IAEA-TECDOC-1444

Optimization of the coupling of nuclear reactors and desalination systems

Final report of a coordinated research project 1999–2003

IAEA
International Atomic Energy Agency

June 2005

IAEA-TECDOC-1444

Optimization of the coupling of nuclear reactors and desalination systems

Final report of a coordinated research project 1999–2003

IAEA
International Atomic Energy Agency

June 2005

The originating Section of this publication in the IAEA was:

Nuclear Power Technology Development Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

OPTIMIZATION OF THE COUPLING
OF NUCLEAR REACTORS AND DESALINATION SYSTEMS
IAEA, VIENNA, 2005
IAEA-TECDOC-1444
ISBN 92–0–102505–X
ISSN 1011–4289

Printed by the IAEA in Austria
June 2005

FOREWORD

Nuclear power has been used for five decades and has been one of the fastest growing energy options. Although the rate at which nuclear power has penetrated the world energy market has declined, it has retained a substantial share, and is expected to continue as a viable option well into the future. Seawater desalination by distillation is much older than nuclear technology. However, the current desalination technology involving large-scale application, has a history comparable to nuclear power, i.e. it spans about five decades.

Both nuclear and desalination technologies are mature and proven, and are commercially available from a variety of suppliers. Therefore, there are benefits in combining the two technologies together. Where nuclear energy could be an option for electricity supply, it can also be used as an energy source for seawater desalination. This has been recognized from the early days of the two technologies. However, the main interest during the 1960s and 1970s was directed towards the use of nuclear energy for electricity generation, district heating, and industrial process heat.

Renewed interest in nuclear desalination has been growing worldwide since 1989, as indicated by the adoption of a number of resolutions on the subject at the IAEA General Conferences. Responding to this trend, the IAEA reviewed information on desalination technologies and the coupling of nuclear reactors with desalination plants, compared the economic viability of seawater desalination using nuclear energy in various coupling configuration with fossil fuels in a generic assessment, conducted a regional feasibility study on nuclear desalination in the North African Countries and initiated in a two-year Options Identification Programme (OIP) to identify candidate reactor and desalination technologies that could serve as practical demonstrations of nuclear desalination, supplementing the existing expertise and experience.

In 1998, the IAEA initiated a Coordinated Research Project (CRP) on Optimization of the Coupling of Nuclear Reactors and Desalination Systems with participation of institutes from nine Member States. The CRP was initiated as a step forward for facilitating an early deployment in developing countries, where nuclear desalination is being considered as an option to cope with fresh water deficit as well as energy in the coming decade.

The CRP has enabled the IAEA and participating institutes to accumulate relevant information on the latest research and development in the field of nuclear desalination and share it with interested Member States. The CRP has produced optimum coupling configurations of nuclear and desalination systems, evaluated their performance and identified technical features, which may require further assessment for detailed specifications of large-scale nuclear desalination plants.

This publication highlights the outcomes of projects under this CRP and draw lessons and suggestions for further investigation for deployment of nuclear desalination.

The IAEA officer responsible for this publication was B.M. Misra of the Division of Nuclear Power.

CONTENTS

CHAPTER 1. INTRODUCTION

1.1. BACKGROUND

The first nuclear reactor designed for electricity generation was constructed in the town of Obninsk in Russia and was operated on June 27, 1954. It was a 5 MW(e) power reactor. The first commercial nuclear power plant in the United States was of the pressurized water Reactor (PWR) type. It was originally designed in 1954 by Westinghouse Bettis Atomic Power Laboratory for military ship applications, and then redesigned by the Westinghouse Nuclear Power Division for commercial applications. The plant (90 MW(e)) was constructed at Shippingport, near Pittsburgh, USA and became operational in 1957. The first gas cooled Reactor (GCR) was a 50 MW(e) reactor constructed also in 1957 in Calder Hall, UK. In 1960, the first boiling water reactor (BWR) was constructed in Dresden, USA with a net electrical power of 200 MW(e). The 22-MW(e) Nuclear Power Demonstration (NPD) was constructed in Canada in 1962 as a prototype for the pressurized heavy water reactor (PHWR).

Nuclear power has been used for five decades and has been one of the fastest growing energy options. By the end of 2003, there were 439 power reactors in operation worldwide, with a total installed capacity of 361 GW(e) [1.1]. There were also 26 reactors under construction, with a total capacity of 22 GW(e). In 2003, about 16% of the world electricity was generated by nuclear power. Although the rate at which nuclear power has penetrated the world energy market has declined, it has retained a substantial share, and is expected to continue as a viable option well into the future.

Seawater desalination by distillation is much older than nuclear technology. Prior to the early 1800s, it was practiced almost entirely in shipboard single stage stills, operated in the batch mode, fired directly from the cooking stove or furnace. These stills were bulky and inefficient. Distillation technology evolved slowly lacking major impetus for change. The developing sugar industry at the beginning of the 19th century was to be the driving force for more efficient systems. By the end of the 19th century, multi-effect systems were commonplace and several land-based desalination units were constructed [1.2].

However, the current desalination technology involving large-scale application, has a history comparable to nuclear power, i.e. it spans about five decades. Major milestones include:

(1) Development in the late 1950s of Multi-Stage Flash distillation (MSF) process by Prof. Silver that replaced the costly and uneconomic submerged tubes Multiple Effect Distillation (MED).
(2) Manufacturing in the early 1960s of cellulose acetate membranes by Sourirajan and co-workers that made it possible to use Reverse Osmosis (RO) desalination process.
(3) Development of hollow fiber membranes by Dupont.
(4) Development of vertical tube evaporators that made the MED process more efficient and competitive with MSF.

Seawater desalination has also been growing very fast. By the end of July 2002, the total contracted capacity of all desalination technologies was about 32.4 million m^3/d, of which 60% capacity was for desalting seawater [1.3]. Figure 1.1 depicts the normalized yearly addition of nuclear and desalination capacities [1.4]. The Figure reveals general similarity in the trends of demand development for both nuclear and desalination technologies in the period 1955–91. The demands for both technologies increased rapidly after the 1973 Arab-Israeli war that increased oil prices and reached their peaks just before the collapse of the oil prices in 1986 and the Chernobyl accident in the same year.

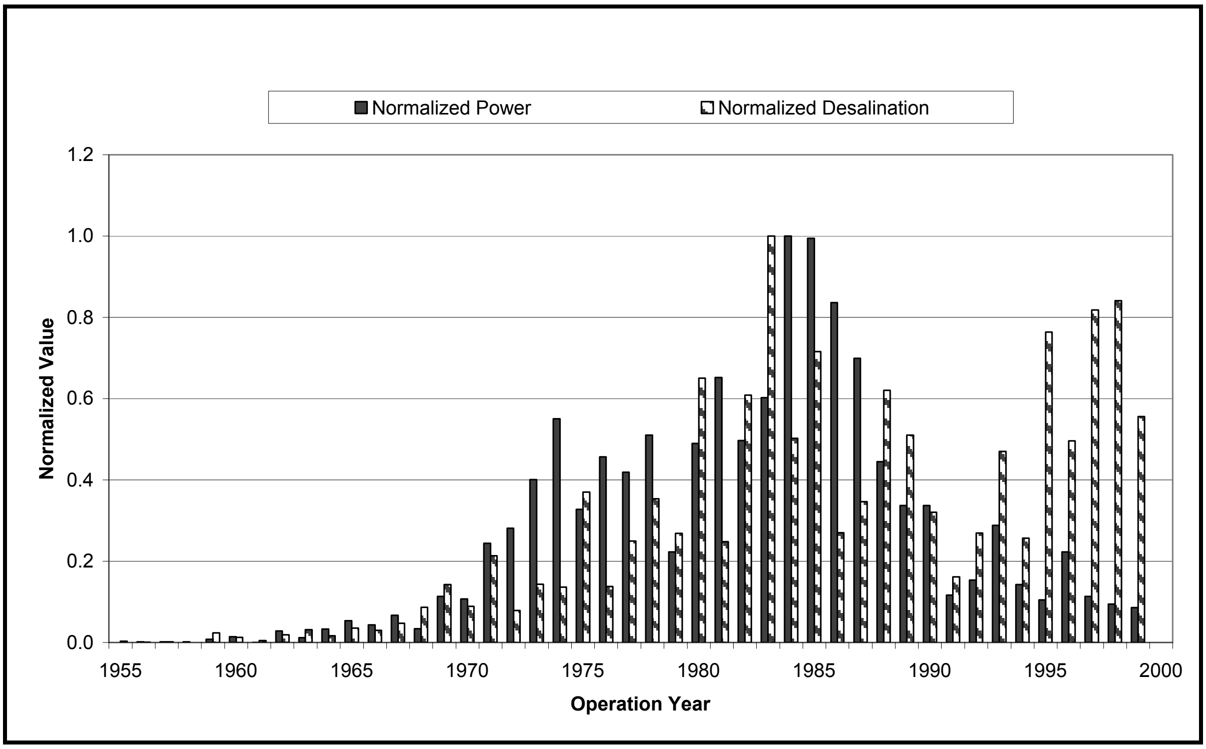


Figure 1.1: Development of Nuclear Power and Desalination Installed Capacity [1.4].

High oil prices encouraged industrialized countries to rely on nuclear energy as a reliable alternative source, and in the same time increased the financial resources of the oil exporting countries in the Middle East, and thus provided them with the means to acquire alternative source of potable water to augment their acute shortage of fresh water resources. With the falling oil prices, annual demand for nuclear power and seawater desalination decreased. However, while this trend persisted for nuclear energy after the second Gulf War in 1991, it was reversed for seawater desalination. Nevertheless, both nuclear and desalination technologies are mature and proven, and are commercially available from a variety of suppliers. Therefore, there are benefits in combining the two technologies together.

Where nuclear energy could be an option for electricity supply, it can also be used as an energy source for seawater desalination. This has been recognized from the early days of the two technologies. As early as in the 1960s, the IAEA surveyed the feasibility of using nuclear reactors for seawater desalination, and has since published a number of reports on the technical and economic aspects of the subject [1.5–1.9] and sponsored an international conference on nuclear desalination in 1968 [1.10]. These studies have drawn attention to the economical advantages of co-generation (combining water and power production into a single system). However, the main interest during the 1960s and 1970s was directed towards the use of nuclear energy for electricity generation, district heating, and industrial process heat.

Renewed interest in nuclear desalination has been growing worldwide since 1989, as indicated by the adoption of a number of resolutions on the subject in the IAEA General Conferences. This has been motivated by a variety of reasons that include: economic competitiveness in areas lacking cheap hydropower or fossil fuel resources, energy supply diversification, conservation of fossil fuel resources, spin-off effects of nuclear technology for industrial development, and environmental protection by avoiding emissions of air pollutants and green-house gases.

Responding to this trend, the IAEA reviewed information on desalination technologies and the coupling of nuclear reactors with desalination plants [1.11], compared the economic viability of seawater desalination using nuclear energy in various coupling configuration with fossil fuels in a generic assessment [1.12], and conducted a regional feasibility study on nuclear desalination in the North African Countries [1.13].

A series of resolutions of the IAEA General Conference have led the IAEA to initiate in 1994 a two-year Options Identification Programme (OIP). The objective of the OIP was to identify candidate reactor and desalination technologies that could serve as practical demonstrations of nuclear desalination, supplementing the existing expertise and experience [1.14]. Demonstration desalination processes need not be implemented at the large-scale commercial production level. Two or three trains or units could provide design and operational characteristics fully representative of larger scale production facilities, as larger plants are simply multiple trains or units operated in parallel. The three options described below were identified as recommendable, practical candidates for demonstration. These options use well-proven water-cooled reactors and desalination technologies [1.14].

Option 1: RO desalination in combination with a nuclear power reactor being constructed or in an advanced design stage, with construction expected in the near term. The preferred capacity of the reactor is in the medium-size range. Two or three RO trains, up to 10,000 m^3/d each, would provide a suitable demonstration. A newly constructed reactor would offer the best opportunity to fully integrate the RO and reactor systems, including feed water preheating and optimization of system design. Such demonstration could readily be extrapolated to larger scale commercial production facilities.

Option 2: RO desalination, as above, in combination with an operating reactor. Some minor design modifications may be required to the periphery of the existing nuclear system. Advantages include a short implementation period, a broad choice of reactor sizes, and the availability of nuclear infrastructures. reactor in the medium-size range is preferred, as it provides a system close to that most likely to be used in commercial facilities.

Option 3: MED desalination in combination with a small reactor. This is suitable for the demonstration of nuclear desalination for capacities of up to 80,000 m^3/d.

The OIP also recommended intermediate steps before or in conjunction with recommended demonstration options, aiming at gradual, partial and progressive confidence building, to reduce unknowns and risks with relatively low cost [1.14].

Such intermediate steps might be for RO:
(a) Small scale preheated seawater desalination with reverse osmosis (RO);
(b) Small scale RO integrated with a NPP;
(c) Larger scale RO integrated with a fossil-fueled power plant.

Similarly, intermediate steps for demonstrating MED might be:
(d) Two or three parallel MED units with a heat source simulating NPP conditions;
(e) A small-scale MED unit connected to a NPP, which is operating, or under construction.

For the ultimate objective of facilitating and promoting commercial deployment of nuclear desalination, programs and projects have to be directed at those issues, which are relevant to large-scale projects. These issues include technical, economic, financial, safety, infrastructure and institutional concepts. Some issues, in particular those technical features, which have a major impact on economic competitiveness and on the overall economics of nuclear desalination, need to be demonstrated in order to confirm assumptions and estimates.

In 1998, the IAEA initiated a Coordinated Research Project (CRP) on “Optimization of the Coupling of Nuclear Reactors and Desalination Systems” with participation of institutes from nine Member States in order to share relevant information, optimize the resources and integrate related research and development in this area. The participating institutions are: INVAP (Argentina), CANDESAL (Canada), INET (China), NPPA (Egypt), BARC (India), KAERI (Republic of Korea), CNESTEN (Morocco), IPPE, OKBM, RDIPE, JSC Malaya Energetica (Russian Federation) and CNSTN (Tunisia).

The CRP has enabled the Agency and participating institutes to accumulate relevant information on the latest research and development in the field of nuclear desalination and share it with interested Member States. The CRP has produced optimum coupling configurations of nuclear and desalination systems, evaluated their performance and identified technical features, which may require further assessment for detailed specifications of large-scale nuclear desalination plants.

This publication highlights the outcomes of projects under this CRP and draw lessons and suggestions for further investigation for deployment of nuclear desalination. The remainder of this Chapter presents the objectives of the CRP, subjects for optimization and investigation and an overview of research projects within the CRP as well as collaboration among participating institutes.

Various reactor and desalination technologies considered within this CRP are presented in Chapter 2. Proposed coupling schemes between nuclear reactors and desalination systems are presented in Chapter 3. Chapter 4 provides description of analytical evaluations and results, including safety, performance and economic assessments, as well as experimental validation of RO and thermal systems. The main conclusions of the CRP regarding the optimization of coupling schemes and relevant technical considerations are summarized in Chapter 5.

1.2. OBJECTIVES

This CRP was initiated as a step forward for facilitating an early deployment in developing countries, where nuclear desalination is being considered as an option to cope with fresh water deficit as well as energy in the coming decade. The main objectives of the CRP are:

- To share relevant information and resources in research activities with respect to those technical features which have major impact on performance, economic competitiveness and reliability of nuclear desalination plant concepts;

- To accumulate relevant information on latest research and development in the field of nuclear desalination and shore it with interested member states;

- To produce the best coupling configurations of nuclear and desalination systems;

- To evaluate their performance; and

- To identify technical features which may require further assessment for detailed specifications of large-scale nuclear desalination plants.

1.3. SUBJECTS FOR OPTIMIZATION AND INVESTIGATION

The overall scope of this CRP is to encompass research and development projects focused on optimized coupling of nuclear and desalination systems in the following major areas:

- Nuclear reactor design intended for coupling with desalination systems.
- Optimization of thermal coupling of NSSS and desalination systems.
- Performance improvement of desalination systems for coupling.
- Advance desalination technologies for nuclear desalination.

The following are brief descriptions of objectives and target products of individual research projects in the CRP. Table 1.1 gives an overview and more details are in the Annexes.

INVAP, Argentina, contributed to the CRP with an objective of developing a flexible modeling tool, DESNU, helpful for evaluating nuclear desalination systems from the safety point of view. Specifically it prepares input files for running RETRAN for evaluating accidental sequences transferring radioactive substances of the reactor cooling system to the water desalination system and eventually to the product water. It contains in its built-in tables optional models of MSF, MED and RO for desalination processes coupled with the balance of plant of a small modular reactor (in principle, a small integral PWR). The scope of all the models were optimized in order to adequately focus on the "residence time per component", as relevant to the convective phenomena over which the contamination is spread, while minimizing the input data requirement from the user.

RO membrane permeability is improved as feed water temperature into the system is increased. This results in the possibility of "preheating" the feed water temperature above ambient seawater temperature, thereby increasing the potential to reduce the cost of water production. *CANDESAL*, Canada, expanded this concept by exploring the possible use of low grade waste heat from the nuclear steam supply system using an advanced RO system design. *CANDESAL* carried out a series of experiments using a newly installed test rig in order to experimentally verify its proposed design and operating methodology.

Institute of Nuclear Energy Technology (INET) at Tsinghua University, China, has been developing a dedicated nuclear heating reactor of varying capacities e.g., 5, 10, 200 MW (th). The objective of the research project in the CRP is to define an optimized coupling scheme, parameters and performances for an integrated nuclear desalination plant using a 200 MW(th) NHR as an energy source. Comparative assessment has led to the selection of a high temperature vertical tube evaporation multi-effect desalination (VTE-MED) scheme for the in-depth examination.

Based on the findings and recommendations of the OIP, *Nuclear Power Plants Authority (NPPA)*, Egypt, decided to construct an experimental RO facility at its site in El-Dabaa to validate the concept of feedwater preheating. NPPA research project has the following objectives:

- Overall: to investigate experimentally whether the preheated feedwater can be realized in actual operation.
- Short-term (~3 years): to study the effect of feedwater temperature and pressure on RO membrane performance over a range of temperatures (20-45°C) and pressures (55-69 bar).
- Long-term: to study the effect of feedwater temperature and pressure on RO membrane performance as a function of time.

TABLE 1.1: SUMMARY OF NUCLEAR DESALINATION PROJECTS WITHIN THE IAEA CRP
"OPTIMIZATION OF THE COUPLING OF NUCLEAR REACTORS AND DESALINATION SYSTEMS"

COUNTRY/ INSTITUTION	**PROJECT**	**OBJECTIVES**	**STATUS at the end of 2003**
1- Argentina/ INVAP	Evaluation of Nuclear Desalination Coupled Systems	- Development of a flexible calculating tool (DESNU) to be used for the safety evaluation of nuclear desalination systems.	- DESNU spreadsheet was developed achieving the goal of producing RETRAN input files for computer modelling of MED, MSF and RO desalination systems, and the BoP of a small NPP. - INVAP also achieved the additional goals of flexibility, review by collaboration and a User's Manual, making DESNU an open product for the Nuclear Desalination community.
2- Canada/ CANDESAL	Optimized Nuclear Desalination/ Cogeneration Using Reverse Osmosis	- To demonstrate through analytical and experimental work the improved performance characteristics and economic advantages as a result of using RO desalination with waste heat from energy generation process used to enhance desalination process.	- Construction of test facility completed and 1st phase of the experimental program carried out successfully. - Results indicated that improvements anticipated in analytical projections could be achieved in practice.
3- China/ INET	Optimization of Seawater Desalination with Nuclear Heating Reactor	- To define an optimized coupling scheme for MED plant with NHR-200 nuclear heating reactor.	- A computer program was developed to investigate thermal hydraulic performance of coupling VTE-MED with NHR-200 heating reactor. - A thermal hydraulic test facility was constructed for validation of the developed computer program. - Experimental work and economic evaluation of coupling schemes were carried out.

TABLE 1.1: (Cont.)

COUNTRY/ INSTITUTION	PROJECT	OBJECTIVES	STATUS at the end of 2003
4- Egypt/ NPPA	Investigation of Feedwater Preheating Effect on RO Performance	- *Overall:* to investigate whether the projected improvements due to preheated feed water can be realized in actual operation. - *Short-term (-3 years):* to study the effect of feed water temperature and pressure on RO membrane performance characteristics over a range of temperatures and pressures. - *Long-term:* to study the effect of feed water temperature and pressure on RO membrane characteristics as a function of time.	- Construction of the experimental facility continues with delays due to multiple reasons. - The overall plan for the research work is expected to be delayed for one year.
5- India/ BARC	Performance Improvement of Hybrid Desalination Systems for Coupling to Nuclear Reactors.	- To couple a 6300 m^3/d hybrid MSF-RO desalination plant with 2 x 170 MW(e) PHWR units at the Madras Atomic Power Station (MAPS). - To carry out basic studies on performance behavior of MSF and RO plants as related to the MAPS project. - Investigation of waste heat Temperature on LTE plant performance.	- Site and installation work for the Nuclear Desalination Demonstration Project completed. - Effects of top brine temperature and seawater inlet temperature on MSF performance were examined. - The effect of elevated feed water temperature on RO membrane performance was studied.
6- Korea, Republic of/ KAERI	Optimal Coupling of SMART and Desalination Plant	- To investigate the optimal coupling of SMART Reactor with MSF & MED for electricity generation and seawater desalination.	- Detailed configuration and performance analyses of coupling MED and SMART completed. - The standard design of the 10000 m^3/d MED-TVC was carried out.

TABLE 1.1: (Cont.)

COUNTRY/ INSTITUTION	PROJECT	OBJECTIVES	STATUS at the end of 2003
7-Morocco/CNESTEN	Optimization of Coupling of Nuclear Reactors and Desalination Systems in Morocco.	- Investigating the optimal coupling of nuclear reactors and desalination systems for Moroccan sites, taking into account the energy and water demand and with emphasis on a new generation SMR	- Two sites were selected, namely Agadir and Laayoune - Economic evaluation utilizing DEEP code was carried out for two scenarios for coupling MED and RO with 600 MW PWR and GTMHR.
8- Russian Federation/ IPPE, OKBM, RDIPE	Russian Small-sized Nuclear Reactors as an Energy Source for Nuclear Desalination Plants: Coupling and Economic Evaluation.	- Development of coupling schemes for nuclear desalination plants using small-sized reactors. - Assessment of performance and cost data for integrated systems. - Evaluation of economic parameters of nuclear desalination plants. - Optimization of nuclear desalination plants design, performance and economic characteristics.	- The designs of RUTA and NIKA reactors were changed to improve performance of coupled desalination process. - Various coupling schemes were developed using KLT-40C reactor aboard a floating power unit. - DEEP was modified to meet Russian needs. The revised version was used in the economic assessment.
9- Tunisia/ CNSTN	Optimization of Coupling Desalination Systems with a Nuclear Reactor.	- Optimization of coupling schemes for various combinations of nuclear reactor and desalination processes. - Economic evaluation of the integrated nuclear desalination systems.	- Three nuclear power plants (PWR-900, SCOR-600 and GT-MHR) and two fossil fuel power plants (steam turbine “SSB-600” and CC 600) were coupled with MED, RO and preheated RO. - Economic evaluation of the integrated systems was carried out using CEA-modified version of DEEP. - Detailed analysis of the calculated results showed that the nuclear option becomes more economic if current gas TEP prices increase by more than 15% .

Bhabha Atomic Research Centre (BARC), India, is undertaking a nuclear desalination demonstration project at Kalpakkam to set up a hybrid 6300 m^3/d (4500 m^3/d MSF and 1800 m^3/d SWRO) sea water desalination plant coupled with two units of 220 MW(e) PHWRs at the Madras Atomic Power Station (MAPS). The requirements of seawater, steam and electrical power for the desalination plants are met from MAPS I & II which are around 1.5%, 1.0% and 0.5% of that available at MAPS. The hybrid plant has provision for redundancy, utilization of streams from one to other and production of two qualities of products for their best utilization. The project has the main objective of demonstrating the capability for design, fabrication and operation of large size future plants.

The research in the framework of this CRP has the main objective of collecting relevant data for improving the hybrid desalination system performance using experimental facilities. The facilities and experiences of 425 m^3/d MSF pilot plant, 30 m^3/d Low Temperature Evaporation (LTE) pilot plants at Trombay are being used for the studies and prediction of performance of a hybrid system to be installed on the Kalpakkam demonstration plant. The LTE facility is connected to the experimental reactor CIRUS for using its low grade waste heat. The 100 m^3/d seawater reverse osmosis (SWRO) pilot plant with an energy recovery turbine provides useful data. It has a provision of testing various types of membrane elements at different temperatures to collect data for preheat RO. An ultra-filtration (UF) pretreatment system has been recently installed in this plant.

The *Korea Atomic Energy Research Institute (KAERI),* Republic of Korea, is setting a set of optimum coupling parameters between its newly designed small size co-generating reactor SMART and the desalination facility for the water production capacity of 40,000m^3/day and the electricity generation of about 90MW. SMART is an integral type advanced pressurized water reactor with a rated thermal capacity of 330MW. The research programme within this CRP is being conducted to define the best coupling option between SMART and desalination system and to establish a set of optimal coupling parameters and their interfacing conditions for the selected coupling option.

The Moroccan *Centre National de l'Energie des Sciences et des Technique Nucleaires (CNESTEN)* joined the CRP in 2001 with the objective of investigating the optimal coupling of nuclear reactors and desalination systems for Moroccan sites, taking into account the energy and water demand and with emphasis on a new generation SMR.

In the Russian Federation *Institute of Physics and Power Engineering (IPPE)* is taking an initiative in coordinating four institutes (IPPE, OKBM, RDIPE and JSC "Malaya Energetica") in a research programme to investigate effective utilization of Russian SMRs for a nuclear desalination complex. Nuclear reactors being evaluated as the energy source for the desalination plant are: (i) a barge-mounted co-generating reactor KLT-40C; (ii) an integrated small PWR NIKA-70; and (iii) a dedicated heating reactor RUTA. Various coupling configurations have been technically and economically evaluated. For economic evaluation the IAEA software DEEP was used with necessary modifications.

Construction of a pilot plant based on a floating power unit (FPU) with KLT-40C reactors is planned for 2005-2006. The co-generation plant will be sited at the shipyard in Severodvinsk, Arkhangelsk Region, in the western North Sea area, where the FPU is being manufactured. The project will provide full-scale demonstration of nuclear technology suitable also for nuclear desalination.

Centre National des Sciences et Technologies Nucleaires (CNSTN) of Tunisia is carrying out a feasibility study of utilizing nuclear desalination in Skhira in the south of Tunisia. The study involves the following tasks:

- Preliminary dimensioning of nuclear reactors and desalination plants.
- Optimization of coupling schemes for various combinations of nuclear reactor and desalination processes.
- Economic evaluation of the integrated nuclear desalination systems.
- Safety studies.

The Tunisian contribution to this CRP is focused on economic evaluation of the integrated nuclear desalination and fossil energy systems.

1.4. COLLABORATION AMONG PARTICIPATING INSTITUTES

In order to increase synergetic effects within the CRP, an attempt was made with the initiative of the chief research coordinator to improve the interacting activities, especially in sharing relevant information and resources in research activities.

Looking at a large potential for enhancing benefits through closer cooperation between the participating countries/institutes to achieve common objectives, divide tasks, share information and avoid duplication, all contracts under the CRP were divided into two major desalination technology groups, namely coupling with distillation processes and coupling with membrane processes. A number of consultants meetings with specific subjects were coordinated at times between Research Coordination Meetings (RCMs) for the enhanced synergy.

One of these meetings was a fruitful and successful Consultants Meeting around DESNU modelling tool, hosted by INVAP (Argentina), in which different users performed an external review and assessment to improve the spreadsheet along its development. Within the interaction/interchange frame amongst participants, KAERI (Korea) and BARC (India) effectively used this tool to carry out a systematic safety assessment, while IPPE (Russia) extended DESNU capabilities to produce a model of the BOP for RELAP code.

NPPA cooperated with CANDESAL to review the design report of El-Dabaa experimental facility. Several modifications were introduced to the report and were reflected in the final design and technical specifications of the facility.

REFERENCES TO CHAPTER 1

[1.1] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Power Reactors in the World Reference Data Series No. 2 (2004).

[1.2] BIRKETT, J.D., Proc., International Desalination Association World Congress on Desalination and Water Sciences, Abu Dhabi, 18–24 November 1995,2, IDA, (1995).

[1.3] WANGNIK CONSULTING/IDA, 2002 IDA Worldwide Desalting Plants Inventory, Report 16, Germany, July 2002.

[1.4] MEGAHED, M.M., “An Overview of Nuclear Desalination: History and Challenges”, International Journal of Nuclear Desalination, Vol. 1, No. 1, pp. 2–18, (2003).

[1.5] INTERNATIONAL ATOMIC ENERGY AGENCY, Desalination of Water using Conventional and Nuclear Energy, Technical Reports Series No. 24, IAEA, Vienna (1964).

[1.6] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Energy for Water Desalination Technical Report Series No. 51, IAEA, Vienna (1966).
[1.7] INTERNATIONAL ATOMIC ENERGY AGENCY, Costing Methods for Nuclear Desalination Technical Reports Series No. 69, IAEA, Vienna (1966).
[1.8] INTERNATIONAL ATOMIC ENERGY AGENCY, Guide to the Costing of Water from Nuclear Desalination Plants, Technical Reports Series No. 80, IAEA, Vienna (1967).
[1.9] INTERNATIONAL ATOMIC ENERGY AGENCY, Guide to the Costing of Water from Nuclear Desalination Plants, Technical Reports Series No. 151, Vienna (1973).
[1.10] Nuclear Desalination, Proc. Symposium on Nuclear Desalination, International Atomic Energy Agency, 18–22 Nov. 1968, Madrid. Suppl. to Desalination, Elsevier, (1969).
[1.11] INTERNATIONAL ATOMIC ENERGY AGENCY, Use of Nuclear Reactors for Seawater Desalination, IAEA-TECDOC-574, Vienna (1990).
[1.12] INTERNATIONAL ATOMIC ENERGY AGENCY, Technical and Economic Evaluation of Potable Water Production through Desalination of Seawater by Using Nuclear Energy and Other Means, IAEA-TECDOC-666, Vienna (1992).
[1.13] INTERNATIONAL ATOMIC ENERGY AGENCY, Potential for Nuclear Desalination as a Possible Source of Low Cost Potable Water in North Africa, IAEA-TECDOC-917, Vienna (1996).
[1.14] INTERNATIONAL ATOMIC ENERGY AGENCY, Options Identification Programme for Demonstration of Nuclear Desalination, IAEA-TECDOC-898, Vienna (1996).
[1.15] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Desalination of Sea Water (Proceeding Series STI/PUB/1025), Vienna (1997).

CHAPTER 2. NUCLEAR DESALINATION TECHNOLOGIES

Both nuclear and desalination technologies are mature and proven by experience, and are commercially available from a variety of suppliers. From the early days of the two technologies, it was realized that nuclear energy could be utilized to overcome two of the challenges to the development of mankind, namely sustainable supply of electricity and water. The adaptation of nuclear energy to desalination has two parts, the selection of the reactor type and the implementation of the nuclear fuel cycle evaluated from the availability of uranium to the disposal of radioactive waste.

This chapter presents information on various types of nuclear reactor and desalination technologies considered within this Coordinated Research Project (CRP) by the participating institutions.

2.1. NUCLEAR REACTORS

Many nuclear reactor concepts have been conceived through the various stages of development of nuclear reactor technology [2.1]. All nuclear reactor types can provide the energy required by the various desalination processes. The amount of energy (electricity and/or heat) needed for desalination can be readily supplied by tapping the low-grade steam and/or electricity produced by the nuclear plant. In this regard, it has been shown in previous studies [2.2-2.6] that Small and Medium Reactors (SMRs) offer a big potential as coupling options for nuclear desalination systems. Nuclear reactors examined for optimal coupling with desalination systems within this CRP are shown in Figure 2.1.

2.1.1. Power reactors

2.1.1.1. CANDU-6 (Canada)

Basic objectives

The basic design objectives of CANDU-6, when developed in the early I 970’s, was to create a safe, reliable, economic and robust nuclear power plant with a net electrical output in the range of 600 MW. The performance of the first four CANDU-6 plants, located in two Canadian provinces (Quebec and New Brunswick) and two foreign countries (Argentina and the Republic of Korea) and which have a total of over forty years of safe and economic reactor operation at average capacity factor of over 84%, confirms that the design objectives were met. Evolutionary improvement have been incorporated in the CANDU-6 since the first units entered service, with the basic objectives of further enhancing safety, reliability, and overall economics. Evolutionary improvements continue to be incorporated in the CANDU-6 design, taking advantage of CANDU operating experience, AECL research and development, and technical advances world-wide, in order to further enhance safety, reliability and economics.

Basic design features

The CANDU-6 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

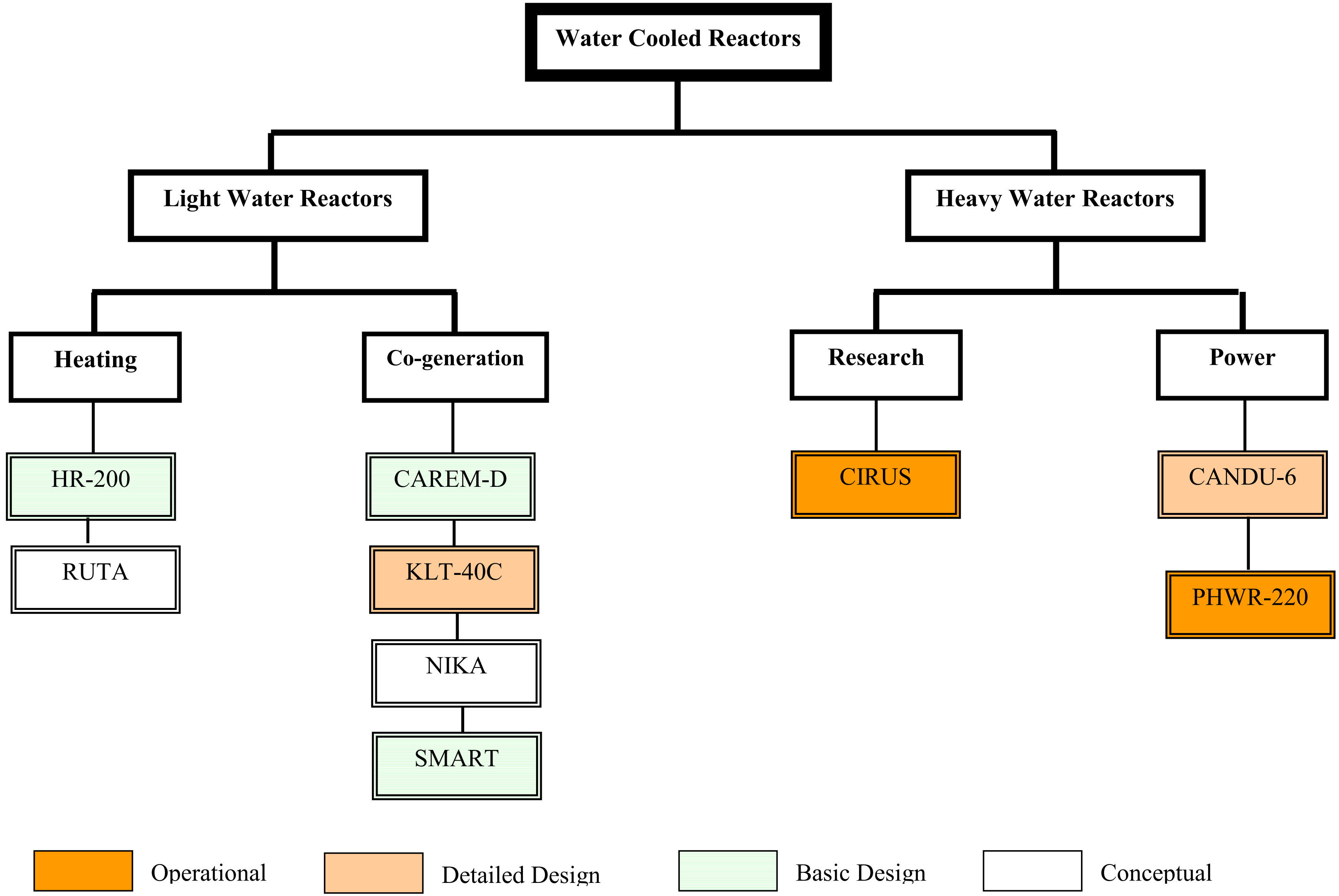


Figure 2.1: Nuclear Reactors Examined for Optimal Coupling with Desalination Systems.

- The standard CANDU 37-element CANDU fuel bundle, and the ability to operate on natural uranium or other low fissile content fuel.
- On-power refueling, to eliminate the need for refueling 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.

Design description

Nuclear steam supply system

The CANDU-6 Nuclear Steam Supply System is illustrated in Figure 2.2, and key design data are listed in Table 2.1. A brief description of CANDU-6 nuclear steam plant systems is provided below. Further details are provided in Reference [2.7].

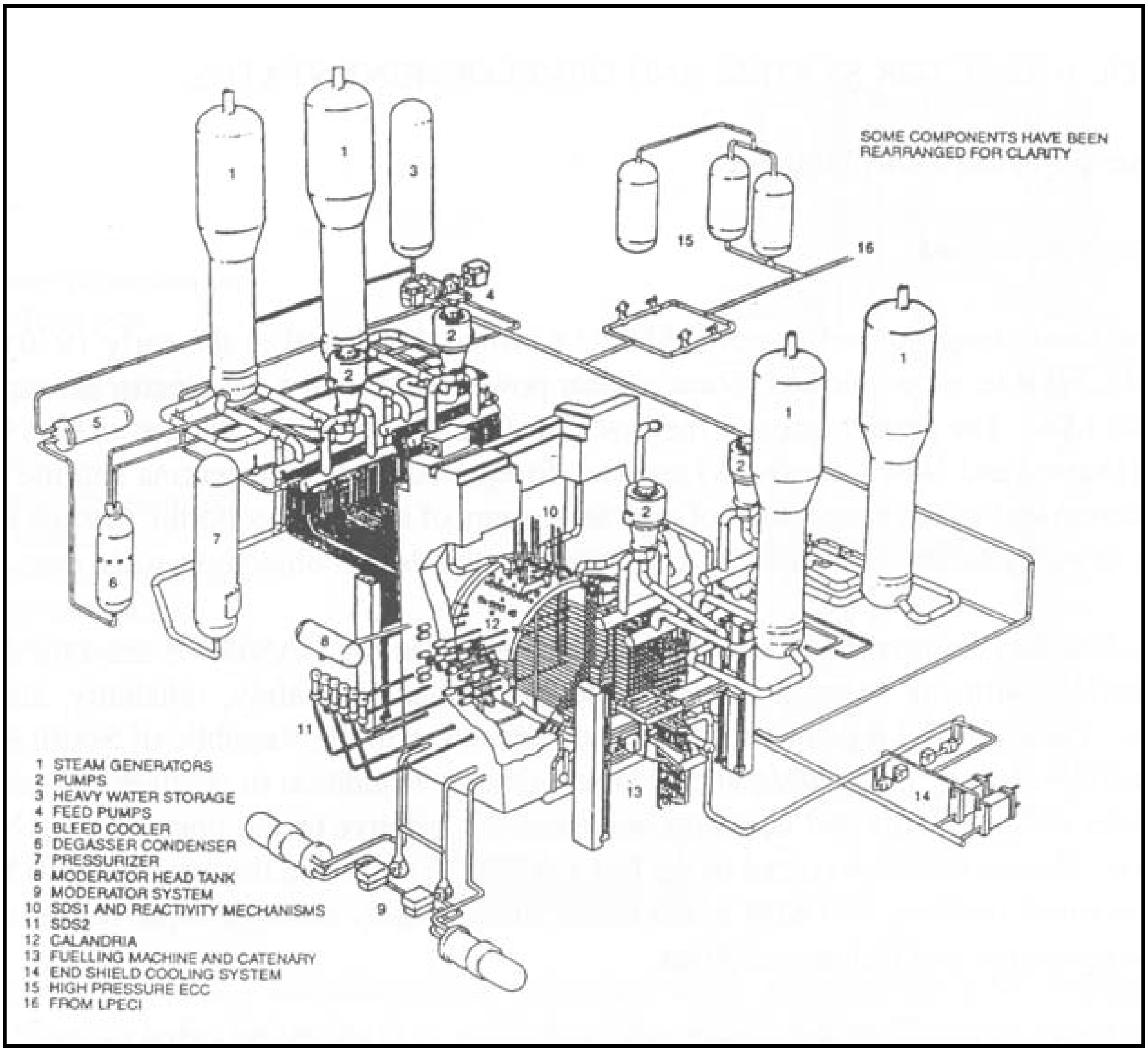


Figure 2.2: CANDU-6 Nuclear Steam Supply System.

TABLE 2.1: CANDU-6 KEY DESIGN DATA [2.7]

General Information	- Reactor name/type - Thermal power (MWt) - Design life time (year)	CANDU-6/PHWR 2158 40
Fuel and Reactor Core	- Fuel type - Active core horizontal length (m) - No. of fuel assembly - Core power density (kW/L) - Type of refueling - Refueling frequency (months)	Natural $U0_2$ 5.94 4560 82.5 On-power 10 month gradual changeover of core with average refueling rate of 15 fuel bundles/full power day.
Reactivity Control	- No. of control bank elements - Control banks/material - Burnable poison material	39 for reactor regulation Cadmium, SS None in fuel or HTS
Reactor Pressure Vessel	- Overall length (m) - Inner diameter of Calandria (m) - Average Calandria thickness (mm) - Vessel material	7.77 7.59 28.6 Calandria: 304 L SS Press. tube: Zr-2.5 wt% Nb
Reactor Coolant System	- Design pressure (MPa) - Operating pressure (MPa) - Core inlet temperature (^{0}C) - Core outlet temperature (^{0}C)	11.5 10 266 310
Secondary System	- Feedwater pressure (MPa) - Feedwater temperature (^{0}C) - Steam pressure (MPa) - Steam temperature (^{0}C)	4.9 187 4.7 260

Fuel
Current CANDU-6 plants are operated with natural uranium fuel (0.7% U-235); they can, however, operate on a variety of other low fissile content fuels including slightly enriched uranium, and recovered uranium from PWR fuel reprocessing plants. The use of natural uranium or other low fissile content fuel in CANDU-6 assures that there is no potential for the fuel (new or irradiated) to achieve criticality outside of the reactor regardless of the storage configuration.

Fuel channels
The CANDU-6 reactor includes 380 horizontal fuel channels. Each fuel channel consists of a zirconium-niobium alloy pressure tube. The pressure tubes, the only components in CANDU that are subject to high levels of radiation and stress, are readily replaced. However, pressure tube replacement is not required in new CANDU-6 plants for 35 or more years; the design of the fuel channels and the layout within the reactor building facilitate fuel channel replacement.

Calandria assembly
The reactor core is contained within a cylindrical low-pressure tank called the calandria. The calandria contains the heavy water moderator at low temperature and at near-atmospheric

pressure. The calandria is positioned within a low-pressure steel lined concrete vault, filled with ordinary water. This tank provides biological shielding from neutron and gamma radiation from the reactor core. The calandria is penetrated vertically and horizontally by flux measurement and reactivity control devices, and by the in-core components of two safety shutdown systems. All reactivity control devices function in the low-pressure moderator. No reactivity control devices penetrate the heat transport system and no chemicals are added to the heat transport system for reactivity control.

Heat transport system
The heat transport system is subdivided into two independent circuits or loops, each including half (190) of the 380 fuel channels. Each fuel channel is connected by individual inlet and outlet feeder pipes to distribution pipes, called headers, at both ends of the reactor. Four heat transport pumps, two in each loop, circulate the heavy water coolant from the inlet header, through the fuel channels, to the outlet header, through the steam generators (where the heat is transferred to ordinary water to generate steam), and back through the reactor again; the coolant flow in adjacent fuel channels is therefore in opposite directions (bi-directional). If all pumps are lost, fuel cooling is maintained by thermosyphoning flow.

Moderator system
Heat is deposited in the heavy water moderator contained within the calandria during normal operation, principally from direct gamma and neutron interaction. This heat is removed by the moderator system, which circulates and cools the heavy water in an external circuit connected to the calandria; the heat is rejected to the recirculated coolant water system.

Fuel handling
On-power refueling is performed by two fuelling machines, located at opposite ends of the reactor. These machines transport new fuel bundles to the reactor fuel channel to be refueled, and load them into the fuel channel while the reactor is operating, and simultaneously removing used fuel bundles from the fuel channel. In the event that a defect occurs in a fuel bundle during reactor operation, the fuelling machines can be used to remove the defective fuel, thereby limiting the release of fission products to the heat transport system coolant.

Reactor control
On-power refueling provides the principal means for controlling reactivity in CANDU-6. Additional reactivity control, independent of the safety shutdown systems, is achieved through use of reactivity control mechanisms. These include light-water zone compartments, absorber rods, and adjuster rods; all are located between fuel channels within the low-pressure heavy water moderator and do not penetrate the heat transport system pressure boundary. The reactor is controlled by the dual redundant computer control system.

Safety systems
The CANDU-6 incorporates four special safety systems, which consist of the two passive, diverse, dedicated reactor shutdown systems; the emergency core cooling system; and the containment system. Each is separated from and is independent of the normally operating plant systems, and of all the 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.

Balance of plant

Turbine-generator
The CANDU-6 turbine consists of a double-flow high pressure cylinder and three double flow low pressure cylinders that exhaust to individual condensers. The turbine-generator operates at 1800 rpm. CANDU plants have operated with turbine-generators supplied by a variety of manufacturers. In the event of a loss of line, the turbine-generator on CANDU-6 runs back to sustain station loads. In addition, a turbine steam bypass system is provided (100% short-term, 70% long-term capacity) to reject steam directly to the condensers in the event that the turbine is unavailable; this allows reactor power to be sustained during the event.

Radioactive waste management
The CANDU-6 incorporates comprehensive systems for the management, disposal and storage of solid, liquid, and gaseous radioactive wastes. The irradiated fuel bays, located adjacent to the reactor building have sufficient capacity to accommodate the fuel from ten years of reactor operation; facilities are included for the transfer of irradiated fuel to dry storage. There is no potential for new or irradiated CANDU fuel to achieve criticality in air or ordinary water, regardless of the storage configuration.

Information and control system
Plant instrumentation, computer control system, control room man/machine interface and the plant information system are provided by ICS 90+ System (Information and Control System 90+). ICS 90+ evolved from the highly automated CANDU control systems incorporated in the operating CANDU-6 and Darlington stations, and take advantage of the rapid developments in digital and communication systems that have occurred in recent years. The result is significant improvements in safety and reduced operating cost.

Safety features

Safety is assured in CANDU-6 through a defense 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 design also contributed to CANDU-6 safety. CANDU design practice places emphasis on the performance of the special safety systems (shutdown systems, emergency core cooling system and containment system).

This overall safety approach is achievable because there are at least two ways of providing the safety functions of shutdown and decay heat removal. Firstly, there are two independent safety shutdown systems, each equally effective in handling accidents. Secondly, the heavy water moderator provides an independent source of cooling water surrounding the pressure tubes. Moreover, every postulated accident that might release radioactivity must be shown to have acceptable consequences even if any one of the special safety systems fail to function.

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.

Passive safety

CANDU-6 incorporates a number of passive safety features. These include:
- The two independent, passive safety shutdown systems.
- A gravity feed supply of water to the steam generators (from the dousing tank), available if normal and backup engineered supplies are lost.
- The ability to cool the fuel even if all coolant (normal and emergency supply) are lost, by the rejection of heat from the fuel to the cool, low pressure moderator.

Engineered safety

A range of engineered systems contribute to CANDU-6 safety. These include:
- The shutdown cooling system, consisting of two independent circuits that, utilizing pumps and heat exchangers, can remove decay heat should the steam generator sink be lost. This system can be brought into service at full heat transport system temperature and pressure.
- The connection of all pipes that can potentially discharge reactor coolant from the heat transport system (HTS) to a high pressure tank (the degasser condenser). The relief valves on this tank are set above HTS operating pressure. Hence, failure of any valve in the above lines in the open position does not lead to a loss of reactor coolant.
- The provision of an emergency water supply to the steam generators to backup the normal (high capacity electric main feed water pumps, and electric and steam turbines auxiliary feed water pumps). A passive water supply is also provided.
- The provision of multiple systems and supplies throughout the plant to ensure reliability. For example, four on-site diesel electric generators, two to back up the normal power supplies and two for emergency power.

With respect to the redundant heat removal paths, the moderator can act as an emergency heat sink even with no water in the fuel channels. Should the moderator heat removal system subsequently fail, the large water-filled reactor vault surrounding the calandria vessel provides an additional line of defense. Its primary purpose is to provide shielding of the concrete reactor vault from neutrons and gamma rays. However it can also act as a passive emergency water reservoir in case of a severe core damage accident; that is, should the primary coolant system (HTS), the emergency core cooling system, and the moderator heat removal system all fail, the water-filled vault will retain the debris inside the calandria, by keeping the outside of the calandria shell cool for a minimum period of 24 hours. This allows time for fission products to decay further and for decay heat to reduce, and for emergency planning.

2.1.1.2. PHWR-220 (India)

Basic objectives

Pressurized Heavy Water Reactors (PHWRs) form the main stay of the Indian nuclear power program. PHWRs are being built in the country to meet the increasing demand for electricity. The PHWRs also form the first stage of nuclear power plant types to utilize the existing uranium reserves in such manner as to lead to the full exploitation of the vast thorium resources available in the country. Indian PHWRs are built in two sizes, namely 220 MW

(PHWR-220) and 550 MW (PHWR-550). However, because the Indian research project within this CRP relates to coupling a hybrid desalination plant to the existing PHWR-220 power plant in Kalpakkam, focus is confined to PHWR-220.

The first two units of PHWR-220 were constructed in Rajasthan as a collaborative work with Atomic Energy of Canada Ltd. (AECL). Work on the two units in Kalpakkam was taken up in 1967 as a totally indigenous effort including engineering, procurement, construction, commissioning and operation. Current designs are largely standardized.

Basic design features

The Indian PHWR is a pressure tube type reactor using heavy water moderator, heavy water coolant (in a separate high-pressure high temperature system) and natural uranium oxide fuel. The unique design concepts of PHWR systems offer certain intrinsic advantages with respect to safety, both during normal operations and accident conditions.

Some of these inherent features are as follows:
1. Relatively large neutron generation time (about 0.001 second);
2. On-power fuelling and hence minimum requirement for reactivity reserve;
3. Reactor core and steam generator configuration is such as to promote and establish thermosyphon mode of cooling the core;
4. Availability of large body of cool heavy water around pressure tubes in the core;
5. Availability of large body of cool light water around the calandria in its vault.

Various safety-related systems and safety support systems are designed to appropriate safety class specifications depending on the functional importance.

Design description

Nuclear steam supply system

The NSSS comprises of various nuclear systems and auxiliaries located in Reactor Building (RB), Reactor Auxiliary Building (RAB), Station Auxiliary Building (SAB), Service Building (SB) and Control Building (CB). Instrumentation and Control for reactor regulation process control and reactor protection are also housed in the nuclear island. Key design data of PHWR-220 are listed in Table 2.2. Typical cross-section of the Reactor Building is shown in Figure 2.3.

Pressurised Heavy Water Reactors (PHWRS) are horizontal pressure tube reactors using natural uranium oxide fuel in the form of 495 mm clusters. The fuel is cooled by a high-pressure, high temperature circulating heavy water system called the primary heat transport (PHT) system. The PHT system of PHWR-220 is a single “figure of eight” loop with two coolant pumps and steam generators at either end of the reactor. Heavy water is also used as moderator in a separate low temperature, low pressure moderator system. Refueling of the reactor is carried out “on-power” by the fuel handling system. The heat from the reactor is carried away by the heavy water coolant in the PHT system and is given away to the secondary side in the steam generators (SG). The steam from SGs is fed to the turbine-generator in the conventional island for production of electricity.

TABLE 2.2: PHWR-220 KEY DESIGN DATA [2.7]

General Information	- Reactor name/type	220 MWe PHWR
	- Thermal power (MWt)	743
	- Design life time (year)	40
Fuel and Reactor Core	- Fuel type	Natural $U0_2$ Pellets
	- Active core horizontal length (m)	5.005
	- No. of fuel assembly	3672
	- Core power density (kW/L)	9.38
	- Type of refueling	On-power
	- Refueling frequency (months)	-
Reactivity Control	- No. of control bank elements	14
	- Control banks/material	Cadmium
	- Burnable poison material	None in fuel or HTS
Reactor Pressure Vessel	- Overall length (m)	7.77
	- Inner diameter of Calandria (m)	4.51
	- Average Calandria thickness (mm)	-
	- Vessel material	Calandria: 304 L SS Press. Tube: Zr-2.5 wt% Nb
Reactor Coolant System	- Design pressure (MPa)	11.5
	- Operating pressure (MPa)	10.3
	- Core inlet temperature (^{0}C)	249
	- Core outlet temperature (^{0}C)	293.4
Secondary System	- Feedwater pressure (MPa)	4.2
	- Feedwater temperature (^{0}C)	170
	- Steam pressure (MPa)	4.06
	- Steam temperature (^{0}C)	250.7

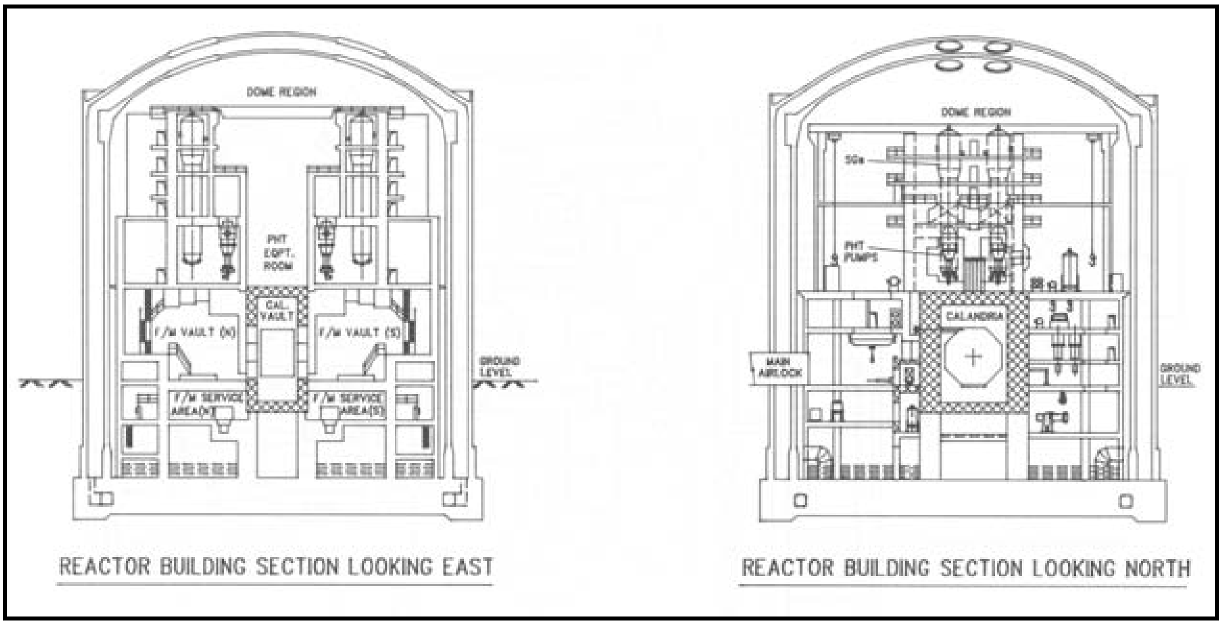


Figure 2.3: Typical Cross Sections of PHWR-220 Reactor Building.

Reactor process system

The PHT system constitutes part of the nuclear steam supply system (NSSS) and is briefly described below. The other reactor process systems are:

- Moderator system in which heat is dissipated due to slowing down of fast neutrons and absorption of gamma rays;
- Reactor shield systems (end shield and calandria vault water) in which heat is dissipated due to attenuation of radiation escaping from the reactor core;
- Closed loop process water systems employing de-mineralized water as a secondary coolant for the above reactor process system heat exchangers. This process water in the closed loop is in turn cooled by service water in a tertiary circuit.

The PHT system circulates high-pressure coolant through the fuel channels to remove the heat generated in fuel. The major components of this system are the reactor fuel channels, feeders, four reactor inlet headers, four reactor outlet headers, four pumps and interconnecting pipes and valves. The headers, steam generators and pumps are located above the reactor and are arranged in two symmetrical banks at both ends of the reactor. The headers are connected to fuel channels through individual feeder pipes. The PHT system is divided into two independent loops, connected to a common pressurizer. Each loop, with its associated equipment, circulates water through its respective half of the reactor core.

The coolant circulation is maintained at all times during reactor operation, shutdown and maintenance. The PHT pumps are provided with flywheels to provide better flow coast down after pump trip. The system layout as discussed above assures adequate flow for decay heat removal from reactor during shutdown by thermosyphoning action. A separate shutdown cooling system is provided to remove reactor decay heat during cold shutdown conditions. This mode of cooling permits the draining of the steam generators and pumps in the PHT system for maintenance. An emergency core cooling system provides adequate coolant flow to prevent overheating of the fuel in the unlikely event of loss of coolant accident.

PHT system pressure control is achieved through feed and bleed and a pressurizer has been added to improve performance during operational transients. System components are protected from over pressure by instrumented relief valves and suitable regulating system and protective system action. Potential heavy water leak sources are kept to a minimum by using welded construction wherever practicable, and bellow sealed valves. Heavy water leakage collection and recovery systems are connected to the locations where potential leak sources exist.

On-power fuelling

On-power fuelling is a feature of all PHWRs which have very low excess reactivity. In this type of reactor, refueling to compensate for fuel depletion and for overall flux shaping to give optimum power distribution, is carried out with the help of two fuelling machines, which work in unison on the opposite ends of a channel. One of the machines is used to fuel the channel while the other one accepts the spent bundles. In addition, the fuelling machines facilitate on-power removal of failed fuel bundles.

Each fuelling machine is mounted on a bridge and column assembly. Various mechanisms provided allow two-directional movement of fuelling machine head and make it possible to align it accurately with respect to channels. Various features have been provided which enable clamping of fuelling machine head to the end fitting, opening and closing of the respective seal plugs and perform various fuelling operations.

Balance of plant

Power generation system
Turbine building generally houses the power generation system, electrical system and common services system. Turbine is a tandem compound machine directly coupled to an electrical generator. The turbine consists of a high pressure, double flow cylinder with external moisture separators and steam re-heaters and two double flow low pressure cylinders. Turbine is provided with necessary supervisory and protection instrumentation and devices. All auxiliary systems are located in the Turbine Building.

Electrical generator directly coupled to the turbine produces electricity and the voltage is stepped up by the generator transformer which in turn is connected to a switch yard. Generated power is thus transmitted to the electrical power grid.

Auxiliary feed water system
Two auxiliary feed pumps (one of them is standby) located in the annex to reactor auxiliary building take suction from condensate storage tanks and supply feed water to the steam generators if the main feed water system is unavailable. The system is totally independent from the main feed water system and is capable of supplying feed water at the rated pressure to the steam generator for decay heat removal. The auxiliary feed water system is designed to be operable under safe shutdown earthquake condition.

Electrical power system
As explained above, power is supplied to the grid from the power station. Station service power is obtained from station startup transformer (SUT) which is connected to the grid. Similarly generator through unit auxiliary transformers (UT) provides supply to auxiliary power supply buses. Four classes of power are used to supply station requirements.

The station electrical power distribution system has been divided into two physically separate divisions to minimize common cause failures. Division-1 and Division-2 supply the systems of the plant dedicated to normal power production, and safety-related loads. These loads are appropriately shared between the two Divisions of power supply. Supplies and distribution systems as well as connections between the two Divisions, are seismically qualified to be operational after an earthquake.

Waste disposal and environmental releases
Facilities are provided at the station for the safe disposal of all radioactive wastes. While all solid wastes are stored at site, releases of liquid and gaseous effluents are so organized that the prescribed dose limits for members of the public are strictly adhered to.

All airborne and gaseous activity from the plant is routed to the atmosphere via a tall stack along with ventilation exhaust air from the Reactor and Service Buildings. Before entry into the stack, the air is passed through high efficiency particulate air filters for removal of most of the particulate radioactivity.

Radioactive effluents released from the stack are continuously monitored for inert gas activity, particulate activity, iodine and tritium, to ensure that releases are within stipulated limits.

The liquid wastes generated at the station are segregated into four categories on the basis of activity level and chemical characteristics of the waste. While no treatment is required for the least active category the other categories are sent to the liquid waste management facility where they are given appropriate treatment. After treatment, the wastes are diluted to achieve the final stipulated discharge concentration limit.

Solid wastes from the station are generally grouped into three categories, based on the level of activity. A solid waste management facility within the plant perimeter provides space for storage of solid wastes of all categories.

Instrumentation, control and electrical systems

The instrumentation and control systems in the PHWR plant include a variety of equipment intended to perform display, monitoring, control, protection and safety functions.

Reactor regulating system (RRS)

Reactor control in PHWR-220 is achieved with the help of control rods, shim rods and adjuster rods. Flux-tilt control is achieved by differential movement of these reactivity controllers. The main functions of RRS are:

1. Automatic control of reactor power between 10 and 100% full power. This covers raising or lowering of reactor power at a desired rate and maintaining it at any desired level within this range (reactor regulation).

2. Maintain the neutron flux profile in the reactor close to its design shape so as to enable operation at the minimum possible power without exceeding the limits on fuel bundle power (flux tilt control).

3. Monitor specified plant parameters and reduce the reactor power, at a predetermined rate up to predetermined low limits, whenever these parameters cross their respective preset limits (reactor setback).

4. Provide limited xenon override capability.

5. Step reduction in reactor power (step back).

Reactor protection system (RPS)

In the PHWR-220 there are two systems consisting of 14 mechanical shut-off rods which fall freely under gravity into the core, and 12 liquid poison stand-pipes into which sodium-penta borate solution is introduced, which provide fast acting redundant and diverse shut-down systems. These are reinforced by slow-acting automatic liquid poison addition into the moderator system for prolonged shutdown

Safety features

The Concept of defense-in-depth is applied in designing safety systems to achieve functional diversity, i.e. by providing diversely functioning systems that can perform same safety function (e.g. two shutdown systems), multi barriers to prevent release of radioactivity, multi-defense system, using physical separation of systems and components which serve as back-up (in safety functions) to each other, and procuring components for different systems from different suppliers, to the extent possible. Such an approach leads to a design of safety systems which will be tolerant to a wide range of human errors and equipment failures.

There are six barriers in the transmission path of radioactive fission products from the fuel to the environment, namely:

1) *Fuel* — diffusion resistant ceramic fuel virtually retains all solid fission products even at the operating temperatures.
2) *Fuel sheath* — sealed to vacuum technology standards.
3) *Primary heat transport system* - designed and maintained to achieve very low leakage.
4) *Containmen*t — double walled and designed to maintain very low leakage under every accident conditions.
5) *Exclusion zone* — provides atmospheric dilution of any fission to product release. An area encompassed within a radius of 1.6 km from the plant constitutes the exclusion zone.
6) *Green belt* — consisting of rows of trees around the plant (no credit taken for this in safety analyses).

The concept of double containment has been adopted for Indian PHWRS. The containment structure consists of a cylindrical pre-stressed cement concrete (PCC) primary containment with PCC dome and a secondary containment of reinforced cement concrete (RCC) structure' completely surrounding the primary containment. Primary containment is designed to withstand the overpressure and higher temperature caused by release of hot fluid from the primary heat transport system under LOCA condition. The secondary containment is basically an additional concrete envelope over the primary containment. This reduces the ground level release of radioactivity by way of holdup and dilution of leakage from the inner primary containment, by a purge and maintaining negative pressure.

The Emergency Core Cooling System (ECCS) is designed to provide enough coolant to the PHT system and to transport heat from the core to the ultimate heat sink in such a way as to ensure adequate core cooling during all phases of LOCA. In the case of small breaks in the PHT system (small LOCA), the ECCS system serves to recover the spilled water and transfer it back to the PHT system after cooling it to ambient temperature. A separate heavy water inventory is also provided for making up losses from the PHT system. In this situation where PHT system inventory of heavy water is maintained, the core cooling function is carried out by the shutdown cooling system.

2.1.2. Co-generation reactors

CAREM-25 is a CNEA (Comisión Nacional de Energía Atómica) project, which is jointly developed by INVAP SE. INVAP activities cover a broad field of technological developments, among which the peaceful use of nuclear energy is paramount.

2.1.2.1. CAREM-D (Argentina)

Basic objectives

The project consists of development, design and construction of a prototype small nuclear power plant (100 MWth, 27 MWe). A design alternative called CAREM-D has been developed for the co-generation of electricity and potable water (modules of 10000 m^3/day).

Basic design features

CAREM is a project for an advanced, simple and small nuclear power plant, conceived with new generation design solutions and standing on the large world wide experience accumulated in the safe operation of Light Water Reactors, especially adequate for nuclear desalination of seawater. It is an indirect cycle Plant with some distinctive and characteristic features that greatly simplify the reactor and also contribute to a higher level of safety. These distinctive features are:

Integrated primary cooling system.
- Primary cooling by natural circulation.
- Self-pressurized.
- Safety systems relying on passive features.
- Coupling system minimizing the risk of contamination.

Design description

Nuclear steam supply system

The primary system is integrated, that means the whole high energy primary system — core, steam generators, absorber rod drive mechanisms and pressurising system — is contained inside a single pressure vessel, as shown in Figure 2.4.

The cooling of the reactor primary system is achieved by natural circulation, producing a flow rate in the core that allows for sufficient thermal margin to critical phenomena. The coolant acts also as moderator.

Self-pressurisation of the primary system in the steam dome is the result of the liquid-vapour equilibrium, i.e. the core outlet bulk temperature corresponds to saturation at primary pressure. Heaters and sprinkles typical of conventional PWRs are thus eliminated.

The main features considered for the fuel design were the use of enriched uranium, the use of burnable poisons, a higher final burn-up (compared to PHWRs), absorbing elements of cluster type and the required low pressure-loss of the assembly.

The core has 61 Fuel Assemblies (FA) of hexagonal cross section. Its components are typical of the PWR fuel assemblies. The fuel is UO_2 enriched at 1.8 and 3.4%. An 8% weight of Gd_2O_3 is used as burnable poison in specific fuel rods. Temperature and density feedback coefficients are strongly negative.

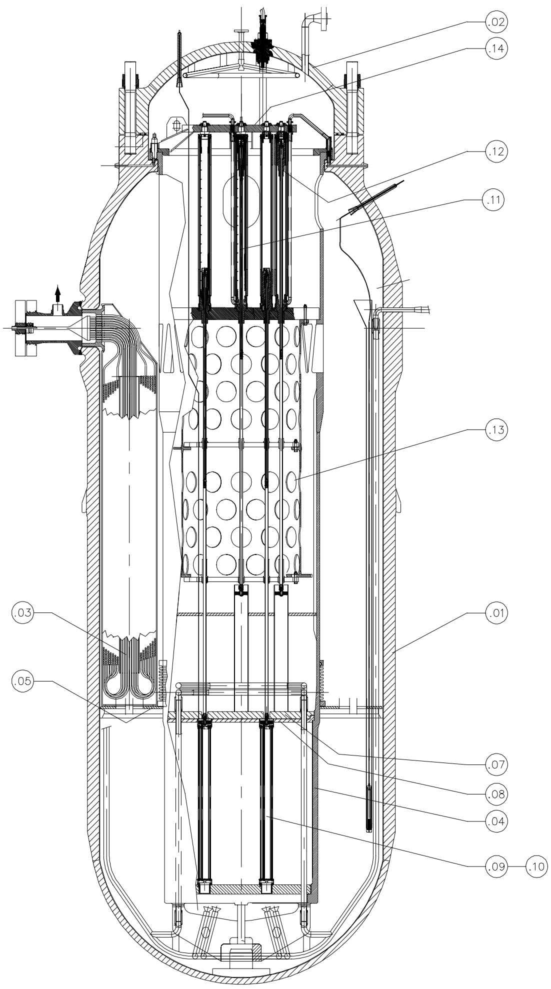
Absorbing Elements (AE) are used for reactivity control during normal operation (Adjust and Control System), and to produce a sudden reactor shutdown (Fast Extinction System). Boron injection is not used for reactivity control during normal operation.

The steam generators are of a ‘Mini Helical’ vertical, ‘once through’ design. The counter-current flow has the secondary coolant flowing upwards inside the tubes, which exits the steam generator with ample superheating.

Technical and economical advantages are obtained with the CAREM design compared to the traditional design:

- No large Loss of Coolant Accident (LOCA) has to be handled by the safety systems due to the absence of large diameter piping associated to the primary system.
- Large coolant inventory in the primary results in large thermal inertia and long response time in case of transients or severe accidents.
- Shielding requirements are reduced by the elimination of gamma sources of dispersed primary piping and parts.
- The large water volume between the core and the wall leads to a lower fast neutron dose over the reactor pressure vessel wall.
- Eliminating primary pumps and pressurizer results in lower costs, added safety, and advantages for maintenance and availability.

The natural circulation of the coolant produces a self-correcting response in the flow rate under different power transients. Self-pressurising of the Reactor Pressure Vessel (steam dome) keeps the pressure very close to the saturation pressure. At all the operating conditions this has proved to be sufficient to guarantee a remarkable stability of the RPV pressure response.



01 Reactor Pressure Vessel

03 Steam Generator

08 Core

09 Absorbing Element

10 Fuel Assemblies

11 Control and adjustment CRD

12 Fast Extintion System CRD

Figure 2.4: CAREM Reactor Pressure Vessel.

The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in case of power ramps. The negative reactivity feedback coefficients and the large water inventory of the primary circuit, combined with the self-pressurisation features, make this behaviour possible with minimum control rod motion.

The reactor is expected to have an excellent behaviour under operational transients. The main design data of the CAREM-D reactor is shown in Table 2.3.

Balance of plant (BOP)

The CAREM Plant has a standard steam cycle of simple design. The steam generators are drum-less “once-trough” boilers without accumulators by which steam is superheated in all plant condition. No blow-down is needed in the steam generators, reducing the waste generation.

The twelve steam generators are connected alternately in two groups to annular collectors. Each branch has its own relief and isolation valves. A single turbine is used, and the exhaust steam at low pressure is condensed in a water-cooled surface condenser. The condensate is pumped and sent to the full stream polishing system in order to maintain water conditions.

High purity water leaving the polishing system is sent to the low-pressure pre-heater using turbine extraction as a heating media. The warm water is delivered to the water accumulator in order to perform degassing operations with temperature control using extraction steam.

TABLE 2.3: CAREM-D DATA SHEET

Reactor Type	Integrated PWR	Max. Electrical Power Output	27 MWe
Desalination technology	Pre-heated RO	Max. Water Output per Module	10,000 m^3/day
Gross Thermal Power	100 MWth	Max. # of Pre-heated Modules	3
Fuel Initial Enrichment	3.4 %	Control Rod neutron absorber	Ag-In-Cd
Refuel Cycle	390 full power days	Additional Shut-down system	Boron Injection
Clad Material	Zircaloy-4	Burnable poison	Gd_2O_3-UO_2
Primary cooling mode	Natural Circulation	Operating Coolant Pressure	12.25 MPa
Coolant Inventory	39 m^3.	Core Intake/Outfall Temperature	284 / 326 ^oC
Primary coolant mass flow	410 kg/sec		
# of **Steam Generators**	12	Feed Water Pressure	4.7 MPa
Type	Once through	Feed Water Temperature	200 ^oC
Configuration	Mini helical	Steam Pressure	4.7 MPa
Tubes Material	Inconel 690	Min. Steam Temperature	290 ^oC
Shell Material	SS-304 L		
Average SW Temperature	15 ^oC	Type of membrane	Spiral wound
Average Salinity of SW	38,000 ppm	Pre-heating temperature	44 ^oC
Module capacity	10,000 m^3/day	Energy recovery system	Pelton turbines
Trains per module	5	Output water standards	W H O

Water is then pumped to the high-pressure pre-heaters (two in tandem using extraction steam) and sent to the steam generators as a feed-water, closing the circuit.

For the co-generation option, there are few small changes in the CAREM plant BOP design. The main change is on the condenser design: in order to optimize the thermal coupling, the cooling-water outlet temperature is taken to 44ºC, while the turbine back-pressure is taken to 0.124 bar. In a first approach, this change assures the pre-heated intake for two 10000-m^3/day RO modules, with a direct viable extension for a third module. Besides, the BOP for co-generation has additional piping for the seawater cooling flow by-passing the condenser. It allows the seawater intake to be used by the seawater reverse osmosis (SWRO) plant when the condenser is under maintenance or reparation.

Safety features

The main criteria used in the design of Safety Systems were simplicity, reliability, redundancy and passivity. Special emphasis has been put on minimizing the dependence on active components and operators' actions.

To extinct the nuclear chain reaction and to keep sub-criticality, CAREM NPP has two safety systems, as shown in Figure 2.5.

First shutdown system: extinguishes the nuclear reaction and maintains the core sub-critical during all shutdown states, by dropping 25 neutron-absorbing elements into the core.

Second shutdown system: It is a gravity driven injection system of borated water at high pressure that actuates automatically when the Reactor Protection System detects the failure of the First Shutdown System or in case of LOCA.

Residual heat removal system: it is designed to reduce the pressure on the primary system and to remove the decay heat in case of loss of heat sink. It is a simple and reliable system that operates condensing steam from the primary system in emergency condensers.

Emergency injection system: it prevents the core exposure in case of LOCA. The system injects borated water flooding the RPV (Reactor Pressure Vessel).

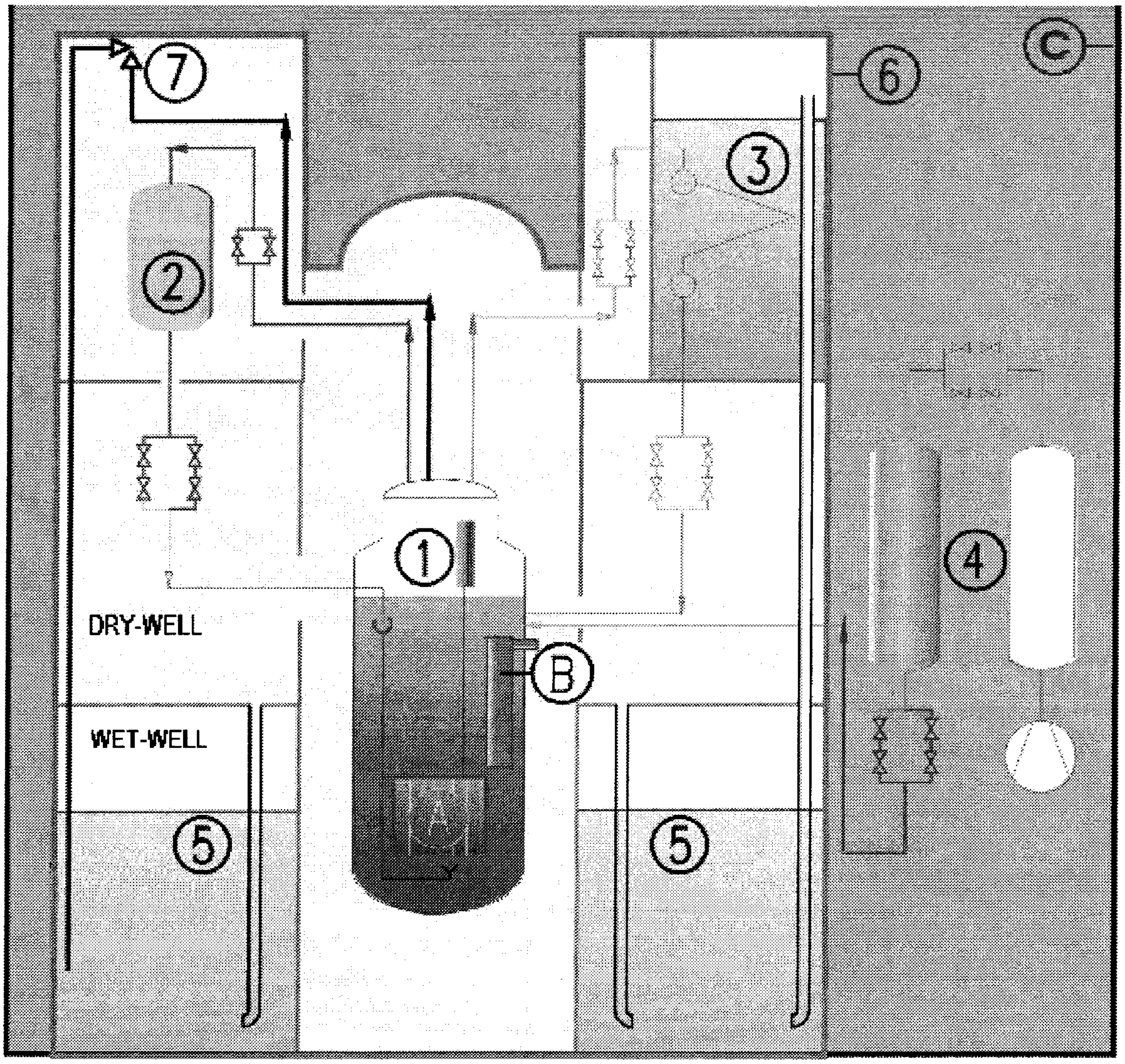
Containment system - where the primary system, the reactor coolant pressure boundary and important ancillary systems are enclosed - is of pressure suppression type and consists of a cylindrical concrete structure with an embedded steel liner.

Pressure relief system: it has three safety relief valves to protect the reactor pressure vessel against overpressure, in case of strong power unbalances.

2.1.2.2. KLT-40C (Russia)

Basic objectives

The KLT-40C is a twin-reactor system intended to produce fresh water and electric power in different proportions. It may also be used for heat production in a co-generation cycle. The KLT-40C design is based entirely on the serially produced marine NSSS being used in the Russian nuclear-powered icebreakers.



1- First Shut-down System
2- Second Shut-down System
3- Residual Heat Removal System (Emergency Condenser)
4-Safety Injection System
5- Suppression pool
6- Containment
A: CORE B: STEAM GENERATORS C: BUILDING (Secondary Containment)

Figure 2.5: CAREM-D Safety Systems.

Basic design features

The KLT-40C has the following original features:

1. Primary piping length is minimized.
2. Natural circulation is used in the primary and secondary circuits for all emergency modes.
3. The containment is designed for high over pressure and includes a passive pressure suppression system.
4. Safety is enhanced through fine-tuning of the engineered features proven by operation of the NSSS-prototype and by the use of systems, which do not require external power sources.

Design description

A floating power unit (FPU) equipped with KLT-40C reactors has been developed for a small floating nuclear co-generation plant (NCGP) dedicated for local power supply of remote coastal areas. National safety regulatory requirements and codes were used and IAEA NUSS documents were also taken into account for reactor and FPU development. NCGP with FPU has essential advantages:

- FPU is constructed at specialized shipbuilding facility. Quality is strictly controlled during all stages of construction and test process cycle.
- FPU is towed to the operation site ready for service and is tied up at the equipped berth and is connected to coastal communications, which are to be built at this moment.
- Storage of accompanying radwastes on the FPU board during operating period and further transportation of them along with the FPU for disposition.
- Power unit transportability that allows its delivery to specialized enterprise for restoration and repair and assurance of high quality.
- Rather simple procedure of NCGP decommissioning by FPU transportation to the site of ship nuclear objects dismounting and cutting.

In view of technical and economic advantages and high safety of KLT-40C RP it is expedient to use it as a part of nuclear desalinating complexes (NDC). Main technical parameters and characteristics of the FPU are presented in Table 2.4.

Nuclear steam supply system

KLT-40S NSSS has been developed on the basis of design experience and long-term (more than 160 reactor-years) operation of the Russian nuclear ice-breakers with similar reactor plants under the conditions of Russian North.

Main technical features of the reactor plant:

- Modular plant with vessel-type PWR.
- Forced circulation of primary coolant. Natural circulation is used in the emergency cooling mode.

The reactor plant equipment comprises the reactor, four steam generators and four primary pumps joined with short loaded bearing nozzles of tube-in-tube type in compact steam generating unit. The unit is installed in a metal-water shielding tank arranged in the containment, as shown in Figure 2.6.

The vessel is made of heat resistant high strength perlitic steel with corrosion-resistant brazing. Coolant layer between the vessel and core provides the integral fluence of neutrons, with energy of ≥ 0.5 MEV on the vessel is not more than $2x10^{20}$ n/cm^2. Vessel-to-cover joint is sealed with self-packing copper wedge gasket.

Steam generator is once-through heat exchanger. SG casing is made of low-alloyed steel lined inside with corrosion resistant brazing. SG tube system is made of titanium alloy as a set of cylindrical spatial helical coils joined in individual sections as for water supply and steam removal. Major rated technical parameters of KLT-40C are given in Table 2.5.

TABLE 2.4: MAIN CHARACTERISTICS OF FPU

Item	**Value**
1. Number of RPs	2
2. RP type	KLT-40C
3. RP thermal capacity, MW	2x 150
4. Steam capacity, t/h not less	2x240
5. Electric power of turbo-generator plant, MW	2x35
6. Electric power supplied to the network during turbine operation in heat extraction mode, MW	2x30
7. Electric power supplied to the network during turbine operation in condensing mode, MW	2x32.5
8. Heat quantity supplied to district heating system, Gcal/h	2x25

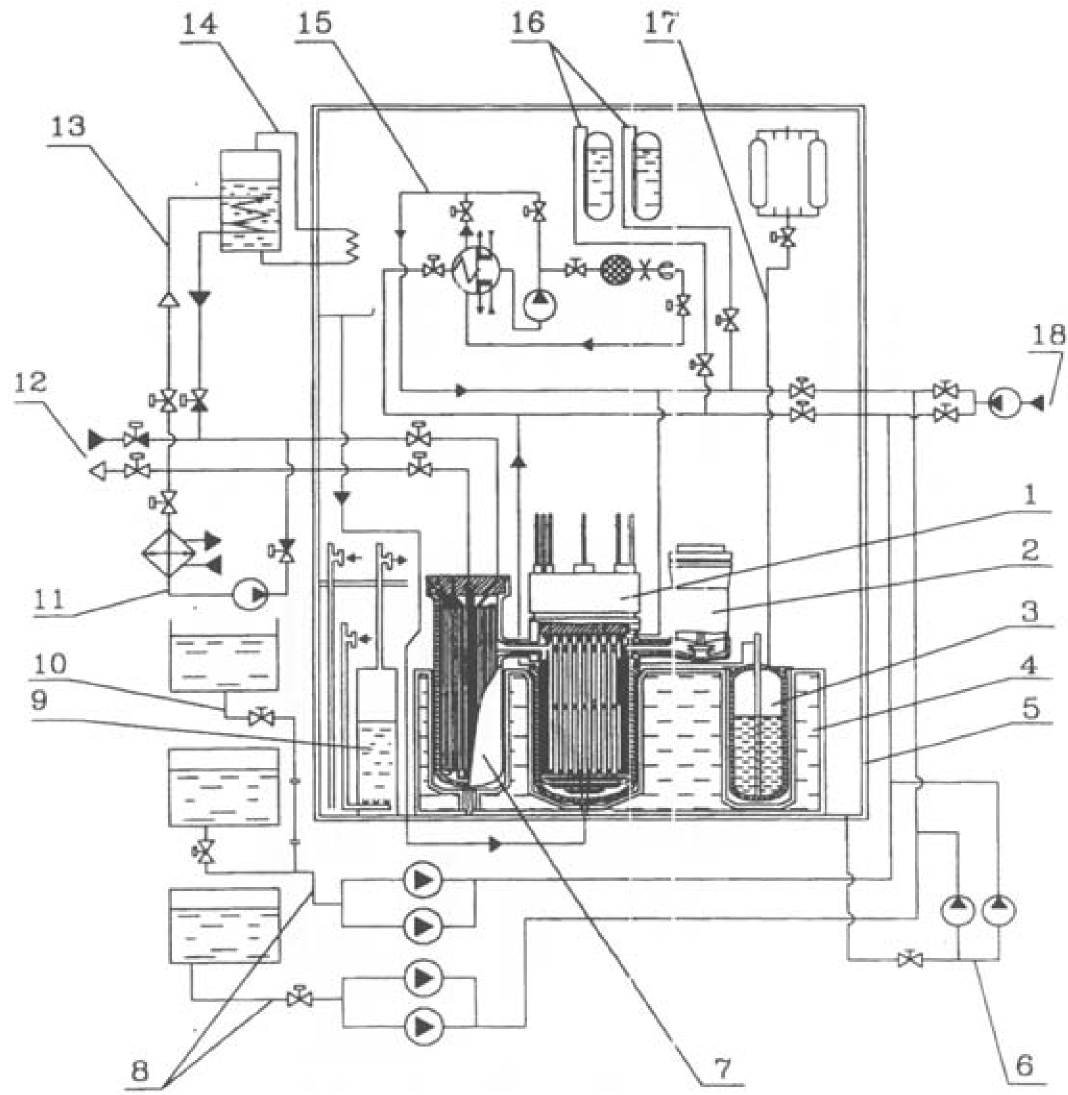
TABLE 2.5: MAJOR RATED TECHNICAL PARAMETERS OF KLT-40C

Item	**Value**
Thermal capacity, MWt	150
Steam-generating capacity, tons/hour	240
Primary circuit pressure, MPa	12.7
Steam pressure at SG outlet, MPa	3.72
Superheated steam temperature, ^{0}C	290
Service life, years	35–40
Overhaul period, years	10-12
Non-removable equipment lifetime, thousand hours	240–300
Removable equipment lifetime, thousand hours	80–100
Continuous operation period, hours	8000

Balance of plant

NSSS KLT-40C is intended to generate steam for power conversion system containing steam turbine, two-sectional horizontal condenser and electric generator. Steam turbine has steam extraction line to supply heat for intermediate circuit of the district heating system. The characteristics of steam supplied to steam turbine and intermediate water heater are shown in Table 2.6.

The closed intermediate circuit provides heat for heaters of district heating system. Both steam extracted from the turbine and live steam extracted directly from the main steam line (peak-mode heat supply) can be used as sources of heat for intermediate circuit. Active, one-cylinder, one-flow condensing turbine has non-controlled and controlled steam extractions to heat up feed water and intermediate circuit water. Turbine flow area consists of 13 pressure stages. District heating takeoff chamber divides the turbine into high and low-pressure parts.



1 – Reactor
2 – Main Primary Pump
3 – Pressurizer
4 – Metal-Water Shielding Tank
5 – Containment Shell
6 – Emergency Recirculation System
7 – Steam Generator
8 – High Pressure ECCS
9 – Bubbler System for Pressure Suppression in Containment Shell
10 – Liquid Absorber Injection
11 – Active Emergency Heat Removal System
12 – to Steam Turbine Plant
13 – Passive Emergency Heat Removal System
14 – Containment Pressure Suppression Emergency Condensation System
15 – Primary Circuit Clean-up and Heat Removal System
16 – Passive ECCS
17 – Pressurization Gas System
18 – Low Pressure ECCS

Figure 2.6: KLT-40C Nuclear Steam Supply System.

TABLE 2.6: THE CHARACTERISTICS OF STEAM SUPPLIED TO STEAM TURBINE AND INTERMEDIATE WATER HEATER

Item	Value
Steam flow rate for one unit, t/h	up to 240
Steam pressure at turbine inlet, MPa	3.5
Steam temperature turbine at turbine inlet, ^{0}C	285
Condenser Cooling Water Parameters:	
Flow rate, m^3/h	5400
Inlet temperature ^{0}C	5-25
Range, ^{0}C	10
Intermediate Coolant Parameters:	
Temperature, ^{0}C	130
Flow rate, m^3/h	100

Safety features

The KLT-40C reactor safety is based on proven engineering solutions:

- PWR-type reactor being the most widely used in the world practice is applied,
- Reliable and proven elements are used in the reactor plant, along with elimination of “weak points’’ revealed by operation experience of analogs,
- Diversified and redundant systems are used for the reactor emergency shutdown and passive and active heat removal.

The KLT-40C reactor plant safety systems are shown in Figure 2.7. Essential engineered features are:

Emergency reactor shutdown:

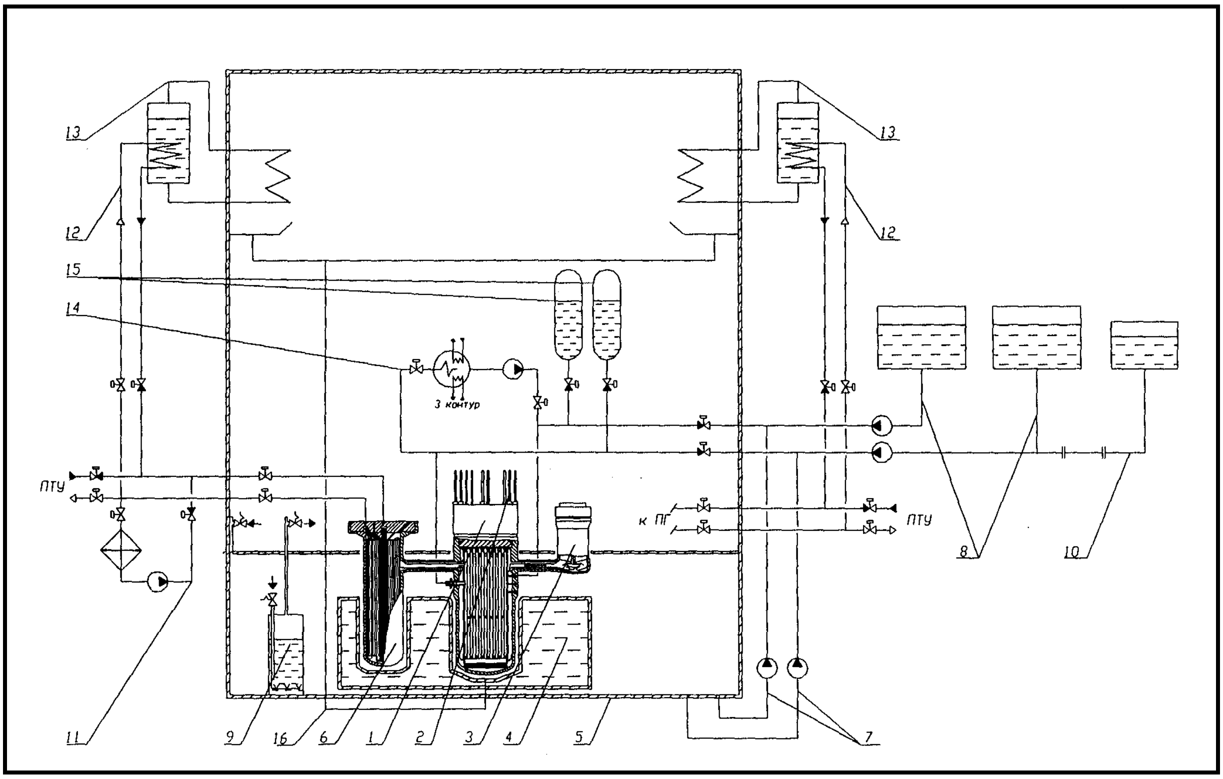
- Two sets of control & protection system (GPS) actuators;
- Two sets of GPS which control the GPS actuator. *Emergency core cooling*
- Two-train passive ECCS with hydraulic accumulators;
- Two-train active ECCS with make-up pumps;
- Two-train active ECCS with re-circulation pumps.

Emergency heat removal from the reactor:

- Two-train active ERHR system acting through SG and a primary-to-tertiary circuit heat exchanger with unlimited heat removal duration;
- Two-train passive EGS for heat removal from the reactor during 24 h without personnel intervention at long-term blackout.

Emergency isolation of radioactive materials:

- Double isolation valves in systems interfacing the primary circuit;
- Five barriers preventing release of radioactive materials to the environment (fuel matrix, fuel element cladding, primary circuit boundary, containment, protective enclosure).



1. Reactor; 2.CPS; 3.Reactor coolant pump; 4. Metallic-water shielding tank; 5. Containment, 6. Steam Generator; 7. ECCS recirculation channels; 8. Active ECCS channels; 9. Emergency condensation bubbling Pressure suppression subsystem in containment; 10. Liquid absorbent injection system; 11. Active ERHR channel through SG; 12. Passive ERHR channel through SG; 13. Emergency condensation pressure suppression subsystem in containment, 14. Active ERHR channel through 1–3 circuit heat exchanger; 15. Passive Eccs channels 16. System for reactor caisson filling with water.

Figure 2.7: KLT-40 C Safety Systems.

Accident management:

- Long grace period (more than 1 h) for personnel actions;
- Diversity and independence of accident management features, which technically eliminate a severe accident with fuel damage, including:
  (a) Reactor shutdown by means of devices actuated directly by primary circuit pressure;
  (b) Reactor shutdown by means of back-up liquid absorbent injection system;
  (c) Actuation of passive ERI-IR system by devices actuated directly by primary circuit pressure.

Mitigation of severe beyond-design accidence consequences:

- Retention of core melt inside the reactor vessel by means of outside heat removal using active and passive systems supplying water into reactor cavity;
- Double localizing barrier for radioactive products, including containment designed for emergency parameters and a protective enclosure with a system of aerosol and iodine filters.
- 500 m emergency planning zone (site boundary) sufficient for all realistically conceivable accidents.

2.1.2.3. NIKA-70 (Russia)

Basic objectives

The Research and Development Institute of Power Engineering (RDIPE) is actively involved in designing the NIKA-70 nuclear reactor, based on the long-term experience of RDIPE in designing mobile nuclear facilities. NIKA-70 is an advanced small integral PWR with enhanced safety capabilities. The main objective is to use NIKA-70 as part of the co-generation floating nuclear plant.

Basic design features

All components of the primary circuit (i.e. core, control rods, steam generator, main coolant pumps, and pressurizer) are located in a single vessel (Figure 2.8). Such reactor design offers the following advantages:

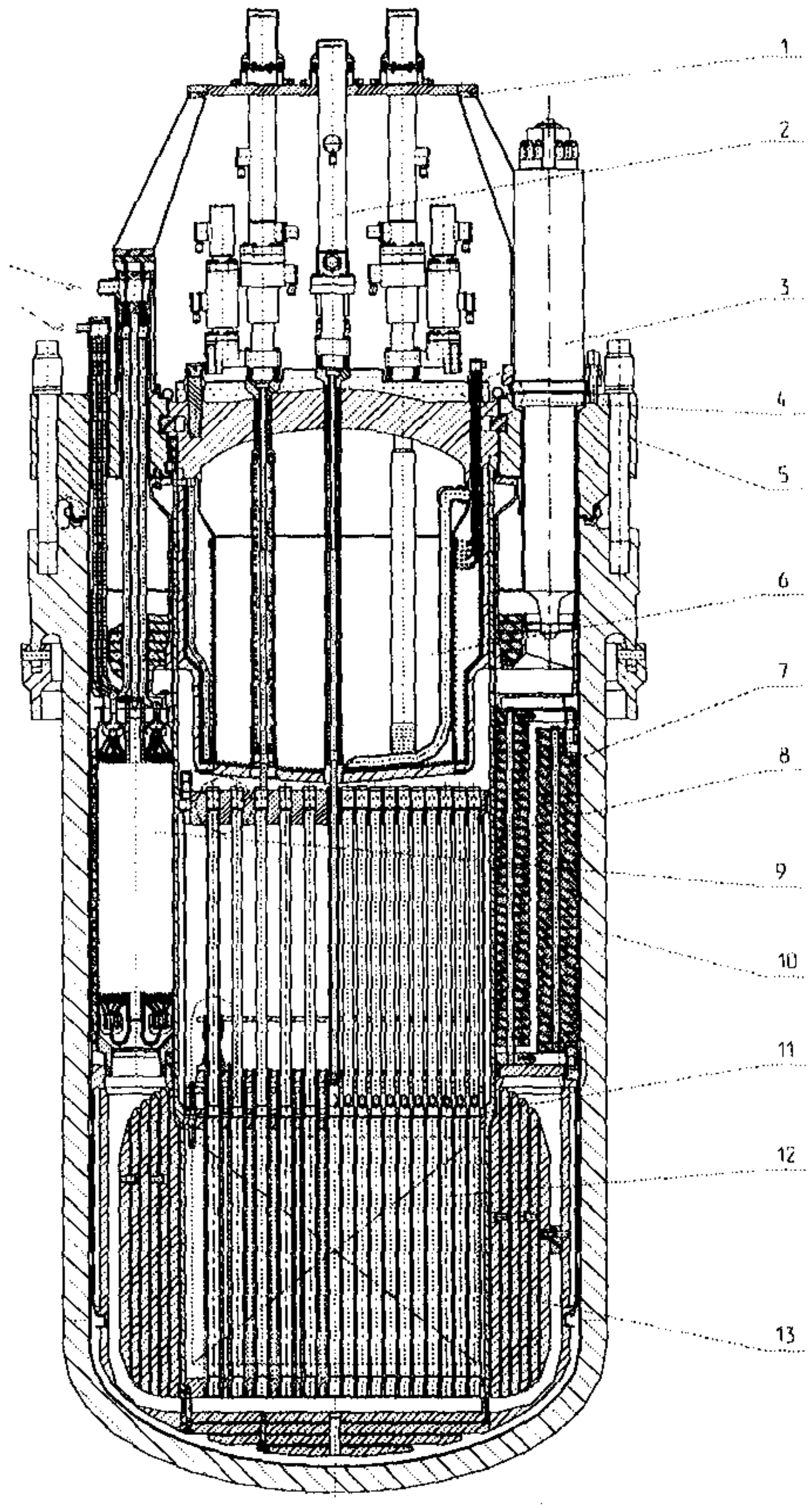
- Almost all primary circuit pipelines can be excluded from NSSS (because of its integral arrangement) and, hence, the probability of a leakage can be substantially reduced;
- High flow rate of natural circulation of primary coolant can be provided; and
- Water inventory above the core can be increased 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.

Design description

NIKA-70 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 circulations 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.



1. Drive fastening frame; 2.Shim rod group drive; 3. MCP; 4. thermal insulation; 5. Annular cover; 6. Presuurizer; 7. Displacers; 8. Metal work with control rod clusters; 9. SG; 10. Vessel; 11. Core barrel; 12. Fuel assembly; 13. Side shield.

Figure 2.8: General View of the Reactor NIKA-7.

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 is depicted in Figure 2.9. 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 dry-out.

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 2.7 summarizes the major design parameters of NSSS NIIKA-70.

TABLE 2.7: DESIGN CHARACTERISTICS OF NIKA-70 REACTOR

No.	Characteristic	Unit	Value
1	Thermal power of the core	MW(t)	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	^{0}C	274
6	Feed water temperature	^{0}C	60
7	Nominal pressure in primary circuit	MPa	15.0
8	Primary coolant temperature at full power:		
	at core inlet	^{0}C	260
	at core outlet	^{0}C	300
9	Operating range of power variations	% Nnom	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

Safety features

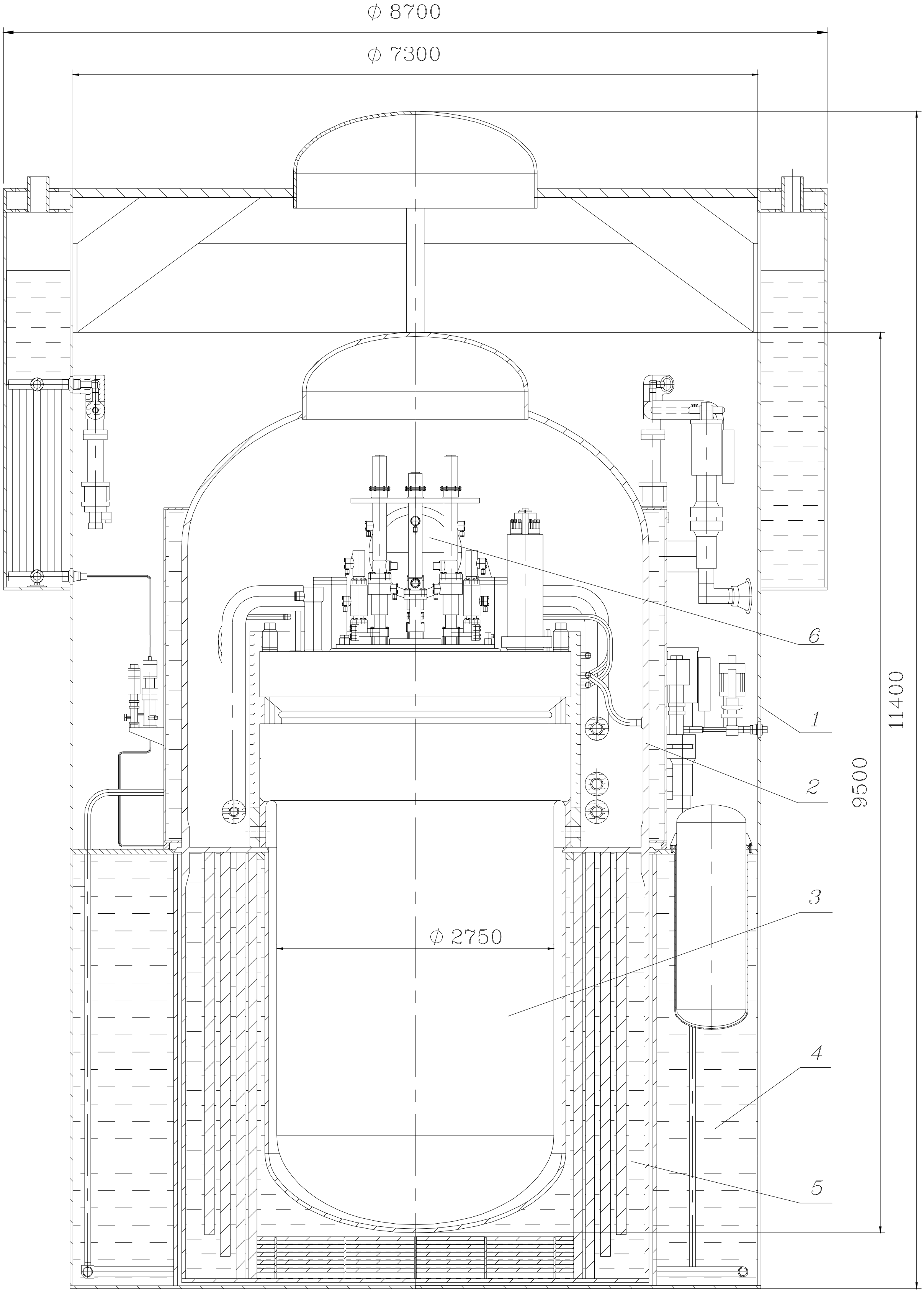
Essential to a high safety level of the NSSS is implementation of the following solutions:

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;



1. Containment 2. Safeguard vessel 3. Reactor 4. Biological shielding external tank
5. Biological shielding internal tank 6. Entrance hatch

Figure 2.9: Reactor Structure of the NIKA-70.

- 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 dry out.

Safety systems reliability 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 x 100%, of ECCS – 4 x 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.

2.1.2.4. SMART (Republic of Korea)

Basic objectives

SMART (**S**ystem-integrated **M**odular **A**dvanced **R**eac**T**or) is a small PWR with a rated thermal power of 330 MW. It is being developed by KAERI (Korea Atomic Energy Research Institute). The main objectives of the project are:

1- Development of a co-generation capable of producing 90 MW(e) and 40,000 m^3/day of desalted water.
2- Enhanced safety through a combination of inherent and passive engineered safety features.
3- Improved economics through system simplification, component modularization and construction time reduction.

Basic design features

The design of SMART is based on existing PWR technology and the fuel designs utilized in currently operating power reactors in Korea. The prominent design feature of SMART is the adoption of integral arrangement. All the primary components such as core, steam generator, main coolant pumps and pressurizer are integrated into a single pressurized vessel. The integrated arrangement of these components enables the elimination of large-sized 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. Table 2.8 summarizes the major design parameters of the SMART system.

Design description

Nuclear steam supply system

SMART Nuclear Steam Supply System (NSSS) is illustrated in Figure 2.10. A brief description of NSSS is provided below. Further details are provided in Reference [2.8].

Fuel and reactor core:
SMART core consists of fifty-seven (57) fuel assemblies, which are based on the industry proven 17x17 square array of the Korea Optimized Fuel Assembly (KOFA). Each fuel assembly holds 264 fuel rods, 24 guide tubes for control rods, and 1 instrumentation thimbles, which are mechanically joined in a square array. Five space grids hold the fuel rods in position.

TABLE 2.8: KEY DESIGN PARAMETERS OF SMART

General Information	- Reactor name/type	SMART/Integral PWR
	- Thermal power (MWt)	330
	- Design life time (year)	60
Fuel and Reactor Core	- Fuel type	$U0_2$ Square FA
	- Active fuel length (in)	2.0
	- No. of fuel assembly	57
	- Core power density (w/cc)	62.6
	- Refueling cycle (year)	>3
Reactivity Control	- No. of control bank elements	49
	- No. of control banks/material	4/Ag-In-Cd
	- Burnable poison material	$A1_20_3$-B_4C, GD_20_3-$U0_2$
Reactor Pressure Vessel	- Overall length (in)	10.6
	- Outer diameter (in)	4.6
	- Average vessel thickness (mm)	264
	- Vessel material	5A508, CL-3
Reactor Coolant System	- Design pressure (MPa)	17
	- Operating pressure (MPa)	15
	- Core inlet temperature (^{0}C)	270
	- Core outlet temperature (^{0}C)	310
Secondary System	- Feedwater pressure (MPa)	5.2
	- Feedwater temperature (^{0}C)	180
	- Steam pressure (MPa)	3.0
	- Steam temperature (^{0}C)	~74
	- Degree of superheating (^{0}C)	≥40

Top and bottom spacer grids are made of inconel, and the three middle space grids are made of zircaloy. A specially designed bottom end piece offers improved resistance to the debris entering the core. Enriched uranium oxide fuel of 4.95 w/o is enclosed in zircaloy cladding and has sufficient reactivity for a three (3) year or longer operation cycle. The fuel assembly is designed to allow power ramp operations during load-following maneuvers.

The SMART core is characterized by the ultra long cycle operation of a single or modified single batch reload scheme, low core power density, soluble boron-free operation, enhanced safety with a large negative Moderator Temperature Coefficient (MTC) at any time during the cycle, an adequate thermal margin, inherently free from xenon oscillation instability, and minimum rod motion for the load follow with coolant temperature control. SMART fuel management is designed to achieve a maximum cycle length for refueling. A simple single batch refueling scheme allows a cycle of 990 Effective Full Power Days (EFPD). This reload scheme minimizes complicated reload design efforts. A modified single batch scheme with 20 peripheral assemblies reloaded at every even numbered cycle is also possible, and thus enhances fuel utilization. The SMART fuel management scheme is highly flexible to meet customer requirements.

Reactor pressure vessel (RPV):
The SMART reactor pressure vessel (RPV) is a pressurized cylindrical vessel containing all major components of the primary system. The RPV consists of a cylindrical shell with an elliptical bottom and an upper flange part welded to the shell. The RPV is closed at the top by

the annular peripheral cover and the round central cover that also serves as the cover of the in-vessel pressurizer. The annular cover is fixed onto the vessel flange by means of a stud bolt joint. The vessel-to-annular cover joint is made leak-tight by welded torus sealing. The central cover is fastened to the annular cover by a flangeless joint, using rack-and-gear mechanism. The annular-to-central cover joint is also made leak-tight by welded torus sealing.

All penetrations to the RPV are limited to the vessel head region. This assures that the RPV forms a leak-tight container under any postulated pipe break accident. On the annular cover there are many nozzles for the twelve SGs, four MCPs, makeup piping, resistance thermometers, branch pipes, etc. On the outer surface of the central cover, there are nozzles for the 49 CEDMs, rack and gear drives, branch pipes, etc. The core barrel with fuel assemblies and shielding tube assemblies are located in the lower part of the RPV.

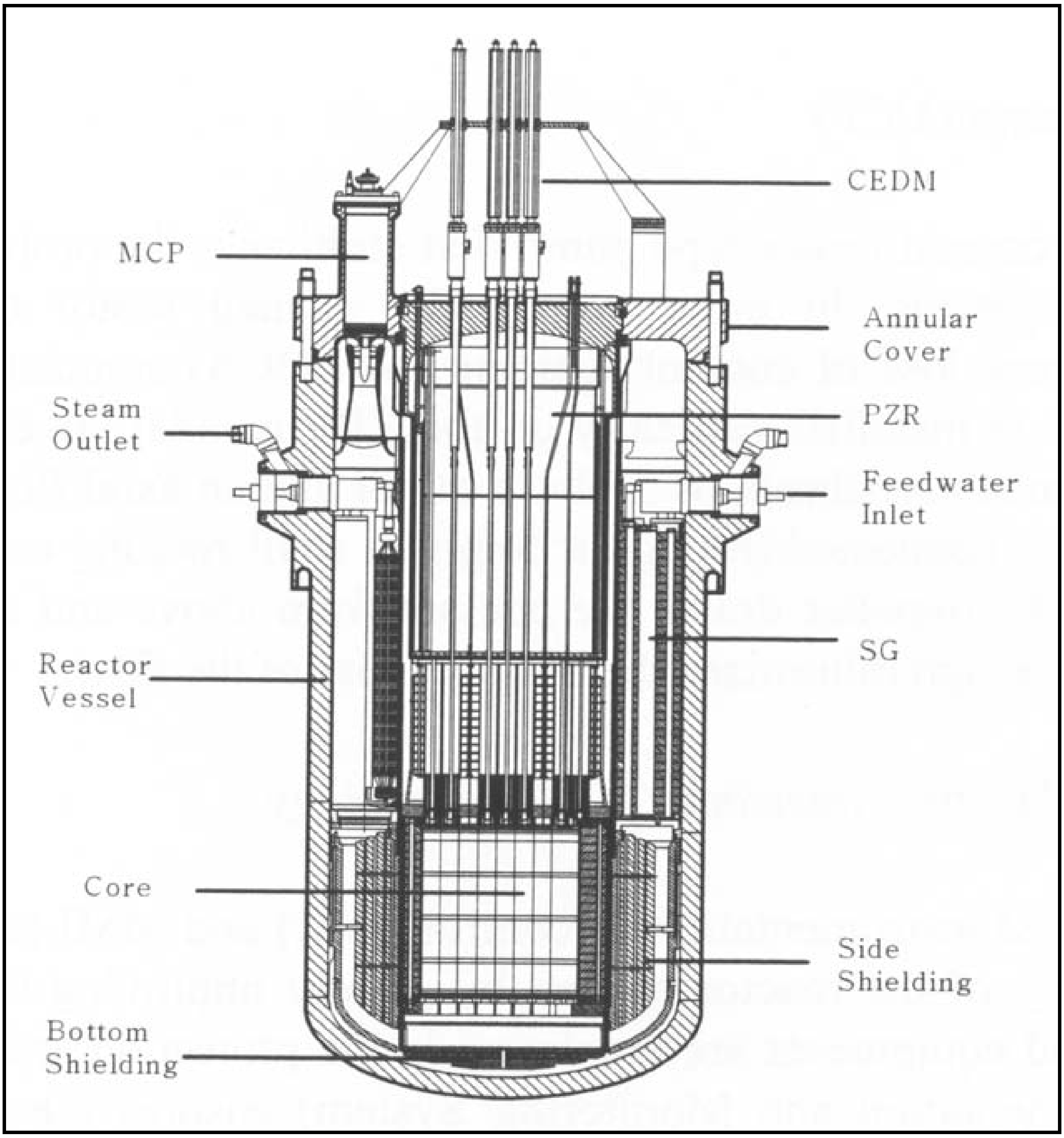


Figure 2.10: SMART Reactor Assembly.

Steam generators (SG):
SMART has twelve identical SG cassettes which are located in the annulus formed by the RPV and the core support barrel. One SG cassette has 324 helically coiled tubes wound around the inner shell through which the six feedwater tubes pass. The primary reactor coolant flows downward in the shell side of the SG tubes, while the secondary feedwater flows upward in the tube side. Steam at 3.0 MPa and superheated by 40^0C exits the SG For the performance and safety aspects, each SG cassette consists of six identical and independent modules with the same number of tubes per module. Each module has one feedwater intake header and one steam outlet header. Six modules from three adjacent SG cassettes - two modules per cassette, are then joined into one nozzle, and three nozzles are again joined to form one section.

This hierarchical grouping of SG modules is designed to minimize the impact of a SG tube rupture accident on the reactor system. To prevent hydrodynamic instability inside the tubes, a throttling device is installed at the feedwater inlet of each helically coiled tube. Each throttling device is located in the tube sheet of the feedwater header, and it is a sleeve with a core connected by a thread joint. Throttling orifices are also installed in the lower part of each cassette on the primary side to provide a uniform primary coolant flow rate to each cassette. In the case of normal shutdown of the reactor, the SG is used as the heat exchanger for the passive decay heat removal system (PRHRS).

Pressurizer (PZR):
The pressurizer is located inside the upper part of the RPV, and it is filled with nitrogen gas, water and steam. The pressurizer is connected to compressed nitrogen gas tanks located outside the RPV. The primary system pressure is determined by the sum of the partial pressures of nitrogen gas and steam. The pressure in the primary system is automatically and passively regulated by the thermo-dynamic interactions of water/steam and gas in the pressurizer. The pressurizer design eliminates the complicated control and maintenance requirements and there is no active pressure control mechanism such as a spray or a heater.

To prevent relatively large variation in the system pressure with the power change, the average temperature variation of the primary coolant is programmed to compensate the water volume change difference between the cold and hot parts of the primary coolant. As a result, the average temperature of the primary coolant slightly decreases as the power level increases. Another unique feature of the SMART pressurizer is the pressurizer cooler. To reduce the effect of the steam partial pressure and thus reduce pressure variations with the power, the cooler maintains the temperature of the water/steam and gas inside the pressurizer low. For the same purpose, a wet thermal insulator is placed between the pressurizer and the primary system to reduce the conductive heat transfer.

Control element drive mechanism (CEDM):
SMART CEDM is designed for fine-step movement and consists of a linear pulse motor (LPM), choke rack with top and bottom limit switches which also act as a control element assembly (CEA) position indicator, hydro-dampers, locking device, and extension shaft connecting the CEDM and CEA. The LPM is a four-phase synchronous DC electric machine with a passive armature in a coolant medium inside a strong sealed body made of magneto-soft corrosion-resistant steel. The stator and armature are the basic units of the LPM. The normal travel length of the control rod is 4mm per pulse, and the travel speed is in the range of 0–50 mm/sec. The drives are located on the reactor central cover and fastened to the cover by means of flange joints with studs and nuts. Between the flange joints, copper gaskets are used for sealing.

Main coolant pump (MCP):
SMART MCP is a canned motor pump which does not require pump seals. This characteristic basically eliminates small break Loss of Coolant Accidents associated with pump seal failure which becomes one of design bases events in the reactors using a conventional pump. The SMART has four MCPs vertically installed on the top annular cover of the RPV. Each MCP is an integral unit consisted of a canned asynchronous three phase motor and an axial-flow single-stage pump. The motor and pump are connected by a common shaft rotating on three radial and one axial thrust bearing. The bearings use a specialized graphite-based material, and the axial bearing performs the function of sealing too. The cooling of the pumps is accomplished with the component cooling water. A sensor installed in the upper part of the motor controls the rotational speed of the pump rotor. To avoid the reverse rotation of the pump rotor, an anti-reverse device is installed at the motor shaft near the middle radial bearing.

Balance of plant

The secondary system of SMART consists of the main steam valves, turbine-generator, condenser, feedwater heaters, pumps, associated piping and valves and instrumentation and control systems. Feedwater is heated by three (3) stages of Feedwater Heaters and supplies to Steam Generator.

The secondary system receives superheated steam from the NSSS. It uses most of the steam for electricity generation, seawater desalination and pre-heaters. The main steam pressure is controlled so as to be constant during power operation. The load change is achieved by changing the feedwater flow rate. Seawater desalination system such as MED, MSF, and RO may be used in conjunction with the secondary system using proper interfacing methods.

The major auxiliary system of SMART primary system is consisted of a component cooling system (CCS), purification system and make-up system. The CCS removes the heat generated in the main coolant pumps, control element drive mechanism, pressurizer, and internal shielding tank. The purification system purifies the primary coolant and controls the water chemistry of the reactor core and all equipments. The make-up system fills the primary coolant in case of the primary system leaks and supplies water to the compensating tanks for the passive residual heat removal System.

The auxiliary system for the secondary system is consisted of the circulating water system, service water system, domestic water system, make-up demineralizer system, condensate storage and transfer system, raw water system, seawater cooling water system, auxiliary steam system, fire protection system, instrument and service air system, chemical feed and handling system, chlorination system, wastewater treatment system, carbon dioxide system, nitrogen system, hydrogen system, fuel oil supply system, building drain system, essential chilled water system, plant chilled water system, breathing air system, HVAC system, etc. Figure 2.11 shows the schematic diagram of SMART BOP design.

Safety features

Besides the inherent safety characteristics of SMART, further enhanced safety is accomplished with highly reliable engineered safety systems. The engineered safety systems designed to function passively on the demand consist of a reactor shutdown system, passive residual heat removal system, emergency core cooling system, safeguard vessel, and containment overpressure protection system.

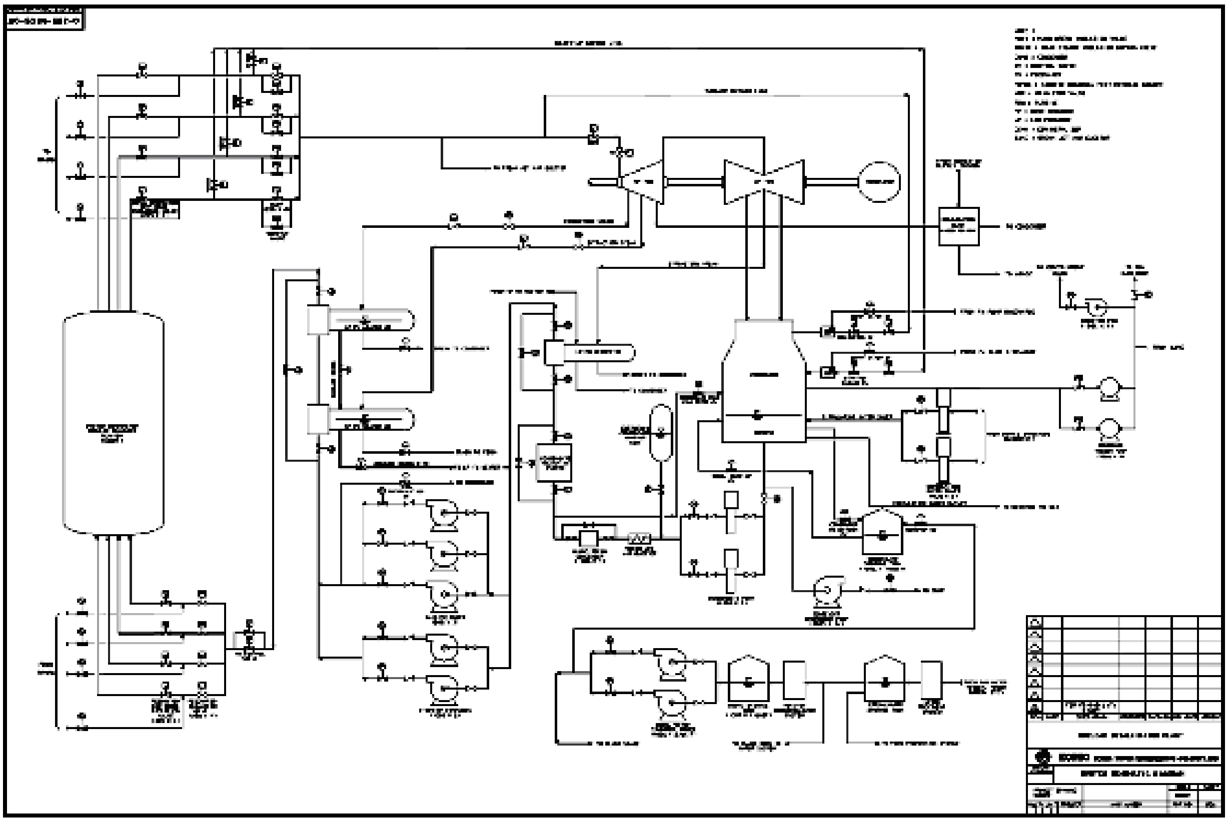


Figure 2.11: Schematic diagram of SMART balance of plant.

Additional engineered safety systems include the reactor overpressure protection system and the severe accident mitigation system. The schematic diagram of the SMART safety system is shown in Figure 2.12.

Reactor shutdown system (RSS):
Shutdown of the SMART can be achieved by a function of one of three independent systems. The primary shutdown system is 32 shutdown banks of CEA of which the absorbing material is B4C. The control banks are dropped into the reactor core by gravity force. These control banks have a sufficient shutdown margin to bring the reactor from hot full power to hot shutdown, even with a most reactive bank stuck out of the core. In the case of failure of the primary shutdown system, the emergency boron injection system is provided as a backup system and consists of two 6m3 tanks, each filled with 30g of boric acid per 1 kg water. One train is able to bring the reactor to the sub critical state. The system is an active system working with a pump.

Passive residual heat removal system (PRHRS):
The system passively removes the core decay heat and sensible heat by natural circulation in case of an emergency such as steam extraction, unavailability of feedwater supply, and station black out. The PRHRS may also be used in case of long-term cooling for repair or refueling works. In those long-term cooling cases, active pumps are used for circulating the cooling water. The PRHRS consists of 4 independent trains with a *50%* decay heat removal capacity each. Two trains are sufficient to remove the decay heat. Each train is composed of an emergency cooldown tank, a heat exchanger and a compensating tank. The system is designed to keep the core un-damaged for 72 hours without any corrective actions by operators at the postulated design basis accidents. In the case of SMART normal shutdown, the residual heat is removed through the SG to the condenser with a turbine bypass system.

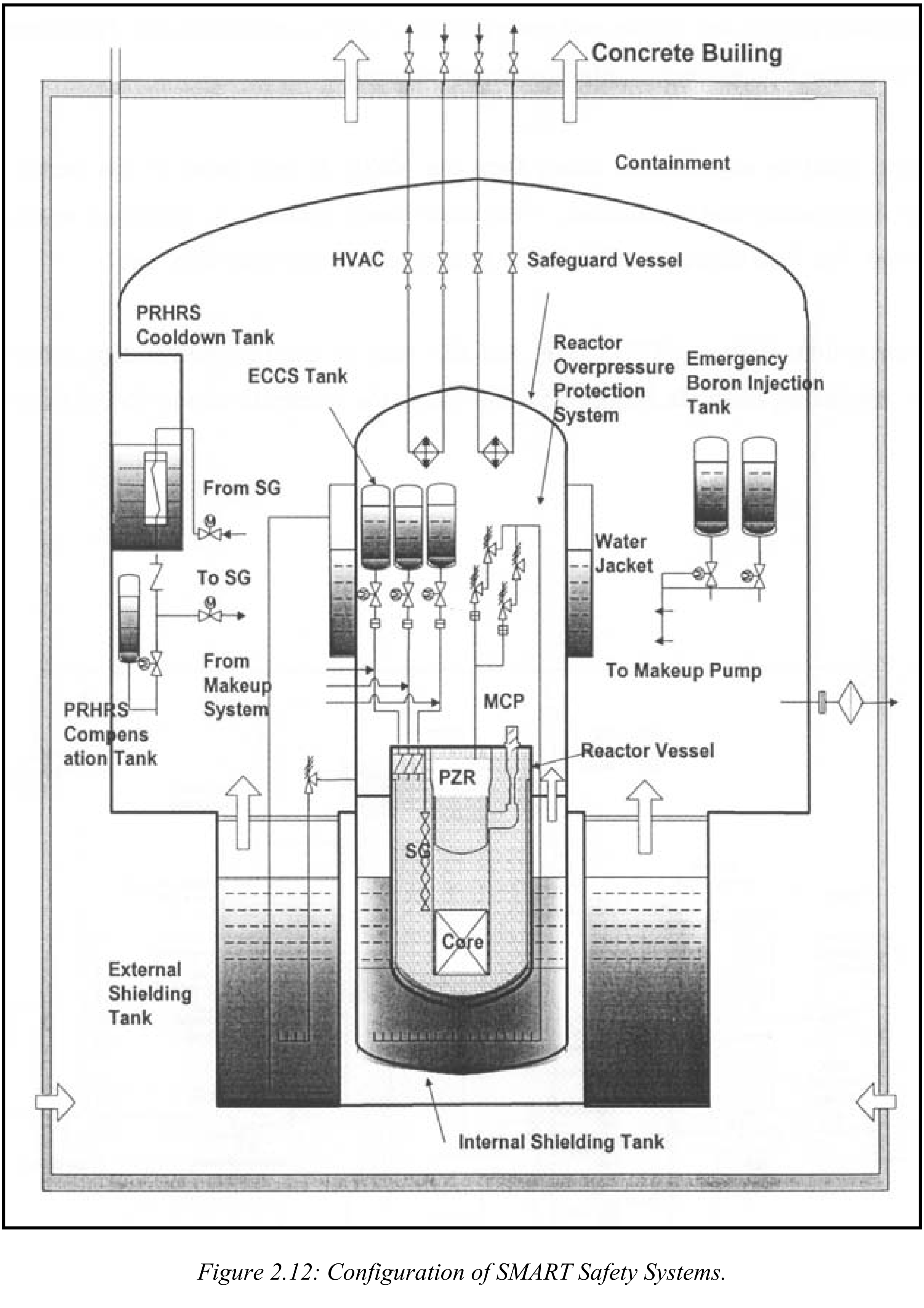


Figure 2.12: Configuration of SMART Safety Systems.

Emergency core cooling system (ECCS):
The SMART design excludes any possibility of large break Loss of coolant accident. The largest size of pipes connected to the outside of the RPV is 20mm. The ECCS is thus provided to protect the core uncovery by mitigating the consequences of design basis events such as

small break LOCA through make-up of the primary coolant inventory. When an initiating event occurs, the primary system is depressurized. The pressure difference between the primary system and the ECCS breaks the rupture disc installed in the pipe of the ECCS and water immediately comes into the core by the compressed gas pressure. The ECCS consists of two independent trains with 100% capacity each. Each train includes a cylindrical water tank of 5 m^3 in volume pressurized with nitrogen gas, isolation and check valves, rupture disc, and a pipe of 20mm in diameter connected to the RPV.

Safeguard vessel
The safeguard vessel is a leak-tight steel-made vessel intended to accommodate all primary reactor systems including the reactor assembly, pressurizer gas cylinders, and associated valves and pipings. The primary function of the safeguard vessel is to confine the radioactive products within the vessel and thus to protect any primary coolant leakage to the containment. The vessel also has a function to keep the reactor core undamaged for 72 hours without any corrective actions at the postulated design basis accidents including LOCA, with the operation of the PRHRS and ECCS. The steam released from the opening of the relief valve of the safeguard vessel at the postulated beyond design basis accidents is sparged into the external shielding tanks and immediately condensed.

Containment overpressure protection system (COPS)
The containment is a steel structure with a concrete building enclosing the safeguard vessel to prevent the release of radioactive products to the outside environment under the postulated beyond design basis accidents relating to the loss of integrity of the safeguard vessel. For any accident causing a temperature rise and thus a pressure rise in the containment, a containment cooling is accomplished in a passive manner. The heat is removed through the steel structure and through the emergency cooldown tanks installed inside the containment. A rupture disc and a filtering system are also provided in the containment to protect the steel structure from overpressure and to purify the released radioactive products at the postulated beyond design basis accidents.

Reactor overpressure protection system (ROPS)
The function of the ROPS is to reduce the reactor pressure in overpressure accidents. The system consists of two parallel trains, which are connected to the pressurizer through a single pipeline. The two trains are also combined to a single pipeline connected to the internal shielding tank. Each train is equipped with a rupture disc and two relief valves. In the overpressure accidents, the rupture disc is broken and the relief valve is open by the passive means of flow thrust. The steam is then discharged into the internal shielding tank through the sparger and condensed.

2.1.3. Heating reactors

2.1.3.1. HR-200 (China)

Basic objectives and features

The HR-200 is a nuclear heating reactor (NHR) with 200MW of thermal power, developed by the Institute of Nuclear Energy Technology (INET), Tsinghua University, China. It is specially designed to provide with heat to seawater desalination, district heating and so on. The State Nuclear Safety Authority of China has issued the construction permit for NHR-200. The basic design objectives of HR-200 are as follows:

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} per year and a significant release frequency <10^{-9} per year.
- Reliable reactivity control and shutdown system.
- Low operating parameter and large safety margin.

Reliability

- Simplified design, operation and maintenance.
- Public radiation exposure at the DaQing plant site within the range of 80 km: $5x10^{-4}$ <man-Sv per year.
- Overall plant availability goal greater than 95% considering forced and planed outage.

Economics

- Net heating output of 196 MW(th)
- The capital cost of a HR-200 is about 110.0 Million US$ in 1991 and the heat generation cost is competitive with coal (or oil) on an average China site.

Major innovative features

- Integrated arrangement, self-pressurization.
- Dual pressure vessel structure.
- Low operating temperature, pressure and low power density in reactor design which provided increases operating margins and improved fuel economy.
- Cooling reactor core with simple, passive measure, which uses natural driving force.
- Adopted 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 workstations, 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.

Design description

The main features of HR-200 are described below and summarized in Table 2.9.

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 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.

TABLE 2.9: MAIN DESIGN PARAMETERS OF THE HR-200

Parameters	Unit	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	^0C	146/186	174/210	153/210
Power density in the core	kW/L	26	23	27
Number of fuel assemblies		16	32	120
Number of control rods		13	13	32
Intermediate circuit pressure	MPa	1.1	3.0	3.0
Intermediate circuit inlet/outlet temperature	^0C	102/142	135/180	135/170
Supplied steam temperature	^0C		125	125

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, 5.0 cm inside diameter. The design concept is shown in Figure 2.13. The design also features a large coolant inventory in RPV (about 200T) which provide for a lower total neutron flux to the RPV (10^{16} n/cm^2 for 40 years lifetime of the reactor). All in-vessel penetrations (only with small diameter) are located on the upper of RPV.

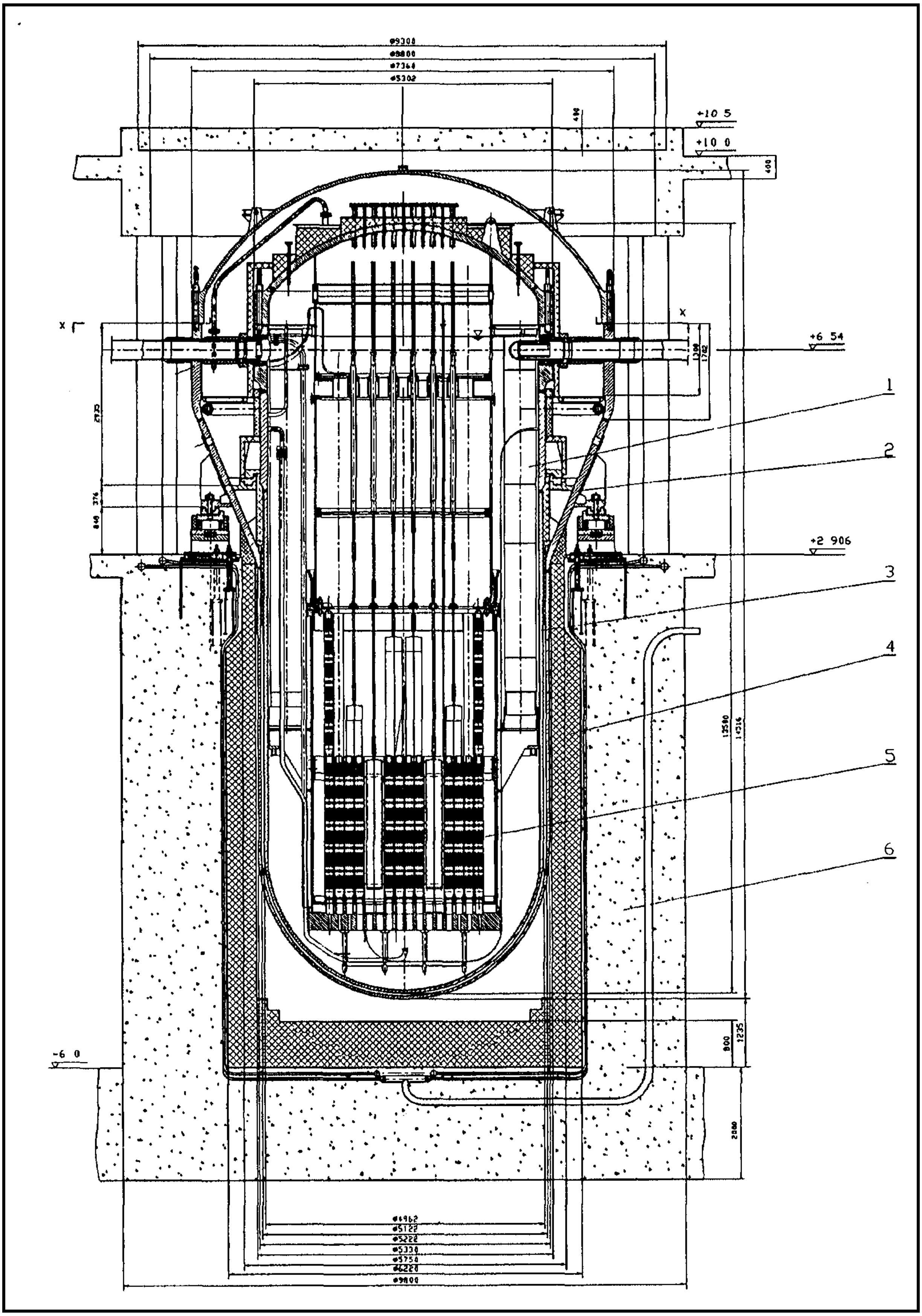
Reactor core
The reactor core of HR-200 consists of 120 assemblies (fuel ducts) and 32 control rods. A long riser is located above the core to enhance the capability of natural circulation. The reactor core stands on the lattice-support structure, which is fixed on the RPV.

The fuel bundle is arranged in 12x12 matrix with an active length of 1.9m and is contained in the duct (or box). The circuit form control rods are placed in the gaps between the square duct. 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/ton.

The equivalent core diameter is 2.6m. 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 results in a negative temperature coefficient of reactivity over the complete core life.

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

Control rod and control rod-driven mechanism
A new type of hydraulic driving facility is used for driving the control rod in 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 valve in the control unit, and it moves the movable part of the step-cylinder step by step. The drive system is very simple in its structure and is designed on “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.



1. Main heat exchanger
2. Containment
3. Reactor pressure vessel
4. Thermal isolation
5. Reactor core
6. Biological shield

Figure 2.13: Nuclear Heating Reactor NHR-200.

Primary heat exchanger (PHE)
On the periphery of RPV upper part the 6 sets of PHE are located. 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 divides 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. The operating temperature of PHE is 210 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. The 30 assemblies will be refueled at one time. The spent fuel will be moved into spent fuel racks around the active core and stored in it. The design greatly simplifies the refueling 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 RPV.

Intermediate circuit

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

Nuclear steam supply system

The primary coolant absorbs the heat from the reactor core, then flows through the primary heat exchangers, where the heat carried is transferred to the intermediate circuit. The nuclear heat then is transported by the coolant in the secondary circuit to the steam supply system, through the steam generators. So that the steam supply system is the third circuit of the nuclear heating reactor plant. The general flow diagram of the plant is shown in Figure 2.14.

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 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 reactor operation safe, and the 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”. 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 system.

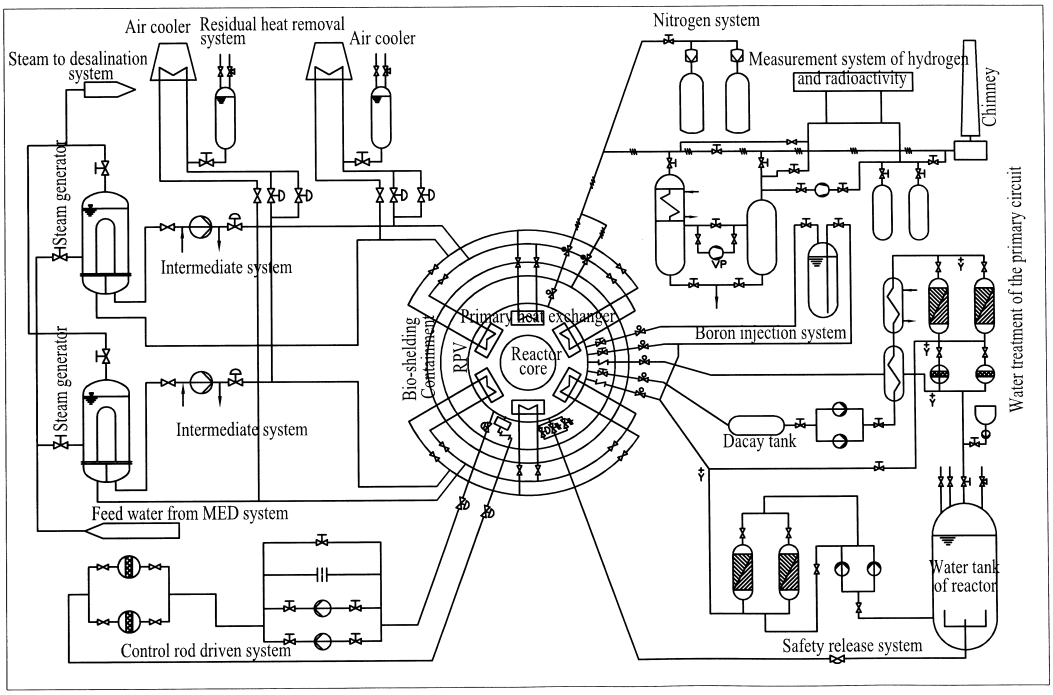


Figure 2.14: General Flow Diagram of NHR-200 Plant.

Safety features

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

(1) Large negative temperature reactivity coefficient over the complete core life.
(2) Integrated arrangement, self-pressurization, minimization of in-vessel penetration and their location at the upper part of the RPV.
(3) Natural circulation of coolant in the primary circuit under all conditions.
(4) Low power density in the core and low operating pressure in the primary system.
(5) Large coolant inventory and double pressure vessel design keeping the core covered by coolant under all conditions and excluding large break LOCA.
(6) Fail-safe principle on hydro-driven control rod drive mechanism design.
(7) Elimination of positive control rod ejection accidents by in-vessel hydro-driven control rod mechanisms.
(8) Long grace period for corrective actions.
(9) Spent fuel in-RPV storage
(10) Passive safety systems.

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 atmosphere properly by means of natural circulation.

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

The safety concept of NHR is based on the higher degree of its inherent safety instead of the engineering safety features. The NHR is operated under low pressure, low temperature, low power density and low radioactivity content in primary coolant. The large sub-cooled water inventory results in a high thermal inertia in the primary system. A large negative temperature reactivity coefficient has been achieved in the core nuclear design, so any uncontrolled reactivity addition and anticipated transients without scram (ATWS) will be counteracted very well. The core decay heat is transferred into atmosphere by natural circulation. Moreover, loss of primary coolant is limited to the content that the core will never be uncovered, therefore, the emergency core cooling system is not necessary for the NHR. Even in the worst case of the beyond design accident, the grace period is long enough for operator to take corrective action to mitigate the consequences of the accidents. As a result, the core meltdown can be excluded.

2.1.3.2. RUTA (Russian Federation)

Basic objectives

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

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

Basic design features

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

Variants of the RUTA reactor with a thermal power of 10 to 70 MW have been developed at the conceptual level. The facility's economic efficiency improves significantly as the reactor power is increased. Reactors with the thermal power of over 50 MW should be considered as most promising. The basic technical characteristics of the RUTA-55 and RUTA-70 reactors are presented in Table 2.10.

TABLE 2.10: PRINCIPAL TECHNICAL CHARACTERISTICS OF RUTA-55 AND RUTA-N-70 REACTORS

Reactor	**RUTA-55**	**RUTA-N-70**
Nominal thermal power, MW	55	70
Circulation mode in primary circuit at nominal power	Natural Circulation	070%N_{nom} - natural circulation 70-100% N_{nom} - forced circulation
Water volume in reactor tank, m^3	700	700
Core dimensions (diameter/ height), m	2.03/1.2	1.72/1.4
Number of fuel assemblies in the core	169	121
Type of fuel elements	RBMK	VVER
Fuel	UO_2	UO_2
Enrichment for U_{235},%	3.6	4.0
Amount of uranium U_{235} in the core, kg	5942/213.9	6622/265.0
Core power density, MW/m^3	14.1	21.5
Average power generation, MW·day/kg h.a.	27.5	26-27
Fuel cycle, effective days	2970	2500-2800
Parameters of primary coolant:		
Flow rate, kg/s (t/h)	525 (1890)	725 (2610)
Temperature in the core (inlet/ outlet), ^oC	75/100	78/101
Pressure at the core inlet, Mpa	0.25	0.27
Parameters of secondary coolant:		
Flow rate, kg/s(t/h)	535 (1926)	795 (2862)
Temperature in primary HX (inlet/outlet), ^oC	66/90	74/95
Pressure, Mpa	0.39	0.39
Parameters of tertiary coolant:		
Flow rate, kg/s (t/h)	525 (1890)	835 (3007)
Temperature in network HX (inlet/outlet) , ^oC	60/85	70/90
Pressure, Mpa	0.59	0.59

The basic design characteristics of RUTA-70 are summarized below.

- The reactor reinforced concrete vessel with inside bimetallic liner is made of two sections: lower section — cylindrical; upper section — a rectangular (in plan) pool with primary heat exchangers;
- Compact reactor core, 1.4 m high and 1.7 m in diameter, incorporates fuel assemblies containing standard VVER fuel elements;
- Geometrical dimensions of the in-pile circulation circuit flow part are determined from the requirement of the core cooling assurance in the natural circulation conditions in the range of loads from minimal power to ~70% N_{nom};
- Switching from one circulation mode to another in the primary circuit (natural/forced), when the pumps are actuated occurs automatically by means of passive closing/opening of the check valve (gate).

Design description

The structural diagrams of the RUTA-55 and RUTA-70 reactors are presented in Figures 2.15-2.17. The reactor is located in a concrete tank vessel with an internal bimetallic lining. The concrete vessel also serves as a radiation shield. Concrete plates are installed over the reactor to protect the equipment against external impacts. The reactor tank is filled with distilled water at the atmospheric pressure in the above-water air space. The primary circuit components with an integral in-tank arrangement operate at the hydrostatic pressure. The primary circuit's hydraulic train is arranged inside the reactor's water volume by a system of the internals and consists of the core, the up-comer section, primary heat exchangers, the reactor pool and the down-comer section. The core is build up of hexagonal fuel assemblies with fuel rods. The fuel is uranium dioxide. A core partial refueling is envisaged with the fuel assembly's partial replacement once in three years. The overall nuclear fuel life is 6–9 years.

The RUTA reactor facility has three circuits. The coolant circulation in the primary circuit is natural and a power boost through forced circulation is possible when pumps are installed. Heat is transferred from the primary circuit to the secondary one and from the secondary circuit (intermediate circuit) to the tertiary one via heat exchanging surfaces. The secondary circuit is intermediate and intended for the heat transfer from the primary to the tertiary circuit consumer circuit). The secondary circuit is also used for the reactor cool down (in normal and emergency modes). For NHP the tertiary circuit means the heating network, for nuclear desalination complex (NDC) the tertiary circuit is a system of steam generating loops for heating the desalination modules.

In the RUTA-55 reactor (Figure 2.15) the core is cooled by means of coolant natural circulation [2.9]. For the RUTA-70 reactor the natural circulation mode was adopted for operation at a power level below 70% N_{nom}, for the range of power 70-100% N_{nom} the operation with forced circulation in the reactor is envisaged. RUTA-70 advanced reactor structural diagram and vertical cut are given in Figures 2.16 and 2.17.

A certain increase in potential of heat supplied to consumer is possible when changing over to forced circulation in the primary circuit along with a 25–30% gain in the RE power without significant increase in the reactor size. Creation of additional head using pumps in the natural circulation circuit permits providing the necessary flow rate in the primary circuit at a lower temperature drop between hot and cold legs of the circuit and increasing the coolant temperature in the down-comer.

Increase in supply water temperature in secondary and tertiary circuits of the RE is attained by reducing temperature head in the primary and intermediate heat exchangers at the expense of plate-finned heat exchangers made of aluminum alloys. The supply/return water temperature in the tertiary circuit of the optimized RE makes up 90/70°C compared to 85/60°C for the RUTA-55 basic variant.

Forced circulation in the reactor circuit is to be considered as optimization problem. Installation of pumps involves additional equipment costs and the relative increase in the operation cost due to increase in required power supply, which will be especially tangible in case of NDC, when actually base-load reactor operation at maximum power is envisaged (capacity factor 0.9).

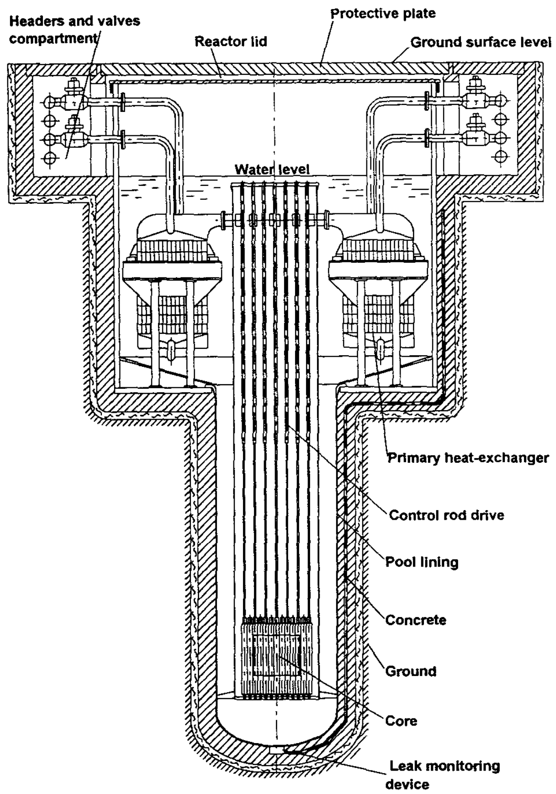


Figure 2.15: RUTA-55 Structural Diagram.

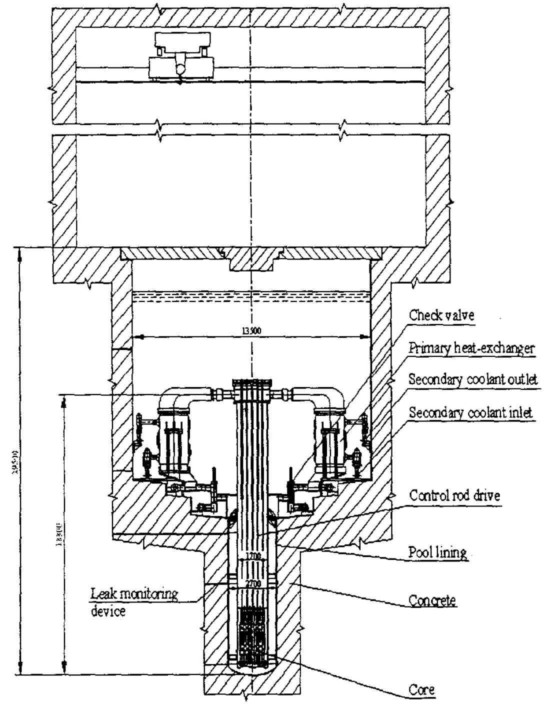


Figure 2.16: RUTA-70 Structural Diagram.

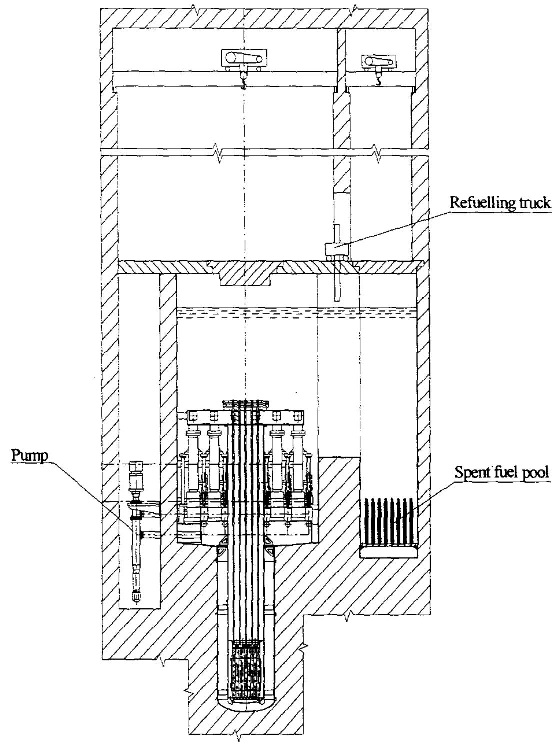


Figure 2.17: Advanced RUTA-70 Reactor Sectional Drawing (pumps are shown).

Safety features

Enhanced safety level of the RUTA reactor is based on the following inherent features:

- Absence of excessive pressure in the reactor tank;
- Large reactor thermal inertia thanks to vast amount of water (700 m^3~ in the reactor tank;
- Integral arrangement of the primary circuit
- Natural circulation of coolant in the primary circuit
- Low power density in the core
- Negative reactivity feedbox result in spontaneous reduction or self-restriction of the reactor power irrespective of the control rod positions
- Two pressure barriers ($P_1 < P_2 < P_3$) prevent ingress of radioactivity from primary into customer circuits.

The combination of these factors makes it possible to exclude deterministically the probability of an accident with a melt or severe damage of the core in the RUTA reactors.

2.1.4. Research reactors

There is only one research reactor considered for coupling with desalination unit within this CRP, which is the CIRUS research reactor. BARC has developed a low temperature vacuum evaporation (LTE) process that can utilize energy sources at low temperature for desalination of seawater. A desalination unit of 30 m^3/day capacity based on this process has been designed and fabricated.

Since nuclear research reactors produce significant quantities of low temperature heat (energy), a scheme has been evolved to integrate this desalination unit with CIRUS reactor at BARC such that the technology of utilizing reactor waste heat for desalination of sea water through low temperature vacuum evaporation process can be practically demonstrated. The salient features of CIRUS reactor are summarized below.

Salient features of CIRUS reactor

CIRUS is a 40 MW thermal research reactor using metallic natural uranium fuel, heavy water moderator, de-mineralized light water coolant and seawater as the secondary coolant. The heat generated in the fuel assemblies is removed by re-circulation of de-mineralized water in a closed loop (Primary Cooling Water System) using four re-circulation pumps operating in parallel. The coolant flow through reactor core is from top to bottom. Control valves installed at the re-circulation pump discharge maintain coolant pressure at the core inlet. Pressure at the core outlet is maintained by keeping constant level in the expansion tank (standpipe) provided at core outlet/ heat exchangers inlet.

The heat from the primary coolant is transferred to the secondary coolant (seawater) in a set of five heat exchangers and ultimately rejected to sea. The relevant process parameters of Primary Cooling Water (PCW) and Seawater (SW) systems are given as case I in Table 2.11. When PCW pumps are not in operation for any reason, shut down cooling flow through the core is established automatically by one pass gravity flow from an overhead water storage tank (ball tank). The core outlet is led to an underground water storage tank (dump tank) from where it is pumped back to the ball tank.

TABLE 2.11: CHANGES IN FLOW AND TEMPERATURE OF CIRUS PCW AND SW SYSTEMS AT 40 MW OPERATION

PARAMETERS			CASE–I	CASE–II	CASE–III
CORE	PCW Flow (lpm)		17700	17700	17700
	PCW Inlet Temp (°C)		45	46	47.3
	PCW Outlet Temp (°C)		76.2	77.2	78.5
PCW/SW HEAT EXCHANGERS	PCW	Pcw Flow (lpm)	3540	3285	3285
		Inlet Temp. (°C)	76.3	77.2	78.5
		Outlet Temp. (°C)	45	44.4	44.8
	SW	Sw Flow (lpm)	6265	6175	6175
		Inlet Temp. (°C)	29	29	29
		Outlet Temp. (°C)	46.2	46	46.5
INTER MEDIATE HEX	PCW Flow (lpm)		-	1280	1280
	Pcw Inlet Temp. (°C)		-	77.2	78.5
	Pcw Outlet Temp. (°C)		-	65.8	78.5
SEA WATER SYSTEM	Gross Flow (lpm)		33600	35150	35150
	Inlet Temp. (°C)		29	29	29
	Outlet Temp. (°C)		45.7	45	45

CASE–I: parameters of existing Cirus reactor
CASE–II: parameters after proposed modifications
CASE–III: parameters considering non-availability of desalination unit

The required heat energy is transferred from PCW system to the intermediate circuit through a plate type heat exchanger (called intermediate heat exchanger). For this purpose PCW at a rate of 1280 lpm is tapped off from the core outlet line and supplied to primary side of the heat exchanger and outlet from the heat exchanger is connected to the main PCW heat exchanger outlet header/re-circulation pump suction header. At the rated power of the reactor, PCW enters the heat exchanger at 77°C and leaves the heat exchanger at 66°C.

The intermediate system uses de-mineralized water as the medium, which is re-circulated in a closed loop. The intermediate heat exchanger is located at the discharge of re-circulation pump to ensure that the pressure of the water in the intermediate circuit is higher than the pressure of primary system water. This ensures that in case any leak develops in the heat exchanger, active primary water will not enter the non-active intermediate circuit. The re-circulation flow through the system is maintained at 1600 lpm.

At the rated power of the reactor intermediate coolant enter the heat exchanger at a temperature of 56°C and leaves at a temperature of 65°C. The heat exchanger outlet is connected to the shell side of the heater section of the desalination unit and outlet from the heater section is led back to the pump suction. A purification circuit consisting of de-oxygenerator and mixed bed ion-exchanger units has been provided in the system to maintain the system chemistry.

2.2. DESALINATION SYSTEMS

Desalination is a process of separating dissolved salts from saline water. Many desalination technologies have been proposed based on different principles of separation. Some of them have been successfully developed and were discussed in detail in Reference [2.10], but few of them reached commercial operation. The commercial seawater desalination processes, which are proven and reliable for large-scale production of are MSF and MED for distillation processes and RO for membrane processes. Combining distillation processes with membrane processes into hybrid systems (e.g. MSF-RO or MED-RO) has certain merits where the specific advantages and disadvantages of each enable mutual compensation. Low temperature waste heat from nuclear research reactors and power plants is an abundant source of low cost energy for those desalination technologies, which can operate effectively at low temperature. Such a system is Low Temperature Evaporator (LTE) which operates at temperatures close to the reject heat temperature of a typical power plant.

Within this Coordinated Research Project (CRP), all of the above desalination systems were considered for optimization. INVAP, Argentina considered a generic MSF-BR, a generic Horizontal Tube Multiple Effect distillation (HTME) and a generic RO plants for coupling with a generic small PWR of the advanced type like the CAREM reactor during the development of the modeling tool DESNU described in Chapter-4. INET, China developed a Vertical Tube Evaporator MED (VTE-MED) with tower design for coupling with NHR-200 reactor. BARC, India designed a hybrid MSF-RO plant for coupling with an existing PHWR-220 at Kalpakkam. KAERI, Republic of Korea, designed a HTME for coupling with its SMART reactor and IPPE and other participating Russian institutions utilized the existing DOU-GTPA Russian designed HTME as well as a number of hybrid systems.

The remainder of this section provides general description of the distillation, membrane and hybrid systems considered within the CRP, with emphasis on specific design features of each project.

2.2.1. Distillation systems

In distillation processes, seawater is heated to evaporate pure water that is subsequently condensed. With the exception of mechanical vapor compression, distillation processes are driven by low-temperature fluid (below 130^0C). This fluid is generally steam, which may be taken from a power plant after partial utilization. From the beginning, distillation processes have been implemented in heat recovery chambers placed in series as a result of the high specific heat required to evaporate water. The performance of distillation processes increases with increasing number of chambers. However, the overall temperature difference between the heat source and the cooling water sink as well as economic considerations limit the number of chambers. Typical temperature differences for commercial distillation plants are 2-6^0C per heat recovery chamber.

2.1.1.1. Multi stage flash (MSF) distillation

Figure 2.18 illustrates the schematic flow diagram of an MSF system. Seawater feed passes through tubes in each evaporation stage where it is progressively heated. Final seawater heating occurs in the brine heater by the heat source. Subsequently, the heated brine flows through nozzles into the first stage, which is maintained at a pressure slightly lower than the saturation pressure of the incoming stream. As a result, a small fraction of the brine flashes forming pure steam.

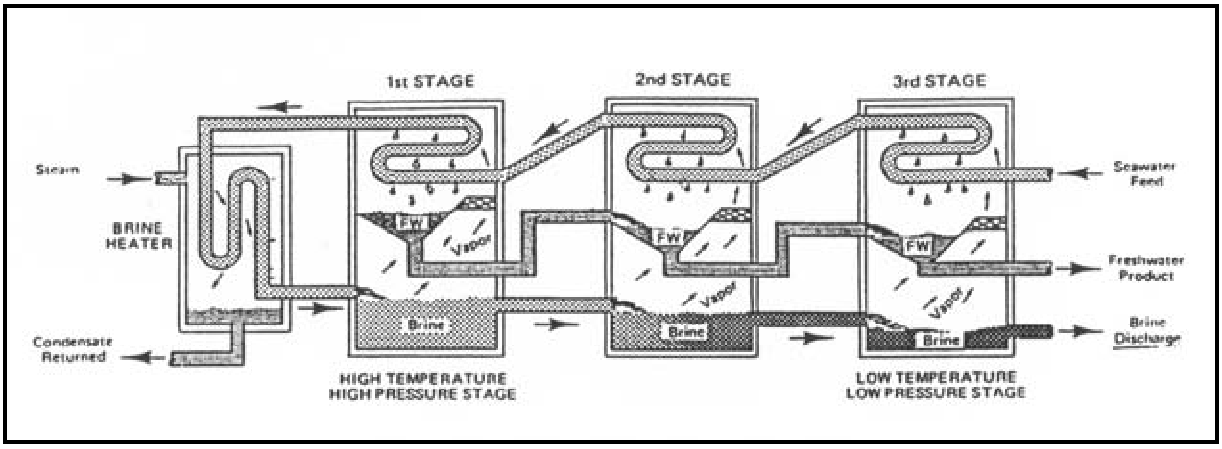


Figure 2.18: Schematic diagram of a simplified MSF system.

The heat needed to flash the vapor comes from cooling of the remaining brine flow, which lowers the brine temperature. Subsequently, the produced vapor passes through a mesh demister in the upper chamber of the evaporation stage where it condenses on the outside of the condensing brine tubes and is collected in a distillate tray. The heat transferred by the condensation warms the incoming seawater feed as it passes through that stage. The remaining brine passes successively through all the stages at progressively lower pressures, where the process is repeated. The hot distillate flows as well from stage to stage and cools itself by flashing a portion into steam which is re-condensed on the outside of the tube bundles.

MSF plants need pre-treatment of the seawater to avoid scaling by adding acid or advanced scale inhibiting chemicals. If low cost materials are used for construction of the evaporators, a separate deaerator is to be installed. The vent gases from the deaeration together with any non-condensable gases released during the flashing process are discharged to the atmosphere. There are two principal arrangements in MSF systems: the brine re-cycle mode (MSF-BR), and the once-through mode (MSF-OT). The majority of MSF plants built use the brine re-cycle mode. The brine re-cycle mode was invented in the early years of desalination when seawater corrosion materials and advanced additives were not available or too expensive. Today, corrosion resistant materials are available at reasonable costs as well as high temperature, cost effective antiscalants. Therefore, MSF-OT plants have already been successfully applied.

The efficiency of MSF systems increases with the number of flash stages. It is easier and less expensive to have a large number of flash stages in a long-tube design than in cross-tube design. In a cross-tube design, the incoming brine flows at right angles to the flashing brine whereas, in a long-tube design, the incoming brine flow path is parallel to the flashing brine. Three to five flashing stages can be accommodated in a single vessel (module). A long-tube MSF requires significantly fewer tube sheets and low pumping power.

BARC, India designed an MSF plant to produce 4500 m^3/day of very pure quality from seawater as part of the 6300 m^3/day hybrid MSF-RO to be coupled with Madras Atomic Power Station (MAPS) at Kalpakkam. The technical specifications of the MSF plant are given in Table 2.12 and the process flow sheet in Figure 2.19.

TABLE 2.12: TECHNICAL SPECIFICATIONS OF THE 4500 m^3/d MSF PLANT

(A) Product water output	187.5 m^3/h
(B) Product water salt content	<50 ppm
Sea water requirements	
(A) Total (cooling)	1544 m^3/h
(B) Makeup feed (part of A)	375 m^3/h
Maximum re-circulation brine temperature	121^0C
Blowdown temperature	40 ^{0}C (sea water)
Concentration ratio of blowdown brine	2 x the normal water concentration
Steam consumption	
(A) Heating in the brine heater	21 t/h (2.8 bar)
(B) Steam jet ejectors	400 kg/h (~7 bar)
Performance ratio (kg water production per kg steam input to the brine heater)	9
Power consumption	600 kW(e)

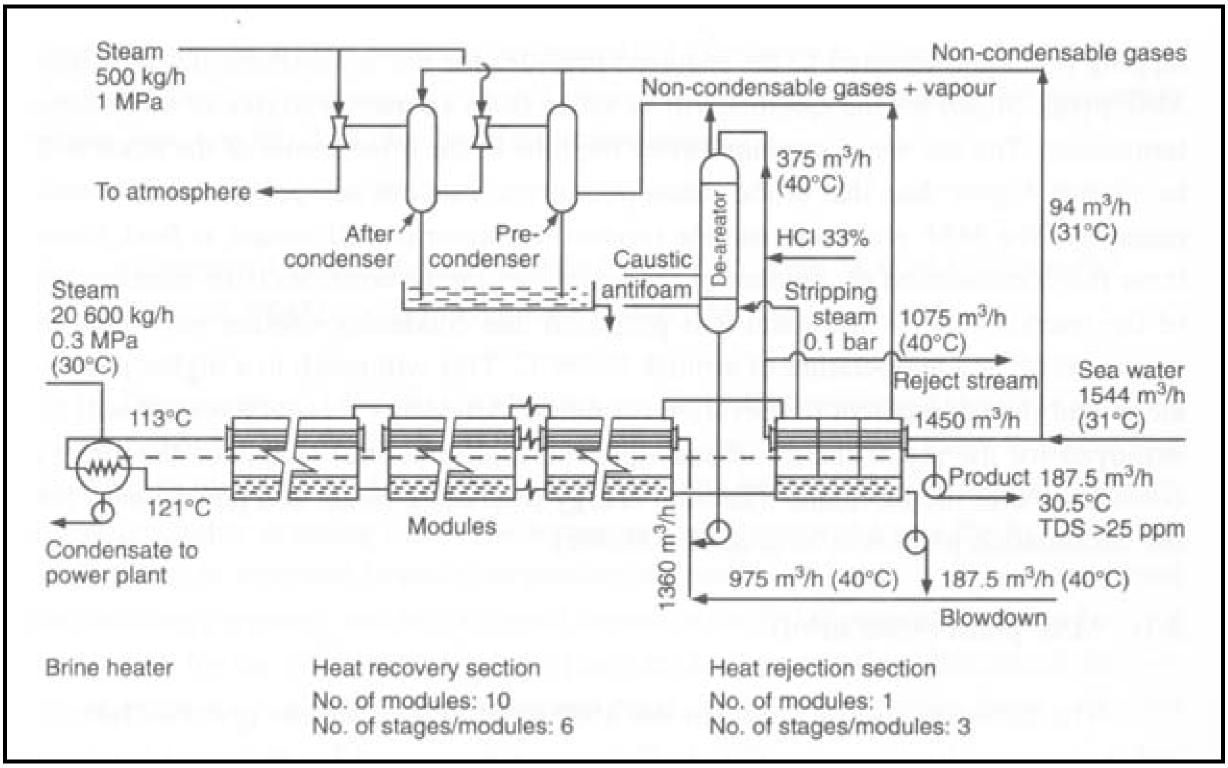


Figure 2.19: Process Flow Sheet of the 4500 m^3/day MSF Plant, Kalpakkam.

The plant is based on long tube design concept, re-circulation type with acid dosing system, top brine temperature around 120^0C and Gain Output Ratio (GOR) of 9. As per design, seawater enters the heat reject section at 30^0C and comes out at 40^0C.

The makeup seawater is first sent to a chemical pretreatment system, where acid treatment is administered to suppress the formation of alkaline scales inside the tubes of the heat transfer equipment. It is then de-aerated to reduce the dissolved oxygen and carbon dioxide in order to minimize corrosion and heat transfer. The makeup feed is mixed with recycled brine and preheated at the heat recovery stages. It is further heated in the brine heater and then subjected to a flashing process at the heat recovery and heat rejection stages, which are maintained at successively lower pressures. The vapor produced during flashing gives its latent heat to the incoming brine flowing inside the condenser tubes at the heat recovery stage. The condensate is collected as product water.

2.1.1.2. Multiple effect distillation (MED)

The MED process has fairly long history. It is the oldest large-scale evaporative process used for the concentration of chemicals and food products, and it was the first process used for producing significant amounts of desalted water from seawater. However, its large-scale application to desalination began only during the past three decades.

The essential features of MED process are: heating steam (or hot water) is fed into the first effect, causing the evaporation of brine water. The vapor formed is introduced into the following effect. In each effect steam is condensed on one side of the tube and the heat of condensation derived from it is utilized to evaporate saline water on the other side of the tube wall. Thus, the heat of vaporization imparted to the water to produce the initial vaporization is efficiently reused through the subsequent exchange of the heats of condensation and vaporization in later effects.

The pretreatment of seawater for MED plants is similar to that in MSF plants. In general, polyphosphate is introduced into the seawater feed to prevent calcium carbonate scale formation on the heat transfer tubes. A steam jet ejector vacuum system is used to remove vent gases from the deaerator and non-condensable gases evolving during evaporation from the system.

In MED systems the evaporators are placed in series, in a descending order with respect to temperature. It is possible to route the seawater feed either in the same direction (forward feed), or to route the seawater feed to enter the plant at the last effect and to progress through the evaporators in a direction opposite to the flow of condense (backward feed). The backward feed gives slightly better steam economy than the forward feed. However, in backward feed maximum brine temperature occurs at maximum brine density and thus tends to promote scale formation. Therefore forward feed is used for MED seawater desalination.

MED plants have been designed with top brine temperatures as high as 120°C. Various designs are applied in the design of MED evaporators, the best known of which are: the vertical tube evaporator (VTE) and the horizontal tube multiple effect (HTME) configurations. These are briefly described.

Vertical tube evaporator (VTE)

The configuration is shown in Figure 2.20. Vertical tubes are suspended above the brine pool, the brine is allowed to flow on the inside of the tubes while vapor is condensed on the outside. The plants are designed so that the liquid flows as a thin film on the tube surfaces, exposing large surface area for heat transfer and evaporation.

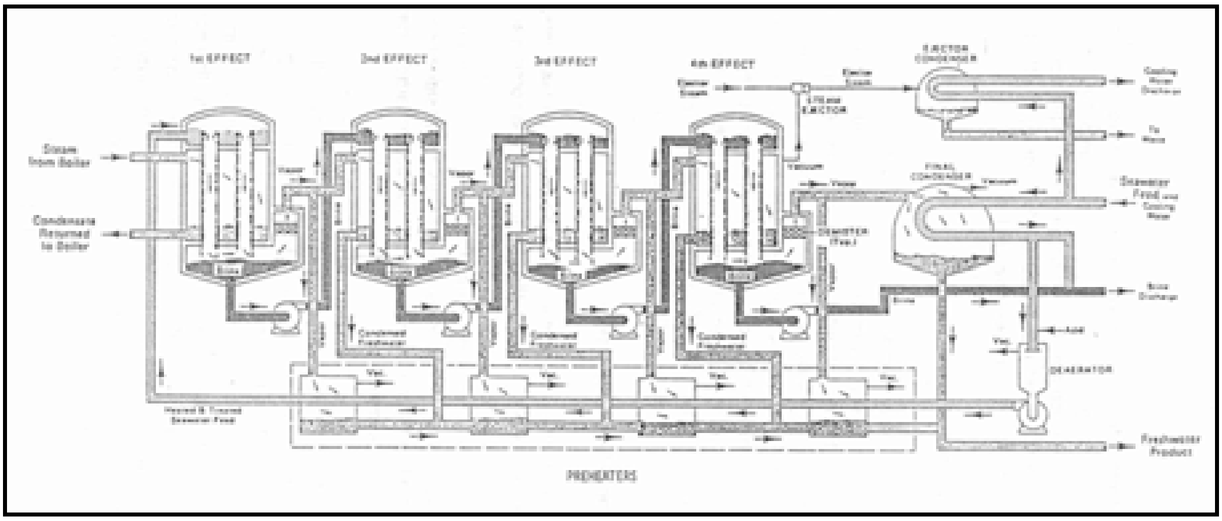


Figure 2.20: Schematic Diagram of a Simplified *VTE System.*

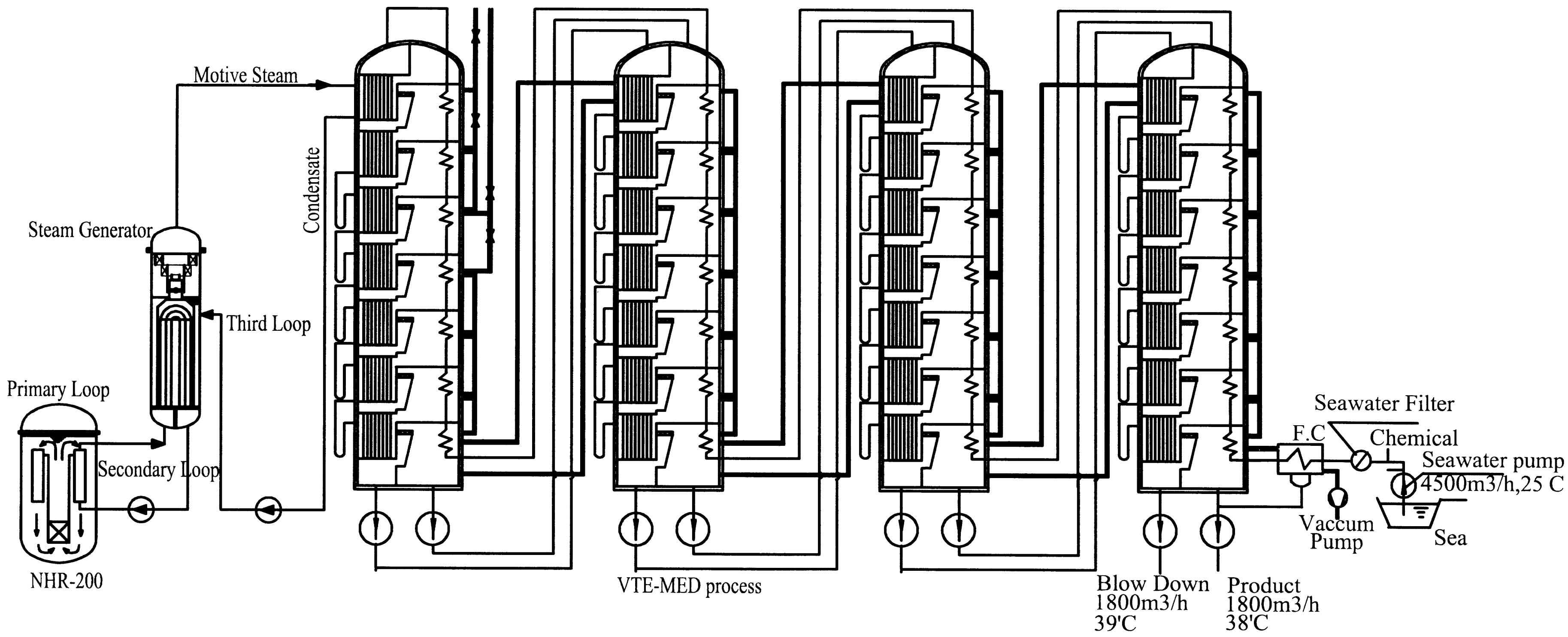
The principle of the process is to promote evaporation from the liquid's free surface and prevent nucleate boiling, and scaling. In present-day MED plants (both vertical and horizontal tube) a feed heater section is added to each effect. This reduces the amount of external steam required to heat the feed to its boiling point. All the large VTE plants that have been built use the falling brine film configuration. These require inter-stage pumps unless the performance ratio is low. On the other hand, there are some small rising film vertical-tube plants operating successfully which do not have any inter-stage pumps.

The optimum operation parameters of NHR-200 reactor for the heat only operation mode provide motive steam in the temperature range 130–150^{o}C to the desalination process. For GORs above approximate 18, the high temperature MED desalination process with Vertical Tube Evaporators (VTE-MED) is more economic. In order to minimize the number of inter-effect pumps and enhance the available gravity pressure drop, the tower arrangement of VTFE-MED with stacked evaporators was selected by INET, China.

Diagram of VTE-MED desalination system with tower design is shown in Figure 2.21. The desalination system consists of 4 identical desalination units. Each of the 4 desalination units consists of 2x2 towers with 9.0 m and 11.0 m in diameter separately and 39m in height.

Seven effects of evaporator are vertically arranged in each tower. The 4 x 4 towers are installed on a concrete platform at a height of 5 m. The inter-effect pumps, the dosing system and other auxiliaries are installed below the platform. There are a common control room and the shift personnel office between the 4 desalination units.

Seawater of 25^{o}C is mixed with sulfuric acid to remove carbonates, filtered, and then sent to the final condenser as its coolant. Then the coolant, after flowing through the preheaters stage by stage and preheated there, functions as feed water to the evaporators, with half of it being recovered as produced distillate.



Schematic diagram of NHR Heating Reactorcoupled with VTE-MED desalination process

Figure 2.21: Stacked VTE-MED Process with Tower Design Coupled with NHR-200.

The warmed, pretreated seawater is pumped upward, and flows through 7 feed water pre-heaters in the 4th tower. The seawater leaving the 4th tower is pumped upward through 7 feed water pre-heaters of the 3rd tower. In sequence, the feed water is pumped through the 2nd and 1st towers, to be gradually heated to 120°C. Then, the brine flows down vertically into the heat transfer tubes in the 1st evaporator and heats up by motive steam with temperature of 125°C and pressure of 2.32 bar. The motive steam is provided from the steam generator in the intermediate circuit of the heating reactor system. Condensate of the motive steam in the first effect is pumped back to the steam generator of NHR-200 as its feed water, therefore to finish close circulation in the motive steam circuit.

The brine, through a spray nozzle, vertically flows down into the heat transfer tubes in every evaporator, forming a thin film on the tube wall and flowing down along the tube wall. As flowing down, the brine water gradually be heated and evaporated, forming a rapidly moving steam and water foamy two-phase flow in the center of the tube. At the exit below each tube bundle the brine and steam mixture flows into a brine reservoir, its floor is just the top plate of the tube bundle of the next effect below. The brine and steam separate in the reservoir. The brine enters the tube bundle of the next effect below through its spray nozzle. The steam separated from the tow-phase flow mixture, passing through a demister, flows through a side-passage downward into the vessel of the next effect as its heating steam. The heating steam flowing around the tube, transfers heat to the brine water in next effect and condense there. The condensate as the produced fresh water collects on the lower tube sheet and passes through a trap into the next effect below. Here it flashes to regenerate part of its latent heat and thermodynamically equilibrium with the vapor in that effect.

Steam entering the effect bundle is not completely condensed there. The remainder passed directly to the vapor side of the feed preheater as its heating steam. Condensate in one stage of preheater flows to and regeneratively flashes in the next lower stage of preheater. At the bottom of each tower, it joins with the main distillate, and the brine and condensate are pumped to the top effect of the following tower. The steam is also led into the top effect of the following tower as its heating steam.

Brine leaves the 28th effect at a temperature of about 39°C and at a concentration twice that of seawater. It is pumped to the plant out-fall and finally to the sea. The combined distillate with temperature of about 38°C is pumped to the post-treatment and storage system. The main design parameters of the desalination plant are shown in Table 2.13. The main components of the desalination plant are briefly described below.

Evaporator
The design parameters of the evaporator are shown in Table 2.14. For each of the 28 effects, the double-fluted (flutes both inside and outside) stainless steel tubes are 50 mm in diameter and 3m in length. The numbers of tubes for each tower are different, 7700 tubes for the 1st tower, 9100 for the 2nd tower, 10500 for the 3rd tower and 11900 for the 4th tower. The tubes are arranged by equilateral triangle mode. And the tube spacing of the 1st and 2nd towers is slightly differed from that of the 3rd and 4th towers.

The tube sheets are of semicircle. Above the upper tubes sheet is a brine reservoir to collect the condensed brine from the effect above, below the lower tubes sheet is the disengagement zone for the vapor and brine. The vapor separates from the brine in the zone, passed through a de-entertainment device (demister) in the middle of the effect, and flows downward to form the heating steam for the bundle below.

TABLE 2.13: DESIGN PARAMETERS OF THE DESALINATION PLANT

Type of basic process	Unit	MED
Process arrangement		VTE
Arrangement of plant		Vertical
Number of units per plant		4
Number of effects per unit		28
Towers per unit		4
Total towers per plant		16
Effects per tower		7
motive steam to desalination plant	t/h	normal range 330
Parameter range of motive steam Temperature,	^{o}C	110~130
Motive steam pressure	bar	1.43~2.70
Design capacity	m^3/d	40,000
Capacity range		minimum % 40 maximum % 120
Product purity, TDS	ppm	20
Product conductivity	m.S/cm	12
Design temperature of seawater	^{o}C	25
Temperature ranges of seawater	^{o}C	minimum 15 maximum 27
Design feed temperature in the first effect	^{o}C	120
Steam temperature at entrance to the first effect	^{o}C	125
Heat input design value	MW_{th}	at startup 4× 46 after 36 months of operation MW_{th} 4×50
Seawater flow	m^3/h m^3/h	in evaporator 4×3400 to plant totally 4×4500
Seawater salinity design value, TDS	ppm	40000
Seawater salinity range, TDS	ppm	36000 ~41000
Product temperature at the design seawater temperature	^{o}C	38
Blow down temperature at the design seawater temperature	^{o}C	39
Electricity consumption of desalination plant		
Maximal consumption of desalination process per unit	MW	3.8
Maximal consumption of desalination plant	MW	15.2
Maximal specific consumption for desalination plant	kWh/m^3	2.3
Steam consumption per unit	t/h	82.5

There are two flow deflectors along the tube bundles to improve the flow of steam among the tubes. The shells of towers are welded with steel board, and there are two manholes for each effect, through which maintenance personnel can enter the evaporator to check heating tubes.

Feed water preheater

For each feed preheater, the plain stainless steel tubes are 16 mm in diameter and 4.4m in length. 7 feed preheater bundles of each tower are jointed to perform as a single long-tube heater on the seawater side with 7 regenerative heating stages on the vapor side. The design parameters of the feed water preheater are shown in Table 2.15.

TABLE 2.14: DESIGN PARAMETERS OF THE EVAPORATOR

Number of evaporators per unit	28
Number of heat transfer tubes per unit	
- tower 1	7700
- tower 2	9100
- tower 3	10500
- tower 4	11900
Diameter of heat transfer tubes, mm	50
Thickness of heat transfer tubes mm	0.35
Material of heat transfer tubes	SS1.4565
Total heat transfer area per unit, m^2	129200
Form of tubes	Double fluted
Diameter of evaporators	
- tower 1 and tower 2, m	30
- tower 3 and tower 4 m	34

TABLE 2.15: DESIGN PARAMETERS OF THE FEED WATER PREHEATER

Number of heat transfer tubes per unit	2900.
Outside diameter of heat transfer tubes mm	16
Thickness of heat transfer tubes mm	0.35
Material of heat transfer tubes	SS1.4565
Total heat transfer area per unit m^2	17900
Form of tubes	Plain

Final condenser and vent condenser

The final condenser is also working as the vent condenser. Steam from the last effect evaporator flows into shell side of the final condenser and transfers heat by condensation to the brine water which flows through inner side of heat transfer tubes. The condensate as part of the produced distillate join into the distillate system. The heated brine water flows into the final stage preheater.

Vacuum system

The vacuum system is comprised of a vent condenser and a liquid seal vacuum pump. Because the operation pressure of the first three effects is higher than that of the atmosphere, the non-condensable gases in these effects vent directly to the atmosphere. The non-condensable gases in the 4th effect to the 28th effect must be abstracted with the vacuum system due to their negative operation pressure.

Horizontal tube multiple effect evaporator (HTME)

A conceptual diagram of a horizontal tube multiple effect (HTME) configuration is shown in Figure 2.22. Although the principle of operation is the same as for the VTE, the brine and steam are applied on opposite sides of the tubes in the two systems. The brine is distributed as a film on the outside of the tubes where it is partially evaporated by heat derived from condensation of vapor (to fresh water) on the inside of the tubes. The effects are amenable to being stacked both vertically and horizontally.

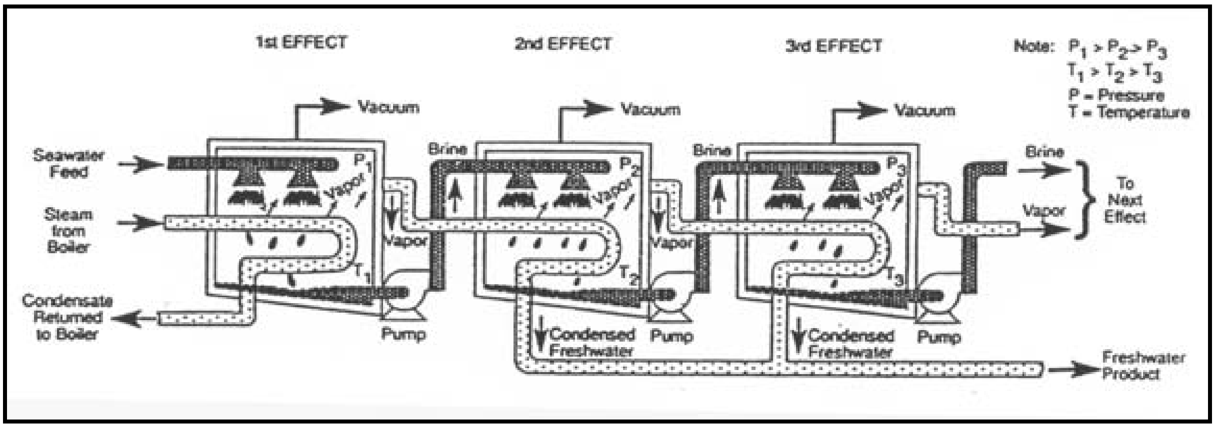


Figure 2.22: Schematic Diagram of a Simplified *HTME System.*

The configuration of the vertical arrangement permits greater use of gravity to move liquid between effects. Limited operational experience with HTME indicates that scale formation and removal is significantly less problematic than in VTE.

Comparison between HTME and VTE-MED plants indicated that HTME plants have better heat transfer in the temperature range 20–70^{0}C, while VTE plants have better heat transfer above 70^{0}C. KAERI carried out an economic evaluation of different coupling schemes with SMART reactor to produce 40,000 m^3/day of desalted water. The results of the economic evaluation indicated that optimum economics are obtained when top brine temperature (TBT) is in the range 60–70^{0}C. Therefore, Low Temperature HTME (LT-HTME) process was selected for coupling with SMART. A Thermal Vapor Compression (TVC) through steam ejector was added to the MED process to improve the efficiency of the distillation plant.

Korean LT-HTME/TVC

The desalination plant coupled with SMART consists of 4x10 000 m^3/day LT-HTME with TVC. Figure 2.23 shows the flow diagram of the selected desalination process. Major design parameters are summarized in Table 2.16. The main components of the desalination plant are briefly described below

Evaporator

The Evaporator is designed for the Performance Ratio of eight to one (8:1), i.e. eight (8) kg of distillate will be produced per 2326 kJ of steam enthalpy entering to the thermo-vapor compressor.

The evaporator tubes are horizontal, straight tubes with vapor flowing through the tubes in two passes. The feed/brine are distributed across the top of the evaporator bundles by even spray distribution. Since the top brine temperature is 6 500, the evaporator is designed such that the inlet steam saturation temperature not exceed 7 000.

The spray nozzles are accessible from outside the evaporator effects and are arranged to be readily removable for inspection and cleaning. A strainer is installed to the plant to limit the total seawater supply or total feed water supply with openings of not more than *25%* of the nozzle diameter. The feed loading across the evaporator bundles is in the range of 2,270 to 4,135 kg/hr/in^2.

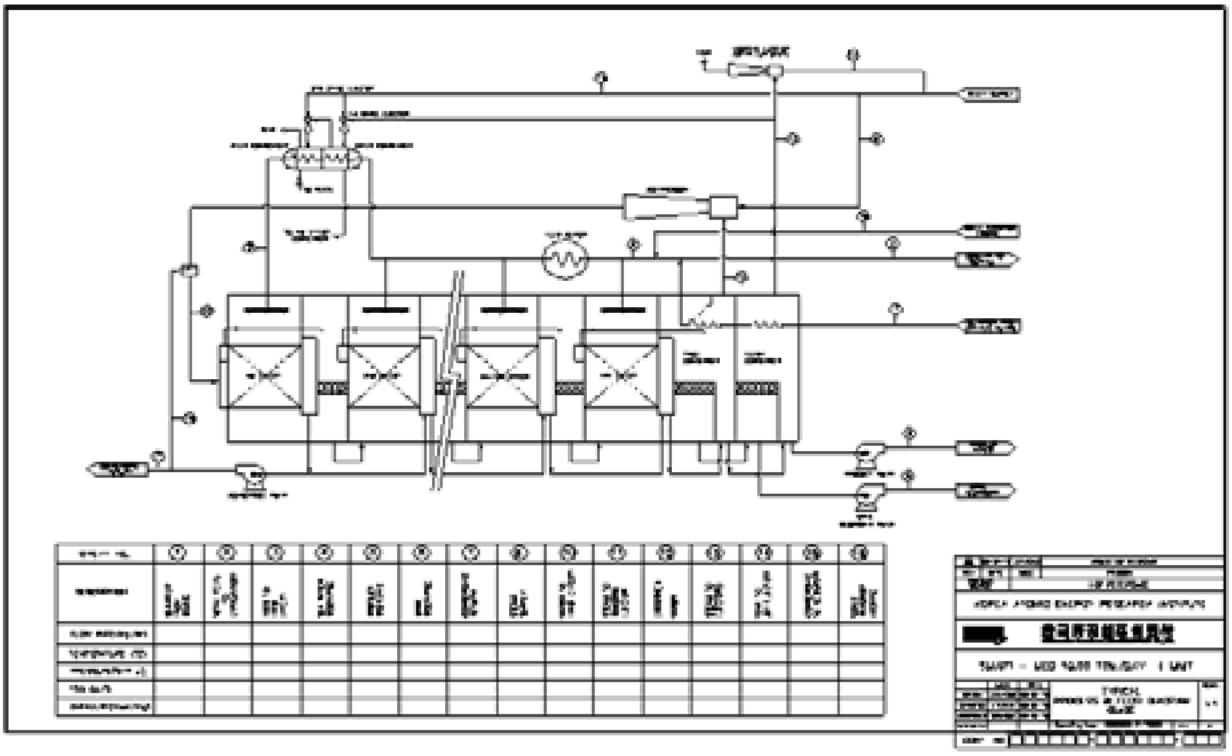


Figure 2.23: Flow diagram of MED-TVC Plant Coupled with SMART Reactor.

Tubes are made of Aluminum Brass IJNSC 68700 with a minimum wall thickness of 0.7 mm and minimum outer diameter of 16 mm and expanded into tube sheets of the same material. The top two rows of evaporator bundles are made of titanium with 0.5 mm minimum wall thickness. The vapor velocities are limited the maximum velocity of 60 m/s and the minimum of 3.0 m/s.

Air ejectors

Air ejectors are provided to maintain vacuum in the evaporator and to remove the non-condensable gas produced in the evaporator. Also, the ejectors are used to decrease the boiling temperature of the water to be treated.

The main steam air ejector system consist of a two (2) stage steam air ejector system equipped with twin element ejector (2 x 100%) per each stage and one inter-condenser and one after-condenser. The steam air ejector system is designed so that non-condensable gases can be removed efficiently under any operational condition of scale control agent, namely, high temperature threshold additive. The following non-condensable gases have been considered in sizing the equipment;

- An air leakage rate of twice of the expected design leakage of the evaporator shell and inter-connecting piping.
- Gases released from steam into the evaporator.
- 150% of the maximum calculated quantity of non-condensable gases released from the feed water.
- 2% of vapor extraction from flash condenser.

TABLE 2.16: MAIN DESIGN PARAMETERS OF MED-TVC UNIT

Parameters	**Data**
Type of the desalination process	MED-TVC
Number of units per plant	4
Arrangement of plant	Horizontal
Number of effects	9
Water production capacity per unit plant	10,000 m^3/day
Total water production capacity	40,000 m^3/day
Minimum controllable unit output	5,000 m^3/day
GOR (Gain Output Ratio)	19.6
Design temperature of seawater	3300
Maximum seawater salinity	45,000 ppm
First effect maximum brine temperature	65^{0}C
Last effect minimum brine temperature	47.9^{0}C
Vapor temperature in the first effect	69.3^{0}C
Vapor pressure in the first effect	0.25 bar
Vapor temperature in the last effect	479^{0}C
Maximum distillate temperature	43^{0}C
Total seawater flow per unit	3,234 m^3/hr
Brine blow down flow rate	1,010 m^3/hr
Brine blow down temperature	41^{0}C
Steam pressure to thermo-compressor	8.0 bar
Steam temperature to thermo-compressor	170.6^{0}C
Suction vapor pressure to thermo-compressor	0.11 bar
Suction vapor temperature to thermo-compressor	47.6^{0}C
Suction vapor flow to thermo-compressor	26,041 kg/hr
Delivery flow from thermo-compressor to the first effect	52,299 kg/hr
Distillate pH at evaporator outlet	6~8
Maximum TDS of distillate at evaporator outlet	25mg/L
Scale control dosing rate related to feed make-up	5.61 mg/L
Total steam consumption per unit	26,258 kg/hr

Steam jet ejector (Thermal Vapor Compressor)
The thermal vapor compressors consists of three parts; a motive steam nozzle, a mixing chamber and a diffuser and designed to be able to operate with a 10% reduction of applied steam pressure at reduced output without “breaking back”. Two jet vapor compressors operating in parallel are provided. The vapor compressors deliver the rated vapor flow at the pressure difference required between the final effect condenser and the first effect evaporator at rated fouling factors. Operation at reduced output is controlled by throttling either the steam supply or vapor output. The components of thermal-vapor compressor, jet nozzles, bodies and diffusers, are designed with 316L stainless steel.

The limit of the compression ratio, for a single jet stage, is dependent on the motive steam pressure and suction pressure. The maximum ratio is about 15, but the optimum ratio for saving of the motive steam is often considerably less. Therefore, the steam jet systems designed with the multiple stages connected in series. The number of stages depends, primarily, on the overall compression ratio and the energy saving needed.

Condensers
The total steam consumption is considerably reduced if some of the steam in vacuum system is condensed. The pre-condensers are used to reduce the flow to the vacuum system by condensing as far as possible the water vapor associated with non-condensable gases. The inter-condensers, where steam condenses, follow each stage of ejectors. The non-condensable part of inlet mixture, coming out from each inter-condenser, is sent to the next stage of ejector system. The last condenser is called the after-condenser and non-condensable gases are discharged to the atmosphere.

Pre-condenser and inter-condenser are of direct contact or surface type. Multi-stage jet ejectors have inter-condensers to take advantage of energy savings. Inter-condensers are used whenever the inter-stage pressure reaches the dew-point of the discharge vapor from the previous stage, so that cooling water can be used for condensing. Heat is removed from the process gases by direct contact or mixing with the cooling medium.

After-condensers are installed after the last stage of a steam jet ejector system. After condensers are used to minimize the steam discharge to the atmosphere. They further reduce the amount of discharge of any condensable from the process into the atmosphere. After condensers also reduce the noise level of the discharge from the last effect by installing a muffler or “silencer” at the stage discharge.

Process pumps
The process pumps of MED-TVC unit are composed of the brine blow-down pump, distillate extraction pump, condensate extraction pump and desuperheater spray water injection pump.

Brine blow-down pump
Each unit has one brine blow down pump, which is a full duty and electric motor driven horizontal centrifugal pump. This pump extracts the concentrated brine from the evaporator under vacuum condition and discharges the brine to the sea. The materials for brine blow-down pump casing, impeller and shaft are fabricated 31 6L stainless steel.

Distillate extraction pump
Each unit is has two full duty (100% capacity) distillate extraction pumps. These are electric motor driven, horizontal, centrifugal pump. In operation, one pump used at full duty and one pump is on stand-by. This pump extracts the distillate from the distillate duct of the flash

condenser under vacuum condition and pumps the distillate to the product water tanks. The 316L stainless steel was selected as a construction material for distillate extraction pumps.

Condensate extraction pump
One full duty motor driven horizontal, single stage pump is used for condensate extraction from the first effect. This pump delivers the steam condensate to the condensate return tanks by taking suction from the first effect.

Desuperheater spray water injection pump
One pump is in operation at full duty and one pump is on stand-by. An electric motor driven, horizontal, single stage pump takes suction from the condensate return line and delivers the steam condensate to the desuperheater. The pumps are constructed of the same material as that of the distillate extraction pumps.

Desuperheater
The steam atomizing desuperheater reduces the steam temperature without the pressure loss. The desuperheater speeds up the evaporation and absorption of cooling water by the powerful preheating atomization of high-speed steam jets. The nozzles arranged in a ring form and faced downstream. The direction of each injection is in parallel with the main steam pipe axis. The cooling water enters the rear center of the head and flows down to the nozzle through a number of separate holes. The atomizing steam comes from the high-pressure source and is drawn into the main steam pipe. The atomized cooling water is sprayed into the main steam pipe. Then, the steam temperature is reduced due to the high-speed vaporization of the sprayed water.

Russian DOU_GTPA HTME

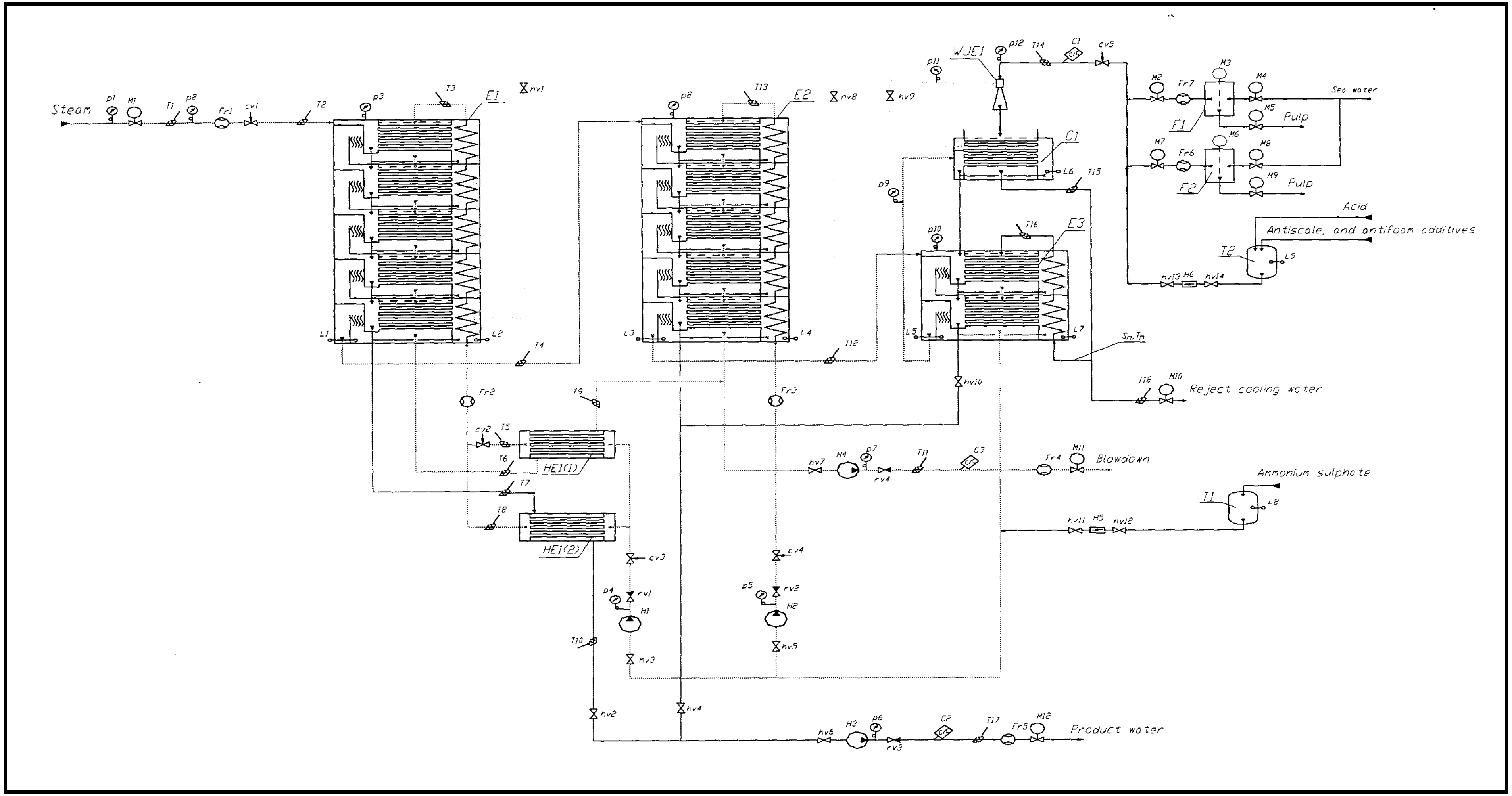
Russian DOU_GTPA HTME desalination units of rated capacities ranging from 240 to 20160 m^3/day (10 to 840 m^3/hour) have been developed by Sverdlovsk Research Institute of Chemical Engineering in Ekaterinburg city and are manufactured commercially in Russia. DOU_GTPA units are multi-column plants with 2 to 8 stages in each column, based on horizontal tube film evaporators. The number of columns and stages in DOU_GTPA is calculated and typically depends on the location and the allowable height of the plant. The Russian institutions participating in this CRP considered various sizes and combinations of DOU_GTPA units for coupling with the Russian reactors described in the previous section. A diagram of the distillation DOU GTPA plant based on horizontal tube film evaporators and 600-m3/h capacity is shown in Figure 2.24. Characteristics of some Russian DOU GTPA's are given in Table 2.17.

DOU GTPA is operated in the following way (a three-column variant with 12 evaporating stages is considered in Figure 2.24). Seawater passes through the mechanical filters F1 and F2, then, after chlorination, through the water-jet ejector WJE 1 comes into the condenser C1, where the seawater is heated up with the steam of the last evaporating stage of the last column E3. A part of the seawater flow after the condenser C1 is discharged into the sea, and the rest of the flow is directed to deaeration and desalination into the column E3, the column consisting of two evaporating stages. The water sequentially passes through the regenerative heaters installed in the evaporating stages of the column E3, where it is heated up with the steam of the corresponding evaporating stages. From the last highest-temperature heater the seawater enters the first stage of the column E3.

TABLE 2.17: CARACHTERISTICS OF SOME RUSSIAN DOU_GTPA PLANTS

	DOU-200	DOU-400	DOU-700
DISTILLATE			
Capacity, m^3/hour	200	400	700
Seawater temperature)	30	30	30
Salt content, mg/l	2	2	5
Pressure, MPa	0.4	0.4	0.6
GOR	14.3	15.4	15.6
INITIAL SEAWATER			
Pressure, MPa	0.2	0.2	0.2
Calculated initial seawater temperature, ^{0}C	25	25	25
First stage boiling temperature, ^{0}C	90–100	90–100	90–100
Flow rate, m^3/hour	550	1110	1940
For desalination, m^3/hour	400	800	1400
Seawater desalination factor	1.6–2.5	1.6–2.5	1.6–2.5
STEAM			
Pressure, MPa	0.15–1.4	0.15–1.4	0.15–1.4
Flow rate, ton/hour	14	26	45
CONDENSATE			
Pressure, MPa	0.5	0.5	0.6
Temperature, ^{0}C	40	30	35
ELECTRIC POWER			
Current, voltage, frequency			
Consumption for desalination, kW	320	840	910
PLANT DIMENSIONS			
Length x width x height, m	9 x 10x 16.5	21 x 26 x 21	38 x 26 x 28
FULL PLANT MASS			
Dried, tons	230	550	1000

There the seawater is uniformly distributed along the tubes of the upper tube bundle. Further the water in a film form goes on the outer sides of the lower horizontal heat-exchanging tubes heated with secondary steam of the 10th and 11th evaporating stages correspondingly. Then the seawater is stripped of corrosive gases in the E3 column and deoxygenated with the use of ammonia sulfate. Seawater is then directed, as two parallel flows, through the regenerating heat exchangers system into the first stages of the columns El and E2. Each column is equipped with five horizontal-tube film-evaporating stages. Prior to entering the El column, the seawater is heated in the HE 1(1) and HE 1(2) heat exchangers with distillate and brine flows leaving that column. In both columns El and E2, the seawater evaporates, sequentially going from the first/upper stage down to the lowest/last one. After the last evaporating stage of the El column, the brine, cooled in the shell-and-tube regenerating heater HE 1(1), is mixed with the brine from the lowest evaporating stage of the column E2 and then discharged into the sea area remote from the seawater intake where it can not blend with the initial seawater.



E1-E3=Evaporators C1=Condenser WJE1=Water-jet ejector HE1-HE2=Heat exchangers F1-F2=Filters T1-T2=Tanks H1-H6=Pumps

Figure 2.24: Diagram of DOU_GTPA Distillation Plant.

Heat for the seawater distillation desalination is transferred to the DOU plant with the steam produced by a nuclear reactor plant or a thermal power plant. That steam heats up the first evaporating stage of the first column El. There, due to the steam condensation in the horizontal heat-exchanging tubes, the secondary steam is produced from the seawater, which is used later on as the heating steam in the second lower evaporating stage. The working temperature of the stage is lower than the first stage temperature. Therefore, the heat for the desalination process is transferred from the first stage to the last one in each of evaporating stage columns. From the last stage of the column El the secondary steam goes into the first stage of the column E2, and from the last stage of the column E2 the steam is discharged into the first stage of the column E3. The secondary steam of the last stage of the column 143 is fed into the condenser C1.

The distillate produced in the each evaporating stage by the condensation of the preceding stage secondary steam, flows from the upper stage to the lower one. From the last stage El the distillate is discharged through the shell-and-tube regenerating heat exchanger HE 1(2) to be blended with the distillate flows of the columns E2 and E3 and of the condenser C1. The distillate blend is pumped with the P3 pump to the consumer. Non-condensable gases leave the plant through the water jet ejector WJE. The working fluid of the ejector is the initial seawater.

The heat exchange surfaces of the equipment are protected against scaling by adding the antiscalant PAF- 1 3A, IOMS or CK- 110 to the initial seawater before the condenser K1. The type of the scales depends on the composition of the seawater. If required, to increase the quality of produced distillate a foam suppressor is used along with the antiscalant.

The heat exchange surfaces of the equipment are cleaned of contamination and scaling at regular intervals while running (i.e. during 8800 hours of uninterrupted work of the plant). Each evaporating stages column is cleaned separately, once or twice a year by adding chlorine acid, sulfuric acid to the initial water, and also under stable conditions, when the plant is in standby mode, in 8800 hours with fresh-water solution of the same acids. The used acids solution is sodium alkali neutralized and then discharged into the sea remote from the water intake such that blending with initial seawater is prevented.

2.1.1.3. Low Temperature vacuum evaporator (LTE)

BARC, India designed and manufactured a Low Temperature vacuum Evaporator (LTE) plant, which can utilize low energy sources at low temperature for the desalination of seawater. The LTE desalination plant was coupled to CIRUS nuclear research reactor at BARC. The plant is operated at hot water temperature ranging from 50 to 65^{0}C. The desalted water produced from this plant is supplied to CIRUS for use as make up demineralized water after polishing. The LTE desalination plant produces 30 m^3/day of fresh water, having less than 10 ppm TDS and silica content as low as 190 ppb, from seawater without any pretreatment of the feed. The LTE and requires about 800 kW of heat energy.

The unit consists of a heater, a vapor separator and a condenser (Figure 2.25). Product water is produced by evaporation of seawater flowing in the tube side of the heater section, by the heat supplied from the hot water flowing in its shell side. The water vapor is condensed and removed from the condenser section located above the heater. The LTE unit is maintained at a vacuum of about 710 mm of Hg to facilitate evaporation of water at low temperature of about

41^{0}C. The seawater before entering the heater section is pre heated in the condenser, which in turn condenses the steam produced in the heater.

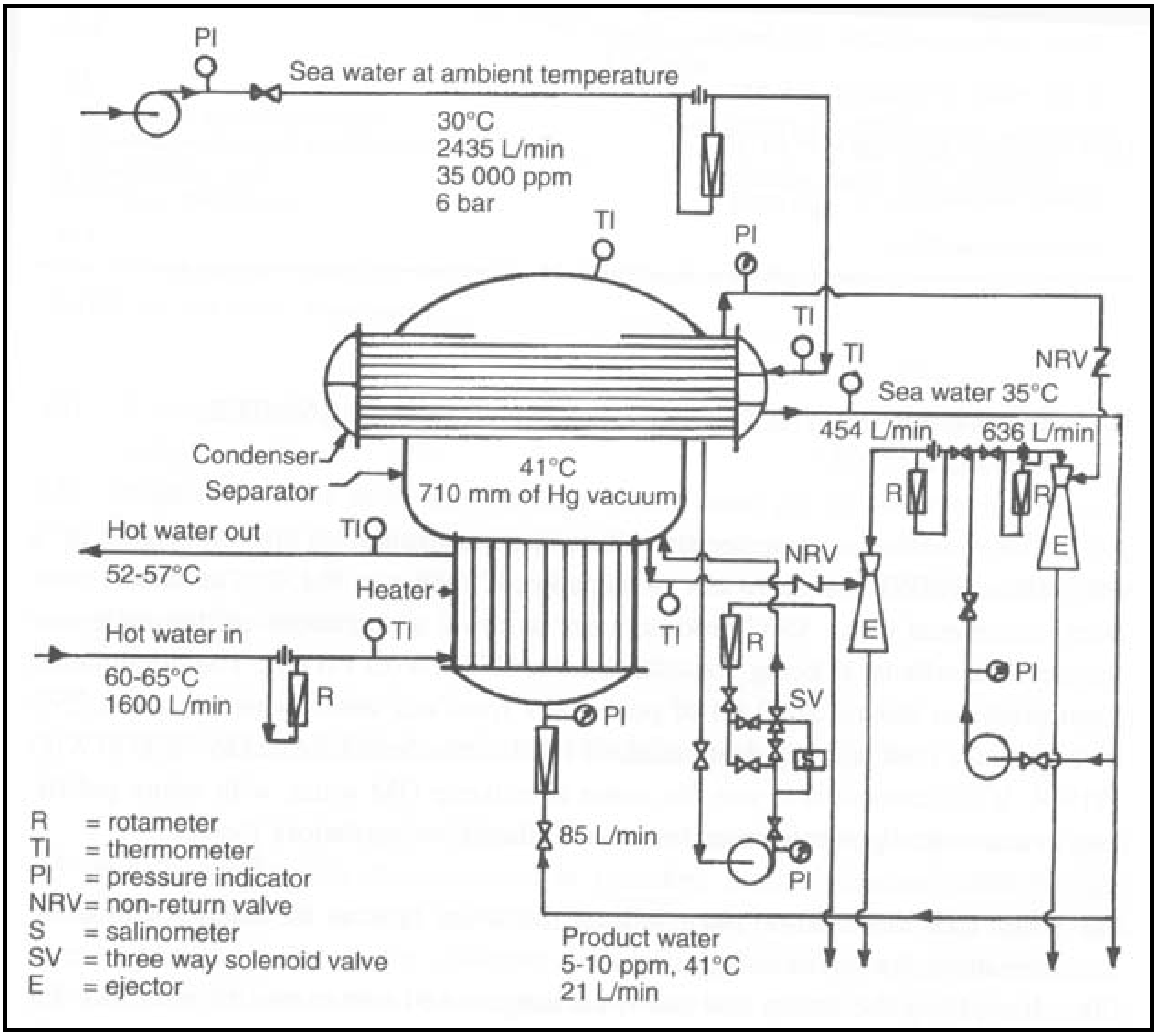


Figure 2.25: Schematic flow diagram of LTE Desalination System.

Seawater jet ejectors are used to create vacuum in the LTE unit as also to remove un-evaporated seawater (brine concentrate) from the outlet of heater section. The product water is taken out from the base of condenser with the help of a pump. The main design parameters of the LTE unit are listed in Table 2.18.

2.2.2. Reverse osmosis (RO)

Osmosis is the natural process by which water flows through a semi-permeable membrane from pure or dilute solution to a more concentrated solution. The flow continues until the resulting osmotic head equals the osmotic pressure of the solution. If a pressure higher than the natural osmotic pressure of the solution is applied to it, the direction of the water flow is reversed.

The solution becomes more concentrated, and purified water is obtained on the other side of the membrane, hence the term Reverse Osmosis (RO). The RO desalination process was developed in the USA in the 1960s, and a first test plant was built in 1965. Since its commercial application to seawater in 1970, plants of larger and larger capacity have been designed, constructed and operated successfully.

TABLE 2.18: MAIN DESIGN PARAMETERS OF BARC LTE PLANT

Product water output, m^3/h	1.25
Seawater salinity, ppm	35000
Product water quality, ppm	20
Total seawater requirements for cooling, m^3/h	144
Feed seawater requirements (part of total requirements), m^3/h	4.5
Hot water temperature, ^0C	65
Operating pressure, bar	0.08
Waste heat requirements, MW(th)	1
Electric power consumption, kW(e)	
If seawater is available at 6 bar	2.25
If seawater is available at 1 bar	15
Heat transfer coefficient, $kW.m^{-2}. K^{-1}$	
Heater section	0.8
Condenser section	1.0

An RO desalination plant consists mainly, as shown in Figure 2.26, of a pre-treatment section, a high-pressure pump section, a membrane module section, and a post-treatment section. In general all modern large-scale RO plants use power recovery turbines where the pressure of the concentrate is utilized to reduce the overall power consumption of the system.

Membranes can be sensitive to pH, temperature, chemicals, etc., and are highly sensitive to fouling and clogging. Proper design of the system and pre treatment of the water can minimize these problems and hence protect the membranes.

The principles governing the operation of RO plants are:

1. The water flux through any given membrane is proportional to the effective pressure difference across it.
2. Salt also diffuses through the membranes. This diffusion is independent of the pressure difference, and depends only on the difference in concentration between the feed and product waters and the nature of the membrane.

In the process, saline water is pumped to pressurize it against a membrane in a container. As desalted water from the feed solution passes through the membrane, the remaining solution becomes more concentrated. A valve allows portion of the feedwater to be discharged without passing through the membrane. Without this discharge (or blow down) the concentration of dissolved salts in the feed solution would continually increase, requiring the pump to add ever-increasing energy to overcome the increased osmotic pressure, and precipitation of supersaturated constituents in the brine would occur.

The product water emerging from the membrane assembly generally needs some type of post-treatment before being distributed as potable water. Such post-treatment includes pH adjustment, usually by the addition of a base, removal of dissolved gases such as H_2S and CO_2 by air stripping, and/or disinfecting.

An advanced concept for the design and operation of RO seawater desalination systems was developed by CANDESAL, Canada. Some of the key features of this advanced approach to reverse osmosis system design and operation are:

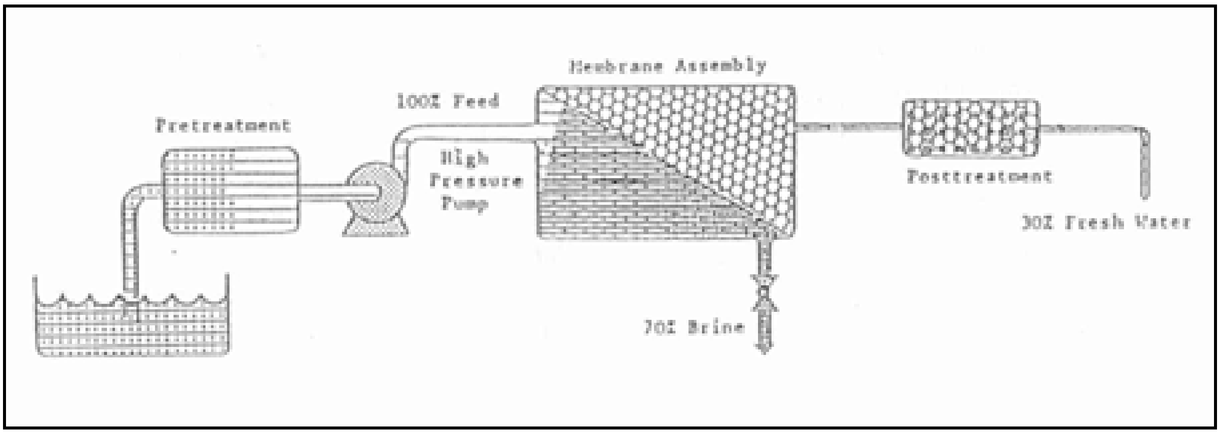


Figure 2.26: Simplified flow diagram of RO System.

Preheating of the feed water

This concept utilizes low grade heat that is otherwise lost in such essential but environmentally unfriendly power production elements as condenser cooling systems where very large quantities of water, typically 10 degrees and more above ambient, are discharged into the environment. The operation of RO systems at higher temperatures increases the efficacy of production. Cost savings are found at all temperatures where waste heat can be used to preheat the feed water. But the overall savings depend on a number of factors that is site specific including the TDS of the feed water, the size of the plant, etc. A criticism of preheated feed water that has sometimes been expressed is the fact that, as the temperature goes up, the TDS of the product water rises. This is of course correct if all other factors remain the same; however, this effect is manageable with correct system design.

Operation at high feed water pressure

High-pressure operation is another key feature of this advanced design concept. While many RO system designers and membrane manufacturers themselves will advocate use of lower pressures to save energy costs in water production, operation at higher pressures leads to significant water production increases. Analytical studies carried out by CANDESAL have shown that the apparent increase in pumping costs due to the higher pressure is offset by increased water production and is ameliorated further by the use of energy recovery. CANDESAL studies has also shown that as the feed pressure increases, with all other factors remaining the same, the TDS of the product water decreases [2.10].

The use of ultra filtration as an effective pretreatment method

Another key feature is the use of ultra-filtration (UF) pretreatment of the feed water rather than the extensive use of mechanical filtration and chemicals. Chemical pretreatment is not only a major cost in the operation of an RO plant, it is also a source of environmental pollutants. Through the use of UF membranes, the system uses a minimal amount of chemical pretreatment and thus lowers the cost of water production, and the associated environmental burden, significantly over time. A second and equally important benefit of UF pretreatment is the broadening of the acceptable performance envelopes for the RO membranes that accrues with the use of very clean feed water that has not been subject to chlorination, has a very low silt density and turbidity, and is virus and bacteria free.

Energy recovery
Energy recovery is used in the system design to take advantage of energy that would otherwise be lost by the direct rejection of the high-pressure brine. Significant progress has been made in energy recovery systems in recent years and it is expected that progress in this area will continue to reduce the cost and raise the efficiency of these systems. Today there are a number of these products on the market that could be incorporated in a plant to increase the level of cost saving.

Site specific optimization
The importance of site-specific optimization cannot be over-emphasized and is a key component in this advanced RO system design. Many factors influence the ultimate cost of the product water and need to be considered when designing a plant. This stresses the importance of considering all of a site's characteristics when developing and optimizing a plant design, (e.g. the make-up of the feedwater, the availability of waste heat, seasonal and other variations in feedwater quality and temperature, the owner's specific needs with respect to water production minimums and maximums, system management requirements. and so on).

Management and control system
One of the most important aspects of the design of an advanced RO desalination plant where minimum life cycle water costs are the goal is the design and configuration of the Management and Control System. Not only does the Management and Control System remove the task of the general determination of optimal operation conditions for the plant from the operators, it also, by being a real time system monitor, allows for the minute by minute adjustment of the system operating parameters to optimally accommodate varying conditions in the feed stream. This real-time capability is especially important on large plants (above 100,000 m^3/day) located on river estuaries and other locations where tidal action causes very large changes in the diurnal levels of dissolved solids, turbidity and silt density.

BARC, India designed an RO plant to produce 1800 m^3/day of very pure quality from seawater as part of the 6300 m^3/day hybrid MSF-RO to be coupled with Madras Atomic Power Station (MAPS) at Kalpakkam. The technical specifications of the RO plant are given in Table 2.19 and the process flow sheet in Figure 2.27.

TABLE 2.19: TECHNICAL SPECIFICATIONS OF THE 1800 m^3/d RO PLANT

(A) Product water output	75 m^3/h
(B) Product water quality	550 ppm TDS
(A) Sea water requirements	214.3 m^3/h
(B) Sea water TDS	35 000 ppm
% recovery	35%
Membrane element	8040-HSY-SWCI
Average salt rejection	98.5%
Operating pressure	58 bar
Operating temperature	36-38 ^{0}C
Energy consumption	6 kW.h/ m^3

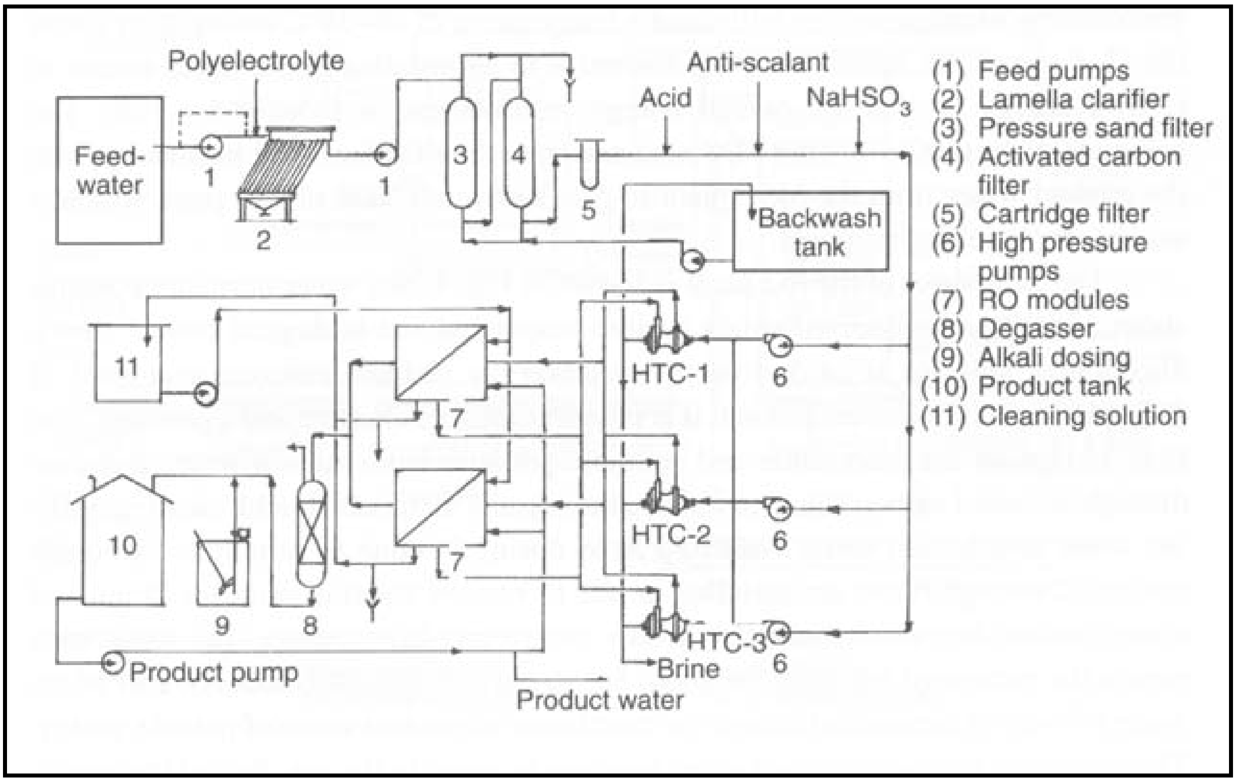


Figure 2.27: Process flow sheet of the 1800 m^3/day RO Plant/

The plant is based on advanced thin film composite membranes that are capable of desalting seawater to produce less than 500 ppm total dissolved solids (TDS) product water. These membranes can withstand a temperature of 36–38^{0}C during their entire life (3–4 years). An energy recovery system is installed in the reject stream to recover energy so that the overall energy consumption is reduced by 30%.

Seawater normally contains about 35 000 ppm dissolved solids besides suspended and biological matter. Shock injection of chlorine is carried out to maintain the residual chlorine at a level of 2–3 ppm to prevent marine growth. It is passed through a clarifier and a pressure sands filter to remove the suspension and colloidal particles from the seawater, and then through activated carbon filters to remove the organic materials. Dechlorination of the seawater is achieved using $NaHSO_3$. Acid dosing is done to minimize carbonate scaling. Cartridge filters are installed in line to remove the fine particles (5 pm and above) before they reach the high-pressure pumps and RO modules. The RO system pumps the pretreated seawater (at about 50–70 bar) through RO modules. Part of the water (30–40%) permeates through the membrane as product water of potable quality. The remaining concentrated seawater emerges as water to the sea. Part of the energy is recovered from the reject seawater by the energy recovery turbine, reducing the specific energy consumption. After degassing, the product water is dosed with soda ash or lime to adjust the pH.

2.2.3. Hybrid systems

There are many ways to improve the efficiency and cost of product water of desalination plants. One mean is to combine two or more desalination systems resulting in a hybrid plant. Hybrid systems can offer performance improvement, savings in pre-treatment and overall cost reduction. A number of hybrid desalination systems have been proposed over the past years such as [2.6]:

- Combination of RO with distillation processes.
- Combination of vapor compression (VC) with distillation processes.
- Combination of RO and VC processes.

In order to gainfully employ the years of experience and expertise in various aspects of desalination activity, BARC (India) has undertaken installation of a demonstration scale hybrid MSF-RO desalination plant coupled to 220 MW(e) PHWR station at Kalpakkam, Madras in the Southeast coast of India (presently operating at 170 MW(e) only). The hybrid plant will have a capacity of 6300 m^3/d, and will thus meet the dual needs of 1000 m^3/d process water for the nuclear power plant and drinking water for the neighboring people.

The hybrid plant aims for demonstrating the safe and economic production of good quality water by nuclear desalination of seawater. It comprises a 4500 m^3/d MSF plant (described in section 2.1.1.1) and an 1800-m^3/d RO (described in section 2.2.2) plant. The MSF section uses low-pressure steam from Madras Atomic Power Station (MAPS).

The objectives of the hybrid MSF-RO are as follows:

- to establish the indigenous capability for the design, manufacture, installation and operation of nuclear desalination plants
- to generate necessary design inputs and optimum process parameters for large scale nuclear desalination plant
- to serve as a demonstration project to IAEA welcoming participation from interested member states.

The hybrid plant is envisaged to have a number of advantages:

- A part of high purity desalted water produced from MSF plant will be used for the makeup demineralized water requirement (after necessary polishing) for the power station.
- Blending of the product water from RO and MSF plants would provide requisite quality drinking water.
- The RO plant will continue to be operated to provide the water for drinking purposes during the shutdown of the power station.

The hybrid MSF-RO requires around 2000 m^3/hr seawater. Process cooling water from MAPS out-fall seal pit is used as a source of seawater supply for hybrid MSF-RO. The out-fall seawater temperature varies from 26–32^{0}C which is about 4^{0}C higher than ambient temperature due to heat being rejected from the process cooling water system. The cooling water discharge has no debris since the intake water passes through the trash rack and travelling water screens. It is reported to have less bio-fouling potential.

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 the nuclear reactors. The moisture content will be removed through a moisture separator and steam will be sent to intermediate isolation heat exchanger (IHX) to produce process steam (using demineralized 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 utilized 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, possibility of an exclusive intake for supply of raw seawater is also being investigated. The return stream of cooling seawater from the reject stages of MSF plant will be blended with raw sea water to bring down the temperature to 36–38^{0}C before the same is sent to the pre-treatment section of RO module.

In addition to the Indian hybrid nuclear desalination plant, currently under construction, other hybrid desalination systems have been investigated in the preliminary stages of other projects within this CRP. INET, China considered the coupling of the hybrid plants (LT-HTME/RO) and (LT-HTME/VC) before deciding on VTE-MED. IPPE and other Russian institutions participating in the CRP considered a hybrid (LT-HTME/RO) among many alternatives for coupling with KLT-40C reactor and hybrid (LT-HTME/RO) for coupling with NIKA-70 reactor. These are discussed with other coupling schemes in Chapter 3.

REFERENCES TO CHAPTER 2

[2.1] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidebook on the Introduction of Nuclear Power, Technical Report Series No. 217, Vienna (1982).

[2.2] INTERNATIONAL ATOMIC ENERGY AGENCY, Use of nuclear reactors for seawater desalination, IAEA-TECDOC-574, Vienna (1990).

[2.3] INTERNATIONAL ATOMIC ENERGY AGENCY, Technical and economic evaluation of potable water production through desalination of seawater by using nuclear energy and other means, IAEA-TECDOC-666, Vienna (1992).

[2.4] INTERNATIONAL ATOMIC ENERGY AGENCY, Potential for nuclear desalination as a source of low cost potable water in North Africa, IAEA-TECDOC-917, Vienna (1996).

[2.5] INTERNATIONAL ATOMIC ENERGY AGENCY, Options Identification Programme for Demonstration of Nuclear Desalination, IAEA-TECDOC-898, Vienna, (1996).

[2.6] INTERNATIONAL ATOMIC ENERGY AGENCY, Introduction of Nuclear Desalination — a Guidebook, Technical Reports Series No. 400, IAEA, Vienna (2000).

[2.7] INTERNATIONAL ATOMIC ENERGY AGENCY, Design and Development Status of Small and Medium Reactor Systems 1995, IAEA-TECDOC-881, Vienna (1996).

[2.8] INTERNATIONAL ATOMIC ENERGY AGENCY, Design Concepts of Nuclear Desalination Plants, IAEA-TECDOC-1326, Vienna (2002).

[2.9] MISHANIA, Yu.D. et al, “RUTA a Pool-Type Reactor for Heat Supply and the Possibility for its Application Area Expansion”, Proc. International Seminar of Small and Medium Sized Reactors: Status and Prospects, held in Cairo, Egypt, 27–31 May 2001, C&S Papers Series 14/p, pp. 593–601, IAEA, Vienna (2002).

[2.10] DAVIES, K., HUMPHRIES, J.R. and ACKERT, J.A., "Reducing the Cost of Desalted Water: A Status Report on an Experimental Demonstration Project", Proc. International Desalination Association World Congress on Desalination and Water Reuse, IDA, (2001).

CHAPTER 3. COUPLING OF DESALINATION AND NUCLEAR PLANTS

The nuclear reactor supplies energy to desalination systems either in the form of heat or of the mechanical/electrical energy. It supplies thermal energy for distillation processes such as MSF or MED and provides mechanical/electrical energy for the processes which are based on mechanical energy such as RO, MVC or MED/VC. Apart from the basic energy for desalination, all desalination processes require electricity for pumping, auxiliaries and other services.

Desalination plants can be coupled as a single purpose plant or a co-generation plant. In the case of a single purpose nuclear desalination plant, energy is exclusively used for the desalination process, and the desalted water is the only product output. The nuclear reactor is fully dedicated to supplying energy for desalting. In case of a co-generation plant, only a part of the energy is utilized for desalting. A co-generation plant produces both electricity and water simultaneously.

When a nuclear reactor is used to supply steam for desalination, the method of coupling has a significant technical and economic impact. The optimum method of coupling depends on the size and type of the reactor, the specific characteristics of the desalination process and the desirability and value of electricity generation as a co-product.

This Chapter presents a brief description of schemes considered within this Coordinated Research Project (CRP) for coupling the nuclear and desalination technologies described in Chapter 2.

3.1. COUPLING DESALINATION PLANTS TO POWER REACTORS

Power reactors are suitable, in general, for desalination processes that require energy in the form of electricity such as RO and MVC. The power is supplied from a dedicated plant/electrical grid to drive the high-pressure pump for the RO process and the main compressor of MVC. Steam can also be bled off at suitable points in the secondary circuit of the power plant for use by the desalination plant. However, protective barriers must be included in all co-generation modes to prevent potential carry-over of radioactivity. The power plant condenser cooling water is usually discharged to the sea as waste heat. In a contiguous plant, it is possible to use this heated seawater as feed to an RO desalination plant, thereby improving the performance of the desalination plant. In this arrangement, waste heat from the power reactor is used to improve the efficiency of the RO plant.

Within the CRP, two power reactors were considered namely CANDU-6 by CANDESAL, Canada and PHWR-220 by BARC, India. The proposed desalination technologies for coupling with these reactors are RO for the former and hybrid MSF-RO for the latter. The coupling schemes are presented below.

3.1.1. Coupling of RO and CANDU-6

An advanced concept for the design and operation of RO seawater desalination systems was first presented by CANDESAL in 1993 [3.1]. Preheated feed water is one of the key features of this concept, which is a feature now being considered for a number of current desalination systems. 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 design of the CANDU reactors allows for the use of waste heat from its moderator cooling water system giving an additional temperature rise to the RO system feed 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 resulting in a maximum feed water preheat available of only about ^{0}C. An analysis carried out by CANDESAL showed that this additional source of waste heat can be used to further increase the temperature rise of the feed water stream by as much as 9^{0}C under the design conditions of the specific case analyzed. This results in significant performance improvement. A schematic of the CANDU-6/RO nuclear desalination system is shown in Figure 3.1.

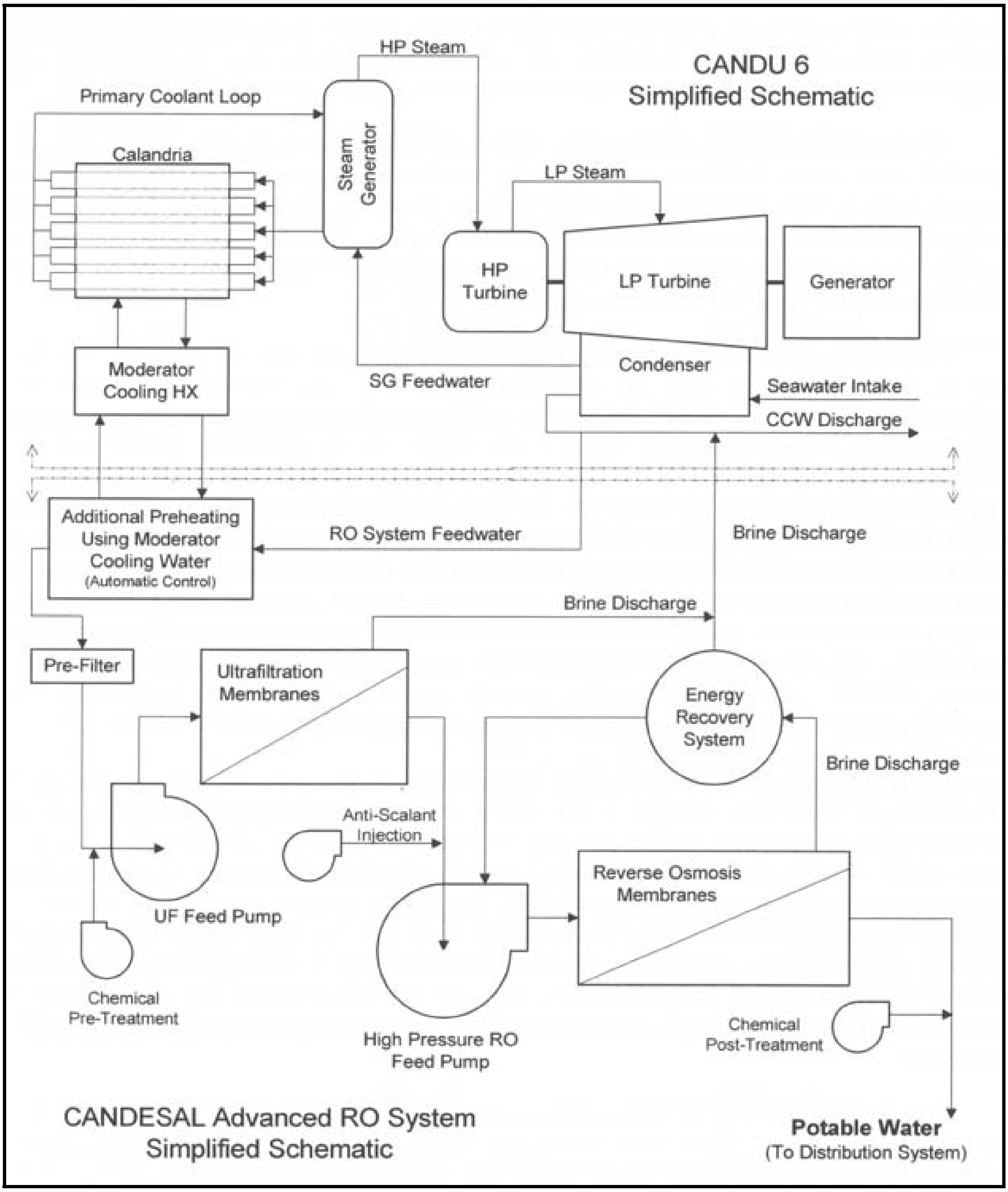


Figure 3.1: Simplified Schematic CANDU-6 CANDESAL Nuclear Desalination System.

A strong emphasis is placed on the integration of the energy and water production systems into a single, optimized design for the co-generation of both water and electricity. This approach has the advantage of maximizing the benefits of integration while at the same time minimizing the impact of physical interaction between the two systems. Transients in the desalination plant do not have a feedback effect on reactor operation.

3.1.2. Coupling of hybrid MSF-RO and PHWR-220

Nuclear Desalination Demonstration Project at Kalpakkam aims for demonstrating the safe and economic production of good quality water by nuclear desalination of seawater comprising of 4500 m^3/d Multistage Flash (MSF) and 1800 m^3/d Reverse Osmosis (RO) plant. MSF plant uses low-pressure steam from Madras Atomic Power Station (MAPS), Kalpakkam.

The 6300 m^3/d combined MSF-RO nuclear desalination project is located in between the existing 220 MW(e) PHWR MAPS and proposed 500 MW(e) FBR at Kalpakkam. The MSF plant uses the required quantity of low-pressure (LP) steam for seawater desalination. In order to avoid any chance of ingress of radioactivity (tritium) to MSF process and product water, it has been decided to incorporate an isolation heat exchanger between MAPS steam supply and the brine heater of MSF. The LP steam is tapped from the manholes in the cold reheat lines after HP turbine exhaust from both the nuclear reactors (MAPS I and II). The moisture content is removed through a moisture separator. The steam is sent to intermediate isolation heat exchanger to produce process steam for brine heater of the MSF plant. It is designed to keep the steam temperature in brine heater below 130°C to avoid scaling on the tube side. The condensate from the isolation heat exchanger is returned back to the deaerator section of the power station. Adequate provisions for monitoring and control have been incorporated for isolation of the steam supply in case of shutdown of the power station or desalination plant. The process flow sheet of coupling MSF plant and isolation HX with MAPS are shown in Figure 3.2.

This heat exchanger is important equipment for the MSF plant. It is the interface between the power plant steam and heat recovery stages of MSF plant. As the brine is very corrosive in nature, the brine is passed through the tube side of the brine heater. Design and selection of material is done in such a way that it can give thirty years of useful life. As the maximum temperature from the brine heater outlet is 121°C, the recycle brine concentration is chosen in such a way that it will not cross the solubility limit of Ca^{2+} and SO_4^{-2}. Brine velocity inside the tube is chosen more than 1.8 m/s. This velocity not only gives more heat transfer co-efficient but also provides very less contact time for scale nucleation. In addition to physical barrier given by LP isolation HX, the pressure of brine in the tube side is kept more than the heating steam pressure to the brine heater to prevent any in leakage of tritium activity in the product water.

There will be loss of net electrical power generation of the power station due to steam extraction for the MSF plant. It has been estimated that about 2.8 MW(e) power will be lost. The power is lost, mainly due to extraction of LP steam and small loss is due to extraction of HP steam. Due to extraction of LP steam there will be reduction of steam flow in LP turbine and decrease in extraction pressure in LP turbine and hence the output power of LP turbine will be reduced. But at the same time the steam flow to the HP turbine will be increased marginally due to lesser requirement of moisture separator reheat steam and HP turbine power output will be increased marginally.

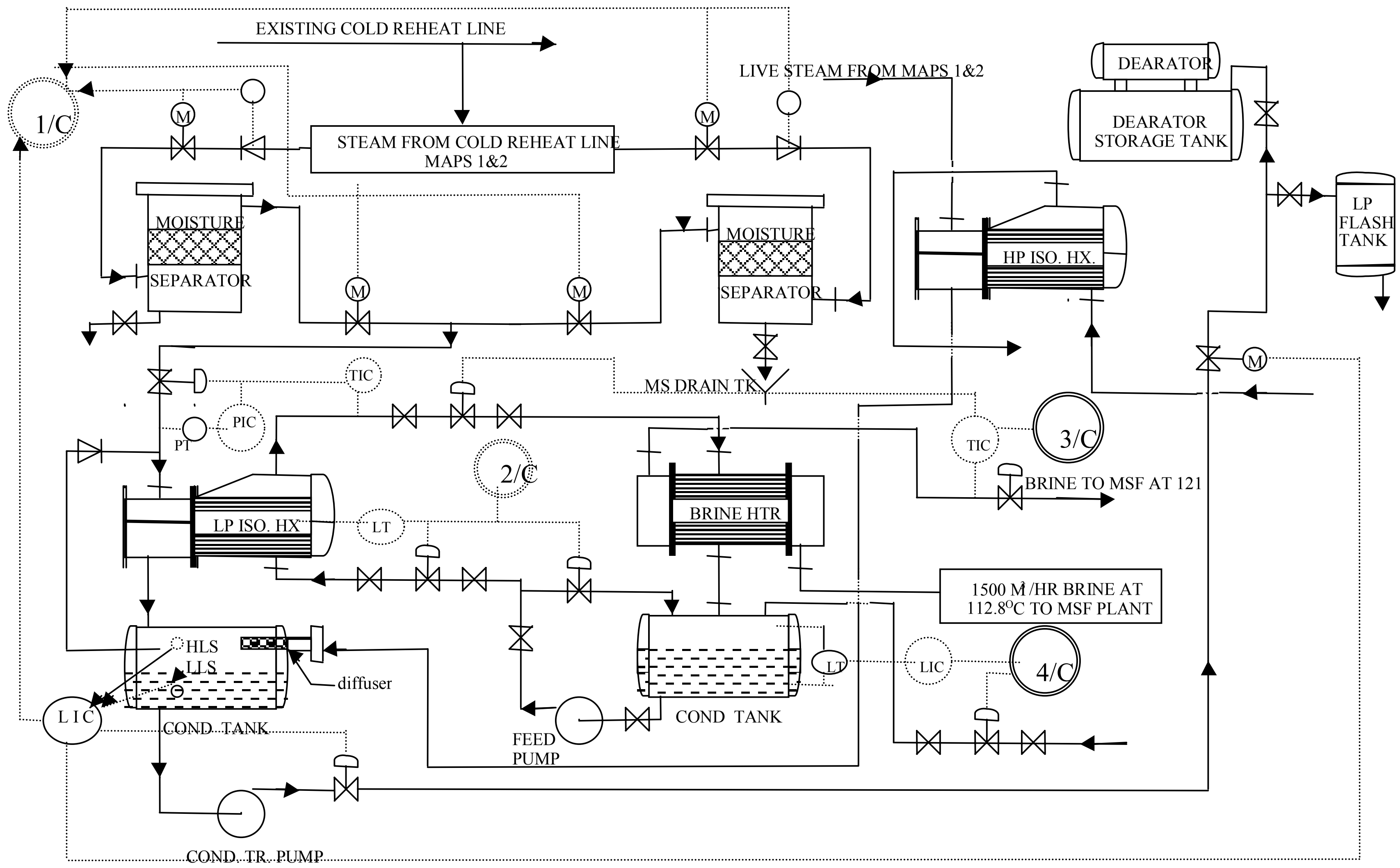


Figure 3.2: Schematic of Coupling 6300 m3/day MSF-RO with 170 MW PHWR at Kalpakkam.

3.2. COUPLING DESALINATION PLANTS TO COGENERATION REACTORS

In the case of nuclear plants that co-generate heat and electric power, the steam can be bled off at suitable points in the secondary circuit of the power plant for use by the desalination plant. There are two types of co-generation mode [3.2]:

- Parallel co-generation,
- Series co-generation.

In parallel co-generation, electricity is produced as co-product along with desalted water by diverting a part of the steam to the turbine to produce electricity in the conventional manner and part of the steam to the desalination plant. This configuration allows increased flexibility in energy use. However, the total energy consumption would be the same as if the steam for desalination and electricity had been produced separately.

In series co-generation, electricity is produced by expanding the steam first through a turbine with an elevated backpressure and then to the desalination process. This form of co-generation results in reduced total energy consumption as compared to parallel co-generation. From the thermodynamic point of view, it is useful to convert most of the steam enthalpy to mechanical/electrical energy in the turbogenerator before using it as a heating medium in a thermal desalination plant for producing desalted water. Raising the turbine backpressure increases the temperature of the heat available to the desalination plant but reduces the amount of electricity generated. Therefore, in series co-generation the turbine backpressure must be optimized relative to overall plant economics.

The coupling between co-generation reactors described in Chapter 2 and various desalination systems is described below.

3.2.1. Coupling of desalination processes and CAREM-D

A generic small PWR of the Advanced Type, with an Integrated Primary Coolant Loop and once through Steam Generators, like the CAREM-D reactor, producing superheated steam is taken as the Nuclear Steam Supply System. The Nuclear Power Plant size was chosen considering a reasonable Project balance (an appropriate relative size to the Desalination Plant). The Thermal-Cycle (Turbo-group) chosen is quite standard for a Small Size NPP, with a single turbine having four vapour extractions. The final heat sink (through seawater) uses different exchange surfaces for condensing and sub-cooling.

Within this CRP, INVAP elaborated simplified models of generic MSF, MED and RO processes in order to identify the relevant variables for the simulation connected to the safety analysis of the co-generation Plant.

3.2.1.1. Coupling with MSF

The Generic MSF Plant consists of a variable number of control volumes, or stages, through which steam from the turbine and sea water couple. The first stage is actually an evaporator while and the last is connected with a Product Water Tank. A connecting pipeline between two sections of stages allows modeling a MSF Plant with brine recycle.

The use of turbine exhaust was disregarded, and a decision had to be made between two alternatives:

1. The direct use of the turbine extraction #3 giving the following thermodynamic conditions, 9.4 bar and 177°C.

2. The use of the outlet of the High Pressure Pre-heater #2 with water at about 10 bar and 180°C.

Both of them have been analysed and the second alternative was chosen for the following reasons:

- The convective transport through the steam lines and turbine is extremely fast. Extraction #4 goes through the shell-side of Pre-heater #2, and placing the coupling HX after it gives a convenient margin for isolation, in case contamination is detected at the outlet of the Steam Generators.

- This coupling scheme has a smaller impact on the overall Plant design. If the turbine extraction #3 is used then Pre-heater #2 has to be completely redefined.

- Direct turbine extractions change strongly with the NPP power. This scheme gives the Desalination plant a greater independence of operation.

In the NPP scheme (Figure 3.3) the coupling is represented by the element "MSF HX" connected to the outlet of Pre-heater #2 and the FW tank (outlet of Pre-heater #1), respectively.

3.2.1.2. Coupling with MED

The Generic MED Plant, most likely Low Temperature MED, consists of a variable number of evaporators, or effects, through which steam from the turbine and sea water couple, either in parallel flows (co-current) or in opposite direction flows (counter-current). Analogous steam conditions as those presented for MSF are adopted for MED coupling. Figure 3.4 shows the coupling scheme.

3.2.1.3. Coupling with RO

As compared to the processes of MED and MSF, RO is simpler in terms that all the relevant components are filled with single-phase liquid. For the co generation option (CAREM-D), there are few small changes in the CAREM Plant BOP design. The main change is on the condenser design: in order to optimize 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 assures the pre-heated intake for two 10 000 m^3/day RO modules, with a direct extension for a third module. Besides, the BOP for co-generation has additional piping for the seawater cooling flow by-passing the condenser. It allows the seawater intake to be used by the SWRO plant when the condenser is under maintenance or reparation. Figure 3.5 shows CAREM-D Balance of Plant and preheated SWRO Desalination System.

This BOP change reduces the mechanical power delivered by the turbine in a small amount, balanced by the performance improvement of preheating in a single module. With a second module in operation, there is a clear benefit from the thermal coupling. This benefit increases with a third module.

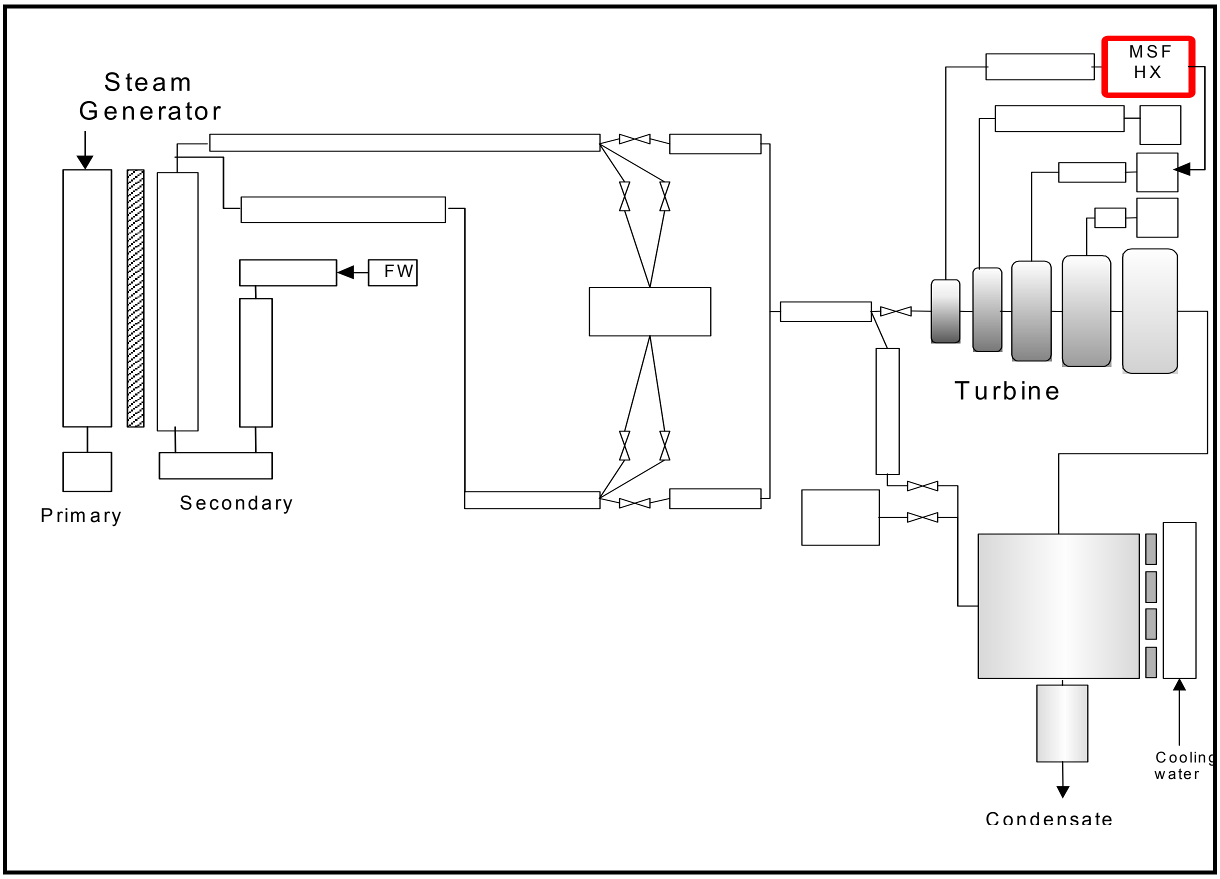


Figure 3.3: Scheme of the CAREM-MSF coupling.

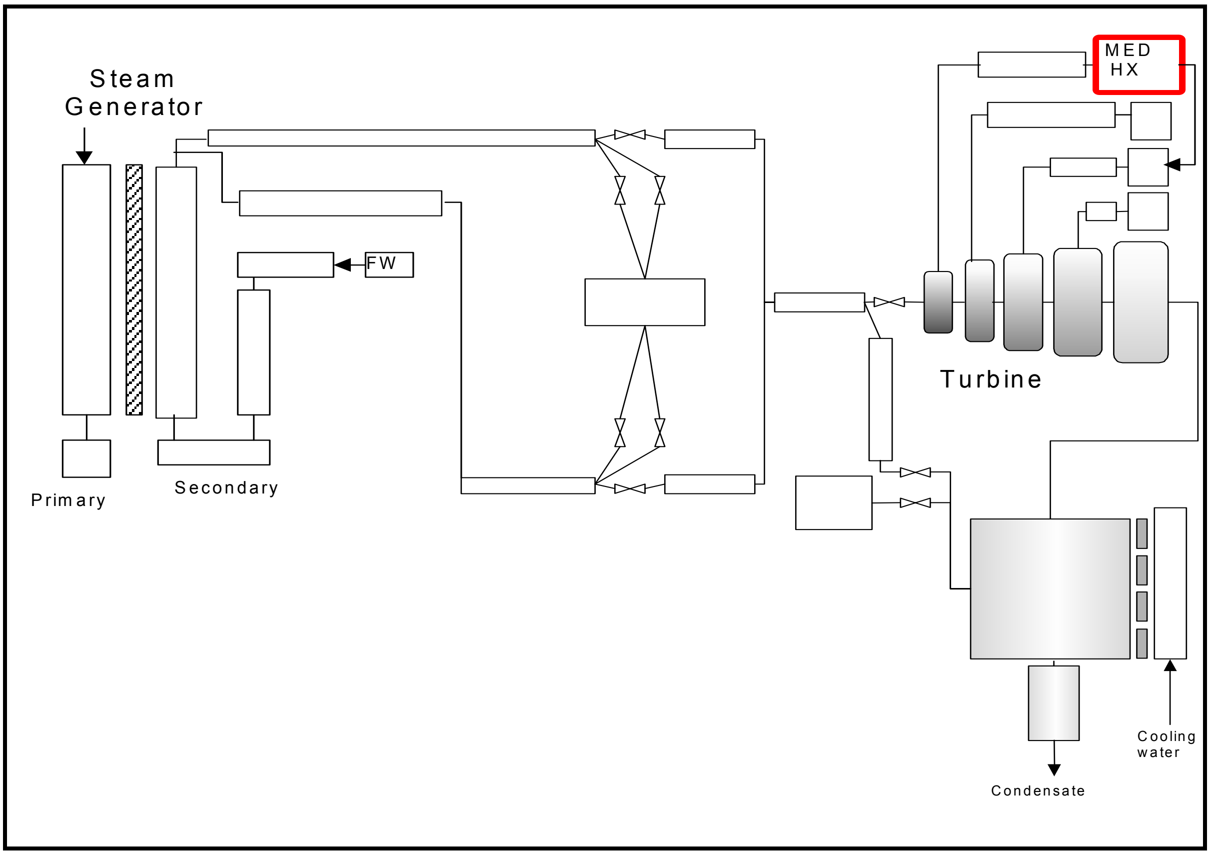


Figure 3.4: Scheme of the CAREM-MED coupling.

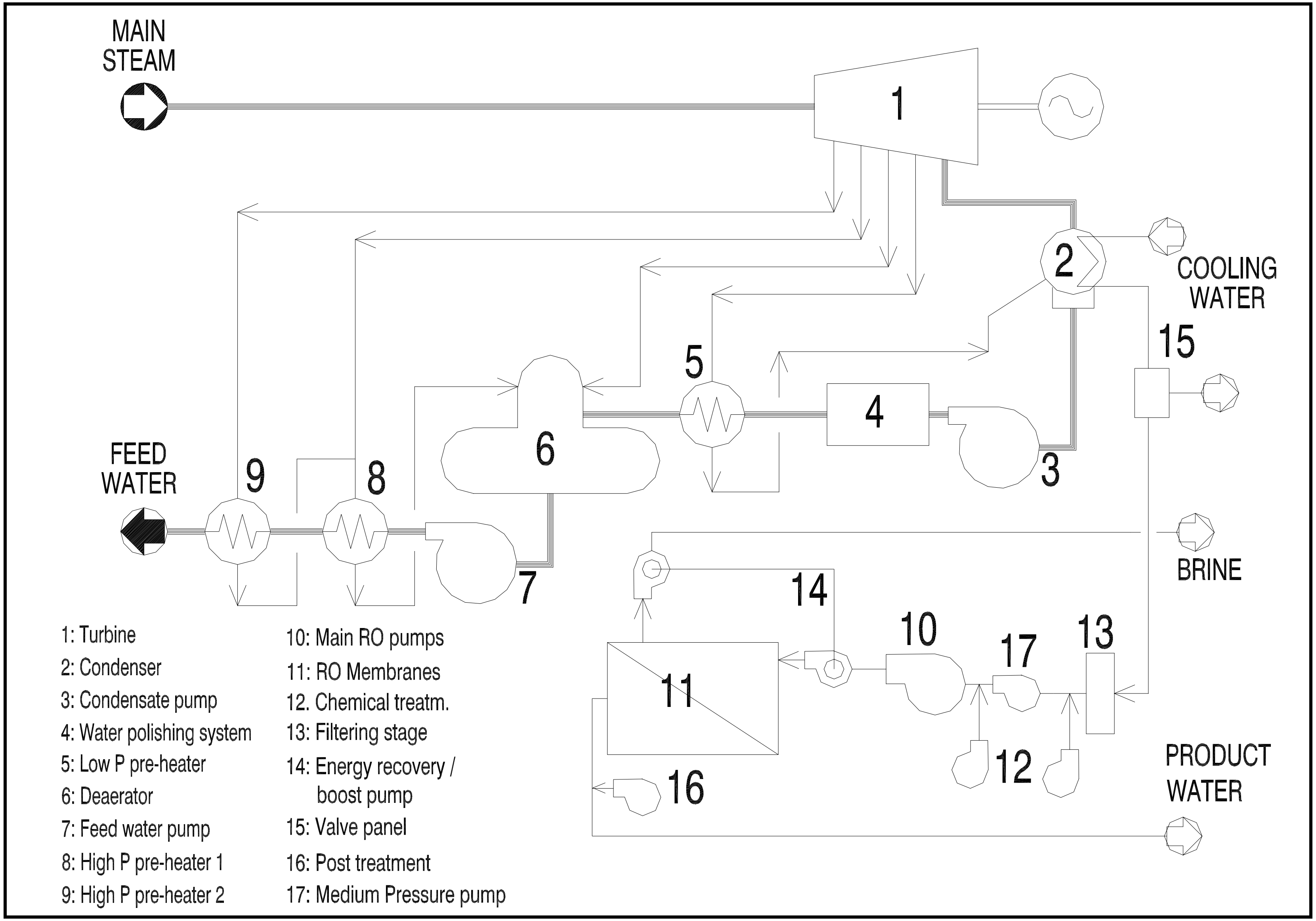


Figure 3.5: CAREM_D Balance of Plant and preheated SWRO Desalination System.

The co-generation plant layout places the desalination plant on the outlet channel, and foresees the connection of both plants by a thermally isolated piping. The make-up building houses two make-up tanks and the isolating system. This, and the membranes building, are designed and placed allowing their walls to be shared by another building (in between) for additional RO modules.

The desalination plant has two buildings. The main one is called the membranes building and houses the intake valves panel, a two-stages (roughing and polishing) filtering system, an electrical and control board, and space available for three modules. A module comprises a chemical dosing cartridge filters and cleaning system, and five membrane trains. Each train has a pumping and energy recovery unit and a membranes array. On the East Side of the building there are four areas aimed for offices, maintenance, post-treatment and ware-house. The routine operation is possible to be performed from a control-room within the office area, with daily check routines in the rest of the desalination plant. The second building houses the make-up tanks, and the isolation system. The latter triggers to avoid the product water contamination, and has independent actuators both in the tanks inlet (coming from the membranes building), and in the plant outlet (connection to the Water Supplier). A general layout of CAREM-D and preheated SWRO integrated plant is shown in Figure 3.6.

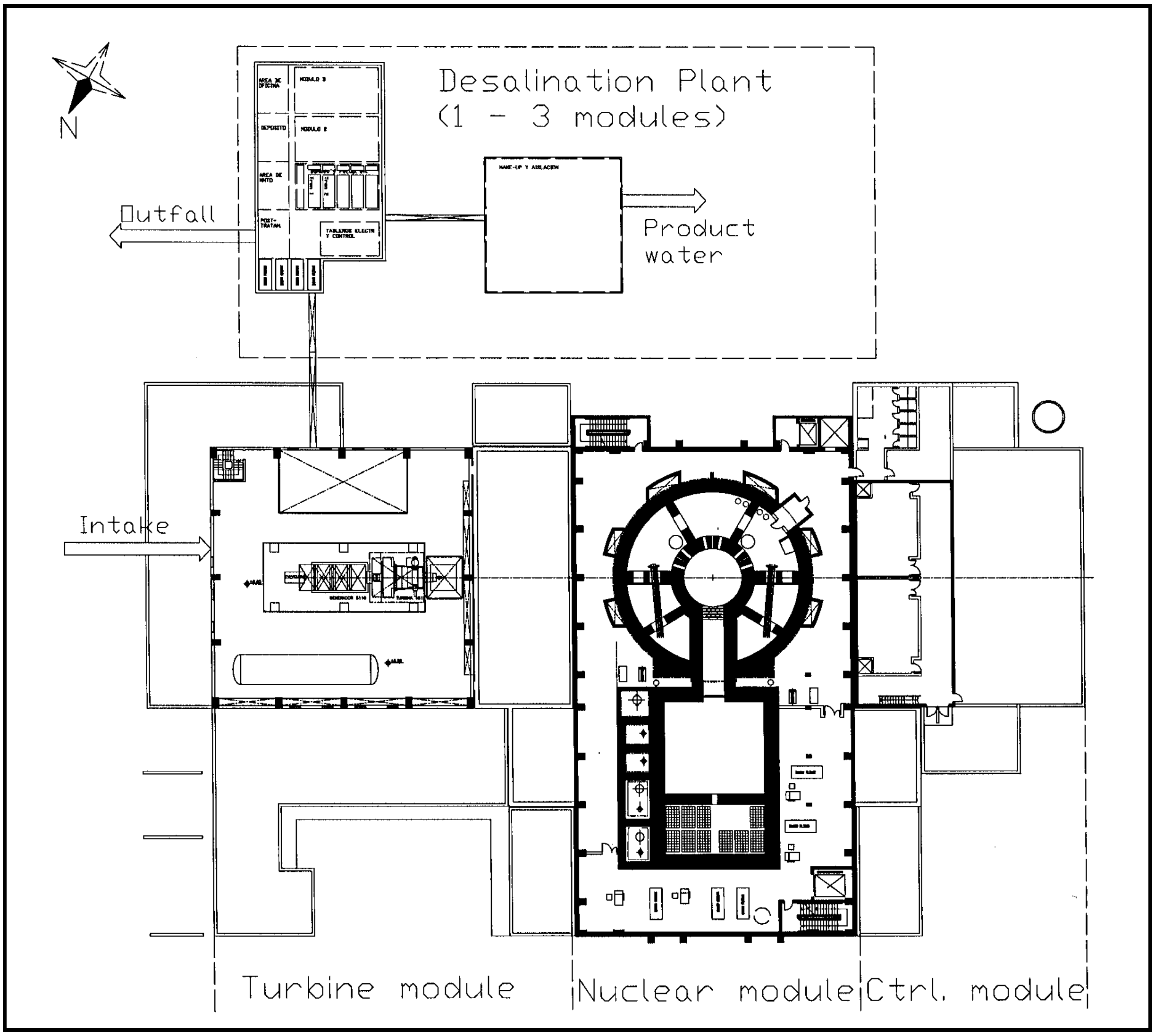


Figure 3.6: General layout of CAREM-D and preheated SWRO integrated plant.

3.2.2. Coupling of desalination processes and KLT-40C

Two approaches to the use of KLT-40C reactor plant with the Nuclear Desalination Complex have been chosen by IPPE, namely:

- Studying the variants of the KLT-40C reactor couplings in a nuclear desalination complex without a strict relation to the project of floating power unit (FPU), which is currently under development.
- Full retention of both the reactor plant and steam turbine generator plant configurations, and the design and parameters of the system of heat supply to the district heating circuit used in the KLT-40C floating nuclear power plant.

For each of these two approaches various coupling schemes with HTME, RO and hybrid MED-RO were investigated. The coupling schemes are briefly discussed below.

3.2.2.1 Coupling with desalination processes and KLT-40C

Coupling KLT-40C with distillation HTME plants

The following variants of coupling schemes were considered for the Complex:
- Coupling with the use of turbine plant steam extraction 2x25 Gcal/hour for the distillation plant (Variant 1). The Co-Generation Complex comprises the reactor plant, the steam extraction turbine, a condenser, the intermediate loop heater and a distillation plant (see Figure 3.7). The reactor plant and the DOU plant are connected not just electrically, but thermally.
- Coupling with the use of turbine plant steam extraction and a peak heat of the intermediate loop water for the distillation plant (full thermal power of the steam extraction Q =2x25 + 2x 18 = 86 Gcal/hour, (Variant 2).
- Coupling with the use of a back-pressure turbine and heat from main condenser for DOU (Variant 3). The co-generation complex comprises the reactor plant, the backpressure turbine; a condenser and DOU plant (see Figure 3.8).

When optimizing the coupling scheme of the reactor and the distillation plant, a steam dump system was considered which allows for independent regulation of the Complex water and electricity outputs.

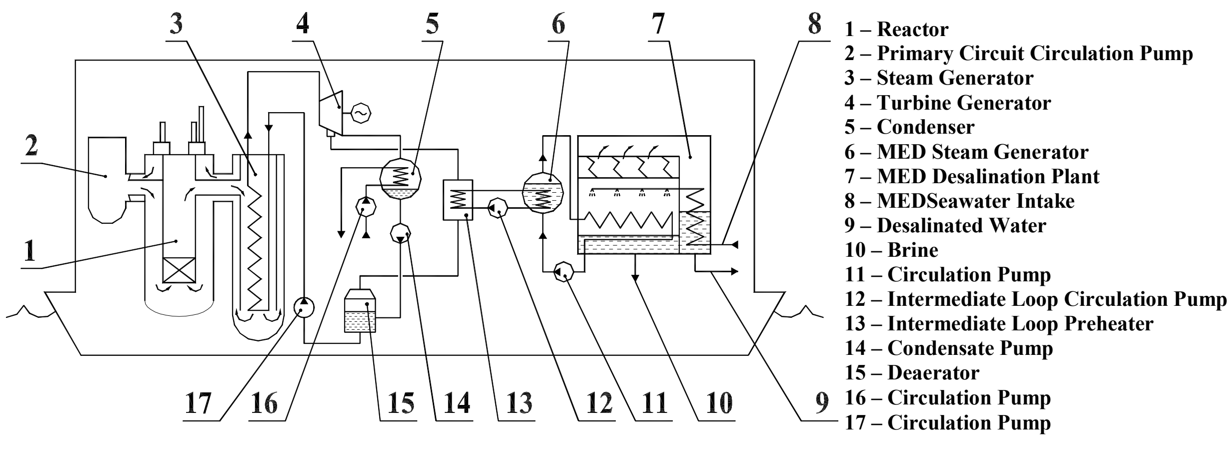
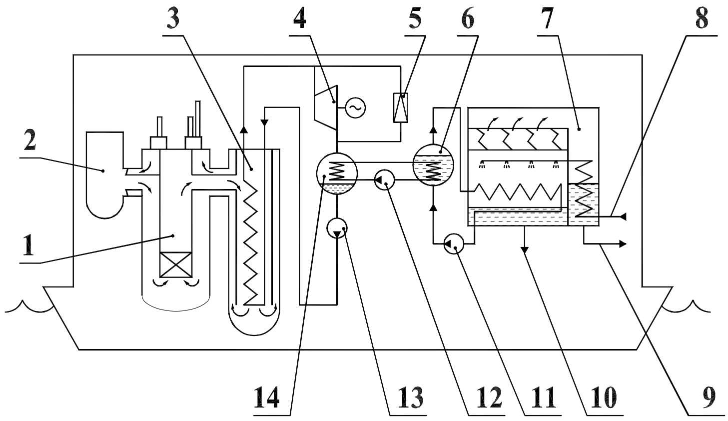


Figure 3.7: Coupling KLT-40 C with MED through Turbine Extractions.



1 – Reactor
2 – Primary Circuit Circulation Pump
3 – Steam Generator
4 – Turbine Generator
5 – Pressure Regulator
6 – MED Steam Generator
7 – MED Desalination Plant
8 – MEDSeawater Intake
9 – Desalinated Water
10 – Brine
11 – Circulation Pump
12 – Intermediate Loop Circulation Pump
13 – Condensate Pump
14 – Condenser

Figure 3.8: Coupling KLT-40 C with MED through Back-Pressure Turbine.

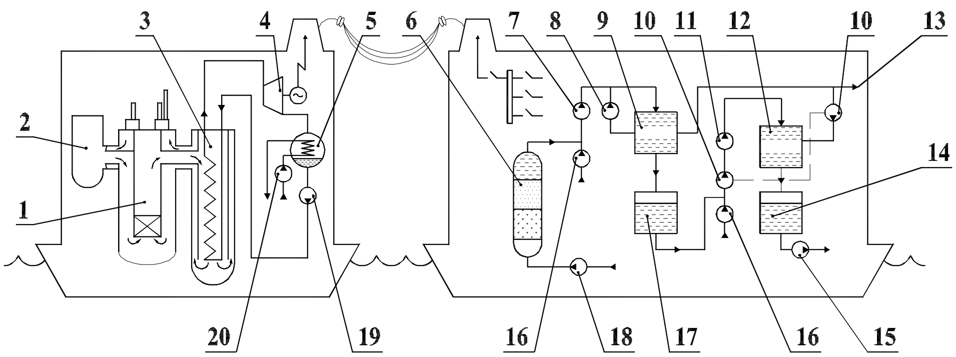
Coupling KLT-40C with reverse osmosis plants

The co-generation complex comprises the reactor plant, the condensing turbine or a steam extractions one, a condenser, the intermediate loop heater using the turbine steam extraction power of 25 Gcal, and RO plant. The following variants have been considered:

(1) No seawater preheat (see Figure 3.9):
- Seawater flow, which is directed to the RO plant, is equal to the cooling seawater flow to the turbine condensers (Variant 4);
- With maximum desalinated water output of RO plant (Variant 7).

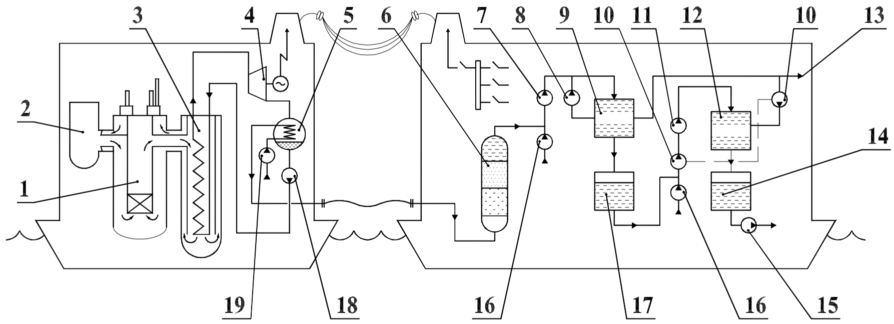
(2) With seawater preheat in condensers of the turbines working in the condensing mode (see Figure 3.10):
- With seawater flow to the RO plant after the turbine condensers (Variant 5).
- With maximum desalinated water output of RO plant (Variant 8) (see Figure 3.11). The distinctive feature is directing a part of seawater flow to the RO plant without preheat. The total preheat value equals to 4.5°C.

(3) With seawater preheat in turbine condensers and with the use of heat form the turbine steam extractions (Variant 6) (see Figure 3.12).



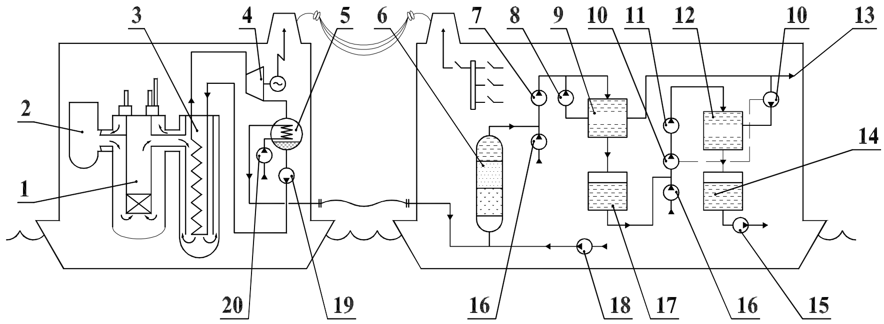
1 – Reactor
2 – Primary Circulation Pump
3 – Steam Generator
4 – Turbine Generator
5 – Condenser
6 – Water Pretreatment Filter
7 – Intermediate Pressure Pump
8 – Recirculation Pump
9 – Ultrafiltration Membranes
10 – Power Recovery System
11 – Booster Pump
12 – RO Membranes
13 – Brine Discharge
14 – Drinking Water Tank
15 – Drinking Water Pump
16 – Chemicals Adding
17 – Ultrafiltration Tank
18 – Seawater Pump
19 – Condensate Pump
20 – Seawater Pump

Figure 3.9: Coupling KLT-40 C with RO without Seawater Preheat.



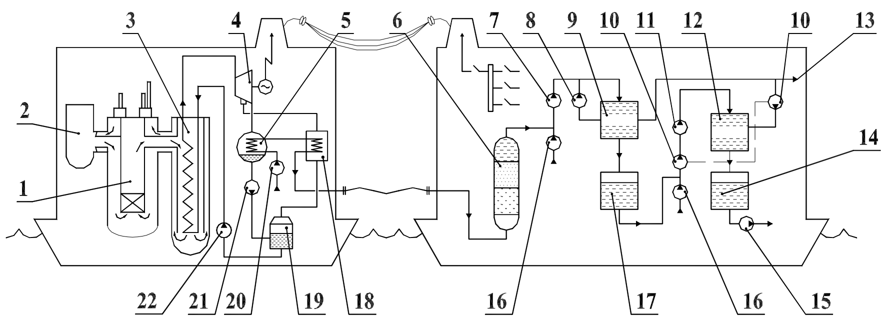
1 - Reactor
2 – Primary Circulation Pump
3 – Steam Generator
4 – Turbine Generator
5 – Condenser
6 – Water Pretreatment Filter
7 – Intermediate Pressure Pump
8 – Recirculation Pump
9 – Ultrafiltration Membranes
10 – Power Recovery System
11 – Booster Pump
12 – RO Membranes
13 – Brine Discharge
14 – Drinking Water Tank
15 – Drinking Water Pump
16 – Chemicals Adding
17 – Ultrafiltration Tank
18 – Condensate Pump
19 – Seawater Pump

Figure 3.10: Coupling KLT-40 C with Preheating RO Feed Water in Plant Condensers.



1 – Reactor
2 – Primary Circulation Pump
3 – Steam Generator
4 – Turbine Generator
5 – Condenser
6 – Water Pretreatment Filter
7 – Intermediate Pressure Pump
8 – Recirculation Pump
9 – Ultrafiltration Membranes
10 – Power Recovery System
11 – Booster Pump
12 – RO Membranes
13 – Brine Discharge
14 – Drinking Water Tank
15 – Drinking Water Pump
16 – Chemicals Adding
17 – Ultrafiltration Tank
18 – Seawater Pump
19 – Condensate Pump
20 – Seawater Pump

Figure 3.11: Coupling KLT-40 C with Preheating Part of RO Feed Water in Plant Condensers.



1 – Reactor
2 – Primary Circulation Pump
3 – Steam Generator
4 – Turbine Generator
5 – Condenser
6 – Water Pretreatment Filter
7 – Intermediate Pressure Pump
8 – Recirculation Pump
9 – Ultrafiltration Membranes
10 – Power Recovery System
11 – Booster Pump
12 – RO Membranes
13 – Brine Discharge
14 – Drinking Water Tank
15 – Drinking Water Pump
16 – Chemicals Adding
17 – Ultrafiltration Tank
18 – Feed Water Preheater
19 – Deaerator
20 – Seawater Pump
21 – Condensate Pump
22 – Circulation Pump

Figure 3.12: ***Coupling KLT-40 C with Preheating RO Feed by Extracted Steam and Plant Condensers****.*

Coupling KLT-40C with hybrid MED-RO plants

The Complex comprises the reactor plant, a turbine, a condenser and a hybrid plant. The reactor plant and the hybrid plant are connected not just electrically, but thermally. The RO plant consumes the electricity produced. The thermal connection is provided by the use of heat of condensers in the course of electricity production. The heat from condensers is removed with seawater used as feed water for the RO plant. The following variants are considered:

(1) Using a back-pressure turbine and heat from the condenser to the DOU plant, and the electricity produced for the RO plant (Variant 9) (see Figure 3.13);

(2) Using steam from the turbine steam extractions for the DOU, and the rest of electricity is used for the RO plant with seawater preheats in the condenser (see Figure 3.14):

- With the use of 2x25 Gcal/hour turbine steam extraction and 2x25 + 2x 18 = 86 Gcal/hour for the DOU plant and seawater flow to the RO plant after turbine condensers (Variant 10 and Variant 11).
- With the use of 2x25 Gcal/hour of turbine steam extraction and 2x25 + 2x 18 = 86 Gcal/hour for the DOU plant, and with maximum capacity of the RO pant (Variant 12 and Variant 13).

Principal characteristics of the Complex calculated for Variants 1 through 13 are summarized in Table 3.1.

3.2.2.2. Coupling with KLT-40C in floating power unit

In the Russia-Canada Project of Nuclear Desalination Complex with KLT-40C reactor plant 4 variants of coupling schemes have been chosen for further studies. Besides, the 5th variant is regarded as additional.

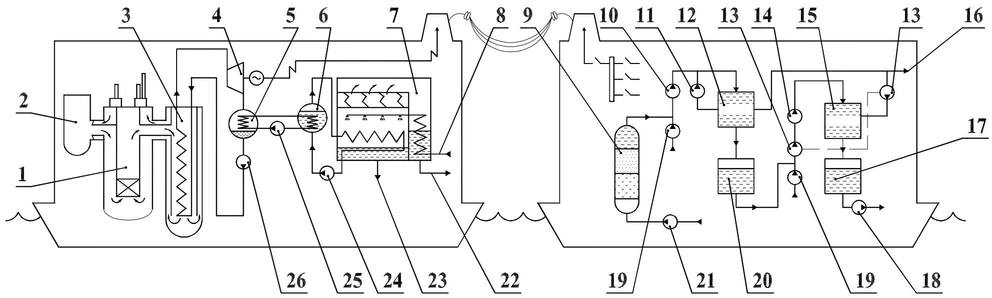
The first coupling scheme for Nuclear Desalination Complex on the basis of FPU and a hybrid desalination plant is presented in Figure 3.15. The turbine plant of the FPU produces both heat with the steam extractions and electricity with the turbine generator. In this case, steam from the second steam extraction is directed to the intermediate loop preheaters. There extracted steam condensates, giving its heat to the intermediate loop coolant. A part of the seawater flow preheated in the DOU GTPA desalination plant is headed to the evaporator of the intermediate loop part. After it has partly evaporated in the evaporator, steam at nearly 65 C with the two-phase jet compressor heats up to 95°C. The working fluid of the jet compressor is water, which is pumped out of the evaporator and heated in the intermediate loop heat exchanger up to 125 C. The RO plant in the preheat mode utilizes the preheated seawater from the FPU condensers and the DOU GTPA-840 condenser.

The second scheme of the Complex with the FPU and a preheat RO plant is presented in Figure 3.16. The turbine plant of the FPU produces both heat with the steam extractions and electricity with the turbine generator. The feed water of the RO plant is preheated first in the condenser and then further in the intermediate loop preheater. This scheme fully utilizes both the condenser heat and the turbine steam extraction heat to preheat the seawater.

The third scheme with the seawater preheat in the turbine plant condenser of the FPU is given Figure 3.17. The turbine plant of the FPU produces both heat with the steam extractions and electricity with the turbine generator. The RO plant feed water intake and preheat is performed after the turbine condensers.

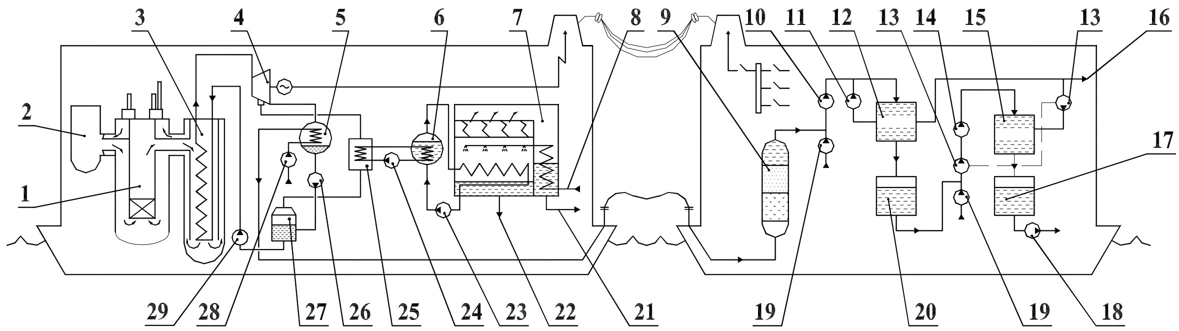
The fourth scheme of a Co-Generation Complex on the basis of the FPU and an RO plant is shown in Figure 3.18. The turbine plant of the FPU produces both heat with the steam extractions and electricity with the turbine generator. Seawater is preheated only in the intermediate loop preheater.

The fifth scheme of the Complex on the basis of the FPU and 22-stage distillation plant is depicted in Figure 3.19. The turbine plant of the FPU produces both heat with the steam extractions for the DOU GTPA-840 plant and electricity with the turbine generator.



1 – Reactor
2 – Primary Circuit Circulation Pump
3 – Steam Generator
4 – Turbine Generator
5 – Condenser
6 – MED Steam Generator
7 – MED Desalination Plant
8 – MEDSeawater Intake
9 – Water Pretreatment Filter
10 – Intermediate Pressure Pump
11 – Recirculation Pump
12 – Ultrafiltration Membranes
13 – Power Recovery System
14 – Booster Pump
15 – RO Membranes
16 – Brine Discharge
17 – Drinking Water Tank
18 – Drinking Water Pump
19 – Chemicals Adding
20 – Ultrafiltration Tank
21 – Seawater Pump
22 – Desalinated Water
23 – Brine Discharge
24 – Circulation Pump
25 – Intermediate Loop Circulation Pump
26 – Condensate Pump

Figure 3.13. Coupling KLT-40 C with Hybrid MED-RO using Back-Pressure Turbine.



1 – Reactor
2 – Primary Circuit Circulation Pump
3 – Steam Generator
4 – Turbine Generator
5 – Condenser
6 – MED Steam Generator
7 – MED Desalination Plant
8 – MEDSeawater Intake
9 – Water Pretreatment Filter
10 – Intermediate Pressure Pump
11 – Recirculation Pump
12 – Ultrafiltration Membranes
13 – Power Recovery System
14 – Booster Pump
15 – RO Membranes
16 – Brine Discharge
17 – Drinking Water Tank
18 – Drinking Water Pump
19 – Chemicals Adding
20 – Ultrafiltration Tank
21 – Desalinated Water
22 – Brine Discharge
23 – Circulation Pump
24 – Intermediate Loop Circulation Pump
25 – Intermediate Loop Preheater
26 – Condensate Pump
27 – Deaerator
28 – Seawater Pump
29 – Circulation Pump

Figure 3.14: Coupling KLT-40 C with Hybrid MED-RO using Extractions from Turbine.

TABLE 3.1: PRINCIPAL CHARACTERISTICS OF VARIOUS COUPLING VARIANTS

Characteristic	Variant												
	1	2	3	4	5	6	7	8	9	10	11	12	13
DOU Plant Capacity, m^3/day	50400	84000	218400						218000	50400	84000	50400	84000
RO Plant Capacity, m^3/day				108000	108000	108000	324000	324000	120000	108000	108000	300000	276000
Number of RO Modules				9	9	9	27	27	10	9	9	25	23
Number of DOU Plants	3	5	13						13	3	5	3	5
DOU Plant Specific Heat Consumption, kWh/m^3	28.78	28.78	28.78						28.78	28.78	28.78	28.78	28.78
DOU Plant Specific Electricity Consumption, kWh/m^3	0.92	0.92	0.92						0.92	0.92	0.92	0.92	0.92
RO Plant Specific Electricity Consumption, kWh/m^3				4.62	4.04	4.03	4.62	4.51	4.49	4.04	4.04	4.49	4.46
Hybrid Plant Specific Electricity Consumption, kWh/m^3									4.55	3.88	3.83	4.35	4.25
Hybrid Plant Electricity Consumption, MW(e)									33.04	20.31	21.91	58.23	54.97
DOU Plant Electricity Consumption, MW(e)	2.11	3.71	10.58						10.58	2.11	3.71	2.11	3.71
RO Plant Electricity Consumption, MW(e)				20.80	18.20	18.12	64.73	63.07	22.46	18.20	18.20	56.12	51.26
Electric Power to the Grid, MW(e)	57.89	52.69	23.42	44.20	46.80	41.88	0.27	1.93	0.96	39.69	34.49	1.77	1.43
Electricity Cost, US$/kWh	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05
Desalted Water Cost, US$/ m^3	0.789	0.765	0.768	0.661	0.583	0.614	0.596	0.586	0.719	0.713	0.708	0.628	0.637

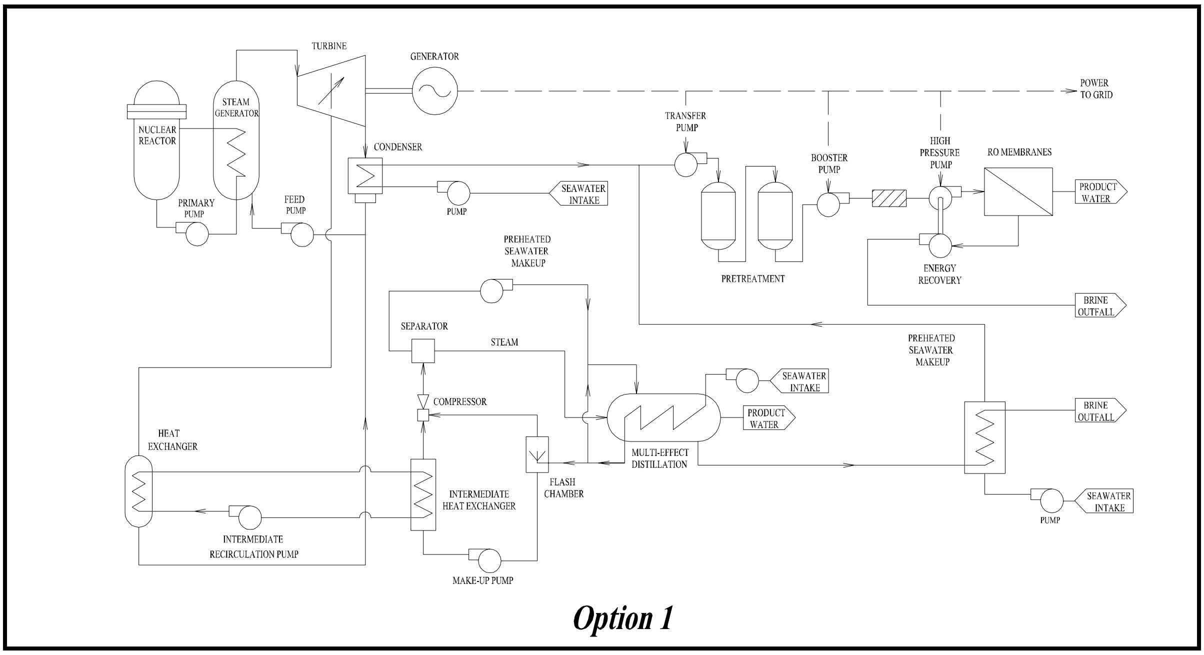


Figure 3.15: Nuclear Desalination Plant on the basis of FPU and a Hybrid MED-RO.

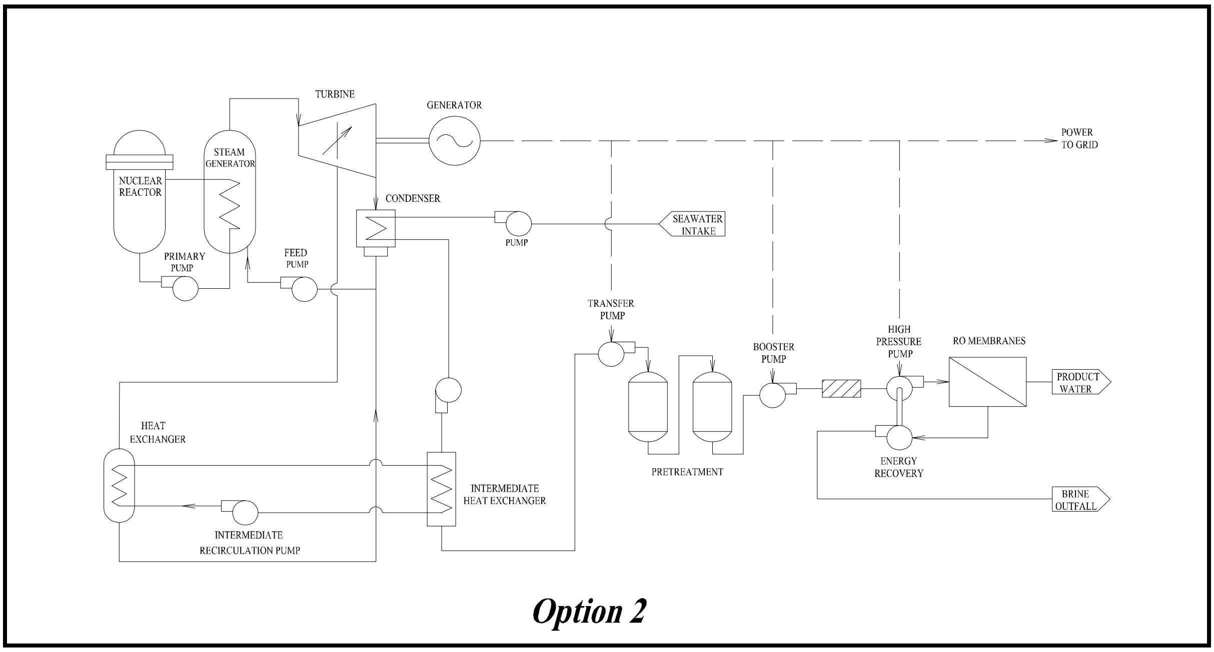


Figure 3.16: Nuclear Desalination Complex on the basis of FPU and a Preheated RO in Condenser and Intermediate Loop.

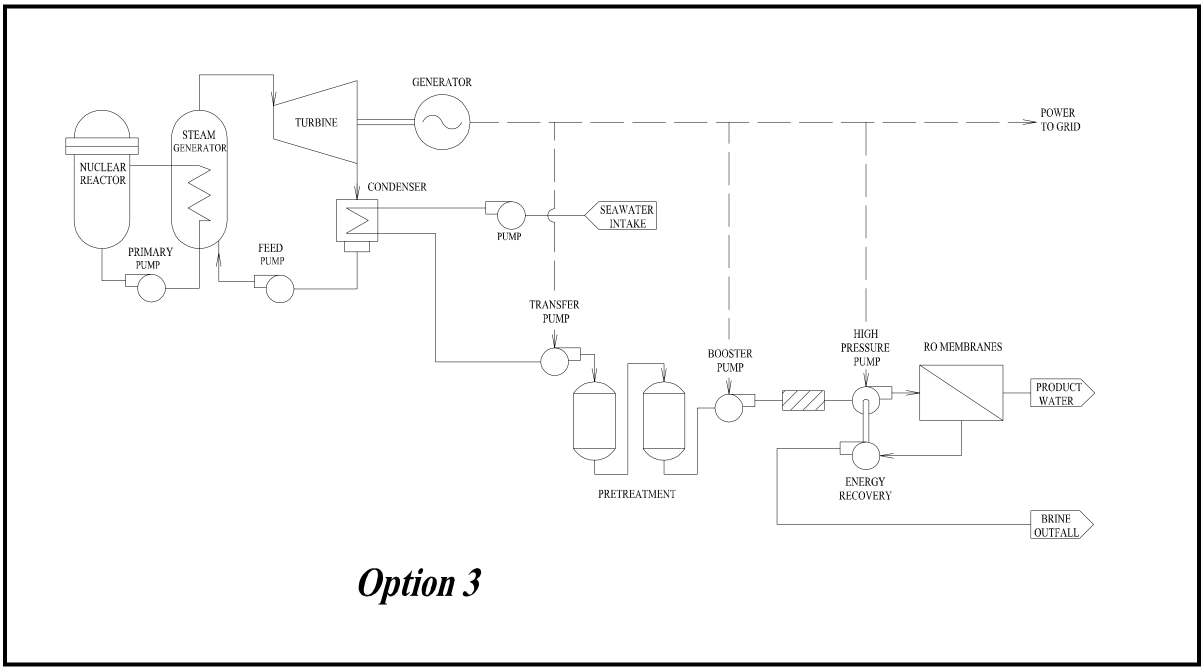


Figure 3.17: Nuclear Desalination Plant on the basis of FPU and Preheated RO in Condenser.

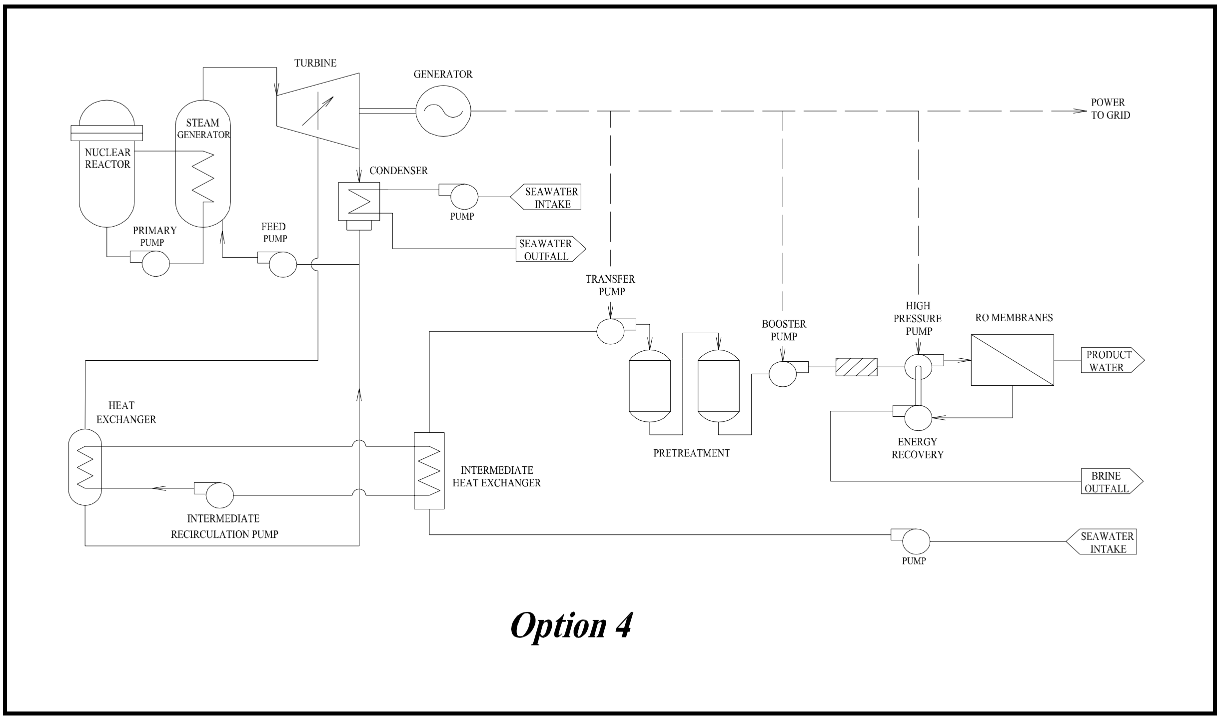


Figure 3.18: Nuclear Desalination Complex on the basis of FPU and Preheated RO in Intermediate Loop.

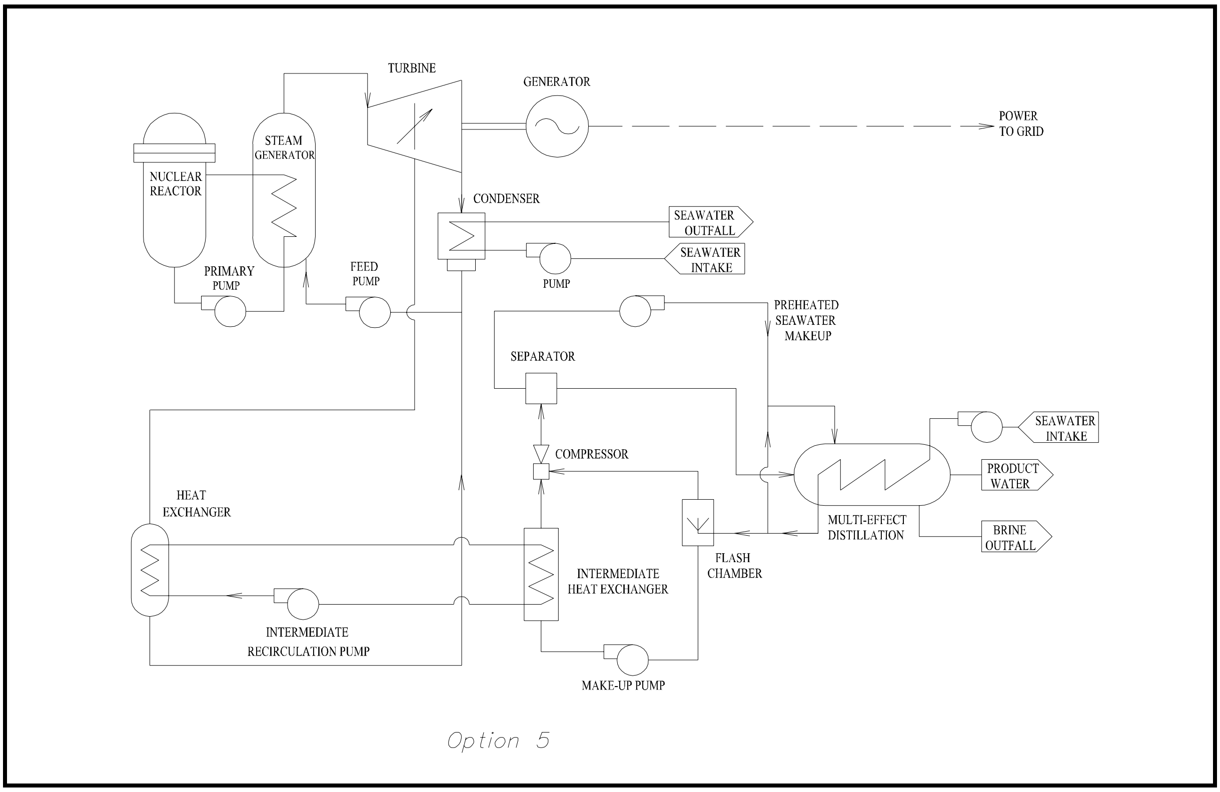


Figure 3.19: Nuclear Desalination Complex on the basis of FPU and 22-stage MED Plant.

3.2.3. Coupling of HTME and NIKA-70

For producing of desalted water NIKA-70 can be coupled with all types of modern desalination plant, i.e. with reverse osmosis (RO) plants, distillation plants and their combinations. It should be noted that multi effect distillation (MED) plants are produced in Russia while RO plants have to be imported.

Hybrid MED-RO can represent special interest for consumers. In this case very pure fresh water is obtained from distillation plant, and potable quality fresh water is obtained from the RO plant at lower cost. Consumer has the choice for optimal ratio between capacity of MED and RO plants. Depending on site condition and customer requirements, desalination plants can be placed at single barge with reactor, at separated barge or on shore.

At this stage, the study of the reactor facility and MED coupling pattern as part of the NIKA nuclear desalination complex has been continued in the following area: integration with Russian made GTPA-230 HTME desalination units recommended by the developer (SverdlNIIKhIMMASh) for coupling with the NIKA-70 reactor facility.

The task of nuclear power and desalination plants coupling is very important especially when coupling with distillation plants. Coupling designs should exclude, firstly, any possibility of radioactive contamination of desalted water, secondly, possibility of penetration of brine water in turbine circuit and, thirdly, they are not being too expensive to facilitate competitiveness of nuclear option.

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.

A variant of coupling through an additional isolated water circuit with a higher pressure 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 electric power losses due to lost shaft work because of a higher temperature in the turbine condenser.

A variant of coupling through an extra intermediate circuit with hot water throttling in the first stage of the distillation plant proposed by the IAEA [3] fits the requirements of both economic effectiveness of the desalination process and radioactive contamination protection.

This variant has a set of 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 case when those take place.

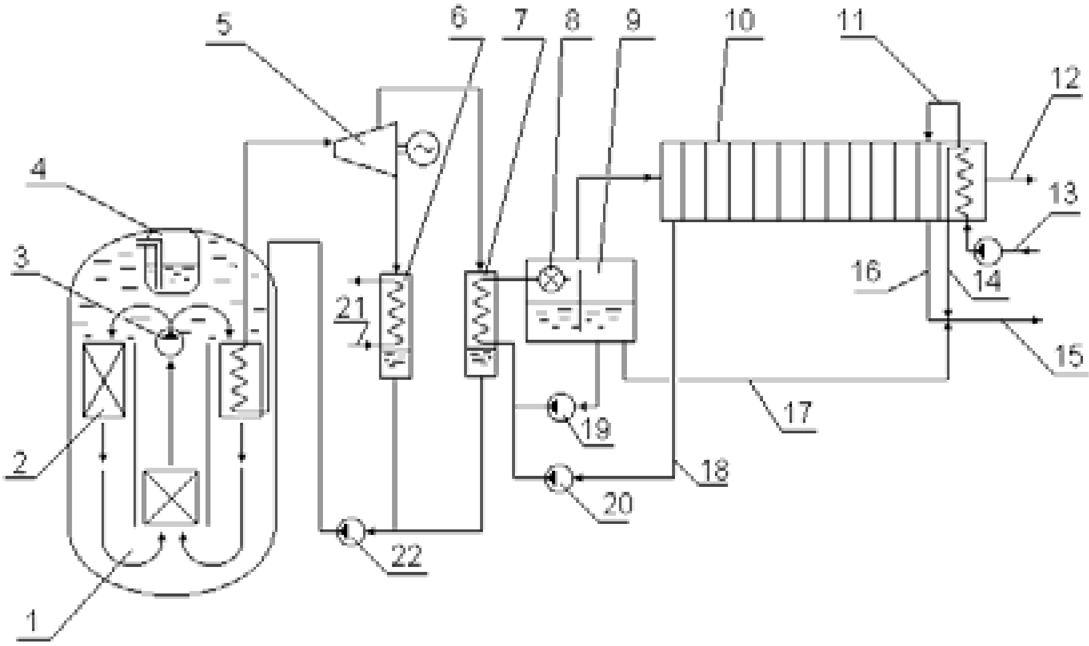
A disadvantage of the above variant is that seawater could enter the secondary circuit of the reactor plant in case there is a leakage in the distillation desalination plant pre-heater. However, the experience of operating nuclear and fossil fuel power plants with seawater-cooled turbine condensers has shown that this disadvantage was not determining. There are various ways reducing the risk of secondary circuit seawater contamination to zero.

This variant is chosen as the most optimal for coupling the NIKA-70 reactor plant and distillation plant. Figure 3.20 shows the optimal scheme of coupling between the NIKA-70 reactor plant and multi-effect distillation plant.

Out of the number of the units serially manufactured in Russia, DDU of the GTPA-230 type (distillation desalination unit with horizontal tube film devices) recommended by the developer (SverdlNIIKhIMMASh) are the most suitable for coupling with the NIKA-70 reactor.

3.2.4. Coupling of HTME and SMART

Three methods of steam extraction were considered for desalination namely, prime steam, turbine extraction, and backpressure turbine. Since the prime interest in application of SMART to desalination is the utilization of steam rather than electricity, only the distillation processes, Multi-Effect Distillation (MED) and Multi-Stage Flashing (MSF) were taken into consideration. The results of economic evaluation showed that the combination of MED with the turbine extraction of steam is the most economical option with respect to the effective use of energy produced from SMART. Therefore, MED and steam extraction from the turbine were selected for coupling with SMART. The most important safety concern in using the nuclear thermal energy for desalination is the radioactivity carry-over into the product water. In the SMART integrated nuclear desalination plant, an intermediate loop (steam transformer) is installed between the secondary cycle of SMART and the desalination plant to protect from any possible radioactive ingress into the product water.



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

Figure 3.20: Schematic Diagram of NIKA-70 coupled with MED.

Figure 3.21 shows the configuration of the coupling scheme for the SMART integrated nuclear desalination plant. The function of the steam transformer is to prevent the contamination of the produced water by hydrazine and radioactive material of the primary steam. In the steam transformer the primary and secondary steam flows are completely separated, thus no mixing occurs between them.

3.3. COUPLING DESALINATION PLANTS TO HEATING REACTORS

In this type of reactor, the steam (or hot water) produced by the reactor is directly supplied to a thermal desalination process without producing electricity. Electricity is required for pumping water and for auxiliaries and, therefore, must be supplied from another source (e.g. the electrical grid). It is desirable to have the nuclear reactor and desalination plant adjacent to each other to minimize piping and heat loss.

In nuclear power reactors, the steam is generated at high temperature and pressure, whereas heating reactors need only produce low temperature steam or hot water for thermal desalination processes. Thermal desalination processes have an upper temperature limit of about 140°C, owing to excessive scaling beyond this temperature. Efforts to operate desalination plants at higher maximum brine temperatures by implementing improved scale control were not found to be cost effective. Thus, heating reactors, which are designed to supply steam at 130°C or lower, have the best potential for coupling to desalination plants [3.3]. Coupling schemes with NHR-200 and RUTA heating reactors are presented below.

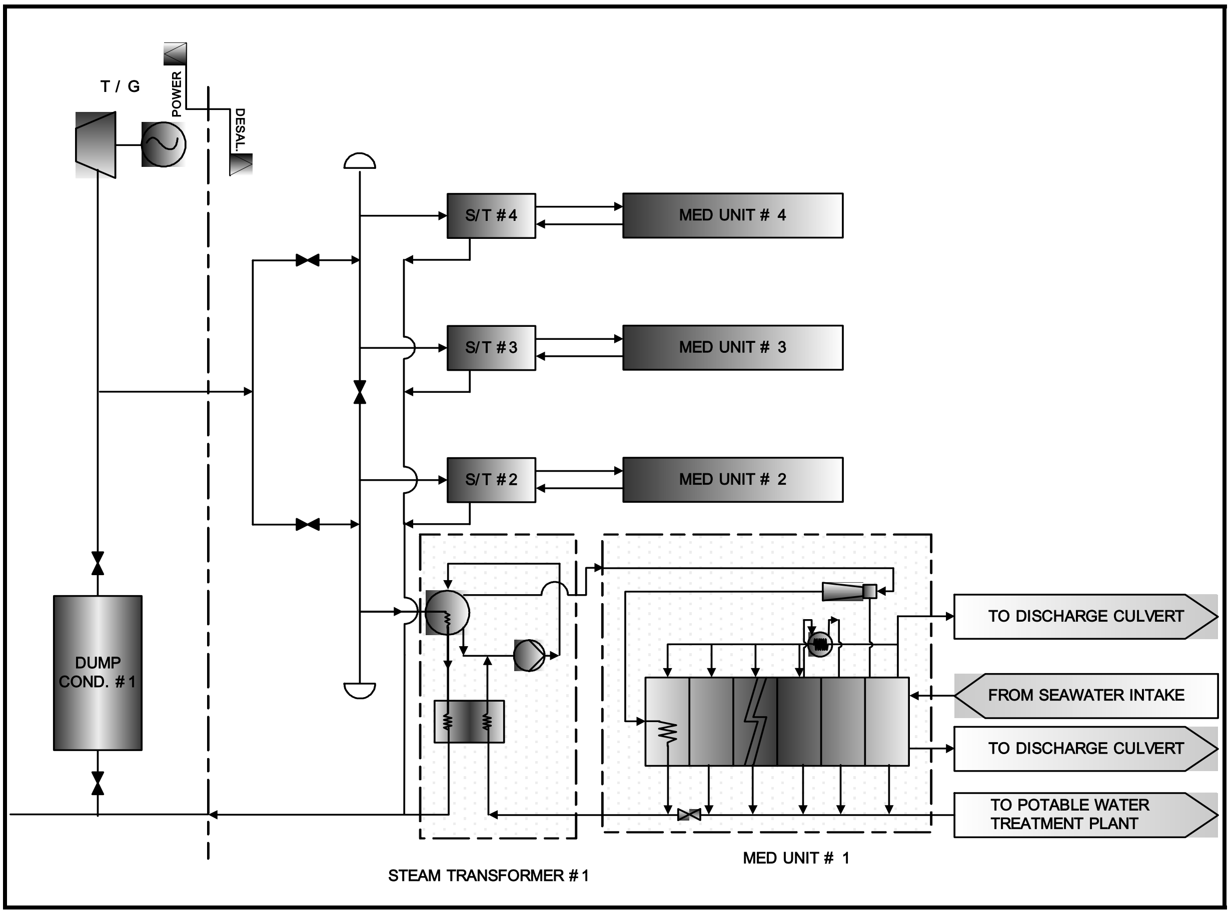


Figure 3.21: Coupling concept of SMART and MED-TVC.

3.3.1. Coupling of desalination processes and NHR-200

3.3.1.1. Coupling of VTE-MED and NHR-200

Coupling scheme of NHR-200 heating reactor with VTE-MED desalination process is shown in Figure 3.22. The desalination system and the nuclear reactor system are mechanically and thermally connected with the main motive steam pipe and the main feed back water pipe in the steam supply circuit (in-between short distance, at a common site) to be build up as the integrated desalination plant Energy required by desalination process is efficiently transferred by steam from the steam generator of NHR to the first effect of the desalination system;

Between nuclear reactor and VTE-MED desalination system, there are three layers of steel wall boundaries (i.e. heat transfer tube wall of the primary heat exchanger, steam generator and the first effect evaporator of desalination system) and two circuits (Intermediate circuit and Steam supply circuit) working as barriers to effectively isolate and prevent the desalination system from radioactive contamination.

Pressures in the primary circuit and the secondary circuit are 25 bar and 30 bar separately. Even in case of rare failure of heat transfer tube 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 due to the appropriate pressure barrier.

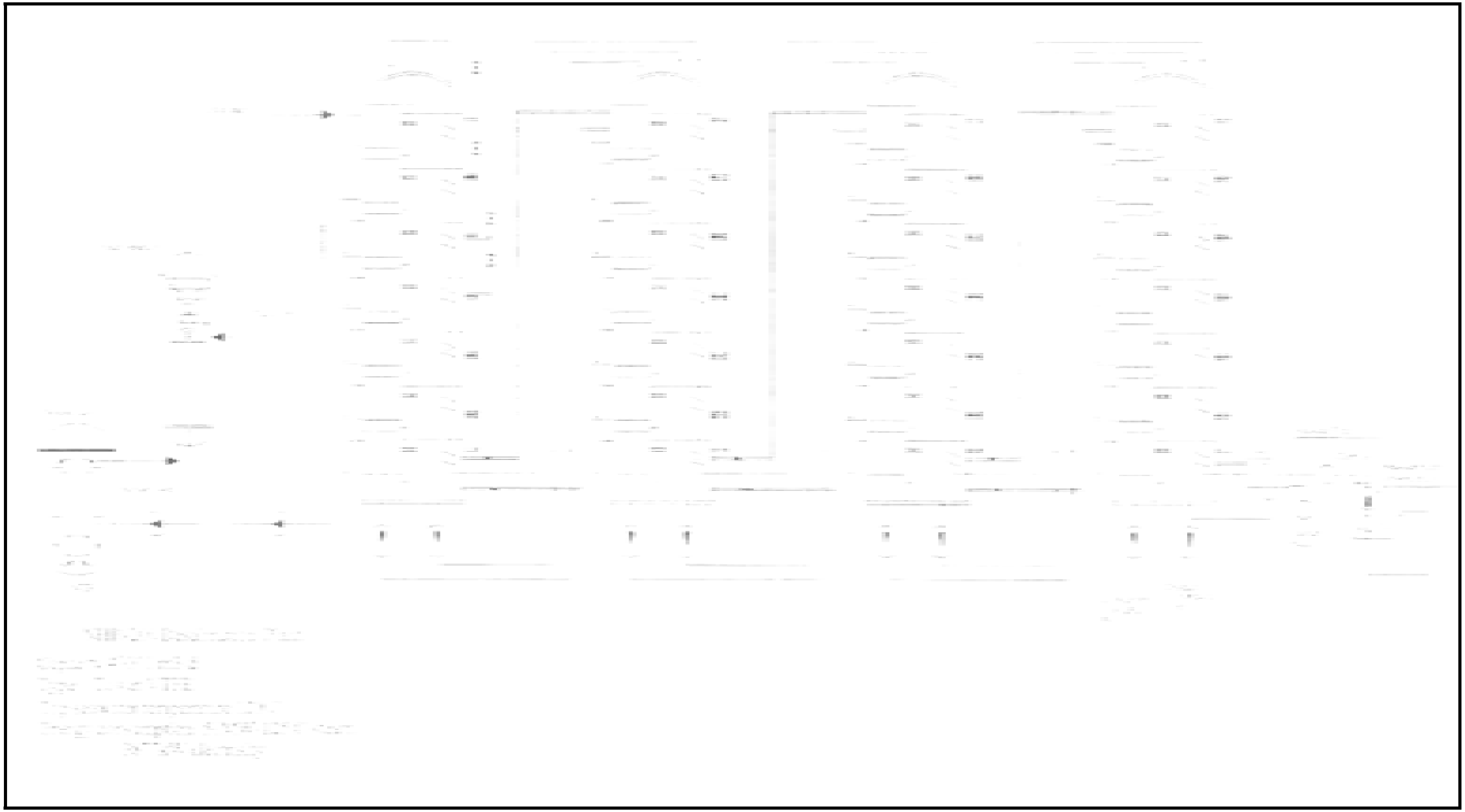


Figure 3.22: Coupling scheme of NHR-200 heating reactor with VTE-MED desalination process.

Both of the heating reactor and the desalination system have excellent self-regulation capability. When the load fluctuates within the range of 70% to 100%, no matter the active perturbation happens in either side, operation of the integrated plant would be very smooth and with perfect self-regulation performance. Adjustment within load range of 40% to 100% is also easy and simple to perform. Therefore the integrated desalination plant has good operability due to suitable operation performance of the MED process and NHR reactor systems.

3.3.1.2. Coupling of hybrid desalination plant and NHR-200

In order to investigate the possible approach to increase in thermal efficiency, decrease in the water production cost and further optimize the coupling design of NHR heating reactor with desalination process, the coupling schemes of NHR reactor with hybrid desalination process were investigated.

Two coupling schemes were selected for the co-generation mode:

a. NHR + Low temperature MED+RO
b. NHR + Low temperature MED+MED/VC

Coupling of hybrid HTME-RO plant and NHR-200

The Coupling scheme of heating reactor NHR-200 with hybrid HTME-RO desalination process is shown in Figure 3.23. It can be seen that there are two barriers between the NHR reactor and the desalination process: intermediate circuit and the steam supply circuit. Saturate steam generated in the steam generator was directly conducted to the steam turbine as the motive steam. The exhausted steam in back-pressure from the turbine was led to the first effect of the LTMED process and the condensate from the first effect was pumped back to the reactor steam generator as its feed water. Several RO units worked parallel with the LTMED process. But the part of the pre-heated brine from LTMED was used as the feed seawater to the RO units. The produced electricity by the turbine-generator of the nuclear plant was used to supply power to the motor of the high-pressure pump in the RO unit and the nuclear power plant itself.

The total capacity of the selected RO process depends on the maximum electricity production of the NHR heating reactor, and the number of RO unit depends on the available unit capacity of RO process. The distribution method for the electricity cost and heat charge in the co-generation plant was considered as following.

Coupling of hybrid HTME-MED/VC plant and NHR-200

The Coupling scheme of heating reactor NHR-200 with LTMED and MED/VC desalination process is shown in Figure 3.24. It can be seen that there are two barriers between the NHR reactor and the desalination process: intermediate circuit and the steam supply circuit. The saturate steam generated in the steam generator was directly conducted to the steam turbine as the motive steam. The exhausted steam in back-pressure from the turbine was conducted to the first effect of the LTMED process and the condensate from the first effect was pumped back to the steam generator as its feed water. Several MED/VC units worked parallel with the LTMED process. But the part of the brine from LTMED was used as the feed seawater to the MED/VC units.

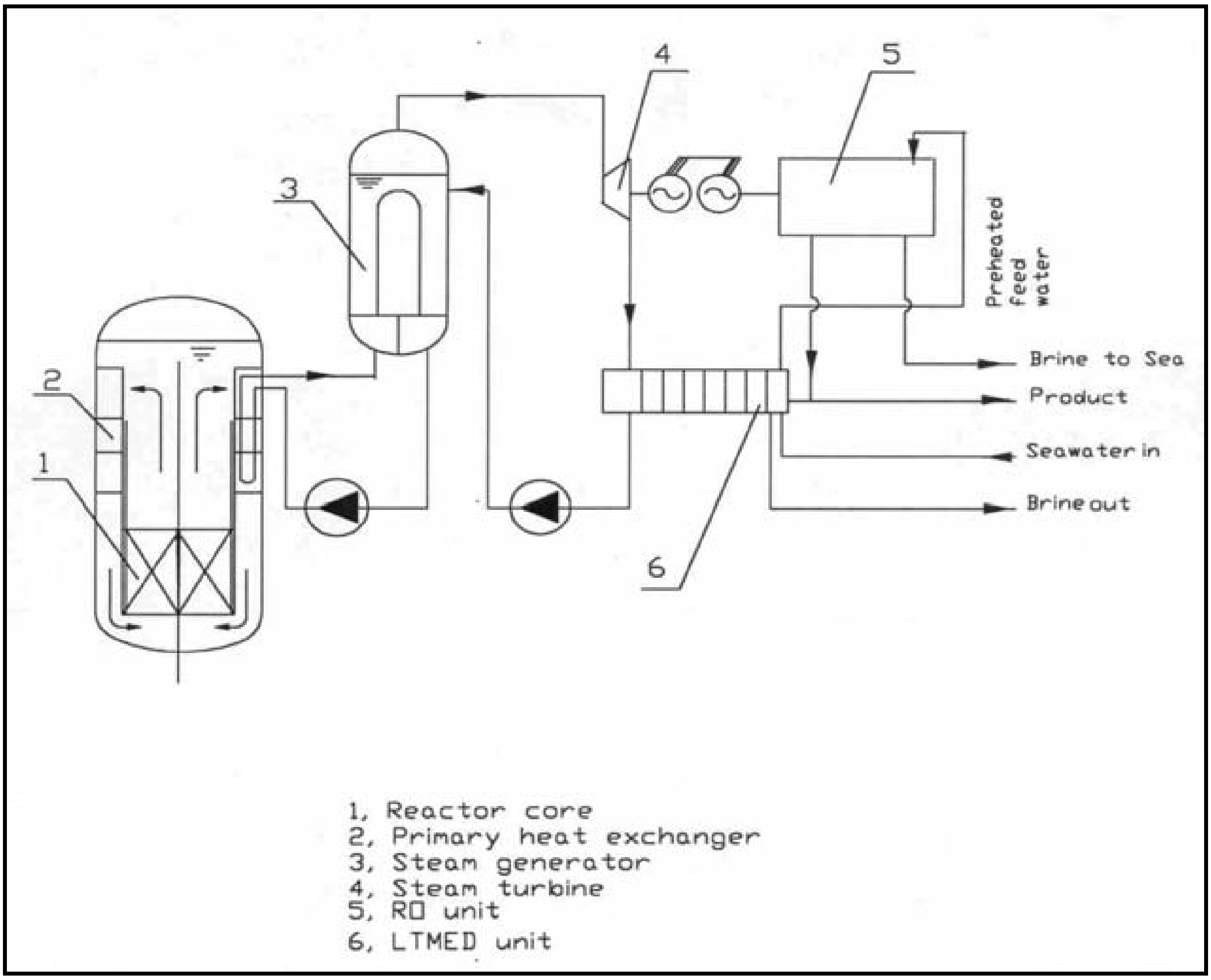


Figure 3.23: Coupling scheme of NHR Reactor with LTMED and RO processes.

The produced electricity by the turbine-generator of the nuclear plant was used to supply power to the motor of the compressor in the VC unit and the nuclear power plant itself. The total capacity of the selected MED/VC process depends on the maximum electricity production of the NHR heating reactor, and the number of MED/VC unit depends on the available unit capacity of MED/VC process.

3.3.2. Coupling of HTME and RUTA reactor

Schematic diagram of the RUTA nuclear desalination complex is illustrated in Figure 3.25. The reactor facility design envisages a two-circuit structure: primary circuit, which is in-pile core cooling circuit, the secondary intermediate circuit, being responsible for heat removal from the reactor and its transfer to the tertiary, consumer's circuit, while for the NDC it is the heating agent circuit for distillation desalination plants of the DOU GTPA type. Heat from the primary circuit to the secondary one and from the secondary to the tertiary one is fed through surface-type heat exchangers. Low parameters of coolant in the circuits permit using heat exchangers manufactured from aluminum alloys. Similar plate-type heat exchangers with intensifying corrugation will be used as the primary (1/2 circuits) and intermediate (2/3 circuits) ones.

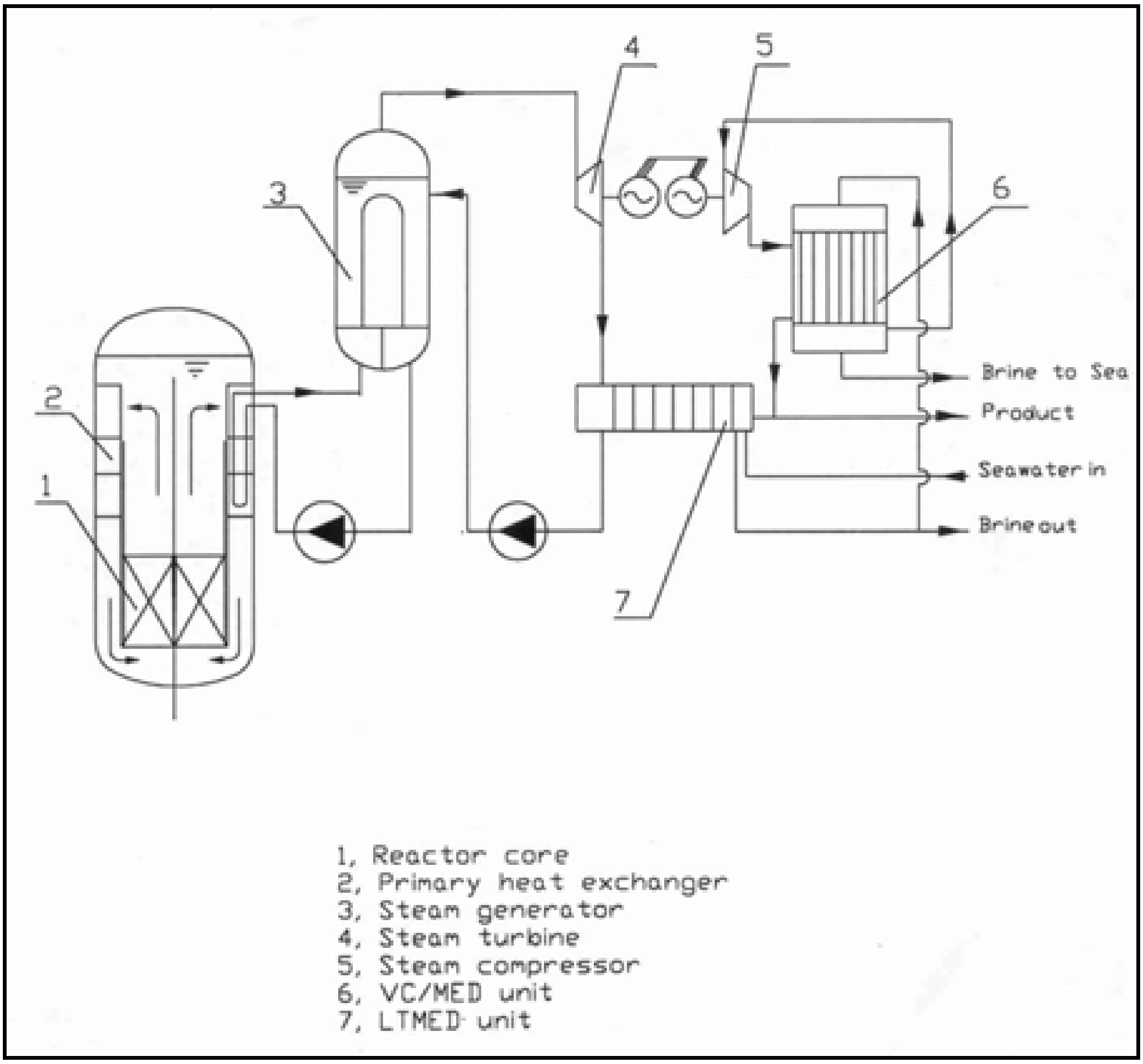
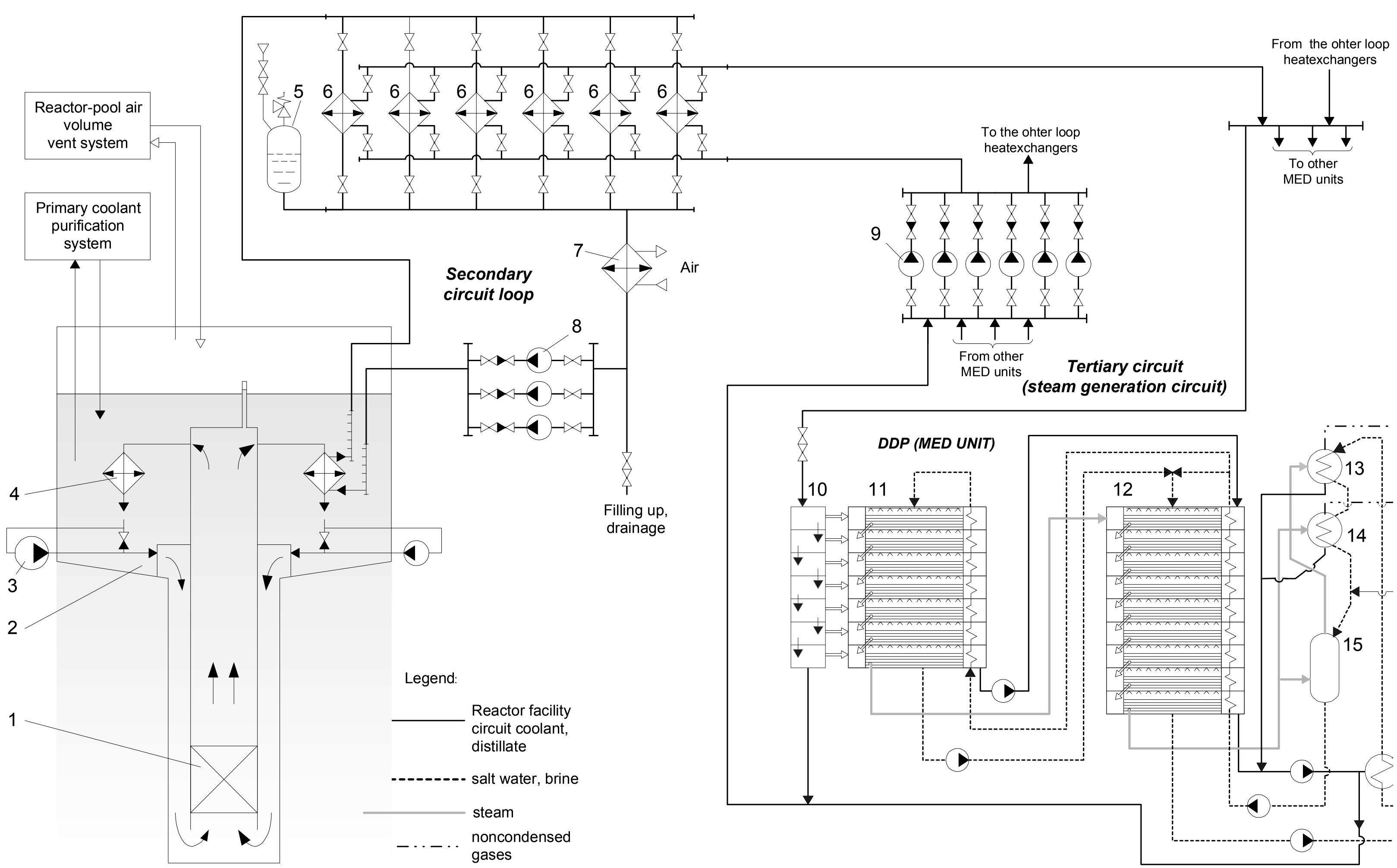


Figure 3.24: Coupling scheme of NHR reactor with LTMED and MED/VC.

The secondary circuit consists of two independent circulation loops. Each loop is a leak-tight closed system integrating primary and network heat exchangers, circulation pumps, pipelines and valves. A gas pressurizer is envisaged for each of the loops to maintain pressure in the secondary circuit and to compensate for temperature-induced coolant expansion. The RF secondary circuit is also used for the reactor cooling down (normal and emergency).

The secondary circuit components are arranged so that ”hot” pipelines are rising, while “cold” ones are downcoming. The 2/3 circuits’ heat exchangers are arranged at elevations providing natural circulation head in the secondary circuit for the reactor cooling down under conditions involving long-term loss of power supply. Decay heat is removed from the secondary circuit convectors to the ultimate heat sink (atmospheric air) using the air system of emergency cooling down (ASEC) with natural convection of the air in the convector rooms. Opening of the ASEC air shutters occurs passively via a direct-acting device.

The RF flow sheet envisages auxiliary systems. A water clean-up system is envisaged for maintaining the primary water quality at a level required. The system incorporates pumps, mechanical and ion-exchange filters, equipment for their regeneration, a heat exchanger.



1 – core; 2 – reactor pool; 3 – primary circulation pump; 4 – primary heat exchanger; 5 – secondary pressurizer; 6 –2/3 circuits heat exchanger; 7 – heat exchanger of the emergency air cooling down system; 8 – secondary circulation pump; 9 – tertiary circulation pump; 10 – self-evaporator; 11,12 – units of evaporating stages of DDP (BT1 and BT2); 13 – deaerator flash steam condenser; 14 – last stage flash steam condenser; 15 – deaerator; 16 – water-ejection unit; 17 – descaler dosing system; 18 – distillate cooler

Figure 3.25: Flow diagram of desalination complex with RUTA reactor.

The equipment is arranged in a separate room beyond the reactor. Intake of water to be cleaned up is envisaged from the lower section of the reactor pool, treated water being returned to the upper section of the pool.

Ventilation of the reactor air volume is envisaged in the flow sheet to remove hydrogen evolved as a result of water radiolysis. Air from the air volume is exhausted and via electric heater is supplied to the detonating mixture burner (DMB), after that the air is cooled in the condensing heat exchanger-condenser. The condensate is returned to the pool, while the air, after passing through iodine and aerosol filters, is returned to the pool air volume. Hydrogen content in the air volume of the pool and DMB is constantly monitored in the course of the system operation. Air activity monitoring is arranged, as well. The ventilation system is also used for sampling of fuel elements for leak-tightness monitoring.

The entire primary circuit components and systems containing radioactive substances are arranged within leak-tight compartments. The tertiary circuit is designed for heat transfer to consumers, therefore, stricter requirements are made for its "cleanliness". Basic equipment of the tertiary circuit within the RF incorporates heat exchangers, pumps, and valves. If more than one RUTA reactor operate within a nuclear plant, the tertiary circuit is common for all the reactors and common-plant pumps are used to circulate water through all heat exchangers connected in parallel.

In order to arrange joint operation of the RUTA reactor facility and desalination equipment, the preliminary analysis of possible integration schemes was made. A variant providing the most efficient use of low-potential heat energy of the reactor was chosen, i.e. as high as possible nuclear desalination complex (NDC) output and the best economic indices.

The DOU GTPA-type HTME desalination plants are used within the RUTA-equipped NDC. Steam is the heating agent in evaporating sections of DOU GTPA. According to the scheme integrating the reactor and desalination plant the tertiary circuit is a steam-generation circuit (SGC), which is a system of steam generation loops connected in parallel, their number corresponding to the number of distillation plants within NDC. Each SGC loop provides steam for the relevant desalination plant. Here we consider:

- 4 DDP units within NDC with the RUTA-55 reactor;
- 5 DDP units within NDC with the RUTA-70 reactor.

Steam is generated as a result of the tertiary circuit water flashing in a multi-stage self-evaporator, whence in parallel flows it is fed for heating the DDP head effects. The self-evaporator is arranged in direct proximity of the DDP evaporating effects cascade, thus reducing to a minimum the temperature head loss for steam transport. There is no stop and control valves in the "self-evaporator stage — DDP effect" steam lines, as the self-evaporator, being integrated in the tertiary circuit, is simultaneously an element of the DOU GTPA.

Remaining water flows of the tertiary circuit downstream of the self-evaporator outlets converge in a collecting header and then the coolant is fed to the inlet of pumps in the tertiary circuit and distributed over heat exchangers. Pipelines, by which the circuit distillate makeup is provided from the desalination plants, are also connected to the collecting header.

The following modifications in the standard design of the DOU GTPA are envisaged for integration with the RUTA reactor:

- each DDP is to be equipped with preliminary multi-stage self-evaporator, where partial water evaporation of the NDC tertiary circuit takes place for feeding heating steam to DDP:
- for NDC with the RUTA-55 the DDP is to be equipped with a 9-stage self-evaporator;
- for NDC with the RUTA-70 the self-evaporator should have 6–7 stages;
- in contrast to a "typical" DOU GTPA, which has only one (first) effect heated by steam from an external source, in the modified plant the heating steam is fed by parallel flows from each of the self-evaporator stages to the relevant DDP effect, i.e. in the head DDP effects operating in the range of the tertiary circuit temperature drop, a mixture of steam received from the self-evaporator and secondary steam from the previous effect is the heating agent;
- the head effects of the modified DDP differ from the standard ones in reduced heat exchange surface area, as they are designed for under load operation due to decreased consumption of heating steam.

Parameters of the self-evaporator operation permit attaining the maximum temperature of brine boiling in the DDP head stage as 80–85°C (the lower value is referred to the RUTA-55 reactor with all-mode natural circulation in the reactor, the upper one corresponds to the maximum feasible temperature in case of forced mode of the RUTA-70 reactor). The optimal configuration of the complex envisages a power source with 2–3 RUTA units (depending on the demand) and a desalination system, featuring a 5–10% margin in terms of specified output.

3.4. COUPLING DESALINATION PLANTS TO RESEARCH REACTORS

The desalination unit working on low temperature vacuum evaporation process (LTE), developed and fabricated in BARC, is being coupled to CIRUS research reactor with a view to practically demonstrate the utilization of research reactor waste heat, available at low temperature, for the desalination of sea water.

The schematic diagram of the coupling arrangement of the LTE desalination plant with the research reactor is shown in Figure 3.26. The heat energy required for desalination of seawater in the desalination unit will be supplied by the PCW system of CIRUS reactor. An intermediate de-mineralized water circuit is provided to transfer heat from active PCW to the in-active desalination unit so as to avoid ingress of radioactive contamination from PCW to the desalination unit in the event of any leakage in the heat transfer equipment (heat exchanger). Seawater requirement of the unit will be met by the secondary coolant system of CIRUS reactor. The product water from the desalination unit will be stored in CIRUS under ground dump tank and utilized for demineralized water make up requirements of the reactor.

The required heat energy is transferred from PCW system to the intermediate circuit through a plate type heat exchanger (called intermediate heat exchanger). For this purpose PCW at a rate of 76.8 m^3/hour is tapped off from the core outlet line and supplied to primary side of the heat exchanger and outlet from the heat exchanger is connected to the main PCW heat exchanger outlet header/ recirculation pump suction header. At the rated power of the reactor, PCW enters the heat exchanger at 77°C and leaves the heat exchanger at 66°C.

The intermediate system uses demineralized water as the medium, which is re-circulated in a closed loop. The intermediate heat exchanger is located at the discharge of recirculation pump to ensure that the pressure of the water in the intermediate circuit is higher than the

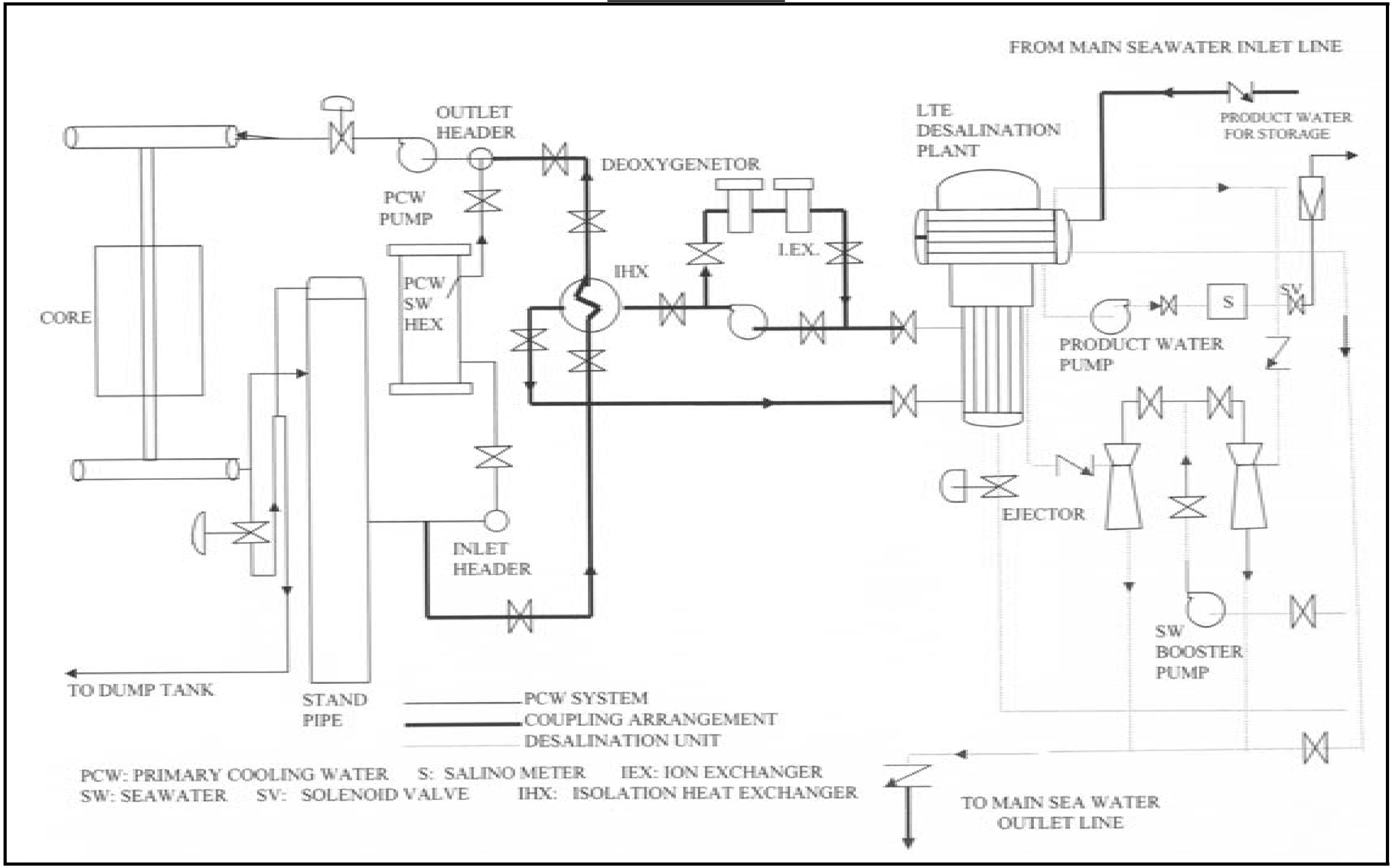


Figure 3.26: Schematic diagram of coupling LTE and CIRUS research reactor.

pressure of primary system water. This ensures that in case any leak develops in the heat exchanger, active primary water will not enter the non-active intermediate circuit. The recirculation flow through the system is maintained at 96 m^3/hour. At the rated power of the reactor intermediate coolant enter the heat exchanger at a temperature of 56°C and leaves at a temperature of 65°C. The heat exchanger outlet is connected to the shell side of the heater section of the desalination unit and outlet from the heater section is led back to the pump suction. A purification circuit consisting of deoxygenerator and mixed bed ion-exchanger units has been provided in the system to maintain the system chemistry.

Seawater requirement to the desalination unit is tapped off from the main seawater inlet line to CIRUS and outlet from the unit is connected to the seawater return line. Seawater at a rate of 120 m^3/hour and ambient temperature (29°C) enters the tube side of condenser where it is pre heated to 35°C by condensing vapors. At the condenser outlet about 66 m^3/hour is pumped to seawater jet ejectors and about 5.1 m^3/hour enters the tube side of the heater section. Remaining seawater directly goes to the seawater outlet header. Seawater entering the tubes of the heater is heated by intermediate coolant entering in the shell side at a temperature of 65°C. The amount of seawater getting evaporated and condensed into product water depends on the temperature of the intermediate coolant entering the heater section. The unevaporated and concentrated sea water from heater outlet along with sea water passing through the jet ejectors and bypass flow after condenser are joined and discharged as sea water outlet from the unit. In the event of failure of seawater jet ejectors, vacuum in the unit cannot be maintained resulting in flooding of the unit with seawater. To prevent the same, provision has been made to automatically cut off sea water supply to the heater section, in case the sea water level in the unit goes beyond a preset level.

The discharge of seawater into the main seawater outlet of CIRUS will develop a back-pressure at the outlet of seawater jet ejectors. Since, by design, the pressure at ejector outlet has to be maintained at atmospheric value, the unit elevation is increased appropriately (1.5 meters) to affect gravity flow from desalination unit to the seawater outlet.

The product water from the desalination unit is pumped to and stored in an under ground storage tank after ensuring that the Total Dissolved Solids (TDS) content is less than 10 ppm. An online conductivity meter will be provided to check the quality of product water. In case conductivity is above the specified limit, the product water will be automatically diverted to seawater outlet. The product water will be passed through CIRUS polishing plant to achieve the desired chemistry parameters and then transferred to another storage tank to meet the make up requirements of CIRUS PCW system. The entire demineralized water make up requirement of the reactor can be met by the supply from this desalination unit. The product water output at the rated reactor power (40 MW) is around 36 Ton/day.

In case of a reactor trip or stoppage of intermediate recirculation pump, no heat is supplied to the unit resulting in stoppage of product water production. Product water level in the condenser will come down resulting in gas locking of product water pump. To prevent this, provision is incorporated to trip the product water pump in case the product water level in the condenser unit goes below a preset value. An alarm annunciation has been provided at high level of product water in the condenser to adjust its flow to match the rate of production in the unit.

Safety implications of coupling the desalination unit to the primary cooling water and sea water systems of CIRUS reactor was analyzed for conditions of normal operation, operational transients and non-availability of desalination unit.

During normal operation, primary cooling water pressure at the inlet to the core is maintained at a constant value with the help of control valves provided at the recirculation pumps discharge and the pressure at the core outlet is maintained by keeping constant level in the stand pipe. This ensures maintenance of constant differential pressure across the core even after introducing a parallel circuit across PCW/SW heat exchangers for supplying heat to the desalination unit because of which the core flow and PCW system flow remains unchanged.

Due to coupling of the desalination unit, the flow through main PCW/SW heat exchangers will reduce by 7% on the primary side and by 1.5% on the secondary side. This will result in an increase in PCW temperature at the core inlet from 45°C to 46°C. The core outlet temperature will also increase by the same extent.

During loss of normal power supply, PCW pumps, sea water pumps and desalination unit pumps will trip along with the reactor and shut down cooling flow through the core is established by gravity flow of water from the ball tank to the dump tank. Since coupling of desalination unit is in a region, which is not a part of the shutdown cooling path, there will not be any change in the shut down cooling flow of the reactor.

The total kinetic energy (KE) of the PCW system comprising of KE of flowing water (1.1×10^5 Nm) and KE transmitted to water by pump motor assemblies (8.7×10^5 Nm) is 9.8×10^5 Nm. Subsequent to the modifications, the KE of flowing water will decrease marginally by 287 Nm while KE transmitted by pump motor assemblies will remain unchanged. Accordingly, net change in KE of the system would be very small (≈ 0.03%) and flow coast down characteristics of the system would remain practically unchanged.

Non-availability of desalination unit can result due to tripping of intermediate system pump, stoppage of seawater flow to desalination unit and loss of vacuum in the evaporator. Under these conditions, PCW flow through the intermediate heat exchanger will continue, but the heat load removed in the desalination unit will also have to be removed by the main PCW/SW heat exchangers resulting in rise in PCW temperatures and SW temperature at heat exchanger outlet. PCW temperature at core inlet will increase from 46°C to 47.3°C, which is acceptable since reactor trip setting is at 51°C. Seawater temperature will also increase marginally by about 0.5°C. However this increase in temperature of PCW can be avoided by stopping PCW flow to the intermediate heat exchanger and diverting the entire flow through the main heat exchangers.

REFERENCES TO CHAPTER 3

[3.1] HUMPHRIES, J.R., NATALIZIO, A., LaFORTUNE, J.F. and MIDDLETON, E., "The CANDESAL Nuclear Desalination System", presented at IAEA Technical Committee Meeting and Workshop on Coupling Aspects of Nuclear Reactors with Seawater Desalination Processes CI-9309, Vienna 13–17 September 1993, Candesal Enterprises Ltd., Ottawa (1993).

[3.2] INTERNATIONAL ATOMIC ENERGY AGENCY, Introduction of Nuclear Desalination. A Guidebook, Technical Reports Series No. 400, IAEA, Vienna (2000).

[3.3] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Applications for Steam and Hot Water Supply, IAEA-TECDOC-615, IAEA, Vienna (1991).

CHAPTER 4. ANALYTICAL EVALUATIONS AND EXPERIMENTAL VALIDATIONS

The use of a nuclear power plant as an energy source for desalination plants has not been extensively exploited although it has been used successfully on a limited scale. Therefore, there are some specific issues related to the coupling between the power plant and the desalination plant that has to be addressed before any large-scale commercial application. These include:

- Technical aspects such as:
    - Desirable features of nuclear desalination plants.
    - Design consideration for coupling with nuclear reactors.
    - Technical protection measures against product water contamination including monitoring and isolation.
    - Reliability and flexibility requirements.
- Safety and environmental considerations.
- Economics of nuclear desalination.

The activities that took place within this CRP have aimed at addressing these issues. Computer programs were developed and/or upgraded to facilitate safety analyses and economic assessments of different coupling schemes. Experimental facilities were constructed and/or experimental programmes were carried out to validate performance, safety and economic assumptions.

This Chapter presents a summary of the analytical evaluations including computer tools/programs developed, and experimental validations including experimental facilities and experimental programmes as well as the results attained analytically and experimentally in the fields of safety, performance and economics.

4.1. ANALYTICAL EVALUATIONS

4.1.1. Safety assessment

Although the coupling of a desalination system to a nuclear power plant is unlikely to introduce any additional risk in terms of reactor safety, it is still a major plant modification and a review of the safety analysis of the plant is needed. This implies verifying that the coupling does not introduce an unacceptable path for radiological contamination towards the critical group and that transients in the desalination system will not impair the safety of the nuclear plant. This requires both deterministic and probabilistic analysis. The deterministic assessment could be made by different procedures, analytical or computer-based, but there are clear advantages by using an approach close in style and tools to the safety assessment of nuclear power plants. Therefore it is reasonable to analyze the use of the codes most widely used in nuclear safety analysis, RETRAN and RELAP for instance.

INVAP, Argentina developed a modeling tool, DESNU, that enables production of RETRAN input files for the modeling of an NDP without requiring specialized knowledge. The input parameters of the spreadsheet are mainly operational and geometrical data easily available, and the model produced has a coherent nodalization allowing conceptual assessment of NDP designs. DESNU was effectively reviewed by its application to specific projects at KAERI (Republic of Korea) and IPPE (Russian Federation). The DESNU modeling tool and results of the safety analyses, as well as other safety analyses are presented below.

4.1.1.1. DESNU Modeling tool for safety analysis

Two of the most accepted codes for analysis of the nuclear reactor safety are RETRAN-02 and RELAP-5. In order to keep compatibility, these codes should be also used for modeling a generic desalination plant coupled with a generic Reactor. Thus enabling the simulation of accidents with loss of isolation barriers between the nuclear power plant and the product water.

When a set of models is involved (several transients of the same system or several similar alternative systems), it is extremely convenient to invest some time to develop a modeling tool that would work as a support for the elaboration of input files. Thus reducing input requirements on geometry and initial conditions, and therefore input errors. The practice of producing a modeling tool for optimizing the production of computer-simulation input-files has been used by INVAP, Argentina for CAREM-plant models, and for the development of models for the high-pressure natural-circulation test-loop (CAPCN).

DESNU is a modeling tool intended for the development of simulation models (input files) of Nuclear and Desalination systems, with a specific technique for considering the coupling. It is a spreadsheet with separate sheets for each component or sub-system, linked with each other for related data, and having, as an output, a final sheet with RETRAN input files.

The final output is the simulation models, that are useful in determining the transport mechanisms of contamination, and therefore, for the definition of requirements on coupling related issues: isolation systems, intermediate loops, safety triggering sequences, etc. Nevertheless, the latter are outputs from a model of a specific plant, elaborated by the End-user. It must be pointed out that in terms of the output for this CRP, the spreadsheet is, in itself, a product valuable for the technical groups working on the evaluation of the safety aspects of nuclear desalination systems. It must be kept in mind that DESNU modeling tool in no way replaces the expertise needed to elaborate safety-quality simulation models, it is only intended to be a support for the user in terms of achieving usable models in a shorter time.

DESNU consists mainly of three sections and an appendix.

- Section 1, “General Data”, is organized in separate sheets for components, where the user defines the input data of each component of the model.
- Section 2, “Nodalization”, is organized in separate sheets for input modules (volumes, junctions and conductors), and a special sheet for the nodalization of the Steam Generator.
- Section 3, “RETRAN and/or RELAP input files”, is organized in separate sheets providing three input files.
- Appendix includes sheets with auxiliary data for unit conversion and physical properties. Details of the sheets within each Section can be found in Table 4.1.

RELAP-5 is the well-known US system code applied to best-estimate transient calculations of LWR coolant systems. The code is capable of simulating two-phase systems for a wide range of accidents and transients. However, the capability of the RELAP-5 transport model is restricted by the cases of boron tracking within the liquid phase of the coolant and tracking of non-condensable gases within the vapor phase. Thus, this code cannot be directly applied to solve contamination transport problems related to nuclear desalination coupled systems (NDCSs), because for such problems contamination transport within both liquid and vapor phases of the coolant should be considered. The possible ways of overcoming this restriction of the RELAP-5 transport model could be:

(1) On-line calculation of contamination transport using the control variable tool of RELAP-5. This approach allows user to perform just simplified conservative estimations of contamination spreading over the NDCS's coolant circuits.

TABLE 4.1: LIST OF CALCULATING SHEETS OF DESNU SPREADSHEET

Name of the sheet	Content or description
Section 1	**Description of component**
Scheme	Diagram of the NPP BOP
Steam line	Geometry of SG outlet, headers, steam lines up to the turbine
Steam valves	Data of safety vales, relief valves and by-pass valves
Turb.	Geometry and operating conditions of stages and extractions
Cond.	Geometry and operating conditions of steam and seawater sides
Cond. Sys.	Data of condensate pump / piping and low pressure pre-heater
FWS	Data of FW pump / tank and high pressure pre-heaters
Fwpip	Geometry of FW piping
MEDsys	Data of MED Desalination Plant
MSFsys	Data of MSF Desalination Plant
ROsys	Data of RO Desalination Plant
SteamGen	Geometry and operating conditions of primary and secondary sides
Section 2	**Specific nodalization modules**
Nodal.Vol.	Control Volumes of BOP, MED and MSF Plants
Nodal.Jun.	Junctions between volumes of BOP , MED and MSF Plants
Nodal.HX	Parameters of pre-heaters and main heat HX of MED and MSF Plants
NodalSG	Parameters of control volumes, junctions and HX surfaces
Section 3	**RETRAN input files**
MEDinp	RETRAN input of isolated MED Plant modelling
MEDinp.R5 (*)	RELAP input of isolated MED Plant modelling
MSFinp	RETRAN input of isolated MSF Plant modelling
MSFinp.R5 (*)	RELAP input of isolated MSF Plant modelling
Roinp	RETRAN input of isolated RO Plant modelling
Roinp.R5 (*)	RELAP input of isolated RO Plant modelling
SECInput	RETRAN input of NPP BOP isolated modelling
SECInput.R5	RELAP input of NPP BOP isolated modelling
Coupled	RETRAN input of coupled NPP-MED Plant modelling
Appendix	**Auxiliary data**
Factors	Units conversion factors
Props	Physical properties of water and materials

(*): The task of elaborating these sheets for extending DESNU in order to produce a RELAP models of the isolated Desalination Plants has been carried out by participants from the IPPE, Russian Federation.

(2) Off-line calculation of contamination transport based on the pre-calculated results of RELAP-5 accident simulation. This approach requires availability of a stand-alone computer code of contamination transport.
(3) On-line calculation of contamination transport using a coupled code system (RELAP-5 + transport code). This approach requires a code-coupling interface to be developed and used in calculations.

Because of the differences in the input requirements of RELAP-5 and RETRAN-02 , IPPE, Russia assessed the efficiency of the DESNU spreadsheet as a tool for generation of the RELAP-5 input deck. The following flexible features of RELAP-S input are important for such consideration:
(1) SI or British units can be used in the RELAP-5 input. Thus, the ability of the DESNU spreadsheet to convert SI units into British units is not needed in the case of RELAP-5 simulations.
(2) Sequential expansion input format of the RELAP-5 pipe component can be used to avoid redundancies in geometrical data and initial conditions.

Some features of RELAP-5 input requirements and models are not relevant to the DESNU efficiency as an input deck generator for NDCS models. but are to be taken into account while creating RELAP-5 modules (sheets) of DESNU. These features concern:
- Nodalization of non-regular shape components of the NDCS model,
- Contamination tracking within TDVs,
- IC (enthalpy as a state variable),
- BC (modeling of the fill components, connections to the time-dependent volumes (TDVs), TDV external references).

IPPE found that DESNU produces a NDCS model, which is quite rigid in nodalization including the interconnection of components. Geometrical and operational parameters of the model can be easily modified, but the model itself can not. NDCS model of DESNU does not contain an intermediate circuit between the secondary side of the nuclear power plant (NPP) and desalination plant (DP), which is included into the Russian design of the floating NDSC with the KLT-40 reactors. Therefore, not all of the RELAP-5 input data for the NDCS model could be produced using the DESNU spreadsheet. Subsequent direct editing of the DESNU-generated file was required to finalize the RELAP-5 input deck prior to calculations. Additionally, the input for the employed model of contamination transport was required.

RELAP-5 input deck for the NPP model has been produced using the DESNU tool and calculation of the steady-state parameters has been performed. The same NDCS model has been used for RELAP-5 simulations as been employed in RETRAN-02 simulations. Several components of the NDCS model have been directly modified according to specific input data requirements of RELAP-5. Table 4.2 shows the comparison of obtained results.

The design features of the SMART integrated desalination plant prevent the ingress of radioactive material into the desalination plant. However, the potential for carryover of radioactive materials from the reactor system to the desalination system must be assessed and reflected to the design. KAERI simulated the migration of the radioactive material from primary reactor coolant to the product water using DESNU modeling tool. Slight modification of the DESNIJ generated input file was needed to accommodate the SMART design.

TABLE 4.2: COMPARISON OF STEADY-STATE PARAMETERS

Parameter	Rated Value	RELAP-5 Calculated Value
SG Heat Transfer, MW	100	95.4
Turbine Inlet Flow. kg/s	48.7	48.0
Turbine Inlet Pressure, bar	47.1	44.7
Extraction 4 Pressure, bar	17.4	16.4
Extraction 3 Pressure, bar	9.47	9.1
Extraction 2 Pressure. Bar	3.18	3.12
Extraction 1 Pressure, bar	0.643	0.694
Condenser Operating Pressure, bar	0.00882	0.00857
Condenser Steam Flow, kg/s	34.961	33.1
Condensate Pump Flow, kg/s	37.91	37.3
SG Inlet Pressure, Mpa	6.0	5.5

The final outputs are the simulation models that are useful to determine the transport mechanism of contamination, and therefore for the definition of requirements on coupling related issues, isolation systems, intermediate loops, safety triggering sequences, and etc.

In the course of this CRP, KAERI's effort was focused on developing a RELAP-5 NPP model of the integrated SMART desalination plant that would simulate the coupling using the steam extracted from turbine. DESNU spreadsheet program was used to generate SMART model for simulation of the contaminant migration and RELAP-5 input file was generated with slight modification of the output sheet of the DESNU output file. The model was used for simulating accidents with loss of isolation barriers between the NPP and the product water.

4.1.1.2. Results of safety analysis

Analysis of contamination path

INVAP, Argentina performed some simulations in order to verify the RETRAN model obtained from the DESNU spreadsheet, in terms of issues as interconnection logic of control volumes, junctions and heat conductors, numerical methods and completeness of RETRAN input file. This allowed detecting and correcting parameters, for example, elevation inconsistencies.

The reference transient consisted in a first period meant to assure the steady state, and then:

- A contamination source is triggered in the steam generation outlet (400 s).
- Contamination spreads on the whole secondary system.
- The coupling heat exchanger is by-passed, simulating a rupture (600 s)
- Contamination migrates on the desalination plant.

The system, as shown in Figure 4.1, achieves the steady state, and the steam generation of the NPP is not sensitive to the coupling HX failure. In terms of contamination, it reaches immediately the turbine (diluted to a half by the flow of the other steam line), and then contaminates gradually the coupling HX. When this HX fails contamination migrates gradually through the effects of the Desalination Plant.

The integrated SMART nuclear desalination plant was modeled by KAERI, Korea, using modeling tool DESNU for the estimation of the transit time of the contaminant. The DESNU generated input file was modified by changing the connecting junction for desalination plant (MED) from volume 980 to volume 981 which represents the steam transformer (Figure 4.2). Except for this change, the nodalization remained same as that of the DESNU. Also, some modifications were made for RETRAN-03 input. For desalination plant, only one unit of MED and its coupling system was modeled. The failure of the isolation barrier of steam transformer was modeled by a direct irruption of contamination in the first effect. The present model does not include the thermal vapor compressor and the flow path of the induced steam from final condenser. Figures in Appendix III show the DESNU modeling used to simulate the transport of the radioactive contaminant of SMART plant.

An analysis on the multiple failure of the isolation barrier, i.e. both of the steam generator tube rupture and steam transformer tube breaks was conducted to estimate the transit time of the contaminant using RETRAN-03. The model of the integrated plant needs some tuning for simulation both of the steady state and the transient. The tuning work and the transient were performed in accordance with the procedure recommended by INVAP.

The path for contaminant migration was modeled by the bypass valve which flows from the primary side of the steam transformer to the first effect of the desalination plant. The flow velocity which is the most important parameter affects the time for contaminant migration was tuned by adjusting junction loss coefficient between each effect.

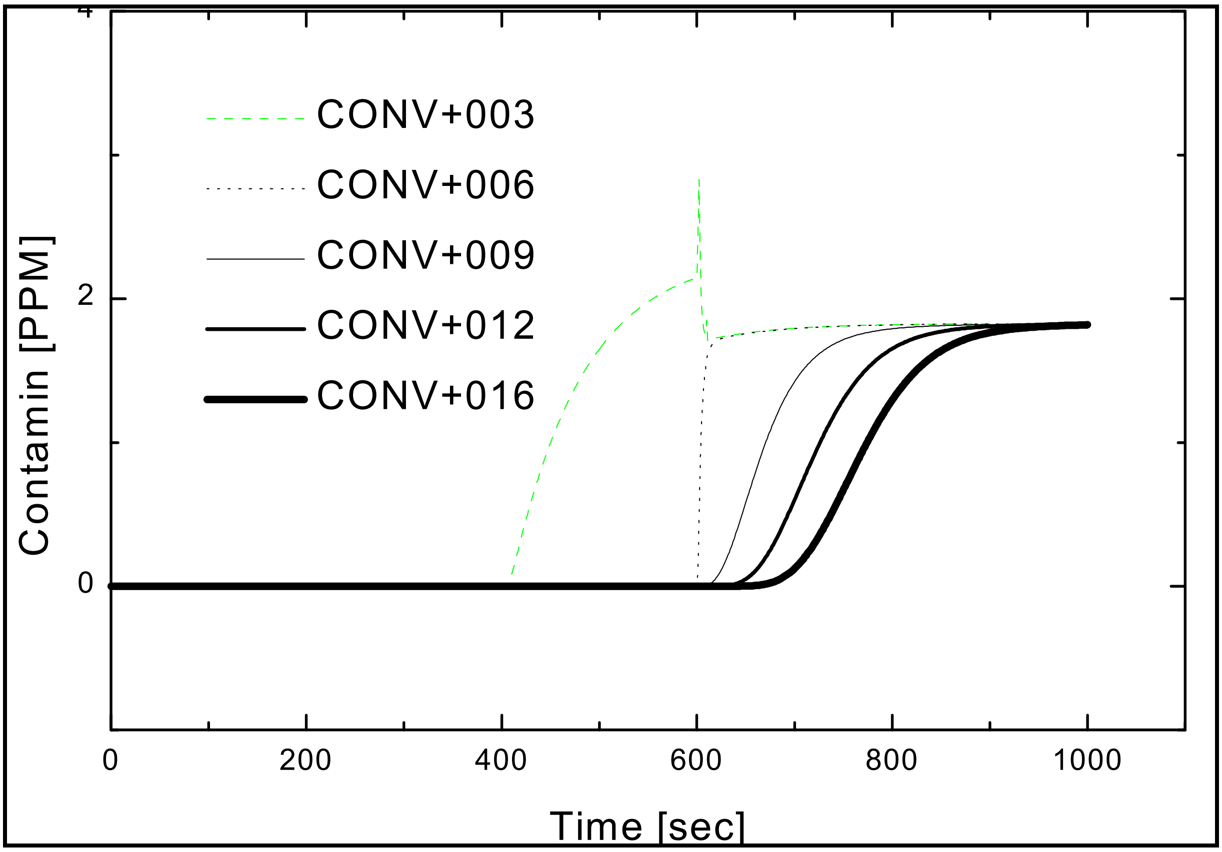


Figure 4.1: Contamination at the HX, 1st , 4th , 7th and 11th effects.

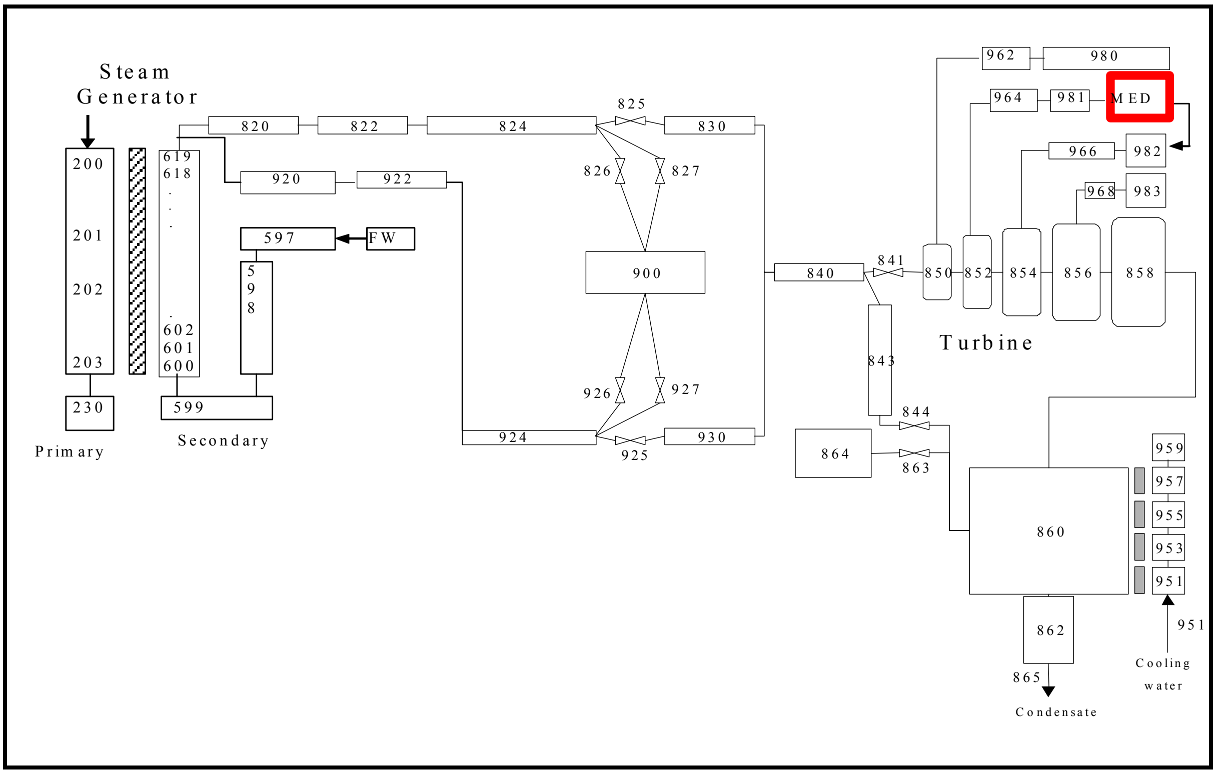


Figure 4.2: Control volumes and connections of Live Steam System and SG models.

A transient scenario similar to that adopted by INVAP was simulated by KAERI. The system achieves the steady state and the simulation was conducted by triggering contamination source into the steam generator outlet (Vol. 820) at 100 sec. After the contamination spreads on the whole secondary system, the contaminant flows into the desalination plant at 250 sec. by simulating a rupture of heat transfer tube of the steam transformer

As shown in Figure 4.3 and with reference to Figure 4.2, the contaminant reaches immediately to the turbine inlet (Vol. 850) at the half of the concentration by the flow of the other steam line and then contaminates gradually the primary side of the intermediate heat exchanger (steam transformer-Vol. 981). When the heat exchanger fails, the contaminant enters into the first effect (Vol. 6) and gradually spreads through the effects (Vol. 6 to 9) of the desalination plant, and then finally reaches to the water distribution grid. The estimated transit time of the contaminant is about 50.0 sec per effect. For the protection of the water distribution grid from the radioactivity contamination, the desalination plant should be isolated before the contaminant reaches to the water distribution grid. The isolation of the desalination plant from water distribution network may be triggered by the safety signal of the nuclear power plant and/or the additional radioactivity measurement in the first effect of the desalination plant. Since the thermal vapor compressor which mixes the supplied steam with the induced steam from the final condenser was not modeled for simulation, the level of concentration in each effects reaches the same level as that of the steam transformer.

Although this is a preliminary qualitative analysis, the results of simulation may provide useful information on the safety of the nuclear desalination, i.e. required time for isolation of desalination plant from water distribution network. For the practical application, more studies on the parameters which affects the contaminant transit time are necessary.

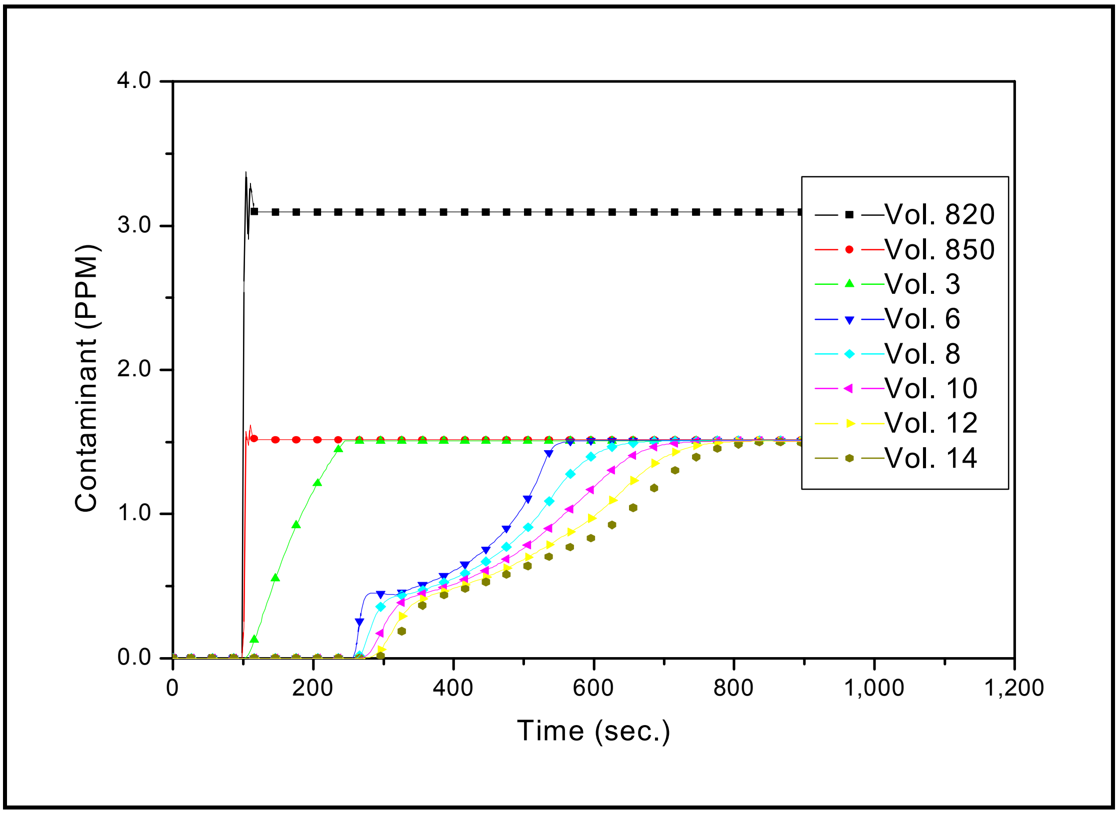


Figure 4.3: Simulation of the Contaminant Migration of SMART Nuclear Desalination Plant.

Transient analysis

The transients that occur in the desalination plant can impact the safety of the nuclear reactor due to the thermal-hydraulic interactions between the reactor and the desalination plant. The potential disturbances of the reactor integrated desalination plant depend on the characteristics of the desalination plant as well as the coupled system. For the current concept of the integrated nuclear desalination plant, the following three events were identified as the major potential disturbances imposed by the desalination plant.

- Turbine trip due to desalination system disturbances
- Excess load due to increased steam flow to the desalination system
- Loss of load due to the desalination system shutdown

The physical feedback of these events may cause the initiation or the design base events of SMART. The impacts of these disturbances on the Design Basis Accidents Events of SMART were evaluated by KAERI through conservative bounding approach of key safety parameters.

A safety analysis methodology was developed, including a set of DBEs based on the ANSI/ANS-51.1-1983 (R1988). A best estimate system analysis code. MARS/SMR 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 as determined for the safety evaluation of the SMART system. The limiting DBEs include steam line break, turbine trip, feed water line break, total loss of flow, SG tubes 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 feedwater line break, steam line break, and the SBLOCA gave the maximum peak pressure, minimum DNBR, and the minimum collapsed core level. It was thus confirmed that the SMART design has a sufficient safety margin.

Turbine trip due to desalination system disturbances

A turbine trip may be caused by the potential desalination system disturbances. This event generates a reactor trip signal either by a high primary pressure or a high secondary steam pressure trip signal. On reactor trip, the PRI-IRS comes into operation automatically to remove the core decay heat. Turbine trip causes the primary RCS pressure and the primary coolant temperature to increase, and thus the RCS peak pressure and the minimum DNBR are the major safety parameters of concern. Figure 4.4 shows the transient RCS pressure and DNBR during the turbine trip, and safety parameters are well within the design limits of 110% design pressure (18.7 MPa) and 1.30 of the SMART SAFDL on minimum DNBR.

Excess load due to increased steam flow to the desalination system

The increased steam flow to the desalination system causes the cool-down of the SMART primary RCS. The cool-down of the primary coolant then causes the core power to increase due to the large negative moderator temperature coefficient. The minimum DNBR is the prime safety concern during the excess load event. The excess load generates the high power reactor trip signal or low secondary steam pressure trip signal depending on the causes of the increase of the steam.

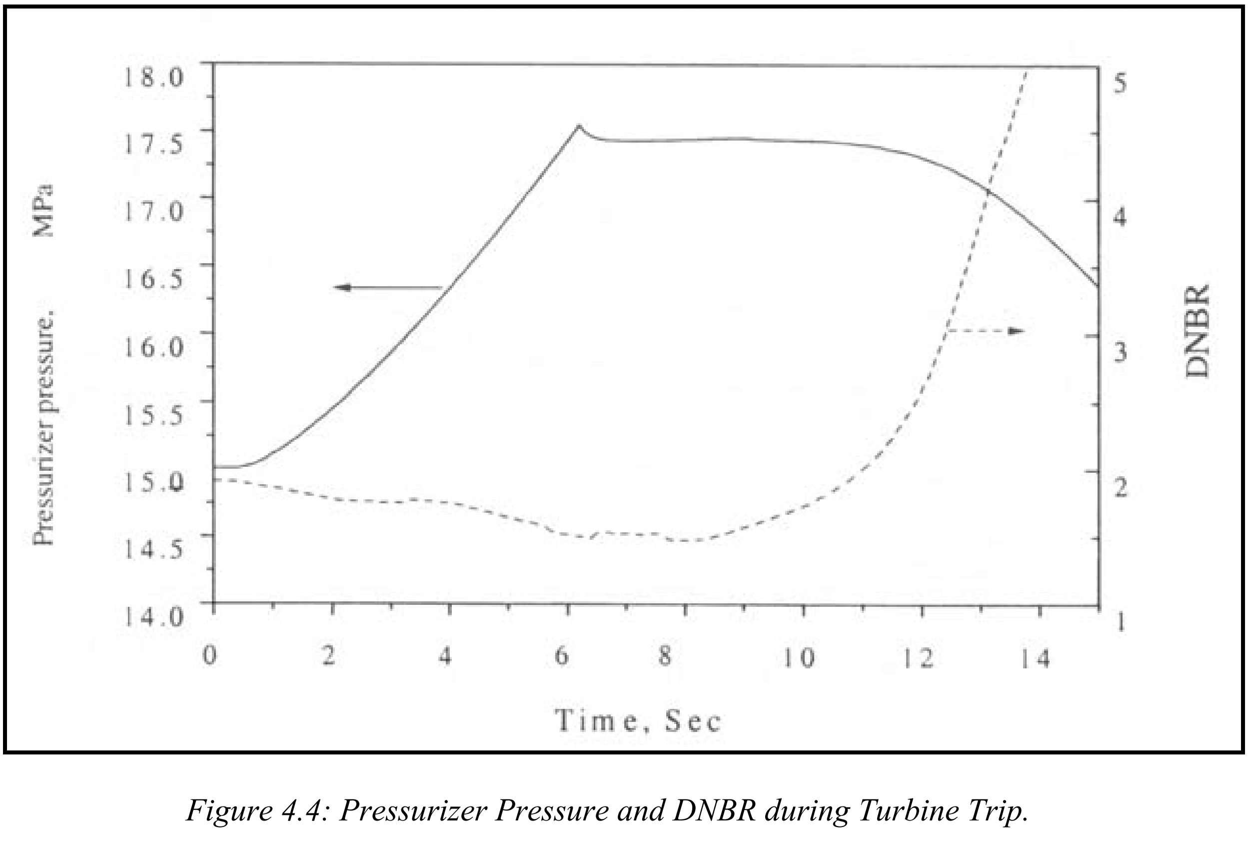


Figure 4.4: Pressurizer Pressure and DNBR during Turbine Trip.

The results of both events, the excess load and the desalination system pipe breaks, were bounded by the main steam line break (MSLB) accident. Figure 4.5 shows the transient minimum DNBR and demonstrates that the minimum DNBR is well within the SMART SAFDL of 1.30.

Loss of load due to the desalination system shutdown

A sudden stop of the steam flow to the desalination system due to the 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. Since the loss of load causes the RCS pressure and temperature to increase due to the degraded heat transfer by the secondary system, the peak RCS pressure is the major safety concern of the event. The loss of load due to the desalination system shutdown is bounded by the total loss of load by the secondary system or the turbine trip. The transient peak RCS pressure is also bounded by the feedwater line break accident, which is well below the safety limit of 110% design pressure.

4.1.2. Performance analyses

An important goal in the design of a nuclear desalination plant is the optimization of power and water production. On the other hand the design of new desalination plants requires detailed information on heat transfer, thermal efficiencies, Gain Output Ratio (GOR), ... etc. Within this CRP two computer programs were developed to study the thermal performance of the integrated nuclear desalination plant and/or the desalination system, namely: PLANTEFF for thermal performance and NHRVM for thermo-hydraulic analysis. These programs, together with the results of the analyses are briefly presented below.

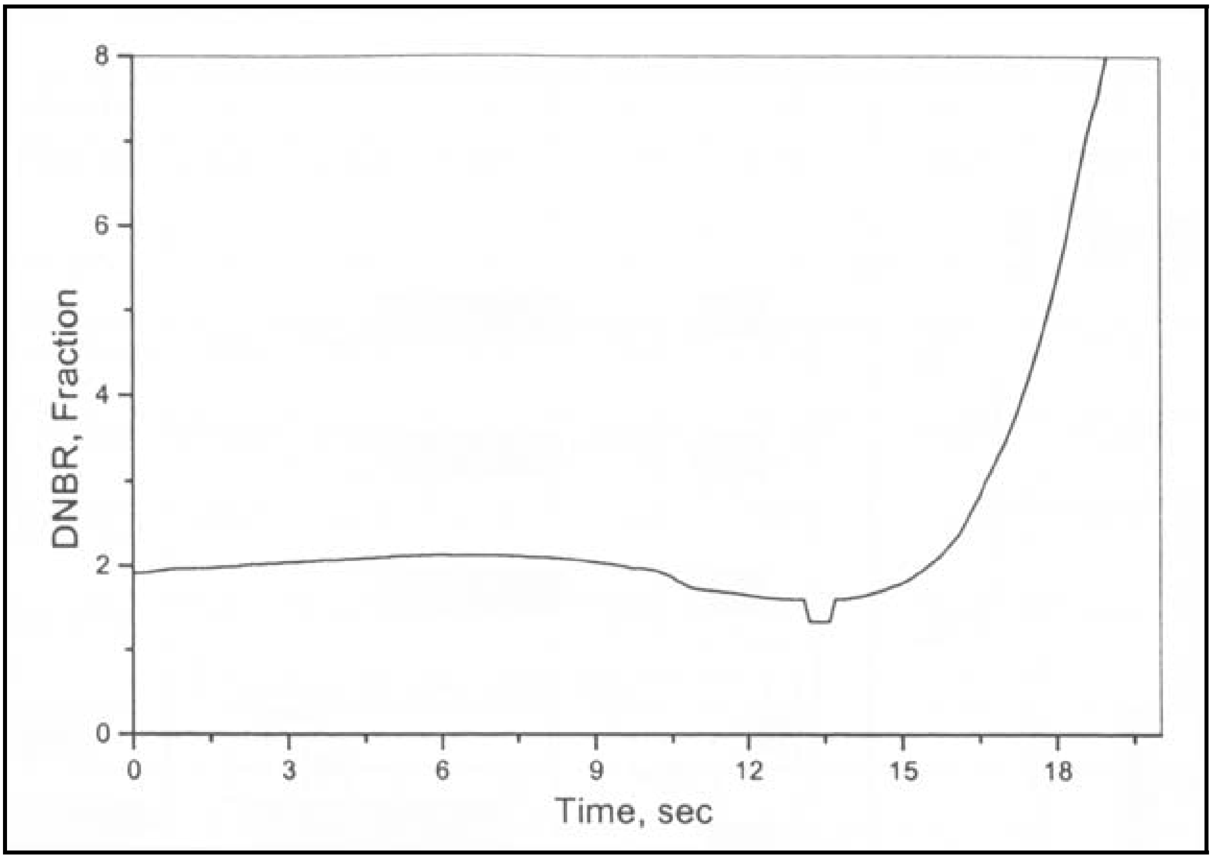


Figure 4.5: Transient DNBR during MSLB.

4.1.2.1. PLANTEFF for thermal performance

Computer program PLANTEFF was developed by KAERI, Korea to be used for evaluation of the heat and mass balance of the SMART Reactor secondary system coupled with desalination plant. The program includes the models for steam turbine and feedwater system, and uses ASME steam table for property calculation. Turbine was modelled with constant stage efficiency and steam extraction. The feedwater system models consist of three close-type pre-heater, pumps and condensation system. Heat losses of components in piping system are not considered in the program. Inputs on the turbine inlet conditions, internal efficiency, extraction pressure for feedwater heater and desalination plant, and the condenser pressure are required for the heat and mass balance calculation of a steam turbine system coupled with desalination plant.

Thermal balance calculations on the turbine cycle of SMART were conducted by PLANTEFF to investigate the optimum coupling scheme between SMART and the thermal desalination system. At the beginning, the base cycle utilizing the thermal energy (330 MW) of SMART to generate electricity. The results indicated that the turbine output at the rated reactor thermal power varied from 84 to 103 MW(e) when the turbine efficiency varied from 70 to 85%. The program was used to estimate the electricity generation and the water production for the various coupling schemes, i.e. MED and/or MSF process coupled through the back-pressure turbine and/or the steam extraction from turbine.

Back-pressure turbine cycle

The basic scheme of the cycle is the same as that of the base cycle except the use of a back-pressure turbine. All the exhaust steam is available to the desalination process. This cycle can produce a larger amount of desalted water than the target capacity (40 000 m^3/day), which make it suitable for situations where more water is needed than electricity**.** Figures 4.6 and 4.7 show the variation of plant output for backpressure turbine cycle coupling with MED and MSF respectively at turbine efficiency of 85%. For MED coupling, the backpressure turbine cycle can produce electric power from 67 to 79 MW(e) and desalted water from 85 000 to 145 000 m^3/day for the temperature range of 70–90°C which is suitable for the low temperature MED. For MSF coupling, the backpressure turbine cycle can produce electric power from 44 to 59 MW(e) and desalted water from 90 000 to 150 000 m^3/day for the temperature range of 110–130°C which is suitable for the MSF processes.

Turbine extraction cycle

This cycle extracts steam for the seawater desalination process from the turbine before it fully expands. Figures 4.8 and 4.9 show the variation of plant output for turbine extraction cycle coupling with MED and MSF respectively at turbine efficiency of 85%.

The calculations were carried out by varying the extraction temperature of steam at the target water production of 40 000 m^3/day. For MED coupling, the turbine extraction cycle can produce electric power from 90 to 92 MW(e) for the temperature range of 70–90°C. For MSF coupling, the turbine extraction cycle can produce electric power from 83 to 85 MW(e) in the temperature range of 110–130°C. This indicates that turbine extraction cycle coupled with MED process is a more suitable choice for the desalination capacity of 40 000 m^3/day.

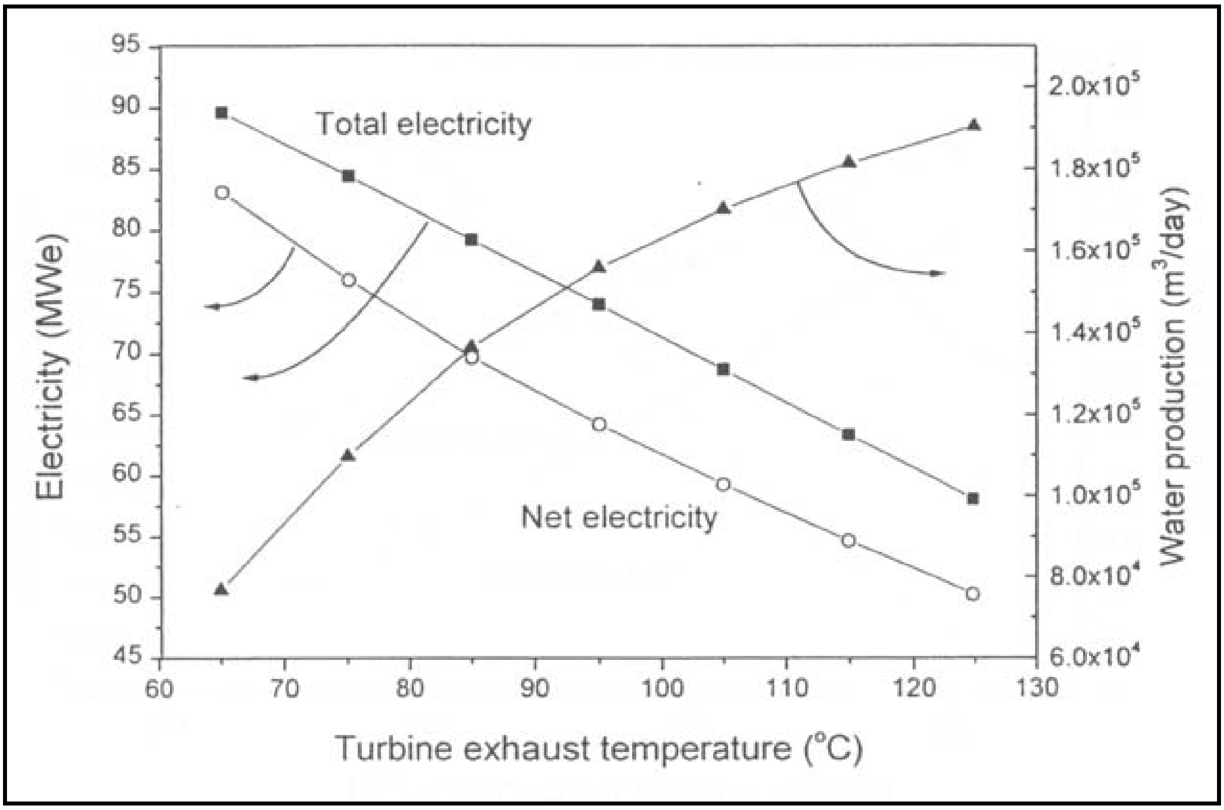


Figure 4.6: Variation of SMART Output Capacity for MED and Backpressure Turbine Cycle (IHX temperature difference 5 °C, turbine efficiency 85%, no bypass flow, 3 preheaters).

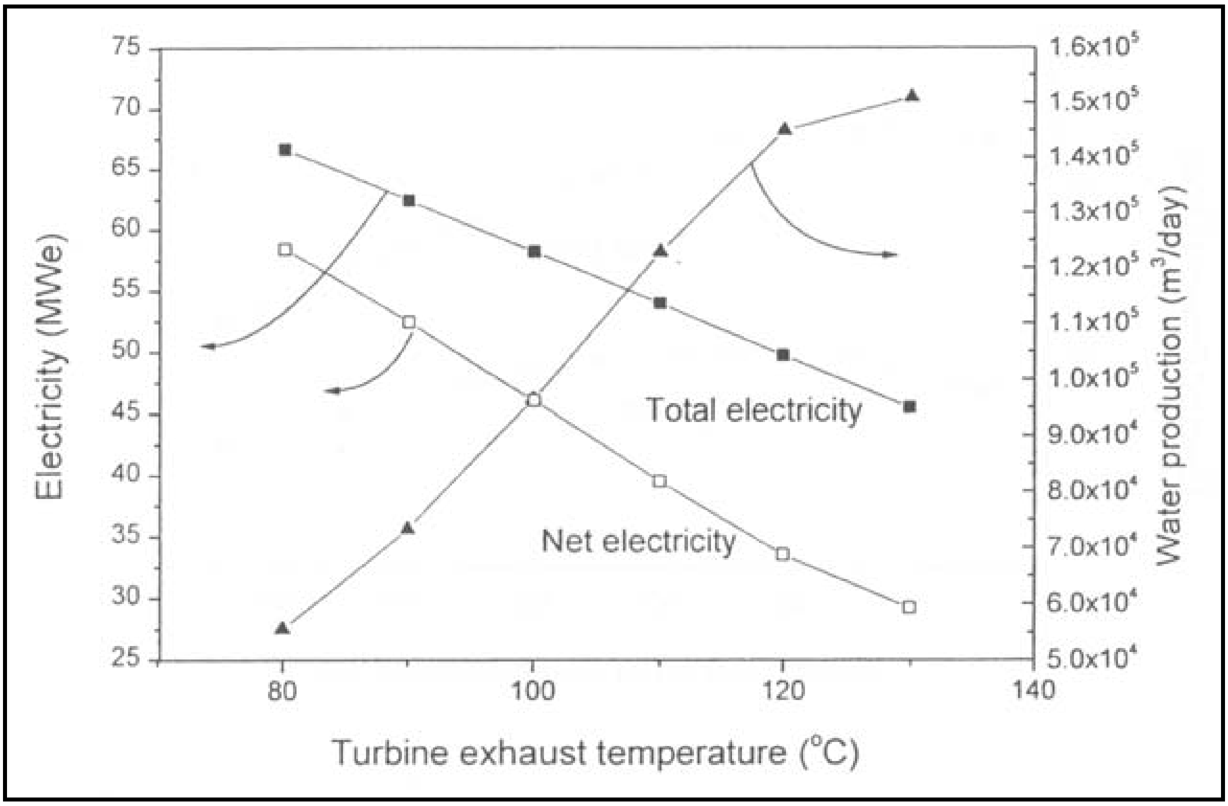


Figure 4.7: Variation of SMART Output Capacity for MSF and Backpressure Turbine Cycle (IHX not included, turbine efficiency 85%, no bypass flow, 3 preheaters).

4.1.2.2. NHRVM for thermo-hydraulic analysis

Due to lack of detailed design data necessary for evaluating thermal hydraulic performance of the selected VTE-MED process, INET, China carried out an extensive theoretical and experimental program to investigate thermal hydraulic performance of VTE-MED such as: heat transfer coefficients, heat transfer enhancement, thermal efficiency, GOR, operation characteristics, …etc. The computer code NHRVM was developed for analysis of the thermal hydraulic performance of the VTE-MED coupled with NHR-200 nuclear heating reactor.

The physical-mathematical model of the program includes governing equations for the evaporators. In the original version of the code, NHRVM-1, the heat transfer module for the evaporator tube was based on Chun and Seban straight smooth round tube correlation [4.1]. In the updated version of the code, NHRVM-2, an enhancement factor ranging from 1.4 to 2.0 was applied to the heat transfer module for the evaporator tube based on the experimental data of IDT Metropolitan concept design [4.2] and Innoshima [4.3–4.4] for the double fluted tubes heat transfer coefficients.

The physical model of the i(th) effect is shown in Figure 4.10. The model is solved numerically effect by effect to obtain the areas of the evaporator and pre-heater at every effect and the product water output flow rate at the conditions specified for inlet feed water, source vapor supplied by NHR-200 and the distribution scheme of temperature difference through an iterative process.

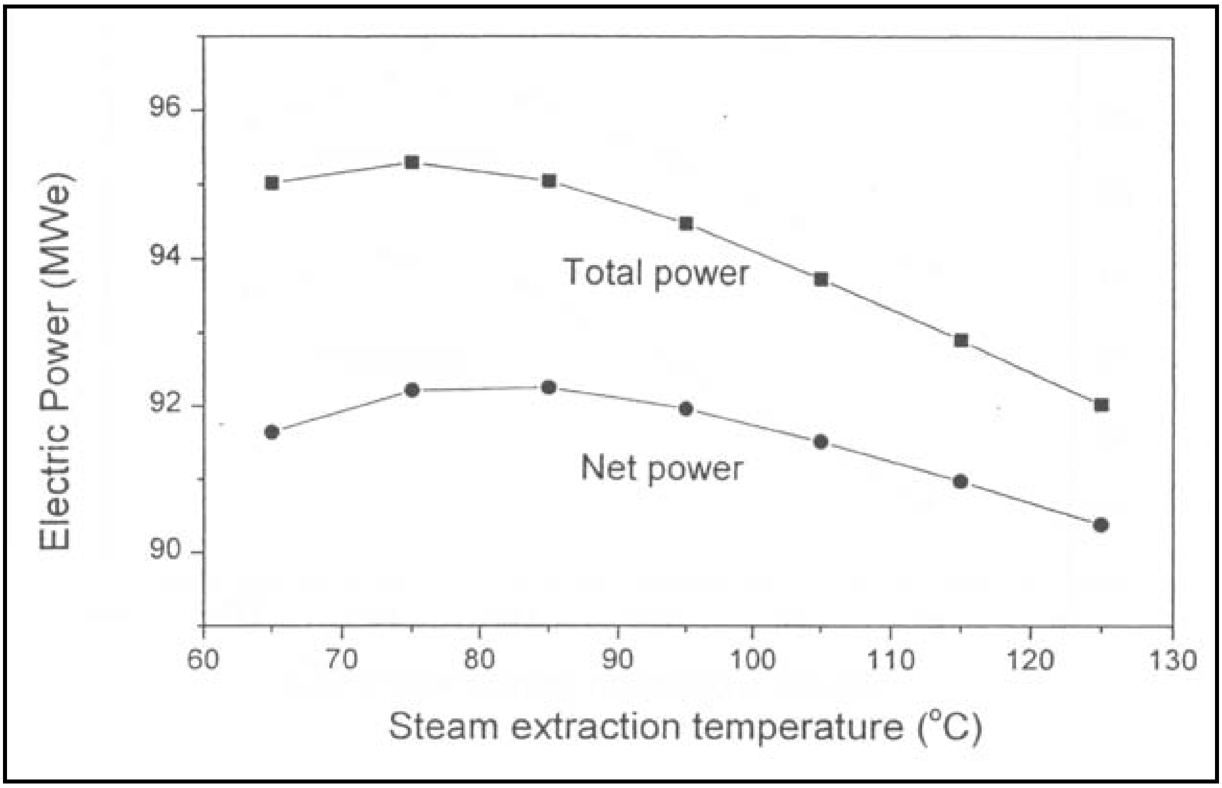


Figure 4.8: Variation of SMART Electric Power for MED and Turbine Extraction Cycle (IHX temperature difference 5°C, turbine efficiency 85%, water production 40 000 m^3/day).

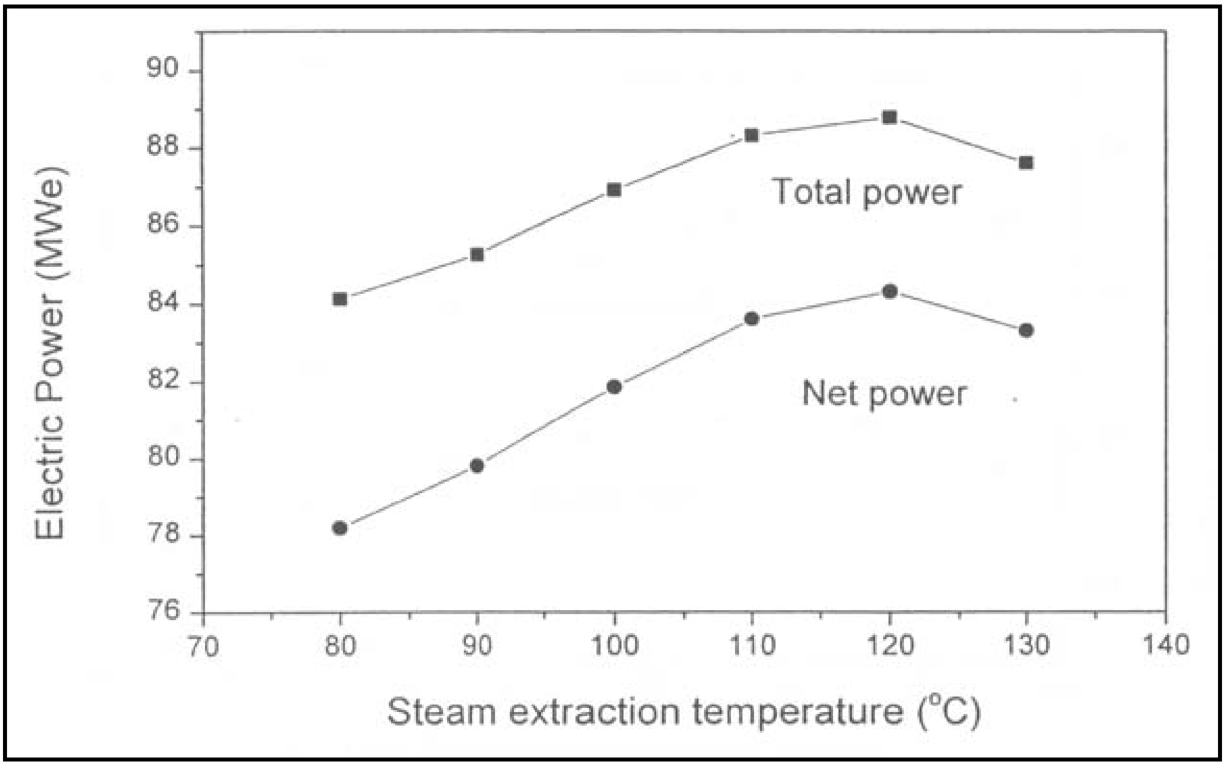


Figure 4.9: Variation of SMART Electric Power for MSF and Turbine Extraction Cycle (IHX not included, turbine efficiency 85%, water production 40 000 m^3/day).

NHRVM-1 was used to perform calculations for two schemes, namely: scheme of equal temperature distribution, and scheme of temperature difference distribution with in-section equal area of heat exchanger, based on input design parameters summarized in Table 4.3. The first scheme assumes uniform distribution of the total temperature difference to all the effects. The second scheme assumes uniform distribution of the total temperature among the four sections comprising the VTE-MED system being considered. Modification of temperature difference distribution was made to realize equal heat transfer area of evaporators and preheaters for every successive 7 effects (comprising a section) with the exception of the first effect. For both schemes it was assumed that scaling would be less than 10 μm in thickness and heat loss in every effect would be less than 0.1% of source heat. The analysis included studying the variation of GOR with the number of effects, variation of brine thermal properties in each effect, average heat transfer coefficient in each effect, number of evaporator and preheater tubes in each effect, effect of scaling on heat transfer coefficient in each effect, flow rate of source vapor and flashing vapor in each effect and pressure and pressure drop distribution.

The analysis indicated that the area of evaporators and preheaters would become increasingly large from top effect to lower effect for the scheme of equal temperature distribution, as shown in Figure 4.11. It was found also that there is no obvious difference in GOR and water production between the two schemes. However, design with temperature difference distribution of equal heat exchanger areas is more suitable for tower structure design.

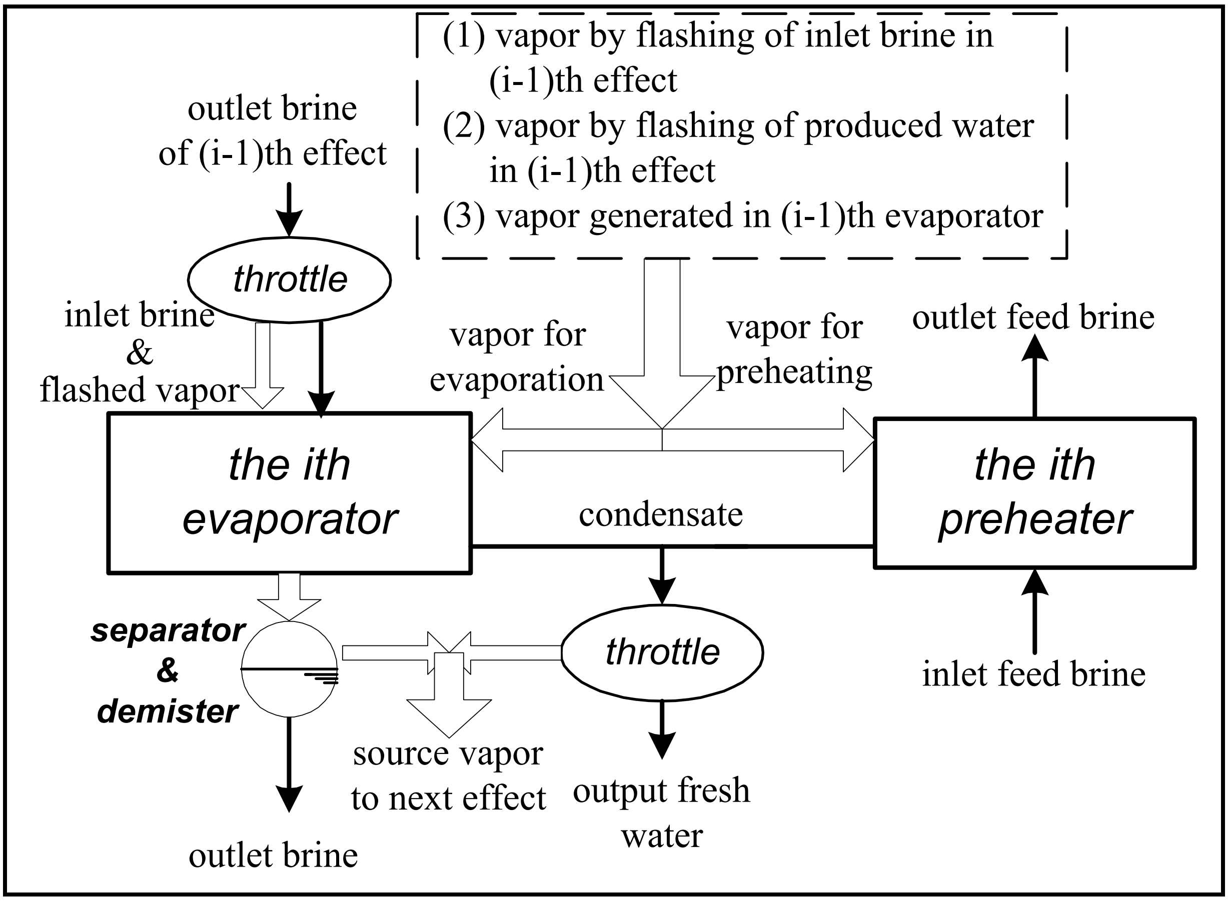


Figure 4.10: Schematic Diagram of NHRVM-Physical Model.

Additional analysis on performance of VTE-MED coupled with NHR-200 reactor using NHRVM-2. The analysis included: effect of system number of effects on GOR and heat transfer area, effect of material of heat transfer tube on heat transfer coefficient and heat transfer area, effect of geometric size of heat transfer tube, effect of seawater temperature, and effect of top brine temperature. Based on the results of additional analysis, design parameters of the VTE-MED system were selected. These are summarized in Table 4.4.

4.1.3. Economic assessment

4.1.3.1. IAEA Desalination economic evaluation program (DEEP)

The DEEP program calculates the cost of water (and power) for single as well as dual-purpose plants. The latter is evaluated using the principle of the power credit method. DEEP is the IAEA's software package for the economic comparison of seawater desalination plants, including nuclear options. It has been validated with reference cases. On the basis of the validated version, a user-friendly version has been prepared along with a manual with information on how to use DEEP [4.5]. The methodology is incorporated in an EXCEL spreadsheet routine, which is available from the Nuclear Power Technology Development Section of IAEA. The methodology is suitable for economic evaluations and screening analyses of various desalination and energy source options. The methodology is embedded in a spreadsheet routine containing simplified sizing and cost algorithms which are easy to implement and are generally applicable to a wide variety of equipment and representative state-of-the-art technologies.

TABLE 4.3: INPUT DESIGN PARAMETERS

Parameters	Value
Temperature of source vapor at the first effect, ^oC	123
Top brine temperature, ^oC	120
Outlet brine temperature of last condenser, ^oC	35
Input rate of source vapor at the first effect, kg/s	89.25
Input rate of feed brine, kg/s	3591.1
Salinity of feed brine, ppm	40000
Inside/outside diameter of evaporator tube [x 10^{-3} m]	49.3/50
Tube length of evaporator, m	3
Inside/outside diameter of pre-heater [x 10^{-3} m]	15.3/16
Tube length of preheater, m	4.4
Conductivity of tube wall, W/ ^oC.m	150
Thickness of scale of brine side [x 10^{-6} m]	0~10
Conductivity of scale, [x 10^{-2} W/ ^oC.m]	6.98

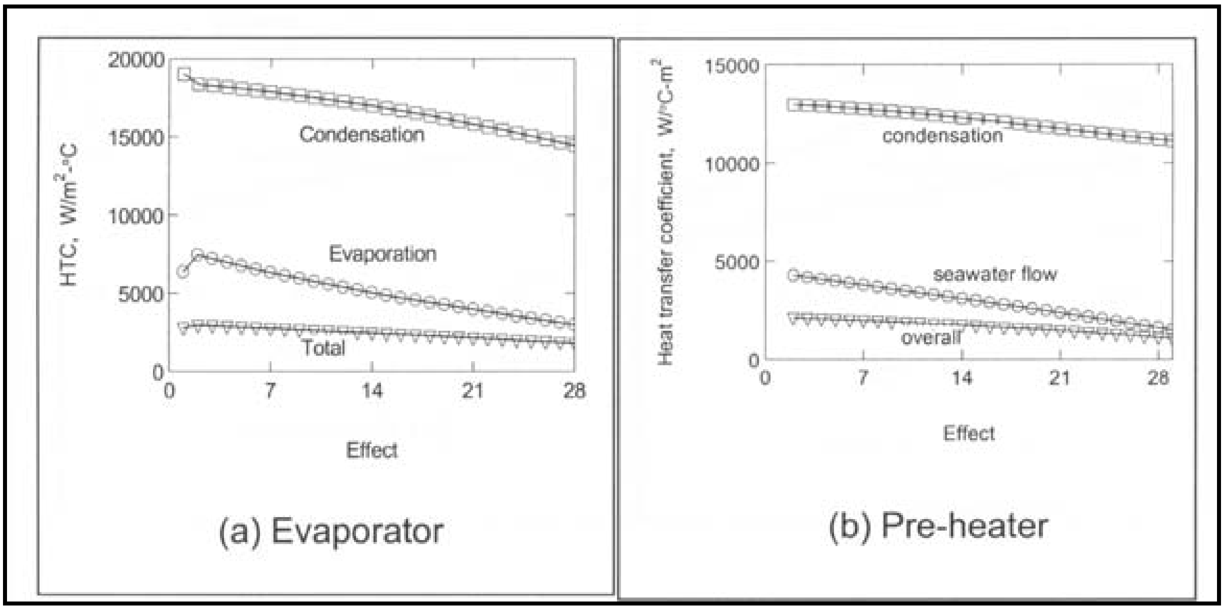


Figure 4.11: Average heat transfer coefficient of each effect.

TABLE 4.4: RECOMMENDED DESIGN PARAMETERS FOR VTE-MED

Parameters	Value
Number of effects	28-30
Material of heat transfer tube	SS 1.4565
Diameter of heat transfer tube, mm	50
Length of heat transfer tube, m	3-4
Thickness of heat transfer tube walls, mm	0.35 for SS 1.4565 1.0 for Aluminum alloy
Top brine temperature, ^oC	120

The spreadsheet methodology was substantially improved to include the capability to model many types of nuclear/fossil electric power and heat sources of varying sizes, depending on site specific demands. It embodies the basic technical and economic principles of power and desalination plant performance and can be adapted to any site conditions. Current cost and performance data are incorporated so that the spreadsheet can be quickly adapted to analyze a large variety of options with very little new input data required.

The output includes the levelized cost of water and power, breakdowns of cost components, energy consumption and net saleable power for each selected option. Specific power plants are modeled by adjustment of input data including design power, power cycle parameters and costs. The desalination systems are modeled to meet the World Health Organization (WHO) drinking water standards. However, modifications and changes can easily be incorporated on the basis of description in the manual.

The spreadsheet serves three important objectives:

(1) It enables side-by-side comparison of a large number of design alternatives on a consistent basis with common assumptions.

(2) It enables quick identification of the lowest cost options for providing a specified quantity of desalted water and/or power at a given location.

(3) It gives an approximate cost of desalted water and power as a function of quantity and site specific parameters including temperatures and salinity.

For planning an actual project, final assessment of project costs should be done more accurately on the basis of more substantive information including project design and specific vendor data.

The use of the spreadsheet methodology is limited to the types of the power or heating plants, desalination processes and coupling models described in the manual. The spreadsheet models the common commercial processes for large scale seawater desalination. The MED process is assumed either to be a low temperature, horizontal tube system or a high temperature, vertical tube evaporator (modification to horizontal tube evaporator is possible). Heat is supplied in the form of low-grade dry-saturated steam, and maximum brine temperature is assumed to be limited to 70°C and 125°C for low and high temperature MED, respectively. MSF is modeled as a once through process. Heat is supplied in the form of hot water or low grade saturated steam and maximum brine temperature is limited to 135°C.

Standalone assumes that the RO plant is only electrically coupled to the power plant. Contiguous RO assumes that the RO plant shares a common seawater intake and outfall with the power plant cooling system and may take the advantage of the power plant reject heat for the feedwater preheating. In addition, there are two membrane options available, hollow fibre membranes and spiral wound membranes.

In the hybrid concept, the thermal desalination and the membrane plants together provide the desired water quality and demand. Feed to RO is taken from the condenser reject water of the thermal desalination plant or power plant condenser. The spreadsheet includes different types of nuclear reactors. Some of them can provide both electricity and heat and are capable of being coupled with any of the desalination plant concepts such as thermal desalination or membrane process or both.

Some of the energy sources are ‘heat only’ systems and are coupled only with thermal desalination systems using hot water or low pressure steam. Some of the energy sources are considered as ‘power only’ systems and are only to be coupled with membrane process using electricity. The input data in the spreadsheet can be adjusted to enable the model to approximate any type of nuclear steam power plant including liquid metal systems. In case of water cooled nuclear reactors, the spreadsheet considers the specification of an intermediate loop for either steam or hot water supply to the desalination system to prevent the possibility of radioactive contamination of the product water. The model also includes a backup boiler for ensuring energy supply to the thermal desalination plant if the main energy source is not operating.

DEEP can be used for comparative calculations, in order to determine under which conditions nuclear desalination is economically competitive. DEEP has been used within this CRP for economic evaluation of selected coupling schemes. However, in some cases it was necessary to update the Program in order to allow for specific calculations that were not included and/or improve correlations and models. These modifications are described below.

DEEP-PH by CANDESAL, Canada

In order to enable an economic analysis that properly accounts for feed water preheat and design configurations leading to improved RO performance characteristics, the DEEP code was modified to include more realistic correlations. CANDESAL renamed its version DEEP-PH to reflect that the modifications to the program were primarily with respect to the addition of preheated (PH) feed water.

Modifications were performed in both the actual Excel file and VBA routines of the sample case. Alterations to the RO sections of the code were based on performance characteristics from actual design code analysis results for a specific case, which allows proper representation of the effect of elevated RO feed water temperature. The DEEP-PH code was developed through RO system performance correlations representing increased water production as a function of elevated feed water temperature. Cost estimates based on water production rate were adjusted accordingly. In addition, DEEP-PH user interface was created through minor cosmetic changes to DEEP 2.0.

These modifications to the DEEP code improved the accuracy in the calculation of performance and cost of water and power production for various power plants and spiral wound based RO desalination plant couplings. However, they were based on very specific membrane analysis calculations carried out over a wide range of RO feedwater temperatures, but limited to single values of seawater salinity and system pressure. The code is still under development and has the following limitations: seawater salinity of 38 500 ppm, and feed water pressure of 69 bar.

Modification of DEEP by IPPE, Russia

The existing version of DEEP 2.0 does not cover a set of specific features of the thermal scheme of the Small Heat and Power Plant. Besides, making new schemes of power source and desalination plant coupling in the program is troublesome, e.g. an RO plant with seawater preheat with the steam extraction, which considerably affects calculations of economic effectiveness of the nuclear co-Generation/desalination complex based on the Floating Power Unit (FPU). Therefore, a number of formulae and technical features of the typical power source (SPWR) and desalination plants (MED/RO) has been changed by IPPE.

Changes in economic characteristics of the power source and desalination plants have been made to put the calculation parameters into correspondence with the FPU and desalination plants project data. Changes in the formulae and project characteristics are summarized in Table 4.5. Files prepared in the course of DEEP modernization were used for the analysis of the updating results (see Section 4.2 of the report), and also both for cost calculations for the nuclear co-generation/desalination complex based on the FPU and sensitivity analysis.

4.1.3.2. Results of economic evaluation

Economic evaluation of the various coupling schemes were carried within this CRP by a number of participating institutions namely: CANDESAL (Canada), INET (China), KAERI (Republic of Korea), IPPE (Russia), CNESTEN (Morocco) and CNSTN (Tunisia). The economic evaluation was carried out using DEEP. The results of these analyses are summarized below.

CANDESAL, Canada

Use of the condenser cooling water as well as CANDU moderator cooling water as preheated feedwater to the RO system improves the efficiency of the RO process and therefore the economics of water production.

To present the performance and economic characteristics from actual design code 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 input data that are not normally allowed by DEEP, as well as modifications to the basic calculation modules themselves. In addition, rather than using the default data supplied by DEEP for calculation of the performance of the nuclear power plant, actual CANDU performance and economic data was entered. Table 4.6 lists those data that differed from the DEEP default for the specific case with 25°C seawater having a TDS of 38 500 ppm and feed flow suitable for producing about 100,000 m^3/day of potable water at ambient seawater temperature.

Economic analysis was carried out using both the standard version of DEEP and the modified version that accounts for the proper treatment of RO systems with feedwater preheating (DEEP_PH). The calculations were done using the same energy plant data for each case and two sets of calculations were done for comparison. Indeed, no significant changes resulted when the modified code was used 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 feedwater.

The first used DEEP as it is currently programmed with the default RO performance calculations. Cases were run 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 feedwater 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 second set of economic analyses evaluated the same four cases (stand-alone, contiguous at ambient seawater temperature, feedwater preheat at 35°C and feedwater preheat at 44°C using the modified version of DEEP, DEEP_PH.

TABLE 4.5: SUMMARY OF IPPE CHANGES MADE TO THE FORMULAE OF DEEP 2.0

DEEP Section	FPU and Desalination Plant Parameters
Technical Parameters	• Factor auxiliary load **Fal=0.05751(1-0.0575)** • Turbine mechanical efficiency **Etm=0.997** • Generator efficiency **Eg=0.98** • Plant economic life **Lep*)=36 ЛеТ** • Discount rate **I=8%** • Interest rate **ir*)=0%** • Purchased electricity cost **Cpe*)=0.037 $/кВт*ч**
Power Plant	*To perform a correct modeling of the Base Plant mode of the FPU, the following dependencies and values have been added to the program:* • Ref power plant unit net output **Qbn=36.2 MWe** • Reference net thermal efficiency **Ebpnr=25.6** • Total base plant thermal power **Qtp=284 MWt** • Condenser range **DTcr** vs. Seawater temperature **Tsw**: **DTcr=0.0555*Tsw-0.0004Tsw^2+17.247** • Condensation temperature **Tca** vs. Seawater temperature **Tsw**: **Tca=0.504*Tsw-0.01l4Tsw^2+1.8673** • Condenser cooling water pump head **DPcp=1 (bar)** • Reference condensing temp. of base power plant **Tcr=34.5** • Planned outage rate **opp*)=0.25** • Unplanned outage rate **oup=0.10** *Project economics data have been put:* • Specific construction cost Ce*)=2062 /**$/кВТ** • Additional site related construction cost **DCrs=0** • Energy plant contingency factor **kec=0** • Construction lead time **Le*)=48 months** • Specific O&M cost **Ceom*)=11.1 $/МВтч** • New variable is introduced **Tctl** — FPU overhal tow **Tctl*)=2 M$/year** • Specific nuclear fuel cost **Csnf =10 $/МВтч** • Specific decommissioning cost **Csde*)=1.9 $/МВтч** • Nuclear fuel annual real escalation **efn*)=0**
	perform a correct modeling of the Base Plant mode of the FPU, the following dependencies and values have been added to the pro gram. • Ref power plant unit net output **Qbnt=32.8 MWe** • Turbine plant thermal efficiency **Ebnrt=23.8** • Condenser range **DTcrt** vs. Seawater temperature **Tsw: DTcrt=0.0305*Tsw+13.116** • Condenser approach temperature vs. Seawater temperature **Tcat=0.0089*Tsw^2-0.583Tsw+12.243** • Reference condensing temp. of base power plant **Tcrt=33.1^{0}C.** • Site specific average condensing temperature vs. Seawater temperature **Tct=Tsw*0.8503+24.495** • FPU reactors thermal power **Qbnt=Nb*Qbnrt/Ebnrt*100** • Total plant auxiliary loads **Palt*)=4.43 Мвт**

TABLE 4.5: SUMMARY OF IPPE CHANGES MADE TO THE FORMULAE OF DEEP 2.0 (Cont.)

	• Total site specific plant gross output **Pegt=Pent+Palt-corfact,** where **corfact=0.0004Tsw^2+0.0012*Tsw+0.0248** • Condenser reject heat load **Qcrt=Qtpt-Pegt-Qcrmt** • Condenser cooling water flow **Fcct=Qcrt* 1000/4.1875/DTcrt**
Distillation Plant	Unchangeable FPU Project Characteristics: • Total heat to water plant **Qcrm** was compared to the project value of thermal power **Qcrmt = 58.2 MWt.** Unchangeable DOU GTPA-840 Project Characteristics: • **Tdc** - Last stage/effect brine/steam temperature **Tdc=40^{0}C** • Number of MED effects **Nemed = 23=const.** • Overall water plant working temperature **Dtao=55.75^{0}C** • Seawater flow **Fsd=1252 kg/s**. • Condenser approach temperature **Dtca=9 ^{0}C** • Intermediate loop temperature drop **DTft=130-95=35 ^{0}C.** • Intermediate loop pressure loss **Dpip=16 bar.** • Seawater total dissolved solids **TDS=3 ppm.** • MED plant condenser range **DTdcr =22/625-0.575*Tsw.** • MED plant condenser approach **DTdca=17/375-0.425*Tsw** • Maximum brine temperature **Tmb=94.27C^0** • Water plant planned outage rate **opd=0.01** • Water plant unplanned outage rate **oud=0.04** • Design condensation temperature in intermediate circuit HX **Tsm=139.7C^0** • Selected unit size **Wdu=2000 m^3/day** *Project Economic data on DOU GTPA have been used:* • Base unit cost **Cdu=1693 \$/m^3/day** • Intermediate loop cost **Cil = 0** • Water plant cost contingency factor **kdc= 0** • Operation costs for DP correspond to the Project data • Average DP management and labour salaries correspond to the Project data
Reverse Osmosis Plant	• Calculation operation temperature **Tsm=20^{0}C=const** • High head pump pressure rise **DPhm=68 bar** • Membrane area factor (over reference) **Fma = 1** • Permeate TDS **dsms=0.11*T^2+1.94*T+191,** where **T=Timc (C-RO) or T=Timh (hybrid plant)**. • RO Plant Feedwater temperature: **if** Fsms <= Fcc, **then** Time (Timh)=Tsw+DTcr, in opposite case, **Timc (Timh) =Tsw+1000*Qcrmtot/Fsmc/4.18,** where **Qcrmto-** total heat to C-RO Plant (or Hybrid paint) depending on the coupling scheme.

TABLE 4.6: INPUT TO REVERSE OSMOSIS SECTION IN DEEP

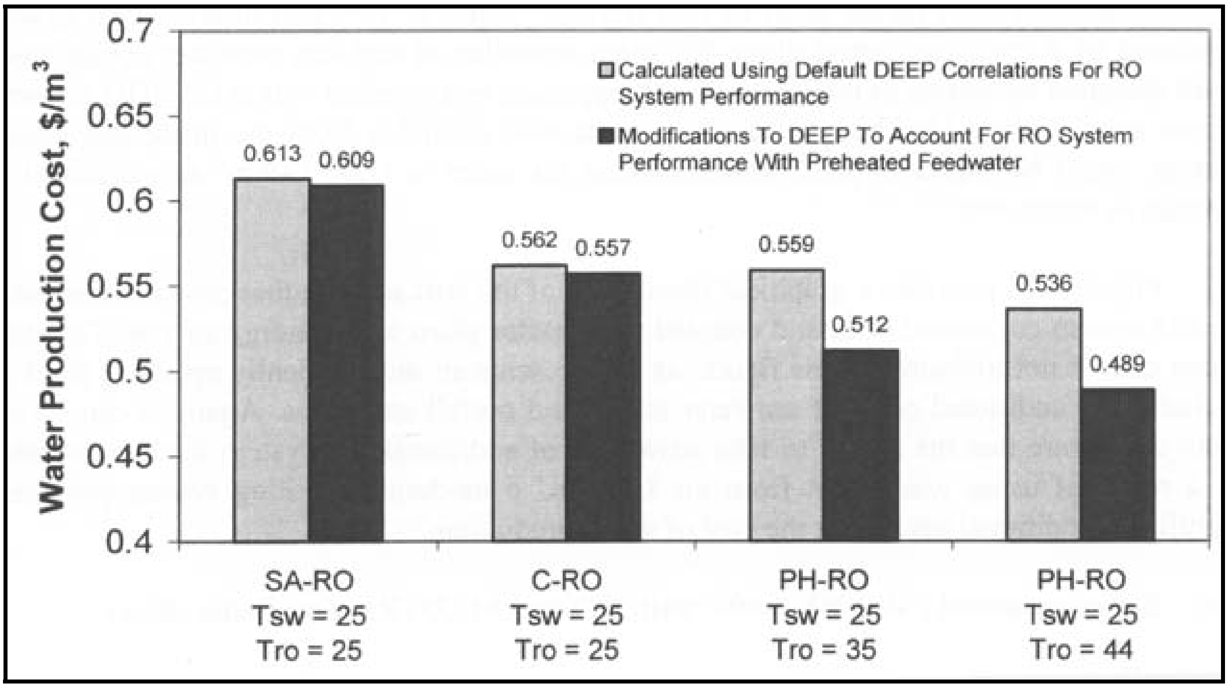
Data Definition	DEEP default data	CANDESAL analysis
Average annual cooling water temperature	21 ^{0}C	25 ^{0}C seawater temperature
Contiguous RO design cooling water temperature	31 ^{0}C	25, 35 and 44 ^{0}C
Desalination plant optional unit size specification	24000 m^3/d	12000 m^3/d
High head pump pressure	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 ^{0}C	25 ^{0}C

The results of both sets of economic analyses are shown in Figure 4.12. Analysis results from the first two 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 feedwater from the intake canal for the power plant at ambient seawater temperature, the RO system would be operated at seawater temperature.

The second two cases, for RO operating temperatures of 35°C and 44°C (leaving all other parameters the same) show that there are some significant changes when proper accommodation has been made for preheated feedwater through modification of the DEEP code. The calculations show significant economic improvement as the feedwater 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.

Figure 4.13 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 moderator cooling system provides a significant additional reduction in the cost of water production.

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.



SA = Stand Alone C = Contiguous PH = Pre heated
Tsw = Seawater temperature in °C Tro = = Feedwater temperature in °C

Figure 4.12: DEEP economic analysis of water production costs.

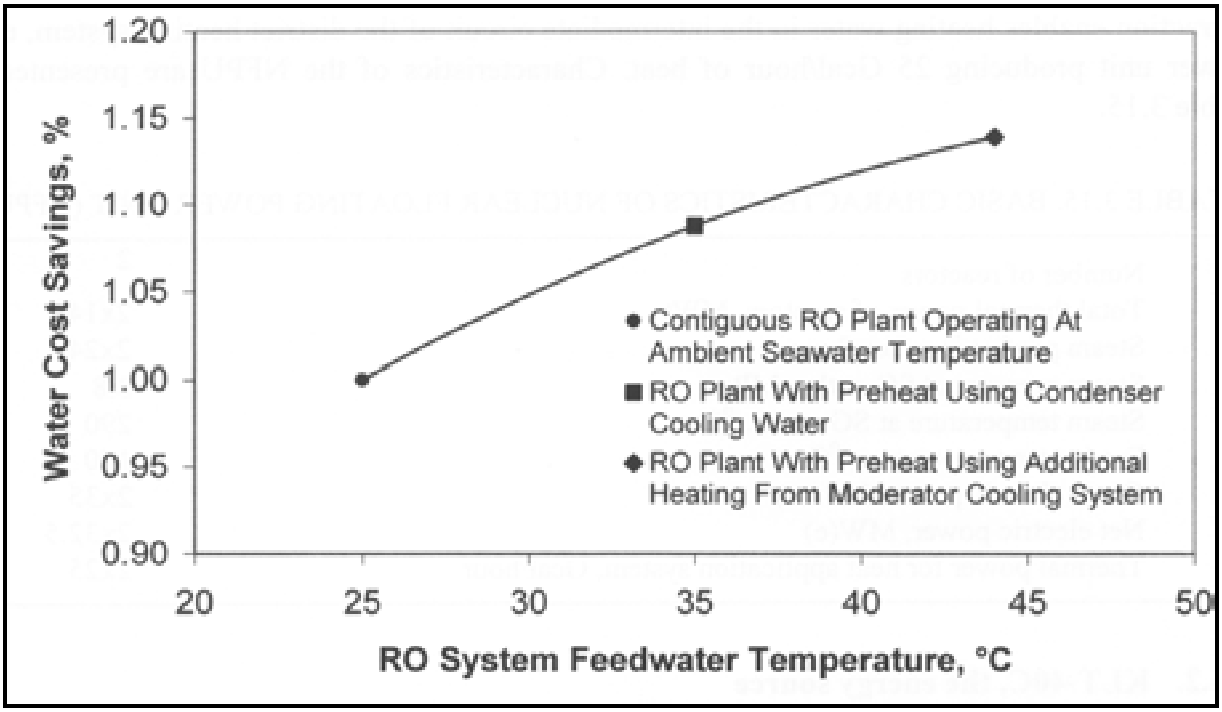


Figure 4.13: Water cost savings calculated using Modified DEEP Version.

It has also shown that an additional source of waste heat that can be used to achieve a higher level of RO feedwater preheat contributes to additional reductions in the cost of water production. A plant designed according to this advanced system design concept and coupled with a CANDU 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, representing an additional 5% savings in water cost.

INET, China

Economic evaluation of coupling NHR-200 with VTE-MED process was carried out using DEEP. The analysis was based on the site-specific case study of Shandong nuclear desalination project, domestic market price in China and innovative technical design. As a result the specific desalted water cost was found to be 0.61 US$/$m^3$. The main economic and performance parameters are shown in Table 4.7.

TABLE 4.7: MAIN ECONOMIC AND PERFORMANCE PARAMETERS FOR SHANDONG PROJECT

Parameter	**Unit**	**Reference Value**
Reference Currency		US dollar
Currency Reference Date		1 January 2003
Operation Reference Date		1 January 2008
Economic Life	Years	30
Discount Rate	%	5.58
Nuclear Plant Data		
NPP Type		NHR-200
Net Output	MW(t)	200
NPP Availability	%	93
Total Specific Construction Cost	$/kW(t)	289
Construction Lead Time	Months	40
Distillation Plant Data		
DP Type		VTE-MED
Design Capacity	m^3/day	2x85,000
DP Availability	%	97
DP Base Cost	$/($m^3$/day)	571
Construction Lead Time	Months	24
Number of Effects per Unit		29
GOR	Kg product/kg steam	21.2
Site Specific Data		
Assumed Site		Shandong, China
Seawater Total Dissolved Solids	ppm	31500
Average Seawater Temperature	^{o}C	20
Economic Evaluation		
Specific Energy Cost	$/ton steam	4.942
Specific Desalted Water Cost	$/$m^3$	0.61

KAERI, R.O. Korea

The economic evaluation of the integrated SMART desalination plant was carried out using DEEP. The total water production capacity of the SMART desalination plant is 40 000 m3/day. The desalination plant consists of 4x10 000 m^3/day MED units.

Construction cost of SMART is one of the major input data having a great influence on the water production cost. However, there is no construction experience on small sized integral reactors in Korea. Therefore, construction cost of SMART was estimated based on the existing 1000 MW(e) PWR data in Korea. Overnight cost was calculated by applying a scaling factor of 0.8 to the conventional nuclear power plant of 1000 MW(e). As a result, overnight cost of SMART with 100 MW(e) was calculated to be 2,442 $/kW(e). However, the calculated overnight cost of SMART is expected to be reduced by up to 25% reflecting the advanced features adopted in SMART design such as integral and modular design. Therefore, the overnight cost of SMART is assumed to be 1, 800 $/kW(e) as a reference case.

Sensitivity analyses of water production cost were carried out regarding major parameters such as capacity factor, discount rate, and construction cost. The major input data necessary for the calculation of water production unit cost came from DEEP computer code. Table 4.8 summarizes the results of water production unit cost. It is clear from Table 4.8 that for plant availability of 80% or higher and discount rate of 8%, the calculated water production unit costs are in the range of 0.74–0.84($/$m^3$). Since the SMART plant is designed for availability higher than 90%, this indicates that SMART can be a competitive choice for nuclear desalination.

TABLE 4.8: SENSITIVITY ANALYSIS OF SPECIFIC DESALTED WATER COST FOR SMART NUCLEAR DESALINATION PROJECT

	Specific water cost, US$/$m^3$		
Overnight cost, ($/kWe)	**2,200**	**2,000**	**1,800**
Capacity factor, %			
70	0.92	0.91	0.90
80	0.84	0.83	0.82
90	0.76	0.75	0.74
Discount Rate, %			
3	0.53	0.52	0.52
5	0.62	0.61	0.60
8	0.76	0.75	0.74
10	0.88	0.86	0.85
12	1.00	0.98	0.96

CNESTEN, Morocco

Morocco has been experiencing a continuous drought for a number of years. This exerted stress on the limited water resources in the northern region and Sahara provinces. Therefore, Morocco has been considering for a number of years seawater desalination for the production of potable water using fossil and nuclear energy.

Currently, MED, RO and VC desalination processes are producing about 10,000 m^3/day of potable water. Additional 13,000 m^3/day of potable water are produced by the phosphate industry through MSF and VC desalination processes.

Within this CRP, coupling options for the two candidate sites Agadir and Laayoune were investigated. The city of Agadir is located on the Atlantic Ocean south of the capital Rabat. Laayoun is the largest city in the Sahara region. The projected population and water demand for the two cities up to the year 2025 is shown in Table 4.9. Moroccan energy projections indicated that electricity generation installed capacity will increase from 4325 MW(e) in the year 2002 to 12982 MW(e) in the year 2025 as shown in Table 4.9. Hence, the electrical grid could easily accommodate a 600-MW(e) nuclear power plant.

DEEP was used for economic evaluation of three nuclear power reactors and a combined cycle plant in the range of 600 MW(e) coupled with the desalination processes: MED, MSF, RO (contiguous and stand alone) and hybrid (MSF-RO and MED-RO). Two scenarios were considered for each site. The first scenario assumes that nuclear desalination will provide only part of the water demand. The second scenario assumes that all of the water demand will be met by nuclear desalination. Table 4.10 summarizes the results of the evaluation for the two sites.

For both sites, the costs of desalted water produced by nuclear and fossil energies are in the same range. Most economic configuration appears to be contiguous RO coupled with a PWR and an adequate amount of electricity to the grid. The installation of a 600-MW(e) PWR in Agadir could produce more than 300 000 m^3/day of desalted water at competitive cost and an average of 517 MW(e) to the grid. In the case of Laayoune, these figures are 72 000 m^3/day and 517 MW(e) respectively.

TABLE 4.9: PROJECTED POPULATION AND WATER DEMAND FOR AGADIR AND LAAYOUNE UP TO THE YEAR 2025

YEAR	2005	2010	2015	2020	2025
Agadir					
Population, ‘000 person	787.2	895.0	1017.6	1160.0	1318.8
Water Demand, ‘000 m^3/day	140.5	201.4	238.3	275.3	312.2
Laayoune					
Population, ‘000 person	182.9	208.1	237.0	269.0	305.7
Water Demand, ‘000 m^3/day	25.3	32.4	43.2	53.6	71.2
Projected Installed Capacity, MW(e)	5600	6810	7970	10172	12982

TABLE 4.10: SUMMARY OF WATER COST FOR AGADIR AND LAAYOUN

Desalination Process		Desalted Water Cost, $/m^3$			
		PWR	PHWR	CC	GTMHR
Agadir Site					
MED	Scenario-1	0.62	0.63	0.64	N/A
	Scenario-2	0.74	0.77	0.88	N/A
MSF	Scenario-1	0.88	0.90	0.91	N/A
	Scenario-2	1.41	1.47	1.69	N/A
S-RO	Scenario-1	0.57	0.58	0.58	0.58
	Scenario-2	0.66	0.68	0.74	0.68
C-RO	Scenario-1	0.56	0.56	0.57	N/A
	Scenario-2	0.63	0.65	0.71	N/A
MED-RO	Scenario-1	0.58	0.58	0.59	N/A
	Scenario-2	0.72	0.74	0.82	N/A
MSF-RO	Scenario-1	0.62	0.59	0.63	N/A
	Scenario-2	1.02	1.05	1.17	N/A
Laayoune Site					
MED	Scenario-1	0.89	0.91	0.94	N/A
	Scenario-2	0.80	0.83	0.92	N/A
MSF	Scenario-1	1.33	1.50	1.55	N/A
	Scenario-2	1.44	1.50	1.70	N/A
S-RO	Scenario-1	0.81	0.82	0.85	0.81
	Scenario-2	0.75	0.77	0.83	0.77
C-RO	Scenario-1	0.75	0.76	0.79	N/A
	Scenario-2	0.69	0.70	0.77	N/A
MED-RO	Scenario-1	0.80	0.82	0.88	N/A
	Scenario-2	0.73	0.74	0.81	N/A
MSF-RO	Scenario-1	0.92	0.94	1.02	N/A
	Scenario-2	0.82	0.84	0.92	N/A

IPPE and other Institutions, Russian Federation

IPPE, OKBM, JSC NPP Mashprom and JSC Malaya Energretica used different versions of DEEP including that modified by IPPE for the economic assessment of coupling Russian Federation small reactors (KLT-40C, NIKA-55, NIKA-70 and RUTA-70) with DOU_GTPA Russian Federation MED plants. The results of the economic assessments are briefly described below.

KLT-40C is designed to work as a constituent part of the Floating Power Unit (FPU) of a nuclear desalination plant. The project is being carried out in cooperation with CANDESAL, Canada. Five coupling options were considered. These are:

- Option 1: a nuclear desalination plant based on FPU and a Hybrid desalination plant (50% MED + 50% RO). In this option, the RO utilizes preheated seawater from FPU condensers and the DOU_GTPA-840 condenser.
- Option 2: a nuclear desalination plant based on FPU and a preheating RO plant. This scheme fully utilizes both the condenser heat and the turbine extraction heat to preheat the RO feedwater.

- Option 3: a nuclear desalination plant based on FPU and seawater prehaheat in steam plant condenser.
- Option 4: a nuclear desalination plant based on FPU and an RO plant. In this scheme, seawater is preheated only in the intermediate loop preheater.
- Option 5: a nuclear desalination plant based on FPU and a 22-stage MED plant. In this scheme, steam is extracted from turbine plant for the DOU_GTPA-840

The results of the calculations for different water complex capacities are depicted in Table 4.11. For all options, the cost of electric power to the grid was 0.059 $/kWh. Option 1 represents the optimum coupling for lowest water cost and maximum power to the grid. The analysis indicated that option 4 is ineffective.

NIKA-70 was developed at the RDIPE for cogeneration of electricity and heat. NIKA-70 is a double-circuit facility with an integral single-unit type reactor. The economics of coupling NIKA-70 reactor with MED, RO and hybrid desalination plants was carried out with DEEP for various configurations. Table 4.12 shows DEEP results for a nuclear desalination complex based on one NIKA-70 reactor coupled with three DOU_GTPA-230 MED plants. Table 4.13 summarizes the calculation results for a nuclear desalination complex based on two power units and different desalination options.

The RDIPE is developing the RUTA reactor plant as a heat source for District Heating Systems (DHS) and desalination systems. RUTA produces low temperature heat. Hot water in the district heating circuit has the pressure of 0.6 MPa and temperature of 85°C. DOU GTPA multi-effect distillation desalination plants are the most suitable for the use with RUTA reactor. The technical and cost indices of nuclear desalination plants based on RUTA-55 and RUTA-N-70 are depicted in Table 4.14.

TABLE 4.11: MODIFIED DEEP CALCULATIONS FOR VARIOUS KLT-40C OPTIONS

Installed Water Plant Capacity, m^3/day	**Coupling Option**	**Parameter**			
		Water Plant Average Daily Capacity, m^3/day	**Water Cost, $/$m^3$**	**Power to Grid, MWe**	**Specific Water Plant Investments, $/($m^3$/day)**
120 000	1	113288	0.69	45.0	1125
	2	110 282	0.67	39.3	968
	3	110243	0.67	44.8	968
	4	114934	0.61	36.8	774
	5	110565	0.74	54.8	1734
180 000	1	167883	059	32.6	730
	2	165283	0.62	27.1	827
	3	165423	0.62	32.6	826
	4	169622	0.60	14.8	755
240 000	1	233 398	0.52	20.0	730
	2	220 350	0.63	14.8	827
	3	220 332	0.63	20.3	826
	4	224 263	0.60	12.2	755
300 000	1	277074	0.51	8.0	431
	2	275270	0.60	2.5	766
	3	275 534	0.60	7.9	766
	4	278886	0.59	-0.1	727

TABLE 4.12: ECONOMICS OF NUCLEAR DESALINATION BASED ON ONE NIKA-70 REACTOR AND THREE DOU_GTPA-230 MED PLANTS

Parameter	units	value
Total installed water production capacity	m^3/day	16560
Average water plant capacity	m^3/day	14367
Number of distillation plants		3
Installed water plant capacity	m^3/day	5520
Electric power to the grid	MW(e)	9.4
Total heat to water plant	MW(t)	44.7
Water plant specific electricity consumption	kWh/m^3	1.4
Steam temperature from steam turbine	^{0}C	94
Overall water plant working temperature	^{0}C	52.4
Average temperature drop between MED effects	^{0}C	4.37
Number of effects		12
GOR (kg water/kg steam)		9.5
Maximum brine temperature	^{0}C	80
Specific O&M cost for reactor plant	$/MWh(e)	10
Specific nuclear fuel cost	$/MWh(e)	20
Specific decommissioning cost	$/MWh(e)	1
Reactor plant operating availability		0.8
Purchased electricity cost	$/kWh(e)	0.09
Fissile fuel cost	$/bbl	20
Reactor plant economic life	Years	30
Distillation plant construction cost	M$	19.6
Reactor plant construction cost	M$	58
Interest rate	%	8
Desalted water cost	$/$m^3$	1.22

TABLE 4.13: DEEP RESULTS FOR NUCLEAR DESALINATION COMPLEX BASED ON TWO POWER UNITS COUPLED WITH VARIOUS DESALINATION PLANTS

Parameter	Units	MED	RO	Preheat-RO	Hybrid (50% MED + 50% RO)
Installed Water Capacity	M3/day	72,000	144,000	144,000	144,000
Electric Power to the Grid	MW(e)	15.7	0	2.7	1.6
Desalted Water Cost	$/$m^3$	0.91	0.86	0.82	0.84

CNSTN, Tunisia

The main objective of the project is to perform a technical and economic feasibility study of potable water production using nuclear seawater desalination in Skhira in the south of Tunisia.

- Preliminary dimensioning of nuclear reactors and desalination plants.
- Optimization of the coupling schemes for various nuclear reactors and desalination process combinations.
- Economical evaluation of the integrated nuclear desalination systems.
- Safety studies.

TABLE 4.14: TECHNICAL AND COST INDICES OF DESLINATION COMPLEXES BASED ON RUTA-55 AND RUTA-N-70 REACTORS

Parameter	Units	RUTA -55 (Basic option)	RUTA-N-70 (Optimized option)
Reactor thermal power	MW(t)	55	70
Specific Capital Cost of Reactor	$/kW	390	330
Auxiliary power consumption	MW(e)	1.1	1.2
Fuel specific cost	$/MW(t)	1.15	0.64
Thermal power cost	$/MW(t)	5.6	4.9
Capacity factor		0.92	0.92
Number of DOU_GTPA units		4	5
Top brine temperature	^{0}C	80	85
Capacity of DOU_GTPA module	m^3/h	220	250
Total capacity of desalination plant	m^3/day	21,000	30,000
Specific cost of desalination plant	$/(m^3/day)	1700	1400
Specific electricity consumption	kWh(e)/ m^3	1.8	1.8
Cost of purchased electricity	$/kWh(e)	0.05	0.05
Discount rate	%	8	8
Cost of desalted water	$/ m^3	1.22	1.0

The Tunisian contribution to this CRP is focused on Economic evaluation of the integrated nuclear desalination and fossil energy systems. The energy systems considered were: generic PWR, 900 MW (PWR-900), the innovative 600-MW reactor, SCOR being developed at CEA, France (SCOR-600), the 300-MW Gas Turbine Modular High temperature Reactor (GT-MHR 300), the conventional 600-MW fossil fuel steam turbine (SSB-600) and the conventional 600-MW gas turbine combined cycle plant (CC 600). The desalination systems were confined to MED, RO and preheated RO (RO_{ph}). Energy production costs were calculated using the code SEMER developed by CEA, France. These costs were used as input to DEEP to calculate the desalted water costs. The major input parameters for power and desalination cost evaluations are shown in Tables 4.15 and 4.16. Base case scenario assumed interest/discount rate of 8%, fossil fuel price 25 $/bbl. Sensitivity analysis was carried out for discount rates of 5 and 10%, and fossil fuel prices of 18 and 30 $/bbl. The resulting water cost for the various combinations are shown in Figure 4.14.

Detailed analysis of the calculated results shows that nuclear option becomes more economic if current gas TEP prices increase by more than 15% (from 130 $/TEP to 150 $/TEP, Tunisian conditions). For example, for a discount rate of 8% and with the standard prices of the fossil fuels (25 $/bbl or 183 $/TEP), the desalination cost with the integrated system SCOR 600 - MED is respectively 25% and 33% lower than those with CC-600-MED and SSB-600-MED. The desalination with the nuclear power plant is also more economic with RO process: 21% and 29% of reduction in comparison with the desalination by CC-600 and SSB-600. The RO_{ph} process brings an additional reduction of about 19% in comparison with the cost of desalination by traditional RO.

4.2. EXPERIMENTAL VALIDATION

4.2.1. Experimental validation of RO preheating effects

Although the proportional relationship between feed water temperature and membrane permeability is well known, the idea of utilizing the condenser's cooling water as a source of feed for RO systems did not appear until early 1994 [4.6-4.7]. This concept was adopted and investigated by the IAEA in all subsequent studies [4.8–4.10]. These and other studies [4.11–4.12] have shown that there is a potentially significant economic and performance benefit through the combined effects of feedwater preheating and system design optimization. These conclusions have been drawn from analyses and preliminary design studies without any experimental validation.

Experimental validation is of extreme importance in the confidence building process, particularly when some experts [4.13] argue that elevated temperatures may result in higher product water salinity, more rapid membrane fouling, greater membrane compaction, reduction in membrane lifetime and that saving in total water cost by elevating temperature from 15~18°C to 30°C would be in the range of 3% only.

TABLE 4.15: MAJOR ENERGY INPUT PARAMETERS FOR LA SKHIRA PROJECT

Power plant type	**PWR-900**	**SCOR-600**	**GT-MHR 300**	**SSB-600**	**CC 600**
Reference year	2003	2003	2003	2003	2003
Net Power, MW(e)	951	600	300	600	600
Number of plants	1	1	2	1	1
Thermal efficiency, %	33.0	30.5	48	39.0	51.0
Planned outage, %	9	5	5	5	5
Unplanned outage, %	4	5	5	5	5
Power plant lead time, years	5	4	4	3	4
Plant economic life, years	40	60	60	30	25
Interest/discount rate, %	8	8	8	8	8
Fossil fuel price, $/bbl	-	-	-	25	25
Fossil fuel real escalation, %/year	-	-	-	2	2
Transport cost, $/Bbl	-	-	-	0.5	0.5
Nuclear fuel price, $/MWh	6.48	6.48	6.48	-	-
Average salary, $/month	2286	2286	2286	1625	1625

TABLE 4.16: MAJOR DESALINATION INPUT PARAMETERS FOR LA SKHIRA PROJECT

Desalination plant type	**MED**	**RO**	**RO-ph**
Reference year	2003	2003	2003
Desalination plant capacity, m^3/day	150,000	150,000	150,000
Seawater salinity, ppm	38,375	38,375	38,375
Seawater temperature, ^{o}c	21	21	21
Unplanned outage, %	6	6	6
Planned outage, %	3.2	3.2	3.2
Desalination plant lead time, months	20	20	20
Plant economic life, years	60	30	25
Specific construction cost, $/(m^3/day)	900	800	800
Average salary, $/year			
- Management:	20,000	20,000	20,000
- Labour:	7,000	7,000	7,000

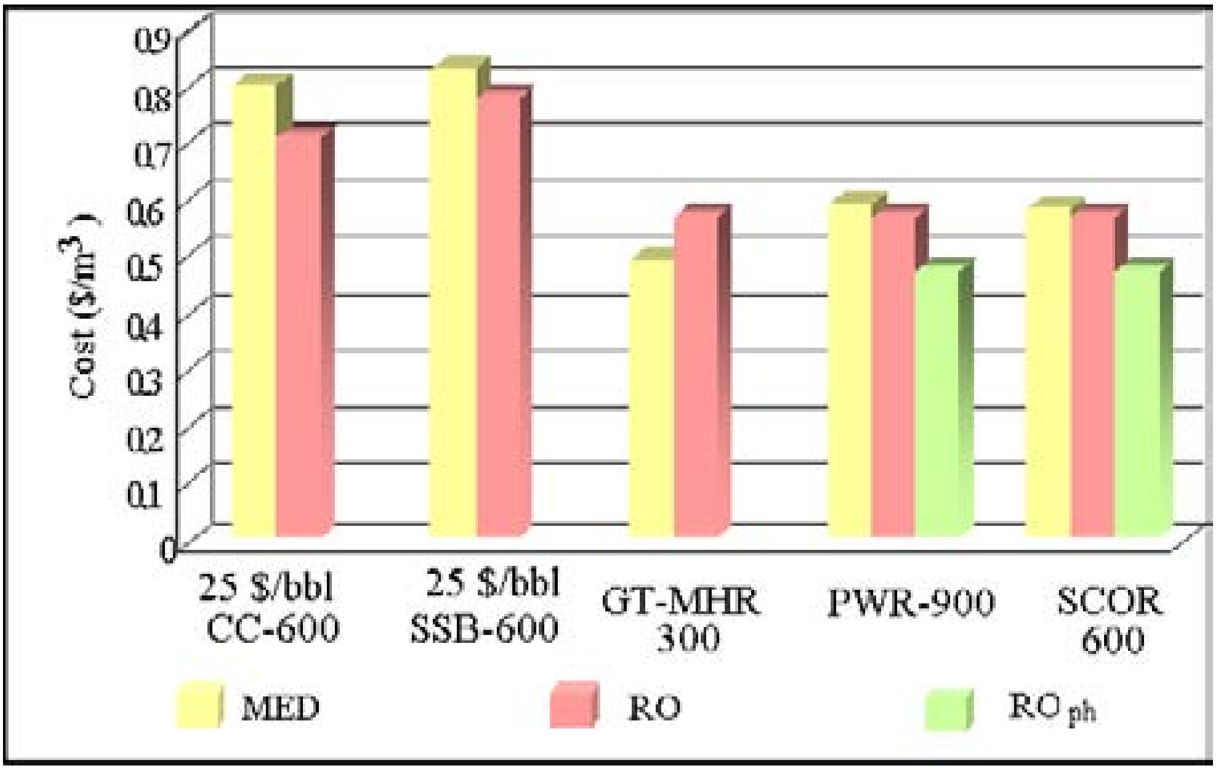


Figure 4.14: Desalted Water Cost for Various Energy Sources – Discount rate = 8%.

In view of the possible role of RO desalination technology in future nuclear desalination programs and the need to validate the concept of RO feedwater preheating, CANDESAL (Canada), NPPA (Egypt) and BARC (India) have decided to carry out research projects and decimate the results through this CRP. The common overall objective of these experimental programs was to investigate experimentally whether the projected performance and economic improvements of preheated feedwater can be realized in actual operation. The intent is to simulate as closely as possible performance characteristics that would be expected to occur in commercial large-scale RO seawater desalination plant. This Section provides a brief description of the experimental facilities and experimental programs of the above Institutions.

4.2.1.1. CANDESAL test rig

One of the key elements of this project was experimental validation of the theoretical concept developed by CANDESAL for RO system design. The CANDESAL concept approaches the application of RO technology in a non-traditional manner and therefore validation of the analytical results is necessary to ensure confidence in the membrane performance and cost evaluation models.

The experimental validation was carried out as a part of a separate project funded in part by the National Research Council of Canada. Although having slightly different long-term objectives, many of the tasks required for the IRAP project were similar to those required for this CRP project, and the existence of that project has provided a substantial framework that allowed this project to be carried to a satisfactory conclusion.

Objective of the test program

- To measure semi-permeable membrane characteristics as a function of temperature and pressure.
- To demonstrate that improved performance at elevated of temperatures and pressures projected by CANDESAL are achieved in practice.

Main features of the test rig

A schematic diagram of the test rig is shown in Figure 4.15. It has the following features:
- Single RO pressure vessel.
- Seven RO membranes in series inside the pressure vessel of Dow Filmtec SW30-8040 seawater membranes.
- 2.7 m^3 seawater reservoir (Ocean).
- Re-circulating system.
- No energy recovery.

Experimental conditions

- Feedwater Flow: fixed flow of 315 m^3/day.
- Feedwater Conditions:
  - TDS: 32400 ppm.
  - Temperature Range: 20–45 ^oC.
  - Pressure Range: 37–69 bar.

Demonstration testing has been carried out using a trail-mounted system producing up to 150 m^3/d of potable water. The facility was commissioned and functional testing was carried out in the spring of 2001. Experimental data on performance characteristics under varying conditions of temperature and pressure were obtained in the summer of 2001.

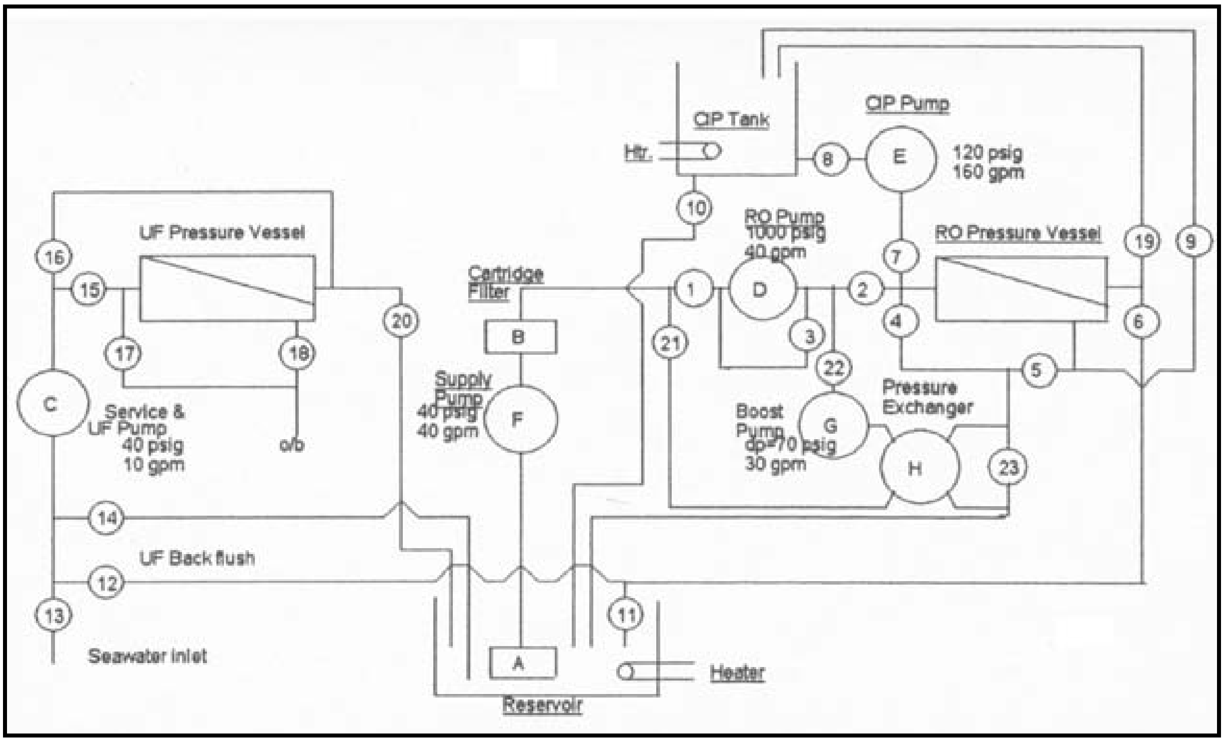


Figure 4.15: Schematic diagram of CANDESAL test rig.

Results of experimental work

Experimental data on performance characteristics under varying conditions were obtained during June 2001. The experimental results are shown in Figure 4.16. The data recorded during the experimental program has addressed feed water temperature in the range 20°C to 45°C over a pressure range up to 69 bar. The results demonstrated clearly that the CANDESAL approach is valid. The expected performance improvements resulting from increased temperature and pressure of the RO system are demonstrable and repeatable.

It is planned outside the scope of this CRP to extend the experimental program to carry out testing at a system level to study parameters not evaluated under this experimental program. The next series of tests will explore the currently proposed limits of membrane performance and evaluate the energy consumption requirement.

4.2.1.2. NPPA experimental facility

In view of the limited Egyptian resources of both primary energy and fresh water, Egypt has been considering for a number of years the introduction of nuclear energy for electricity generation and seawater desalination. NPPA decided to construct an experimental RO facility at its site in El-Dabaa to validate the concept of feed water preheating. The results of this experimental work could have a strong influence on how the international nuclear desalination community perceives the value/benefit of feed water preheating, and hence there is a common international interest in this project. Therefore, NPPA proposed the research project as part of this CRP.

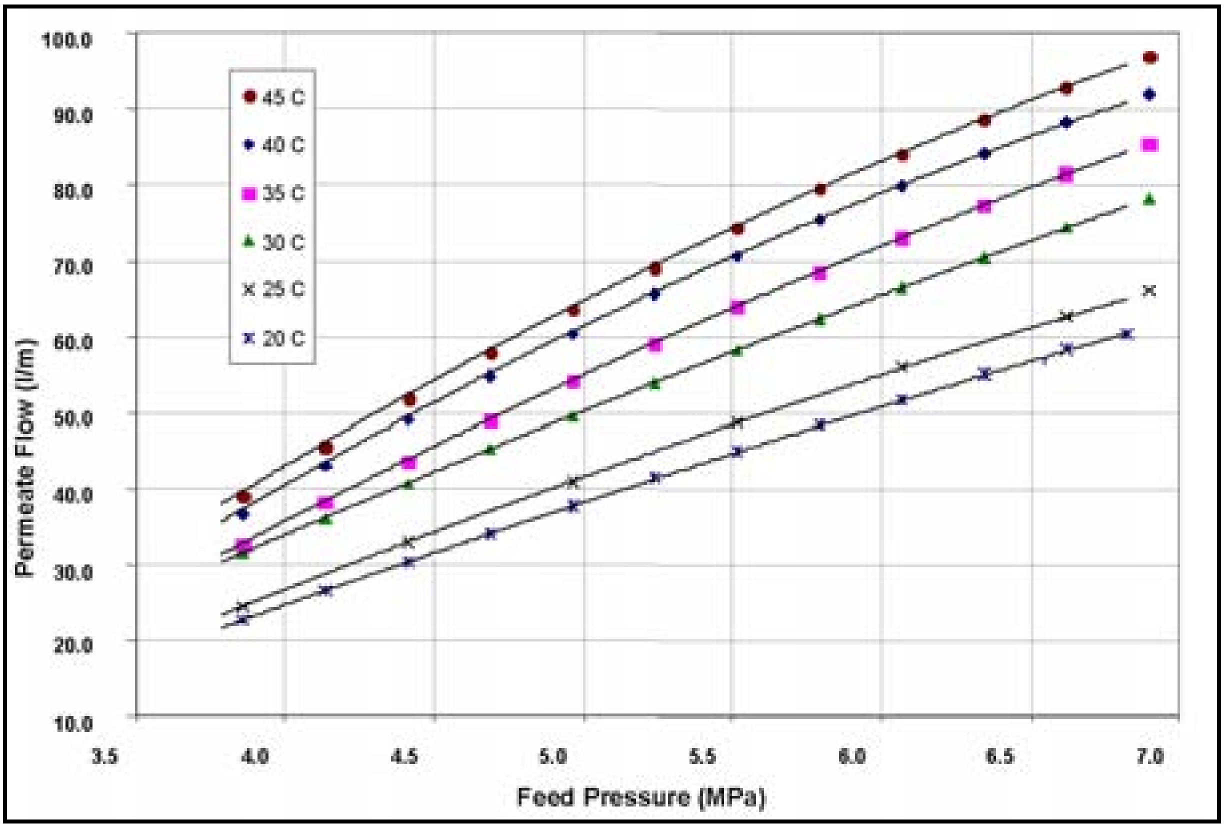


Figure 4.16:Mmeasured permeate flow rate as a function of pressure and temperature.

Objectives

In view of the possible role of RO desalination technology in any future Egyptian nuclear desalination program and the need to validate the concept of RO feedwater preheating, NPPA has decided to carry out this research project, with the following objectives in mind:

(A) ***Overall***: to investigate experimentally whether the projected performance and economic improvements of preheated feedwater can be realized in actual operation. The intent is to simulate as closely as possible performance characteristics that would be expected to occur in commercial large-scale RO seawater desalination plant.
(B) ***Short-term (~3 years):*** to study the effect of feedwater temperature and pressure on RO membrane performance characteristics over a range of temperatures (20–45°C) and pressures (55–69 bar). The intent is to gather data on all aspects of system operation, utilizing membranes from three different manufacturers, so that sufficient data analysis is possible to determine if the performance and economic benefits suggested by the analytical models can in fact be demonstrated by experiments, and to determine the possible differences in results due to materials and type.
(C) ***Long-term***: to study the effect of feedwater temperature and pressure on RO membrane performance characteristics as a function of time. The intent is to select one of the membranes used during the short-term program for extended study to investigate possible reduction in membrane lifetime due to effects such as increased fouling or membrane compaction.

Design of experimental unit

The facility was designed by one of the most reputable and experienced desalination consulting firms in the Arab World, the Consulting Engineering Company (CEC) and reviewed by Dr. J. R. Humphries of CANDESAL Enterprises Ltd under technical assistance from the IAEA. The cooperation between NPPA, CEC and IAEA resulted in successful completion of the design. Major design parameters are outlined below.

(a) Number and size of membranes

To determine the minimum number of membranes needed to ensure statistical relevance of the results, a statistical analysis was carried out to determine sample size for any one type of membranes tested. The analysis indicated that a sample size of five membranes in parallel is optimal. The size of the experimental unit will be equal to the number of parallel membranes multiplied by the capacity by the particular membrane to be tested.

Because the objective of the experimental facility is to investigate membrane performance, rather than the production of potable water, it is important for economic reasons to have the smallest capacity capable of representing performance characteristics under investigation.

- Commercial membranes are produced in various sizes; the most common of which are the 4” and the 8” diameter membranes. From the economic and practical point of view, the 4” membrane seems to be a more attractive choice. However, the primary risk introduced by the use of smaller diameter membranes is the potential that the performance characteristics of the 4” membranes may not be representative of the performance characteristics of 8” membranes under operating conditions expected during the experimental program.

- In order to assess this risk, a number of “membrane equivalency comparisons” of the performance characteristics of 4” and 8” membranes have been made using the ROSA code. The analysis indicated that the 4” membrane provides a very close equivalence to the 8” membrane in its essential performance characteristics. However, the feed water flow rate and the daily energy consumption are significantly less for the 4” membrane. Therefore, the experimental program can be carried out with 4” membranes without adverse impact on its representation of essential performance characteristics. Moreover, potentially significant savings in program capital and O&M costs can be realized.

Different commercial membranes have different performance characteristics. In particular, they have different nominal permeate flows and conversions at some standard test conditions. Because the high-pressure pump should accommodate the different commercial membranes to be tested, it will be based on the highest anticipated feed flow.

The short term experimental program shall be based on the three 4” spiral wound membranes manufactured by Filmtec, Fluid Systems and Hydranautics. The rational for this selection is:

- ♦ The capability to operate at high feed water temperature (45°C).
- ♦ Similar permeates flows and recovery ratios. Thus, limiting the operational range of the high pressure pump, this will facilitate the pump selection.
- ♦ Similar dimensions and materials. This should facilitate racking requirements and changing from one commercial membrane to another as well as direct comparison of performance.

(b) ***Configuration of the test facility***

The test facility consists of two identical units: one unit operating with preheated feedwater at 25, 30, 35, 40 and 45°C (Train A) temperatures and the other at ambient seawater (Train B). This configuration is considered practical with 4” membranes, and has the benefit of giving direct comparison of performance characteristics for the preheated and no-preheated cases at all values of preheat temperature. The test facility consists of the following main components, as shown in Figure 4.17:

(i) ***Beach wells and pumps:*** To ensure clean feedwater with minimum pretreatment requirements and lower operational costs, beach wells will be used for the feedwater rather than open seawater intake.

(ii) ***Pretreatment system:*** The pretreatment system is designed to allow for the various pretreatment requirements for the different commercial membranes to be tested.

(iii) ***Water heating system (for one unit only):*** The feedwater will be heated by freshwater/seawater heat exchanger. The hot fresh water shall be obtained from an electric water heater. The hot brine and permeate shall be used to preheat the feedwater, utilizing permeate/seawater and brine/seawater heat exchangers.

(iv) ***High pressure pump with energy recovery and hydraulic coupling***: The experiments involve different types of membranes, requiring different operating pressures and feed flows. Therefore, the high-pressure pump is coupled with a hydraulic coupling to obtain the required pressure-flow.

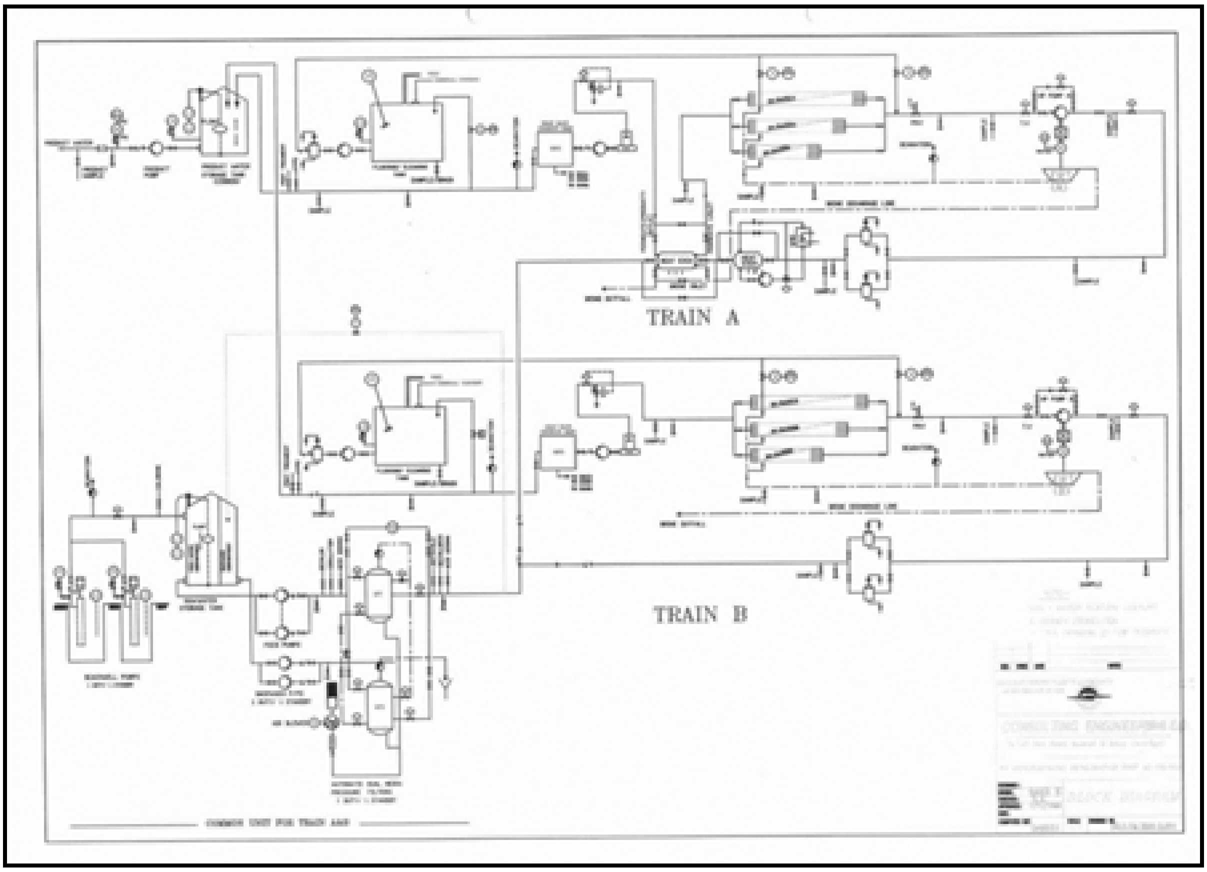


Figure 4.17: Schematic diagram of El-Dabaa experimental facility.

*(v) Racking***:** To accommodate the different dimensions and piping arrangements of the membranes and to reduce the time required for switching to a different experimental Phase, each unit is provided with three racks with connections suitable for a particular membrane.

(vi) Other systems common to commercial RO.

The experimental program

The experimental program is intended to produce data that can be used to validate the performance characteristics of RO feedwater preheating. The planned sequence and timing of testing (Steps 1–5 below) is provided below is depicted in Figure 4.18 for a one month cycle.

On the assumption operating parameters, including feed temperature, can be changed and a stable plant condition reached within a time period of 12 hours, the above test sequence should take on the order of 24 days to complete. Having completed one full test cycle (Steps 1–4) and returned the plant to maximum pressure and temperature operating conditions (Step 5), operation should continue for the balance of the month (approximately 6–7 days).

Following one full month of operation (including testing), the next test cycle (Steps 1–5) should be carried out. This pattern should be repeated for the duration of the test phase. The experimental program is to consist of 3 phases, each phase taking approximately one year and consisting of testing with one of the three membranes being evaluated. Following this schedule will provide a series of data sets taken at monthly intervals over the duration of a year, for each membrane type. This cycle is repeated each month for the full year of testing.

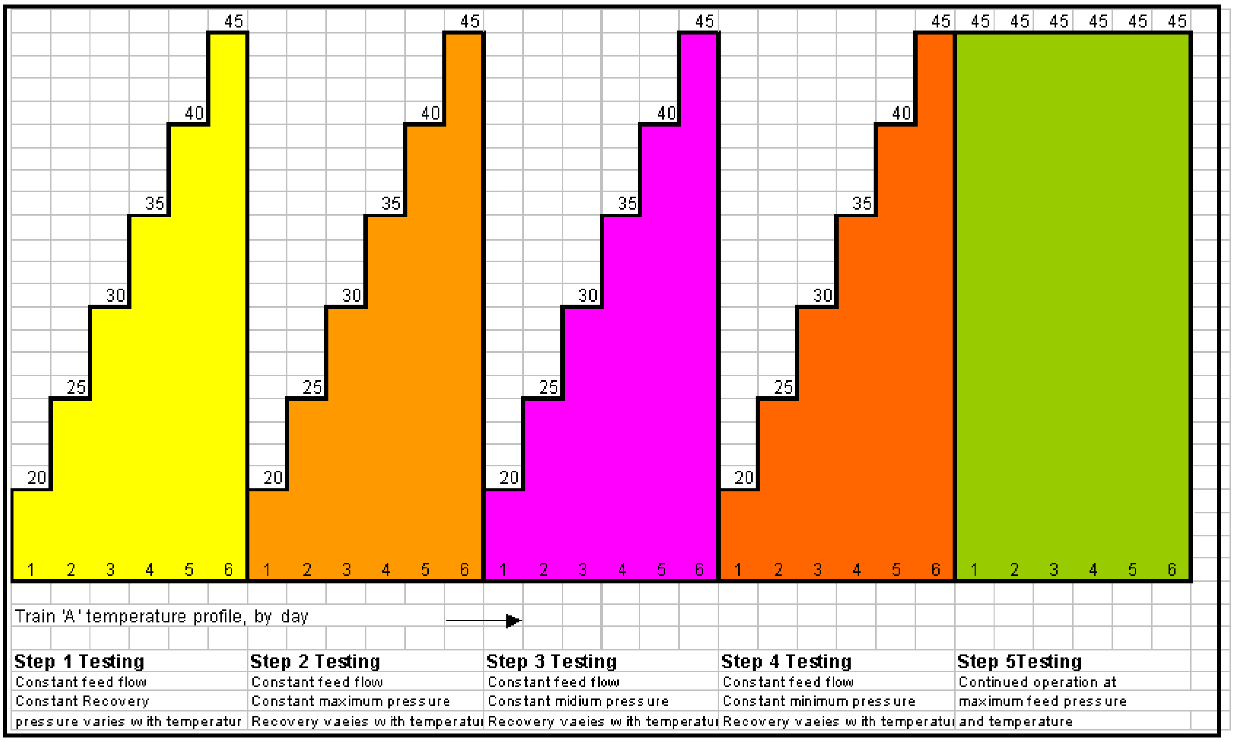


Figure 4.18: One-month test cycle.

Results of experimental work

Based on the pre-construction activities already performed at the beginning of this project, it was envisaged that the experimental program could start at the beginning of January 2000.

However, delays were encountered due to the following:

- Bid evaluation took longer than expected.
- Re-bidding was necessary due to legal and financial constraints.
- Start of construction was delayed due to delays in submitting detailed design.
- Devaluation of Egyptian currency imposed difficulties on importing equipment.
- Problems in executing civil works.

As a result the construction of the facility was not completed within the period of this CRP. However, preliminary calculations carried out at the beginning of this project indicated that the utilization of preheated feed water will lower the operating pressure necessary to produce certain permeate flow rate, as shown in Figure 4.19. Calculations have also shown that despite of the increase in product water salinity as a result of preheating feed water, product water quality will still meet WHO standards, as shown in Figure 4.20.

These anticipated results have been confirmed for the short-term use of preheated feed water by the experimental work carried out by BARC and CANDESAL. However, NPPA experimental program will mainly address the long-term impact of preheating feed water on membrane performance and life time, and hence, will have a strong influence on how the international nuclear desalination community perceives the value/benefit of feed water preheating.

The facility is expected to be operational before the publication of this Technical Document. Therefore, some experimental results will be included. However, NPPA remains committed to making the results of the experimental program available to Member States.

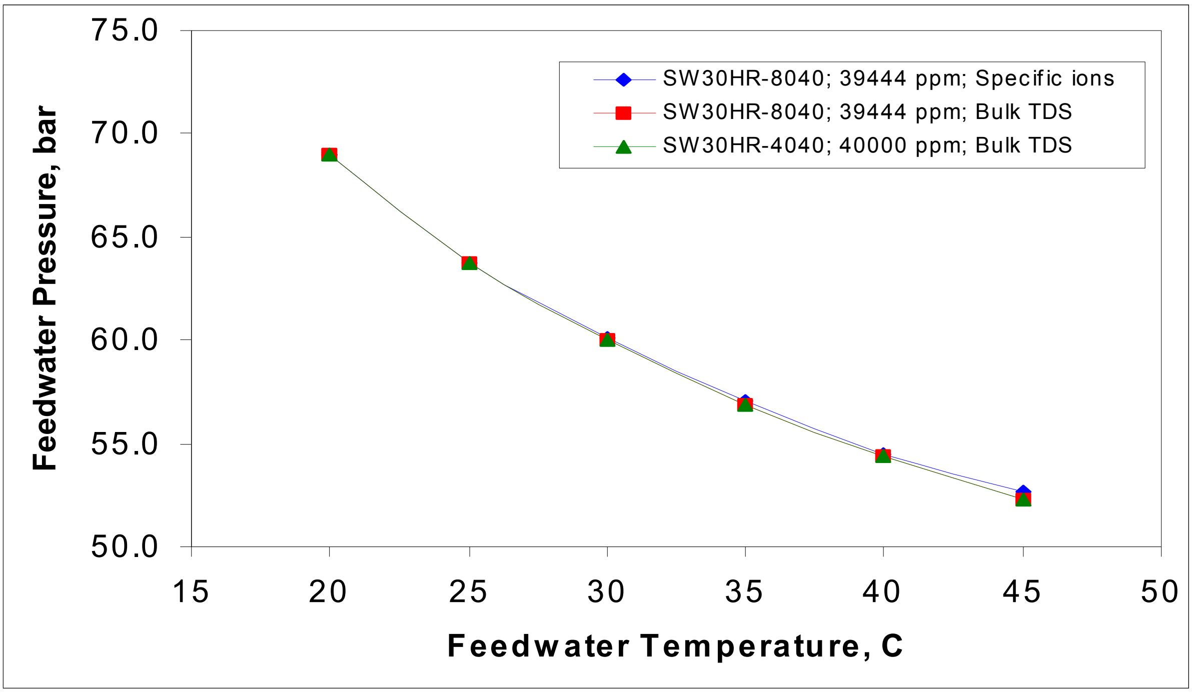


Figure 4.19: Comparison of feedwater pressure versus temperature.

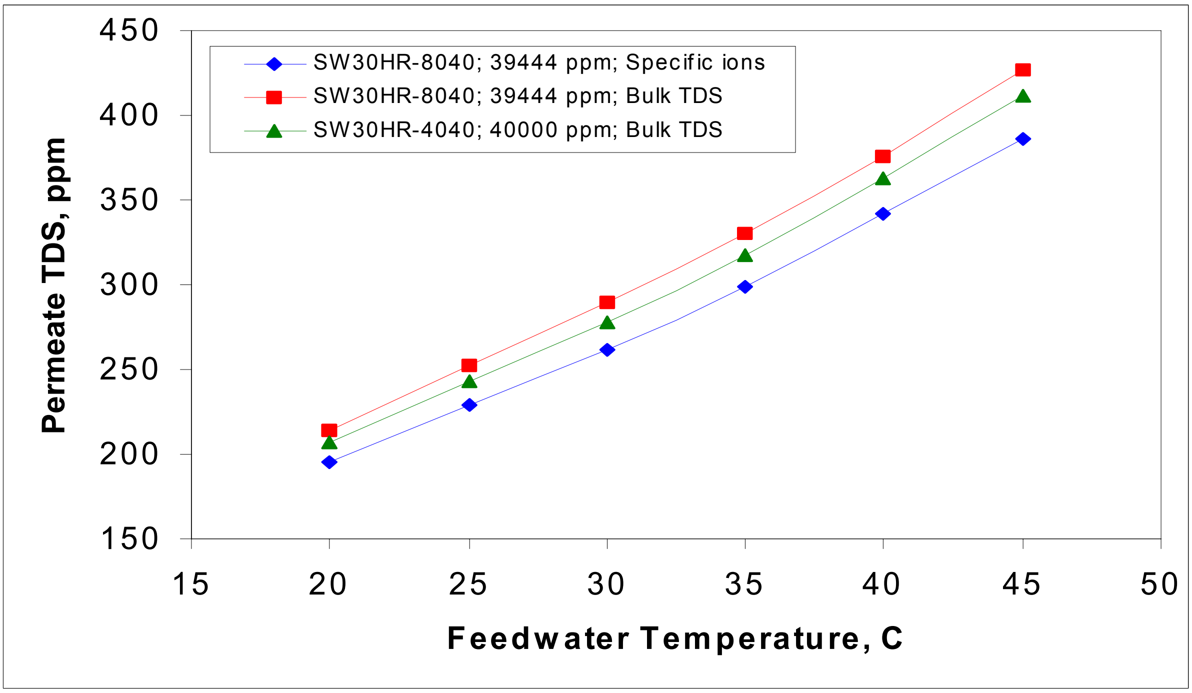


Figure 4.20: Comparison of permeate TDS versus feedwater temperature.

4.2.1.3. BARC test loop

The performance data of the 100-m^3/day seawater reverse osmosis plant at Trombay during various seasons have indicated that the water production rate shows a definite increase with temperature. The studies carried out have indicated that at higher temperatures the throughput of water through the membrane has shown an increase. The solute rejection has also shown minor changes. The theoretical projections using Film Tech membrane (SWHR-380 and SWHR-8040) have indicated a gain in the water permeation with temperature albeit with a minor penalty on permeate quality. In practice, the membrane performance is concurrently controlled by feed flow rate, operating pressure and temperature besides the compaction effect on the membranes due to temperature, pressure and time. Therefore, the experiments were carried out using three 4040 spiral RO elements (Film Tec, Koch and Hydranautics make) at different feed velocities and pressures.

Objectives

- To study the performance of SWRO plant at varying feed salinity, temperature and pressure and utilization of data for preheat RO.
- To study the dependence of solute rejection and product water flux on seawater temperature.

Main features of the test loop

The test loop as shown in Figure 4.21 consists of:
- A 1-m^3 capacity feed tank.
- A high pressure pump of 40 1pm flow and 50 bar head.
- Provisions were made to connect individual 4040 elements one after the other.
- The reject and permeate stream were connected to the feed tank.
- The temperatures of the feed at the point of entry to the pump, reject and permeate streams as they fall into the feed tank were measured within an accuracy of ± 0. l C.
- Rotameters were fitted in the reject and permeate line to measure the flow rates.
- Flow rates were measured manually also using calibrated containers, whenever required.

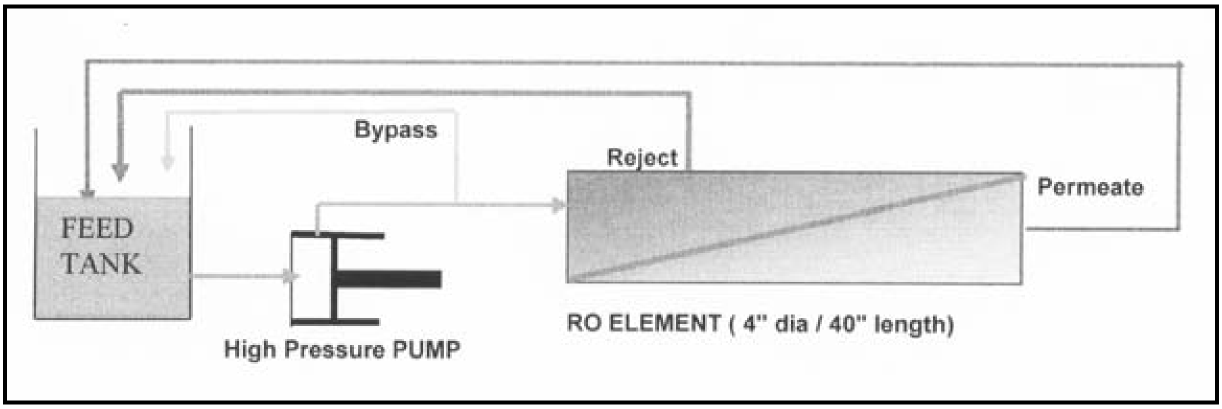


Figure 4.21: BARC Experimental Test Loop.

Experimental conditions

Since ambient water temperature was always around 27–28°C, data could not be collected beyond this temperature under the present experimental conditions. Using one of the elements experiments were conducted as a function of feed flow rate, varying the value from about 6 L/min to about 40 L/min. The experimental conditions are summarized below.

- Source of seawater:	Trombay
- Feed TDS	35000 ppm
- Membrane elements:	2512 Spiral SP4 BARC
	4040 Spiral SP5 BARC
	4040 FilmTech
	4040 Fluid System (Koch)
	4040 Hydranautics
- Temperature range	25 – 40^{0}C
- Pressure range	50 – 60 bar
- Feed flow rate	6 – 40 lpm

Results of experimental work

BARC carried out the experimental program in three Phases. In Phase-1, the behavior of solute rejection and permeate flux were studied as a function of temperature from 25 to 40°C; at different operating pressures ranging from 50 to 60 bar, using BARC wound commercial size 4040 spiral wound element at low recovery (~7%) conditions. The experiments were carried out using a re-circulation mode thereby allowing the temperature to rise constantly with time.

The experimental results, as shown in Table 4.17, indicated that solute rejection increases with temperature up to 32°C and thereafter it decreases. However, the change is only marginal. Permeate flux increases by about 1.5–2.3% for every degree rise in temperature.

Although the above results were considered stable within the errors of measurements, BARC decided to carry out more experiments to confirm the observed behavior (Phase-2). Two sets of experiments were conducted in Phase-2. The first set of experiments was conducted with a 4040 spiral element connected to a test loop. A flow rate of 30 1pm was maintained and the requisite temperature was maintained by the re-circulation of the reject stream. Data were collected at three different pressures namely 50, 55 and 60 bars. The results obtained confirmed earlier observation that the solute rejection improves only up to 32°C and beyond it marginally declines. The reduction in solute rejection was also very small.

TABLE 4.17: PERFORMANCE EVALUATION OF BARC SPIRAL ELEMENT

Pressure, bar	**Performance**	**Temperature, ^{0}C**				
		26	**28**	**32**	**36**	**40**
50	SR, %	92.5	92.9	93.3	93.2	93.1
	Flux, $lm^{-2}d^{-1}$	440.6	468.5	499.2	543.3	583.7
55	SR, %	93.1	93.1	93.9	93.8	93.6
	Flux, $lm^{-2}d^{-1}$	487.4	529.9	568.3	606.7	645.1
60	SR, %	93.8	94.4	94.7	94.5	94.3
	Flux, $lm^{-2}d^{-1}$	529.4	578.4	622.4	670.5	710.4

Possible errors in the above experiment might be due to the constant change in temperature and interpretation of the data in terms of solute rejection, which is less sensitive for small changes in solute passage. Accordingly, the second set of experiments was conducted in a controlled environment. A 200 litre tank was fitted with a 2 kW heating element and a stirrer. Initially the solution was heated to the requisite temperature and stopped. Later, the temperature could be maintained at ± 0.5°C. In order to sustain the temperature within the limits, we were constrained to use a maximum of 16 1pm flow rate. In view of this the experiments were carried out with 2512 elements at two different pressures 50 and 55 bar.

Solute flux was determined at five different temperatures starting from the ambient temperature. They were fitted into a second-degree polynomial function and values at 20°C were extrapolated. Using the solute flux at 20°C as a standard (NAo), all the other solute fluxes were normalized as dimensionless ratios (NA/NAo). Similarly water fluxes were also normalized with respect to 20°C value as (NB/NBo). Plots of these two ratios as a function of temperature are shown in Figures 4.22 and 4.23. The water flux shows a near linear relationship while the solute flux shows a slow decline initially followed by a significant rise, with the point of inflexation being observed at about 32°C. These observations confirmed that solute rejection would improve up to about 32°C and later would decrease.

These experiments were repeated in Phase-3 with three commercial membranes (FilmTec, Fluid System and Hydranautics) to confirm the behavior and ascertain the factors responsible. Phase-3 experiments indicated that Hydranautics membrane exhibits the best solute rejection at both the flow rates. The solute rejection is about 99.2% for a 6 lpm flow rate, while it is about 99.6% at 25 lpm flow rate. Film Tec membranes exhibit marginally lower rejection of about 99.3% at higher flow rate, while the solute rejection is significantly lower at about 98%.

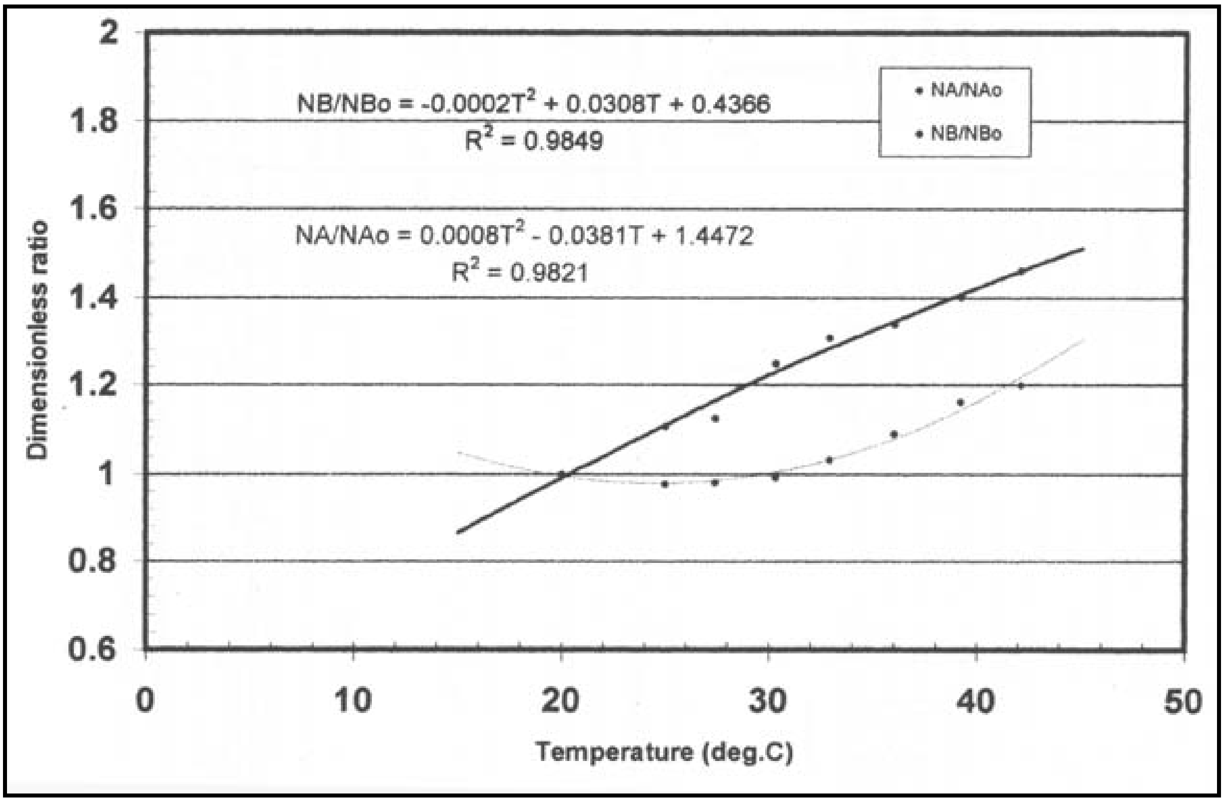


Figure 4.22: Effect of temperature on membrane performance.
(50 bar, 16 lpm, 2512 element).

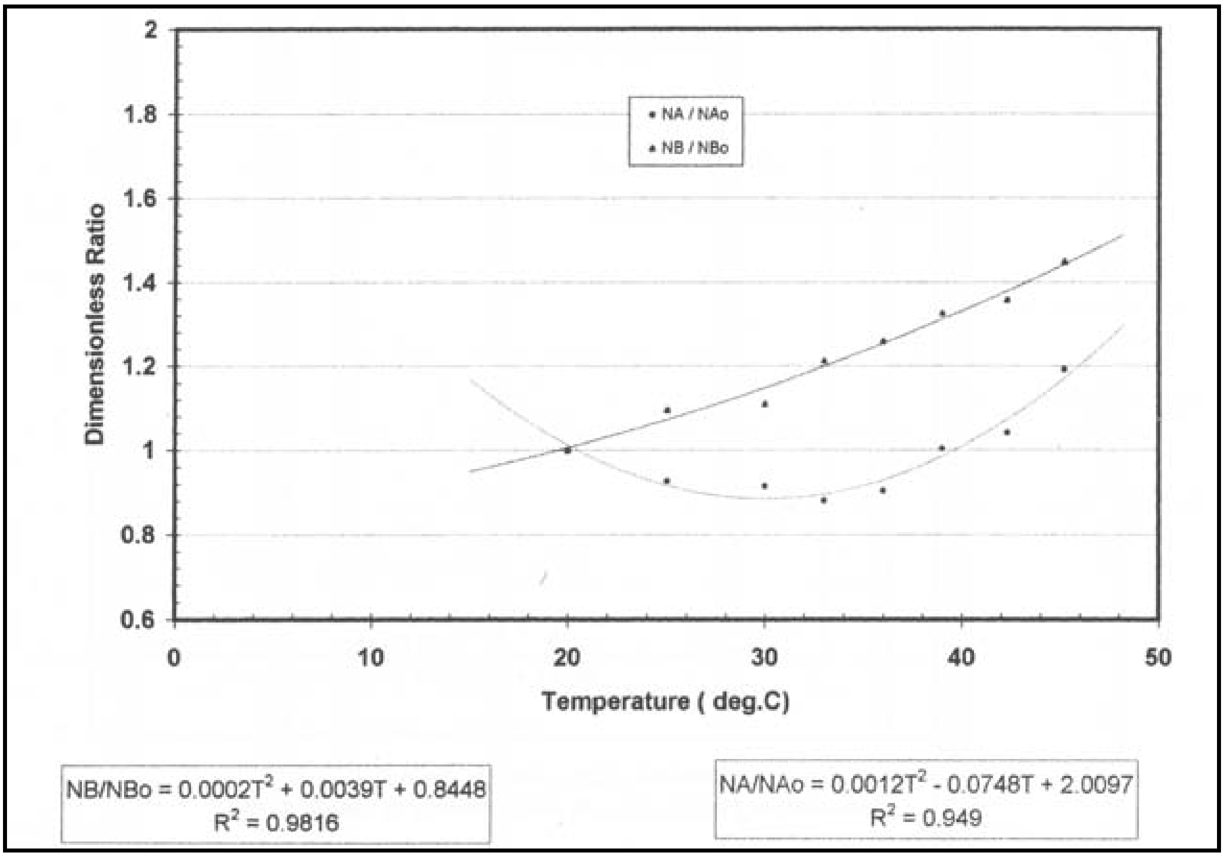


Figure 4.23: Effect of temperature on membrane performance.
(55 bar, 16 lpm, 2512 element).

Fluid system element exhibits a lower solute rejection compared to other two membranes with around 98% for 6 1pm flow rate and 99% at 25 1pm flow rate. The effect temperature on solute rejection is more pronounced for all the cases at lower flow rates. The behavioral pattern is also different for all the three membranes, as seen in Figure 4.24, indicating the temperature effect is dependent on membrane material. Film Tech membranes show a marginal decrease in solute rejection with temperature while the other two membranes exhibit a maximum in solute rejection at around 35°C.

Film Tech membrane element, as seen in Figure 4.25, shows a higher water flux compared to the other two membrane elements exhibiting on an average about 1% rise per degree Celsius in the temperature range of 32–40°C. The water flux of Hydranautics membrane element is only about 50% of FilmTec but shows an average increase of about less than 1% per degree Celsius up to 35°C and thereafter about 3 - 4% per degree rise. Fluid system exhibits about 80% of the FilmTec membrane flux and exhibits an average increase of about 2% per degree Celsius. Except in the case of Hydranautics membrane element, the rate of flux increases with temperature decreases at higher temperature.

4.2.2. Experimental validation of thermal systems

Commercial thermal desalination technologies include MED and MSF. Within this CRP experimental work has been carried out on these processes by INET, China and BARC, India to study performance under varying conditions and/or acquire design information data. INET, China designed a test loop simulating 4 effects of a VTE-MED process to obtain the necessary design data and experience. BARC, India utilized existing MSF and LTE facilities to acquire needed performance and design information for coupling a 6300 m^3/day hybrid seawater desalination plant (4500 m^3/day MSF and 1800 m^3/day RO) with PHWR at MAPS, and for coupling LTE with CIRUS research reactor. This Section provides a brief description of the experimental facilities and experimental programs for validation of thermal systems.

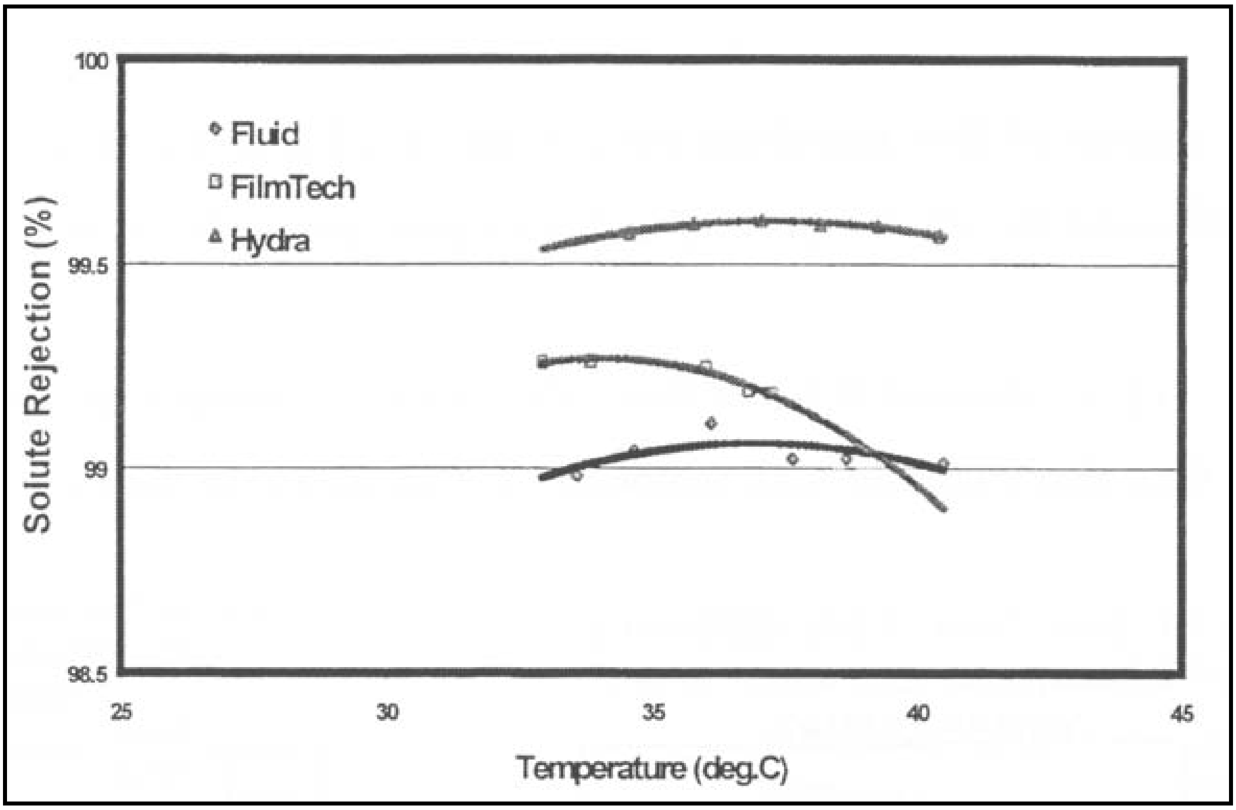


Figure 4.24: Variation of solute rejection with temperature for commercial membranes. (50 bar, 35 lpm, 35000 ppm).

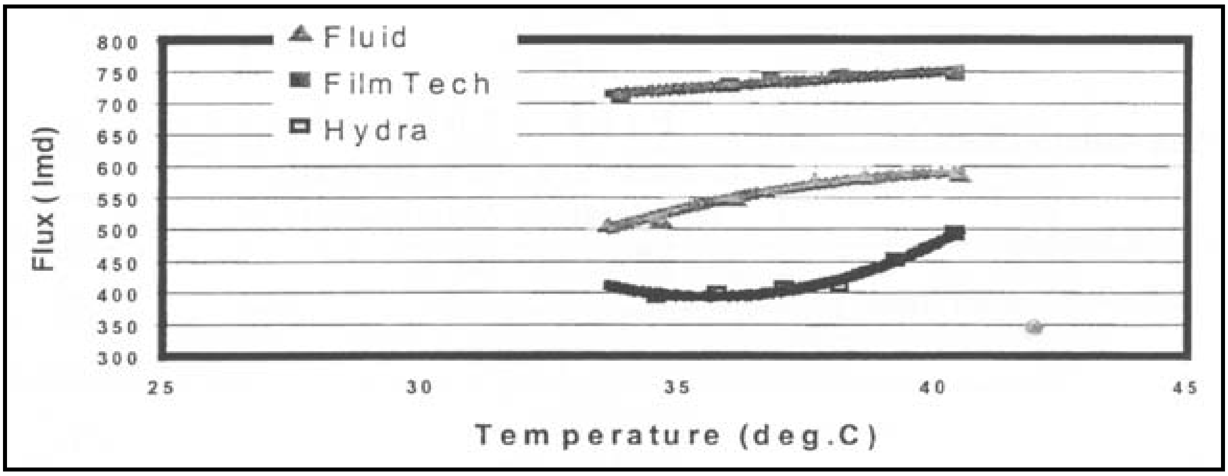


Figure 4.25: Variation of water flux with temperature for commercial membranes. (50 bar, 25 lpm, 35000 ppm).

4.2.2.1. INET experimental facility

Considering the importance of design data and experience for the design of VTE-MED process, an experimental investigation program was developed by INET. A test facility with 4 effects was designed and constructed. The facility can simulate any successive 4 effects of the 28 effects in the VTE-MED process, by adjusting the operation parameters.

Objective of the experimental program

The main objectives of the experimental investigation were to gather experience and accumulate necessary database for the design and operation of VTE-MED process:

- Conduct an investigation in operation performance of the VTE-MED process, including suitability of the desalination process to NHR reactor, system self-regulation and load following performance,
- Investigate the thermo-hydraulics of VTE-MED process, including heat transfer performance, characteristics of orifice plate, thermo-fluid dynamics in the system.

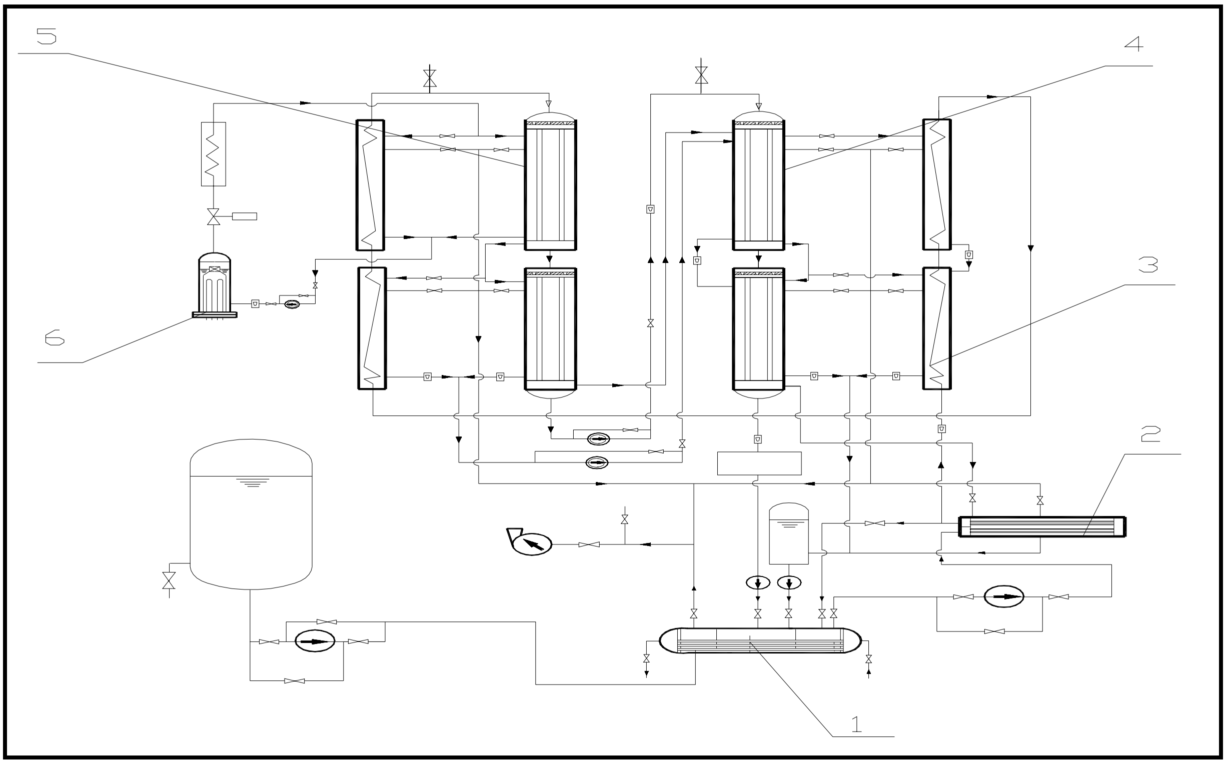
Design of experimental unit

The VTE-MED experimental facility consists, as shown in Figure 4.26, of motive steam system, raw seawater supply system, product fresh water circuit, vacuum system, seawater sub cooling control system, equipment cooling system and measurement system. The main equipment includes an electrically heated motive steam generator, a raw sea water tank, 4 pre-heaters, 4 evaporation effects which are arranged in two parallel towers, a final condenser, auxiliary feed system, pumps, valves and some connecting pipes. The electrically heated steam generator is the power source, which supplies the motive steam for the test unit; it simulates the steam generator of the heating reactor.

The raw seawater tank is a heat exchanger whose shell side is used as seawater tank and the cooling water flows inside the heat transfer tubes, so the temperature of the inlet raw seawater can be easily changed in order to model the season change. The pre-heater and its counterpart evaporator are not included into one vessel in the test loop, four pre-heaters are connected with its counterpart evaporators at the steam side (shell side) and their tube side are connected in series. Brine flows inside the tube, and the vapor condenses on the outside of the heat transfer tubes. The pre-heaters are installed stand-by-stand with their counterpart evaporators.

Four evaporators are installed in two towers and each tower includes 2 effects, which are vertically stacked up in series, the exit of the top effect is the inlet of the below effect in each tower. At the bottom of each effect there is a brine reservoir, which collects the brine flowing down form the above evaporator and also to provide a region for separating vapor from brine.

There are double shells around the reservoir, its outer shell is the pressure seal vessel and the inner shell is the shroud of the reservoir. On the top periphery of the inner shroud six windows with multi-layer net made of compressed stainless wire are used as the demisters. The vapor separated from the brine, passes through the demister window and flows downward along the alleyway between two shells to the shell side of the below effect and is used as heating steam for the effect.



1	**Seawater tank**	**2**	**Final condenser**	**3**	**Pre-heater**
4	**Main evaporator**	**5**	**Seawater heater**	**6**	**Steam generator**

Figure 4.26: Schematic diagram of INET test facility.

The first effect is the seawater heat exchanger, which transfers the heat from the motive steam to the brine flowing and preheated in the pre-heaters. The following three effects can be used to simulate any successive 3 effects of the 28 effects in the VTE-MED process by adjusting the operation parameters.

The final condenser is also working as the vent condenser. Vapor from the last effect flows into the shell side of the final condenser, condenses on the outside of the heat transfer tubes and transfers heat to the seawater which flows inside the final condenser heat transfer tubes. As part of the production water the condensate flows into the distillate system. The preheated seawater flows upward via pump as feed water into the final stage pre-heater. The non-condensable gas separated from the vapor and extra vapor are vented from the final condenser. A water sealed vacuum pump is used as the vacuum equipment, and all effects, pre-heaters, seawater tank, final condenser and steam generator are connected with the vacuum pump, the vacuum of every equipment can be adjusted separately.

The nominal design capacity of water production is about 4 m^3/d for the test unit. In order to simulate the process in an actual seawater desalination plant, the dimensions of the heat transfer tube and operation parameters in the test unit are the same as those in the real prototype plant. This resulted in the big size and the complexity of the test unit. The full height of the test unit is about 10 m.

Results of experimental work

The VTE-MED test system can be operated steadily in the seawater flow range from 1000 to 2600 kg/h. The temperature differences between the two sides of the heat transfer tubes, can be achieved and maintained during the tests. The test unit was operated stably under different pressure level. The results showed that the present VTE-MED test unit is successful both in design and in its stable operation. Design and operation experiences were accumulated.

Pressure drop and temperature distributions in the system, mass flow rate of steam, brine and fresh water were recorded. All the recorded data were presented and provided with figures and data documents, which provide the data base for verification of the thermal hydraulic design of the evaporators and pre-heaters. Some recorded test data are presented below as examples. Figures 4.27 and 4.28 show the make-up seawater mass flow rate and temperature differences between two sides of heat transfer tube, respectively.

In the VTE-MED test unit, the make-up seawater is pumped from a limited volume seawater tank which is not open to the atmosphere, as a result, the pressure of seawater tank, water level and the balance of mass flow rates between two towers are greatly impacted by other systems if the pressure feedback, particularly the vacuum effectiveness is not considered and solved properly. The measures taken to overcome the unstable make-up seawater flow were:

(1) to add pumps between exits of brine and fresh water flow ways and raw seawater tank;
(2) to adjust the pump rolling speed by frequency adjusters which were controlled by feedback signals of seawater levels in tank and seawater ponds.

The experiment results also shown that the brine temperature dropped more than vapor temperature did when they were transferred from one tower to another one along different ways in the test, the inconsistent temperature drop of brine and water means some extra heat from the vapor will be used to increase the brine temperature, not to produce the vapor and fresh water in effect No. 3. The result reminds us more attention should be paid to the thermal isolations in actual plant as the brine will be pumped over several decade or a hundred meters from the bottom of the first tower to the top of the second tower, the temperature drop of 2 to 3 degree centigrade means the loss of production water ability of an effect evaporator.

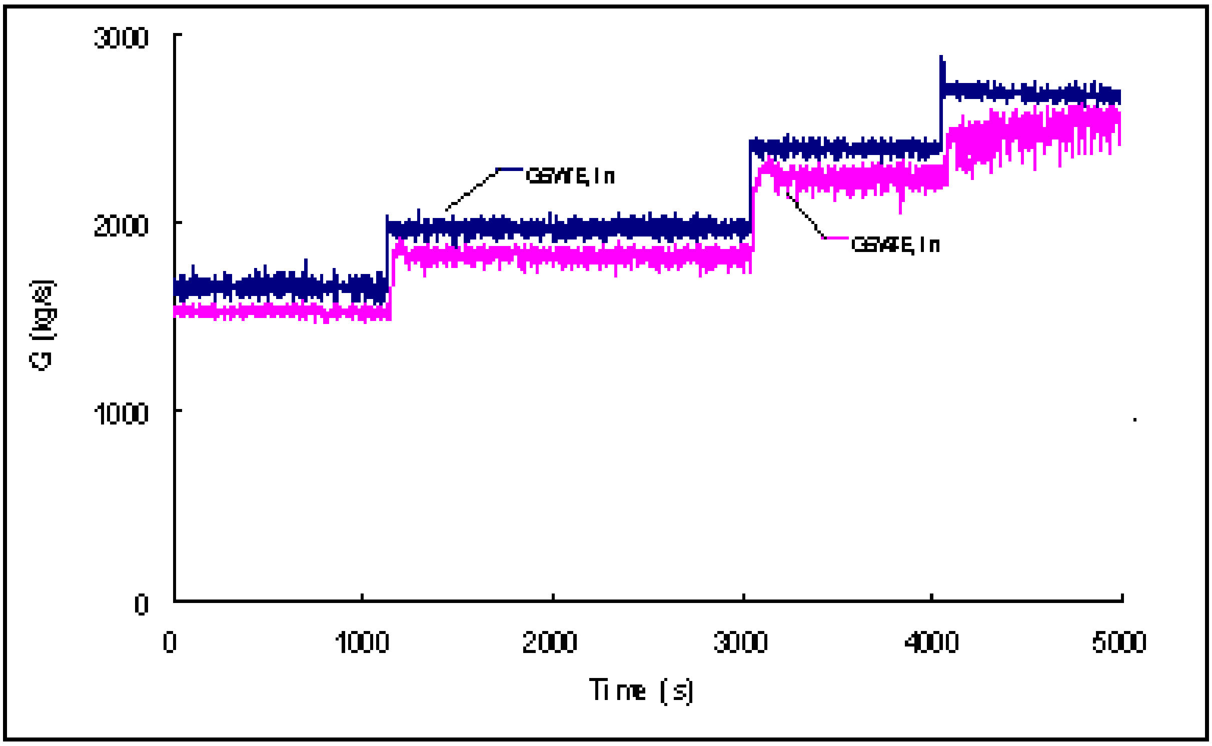


Figure 4.27: Mass flow rates of make-up seawater and brine.

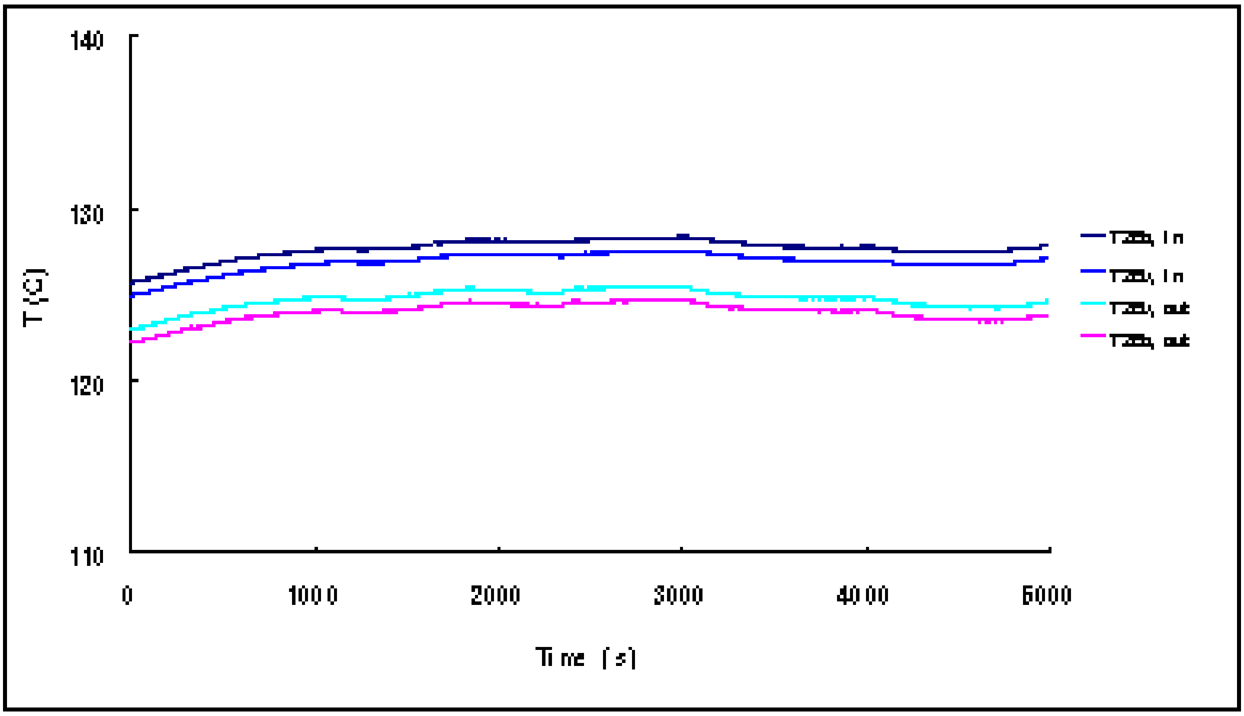


Figure 4.28: Inlet and outlet temperatures of brine and vapour in effect No. 2.

The vacuum system was also very important in the VTE-MED test unit, the incorrect intake positions of non-condensable gas exhaustion tubes would result in the failure to exhaust the non-condensable gas completely. The vacuum in the system also would cause some problem for differential pressure measurement meters because the vacuum would lead to exhaustion of pressure transfer liquid in the pressure transducer tube.

The pressure P_{SG} in motive steam generator was the highest pressure and the pressure P_{fL} in final condenser was the lowest one in the tests, the seawater pressure decreased step by step from entrance of effect No.1 to exit of effect No.4. The experimental pressure records presented in Figure 4.29 shows that the above described correct pressure distribution in the system could be constructed and be maintained for an enough long time during the tests. The VTE-MED test unit has been steadily operated under the seawater pressure range from 1.1 bar to 1.8 bar, the corresponding highest brine temperature is over 115°C.

Figure 4.30 shows the test results of the fresh water mass flow rate in effect No. 4 when the temperature drop across the heat transfer tubes was increased. From the figure, it is clear that the mass flow rate of the produced fresh water increases with the increase of the temperature drop across the heat transfer tubes in effect No.4.

The orifice plate, which is located at the entrance in every effect, is very important in the VTE-MED unit. When the brine from the seawater pool flows through them, the pressure of the brine will decrease and the seawater will flash with the decrease of the pressure, so some bubbles appear in the brine and the brine become the foaming flow. The foam fills the space between the plate and the bundle sheet of the heat transfer tube, so that the foaming flow is regulated by an orifice plate and flows into the individual tubes evenly, the orifice plate acts as an efficient distribution plenum for the brine. The stable and correct pressure drops across the orifice plate are the basis for the efficient fresh water production; the pressure drops determine the temperature differences between successive effects. The pressure drop and mass flow rate across the orifice plate were recorded, which provide the data for verification and correction of the design of flow resistant coefficient of the orifice plate.

More experiment investigation should be performed in the future. It should be noted that the data accumulated up to now are still not enough to be used to verification of developed code.

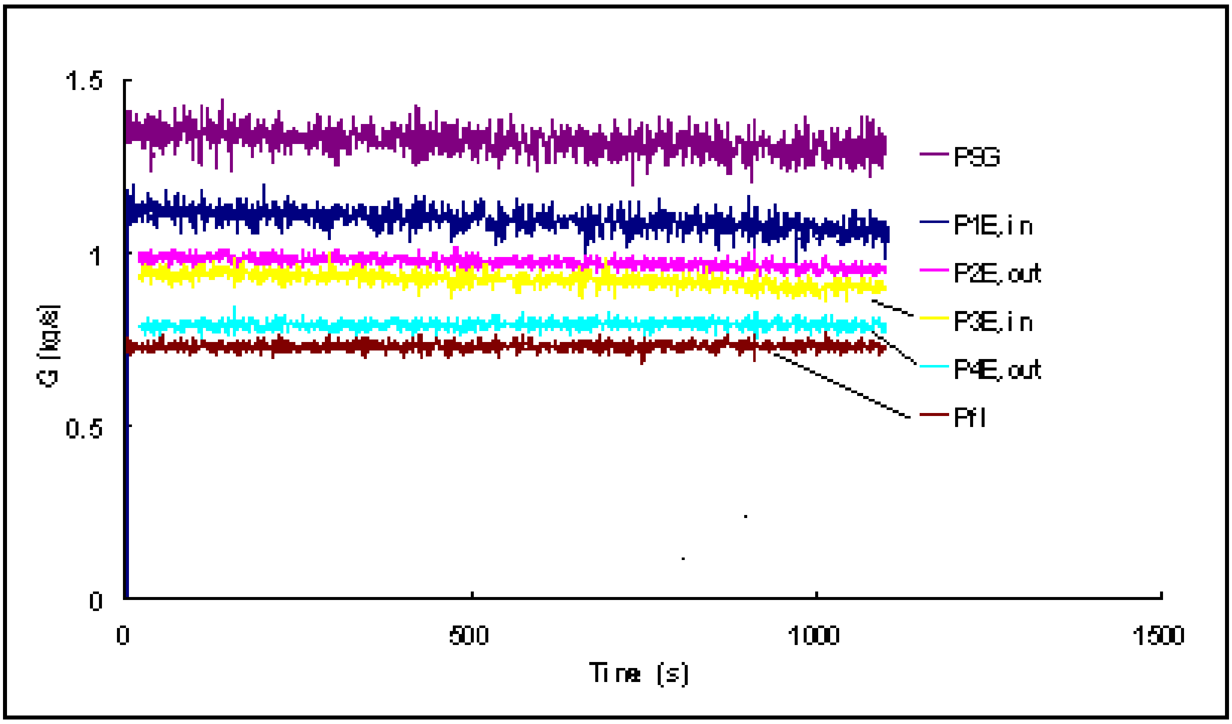


Figure 4.29: Pressures on the different positions in the system.

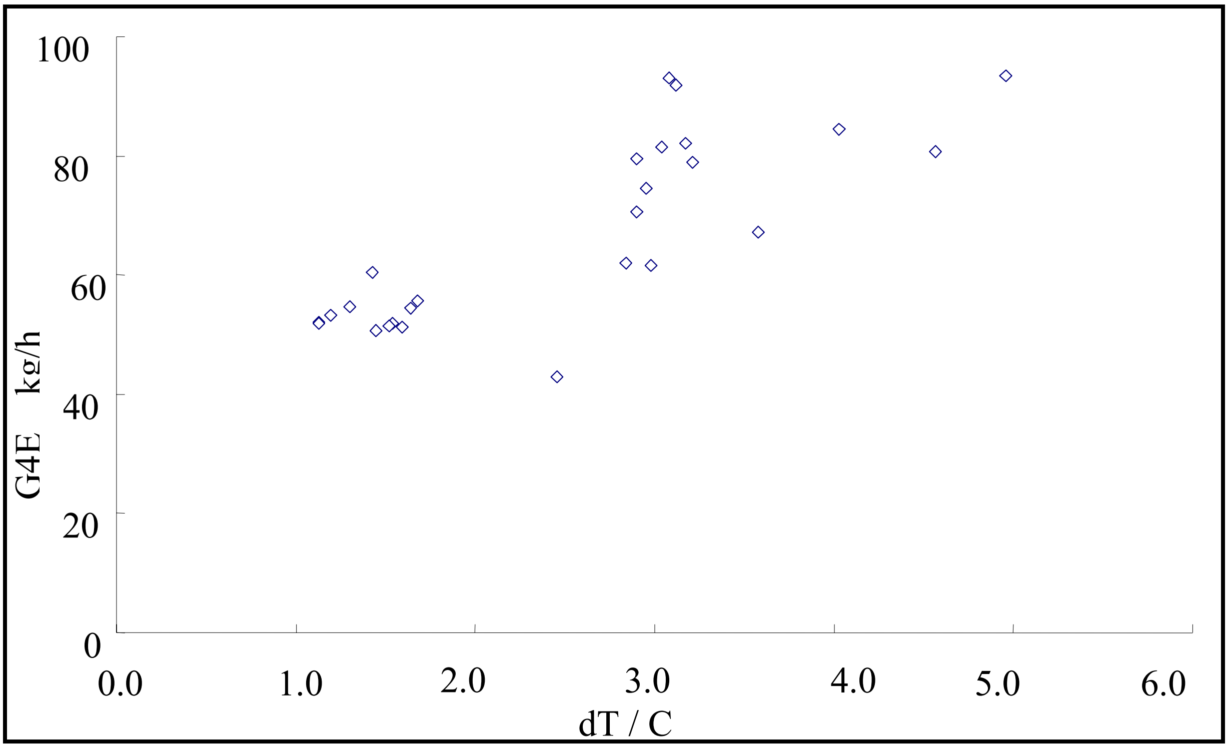


Figure 4.30: Fresh water mass flow rate in effect No. 4.

4.2.2.2. BARC Thermal experimental units

BARC has been engaged in research and development work in the field of evaporative and membrane desalination for about two decades. Beside the 40 m^3/day SWRO plant described above, a 425 m^3/day MSF seawater desalination plant was designed, manufactured, installed and operated successfully by BARC. A 30 m^3/day Low Temperature Evaporator (LTE) desalination plant was also installed at BARC for utilizing waste heat from the research reactor CIRUS. These units were used to obtain data needed for the optimization of the hybrid desalination plant (4500 m^3/day MSF and 1800 m^3/day RO) to be coupled with Madras Atomic Power Station (MAPS), and the coupling of LTE with CIRUS reactor. The experimental work carried out in relation to these goals is described below.

Experimental work on MSF plan.

The operational data of the MSF pilot plant at Trombay at different top brine temperatures were utilized for performance evaluation of MSF process. Two variants are selected for studying the performance of MSF plant for coupling to a nuclear reactor. The first is the effect of raw seawater temperature, as it would vary if it is taken directly from sea and if taken from out fall streams of nuclear power station. The second is the effect of top brine temperature variation, likely to occur due to transients, on the performance of MSF plant. A typical data sheet of the 425 m^3/d MSF pilot plant from a trial run conducted during the year is given in Table 4.18. The results of the experimental work indicated that increasing seawater would have the following impact.

(a) The amount of raw seawater required in the reject stages has to be increased to maintain the blow down temperature. This increases the pumping power requirement for the same production level.
(b) If seawater flow in reject module is not increased, blow down temperature will go up and total flash drop ranges will come down. This reduces the output and the performance ratio.
(c) With higher blow down temperature, however, the product quality improves slightly.

On the other hand, lower top brine temperature would have the following impact:
(a) Pumping power requirement increases for the same production
(b) Production and performance ratio decreases for the same pumping power
(c) Lower product quality with lower blow down temperature.
(d) Scaling/corrosion problem reduces.
(e) Plant stability decreases.

Data was also collected on combined post treatment of product water from MSF and RO plant at Trombay. A test facility was installed to collect the product water from the plants in different proportions and dosing of milk of lime (Figure 4.31).

The treatment scheme has been designed for carrying out two different types of experiments namely (i) alkalinization of RO permeate in limestone column and (ii) combined post-treatment of MSF distillate and SWRO permeate by milk of lime dosing in a mixing ratio of 2.5: 1. Hydrated lime of 0.5% solution at a pH of 12.2 was dosed. Combined post-treatment facility of MSF distillate and SWRO permeate in a mixing ratio of 2.5: 1 (100 lpm/40 lpm) have been run for about 100 hrs and the results are mentioned in Table 4.19. MSF distillate TDS and temperature was in the range of 10–50 ppm and 44°C respectively while SWRO permeate was in the range of 380–500 ppm and the temperature was about 28 °C.

TABLE 4.18: OPERATIONAL DATA OF BARC MSF PILOT PLANT

Seawater		**Blow down**		**Product**	**Top brine**
Concentration (ppm)	**Temperature (^0C)**	**Concentration(ppm)**	**Temperature (^0C)**	**Concentration (ppm)**	**Temperature (^0C)**
38,000	23	54,000	44	10	121.0
38,000	24	53,000	44	10	120.8
40,000	23	64,000	42	10	121.0
42,000	24	-	44	10	120.6
32,000	25	-	43	10	120.2
25,000	24	31,000	45	10	121.0
40,000	24	62,000	44	10	121.7
37,000	26	-	48	10	121.6

The results of combined post-treatment facility are quite satisfactory. The TDS, alkalinity and LSI value obtained by the experiments are well within the desirable limit. However, the mixed water coming out of the process is slightly turbid. Attempts are to be made further to eliminate the turbidity either by increasing the carbon dioxide concentration or a suitable cartridge filter may have to be put before final delivery to the water reservoir.

Experimental work on LTE plant

A 30 m^3/day LTE desalination unit using waste heat for producing pure water from seawater was installed for coupling to nuclear research reactor CIRUS. The unit operates at 41°C and 710 mm Hg vacuum. The LTE unit was operated at different temperatures of heating medium ranging from 50 to 65°C. Production rate of fresh water for different temperatures of heating medium are shown in Table 4.20

The calculated heat transfer coefficient (U_{cal}) for the heater and condenser section and the experimental values of heat transfer coefficient (U_{exp}) are also shown in the table. The product water is of high quality (10 µSiemens conductivity) with low silica content of 190 ppb. Following this experimental work, the LTE unit was connected to CIRUS reactor and will be operational when the reactor is refurbished.

TABLE 4.19: RESULTS OF COMBINED POST TREATMENT OF MSF DISTILLATE AND SWRO PERMEATE (Blending ratio 2.5:1, Temperature 40 °C)

No.	**TDS (ppm)**	**pH**	**Calcium (ppm $CaCO_3$)**	**Alkalinity (ppm $CaCO_3$)**	**Corrosion potential**	
					LSI	**RI**
1	172	8.22	86.0	57.6	0.37	7.48
2	154	7.85	80.0	51.0	-0.05	7.95
3	146	7.67	76.0	45.0	-0.32	8.37
4	173	7.53	70.0	55.0	-0.41	8.35
5	179	7.80	82.0	50.0	-0.11	8.02

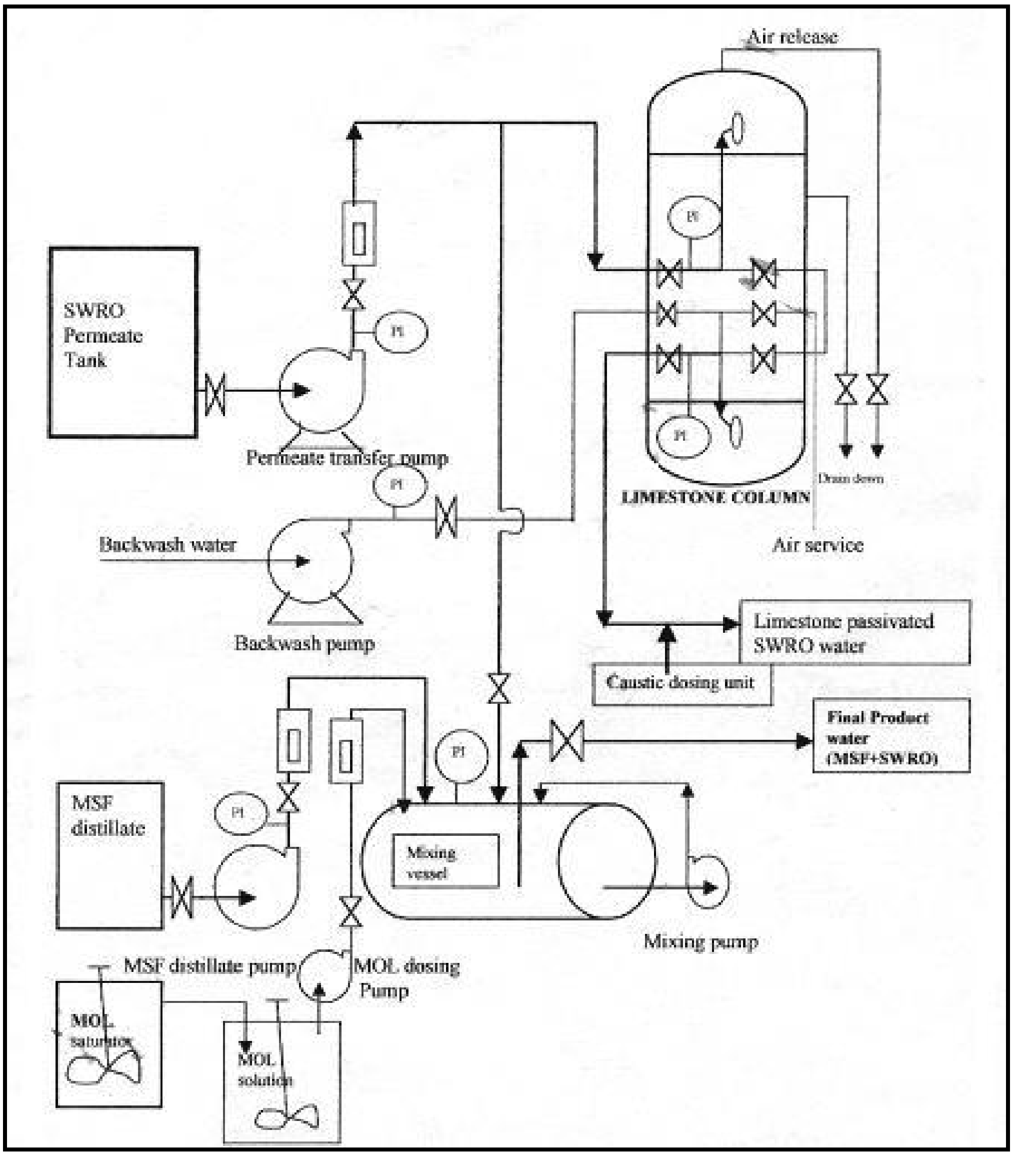


Figure 4.31: Experimental setup of pilot plant for studying passivation of desalted water.

TABLE 4.20: OPERATIONAL DATA OF BARC LTE UNIT

No	Production rate of desalted water, m^3/d	Temperature of Heating Medium, ^oC	Heat load kcal/hr	U_{cal} kcal/hr.m^2. ^oC	U_{exp} kcal/hr.m^2. ^oC
Heater section					
1	9.0	50	215300	475	460
2	15.8	55	378950	550	530
3	25.2	60	602900	670	645
4	31.7	62	757900	785	760
Condenser section					
				1800	1500

REFERENCES TO CHAPTER 4

[4.1] CHUN, K.R. and SEBA, "Heat Transfer to Evaporating Liquid Film", ASME J Heat Transfer, Vol. 9, pp. 391–386, (1971).

[4.2] BEDANARSKI, J. and PEPP, F. "Implementation of an Enhanced Multi-Effect Distillation Process for Seawater Desalination", International Desalination Technologies, Pasadena, CA91124, USA.

[4.3] SATONE, H., FUNAHASHI, S. and INOUE, S. "Experience on VTE Distillation Plants in Japan", Desalination, Vol. 22, pp. 169–179, (1997).

[4.4] MITSUMORI, T., IKEGAMI, Y., and UEHARA, H. "Comparison of OTEC Power Plant using Plate Type Heat Exchanger and One using Double Fluted Tube Type", Proceedings of the International Offshore and Polar Conference, Vol. 1, pp. 147–154, May 24–29, (1998).

[4.5] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the Economic Evaluation of Co-generation/Desalination Options, Computer Manual Series No. 12, IAEA, Vienna (1997).

[4.6] HUMPHRIES, J.R., MIDDELTON, E., "CANDESAL: a Canadian Nuclear Desalination System", paper presented at US/Middle East Joint Seminar on Desalination Technology", CI-9404, Washington D.C, 10–14 April 1994, Candesal Enterprises Ltd., Ottawa (1994).

[4.7] CANDESAL INC., "CANDESAL an Advanced Water Energy Production System: Egypt Application Study", Report CI-9403. Ottawa (1994).

[4.8] INTERNATIONAL ATOMIC ENERGY AGENCY, Potential for Nuclear Desalination as a Source of Low Cost Potable Water in North Africa, IAEA-TECDOC-917, Vienna (1996).

[4.9] INTERNATIONAL ATOMIC ENERGY AGENCY, Options Identification Programme for Demonstration of Nuclear Desalination, IAEA-TECDOC-898, Vienna (1996).

[4.10] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the Economic Evaluation of Cogeneration/Desalination Options: A User's Manual, Computer Manual Series No. 12, IAEA, Vienna (1997).

[4.11] ACADEMY OF SCIENTIFIC RESEARCH AND TECHNOLOGY, "Utilisation of Nuclear Technology in Seawater Desalination", Project No. 53-1992/97, Reports 1–4, Cairo, Egypt, (1995–1997).

[4.12] HUMPHRIES, J.R. et al, "A Technical and Economic Evaluation of the CANDESAL Approach in Indonesia Using Reverse Osmosis and Waste Heat from the CANDU-6 Nuclear Power Plant", Report, rev. 1, Candesal Enterprises Ltd., Ottawa (March 1998).

[4.13] FURUKAWA, D.H., "Desalination Processes and Technology", IAEA-TECDOC-898, IAEA, Vienna (1996) 82–93.

CHAPTER 5. CONCLUSIONS

5.1. MAIN CONCLUSIONS

In 1998, the IAEA initiated a Coordinated Research Project (CRP) on "Optimization of the Coupling of Nuclear Reactors and Desalination Systems" with participation of institutes from nine Member States in order to share relevant information, optimize the resources and integrate related research and development in this area. The overall scope of the CRP was to encompass research and development projects focused on optimized coupling of nuclear ractors and desalination systems in the areas: nuclear reactor design intended for coupling with desalination systems; optimization of thermal coupling of NSSS and desalination systems; performance improvement of desalination systems for coupling; and advanced desalination technologies for nuclear desalination. The CRP was completed by the end of 2003 and has in general achieved its objectives.

The overall conclusion of the CRP is that it has enabled the IAEA and participating institutes to accumulate relevant information on the latest research and development in the field of nuclear desalination and share it with interested Member States. The CRP has produced optimum coupling configurations of nuclear and desalination systems, evaluated their performance and identified technical features, which may require further assessment for detailed specifications of large-scale nuclear desalination plants. The main conclusions of the CRP are summarized below.

- All nuclear reactor types can provide the energy required by the various desalination processes. A total of nine nuclear reactors were examined for optimal coupling with desalination systems within this CRP. They are all of the water-cooled reactor type and are in various degrees of development. These reactors were categorized based on design status into: operational (PHWR-220 and CIRUS), in detailed design stage (CANDU-6 and KLT-40C), in basic design stage (CAREM-D, SMART and HR-200), and in conceptual design stage (NIKA and RUTA).

- The commercial seawater desalination processes, which are proven and reliable for large-scale production of desalted water are MSF and MED for distillation processes and RO for membrane processes. All these processes and hybrid technologies such as MSF-RO and MED-RO, as well as the Low Temperature Evaporator (LTE) which operates at temperatures close to the reject heat temperature of a typical power plant were considered for coupling within nuclear desalination systems.

- Different coupling options between the above nuclear and desalination technologies have been investigated within the CRP and were optimized with respect to safety, operational flexibility, reliability/availability and economics. Nuclear desalination has been demonstrated in India through the coupling of a 6300 m^3/day Hybrid MSF-RO desalination plant with an operating PHWR plant at Kalpakkam and a 30 m^3/day LTE desalination unit with CIRUS research reactor at Trombay.

5.2. SPECIFIC PROJECTS' CONCLUSIONS

The specific conclusions of each research project are presented country wise below.

5.2.1. Evaluation of nuclear desalination coupled systems, INVAP, Argentina

The main objective of this research project was the development of a flexible modeling tool, namely DESNU, for the evaluation of nuclear desalination systems from safety point of view. More specifically, computer models for the simulation of failure transients that describes the transport mechanisms of contamination.

DESUNU was developed to support the production of RETRAN input files for the modeling of nuclear desalination plant consisting of a small integral PWR coupled with MED, MSF or RO desalination plants, and the balance of plant. Use of DESNU does not require specialized knowledge. The input parameters are mainly operational and geometrical data easily available. The model produced has a coherent nodalization. However, the verification and approval of a model for a safety relevant simulation is still a task requiring experience and specialization in both modeling and safety.

A fruitful and successful collaboration took place between INVAP and other participating institutions within this CRP regarding the development of DESNU modeling tool. The review of DESNU was performed through submitting the updated versions other participants for external review. This practice of using DESNU tool for the production of models for different projects has proved to be adequate mechanism for improving DESNU as an open product for the nuclear desalination community. The user-friendliness of DESNU was improved by the preparation of a brief User's Manual with a summary of guidelines, rules and hints for using DESNU.

5.2.2. Optimized nuclear desalination/co-generation using reverse osmosis with feed water preheat, CANDESAL, Canada

The work carried out in Canada focused on the development and optimization of an innovative approach to the coupling of RO desalination systems to nuclear power plants and to the experimental demonstration of the key concepts underlying that coupling approach. A coupling scheme was developed that made use of waste heat from the reactor (discharged primarily through the condenser cooling system in water-cooled reactors) to preheat the RO system feedwater above ambient seawater temperature.

Analytical studies were carried out investigating the coupling of such an RO desalination system to the CANDU reactors. The anticipated benefits were observed in the studies. It was also found that with the Pressurized Heavy Water Reactors (PWHR~) the availability of additional waste heat in the form of moderator cooling can allow for a higher level of feedwater preheat, resulting in greater reductions in product water cost than for systems using waste heat from the condenser cooling system alone.

A demonstration program was carried out to experimentally confirm the increased water production rates at increasing system temperatures and pressures. Experimental results from the demonstration testing behaved as expected based on analytical performance models, validating the advanced design concept and confirming that the performance improvements indicated by the analyses can be achieved in operating systems.

Although initially developed in conjunction with the CANDU reactor, this coupling scheme is sufficiently flexible that it can be applied with other nuclear (and non-nuclear) energy sources. Hence the results of the Canadian work can be of benefit to other IAEA Member States who already have their own nuclear desalination programs underway or who are interested in instituting a nuclear desalination program.

5.2.3. Optimization of system of seawater desalination with 200 MW nuclear heating reactor, INET, China

The objective of this research project was to define an integrated nuclear desalination plant based on coupling the nuclear heating reactor, NHR-200, with a selected desalination process that is capable of safe, stable and economic production of potable water for one million people. The work carried out within the research contract included development and analysis of coupling NHR-200 heating reactor with high temperature MED, low temperature MED and hybrid (MED-VC and MED-RO). This was followed by estimation of GOR vs. investment cost for each coupling scheme, and optimization of water production cost.

The results indicated that the optimum coupling scheme with MED process for heat only mode is an NHR-200, steam generator and VTE-MED. Due to lack of detailed design data necessary for the selected VTE-MED process, an extensive theoretical and experimental program was carried out. This included development of the computer code NHRVM for thermal hydraulic analysis and construction of an experimental facility to compile the necessary database for design and operation on VTE-MED desalination plant.

Based on the results of the investigation, the integrated coupling of NHR-200 and VTE-MED with stacked tower design was recommended. Economic calculations based on input data established at the beginning of 1990s produced specific water costs in the range of 0.8-1.0 US$/m^3. However, recent calculations based on innovative technical design and domestic market prices in China indicated that the specific water cost could be reduced to 0.6 US$/m^3.

5.2.4. Investigation of feedwater preheating effect on RO performance, NPPA, Egypt

The objective of this research project was to study the performance characteristics of three commercial seawater RO membranes over the ranges of temperatures and pressures allowed by the manufactures. To this end, an experimental facility was designed by an Egyptian consultant and reviewed by an international expert. The facility consists of two identical units, one equipped with a heater to simulate different preheating conditions and the other operates at ambient conditions. Both units are fully instrumented to measure various parameters. An extensive experimental program was developed for a period of three years (one year for each commercial membrane) to produce data that can be used to validate the performance characteristics of RO feedwater preheating. A contract was signed with an Egyptian contractor to construct the experimental facility in June 2000, with a total construction period of 9 months. However, the construction was delayed due to a number of reasons and the facility, albeit in advanced stage, was not completed by the end of the CRP.

The construction of the facility is expected to be completed in the near future and will provide valuable data to the international desalination community on the long-term effect of feedwater preheating on RO performance. NPPA remains committed to making the results of the experimental program available to Member States.

5.2.5. Performance improvement of hybrid desalination systems for coupling to nuclear reactors, BARC, India

The objective of the research project was to study the behavior of Multi-Stage (MSF), Reverse Osmosis (RO) and Low Temperature Evaporation (LTE) desalination plants under different operating conditions and to utilize data for improving the performance of hybrid MSF-RO plant coupled to PHWR and LTE plant coupled to a nuclear research reactor.

To this end, the performance of SWRO using seawater feed at different temperatures was evaluated. Data collected for different membrane elements were analyzed and the operation data in the 100 m^3/d RO pilot plant was collected. The performance of MSF plant under different operating conditions of seawater temperature was studied through analysis of operational data of a 425 m^3/d MSF plant at BARC. The RO and MSF data were utilized for studying the coupling aspects of the hybrid MSF-RO nuclear desalination demonstration plant at Kalpakkam, which is in advanced stage of construction.

The performance of a 30 m^3/d LTE plant at different temperatures of hot water required for seawater evaporation was evaluated. Coupling of LTE plant set up at CIRUS has been completed.

5.2.6. Optimal coupling of SMART and desalination plant, KAERI, Republic of Korea

The objective of this research project was to establish a set of optimum coupling parameters between SMART and the desalination plant which satisfies the diverse requirements imposed by water production and electricity generation. The coupling scheme between SMART and distillation plant MED-TVC was selected and optimized. The coupling scheme was evaluated from the safety and economic points of view.

The transients of SMART induced by the desalination system were evaluated for the major potential disturbances that may affect the safety of SMART. KAERI identified three events as the major potential disturbances imposed by the desalination plant, namely, turbine trip due to desalination system disturbances, excess load due to increased steam flow to the desalination system, and loss of load due to the desalination system shut-down. The analysis of these events showed that the desalination plant disturbances would not have safety impacts on SMART reactor.

Simulation of contaminant migration was performed utilizing the modeling tool DESNU developed by INVAP, Argentina. Although this was a preliminary qualitative analysis, the results provide useful information regarding the time required for isolating the desalination plant from the distribution network.

The economic evaluation of the integrated SMART desalination plant was carried out using DEEP. The analysis result on water production costs are in the range of 0.74–0.84 US$/$m^3$ for the plant availability of 80% or higher with the discount rate of 8%. Since the SMART plant is designed for availability higher than 90%, it would be a competitive choice for nuclear desalination

5.2.7. Optimization of the coupling of nuclear reactors and desalination systems, CNESTEN, Morocco

The objective of this study was to evaluate the economics of coupling various nuclear reactors and desalination systems for the two candidate sites of Agadir and Laayoune. DEEP was used for the economic evaluation of three nuclear reactors and a combined cycle plant in the range of 600 MW(e) coupled to the desalination processes MED, MSF, RO (contiguous and stand alone), and Hybrid (MSF-RO and MED-RO).

For both sites, the costs of desalted water produced by nuclear and fossil energies are in the same range. Most economic configuration appears to be contiguous RO coupled with a PWR and an adequate amount of electricity to the grid. The installation of a 600 MW(e) PWR in Agadir could produce more than 300 000 m^3/day of desalted water at competitive cost and an average of 517 MW(e) to the grid. In the case of Laayoune, these figures are 72 00 m^3/day and 517 MW(e) respectively.

5.2.8. Russian small nuclear reactors as energy sources for nuclear desalination plants: coupling schemes and economic evaluations, IPPE, Russian Federation

The objectives of this project was to study the technical and economic effectiveness of using the Russian small-sized reactors, KLT-40C, NIKA and RUTA, and Russian HTME desalination plants, DOU GTPA, as constituents of nuclear desalination complexes, and the economic effectiveness of these nuclear desalination systems. Several coupling schemes have been proposed and investigated within this project including coupling the various Russian small reactors with HTME, RO and Hybrid MED-RO.

Detailed design characteristics and features as well as performance and economic data relevant to nuclear desalination systems of Russian small reactors and HTME distillation plants were presented. DEEP was modified and used for the economic evaluation of floating power unit based on KLT-40C reactor coupled with DOU GTPA, RO and Hybrid desalination plants. The results of the economic evaluation indicated that if the lowest water specific cost and maximum power to the grid were use as criteria, the optimum coupling scheme would be floating power unit and Hybrid desalination plant.

Preliminary safety assessment of the nuclear desalination complex was carried out utilizing RELAP5/MOD3. The input deck for RELAP5/MOD3 was produced using the DESNU tool and calculations for the steady state parameters were performed. Several components of the nuclear desalination complex system proposed by INVAP were modified according to the specific input data requirements of RELAP5/MOD3.

5.2.9. Optimization of coupling desalination systems with nuclear reactors, CNSTN, Tunisia

The work carried out within the framework of this CRP can be divided into:

- The economical evaluation of the energy production systems,
- The economical evaluation of the integrated systems for combined seawater desalination and electricity production.

The economical evaluation of the power production systems was performed with the help of the SEMER code, recently developed by the CEA. The power plants considered were:

- PWR 900,
- SCOR600,
- GT-MHR300,
- Super Steam Boiler, (SSB 600)
- Gas turbine combined cycle (CC 600).

Desalination costs were calculated using the IAEA DEEP code with SEMER results as input for three types of desalination processes:

- MED (Multiple Effect Distillation)
- RO (Reverse Osmosis)
- RO-ph (Reverse Osmosis with preheating of feed water)

Detailed analysis of the calculated results shows that nuclear option becomes more economic if current gas TEP prices increase by more than 15% (from 130 $/TEP to 150 $/TEP, Tunisian conditions). For example, for a discount rate of 8% and with the standard prices of the fossil fuels (25 $/bbl or 183 $/TEP), the desalination cost with the integrated system SCOR 600 - MED is respectively 25% and 33% lower than those with CC-600-MED and SSB-600-MED. The desalination with the nuclear power plant is also more economic with RO process: 21% and 29% of reduction in comparison with the desalination by CC-600 and SSB-600. The RO_{ph} process brings an additional reduction of about 19% in comparison with the cost of desalination by traditional RO.

ABBREVIATIONS

ATWS	Anticipated Transients Without Scram
BOP	Balance of Plant
BWR	Boiling Water Reactor
CB	Control Building
CCS	Component Cooling System
CEDM	Control Element Drive Mechanisms
COPS	Containment Overpressure Protection System
CRP	Coordinated Research Project
DBA	Design Base Accident
ECCS	Emergency Core Cooling System
ERHR	Emergency Residual Heat Removal system
FA	Fuel Assemblies
FPU	Floating Power Unit
GCR	Gas Cooled Reactor
GOR	Gain Output Ratio
HTME	Horizontal Tube Multiple Effect distillation
HVAC	Heat, Ventilation, and Air Conditioning system
LOCA	Loss Of Coolant Accident
LTE	Low Temperature vacuum Evaporation
LT-HTME	Low Teperature-Horizontal Tube Multiple Effect distillation
LT-HTME	Low Temperature- Horizontal Tube Multiple Effect distillation
MAPS	Madras Atomic Power Station
MCP	Main Cooling Pump
MED	Multiple Effect Distillation
MSF	Multi-Stage Flash distillation
MSLB	Main Steam Line Break
NDCS	Nuclear Desalination Coupled System
NHR	Nuclear Heating Reactor
NPD	Nuclear Power Demonstration reactor
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
OIP	Options Identification Programme
PCC	Pre-stressed Cement Concrete
PCW	Primary Cooling Water
PHE	Primary Heat Exchanger
PHT	Primary Heat Transport system
PHWR	Pressurized Heavy Water Reactor
PRHRS	Passive Residual Heat Removal System
PSA	Probabilistic Safety Analysis

PWR	Pressurized Water Reactor
RAB	Reactor Auxiliary Building
RCC	Reinforced Cement Concrete
RD	Reactor Building
RO	Reverse Osmosis desalination
ROPS	Reactor Overpressure Protection System
RPS	Reactor Protection System
RPV	Reactor Vressure Vessel
RRS	Reactor Regulating System
RSS	Reactor Shutdown System
SAB	Station Auxiliary Building
SAFDL	Specified Acceptable Fuel Design Limit
SB	Service Building
SBLOCA	Small Break Loss Of Coolant Accident
SG	Steam Generators
SMART	System-integrated Modular Advanced ReacTor
SWRO	Sea Water Reverse Osmosis
TBT	Top Brine Temperature
TDS	Total Dissolved Solids
TDV	Time-Dependent Volume
TVC	Thermal Vapor Compression
UF	Ultra-Filtration
VC	Vapor Compression
VTE	Vertical Tube Evaporator

ANNEXES

Summaries of Projects within the Coordinated Research Project on Optimization of the Coupling of Nuclear Reactors and Desalination Systems

ANNEX I

EVALUATION OF NUCLEAR DESALINATION COUPLED SYSTEMS

N.A. Masriera, A.S. Doval
Nuclear Engineering Department, INVAP S.E., Argentina

I.1. OBJECTIVE

The global objective of INVAP's contribution to this CRP is the development of a flexible modelling tool, namely DESNU, helpful for the evaluation of Nuclear Desalination systems, from the safety point of view. More specifically, computer models for the simulation of failure transients should clarify the description of the transport mechanisms of contamination, and therefore would be useful for assessing the consequences of accidents, and for the definition of requirements on coupling related systems.

I.2. SCOPE

The Spreadsheet DESNU is a tool for the development of simulation models (input files) of the Nuclear and Desalination systems. The scope of the models is in accordance with the goal of assessing contamination transport, i.e. a thermal-hydraulic description of the safety relevant process loops. The computer code used for the simulations is RETRAN-02. By collaboration with IPPE as a co-participant of this CRP, the extension to produce also RELAP-5 input files is foreseen. The software used for the spreadsheet, as modelling tool for the preparation of the input files, is EXCEL. The provision of software itself is beyond the scope of this CRP.

I.3. INTRODUCTION

In order to evaluate different coupling systems from the safety point of view, information about their performance during accidental transients should be obtained. The complete safety analysis of incidental events should include several steps or levels. In a simplified description:

- Possible accidental events should be determined and their consequences assessed (deterministic) in terms of harm to the public (collective dose).
- For every accidental sequence the chance of occurrence should be calculated (probabilistic) out of the failure probability of components and equipment.
- A complete "map" of accidents (collective dose) vs. Probabilities of occurrence should be elaborated, and their acceptability according to regulations and guidelines should be verified.

The goal of the work is the elaboration of a modelling tool useful for the deterministic safety analysis of Nuclear Desalination coupled systems.

The calculating tools most commonly accepted for Nuclear Safety Analysis are RETRAN and RELAP computer codes. In order to keep compatibility, these same tools should be applied for modelling a Generic Desalination Plant, coupled with a Generic PWR. The model would be used for simulating accidents with loss of isolation barriers between the NPP and the product water.

It is widely known that the codes with stress on reliability of results are, in terms of the input file information, very inelastic, and the elaboration of each model generally implies an important amount of systematic-routine work by a thermal hydraulics specialist. Our experience shows that when **a set of models** are involved (several transients of the same system, for example), it is extremely convenient the investment of some preliminary work on a **modelling tool.** It would work as a support for the elaboration of input files, reducing input requirements on geometry and initial conditions.

For the scope of this CRP, INVAP´s contribution is to provide RETRAN models of generic Nuclear Desalination Systems, produced by an EXCEL spreadsheet that allows adapting the model for different small Nuclear Power Plants (NPP) and Desalination Plants (DP).

The final output initially foreseen was a set of simulation models, useful to determine the transport mechanisms of contamination, and therefore for the definition of requirements on coupling related issues: Isolation Systems, Intermediate Loops, Safety Triggering sequences, etc. Nevertheless, the latter are outputs from a model of a specific Plant, elaborated by the End-user. It must be pointed out that in terms of the output for this CRP, the spreadsheet is itself a product valuable for the technical groups working on the evaluation of Nuclear Desalination Systems.

The work plan presented in 1998 specifically took as a relevant issue the collaboration with other institutions participating of this CRP, that are in position to provide a complement and a feedback for improving INVAP's contribution. This collaboration effectively took place, and provided a feedback that allowed tuning the goals of the contribution to better cater the need of the Nuclear Desalination Community.

Within the frame of our support to IAEA's activities on Nuclear Desalination a parallel development was carried out concerning the safety aspects of nuclear desalination in connection not only with deterministic analysis but with the contents of the Safety Analysis Report, as well.

The progress in INVAP's work for this CRP will be presented chronologically, including the adjustment of the original Work-plan.

I.4. ORIGINAL WORK PLAN

The participation in this CRP has INVAP S.E. as the contracting institution, by the Nuclear Engineering Department, Thermal-hydraulics Group. The manpower committed for the work is composed of 200 ManHour/year of a senior specialist plus 400 ManHour/year of a junior specialist on Thermal-hydraulics, modelling and computer simulation. However, the dedication has been kept increased up to two senior specialists with a third senior specialist reviewing, according to the support decided by INVAP for IAEA Nuclear Desalination activities.

The original work plan, as presented in November 1998 (reference I.1), included the development of three levels of Plant Models (described in the following paragraph), transient modelling and assessment, the analysis with different codes and/or different desalination processes, optimized transients and the final report. The Plant Modes are:

- Model 0: Is a generic model describing the process; it is used mainly for the elaboration of the spreadsheet, and for verification of consistency and reasonability of steady states and trends in transients (it gives no quantitative results).
- Model 1: Is a model describing the coupling of a specific NPP and a DP (either built or with “frozen” design); results should allow a quantitative analysis.
- Models 2: Set of models describing how to work on alternatives for the designs of Model 1 (requirements, countermeasures, etc.).

For the successful implementation of Model 1, specific data from a DP should be provided by other participant/s of this CRP. Models of level 2 are actually part of DESNU utilisation by End-Users, and therefore will be presented as an “example” in this work.

I.5. PROGRESS UP TO MAY 1999

I.5.1. Desalination plant model

The goal of this stage was to produce a set of RETRAN models with a variable number of effects keeping the consistency in the description of the process, while producing an input file available for upgrading to “Model 1” once real data were received. It should be possible to make this upgrading changing geometrical and physical parameters, without re-defining number and interconnection of control volumes, junctions, conductors and components.

Work progressed with certain delay for increasing the number of effects. It was found difficult to produce this set of models, because the RETRAN self-initialisation algorithm works solving first a mass-balance, then momentum and finally energy equations. In order to overcome this problem, work proceeded with the calculation of steady states with a different computer code (CHEMCAD, see reference I.7), and then using the operating parameters for RETRAN models.

As for the DP modelling itself, at this stage, models included the thermal power from the NPP as a fixed power inlet by a non-conductive heat exchanger.

I.5.2. Nuclear power plant model

In this stage, the effort was focussed in developing a set of NPP models (RETRAN input files) that would simulate the coupling using steam from different turbine vapour extractions. Here again, RETRAN self-initialisation option does not seem to work adequately for a flexible approach on several turbine vapour extractions lines working in parallel. The same alternative of using CHEMCAD code (reference I.7) was investigated. The model developed includes the following features:

- An Advanced Type, Integrated Primary Coolant Loop and Once Through Steam Generators, for the Generic PWR.
- A Standard Thermal-Cycle (Turbo-group) for a Small Size NPP, with a single turbine and several extractions.
- Different exchange surfaces for condensing and sub-cooling for the final heat sink (seawater).

I.5.3. DESNU spreadsheet

Our experience shows that for tasks involving a set of models (several transients of the same system or several similar alternative systems), it is extremely convenient to work on a modelling tool as support for the elaboration of RETRAN input. This allows reducing input requirements on geometry, measuring units handling and initial conditions, rendering less mistakes. This modelling tool may be a spreadsheet with separate sheets for each component or sub-system, linked with each other, and having as an output a sheet resembling a RETRAN input file.

The level of detail of the spreadsheet elaboration has to be defined to optimize the amount of work. For example, the spreadsheet could be developed for producing any of the following calculations (ordered by priority according experience):

1. Units conversion (RETRAN input uses British measuring units)
2. Geometrical data (unification and avoidance of redundancies): control volumes parameters, heat conductors parameters, interconnections in surfaces, elevations, junctions, etc.
3. Control system parameters, and pseudo-control algorithms.
4. Initial conditions: mass-flow distribution, heat fluxes.
5. Initial state variables: enthalpy, pressure.
6. Interconnections among control volumes, junctions, heat conductors, components.
7. General modelling parameters (for example correlation choosing)

In our experience, the spreadsheet produces the calculations including item 4 of the list while the initial conditions calculation for enthalpy and pressure imply state functions algorithms that should be included in the spreadsheet, and for models that share nearly the same initial steady state it was considered unnecessary. Similarly, items 6 and 7 were considered beyond the optimal point as long as the models to evaluate keep qualitatively constant.

I.5.4. Collaboration plan

INVAP, clearly a nuclear vendor with certain experience in different plant process design, allows us to conceive a NDP by coupling a NP with an RO DP. Nevertheless, there was a broad consensus regarding thermal desalination process as a more interesting issue for the study in this CRP, and INVAP agreed to focus initially on them. We were sure that the final output could be enhanced by external collaboration on issues of thermal desalination like the following ones:

- Data of desalination process and engineering parameters needed for modelling.
- Verification of DP models in terms of steady states and dynamic trends.
- Review of the overall approach of our work in the CRP (feedback for tuning the goals).
- Discussion of correctness / reasonability of simulated transients.

Additionally, and according to the standard practice as a Technical Group within INVAP, we expected external review of partial reports, results and conclusions.

I.6. PROGRESS UP TO FEBRUARY 2000

I.6.1. Desalination plant model

The simulations of previous stage allowed us to identify the variables and phenomena relevant for the adequate simulation of the coupling when performing the NDP assessment on contamination transport. This study was presented by INVAP in the Consultancy Meeting of reference I.3, and the conclusions obtained after modelling with RETRAN a MED device with few effects are summarised in the following paragraphs.

- RETRAN code is able to simulate properly the processes of evaporation and condensation in complicated schemes, reproducing steady states and soft transients.
- If these same models are run to simulate accident conditions (implying massive irruption of water and sudden pressurisation of the distillation chambers) the code fails.
- Transit time of fresh water during normal operation is large, as it is related with liquid convection through flash chambers from one stage to the following.
- There is a delicate equilibrium during the operation of a real MED Plant that may be broken by even small amounts of water entering the system resulting in an automatic shutdown.
- The simulation of steady states and soft transients are not specifically relevant for a safety analysis, but provided the insight of the fluid paths in case of a severe accident.
- Once these paths are defined, it is possible to develop a thermal-hydraulic model of connected chambers to simulate the “contamination wave” moving from the secondary side of the NPP through the DP.
- Evaporator tube connections to their tube plate are generally not designed to withstand important pressure differences, so the most likely “path” during a severe transient seems to be straight from one effect to the following through the empty tube plates.

Therefore it was decided to proceed modelling the MED Plant distillation effects by a series of vapour chambers (close to saturation) connected to each other from the beginning of the simulation. In the initial steady state the steam flows from the first to the last effect, not describing the MED chambers steady state operation, but defining a “steam residence time per effect” that is relevant to the convective phenomena over which the contamination is spread through the Plant. So the steady state steam velocity should be “tuned” to a conservative estimation of the steam velocity inside the hottest effect. As mass flow in this model is not a really meaningful parameter it should not be compared with the total water flow of a MED system. Boundary conditions were simplified coherently:

- The first effect was modelled as an evaporator of the Feed-Water and it was considered as a flow and enthalpy boundary condition.
- The last effect is modelled as a pressure boundary condition towards which the process steam flows. This mass-flow (turned to water) is the inlet of a Product Water Tank and carries the contamination of the steam.

In a DP isolated model this first effect evaporator is modelled by a tube and vessel heat exchanger. The tube side has steam coming from a steam source and going to a steam sink (as pressure and enthalpy Boundary Conditions). The failure of the main heat exchanger would produce a by-pass connecting the turbine steam (potentially contaminated) with the desalination process steam.

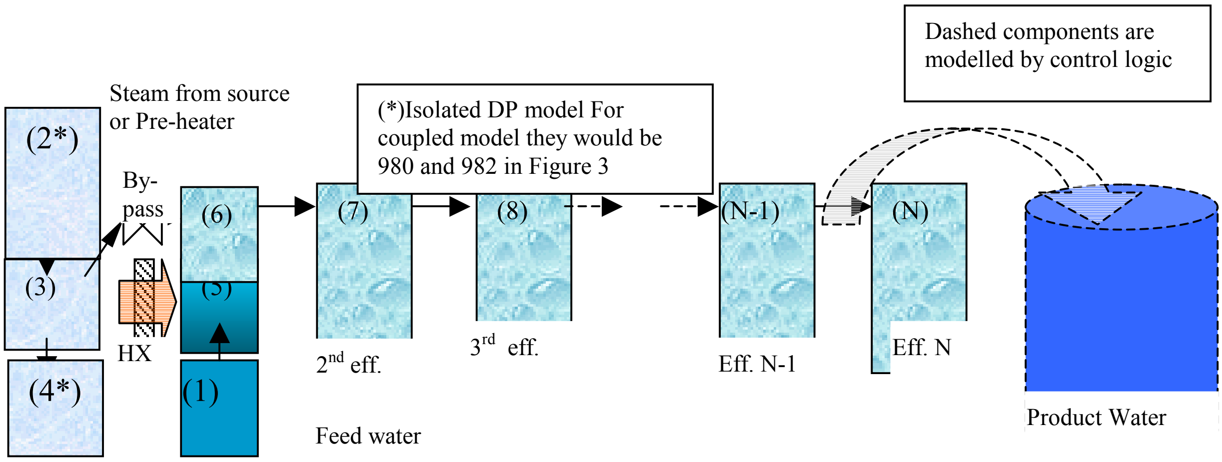


Figure I.1: Scheme of the control volumes of MED Desalination Plant model.

As the input data for the DP model are kept as summarised as possible, the initial conditions are specified only in the boundary volumes, first and last effects. Figure 1 is a simplified scheme of the control volumes of MED DP model. Junction codification keeps the rule of using the same number as the "from" volume, except junction 4, which is a valve allowing the by-pass of main heat exchanger while numbers in brackets correspond to Control Volumes (CV) identification.

I.6.2. Nuclear power plant model

In this stage, it was decided to proceed using a secondary loop model with a degree of complexity similar to the one developed at INVAP for CAREM NPP. As for the primary loop, it was considered mainly as a Boundary Condition and therefore modelled by few elements. Nevertheless, the Steam Generator has the degree of detail needed to consider the heat transfer phenomena occurring with a sub-cooled feeding, a two-phase region, and a superheated outlet. A minimum of 20 CV for the secondary side, and 4 in the primary side was evaluated to be correct. The condenser, safety valves, relief valves and final heat sink are modelled in the "standard" way for engineering models.

There is an important issue related with RETRAN limitations, i.e. the contamination transport modelling is only possible **within** one fluid phase. The user may choose if it is steam or liquid, but always the other phase remains modelled as free of contamination. The changes in phase in the Steam Generator and in the Condenser define two separated regions in the BoP of the NPP.

According to RETRAN limitations, it is possible to describe the irruption of contamination in the Steam Generator, and its transport through the piping towards the turbine, and to the turbine extractions. But it is not possible to represent the transport to the Condensate and to the Feed Water Systems (FWS). Therefore, the modelling of the Condensate System and FWS does not contribute directly to the simulation at this stage of development of the safety analysis.

This could lead to simplify the NPP modelling replacing the Condensate and Feed-water Systems by adequate flow, enthalpy and pressure boundary conditions. Nevertheless it was decided to keep the general scope of the BoP model for two reasons:

- It had been suggested as a future task in this CRP the extension of DESNU spreadsheet for the production of inputs for other codes. This may overcame RETRANs limitations.
- It is always possible to implement, by pseudo control logic cards, an algorithm calculating the contamination transport phenomena in the liquid phase.

Preliminary simulations show that the condensate system and FWS would turn relevant only if contamination could complete at least one loop returning to the Steam Generator after the safety transient beginning (coupling HX rupture). This would occur in transients lasting more than the transit time through the Feed-water Tank residence time (≈ 1000 seconds).

The modelling of turbine extractions deserves some further description. The pre-heaters are “tube and shell” heat exchangers, with the FW through the tube side, and the steam coming from extractions condensing on the shell side. The thermal-hydraulic modelling of the behaviour of these equipment, working at quite a constant pressure, is possible with a detailed nodalization. But the purpose of including these components in the NPP model is only to take into account the energy balance in a first order approximation, so an internal modelling is out of scope.

Giving credit to the pressure control at the steam side of the pre-heaters solves this issue. The pre-heaters are represented as pressure boundary conditions (TDVs 980 to 983 in RETRAN model) for the hydraulic modelling of the turbine extractions. The energy balance at each pre-heater is then represented by a non-conductive heat exchanger with the power defined as a function of correspondent turbine extraction.

The mass balance of the secondary loop is kept by adequate flow boundary conditions, i.e. by “FILL” elements injecting in the FW tank, with flows defined by the extractions flows. The scheme of the control volumes of the NPP model is presented in Figures 2 and 3.

I.6.3. Coupling system

The user does not modify the CV interconnection logic through the spreadsheet. Therefore, when developing a “Model 0” of a Desalination System coupled with a NPP, a concrete coupling scheme had to be selected.

The steam conditions (enthalpy, pressure and flow) of the coupling HX depend on the coupling scheme chosen, so it is not possible to keep open to the user the definition of the steam conditions at the coupling HX.

From the coupling schemes evaluated the use of turbine exhaust was disregarded, and the use of the outlet of the High Pressure Pre-heater #2 (10.55 bar and 179.3 C) was chosen for the following reasons:

- The convective transport through the steam lines and turbine is extremely fast. Extraction #4 goes through the shell-side of Pre-heater #2, and placing the coupling HX after it gives a convenient margin for isolating it, in case contamination is detected at the outlet of the Steam Generators.

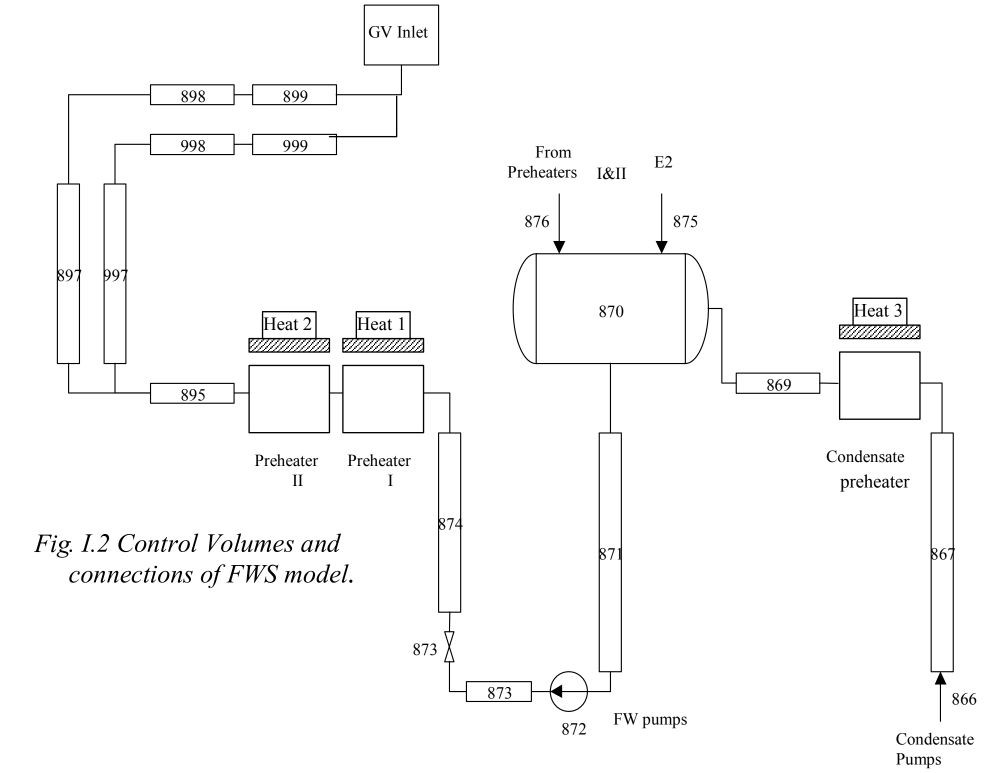


Fig. I.2 Control Volumes and connections of FWS model.

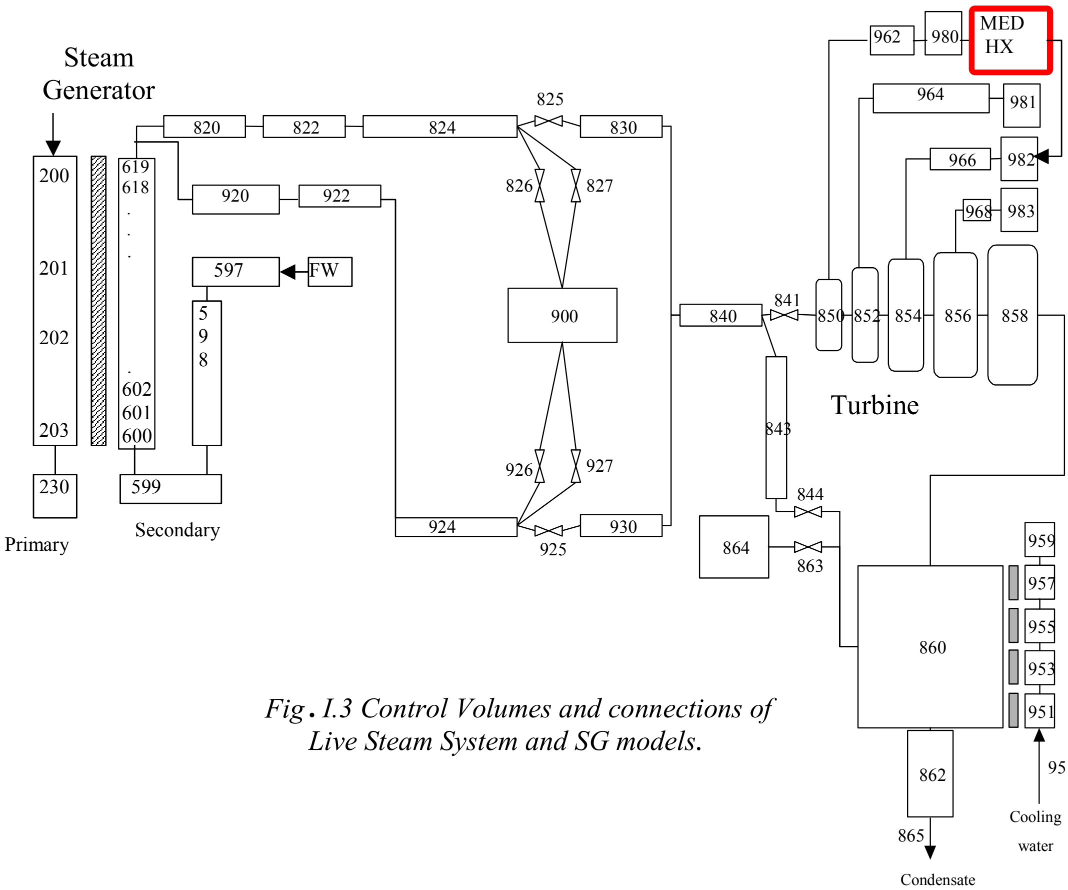


Fig. I.3 Control Volumes and connections of Live Steam System and SG models.

- This coupling scheme has a smaller impact on the overall Plant design. If the turbine extraction #3 is used then Pre-heater #2 has to be completely redefined.
- The turbine extractions change strongly with the NPP power. This scheme gives the Desalination plant a greater independence of operation.

In the NPP schemes (Figures 2 and 3) the coupling is represented by the element "MED HX" connected to the volumes 980 and 982, which in terms of mass transport represent the outlet of Pre-heater #2 and the FW tank (outlet of Pre-heater #1), respectively.

The model of the integrated Plant requires some "tuning" work in order to produce an optimized simulation of both the steady state and the safety transient. The adjusting procedure suggested is:

1. The area of junctions connecting to and from the volume of the coupling HX should be adjusted to produce a mass flow equal to the flow of extraction #4.
2. The junctions area between effects should be adjusted to produce the desired steam velocity.
3. The heat exchange surface of the coupling HX should be adjusted to reproduce the enthalpy value in the first effect.
4. The steps 1 to 3 may require iteration for slight corrections.
5. The generated power (related with the turbine shaft) should be corrected in order to take into account that the power transferred by coupling changes the energy balance.

I.6.4. DESNU Spreadsheet

The elaboration of DESNU spreadsheet was completed in this stage to a level producing the first useful models. The spreadsheet has three chapters and an appendix.

- Chapter 1, "General Data", is organised in separate sheets for components, where the user defines the input data of each component of the model.
- Chapter 2, "Nodalization", is organised in separate sheets for input modules (volumes, junctions and conductors), and a special sheet for the nodalization of the Steam Generator.
- Chapter 3, "RETRAN input files", is organised in separate sheets providing three RETRAN input files.
- Appendix, it includes sheets with auxiliary data for unit conversion and physical properties.

I.6.5. Preliminary transients

Some simulations were performed with the models developed till this stage, in order to verify the RETRAN model obtained from the spreadsheet, in terms of issues as interconnection logic of control volumes, junctions and heat conductors, numerical methods and completeness of RETRAN input file. This allowed detecting and correcting parameters, for example, elevation inconsistencies. The reference transient consisted in a first period, meant to assure the steady state, and then:

- A contamination source is triggered in the Steam Generation outlet (400 s).
- Contamination spreads on the whole secondary system.
- The coupling heat exchanger is by-passed, simulating a rupture (600 s)
- Contamination migrates on the Desalination Plant.

The system achieves the steady state, and the steam generation of the NPP is not sensitive to the coupling HX failure. In terms of contamination, it reaches immediately the turbine (diluted to a half by the flow of the other steam line), and then contaminates gradually the coupling HX. When this HX fails contamination migrates gradually through the effects of the Desalination Plant (see Figure 4).

I.7. PROGRESS UP TO AUGUST 2001

During this stage the work-plan was reviewed and adjusted according to the actual practice of collaboration among CRP participants, with the following guidelines:

- The advance on the modelling and analysis of specific transients is justified by the use of specific "real" data, coming from a NDP either built or with "frozen" design. The Nuclear Desalination scenario still has very few projects complying with this condition, they have relevant differences, and in some cases may not be willing to open design data. Therefore the simulation and analysis of specific transients may not produce the results most useful for the Nuclear Desalination community.
- There has been a wide agreement on shifting the effort from the modelling and analysis of specific transients to the extension of the modelling capabilities on different desalination technologies and different simulating codes (see reference I.5). Therefore the tasks "Transients modelling", "Transients Analysis" and "Optimized transients" will not be developed further. In turn, the task "Analysis with different codes and different Desalination Process" has been expanded in scope. Table 1 shows the scheme of the new work plan.

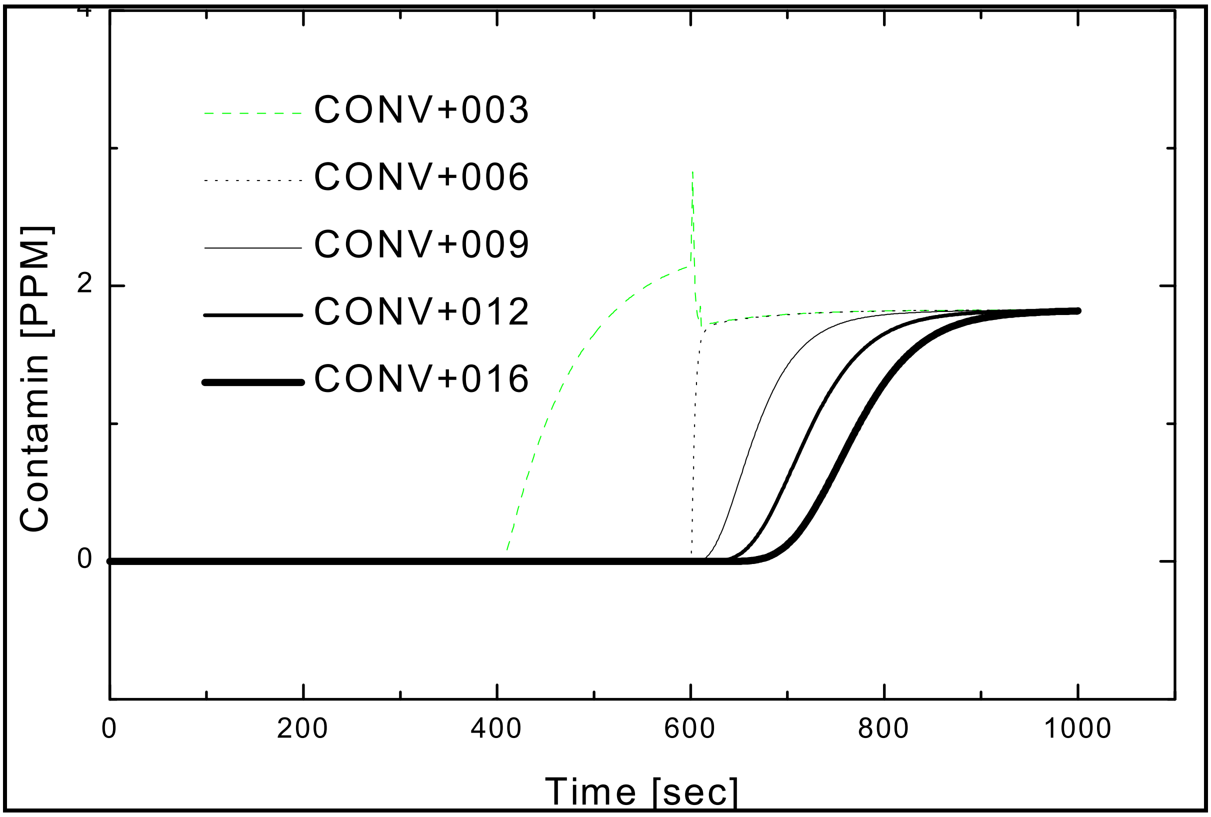


Figure I.4: Contamination at the HX, 1st , 4th , 7th and 11th effects.

TABLE I.1: NEW SCHEME OF THE WORK-PLAN

Year	**2000**		**2001**		**2002**		**2003**	
Task	1st ½	2nd ½	1st ½	2nd ½	1st ½	2nd ½	1st ½	2nd ½
Extension of DESNU including MSF/RO	**X**	**X**	**X**	**X**	**X**			
Data collection	**X**							
Models elaboration	**X**	**X**	**X**	**X**				
Spreadsheet elaboration		**X**	**X**	**X**	**X**			
External review	**X**	**X**			**X**	**X**		
DESNU modification according review			**X**			**X**	**X**	
Turbine steam extraction evaluation	**X**	**X**	**X**					
Extension to RELAP (&)	**X**	**X**	**X**			**X**	**X**	
Review (&)				**X**		**X**	**X**	
Case study		**X**	**X**					
Final report						**X**	**X**	**X**

(&) Depending on tasks by IPPE, Russian Federation

I.7.1. Desalination plant model

The starting point on DP modelling for this stage was a RETRAN model of a MED simplified to an optimum regarding the relevance on the convective phenomena over which the contamination is spread through the Plant. The progress for this period was mainly the elaboration of a RETRAN model of a MSF DP (complete). Additionally, a model of a RO Plant was begun.

Essential MED-MSF differences regarding modelling:

- In MED systems, evaporator tubes connections to their tube plate are generally not designed to withstand important pressure differences, so the most likely "path" during a severe transient seems to be straight from one effect to the following through the empty tube plates.
- In MSF systems, the heat exchangers are generally continuos pipes, and therefore the design excludes the possibility of a break leading to a steam path from one stage to the following. There seems to be no possible sudden event that connects vapour chambers with each other.

Therefore it was decided to proceed modelling MSF stages by liquid chambers connected to each other from the beginning of the simulation. These chambers have a different fluid condition specification (initialisation parameters) and heat transfer model. As another difference, the configuration of chambers may include a brine recycle connection.

Feature in common MED-MSF: Water flow does not describe MSF chambers steady state operation, but a "residence time per stage" relevant to convective phenomena by which the contamination is spread.

Main features in the General Transport Model of the MSF: RETRAN limitation

As already described, the contamination transport modelling with RETRAN is only possible **within** one phase, and the other phase remains modelled as free of contamination.

For MSF systems, the contamination transport in liquid is the one to be considered. This introduces two more differences:

- It is no longer possible to make a straightforward model of the contamination starting in the Steam Generator, and its transport through the live steam piping, turbine, turbine extractions, and then to the Desalination System through the coupling HX.
- In return, it is possible to model a direct transport from the DP to the product water tank.

It would be possible to simulate contamination bypassing the main HX, using analytical algorithms of the phenomena built in the "control cards" structure. But in our criteria this would go somehow against the good practice of keeping safety modelling as simple as possible. Therefore it was decided not to produce a NPP-MSF coupled model, but separated models that can be run in a two step procedure.

General features of the MSF model:

- MSF DP is only simulated in an isolated model (connected to a steam source and a sink as pressure and enthalpy boundary conditions).
- The HX failure is modelled by a direct irruption of contamination in the first stage.
- The contamination through the BoP of the NPP may be modelled separately and included in the analysis.
- The MSF Plant stages are divided in two "sections" separated by a connecting piping, representing an approximation to the configuration of a MSF Plant with brine recycle.
- Introducing negligible dimensions to the connecting piping, the model reduces to a MSF Plant without brine recycle.
- The first stage is actually an evaporator, but was simplified by keeping the water as single phase and using a heat transfer of the same order of magnitude of the real heat exchanger.
- The feed-water is a flow and enthalpy Boundary Condition, with an artificially low enthalpy in order to assure single-phase conditions.
- The last stage is connected with a Product Water Tank by a simple pipe.
- DESNU spreadsheet may produce models with different number of stages, keeping the possibility of considering two sections equal in number of stages (N must be even).

For the MSF steady state model, defining a "residence time per stage" that is relevant to the convective phenomena over which the contamination is spread through the Plant. So the steady state water should be "tuned" to a conservative estimation of the velocity inside the hottest stage. As the mass flow in this model is not a really meaningful parameter, it should not be compared with the total water flow of a MSF system. The mass balance (inlet and outlet) is simplified in coherence with previous approximation.

In a similar way as for MED model, the spreadsheet was developed for producing models of MSF plants with different number of stages and initial conditions are specified only in the boundary volumes, first and last effects. Figure 5 is a simplified scheme of the control volumes of MSF Desalination Plant model.

I.7.2. Nuclear power plant model

The development of NPP Models was completed according to the Work plan during previous stages of the project. The modelling of the component producing the thermal coupling was dropped as a goal, while focused in the contamination transport, according to the shift of effort decided in the meeting of reference (I.5).

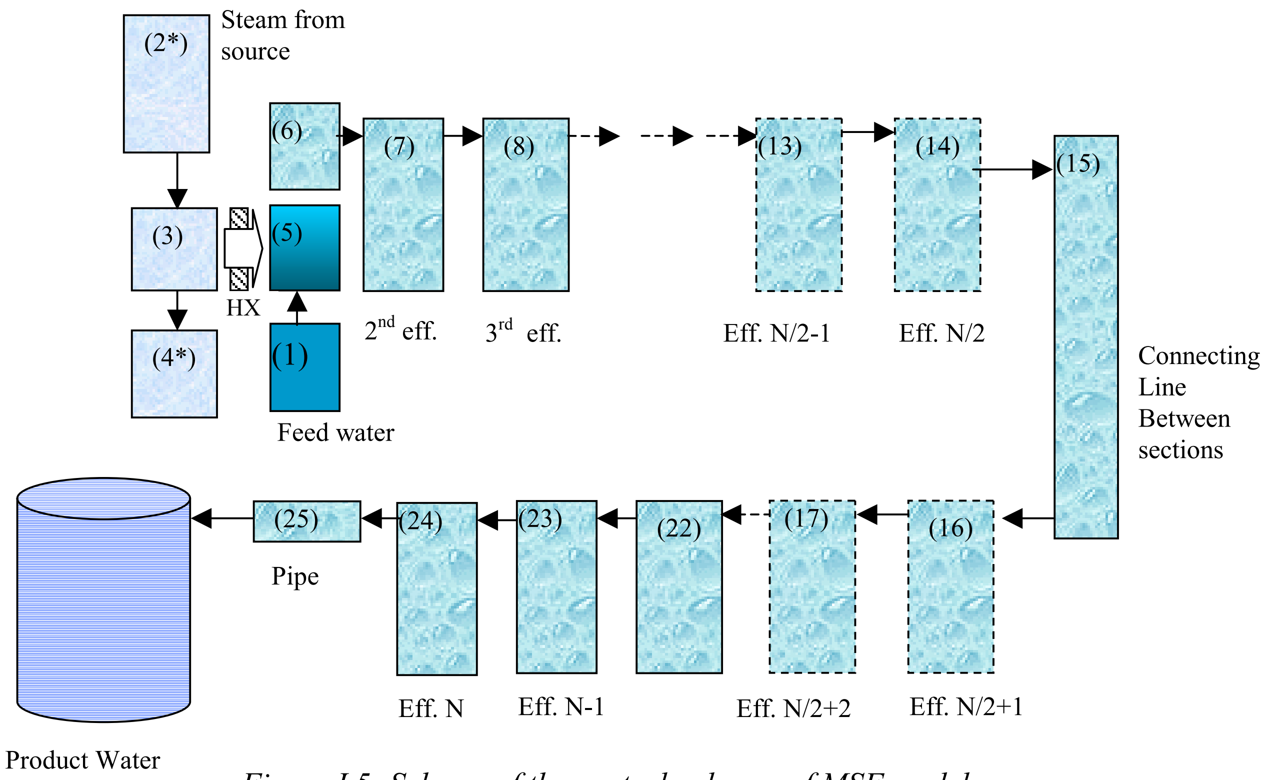


Figure I.5: Scheme of the control volumes of MSF model.

I.7.3. Coupling system

As previously described, the user does not modify through the spreadsheet, the Control volumes interconnection logic. For the direct modelling of a Desalination System coupled with the NPP, a concrete coupling scheme has to be selected. Even more, the steam conditions (enthalpy, pressure and flow) of the coupling Heat Exchanger depend on the coupling scheme chosen, so it is not possible to keep open to the user the definition of the steam conditions at the coupling Heat Exchanger.

For the NPP-MED coupling, a particular scheme was chosen (the outlet of the High Pressure Pre-heater #2) and modelled, but this was not repeated for different coupling schemes and different desalination technologies. In order to cope with acceptable flexibility with models of MED, MSF and RO desalination systems, an alternative approach was defined for the coupling treatment, using isolated models of the BoP of the NPP and the DP, by a two-step procedure:

(1) The contamination from the steam generator through the coupling points is simulated, and the “Contamination vs. Time” tables are obtained for all the points of possible thermal coupling.
(2) The “Contamination vs. Time” table corresponding to the case is taken and used for elaborating the Boundary Condition for the contamination source in the isolated DP model. Then the contamination from the coupling HX to the product-water tank is simulated.

The implementation of this approach for RETRAN modelling of the NPP, MED and MSF has been completed. For the modelling of RO, and all the cases with RELAP, it is foreseen that the task will be completed during the following stages.

I.7.4. DESNU spreadsheet

The advances on the modelling tool in this stage were mainly in two lines.

- The extension in order to produce a RELAP model of the original NPP BOP (see reference I.6). This task should be presented in the report from the IPPE, Russian Federation.
- The increase in the possible number of effects/stages (Control Volumes), according to the conclusions of the meeting of reference (I.5). The number of effects/stages should be increased up to the theoretically possible value of 50 and they could be defined out of the DT value provided by the USER preserving the limiting value of DT=2.3 C. These values are valid for both MED and MSF.

The implementation of these features in DP models was completed for RETRAN MED and MSF models. A separate spreadsheet improvement, from the conclusions of the meeting of reference (I.5), was the inclusion of a scheme of the BoP presenting to the USER a first idea of the nodalization.

I.8. PROGRESS UP TO DECEMBER 2002

During this period the spreadsheet (DESNU), has been upgraded to include an RO Desalination System, completing the goals of the work-plan. This version of DESNU covers a small PWR BoP, a MED system, a MSF system, an RO system, and a PWR-MED coupled plant, (all of these as input files for RETRAN code).

With this, INVAP's work on this CRP has achieved most of the goals of the original Work Plan, and additionally the new goals that were defined in RCM's and CM's, according to specific needs and the possibilities of collaboration of participant institutions.

Besides the standard practice of internal review and approval already performed, the verification and review of the spreadsheet by other participants will continue.

As a result of the work performed in the previous stages, and based on one of the important issues of the ND Coupling, i.e. the safety aspects, we considered that pointing out those generic features that could be included in the Safety Analysis Report of the co-generation plant would be a fruitful contribution. Details of this analysis are given in reference (I.8).

I.8.1. Desalination plant model

The starting point on DP modelling for this period was RETRAN models of both MED and MSF processes simplified to an optimum.

For the RO DP simulation, it is necessary to focus on the "residence time" on every component relevant to the convective phenomena over which the contamination is spread through the Plant. Therefore, the piping related with the energy recovery part of the system is not modelled as it is in the rejection lines (not in the potential contamination path). As compared to MED and MSF processes, RO is simpler in terms that all the relevant components are single-phase liquid.

I.8.1.1. General features of the RO model:

- The RO DP is only simulated in an isolated model (connected to a pre-heated line coming from the coupling with the NPP as a pressure and enthalpy boundary condition).
- The thermal-coupling HX failure is modelled by a direct irruption of contamination in the first volume.
- The contamination through the BoP of the NPP may be modelled separately and included in the analysis as a boundary condition.
- It is left as an option the arrangement of RO separating units in two "stages", representing an approximation to the configuration of a Plant with re-circulation of the second stage reject.
- If this option is not used, introducing negligible dimensions to the second stage piping, the model may be reduced to an RO Plant of one stage.
- The outgoing line is connected with a Product Water Tank by a simple pipe.

The actual pressure difference around the separating units is due to the osmosis effect (difference of salinity) plus a driving pressure difference. The modelling of the Separating Units (SU) is made using hydraulically equivalent pressure losses, in order to reproduce the pressures and flows in the nominal steady state. Nevertheless, the membrane coefficients and the pumps parameters are calculated by DESNU by a first order approach considering the system boundary pressures and the pumps elevation, but neglecting the pressure losses in pipes. This produces a steady state with an approximation in flows within the 10%, which is compatible with the scope of the models. If it is required to reproduce the flows more accurately, the pressure loss coefficients and the pumps parameters should be tuned taking the result of running the DESNU model as a reference value.

Similarly as for MED and MSF models, the spreadsheet was developed to produce models with different number of components. Due to the simpler configuration of RO, because within each stage all the SU may be considered in parallel, they can be lumped in an equivalent component with a cross section and flows equal to the sum of individual SU's.

Initial conditions are specified only in the boundaries, i.e., first and last volumes. Figure 6 shows a simplified scheme of the RO-DP model.

I.8.2. Nuclear power plant model

The development of NPP Models was completed according to the Work plan during previous stages of the project. The modelling of the component producing the thermal coupling has been dropped as a goal, according to the shift of effort decided in the meeting of reference [I.5].

I.8.3. DESNU spreadsheet

The advances on the modelling tool were centred on the implementation of the RO Plant modelling. This upgrading of DESNU spreadsheet implied changes in some sheets of the three chapters. From the USER's point of view, DESNU upgrading from its first version has been transparent regarding the original features, and new options have been added as new sheets, or new lines in existing sheets. With this extension, the scope of tasks on DESNU defined for INVAP's contribution is completed. It is foreseen a further expansion by collaboration with the IPPE (Russian Federation) consisting in producing the RELAP input files for the three technologies of desalination.

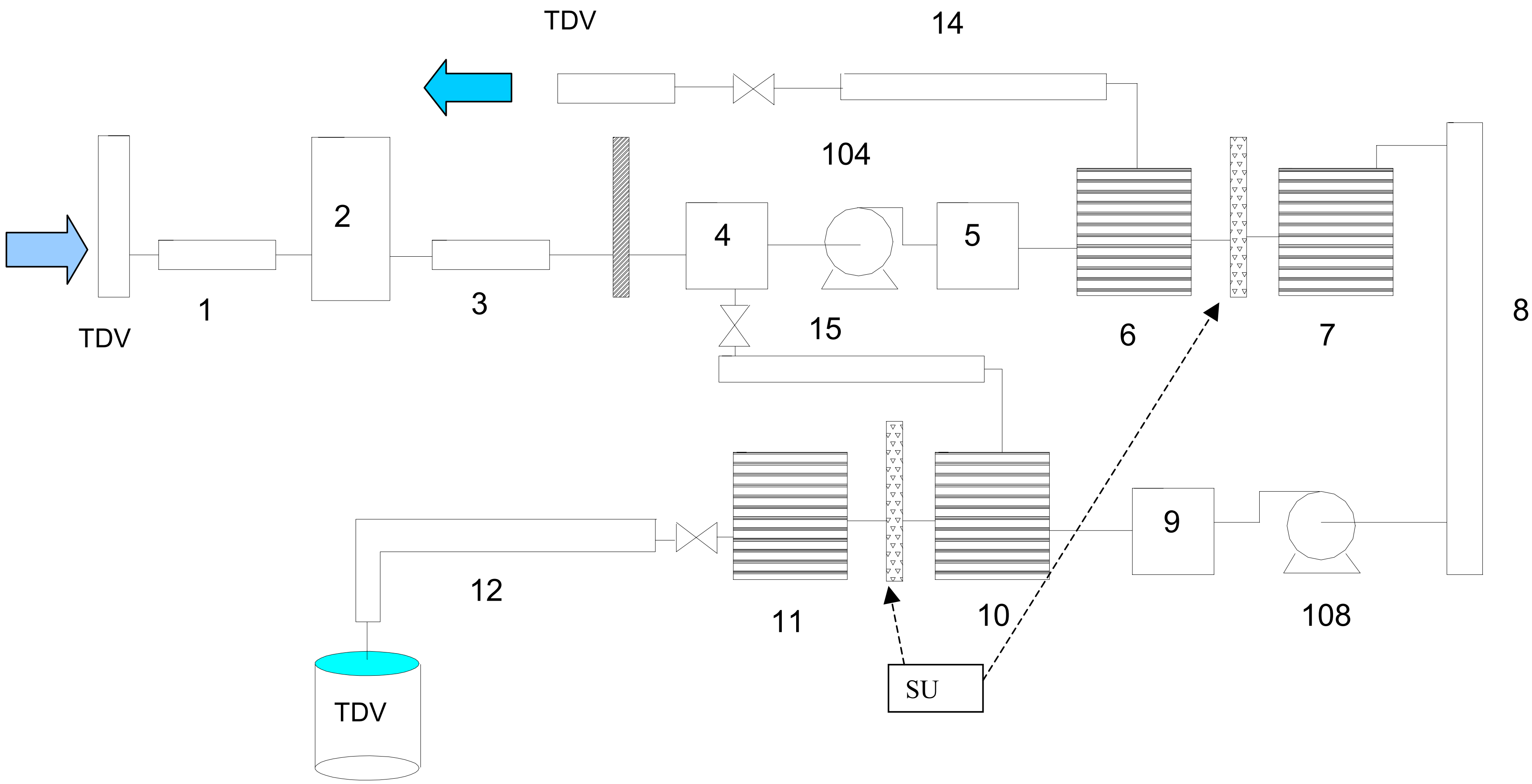


Figure I.6: Scheme of the control volumes of RO model .

I.9. PROGRESS UP TO CRP COMPLETION

During this period a few tasks pending after the last RCM were completed. Other participants were contacted in the frame of improving DESNU by the feedback of external review, and a brief User's Manual was prepared for DESNU.

This User's Manual presents an introduction to the modelling tool and a detailed description of DESNU structure, at different levels. It describes the modelling guidelines, the block scheme of possible models and explain the input sheet for the three desalination technologies as well as the modelling of the BoP with a more limited scope.

A chapter is dedicated to summarise guidelines, rules and hints for using DESNU in terms of EXCEL spreadsheet, producing a RETRAN input file.

I.10. FINAL PRODUCT

The final output, as foreseen for this CRP, is a version of DESNU spreadsheet capable of producing input files for computer modelling of MED, MSF and RO desalination systems, and the BoP of a small NPP.

The goal of DESNU is to support the production of RETRAN input files for the modelling of a NDP without requiring specialised knowledge. The input parameters of the spreadsheet are mainly operational and geometrical data easily available, and the model produced has a coherent nodalization. However, it must be clearly taken into account that the verification and approval of a model for safety relevant simulations is still a task requiring experience and specialisation, both on modelling and on safety. Within this overall goal, INVAP has completed the extension of DESNU for producing RETRAN input files for all the foreseen models, and making it a user-friendly modelling tool.

I.11. CONCLUSIONS

It may be concluded that INVAP's work on this CRP has achieved the goals of the original Work Plan besides the new goals that were defined in RCM's and CM's, according to specific needs and the possibilities of collaboration of participant institutions.

Particularly, a fruitful and successful collaboration took and is taking place around DESNU modelling tool, an expression of which was the Consultants Meeting hosted by INVAP in Bariloche, Argentina from 5 to 7 December, 2000.

The review of DESNU spreadsheet at different stages was performed by the practice of submitting its updated versions to other participants for external review. This practice of using it for the production of a model of different Projects has proved to be adequate to improve DESNU as an open product for the Nuclear Desalination community.

As an additional work, and based on INVAP´s expertise in performing and preparing Safety Analysis Reports for different types of nuclear facilities, the main features regarding the safety aspects of co-generation plants for nuclear desalination could be identified.

The user-friendliness of DESNU was improved by the preparation of a brief User's Manual with a summary of guidelines, rules and hints for using DESNU.

REFERENCES

[I.1] MASRIERA, N., DOVAL, A., "Evaluation of Nuclear Desalination Coupled Systems," Nuclear Engineering Department, INVAP S.E. Work-Plan Report presented in RCM held in Vienna (1998).

[I.2] MASRIERA, N., DOVAL, A., "Evaluation of Nuclear Desalination Coupled Systems", Nuclear Engineering Department, INVAP S.E. First advance report (1999).

[I.3] "Review of Desalination Technologies for the Agency's CRP on Optimisation of the Coupling of Nuclear Reactors and Desalination Systems", Consultancy held at San Diego, USA (1999).

[I.4] MASRIERA, N., DOVAL, A., "Evaluation of Nuclear Desalination Coupled Systems", Nuclear Engineering Department, INVAP S.E. Second advance report (2000).

[I.5] "Analytical modelling of coupling configurations of nuclear desalination plants", Meeting Report of Consultants Meeting held in Bariloche, Argentina (2000).

[I.6] "Development of RELAP5 models for nuclear desalination coupled systems using DESNU tool", presentation by Yaroslav K. Kozmenkov, Department of Nuclear Power Plants, IPPE at the Consultants Meeting held in Bariloche, Argentina (2000).

[I.7] CHEMCAD – Process Flow-sheet Simulation, Chempute Co.

[I.8] "Specific Contents of the SAR for a Nuclear Desalination Plant", International Conference on Nuclear Desalination: challenges and options", Marrakech (2002).

ANNEX II

OPTIMIZED NUCLEAR DESALINATION/COGENERATION USING REVERSE OSMOSIS WITH FEED WATER PREHEAT

J.R. Humphries
CANDESAL Enterprises Ltd, Ottawa, Canada

II.1. BACKGROUND

The International Atomic Energy Agency (IAEA) has been investigating the technical viability and economic competitiveness of nuclear desalination for the past several years. This work has been carried out at the request of, and in conjunction with, a number of Member States to address the increasingly severe problem of water shortages faced by many developing and developed countries throughout world. These Member States have recognized that the lack of adequate supplies of safe drinking water is one of the most pressing problems facing the world today. The desalination of seawater, using a nuclear reactor as the energy source for the desalination process, is one of the alternatives available for the large-scale production of potable water. However, to become attractive as a commercially viable technology, nuclear desalination, as it has come to be called, must offer performance and economic advantages relative to other desalination options. In addition, because shortages in energy supply and fresh water availability often occur together, the production of potable water by desalination must not place an additional burden on already strained energy resources.

Studies [II.1–II5] carried out by CANDESAL prior to this project have indicated the potential for significant performance and economic improvements by using waste heat discharged from a nuclear reactor through the condenser cooling water circuit. A portion of the condenser cooling water discharge stream is taken as preheated feedwater to the RO system, resulting in improved potable water production efficiency by the RO membranes. The IAEA *Options Identification Programme* [II.6] has identified RO with feedwater preheat (coupled to a medium sized reactor such as the CANDU 6) as one of the most practical technical options for the near term demonstration of nuclear desalination. Annex 1.2 *(Desalination Processes and Technology)* of TECDOC-898 did identify some concerns with the operation of RO membranes at elevated temperatures. These included the potential for more rapid membrane fouling and greater membrane compaction, both of which would lead to reduced membrane lifetime. Nevertheless, the potentially significant economic benefits to be gained from RO system operation at elevated temperature warrant a more detailed technical and economic evaluation, including a technical assessment of perceived concerns. Validation of the analytical results is also necessary to ensure confidence in predicted performance and cost calculations.

In view of the above, this project on "Optimized Nuclear Desalination/Cogeneration Using Reverse Osmosis With Feedwater Preheat" was proposed to the IAEA as a part of the Coordinated Research Programme (CRP) on Optimization of the Coupling of Nuclear Reactors and Desalination Systems. The project was approved with a 1998 November 1 start date. The project has reached the conclusion of its work as of 2001 November 30. This report documents the results achieved from the work.

II.2. OBJECTIVE

The long-range objective of the project is to demonstrate the superior performance characteristics and economic advantages of nuclear desalination/cogeneration using reverse osmosis (RO) desalination, with waste heat from the energy generation process used to enhance the characteristics of the desalination process.

II.3. WORK SCOPE

The project on "Optimized Nuclear Desalination/Cogeneration Using Reverse Osmosis With Feedwater Preheat" had several detailed elements that constituted the overall scope of work. Of these, the two most significant were:

- Analyze the performance and economic characteristics of selected systems using RO feedwater preheat and RO system design optimization.
- Validate the analytical models used to carry out the system performance and economic calculations.

Work carried over course of the project was intended to address these issues. The results of this work were intended to support the activities of the IAEA in nuclear desalination by providing information that could be of benefit to those CRP projects and Demonstration projects considering the use of RO with preheated feedwater as their desalination technology.

II.4. ACHIEVEMENTS

Work carried out during this project has resulted in the following significant achievements:

- Performance calculations were carried out for a variety of membrane configurations and types, over a wide range of system input conditions and operating parameters.
- These analysis results demonstrated consistently that improved water production capability and reduced energy consumption can be achieved by taking advantage of the waste heat from an energy plant to provide preheated feedwater to the RO system.
- An experimental facility was constructed and an experimental program carried out to investigate these analytical conclusions. The data obtained has been of very good quality, has been extremely consistent over the full range of pressure and temperature conditions, and has been highly repeatable.
- The results of this experimental program have shown, without a shadow of a doubt, that the performance improvements that have been projected in the analytical studies are achievable in practice.

These results represent the first experimental validation that taking advantage of RO system feedwater preheat can yield improved performance characteristics and water production rates. This important conclusion supports work being carried out in Egypt and India, where experimental investigations of RO preheat is also being carried out. Based on these results, it is judged that there is a high likelihood of success in those programs.

II.5. DISCUSSION

II.5.1. Literature review

A literature review was carried out to survey the technical literature on RO seawater desalination and to identify state-of-the art information regarding experience in RO processes with preheating, in either laboratory scale or operational systems. The literature search was carried out using the services of the technical library at Canada's National Research Council (NRC), local university libraries and the Internet, since much technical literature is now made available on-line.

Aside from CANDESAL and IAEA reports, very few other relevant publications were found relating to high temperature RO, theoretical modeling of temperature effect, or data of similar experimental work. The literature search did identify a few, relatively recent, papers addressing the topic of preheating RO feedwater as a means to improved system performance. (Some, in fact, have referenced previous CANDESAL work on the subject.) None of the papers identified at this point present the results of analysis work; they are all general discussion papers identifying the nature of the expected benefits. Likewise, there appears to be still no operating systems in which this principle has been applied.

It can be concluded from this that CANDESAL still has a technological lead, and is the only organization to have done any significant development work on the concept. Nevertheless, it is clear that others are beginning to carry out work in this field.

II.5.2. Reverse osmosis system performance analyses

Analysis of prospects for performance improvements and economic optimization in RO systems has been carried out with preheating based on current and foreseen membrane technologies

Information on the latest RO seawater desalination membranes was obtained from Dow Filmtec. Their newest membrane, the SWHR-380, has a higher membrane surface area than previous membranes studied by CANDESAL, and represent an advance in membrane manufacturing technology. An increase in membrane surface area from 300 ft^2 (28 m^2) to 380 ft^2 (35 m^2) within the same membrane element dimensions offers a significant performance improvement that is likely to be taken advantage of in more traditional approaches to RO system design by allowing operation at reduced pressures.

Other membrane manufacturers were also surveyed to identify those that may have supply suitable membranes for the CANDESAL system. Hydranautics, Koch Fluid Systems and Osmonics were identified as three of the key competitors to Dow in the supply of RO membranes. While there appear to be a large number of smaller suppliers of RO membranes, serious competition to Dow FilmTec is limited. Due to the eventual requirement for supply of membranes in large quantity consideration will be restricted to these larger competitors, and efforts were initiated to obtain design information on these membranes. Preliminary information indicates that each of these suppliers is also now manufacturing larger surface area membranes similar to those from FilmTec.

In addition to larger surface areas within the same outer dimensions, various manufactures are now providing (on a specialty basis), or are in the process of developing, 16 inch diameter elements and elements that are 60 inches long rather than the standard 40 inch length. Higher temperature membranes and alternative membrane materials are also under development.

All of these advances offer the potential for further performance improvements and economic optimization. For the purposes of this study, attention was limited to the larger surface area membranes that have recently been introduced to the marketplace. A comparison of the FilmTec SWHR-8040 (28 m^2) and the SWHR-380 (35 m^2) was carried out for both single vessel cases with 7 membrane elements per vessel and for multiple vessel (large system) cases. The results are illustrated in Figure II.1 for the single vessel cases. Results for the multiple vessel cases show similar behaviour.

The results shown in Figure II.1 illustrate that for fixed input conditions (such as feed flow, seawater TDS, operating pressure) the SWHR-380 membranes offer significant performance enhancement relative to the SWHR-8040 membranes. The figure also shows that there is a penalty in terms of increased permeate TDS, and so attention will need to be paid in the system design and optimization process to ensuring the permeate quality requirements are satisfied. (Note that unit energy consumption is calculated for these cases without energy recovery.)

The results shown in Figure II.1 for the two membrane types are entirely consistent with the general analytical results that have been obtained during various performance analysis studies, which are shown in Figures II.2 and II.3.

II.5.3. Experimental validation of the analytical model

One of the key elements of this project was experimental validation of the theoretical concept developed by CANDESAL for **RO** system design. The CANDESAL concept approaches the application of **RO** technology in a non-traditional manner, and therefore validation of the analytical results is necessary to ensure confidence in the membrane performance and cost evaluation models.

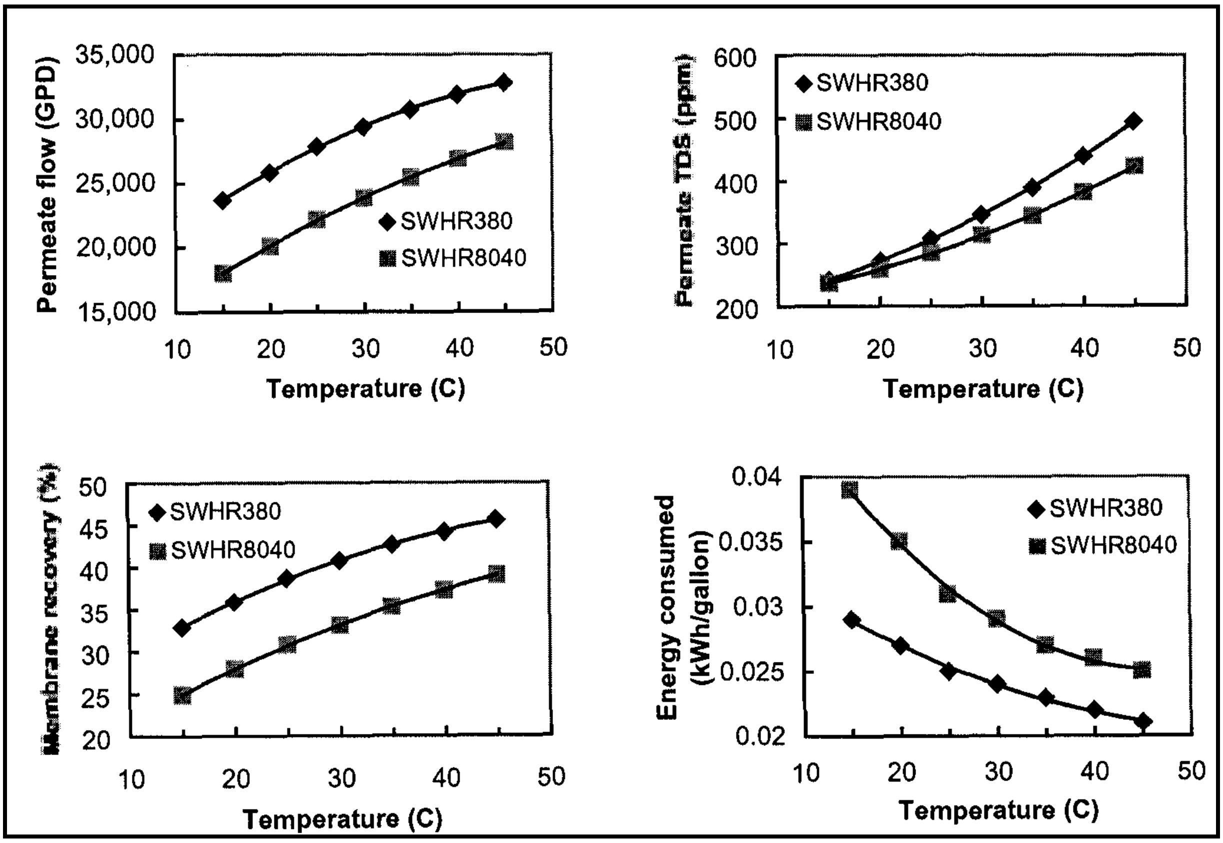


Figure II.1: Single Vessel Cases.

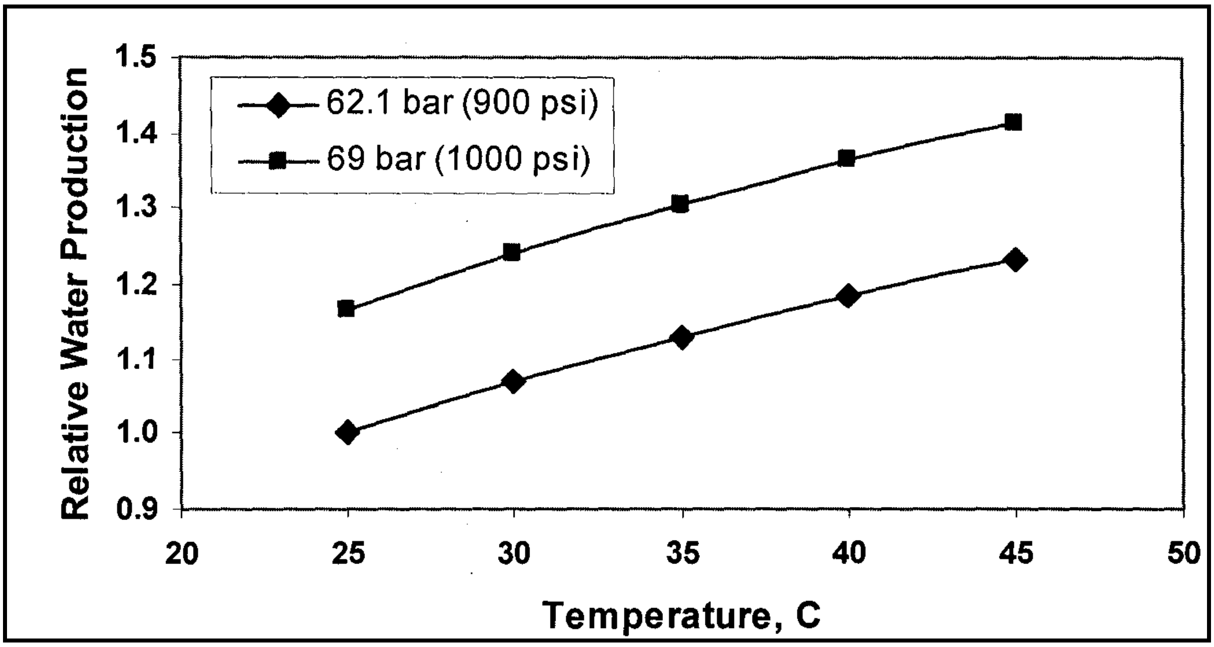


Figure II.2. Relative Water Production.

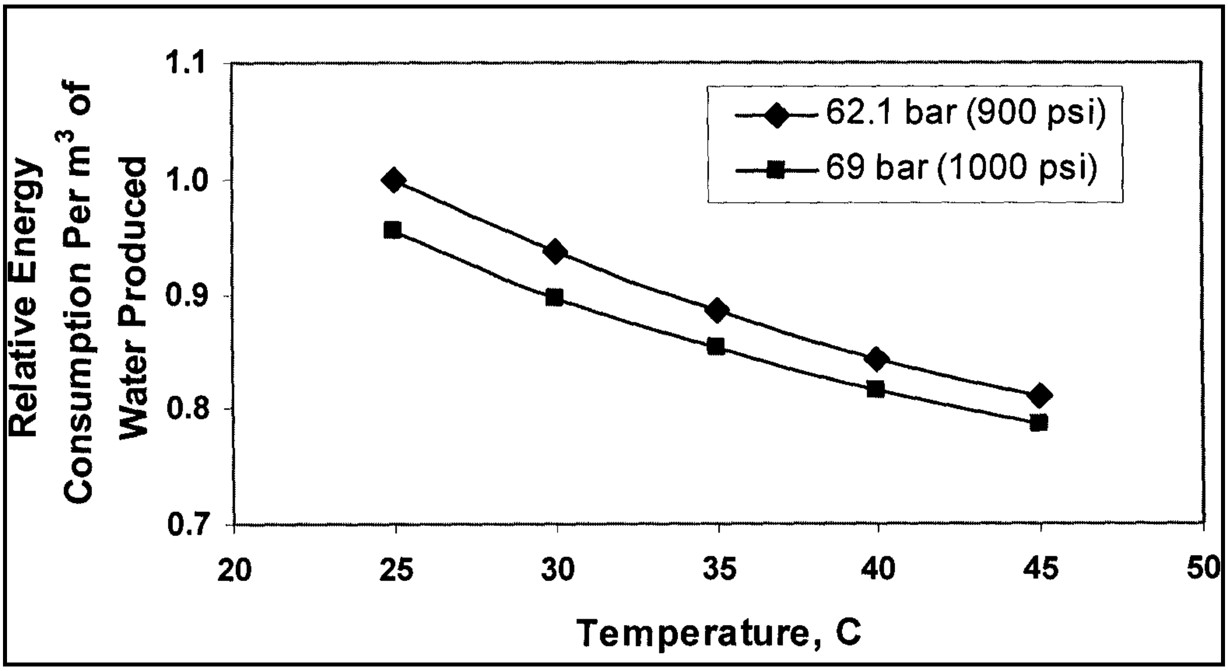


Figure II.3. Relative Energy Consumption.

Because of the high cost of experimental pilot plant equipment, the experimental validation was carried out as a part of a separate project funded in part by the National Research Council of Canada, through its Industrial Research Assistance Program (TRAP). Although having slightly different long-term objectives, many of the tasks required for the TRAP project were similar to those required for this CRP project, and the existence of that project has provided a substantial framework that allowed this project to be carried to a satisfactory conclusion. Approval for the TRAP project was granted in May 1999, and work on the project started as of 1 June 1999.

The design of the experimental demonstration unit was carried out during the year 2000, and procurement of the parts and equipment was undertaken shortly thereafter. The experimental demonstration unit was built and commissioned in early 2001, and the initial commissioning trials program was successfully completed in the spring of 2001. The experimental program itself was carried out in mid-2001. The experimental test runs were conducted over the range from 20°C to 45°C and for system pressures up to 1000 psig (69 bar). Feed flow for all runs was fixed at 40 gallons per minute. The data obtained are illustrated in Figure II.4.

The results of these experiments demonstrate clearly the increasing potable water production achieved for constant system feed flow, with increasing system pressure and temperature. Although such large ranges in temperature and pressure would not be expected in actual operating environments, it is notable that a nearly five-fold increase in production rate was observed from the low temperature, low pressure conditions to the high temperature, high pressure conditions.

Finally, this work has shown the fundamental concepts underlying the CANDESAL design approach are sound and that the expected performance increases suggested by analytical studies can be realized in practice.

II.6. IMPLICATIONS FOR OTHER CRP PROJECTS

Several programs related to the coupling of nuclear reactors and desalination systems are being carried out within the framework of the Coordinated Research Programme. Projects undertaken by Egypt and India include experimental projects that are related to the Canadian project in that they also utilize RO systems that incorporate the principle of RO preheat.

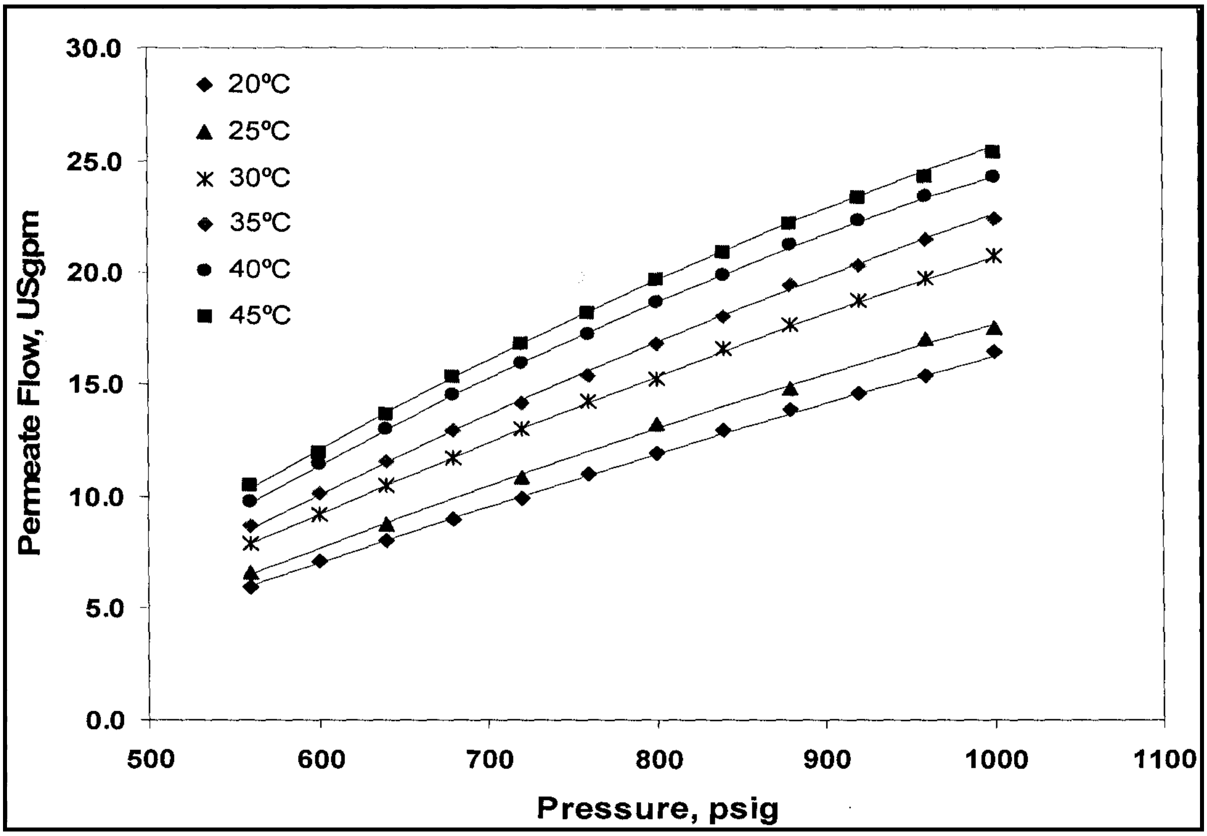


Figure II.4. Permeate Flow as Function of Pressure and Temperature.

In the case of Egypt, and experimental investigation of RO membrane behaviour under varying conditions is being carried out under conditions appropriate to a nuclear desalination plant in that country. In the case of India, a nuclear desalination project is being undertaken in which both MED and RO systems are being coupled to an existing nuclear reactor.

The results obtained in this program lead to a high level of confidence that these other projects can anticipate a successful outcome in their experimental work.

II.7. CONCLUDING REMARKS

Approval was granted for a Technical Project carried out under the auspices of the CRP, entitled "Optimized Nuclear Desalination/Cogeneration Using Reverse Osmosis With Feedwater Preheat," with a project start date of 1 November 1998. The project has successfully achieved its primary objectives. Analytical results obtained have enhanced the level of confidence in the CANDESAL design methodology as an RO system design approach with a potential for significant performance and economic benefits. Experimental validation of the basis for this approach provides confidence that the performance improvements obtained in analytical studies can be realized in practice. In addition, the work provides a high level of confidence that other CRP projects incorporating the RO preheat concept are likely to achieve successful outcomes.

REFERENCES

[II.1] HUMPHRIES, J.R. AND MIDDELTON, E., CANDESAL: A Canadian Nuclear Desalination System, paper presented at US/Middle East Joint Seminar on Desalination Technology", CI-9404, Washington D.C, 10–14 April 1994.

[II.2] CANDESAL Inc., CANDESAL An Advanced Water and Energy Production System: Egypt Applications Study, Report CI-9403, Ottawa (1994).

[II.3] HUMPHRIES, J.R., DAVIES, K., A Floating Cogeneration System Using The Russian KLT-40 Reactor and Canadian Reverse Osmosis Water Purification Technology, paper presented at the IAEA Technical Committee Meeting on "Floating Nuclear Plants for Seawater Desalination", Obninsk, 1995.

[II.4] HUMPHRIES, J.R. et al, A Technical and Economic Evaluation of the CANDESAL Approach in Indonesia Using Reverse Osmosis and Waste Heat from the CAND U-6 Nuclear Power Plant, Rev. 1, CANDESAL Enterprises Ltd, Ottawa, 1998.

[II.5] HUMPHRIES, J.R., DAVIES, K., A Floating Desalination/Cogeneration System Using the KLT-40 Reactor and Canadian RO Desalination Technology, paper presented at the IAEA Advisory Group Meeting on "Propulsion Reactor Technologies For Civilian Application", Obninsk, 1998.

ANNEX III

OPTIMIZATION OF SYSTEM OF SEAWATER DESALINATION WITH 200 MW NUCLEAR HEATING REACTOR

H. Jia
Institute of Nuclear Energy Technology, Beijing, China

III.1. BACKGROUND

With the increasing demand for fresh water reaching critical situation in several areas of the world and China, it is of great significance to develop technologies for increasing global availability of fresh water with consideration of energy strategy, environment protection and safety at the same time. One of the technical schemes is seawater desalination using nuclear heating reactor coupled with MED process.

The NHR type of nuclear heating reactor, developed by Institute of Nuclear Energy Technology, Tsinghua University of China, was designed with excellent safety features, capacity compatibility with reasonably sized desalination plant, thermal-hydraulic parameters suitable for coupling with MED process and economically competitive.

The Nuclear Heating Reactor is a suitable thermo-energy source for use in potable water production with seawater distillation possesses due to its excellent inherent safety features and its appropriate parameters for match with the desalination process. In order to achieve the optimum technical, economical and safety objectives, an investigation project entitled as Optimization of System of Seawater Desalination with NHR-200 was executed as one of the CRP Project of IAEA.

This investigation project was encouraged and supported by IAEA, via a research contract 10243/R0/Regular Budget Fund, and as a part of the IAEA Coordinated Research Project: Optimization of the coupling of Nuclear Reactors and Desalination Processes. This investigation project was entered into force on May 19, 1998. And the research contract was successively renewed as contract 10243/R1, R2, R3 and R4 in the period from July 01, 1999 to Dec. 31, 2003.

During the first research year, preliminary analyses on optimum coupling scheme and optimum coupling parameters were performed. The analysis results indicate that The optimum coupling scheme between NHR-200 and MED process for heat only mode is Scheme A: NHR-200+ Steam generator coupled with MED process;

During the second research year, analysis on coupling schemes with hybrid desalination process, further optimization analysis on coupling parameters of NHR heating reactor and desalination process and Investigation on selected VTE-MED process were performed.

During the third research year, an investigation program on thermal-hydraulic performance of selected VTE-MED process coupled with NHR-200 was implemented, including design and build up of a VTE-MED process test system and development of a computer code for analysis on thermal-hydraulic performance of the stacked VTE-MED process.

During the fourth research year, experimental investigation on thermal-hydraulic performance of the selected VTE-MED process coupled with NHR-200 (Commissioning and Running of the test system), improvement of the computer code for analysis on thermal-hydraulic performances and result of the additional analysis on thermal-hydraulic performances of stacked VTE-MED process coupled with NHR-200 were performed.

Based on the results of the investigation, an integrated seawater desalination plant with nuclear heating reactor NHR-200 coupled with high temperature MED desalination process with stacked VTE-MED process with tower design was recommended.

It should be noted that all the water cost analyses performed during the period (1998-2001) of this contracted project and all the results of the calculated water cost presented in the relevant progress reports of CRP-10243 were based on the old economic input data, which were established before nineties in accordance with the international market price. So that the calculated water cost was about 0.8–1.0 US$/$m^3$ that is higher than the present updated data.

From the year of 2002 on, especially via the "Economic research on the Desalination Plant with NHR reactor coupled with high temperature VTE-MED process and the site-specific investigation", the later on economic analyses were based on innovative technical design and domestic market price in China. The construction cost, investment, M&O cost etc. were updated, and therefore the water cost was also updated. In that cases, the water cost is decreased to about 0.6 US$/$m^3$. (The results of this part of economic analysis may be presented in the relevant report of CRP-11863 project). But in order to reflect the history of the Contracted Research Project CRP-10243, all the economic data listed in this Final Report still maintain the original values as that in the relevant progress report of the CRP-10243 project.

III.2. OBJECTIVES

The ultimate objective of this research project is to define a integrated seawater desalination plant with NHR-200 Nuclear Heating Reactor coupled with a selected desalination process, which is able to safely, stably and economically provide high quality potable water to a medium size of town with about one million population. Design of the desalination plant should satisfy the optimum criteria defined in the investigation project.

The objective of the contracted research project is to:

— Define optimum criteria;
— Establish a optimum coupling scheme and optimum operation parameters;
— And therefore finally lead to a optimum water production cost

III.3. SCOPES OF THE INVESTIGATION

The scope of this research project for the first research year was to study the following aspects in the optimized coupling, include both technical and computational analysis on the coupled system of NHR-200 Heating reactor and selected desalination process:

(1) Development of coupling scheme of following desalination process with NHR-200 reactor:
    (a) High temperature MED
    (b) Low temperature MED and MED/VC
    (c) Low temperature MED and RO

(2) Estimation of parameters and GOR vs. investment cost for each process.
(3) Optimization of water production cost achieved by the coupling and variation of parameters.

The scope of this research project for the second research year was to study the following aspects in the optimization analysis of coupling schemes of NHR-200 Heating reactor and selected desalination process, include:

1. Analysis on the selected hybrid desalination systems:
   - Coupling scheme of heating reactor NHR-200 with LTMED and MED/VC desalination process.
   - Coupling scheme of heating reactor NHR-200 with LTMED and RO desalination process. Part of this work was developed during the first research year.
2. Investigation on further optimization of the selected coupling scheme.
3. Further analysis on optimization of coupling parameters and water production cost.
4. Sensitivity analysis on the water production cost for the coupling scheme NHR-200 and high temperature MED.
5. Investigation on tower design with selected VTE-MED process coupled to NHR-200.

The scope of this research project for the third research year was to study the following aspects in the optimization of coupling schemes of NHR-200 Heating reactor and selected desalination process, include:

1. Additional analysis on coupling scheme
2. Development of the computer code for analysis on the thermal-hydraulic performance of the stacked VTE-MED process.
3. Design of the VTE-MED test system:
4. Build up of the test system.

The scope of this research project for the fourth and fifth research year was to study the following aspects in the optimization of coupling schemes of NHR-200 Heating reactor and selected desalination process, include:
1. Improvement of the developed computer code for analysis on thermal-hydraulic performance of the VTE-MED process coupled with an NHR nuclear heating reactor;
2. Performing experimental investigation on performance of the designed VTE-MED process and analysis on the test results;
3. Verification of the developed computer code with the experimental data;
4. Additional analysis on performance of the designed VTE-MED coupled with the NHR reactor;
5. Preparation of the materials for the draft TECDOC and the Final Report of the contracted research project.

III.4. TECHNICAL CONSIDERATIONS

III.4.1. Advantages of NHR-200

Advantages in technical, safety and economical aspects of seawater Desalination plant by using a modest size of Nuclear Heating Reactor

(1) Favorable capacity-compatibility of seawater desalination plant with nuclear power reactor.
Medium scale of seawater desalination plant with (50,000 ~100,000) m^3/d could possess broad application prospects, which matches the potable water demand of a city with tens thousand to hundreds thousand of population. This size of desalination plant would be supplied with (20~40) MW(e) of electricity if RO process is adopted, or be supplied with (60~125) MW(th) of heat if MED process is used. The energy requirement of above mentioned medium size of desalination plant is only a small portion of the capability of a conventional nuclear power plant with (600~1000) MW(e). A small nuclear power facility would be much more suitable in capacity compatibility for the medium size of seawater desalination plant.

(2) Advanced technical and safety features of the nuclear heating reactor.
The 5MW nuclear heating reactor (NHR-5), developed by Institute of Nuclear Energy Technology (INET) of Tsinghua University, Beijing, China, has been successfully operated for 10 winter seasons since put into operation in 1989. An experiment program was performed to demonstrate transient behavior under accident conditions directly on the heating reactor NHR-5. Excellent safety characteristics of the nuclear heating reactor were proven by the results from the demonstration experiments on transient behavior under accident conditions directly on the heating reactor NHR-5.

NHR is a vessel type of pressurized water reactor with integrated design of primary circuit, natural circulation of coolant for full power operation and self-pressurization. According to the special safety requirement to be located near to the consumer, the heating reactor is designed as a new generation of nuclear reactor with inherent and passive safety features. The reliability and economy of the heating reactor are notably improved by simplified system design due to above-mentioned technical and safety features.

(3) Economy of the small desalination plant using nuclear heating reactor.
Generally, the smaller the size of nuclear power plant, the higher the energy cost and therefore the higher the produced water cost. But for the heating reactor case, the size increased energy cost would be partly compensated by the design features of simplified systems and lower system design parameters.

The preliminary analysis shown that for a desalination plant using MED process coupled with a 200 MW heating reactor, the potable water production could reach 160,000 m^3/d and the water cost could be about US$ 0.8–1.0/m^3.

The small heating reactor would not be confronted with such serious difficulties in funding, site selection and infrastructures as that the large nuclear power plant usually faced, especially in developing countries. There are some markets potential for their deployment as integrated nuclear desalination plant in several countries and areas with only small electric grids

(4) Favorable for local participation

- Simplified system and passive safety performance would less require operation and maintenance techniques.
- Less manufacturing infrastructure is required due to the lower design parameters for the systems and components of NHR.
- Nuclear safety Codes and guidelines adopted in design of heating reactor are compliant with that international prevailing and approved by IAEA. Codes adopted in manufacturing and design of NHR components is in compliance with prevailing ASME code.

III.4.2. Nuclear heating reactor NHR-200

NHR Nuclear Heating Reactor is a vessel type of pressurized water reactor with integrated design of primary circuit, natural circulation of coolant for full power operation and self-pressurization. The structure of NHR-200 is shown in Fig.1. The design parameters of NHR-200 are shown in Table III.1.

According to the special safety requirement to be located near to the consumer and directly connected with desalination plant, NHR is designed as a new generation of nuclear reactor with inherent and passive safety features quite different from that of general nuclear power plant that is strongly based on engineering safety facilities. The inherent safety features of the heating reactor are mainly as follows:

- strong negative temperature coefficient of reactivity
- residual heat removal passively by natural circulation
- lower operating pressure in primary system and lower power density in the core
- large break LOCA is excluded,
- long enough grace period for corrective action
- fail-safe principle on design of hydro-driven control rod and elimination of possibility of control rod ejection.

The reliability and economy of the heating reactor are notably improved by simplified systems due to above-mentioned technical and safety features.

III.4.2.1. Basic design objectives of NHR-200

The Nuclear Heating Reactor NHR-200[2] (with 200MW of thermal power) is designed for district heating, seawater desalination and district refrigeration and so on.

(A) The basic design Objectives on Safety of NHR-200 are as follows:

- Reactor core is cooled with natural circulation.
- Reactor core is prevented from uncovering of coolant under any accident conditions.

TABLE III.1: MAIN DESIGN PARAMETERS OF NHR-200

Design parameters	Heat only	Co-operation
Reactor power, MW(th)	200	200
Pressure in primary circuit, MPa	2.5	3.0
Core outlet temperature , ℃	213	223
Core Inlet temperature , ℃	154	164
Active core height, m	1.9	1.9
Number of fuel assemblies	156	156
Enrichment of initial core, %	3.0/2.4/1.8	3.0/2.4/1.8
Diameter of fuel element, mm	10	10
Average power density, kW/L	37	37

– Integral primary circuit arrangement.
– Self-pressurization of the primary circuit.
– 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 the accident, and maintain core cooling indefinitely without AC power.
– Predicted core damage frequency $<10^{-8}$ per year and a significant release frequency $<10^{-9}$ per year.
– Reliable reactivity control and shutdown system.
– Low operating parameter and large safety margin.

(B) The basic design Objectives on Reliability of NHR-200 are as follows:
– To simplify design, operation and maintenance.
– Public radiation exposure at the plant site within the range of 80Km: $<5\times10^{-4}$ man–Sv per year.
– Overall plant availability goal considering forced and planed outage: greater than 95%.

III.4.2.2. The major innovative design features of NHR-200

– Integrated arrangement, self-pressurization.
– Dual pressure vessel structure.
– Low operating temperature, low pressure and low power density in reactor design which provide increased operating margins and improve fuel economy.
– Cooling reactor core with simple, passive measure which uses natural driving forces only.
– Adopting innovating 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.

III.4.2.3. Brief description of NHR-200 reactor system design

The design of main systems of NHR-200 is summarized as the following:

(1) Primary Circuit
The primary coolant receives heat from the reactor core, then flows up through the riser and enters the primary heat exchangers, where the heat carried is transferred to the coolant of the secondary circuit. Integrated arrangement of the primary circuit is adopted and all main components of the primary circuit are contained in the RPV to decrease the possibility of LOCA.

(A) The reactor pressure vessel
The HR-200 is designed with integral, vessel-type structure. The RPV is the pressure boundary of the reactor cooling. It is 13.62m inside height, 5.00m inside diameter, shown in Figure 1. The design also features a large coolant inventory in the RPV (about 200T) which provide for a lower total neutron flux to the RPV ($<10^{16}$n/cm^2 for 40 year's lifetime of the reactor). All in-vessel penetrations (and only with small diameter) are located on the upper part of the RPV.

(B) Reactor core
The reactor core of NHR-200 consists of 96 assemblies (fuel ducts) and 32 control rods. The reactor core stands on the lattice-support structure, which is fixed on the lower position of the RPV. A long riser is designed and located above the core to enhance the natural circulation capability of coolant.

The fuel bundle arranged in 12×12 with an active length of 1.9m is contained in the fuel box. The crucial-type control rods are placed in the gaps between the square boxes. 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 2.3m. Spent fuel assemblies are stored in the rack around the active core. Burnable poison (Gd2O3) is used to partly compensate the fuel burn-up reactivity, and soluble boron is utilized for reactor shutdown only. This results in a negative temperature coefficient of reactivity over the complete core life.

A low core power density (～36kw/l) provides thermal reliability during normal and accidental operating conditions. This solution greatly simplifies the refueling equipments and eliminates the necessary space in the reactor building.

(C) Control rod and control rod drive mechanism
A new type of the hydraulic driving facility is used for driving the control rod in 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 with controlling magnetic valve in the control unit, and then it moves the movable part of the step-cylinder step by step. The drive system is very simple in its structure and is designed on "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.

(2) Intermediate circuit
An intermediate circuit is arranged between the NHR-200 and the steam supply system. Operating pressure in the intermediate circuit is higher than that in the primary circuit. So that the intermediate circuit would also works as a barrier to prevent the desalination system from radioactive contamination.

The passive residual heat removal system with three trains of natural circulation systems is connected on the intermediate circuit in parallel. It is able to properly disperse core decay heat into atmosphere, the final heat sink, by means of natural circulation, when the desalination system is in shutdown condition. A steam generator is located on the intermediate circuit, in which the motive steam is generated and sent to the seawater desalination system.

III.4.3. Preliminary optimization analysis of coupling of NHR-200 with selected seawater desalination process

According to the design features and the parameters of NHR-200, match able desalination processes were selected, and several coupling schemes were established as possible candidates. Technical optimization analysis and economical optimization analysis were performed for every candidate-coupling scheme.

III.4.3.1. Reasons for choosing MED desalination process

IAEA has organized widely studies on the various desalination process coupled with various energy system during the last 30 years [3]. In the “Option Identification Program” of IAEA, the pro and cons for the various combinations are summarized [4]. The necessary selection criteria for the available desalination processes were provided in the “Options Identification Program” and a most suitable desalination process, Multi-Effect Distillation (MED) system coupled with the heating reactor, was proposed.

MED is a mature desalination process and energy efficiency is an important economic factor as well an environmental factor, the energy efficient MED is competitive against MSF.

MED systems exist in different arrangement. The main difference is the tube arrangement, which can be either vertical or horizontal. The horizontal tube arrangement is called HTME (Horizontal Tube Multiple Effect). The advantage of a HTME desalination plant is that in the temperature range of 40~100°C with plain tubes high heat transfer coefficients can be obtained and low temperature drop ΔT per effect in the range of 2.5~3.0°C is achievable. This permits to obtain GOR’s of 10~12 with top temperatures as low as 65~70°C and GOR’s of 15~17 with top temperatures in the range of 90~100°C and seawater temperature of 25°C. The disadvantage of HTME is that at approximate 90°C the heat transfer coefficient is not increasing any more. There is even a decline in heat transfer coefficients above 100°C.

Therefore, for required GOR’s above approximate 18, the high temperature MED desalination process with Vertical Tube Evaporators (VTE-MED) should be used. Especially with heat transfer tubes with enhanced surface (such as double fluted tubes) and with additive creating two phase flows in the tubes, called as Vertical tube Foamy Flow Evaporation (VTFE), high GOR’s up to 30 and top brine temperatures of 135℃ are obtainable.

For the Desalination Plant with NHR-200 nuclear reactor, temperature of motive heating steam provided by the heating reactor is (130–150)°C and GOR above 22 is required, which makes the plant reach high efficiency of water production. Therefore high temperature MED desalination process, say VTFE-MED process, should be at the first priority in selection from economical and technical point of view.

In order to minimize the number of inter-effect pumps and enhance the available gravity pressure drop, the tower arrangement of VTFE-MED with stacked evaporators is a good and economic solution. A single tower arrangement without any inter-effect pump is also possible, however the height of nearly 130m above the ground for GOR 22 makes this arrangement only feasible for large capacities requiring large tower diameters (i.e. for NHR-200 nuclear desalination plant with 170 000 m^3/d capacity and GOR 25, with a tower diameter of approximate 20m and tower height of approximate 150m).

III.4.3.2. Selection of optimum coupling scheme of NHR-200 nuclear heating reactor to MED process

In order to reach the optimum technical, economical and safe objective, a optimization analysis was performed for a desalination plant with NHR-200 as energy source and coupled to MED process with different coupling schemes.

III.4.3.2.1. Criteria for selection of optimum coupling scheme

Special considerations for selection criteria of optimum coupling scheme of desalination process and Nuclear Heating Reactor are as follows:

(1) **Reliability**
Both of the Nuclear Heating Reactor and the selected desalination process should be proven technology. 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 desalination process is efficiently transferred with steam from the steam generator to the first effect of the desalination system. Both of nuclear heating reactor and MED process system have perfect reliability, which leads to higher availability of the integrated desalination plant.

(2) **Safety**—Multi-barriers isolation and continuous inspection of the radio activity in the Intermediate circuit; Between nuclear heating reactor and desalination system, there are three layers of steel wall boundaries, i.e. heat transfer tubes of the primary heat exchanger, steam generator and the first effect of evaporator of desalination system) and two circuits (Intermediate circuit and Steam supply circuit) working as barriers to effectively prevent the desalination system from radioactive contamination.

(3) **Inherent safety**—Appropriate pressure barrier;Pressure in the primary circuit and the secondary circuit are 25 bar and 30 bar respectively. Even in case of very rare failure of heat transfer tube 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) **Efficiency**—Perfect match of parameters; Coupling of a high temperature MED process with the nuclear heating reactor leads to a higher energy efficiency of the integrated desalination plant; Nuclear Heating Reactor is unitarily dedicated to supply heat to the desalination process. Both sides can share a common planed outage for maintenance, which leads to higher availability for the integrated plant.

(5) **Operability**—Suitability of operation performance of the MED process and Nuclear Heating Reactor systems; Both of the nuclear heating reactor and the desalination system has excellent self-regulation capability. When the load fluctuates within the range of 70% to 100 %, no matter the active perturbation happens in either side, operation of the integrated plant would be very smooth and with perfect self-regulation performance. Adjustment within load range of 40% to 100% is also easy and simple to perform.

(6) **Simplicity**—Simplicity and maintainability of design of the main components of both heating reactor and MED process result in the easy operation and maintenance of the integrated desalination plant;

(7) **Economical**—The integrated desalination plant with the selected coupling scheme would produce drinking water with optimum water production cost and high water quality. The optimization of coupling scheme should eventually be economical optimization.

III.4.3.2.2. Optimization analysis on coupling scheme with heat only mode

Several schemes were selected as candidates for comparison in the investigation:

- The optimum-coupling scheme was defined based on assessment of the production of potable water and technical optimization analysis according to the selection criteria mentioned above.
- The analysis results indicate that the optimum coupling scheme between NHR-200 and MED process is Scheme A: NHR-200+ Steam generator +MED process.

III.4.3.3. Optimum design parameters of seawater desalination plant with nuclear heating reactor NHR-200 coupled to MED process

Based on the coupling scheme optimization analysis mentioned above, the design parameter optimization was performed for the optimized scheme.. For the defined optimum coupling scheme, preliminary estimation of water production cost was made with variation of its operation parameters.

III.5. ANALYSIS ON THE SELECTED HYBRID DESALINATION SYSTEMS

In order to investigate the possible further optimization of the coupling design of NHR heating reactor with desalination process, the coupling schemes of NHR reactor with hybrid desalination process were investigated.

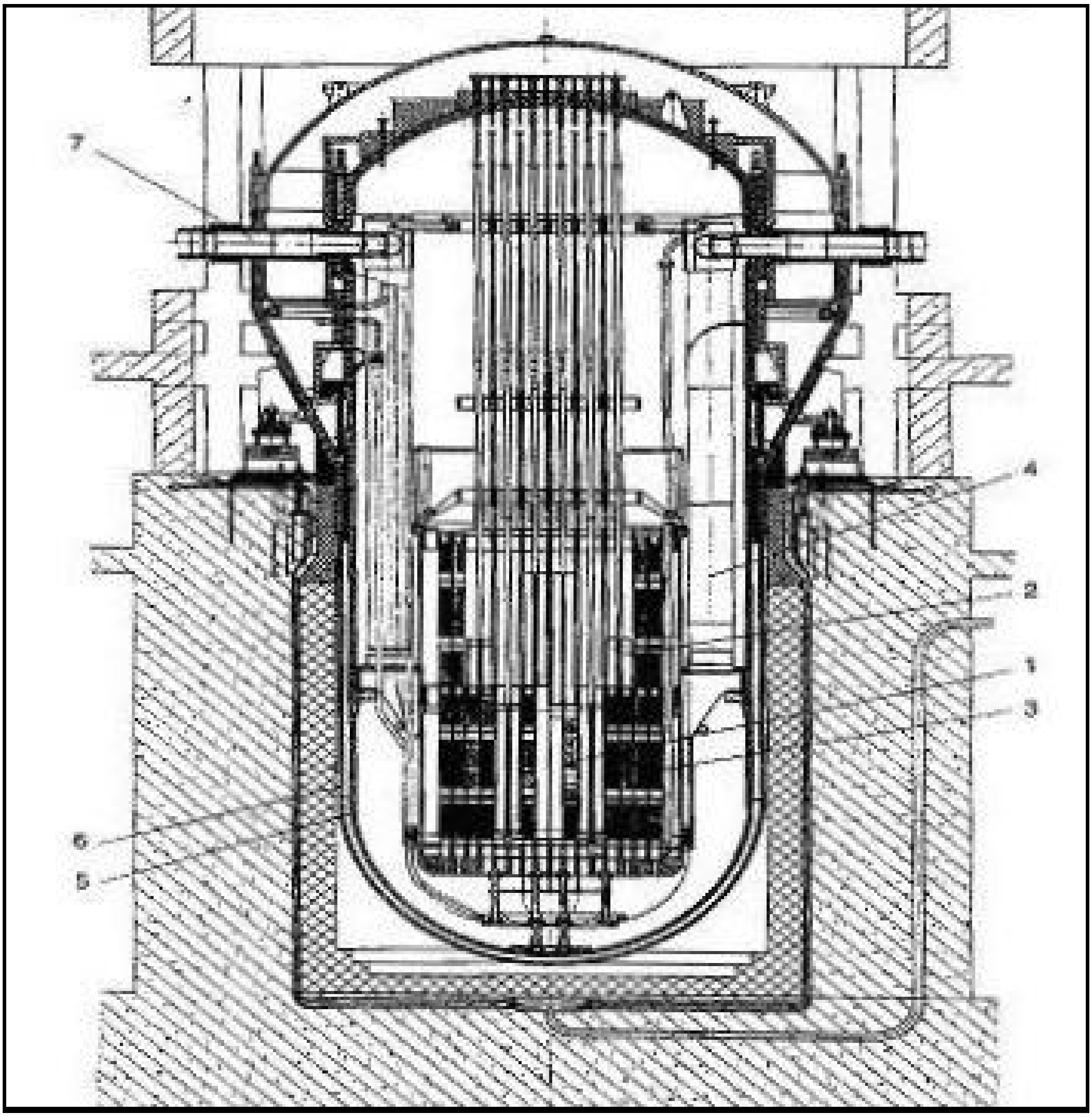
Two coupling schemes were selected for the cogeneration mode:

a. NHR+Low temperature MED+MED/VC
b. NHR+Low temperature MED+RO

In these cases, the design parameters of the NHR heating reactor with cogeneration mode are listed in Table III.2.

III.5.1. Coupling scheme of heating reactor NHR-200 with LTMED and MED/VC desalination process

The Coupling scheme of heating reactor NHR-200 with LTMED and MED/VC desalination process is established. The design of the coupling scheme is accorded with the defined criteria for selection of optimum coupling scheme. The saturate steam generated in the steam generator was directly conducted to the steam turbine as the motive steam. The exhausted steam with backpressure from the turbine was conducted to the first effect of the LTMED process and the condensate from the first effect was pumped back to the steam generator as its feed water. Several MED/VC units worked parallely with the LTMED process.



1. Reactor Core 2. Control Rod 3. Spent Fuel Storage 4. Primary Heat Exchanger
5. Reactor Pressure Vessel 6. Tight Containment 7. Pipes for Secondary Loop

Fig III.1: Structure of Nuclear Heating Reactor NHR-200.

But the part of the brine from LTMED was used as the feed sea water to the MED/VC units. The total capacity of the selected MED/VC process depends on the maximum electricity production of the NHR heating reactor, and the number of MED/VC unit depends on the available unit capacity of MED/VC process. The preliminary economic analysis for the coupling scheme of heating reactor NHR-200 with LTMED and MED/VC desalination process was performed.

Based on the analysis results, It is noted that the cost of potable water produced with the hybrid coupling scheme (NHR+LTMED+MED/VC) is relatively low, 0.73 US$/ m^3 and 0.77 US$/ m^3, respectively.

TABLE III.2: DESIGN PARAMETERS OF HEATING REACTOR NHR-200 FOR COUPLING WITH SELECTED HYBRID DESALINATION PROCESS

Items	Design parameters
Operation mode	Cogeneration
Reactor power, MWt	200
Core outlet temperature, ℃	223
Core inlet temperature, ℃	164
Outlet temperature of the secondary circuit, ℃	163
Inlet temperature of the secondary circuit, ℃	150
Outlet steam temperature of the steam generator, ℃	147
Inlet steam pressure in turbine, Mpa	0.416
Inlet steam temperature in turbine, ℃	145
Back pressure of the turbine, Mpa	0.101
Inlet steam temperature in LTMED, ℃	100
Electricity production, MW	14.42

III.5.2. Coupling scheme of heating reactor NHR-200 with LTMED and RO desalination process

The Coupling scheme of heating reactor NHR-200 with LTMED and RO desalination process is established. The saturate steam generated in the steam generator was directly conducted to the steam turbine as the motive steam. The exhausted steam at backpressure from the turbine was conducted to the first effect of the LTMED process and the condensate from the first effect was pumped back to the steam generator as its feed water. Several RO units worked parallel with the LTMED process. But the part of the brine from LTMED was used as the feed seawater to the RO units. The total capacity of the selected RO process depends on the maximum electricity production of the NHR heating reactor, and the number of RO unit depends on the available unit capacity of RO process.

III.5.3. Analysis on optimization of coupling parameters for the selected coupling scheme

The analyses on influence of coupling parameters on water production for heat only mode and water & electricity production for cogeneration mode were carried out.

The effect of motive steam temperature on water production for coupling of heat only mode (NHR-200+SSG+MED) was analyzed. With increasing in motive steam temperature up to 130℃, the water production increases, but the mass flow rate in the secondary circuit must also increases due to the decrease of useful temperature deference in the flasher, and so that the pumping power consumption must also increase.

The effect of motive steam temperature on water production, for coupling scheme of heat only mode (NHR-200+OTSG+MED) was analyzed. In this coupling scheme, the once-through steam generator was adopted instead of the primary heat exchanger in the integrated reactor vessel and the separate steam generator for coupling scheme A. With increasing in motive steam temperature up to 130℃, the water production increases.

III.5.4. Sensitivity analysis on the water production cost for the coupling scheme NHR-200 and high temperature MED

The results of both sensitivity analysis and analysis on optimization of coupling parameters and water production cost were presented. The effect of motive steam temperature to MED process on the water cost for heat only mode and the effect of motive steam temperature to turbine and exhaust steam temperature to MED process on the water cost for cogeneration mode were also investigated. Sensitivity analysis was carried out in order to understand the effect of some key parameters on the final water cost.

III.5.5. Investigation on stacked VTE-MED process with tower design

Concept design of the VTE-MED desalination system was presented. Flow diagram of the VTE-MED desalination system coupled to NHR-200 was given. Descriptions of the concept design of the VTE-MED desalination system, including design parameters, thermal hydraulic mechanisms, main components equipments, seawater supply and pretreatment, etc. of the desalination plant were provided in detail in the relevant section of the Final Report.

III.6. ADDITIONAL ANALYSIS ON COUPLING SCHEMES

In order to investigate the possible further optimization of the coupling design of NHR heating reactor with desalination process, the coupling schemes under heat only operation mode of NHR reactor with different distillation desalination process were comparatively investigated.

Two coupling schemes were selected for the comparative analysis:

(a) NHR coupled with HT-MED (Low temperature distillation process with Horizontal tube Evaporators)

(b) NHR coupled with VTE-MED (High temperature distillation process with Vertical tube Evaporators)

In these cases, the design parameters of the NHR heating reactor with heat only mode are the same for these two coupling schemes. The design parameters of the NHR heating reactor and the selected desalination process are listed in Table III.3.

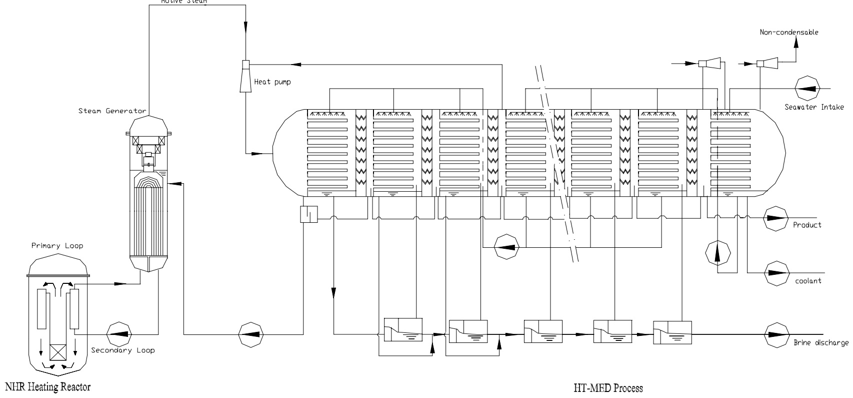
III.6.1. Coupling scheme of heating reactor NHR-200 with HT-MED process

The Coupling scheme of heating reactor NHR-200 with HT-MED and desalination process is shown in Fig. III.1. The design of the coupling scheme is accorded with the criteria for selection of optimum coupling scheme (see “Selection of criteria for selection of optimum coupling scheme”, Section 3.2.1, Progress Report, May. 1998). On Figure III.2, it can be seen that the saturate steam with higher temperature (125°C) generated in the steam generator of the NHR-200 was directly conducted to the inlet of the heat pump (a steam ejector) as its motive steam. Some amount of steam with lower pressure is drawn out from a middle effect of the HT-MED process and blended with the motive steam in the mix chamber of the ejector. At the exit of the heat pump, The mixed steam reach temperature of 73°C. This mixed steam is conducted to the first effect of the HT-MED desalination process as the motive steam of the HT-MED process.

Part of condensate from the first effect was pumped back to the steam generator of heating reactor as its feed water. Meanwhile another part of the condensate from the first effect must be conducted to the produced water line to keep the total mass balance of the system.

TABLE III.3 DESIGN PARAMETERS OF HEATING REACTOR NHR-200 AND THE SELECTED DESALINATION PROCESS

Design parameters	Heat only	Heat only
Reactor power, MWt	200	200
Core outlet temperature, ℃	213	213
Core inlet temperature, ℃	154	154
Pressure at the primary circuit, MPa	2.5	
Outlet temperature of the secondary circuit, ℃	170	163
Inlet temperature of the secondary circuit, ℃	135	135
Pressure at the secondary circuit, MPa	3.0	3.0
Outlet steam temperature of the steam generator, ℃	130	130
Outlet steam pressure, of the steam generator, MPa	0.27	0.27
Flow rate of motive steam to the MED process, t/h	330	330
Specific construction cost of the reactor, US$/kWt	552	552
Water plant:	VTE-MED	HT-MED
Inlet steam temperature in MED process, ℃	125	125/73
Top brine temperature, ℃	120	70
Unit capacity of MED process, m^3/d	40,000	21000
Unit number	4	6
Number of effect	28	15
GOR	20.2	16



Simplified schematic diagram of NHR-200 coupled with HT-MED process

Fig III.2: Coupling scheme of NHR-200 with HT-MED desalination Process.

TABLE III.4: RESULTS OF PRELIMINARY ECONOMIC ANALYSIS FOR THE COUPLING SCHEME OF HEATING REACTOR NHR-200 WITH HT-MED AND VTE-MED PROCESS

Design parameters	**Heat only mode**	**Heat only mode**
Selected Desalination Process	VTE-MED	HT-MED
Reactor power, MWt	200	200
Core outlet temperature, ℃	213	213
Core inlet temperature, ℃	154	154
Outlet temperature of the secondary circuit, ℃	163	163
Inlet temperature of the secondary circuit, ℃	135	135
Outlet steam temperature of the steam generator, ℃	130	130
Economic aspect		
Nuclear Heating Reactor		
Specific construction cost of the reactor, US$/kWt	552	552
Total construction cost (incl. Site rel. cost), M$	121.44	121.44
Interest rate	8%	8%
Construction lead time, month	48	48
Interest during construction, M$	12.38	12.38
Total investment for Nuclear Plant, M$	133.82	133.82
Annual levelized capital cost M$/a	8.71	8.71
Annual l fuel cost, M$/a	4.1	4.1
Annual electricity cost, M$/a	0.63	0.63
Annual O & M cost, M$/a	4.05	4.05
Annual levelized decom. cost, M$/a	0.52	0.52
Annual water plant heat charge., M$/a	18.01	18.01
Steam price, $/t	6.92	6.92
Water plant	VTE-MED	LT-MED
Inlet steam temperature in MED process, ℃	125	125/73
Unit capacity of MED process, m^3/d	40,000	21000
Unit number	4	6
Number of effect	28	15, with heat pump
GOR	20.2	16
Total construction cost for MED plant**, M$	192.0—255.6	96.0—151.2
Construction lead time, month	24	12
Total investment cost for water plant M$	207.36—276.05	102—160.6
Annual water plant fixed charge, M$	14.41—19.18	9.62—15.15
Annual water plant O&M cost, M$	4.55	3.06
Annual water plant electricity cost, M$	4.61	2.62
Annual water plant heat charge., M$/a	18.01	18.01
Annual water plant total cost, M$/a	41.58	33.20
Maximum water production, m^3/d	160,000	126,000
Water cost, US$/$m^3$	0.8—0.92	0.76—0.89

**: Strongly depends on the specific construction cost provided by the provider

The coupling scheme would possess higher efficiency due to use of heat pump. But meanwhile, the produced water circuit would directly link up with the steam circuit of NHR reactor. Therefore, the produced water would loss one barrier circuit, compared with the coupling scheme without use of heat pump. The total water production of the coupling scheme (NHR+HT-MED) is 126,000 m^3/d, which is lower than that of the coupling scheme (NHR+VTE-MED) due to its lower temperature of motive steam (73 ℃) and therefore lower GOR (16).

Preliminary economic analysis for the coupling scheme of heating reactor NHR-200 with HT-MED desalination process was performed. During the economic analysis for the HT-MED process, some of the input data are consulted with IDE limited co., Israel. The estimated economic results are listed in Table III.4.

Based on the analysis results, It is noted that the cost of potable water produced with the coupling scheme (HNR+HTMED) is relatively lower (0.76-0.89) US$/ m^3 and this result would strongly depend on the specific construction cost provided by the vendor. This water cost is competitive with that of coupling scheme (HNR+VTE-MED), even though the total water production (126,000 M^3/d) would be less than that of coupling scheme (HNR+VTE-MED). The main reason is that the used input data of specific construction cost for the HT-MED process is relatively favorable.

III.6.2. Coupling scheme of nuclear heating reactor NHR-200 with VTE-MED desalination process

The Coupling scheme of heating reactor NHR-200 with VTE-MED desalination process is shown in Fig. 1.2. Design of the coupling scheme is also accorded with the criteria for selection of optimum coupling scheme. On Figure III.2, it can be seen that the saturate steam with temperature of 125°C, generated in the steam generator is directly conducted to the first effect of the VTE-MED process and the condensate from the first effect was pumped back to the steam generator as its feed water. Therefore, the motive steam circuit works also as an additional barrier between reactor and desalination process. The coupling scheme (NHR-200+VTE-MED) possesses higher efficiency (GOR 20) due to higher top brine temperature (120°C) and therefore higher useful total temperature deference. Design parameters of the coupling scheme (NHR-200+VTE-MED) are listed in Table III.3 too.

Preliminary economic analysis for the coupling scheme of heating reactor NHR-200 with VTE-MED desalination process was performed. The estimated economic results are listed in Table III.4 also.

Based on the analysis results, It is noted that the cost of potable water produced with the coupling scheme (HNR+VTE-MED) is in the range of (0.8–0.92) US$/ m^3 and also strongly depends on the specific construction cost. The total water production may reach 160,000 M^3/d, which is larger than that of coupling scheme (HNR+LT-MED, 126 000 M^3/d). Therefore, If only the specific construction cost for the VET-MED process could be reduced by optimization of design of desalination system, the VTE-MED process coupled with NHR nuclear heating reactor would still occupy superiority in the competition.

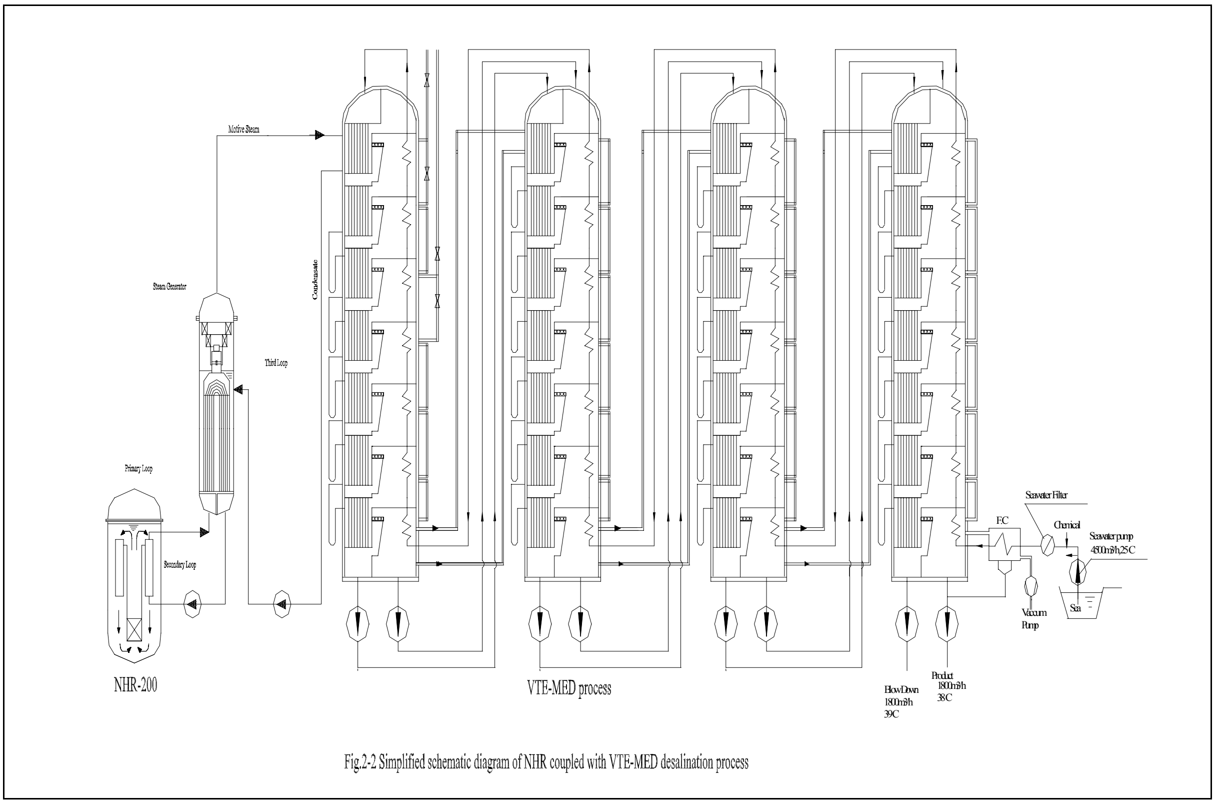


Fig. III.4 Coupling scheme of heating reactor NHR-200 with VTE-MED desalination process.

III.7. DEVELOPMENT OF THE COMPUTER CODE NHRVM-1

The results of economic analysis for desalination plant with nuclear heating reactor indicate that the investment of desalination system occupy the major part (more than 65%) of the total investment of the nuclear desalination plant. And the water plant fixed charge is also the major component (about 50%) of the total water cost. Therefore, improvement of thermal hydraulic performance and optimization of thermal hydraulic design of the seawater desalination system coupled with Nuclear Heating Reactor would be of great significance for decreasing in the produced water cost. Meanwhile the detail design data and the first hand materials necessary for understanding the thermal hydraulic performance (including heat transfer coefficient, heat transfer enhancement, thermal efficiency, GOR, operation characteristics etc.) of the selected VTE-MED process were not available on hand. An investigation program on the thermal hydraulic performance was implemented both experimentally and theoretically. The computer code named **NHRVM-1** was developed for analysis on the thermal hydraulic performance of the VTE-MED process coupled with NHR nuclear heating reactor.

III.7.1. Physical-mathematical model

Every effect of the VTE-MED seawater desalination system includes one evaporator and one pre-heater with tube-in-shell arrangement. The feed brine is pumped through the serial pre-heaters and is preheated to the required temperature by the source vapor outside the tube, and then, flows in the tubes of evaporators by gravity from the top effect to the successive effect.

At the inlet of evaporators, a properly sized distributor is designed for every tube to guarantee a homogenous film flow of brine on tube wall. Condensing source vapor at shell side evaporates the liquid film. The generated vapor is separated from liquid in the separator and demister located at the outlet of evaporator, and then is induced to the shell side of evaporator and pre-heater of the next effect as heat source. The residual brine flows through the throttle element for establishing pressure drop and temperature difference between successive effects, and then, enters the next evaporator. At the first effect, motive steam at designed temperature is supplied by nuclear heating reactor and the condensate flows back to the reactor system as feed water. Condensate in i-th effect accumulated as produced fresh water and added to the water output of (i-1)-th effect after throttling. The physical model of an (i-th) effect (a middle effect) is shown in Fig. III.5.

The number of effects in the desalination process is limited by useful temperature difference between motive steam temperature at the first effect and condensed steam temperature at the final condenser. For the present analysis, saturated steam of 123°C is provided by NHR-200 as the motive steam to the first effect, while the top brine temperature (TBT) reaches to 120°C. The outlet temperature of brine from the final condenser is designed as 35°C. In this case, the system may include 28 effects with temperature difference between successive effects being not less than 3°C. Based on the physical model, the Governing equations for the evaporators and Pre-heaters were established and presented.

During the solution of governing equations, the thermal properties of brine are thought to be constant at every integral grid. However, the saturation temperature and other properties should be updated according to new acquired concentration of brine before solving equations at the next grid. Flashing of brine and produced fresh water were considered.

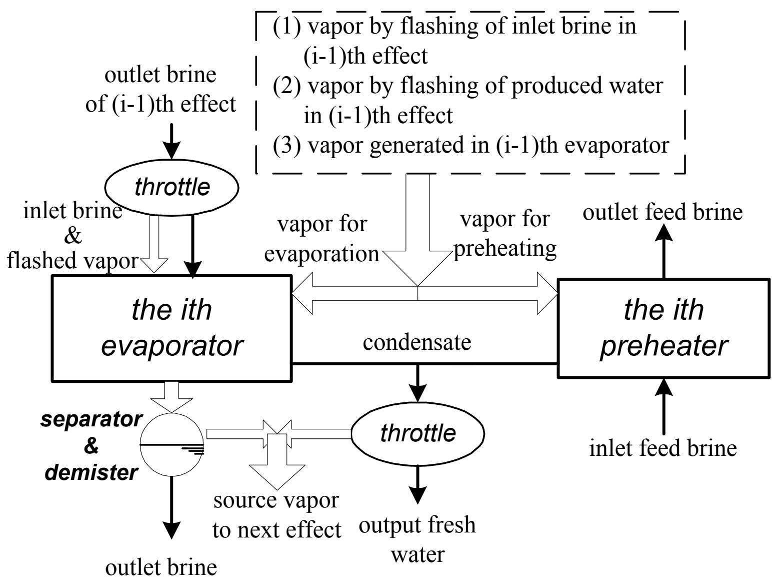


Fig. III.5 Schematic diagram of physical model.

The physical-mathematical model of VTE-MED system is solved numerically effect by effect. The purpose of numerical scheme is to obtain the areas of evaporator and pre-heater at every effect and the output flow rate of produced water at the conditions of determined inlet feed brine, source vapor supplied by NHR-200 and the distribution scheme of temperature difference.

The system with equal temperature difference distribution scheme was analyzed firstly, i.e., the total temperature difference between TBT and outlet feed temperature at the final condenser is uniformly distributed to all the 28 effects, so the temperatures of brine at each evaporator are predetermined. The outlet feed temperature of the i-th pre-heater is designed with the same value as that of current evaporator.

The thermal hydraulic model was proposed to analysis the performances of the system. The system, with TBT as high as 120°C, was composed of 28 effects and with GOR up to 20. The potable water production may reach 160,000 m^3/d. Two kinds of temperature difference distribution schemes (so called "scheme of equal temperature difference distribution" and "scheme of temperature difference distribution with in-section equal area of heat exchanger") were applied for analysis on thermal-hydraulic performances of the system.

Results of analysis on thermal hydraulic performance and discussions for both temperature difference distribution schemes were provided with several tens of Figures, including variation of thermal properties of brine in each effects, GOR in each effect, average heat transfer coefficients for evaporator and pre-heater at every effect, required heat transfer area for evaporator and pre-heater at every effect, influence of scaling on HTC, mass flow rate of heating vapor at each effect, heat flux in evaporators and pre-heaters, flow rate of flashing vapor at each effect and pressure drop at successive effects, et cetera. All the results were discussed in detail. Energy and Mass balance of the desalination system VTE-MED coupled with NHR-200 at nominal design condition was calculated using the developed code NHRVM-1. The calculation results were presented.

Based on the analysis results, it is indicated that there is no obvious difference in GOR and water production for the selected two kinds of temperature difference distribution schemes. However, design with temperature difference distribution of equal heat exchanger areas is favorable for tower structure design.

III.7.2. Improvement of the developed computer code for analysis on thermal hydraulic performance of the VTE-MED process coupled with an NHR nuclear heating reactor

The main Improvement of the developed computer code for analysis on thermal hydraulic performance of the VTE-MED process coupled with an NHR nuclear heating reactor was concentrated on the consideration for calculation of enhanced heat transfer. In the original version of the developed code NHRVM-1, the established module of the heat transfer tube of evaporators was for the straight smooth round tube. Therefore, all the reported results in the progress report of the contracted project (CRP-10243, Progress Report 2000) were based on the calculation for the straight smooth round tube. The correlation of heat transfer coefficient for the straight smooth round tube referenced to Chun K R, Seban R A [8]. In that case, at the top brine temperature of 120°C, the predicted heat transfer coefficient was about 4500 W/m^2-°C in the evaporator.

In the improved new version of the developed code NHRVM-2, Double fluted round tube was used as the heat transfer tube of the evaporators in order to improve the economics of the selected VTE-MED system, But the correlation of heat transfer coefficient for the double fluted tube is unavailable. So that the heat transfer coefficient data for the double fluted tube of IDT Metropolitan's concept design [6] was adopted as the comparative reference in the calculation and it is also listed the original report. The actual measured performance test data collected at Metropolitan's Test Unit has confirmed those values. At the top brine temperature of 230 ^{0}F (110^{O}C), the predicted heat transfer coefficient was 1861 Btu/h-ft^{2}-^{0}F (10533.3 W/m^{2}-^{O}C), while that actually measured was in excess of 1900 Btu/h-ft^{2}-^{0}F (10754.0 W/m^{2}-^{O}C). The comparison of IDT expected heat transfer coefficient data with the test results were given. It can be seen that for the total VTE-MED system the average heat transfer coefficient may reach up to 9000 W/m^{2}-^{O}C.

Another useful reference for the heat transfer coefficient data for the double fluted tube is that of the research results of INNOSHIMA of Japan [9, 10]. It was also adopted as the comparative reference for the calculation of heat transfer coefficient in the improved version of the developed code. the research results of INNOSHIMA was also referenced. It can be seen that at the top brine temperature of 120^{O}C, the measured heat transfer coefficient may reach at 10000 Kcal/h-m^{2}-^{O}C (11630 W/m^{2}-^{O}C) for the double fluted tube, while that the heat transfer coefficient is about 4500 10000 Kcal/h-m^{2}-^{O}C (5233 W/m^{2}-^{O}C), which means that compared with round straight tube, the enhancement factor of the heat transfer coefficient may enhanced by a factor of 2 for the double fluted tube. And it is also indicated that the enhancement of heat transfer gradually decreased with decreasing in brine temperature (increasing in system effect number).

In the improved module of heat transfer calculation in the developed code NHRVM-2, an enhancement factor (ranged from 2.0 to 1.4) was adopted, based on the IDT and INNOSHIMA research results above mentioned. The result for the double fluted heat transfer tube by using the improved module of heat transfer calculation in the developed code NHRVM-2 was provided. The detail calculation result of energy and mass balance was presented.

III.8. EXPERIMENTAL INVESTIGATION ON VTE-MED DESALINATION PROCESS

Considering the importance of necessary data and experience for the design of VTE-MED process, a experimental investigation program was implemented. A test system with 4 effects was designed, which was planed for simulating to any successive 4 effects of the 28 effects in the VTE-MED process by adjusting the operation parameters. Diagram of the test system, the Test Section with 4 effects, the design parameters of the test system were presented. Description of design of the test system was presented in detail

Construction work of the VTE-MED test unit built at the Institute of Nuclear Energy Technology (INET), Tsinghua University has been finished in December 2001, and the cold and hot adjustment work has been carrying out since January 2002.

Considering the complexity and difficulties to operate the VTE-MED test unit, as the first step the present adjustment work is done with fresh water. Real seawater desalination investigation will be carried out after the features of the test unit are known well and some operation experiences are achieved from the VTE-MED test unit.

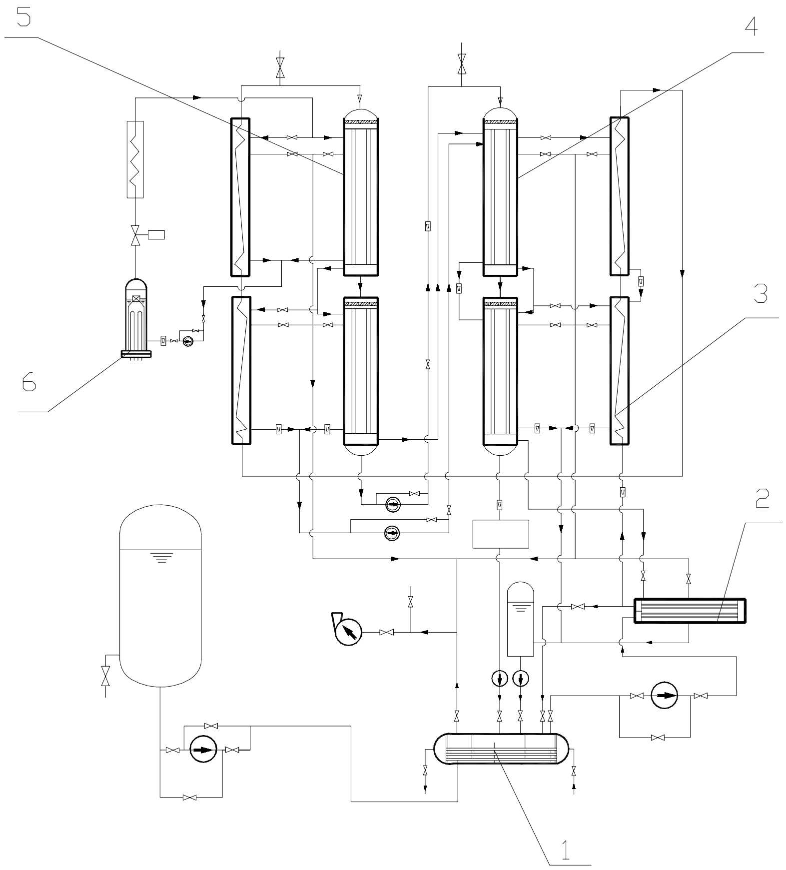
The objectives of the present experimental investigation are to gather experience and accumulate basic database for the design and operation of VTE-MED process:

– Conduct an investigation in operation performance of the VTE-MED process, including suitability of the desalination process to NHR reactor, system self-regulation and load following performance,

– Investigate the thermo-hydraulics of VTE-MED process, including heat transfer performance, characteristics of orifice plate, thermo-fluid dynamics in the system.

III.8.1. Test unit

The VTE-MED test unit mainly consists of motive steam system, raw seawater supply system, product fresh water circuit, vacuum system, seawater subcooling control system, equipment cooling system and measurement system. The main equipment includes an electrically heated motive steam generator, a raw sea water tank, 4 pre-heaters, 4 evaporation effects which are arranged in two parallel towers, a final condenser, auxiliary feed system, pumps, valves and some connecting pipes. Schematic diagram of the test unit is shown in Fig. III.6. The picture of the test unit is shown in Fig. III.7.



1 Seawater tank 2 Final condenser 3 Pre-heater
4 Main evaporater 5 Seawater heater 6 Steam generator

Fig.III.6: Diagram of the test unit.

The electrically heated steam generator is the power source, which supplies the motive steam for the test unit; it simulates the steam generator of the heating reactor.

The raw seawater tank is a heat exchanger whose shell side is used as seawater tank and the cooling water flows inside the heat transfer tubes, so the temperature of the inlet raw seawater can be easily changed in order to model the season change.

The pre-heater and its counterpart evaporator are not included into one vessel in the test loop, four pre-heaters are connected with its counterpart evaporators at the steam side (shell side) and their tube side are connected in series. Brine flows inside the tube, and the vapor condenses on the outside of the heat transfer tubes. The pre-heaters are installed stand-by-stand with their counterpart evaporators.

Four evaporators are installed in two towers and each tower includes 2 effects which are vertically stacked up in series, the exit of the top effect is the inlet of the below effect in each tower. At the bottom of each effect there is a brine reservoir, which collects the brine flowing down form the above evaporator and also to provide a region for separating vapor from brine. There are double shells around the reservoir, its outer shell is the pressure seal vessel and the inner shell is the shroud of the reservoir. On the top periphery of the inner shroud six windows with multi-layer net made of compressed stainless wire are used as the demisters. The vapor separated from the brine, passes through the demister window and flows downward along the alleyway between two shells to the shell side of the below effect and is used as heating steam for the effect.

The first effect is the seawater heat exchanger, which transfers the heat from the motive steam to the brine flowing and preheated in the pre-heaters. The following three effects can be used to simulate any successive 3 effects of the 28 effects in the VTE-MED process by adjusting the operation parameters.

The final condenser is also working as the vent condenser. Vapor from the last effect flows into the shell side of the final condenser, condenses on the outside of the heat transfer tubes and transfers heat to the seawater which flows inside the final condenser heat transfer tubes. As part of the production water the condensate flows into the distillate system. The preheated seawater flows upward via pump as feed water into the final stage pre-heater. The non-condensable gas separated from the vapor and extra vapor are vented from the final condenser.

A water sealed vacuum pump is used as the vacuum equipment, and all effects, pre-heaters, seawater tank, final condenser and steam generator are connected with the vacuum pump, the vacuum of every equipment can be adjusted separately.

The nominal design capacity of water production is about 4 m^3/d for the test unit, In order to simulate the process happened in actual seawater desalination plant as real as possible, the dimensions of the heat transfer tube and operation parameters in the test unit are same as ones in the real prototype plant, this results in the big size and the complexity of the test unit. The full height of the test unit is about 10m. The picture in Fig. 3.1-2 just shows the upper part of the test unit and the inter-tower pumps, brine tank, final condenser, the feed system and other auxiliaries are installed below the platform.

Fig. III.7: Picture of the VTE-MED test unit.

III.8.2. Description of the process in the test unit

The pretreated raw seawater is stored in the seawater tank from where it flows into the tube side of the final condenser via seawater pump No.1. The seawater is the coolant, which leads the shell side vapor from the effect No. 4 to be condensed into condensate. At the same time, the seawater temperature ascends a little in the final condenser. Leaving from the final condenser, the seawater will flow through all four pre-heaters in series, then its temperature arrives at the required value and it enters the first effect as make up water.

In the first effect the preheated seawater is accumulated on the top pond and distributed into the heat transfer tubes after passing through the spray nozzles at the heat transfer tube entrance of effect No. 1. The patented spray nozzles guarantee to form a thin downward flow seawater film along the tube inner wall, with its flow down the seawater absorbs the heat from the motive steam condensed on the outside of heat transfer tubes and at the same time the vapor is produced. The mixture of vapor and brine form a rapidly moving two-phase flow inside the tube, and it flows down into the reservoir on the top of effect No.2. Condensate of the motive steam on the shell side in effect No. 1 is pumped back to the steam generator and will be used again as its feed water.

In the reservoir on the top of the effect No.2 the mixture of brine and vapor is accumulated and the vapor is separated, the separated vapor passes through the demister and flows downward through alleyway into the shell side of the effect No.2. Because there is almost no pressure drop and heat loss, the vapor in the shell side of effect No.2 has the same temperature as in the reservoir region. The floor of the reservoir is just the inlet orifice plate and top tube sheet of effect No. 2 below. Passing through the orifice plat the brine pressure drops a certain value and at the same time the brine temperature will also decreases to its saturation value. Therefore, a heat transfer temperature difference occurs between the outside and inside of the heat transfer tubes. Then the heat transfer process described in first effect will be repeated again in effect No. 2. The shell side condensate is collected on the lower tube sheet as the produced fresh water and pumped into shell side of effect No.3 where it flashes and releases part of its latent heat, at last it arrives thermodynamic equilibrium with the vapor in the effect No.3.

The brine out of the heat transfer tubes of effect No.2 will flow into another brine pond at the exit of the effect No.2, the pond is the bottom cover of tower No.1. From there the brine is pumped into the brine pond on the top of effect No.3 via seawater pump No. 2 and the vapor separated from the mixture will flow upward automatically into the shell side of effect No. 3.

In effects No.3 and No. 4, the same brine flowage, heat transfer and fresh water production processes will repeat for other two times. At last, the brine and vapor separated in the brine pond of effect No.4 which is the bottom cover of tower No.2 will flow back to the seawater tank or final condenser along the different ways. In the test unit the produced fresh water will flow back to the seawater tank at last.

In every effect the steam or vapor is divided into two parts, most of them flows into the shell side of the evaporator bundle region, the remainder flows into the shell side of the feed pre-heater as its heating steam. Except for in pre-heater No. 1, the condensate in any stage pre-heaters flows to the next stage pre-heater via pump or gravity and flashes in the next stage pre-heater.

III.8.3. Commissioning and running of the test

Some experiences and results, which were obtained from the VTE-MED test, will be briefly presented in this section. There are four stacked evaporation effects arranged in two towers and nine different types of pumps, and the vacuum system and other systems interconnect each other in the test unit, also there is no any design and operation experiences in the fields. All these factors make it be a very difficult job to operate the VTE-MED test unit steadily at first.

Based on lots of cold and hot commissioning work and modifications to the different systems which revealed various issues during commissioning process, the VTE-MED test system can be operated steadily now among the seawater flow range from 1000 to 2600 kg/h, the temperature differences between two sides of the heat transfer tubes, which are necessary conditions in order to produce fresh water in evaporators, are also constructed and maintained during the tests.

The results show that the present VTE-MED test unit is successful both in design and in its stable operation. Design and operation experiences have been obtained. During the desalination process tests the first big difficulty was how to maintain the make-up seawater flow steadily. In actual seawater desalination plant, because the make-up seawater is pumped into the desalination equipment from a seawater pool whose volume is large enough. And the plant pressure would not impact the pressure of feed water. Therefore, to maintain stable make-up seawater flow is not a big problem. But in the VTE-MED test unit, the make-up seawater is pumped from a limited volume seawater tank which is not open to atmosphere, so the seawater tank pressure and water level were greatly impacted by other systems if the feedback problem, particularly the vacuum effectiveness was not considered and solved properly, although these issues maybe were not present in actual seawater desalination plant.

The major measures to overcome the unstable make-up seawater flow were (1) to add pumps between exits of brine and fresh water flow ways and raw seawater tank; (2) to adjust the pump rolling speed by frequency adjusters which were controlled by feedback signals of seawater levels in tank and seawater ponds.

As described above, the double tower structure VTE-MED test unit can simulate the actual seawater desalination plant designed by INET, but it causes a lot of problems for the experimental research. For instances, the seawater pump No. 1 pumped the seawater from limited volume seawater tank to the top of the first tower, then the seawater flowed downward to the bottom of the tower from where the brine was pumped by seawater pump No. 2 to the top of the second tower, therefore there were many factors which affected the balance of mass flow rates between two towers.

The experiment study also shown that the decrease of the brine temperature would become another big problem in actual plant as the brine was pumped over several decade or a hundred meters from the bottom of the first tower to the top of the second tower if the thermal isolation was not very well. The temperature drop of 2 to 3 degree centigrade means the loss of production water ability of an effect evaporator. Usually the brine temperature dropped more than vapor temperature did when they were transferred from one tower to another in the test, the inconsistent temperature drop of brine and water means some extra heat from the vapor will be used to increase the brine temperature, not to produce the vapor and fresh water in effect No. 3.

The vacuum system was also very important in the VTE-MED test unit, the incorrect intake positions of non-condensable gas exhaustion tubes resulted in the failure to exhaust the non-condensable gas completely at first. Based on the experimental results the modification plan has been scheduled. The vacuum in the system also caused some problem of differential pressure measurement meters before some improvements were performed. The vacuum would lead to exhaustion of pressure transfer liquid in the pressure transducer tube, so that the meters could not correctly feel the pressure change.

III.8.4. Preliminary experiment results

Pressure drop and temperature distributions in the system, mass flow rate of steam, brine and fresh water were recorded. All the recorded data were presented and provided with figures and data documents, which provide the data base for verification of the thermal hydraulic design of the evaporators and pre-heaters.

The VTE-MED test unit was operated stably under different pressure level. This shows that the test system can used to simulate to the behavior in deferent section of a successive effects of a desalination plant with 28 effects.

The orifice plate, which is located at the entrance in every effect, is very important in the VTE-MED unit. When the brine from the seawater pool flows through them, the pressure of the brine will decrease and the seawater will flash with the decrease of the pressure, so some bubbles appear in the brine and the brine become the foaming flow. The foam fills the space between the plate and the bundle sheet of the heat transfer tube, so that the foaming flow is regulated by an orifice plate and flows into the individual tubes evenly, the orifice plate acts as an efficient distribution plenum for the brine. The pressure drop and mass flow rate across the orifice plate were recorded, which provide the data for verification and correction of the design of flow resistant coefficient of the orifice plate.

More experiment investigation should be performed in the future. It should be noted that the data accumulated up to now are still not enough to be used to verification of the developed code.

III.9. ADDITIONAL ANALYSIS ON PERFORMANCE OF VTE-MED COUPLED WITH NHR REACTOR

In order to investigate the optimization of the design parameters of the VTE-MED system coupled with NHR-200, effects of system design parameters (including system effect number, material used of heat transfer tube, geometric size of heat transfer tube, feed seawater temperature and top brine temperature) on performance of the VTE-MED system were analyzed by using the developed code. Results of the analysis were presented in the original technical report.

Based on the results of the additional analysis on thermal hydraulic performance, the recommended design parameters for the selected VTE-MED system are abstracted and listed in Table III.5.

TABLE III.5:
RECOMMENDED DESIGN PARAMETERS FOR THE SELECTED VTE-MED SYSTEM

Parameters	**Unit**	**Values**
System effect number		28-30
Material of heat transfer tube		SS1.4565 Or Aluminium alloy
Diameter of heat transfer tube	mm	50
Length of heat transfer tube of evaporator	m	3.0-4.0
Thickness of the heat transfer tube wall	mm	0.35 for SS1.4565 1.0 for Aluminium alloy
Top brine temperature	℃	120

III.9. GENERAL CONCLUSIONS

(1) Results based on the preliminary analysis indicate that: The optimum coupling scheme between NHR-200 and MED process for heat only mode is Scheme A: NHR-200+ Steam generator+ MED process.

(2) Analysis on coupling schemes with hybrid desalination process shown a possible approach to decrease water cost.

(3) Optimization analysis on coupling parameters of NHR heating reactor established a set of optimum parameters.

(4) Selected VTE-MED process coupled with NHR-200 is a higher efficiency system.

(5) A VTE-MED process test system was build up, commissioning and running for experimental investigation on thermal-hydraulic performance of the selected VTE-MED process.

(6) Development of a computer code for analysis on thermal-hydraulic performance of the stacked VTE-MED process was established.

(7) Thermal-hydraulic performances of stacked VTE-MED process coupled with NHR-200 were analyzed using the developed code.

(8) Based on the results of the investigation, integrated seawater desalination plant with nuclear heating reactor NHR-200 coupled with high temperature MED desalination process with stacked VTE-MED process with tower design was recommended.

(9) It should be noted that all the water cost analyses performed during the period (1998-2001) of this contracted project and all the results of the calculated water cost presented in the relevant progress reports of CRP-10243 were based on the old economic input data, which were established before nineties in accordance with the international market price. So that the calculated water cost was about 0.8–1.0 US$/$m^3$ that is higher than the present updated data.

(10) From the year of 2002 on, especially via the "Economic research on the Desalination Plant with NHR reactor coupled with high temperature VTE-MED process and the site-specific investigation", the later on economic analyses were based on innovative technical design and domestic market price in China. And therefore the water cost was also updated. In that cases, the water cost is decreased to about 0.6 US$/$m^3$. (See relevant report of CRP-11863 project).

ANNEX IV

INVESTIGATION OF FEEDWATER PREHEATING EFFECT ON RO PERFORMANCE

Mohamed M.M. Megahed,
Nuclear Power Plants Authority, Cairo, Egypt

IV.1. BACKGROUND

In view of the limited Egyptian resources of both primary energy and fresh water, Egypt has been considering for a number of years the introduction of nuclear energy for electricity generation and seawater desalination. In this regard, the Nuclear Power Plants Authority (NPPA) participated actively in a number of national and international studies to investigate the prospects of using nuclear energy for the simultaneous production of electricity and potable water [IV.1-3]. These studies concluded that:

- In general, there are no technical impediments to the use of nuclear reactors for the supply of energy to desalination plants;
- The costs of electricity and desalted water for the most economic fossil and nuclear driven power/desalination plants were in the same range;
- Nuclear reactors which offer the best prospects for near term commercial deployment are: PWRs, PHWRs and NHRs;
- The most economic desalination process seems to be contiguous RO plants with preheated feedwater.

In response to the increasing interest in nuclear desalination, the IAEA has performed a two-year Options Identifications Programme (OIP). The OIP identified three possible approaches to the demonstration of nuclear desalination technology, namely [IV.3]:

♦ Demonstration through the design and construction of a nuclear desalination facility.
♦ Demonstration of nuclear desalination as an addition to an existing NPP.
♦ Demonstration based on simulation of nuclear desalination.

The OIP recommended also a number of intermediate steps to reduce unknowns and risks aiming at gradual, partial and progressive confidence building. The recommended intermediate steps for RO were [IV.3]:

⇒ Small scale preheated seawater desalination with RO;
⇒ Small scale RO integrated with NPP;
⇒ Large scale RO integrated with a fossil-fueled power plant.

Based on the above findings and recommendations, the fact that Egypt does not posses any nuclear power plants, and the existing financial limitations, NPPA decided to construct an experimental RO facility at its site in El-Dabaa to validate the concept of feedwater preheating. The results of this experimental work could have a strong influence on how the international nuclear desalination community perceives the value/benefit of feedwater preheating, and hence there is a common international interest in this project. Therefore, NPPA proposed in January 1998 the research project “Investigation of feedwater preheating on RO performance” as part of the IAEA coordinated research project “Optimization of the coupling of nuclear reactors and desalination systems”. In May 1998, the IAEA agreed to fund the project and a research contract was concluded with NPPA for one year and subsequently, renewed to the end of 2003.This report summarizes the work carried out under this research contract.

IV.2. JUSTIFICATIONS

Although the proportional relationship between feedwater temperature and membrane permeability is well known, the idea of utilizing the condenser's cooling water as a source of feed for RO systems did not appear until early 1994 [IV.4,5]. This concept was adopted and investigated by the IAEA in all subsequent studies [IV.2,3,6]. These and other studies [IV.1-7] have shown that there is a potentially significant economic and performance benefit through the combined effects of feedwater preheating and system design optimization. These conclusions have been drawn from analyses and preliminary design studies without any experimental validation.

Experimental validation is of extreme importance in the confidence building process, particularly when other experts [IV.8] argue that elevated temperatures may result in higher product water salinity, more rapid membrane fouling, greater membrane compaction, reduction in membrane lifetime and that saving in total water cost by elevating temperature from 15~18 C to 30 C would be in the range of 3% only.

IV.3. OBJECTIVES

In view of the possible role of RO desalination technology in any future Egyptian nuclear desalination program and the need to validate the concept of RO feedwater preheating, NPPA has decided to carry out this research project, with the following objectives in mind:

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

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

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

IV.4. SCOPE OF WORK

The work plan for the duration of the research project (1 July 1998–31 December 2003), included the following activities:

(i) Design of Experimental Unit
(ii) Development of Detailed Experimental Program
(iii) Preparation of Technical Specifications

(iv) Call for Bids, Bid Evaluation and Contracting
(v) Construction and Commissioning of the experimental Facility
(vi) Carrying Out the Experimental Program.

The duration of the experimental program was expected to be 3 years divided into three Phases (I–III). Each Phase is based on a particular membrane make, with a matrix of fixed operational conditions such as:

- Feedwater Temperature
- Feed Pressure
- Chemical dosing and type
- Operating time
- Etc.

The experimental program was developed with IAEA technical assistance in September 1998, the design of the experimental facility was completed in December 1998, and preparation of the technical specifications and tender documents was completed in May 1999. It was envisaged that the experimental program could start at the beginning of January 2000. However, delays were encountered, in the following activities:

1. *Contracting*: bid evaluation took longer than expected, because there were many details that needed clarifications from the bidders. Finally, when the financial envelopes were opened, it was found that the prices of the successful bidders exceeded the allocated budget. Bidding process was repeated and contract concluded with a main contractor at the beginning of July 2000.
2. *Construction*: The construction work started in January 2001 due to delays in submitting the final drawings and designs. The situation was further complicated by devaluation of the Egyptian pound (currently US$ 1= LE 6.4 compared to US$ 1= LE 3.4 in 2000). This lead to difficulties in importing the equipment and increase in prices. In addition, some problems occurred during the execution phase between NPPA and the Contractor, which lead to stopping the civil work several times and financial problems between the main contractor and the Electro-mechanical sub-contractor. As a result the construction of the experimental facility was not completed.

The current status of the project is summarized below.

1- Design and Engineering	Completed
2- Civil Work	90% completed
3- Beach Wells	Completed
4- Electrical Power Supply Infrastructure	Completed
5- Procurement	Completed, equipment stored on site
6- Installation of Electro-mechanical Equipment	Pending
7- Commissioning	Pending

There are attempts to complete the experimental facility in the near future. However, despite the fact that the experimental facility was not completed during this CRP, NPPA remains committed to making the results of the experimental program available to Member States. Details of technical and scientific work carried out within the CRP are outlined below.

IV.5. STATUS OF THE PROJECT

IV.5.1. Review of literature

The main objective of the literature review is to provide the state-of-the art information - based on theoretical analysis and operational experience – to enable: (a) the design of a trouble free, efficient and flexible test facility; (b) development of experimental program; and (c) analysis of the test results.

Apart from CANDESAL and IAEA reports, the team could not find any relevant publications on high temperature RO, theoretical modeling of temperature effect or data of similar experimental work.

IV. 5.2. Design of experimental unit

In order to design a trouble-free, efficient and flexible test facility, NPPA carried out a detailed screening and qualifying process of potential project consultants. In April 1998, qualified consultants were invited to submit technical and financial proposals to design the test facility, prepare technical specifications, supervise construction and commissioning, and supervise training of personnel. Subsequently, in June 1998 one of the most reputable and experienced desalination consulting firms in the Arab World, the Consulting Engineering Company (CEC), was selected to carry out the required engineering services.

Since the concept of preheating is proposed as one of the options for coupling RO desalination systems with nuclear power plants, it is important to ensure that the experimental RO facility is properly designed to perform the intended tasks. Hence, NPPA requested expert assistance from the IAEA in reviewing the preliminary Design Report submitted by CEC. Consequently, an expert from CANDESAL Enterprises Ltd. carried out an expert mission in the period 18–27 September 1998 to assist in the design review. The cooperation between NPPA, CEC and IAEA resulted in successful completion of the design. The details of the experimental facility were published for the benefit of the international community [IV.9]. Major design parameters are outlined below.

IV.5.2. 1 Number and size of membranes

To determine the minimum number of membranes needed to ensure statistical relevance of the results, a statistical analysis was carried out to determine sample size for any one type of membranes tested. The analysis indicated that a sample size of five membranes in parallel is optimal. The bases of the analysis were as follows:

- Membranes manufactured follow the normal distribution.
- Sample size was varied from 1 to 9.
- Level of confidence was calculated for each sample size.
- Relevance tolerance was set at 5%.

The size of the experimental unit will be equal to the number of parallel membranes multiplied by the capacity by the particular membrane to be tested. Because the objective of the experimental facility is to investigate membranes performance, rather than the production of potable water, it is important for economic reasons to have the smallest capacity capable of representing performance characteristics under investigation.

Commercial membranes are produced in various sizes; the most common of which are the 4” and the 8” diameter membranes. From the economic and practical point of view, the 4” membrane seems to be a more attractive choice for the following reasons:

- The cost of the experimental facility is likely to be reduced significantly. Although the cost of membranes would be reduced, that is likely to be one of the least significant impacts. More significant is the reduction in feed flow required, and hence a reduction in the size of high pressure feed pumps, storage tanks, piping, membrane cleaning system, etc. In general, it can be said that the entire facility would be reduced in size. In addition, significant reductions could be expected in O&M (operating and maintenance) costs for a system based on 4" membranes.
- More flexibility is likely in the selection of membrane configurations and in the number of membranes used in the experimental facility.
- Reconfiguration of the experimental facility to allow changeover from one membrane type to another is likely to be easier with smaller diameter membranes.

However, the primary risk introduced by the use of smaller diameter membranes is the potential that the performance characteristics of the 4" membranes may not be representative of the performance characteristics of 8" membranes under operating conditions expected during the experimental program. In order to assess this risk, a number of "membrane equivalency comparisons" of the performance characteristics of 4" and 8" membranes have been made using the ROSA code. For a fixed feed flow and recovery, the highest feed pressure occurs at the lowest temperature. For any given feedwater TDS, an iterative analyses is required to establish the specific feed flow and recovery that will correspond to the maximum allowed operating pressure (69 bar), without violating minimum brine flow requirements. Feed flow and recovery were then held constant in each case, and the membrane performance characteristics were calculated over the temperature range 20–45°C, the results are shown in Figures IV.1 and IV.2 below.

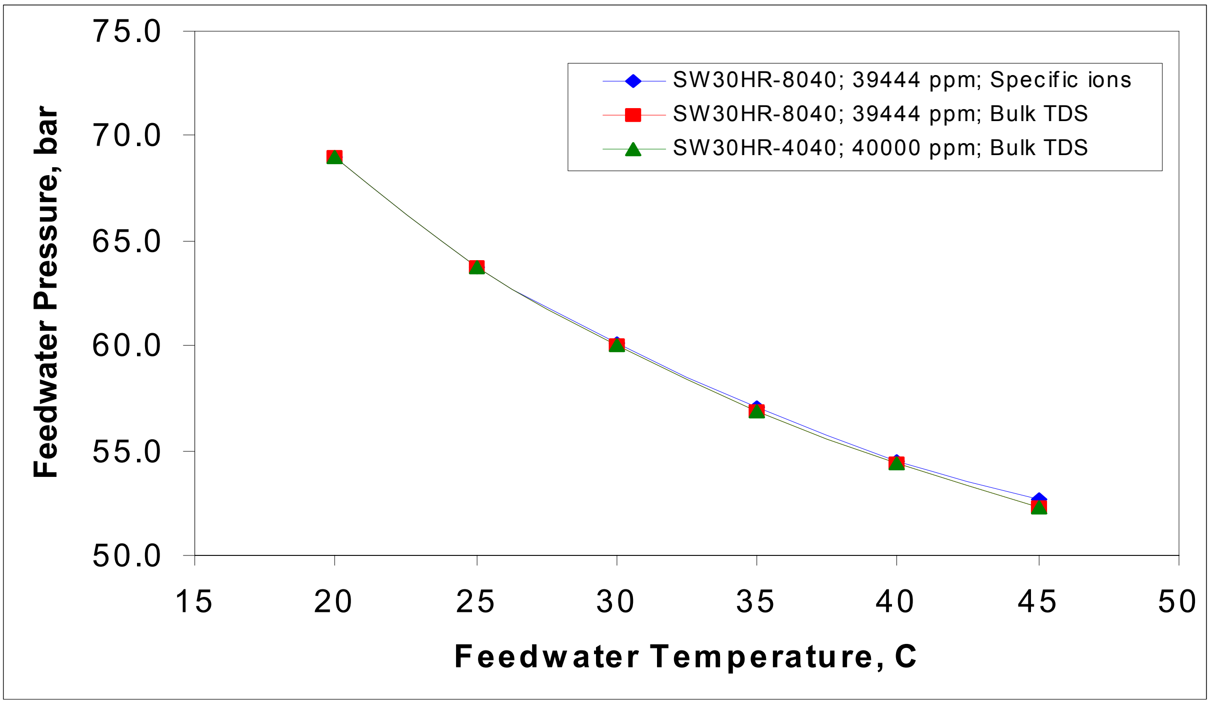


Figure IV.1: Comparison of Feedwater Pressure versus Temperature.

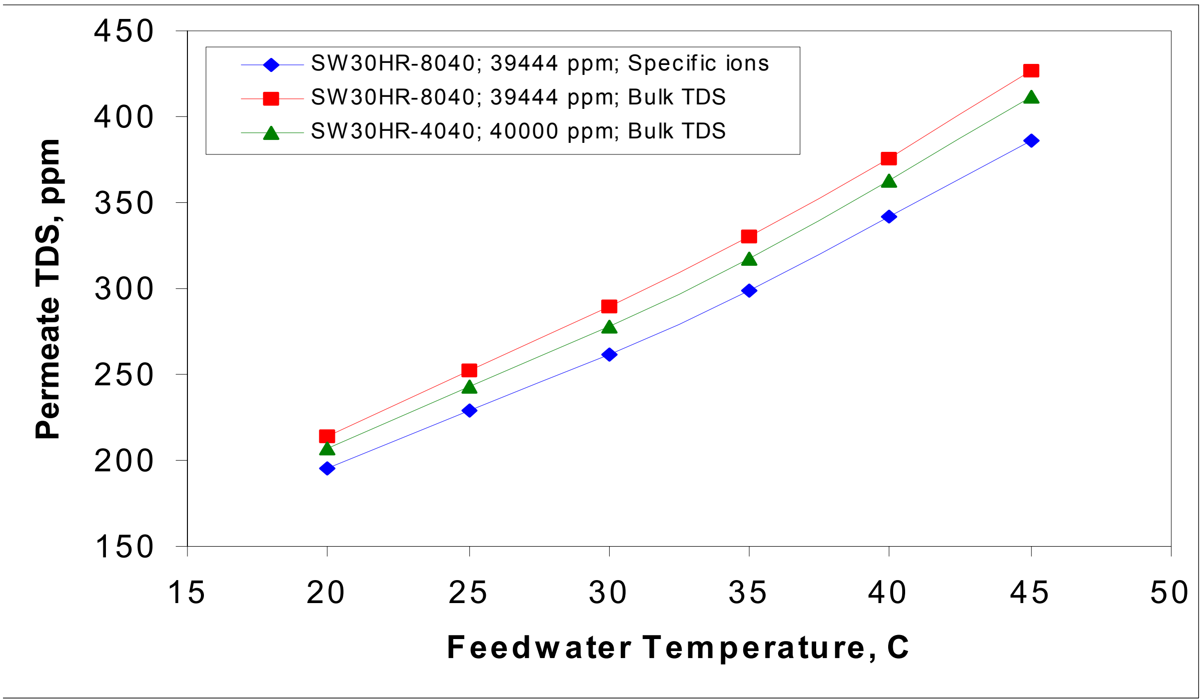


Figure IV.2: Comparison of Permeate TDS versus Feedwater Temperature.

As can be seen from Figure IV.1. the feedwater pressure as a function of temperature is essentially identical for the 4" and 8" membranes. The permeate TDS for the 4" and 8" membranes, as shown in Figure IV.2, is not identical, albeit, very close. Other parameters of interest in the comparison of the 4" and 8" membranes are shown in Table IV.1.

The conclusion that can be drawn from these results is that the 4" membrane provides a very close equivalence to the 8" membrane in its essential performance characteristics. It has virtually the same pressure profile with temperature and a very similar permeate TDS profile with temperature. Of even more significance, it has a nearly identical flux through the membrane surface, as would need to be the case in order to expect equivalent performance characteristics. However, the feedwater flow rate and the daily energy consumption are significantly less for the 4" membrane, as shown in Table IV.1. Therefore, the experimental program can be carried out with 4" membranes without adverse impact on its representation of essential performance characteristics. Moreover, potentially significant savings in program capital and O&M costs can be realized.

TABLE IV.1: COMPARISON BETWEEN 4" AND 8" MEMBRANE CHARACTERISTICS

Membrane	Feed Flow	Average Flux	Energy Consumed*
	m^3/h	$l/m^2/h$	kWh/d
SW30HR-8040	4.18	19.3	256.4
SW30HR-4040	1.146	19.6	70.4

**** Based on ROSA calculation at 20°C without energy recovery***

IV.5.2.2. Unit capacity

Different commercial membranes have different performance characteristics. In particular, they have different nominal permeate flows and conversions at some standard test conditions, as shown in Table IV.2. Because the high-pressure pump should accommodate the different commercial membranes to be tested, it will be based on the highest antfeed flow.

The short term experimental program shall be based on the three 4” spiral wound membranes manufactured by Filmtec, Fluid Systems and Hydranautics. The rational for this selection is:

- The capability to operate at high feed water temperature (45 ^oC).
- Similar permeates flows and recovery ratios. Thus, limiting the operational range of the HP pump, this will facilitate the pump selection.
- Similar dimensions and materials. This should facilitate racking requirements and changing from one commercial membrane to another as well as direct comparison of performance.

Although Dupont membranes allow lower feedwater temperature, they constitute about 40% of the worldwide RO membranes. Therefore, Dupont membranes should be tested at a later stage.

Based on information provided in Table IV.2 and the results of the statistical analysis, the nominal size of the experimental unit will be about 21 m^3/d.

TABLE IV.2: PERFORMANCE CHARACTERISTICS OF SOME COMMERCIAL MEMBRANES

Manufacturer		Dupont	Filmtec	Fluid Systems	Hydranautic	Toyobo
Type		Hollow Fiber	Spiral Wound	Spiral Wound	Spiral Wound	Hollow Fiber
Model		6410	SW30HR-4040	TFC 1820SS	SWC1-4040	HR3155
Material		Aramid HF	Thin Film Composite	Composite Polyamide	Composite Polyamide	CTA
Dimensions - Diameter - Length	 mm mm	 154 587	 99 1016	 102 1016	 100 1016	 104 400
Permeate Flow	m^3/d	2.46	3.8	4.2	4.2	0.4
Salt Rejection	%	99.2	99.4	99.3	99.6	99.4
Test Conditions - Salinity - Pressure - Recovery - Temperature	ppm bar % ^oC	35000 69 30 25	35000 55.2 8 25	32800 55.2 7 25	32000 55.2 10 25	35000 55.2 30 25

IV.5.2.3. Configuration of the test facility

The test facility consists of two identical units: one unit operating at ambient seawater temperature and the other with preheated feedwater at 25, 30, 35, 40 and 45°C, as called for by the experimental sequence. This configuration is considered practical with 4” membranes, and has the benefit of giving direct comparison of performance characteristics for the preheated and no-preheated cases at all values of preheat temperature. The test facility consists of the following main components:

(a) Beach wells and pumps
To ensure clean feedwater with minimum pretreatment requirements and lower operational costs, beach wells will be used for the feedwater rather than open seawater intake. Although the required feed capacity is small, two beach wells will be used (one working and one standby) in order to ensure wells reliability and durability. The beach well pumps shall be operated intermittently in cases of unit’s lower capacity. As for the filter feed pump, it shall be provided with variable speed motor or hydraulic coupling to cater for the required capacity variation.

(b) Pretreatment system
The pretreatment system is designed to allow for the various pretreatment requirements for the different commercial membranes to be tested. This includes the various chemicals and dosing points recommended by the manufacturers.

(c) Water heating system (for one unit only)
Direct heating of the raw seawater is not recommended due to scale formation problems. Therefore, the feedwater will be heated by freshwater/seawater heat exchanger. The hot fresh water shall be obtained from a diesel oil fired or electric water heater (not decided yet). To reduce fuel consumption during continuous operation, the hot brine and permeate shall be used to preheat the feedwater, utilizing permeate/seawater and brine/seawater heat exchangers. This will give the following advantages:

- Reduction of fuel consumption in the water heater.
- Reduction in permeates temperature.
- Reduction in the brine temperature before disposal which will be advantageous from the environmental point of view.

(d) High pressure pump with energy recovery and hydraulic coupling
The experiments involve different types of membranes, requiring different operating pressures and feed flows. Therefore, the high-pressure pump is coupled with a hydraulic coupling to obtain the required pressure-flow. For fine-tuning, throttling and back-pressure valves are provided. To recover the brine kinetic energy, energy recovery turbine (ERT) is provided.

(e) Racking
As mentioned in section 5.2.2 above, the short-term experimental program is based on three 4” spiral wound membranes manufactured by Filmtec, Fluid Systems and Hydranautics. To accommodate the different dimensions and piping arrangements of the membranes and to reduce the time required for switching to a different experimental Phase, each unit is provided with three racks with piping and connections suitable for a particular membrane. The racks are required to be designed for easy operation, maintenance and membrane replacement when necessary.

(f) Other systems common to commercial RO are also included, such as:

- Cleaning/flushing system
- Post-treatment system
- Chemical treatment system
- Raw and product water tanks
- Etc.

IV.5.3. The experimental program

The specific approach to application of RO technology that this experimental program is intended to investigate is based on several important principles. These include:

- Operations at temperatures above ambient seawater temperature results in increased permeate production relative to that same plant operated at seawater temperatures.
- Operation at the highest pressure allowed by the membrane design limitations results in the most efficient operation. Permeate production is maximized, and design configurations can be establish which allow such operation without exceeding permeate TDS limits.
- Operation at high pressure results in unit energy consumption being minimized. Although power consumption is increased to pump feedwater to higher pressures, energy consumed per cubic meter of permeate produced is reduced.

The experimental program is intended to generate data that can be used to validate these performance characteristics. It is assumed that test runs for the experimental program do not begin until the experimental unit has completed all its commissioning trials and has demonstrated the ability to maintain a stable operating state.

IV.5.3.1. Test sequence and timing

The planned sequence and timing of testing (Steps 1–5 below) is provided below. Actual test sequencing and timing will need to be adjusted as experience is gained throughout the experimental program. In the discussion that follows, Train 1 is taken to be the unit that operates at ambient seawater temperature. Train 2 is the unit in which the RO membrane feedwater will be preheated.

Step 1

The traditional approach to RO system design is to minimize the feed pressure, consistent with the required permeate flow. In the first phase of testing, a set of “reference” operating profiles would be established consistent with this approach. These tests would hold feed flow and recovery constant while feed pressure was allowed to drop with increasing temperature.

Step 2

The goal of this step is to collect data for operation at the maximum operating pressure and a fixed feed flow for all values of RO feedwater preheat. Data collected from this step should give an indication of the performance benefits achieved due to feedwater preheat. The tests performed in this step hold feed flow and pressure fixed, allowing recovery to vary with temperature.

Step 3
In order to assess the impact of feed pressure, it is necessary collect data at various feed pressures below the maximum allowed membrane pressure. In this step, data is collected at the first of these reduced pressure plateaus.

Step 4
The lowest operating pressure normally used in large seawater RO systems is on the order of 55 bar. Data is collected during this step to represent operation at that pressure.

Step 5
Having completed data collection at a feed pressure of 55 bar, the system should be returned to a steady state operating condition. This steady state operating condition is one for which:

- Feed flow and recovery ratio for both Trains 1 and 2 remain fixed at values that allow operation with a feed pressure of 69 bar.
- Train 1 operates with a feed temperature at ambient seawater temperature.
- Train 2 operates with a feed temperature of 45°C.

IV.5.3.2. Test cycle

On the assumption operating parameters, including feed temperature, can be changed and a stable plant condition reached within a time period of 12 hours, the above test sequence should take on the order of 24 days to complete. Having completed one full test cycle (Steps 1–4) and returned the plant to maximum pressure and temperature operating conditions (Step 5), operation should continue for the balance of the month (approximately 6–7 days).

Following one full month of operation (including testing), the next test cycle (Steps 1–5) should be carried out. This pattern should be repeated for the duration of the current test phase. In accordance with the current schedule, the experimental program is to consist of 3 phases, each phase taking approximately one year and consisting of testing with one of the three membranes being evaluated. Following this schedule will provide a series of data sets taken at monthly intervals over the duration of a year, for each membrane type. This test cycle is illustrated in Figure IV.3, which shows one complete monthly test cycle. This cycle is repeated each month for the full year of testing.

IV.5.3.3. Data collection

At each test plateau a complete set of data should be taken, consisting of:

- Feedwater temperature.
- Feedwater TDS (based on conductivity and water analysis).
- Feedwater SDI, before and after pretreatment.
- Feedwater flow rate.
- Feedwater pressure
- Permeate flow rate.
- Permeate TDS
- Permeate pressure
- Permeate temperature
- Brine flow rate

Figure IV.3: One-Month Test Cycle, Steps 1 through 5.

- Brine pressure
- Brine temperature
- Brine TDS
- Chemicals used for feedwater pretreatment
- Electrical consumption
- Other process parameters as per the Design Report by CEC.

IV.5.3.4. Data analysis

Data analysis requirements will be established, to some extent, by the nature of the data taken and its indication of membrane performance characteristics. As a minimum, essential membrane performance characteristics should be derived from the data.

IV.6. COOPERATION WITH MEMBER OF THE CRP

As mentioned earlier, it was important to ensure that the experimental RO facility is properly designed to perform the intended tasks. Therefore, NPPA requested expert assistance from the IAEA in reviewing the preliminary Design Report. Consequently, an expert from CANDESAL Enterprises Ltd., which introduced the concept of feedwater preheating in 1994, carried out an expert mission in September 1998 to assist in the design review.

Another aspect of cooperation was training of NPPA personnel. The ability of personnel involved in the experimental program to operate and maintain the test facility, as well as to collect and analyze experimental data is of paramount importance to the success of the program. Training on operation and maintenance is included in the scope of work of the construction stage i.e. it will be provided by the contractor. However, data collection and analysis could only be obtained through practical (on-the-job) training in a similar facility with the context of design optimization of coupling the nuclear power and desalination plants. Therefore NPPA approached the IAEA through another project for a fellowship to provide on-the-job training on coupling optimization of nuclear power and RO desalination plants. The fellowship was approved by the IAEA and one of NPPA engineers stayed for 3 months (1 February–30 April 2001) at CANDESAL facilities in Canada. The training program included formal training through lectures and/or relevant and On-the Job Training, through participation, as a team number, in the experimental program that was going on at CANDESAL.

IV.7. CONCLUSIONS

1. All pre-construction activities have been completed successfully, including design of the experimental facility, preparation of the technical specifications and bidding as well as development of the experimental program.
2. Contract was awarded to a main Contractor in July 2000. However, due to reasons beyond NPPA construction work was not completed during the CRP.
3. There are attempts to complete the experimental facility in the near future. However, despite the fact that the experimental facility was not completed during this CRP, NPPA remains committed to making the results of the experimental program available to other Member States.
4. A useful cooperation between NPPA and CANDESAL has been achieved in the design of the unit and training of personnel. Further cooperation with other countries/institutions within this CRP is expected when the construction is completed.

REFERENCES

[IV.1] ACADEMY OF SCIENTIFIC RESEARCH AND TECHNOLOGY, “Utilization of Nuclear Technology in Seawater Desalination”, Project No. 53-1992/97, Reports 1–4, Cairo (1995–1997).

[IV.2] INTERNATIONAL ATOMIC ENERGY AGENCY, Potential for Nuclear Desalination as a Source of Low Cost Potable Water in North Africa, IAEA-TECDOC-917, Vienna (1996).

[IV.3] INTERNATIONAL ATOMIC ENERGY AGENCY, Options Identification Programme for Demonstration of Nuclear Desalination, IAEA-TECDOC-898, Vienna (1996).

[IV.4] HUMPHRIES, J.R., MIDDELTON, E., “CANDESAL: A Canadian Nuclear Desalination System”, paper presented at US/Middle East Joint Seminar on Desalination Technology”, CI-9404, Washington D.C, Candesal Enterprises Ltd., Ottawa (1994).

[IV.5] CANDESAL INC., “CANDESAL An Advanced Water Energy Production System: Egypt Application Study”, Report CI-9403. Ottawa (1994).

[IV.6] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the Economic Evaluation of Cogeneration/Desalination Options: A User’s Manual, Computer Manual Series No. 12, IAEA, Vienna (1997).

[IV.7] HUMPHRIES, J., et al, “A Technical and Economic Evaluation of the CANDESAL Approach in Indonesia Using Reverse Osmosis and Waste Heat from the CANDU-6 Nuclear Power Plant”, Report, rev. 1, Candesal Enterprises Ltd., Ottawa (March 1998).

[IV.8] FURUKAWA, D.H., “Desalination Processes and Technology”, Annex I.2, pp. 82–93, IAEA-TECDOC-898, Vienna (1996).

[IV.9] EL-SAIE, M.H.A., et al, “Experimental RO Facility to Study the Heating Effect of Raw Water on the Varying Main Parameters”, Desalination Vol. 134, pp 63–76, 2001.

ANNEX V

PERFORMANCE IMPROVEMENT OF HYBRID DESALINATION SYSTEMS FOR COUPLING OF NUCLEAR REACTORS AND DESALINATION SYSTEMS

P.K. Tewari,
Bhabha Atomic Research Center (BARC), India

V. 1. OBJECTIVES

The objective of the research project is to study the behavior of Multi-Stage (MSF), Reverse Osmosis (RO) and Low Temperature Evaporation (LTE) desalination plants under different operating conditions and to utilize data for improving the performance of hybrid MSF-RO plant coupled to PHWR and LTE plant coupled to a nuclear research reactor.

RO

(i) Study the performance of SWRO using sea water feed at different temperatures
(ii) Analysis of data collected on different membrane elements and projection (iii) Collection of data in the 100 m^3/d RO pilot plant

MSF

(i) Operation of 425 m^3/d MSF plant and collection of data
(ii) To study the performance of MSF plant under different operating conditions of sea water temperature
(iii) Analysis of data

LTE

(i) To study the performance of LTE plant at different temperatures of hot water required for sea water evaporation
(ii) Coupling of LTE plant set up at CIRUS

Utilization of the data for study of coupling aspects of the hybrid MSF-RO nuclear desalination demonstration plant, Kalpakkam.

V.2. DESCRIPTION OF REACTOR AND DESALINATION SYSTEMS CONSIDERED

V.2.1. PHWR at MAPS

PHWRs of 220 and 500 MWe appear to be the suitable choice as cogeneration reactors for producing desalted water and electrical power. Uranium in the form of uranium oxide is used as the fuel. Heavy water is used as moderator and in a separate circuit as coolant. Steam is produced at 250°C and 4.0 MPa in PHWR which is somewhat lower than the light water reactor. The PHWR at MAPS is operated at 170 MWe. The design parameters of a 220 MWe PHWR nuclear power station is given in Table V.1.

The Multi Stage Flash process involves heating of sea water to an appropriate temperature and flashing the heated brine under reducing pressure in a number of stages connected in series and maintained at successively lower pressures. The vapour flashed off in each stage is condensed outside the tubes in which recycle brine is flowing for heat recovery thereby raising its temperature.

TABLE V.1: DESIGN PARAMETERS OF A 220 MWE PHWR NUCLEAR POWER STATION

1.	Thermal Power Gross Electrical Output Net Electrical Output	790 MW 235 MW 220 MW
2.	Moderator	D20, Pressure: 7.5 kg/cm^2 Temperature: 44°C(in), 65°C(out)
3.	Primary Heat Transport System	D20, Pressure: 87 kg/cm^2 Temperature: 249°C(in), 293°C(out)
4.	Secondary Heat Transport System	H2O/Steam, Steam Pressure: 40.7 kg/cm^2, Temperature: 253.3°C, 0.28% wet
5.	Exhaust Steam from High Pressure (HP) Turbine	Pressure: 6 kg/cm^2, 11.2 % wet
6.	Inlet Steam to Low Pressure Turbine	Pressure: 5.7 kg/cm^2, Temperature: 233.3°C
7.	Condenser	Coolant : Sea water, Temperature: 30°C(in), 40°C(out)
8.	Steam Pressure Inlet to Condenser	0.087 kg/cm'

The condensate is collected and withdrawn as product. The latent heat given off by the condensing vapour is utilized to re-heat the recycle brine before going to the brine heater. The MSF process of brine recycle type has three distinct sections, namely, *(i) Heat Input Section i.e. Brine Heater, (ii) Heat Recovery Section, and (iii) Heat Reject Section.*

V.2.2. Distillation systems — MSF Plant

Steam economy of MSF plant is expressed in terms of Gain Output ratio(GOR) which is defined as kg water produced per kg of steam used in the brine heater. In India, energy cost is high and hence plants are designed for high GOR. With increasing GOR, the plant requires more heat transfer surface area for fixed number of stages due to decrease in log mean temperature difference (LMTD) for the condensers. Increasing the number of stages can reduce the heat transfer area. Increasing the number of stages, LMTD for the condenser increases. The ideal condition will be of infinite number of stages. But after certain number of stages, the saving due to surface area will be offset by increasing stage partition plates, orifices and etc. So, an optimum is obtained between GOR and number of stages with minimum cost. For MSF plant number of stages and GOR can independently be chosen subject to the following condition:

$$n > GOR[1 + Cp(To - Tn)]/2\lambda \quad \text{----------------------(1)}$$

And the GOR is related to the number of stages as

$$GOR = \frac{Md}{Ms} = \left(1 - \frac{Md}{2Mr}\right)\left(\frac{\lambda s}{\lambda}\right)(To - Tn).\left\{\frac{1}{\left(\frac{To - Tn}{n}\right) + (T1 - t1)}\right\} \quad \text{----------------------(2)}$$

The specific surface area is given by

$$\frac{A}{Md} = \left(\frac{MrCp}{Md}\right)\left[\left(\frac{n - j}{Ur}\right).\ln\left(1 + \frac{\Delta Tr}{\delta cr}\right) + \frac{jMsw}{UjMr}.\ln\left(1 + \frac{Mr}{Msw}\frac{\Delta Tj}{\delta cj}\right)\right] \quad \text{----------------------(3)}$$

where, δtc_i is condenser approach temperature given as

$$\delta ci = \frac{\Delta Ti}{\left[\exp\left(\frac{UiAi}{MrCp}-1\right)\right]} \qquad \text{----------------------(4)}$$

Also for a given capacity , if the top brine temperature increases the total flash drop increases thereby decreasing the re-circulation flow and hence reduces the pumping power according to the relation,

$$\frac{Mr}{Md} = \left[\frac{\lambda}{Cp.(To-Tn)}\right]+\frac{1}{2} \qquad \text{---------------------(5)}$$

Further, pumping power can be reduced by providing the long condenser tubes instead of cross tubes. Again, the brine blowdown concentration plays an important role for selecting top brine temperature(TBT). As TBT increases, GOR increases but there is a limit of TBT beyond which CaSO4 precipitation occurs.

India is setting up a 4500 m^3/d MSF desalination plant as a part of the hybrid nuclear desalination plant at Kalpakkam. The MSF plant a scale up of our pilot plant is based on long tube design, re-circulation type with acid dosing. It consists of 9 recovery modules each of 4 flash stages and one recovery module of 3 flash stages. Maximum brine temperature is limited to 121°C to avert calcium sulfate scaling on cupronickel heat transfer tubes. The pumping power requirement is 2.5 kWh/m^3 of water produced. It will produce high quality product water from seawater. The salient features of MSF plant are given in Table V.2.

In the long tube design the incoming brine flow path is parallel to the flashing brine. It is easier and less expensive to have a large number of flash stages in a long tube design than in a cross tube design. The long tube design has been chosen due to low capital cost and less pumping energy requirement The pretreatment scheme for the MSF plant involves acidification, vacuum de-aeration for control of O_2/CO_2 concentration, pH control by alkali neutralization followed by antifoam dosing.

TABLE V.2: SPECIFICATIONS OF 4500 m^3/d MSF PLANT

1. Product water output	187.5 m^3/hr
2. Product water salt content	< 20 ppm
3. Sea Water Requirement	
3.1 Total (cooling)	1544 m^3/hr
3.2 Make-up Feed (Part of 3.1)	375 m^3/hr
4. Brine Temperature	
4.1 Max. recirculation brine temperature	121°C
4.2 Blow down temperature	40°C
5. Concentration ratio	2
6. Steam Consumption	
6.1. Heating in the brine heater	20.6 Te/hr (2.8 bar)
6.2 Steam jet ejectors	400.0 kg/hr (7 bar)
7. Gain Output Ratio	9
8. Power Consumption	475 kWe

V.2.3. LTE Plant

The nuclear research reactors produce significant quantities of waste heat. A scheme has been evolved to integrate a desalination unit with primary coolant system of a nuclear research reactor such that the technology of utilizing reactor waste heat for desalination of seawater by low temperature evaporation process can be demonstrated. For conducting practical demonstration of Low Temperature Evaporation (LTE) technology, a 30 m^3/d LTE desalination system was designed and developed. The product water from the desalination plant is sufficient to meet the make-up requirement of the demineralised water for the reactor. It is a demonstration of low temperature evaporation process utilizing a part of the low temperature waste heat from nuclear research reactor for producing high quality desalted water and also the safety and economics of nuclear desalination technology as a viable alternative to produce low cost demineralised water from seawater. The experience gained will be useful in designing and setting up of larger scale desalination plants utilizing waste heat from nuclear research reactors and power plants.

LTE desalination plant is a self-contained unit with heater, separator and condenser housed in a casing. It utilizes waste heat in the form of hot water (50°C) or low pressure steam (0.13 bar) to produce pure water from seawater. Apart from the electric power requirement for the pump, no other energy or fuel is required. The probability of scale formation is practically eliminated by operating at low temperature and permissible brine concentration. Vacuum is maintained and excess brine is drained by water jet ejectors having no moving parts. It produces good quality product water (10 μS conductivity) directly from seawater. It does not need any chemical pretreatment of seawater except chlorination.

V.2.4. Reverse osmosis plant

The reverse osmosis desalination involves the separation of relatively pure water from sea water through a membrane under the application of pressure. The performance of the process depends on seawater salinity, temperature, extent of feed pretreatment, type of membrane element, feed flow rate, recovery and operating pressure. Depending on the critical design parameters governed by the site conditions, the reverse osmosis plants have operating pressures varying from 50 to about 75 bar and recoveries ranging from about 30 to 45%.

SWRO plant at NDDP, Kalpakkam is designed to produce water for blending with MSF water at the minimum energy cost. Accordingly the operating pressure is fixed at 52 bar. The plant has an energy recovery system with provisions to operate with MSF heat reject stream at 36–38°C. With a view to ensure longer membrane life an elaborate pretreatment scheme is incorporated involving lamella clarifier, particulate filters, dosing systems for prevention of scalants, biofouling and membrane degradation. As the product is to blended with the MSF product water the design permeate concentration has been fixed at 500–600 ppm. The salient parameters of RO plant are given in Table V.3.

TABLE V.3:
TECHNICAL SPECIFICATIONS OF 1800 m^3/d SWRO DESALINATION PLANT

1. A. Product water output	75 m^3/hr
B. Product quality	550 ppm TDS
2. A. Sea Water Required	214.3 m^3/hr
B. Sea Water TDS	35,000 ppm
3. % Recovery	35%
4. Membrane element	8040-SWHR 380
5. Design Flux	400 Lm 2_d-1
6. Average salt rejection	98.5%
7. Operating pressure	52 kg/cm^2
8. Operating temperature	36–38°C
9. Energy consumption	4.5 kWh/m^3

V.3. COUPLING OF DESALINATION AND NUCLEAR PLANTS

V.3.1. Hybrid systems — MSF/RO

Nuclear Desalination Demonstration Project (NDDP) at Kalpakkam aims for demonstrating the safe and economic production of good quality water by desalination of seawater comprising of 4500 nn^3/d Multi Stage Flash (MSF) and 1800 ni^3/d Reverse Osmosis (RO) plant. The MSF plant based on long tube design requires lesser energy.

The effect on performance of MSF plant due to higher seawater intake temperature is marginal. The Preheat RO system, part of the hybrid plant, uses reject cooling seawater from MSF plant. This allows lower pressure operation, resulting in energy saving. The two qualities of water produced are usable for the power station as well as for drinking purposes with appropriate blending. The post treatment is also simplified due to blending of the products from MSF and RO plants.
The hybrid plant has a number of advantages:

- A part of high purity desalted water produced from MSF plant will be used for the makeup demineralised water requirement (after necessary polishing) for the power station.
- Blending of the product water from RO and MSF plants would provide requisite quality drinking water.
- The RO plant will continue to be operated to provide the water for drinking purposes during the shutdown of the power station.

The nuclear desalination plant is located between the existing PHWR, MAPS and the proposed PFBR. The desalination plant utilizes power steam and seawater from MAPS. The location of the desalination plant at an existing nuclear reactor has resulted in many experiences not encountered in a desalination plant to be coupled to a new nuclear power plant.

- The steam tapping for NDDP could not be done in either back pressure or extraction mode from LP turbine as it required modification in the station which were not desirable. Steam is therefore taken from cold reheat lines after the HP turbines exhaust and employing adequate moisture separation.

- No additional seawater from the existing seawater intake for the station was available. As the site is known to have excessive sand movement, a new seawater intake was not considered suitable. The Kalpakkam site is also prone to excessive biofouling from raw seawater. It was therefore decided to utilize the process seawater coolant outfall as feed to the desalination plants. The condenser cooling seawater was not considered as it is at higher temperature than the process seawater outfall temperature. Further, the process seawater coolant is available even during the long shut down of the reactor.

- The coolant seawater from MSF plant which is disposed otherwise is blended with some raw sea water to bring it to a temperature of 35–36°C to be used as feed to SWRO plant. This would help operate the RO plant at relatively lower pressure, however, with slightly higher salinity of product. This is tolerable as the product water from RO plant is blended with the product of MSF plant having low TDS. The blending also simplifies the post-treatment and the combined product with appropriate LSI is obtained with minimal treatment.

Figure V.1 shows the coupling scheme of the hybrid MSF-RO desalination plant with the nuclear power plant for utilizing seawater, steam and power. The hybrid desalination plant requires around 2000 m^3/hr seawater. Different alternatives for seawater intake were considered and evaluated. The bathymetric and morphological survey was carried out. The geology of seabed has been identified. The tests were carried out for the soil bearing capacity, erosion and depositional characteristics of the seabed. From the bathymetry and seabed soil characteristics, it is observed that the seabed primarily consists of silty sand. Morphological changes along the beach are very high with respect to seasonal changes. After detailed studies, it has been decided to use process cooling water from the outfall seal pit of the power station as a source of seawater supply.

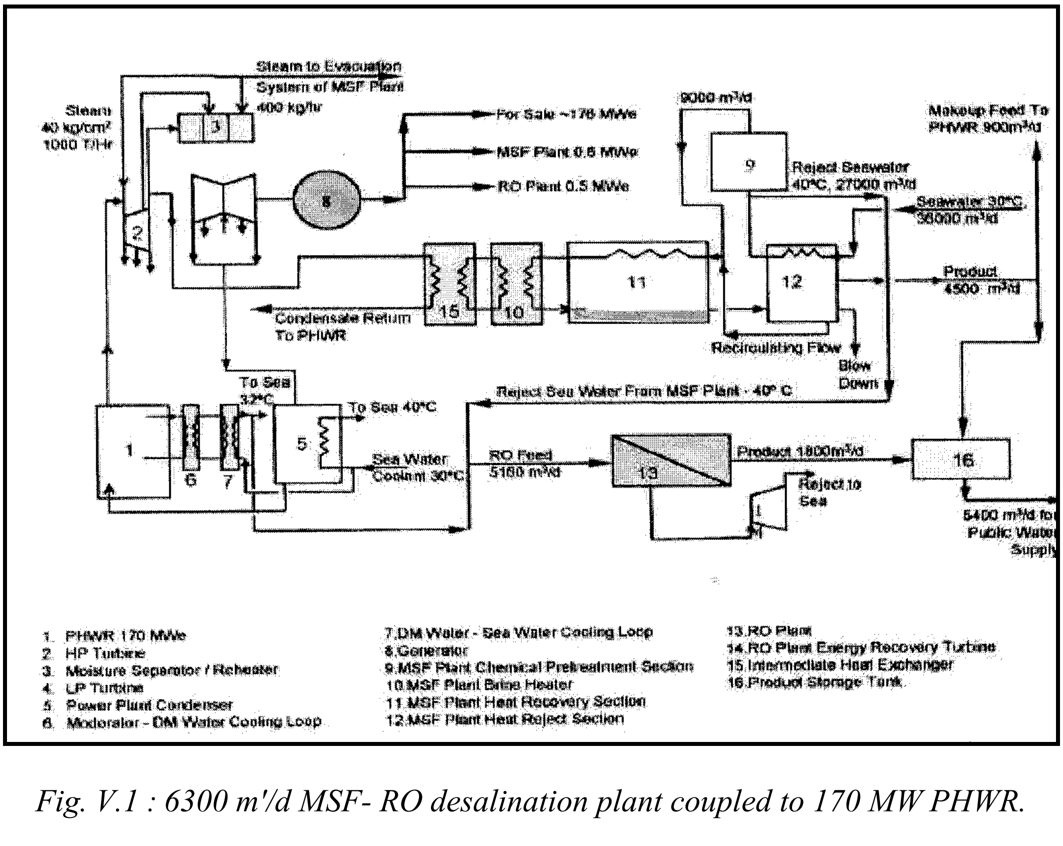


Fig. V.1 : 6300 m'/d MSF- RO desalination plant coupled to 170 MW PHWR.

The outfall seawater temperature varies from 26–32°C which is about 4°C higher than ambient temperature due to heat being rejected from the process cooling water system. In order to avoid any chance of ingress of radioactivity (tritium) to MSF process and product water, it has been decided to incorporate an isolation heat exchanger between MAPS steam supply and the brine heater of MSF. The LP steam is tapped from the manholes in the cold reheat lines after HP turbine exhaust from both the nuclear reactors (MAPS I and II). The moisture is removed through a moisture separator.

The steam is sent to intermediate isolation heat exchanger to produce process steam for brine heater of the MSF plant. Adequate provisions for monitoring and control have been incorporated for isolation of the steam supply in case of shutdown of the power station Q' desalination plant. Small amount of HP steam is withdrawn and sent to HP intermediate heat exchanger before reaching to the ejectors. Fig. V.2 shows the flow details of both the isolation heat exchangers. The loss of steam pressure in both the IHX is due to a distance of 600 meters between the reactor and the desalination plant.

Sea water is passed through the tube bundles of the *Heat Reject Section* where it is heated up by 10°C. A major part of this water is used for condenser cooling water and is rejected back to the sea. The rest part of the heated brine equivalent to about twice the product rate forms the process make up feed. It is acidified to decompose bicarbonates present in sea water and then sprayed over in a vacuum deaeration tower for removal of dissolved oxygen and carbon dioxide. The deaerated brine is then neutralised by alkali and the makeup feed is then mixed with concentrated brine in the last reject stage.

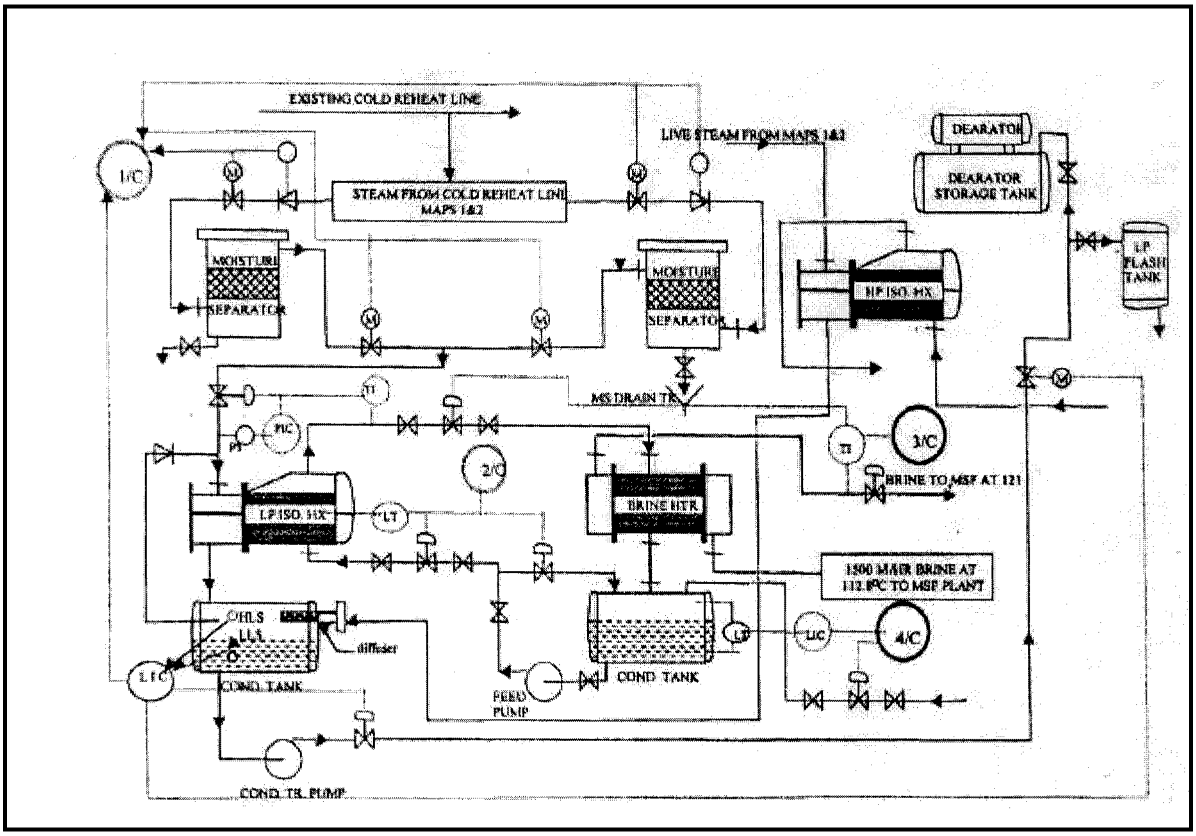


Fig. V.2 Flow details of isolation heat exchangers.

The bulk of the brine from the last reject stage is recycled and passed through tube bundles of the *Heat Recovery Section* where it gets heated up to about 113°C by the condensation of vapours produced due to flashing of brine in those stages. The brine is then further heated up to a maximum of 121°C by steam in the *Brine Heater.* The hot brine is then fed to the first stage of the flash chamber through an orifice. The first stage is maintained at a pressure, which is slightly below the saturation pressure corresponding to 118.3°C. As a result, the brine flashes and gives off water vapour in order to attain equilibrium. The vapour so formed is passed through a wire-mesh demister to remove entrained droplets of brine. It then gets condensed on the outside of the tube bundles to give product water. The flashing process is repeated through all the 39 stages. The last 3 flashing stages are coupledd to *Heat Reject Section.* The last stage is maintained at –0.08 bar by means of a vacuum system consisting of a two stage steam jet ejector with pre, inter and after stage condensers with larometric legs. The distilled water from all the stages flows in cascade to the last stage and is pumped out as product. A part of the concentrated brine from the last stage is discharged as blow down and the remaining is mixed with fresh deaerated feed and pumped back as recycle brine into the tube bundles of *Heat Recovery Section,* entering at the last recovery stage i.e. 36^{th} by brine recycle pumps.

Our RO plant design differs from the seawater RO plants set up internationally in a relatively lower pressure operation to save energy. The high temperature operation results in lowering the operational pressure to about 52 kg/cm^2. Slightly high TDS of RO permeate of 500–600 ppm is tolerated because this product water is blended with product water from MSF. The low pressure operation may also contribute to the long life of the membranes due to lesser possibility of compaction.

The return stream of cooling seawater from the reject stages of MSF plant is blended with the seawater to bring down the temperature to 36–38°C before the same is sent to the pretreatment section of RO. The pretreatment is presently conventional. The seawater is pumped through the clarifier and pressure sand filter. It is then passed through the cartridge filter to remove the particles up to 5 micron size. Since the membranes are polyamide, de-chlorination of seawater is carried out by adding NaHSO3. Acid dosing is given to minimize carbonate scaling followed by addition of antiscalant or SHMP. The pretreated seawater is pumped n to two parallel streams through the modules at a rate of 110 m^3/h each. Pumps are fitted with energy recovery hydraulic turbocharger. The seawater membrane element is 8040 spiral polyamide TFC with 400 lmd flux and 98.8% salt rejection. The specific energy requirement is around 4.5 kWh/m^3 of water produced.

V.3.2. Trombay research reactor/LTE

Figure V.3 gives the schematic diagram of the LTE desalination plant coupled to nuclear reactor (CIRUS). CIRUS research reactor is a 40 MW_{th} nuclear research reactor using metallic natural uranium fuel, heavy water moderator, demineralised light water coolant and seawater as the secondary coolant. The heat generated in the fuel assemblies is removed by recirculation of demineralised water in a closed loop called Primary Cooling Water (PCW) System using recirculation pumps. The heat from the primary coolant is transferred to the secondary coolant (seawater) in a set of heat exchangers and ultimately rejected to sea.

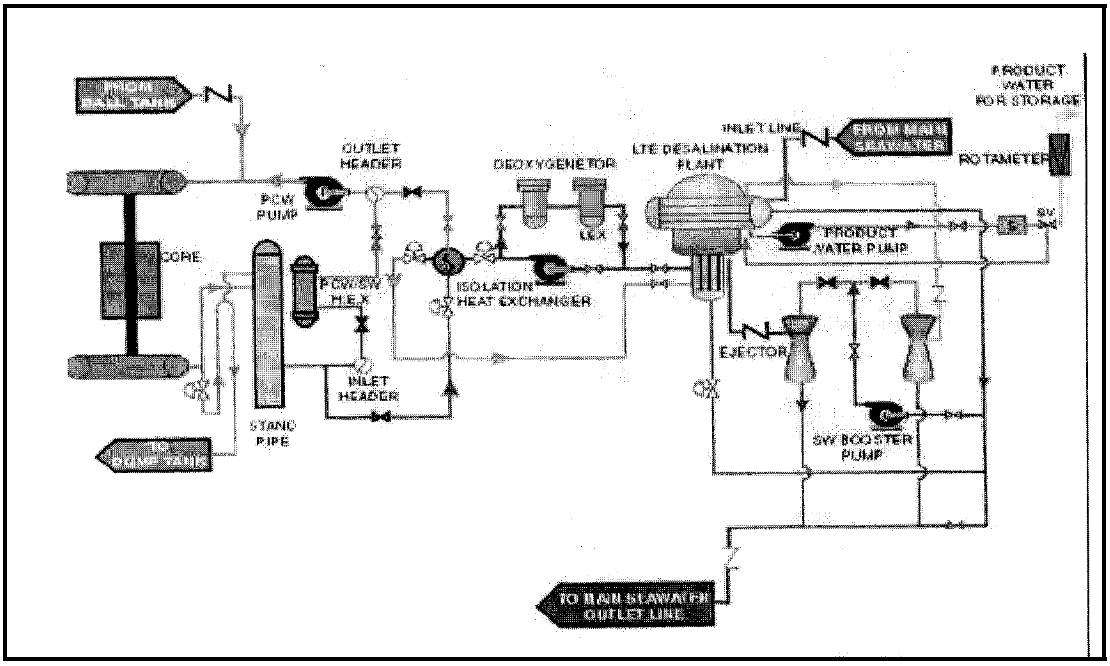


Fig. V.3. Schematic Diagram of the LTE Desalination Plant Coupled to Nuclear Reactor (CIRUS).

The changes in process parameters of PCW and seawater (SW) systems for different cases are shown in Table V.4. When PCW pumps are not in operation for any reason, shut down cooling flow through the core is established automatically by one pass gravity flow from an overhead water storage tank. The core outlet is led to an underground water storage tank (dump tank) from where it is pumped back to the storage tank.

An intermediate heat exchanger is incorporated between the primary coolant water (PCW) and the desalination plant to ensure that no radioactive material reaches the desalted water. The intermediate circuit consists of a booster pump, the intermediate heat exchanger, the desalination plant and the associated piping and isolation valves. The intermediate circuit water is maintained at a pressure higher than the PCW pressure in the intermediate heat exchanger so that ingression of activity to the inactive intermediate circuit is prevented in the event of leakage in the heat exchanger tubes. It ensures an extremely low probability of radioactive contamination and high protection of product water. Flow controls for the feed seawater and product water have been provided to take care of the sudden changes in the power rating of the nuclear reactor. The sudden change rating leads to fluctuations in temperatures of heating medium resulting in changes in production rate of fresh water.

V.4. ANALYTICAL EVALUATION AND EXPERIMENTAL VALIDATION

V.4.1. Experimental evaluation of preheat RO

The membrane performance is concurrently controlled by feed flow rate, operating pressure and temperature besides the compaction effect on the membranes due to temperature, pressure and time. In order to assess the performance of the membranes, commercial seawater membrane elements of M/s Hydranautics, Film Tech and Fluid systems were used in the experiments. It was decided to use 4040 elements to collect the data as the data can be used for the prediction under scale up conditions.

TABLE V.4:
CHANGES IN FLOW & TEMPERATURE OF PCW & SW SYSTEMS AT 40 MW OPERATION

Parameters			CASE - I	CASE - II	CASE - III
CORE	PCW flow (lpm)		17700	17700	17700
	PCW inlet Temp. (°C)		45	46	47.3
	PCW outlet Temp. (°C)		76.2	77.2	78.5
PCW/SW HEAT EXCHA-NGERS (HX)	PCW	PCW flow (lpm)	3450	3285	3285
		Inlet Temp (°C)	76.3	77.2	78.5
		Outlet Temp (°C)	45	44.4	44.8
	SW	Flow (1pm)	6265	6175	6175
		Inlet Temp (°C)	29	29	29
		Outlet Temp (°C)	46.2	46	46.5
INTER-MEDIATE HX	PCW flow (fpm)		---	1280	1280
	PCW inlet Temp. (°C)		---	77.2	78.5
	PCW outlet Temp. (°C)		---	65.8	78.5
SEA WATER SYSTEM	Gross flow (1pm)		33600	35150	35150
	Inlet Temp (°C)		29	29	29
	Outlet Temp (°C)		45.7	45	45

CASE - I: Parameters of existing reactor
CASE - II: parameters after proposed modifications
CASE - III: Parameters considering non-availability of desalination unit

V.4.1.1. Experimental work

The test loop consists of a 1000 litre capacity feed tank, a high pressure pump of 40 lpm flow and 50 bar head (Fig. V.4). Provisions were made to connect individual 4040 elements one after the other. The reject and permeate stream were connected to the feed tank. The temperatures of the feed at the point of entry to the pump, reject & permeate streams as they fall into the feed tank were measured within an accuracy of ± 0.1 deg C. Rotameters were fitted in the reject and permeate line to measure the flow rates. Flowrates were measured manually also using calibrated containers, whenever required.

V.4.1.2. Performance of different commercial membrane elements

The commercial sea water elements of M/s Film Tech, Hydranautics and Fluid System were fitted into the test loop one after the other and their performance was evaluated as a function of temperature at 251pm.

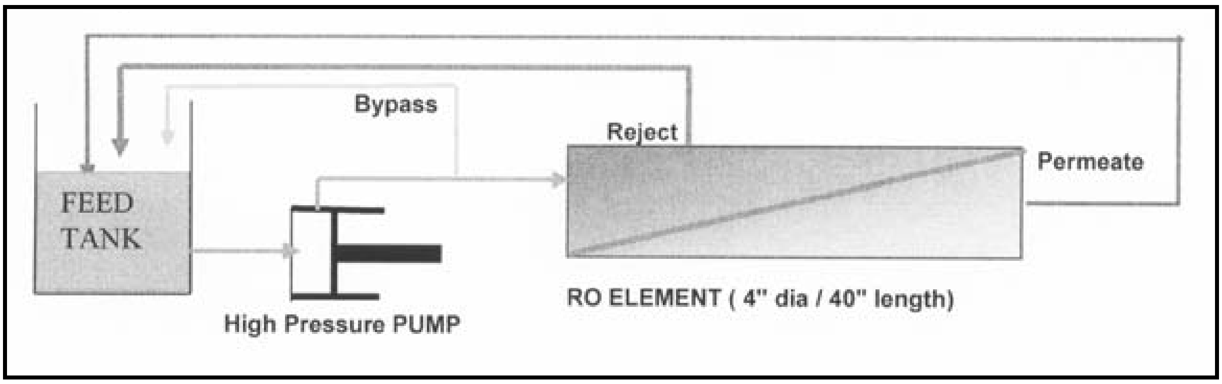


Fig. V.4. Experimental Test Loop.

Solute Rejection : Hydranautics membrane exhibits the best solute rejection of about 99.6 at 25 lpm flow rate. Film Tech membranes exhibit marginally lower rejection of about 99.3%. Fluid system element exhibits a lower solute rejection compared to other two membranes with around 99%. The behavioural pattern is different for all the three membranes as seen in Fig. V.5 indicating the temperature effect is dependent on membrane material.

Film Tech membranes show a marginal decrease in solute rejection with temperature while the other two membranes exhibit a maximum in solute rejection at around 35°'C.

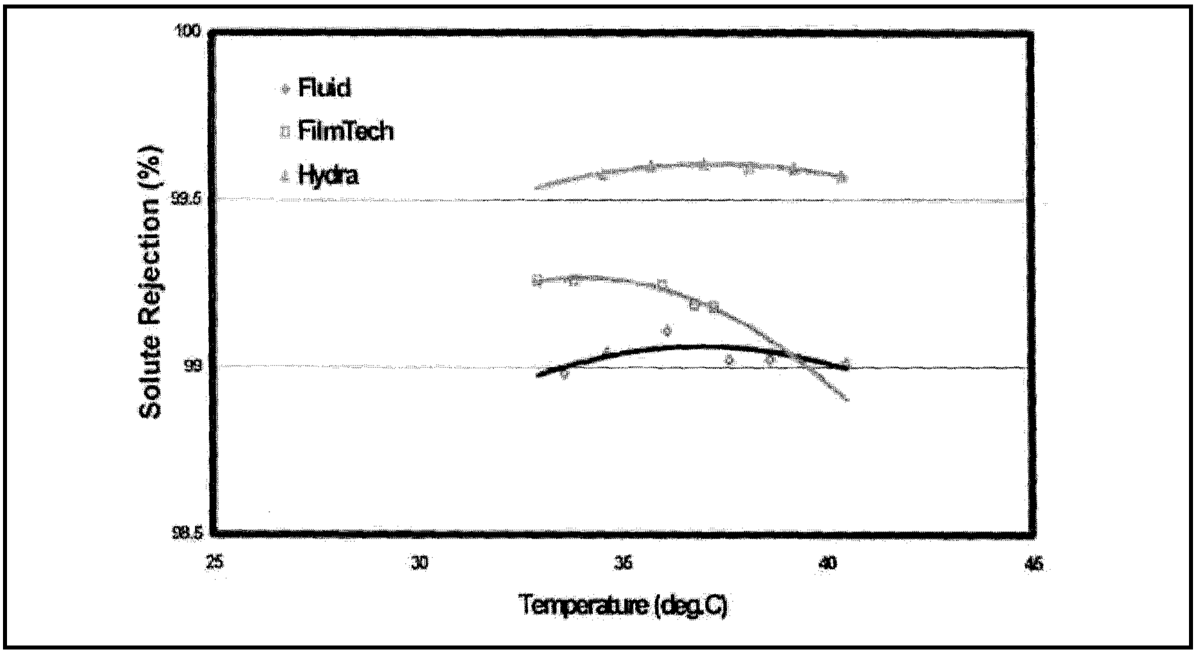


Fig V.5. Variation of Solute ejection with Temperature for Commercial. Seawater membranes (35 lpm - 50 bar - 35000ppm)

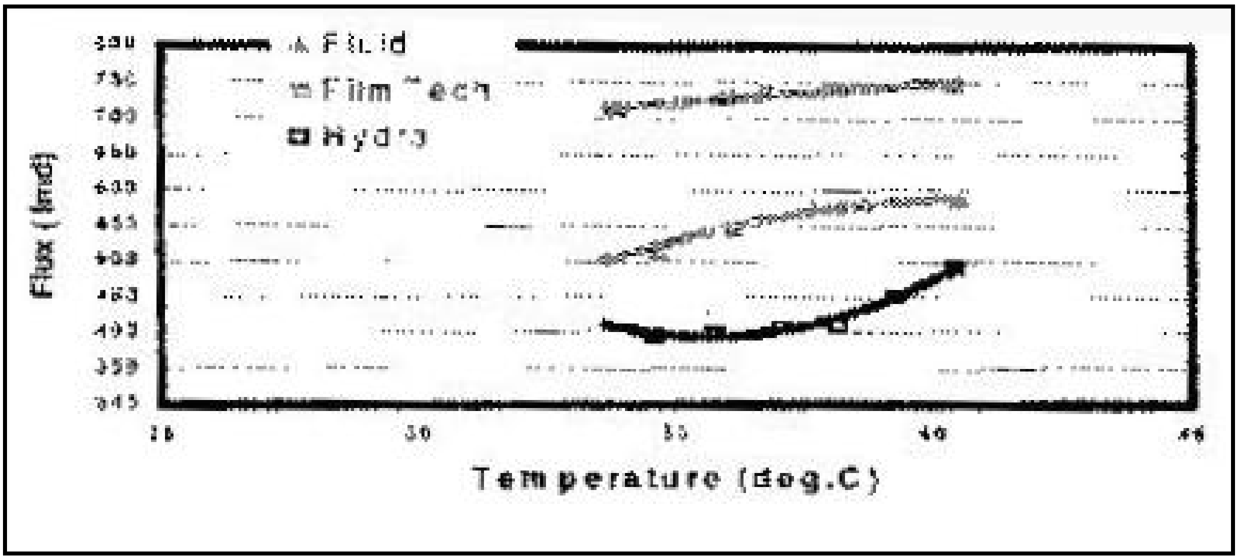


Fig. V.6. Variation of Water Flux with Temperature for Commercial seawater elements. (25 lpm -50 bar – 35000 ppm)

Permeate Water Flux: Film Tech membrane element, as seen in Fig. V.6 shows a higher water flux compared to the other two membrane elements exhibiting on an average about 1% rise per degree Celsius in the temperature range of 32–40°C . The water flux of Hydranautics membrane element is only about 50% of Film Tech but shows an average increase of about less than 1% per deg. Celsius up to 35°C and thereafter about 3–4% per every degree rise. Fluid system exhibits about 80% of the Film Tech membrane flux and exhibits an average increase of about 2% per °C.

General conclusions

Film Tech membrane element appears to be more suitable for seawater applications under varying temperature conditions because of higher flux and fairly high solute rejection. Hydranautics membranes even though exhibit marginally higher solute rejection the flux is much less. The performance of Fluid System element is marginally lower compared to Film Tech both with respect to solute rejection and permeate flux.

V.4.2. Experimental evaluation of thermal systems - MSF Plant

The variation in seawater temperature may have some effect on the performance of MSF. Experiments were carried out in our pilot plant to study the effect of seawater temperature and the temperature rise of seawater coolant in the heat reject section of the MSF. The results collected for sea water temperature varying from 30-36°C is given in Table V.5. The higher sea water temperature for MSF plant leads to loss in production rate and GOR. Analysis has been carried out for operating the plant at lower reject seawater temperature by increasing the seawater flow rate in the heat reject section. It has been done to maintain lower seawater discharge temperature in case required due to environmental considerations. It is noted that the production of desalted water can be maintained or increased slightly within the design limits by increasing the coolant flow rate.

V.4.3. LTE Plant

The hot water temperature to the desalination plant varies from 53–65°°C as the power rating of the reactor operation varies from 20–40 MWth. Experiments were carried out at 41°C operating temperature and 700 mm Hg vacuum for hot water temperature varying from 50–65°C. Table V.6 gives the performance of the desalination plant at various hot water temperatures for reactor rating varying from 20–40 MWth. The conductivity of the product water was obtained in the range of 10–14 µS (4–5.5 ppm TDS) throughout the operation of the desalination plant. The desalted water produced from this plant was supplied to nuclear research reactor (CIRUS) for use as makeup demineralised (DM) water after polishing. The desalted water obtained was high quality with silica content close to nil.

V.5. STATUS AT THE END OF 2003

The research in the framework of this CRP has the main objective of collecting relevant data for improving the hybrid desalination system performance using experimental facilities. The facilities and experiences of 425 m^3/day pilot plant, 30 m^3 /day Low Temperature Evaporation(LTE) pilot plant at Trombay are being used for the studies and the prediction of hybrid system being installed at Kalpakkam demonstration plant. The LTE facility is connected to the experimental reactor CIRUS for using its low grade waste heat. The 100 m^3/day sea water reverse osmosis plant with energy recovery turbine provides useful data. It has provision for testing various types of membranes elements at different temperatures to collect data for pre-heat RO. An UF pretreatment system has been recently installed in this plant.

TABLE V.5:
EFFECT OF SEA WATER TEMPERATURE AND HEAT REJECT CONDENSER COOLANT TEMPERATURE RISE ON DESALTED WATER PRODUCTION

Sr. No.	Coolant seawater Temperature (°C)		Coolant temperature rise (°C)	Flashing brine temperature entering heat reject section (°C)	Sea water coolant flow rate (m^3/hr)	Production rate (m^3/hr)
	In	Out				
1.	30	40	10	47.8	1160.9	187.88
2.	32	42	10	49.8	1160.9	183.76
3.	34	44	10	51.8	1160.9	179.63
4.	36	46	10	53.8	1160.9	175.47
5.	30	39	9	46.8	1289.91	189.93
6.	32	41	9	48.8	1289.91	185.85
7.	34	43	9	50.8	1289.91	181.69
8.	36	45	9	52.8	1289.91	177.56
9.	30	38	8	45.8	1451.15	191.97
10.	32	40	8	47.8	1451.15	187.87
11.	34	42	8	49.8	1451.15	183.77
12.	36	44	8	51.8	1451.15	179.63
13.	30	37	7	44.8	1658.46	194.0
14.	32	39	7	46.8	1658.46	190.0
15.	34	41	7	48.8	1658.46	185.85
16.	36	43	7	50.8	1658.46	181.71

TABLE V.6:
LTE DESALINATION PLANT PERFORMANCE AT DIFFERENT HOT WATER TEMPERATURES

Sl. No.	Hot water Temperature (°C)	Product water flow rate		Total dissolved solids in the product water (ppm)
		LPM	Te/d	
1	50	7.0	10.0	5
2	55	10.5	15.1	5
3	60	15.0	21.6	5
4	62	17.0	24.5	5
5	65	21.0	30.2	5

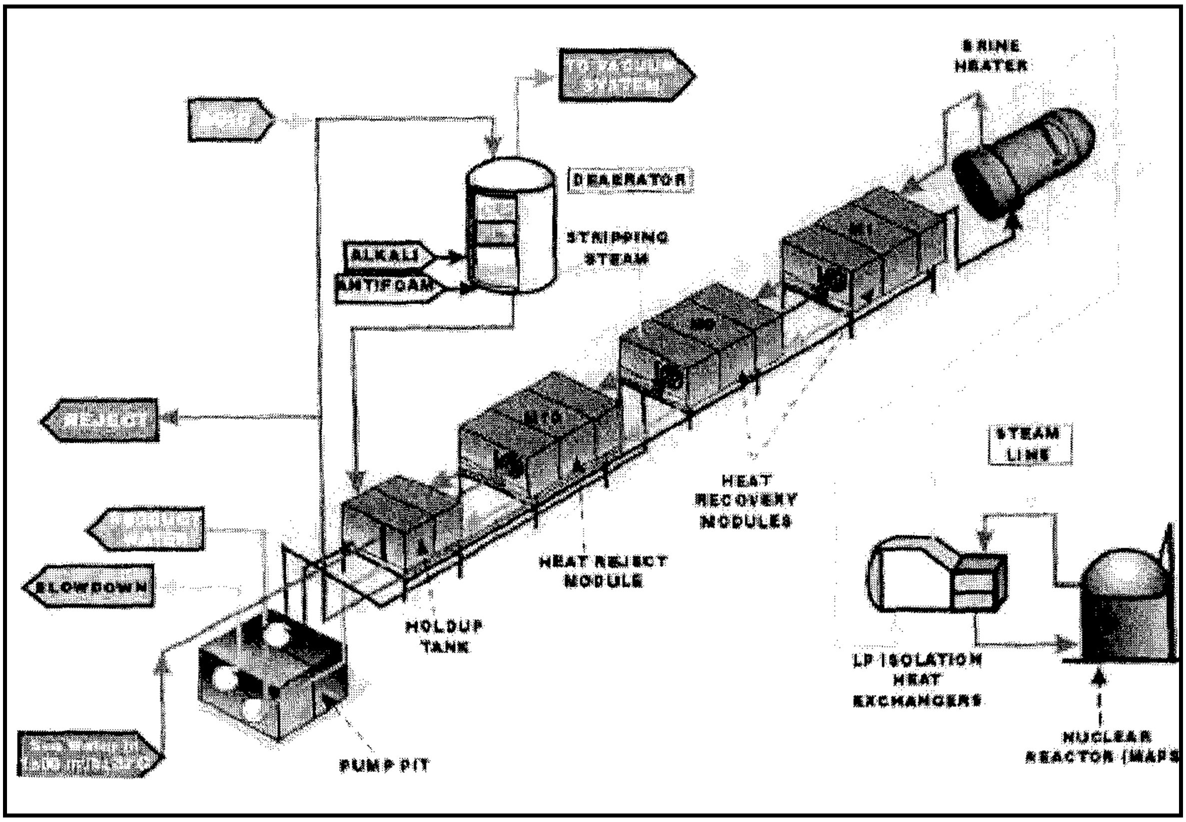


Fig. V.7: Layout of 4500 m^3/d MSF desalination plant, Kalpakkam.

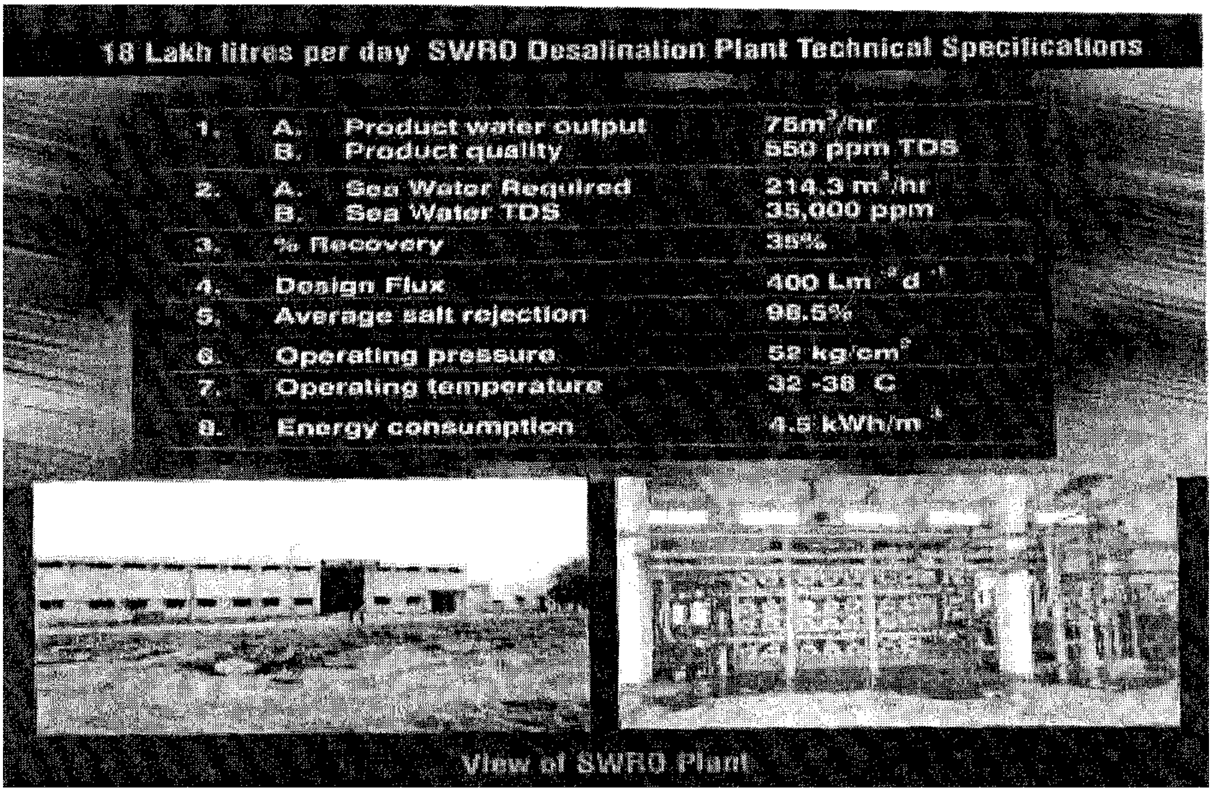


Fig. V.8: A view of 1800 m^3/d SWRO Plant, Kalpakkam.

V.6. CONCLUSIONS

BARC is in the process of setting up a hybrid 6300 m^3/day (4500 m^3/day MSF and 1800 m^3/day SWRO) nuclear desalination demonstration project at Kalpakkain coupled with two units of 170 MW(e) PHWRs at the Madras Atomic Power Station(MAPS). The requirements of sea water, steam and electrical power for the desalination plants are met from MAPS I & II which are around 1.5%, 1.0% and 0.5% of that available at MAPS. The hybrid plant has provision for redundancy, utilisation of streams from one to the other and production of two qualities of products for their best utilisation. The project has the main objective of demonstrating the capability for design, fabrication and operation of large size future plants. Fig. V.7 shows the layout of 4500 m^3/day MSF desalination plant, Kalpakkam which is at the advanced stage of commissioning. Fig. V.8 shows a view of the 1800 m^3/day SWRO plant which has been commissioned in Oct. 2002. A typical performance data is shown in Table V.7.

TABLE V.7:
PERFORMANCE DATA OF (STREAM-I) SWRO PLANT AT KALPAKKAM

Particulars	**I**	**II**	**III**		
Feed TDS (ppm)	34969	34152	33826	35296	34560
Feed flow rate (m^3Ih)	**110**	110	110	**110**	110
Water temp. (°C)	30.5	31.5	30.8	32.5	31.0
Raw water pH	8.18	7.83	7.82	8.01	8
Treated water pH	6.56	6.58	6.42	6.52	6.54
Operating pressure (bar)	44.2	43.4	45.6	47.9	48.7
Permeate TDS ppm	370	455	450.3	444.9	418.0
Permeate Flow (m^j/h)	39.35	37.95	38.17	42.84	43.7
Permeate pH	8.4	8.4	8.32	8.34	8.35
Product water TDS (ppm)	434.1	477.3	482.7	461.1	510.0

COUPLING OF SMART AND DESALINATION PLANT

Y.D. Hwang
Korea Atomic Energy Research Institute, Taejon, Republic of Korea

VI.1. INTRODUCTION

SMART is an integral type advanced pressurized water reactor with a thermal capacity of 330MW which is developed for the dual application of seawater desalination and small-scale electricity generation. The integrated SMART nuclear desalination plant aims to produce 40,000 m^3/day of potable water and about 90 MW of electricity. The objective of this CRP is to establish a set of optimal coupling parameters between SMART and the desalination plant which satisfies the diverse operational and safety requirements imposed by water production and electricity generation and while retaining economic advantages of the use of nuclear energy.

During the previous 4–year period of the project (1998~2002), the major coupling parameters were identified and the sensitivity analysis of these coupling parameters on plant performance were conducted. Also, a series of economic evaluation was carried out for the various combinations of the coupling schemes. Based on these results, the desalination process and the coupling scheme were selected and the optimization study on the coupling parameters has been conducted for the selected coupling scheme. In addition, the safety of the integrated SMART nuclear desalination plant was evaluated through the bounding approach of the key safety parameters of SMART and the work to simulate the contaminant migration from reactor to the product water was conducted. During the final year (2003), the works to finalize the optimal coupling between SMART and the desalination plant was conducted. The major coupling parameters and the interfacing conditions were confirmed by reviewing the basic design data of the integrated SMART nuclear desalination plan. Also, a simulation of the contaminant migration from reactor to the product water was conducted utilizing modeling tool DESNU developed by INVAP. General-purpose thermal hydraulic computer code RETRAN-03 was used to simulate the postulated multiple failure of isolation barriers between SMART and the desalination plant. In addition, some works for final report preparation have been conducted.

VI.1.1. Background

An advanced small integral reactor, SMART has been developed in Korea for the dual-purpose energy utilization. For the demonstration of its application, a nuclear desalination system thermally coupled with SMART has also been developed. The integrated SMART nuclear desalination plant aims to produce 40,000 m^3/day of potable water and about 90 MW of electricity.

Since the integrated nuclear desalination plant uses a large portion of the steam produced by SMART for seawater desalination, the coupling system between SMART and desalination system should be designed to satisfy the diverse operational requirements imposed by power and water production while retaining the inherent economic advantages of the co-generation. Thus the proper establishment of optimum coupling parameters between SMART and the desalination process is essential.

The objective of this study is to establish a set of optimum coupling parameters between SMART and the desalination process, which meets the requirements of nuclear desalination using SMART. For this purpose, the study has been conducted to investigate optimal coupling between SMART and desalination plant, which meets the electricity and water production requirements. As results of the study a set of the major coupling parameters was established for the selected coupling scheme of the integrated SMART nuclear desalination plant. During this project year, the work has been focused on the confirmation of the major coupling parameters, which influence the water and electricity production. Final report has been revised with confirmed data and compiled in the Business Collaborator Common Workplace of the Nuclear Power Technology Development Section, IAEA. In addition, a qualitative analysis on the contaminant migration was performed during this project year.

VI.1.2. Work scope

The major tasks for this project year (Nov. 2002–Oct. 2003) are as follows:

1. Finalization of the major coupling parameters which influence the performance of the integrated SMART nuclear desalination plant
2. Preparation of the final report
3. Simulation of the contaminant migration

This report presents the results of the studies conducted in this project year. Final report was presented in the Business Collaborator Common Workplace of the Nuclear Power Technology Development Section, IAEA.

VI.2. PROGRESS OF WORK

A study on optimal coupling between SMART and the desalination plant was conducted for the requirement of water production capacity of 40,000 m^3/day and the electricity generation of 90MW. During this project year, the major coupling parameters, which had been optimized during the previous project years, were revised based on the basic design data of SMART and the desalination plant. In addition, a qualitative analysis on contaminant migration was performed for the postulated multiple failure sequence of isolation barriers between the nuclear reactor and the desalination plant. The results of the study are presented in the following sections of this report.

VI.2.1. Design characteristics of SMART

SMART is an integral type power reactor with a rated thermal power of 330MWt. Different from the conventional loop-type reactors, 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. The integrated arrangement of these components enables the elimination of large-sized 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. Appendix I summarizes the major design parameters of SMART.[1]

VI.2.2. Design characteristics of SMART desalination plant, MED-TVC unit

The desalination process employed in the integrated SMART nuclear desalination plant is the type of falling film, multi-effect evaporation with horizontal tubes combined with thermal vapor compressor. Four distillation units will be coupled with SMART and each distillation unit has the production capacity of 10,000m^3/day of distilled water at the top brine temperature of 65^{o}C using the seawater supplied at the design temperature of 33^{o}C. Thermal Vapor Compressor was added to the MED process to improve the energy utilization. The thermal vapor compressor induces the vapor from the final condenser and discharges the mixture of the vapor and the motive steam to the first effect. The advantages of this design are high heat transfer coefficients and relatively simple system operation.

The MED-TVC unit is designed with performance ratio (PR) of 19.6 and the motive steam (supplied steam to the thermal vapor compressor) to load ratio of one (1.0) and the number of effects 10. The SMART desalination plant is designed to be capable of operating at reduced output down to 50% of the rated design output. Scale is controlled by injection of the dosing additive into the feed seawater. The plant is designed to be capable of operating at rated output for seawater temperatures from 20 to 33°C.

The general arrangement of the plant was designed to facilitate an operation by the minimum number of personnel, ease of maintenance, cleaning, inspection, replacement of pumps and other auxiliaries. The major design parameters of the SMART desalination plant are summarized in Appendix II. [2,3]

VI.2.3. Coupling between SMART and Desalination Plant

The most important safety issue on nuclear desalination is the exclusion of the radioactivity carry-over into the product water. In the SMART integrated nuclear desalination plant, an intermediate loop (steam transformer) is installed between the secondary cycle of SMART and the desalination plant in order to protect the product water from the radioactive contamination.

The integrated SMART desalination plant consists of 4 units of MED-TVC (water production capacity of 10,000 m^3/day per unit) and connected with SMART plant through steam transformer. The steam transformer produces the process steam for desalination using the steam extracted from turbine. The major safety function of the steam transformer is to prevent the contamination of the produced water from the radioactive material produced in reactor. In addition, the radioactivity monitoring system was installed to check radioactivity level in the primary side of the steam transformer and the desalination plant. The monitoring system checks the symptom of the radioactivity in the system and triggers the signal for proper action such as system isolation.

Figure VI.1 shows the configuration of the coupling scheme for the SMART integrated nuclear desalination plant. For this coupling configuration, the major coupling parameters and interfacing conditions were optimized for the water production requirements of 40,000 m^3/day and electricity generation of 90MW.

Major coupling parameters were optimized based on the basic design data of SMART secondary system, intermediate heat transfer loop (steam transformer) and desalination plant as presented in the previous section of this report. The thermal balance calculations on SMART secondary cycle were performed for the various conditions of steam extraction and the suitable extraction point for desalination was identified based on the calculation results. It was decided to extract 30.0kg/sec (4x7.5kg/sec) of steam from turbine at the pressure of 9.0 bara. The flow rate of extracted steam was determined by taking into major coupling parameters, such as water production and electricity generation requirements, performance ratio, steam to load ratio, thermal performance of the steam transformer and plant economy. The optimized coupling of the integrated SMART desalination plant produced net electricity generation rate of about 85Mw while producing potable water 40 000m^3/day.

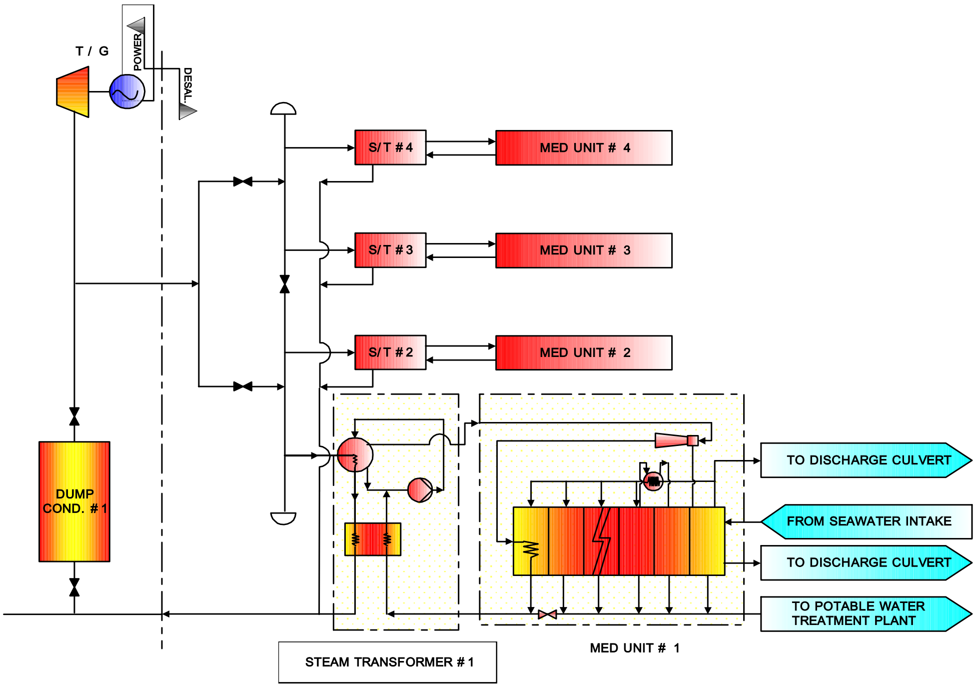


Figure VI.1. Coupling Concept between SMART and Desalination Plant.

The performance ratio (PR), the motive steam to load ratio and the number of effects of MED units are the major coupling parameters influencing the amount of process steam required for desalination and thus the economy of the plant. SMART desalination plant was optimized performance ratio of 19.6, motive steam to load ratio of one (1) and number of effects 10.

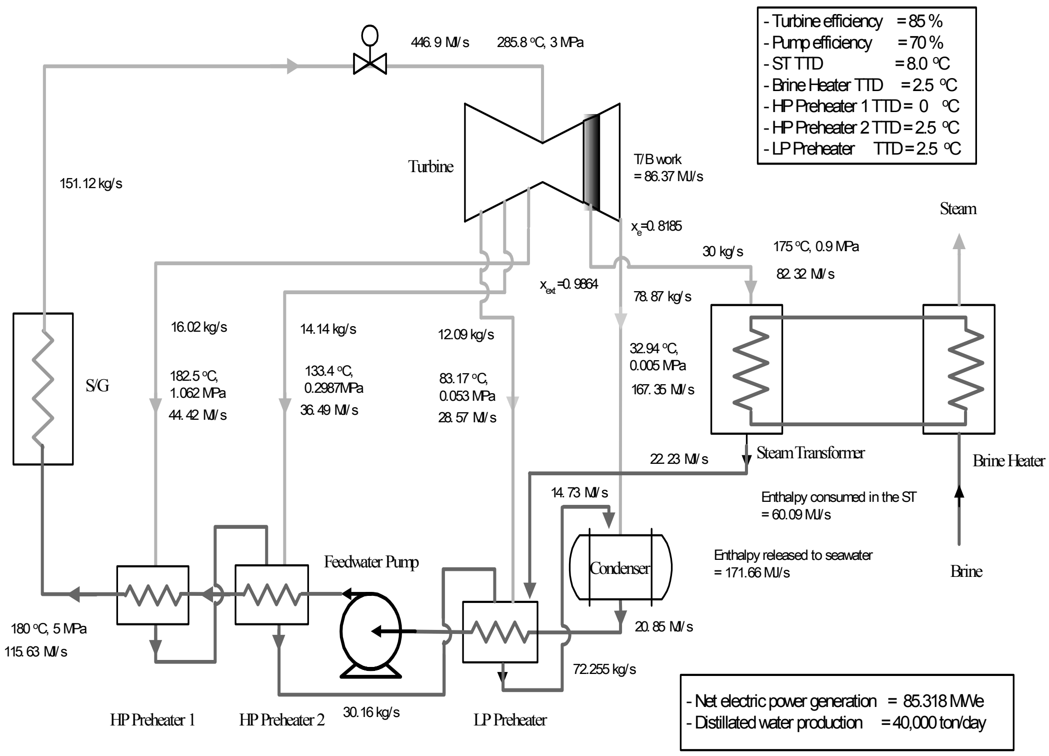
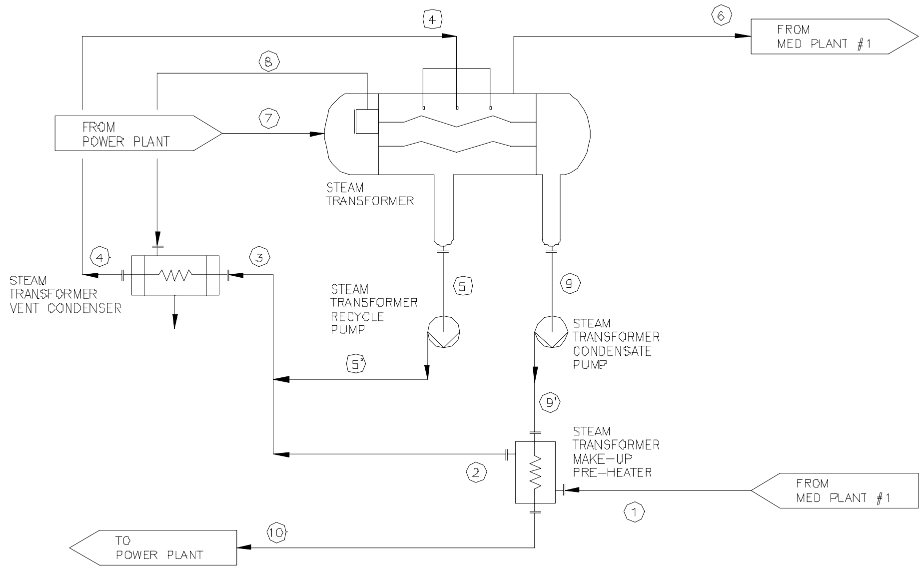


Figure VI.2. Thermal Balance of the SMART Secondary System.



Stream No.		1	2	3	4	5	5'	6	7	8	9	9'	10
Fluid	Unit	Water	Water	Water	Water	Water	Water	Steam	Steam	Vent	Cond	Cond	Cond
Flow	kg/s	5.93	5.93	17.21	17.21	11.28	11.28	5.93	7.54	0.15	7.39	7.39	7.39
Temp.	C	39.60	43.78	126.90	131.90	170.60	170.60	170.60	175.36	175.36	175.21	175.21	171.95
Press.	bara	10.00	10.00	10.00	10.00	8.00	10.00	8.00	9.00	9.00	9.00	11.00	11.00
Enthalpy	kJ/kg	166.7	184.1	533.6	554.9	721.7	721.9	2767.9	2772.1	2772.1	742.0	742.1	727.8

Figure VI.3. Thermal Balance of the Steam Transformer.

VI.2.4. Simulation of the contaminant migration of the SMART desalination plant

In nuclear desalination, the most important issue with respect to safety and public acceptance is the prevention of the radioactive contamination of the product water. Thus the design of the nuclear desalination plant should pay proper attention to the multiple failure of accidental-sequence that would end up contaminating the product water. Although the probability of multiple failure occurrences is very low, some sequences can be shown as unlikely enough to be considered unrealistic. This may be introduced a probabilistic safety analysis (PSA) specific requirements related to the possibility of spreading contamination through the distribution of product water.[4] In order to meet the requirements, 1) a measurement of radiation level in the process steam supplied to the MED plant and 2) an isolation of desalination plant from SMART plant are necessary. Both this measurement and actuator system should comply with response time, sensitivity and reliability requirements applied to specific equipment. For the integrated SMART nuclear desalination plant, an analysis on the multiple failure of the isolation barrier, i.e. both of the steam generator rupture and steam transformer tube breaks may provide a useful information on the transit time of contaminant from the source up to the inlet of distribution network.[4,5]

An analysis on the transport of the radiactive material from primary reactor coolant to the product water was performed for the selected scenerio of the multiple failure of the integrated SMART nuclear desalination plant. Three dimensional thermal hydraulic analysis computer code RETRAN-03 was used for this simulation. The modeling tool DESNU developed by INVAP was utilize to generate a input for simulation. DESNU spreadsheet program generates RETRAN or RELAP models of generic nuclear desalination systems for the different combination of small nuclear power plant and desalination plant.

VI.2.4.1 Modeling of the integrated SMART nuclear desalination plant

Thermal cycle of SMART is quite standard with a single turbine having for steam extractions. The integrated SMART desalination plant consists of 4 units of MED-TVC and coupled with SMART through steam transformer. The steam transformer directly uses the steam of turbine extraction #3 to produce the process steam for desalination. To prevent the radiactive contamination of the product water from the contaminant produced in reactor, shell side of the steam transformer maintains higher pressure (pressure of process steam used for desalination) than that of the tube side (pressure of the extracted steam from turbine). Therefore, the postulated leak flow always directed from desalination plant to the secondary system of reactor. However, this protection mechanism may be failed for the accident condition, such as steam generature tube rupture which increases the secondary system pressure.

The steam generature tube rupture is one of the design base accident (DBA) of SMART. When the steam generature tube rupture accident occurs, the pressure of the secondary systm rapidly increases because of the higher pressure primary coolant enters through the ruptured tube (Figure VI.4). The pressure reversal condition between the shell and tube side of steam transformer may not be maintained because of the pressure increase of the secondary system. The radioactive contaninant produced in the reactor may be ingressed into the desalination plant through the damaged heat exchager tube of steam transformer. Therefore, the proper isolations of the desalination plant from the water distribution grid need to be provided for the protection of the public from radioactivity exposure for this accident. Since the reactor trip and the isolation of desalination plant are triggered by the high pressure and/or contamination detection in the secondary loop and by a pressure decrease in the primary system, an estimation of the transit time of contaminant from reactor to the inlet of distribution network may provide important information for the safety aspect of the nuclear desalination.

The integrated SMART nuclear desalination plant was modeled using modeling tool DESNU for the estimation of the transit time of the contaminant. DESNU program allows adapting the model for different particular combinations of small nuclear power plant and desalination plant. The DESNU generated input file was modified by changing the connecting junction for desalination plant (MED) from volume 980 to volume 981 which represents the steam transformer (Figure VI.5).

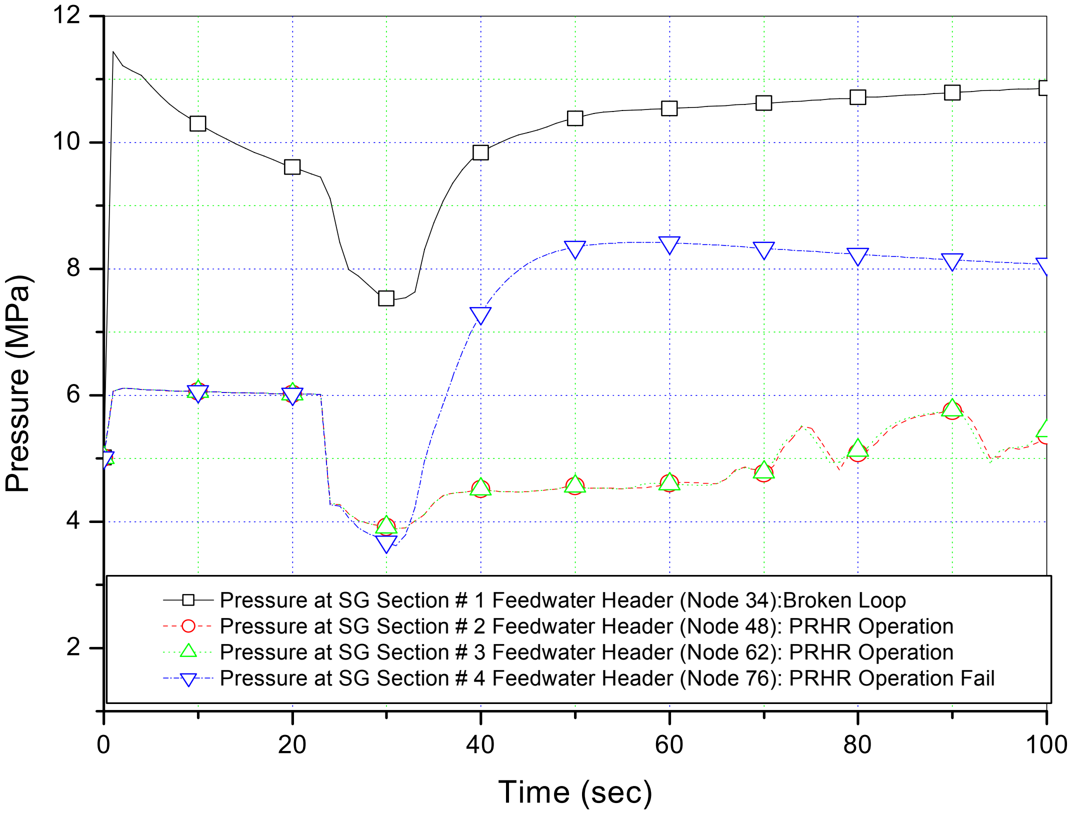


Figure VI.4. Pressure of SMART secondary system pressure during SGTR.

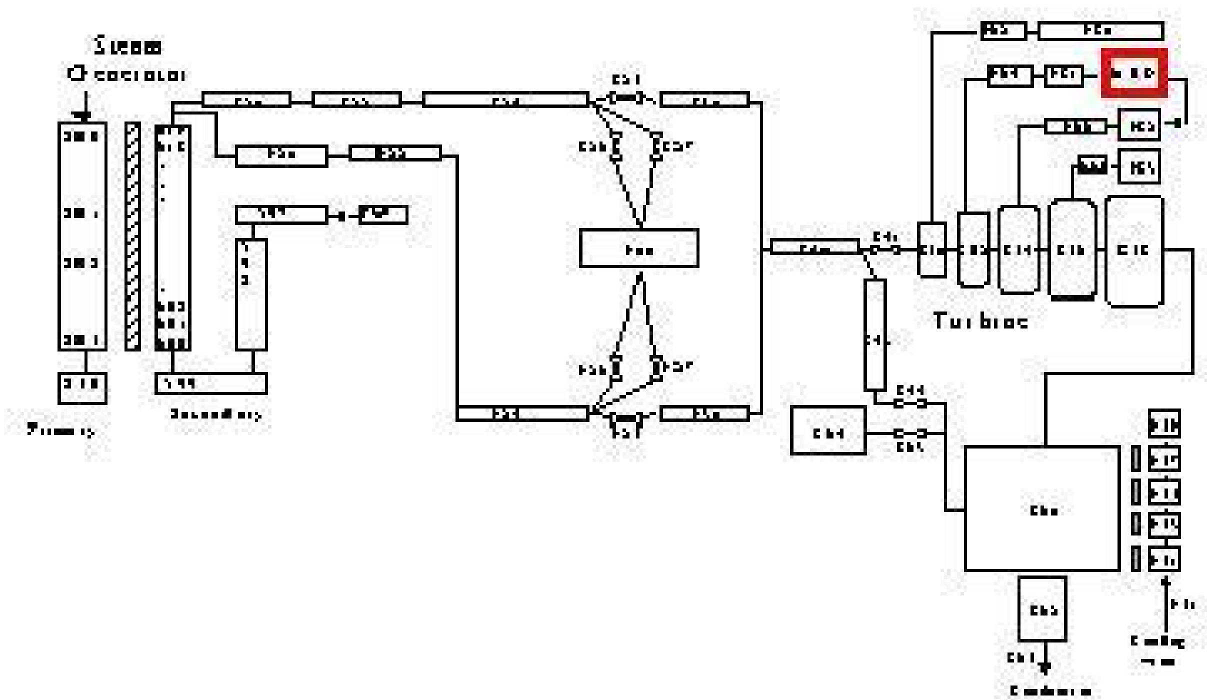


Figure VI.5: Control Volumes and Connections of Live Steam Systems and SG Modules.

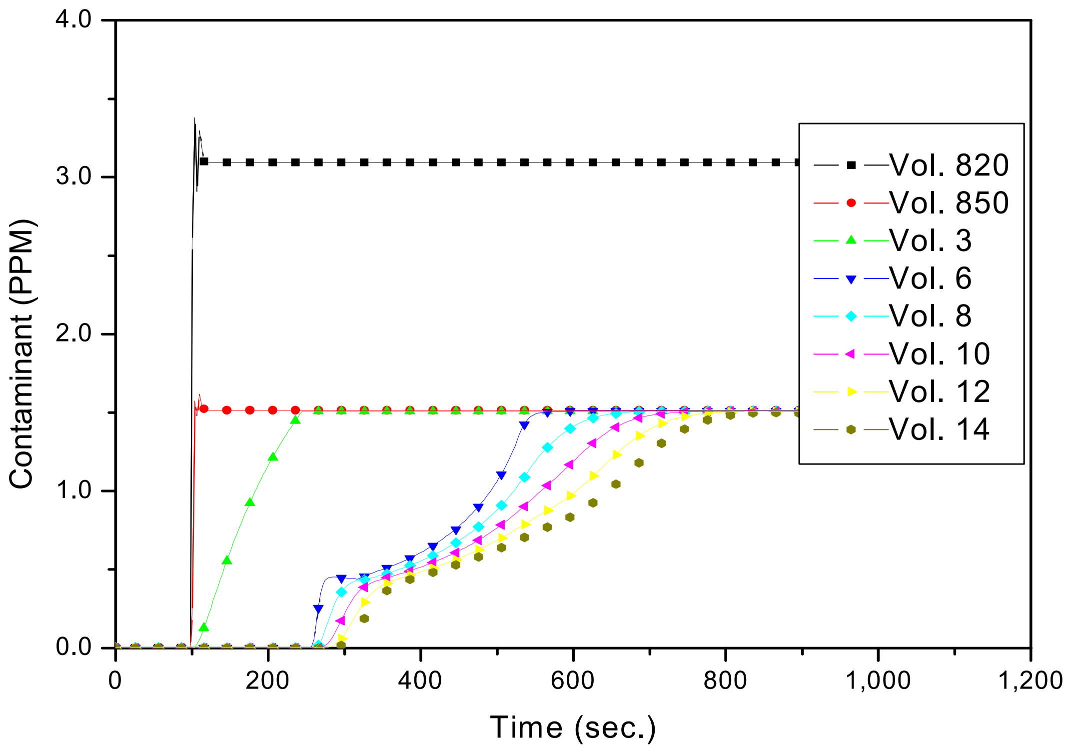
Except for this change, the nodalization remained same as that of the DESNU. Also, some modification was made for RETRAN-03 input. For desalination plant, only one unit of MED and its coupling system was modeled. The failure of the isolation barrier of steam transformer was modeled by a direct irruption of contamination in the first effect. The present model was not included the thermal vapor compressor and the flow path of the induced steam from final condenser. Basically the modeling scheme of the SMART desalination plant remained the same as that of the Reference [4]. Figures in Appendix III show the DESNU modeling used to simulate the transport of the radioactive contaminant of SMART plant.

VI.2.4.2 Simulation of the contaminant migration

An analysis on the multiple failure of the isolation barier, i.e. both of the steam generator tube rupture and steam transformer tube breaks was conducted to estimate the transit time of the contaminant using RETRAN-03. The model of the integrated plant needs some tuning for simulation both of the steady state and the transient. The tunning work and the transient were performed in accordance with the procedure described in the Reference[5].

The path for contaminant migration was modeled by the bypass valve which flows from the primary side of the steam transformer to the first effect of the desalination plant. The flow velocity which is the most important parameter affects the time for contaminant migration was tuned by adjusting junction loss coefficient between each effects. The transient senario similar to that of INVAP[4,5] was simulated. The system achieves the steady state and the simulation was conducted by triggering contamination source into the steam generator outlet (Vol. 820) at 100sec. After the contamination spreads on the whole secondary system, the contaminant allows into the desalination plant at 250 sec by simulating a rupture of heat transfer tube of the steam transformer

Figure VI.6 shows the result of the simulation. As shown in Figure, the contaminant reaches immediately to the turbine inlet (Vol. 850) at the half of the concentration by the flow of the other steam line and then contaminates gradually the primary side of the intermediate heat exchanger (steam transformer-Vol. 981). When the heat exchanger fails, the contaminant enters into the first effect (Vol. 6) and gradually spreads through the effects (Vol. 6 to 9) of the desalination plant, and then finally reaches to the water distribution grid. The estimated transit time of the contaminant is about 50.0 sec per effect. For the protection water distribution grid from the radioactivity contamination, the desalination plant should be isolated before the contaminant reaches to the water distribution grid. The isolation of the desalination plant from water distribution network may be triggered by the safety signal of the nuclear power plant and/or the additional radioactivity measurement in the first effect of the desalination plant.



FigureVI.6. Simulation of the Contaminant Migration of SMART Nuclear Desalination Plant.

Since the thermal vapor compressor which mixes the supplied steam with the induced steam from fhe final condensor was not modeled for simulation, the level of concentration in each effects reaches the same level as that of the steam transformer.

Although this is a preliminary qualitative analysis, the results of simulation may provide useful information on the safety of the nuclear desalination, i.e. required time for isolation of esalination plant from water distribution network. For the practical application, more studies on the parameters which affects the contaminant transit time are necessary.

VI.3. CONCLUSIONS

The objective of this CRP is to establish a set of optimal coupling parameters between SMART and the desalination plant which satisfies the diverse requirements imposed by water production and electricity generation. Through this project, the coupling scheme between SMART and desalination plant was selected and the major coupling parameters were optimized for the selected coupling scheme. Also the coupling scheme has been evaluated against the issues on the safety and economics of the nuclear desalination.

During this project year, the final report has been revised based on the design data of the SMART and the desalination plant. The Final report has been compiled in the Business

Collaborator Common Workplace of the Nuclear Power Technology Development Section, IAEA. In addition, a simulation of the contaminant migration was performed utilizing modeling tool DESNU developed by INVAP in the frame of CRP. Although the analysis is preliminary qualitative analysis, the results may provide useful information for the safety issues on nuclear desalination.

REFERENCES

[VI.1] CHANG, M.H., et al., Basic Design Report of SMART, KAERI/TR-2142/2002, Korea Atomic Energy Research Institute, Taejon(2002).

[VI.2] HWANG, Y.D., et al., Design concept and its Requirements of the Integrated SMART Desalination Plant, KAERI/TR-1761/2001, Korea Atomic Energy Research Institute, Taejon(2001).

[VI.3] HWANG, Y.D., et al., Optimization Study of the SMART Desalination Plant, KAERI/TR-2145/2002, Korea Atomic Energy Research Institute, Taejon(2002).

[VI.4] ALICIA, S., et al., "Analysis of Desalination System Models Relevant for the Safety Evaluation of a Nuclear Desalination Plant," Proc. of IDA World congress on Desalination and Water Reuse, San Diego, USA, Aug. 29–Sept. 3, 1999, IDA (1999).

[VI.5] WORKING MATERIAL, Optimization of the Coupling of Nuclear Reactors and Desalination Systems, IAEA-RC-719 (no. 7), IAEA, Vienna, 2003.

TABLE 1. SUMMARY OF MAJOR DESIGN PARAMETERS OF SMART

Reactor Name	: SMART
Reactor Type	: Integral PWR
Thermal Power	: 330 MWt
Electric Power	: 100 MWe
Design Lifetime	: 60 years
Fuel Material	: 4.95 w/o UO_2
Refueling cycle	: > 3 years
Core power density	: 62.6 W/cm^3
Reactor Pressure Vessel	
• Overall Height	: 10.6 m
• Outer Diameter	: 4.6 m
• Average Vessel Thickness	: 264 mm
Primary System	
• Cooling Mode	: Forced circulation
• Design Pressure	: 17 MPa
• Operating Pressure	: 15 MPa
• Core Inlet Temperature	: 270°C (291.6°C for PWR)
• Core Outlet Temperature	: 310°C (326.8°C for PWR)
Steam Generator	
• Type	: Helically-coiled once through
• Number of S/G Cassettes	: 12
• Feedwater Temperature	: 180°C
• Steam Pressure	: 3.0 MPa
• Steam Temperature	: ≥ 274°C
• Superheat	: ≥ 40°C
Pressurizer	
• Type	: Self-controlled
• Total volume	: 16.0 m^3
Control Element Drive Mechanism	
• Type	: Linear Pulse Motor Driven
• Number of CEDM	: 49
Main Coolant Pump	
• Type	: Glandless Canned Motor Pump
• Number of MCP	: 4
• Flowrate	: 2006 m^3/h
Secondary System	
• Main Steam Flow Rate	: 549,270 kg/h
• Feedwater Pressure	: 5.2 MPa
• Type of Feedwater Pump	: Multistage
• Type of Condenser	: Shell and Tube
Containment	
• Type	: Pressurized Concrete with steel lining
• Design Pressure	: 0.3 MPa
• Design Temperature	: 120°C

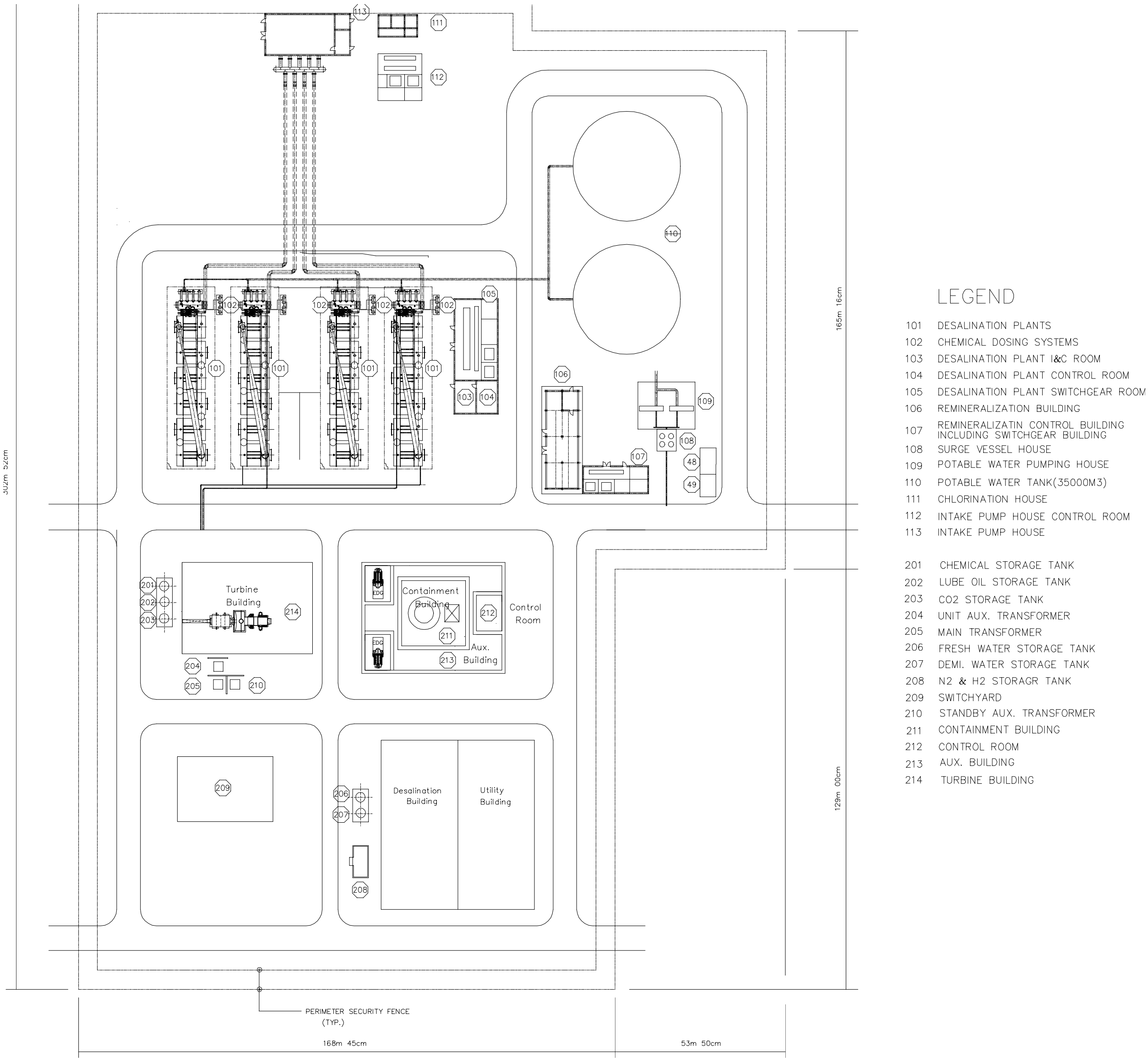


Figure 1. Site Plot plan for the Integrated SMART Nuclear Desalination Plant.

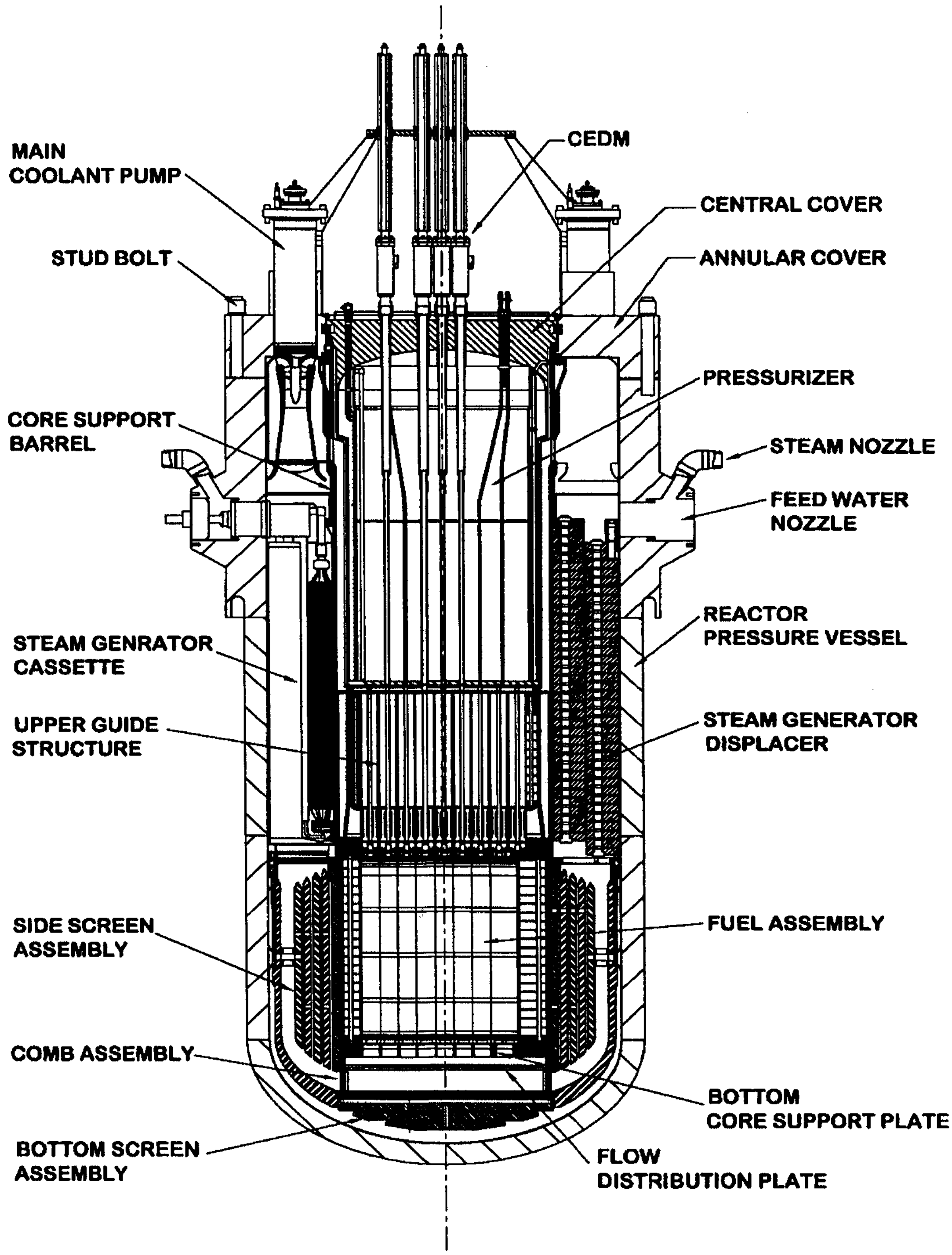


Figure 2. *SMART Reactor Vessel Assembly.*

TABLE 2. SMART FUEL / FUEL ASSEMBLY DATA

Design Parameter	Unit	KOFA	SMART
Cladding Tube			
• Outer Diameter	mm	9.5 ± 0.05	9.5 ± 0.05
• Inner Diameter	mm	8.22 ± 0.04	8.22 ± 0.04
• Min. Clad Thickness	mm	0.57	0.57
• Material	-	Zry-4	Zry-4
• Volume	cm^3	68.164	38.59
• Mass	kg	0.450	0.253
Active Length	mm	3,658 ± 6	2,000 ± 6
Fuel Rod Length	mm	3,847 - 2	2,189 – 2
Active Length Position	mm	11.5 – 0.8	11.5 – 0.8
Plenum Length	mm	166 ± (+7.6,-8)	166 ± (+7.6,-8)
Pellet Diameter	mm	8.05 ± 0.01	8.05 ± 0.01
Diametral Gap	μm	170 ± 50	170 ± 50
Pellet Density	g/cm^3	10.4 ± 0.15	10.4 ± 0.15
Fuel Material / Enrichment	w/o	UO_2 / 3.7	UO_2 / 4.95
Plenum Volume	cm^3	6.4 ± 0.6	6.4 ± 0.6
End Plug Material		Zry-4	Zry-4
Fill Gas Composition			
• He	%	-	≥ 96
• Ar	%	-	≥ 4
Total per FA	-	264	264

Fuel Rod / Assembly Thermal Hydraulic Data

Fuel Maximum Temperature:	°C	-	
- BOC			1881 (centerline)
- Cycle average			1255 (centerline)
Cladding Maximum Temperature	°C	-	396
Average Linear Heat Generation Rate	W/cm	178 (commercial PWR)	120

Fuel Assemblies Specifications

Fuel rod array, square	: 17 x 17 – 24 – 1
Total weight of fuel assembly, kg	: 370.842
Fuel rod to fuel rod, cm	: 21.11 x 21.11

Type of SMART Fuel Assemblies

Assembly Type	No. of Assembly	Fuel Enrichment (w/o U-235)	No. of Fuel Rods	No. of Al_2O_3-B_4C Rods	No. of Gd_2O_3-UO_2 Rods
A	8	4.95	224	28	12
B	28	4.95	240	20	4
C	21	4.95	236	24	4

Control Element Assemblies

Absorber elements, No. per assembly	: 24
Type, clad material	: Cylindrical, stainless steel
Poison material of CEA fingers	: Ag-In-Cd

Burnable Absorbers

Absorber material	Al_2O_3-B_4C	Gd_2O_3 - UO_2
Pellet diameter, cm	0.7646	0.805
B-10 contents, g/cm		0.01110 for FA Type A
		0.01588 for FA Type B and C
		0.02900 for FA type C

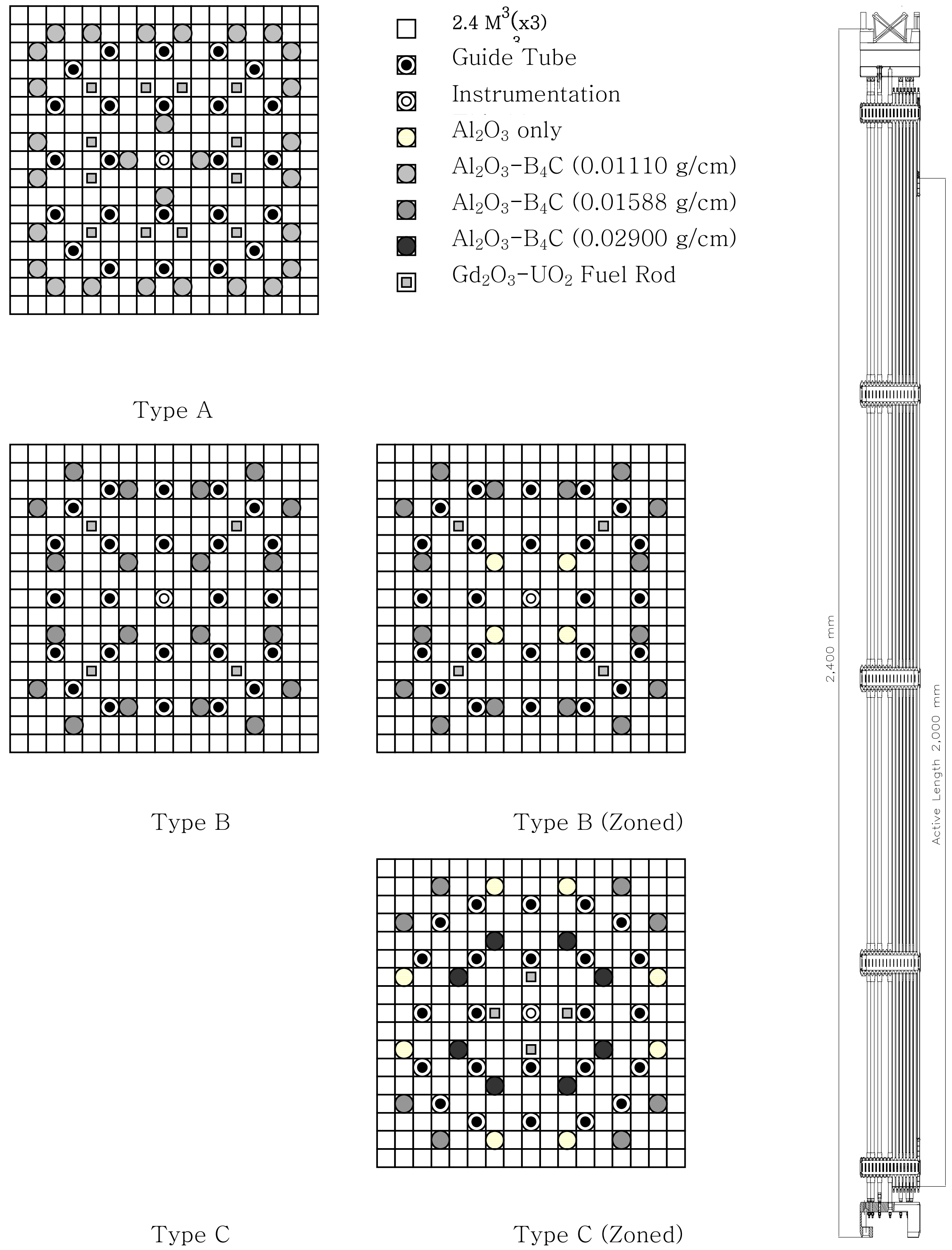
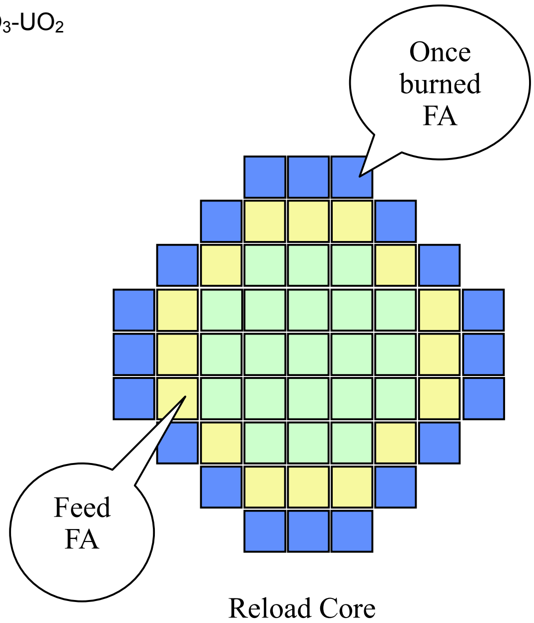


Figure 3. *Type of SMART Fuel Assembly.*

	J	H	G	F	E	D	C	B	A
1				A	B	A			
2			B	B	B	B	B		
3		B	B	C	C	C	B	B	
4	A	B	C	C	C	C	C	B	A
5	B	B	C	C	C	C	C	B	B
6	A	B	C	C	C	C	C	B	A
7		B	B	C	C	C	B	B	
8			B	B	B	B	B		
9				A	B	A			

A	4.95w/o U-235 / 28 Al_2O_3-B_4C Shim / 12 Gd_2O_3-UO_2
B	4.95w/o U-235 / 20 Al_2O_3-B_4C Shim / 4 Gd_2O_3-UO_2
C	4.95w/o U-235 / 24 Al_2O_3-B_4C Shim / 4 Gd_2O_3-UO_2

Single Batch Refueling Scheme:
All previous fuels are replaced by the new, fresh fuels.
Modified Single Batch Refueling Scheme:
Only feed FA are replaced, peripheral fuels (once burned FA) are re-positioned / reuse in the opposite direction.



Reload Core

Figure 4. *SMART Core Loading Pattern.*

TABLE 3. SMART CORE DESIGN DATA

Core Arrangement	
Number of fuel assemblies in core, total	: 57
Number of Control Element Assemblies (CEAs)	: 49
Number of fuel rod locations, total	: 15048
Number of fuel rods	: 13468
Number of Al_2O_3-B_4C poison rods	: 1288
Number of Gd_2O_3-UO_2 fuel rods	: 292
Total number of fuel rods (including gadolinia fuel rods)	: 13760
Spacing between fuel assemblies, fuel rod surface to surface, cm	: 0.394
Spacing, outer fuel rod surface to core shroud, cm	: 0.295
Hydraulic diameter, nominal channel, cm	: 1.18
Total flow area (excluding guide tubes), m^2	: 1.40
Total core area, m^2	: 2.64
Core equivalent diameter, cm	: 183.2
Core circumscribed diameter, cm	: 204.0
Fuel loading in active core, kgU	: 12470.9
Weight of zircaloy in active core, kg	: 4023.7
Fuel volume (including dishes and chamfers), m^3	: 1.40
Core T/H Parameters	
Core thermal power, MWt	330
Core avg. heat flux, kW/m^2	394.1
Core avg. LHGR, W/cm	199.9
Pressure, MPa	15.0
Core inlet temperature, °C	270
Vessel temperature rise, °C	40
Core avg. mass flux, $kg/m^2/s$	1010
Specific power, kW/kgU	26.462
Volumetric power density, kW/lt	62.60
3D-power peaking factor, HFP (based on LPD = 394.50 W/cm	3.29
Fuel thermal margin, %	15
Heated length, m	2.0

TABLE 4. SMART REACTOR COOLING SYSTEM AT FULL POWER

Thermal power of reactor, MWt	: 330
Coolant temperature at core outlet / inlet, °C	: 310.0 / 290.0
S/G outlet temperature, °C	: 268.5
PZR end cavity temperature, °C	: 74.0
Coolant flow rate via MCP, kg / s	: 1585.0
Coolant flow rate via S/G cassettes, kg / s	: 1540.0
Coolant flow rate via core, kg / s	: 1550.0
Developed head of MCP, MPa	: 88.57
Pressure in primary circuit (nominal / design), MPa	: 15.0 / 17.0
Coolant temperature rise at core, °C	: 40.0
Steam output, kg / s	: 152.4
Steam / feedwater pressure, MPa	: 3.0 / 5.0
Steam / feedwater temperature, °C	: 274 / 190

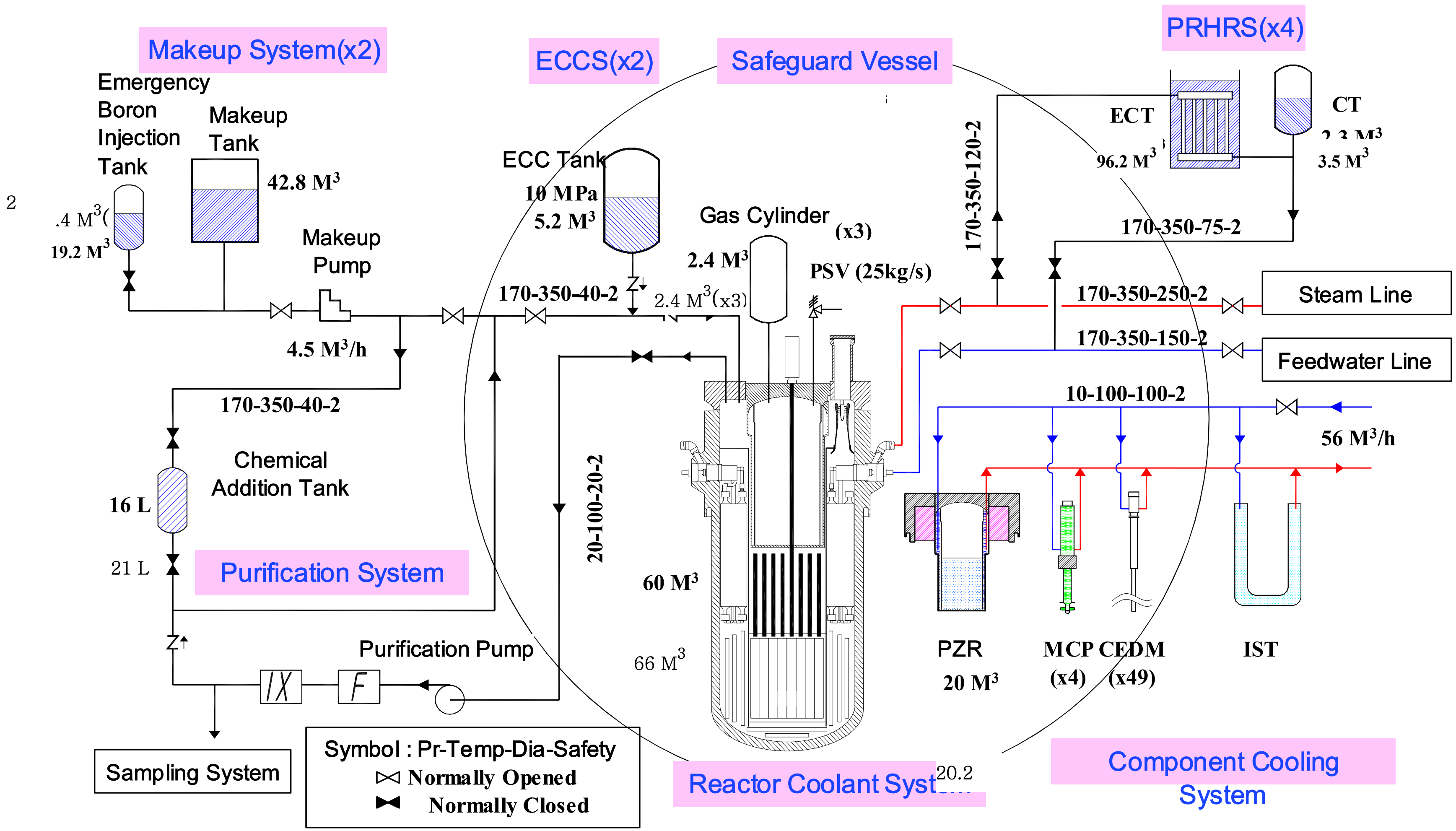


Figure 5. *SMART Nuclear Steam Supply System Schematic Diagram.*

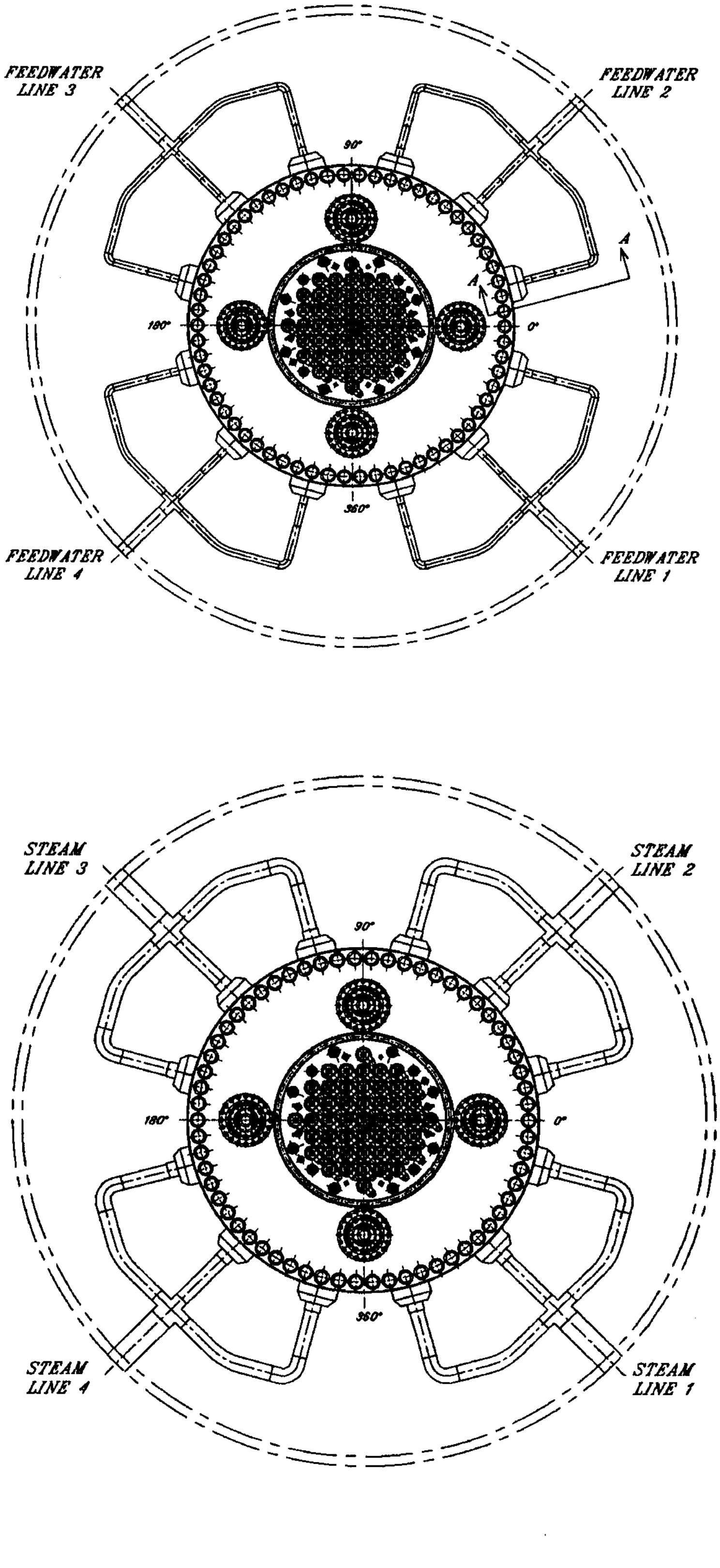


Figure 6. *Layout of Feedwater and Steam Pipelines at the SMART.*

TABLE 5. SMART MAIN COOLANT PUMP DESIGN SPECIFICATIONS

Type	: Canned motor axial pump
Design temperature	: 350 °C
Design pressure	: 17 MPa
Flow rate at 310°C, 15 MPa	: 2006 m^3/h, 392 kg/s (at 3600 rpm) 501.5 m^3/h, 98 kg/s (at 900 rpm)
Efficiency	: 85 % (at 2006 m^3/h)
Head	: 17.5 m (at 2006 m^3/h)
MCP power	: 225 kW
Motor type	: 2 pole squirrel cage induction motor
Rotational speed by inverter	: 900 rpm (in 10 sec.), 3600 rpm (in 40 sec.)
Weight	: 3500 kgf
Main material	: STS 321 (pressure barrier-ASTM standard)
Grid power	- 3 phase - 60 Hz, 440 VAC input - 15 Hz, 400 VAC output (900 rpm) - 60 Hz, 400 VAC output (3600 rpm)
Bearing type	: 2 journal and 1 thrust bearing
Bearing material	- Self-lubrication at high temperature and pressure in coolant - Silicon graphite

TABLE 6. SMART CEDM AND MAIN ASSEMBLIES DESIGN SPECIFICATIONS

Design temperature	: 350°C (operating temperature inside pressure vessel: 180°C)
Design pressure	: 17 MPa
Number of CEDM	: 49
Total stroke	: 2000 mm (active core height)
Min. moving distance	: 4 mm per pulse
Load carrying capacity	: 1000 N
Moving velocity	: 0–50 mm/sec and 250 mm/sec (gravity fall) for safety analysis group
CEA position indicator	: 4 channels of safety class

TABLE 7. SMART-S/G CASSETTE DESIGN SPECIFICATIONS

S/G Type	: Once-through, helically coiled, modular cassette for superheated steam production
Number of cassettes	: 12
Thermal capacity	: 27.5 MWt / cassette
Design temperature / pressure	: 350°C / 17 MPa
Feedwater temperature / pressure	: 180°C / 5.5 MPa
Steam temperature / pressure	: 284°C / 3.5 MPa
Structural material	: Titanium alloy (tubes) and STS 321 (header)
Primary / secondary coolant flow rate	: 128.3 / 12.7 kg / s
Heat transfer area	: 168.8 m^2
Axial height	: 2.8 m
Shell-side / Tube side pressure drop	: 2.57E-02 / 0.3 MPa
Cassette weight	: 2000 kgf
Tube bundle	
Material	: Titanium alloy
OD x thickness	: 12 x 1.5 mm
Average length	: 15.8 m
Number of tubes per cassette	: 324 (54 tubes each header)
Tube bundle design each module header	: 17 rows (different tube number each row)
Radial pitch	: 17 mm
Axial pitch	: 13.5 mm
Innermost coil diameter	: 0.182 m
Outermost coil diameter	: 0.726 m
Titanium alloy performance	Corrosion and Hydrogenation resistance
Module feedwater and steam header	
Number of module feedwater headers	: 6 in the bottom of each cassette via 6 tubes arranged inside a cassette central tube
Number of module steam headers	: 6 on top of each cassette
Material	: STS 321

TABLE 8. PRESSURIZER DATA AT FULL POWER CONDITION

Water temperature:	
- end cavity	: 121°C
- intermediate cavity	: 118°C
- upper annular cavity	: 308°C
Primary pressure	: 15.3 MPa
Required nitrogen mass	: 1130 kg
Nominal water level in end cavity	: 1.1 m
Volume of end/intermediate/upper annular cavity	: 12.46 / 1.54 / 7.70 = 20.2 m^3

TABLE 9. SMART SAFETY SYSTEMS DESIGN SPECIFICATIONS

Reactor shutdown system (RSS)	
Primary RSS: CEDM	
- Number of shutdown bank	: 16
- Material	: B_4C
Back-up RSS: Emergency Boron	
Injection System	: 2 x 100%
- Number of trains	: 6 m^3
- Volume of tank each train	: 30 kg / 1 kg water
- Density of boric acid each tank	: active (1 pump for 2 trains)
- Mode of operation	
Emergency Core Cooling System (ECCS)	
Number of trains	: 2 x 100%
Volume of tank each train	: 5.2 m^3
Pressurized gas	: Nitrogen
Gas pressure in tank	: 10 MPa
Mode of operation	: passive (rupture disc)
Passive Residual Heat Removal System (PRHRS)	
Number of trains	: 4 x 50%
Major components each train:	
- Emergency cooldown tank	: 50 m^3
- Heat exchanger	: -
- Compensating tank	: 2.3 m^3
Mode of operation	- Passive (natural circulation in emergency) - Active (pump in long-term cooling for refueling)
Containment Overpressure Protection System (COPS)	
1st physical barrier	: Steel containment structure
2nd physical barrier	: Concrete building
- design pressure	: 0.3 MPa
- design temperature	: 120°C
Mode of operation	: Passive - heat removal via cooldown tank - rupture disc and filtering system for overpressure
Safeguard Vessel (SV)	
Material	: Leak-tight steel made vessel
1st cooling method	: Water jacket in the upper part and HVAC
2nd cooling method	: Internal shielding tank
Design pressure, temperature	: 3 MPa, 250°C
Reactor Overpressure Protection System (ROPS)	
Number of trains	: 3
POSRV set point	: 17.5 MPa (open), 13.8 MPa (close)
Operation mode	: Steam discharging into the external shielding tank
Severe Accident Mitigation System (SAMS)	
Physical barrier	- Small air gap filled with water from make-up system - Internal and external shielding tank
Hydrogen removal	: Hydrogen igniters inside the safeguard vessel

TABLE 11. EVENT CLASSIFICATION OF SMART

Initiating Event	PC
1. Increase in the Heat Removal by Secondary System:	
- Decrease in Main Feedwater Temperature	PC-2
- Increase in Main Feedwater Flow	PC-2
- Increase in Main Steam Flow	PC-2
- Steam Line Break inside and outside Containment	PC-5
- Steam Line Break in Safeguard Vessel	PC-5
- Malfunction in Residual Heat Removal System	PC-2
2. Decrease in the Heat Removal by Secondary System	
- Loss of External Load	PC-2
- Turbine Trip	PC-2
- Loss of Condenser Vacuum	PC-2
- Inadvertent Closure of Main Steam Isolation Valve	PC-2
- Loss of Non-Emergency AC for Plant Auxiliary System	PC-2
- Loss of Normal Feedwater Flow	PC-2
- Main Feedwater Line Break inside and outside Containment	PC-5
- Main Feedwater Line Break in Safeguard Vessel	PC-5
- Loss of Component Cooling System	PC-2
- Component Cooling System Pipe Break	PC-4
3. Decrease in Reactor Coolant Flow	
- Complete Loss of Reactor Coolant Flow	PC-3
- Single Main Coolant Pump Rotor Seizure	PC-4
- Reactor Coolant Pump Shaft Break	PC-4
4. Reactivity and Power Distribution Anomalies	
- Uncontrolled Rod Control Cluster Assembly Bank Withdrawal from a subcritical or Low Power	PC-2
- Uncontrolled Rod Control Cluster Assembly Bank Withdrawal at Power	PC-2
- Single Rod Control Cluster Assembly Withdrawal	PC-3
- Single Rod Control Cluster Assembly Drop	PC-2
- Rod Ejection	PC-5
- Start-up of an Inactive Main Coolant Pump	PC-2
- Fuel Misloading	PC-3
5. Increase in RCS Inventory	
- Inadvertent Operation of the Make-up System	PC-2
6. Decrease in RCS Inventory	
- Inadvertent Opening of a Pressurizer Safety Valve	PC-3
- Small Break Loss of Coolant Accident	PC-4
- Steam Generator Tube Rupture	PC-3
- Feed / Steam Line Break inside Steam Generator	PC-4
- Abnormal Operation of ECCS	PC-2
7. Radioactive Release from a Subsystem or Component	
- Release or Break of Radioactive Waste System	PC-4
- Fuel Handling Accident	PC-5
8. Anticipated Transients without Scram (ATWS)	

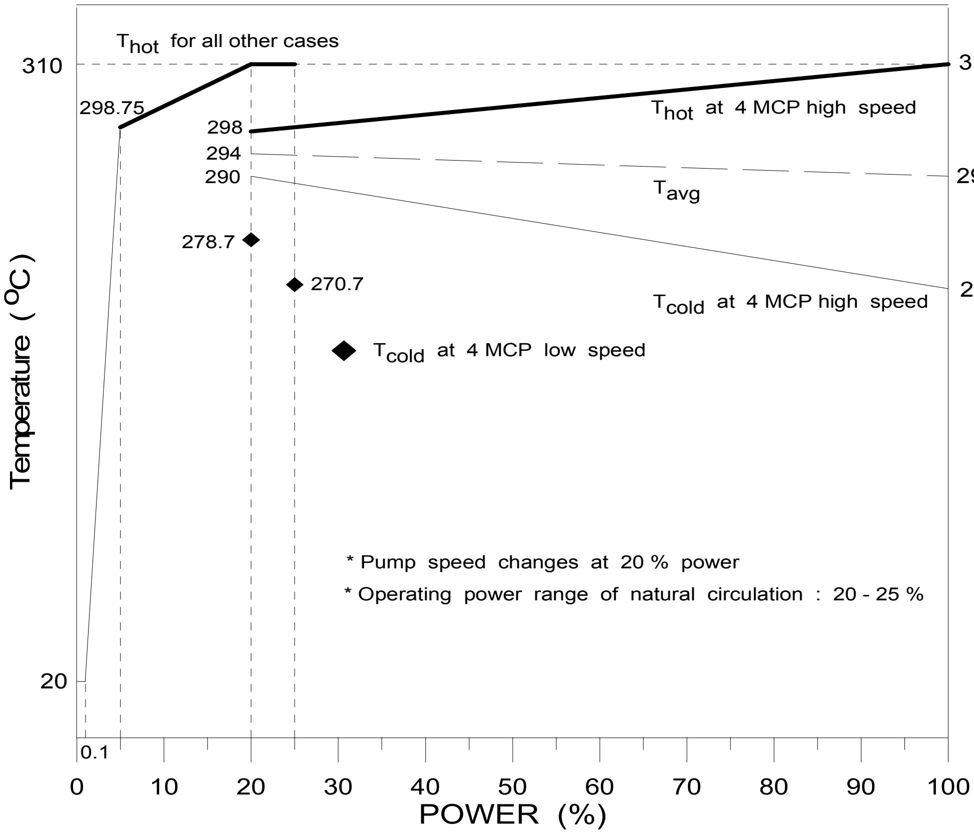


Figure 7. Control Logic of NSSS Programmed T_{hot} Temperature.

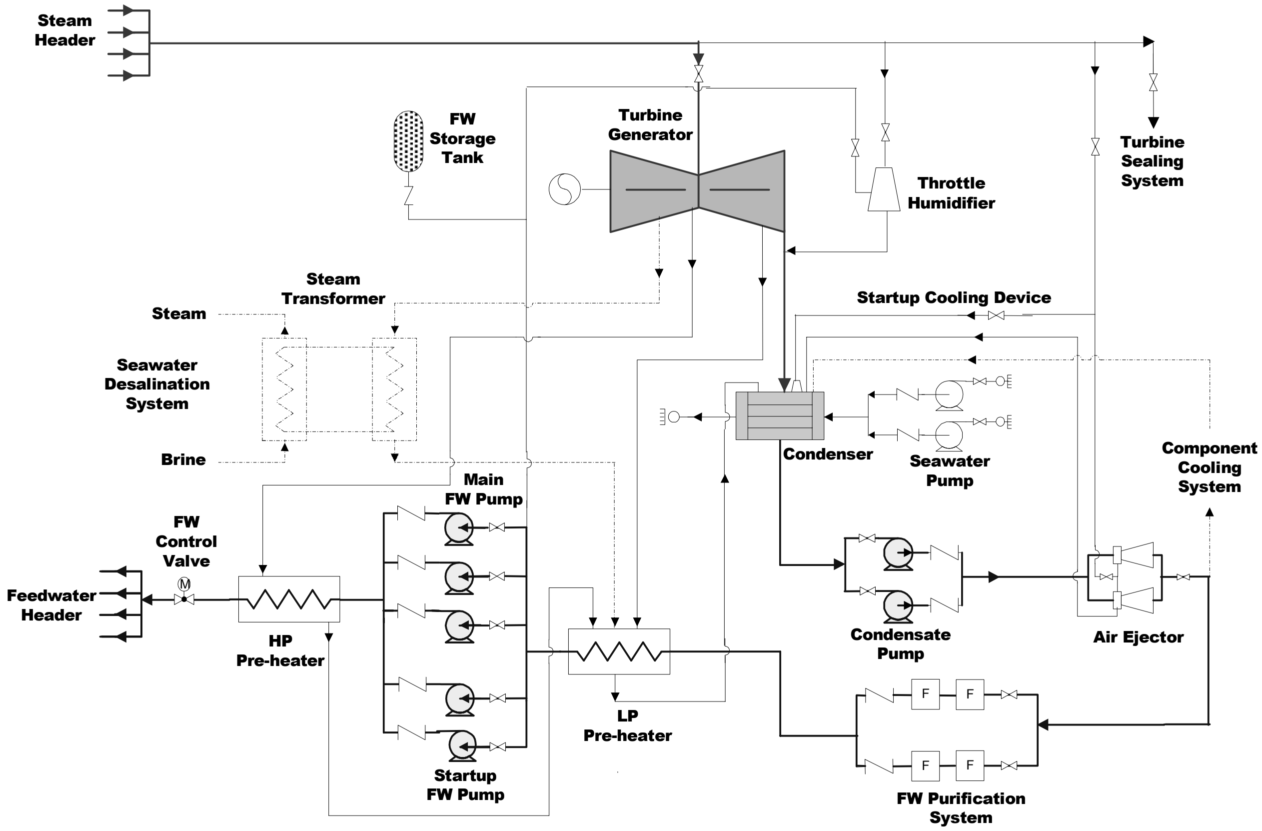


Figure 8. SMART Secondary System Diagram.

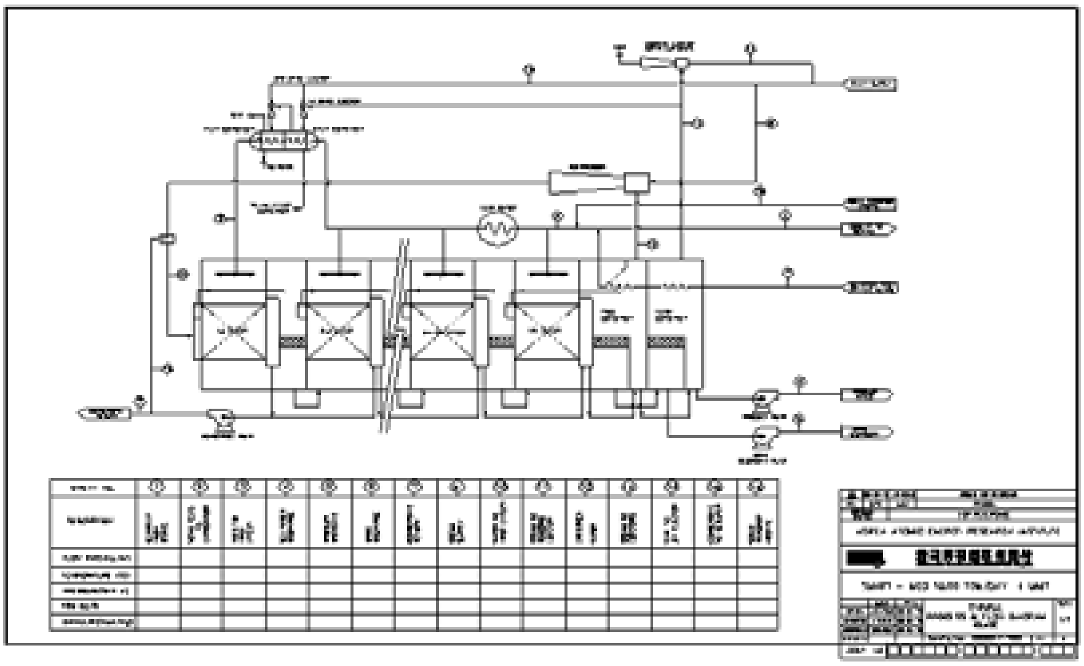


Figure 1. Flow Diagram of MED-TVC Process.

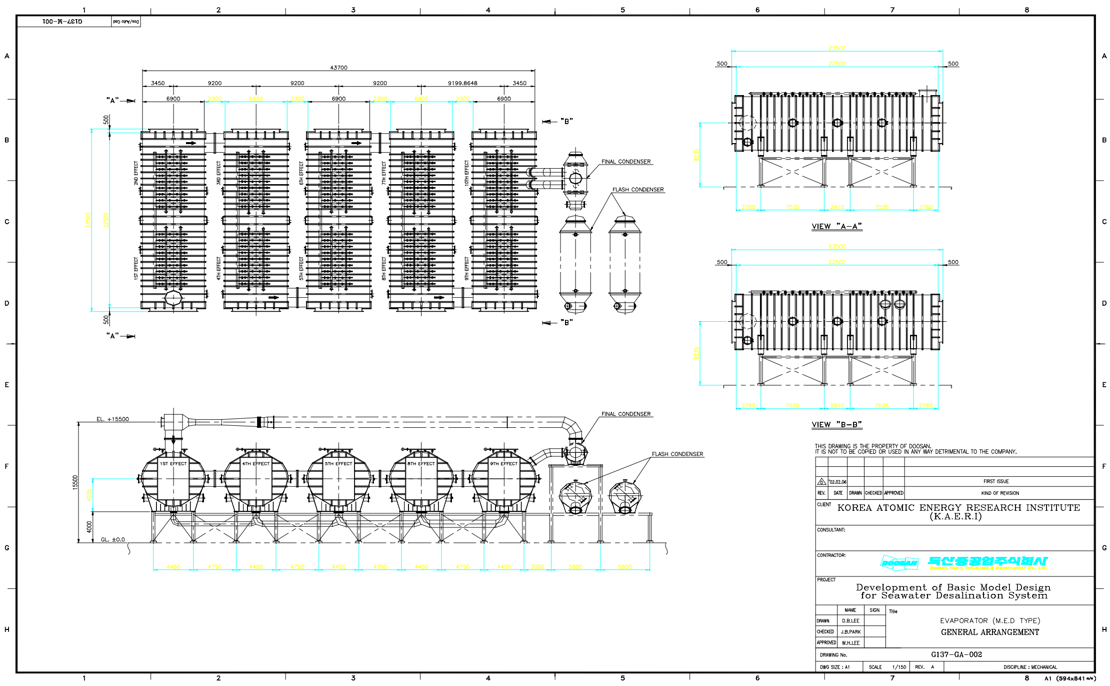


Figure 2. General Arrangement of the Desalination Unit of the SMART Plant.

Design Specification of SMART Nuclear Desalination Plant

The SMART desalination plant was designed to produce the net output of 10,000 m3/day of desalinated water per unit at the following conditions:

• Number of MED-TVC desalination units	4	
• Seawater Temperature	33	°C
• Maximum Seawater Salinity	40,000	ppm
• Maximum Brine Temperature	65	°C
• Maximum Distillate Temperature	40.0	°C
• Net output of distillate measured at the discharge of the distillate pump	10,000	m^3/day/unit
• Total net output of distillate measured at the discharge of the distillate pump	40,000	m^3/day
• Total steam required to produce distillate for a) & b) below, which is the sum of:	5.93	kg/sec
a) Steam from the steam header to the Thermo-compressor	5.66	kg/sec
b) Steam to air ejectors	0.27	kg/sec
• Performance Ratio, as defined in specification	19.6	kg of distillate per 2326 kJ of heat
• Minimum controllable output	5,000	m^3/day/day
• Total flow requirement of seawater per unit	2,049	m^3/hr/unit
• Feed makeup flow rate	1,855	m^3/hr/unit
• Feed makeup flow rate at reduced load	928	m^3/hr/unit
• Seawater flow rate to ejector condenser	185.5	m^3/hr/unit
• Brine blow down flow rate	1,427	m^3/hr/unit
• Maximum/Minium fluid velocity in tubes at the net output		
- Evaporator condenser	2.1/1.1	m/s
- Distillate cooler/preheater	2.1/1.1	m/s
- Ejector condensers	2.1/1.1	m/s
- Heat gain exchangers	2.1/1.1	m/s
• Seawater temperature	33.0	°C
• First effect maximum brine temperature	65.0	°C
• Last effect minimum brine temperature	52.7	°C
• Brine blow down temperature	41.0	°C
• Maximum distillate temperature	40.0	°C
• Condensate return temperature	66.8	°C

- Maximum temperature loss due to pressure drop through demisters

Effect	^{o}C
1	0.14
2	0.14
3	0.15
4	0.16
5	0.16
6	0.18
7	0.19
8	0.20
9	0.21
10	0.22

• Vapor pressure in first effect	0.25	bar a
• Steam pressure to thermo compressor	8.0	bar a
• Steam temperature to thermocompressor	170.6	^{o}C
• Suction vapour pressure to thermocompressor	0.139	bar a
• Suction vapour temperature to thermocompressor	52.4	^{o}C
• Suction vapour flow to thermocompressor	25,371	Kg/hr
• Delivery pressure from thermo compressor to effect no. 1	0.275	bara
• Delivery temperature from thermo-compressor to effect no. 1	66.9	^{o}C
• Delivery flow from thermo compressor to effect no. 1	46,229	kg/hr
• Maximum pressure of seawater at inlet of the 2nd flash condenser	5.77	bar a
• Minimum pressure of seawater at the evaporator condenser inlet	3.0	bar a
• Average brine level in each effects	150	mm

- Heat transfer coefficient in each effect

Effect	Wh/m2 K
1	3,909
2	3,783
3	3,818
4	3,790
5	3,760
6	3,730
7	3,700
8	3,669
9	3,637
10	3,579

• Maximum TDS of distillate at evaporator outlet	25	mg/l

• Maximum TDS of brine in first effect	52,000	mg/l
• Maximum TDS of brine blowdown	52,000	mg/l
• Distillate pH at evaporator outlet	6~7.5	
• Scale control dosing rate related to feed Make up	5.54	mg/l

EVAPORATOR

Type of evaporator		Circlular
Number of evaporator modules		5
Overall dimensions including galleries:		
- Length	m	38.0
- Width	m	32.0
- Height	m	12.0
Number of effects		10

Evaporator Shell

Material		A240-316L
Effect length	m	8.2
Effect width	m	8.0
Effect height overall	m	12.0

Tubes

Number per effect		27,726
Material		
Top three tube rows		B338-Gr.2
Remainder of tubes		B111-C68700
Length (effective)	m	7.32
Outside diameter	mm	19.05
Effective tube surface area per effect	m^2	12,138

CONDENSERS

Final Condenser

Material		
Shell		A240-316L
Tubes		B338-Gr.2
Tube thickness	mm	0.5
Tube plate thickness	mm	20
Tube length (effective)	mm	9,060-9,080
Tube outside diameter	mm	28.57
Tube surface area	m^2 /unit	457

Flash Condenser

Material		
Shell		A240-316L
Tubes		B338-Gr.2
Tube thickness	mm	0.5
Tube plate thickness	mm	20
Tube length (effective)	mm	6035/6035
Tube outside diameter	mm	31.75/31.75
Tube surface area	m^2/unit	1950/1650

Ejector Condensers

Material		
Shell		SS904L
Tubes		ASTM B338, Gr2
Tube thickness	mm	0.5
Tube length (effective)	mm	3657.6 / 1219.2
Tube outside diameter	mm	19.05
Tube surface area	m^2	35.3/6.5

THERO-COMPRESSOR

Design steam pressure	bar a	8.0
Design steam Temperature	°C	170.6
Rated steam flow	kg/h	46,230
Suction vapour pressure	bar a	0.1397
Suction vapour Temperature	°C	52.46
Suction vapour flow	kg/h	25,371

AIR EJECTOR

		First stage	Second stage
Number per stage		1	1
Design steam pressurebar	Bar	8.0	8.0
Design steam temperature °C	°C	170.6	170.6
Vapour/non-condensable gas ratio		5.0	2.49

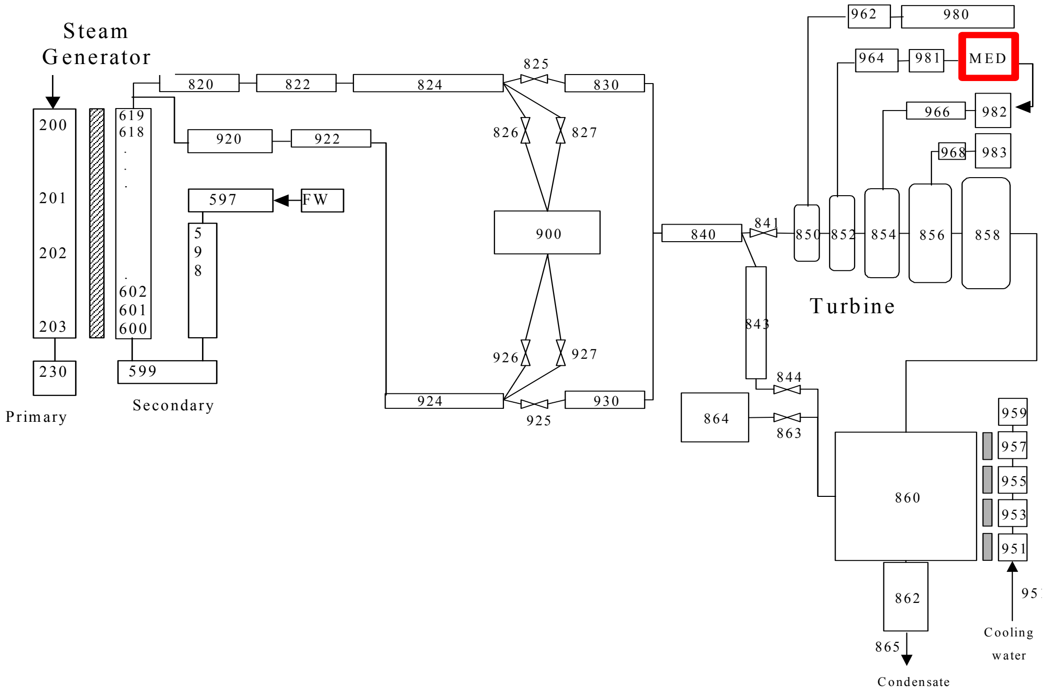


Figure 1: Scheme of the NPP-MED coupling.

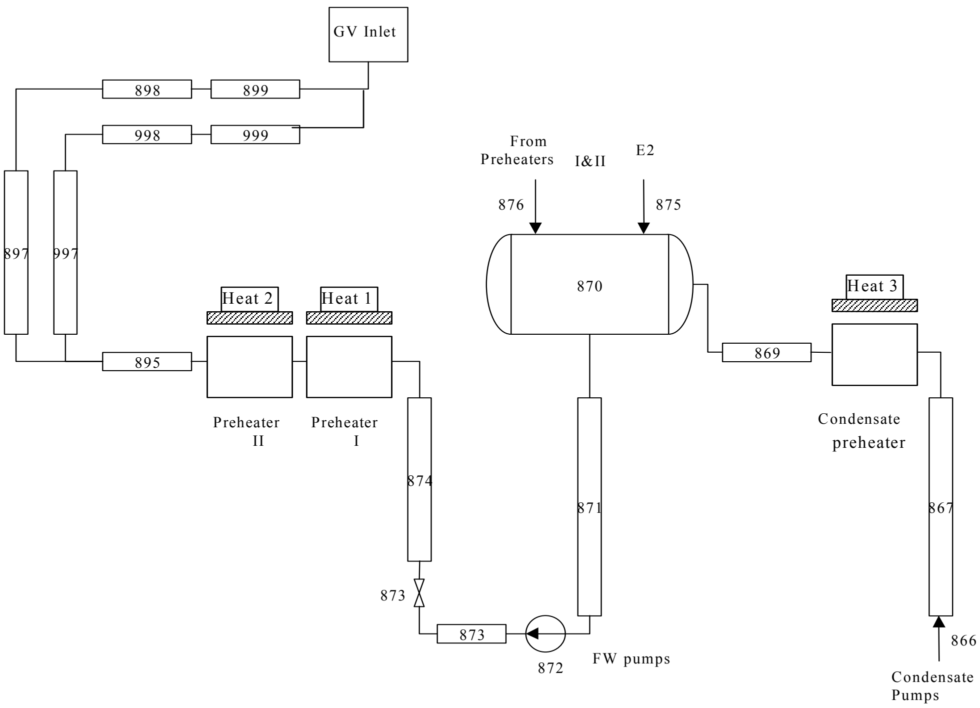


Figure 2: Scheme of the feed-water loop.

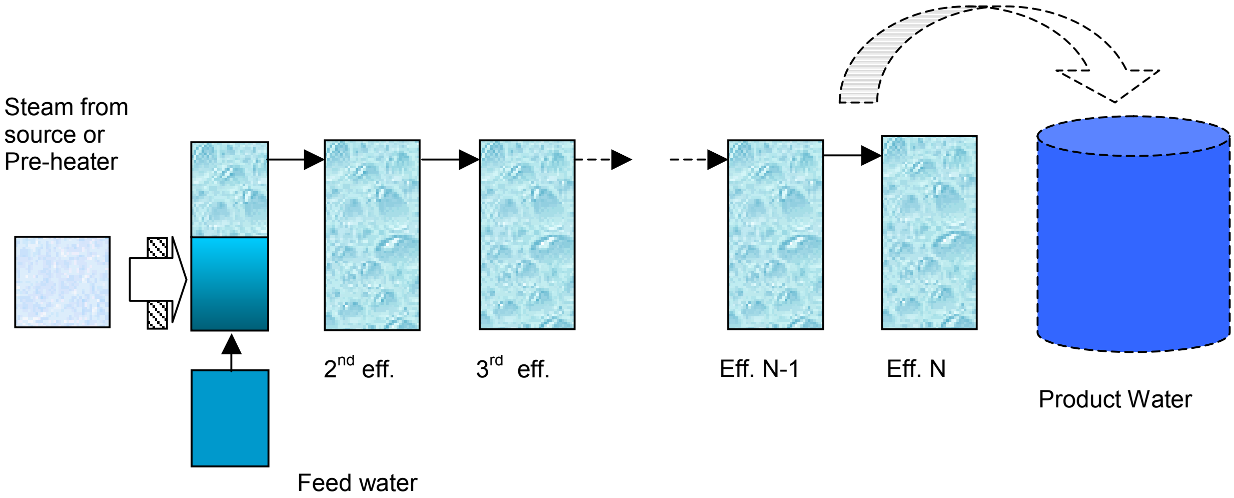


Figure 3: Scheme of the control volumes of MED Desalination Plant model.

REFERENCE

[VI.1] WORKING MATERIAL, Optimization of the Coupling of Nuclear Reactors and Desalination Systems, IAEA-RC-719 (no. 7), IAEA, Vienna (2003).

OPTIMISATION OF THE COUPLING OF NUCLEAR REACTORS AND DESALINATION SYSTEMS IN MOROCCO

M. Abet
Centre National de l'Energie des Sciences et
des Techniques Nucléaires, Morocco

VII.1. BACKGROUND

In the last years Morocco experienced a continuous dry seasons which put more stresses on his limited water resources mainly in the northern regions and in the Sahara provinces. In the later region population are supplied mainly with desalted water through desalination units operated by the potable water utility (ONEP).

Therefore Morocco has been considering for a number of years the desalination of seawater for the production of the potable water. bout 10 000 m^3/d of drinking water are produced using MED, RO and VC desalination processes. The phosphate industry are producing 13 000 m^3/d using MSF and VC processes.

Concerning the energy supply, the limited energy resources has encouraged Morocco in the early eighties to consider other alternatives and launched a feasibility study for the introduction of a nuclear power plant into the national electrical grid. The study showed that the commercially proven large size reactors at that time could not be integrated in the grid due to its limited capacity.

Nevertheless the national electrical utility is pursuing its efforts to introduce nuclear energy in the country. Presently the feasibility study is being updated and a bid invitation specifications is being prepared with the help of the IAEA experts.

In response to the increasing need for energy and water, Morocco participated together with the north African countries to the IAEA regional project on the feasibility study on seawater desalination by using nuclear energy.

Subsequently, Morocco carried out a feasibility study for the construction of a demonstration plant for seawater desalination using a 10 MW Nuclear Heating Reactor with China and IAEA. The study was completed in 1998 and showed that the water production cost is a bit higher due to a small scale of the plant and high discount rate.

VII.2. CRP PROJECT

As part of the interest given to the nuclear energy, our participation to this CRP seeks to investigate all the possibilities to achieve an optimal coupling with emphasis on the others possibilities which was not taken into considerations in the previous studies taking into account the recent development in nuclear reactors and desalination technologies.

To that end we selected two sites: Agadir and Laayoune, which have a growing need for potable water. Projections of the population and water demand have been estimated for the years beyond 2015.

At a first stage, we studied the effect of site specific considerations on the water cost and found that an increase of seawater temperature increases the cost in case of RO and decreases it in case of MED. On the other hand the change in TDS would rise slightly (around 4%) the water cost in case of RO. Finally the size of the installation affects the water cost.

Secondly we performed an economic evaluation using DEEP code by adopting two scenarios for both sites. The first scenario consists of considering nuclear option to supply a portion of water need while the second one to supply all the water need.

For both sites, the water produced by the nuclear reactors and by fossil fuel is in the same range. Among the nuclear configurations, and as consequences of the limitations of the DEEP code, we found that a 600 MWe PWR coupled to RO process gives the most economic water cost and an adequate amount of energy which could be easily integrated in the electrical grid. The coupling of GTMHR reactor with a distillation unit is planned for more investigation with the French Atomic Energy Commission (CEA).

The limited energy resources of the country and the estimated energy demand beyond 2020 are favourable for the introduction of a 600 MWe SMPR into the electrical grid. This figure has been used in the estimation of the water need and the investigation of using nuclear reactors in the two scenarios. Consequently the results obtained in the second part of this study showed that the installation of a 600 MWe PWR in the site of Agadir could produce more than 300 000 m^3/d at competitive cost and provide an average of 517 MWe while in case of site of Laayoune the figures are respectively 72000 m^3/d and 598 MWe.

The use of nuclear energy to produce all the water need could be advantageous in case where a part of desalted water are produced by conventional means. Also it could be an alternative to supply partially or totally the water need.

ANNEX VIII

UTILIZATION OF RUSSIAN SMALL-SIZED NUCLEAR REACTORS AS AN ENERGY SOURCE FOR NUCLEAR DESALINATION COMPLEXES: OPTIMIZATION OF COUPLING DESIGN, PERFORMANCE AND ECONOMIC CHARACTERISTICS

Yu.D. Baranaev, A.A. Pekanov, D.E. Skripnikov (SSC RF IPPE, Obninsk)
A.A. Romenkov, V.A. Shishkin, Yu.A. Mishanina, A.N. Achkasov, V.N. Pepa (RDIPE, Moscow)
V.I. Polunichev, S.A. Fateev (OKBM, Nizhny Novgorod)
V.G. Shamanin (JSC MALAYA ENERGETICA, Moscow)
V.L. Podberezny, L.V. Melinova (JSC NPP MASHPROM, Ekaterinburg)

VIII.1. INTRODUCTION

In the framework of the CRP Optimization of the Coupling of Nuclear Reactors and Desalination Systems, since 1988 Russian participants carried out the project Utilization of Russian Small-Sized Nuclear Reactors as an Energy Source for Nuclear Desalination Complexes: Optimization of Coupling Design, Performance and Economic Characteristics.

The purpose of the project is to study the technical and economic effectiveness of the use of Russian small nuclear reactors and desalination plants as constituents of nuclear desalination complexes, and economic effectiveness of these nuclear desalination systems.

In the framework of the project the principal areas were:

- optimization of reactor plant technical parameters important from the standpoint of desalination plants coupling
- assessment and upgrading of the available methodology and DEEP code for technical and economic evaluation of nuclear desalination complexes
- analysis of water and electricity cost sensitivity to variations of design and performance parameters.

Nuclear reactor considered in the study are Russian small reactors being now under development: heating reactor RUTA, co-generation PWRs KLT-40C and NIKA-70.

VIII.2. NUCLEAR DESALINATION TECHNOLOGIES WITHIN THE CRP

VIII.2.1. Nuclear desalination complexes with RUTA reactors

VIII.2.1.1. Main characteristics of RUTA-55 and RUTA-70 reactor plants

RUTA is a water-cooled and water-moderated pool-type reactor with the gas plenum under atmospheric pressure in the upper volume of the vessel. All primary circuit components are integrated in the reactor tank.

Heat with hot water is supplied to the consumer through the intermediate circuit and is being further used for heating in network heat exchangers of the tertiary circuit water. The secondary, or intermediate, circuit of the RP consists of two independent loops. The tertiary circuit is the consumer circuit: for NHP it means the heating network, for nuclear desalination complex (NDC) the tertiary circuit is a system of steam generating loops destined for heating the desalination modules.

In RUTA-55 reactor the core is cooled due to coolant natural circulation. For the RUTA-70 reactor the natural circulation mode was adopted for operation at a power level below 70% N_{nom}; for the range of power 70–100 %N_{nom} the operation with forced circulation in the reactor is envisaged. Forced circulation in the reactor circuit is to be considered as an optimization problem.

VIII. 2.1.2. RUTA-70 Reactor plant safety concept

Enhanced safety level of the RUTA reactor is based on the following inherent features:

- no excessive pressure in the reactor tank;
- large reactor thermal inertia thanks to vast amount of water (700 m^3) in the reactor tank;
- integral arrangement of the primary circuit;
- natural circulation of coolant in the primary circuit;
- low power density in the core;
- negative reactivity feedbacks result in spontaneous reduction or self-restriction of the reactor power irrespective of the control rod positions;
- two pressure barriers ($P_1<P_2<P_3$) prevent ingress of radioactivity from primary into customer circuits.

VIII.2.1.3. Coupling schemes of nuclear desalination complex on the basis of RUTA reactor plant

The DOU GTPA-type desalination plants operating in accordance with the principle of multi-effect distillation (MED technology) are used within the RUTA-equipped NDC. Steam is the heating agent in evaporating sections of DOU GTPA. In relation to the scheme integrating the reactor and desalination plant the tertiary circuit is a steam-generation circuit (SGC), which is a system of steam generation loops connected in parallel, their number corresponding to the number of distillation desalination plant (DDP) within NDC. Each SGC loop provides steam for the relevant desalination plant

Steam is generated as a result of the tertiary circuit water flashing in a multi-stage self-evaporator, whence in parallel flows it is fed for heating the DDP head effects. The self-evaporator is arranged in direct proximity of the DDP evaporating effects cascade, thus reducing to a minimum the temperature head loss for steam transport. There are no stop and control valves in the “self-evaporator stage — DDP effect” steam lines, as the self-evaporator, being integrated in the tertiary circuit, is simultaneously an element of the DOU GTPA.

VIII.2.2. Nuclear desalination complexes with KLT-40C reactors

VIII.2.2.1. Floating power unit with KLT-40C reactors

At present in the Russian Federation the work is nearing completion on the project of Small Nuclear Heat and Power Plant (ATES) which is expected to be used as a power and heat source in Russian regions with arctic and mild climatic conditions (the basic project). The ATES could be also employed as a power and heat source for desalination complexes in freshwater-shortage regions outside the Russian Federation. The ATES consists of the Floating Power Unit (FPU) on based KLT-40C reactor and shore facilities and hydraulic structures. The key component of the ATES is the FPU, which generates electricity and heat and transfers the products to the consumers through the shore infrastructure.

VIII.2.2.2. KLT-40C Nuclear reactor plant

KLT-40S NSSS had been developed on the basis of design experience and long-term (more than 160 reactor-years) operation of the Russian nuclear ice-breakers with similar reactor plants under the conditions of Russian North.

The reactor plant equipment comprises the reactor, four steam generators and four primary pumps joined with short load-bearing nozzles of “tube-in-tube” type in a compact steam generating unit. The unit is installed in a metal-water shielding tank arranged in the containment. Major rated technical parameters of KLT-40C are given in Table VIII.1.

VIII.2.2.3. Safety concept

The KLT-40C reactor safety is based on proven engineering solutions:

- PWR-type reactor being the most widely used in the world practice is applied;
- reliable and proven elements are used in the reactor plant, along with elimination of “weak points” revealed by operation experience of analogs,
- diversed and redundant systems are used for the reactor emergency shutdown and passive and active heat removal.

2.2.4. Variants of nuclear desalination complexes with KLT-40C reactor plant

Two approaches to the use of KLT-40C reactor plant with the Nuclear Desalination Complex have been chosen by Russian participants of the CRP:

- Studying the variants of the KLT-40C reactor use with the Desalination Complex without a strict relation to the project of floating Nuclear Power Plant which is currently under development
- The principal condition presumed in the course of coupling schemes examinations is full retention of both the reactor plant and steam turbine generator plant configurations, and the design and parameters of the system of heat supply to the district heating circuit used in the FPU with KLT-40C.

TABLE VIII.1. MAJOR RATED TECHNICAL PARAMETERS OF KLT-40C NSSS

Thermal Capacity, MWt	150
Steam-Generating Capacity, tons/hour	240
Primary Circuit Pressure, MPa	12.7
Steam Pressure at SG Outlet, MPa	3.73
Superheated Steam Temperature, ºC	290
Service Life, years	35 ÷ 40
Overhaul Period, years	10 ÷ 12
Non-Removable Equipment Lifetime, thousand hours	240 ÷ 300
Removable Equipment Lifetime, thousand hours	80 ÷ 100
Continuous Operation Period, hours	8000

VIII.2.2.4.1. Coupling schemes of nuclear desalination complexes with KLT-40C

Coupling with distillation desalination plants

The following variants of coupling schemes were considered for the Complex:
- Variant of coupling with the use of turbine plant steam extraction 2×25 Gcal/hour for the distillation plant (**Variant 1**);
- Variant of coupling with the use of turbine plant steam extraction and a peak heat of the intermediate loop water for the distillation plant (full thermal power of the steam extraction Q = 2×25 + 2×18 = 86 Gcal/hour, **Variant 2**);
- Variant of coupling with the use of a back-pressure turbine and heat from main condenser for DOU (**Variant 3**). The Co-Generation Complex comprises the reactor plant, the back-pressure turbine, a condenser and DOU plant.

Coupling with membrane technologies

The following variants have been considered:

1) No seawater preheat:
   - Seawater flow, which is directed to the RO plant, is equal to the cooling seawater flow to the turbine condensers (**Variant 4**);
   - With maximum desalinated water output of RO plant (**Variant 7**).
2) With seawater preheat in condensers of the turbines working in the condensing mode:
   - With seawater flow to the RO plant after the turbine condensers (**Variant 5**).
   - With maximum desalinated water output of RO plant (**Variant 8**). The distinctive feature is directing a part of seawater flow to the RO plant without preheat. The total preheat value equals to 4.5^{o}C.
3) With seawater preheat in turbine condensers and with the use of heat form the turbine steam extractions (**Variant 6**).

Coupling with hybrid technologies

The following variants are considered:

1) Using a back-pressure turbine and heat from the condenser to the DOU plant, and the electricity produced for the RO plant **(Variant 9**);
2) Using steam from the turbine steam extractions for the DOU, and the rest of electricity is used for the RO plant with seawater preheat in the condenser:
   - With the use of 2×25 Gcal/hour turbine steam extraction and 2×25 + 2×18 = Gcal/hour for the DOU plant and seawater flow to the RO plant after turbine condensers (**Variant 10** and **Variant 11**).
   - With the use of 2×25 Gcal/hour of turbine steam extraction and 2×25 + 2×18 = Gcal/hour for the DOU plant, and with maximum capacity of the RO pant **(Variant 12** and **Variant 13).**

Principal characteristics of the Complex calculated for **Variants 1** through **13** are summarized in Table VIII.2.

TABLE VIII.2. PRINCIPAL CHARACTERISTICS OF VARIANTS OF NUCLEAR DESALINATION COMPLEXES WITH KLT-40C REACTOR PLANT

Characteristic	Variant												
	1	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**	**13**
DOU Plant Capacity, m^3/day	50,400	84,000	218,400						218,000	50,400	84,000	50,400	84,000
RO Plant Capacity, m^3/day				108,000	108,000	108,000	324,000	324,000	120,000	108,000	108,000	300,000	276,000
Number of RO Modules				9	9	9	27	27	10	9	9-	25	23
Number of DOU Plants	3	5	13						13	3	5	3	5
DOU Plant Specific Heat Consumption, kW-hour/m^3	28.78	28.78	28.78						28.78	28.78	28.78	28.78	28.78
DOU Plant Electric Power Specific Consumption, kW-hour/m^3	0.92	0.92	0.92						0.92	0.92	0.92	0.92	0.92
RO Plant Electric Power Specific Consumption, kW-hour/m^3				4.62	4.04	4.03	4.62	4.51	4.49	4.04	4.04	4.49	4.46
Hybrid Plant Electric Power Specific Consumption, kW-hour/m^3									4.55	3.88	3.83	4.35	4.25
Hybrid Plant Electric Power Consumption, Mwe									33.04	20.31	21.91	58.23	54.97
DOU Plant Electric Power Consumption, Mwe	2.11	3.71	10.58						10.58	2.11	3.71	2.11	3.71
RO Plant Electric Power Specific Consumption, Mwe				20.80	18.20	18.12	64.73	63.07	22.46	18.20	18.20	56.12	51.26
Electric Power to Grid, Mwe	57.89	52.69	23.42	44.20	46.80	41.88	0.27	1.93	0.96	39.69	34.49	1.77	1.43
Levelized Electricity Cost, $/kW-hour	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05
Total Water Cost, $/$m^3$	0.789	0.765	0.768	0.661	0.583	0.614	0.596	0.586	0.719	0.713	0.708	0.628	0.637

VIII.2.2.4.2. Coupling schemes of nuclear desalination complexes with FPU: Russian-Canadian nuclear desalination complex joint project

In the Russian-Canadian Project of Nuclear Desalination Complex with KLT-40C reactor plant 4 variants of coupling schemes have been chosen for further studies [1]. Besides, the 5th variant is regarded as additional. The principal condition presumed in the course of coupling schemes examinations is full retention of reactor plant and steam turbine generator plant configurations, design and parameters of the system of heat supply to the district heating circuit used in the FPU with KLT-40C.

***The first coupling scheme*:** for Nuclear Desalination Complex on the basis of FPU and a hybrid desalination plant. The turbine plant of the FPU produces both heat with the steam extractions and electricity with the turbine generator. In this case, steam from the second steam extraction is directed into the intermediate loop preheaters. There the extracted steam condenses giving its heat to the intermediate loop coolant. A part of the seawater flow preheated in the DOU GTPA desalination plant is headed to the evaporator of the intermediate loop part. After it has partly evaporated in the evaporator, steam at nearly 65°C with the two-phase jet compressor heats up to 95°C. The working fluid of the jet compressor is water, which is pumped out of the evaporator and heated in the intermediate loop heat exchanger up to 125°C. This scheme was described in detail in [3]. The RO plant in the preheat mode utiizes the preheated seawater from the FPU condensers and the DOU GTPA-840 condenser.

The second scheme of the Complex with the FPU and a preheat RO plant. The turbine plant of the FPU produces both heat with the steam extractions and electricity with the turbine generator. The feed water of the RO plant is preheated first in the condenser and then further in the intermediate loop preheater. This scheme fully utilizes both the condenser heat and the turbine steam extraction heat to preheat the seawater

The third scheme with the seawater preheat in the turbine plant condenser of the FPU. The turbine plant of the FPU produces both heat with the steam extractions and electricity with the turbine generator. The RO plant feed water intake and preheat both are performed after the turbine condensers.

The fourth scheme of a Co-Generation Complex on the basis of the FPU and an RO plant. The turbine plant of the FPU produces both heat with the steam extractions and electricity with the turbine generator. Seawater is preheated only in the intermediate loop preheaters.

The fifth scheme of the Complex on the basis of the FPU and 22-stage distillation plant. The turbine plant of the FPU produces both heat with the steam extractions for the DOU GTPA-840 plant and electricity with the turbine generator.

The above mentioned coupling schemes were analyzed with the use of IAEA's DEEP 2.0 program. However, the program required modernization to thoroughly account for the peculiarities both of the thermal scheme of TK-35/38-3.4 turbine plant and DOU GTPA-840, and to take into account the details of CANDESAL desalination systems. The results of the program modernization along with the analysis results are presented below in Section VIII.4.

VIII.2.3. Nuclear desalination complexes with NIKA-70

VIII.2.3.1. Nuclear steam supply system NIKA-70

NSSS NIKA-70 is a water-cooled integral reactor. Such reactor design offers the following advantages:

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

All primary circuit equipment does not require maintenance during power plant operation and is located in a leakproof strong safeguard vessel which confines radionuclide releases from the primary circuit during all DBAs.

VIII.2.3.2. NIKA-70 Safety concept

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

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: a multi-barrier system preventing release of radioactive products and implementation of a package of engineering and organizational measures protecting these barriers from internal and external impacts.

Use of passive systems and safety features based on natural processes with no need in external power supply:

- CPS drives whose 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 severe 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 its 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.

VIII.2.3.3. Coupling schemes based on NIKA-70 reactor plant

An optimum variant of coupling through an extra intermediate circuit with hot water throttling in the first stage of the distillation plant meets the requirements of both economic effectiveness of the desalination process and radioactive contamination protection. The optimal scheme of coupling between the NIKA-70 reactor plant and three DOU GTPA-230 distillation plants with total installed capacity 16,560 m^3/day.

VIII.3 ANALYTICAL EVALUATION AND EXPERIMENTAL VALIDATION

VIII.3.1. Simulation of contamination transport in nuclear desalination coupled systems using DESNU modelling tool and RELAP5/MOD3 system code

RELAP5/MOD3 [2] is the well known US system code applied to best-estimate transient calculations of LWR coolant systems. The code is capable of simulating two-phase systems for a wide range of accidents and transients.

The DESNU spreadsheet has been initially developed by INVAP (Argentina) as an EXCEL-based computer tool supporting elaboration of RETRAN-02 input for evaluation of NDCSs [3]. This tool is aimed to reduce the cost of the RETRAN-02 input deck preparation.

DESNU produces a NDCS model which is quite rigid in nodalization including the interconnection of components. The DESNU uses the NDCS model having no intermediate circuit between the secondary side of the nuclear power plant (NPP) and desalination plant (DP), which is included into the Russian design of the floating NDSC with the KLT-40 reactors.

Not all of the RELAP5 input data for the NDCS model can be produced using the DESNU spreadsheet. Subsequent direct editing of the DESNU-generated file is required to finalize the RELAP5 input deck prior to calculations.

RELAP5 input deck for the NPP model has been produced using the DESNU tool and calculation of the steady-state parameters has been performed. The same NDCS model has been used for RELAP5 simulations as been employed in RETRAN-02 simulations [3]. Table VIII.3 compares the obtained results.

TABLE VIII.3. COMPARISON OF STEADY-STATE PARAMETERS

Parameter	Rated value	RELAP5 Calculated value
SG Heat Transfer, MW	100	95.4
Turbine Inlet Flow, kg/s	48.7	48.0
Turbine Inlet Pressure, bar	47.1	44.7
Extraction 4 Pressure, bar	17.4	16.4
Extraction 3 Pressure, bar	9.47	9.1
Extraction 2 Pressure, bar	3.18	3.12
Extraction 1 Pressure, bar	0.643	0.694
Condenser Operating Pressure, bar	0.00882	0.00857
Condenser Steam Flow, kg/s	34.961	33.1
Condensate Pump Flow, kg/s	37.91	37.3
SG Inlet Pressure, MPa	6.0	5.5

VIII.3.2. DEEP Modernization for modeling nuclear desalination complex with the FPU

The existing version DEEP 2.0 [4] does not cover a set of specific features of the thermal scheme of the Small Heat and Power Plant. Besides, making new schemes of power source and desalination plant coupling in the program is troublesome.

Being so, a number of formulas and technical features of the typical power source (SPWR) and desalination plants (MED/RO) has been changed for correct modeling the Base Plant and Dual-Purpose Plant modes of the FPU. Changes in economic characteristics of the power source and desalination plants have been made to put the calculation parameters into correspondence with the project data on the FPU and DOU GTPA desalination plants. CANDESAL specialists have modernized the RO Plants Section. Files prepared in the course of DEEP modernization were used for the analysis of the modernization results, and also both for cost calculations for the Nuclear Co-Generation/Desalination Complex based on the FPU and Sensitivity Analysis (see corresponding Sections).

VIII.3.2.1. Analysis of DEEP modernization results

DEEP modernization results have been analyzed with the hybrid scheme of coupling the FPU with the desalination plant as an example. Calculation results are summarized in Table VIII.4.

VIII.3.2.2. Using modernization version of DEEP program for cost estimates for FPU-based nuclear co-generation/desalination complex

Files for calculations of Desalination Complex economics and performance have been prepared with due regard to the DEEP 2.0 modernization for the conditions of the thermal hydraulic scheme of FPU with KLT-40C reactor plants. The upper technical limit for the MED DOU GTPA Desalination Complex capacity is 120,000 m^3/day. For the hybrid scheme the DOU GTPA plants could produce 40,000 m^3/day in all the compared variants. Here, RO_1 is the RO plant feed water preheat in the FPU condensers. RO_2 represents the RO plant feed water preheat both in the FPU condensers and with the steam turbine extractions. RO_3 means the RO plant feed water preheat with the steam turbine extractions. The calculations results for different water capacities are given in Table VIII.5.

TABLE VIII.4. CALCULATION RESULTS

Characteristic	**Seawater Temperature**	**Calculated Value**	
		DEEP 2.0	**Modernized DEEP**
Total Water Cost, $/m^3	15	0.948	0.700
	20	0.954	0.700
	25	0.960	0.699
Net Saleable Power, MW	15	44.1	46.5
	20	42.2	45.0
	25	40.3	43.3
Levelized Power Cost, $/kW-hour	15	0.082	0.057
	20	0.085	0.059
	25	0.087	0.061

TABLE VIII.5. ECONOMIC AND TECHNICAL CHARACTERISTICS OF FPU-BASED NUCLEAR DESALINATION COMPLEX

Installed Water Plant Capacity, m^3/day	Desalination Process	Value	
		Water Cost, $/m^3$	Power to Grid, MWe
120 000	HIB	0.69	45.0
	RO_1	0.67	44.8
	RO_2	0.67	39.3
	RO_3	0.61	36.8
	MED	0.74	54.8
240 000	HIB	0.52	20.0
	RO_1	0.63	20.3
	RO_2	0.63	14.8
	RO_3	0.60	12.2
300 000	HIB	0.51	8.0
	RO_1	0.60	7.9
	RO_2	0.60	2.5
	RO_3	0.59	-0.1

NOTE*: **Bold** are parameters included in the basic file for FPU-based Desalination*

VIII.3.2.3. Using modernization version of DEEP program for sensitivity of water cost to project data variations

The sensitivity analysis has been done for the scheme of the FPU and hybrid desalination plants (120,000 m^3/day). The hybrid coupling scheme is the most complicated, and in DEEP 2.0 this scheme has the most complex dependencies in the corresponding module. The calculations results are summarized in Table VIII.6.

The described methodology of sensitivity analysis could be corrected provided that one has correct site characteristics, and could also be applied to other coupling schemes.

3.3. Cost estimates for RUTA-based desalination complex

Table VIII.7 compares technical and economic parameters of two Desalination Complexes: a one with RUTA-55 reactor plant and an optimized one with RUTA-N-70 (Installed capacity factor 0.92). Economic assessments showed that the decrease in distillate cost by 15–20% is reached as a result of optimization.

VIII.3.4. Cost estimates for NIKA-based co-generation/desalination complex

Technical and economic characteristics of a nuclear desalination complex based on the new generation integral NIKA-70 reactor plant with GPTA-230 distillation units are presented in Table VIII.8.

TABLE VIII.6. CALCULATIONS RESULTS FOR 120,000 M3/DAY WATER PLANT CAPACITY

Parameter	DEEP Designation	Units	Interval	Water Cost, $/m^3$
Interest Rate	ir	%	**0** 12	**0.691** 0.739
Plant Economic Life	Lep	Years	**36** 45	**0.691** 0.677
Purchased Electricity Price	Cpe	$/kW-hour	**0.037** 0.072	**0.691** 0.721
Construction Lead Time	Le	months	**48** 72	**0.719** 0.726
FPU Specific Construction Cost	Ce	$/kWe	1800 **2062** 2800	0.675 **0.689** 0.735

NOTE *Boldare parameters in the basic file for FPU-based Desalination Complex cost calculations.*

TABLE VIII.7. TECHNICAL AND ECONOMIC PARAMETERS OF DESALINATION COMPLEXES WITH RUTA-55 AND RUTA-N-70 REACTOR PLANTS

Parameter	Basic Option of RUTA-55 NDC	Optimized Option of RUTA-N-70 NDC
Reactor Thermal Power, MWt	55	70
Specific Power Source Construction Cost*, $/kW	390	330
Fuel Cost*, $/MWt	1.15 (1.9)	0.64 (1.9)
Cost Price of Thermal Power , $/MWt with Installed Capacity Factor = 0.92	5.6 (6.9)	4.9 (6.2)
Number of DOU GTPA Plants within NDC	4 (5***)	5 (6***)
Capacity DOU GTPA Module, m^3/h	220	250
Total Capacity of NDC, m^3/day	21000	30000
Specific Cost of Desalination System, $/($m^3$/day)	1700	1400
Discount Rate,%	8	8
Total Water Cost, $/$m^3$	1.22	1.0

** The value of specific cost for construction of mono-unit RUTA without replacing power source is indicated.*
*** For basic option the mode of partial refueling once per 3 years is envisaged (total duration of fuel campaign - 9 years), for optimized core the total refueling of the core is assumed in calculations under duration of fuel campaign ≈7 years.*

VIII.4. CONCLUSIONS

1. Has been carried out of the analysis technological and design features of reactor units KLT-40C, NIKA-70, RUTA-55 and RUTA-70.
2. Has been selected optimum schemes and parameters of coupling nuclear plants and desalination units.
3. Has been calculated of technical and economical characteristics of NPDC for nuclear plants on based of KLT-40C, NIKA-70 and RUTA with the IAEA's program DEEP.

4. Has been executed of modernization of the DEEP program with aimed to achieve more accurate description of a design and economical parameters of the FPU with KLT-40C reactors and DOU-GTPA. Has been carried out of the analysis of modernized version of DEEP and analysis of sensitivity of the water cost to the critical design data.
5. Has shown non-effectiveness of usage of extracted steam for a heating of feed water of RO units. Also has been showed that at high capacities of desalination plants the hybrid scheme of coupling has obvious economical advantage.
6. RELAP-5 input deck for the NPP model has been produced using the DESNU tool and calculation of the steady-state parameters has been performed. The same NDCS model has been used for RELAP5 simulations as been employed in RETRAN-02 simulations. Several components of the NDCS model have been directly modified according to specific input data requirements of RELAP5.

TABLE VIII.8. MAIN DESIGN CHARACTERISTICS OF NUCLEAR DESALINATION COMPLEX WITH NIKA-70 AND THREE DOU GTPA-230

Average Desalinated Water Output, m^3/day	14367
Power to the Grid, Mwe	9.4
Specific Reactor Plant O&M Cost, $/Mwe	10
Specific Nuclear Fuel Cost, $/MWe	20
Specific Reactor Plant Decommissioning Cost, $/MWe	1
Reactor Plant Capacity Factor	0.8
Distillation Unit Construction Cost, M$	19.6
Reactor Plant Construction Cost, M$	58
Total Water Cost, $/$m^3$	1.22

REFERENCES

[VIII.1] ZELENSKI D.I., SHAMANIN V.G., POLUNICHEV V.I., HUMPHRIES J.R., DAVIES K "The joint Canadian/Russian Floating Nuclear Desalination project". International Seminar on Status and Prospects for Small and Medium Sized Reactors. Cairo, Egypt, 27–31 May 2001.
[VIII.2] ALLISON, C.M., SCDAP/RELAP5/MOD3.1 CODE MANUAL, INEL, 1992.
[VIII.3] MASRIERA, N.A., DOVAL, A.S., "Evaluation of Nuclear Desalination Coupled Systems", Progress report of Co-ordinated Research Project on "Optimisation of the Coupling of Nuclear Reactors and Desalination Systems", RCM#3, Mumbai, India, 14–18 of February, 2000.
[VIII.4] Examining the economics of seawater desalination using DEEP code. IAEA–TECDOC–1186, Vienna, 2000.

ANNEX IX

OPTIMIZATION OF COUPLING DESALINATION SYSTEMS WITH A NUCLEAR REACTOR

Nafaa Reguigui
National Center of Nuclear Sciences and Technologies, Tunisia

IX.1. INTRODUCTION

In 2002, CNSTN (Tunisian National Centre for Nuclear Sciences and Technologies) and CEA-France signed an agreement to jointly pursue, under the aegis of IAEA's Interregional Cooperation Programme INT/4/134, nuclear desalination feasibility studies for the Skhira site in the south of Tunisia. To carry out the technical programme of the agreement, known as the TUNDESAL Project, a mixed team of experts from CEA and from CNSTN as well as from the Tunisian electric utility (STEG) and the water utility (SONEDE) was formed. It was further agreed that the Tunisian experts would partially complete their work at CEA laboratories with IAEA financial assistance (at the Advanced Water Reactor Studies Laboratory, LFEA, of the Innovative Reactor Studies Service, SERI, in the Nuclear Energy Division at the Cadarache Atomic Centre). This feasibility study considers the following tasks:

- Pre-dimensioning of the nuclear reactor and desalination processes, compatible with Tunisian electricity needs and required water production capacity at The Skhira site.
- Coupling of the selected nuclear reactor to desalination processes and system optimisation.
- Economic evaluation of the integrated systems elaborated above.
- Safety verification studies of coupled systems.

The first three tasks are completed. The fourth task related to safety analysis will be finalized by 2004 end.

IX.2. MOTIVATION

Water is essential for life. However, it is destined to cause major shortage problems in the future. Currently, Tunisia is one of 80 countries making up 40% of the world population having physical water scarcity problems.

In Tunisia, water resources are estimated to be around 4.5 millions of m^3/year (i.e., 450 m^3/year per person which is far below the poverty line). 40% of these resources are ground waters, the salinity of which varies from 0.5 to 3.5 grams per liter, only 54% of those have a salinity less than 1.5%. Furthermore, 84% of these resources are located in the north of the country.

These quantities may be sufficient at the time being, however, given the recurring droughts, the population increase and the industrial development in the country, desalination will be a must around the year 2020. In particular, sea water desalination seems to be an attractive and economical alternative that will contribute to the sustainable development of the country.

Current studies indicate that water demand will reach 300 000 m^3/day in 2010 for the southern region of Sfax and Gabes alone (more than 371.200 m^3/day for 2020).

To satisfy these water needs, sea water desalination seems to be a plausible option given the immense reserves available, the continuous decrease in desalted water cost and due to the fact that desalination makes potable water and/or pure water available to several sectors at the same time (industry, agriculture, domestic needs, etc.…)

The desalination experience in Tunisia has started since the early 80's with the commissioning of the station of Gabes with a daily production of 22,500 m^3 of water with salinity less than 1 g/l (from 3.2 g/l at the input). A second water desalination station was installed in 1983 on the island of Kerkenah with a limited capacity of 4000 m^3 daily. Later on, two more desalination stations were commissioned, one in Zarzis (1999) and the other in Jerba (2000). This brings the overall desalted water by the water utility alone to 58,800 m^3/day.

These stations use reverse osmosis to reduce water salinity to less than 1 g per liter using electrical energy. However, this technique presents several disadvantages, such as the short lifetime of the membrane, and the elevated energy cost that result in an expansive product (0.850 Tunisian Dinars/m^3 ~ 0,67 US$/ m^3). For these reasons, there has been a continuous effort to search for better and more economical ways to generalize desalination to the rest of the southern region.

Because desalination is an energy intensive process, the use of fossil fuels is not foreseen in the future because the national reserves are limited and they can not meet the increasing energy demand for the future (this is besides their polluting nature). Moreover, the fuel prices are worrisome, especially recently when prices reached 50 US$ per barrel of crude oil. That is why the nuclear option needs to be included in any long term planning. The use of nuclear reactors for sea water desalination and electricity production is a promising alternative.

During the 90's, Tunisia and with the help of the IAEA started studying the possibility of using nuclear technology as an alternative source of energy. In particular, Tunisia had conducted a site survey study for the installation of a nuclear power reactor (project TUN/9/007, 1991–1995) and Tunisia participated in the feasibility study for the use of nuclear energy for desalination in the north African states (project RAF 4/013, 1992–1996).

The technology transfer and mastering and the objective and scientific evaluation of the ifferent desalination scenarios will be the principal arguments and base to aid decision makers to decide on the best option that is suitable for the Tunisian conditions.

IX.3. SITE SPECIFIC DATA

Site specific preliminary study of the skhira site (Figure IX.1) was carried out based on IAEA methodology as defined in the Safety Guide. This study, will be refined during the final stages of the project through geo-technical, geological and seismic reconnaissance of The Skhira site. The site of The Skhira is situated on the south eastern cost of Tunisia in the Gulf of Gabès, 6 km away from the town of The Skhira. At its North is the Sahel region,in the South East is the arid region of The Djeffara, while its western limits are the Atlas of Gafsa-Meknassy-Mezzouna. The site is very large and is relatively high with respect to sea level (between 5 and 10 m)

Previous geological studies did not show any particular engineering or geotectonic problems. On the seismic map of the region, the site of The Skhira was attributed a maximum seismic magnitude of VII on the MSK scale.

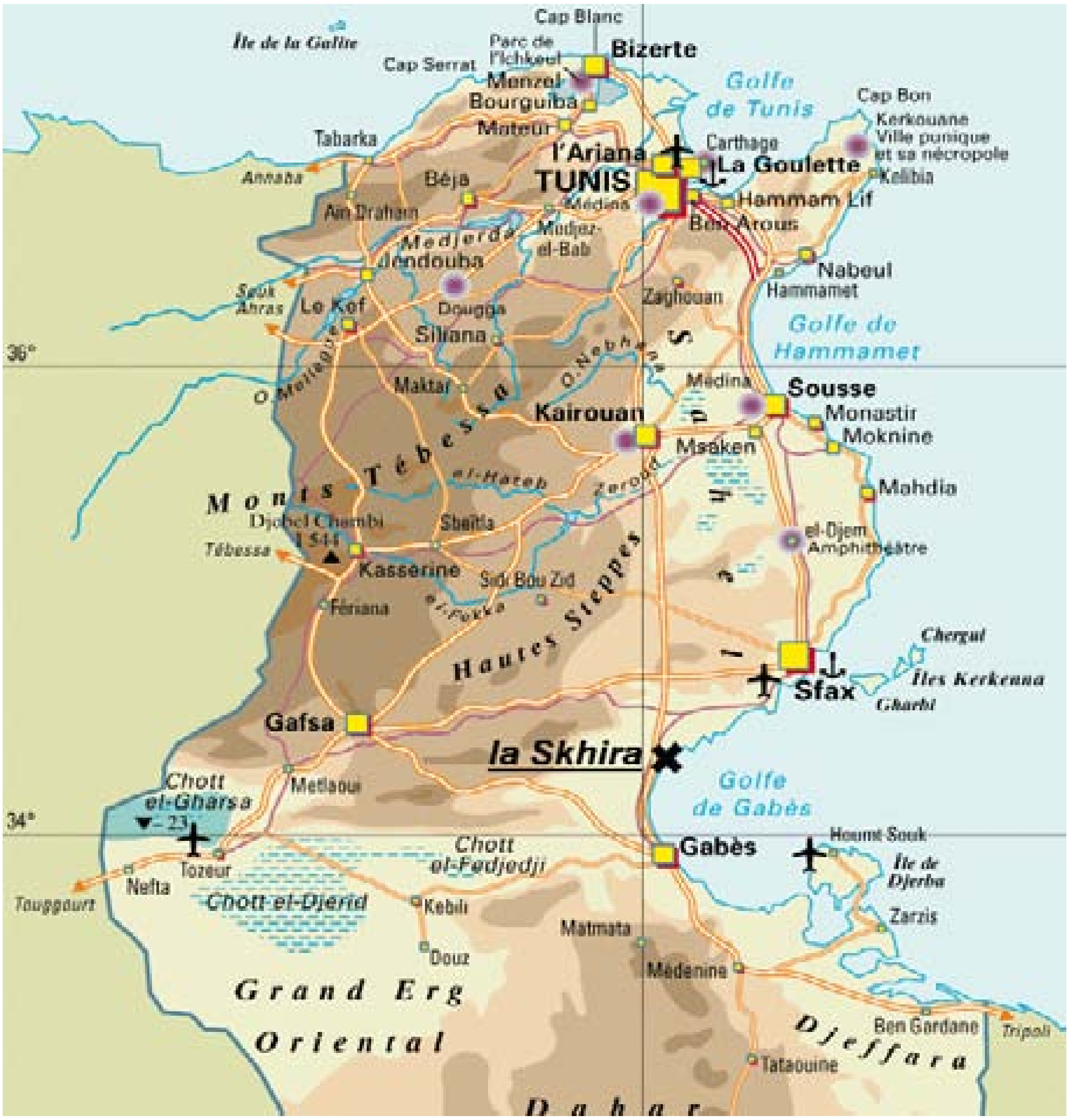


Figure IX.1: The Skhira Site.

From the socio-economic standpoint, The Skhira is easily accessible between two large industrial towns, both facing acute water shortages: The town of Sfax, about 80 km to the North, with more than 830 000 inhabitants and the town of Gabès, about 50 km to the South, with more than 330 000 inhabitants.

IX.4. WATER DEFICIT BY THE YEAR 2020

Two scenarios were considered to estimate water deficit in 2020. The first scenario estimates the expected population for the region using the current population times a growth rate of 1.6%. The water demand is computed based on a 4% water consumption increase until 2010 and 3% afterwards. This approach results on a deficit of 148,751 m^3 per day for the year 2020. To simplify matters, we approximated this figure to 150 000 m^3/day. This is what we call the "the high hypothesis".

The second scenario takes into account water supply projects that are under way or planned for the region by the year 2006 [transfer of a 129 600 m^3/day line from the North up to Sfax (2005) and the installation of the 4th brackish water RO desalination line (8 500 m^3/day at Gabès (2006)],. Estimations give a deficit of 40,000 m^3 per day for the year 2020. Taking into account the usual uncertainties, we retained a value of 48 000 m^3/day. This is what we call the « low hypothesis ».

IX.5. ELECTRICITY NEEDS

According to the Tunisian Institute of Quantitative Economics, (IEQ), the increase in the gross internal product (GIP) of Tunisia will be about 6.8%/year. Assuming that the grid would have losses of about 11%, the electrical consumption would grow at the rate of 6.5%. Making use of the previsions of the 10th to 13th Tunisian national development plans, the required capacity of the grid would be about 35 120 MWh in the year 2020. The peak power demand will then be 5 920 MWe.

IX.6. RESULTS OF PRE-DIMENSIONING

Applying the well known convention that the maximum size of the power plant should not exceed about 10% of the demand, we conclude that the Tunisian grid would require a plant of about 600 MWe in 2020.

The thermal and electrical powers to be furnished by a power plant for an integrated system depend principally on the desalination technology used (its specific consumption) and the required water production capacity.

Table IX.1 indicates, for a typical 600 MWe PWR, the values of these two parameters for the desalting capacity of 150 000 m^3/day. Those corresponding to 48000 m^3/day are given in parentheses. The desalination processes considered are the Multiple effect Distillation, (MED), the Reverse Osmosis (RO) and RO with preheating of the feed water (ROph).

TABLE IX.1: REQUIRED ELECTRICAL AND THERMAL POWERS FOR DESALTING 150 000 (48 000) M^3/DAY

Process	Required thermal Power	Required electrical Power	Electrical power to the grid
	MWth	MWe	MWe
MED	312.5 (89,7))	18.8 (6)	478 (564)
RO	0	28 (9)	572 (591)
Roph	0	28 (9)	572 (591)

In conclusion, and depending upon the desalination process used, and the required capacity, an integrated desalination system based on a 600 MWe plant would provide to the grid from 591 to 375 MWe.

For the purpose of the economic study, five types of solutions were considered for this power range, two are fossil power plants (Super Steam Boiler and Gas turbine combined cycle) and three nuclear power plants, namely:

- An innovative reactor of the type SCOR-600, currently being studied at CEA, DER/SERI/LFEA.
- Two modules of the GT-MHR whose conceptual design studies are being carried out by a consortium comprising FRAMATOME (France), GENERAL ATOMICS (USA), MITSUBISHI (Japan) AND MINNATOM (Russian federation).
- A 900 MWe PWR of the type operating in France, in case by 2020, the Tunisian grid is interconnected to the European grid and could thus support reactors of higher power.

IX.7. ELABORATION AND OPTIMIZATION OF COUPLING SCHEMES

When an integrated system is based on the utilization of a contiguous RO process, the coupling between the reactor and the process does not require any further optimization, except for the adaptation to local site conditions.

However, with a thermal process such as the MED, the coupling is more complex. It is necessary that the thermal-hydraulic characteristics of such a coupling be determined precisely.

In order to elaborate and optimize a given coupling scheme, it is necessary to determine the conditions which allow the transfer of the right amount of heat to the process with the least possible impact on the plant performances.

The formulation of thermodynamic equations concerning the heat and mass balances at different nodes of the coupling and their subsequent resolution, with appropriate boundary conditions then leads to the determination of the basic parameters such as mass flow rate of the extracted vapour, and its ideal temperature for a given amount of desalted water production.

It is to be recalled that in accordance with the safety analysis an intermediate circuit, comprising of a heat exchanger, a Flash Tank and a re-circulation pump, is absolutely essential to the coupling. The dimensioning of these components was then obtained by resolving appropriate heat and mass balance equations

A great advantage of the GT-MHR is that its design allows the utilisation of waste heat from its intercooler and pre-cooler exchangers at ideal temperatures for desalination (80 to 100°C). Because this heat is sent to the heat sink anyway, it is considered virtually free for desalination. However, calculations show that at the Skhira site, for seawater temperature of 21°C and helium temperature of 26°C at the output of the intercooler or the pre-cooler, the maximum amount of heat available to the MED plant is about 43 MWth for two GT-MHR modules. This would correspond to a production of 43 500 m^3/day.

IX.8. SAFETY VERIFICATION

Preliminary safety analysis of the selected nuclear reactors and of the interface coupling them to the desalination processes, such as MED and RO, leads to the following conclusions:
- All the coupling schemes studied, maintain the number of static confinement barriers.
- However, it is necessary to maintain a dynamic barrier (pressure on the process side higher than on the reactor side) through appropriate design and operating conditions, which should minimise any leakages in the process heat exchanger:

- The coupling should ensure an instantaneous cut off of the desalination plant from the reactor through the action of fast acting valves.
- The desalted water should be continuously monitored against any hypothetical radioactive contamination
- The integrated system using MED should be modular, with each module having its own intermediate circuit. The Flash tank of these circuits must be situated on the process side.
- The loss of load produced by the loss of several RO or MED modules is negligible compared to the loss of load in transients studied for the reactor safety.

IX.10. POWER PRODUCTION COST EVALUATION

The economical evaluation of the power production systems was performed with the help of the SEMER code, recently developed by the CEA, and the STEG approach that is based on the method adopted by EURELECTRIC (ex UNIPEDE) for the fossil based power plants. For the GT-MHR, we do not yet have cost models in SEMER. We were thus obliged to use the data as published by its developers.

At present Tunisia has some natural gas resources. It also takes out a certain percentage of the gas transiting from its territory. It is for this reason that present gas prices in Tunisia are of the order of 130 $/tep (or 17.73 $/bbl).

For this price and a discount rate of 8%, and using the STEG approach for calculating the kWh costs of the different systems, the combined cycle plant (2 X 280 MWe) has the lowest kWh cost. This is followed by the nuclear plant PWR900, which could be also SCOR600, and the simple boiler for vapour production (figure IX.2).

Nuclear plants have the advantage of much lower fuel cycle costs than that of the combined cycle. However their investment +O&M costs are much higher. Their kWh cost is about 6% higher than the combined cycle plant with the current gas prices and 8% discount rate.

The simple boiler for vapour production has the highest fuel cost and even at the low gas price of 17.7 $/bbl, it does not compete with the combined cycle plant. Therefore, as long as the gas prices remain about 130$/tep, the nuclear option will not be economic for Tunisia.

However, in the years to come, primary energy production in Tunisia will decline because of the limited reserves whereas electrical consumption will continue to increase as the country develops. The present equilibrium between production and consumption would no longer exist. There would thus be shortage of electrical power unless new natural gas resources are discovered, which is not at all certain.

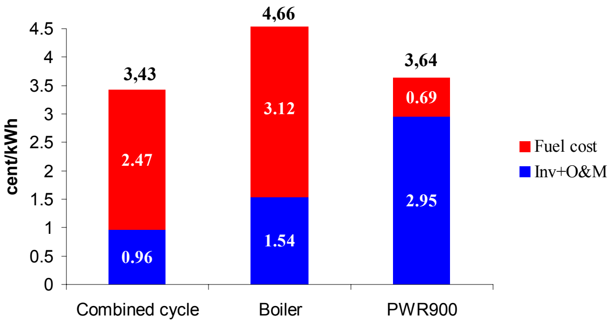


Figure IX.2 : kWh costs by STEG for different power plants; gas price 130$/tep.

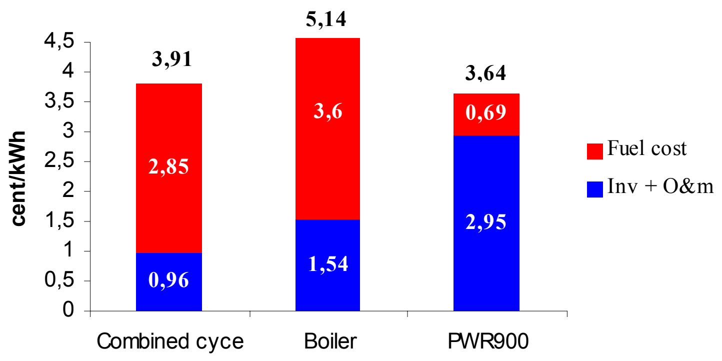


Figure IX.3 : kWh costs by STEG for different power plants; gas price 150$/tep.

If the gas prices were to increase even slightly say, from 130$/tep to 150$/tep then as shown in figure IX.3, the production cost by nuclear power plants would become lower than that by the combined cycle plant. For the year 2020, the deployment of nuclear energy thus seems to be a viable option.

IX.11. DESALINATION COST EVALUATION

Desalination costs were calculated using a CEA-modified version of IAEA's DEEP code with SEMER results as input for three types of desalination processes: MED, RO, and ROph. Corresponding desalination costs are presented in tables IX.2 to IX.4. Typical field parameters for The Skhira site are:

Sea water temperature	: 21°C
Salinity	: 38 375 ppm

A careful inspection of these tables leads to the following observations:

- Whatever the desalination process, production capacity or the discount rate, the desalination costs by the two nuclear options PWR900 and SCOR600 are almost the same, since the relative difference between their costs does not exceed 2 to 4%.
- There is a distinct size effect because the desalination costs by all combinations, for the production capacity approaching 150 000 m^3/day, are systematically lower than corresponding costs for the production capacity approaching 48 000 m^3/day.
- Among the fossil fuelled systems, the desalination costs by the combined cycle integrated plant are much lower. Thus for example, for gas price of 18$/bbl (132$/tep), and a discount rate of 8%, the CC600 +MED system gives 10% lower desalination costs (for the high production capacity) than the SSB600 + MED system. For other fuel prices and discount rates this difference is of the order of 12 to 13%.
- With the RO process, and under similar conditions, the desalination cost by the CC600 +RO system is 9% lower than that by the SSB+RO system. For other discount rates and fuel prices, this difference is 10 to 11%.
- For the two desalination processes (MED and RO) the desalination costs by nuclear systems are significantly lower than corresponding costs by CC600 or SSB600. Thus for example, for 8% discount rate and fossil fuel price of 25$/bbl (183$/tep), the desalination cost by SCOR600 +MED is respectively 25 and 33% lower than that by CC600 +MED and SSB600 +MED systems. For fossil fuel prices of 30$/bbl, these differences are still higher: 30 and 39%.

TABLE IX.2: DESALINATION COST WITH FOSSIL ENERGY BASED PLANTS

Parameters	Units	SSB600										CC600									
Total power	MWe	600										600									
Desalination process		MED					RO					MED					RO				
Year of operation		2020																			
Production	m^3/d	157 155					152 867					157 155					152 867				
Fossil fuel price	$/bbl	18	25			30	18	25			30	18	25			30	18	25			30
Discount rate	%	8	5	8	10	8	8	5	8	10	8	8	5	8	10	8	8	5	8	10	8
Construction cost of desalination unit	M$	167.1.					161.8					171.4					161.8				
Specific investment cost	$/m^3/d$	952	921	952	972	952	1047	1016	1047	1068	1047	977	945	977	998	977	1047	1016	1047	1068	1047
Seawater desalination cost	**$/m^3$**	**0.75**	**0.76**	**0.85**	**0.93**	**0.93**	**0.75**	**0.74**	**0.83**	**0.90**	**0.88**	**0,68**	**0.68**	**0.76**	**0,83**	**0.82**	**0.69**	**0.67**	**0.75**	**0.81**	**0.79**
Production	m^3/d	49 111					54 595					49 111					54 595				
Construction cost of desalination unit	M$	58.5					67					60.2					67				
Specific investment cost	M$/y	1030	1010	1030	1044	1030	1179	1156	1179	1195	1179	1060	1039	1060	1073	1060	1179	1158	1179	1195	1179
Seawater desalination cost	**$/m^3$**	**0.78**	**0.79**	**0.88**	**0.96**	**0.96**	**0.79**	**0.77**	**0.87**	**0.94**	**0.93**	**0.71**	**0.69**	**0.78**	**0.85**	**0.84**	**0.73**	**0.70**	**0.79**	**0.86**	**0.84**

TABLE IX.3: DESALINATION COSTS WITH NUCLEAR POWER PLANTS

Parameters	Units	PWR900									SCOR600								
Total power	MWe	951									600								
Desalination process		MED			RO			ROph			MED			RO			ROph		
Year of operation		2020																	
Production	m^3/d	152 123			152 867			150 461			157 155			152 867			150 461		
Discount rate	%	5	8	10	5	8	10		5	10	5	8	10	5	8	10	5	8	10
Construction cost of desalination unit	M$	182.9			161.8			115.2			183.8			161.8			115.2		
Specific investment cost	$\$/m^3/d$	1008	1042	1064	1016	1047	1068	731	752	768	1014	10470	1070	1016	1047	1068	731	752	766
Seawater desalination cost	**$\$/m^3$**	**0.44**	**0.58**	**0.68**	**0.47**	**0.59**	**0.67**	**0.39**	**0.48**	**0.54**	**0.43**	**0.57**	**0.68**	**0.46**	**0.59**	**0.68**	**0.38**	**0.48**	**0.56**
Production	m^3/d	47 538			54 595			50154			49 111			54 595			50154		
Construction cost of desalination unit	M$	63.3	63.3	63.3	67	67	67	42.8	42.8	42.8	63.7	63.7	63.7	67	67	67	42.8	42.8	42.8
Specific investment cost	M$/y	1092	1114	1129	1158	1178	1195	803	818	828	1098	1120	1135	1156	1179	1195	803	818	828
Seawater desalination cost	**$\$/m^3$**	**0.47**	**0.60**	**0.71**	**0.51**	**0.63**	**0.72**	**0.41**	**0.50**	**0.57**	**0.45**	**0.60**	**0.71**	**0.49**	**0.63**	**0.73**	**0.40**	**0.50**	**0.58**

TABLE IX.4: DESALINATION COSTS WITH A GT-MHR

Parameters	Units						
Total electrical power power	MW	571 (with 2 modules)					
Desalination process		MED			RO		
Date of operation		2020					
Production capacity	m^3/day				152 867		
Discount rate	%	5	8	10	5	8	10
Construction cost of desalination plant	M$				162	162	162
Specific investment cost	$/$m^3$/day				1017	1048	1069
Seawater desalination cost	**$/$m^3$**				**0.49**	**0.59**	**0.67**
Production capacity	m^3/day	43538			54 595		
Construction cost of desalination plant	M$	70.3	70.3	70.3	67.1	67.1	67.1
Specific investment cost	$/$m^3$/day	1214	1238	1254	1157	1180	1196
Seawater desalination cost	**$/$m^3$**	**0.45**	**0.58**	**0.67**	**0.52**	**0.63**	**0.72**

- In similar conditions with the RO process, SCOR 600 +RO gives desalination costs, which are respectively 21 and 29% lower than those by the fossil energy based systems. For fossil fuel price of 30$/bbl, the differences are 33 and 49%. These conclusions remain valid for the lower desalination capacity approaching 48000 m^3/day. The main reason for these differences arise from the fact that desalination costs are strongly influenced by the energy costs. It should be recalled that energy costs by nuclear power plants are 40 to 60 % lower than those by CC600 and SSB600, for fossil fuel prices higher than 18$/bbl,
- ROph systems appear to be the cheapest. With SCOR600 + ROph, the desalination cost is respectively 16 and 19% lower compared to SCOR600 +MED and SCOR600 +RO.
- The GTMHR +MED system produces only about 43 500 m^3/day. However, the desalination cost by GT-MHR+MED is still 3% lower than SCOR600 +MED.
- This advantage does not exist for GT-MHR+RO (48000 m^3/day), since its desalination cost is respectively 16, 9 and 8% higher than the GT-MHR+MED system, for discount rates of 5, 8 and 10%. In fact the desalination cost by this later system is even lower than GT-MHR+RO, producing about 150 000 m^3/day.

IX.12. CONCLUSIONS

The main objective of the project was to provide a choice of global technical options for the eventual deployment of an integrated nuclear desalination system, producing electricity for the Tunisian grid and desalted water to cover the needs of the Sfax-Skhira-Gabès region in 2020.

We have seen that for the entire Sfax – Skhira – Gabès region the additional required desalination capacity in 2020 would be about 150 000 m^3/day in the so called “high hypothesis”. In the « low hypothesis », where all the currently envisage projects for water supply to this region are indeed realised, the required additional desalination capacity would be about 48 000 m^3/day.

If therefore the choice of Tunisia is in favour of the GT-MHR for electricity production, a GT-MHR+MED system would be sufficient to satisfy the water needs in the low hypothesis. The desalination cost with the GT-MHR+RO system would not be very different from those by the other two systems, using RO.

If Tunisia decides in favour of a PWR, the most attractive option could be the SCOR600 + ROph.

It is important to note that the maximum water production capacity with 2 GT-MHR modules (571 MWe) is about 48 000 m^3/day. Therefore, the coupling of a SCOR 600 MWe reactor to a desalination station using preheating RO is the most attractive option no matter what the water capacity production is (water cost starting at US$ 0.486 $/$m^3$). The cost of water produced by the ROph process is the lowest one what ever the energy type was.

It is important to point out here that neither of the two recommended nuclear reactors is in an advanced state of industrial deployment. The conception of the GT-MHR is indeed rather advanced, thanks to the continued effort of the international consortium that is promoting it. SCOR600 is at preliminary conceptual stage in the context of research work being carried out at CEA.

The choice of options and the conclusions that we have presented are merely the result of a preliminary feasibility study, indicating the possible interest of nuclear energy for Tunisia under certain conditions.

CONTRIBUTORS TO DRAFTING AND REVIEWING

Baranaev, Y.D.	Institue of Physics and Power Engineering, Russia
Ben Kraiem, H.	Centre National des Sciences et Technologies Nucleares, Tunisia
Doval, A.	INVAP SE, Argentina
Faibish, R.S.	International Atomic Energy Agency
Gowin, P.J.	International Atomic Energy Agency
Hanara, M.S.	Bhabha Atomic Research Center, India
Humphries, J.R.	CANDESAL Enterprises Ltd., Canada
Hwang, Y.D.	Korea Atomic Energy Research Institute, R.O. Korea
Jia, H.	Institute of Nuclear Energy Technology, China
Konishi, T.	International Atomic Energy Agency
Kupitz, J.	International Atomic Energy Agency
Masriera, N.A.	INVAP SE, Argentina
Megahed, M.M.	Nuclear Power Plants Authority, Egypt
Misra, B.M.	International Atomic Energy Agency
Prabhakar, S.	Bhabha Atomic Research Center, India
Rusli, A.	National Atomic Energy Agency, Indonesia
Saxena, A.K.	Bhabha Atomic Research Center, India
Scripnicov, D.E.	Institue of Physics and Power Engineering, Russia
Srivastava, V.K.	Bhabha Atomic Research Center, India
Tabet, M.	Centre National de l'Energie des Sciences et des Techniques Nucleaires, Morocco
Tewari, P.K.	Bhabha Atomic Research Center, India
Thalor, K.L.	Bhabha Atomic Research Center, India
Verma, R.K.	Bhabha Atomic Research Center, India
Wu, S.	Institute of Nuclear Energy Technology, China

Research Coordination Meetings

Vienna, Austria: 16–20 November 1998
Mumbai, India: 14–18 February 2000
Vienna, Austria: 16–18 October 2001
Vienna, Austria: 24–26 February 2003

Consultants Meetings

San Diego, USA: 2–4 September 1999
Bariloche, Argentina: 5–7 December 2000
Vienna, Austria: 1–3 December 2003