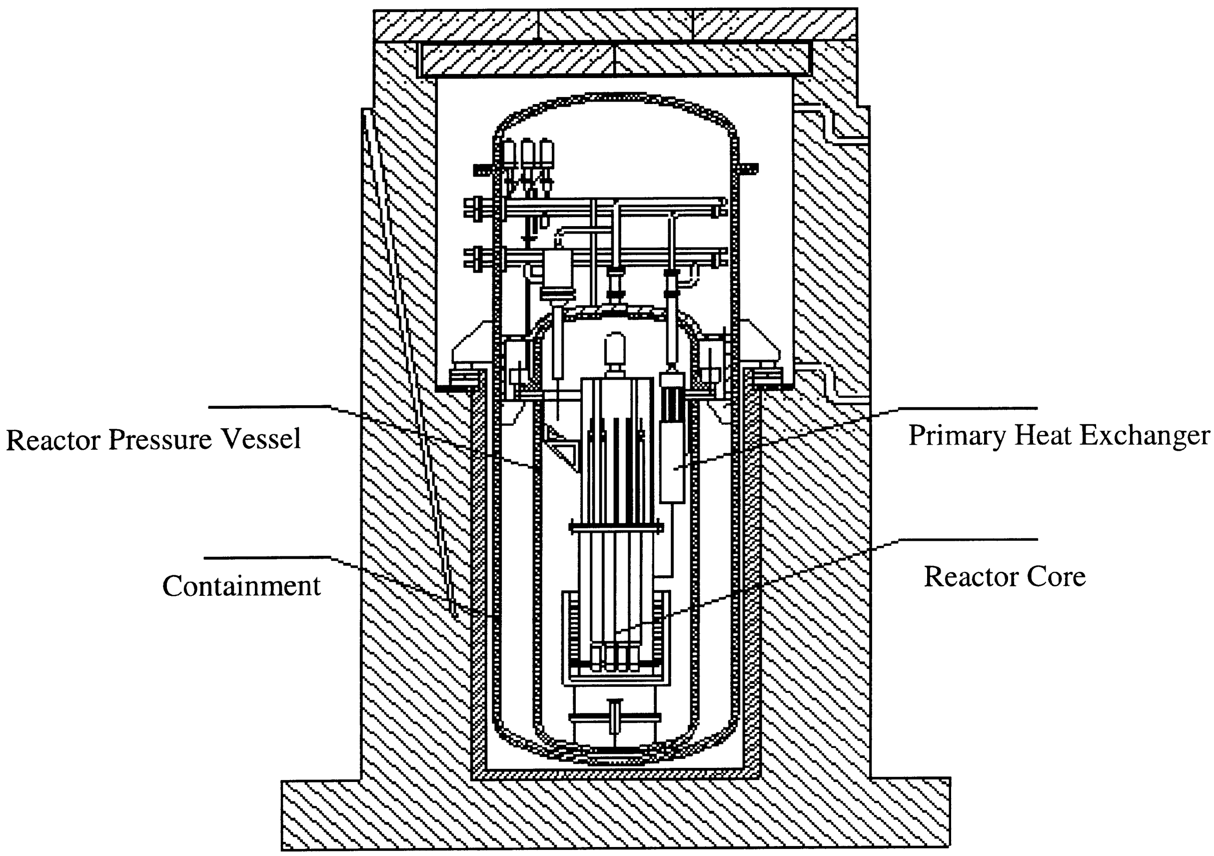
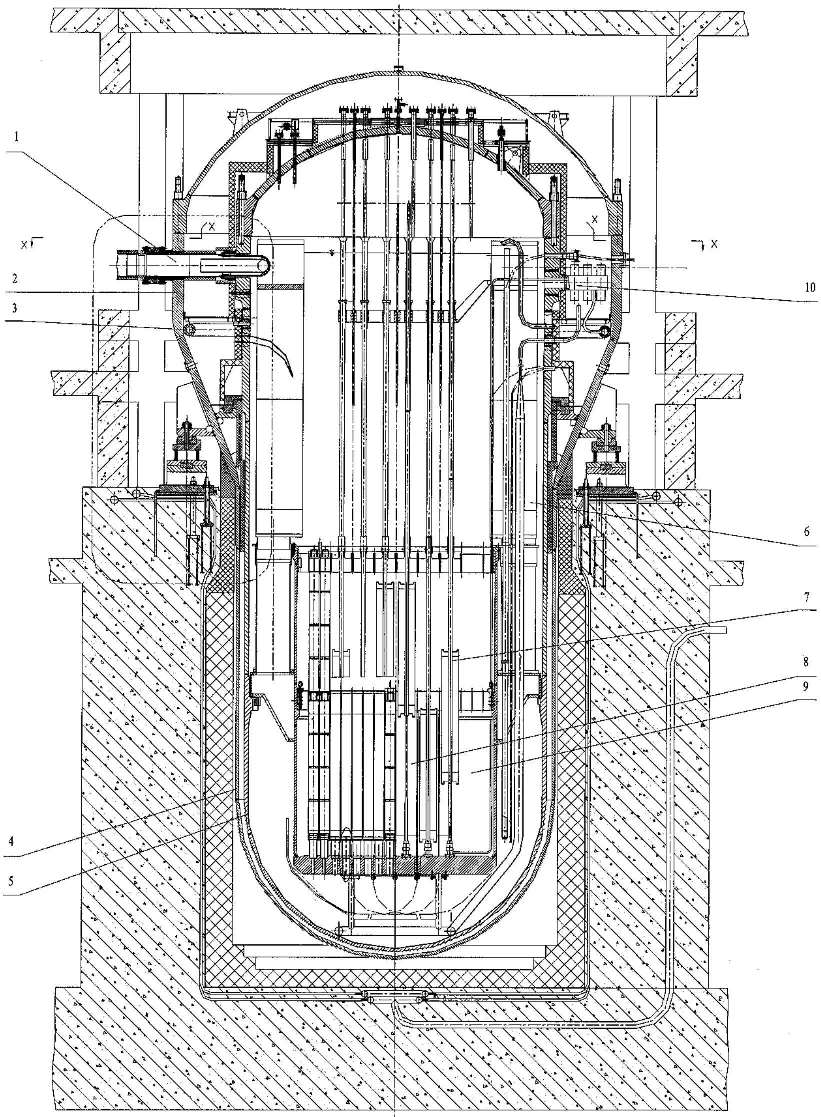


FIG. 4.1a. Reactor structure of the NHR-5.



1. Inlet to primary heatexchanger
2,4. Containment
3. Inlet to control rod driven system
5. Pressure vessel
6. Primary heat exchanger
7. Control rod
8. Reactor core
9. Used fuel assemblies
10. Valves assembly of control rod driven system

FIG. 4.1b. Reactor structure of NHR-200.

(b) Reliability

- Simplified design, operation and maintenance.
- Public radiation exposure at the DaQing plant site within the range of 80 km: ca. 5×10^{-4} man-Sv per year.
- Overall plant availability goal greater than 95 percent considering forced and planed outage.

TABLE 4.1. 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	ºC	146/186	174/210	153/210
Volumetric power density	kW/L	26	23	27
Number of fuel assemblies		16	32	120
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	2.17
Initial inventory of UO_2	t	0.51	1.4	16.7
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	ºC	102/142	180/135	135/170
Heat grid temperature (Steam temp.)	ºC	90/60	130***	130***

* NHR for district heating
** NHR for seawater desalination with MED process
*** Temperature in the steam circuit

(c) Major innovative features

– Integrated arrangement, self-pressurization.
– Dual pressure vessel structure.
– Low operating temperature, pressure, and low power density in reactor design, which provides increased operating margins and improved fuel economy.
– Cooling reactor core with simple, passive measure which uses natural driving force only.
– Adopted an innovative hydraulic system to drive control rod.
– State-of-the-art digital instrumentation and control systems and an advanced man-machine interface control room, console-type work stations, soft controls and integrated, prioritized alarms and procedures.
– Enhanced overall plant arrangement design and advanced construction concept adopted to minimize cost and construction schedule and to meet safety, operational and maintenance criteria.
– Multi-purpose applications.
– NHR can be used for seawater desalination, district heating and central refrigeration

System Design of NHR-200

The main systems of HR-200 are summarized as follows:

Nuclear Heat Supply System

- Primary circuit

The primary coolant absorbs the heat from the reactor core, then passes through the riser and enters the primary heat exchangers, where the heat carried is then transferred to the intermediate circuit. An integrated arrangement is adopted to decrease the possibility of LOCA. All main parts of the primary circuit are contained in the RPV.

- Reactor pressure vessel

The HR-200 is designed with an integral, vessel-type structure. The RPV is the pressure boundary of the reactor cooling. It is 13.62 m inside height, and 5.00 m inside diameter (see Figure 1). The design also features a large coolant inventory in the RPV (about 200 T), which lowers the total neutron flux to the RPV (ca.10^{16} n/cm^{2} for the 40-year reactor lifetime). All in-vessel penetrations (only with a small diameter) are located on the upper part of the RPV.

- Reactor core

The reactor core of the HR-200 consists of 120 assemblies (fuel ducts) and 32 control rods. A long riser is located above the core to enhance the capability of natural circulation. The reactor core stands on the lattice-support structure, which is fixed on the RPV.

The fuel bundle is arranged in a 12 × 12 matrix with an active length of 1.9m and is contained in the duct (or box). The cruciform type control rods are placed in the gaps between the square ducts. There are 3 kinds of enrichments in the initial loading: 1.8%, 2.4% and 3.0% of uranium dioxide. The discharge burn up is about 30 000 MWD/tu.

The equivalent core diameter is 2.6 m. spent fuel assemblies are stored in the rack around the active core. Burnable poison (Gd_2O_3) is used to partly compensate the fuel burn up reactivity, and soluble boron is utilized for reactor shutdown only. This result is in a negative temperature coefficient of reactivity over the complete core life.

A low core power density (ca. 27 kw/l) enhances thermal reliability during normal and accidental operating conditions. This simplifies greatly the refuelling equipment and eliminates the necessary space in the reactor building.

- Control rod and control rod driving mechanism

A new type of hydraulic driving facility is used for driving the control rod in the HR-200. In the drive system the reactor coolant (water) is the actual medium. The water is pumped into the step-cylinders of which the movable parts contain the neutron absorber. A pulsed flow is generated by controlling magnetic valves in the control unit, and it moves the movable part of the step-cylinder step by step. The drive system is very simple both in structure and its design on the “fail-safe” principle, i.e. all control rods will drop into the reactor core by gravity under loss of electric power, depressurization, postulated breaks in its piping systems and pump shut down events.

- Primary Heat Exchanger (PHE)

Six sets of PHE are located on the periphery of the RPV upper part. The triangular-pitch, U-tube-shaped and vertically placed bundles are adopted for easy onsite repair. The coolant enters the upper plenum of the exchangers, and then is divided into two streams to flow downward in the tubes. The flow distribution baffles are installed to make an optimum heat transfer efficiency. The total heat transfer area is approximately 2982 m^2. The operating temperature of PHE is 210^{o}C and the operating pressure is 3.0 MPa.

- Fuel Handling and Storage

The initial core is divided into 4 fuel regions and contains 120 fuel assemblies. Thirty assemblies are refuelled at one time. The spent fuel is then moved into spent fuel racks around the active core and stored there. This design greatly simplifies the refuelling equipment and eliminates the necessary space in the reactor building.

Second pressure vessel

A second pressure vessel made of steel is fitted tightly around the RPV as a guard vessel so that the core will not be uncovered under any postulated coolant leakage within the RPV.

Intermediate circuit

To keep the desalination system free from radioactive contamination, an intermediate circuit is needed in the HR-200 and its operating pressure is kept higher than that in the primary circuit.

Balance of plant systems
- Heat sink

A steam supply system transports heat from the nuclear plant to the seawater desalination system, and the coolant of the final condenser of the seawater desalination process disperses heat to the sea.

Instrumentation, control and electrical systems

The design of the instrumentation, control and electrical systems corresponds with the safety concept for the safe reactor operation. Advance in electronic and information processing technology has been incorporated in the design. The plant is automated to a high degree, and all safety precautions are taken into account. The plant control scheme is based on the “reactor follows plant loads” system.

The computer system is used intensively in the plant control and data acquisition and takes the place of hardware analogue control. This results in a significant reduction in the amount of cabling. In case of unsafe conditions the reactor protection system can automatically scram the reactor and actuate the relevant safety systems.

Safety considerations and emergency protection

Safety of the HR-200 is provided through two key mechanisms: development of the plant self-protection features, and creation of a multi-barrier system of functional and physical protection (defence-in-depth). A number of advanced features have been incorporated into the NHR design to achieve its safety goal, which can be summarized as follows:

– large negative temperature reactivity coefficient over the complete core life;
– integrated arrangement, self-pressurization, minimization of in-vessel penetrations and their location at the upper part of the RPV;
– natural circulation of coolant in the primary circuit under all conditions;
– low power density in the core and low operating pressure in the primary system;
– large coolant inventory and double pressure vessel design keeping the core covered by coolant under all conditions and excluding large break LOCA;
– fail-safe principle on hydro-driven control rod drive mechanism design;

– elimination of possible control rod ejection accidents by in-vessel hydro-driven control rod mechanisms;
– long grace period for corrective actions;
– spent fuel in-RPV storage; and
– passive safety systems.

There is no emergency core cooling system in the NHR design. The residual heat removal system 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.

- *Residual heat removal system (RHRS)*

RHRS is connected with the intermediate circuit and consists of two independent trains, each of which is able to disperse core decay heat into the atmosphere properly through natural circulation.

- *Boric acid injection system*

A boric acid injection system, as a secondary reactor shutdown system, is designed to function by gravity.

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 driven 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 plant power supply, depressurization, pipe break or pump shutdown events.

Coupling of the NHR with selected desalination processes

A simplified schematic diagram of the NHR coupled with a desalination process is shown in Figures 4.2 and 4.3. The nuclear heat supply system contains triple loops. The primary coolant absorbs heat from the reactor core, flows through the riser and enters the PHRs where the heat is transferred to the intermediate circuits. Finally, heat is delivered to the desalination plant via the steam generator. An intermediate circuit is needed in the NHR to insure that the desalination plant is free from radioactive contamination.

Design precautions for the coupling interfaces

In the integrated nuclear desalination plant, energy is supplied to the desalination plant 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 heat transport media to the product water in the desalination system. To this end, the following design precautions have been taken for the coupling interfaces between the NHR and the 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 Figure 4.2, 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. When 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 heating 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 favours keeping the water in the intermediate circuit free of contamination from the heating grid.

(3) The pressure and radioactivity of the intermediate circuit are monitored continuously. When 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 product water against radioactive contamination.

Suitability of the NHR-200 design for desalination

The performance of NHR-200 is suitable for coupling with a seawater desalination plant. Among the various existing desalination processes worldwide, MED has been selected as the most suitable desalination technology for coupling with NHR-200 based upon the following criteria.

(1) *Reliable connection between the reactor system and desalination system (Reliability)*

The desalination system and the nuclear reactor system are mechanically and thermally connected via the main steam pipe and the main feed water pipe in the steam supply circuit (in-between short distance, at a common site). Thermo-energy required by the desalination process is efficiently transferred by steam from the steam generator to the first effect of the desalination system;

(2) *Multi-barriers isolation (Safety)*

Between the nuclear reactor and the desalination system, there are three layers of steel wall boundaries: heat transfer tubes of the primary heat exchanger; a steam generator and the first effect of evaporator of the desalination system; and two circuits (Intermediate circuit and Steam supply circuit) working as barriers to effectively prevent the desalination system from radioactive contamination.

(3) *Appropriate pressure barrier (Inherent safety)*

Pressures in the primary circuit and the secondary circuit are at 25 and 30 bar, respectively. Even in the case of a rare failure of heat transfer tubes or its welding on the tube plate of the primary heat exchanger, the coolant in the primary circuit would not leak into the intermediate circuit;

(4) Perfect match of parameters (Efficiency)

Coupling of a high temperature MED process with the nuclear heating reactor leads to a high energy efficiency of the integrated desalination plant.

(5) *Suitability of operation performance (Operability)*

Both the heating reactor and the desalination system have excellent self-regulation capabilities. When the load fluctuates within the range of 70% to 100%, even with minor active perturbations on either side, operation of the integrated plant would be very smooth and with perfect self-regulation performance. Adjustment within the load range of 40% to 100% is also easy and simple to perform.

(6) *Simplicity and maintainability of the integrated system (Simplicity).*

(7) *The integrated desalination plant with the selected coupling scheme would produce drinking water with optimum water cost and high quality.*

Both single-water production and cogeneration scenarios were studied. The scale of the NHR-200 can satisfy a potable water demand of about 100 000 m^3/day. Under suitable conditions, 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 freshwater requirements. The combined NHR-200 desalination plant can not only ensure the continuity of water production but also improve the economy by sharing of common facilities and service systems such as infrastructure, maintenance facilities, reduction of staff, and so on [42].

4.1.3. Economic perspectives

The capital cost of NHR-200 was estimated at about US$100 million (back in 1991). The heat cost is competitive with a coal (or oil) fuelled plant for a typical site in China. The water cost of the NHR-200 desalination plant was estimated at about (0.9–1.0 US$/$m^3$), and is expected to be strongly affected by the specific capital cost of the coupled MED process.

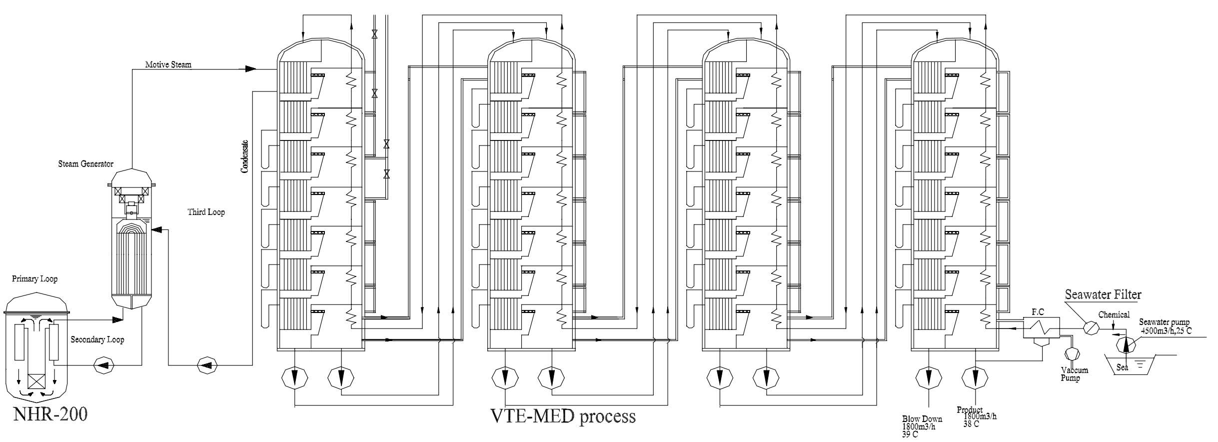


FIG. 4.2. Simplified schematic diagram of NHR-200 coupled with VTE-MED process.

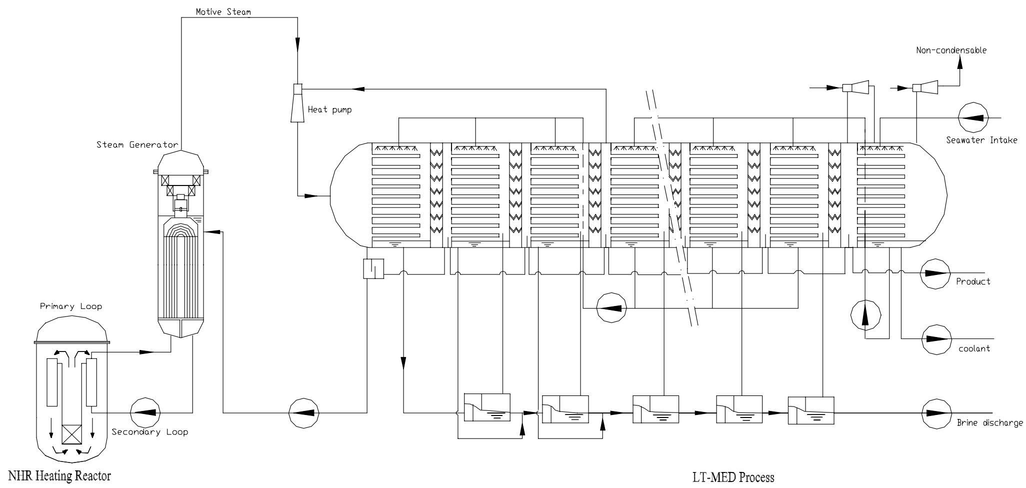


Fig 4.3. Simplified schematic diagram of HHR-200 coupled with LT-MED process.

4.2. Heating reactor NHR-10 with MED (Morocco)

4.2.1. Background

During the last two years Morocco and China has performed joint technical co-operation project under the umbrella of the IAEA. A pre-project is being carried out to study nuclear desalination demonstration plant with a 10 MW (th) Nuclear Heating Reactor (NHR-10) from China to be built in Tan-Tan, Morocco (Figure 4.4). The plant will have a capacity of 8000 m^3/day of potable water and will provide the basis for future introduction of large-scale desalination units using 200 MW (th) heating reactors having a capacity of 160 000 m^3/day. The basic tasks of the pre-project study are to lay down the technical basis for the reactor and the desalination plant, expound and verify its technology, safety, reliability, economy and availability. Its major objectives are as follows:

- Specify the concept design of the reactor and the desalination plant in the demonstration plant, and expound and verify its technical feasibility.
- Expound and verify the safety of the demonstration plant pursuant to international standards.
- Estimate the investment cost of the project, and assess the cost of the fresh water produced in order to expound and verify the economic feasibility of the demonstration plant.
- Conduct a comprehensive study regarding necessity, safety, technical feasibility and economic viability of the demonstration plant, and provide the decision-makers with feasibility for the establishment of such a demonstration plant.

4.2.2. Design description of the nuclear seawater desalination demonstration plant

As the heat source for desalination, the NHR-10 reactor provides the desalination plant with 105–130ºC-saturated steam via the primary circuit; the intermediate circuit and the steam supply circuit. The desalination plant with a vertical tube foamy flow evaporation (VTFE) system consists of two units with four towers each. Each tower includes seven effects. The towers are connected by piping. In each desalination unit there are 28 effects. The design capacity of each unit is 4000 m^3/d, thus the total design capacity of the water production plant is 2 × 4000 m^3/d. The first two towers containing effects 1–14 have a diameter of 3.0 m and the second two towers containing effects 15–28 have a diameter of 3.4 m. All towers have a height of 34 m.

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

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

The vertical tube foamy flow evaporation (VTFE) system

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

In the VTFE first effect, the steam is cooled and condensed by seawater, and then flows back to the steam generator as feed water. The seawater, which is heated by steam in the first effect, becomes secondary steam, which goes into the next effect. This evaporation-condensation process repeats further until the last effect, where the produced steam will be cooled and condensed by the feed seawater to the desalination plant. The feed seawater, taking a counter flow pattern, flows through every effect, arriving pre-heated into the first effect. Through evaporation at every effect, the salt content of the seawater becomes increasingly higher. Finally, seawater with a high salt content will be discharged into the sea. The fresh water coming from one effect to the next one will flash thus increasing the heat recovery. To maintain the heat transfer efficiency of the evaporation-condensation process, vent facilities are designed for removal of non-condensable gases.

Nuclear steam supply system

The NHR-10 is a vessel type light water reactor, which features an integral arrangement, natural circulation, self-pressurisation and passive safety, with the NHR-5 as its prototype.

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

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

The key design parameters of the nuclear seawater desalination plant are listed in Table 4.2.

TABLE 4.2. GENERAL DESIGN PARAMETERS OF NDDP

Reactor thermal power	MWth	10
Pressure	Mpa	2.5
Reactor outlet/inlet temperature	ºC	210/180
Inter-mediate circuit		
pressure	Mpa	3.0
water temperature at steam generator outlet/inlet	ºC	135/180
Steam temperature at outlet	ºC	130
Steam flow rate	kg/s 4.37	
Seawater desalination plant		
design capacity	m3/d	8,000
process		VTFE
number of units		2
number of effects of every unit		28
capacity per unit	m3/d	4080
GOR		21.6
seawater inlet temperature	ºC	25
flow rate	t/h	2 × 340
water quality (TDS)	ppm20	
Reactor operation cycles: first cycle	EFPa *	5.89
second cycle	EFPa *	1.92
Maintenance period of the desalination plant	a	3
Planned reactor outage	%	3
Unplanned reactor outage	%	4
Planned desalination plant outage: arranged at the time with reactor planned outage	-	-
Unplanned desalination plant outage	%	4
Total availability	%	89
Reactor refueling cycle	EFPa	7.81
Reactor-Desalination Plant cooperative maintenance plan	a	3+3+2

* Equivalent Full Power Years

1. ***Design features***

Reactor safety

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

The NHR-10 is designed to have a relatively high negative temperature coefficient of reactivity, which inherently suppresses diverging power and temperature transients. Because the NHR-10 has low power densities, it has a relative large margin against the Departure of Nucleate Boiling (DNB). Therefore the integrity of fuel claddings will be ensured.

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

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

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

There are two reactor shutdown systems in the NHR-10: a control rod system and a boric acid solution injection system. The hydraulic control rod drive system meets the fail-safe principle and eliminates the possibility of control rod ejection, which has been demonstrated in a special experiment. Therefore, there is no unexpected large reactivity insertion under any conditions.

The boric acid solution injection system is the second shutdown system, which has two driving modes pump and high-pressure nitrogen.

There is no emergency core cooling system in the NHR-10. The residual heat removal system (RHRS) is the most important safety system and is designed as a passive system. The decay heat will be dispersed into the ultimate heat sink by natural circulation.

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

- *Safety in product water production*

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

Reliable multi-barriers are designed between the NHR-10 and the desalination system. In the reactor system there is an intermediate circuit whose pressure is higher than the primary circuit. This circuit separates the reactor coolant circuit and the steam supply circuit via the main heat exchangers and the steam generators. The steam supply circuit itself is separated from the freshwater system by the evaporator in the VTFE first effect. It is unlikely for the radioactive reactor coolant to enter the fresh water systems unless all the three barriers fail simultaneously, with concurrent failures of the pressure barrier of the intermediate circuit and isolation.

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

4.2.3. Economic perspectives

The total base investment cost of the project based on the price level in January 1998 is 38.00 MUS$, where the 10 MW (th) nuclear heating reactor base investment cost is 22.50 MUS$, and the 2 × 4000 m^3/d VTFE desalination system base investment is 15.50 MUS$.

The national participation taken into account is around 40%. As the scale of the NHR-10 is very small, the specific investment cost is much higher than that of a commercially sized heating reactor.

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

For the purpose of economic study of a large commercial nuclear heating reactor desalination plant, the construction cost and the potable water cost of a 200 MW (th) heating reactor desalination plant were estimated based on the results of the 10 MW (th) demonstration plant. Table 4.3 shows an economic data summary for the reference cases of the 10 MW (th) and 200 MW(th) nuclear desalination plants. The calculation results indicate that such a nuclear desalination plant should be economically competitive with a fossil one.

TABLE 4.3. ECONOMIC DATA SUMMARY OF REFERENCE CASE

10 MW(th) nuclear desalination plant

Base construction cost of NHR- 10	MUS$	22.50
Base construction cost of 2 × 4000 m^3/d MED water plant	MUS$	15.50
Interest rate during construction period	%	6.5
Water plant capacity	m^3/d	8000
Construction lead time	month	36
Economic life time	year	30
Load factor	%	89.6
Discount rate	%	10
Levelized water production cost	US$/$m^3$	2.79

200 MW(th) nuclear desalination plant

Base construction cost of NHR-200	MUS$	97.28
Base construction cost of 4 × 43200 m^3/d MED water plant	MUS$	163.8
Interest during construction period	%	6.5
Water plant capacity	m^3/d	160 000
Construction lead time	month	42
Economic life time	year	30
Load factor	%	91.5
Discount rate	%	10
Levelized water production cost	US$/$m^3$	0.998

4.3. Dedicated heating reactor RUTA-55 with MED (Russian Federation)

4.3.1. Background

The RUTA reactor facility, originally designed for domestic heat supply, can be used as a heat source for nuclear desalination. It is a single-purpose heating plant generating low-temperature heat power in the form of hot water circulating in the tertiary circuit.

The range of thermal power design, varying from 10 to 55 MW makes it possible to match the capacity of the desalinating facility depending on the demands in a specific region. A thermal power of 55 MW is proposed here.

FIG. 4.4. Location of the proposed Tan-Tan site for NHR-10

4.3.2. Design description

Nuclear reactor

RUTA is a water-cooled water moderated pool-type reactor with atmospheric air pressure in the above-water volume of pool (Figure 4.5). The primary components are integrated within the reactor tank and operated under almost hydrostatic pressure. The primary heat exchangers are of the plate type and made of aluminum. The reactor may operate under natural and forced circulation of the primary coolant.

The secondary (intermediate) circuit of the reactor facility includes two independent loops. The tertiary circuit provides the interface to the heat application. The main technical characteristics of RUTA-55 in natural circulation mode are summarized in Table 4.4.

Desalination System

It is possible to implement only distillation type systems for coupling with RUTA reactors. Seawater desalination takes place in multi-effect distillation (MED) plants with horizontal-tube film apparatus designed and manufactured in the Russian Federation. The design characteristics of this type MED-plant are oriented to using steam from the low-pressure extractions of steam turbine plants. Pressure ratings from 0.15 to 1.0 MPa furnish boiling temperature in the region of 85 to 105°C in the first effect of the plant. A relatively low thermal potential of reactor-generated heat brings about the need for the special purpose design solutions so as to achieve maximum capability of the facility in terms of distillate production. As a result, corrected capacity MED-unit is nearly 70% of the typical unit nominal capacity. Coupling was studied between RUTA–55 and four MED plants of 220 m^3/hour with plant developed on the basis of a standard produced in the Russian Federation MED unit (capacity 300 m^3/hour) and adapted to the parameters of the tertiary coolant of RUTA. Basic technical and economic characteristics of the developed MED units are given in Table 4.5.

TABLE 4.4. RUTA-55 MAIN TECHNICAL CHARACTERISTICS

Rated Thermal Power, MWt	55
Primary Coolant Parameters:	
Coolant	Water
Flow Rate, kg/s (t/h)	522 (1879)
Temperature (Core inlet/outlet), °C	75/100
Pressure, MPa:	
- Core inlet	0.250
- Core outlet	0.229
- Primary HX inlet	0.141
Secondary Coolant Parameters:	
Coolant	Water
Flow Rate, kg/s (t/h)	535 (1926)
Primary HX Temperature, °C	90/66
Pressure, Mpa	0.39
Tertiary Coolant Parameters:	
Coolant	Water
Flow Rate, kg/s (t/h)	525 (1890)
Main Water HX Temperature (inlet/outlet), °C	60/85
Pressure, Mpa	0.59–0.98
Reactor:	
Reactor Pool Water Volume, m^3	700
Core Dimensions (height/equivalent diameter), m	1.2/2.03
Core Power Density, MW/m^3	14.1

TABLE 4.5. BASIC TECHNICAL AND ECONOMIC CHARACTERISTICS OF DEVELOPED MED UNIT

Rated Distillate Capacity, m^3/hour	220
Installed daily capacity, m^3/day	5300
Maximum brine temperature	80
Average temperature drop between effects	2.6
Number of MED-effects	16
Electric power consumed, kW (no tertiary circuit pumps)	350
Seawater flow rate, tons/hour	
for desalination	420
for cooling (in addition) at t = 26°C	700
Equipment metal consumption, tons	600
Equipment estimated price, US$ million	8–9

Coupling scheme

To couple the RUTA reactor and a MED desalination plant, it is necessary to choose an economic and technical optimal scheme of using the heat power from the reactor. Possible variants of coupling between the reactor and a desalination plant have been analysed Fig 4.6. The coupling scheme of the reactor and desalting plants includes steam-generating equipment. A circuit for steam generation is designed as a loop within the tertiary circuit of the reactor facility. The number of such loops being connected in parallel to the common header will depend on the number of MED-units. Each unit is equipped with upstream multistage self-evaporator where the tertiary coolant is partially evaporated to heat the lead stages of MED-unit. The self-evaporator is placed immediately at the evaporating effects.

There are no any stop or control valves in the piping between the self-evaporating stages and MED effects, i.e. the self-evaporator is the constituting part of the MED-unit. Temperature losses during steam transport are minimal in the secondary heat exchanger. It is expected that steam with a maximum temperature of 82^{o}C would be obtained from the secondary heat exchanger. A multi-effect self-evaporator therewith could ensure the boiling temperature of the order of 80^{o}C in the first effect of the desalination plant. The tertiary coolant should be replenished by distillate. Of all considered variants, this variant of coupling between the reactor and the MED plant seems to be the most preferable. The total capacity of Nuclear Desalination Complex (NDC) with one reactor RUTA-55 is about 20 000 m^3/day.

The main characteristics of NDC RUTA are presented in Table 4.6.

TABLE 4.6. MAIN CHARACTERISTICS OF NUCLEAR DESALINATION COMPLEX RUTA

Parameter	Value
Nuclear reactor	RUTA-55
Total heat to water plant, MW(th)	53.5
Number of MED units with capacity 220 m^3/hour each	4
Installed water plant production capacity, m^3/day	21 120
Average daily water production, m^3/day	19 000

The coupling scheme also offers the flexibility of isolating separate MED plants for repair or maintenance, and in the case of decreased desalinated water consumption (i.e. with provision for the power source to work at a reduce power level). In the RUTA-55 design the heat is transferred from the reactor through two symmetric and autonomous loops of the secondary circuit. To avoid a distortion of power distribution in the reactor core in the course of controlled power drop, when asymmetric heat consumption by the desalination units takes place, there are an inlet and an outlet collector in the hydraulic scheme of the Nuclear Desalination Complex. The steam generation loops of all MED plants and all heat exchangers of the secondary/tertiary circuits are connected to those headers. Such a configuration of the Nuclear Desalination Complex with common tertiary circuit enables to uniformly distribute the heat load between the loops of the secondary circuit with the retention of their independence.

Safety

Enhanced safety and reliability of the NHP with RUTA reactors is ensured by the following:

- development and utilization of the inherent safety characteristics;
- implementation of passive safety features;
- realization of the defense in depth principle with multi-barrier protection against radioactivity release into the environment;
- safety oriented selection of the reactor design features, performance and layout.

The inherent and passive safety features of the reactor are based on:

- negative temperature (of coolant and fuel), power and water density reactivity coefficients over the whole range of parameter changes;
- large specific (per unit of power) heat capacity;
- natural primary coolant circulation under all modes of operation.

Safety functions fulfilled by passive systems:

- Reactor scram is ensured by gravity driven insertion of absorber rods into the core;
- Decay heat removal is ensured by passive systems under natural primary and secondary coolant convection. Atmospheric air and/or ambient ground is used as an ultimate heat sink;
- In the event of a rupture of heat exchange surface there is no leakage from the reactor tank, because the pressure in the primary circuit is lower than in the secondary one;
- Barriers to the release of radioactivity;
- Fuel matrix (UO_2) whose temperature under the normal conditions does not exceed 640°C;
- Zirconium fuel rod cladding;
- Leak tight reactor vessel with a closed system of ventilation of the gaseous space above the water level in the reactor tank;
- Containment where all primary circuit equipment is located.

Exclusion of radioactivity release to the network water is achieved by the use of an intermediate circuit with the following circuit pressure distribution:

$$(P_1 = 0.1 \text{ MPa}) < (P_2 = 0.4 \text{ MPa})$$

where P_1 is the primary pressure, P_2 is the secondary (intermediate) pressure and P_3 is the pressure in the heating network.

The pressure difference ($P_1 < P_2$) in the reactor circuits prevents the penetration of radioactive substances from the primary circuit to the secondary one in the case of loss of sealing of heat transfer surfaces, or in-reactor piping depressurization of the secondary circuit. That is, the secondary circuit remains clean even after loss of sealing of the primary heat exchangers. Duplication of the preventive effect of the pressure ratio with an elevated pressure in the tertiary circuit is not of necessity. Keeping $P_2<P_3$ in the 2nd and 3rd circuit heat exchangers of the Desalination Complex would require extra power with a sizeable lowering of the distillation process parameters.

The following features contribute to enhancing the RUTA safety:

- integral layout of the reactor without excessive pressure in the primary circuit;
- utilization of burnable absorbers to reduce excess reactivity. Presence of additional reactivity compensation system;
- low power density and parameters (temperature, pressure) of the coolant in the circuits;
- two-loop arrangement of heat removal from the reactor.

4.3.3. Economic perspectives

The following DEEP calculation results have been obtained:

Variant 1. On average, the cost of fresh water production with a single RUTA-55 reactor nuclear desalination complex is 1.42$/m^3. When increasing the nuclear desalination complex capacity two to four times, that is using 2, 3 or 4 RUTA-55 reactors, the distillate cost decreases to 1.26–1.2–1.17 $/m^3 correspondingly.

Analysis of competitiveness of Nuclear Desalination Complex in comparison with fissile fuel power source Desalination Complex has been performed for the case of most expensive fuel oil (20$/bbl ≈ 120$/ton). Cost of distillate in the considered power region is 1.25 to 1.18$/m^3.

Hence, at fuel oil price 20 $/bbl and higher the economic characteristics of a Nuclear Desalination Complex with 3 or 4 RUTA-55 reactor plants could compete with those of a distillation plants using heat produced by fuel oil burning. At the present-day prices, desalination plants with power sources using cheaper fossil fuels such as gas or coal remain advantageous than nuclear option.

Variant 2. For the Russian and CIS economic environment, cost of fresh water comes to nearly 1.0 $/m^3 for a single RUTA-55 reactor Nuclear Desalination Complex. For a Nuclear Desalination Complex with 2 through 4 reactors, the distillate cost ranges from 0.88 to 0.79 $/m^3.

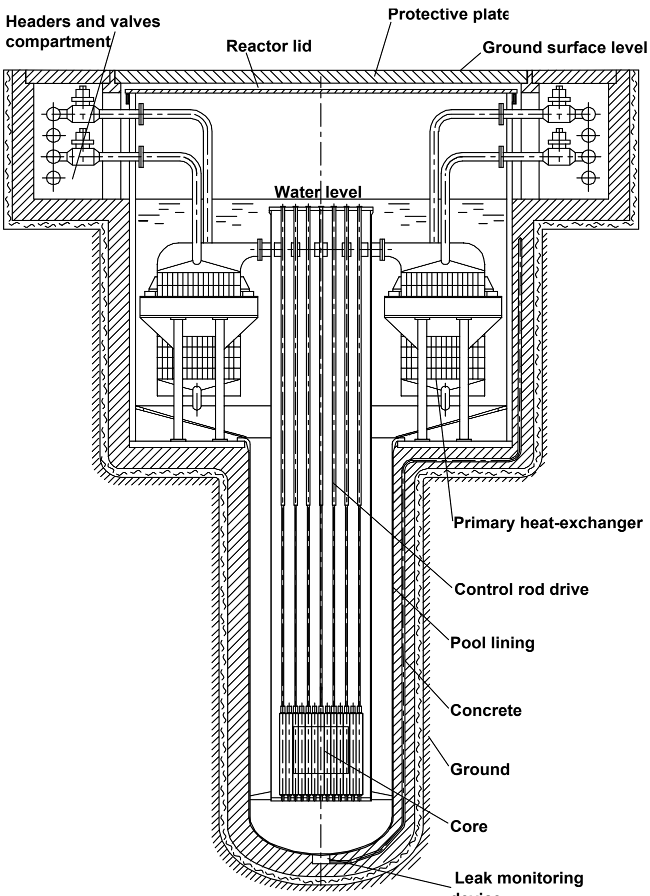


Fig 4.5. Reactor RUTA 55.

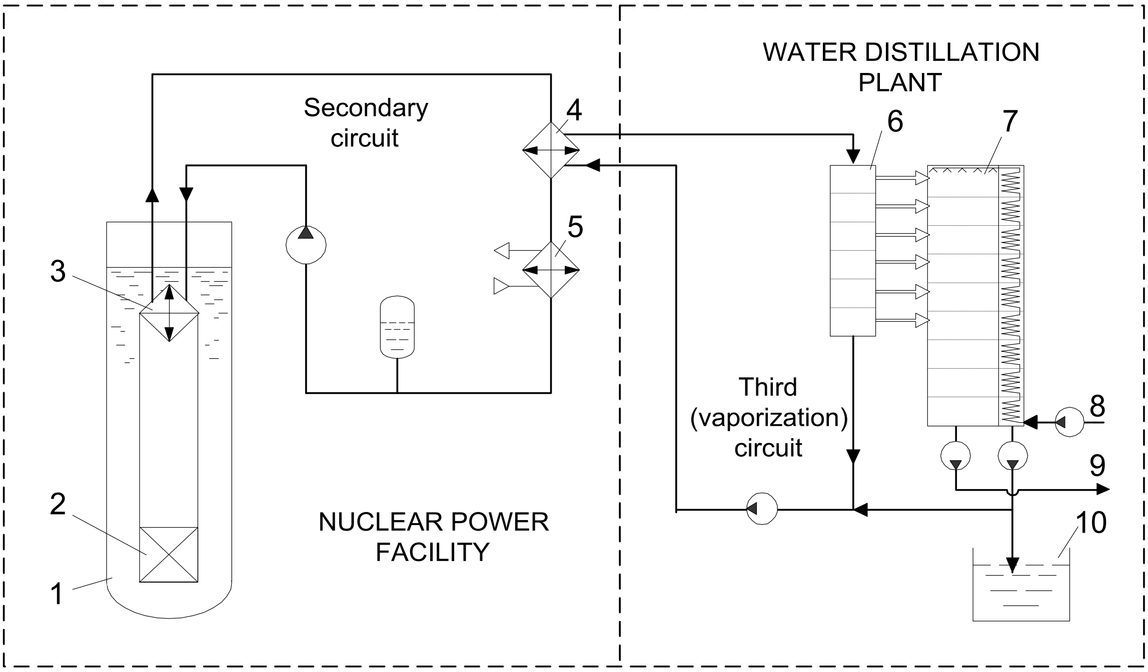


FIG. 4.6. Nuclear desalination complex RUTA. Coupling between the RUTA reactor and a MED desalination plant.

At rather low prices for fossil fuel in the Russian Federation the use of fuel oil power source would gives distillate price about 0.8 $/m^3$. The fuel oil price 50$ per ton used in calculations corresponds to the fuel oil price in the Central European part of the Russian Federation. For remote areas the price could prove to be somewhat higher due to transportation expenses. The nuclear power source becomes competitive at fissile fuel prices rise up to 80 to 100 $/ton.

The economic estimates have demonstrated that the nuclear desalinating facilities with RUTA reactors would be competitive against the similar but fossil fuel-fired facilities in the regions relying on expensive imported fuel. It would be optimal to combine within the desalinating facility two or three reactors of the similar design with the corresponding set of equipment for desalination. The quantity and per-unit capacity of MED-units should be identified with account for needed redundancy.

1-reactor; 2-core; 3-primary heat exchanger; 4- secondary heat exchanger; 5-decay heat removal system; 6-vaporizer; 7-multieffect distillation unit (MED); 8-sea water; 9-brine;
10-fresh water storage

4.4. Small PWRs for heat supply

4.4.1. Background

Nuclear energy in Japan has been penetrating the electricity market in the past three decades and now 52 units of nuclear power plants account for about 36% of the nationwide electricity production. A major role of nuclear energy in Japan has been to produce electricity and for this reason reactor unit size has increased for effective use of site and cost reduction. This environment will continue to be sustained in general, but in the "Long-Term Program for Research, Development and Utilization of Nuclear Energy in Japan", which was established at the end of the year 2000, development of more innovative reactor technologies including small reactors was proposed. Several organizations have begun to assess the possibility of small reactors for electric power supply and other purposes. The science Council of Japan is reviewing the potential seawater desalination with nuclear energy to supply fresh water for agriculture purposes.

Japan's experience with seawater desalination by nuclear energy, accumulated in several sites of nuclear power plants provides valuable operational and maintenance information, in spite of low water production capacity at all sites (below 4000m^3/day). No technical problems for coupling desalination units to nuclear power plants have been encountered during the operating period of more than 100 reactor years.

Japan also has experience in the design, construction and operation of nuclear power plants in the small and medium size range SMRs. One potential prototype of such reactors for application to seawater desalination would be a small PWR developed for the experimental nuclear ship "Mutsu". Based on this experience, two types of small PWRs for heat supply are currently being studied: The first is a Passive Safe Reactor for Distributed Energy System (PSRD) by JAERI, and the second is a Small Heating Reactor by Mitsubishi/CRIEPI.

4.4.2. Design description

Passive safe reactor for distributed energy system (PSRD)

PSRD is a distributing energy supply system, such as heat supply (district heating, seawater desalination, etc.). The reactor is a small size LWR with passive safety capability, which expands the possibilities for various non-conventional applications.

JAERI has continued the design study of an advanced marine reactor MRX (Marine Reactor X [43]. The results of this study will surely be contributed to the design of PSRD.

(a) Nuclear Reactor

The PSRD (see Fig 4.7) is an integral type reactor in which all components of the primary and secondary systems are installed inside the reactor vessel in order to decrease the possibility of Loss of Coolant Accident (LOCA) and to eliminate the possibility of a rod ejection accident. The pipes penetrating the pressure vessel include only the feed water pipes, the steam pipes and the pipe for the safety valve. These pipes are connected at the upper cover of the reactor vessel. To realize this design concept, possibilities of eliminating the volume control system and adopting an emergency decay heat removal system without the penetrating pipe are now being examined in a detailed design study.

The PSRD core consists of 37 assemblies with Zyrcaloy cladding UO_2 fuel rods, with specification similar to those used in the 17 × 17-type fuel assembly for the current PWR. The core life cycle is estimated as 8 years with ^{235}U of about 4% enrichment, by assuming a core load factor of 50%. The core is cooled by natural circulation and the primary coolant is self-pressurized. The temperature at the core outlet of the primary coolant is 233ºC, and its pressure is 3 MPa. The temperature and pressure of the steam produced by the SG are 180ºC and 0.88 MPa, respectively.

(b) Safety

The space inside the containment vessel is maintained at a vacuum to thermally shield it during normal operation, preventing heat transfer through the space in the containment. In a case of a LOCA emergency, the water in the water tank will fill the containment zone. The decay heat in the core can be passively removed by conduction through the reactor vessel wall to the water filled in the containment. The heat transferred to the water can be rejected through the emergency cooler in the containment zone. The coupling with desalination systems and the related safety issues, which are common in heating reactors, are described below

Small Heating Reactor

A small heating reactor (36 MW(th) for multi-applications, such as seawater desalination and district heating, is currently under conceptual design.

(a) Nuclear Reactor

The marine reactor mounted on the experimental nuclear ship "Mutsu" had been adapted for land use [45] at lowered, mid-range pressure and temperature.

The fuel assemblies of the original "Mutsu" reactor are replaced with 17 × 17 (short-length type assemblies) similar to those of commercial PWR. Control rods have been changed from the cruciform type to the cluster type. The follower installed at the bottom of the "Mutsu" control rods has been removed, and as a result, the height of reactor vessel has been reduced by 1 m.

TABLE 4.7. SMALL HEATING REACTOR
Principal design parameters

Parameters		Parameters	*Seawater Desalination*	*Distinct Heating*
Electrical Output (MW(e))	-	Reactor Coolant System		
		Number of Loops	2	
NSSS Thermal Output (MW(th))	36	Operating Pressure (MPa)	4.4	
		Temperature		
		Reactor Outlet (^{o}C)	240.0	
		Reactor Inlet (^{o}C)	224.5	
Reactor Type	PWR			
Reactor Core	Low Core Power Density			
Fuel Assemblies		Steam Generators		
Array	17 × 17	Number	2	
Number	21	Type	Vertical, U-Tube	
		Steam Pressure (MPa)		
Turbine	-		2.1	0.9
Containment Vessel Type	Cylindrical Steel Containment with Hemispherical Head	Reactor Coolant Pumps		
		Number	2	
		Type	Canned Motor Pump	

The reduced operating pressure in the reactor vessel makes it possible not only to improve transportability performance of reactor vessel, but also to use standard-grade flanges for isolation between tie-ins isolation valves of the reactor vessel and main coolant piping. Furthermore, weight reduction of the whole primary system contributes to reducing initial plant construction costs. Principal design parameters, conceptual drawing of reactor vessel and overall primary system diagram are as shown in Table 4.7, Figure 4.8 and Figure 4.9, respectively.

(b) Desalination system

Reverse osmosis (RO) membranes are connected to high-pressure pumps driven by a steam turbine [46]. Fresh water can be produced efficiently using the method of driving a high-pressure pump for membranes directly with the steam generated by steam generators. If the concentrated seawater that does not pass through the reverse osmosis membrane (brine) is fed into a power recovery turbine, then a source of unit auxiliary electric power of nuclear desalination system will also be temporarily supplied. Of course it will be available to make use of recovery power to reduce a load of a high-pressure pump. The system configuration of the small heating reactor coupled with a reverse osmosis process is as shown in Figures 4.10.

In general, electric energy is used to drive a high- pressure pump in the present Reverse Osmosis (RO) desalination systems. In an advanced RO system proposed here, the high-pressure pump is driven by a steam turbine with steam fed directly from the SG. It is also technically possible to combine this reactor with a distillation process (MED or MSF). A possible system configuration of the small heating reactor coupled with a multi-effect distillation process is shown in Figure 4.11.

(c) Safety

Three aspects of nuclear desalination-related safety issues are discussed:

– Preventing contamination of product fresh water by radioactive substance;
– flexibility in meeting seasonal fluctuation of water demand; and
– proliferation resistance.

Various design measures were considered in order to prevent release of radioactive substances into the secondary loop due to tube rupture of a steam generator. The secondary loop (enclosed by a dotted line in Figure 4.9) is designed for the same design pressure detected by primary loop. The main steam and feed water isolation valves will be closed immediately when the radioactive monitoring device located in main steam pipe detects radioactivity. Then secondary loop will be pressurized up to the same operating pressure as that of the primary loop and a leak will be halted by the equalizer effect. Radioactivity will be quickly detected in the case of small and medium reactors because of increase sensitivity due to smaller pipes. The feed seawater, which enters into desalination system, is protected from radioactive substances by use of an intermediate loop between the reactor primary loop and the desalination system.

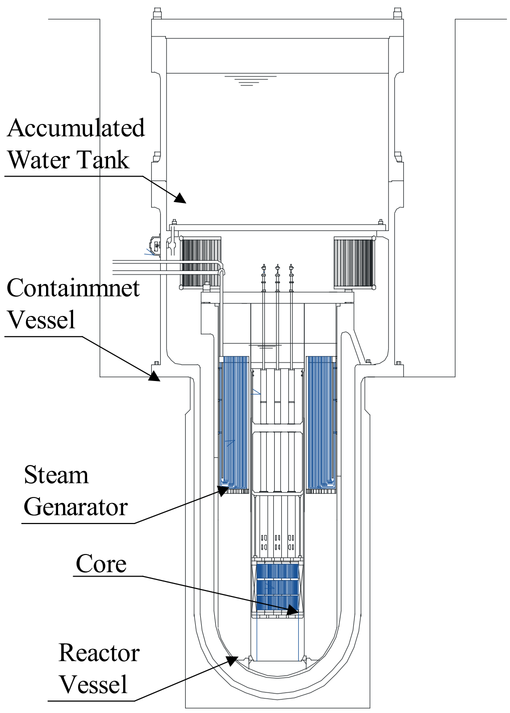
As for load following fresh water storage facility will have major roll to absorb fluctuation of water demand. Nevertheless small and medium water reactor have more flexibility than large reactor for load following characteristics, it had better to operate reactor at a constant power level (base load operation) and desalination process are also suitable for continuous operation especially RO process.

By adapting the exterior fuel exchange mentioned above, safeguard against nuclear proliferation increase at a plant site because the reactor vessel head is in principle never opened on site, in principle. The reactor vessel is transported from the plant site to a fuel exchange base in the lay-down condition within a transportation cask after long-term operation with one batch. And vessel head is opened for refuelling and inspection at the fuel exchanged base. A cooling method against core decay heat after plant shut down and radioactive protection against dosage are required.

4.4.3. Economic evaluation

Initial plant construction costs are reduced, since operating conditions of primary loop reduced into mid-range pressure and temperature specifications and fuel handling facilities at the site become unnecessary through centralizing them at a fuel exchange base. Plant operation costs are also reduced, since numbers of refueling time decrease accompany with long term operation. In addition, the reactor vessel is repeatedly used until its service life has expired.

The overnight construction costs of small PWRs were estimated in accordance that of the currently small power reactor (high pressure and temperature service), and then the costs of desalted water by RO could achieve target 1US$/m^3 easily without mass production effect of small PWRs. Fuel exchange base, fresh water storage tank and transportation costs were excluded in this estimation.



Major parameters

Thermal output (MW)	100
Core outlet temp.(℃)	233
Core pressure (Mpa)	3
Core life (year)	8
No. of fuel assembly	37
Fuel arrangement	square lattice
Fuel	UO_2 (4% enrichment)
Fuel cladding	Zry-4
No. of CRDMs	21
Steam generator	
・steam flow rate(t/h)	150
・temperatures of steam/feed water (℃)	180/90
・pressure of steam(MPa)	0.88
Diameter of containment(m)	8.5
Height of containment(m)	22

FIG. 4.7. Concept of PSRD.

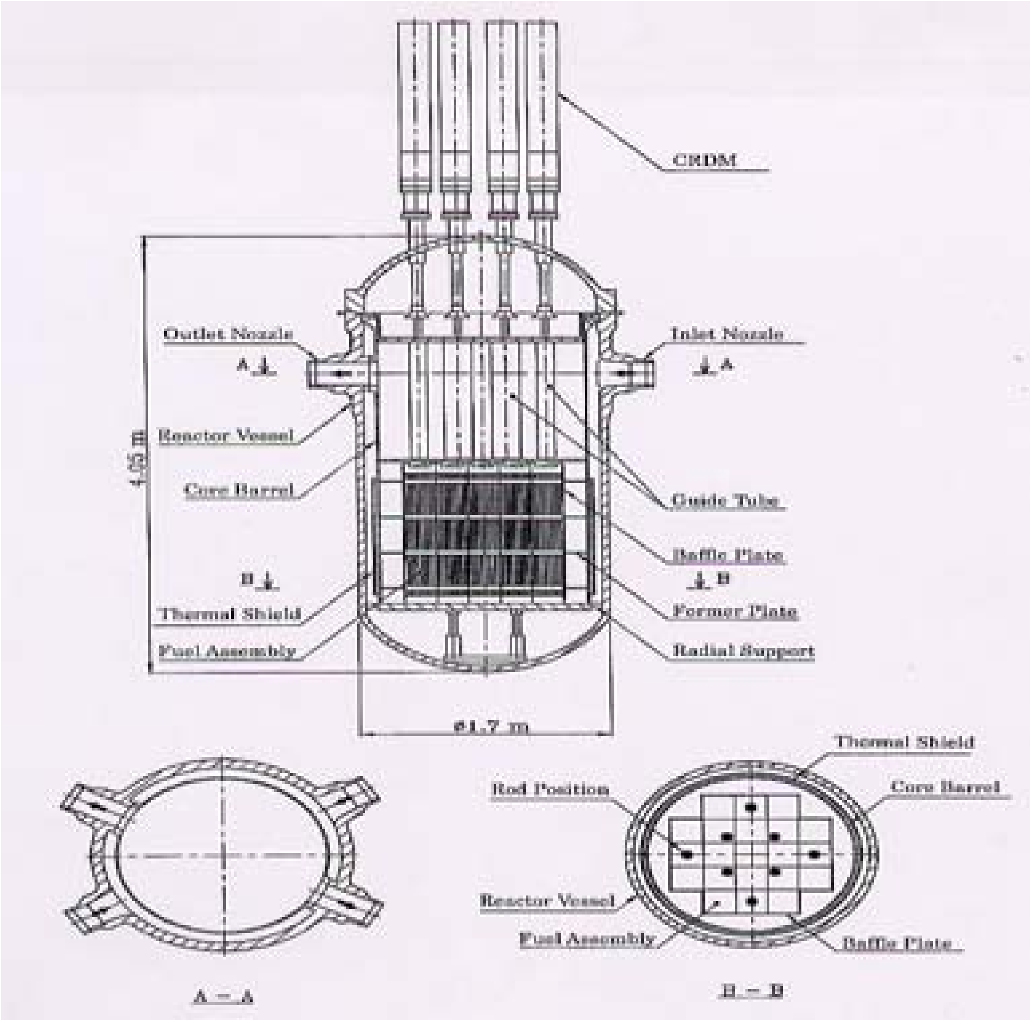


FIG. 4.8. Conceptual drawing of reactor vessel — small heating reactor.

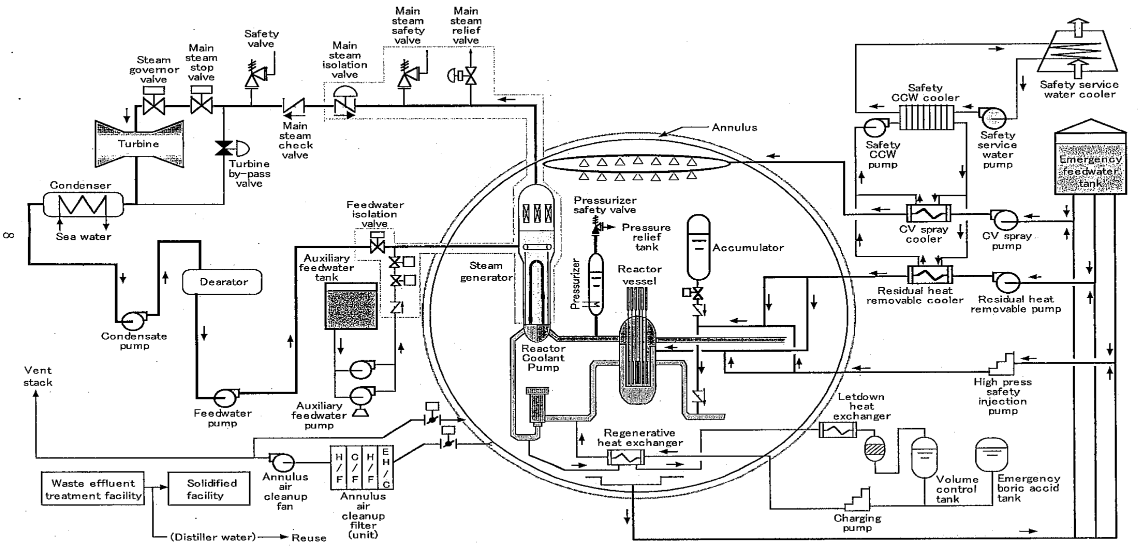


FIG. 4.9. Small heating reactor – overall primary system diagram.

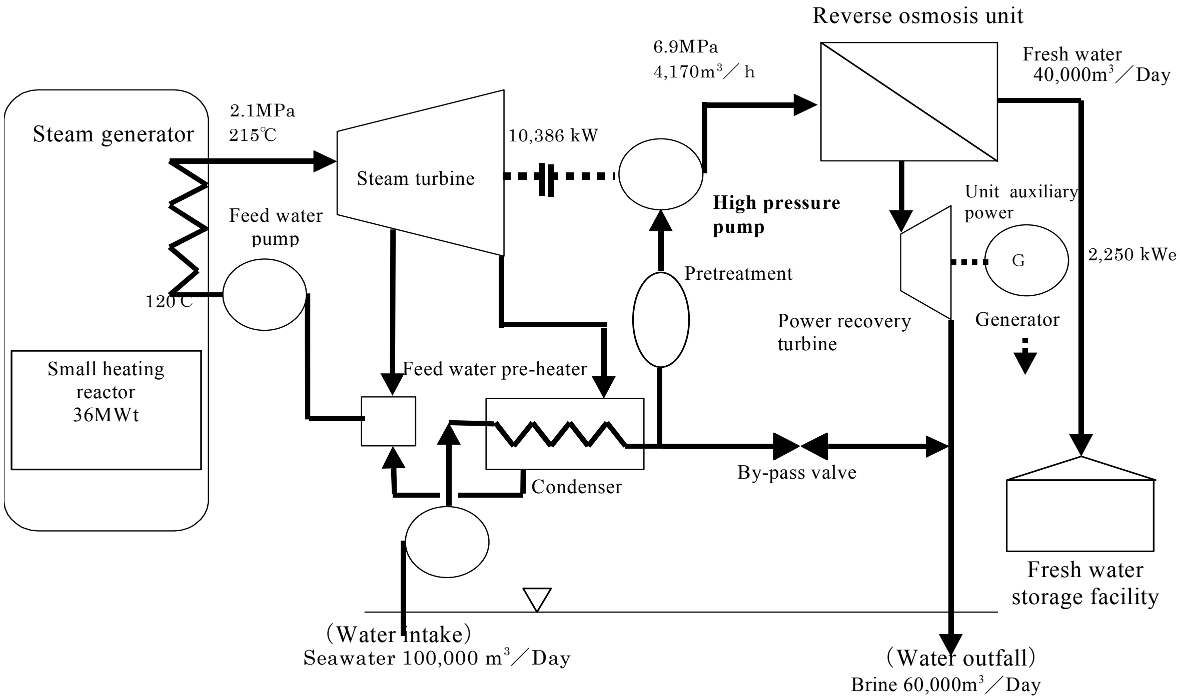


FIG. 4.10. Desalination system with small heating reactor in case of RO. (Steam driven type RO system)

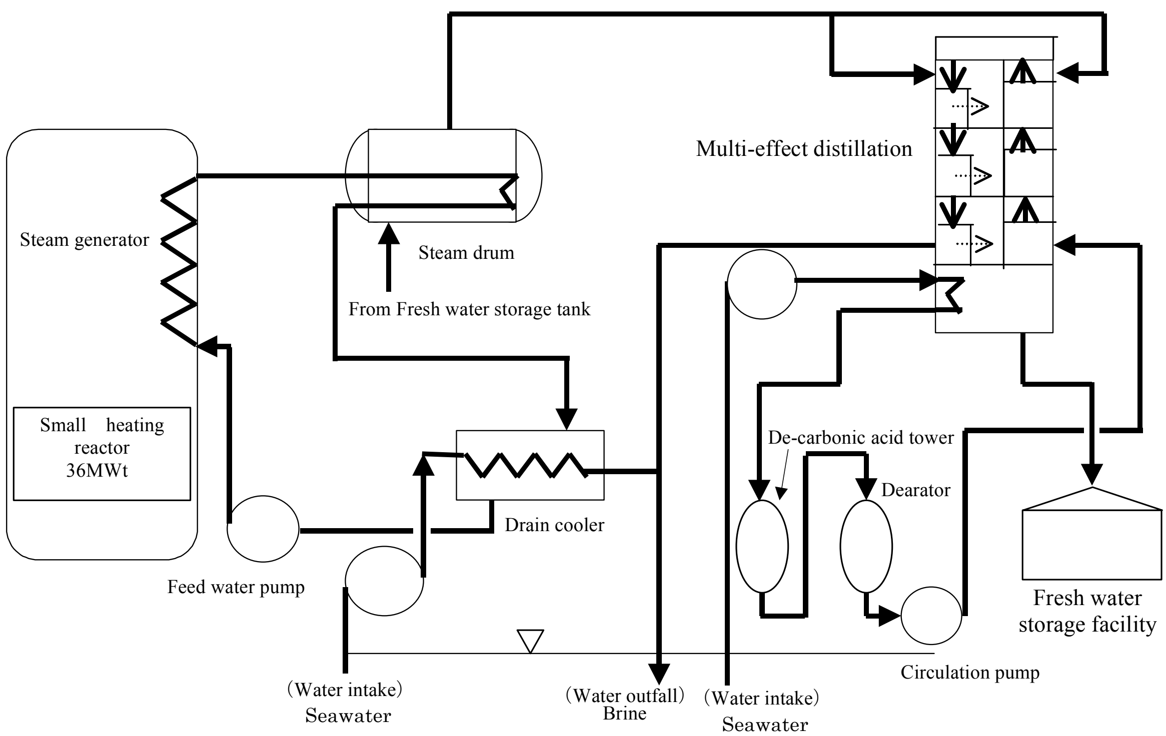


FIG. 4.11. Desalination system with small heating reactor in case of MED.

CONCLUSION

Interest in nuclear desalination has grown in many Member States over the past decade. Some of the States have, therefore, decided to launch nuclear desalination demonstration programmes, which are currently underway or being planned for the near future.

Energy required for desalination could be provided by nuclear reactors in the form of heat and/or electricity. A number of factors contribute to promoting nuclear desalination projects, which include: growing concerns about the environmental effects of burning fossil fuels; recognition of the benefits of diversification of energy sources; expected spin-off effects in industrial development; and the development of new advanced reactor concepts in the small- and medium-power range.

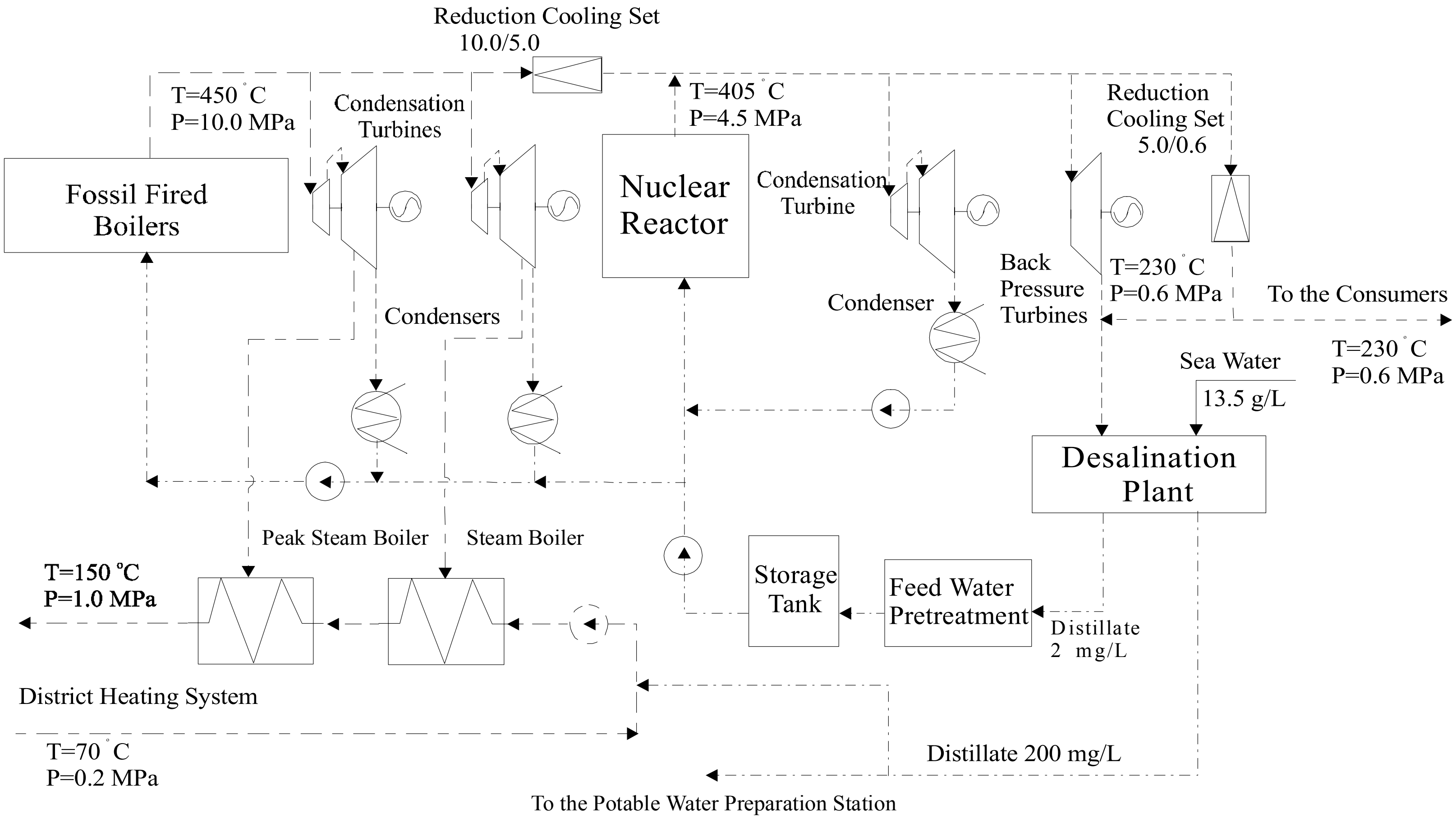
All nuclear reactors can provide sufficient energy to the energy-intensive desalination processes. Operating experience with nuclear desalination has been accumulated via a liquid-metal cooled fast reactor BN-350 in Kazakhstan and several PWR units in Japan. India is currently connecting a desalination facility to two existing PHWR units for demonstration. Other reactor types are also being evaluated for application such as, integral type PWRs, nuclear heating reactors, HTGRs, and BWRs. Operating experience of nuclear desalination complexes and relevant experience in district heating have shown technical feasibility of using nuclear energy for seawater desalination. The basic requirement for preventing radioactive contamination of the desalted water and/or the atmosphere is provided by at least two mechanical barriers and pressure reversal between the reactor primary coolant and brine, which is the case for existing operating facilities. Three mechanical barriers exist in some applications (in operation and under evaluation).

The safety, regulatory and environmental relating to nuclear desalination are those directly related to nuclear power plants (NPPs), with due consideration given to the coupling between the NPPs and the desalination plants. The reactors used for desalination purposes will be designed, constructed and operated in accordance with internationally recognised safety standards for NPPs. Revised IAEA requirements for the design of NPPs do include specific requirements for nuclear reactors used for co-generation. One of the most important requirements, which specifically addresses nuclear desalination plants, is the prevention of radioactive contamination of product water.

One of the most decisive factors for a successful large-scale deployment of nuclear desalination plant complexes is their economic competitiveness. Operating experience in Kazakhstan using an LMR and in Japan with PWRs may not be a good indicator of economic viability in many developing countries, which are now considering nuclear desalination. Demonstration of economic viability under local conditions in such countries will therefore be indispensable in such countries. Indeed, in the coming years, design efforts of advanced SMRs with enhanced safety features and low cost will be an encouraging element for strengthening advantages of nuclear desalination and incentives for its wider deployment.

APPENDIX

1. Principal flow diagramme of the desalination complex at Aktau, Kazakhstan
2. Evaporators at Aktau, Kazakhstan, powered by BN-350 since 1973 till 1999 producing about 80 000 m^3/d of fresh water for industrial use and drinking
3. Water distribution line at Aktau, Kazakhstan
4. Water product characteristics of Aktau, Kazakhstan
5. Schematic illustration of connection between NSSS and desalination facility at Ohi, Japan
6. Multi-Stage Flash Unit connected to the nuclear power plant at Ohi, Japan
7. Multi-Effect Distillation Unit connected to the nuclear power plant at Genkai, Japan
8. Key characteristics of seawater desalination systems at the Genkai Nuclear Power Station, Genkai, Japan (measured between 4.1999 and 3.2000)
9. Kashiwazaki-Kariwa Unit 1 with MSF seawater desalination facility, Japan (desalination facility dismantled in 1999)
10. Kashiwazaki-Kariwa Nuclear Power Station Unit 1
11. MSF Facility coupled with Kashiwazaki-Kariwa Unit 1
12. Flow Diagram of MSF facility in Kashiwazaki-Kariwa Unit 1
13. Construction site of the nuclear desalination plant at Kalpakkam, India (February 2001)



Principal flow diagramme of the desalination complex at Aktau, Kazakhstan

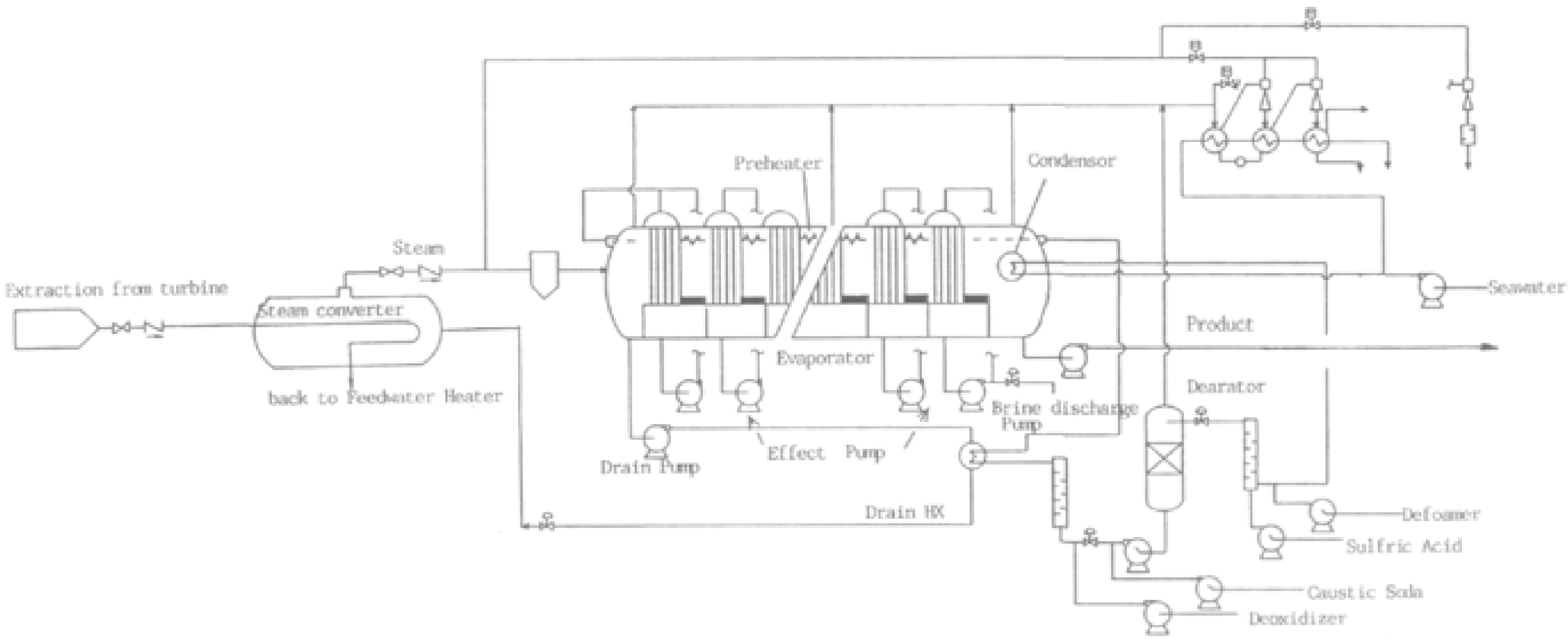
Evaporators at Aktau, Kazakhstan, powered by BN-350 since 1973 till 1999 producing about 80 000 m^3/d of fresh water for industrial use and drinking

Water distribution line at Aktau, Kazakhstan

Characteristics	WHO guideline Values	Distillate «G» For boiler feed water	Distillate «A» For drinking water
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
Sulphate (mg/l)	400	0.31	33.2
Calcium (mg/l)	NG	0.08	7.6
Magnesium (mg/l)	NG	0.09	8.2
Sodium (mg/l)	200	0.18	48.5
Aluminium (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	-	-
B-activity (Bq/l)	1.0	-	-

NG - no guideline value set

Water product characteristics of Aktau, Kazakhstan



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

Multi-Stage Flash Unit connected to the nuclear power plant at Ohi, Japan

Multi-Effect Distillation Unit connected to the nuclear power plant at Genkai, Japan

	Performance			
	Capacity (ton/d)	Purpose	Elect. Conductivity (μs/cm)	Dissolved Solids (mg/l)
RO system	1000	plant use	about 200	about 95
Evaporation type	1000	plant use	about 3	-
Evaporation type	800	drinking	about 1–2	-

FIG. 8. Key characteristics of seawater desalination systems at the Genkai nuclear power station, Genkai, Japan (measured between 4.1999 and 3.2000).

General

Location	Niigata-Pref., Japan
Owner	Tokyo Electric Power Co.
Operator	Tokyo Electric Power Co.
Main suppliers	Toshiba Co.
Type	BWR
Construction Started	1980/06/05
Connected To Electricity Grid	1985/02/13
Commercial Operation	1985/09/18
Net Capacity	1,067 MW(e)
Lifetime Generation	109,457.15 GWeh as of 1999
Cumulative Energy Avail. Factor	80.94% as of 1999

Desalination facility[12]

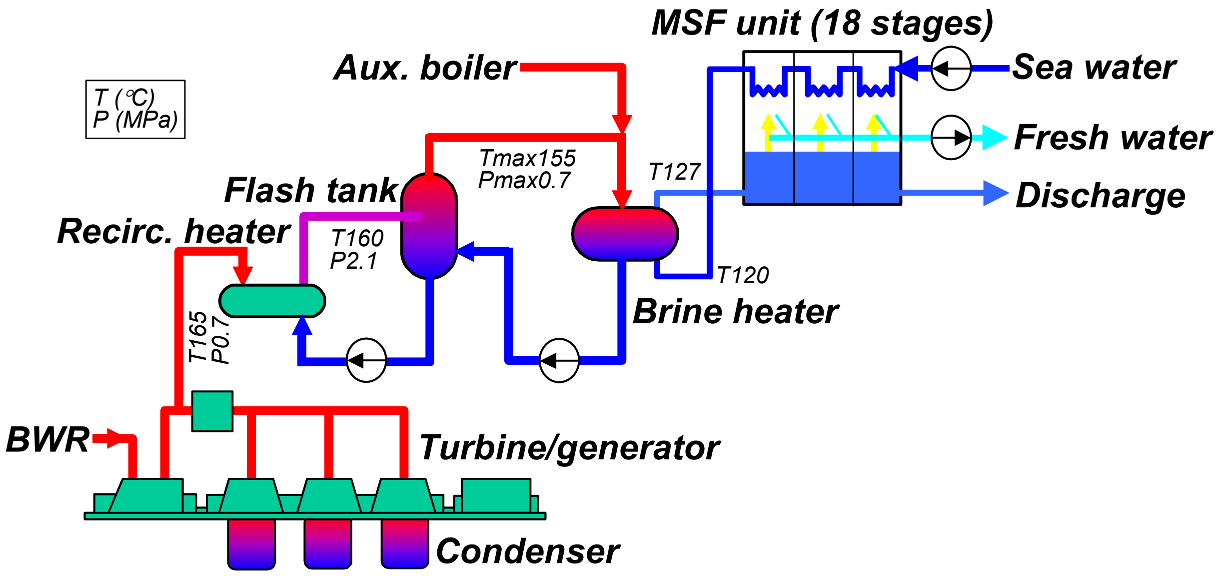
Type	Multi-stage flash
Number of units	2
Water production rate	500 m^3/d/unit
Purpose	Fresh water for internal uses
Design condition	
Seawater temperature	29°C
Discharge temperature	Less than (seawater temp. + 7°C)
Discharge pH	7.8–8.5
TDS of fresh water	Less than 20 ppm

FIG. 9. Kashiwazaki-Kariwa Unit 1 with MSF seawater desalination facility, Japan (desalination facility dismantled in 1999).

[12] This facility was not put into service after construction completed because local fresh water resources were made available.

Kashiwazaki-Kariwa Nuclear Power Station Unit 1
(most right hand side)

MSF Facility coupled with Kashiwazaki-Kariwa Unit 1



Flow Diagram Of MSF facility in Kashiwazaki-Kariwa Unit 1

ABBREVIATIONS

ARCIC	advanced reactor core isolation cooling system
BWR	boiling water reactor
CDF	core damage frequency
CEDM	control element drive mechanisms
CUW	reactor water clean-up system
DBA	design basis accident
DG	standby diesel generator system
ECCS	emergency core cooling system
EFPM	effective full power month
FA	fuel assembly
FPC	fuel pool cooling and filtering system
GTG	standby gas turbine generator system
HP	high pressure
LOCA	loss of coolant accident
LP	low pressure
LPFL	low pressure flooder (system)
MCP	microchannel plate
MD	motor drive
MED	multi-effect distillation
MSF	multi-stage flash
NPP	nuclear power plant
NSSS	nuclear steam supply system
PCC	passive containment cooler
PCCS	passive containment cooling system
PCV	primary containment vessel
PWR	pressurized water reactor
RHR	residual heat removal (system)
RO	reverse osmosis
RPV	reactor pressure vessel
SLC	standby liquid control system
SPCU	suppression pool clean up (system)
TB	station blackout
TC	anticipated transient without scram
TD	theoreticaldensity
TDS	total dissolved salts
TQUV	loss of high and low pressure core cooling
TQUX	loss of high pressure core cooling and depressurization
TW	loss of decay heat removal

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