IAEA-TECDOC-999

Introduction of small and medium reactors in developing countries

Proceedings of two Advisory Group meetings held in Rabat, Morocco, 23–27 October 1995 and Tunis, Tunisia, 3–6 September 1996

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

The IAEA does not normally maintain stocks of reports in this series. However, microfiche copies of these reports can be obtained from

INIS Clearinghouse
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria

Orders should be accompanied by prepayment of Austrian Schillings 100,— in the form of a cheque or in the form of IAEA microfiche service coupons which may be ordered separately from the INIS Clearinghouse.

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

INTRODUCTION OF SMALL AND MEDIUM POWER REACTORS
IN DEVELOPING COUNTRIES
IAEA, VIENNA, 1998
IAEA-TECDOC-999
ISSN 1011-4289

Printed by the IAEA in Austria
February 1998

FOREWORD

In the light of renewed interest in the utilization of small and medium reactors (SMRs) in developing countries for both power generation and heat applications, the IAEA has surveyed the reactor designs in the small and medium size range that are currently deployed or under development. The following TECDOCs on the design and development aspects (including passive safety and integral design concepts) have been recently published:

-Review of Design Approaches of Advanced Pressurized LWRs (IAEA-TECDOC-861);

-Design and Development Status of Small and Medium Reactor Systems 1995 (IAEA-TECDOC-881); and

-Status of Advanced Light Water Cooled Reactor Designs 1996 (IAEA-TECDOC-968).

Since deployment aspects such as lessons learned and technology transfer were not covered in these publications, developing Member States have shown interest in further information on these subjects.

This publication presents material submitted both by vendor and interested buyer organizations and conclusions drawn from the discussions of these contributions at two Advisory Group meetings on the SMR introduction in developing countries. A few papers were prepared as follow-up contributions to the proceedings. The summary presents a review of the main areas related to SMR introduction and of relevant situations and activities in both industrialized and developing countries. It includes an assessment of the expected potential market and of relevant experience that may help developing countries in their efforts to introduce SMRs. Owing to the inclusion of several new designs, this TECDOC provides an update of the SMR status report (IAEA-TECDOC-881) published in 1996. It also reviews real time compact nuclear power plant simulators.

The IAEA is grateful to the experts who have contributed to the publication. The IAEA officers responsible for the compilation of the report were M. Al-Mugrabi and G. Woite of the Division of Nuclear Power and the Fuel Cycle

EDITORIAL NOTE

CONTENTS

PART III: SMALL AND MEDIUM REACTORS POTENTIAL MARKET AND APPLICATIONS

PART IV: DESIGN DESCRIPTIONS

PART V: SIMULATION OF NUCLEAR REACTORS

SUMMARY

1. INTRODUCTION

Energy represents an important factor of everyday life, and techno-economic studies have shown that there is a distinct correlation between the energy consumption and national economic output in a wide range of countries. Electricity is the most convenient and versatile form of energy. Electricity consumption is playing an important role in modernization efforts to bring the developing world into a new era. This era will mainly be characterized by the electrification of rural areas where two billion people still do not have access to electricity in their homes and to modern information technology. Both would require a reliable source of electricity. The share of electricity in the global energy consumption has grown from 17% to 30% and the average annual consumption of electricity per capita has almost tripled in the period from 1960 to 1990. In spite of this increase, developing countries still consume much less electricity per capita than developed countries. The witnessed aspirations for economic growth and improved standard of living of developing countries is only possible if a large expansion of electricity production can be realized.

The world population is expected to increase from the present 5.8 billion to 8 billion by the year 2030, a rate of 70-80 million per year. Developing countries will account for over 95% of this increase. This will drastically increase the demand for electricity, requiring substantial additional power production capacities. Hence, the future electricity growth will likely be dominated by developing countries.

Power generation using fossil fuels or hydropower is the dominant means of electricity production in the world today, contributing approximately 82% of the total of 12913 TWh electricity generated in 1995. Nuclear is currently providing about 17% of the power generation in the world, making it one of the three major important energy sources in many countries. Having been introduced only four decades ago, the nuclear option has grown very fast. Nuclear electricity generation expanded at an average of 24% per year in the period 1970-1980, and 16.5% per year in the period 1980-1985. Later on, this rapid expansion was affected by two severe nuclear accidents and the improved economics of alternate electricity sources, particularly combined cycle gas turbines, and the growth was only 5.6% in the period 1985-1990 and 3% in the period 1990-1995. In 1995 there were 437 NPPs operating in 32 countries. Several facts could be concluded from such status and contribution of nuclear power to the world power generation:

- A significant part of power generation in the world is produced by nuclear power;
- Nuclear power is well developed and proven;
- It is economical and competitive to other forms of power generation in a range of situations;
- Nuclear power generation is environmentally benign. There is an incentive for nuclear power growth if the environmental burdens are to be more effectively controlled in the future.

On the other hand, there are unfavourable factors and possible constraints which currently limit the further growth of nuclear power, including:

- Public opposition;
- Difficulty of new nuclear power plants (NPPs) to compete economically;
- Saturation of the electricity market in many industrialized countries;

- Lack of nuclear infrastructure, technical and financial resources in many developing countries;
- Financing difficulties.

Although the importance of the issues varies among individual countries, they have contributed to limiting nuclear power growth to a few countries.

Most of the nuclear power is generated in developed countries, but a number of developing countries have also deployed nuclear power plants. There were 113 nuclear power plants operating in developing countries in 1995, with a net generating capacity of 68 GW(e). The accumulated operating experience of these plants is over 800 reactor years. There are 20 developing countries that have experience with nuclear power generation.

2. THE CASE FOR SMRs IN DEVELOPING COUNTRIES

It can be noted that in most of the developing countries the unit size has been mainly the small and medium reactor (SMR) size range. With the exception of China, the Republic of Korea, and some of the Republics that were part of the former Soviet Union which operate both medium sized and large nuclear plants, the remaining units are of 700 MW(e) or below. This is mainly due to the size of their grid, total power demand, electricity demand growth rate, and infrastructure and financing capabilities. Concerning new nuclear power plants, it is expected that SMRs will be the preferred choice for developing countries in the next decades, including developing countries that have not yet deployed nuclear power.

Apart from nuclear power generation, some developing countries could utilize nuclear energy for heat applications. Seawater desalination, district heating and process heat application are important areas for many developing countries where nuclear power can play an important role. These applications could be served using small or medium reactors. In addition, some of the developing countries have remote areas and/or isolated islands where nuclear power could have an advantage over conventional power generation to supply electrical power and/or heat for various applications. The total energy demand for this application for a given site is usually relatively small ($\leq$100 MW(e)).

Although nuclear energy is not the only means of providing power and process heat, it is a relevant option, especially if the environmental issues are to be properly accommodated. In addition, countries that are not blessed with fossil natural resources could find the nuclear option attractive on the long run. Future nuclear power utilization will likely deploy advanced nuclear power plants that are currently under design.

Building on four decades of operating experience, which amounts to about 7000 reactor-years, and applying lessons learned from over 500 plants, a new generation of nuclear power plants has been or is being designed and developed. These new generation reactors have incorporated improved safety concepts that will provide better protection against possible releases of radioactivity to the environment. During their design, the requirements for construction, operation, maintenance and repair were taken into consideration. Although this will ensure technical and safety advantages, their economics could only be proven once they have been deployed. A nuclear power plant is a capital intensive project; financing has to be secured.

The deployment of nuclear power in developing countries, especially in those without nuclear power experience, requires special attention to several areas by both vendor and purchasing organizations. These areas were presented and discussed during the Advisory Group Meetings and can be classified in the following three areas:

- design and development status of SMRs
- lessons learned from the introduction of SMRs in developing countries
- assessment of market potential for SMRs.

3. DESIGN AND DEVELOPMENT STATUS OF SMRs

Over the last three decades, about 7000 reactor years of operating experience have been accumulated with the current nuclear energy systems. Building upon this background of success, new small and medium reactor systems are being built or developed. These SMRs generally incorporate improvements of the safety concepts, including features that will allow operators more time to perform safety actions and that will provide increased protection against any possible releases of radioactivity to the environment. The new SMR systems have also incorporated features to make them simpler to build, operate, inspect, maintain and repair. Descriptions of these systems and their development status have been documented in the IAEA-TECDOC-881 [1][1]. The design and development efforts of SMRs have been very active and some new designs have emerged over the last year. These new designs have been presented during the Advisory Group Meetings on the "Status and introduction of small and medium power reactors into developing countries", in Rabat, Morocco, 23-27 October 1995 and in Tunis, Tunisia, 3-6 September 1996. Overviews of these new designs are included in this technical document for completeness (Table 1).

4. SMR MARKET

The IAEA has performed a questionnaire survey on the SMR market potential in 1996. The questionnaire consisted of a supplier's part and a buyer's part. By the end of 1996, the IAEA had received responses to the buyers' questionnaire from the following countries: Bulgaria, Chile, Finland, France, Hungary, Indonesia, Pakistan, Thailand, Tunisia, Turkey and Viet Nam. Several countries showed interest in nuclear power for electricity generation and seawater desalination; they project their next NPPs to be in operation before the year 2020.

The largest plant sizes will be 300-1500 MW(e) in 2000 and 360-1500 MW(e) in 2015. Six countries foresee that their largest power plant sizes will be under 700 MW(e) in 2000; three countries foresee that their largest plant sizes will be under 700 MWe in 2015. The other countries are interested in larger plants.

Some Member States listed factors which they consider important for the NPP introduction in their country: (a) energy generation cost, (b) investment costs, (c) safety and licensability, (d) proveness of the technology used, (e) guarantees concerning costs, construction schedule and performance of the NPP, and (f) local participation capability in the NPP construction.

The following countries provided responses to the suppliers'questionnaire: Belgium France, India, Italy, Japan, Republic of Korea, Romania and the Russian Federation. The

---

[1] The technical document has included 29 design descriptions and Table 1 summarizes the main data and development status of SMRs.

TABLE 1. SMR DESIGNS [1]

1. Reactors being deployed or in the detailed design stage

Design Name	BWR-90	AP-600	SBWR	QP300	AST-500	KLT-40	CANDU-6	CANDU-3	PHWR-500	PHWR-220
Designer / Supplier	ABB	W	GE	SNERDI	OKBM	OKBM	AECL	AECL	NPC	NPC
Reactor Type	BWR	PWR	BWR	PWR	PWR	PWR	PHWR	PHWR	PHWR	PHWR
Gross Thermal Power MW(th)	2350	1940	2000	999	500	Up to 160	2158	1441	1673	743
Net Electrical Power MW(e)	720 - 820	600	600	300	not relevant	Up to 35	666	450	500 (gross)	194

2. Reactors in the basic design stage

Design Name	PIUS	HR-200	CAREM 25	MRX	ABV	GT-MHR	MHTR
Desinger / Supplier	ABB	INET	CNEA/INVAP	JAERI	OKBM	GA	ABB/Siemens
Reactor Type	PWR	-	PWR	PWR	PWR	HTR	HTR
Gross Thermal Power MW(th)	2000	200	100	100	38	600	200
Net Electrical Power MW(e)	610 - 640	-	27	30	6	286	85.5

TABLE 1 (Cont.)

3 Reactors in the conceptual design stage

Design Name	BWR-600	VPBER-600	HSBWR	SPWR	SIR™	ISIS	ATS-150	MARS	RUTA-20	SAKHA-92	MDPR	4S
Designer / Supplier	Siemens-AG	OKBM	HITACHI	JAERI	Consortium ---	ANSALDO	EMBDB	Univ of Rome, ENEA	RDIPE	OKBM	CRIEPI	CRIEPI
Reactor Type	BWR	PWR	BWR	PWR	Integrated PWR	PWR	PWR	PWR	Pool type	PWR	LMR	LMR
Gross Thermal Power MW(th)	2200	1800	1800	1800	1000	650	536	600	20	7	840	125
Net Electrical Power MW(e)	750	630	600	600	320	205	Up to 180	Up to 170	not relevant	Up to 1	325	50

Note to designer/supplier		
	ABB	ABB Atom AB, Sweden
	AECL	*Atomic Energy of Canada Ltd*
	CNEA	Comisión Nacional de Energia Atomica, Argentinia
	CRIEPI	Central Research Institute of Electric Industry, Japan
	EMBDB	*Experimental Machine Building Design Bureau, Russian Federation*
	GA	General Atomic, USA
	GE	*General Electric, USA*
	HTR	High Temperature Reactor
	INET	*Institute of Nuclear Energy and Technology, Tsinghua University, China*
	INVAP	INVAP Company, Argentnia
	JAERI	*Japan Atomic Energy Institute*
	NPC	National Power Corporation, India
	OKBM	*Special Design Bereau for Mechanical Engineering, Nejninovgarad, Russian Federation*
	RDIPE	Reserch and Development Institute of Power Engineering, Russian Federation
	SNERDI	*Shanghai Nuclear Engineering Research & Design Institute, China*
	W	Westinghouse, USA

responses show a similar trend in conditions for supplying NPPs. Suppliers are willing to offer technical assistance to buyer countries. Domestic infrastructure requirements in the buyer's country are not critical, but a sound technical base in staff and industry is preferred depending on contractual agreements. A regulatory body must be established in the buyer country.

The suppliers need a firm commitment from the buyer country to the Non-Proliferation Treaty (NPT) and to international/national regulations before transporting any nuclear materials. Nuclear waste management and disposal have to be arranged by the buyer. Some suppliers are prepared to offer a build-operate-transfer (BOT) contract type.

The responses also update information on SMR design status. Most nuclear suppliers are interested in advanced SMR development for a wide range of applications; some of the advanced SMRs are already in the detailed design stage.

The market potential of SMRs was discussed at the meeting in Tunis, where the questionnaire responses were taken into account. The initial estimate was revised by the IAEA Secretariat after the meeting and is included in Part III.

The market for SMRs until 2015 was assessed by individual countries, taking into account energy demand and supply patterns, growth rates, energy resources, economic and financial resources, electric grids, industrial and technical development, infrastructure availability, environmental and nuclear safety concerns and other policy issues. The market assessment includes all applications of these reactors, that is electricity generation as well as the supply of process heat and district heating.

It is expected that SMRs will be deployed primarily in countries which have already started nuclear projects, in particular in countries which have developed SMR designs themselves. Thus, projects would be supplied predominantly by domestic sources in the years ahead; later, the export market is expected to attain more importance. It is further expected that over two thirds of the SMR units would be in the medium size range, i.e. from 300 to 700 MW(e), the rest would be smaller.

About one third of the SMRs to be implemented are expected to supply heat or electricity or both to integrated seawater desalination plants. More than half of these reactors would be below 300 MW(e) or 1000 MW(th).

The overall market is estimated at about 60 to 100 SMR units to be implemented up to the year 2015. It is recognized that forecasts, just like national development plans, tend to err on the optimistic side. Therefore, an overall market estimate of 70 to 80 units seems reasonable.

5. LESSONS LEARNED FROM NUCLEAR POWER INTRODUCTION

The utilization of experience from the construction and operation of nuclear power plants is important to developing countries in planning and utilizing nuclear energy for power generation or heat applications. Taking advantage of the huge investment already put into nuclear power development and avoiding excessive local expenditure in trying to develop nuclear technologies independently is important to save time and money and to have a sound programme. The general approach as well as the specific steps taken with regard to specific areas such as infrastructure requirements, technology transfer, cooperation and financing by countries that have introduced SMRs provide useful information for other interested countries. Learning from experiences will help to avoid costly mistakes. Worldwide, over 20 countries have been involved in the purchase, construction and operation of SMRs. Fourteen of these are developing countries. This makes the feedback of these experiences quite relevant to other developing countries. Experiences concerning infrastructure requirements, technology transfer and local participation were discussed at the meetings and are summarized below. It was recognized that also the experiences concerning financing and project management are quite relevant. They were not discussed extensively at the meeting but are basically addressed in the conclusions. More detailed information on these areas is contained in other IAEA publications [2, 3].

5.1. Infrastructure requirements

The term nuclear infrastructure refers to the organizations, systems and resources required to implement a nuclear power programme. This includes:

(i) Technical infrastructures
- electric grid
- transportation
- telecommunication

(ii) Organizations for
- regulation and legislation
- programme planning
- project implementation
- plant operation and maintenance

(iii) Qualified manpower
(iv) Industrial support
(v) Financial resources.

A sufficiently strong infrastructure is one of the most important requirements for nuclear power introduction. The infrastructure in the buyer country should be sufficiently well developed to support the introduction of nuclear power with appropriate technology transfer. The effort required to establish the minimum infrastructure should not be underestimated. While the infrastructure requirements for the introduction of a very small reactor are less demanding than for a large NPP and may be developed partly during the plant construction period, the infrastructure requirement for a nuclear programme is quite extensive and requires substantial effort and planning. Institutional and regulatory infrastructure require bilateral as well as multilateral agreements between relevant organizations. These agreements require governmental support, must be long term in nature and should be at all levels in the relevant organizations. Organizations concerned with engineering, development, manufacturing, project management and plant operation must have similar agreements for technology transfer. A broad based relationship between counterparts in vendor and buyer countries must be established. The programme must have manpower development as an important component and should provide mutual benefit to both buyer and vendor countries. In addition to internal training and bilateral cooperation, multilateral cooperation is important to build up the infrastructure.

5.2. Technology transfer and local participation

In order to introduce SMRs successfully into a developing country, a certain level of technology must be present or has to be introduced. These technological capabilities (existing or acquired) should be used for appropriate local participation aiming at a technically sound and economical nuclear programme. This will imply an upgrading of the local capabilities and may lead to a spin off effect to other industries. Identification of fields of interest, the relevant industries and priorities for the selected technologies must be established in the context of the size of the programme (e.g. number of units to be constructed). There are many areas which may be considered for technology transfer in the nuclear field. The topics and the scope indicated below, while not exhaustive, are to be seen as important for staffing a nuclear programme intended to form a major component in the national energy supply:

1. Design and development
   - Basic and on the job training in the design and development areas of interest
   - Joint engineering
   - Transferring and/or sharing of design tools
   - Joint R&D programmes
2. Manufacturing
   - Identification of local capabilities relevant to the nuclear power programme
   - Evaluation of areas of industries which would participate in the nuclear programme
   - Identification of potential constraints in achieving the manufacturing objectives
   - Relevance of technology transfer areas to the overall national plan of industrialization
3. Project management
   - Management structure
   - Tools
4. Quality assurance and quality control
   - Tools and techniques
   - Codes and standards
   - Implementation and commitment of QA/QC programme
   - Application of QA/QC (across the board)
   - Cost and manpower implications
5. Safety and licensing
   - Safety approach
   - General approach to licensing procedures
   - Structure of licensing organization
   - Licensing criteria and regulations
   - Safety culture

It is important to emphasize technologies that are applicable in the short term. The scope of these possibilities should be jointly defined by the vendor and the buyer. This selection process should be built on strengths within local industries and should look for diverse applications of the technology gained. The management of technology transfer should be delegated to dedicated organizations, which are elements in the total management of the programme. Important areas include non-destructive testing (NDT) and in-service inspection (ISI). Another area which is not only important for a particular sector of the programme but rather to the whole of the programme is quality assurance(QA). Quality assurance requires training, early introduction into the programme and the involvement of the user in the vendor's QA programme. The desire for technology transfer at a level for which local industries are not ready, may cause some delay in the construction project and should be avoided.

6. COUNTRY-SPECIFIC SITUATIONS

Argentina

Large gas reserves have been discovered since the start of the nuclear programme. There are large hydroplants coming into operation. ATUCHA-2 (the third NPP in the country) has been under construction for 15 years. The electricity industry has been privatized with the new generation companies seeking a 13% rate of return. All these factors contribute to a situation where nuclear may not be able to compete and orders for new plant are unlikely for many years.

The current increase in demand for power in developing countries will lead to more countries introducing nuclear power but this may not be immediately ahead. The traditional path into nuclear energy is to start with a research reactor and then to buy a larger reactor. The difference in resources needed for these two projects is very large. This applies to human resources, licensing capability, industrial infrastructure, finance and time. An intermediate step utilizing a reactor in the 100-500 MW(th) range which produces something to sell may be an appropriate way to bridge the gap. The CAREM reactor is designed to provide this bridge.

The CAREM project justifications are to maintain national nuclear capabilities and as a possible export to countries proposing to initiate a nuclear power programme. It would form an important step between a low power research reactor and a commercial power plant. CAREM is a 25 MW(e) self-pressurized integral PWR. The Preliminary Safety Analysis Report (PSAR) for CAREM has been presented to the licensing authority in 1996. Basic design and much of the experimental programme is complete. A siting decision for the prototype is expected in 1997. Funds for construction have been requested in the budget for 1997.

Canada

In Canada, the design and development programme is focused on the CANDU family of reactors consisting of:

CANDU 3	450 MW(e)	Market ready
CANDU 6	700 MW(e)	In operation
CANDU 9	900-1300 MW(e)	In detailed design

The next generation plant is CANDUX which will have an alternative coolant at higher temperature and which may incorporate a direct cycle. The CANDU 80 is the latest design in the CANDU line. Among its design requirements were:

- power <100 MW(e).
- Specific costs to be <CAD 3000/kW(e).
- Use of proven systems.
- New features to be applicable to larger plant.

In the design, some basic re-thinking has led to simplifications.

Key features of the CANDU 80 include the use of fully proven systems and technologies from operating CANDU plants, low power density (peak fuel bundle power is 60% of that of current larger CANDU plants), and the incorporation of a number of passive heat removal systems.

There is a double containment with an inner steel containment, serving as a support for major components. It has very little internal concrete. There is no need for a zone control system due to the reduced size and rating of the core. Power control is by absorbing chains which are inserted into the core to a greater or lesser extent. There is no dump tank as was used in previous small CANDU plants. The shutdown systems utilize liquid absorber rods. The basic design has been established, but further design work was stopped in January 1996 pending interest by a potential buyer. There is also the Slowpoke based SES 10 heating plant.

Canada has experience of successful technology transfer to five countries where there have been increasing levels of local participation. A long term commitment on both sides is needed for the process to be successful. It involves all levels of the project from design and analysis down to manufacturing and operation. Both "know-how" and "know-why" are important and occur in different mixes at construction workers and program managers.

For purchasing countries, technology transfer is necessary to take advantage of the huge investment already put into nuclear power development, and to avoid excessive local expenditure in trying to catch up independently. A single co-ordinating organization in the receiving country is recommended. The framework for success requires:

- Comprehensive national planing.
- Organization to develop infrastructure.
- Committed resources.
- Firm management.
- Development of relationships.

China

China is a large consumer of energy. Its total energy consumption ranking is third in the world and the first in coal consumption. Nevertheless, the country has an energy shortage and has a vigorous nuclear programme. The coal deposits are in the north-west and the hydro resources in the south-west of the country. A strong incentive for nuclear is the coal transport problem; 42% of railway and 30% of ship capacity is currently taken up by coal transportation. There is also a serious air quality problem in many cities due to coal burning.

There are 3 power reactors operating, 9 under construction and 20 more are expected by 2010. A large fraction of energy consumption is for heating purposes and China has two reactor development lines to meet these needs:

1. A 10 MW(th) high temperature gas reactor for process heat
2. Nuclear heating reactors NHR 5 (5MW(th)) and NHR 200 (200 MW(th)).

For the NHR 200, the PSAR was submitted in October 1995 and a construction permit has been issued in September 1996. The first reactor will be built in Da Qin City in the North East and is scheduled for completion by the year 2000. Pre-construction work on site has been started. More than 10 other places in China are interested in also installing a NHR 200. China also has isolated areas which could be supplied by SMRs.

The NHR 200 design has been supported by operation of the NHR5 reactor, which has been running for some years, and by an extensive test programme including simulation of some challenging accident sequences. NHR200 is similar in design to NHR5 being an integral low pressure design with natural circulation of the primary coolant, internal control rod mechanisms and passive residual heat removal systems.

The NHR200 can be coupled to a desalination plant to produce 120 000 m^3 per day at a cost of $1.3/$m^3$. The reactor capital cost is estimated at $110 million at 1991 prices. It is expected that the safety features of the reactor would allow its construction within 2 km of a residential area.

France

A newly organized activity on SMRs in France has been started by the Commissariat à l'Energie Atomique (CEA). A study is proposed to look at SMRs with new approaches, fundamentally evolutionary, but possibly with some innovative components. There is a recognition that there are technology thresholds which allow some technical solutions within limits of size and which contribute to low cost and increased safety but which cannot be used at larger sizes. The objective is to study the interactions and relationships between size, modularity, availability, maintainability, threshold effects, safety, risks and economics.

Germany

The objectives of SMR development in Germany are low capital cost and a high safety level. Factors which could help to offset the economies of scale are: use of existing power reactor materials, technology transfer from other projects and prefabrication. A large coolant volume and passive safety systems contribute to the high level of safety.

One design considered a 200 MW(e) integral reactor with hydraulic rod drives and a close fitting containment to ensure core cover at all times. The reactor was designed for 40 years, for which two core loads are needed. Within the vessel is storage capacity for the spent fuel. The design has U-tube heat exchanger and an intermediate circuit.

Utilization of this system for desalination can yield 40000 to 150000 m^3/day but the higher levels need net input of electricity from the grid. Construction in Germany is unlikely and feasibility studies carried out in the Czech Republic and China did not result in orders.

There is also work on SWR600, a 750 MW(e) BWR for which basic design has started with possible extension to 1000 MW(e). The plant is designed for 60 years with spent fuel storage for 40 years. There are passive safety systems with auto-initiation. The basic design and the preliminary safety analysis report are scheduled for completion in 1999.

India

The Indian programme started in 1948 and has three major components.

1. use of natural uranium in PHWRs
2. use of U/Pu/Th in fast reactors
3. use of U-233/Th in fast reactors

For the first stage, there are currently 8 units, mostly of 220 MW(e), in operation. There has been a policy to carry out design, construction, operation and maintenance with Indian personnel only after the first two 220 MW(e)plants. A 14 MW(e) breeder demonstration reactor is under construction as part of the second phase of the programme.

There have been problems with indigenous production but these have been overcome and a degree of standardization has been established. Construction of a 220 MW(e) plant should now require 7 years from first concrete pouring. For Tarapur 3 and 4, a new 500 MW(e) design has been established but this now awaits financial support.

Italy

Italy is looking for innovation to re-establish a reactor programme. There are two concepts under consideration, the European passive reactor, a 1000 MW(e) variant of AP 600, and the ISIS reactor. ISIS has followed the concepts of simplicity to reduce costs and passivity to ensure safety.

Nuclear power costs have been challenged by low fossil fuel costs and an increase in efficiency of fossil fuel plant from 33% in 1963 to 54% today and possibly 60% (by the year 2000) by the use of high temperatures and dual cycle turbines. Co-generation in a nuclear plant could overcome the cost reduction of the dual cycle and ISIS is suited to this application.

Japan

The Ministry of Science and Technology is giving strong support to advanced reactor design, and the proposed national programme has some scope for development work. MRX and the smaller DRX are integral reactors designed for marine use. They have control rod drives within the reactor and have a close fitting water filled containment. The thermal power of the MRX is 300 MW(th), and 500 kW(th) of the DRX. The passive cooling systems make use of the containment water. Results of safety analysis demonstrate satisfactory behavior in the case of design basis accidents and beyond. The design is supported by an extensive R&D programme including a thermal hydraulics rig and a control rod drive development rig.

A one piece removal system is proposed for refuelling and maintenance. The entire reactor, including its water filled containment would be lifted out of the ship and replaced with another one which had already been refueled and maintained.

On economics, a ship carrying 6000 containers as part of a 20 ship fleet operating across the pacific at 30 knots would be economic in comparison with diesel. High speed favors the nuclear option.

A proposal of the utilization of small lead and lead-bismuth reactors in developing countries was presented by Japan. Reactors of 150 MW(th) with a 12 year period between refuelling and with a B_4C shield have been studied. They have 0.1% k/k reactivity swing over 12 years, negative temperature coefficient for the core as a whole and a peak burnup of 9%. There are differences in void coefficient with the lead/bismuth coolant and metal/nitrite fuel giving a high negative coefficient. Accident analyses with all four combinations (Pb, Pb-Bi, metal fuel, nitride fuel), show very satisfactory responses.

Japan has 13 SMR designs developed by 6 organizations, 3 of which are research organizations and 3 are manufacturers. The electrical outputs range from 4 kW(e) to 100 MW(e).

Republic of Korea

The Republic of Korea has started development work on a medium size integral reactor for co-generation, to be used for power generation and seawater desalination. The goal of this project is to complete the design work by 2005. The stages are:

complete conceptual design	1997
complete basic design	2000
complete detailed design	2005

The size is to be in the range of 100-600 MW(e) with initial work concentrating on 300 MW(t) size. The reactor is an integral design with no dissolved boron, except for emergency shut down, and a low power density. It is self-pressurized with a mixture of nitrogen and steam and no pressurizer heater. The steam generators are of helical coil type. There is a guard vessel half filled with water and an additional containment. It has passive safety systems and the valves which must open to initiate the decay heat removal system are opened by passive means.

Morocco

Morocco is a country of 27 million inhabitants of which 51% are urban dwellers. 80% of the rural regions are without electricity. The present sources of electrical energy are: oil 74%, coal 24%, both imported, hydro 2% and gas 0.25%. Estimates based on 7% growth to the year 2005 and 5% thereafter lead to the following figures for the installed capacity:

1994	3500 MW(e)
2000	5300 MW(e)
2005	7000 MW(e)

There are grid links with Spain and Algeria and there is continued prospecting for oil. A serious water deficit is expected in several places unless action is taken. There are several more plans for desalination facilities to come on stream at various dates after the year 2000. A feasibility study for nuclear desalination has commenced as a joint exercise between Morocco, China and the IAEA based on a 10MW heating reactor. A move to a bigger plant could be around 2010.

There is some existing and planned desalination capacity in Morocco. The first plant at Boujdour was commissioned in 1977 and yields 250 m^3/day by Mechanical Vapour Compression at a cost of 50 DH/m^3 ($6/$m^3$). The next two plants came into operation in November 1995 and yield 800 m^3/day and 7000 m^3/day using Reverse Osmosis and costing 43 and 21 DH/m^3 respectively. Water prices to consumers in Morocco, where available, are 2-6 DH/m^3 depending on location and quantity consumed.

The first priority is the development of trained manpower. Nuclear Science is taught in 12 universities and research centers. Post graduate studies, on related subjects, started at the university of Rabat in 1978. There are now activities in other universities as well. Training is given in all nuclear application sciences including industrial, medical, agriculture and food science applications.

Regulation in Morocco is currently a joint responsibility of several ministries. CNESTEN is linked to the ministry of education and is currently planning a nuclear research center (CEN project) with assistance by France on a 25 ha site. It will have a 2 MW TRIGA reactor and 250 staff.

A proposed nuclear programme over the next 40 years is to install 8-12 units. The first should be commissioned in 2008. The sizes would be either 4 × 300 MW(e) to 2020 followed by 4 × 600 MW(e) or 8 × 450 MW(e). The reactor type is not yet chosen.

Pakistan

Pakistan started its nuclear power programme in the sixties. KANUPP is the first nuclear power reactor; it is a 137 MW(e) CANDU type and has been in operation since 1972. CHASNUPP is a 300 MW(e) Chinese reactor planned to operate from 1998.

The first plant had many operational problems due to lack of vendor support. It has resulted, on the other hand, in the development of self-reliance in fuel, spare parts, D_2O production, fuel management and technical support. KANUPP is now loaded entirely with locally fabricated fuel. Much work has also been done to counter the effects of ageing and obsolescence. Some of the tasks undertaken are:

- A new control and instrumentation system
- Techniques developed for dealing with steam generator leaks including a plugging capability.
- Dealing with valve failures.

As a result of this development of local infrastructure, reactor life is expected to be extended ten years beyond the original design life.

In Pakistan, population is concentrated in the central region around the Indus river and its tributaries. The western part of the country is arid and has potential for desalination along the coastal area. There are also gas and oil fields. The population is 128 million and the per capita GNP is US $427. Electricity consumption per capita is 420 kWh and it is available to only 57% of the population. At present 20% of export earnings are spent on imported oil. Coal has recently been discovered but the proven reserves of all fossil fuels are small. About 15% of the hydro potential has been exploited. The electricity production capacities in 1995 were:

Hydro	4825 MW(e)
Fossil	7572 MW(e)
Nuclear	137 MW(e)
Total	12 535 MW(e)

By 2020 the gap between production from indigenous supplies and demand is estimated at 43,500 MW(e). It is proposed to construct 4050 MW(e) by 2010 and 11125 MW(e) by 2020. In the short term, Pakistan will buy plants from abroad but in the long term, it is hoped to develop a complete design and supply organization.

To help reduce the shortfall in electricity production, foreign companies have been invited, under favorable trading terms to construct their own electricity generating plant. So far 3000 MW(e) has been agreed for operation during the next two years.

Romania

In 1979, Romania signed a contract to build 4 CANDU 6 reactors on one site and in 1982 a fifth unit was added. Criticality of the first unit was reached in April 1996. It was synchronized in June and reached full power in September 1996. It will save at least US $100 million p.a. in fossil fuel costs and this money can be put towards completing unit 2 by the year 2000. Romania has its own natural uranium resources. During the period since 1979, a heavy water production plant and a CANDU fuel fabrication facility were built and are operating well. The D_2O plant has produced 600 tons of D_2O reactor grade quality. The fuel plant has produced many fuel elements of which 200 will be used in the first reactor. Commissioning of the first two reactors will result in savings of at least US $ 200 million p.a. on fossil fuel purchases.

Technology transfer has been emphasized and personnel training both in Canada and Romania has been extremely important. Deficiencies in management were recognized in 1990, and a joint company with AECL and Ansaldo was formed. This company will initially operate the first reactor and is contributing further to technology transfer.

As a result of the process of technology transfer there is a proper understanding of the need for national competence, firm management, independent regulation and of the importance of quality control as well as of the basic technology.

This programme was presented as an example to other developing countries of what is necessary if they wish to enter the nuclear field. Lessons learned include the need for a good organizational structure with an independent licensing body, the need to develop competence and the merits of selecting a good partner vendor.

Russian Federation

Russia has experience in the design, construction and operation of several SMRs particularly at the lower end of power. At the Institute of Physics and Power Engineering (IPPE) current development is concentrated on KLT40c and ABV67. The former is based on the successful icebreaker reactor design. Both are integral designs, used in a co-generation mode. The following outputs are possible in a barge mounted plant:

Reactor	Reactor power MW(th)	Electric output MW(e)	Heat output MW(th)
KLT 40c	2 × 150	2 × 35	2 × 29
ABV67	2 × 38	2 × 6	2 × 14

An ABV is constructed for military use VOLNOLOM project. There is also a KLT 40c based project. There are many potential sites on the north coast of Russia but construction can also take place further south where climatic and industrial conditions are more favorable.

A desalination version in which all the plant is barge mounted has been conceived with an expected water cost of US $ 2/m^3. In this case, a facility can be relocated from one site to another prepared site and be operational again in two months. There is lengthy experience of nuclear desalination at a plant in Aktau, Kazakstan using the MED process.

Also an OKBM integral design operating at 500 MW(th) for district heating was constructed but later dismantled. A second plant has been under construction for 10 years.

In the small reactor range, the Russian Nuclear Society organized a contest to select the best of the many Russian designs in different energy and application ranges. Those that were chosen ranked in 3 levels of merit.

The results were:

Power range	<10 MW(th)		10-15 MW(th)			>50 MW(th)	
Application				Co-Gen		Co-Gen	
	Heat	Co-Gen	Heat	Land	Floating	Land	Floating
1	Elena	-	Ruta	Angstrem	ABV6	ATS-80	KLT-40
2	-	Shakha 92		ABV 6	NICA 120		NICA 500
3	-	TES-M		JTEV-M		VK-50	

Elena is a 3-4 MW(th) plant with thermo-electric generation, using technology from the space reactor programme, giving 100 kW of electricity. It is designed for unattended operation for 30 years without refuelling.

Angstrem is a modular mobile Pb/Bi cooled reactor and ATS-80 is a modification of ATS-150 (the design description is included in TECDOC-881).

Russia has experience in the design, construction and operation of several reactors in the medium range. The medium size reactors are

WWER 440 (V213) developed by GIDOPRESS:
There are 16 units operating in 7 countries.

WWER 640 (V407) also developed by GIDOPRESS:

This design was initiated after Chernobyl. It is essentially an evolutionary development of previous WWERs. It has a core catcher, double containment and high level water tanks, and is in the final stage of design. It has been licensed for construction at two sites, but construction is delayed due to financial problems.

VPBER 600

Integral design developed by OKBM. It is in the basic design stage but work on it has been suspended.

Syrian Arab Republic

The Syrian Arab Republic has a population of 14.3 million increasing at 3.3% per year. The increase in energy production is 13% per year. Electricity production capacity is 5000 MW(e) available to 96% of the population. The projected increase in demand will require an installation programme of 500 MW(e) per year up to the early part of the next century.

There are hydro, gas and oil resources but the gas and oil are expected to run out around 2015. By 2010, the estimated energy deficit based on indigenous supplies will be about 4 million tons per year. No decisions have been made on going nuclear, importing electricity or other energy options.

Tunisia

The country has rising energy needs and falling supplies of indigenous fossil fuels. There is a new off-shore gas field which is expected to last until 2015. The main pipeline from Algeria to Europe passes through Tunisia and some supplies are taken from it. The nuclear option is considered both for electrical energy and for desalination.

As far as siting is concerned, the demand is centered in the North but this is an area of seismic activity. It is essential to site any dual purpose facility by the sea. By 2015, a 600 MW(e) plant will be acceptable on the electric grid.

The water supply and demand in the next 20 years is another important area. Average water consumption is 100 liter per person per day, but it is 550 liter/person-day in the tourist areas.

The total annual water consumption is 2.2 x 10^9 m^3. By 2010, the deficit is estimated to be about 15 x 10^6 m^3/year, i.e. about 40,000 m^3 per day. A combined desalination (2000m^3/day) and electricity (120 MW(e)) pilot plant is under construction and a solar powered desalination project is planned.

7. REAL-TIME COMPACT SIMULATORS OF LWRs

Simulators are tools to represent the dynamic behavior of physical and technical systems. Simulators are widely used for various real systems (e.g. airplanes, cars, power plants). Nuclear power plant simulation is used for operator training, design, development and safety aspects. Depending on the objectives, the characteristics of the simulator may differ. In the course of the IAEA activity on the introduction of small and medium power reactors in developing countries, it was found that nuclear power plant simulators would be of great benefit to countries who are interested in the introduction of nuclear power for electrical power generation. A simulator package could give them a tool to become acquainted with operational and safety systems of a nuclear plant. It would be convenient to have the major reactor types simulated in one package before deciding on the most suitable reactor type. It is understood that the package could be of interest to other Member States who are upgrading or expanding their existing nuclear capacity. Some background work by the IAEA has already been conducted to initially identify the main objectives, scope of simulation, training function and material and the intended target group. The result of this work by the IAEA was presented to the Advisory Group. Highlights of presentations on the subject at the Advisory Group meeting are:

In Japan, several activities were carried out in various organizations and research institutes. There are three levels of simulators:

1. Compact simulators for basic studies
2. Engineering simulators for plant behavior including abnormal events
3. Replica simulators for plant operator training.

Three compact simulators were described:

1. The Plevis simulator from TOSHIBA
2. The MPS simulator from EUROSIM/CRC
3. The ANPP simulator from the Ship Research Institute.

The first two are available at a price of US $ 0.5-0.6 million including hardware. The ANPP simulator is still under development.

In Egypt a simulator is being used at the Nuclear Power Plant Authority. The simulator is set up for PWRs in normal operation and for abnormal conditions including large break LOCA. It can also simulate PHWRs of the CANDU type. It has been used for a wide range of training activities. Xenon effects are modelled and many accidents can be simulated. The simulator uses work station computers and was bought from France for about FRF 5 million in 1991.

Major work of the Canadian Company CAE which is an important company in the field of simulation, covering aeroplanes, power plants and chemical process plants, involves the simulation of all major lines of water-cooled reactors. They supply full scale simulators as well as compact desk top type simulators. The desk top simulators currently operate on work stations but are now being adapted for 486/pentium processors. This process will be completed by March 1997. The simulator software will be made available to the IAEA.

In France, there are four organizations that are active in simulation. CEA produces OASES for LMRs and CORIANDRE 2 for LWRs under normal and abnormal operation. There are other lower order codes.

EDF SIPA simulator includes post-accident simulation. SIPA compact is planning to reproduce the simulator capability in a desk top environment. SCAR is being developed to incorporate more capabilities into SIPA. It is planned to be ready by the year 2000. It will give greater flexibility in modelling. CORIS and THOMSON are French companies who are also active in simulation with activities in nuclear power plants simulation, software workshops, tools for training, maintenance of industrial process simulators and control room simulators.

For CHASNUPP in Pakistan, a full scope simulator is being developed including a replica control room. The simulator will be available in late 1997. The KANUPP reactor is being upgraded with extensive re-instrumentation of the control room. The simulator, which is not a full scope one, is being upgraded in parallel.

Based on the input of these presentations, the working session discussion and the proposal provided by the IAEA to develop a reactor simulator to be distributed to interested Member States for training purposes, the main technical specifications of such a simulator were identified. Adoption of the IAEA proposal for a PC based simulator with PWR, BWR and PHWR simulation capability was agreed as a first step. It was expected that further simulators to cover all reactor types would become available later. A general consensus on the purpose of the package is to familiarize the user with the characteristics of nuclear plant and to appreciate the difference between various reactor types. The target audience would be:

- R&D personnel;
- University students at the post-graduate level.

A training programme will include the following areas:

- Basic training on the reactor characteristic of the types simulated;
- Operational aspects, initiation of transients and scenarios to identify transient conditions as they occur.

8. CONCLUSIONS

(1) There is a large experience with SMRs and many lessons learned on SMR introduction were discussed. There is a clear need by developing countries for information in several areas connected to SMR deployment.

(2) A sufficiently strong infrastructure is one of the most important requirements for the introduction of nuclear power. This comprises qualifed manpower, technical and organizational infrastructure, industrial support and financial resources. Local participation and technology transfer were emphasized as important elements in introduction of nuclear power.

(3) Infrastructure requirements for the introduction of a very small reactor are less demanding than for a large NPP. The infrastructure for a very small reactor may be partly developed during the plant construction period.

(4) There are many SMRs under development or design by vendors, governmental organizations and research centers, which indicate a large international interest. A number of SMR designs is suitable for both power generation and co-generation. A consolidation of SMR development efforts may be useful.

(5) Cost reduction should be a major objective of reactor development to improve the economical feasibility of SMRs.

(6) A governmental commitment to utilize nuclear energy for power production is essential for nuclear power to be introduced in a developing country.

(7) The introduction of nuclear power can make a substantial contribution to technological and manpower development in a developing country.

(8) The availability of financing at reasonable terms is a key factor for the feasibility of a nuclear power project. As much as possible of the local cost component of the project should be financed in local currency from sources within the host country. Lack of financing in many developing countries is the most important constraint for more extensive SMR deployment.

(9) Efficient control of quality, costs and schedule are vital for a successful project.

(10) It is recommended that an activity on the user requirements for the SMR range be carried out on regional and global level. This activity should include also requirements on the infrastructure and financing.

(11) The market potential is estimated at about 70 to 80 SMR units to be implemented up to the year 2015.

REFERENCES

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

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Financing Arrangements for Nuclear Power Projects in Developing Countries: A Reference Book, Technical Reports Series No. 353 (1993).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Power Project Management: A Guidebook, Technical Reports Series No. 279 (1988).

BIBLIOGRAPHY

INTERNATIONAL ATOMIC ENERGY AGENCY, Guidebook on the Introduction of Nuclear Power, Technical Reports Series No. 217, IAEA, Vienna (1982).

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

INTERNATIONAL ATOMIC ENERGY AGENCY, Non-Electric Applications of Nuclear Energy, IAEA-TECDOC-923, Vienna (1997).

PART I

STATUS AND INTRODUCTION OF SMALL AND MEDIUM REACTORS

ENERGY DEVELOPMENT AND NUCLEAR PROGRAM IN CHINA

XUE DAZHI
Institute of Nuclear Technology,
Tsinghua University,
Beijing, China

XA9846701

Abstract

In this paper the current situation of energy consumption in China is provided. Coal-burn as a dominant sector of energy consumption causes heavy burden on transportation and serious environmental pollution. The roles of nuclear energy in the future energy supply are discussed. The situation of nuclear development, especially heating reactor is introduced.

1. DEVELOPMENT OF ENERGY INDUSTRY IN CHINA

The energy industry in China was developed very fast in the past 20 years. The annual average growth rate of primary energy output and consumption was 10.4% and 11% respectively (refer to Fig.1 [1]). Especially, remarkable results have been achieved since reform and opening to the outside world. The total output and consumption of primary energy had reached 1,188 million and 1,227 Mtce respectively in 1994, ranking among the first three in the world. Among them, the output of coal was 1,240 million tons, ranking the first in the world; the output of crude oil was 146.08 million tons, ranking the sixth in the world; that of natural gas was 17 billion m^3; electricity production was 927.8 billion kwh, ranking the fourth in the world. (refer to Fig.2[1]). Besides the above commercial energy additional 230-250 Mtce of non-commercial energy were consumed in the rural areas in 1994.

2. THE FEATURES AND MAIN PROBLEMS

In spite of the great achievements of world interest, Chinese energy industry falls far short of the needs of socio-economic developments and is one of the main factors restraining a sustained developments of the economy. Also, coal-burn as a dominant sector of energy consumption causes heavy burden on transportation and serious environmental pollution.

2.1 Shortage of primary energy

In the past years the total supply of energy can basically meets the needs of the economic development. But in light of the economic development target proposed by Chinese government the shortage of primary energy will become gradually outstanding. In order to forecast the prospect of energy supply and demand in next century a estimation of primary energy consumption and composition has been carried out in INET and other institutions. Two results respected high and low estimation are shown in Table 1[2] . To meet this demand a great efforts should be paid to energy conservation and production.

2.2 The structure of energy consumption

With a speedy development of petroleum and natural gas industry, hydropower and other energy resources, China's energy structure improved gradually and a structure of energy production and consumption taking coal as the dominant and mutually supplemented by various energies was initially formed. By 1994, the proportion of coal in the production and consumption of energy had gradually dropped to around 75%, while that of oil, natural gas and hydropower (nuclear power) increased to 17.5%, 2% and 5.5%, respectively. (refer to Fig.3[1])

Coal-burn as the dominant energy consumption causes great pressure on energy transportation and environmental pollution. The coal resources in China is quite geographically uneven distribution (refer to Fig.4[1]). Coal resources mainly locate in middle and west areas, coastal

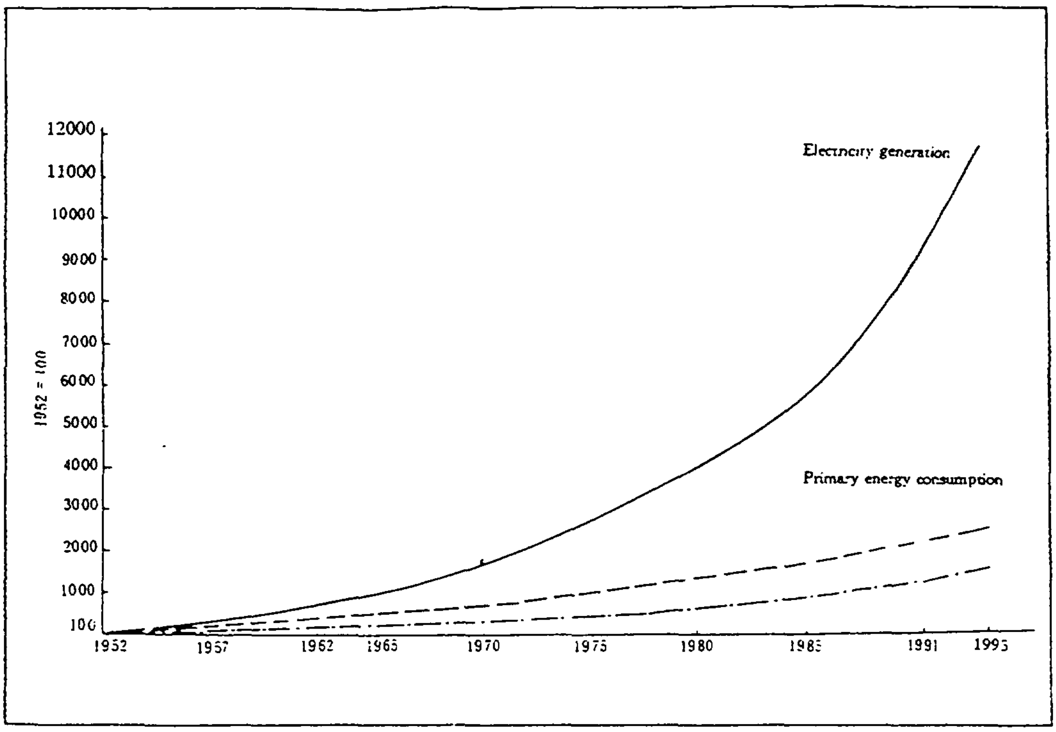


Figure 1. Energy Consumption and Economic Growth, 1952~1993

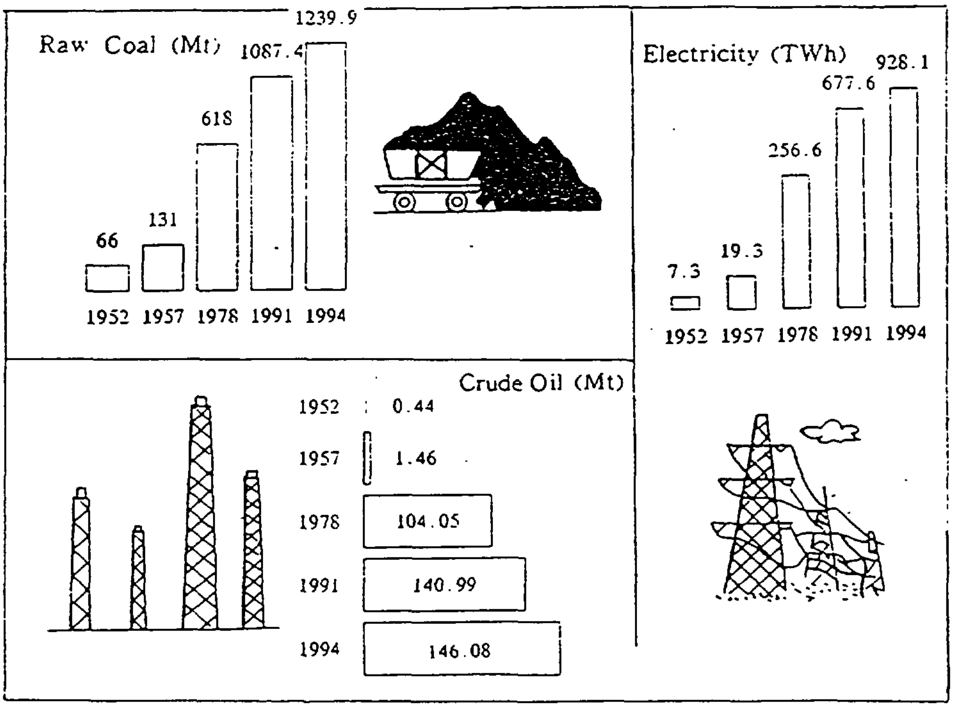


Figure 2. Energy Output, 1949~1994

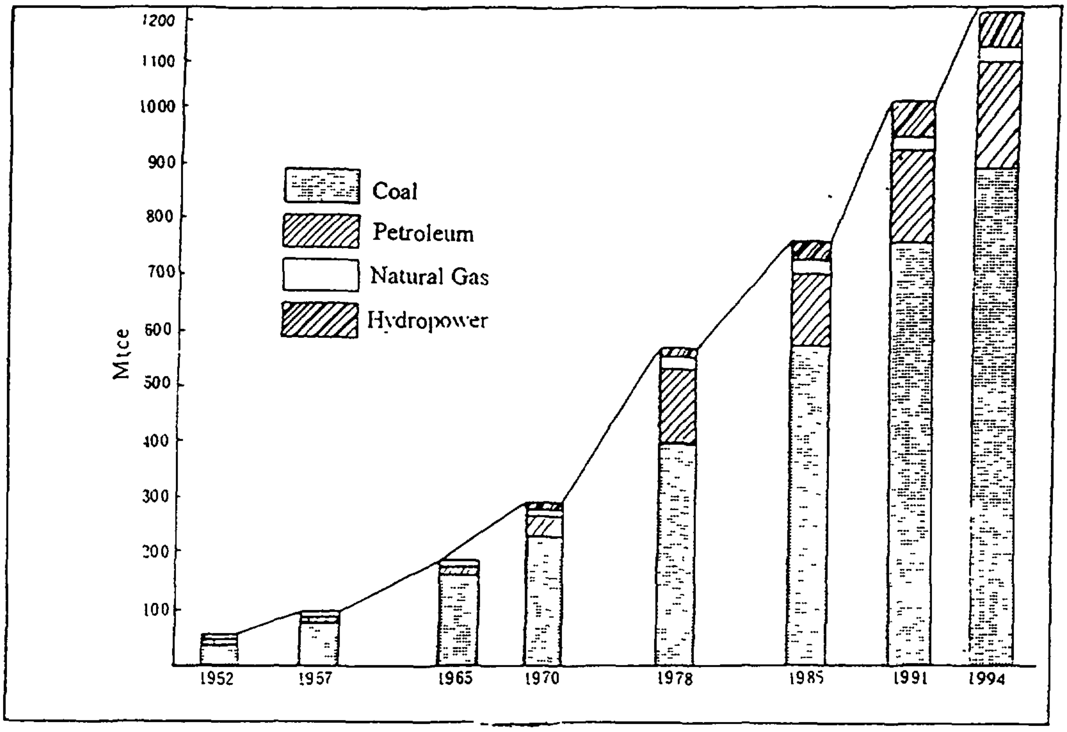


Figure 3. Primary Energy Consumption Mix, 1952~1994

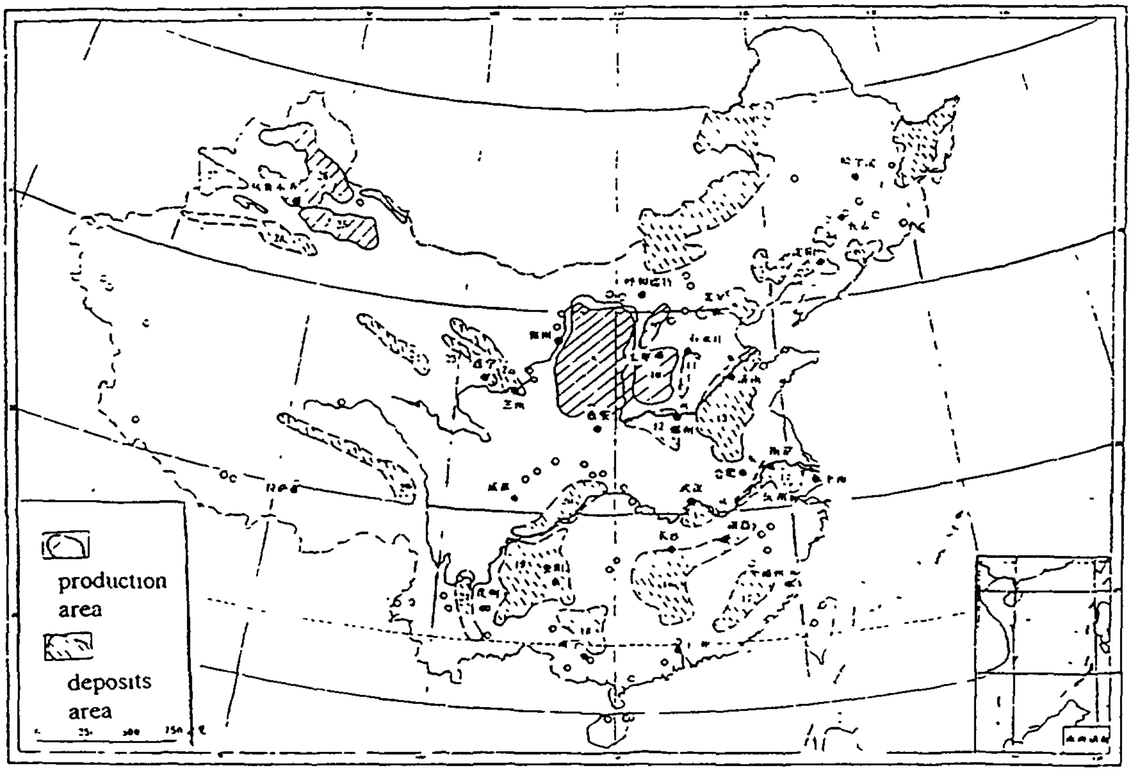


Figure 4. Distribution of China's coal resources

TABLE 1. ESTIMATION OF PRIMARY ENERGY CONSUMPTION AND COMPOSITION

Version		1990		2000		2010		2030	
H_A_P	GDP ($\times 10^8$ RMB)	17681		45861		99010		317468	
		Consumption (Mtce)	Fraction (%)	Consumption (Mtce)	Fraction (%)	Consumption (Mtce)	Fraction (%)	Consumption (Mtce)	Fraction (%)
	Total	987.0	100.0	1433.7	100.00	1937.4	100.00	2912.8	100.00
	Coal	734.0	74.37	1056.2	73.67	1321.7	68.22	1574.5	54.05
	Oil	168.2	17.04	225.4	15.72	288.9	14.91	401.0	13.77
	Gas	20.3	2.06	38.4	2.68	80.4	4.15	177.6	6.10
	Hydro	50.3	5.10	105.6	7.36	179.0	9.24	287.4	9.87
	Nuclear	0.0	0.00	6.7	0.46	64.1	3.31	392.4	13.47
	New Energy	0.7	0.07	1.5	0.10	3.3	0.17	80.0	2.75
L_A_P	GDP ($\times 10^8$ RMB)	17681		41858		82341		218426	
		Consumption (Mtce)	Fraction (%)	Consumption (Mtce)	Fraction (%)	Consumption (Mtce)	Fraction (%)	Consumption (Mtce)	Fraction (%)
	Total	987.0	100.0	1367.6	100.00	1776.3	100.00	2495.9	100.00
	Coal	734.0	74.37	998.7	73.02	1206.8	67.94	1373.4	55.03
	Oil	168.2	17.04	217.0	15.87	270.0	15.20	353.1	14.15
	Gas	20.3	2.06	38.6	2.82	74.2	4.18	164.0	6.57
	Hydro	50.3	5.10	105.1	7.69	162.6	9.16	275.1	11.02
	Nuclear	0.0	0.00	6.7	0.49	59.3	3.34	278.7	11.16
	New Energy	0.7	0.07	1.5	0.11	3.3	0.18	51.5	2.06

area, east and south China are economic developed, densely populated and high energy consumption but short of energy resources. So, a big amount of coal transportation and long transport distance for coal (refer to Table 2) is necessary and leads the already serious transportation situation even worse.

TABLE 2. TRANSPORTATION OF COAL IN 1994

	By Train	By Ship
Freight Transport of coal (Mt)	659.4	94.7
Fraction of Freight Volume (%)	42.0	29.7
Average Transport mileage (Km)	544	2622

Coal-burn has caused serious air pollution in many cities. Some example are shown in Table3. Moreover the long-term CO_2 problem also have to be dealt with.

TABLE 3. AIR QUALITY ($\mu g/M^3$)

	Particle	SO_2	NO_x
Recommendation value by WHO	60-90	40-60	
National Standard			
(I)	150	20	50
(II)	300	60	100
Average daily concentration			
northern cities	429	92	
southern cities	225	88	
Shen Yang City			
Summer	560	45	47
Winter	744	295	92
Chong Qing City			
Summer	600	260	50
Winter	870	660	110

For a long time past the liquefied fuel is not sufficient in China and shortage of liquefied fuel will become more serious in the future.

In light of above, the Chinese government pays attention to the development and application of the clean coal technology in one hand, and in the other hand, various energy sources are explored as a supplement.

3. NUCLEAR ENERGY IN CHINA

During the 1980s the first two nuclear power plants were started for construction. The first phase of Qingshan plant (300 MWe PWR) is the first nuclear power plant designed and built by China. In was completed and interconnected with power grid in 1991 and went into commercial operation on 1994. Daya Bay plant (2x900MWe PWR) was imported from France and put into operation on 1994. In 1990s there are additional 4 plants (8 units) are planned or started for construction (refer to Table 4). It is expected that a large development of nuclear program will be appeared in the beginning of next century. Besides that, some research programs on new type of reactors are being conducted, including a 65 MWth experimental fast breed reactor; 10 MWth high temperature gas cooling reactor etc.

TABLE 4. NUCLEAR POWER PLANT IN CHINA*

Nuclear Power Plant			
NPP	Type	Capacity (MWc)	Remark
Qingshen 1#	PWR	1×300	Commercial Operation in 1994.4
Guangdong 1#	PWR	2×900	Commercial Operation in 1994.2 & 1994.5
Qingshan 2#	PWR	2×600	Under Construction
Guangdong 2#	PWR	2×900	Under Construction
Liaoning	VVER-1000	2×1000	Under Negotiation
Qingshan 3#	CANDU	2×700	Site Preparation

* Another 20GW. NPP are planed by the year 2010. Some early stage work are under way for some of them.

In order to achieve the fixed goal of China's economy development, an ambitions development of nuclear energy is an indispensable way. The important roles of nuclear energy in future energy supply is as follows:

- Nuclear energy is a sole energy resource that could substitute coal at large scale with competitiveness economically to make up the huge gap of future energy supply.
- Nuclear energy-coal conversion to produce liquefied fuel is a feasible way to overcome the liquefied fuel shortage.
- Nuclear energy is the important basis of the future clean energy system to solve the long term energy-environment issue.

4. DEVELOPMENT OF NHR

Among the end user of coal consumption, around 10% is for domestic use. In order to enlarge the utilization of nuclear energy, the research work on the application of nuclear heat was initiated in early eighties in Institute of Nuclear Energy Technology (INET), Tsinghua University. As a result, a 5MWth nuclear heating test reactor (NHR-5) with an integrated vessel type was designed and built during 1986-1989. Since 1989 the NHR-5 has continuously operated for three winters successfully. After that a series of experiments and operation tests were carried out.[3] It has been shown that the NHR-5 possesses excellent safety characteristics and a high operation availability In order to extend the application of heating reactor, some experimental equipment have been installed and tested. The results show that the NHR can be used for district heating, air conditioning and sea-water desalination and other industrial processes.

For speeding up the process of the NHR commercialization, it is decided to build a demonstration plant with output 200 MWt in northeast of China. It is expected that several NHRs will follow up after the first one successes.

REFERENCES

[1] "95 Energy Report of China", Department of Communications & Energy, State Planning Commission of P.R. China.

[2] He Jiankun, "The Estimation of Long-Term CO_2 Emission in China." (in Chinese) Internal Report, Dec. 95.

[3] Wang Dazhong, et al., "Experimental Study and Operation Experiences of the 5 MW Nuclear Heating Reactor", Nuclear Engineering and Design, 143 (1993)

NEXT PAGE(S) left BLANK

XA9846702

PROPOSAL FOR THE CEA/DRN ACTIVITY ON SMALL AND MEDIUM SIZE REACTORS RESEARCH ON THRESHOLD EFFECTS

G.L. FIORINI
Centre d'études de Cardache,
Saint-paul-lez Durance Cedex,
France

Abstract

The discussion on Small and Medium Size Reactors - SMR is difficult considering the presumptions, justified or not, that affect the debate. Nevertheless, within this context, the CEA/DRN/DER generic objective is the achievement of an exhaustive identification and assessment of the problems that are specific for the SMR

The paper shows the proposals for the activities that are actually **under discussion** at the CEA/DRN

Among these activities, the research on **threshold effects** is an essential stage in the **assessment of the choices in innovative concepts** This research, as well as the assessment itself, must cover, in an exploratory way, the aspects of operation, safety, economy, fuel cycle, etc.

Before starting or, in some cases, continuing this research work, it seems interesting to define a **general outline** which, by systematising the approach, provides a helpful tool to the designer

The document is a potential starting point (among others) for the discussions

1 - INTRODUCTION AND OBJECTIVES

As a preamble it must be pointed out that the discussion on Small and Medium Size Reactors - SMR is difficult. This established fact can be understood considering the presumptions, justified or not, that affect the debate. They concern cost, safety, availability etc.. Within this context, the CEA/DRN/DER generic objective is the achievement of an exhaustive identification and assessment of the problems that are specific for the SMR This assessment must covers the safety aspects as well as the economic ones taking into account the different items: design, construction, utilisation (electricity production, cogeneration, district heating, desalination, etc), operation, availability, maintenance and dismantling

The corresponding technical objective is formulated as follow: Achievement of a technical-economical study which leads to the setting up of a motivated set of plant specifications (*"cahier de charges")* for an SMR The details of the activities are actually **under discussion** at the CEA/DRN (see section 2 and 3).

Among these activities, the research on **threshold effects** is an essential stage in the **assessment of the choices in innovative concepts** This research, as well as the assessment itself, must cover, in an exploratory way, the aspects of operation, safety, economy, fuel cycle, etc CEA/DRN will include such a work within the frame of the activities of the **Innovative R&D Program**

Before starting or, in some cases, continuing this research work, it seems interesting to define a **general outline** which, by systematising the approach, provides a helpful tool to the designer (see section 4).

2 - MOTIVATION FOR THE ACTIVITIES

For future reactors, the safety and economic improvements are the objectives that lead to define and justify the guidelines of the CEA/DRN Innovative Programme.

The **safety improvement** can be achieved (the list is not necessarily exhaustive):

- Searching for design simplification in order to improve the plant transparency
- Incorporating more intrinsic safety characteristics for the accident prevention and management as well as for the consequences mitigation.
- Increasing the role of the passive safety systems especially for the accident management and consequences mitigation. *Nota Bene: this implementation must be envisaged only after the verification of adequate criteria that concern the system performances, its reliability and the economy aspects.*
- Improving the man-machine interface in order to reduce human factor.
- Improving the plant inspectability, maintenability and repairability.
- Increasing the components/systems standardisation as well as the concept - as a whole - standardisation. *Nota Bene: concerning the components/systems, the standardisation must verify diversification criteria which are essential for an acceptable management of Common Mode risks.*
- Reducing the frequency of plant abnormal conditions (Increase the plant availability).

The plant economy improvement can be achieved pursuing the following technical objectives (the list is not necessarily exhaustive):

- Increasing the plant life.
- Optimising the fuel cycle both for fuel manufacturing, fuel flexibility versus its use and for the fuel cycle end.
- Ensuring the design stability to simplify the licensing procedures.
- Minimising the components manufacturing costs.
- Shorting the construction delays.
- Minimising the operation and maintenance costs.
- Improving the plant availability.
- Minimising the dismantling costs.
- Reducing the risks for the loss of the investment.

The achievement of the suggested technical-economical studies, must try to take into account all the guidelines-objectives listed before. Within the frame of a SMR assessment, these objectives became criteria that lead to judge the interest for a given solution. This can concern a design choice (component/system), an operation mode (base, follow-on) or a given plant use (electricity, cogeneration, etc.).

Some recent DRN studies (boron free PWR, relationship among Modularity and Safety/Availability/Economy) show that SMR have advantages - and certainly drawbacks - that must be analysed as a whole to achieve a motivated and exhaustive assessment.

3 - TASKS

The basis for the discussion is organised through the following tasks

3 1 ▫ Establishment of the economical objectives (requirements) that must be achieved by the SMRs versus their planned use The comparison terms can be founded using the data coming from the other competing technologies (e g Gas-turbines or Coal plants)

3 2 ▫ Review of the safety objectives and of the possible safety options

3 3 ▫ Detailed analysis about the global plants cost sharing either for the design, the construction, the operation/maintenance and the dismantling

3 4 ▫ In conjunction with the previous item: analysis of the relationships among the size, the modularity and the availability taking into account the possibilities for the resources sharing

3.5 ▫ In conjunction with the item 3.3: analysis of the relationships among the size and the threshold effects[1].

3 6 ▫ Still in conjunction with the item 3.3 and taking into account the safety objectives identified under the item 3 2 analysis of the relationships among the size/modularity and the safety options as well as the global plant risks

3.7 ▫ Conclusions and propositions for a detailed ad-hoc R&D programme.

4 - RESEARCH ON THRESHOLD EFFECTS

4.1 - In general

The relationship between the specific cost and the plant size is currently considered as a decreasing function (the specific cost decreases with the increase of plant size) This is a major handicap for the development of **Small and Medium Reactor (SMR)**

Among the tasks listed above (see section 3), the detailed analysis of the relationships between the size and the threshold effects covers, at least partially, this issue

Threshold effects are defined as the ways to introduce **discontinuity** in such a relationship in order to maintain the specific cost below acceptable values Such threshold effects can correspond to the "**end point of a given technology**" [1] Generally speaking, they practically introduce the **possibility to change the way to achieve the plant needed functions** (for operation, safety, fuel cycle, etc) The Fig.1 [1] summarise the principle

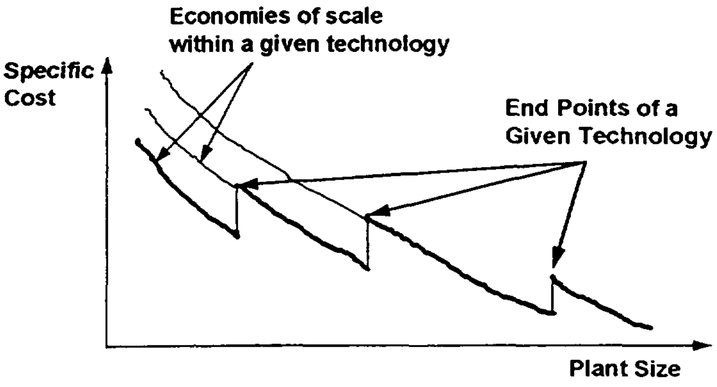


Fig. 1 - Specific Cost as a function of Size for a given point in time [1]

---

[1] The threshold effects correspond to the "**end point of a given technology**" [1] and they practically introduce the possibility to change the **way to achieve** the plant needed functions (operation, safety, fuel cycle, etc , see section 4)

As a **generic activity on Innovative Reactors** (and on **SMR** in particular), research on **threshold effects** is an essential stage in the **assessment of the choices in innovative concepts** This research, as well as the assessment itself, must cover, in an exploratory way, the aspects of operation, safety, economy, fuel cycle, etc CEA/DRN will include such a work within the frame of the activities of the **Innovative R&D Program**

Before starting or, in some cases, continuing this research work, it seems interesting to define a **general outline** which, by systematising the approach, provides a helpful tool to the designer, a tool which is vital for the motivation of the R and D planned or performed on a given design choice

The following sections are the potential starting point for discussions They present a set of proposals which are all to be discussed These proposals are

- the **terminology** to be used
- the **work objectives**
- a **research approach** on threshold effects
- an outline for a **Plant Functional Breakdown** necessary to the application of the approach in question
- a **simplified example**

Once the approach is adopted, practical research work on threshold effects would continue within adequate framework of ad-hoc work groups by large themes (operation, safety, economy, cycle, etc) or by large areas (ALWR, LMFBR, HTGR, etc)

4.2. Definitions, terminology

Product
The "product" is defined as an Innovative design option (component, system, inherent characteristic) implemented in a project (innovative design concept), or proposed individually (ex Steam injector) It is in charge of a given mission or a given task within the frame of a mission ("mission" hereafter) The mission itself is achieved using a given "Principle" (e g passively) For the purposes of this document, the notion of "product" must be generalised to the "principle" that is concretised by the product itself (e.g "Steam injector" for the "passive safety injection")

Project
The "project" is the technical environment or architecture within which the product under examination must achieve the requested mission Once more the "main options" for the design are fixed The assessment of several products will lead to the identification of the different alternative detailed options to achieve, for example, an objective of cost competitiveness optimisation

Context
Set of design, technical-environmental, economic and sociological restrictions (or constraints) which condition the project specifications

* Design restrictions e g implementation of passive systems
* Technical-environmental restrictions e g energy utilisation (electricity production, co-generation, desalination, etc), network problems
* Economic restrictions e g economic competitivity, financing problems, indigenous supply potential
* Sociological restrictions e g reduction/elimination of evacuation plan, adoption of passive systems for safety "transparency"
* Others

Plant Functional Breakdown

The Plant Functional Breakdown (**functional groups, functions** and **sub-functions)** allow to identify, in an exhaustive way, all the objectives which must be met by the project in a defined context as well as the associated constraints

Note: the Plant Functional Breakdown is used to establish the design ***specifications***.

Functional Groups (F.G.)

The Functional Groups are the first level of the Plant functional breakdown They represent a set of complementary study areas for which general objectives can be identified

- operability/reliability
- safety
- economy
- fuel cycle
- market requirements

These functional groups must be identified taking into account the IAEA objectives for Advanced Nuclear Plants ([2], see tab 1) This guarantees the coherence with current requirements for future plants

Functions

The Functional Groups identified above are in turn broken up into **functions** which correspond to complementary and non-redundant study themes of which the objectives have a technical character which is easier to define Going through the functional breakdown two types of functions can be identified those that can be considered as requested performances (e g cooling rate) and others that identify constraints imposed by the context (e g seismic resistance)

Sub-function (from 1 to n)

The breaking up of functions into sub-functions is motivated by the need to define in a more detailed way an objective or a technical criterion until the identification of technical requirements useful for the designer The order 1 to n represents the degree of detail

Criterion

Considering the product characteristics and performances, the **criterion** is a variable (physical, economical, etc) that identify, versus a given function/subfunction, the term of comparison of which the respect is the necessary condition for the allowable achievement of the mission which the product must achieve The criterion (A) can be a correlation between several variables $A = f(a,b,c, \;)$, where a,b,c are single physical variables

The criterion is defined through its **qualitative nature** (e g the integrity of a structure) one or more **allowable limits** (e g temperature and pressure) and through its **flexibility**

Allowable limit

This notion quantifies the criterion identifying the simple or double ended range which border the operability and the acceptability of a product, in a defined context, and for a given project Such thresholds can correspond to an inherent product characteristic (e g mechanical loop resistance maximum operability range) or be induced by the environment (e g maximum size for factory construction or transport maximum acceptability range)

Flexibility

The notion of flexibility is needed to take into account the possibility for different degrees of acceptability For example, within the frame of safety analysis, the acceptable value or range, for a given criterion, depends on the category in which the sequence (or the event), that asks for the missions, is classified The allowable range is larger for accidental operating conditions

Thresholds effects

Threshold effects (**end point of a given technology** [1]) are discontinuities, induced by the limits described above, on the relationships that describe, for example, the specific cost versus

Tab. 1 - IAEA OBJECTIVES FOR ADVANCED NUCLEAR PLANTS AND GUIDELINES TO ACHIEVE THEM

Objectives related to improving Reliability
- Improving inspectability and maintenability
- Improving provision for repair and replacement
- Gaining greater simplicity
- Attaining standardisation
- Using new technologies only after adequate testing
- Improving availability/capacity factors

Objectives related to enhancing Safety
- Assuring stability of the reactor core
- Assuring the removal of residual heat
- Taking advantage of inherent safety characteristics, utilising passive safety systems
- Improving man machine interface
- Reducing on-site impacts
- Reducing off-site impacts of normal operations
- Reducing off site impacts of accidents
- Reducing impacts of external events and internal intervention

Objectives related to gaining better Economics
- Attaining long plant lifetime
- Assuring design stability
- Assuring regulatory licensing stability
- Assuring construction schedules
- Minimising operation and maintenance costs
- Minimising decommissioning costs
- Providing enhanced investment protection

Objectives related to assuring Fuel Cycle
- Assuring fuel qualification
- Providing fuel cycle flexibility
- Providing adequate spent fuel storage

Objectives related to expanding the Market for Nuclear Power
- Expanding the range of plant output
- Investigating indigenous supply
- Assuring infrastructure readiness
- Expanding technology transfer

---

the plant size (see fig.1). Analogous "thresholds" could be identified on the relationships that represent the specific cost versus the safety margins as well as for those that show the safety margins versus the plant size. So these threshold effects can be of different natures: **physical, economic, others**.

Generally speaking, these "thresholds" correspond to the boundaries of the **acceptable domain for the product implementation**. This domain is identified through the comparison of several criteria (intersection of several acceptable areas which are defined separately). Within this acceptable domain the requested function can be achieved by the product under examination At the same time, seeing the meaning of these "**end point of a given technology**" differently, they represent, for a given project the possibility for **changing the way to achieve the function under examination using an innovative - and perhaps less expensive - product**

4.3. - Objectives of the Threshold Effects activities

In a defined context and for a given project, the general objective is the assessment of the potential interest and the possibilities for incorporating a given design option (products)

The first step of the activities is to set up an approach which will guide the research and the identification of possible design option **threshold effects**

Nota· Research on the inherent limits of the product is not a part of the objectives. These limits are identified by the designer (acceptable area of use).

4.4 - The approach - Generics

For a given product (design option) it can be considered that the domain for its allowable implementation results from the comparison between the **requested mission (performances)** and the **constraints** linked to its construction, and or its implementation, and/or its operation-maintenance-repair and/or its dismantling Generally speaking, it can be considered that the performances are the specifications requested for the good behaviour of the plant, and the constraints are imposed, by the context, to ensure that the good behaviour is achievable

Considering the mission, it can be considered that the requested performances - in a first approach considered as directly linked to the "product size" (PS) - depend on the Reactor Size (RS) and are described with a generic function like PS ≥ PSmin(RS) On the other hand the constraints induce a relationship like PS ≤ PSmax(RS)

As pointed out before, several parameters, identified with the generic term of "context" (see § 2), can affect these correlations inducing discontinuities Such discontinuities can result from threshold effects between the parameters that characterise the product (for example the transport costs and the product size) So, real figures will show complex relationships

The figure 2 summarises the principle.

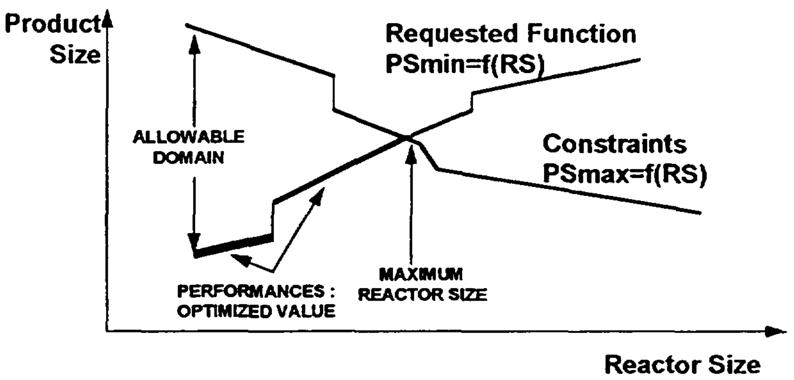
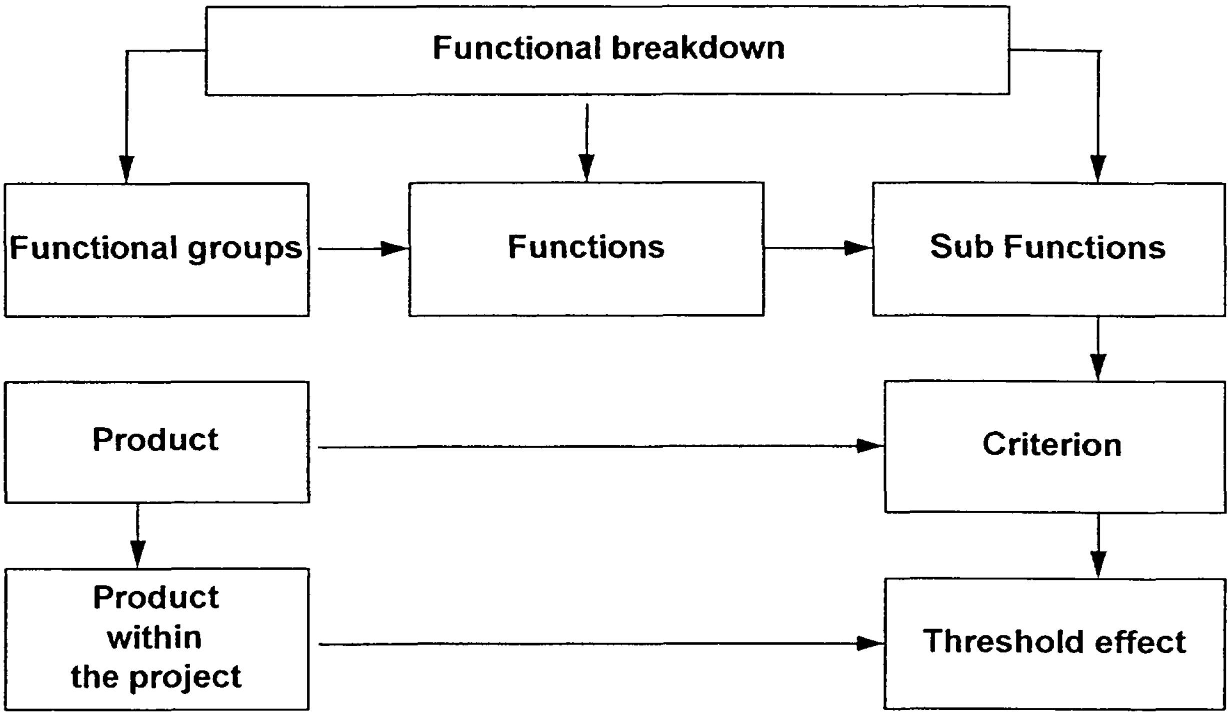


Fig. 2 - Comparison between the product performances and the product constraints versus the reactor size

On this figure the maximum reactor size coherent with the implementation of a given design option is showed

The challenge that the searched approach must cope with, is the systematic take into account of the correlations between the parameters that affect the implementation of a design option within a given reactor plant.

The approach aimed at is summarised in the graph below.



Starting from the general functional groups (standard and applicable to different plant types, see table 1 or analogous) the functions and sub-functions are identified and classified in order to establish a complete, ad-hoc, functional grid for the project. The degree of detail of the breakdown (n order) allows, for each of the sub-function and by comparison with the performances of the product, to identify the nature of the criterion(a). As pointed out in section 2, this criterion is a variable of which the respect is the necessary condition for the allowable achievement of the function. The corresponding allowable limit(s) can be defined as well as the criterion(a) flexibility.

Versus a given sub-function, a series of parametric studies leads to evaluate the product implementation repercussions in the project. The availability of the criterion(a) allows to determine the limits for the implementation. Possible discontinuities (see fig 2) can also be identified.

The systematic comparison versus all the sub-functions leads to identify the overall **threshold effects** and thus to identify the **design sub-function** which corresponds to that which is the most restrictive.

4.5. Functional grid for a PWR (Tentative)

4.5.1 ▫ Functional groups

Below only a tentative functional breakdown is showed for the **operational aspects**. Three main functional groups can be considered to develop the functional grid:

▫ To **Produce energy safely**,

▫ To **Realise the maintenance functions**,

▫ To **Integrate the plant within its environment**

Other aspects, like the economy, fuel cycle, or the coherency with the nuclear market, must be analysed to obtain the corresponding functional lists (see table 2 for tentative guidelines).

The functional grid for the **operational aspects** can be detailed as follow.

▫ To **Produce energy safely**
- Assuring Stability of the Reactor Core
  - ⇨ Knowing the core reactivity
  - ⇨ Implementing the means to change the core reactivity
  - ⇨ Managing the interactions between the reactivity and the other core physical characteristics (e g temperature)
- Removing the Produced Energy during normal operation
  - ⇨ Removing the core power (behind the first barrier)
  - ⇨ Transferring the energy to the secondary circuit (behind the second barrier)
  - ⇨ Distributing the energy to the different users (e g turbines, turbopumps, etc)
- Assuring the Decay Heat Removal
  - ⇨ Removing the heat from the core
  - ⇨ Removing the heat from the primary circuit (second barrier)
  - ⇨ Removing the heat from the confinement (third barrier)
- Assuring the radiological protection and the Radioactive Products Confinement
  - ⇨ Maintaining the fuel element integrity (first barrier integrity)
  - ⇨ Maintaining the primary circuit integrity (second barrier integrity)
  - ⇨ Maintaining the confinement integrity (third barrier integrity)
- Assuring the Reactor Operability
  - ⇨ Following the load specifications
  - ⇨ Monitoring the plant status
  - ⇨ Supervising the monitoring system status
  - ⇨ Reacting to the abnormal situations
  - ⇨ Storing the data concerning the plant operation and maintenance
  - ⇨ Storing the data concerning the plant ageing

▫ To **Realise the maintenance functions**
- Following the ageing and the fatigue of the reactor materials
- Assuring the fuel load and unload
- Assuring inspections and tests
- Assuring diagnostics and repairs
- Allowing the replacement of the consumable materials
- Allowing the controlled waste release
- Allowing the decommissioning

▫ To **Integrate the plant within its environment**
- Taking into account the normal boundary conditions on the design
- Taking into account the potential abnormal boundary conditions on the design (internal and external hazards)
- Using the support functions

4.6 - Practical approach (use of functional grid)

The study of a new product must systematically start by its analysis related to the overall sub-functions defined in the functional grid.

TAB. 2 - DRAFT TENTATIVE FOR THE FUNCTIONAL DEVELOPMENT CONCERNING THE ECONOMY, THE FUEL CYCLE AND THE NUCLEAR MARKET.

➔ IMPROVE ECONOMY

★ **Minimise plant investment**
- ⇨ **Control design**
- ⇨ ***Control certification***
- ⇨ **Control construction**
  - Simplify the material construction
  - Simplify the material qualification
  - Simplify the material transport
  - Simplify the material implementation
- ⇨ **Improve investment protection (Improve the three first Defence in Depth Levels)**
  - Improve safety at prevention level
  - Improve safety at control level
  - Improve safety at protection level
  - Integrate the principles of the defence in depth balanced, gradual and extended defence
- ⇨ ***Minimise decommissioning costs***
  - Improve the quality of the information
  - Minimise the materials activation (Minimise the contact dose)
  - Simplify and automates the procedures for the plant decommissioning conditions
  - Minimise the maintenance times for dismantling
  - Minimise radioactive waste due to the dismantling operations
  - Minimise the quantity of radioactive waste for permanent storage
- ⇨ **Optimise resource sharing (modularity)**

★ **Improve plant availability**
- ⇨ **Simplify every day operations**
- ⇨ **Simplify maintenance**
- ⇨ **Optimise modularity (partial plant availability)**
- ⇨ **Improve man-machine interface**

★ **Improve plant reliability**
- ⇨ **Improve the In Service Inspection and Repair (ISIR)**
- ⇨ **Simplify architecture**
- ⇨ **Standardise components**
- ⇨ **Facilitate retrieval**

➔ IMPROVE FUEL CYCLE (subjects to be developed)

Qualification
Flexibility
Reprocessing
Storage
Proliferation

➔ ADAPT TO NUCLEAR MARKET (subjects to be developed)

Level of power
Unit power
Modularity
Operator participation
Infrastructures
Transfer of technology

A first step allows to identify in a qualitative way (engineer meaning) the functions and the sub-functions which interact with the product

Using adequate R and D, the second step quantifies the correlations between the product characteristics/potential and the requested functions The global analysis of these correlations leads to identify those which are essential for the design and that show threshold effects

Nota - The example below ıs ıntentıonally lımıted to the examınatıon of two sıngle functıons Exhaustıvıty can be reached by the systematıc analysıs of all the functıons of the grıd that ınteract wıth the desıgn of the product under examınatıon

4.7. Simplified example

4.7.1 Generics

Product **Passive Containment Cooling System with a metallic containment (AP600 type: metal vessel cooled on the outside by a flow of gravity water with external concrete building)**

Project - **PWR**

Functional group ▫ To **Produce energy safely**

Functions ⇨ **Removing heat from the confinement (third barrier)**
⇨ **Maintaining confinement integrity (third barrier integrity)**

Sub-functıons

⇨ To Remove heat from the confinement (third barrier)

SF 1 - Transferring heat to outside

SF 2 - Maintaining cold source on the outside for 72 h (water in pool located on the external concrete building: Passive Containment Cooling System - PCCS- in AP600).

⇨ To Maintain confinement integrity (third barrier integrity)

SF 3 - Resisting to internal loads

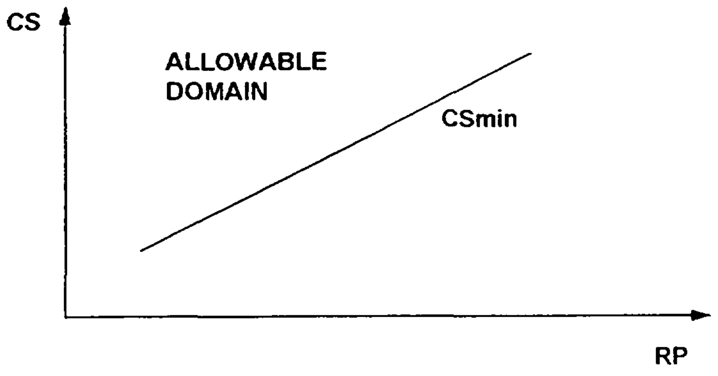
SF 4 - Maintaining a non-deformed geometry (resistance to buckling to ensure good and uniform heat exchange with PCCS water)

4.7.2 Analysis by sub-functions : Identification of correlations between design variables

SF1 - To Transfer heat to outside

This sub-function leads to a correlation which links the metallic Containment Surface (CS) (and thus its volume - CV - to be optimised by the choice of adequate forms) to the reactor residual power (Decay Heat - DH) and thus to its nominal Reactor Power (RP)

$$CS \geq CS_{min} = f(RP) \quad (1$$



SF2 - To **Maintain cold source on the outside for 72 h**

The sub function must be broken up into three functions of second order

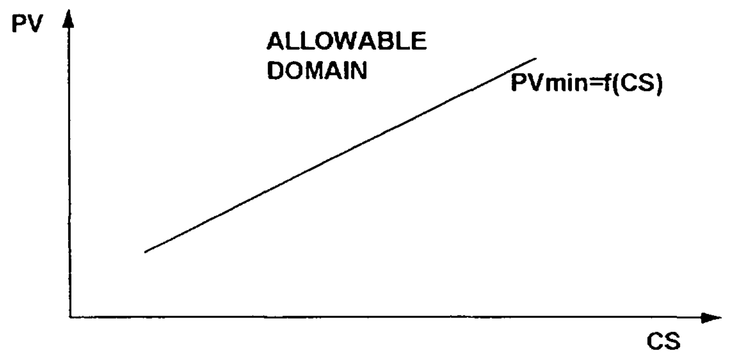
SF2 1 - Ensuring the availability of necessary water (volume of PCCS pools installed on the external concrete building)

SF2 2 - Ensuring the resistance of PCCS pools to external aggressions

SF2 3 - Ensuring the air flow rate between the metal vessel (internal) and the external concrete building

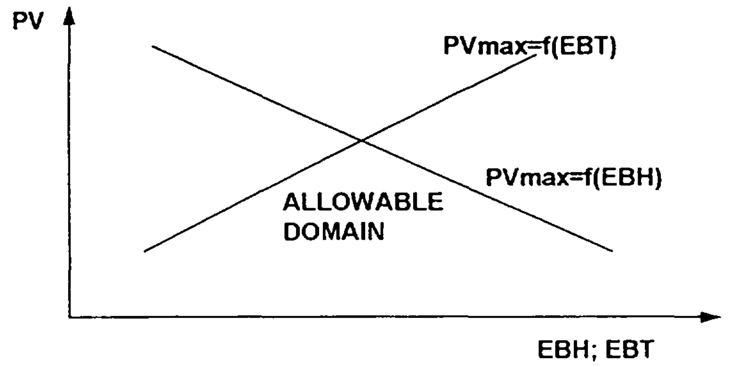
SF2 1 (Ensuring the availability of necessary water) induces a correlation between the PCCS pools Volume (PV) and the Containment Surface (CS) (m^3 water/m^2 surface) (so indirectly with the reactor nominal power (RP))

$$PV \geq PV_{min} = f(CS) \qquad (2\ 1$$



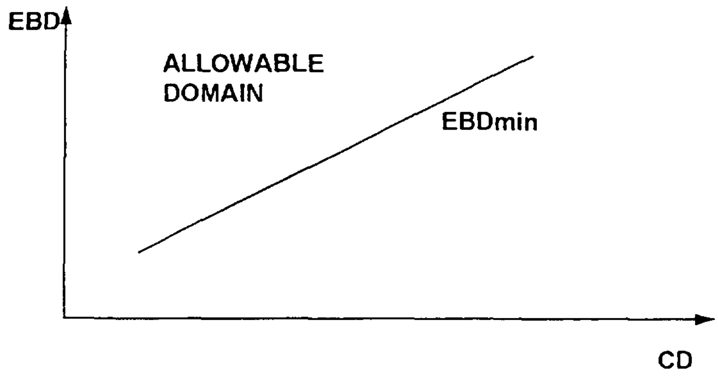
SF2 2 (Ensuring the resistance of PCCS pools to external aggressions) induces correlations between the PCCS pools Volume (PV) and the External concrete Building Height (the elevation - EBH) as well as its Thickness (EBT) These two last variable can be considered as representative of the External Building Size (EBS).

$$PV \leq PV_{max} = f(EBH, EBT) \qquad (2\ 2$$



SF2 3 (Ensuring the air flow rate) induces a correlation between the External concrete Building Diameter (EBD) and the metallic Containment Diameter (CD)

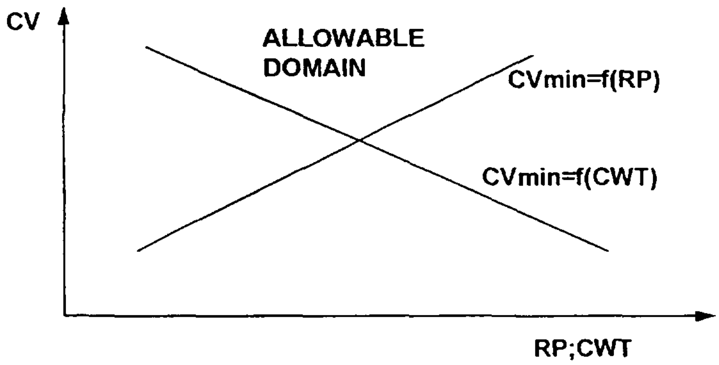
$$EBD \geq EBD_{min} = f(CD) \quad (2.3$$



SF3- To **Resist to internal loads (ex. Loads of Design Basis LOCA)**

SF3 gives a correlation which links the Containment Volume (CV), the metallic Containment Wall Thickness (CWT) and the characteristics of the primary (water volume, size of pipes, etc.: LOCA consequences) and thus generically, for a given specific power, to the Reactor Power (RP).

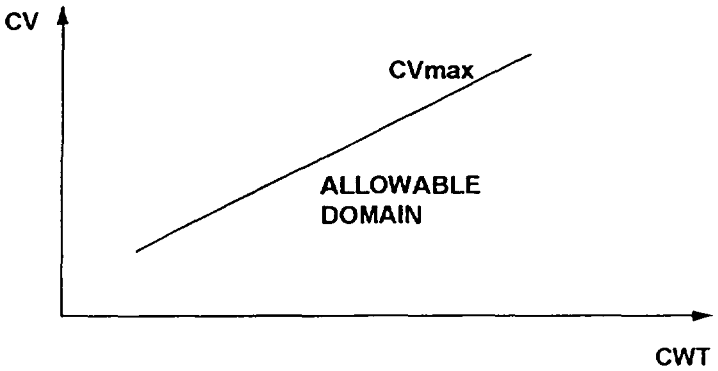
$$CV \geq CV_{min} = f(CWT;RP) \quad (3$$



SF 4 - To **Maintain a non-deformed geometry (resistance to buckling)**

SF4 Induces, for a given form, a correlation between volume of containment (CV) and thickness of the containment wall (CWT).

$$CV \leq CV_{max} = f(CWT) \quad (4$$



Nota - The possible implementation of strengthening structures could easily allows to modify this correlation.

Determination of threshold effect

In the case of a metal vessel, a threshold effect is induced by, on the one hand, the missions which it must face and on the other hand the stresses it must withstand. This comes down to identify an acceptable area which is, in reality, limited.

In order to do this, one must be able to compare two correlations which use the same variables

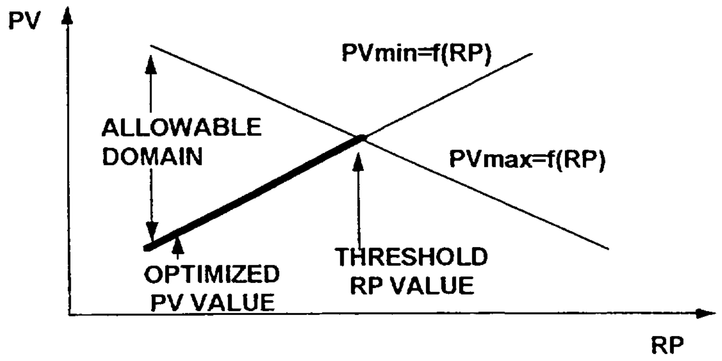
Thus, the correlations (2.1 can, by integrating (1*, be put in the form

$$PV \geq PV_{min} = f(RP) \qquad (5$$

As well (2.2, by integrating (2.3 and (1*, is reduced to the form

$$PV \leq PV_{max} = f(RP) \qquad (6$$

The direct comparison between (5 and (6 identifies the researched threshold effect.



* The compatibility with correlations (3 and (4 must be verified simultaneously and leads to restrictions on thickness

Analogous correlations must be identified considering other key variables like the containment volume versus the reactor power, or the external building dimensions, still versus the reactor power.

4.8. - CONCLUSIONS

Thought is needed to motivate the R&D effort on **Small and Medium Reactor (SMR)**.

The CEA/DRN/DER generic goal is the achievement of an **exhaustive identification and assessment of the problems that are specific for the SMR**.

The technical objective is the achievement of a technical-economical study which leads to the setting up of a motivated set of plant specifications for an SMR. The details of the activities are actually **under discussion** at the CEA/DRN.

Among the **generic activities on Innovative Reactors**, the research on **threshold effects** is an essential stage in the **assessment of the choices in innovative concepts**. This research, as well as the assessment itself, must cover, in an exploratory way, the aspects of operation, safety, economy, fuel cycle, nuclear market. This will leads to have a systematic complement to the **system behaviour assessments**, that today monopolise the effort.

CEA/DRN will include such a work within the frame of the activities of the **Innovative R&D Program**.

The first objective is the definition of a **general outline** which, by systematising the approach, will provides a helpful tool to the designer

The document is a very preliminary proposal which objective is to be the starting point for the discussion The basis for the approach must be discussed to clearly identify its **potential and the difficulties for its practical application**

The development of agreed functional grids either for operational, safety aspects as well as for economic and fuel cycle items, must be the first task

Examples must be developed and quantified to show the potential and to evaluate the real difficulties

REFERENCES

[1] BITTERMANN,
Status of Development Work on Small and Medium Sized Reactors at Siemens KWU presented at the IAEA-AGM meeting in Rabat (Morocco), on October 23-27 1995

[2] INTERNATIONAL ATOMIC ENERGY AGENCY,
Objectives for the development of Advanced Nuclear Plants, IAEA TECDOC - 682, Vienna January 93

NEXT PAGE(S) left BLANK

STATUS OF DEVELOPMENT WORK ON SMALL AND MEDIUM SIZED REACTORS AT SIEMENS/KWU

XA9846703

D. BITTERMANN
Siemens AG/KWU,
Erlangen, Germany

Abstract

Basic principles of SMR development are discussed and relevant design features of a 200 MW(th) heating reactor and of a 600 MW(e) BWR are presented.

1. Introduction

There are two major development tasks to be achieved which are the prerequisites of the success of any development of Small and Medium sized Reactors (SMR). They are low capital cost, and a high safety level. Both tasks are to be seen as equivalent in importance yet both are different in the difficulty to be achieved. Since the safety level of the plants is already high and a higher safety level is easier to achieve the smaller the plant, the goal to achieve acceptable capital cost for a small plant is a much bigger challenge. It is the aim of this paper to discuss some principles, how one can meet these challenges. In addition the current status of the design studies is shortly described.

2. Approach to low heat and/or electricity generating costs for small units

In order to reach competitive heat and/or electricity generating costs the most important task is to concentrate as much as possible on the question how to reach low capital costs. Fuel costs are of second importance only. In addition an approach how to reduce the plant erection time and how to guarantee high reliability and availability of the plant should be considered.

As we know from the scaling laws and as it is illustrated in Fig. 1 qualitatively, one would move along the the upper branch of the capital cost curve without changing of the concept of a plant [1]. With this strategy it is impossible to reach the cost target if one considers smaller units. One only can break through this laws if one would look for advantages which result from the fact that the individual design features of most components and systems have technological discontinuities even though the basic concept remains unchanged regardless of size and duty. For this phenomenon the turbine is a simple example. At a certain size one switches from full speed to half speed designs. The half speed design is physically larger and thus more cost expensive. At the point of cross over, the specific cost of the half -speed design is higher than that of ist full-speed counterpart. Only on account of the increase in output now possible the specific cost can be lowered to a new minimum.

Another example is the use of passive systems or components which are much better to achieve at low power and small sizes. Such behavior is displayed by many technical components and systems. When designing a small nuclear plant, it is for precisely such discontinuities that one must look to achieve a jump to a lower specific cost level. This

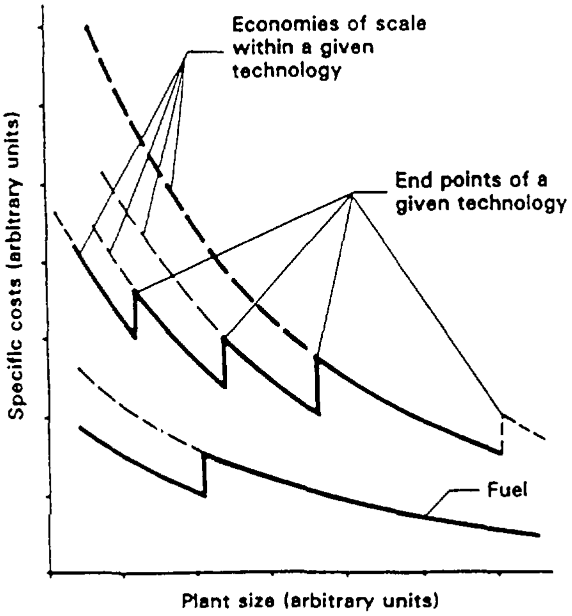


FIG. 1. Specific cost as a function of size for a given point in time.

new lower level represents a technological solution that is only possible with the smaller size or duty of the unit in question.
A second consideration would require the following: in attempting to cut costs, it is better to eliminate certain components and systems altogether, even if those remaining become larger, than to attempt to make everything a little less expensive. The marginal cost increase associated with the remaining equipment is smaller than the savings represented by what is eliminated.

As a third design rule we propose to use the experience gained with components, materials and equipment as much as possible also if one designs new components. This does not mean that one only should use existing designs, but one should carefully decide whether one leaves proven designs and equipment especially such where specific nuclear technology experience has be accumulated like fuel, fuel elements, materials or design and safety principles. If one considers this, high availability of new plant designs can be expected with the benefit to be able to contribute to the goal to reach competitiveness in cost. In case that developing countries want to introduce SMR's which have not already gained this experience, they can participate of this experience by cooperation and technology transfer with the corresponding companies and institutions which have this experience.

For SMR plants, the components and equipment are expected to be physically smaller compared to larger plants. Therefore the possibility of prefabrication of systems and prearrangement can be considered in a larger extend compared to the big plants. This could considerably reduce the erection cost of the entire plant.

In case that from a country like Germany SMR's will be exported an appropriate portion of national supply and services should be considered in order to be able to reduce the cost. In such a case one should optimize the import and national portion together with the cost for technology transfer.

Another and very fundamental general design consideration becomes evident when looking at the fuel cost component in the unit generating cost versus plant size relationship like it is shown in Fig. 1. At small unit sizes, this part of the generating cost is even less important than with a standard size nuclear power plant which is in contrast to fossil fired stations. Nuclear fuel assemblies exhibit all the characteristics of high-technology mass-produced goods. Their scaling relationship tends to be much flatter than that for the custom-tailored plant proper. This fact should have for the designer the consequence that one should not try to minimize the fuel cycle cost as a single design goal. One should rather look how one can reduce the capital cost even if this specific design feature increases the fuel cost.

3. Approach to a high safety level

It is one of the basic requirements of SMR's that this kind of plants from the size of view are fitted to specific applications which normally are also very closely situated to densely populated areas. Therefore the necessity exists to increase the degree of safety level for such kind of plants. This objective is also in line with the ongoing development of large power reactors.
In case of SMR's there are two design features which can be introduced and which are the easier to be achieved the smaller the plant is. One is to use large water volumes in order to increase the grace periods in case of malfunctions. The other one is to use passive systems and components The reason for this is the fact that natural circulation which is one of the important phenomena that must come into force is much easier achievable for smaller sizes and consequently smaller decay heat ratios. In line with the longer grace periods and the higher water volume normally the core power density is also reduced. This again enables longer reaction times in case of malfunctions and consecutively simpler designs for the actuation of safety systems with very high reliability to be expected.

4. Characteristic features and application potential of a nuclear heating reactor

Under consideration of the above described design principles the following design features of a heating reactor with an power of 200 MWth were fixed (Fig. 2):

- the primary system is fully integrated for compactness and for greatly reducing the requirements for engineering safeguards. Boiling is permitted for self-pressurization and enhancement of natural circulation. Hydraulic driven control rods are located inside the reactor pressure vessel (RPV)

- the reactor has an intermediate circuit to keep the distribution grid free of radioactivity

- natural circulation has been selected for the primary circuit and for those parts of the secondary circuit which are needed for decay heat removal

- the reactor containment fits tightly around the RPV so that any postulated coolant leak is limited in such a way that the core cannot become uncovered

- the primary system has a very high thermal heat capacity, eliminating most of the usual engineered safeguard systems

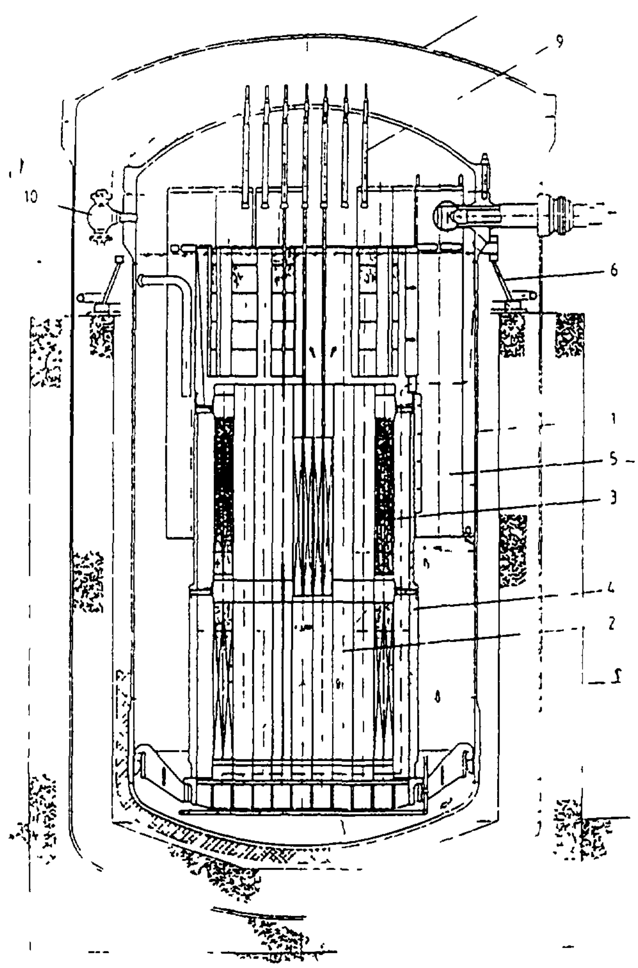


FIG 2. Nuclear heating reactor. Primary circuit.

- the reactor requires only two cores for the entire lifetime of the plant. This reduces significantly the requirements for fuel handling and storage, which in turn leads to a very compact small reactor building

The main data are summarized in Table 1.

The application for which the concept was originally designed has been seen mainly in district heating (Fig. 3). But there is also a potential to use it as a heat source for desalination plants. For this puropose also rough analyses have been performed. A flow scheme and different concepts for the design of a desalination plant [3] have been analysed (Fig. 4) considering f. i. the multi stage flash process (MSF).

Outstanding safety features

The large water volume and the relatively wide subcooling margin result in a high heat capacity of the primary system. Since the primary heat sink of the reactor is in the immediate vicinity of the core, the heat transport mechanism, by means of natural circulation, is maintained during all typs of events, including loss of coolant accidents (LOCAs). The relatively small amount of decay heat can easily be transferred via the intermediate circuit and aircoolers to the environment by natural circulation. The high

TABLE 1. SELECTED DATA OF THE NHR THERMOHYDRAULIC AND NUCLEAR PHYSICS DESIGN

Pressure	(bar)	15
Thermal power	(MW)	200
Core mass flow	(kg/s)	1030
Number of fuel assemblies		180
Active core height	(mm)	2350
Core subcooling	(K)	40
Average volumetric steam content at core outlet	(%)	26
Maximum linear power rate	(W/cm)	75
Average power density	(kW/l)	20
Average specific power	(kW/kg)	10
Number of control rods		45
Control rod type		BWR

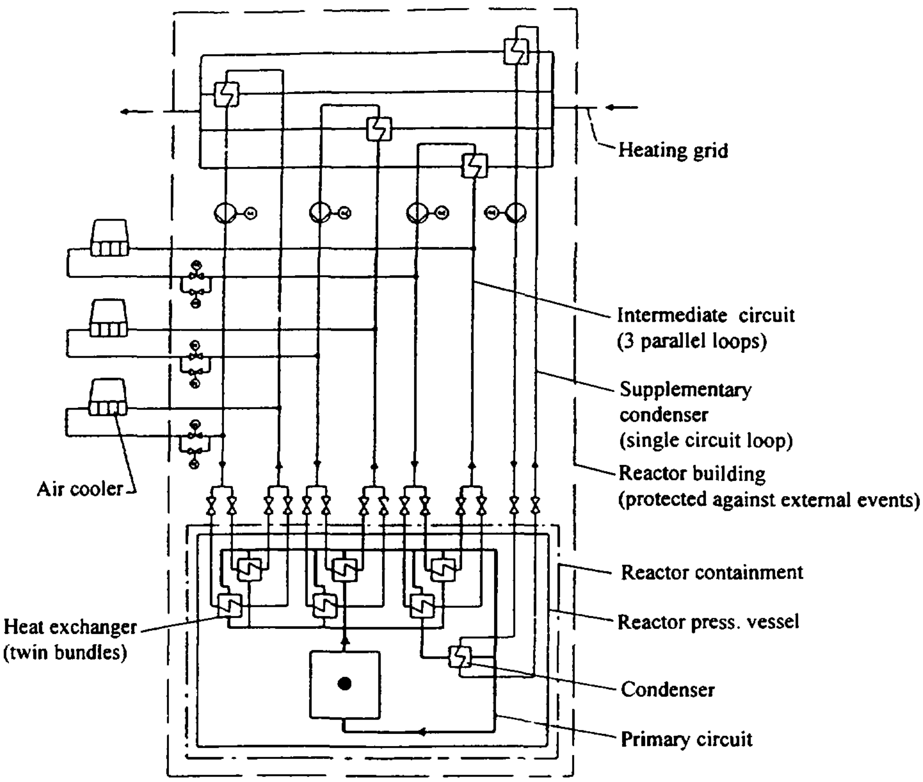
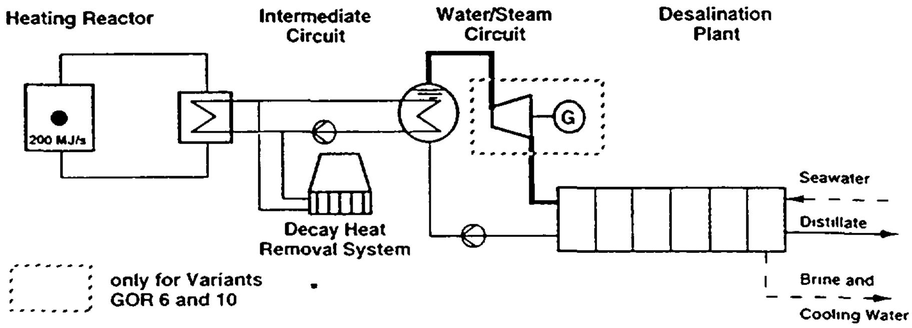


FIG. 3. Nuclear heating reactor: Flow scheme.

fission product retention of the low-rated fuel, together with the low corrosion rates, leads to low primary coolant radioactivity content.

For reactor shutdown two independent and diverse systems are installed. Each of the systems is able to bring the reactor into a cold subcritical condition. The control rods are used as the first and principal shutdown system. This hydraulically driven system is fail safe in a way that after loss of electric power and postulated breaks in its piping



Performance Ratio	Distillate Production m^3/d	Plant Auxiliary Power: Desalination Plant MW	Plant MW	total MW	Electricity Production MW	Electricity Balance MW
GOR 6	38440	5,9	1,6	7,2	23,5	+ 16,3
GOR 10	67500	9,8	1,6	11,4	16,3	+ 4,9
GOR 16	121846	17,8	1,6	19,4	0	- 19,4
GOR 20	148400	23,1	1,6	24,7	0	- 24,7

FIG. 4. Seawater desalination with heating reactor (Q=200 MJ/s).

system, the rods will fall into the reactor core by gravity. A regular shutdown initiated by the reactor protection system is performed by interrupting the electric power supply to the running pump and by actuating the insertion valve. Each of these measures is in itself sufficient to cause the rods to fall into the core. As a second shutdown capability, a boron injection system is installed, which is designed to shut down the reactor after anticipated transients without scram ATWS).
In case of applications for district heating, the main heat sink is normally available after reactor shutdown. This is especially true for plant internal events, including LOCAs. However, it does not apply to events leading to failure of the heat transport capacity of the heating grid, such as loss of electric power or external events. For those cases in which the main heat sink is not available, a separate decay heat removal system is provided. As already described, this system is connected to the intermediate circuit and works by means of natural circulation. During normal operation the system can operate in bypass to the intermediate circuits in order to prevent freezing of the outside heat exchangers. On demand, the system will start up after one of the two parallel valves installed in each train is opened.

5. Development status of the NHR

The NHR concept was originally developed for the application in Germany in the first half of the 80th. The status of development reached until the end of the decade can be characterized as Basic Design quality for the essential components of the primary circuit, the nuclear physics and thermohydraulic design and the safety analyses. For the hydraulic control rod drive which is a complete new design a large test program was performed. A drive of original size and material has performed about 500000 steps under operating pressure and temperature and about 2000 scrams were sucessfully performed. During these tests, ultrasonic position indicators have proven their function ,too. An expertise performed by TÜV Bayern used the criteria given in the appropriate nuclear standard (KTA 3103) had the result that no aspects were found, which indicate malfunctions of the hydraulic control rod drive or a loss of the shutdown margin.

Prefeasibility studies for some German cities have been performed which showed the potential of econonic feasibility. But no order could be placed in Germany up to now and cannot be expected within the next future.The reasons are as follows:

1. Within about 10 years from the first idea of a NHR all fossil fired power plants had to implement expensive backfitting measures caused by new emission standards. At the time when the decisions had to be made, the NHR was not at a design status able to be discussed as a serious competitor.
2. The German NHR development was also negatively affected by the slowdown of the nuclear business in general. This made potential customers hesitate to engage themselves in a new and controversary technology. This in particular because many district heating facilities are owned by communal utilities which thus far have very little, or no nuclear experience,and which are much more exposed than private utilities to the split that runs through the big political parties on the application of nuclear power in Germany.

In the framework of the German-Chinese cooperation in the field of nuclear technology which was envisaged at that time, prefeasibility studies together with the Institute of Nuclear Technology of the Tsinghua University in Beijing were performed. In 1991 a feasibility study with different Czechoslovakian partners especially Skoda, Pilzen was performed for an NHR application for the city of Pilzen. Up to now no final decision has been made on this project.

6. Characteristic design features of a small boiling water reactor

In cooperation with the German utilities a boiling water reactor concept is under development since the beginning of the 90th which mainly uses the principles outlined above. Special emphasis is put on an improved safety concept, the function and availability of which shall be assured by simple and unsensitive safety features. The concept is based on the extensive range of experience which has been gained from boiling water reactor (BWR) plants currently in service. It makes use of systems and component designs which have proven their reliability in operation. Certain systems will be simplified on the basis of proven operating experience. The new safety concept which supplements the active safety systems implemented to date, is characterized by the following four main accident control features:

1. A higher degree of safety is achieved through the introduction of passive systems for accident prevention and control
- these systems function according to basic laws of physics, such as gravity and heat transfer, without need of operator intervention or power supply.

2. Good plant behavior is ensured in the event of transients or accidents.
- this is achieved by means of a lower core power density, a large coolant inventory in the RPV and additional water inventories stored inside and outside the containment. Core cooling is thereby ensured for several days without any need for external intervention

3. Accident control requires no active intervention
- in the event of an accident, the plant can be left to manage itself. Only after several

days, external intervention in the form of simple actions becomes necessary. Human error under accident conditions has no negative effect on plant behavior.

4. Compared to existing nuclear power plants, core melt probabilities are reduced even further
- this is the result of combining the operation of active non-safety-related systems and passive safety systems. Despite this high degree of safety, features for core melt accident control are provided so that evacuation of the population within the immediate vicinity of the plant would not be necessary.

In Fig. 5 a view of the primary circuit and the containment is depicted. Table 2 shows some major plant data.[3]

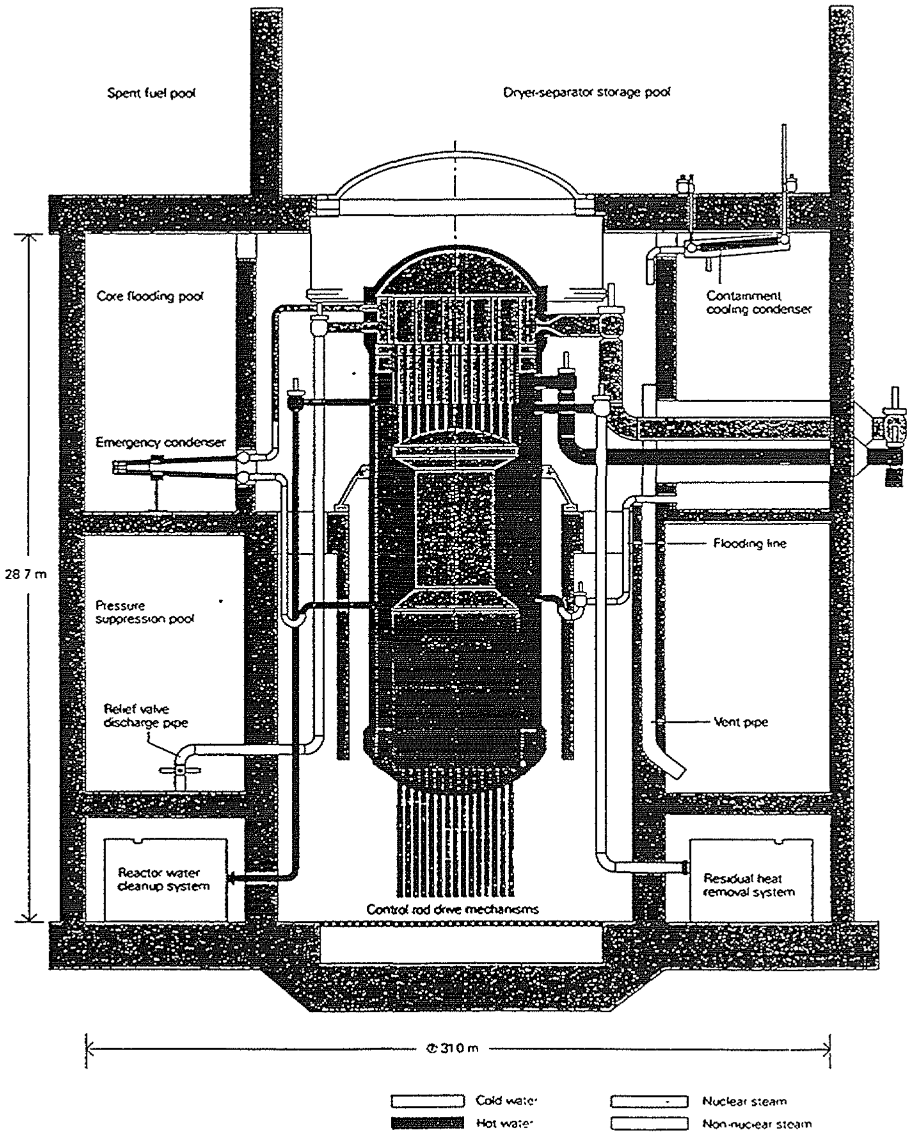


FIG. 5. Containment with internal structures.

TABLE 2. SWR 600 - MAJOR PLANT DATA

Overall Plant		
Thermal output	MW	2200
Net electric output	MW	750
Reactor Core		
No. of fuel assemblies		840
Total uranium weight	t	109
Active core height	m	2.8
Average power density	kW/l	42
Discharge burnup	GWd/t	65
Average enrichment	wt. %	4.8
Coolant flow rate	kg/s	8,000
Reactor Pressure Vessel		
Inside height	m	22,4
Inside diameter	m	6.8
Design pressure	bar	86
Plant Operation		
Spent fuel storage capacity	years	40
Plant design life	years	60

Outstanding passive safety systems

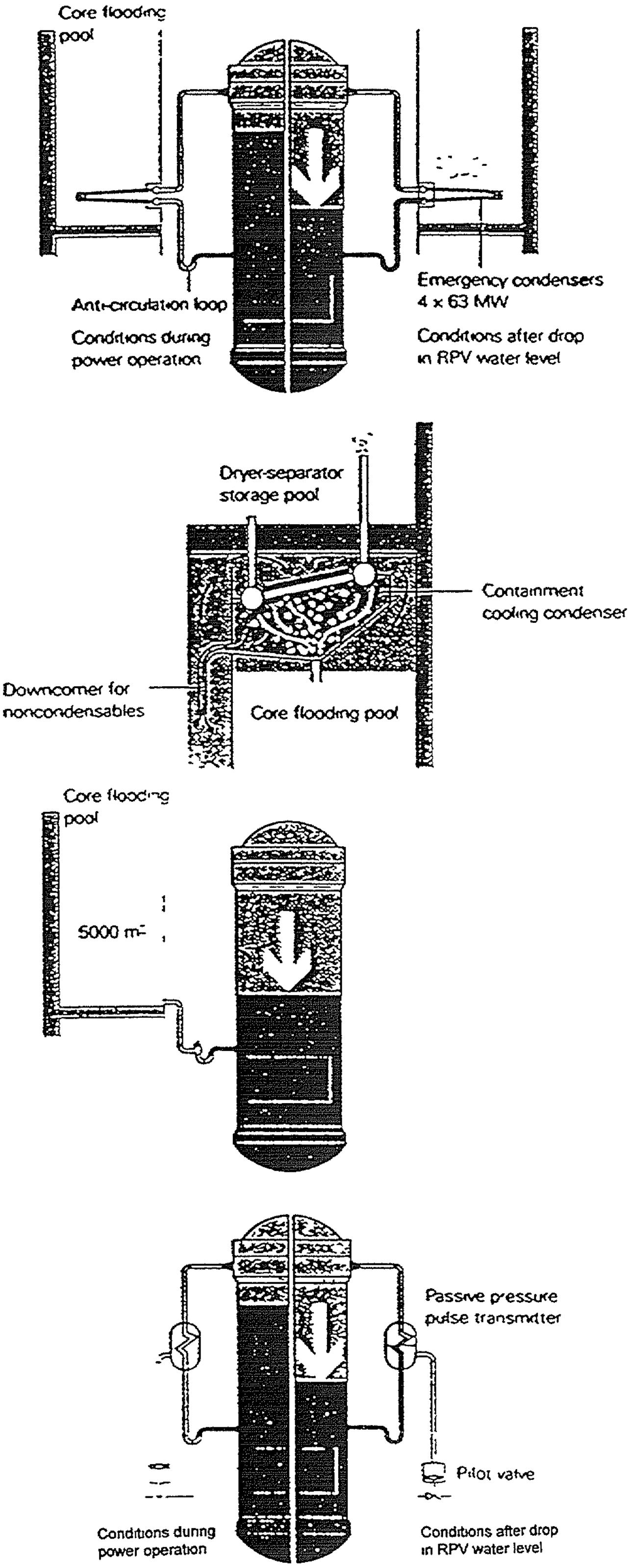
As already mentioned before, the essential task for the development of the safety concept of the BWR 600 is to be seen in the improvement of the plant's safety quality via the introduction of additional passive systems which can take over safety functions during transients and accidents.

Their systems engineering is in contrary to existing reactors much more simplified, their function is independent of electricity supply and control by I&C systems. The characteristic feature of these passive systems, is the use of laws of nature (f.i. gravity) to fulfill their safety functions without use of any active component. One well-known example is the flooding of a depressurized RPV from a water pool using only the static head between the vessels.

The most frequent requirements for the need of flooding and decay heat removal result from transients. After most of the LOCA's the decay heat is transferred via the leak into the containment atmosphere. In order to avoid an uncovered core, measures must be foreseen to inject water or flood the RPV. After most of the transients and other events the following safety related functions have to be performed:

- shutdown of the reactor
- containment isolation
- RPV pressure limitation or depressurization
- decay heat removal from the RPV
- RPV flooding or water injection
- decay heat removal from the containment

The passive systems which can fullfil these functions are briefly described in the following chapters (Fig. 6).



Emergency Condensers

Remove decay heat from reactor as RPV water level drops, requiring neither electric power nor activation by I&C systems

Containment Cooling Condensers

Remove heat after steam formation inside the containment, requiring neither electric power nor activation by I&C systems

Gravity-Driven Core Flooding System

Floods reactor by gravity flow as reactor pressure drops. Shutoff valves self-actuate by deadweight when elevation head of core flooding pool overcomes reactor pressure

Pressure Pulse Transmitter for Switching Operations

Generates pressure passively in a heat-exchanger secondary circuit as reactor water level drops. This pressure is used for actuation of safety systems (reactor scram, RPV depressurization and containment isolation), requiring neither electric power nor I&C signals

FIG. 6. The passive safety system.

Pressure pulse transmitter for switching operations

Safety related functions like reactor scram, RPV depressurization and containment isolation are actuated by a passive device which needs no electricity supply and instrumentation and control equipment. The function is described as follows: the reactor is connected to a small heat exchanger via a pipe which cannot be shut-off. This heat exchanger acts as a passive pulse transmitter. In case of normal water level within the RPV the tube side of the heat exchanger is filled with water and consequently no heat is transferred. If the water level within the RPV decreases and as a consequence also within the heat exchanger tubes, steam will be condensed within the tubes. The heat exchanged to the shell side leads to a pressure increase which acts via pilot valves on the main valves in order to perform the required functions.

Emergency Condensers

One important passive component which esspecially can be used for control of transients are to be seen in the four emergency condensers (4 x 63 MW at 70 bar) which are located within the core flooding pool and which are connected to the RPV via steam and condensate pipes which can not be shut off. The emergency circuit is equipped with a so-called cold siphon which prevents circulation of condensate during normal operation. As consequence of the drop of the water level, steam will condense within the condenser and the condensate is discharged into the RPV again. The heat transferred to the water within the flooding pool will increase its temperature slowly. Even after more than 12 hours the water will reach the evaporation temperature and consequently lead to a pressure increase within the containment.

Gravity - driven core flooding system

After LOCA's, steam and water/steam mixture is discharged into the containment. In order to avoid uncovering of the reactor core, a passive core flooding is actuated. To be able to initiate this type of flooding mechanism a depressurization of the the RPV has to be initiated first. Afterwards a flooding of the RPV with water from the core flooding pool via four pipes can take place. The water content of the flooding pool is about 5000m³. After LOCA this inventory is sufficient to flood the RPV together with the pressure compartment of the containment until an equilibrium level is reached which is above the feedwater lines. This has the consequnce that potential locations of leakages will be flooded and enables the potential of an outside cooling of the RPV, too.

Containment Cooling Condenser

Both after transients and LOCA's the potential of temperature and pressure increase in the late phase exists. In order to limit the pressure and temperature increase specific condensers are provided. The condensate will be discharged into the core flooding pool. The heat sink of the condensers is the water inventory of the dryer-separator storage pool located above the containment. These condensers close the circuit of a passively actuated cooling system for decay heat removal within the containment. The containment condensers enable a heat removal for about seven days without additional water supply for the storage tank outside the containment. This can be performed if necessary with relatively simple measures.

As a result of the above described features and systems this kind of reactor concept will reach stable conditions even after transients and LOCA's independent from the availability of I&C systems and the supply of auxiliary power. Active systems which are installed from operational reasons complete the passive sxstems for accident control.

Provisions to control Severe Accidents
In order to be able to mitigate and control the effects of Severe Accidents which are postulated, different measures are provided. Hydrogen, which is produced from the zirconia - water reaction within the RPV in this hypothetical case, can neither burn nor deflagrate within the containment since it is inerted already during normal operation with nitrogen. The design of the containment against the resulting pressure can cope even with a hydrogen mass according to 100% zirconia water reaction and the resulting pressure within the containment. A containment venting is not considered.
A hypothetical core melt is expected to be stabilized within the RPV by outside cooling.

7. Development Status

For this new reactor concept the conceptual design is finished and the Basic Design is now under way. The development was started for a plant size of about 600 MWe; according to the requirements of the customer (German utilities) the work will be continued for a size of 1000 MWe. In addition it is turned from natural circulation within the RPV to forced circulation with internal pumps. Nevertheless the basic concept will be applicable for the smaller size, too. The Basic Design is considered to be finished at the end of 1999. It includes also a safety report and a technical tender which is seen to be the prerequisites for the decision on a project.
The cost analyses which have been performed during the conceptual work have shown that caused by the simple arrangement and simultaneous reduction of the expenditure for active systems a reduction of the capital costs of the plant can be achieved. Together with an envisaged reduction of the erection time and the higher fuel utilization in connection with a two years fuel cycle the electricity generation cost are expected to be only marginallyhigher compared with a large 1300 MWe BWR. The economic calculations have shown that such a medium sized plant is competitive with a large plant.

8. Conclusion

There is a lot of positive experience available from more than 7000 accumulated reactor years of operation with nuclear power plants. Using this experience and considering the priciples which were explained, one can be confident to be able to reach the goal to design and build economic and safe SMR's. The ongoing development work shows encouraging results.

REFERENCES

[1] Goetzmann, Bittermann, Göbel
Design principles of a simple and safe 200 MWth nuclear district heating plant
Nuclear Technology, Vol. 79, Nov. 1987

[2] Gutachten zum Konzept eines hydraulischen Steuerantriebes für einen 200 MW Heizreaktor
TÜV Bayern 1988

[3] Lezuo
Siemens/KWU internal document

[4] SWR 1000 The Boiling Water Reactor with a New Safety Concept
Siemens Power Generation 1995

NEXT PAGE(S) left BLANK

SOME JAPANESE ACTIVITIES ON SMALL AND MEDIUM NUCLEAR POWER REACTORS

H. SEKIMOTO
Research Laboratory for Nuclear Reactors,
Tokyo Institute of Technology,
Tokyo, Japan

Abstract

Some Japanese research activities on small and medium nuclear reactors are reviewed. The increasing role of nuclear reactors in the future is well understood by the government. The increasing energy consumption by Asia and developing countries is also well understood. The small and medium nuclear reactors may be suited well to these areas. Though several researches of small and medium reactors are under intensive progress on LWR, HTGR, LMR and MSR based reactors, projects for building these reactors have not yet been established.

1. JAPANESE LONG-TERM PROGRAM FOR RESEARCH, DEVELOPMENT AND UTILIZATION OF NUCLEAR ENERGY

At the 39 Regular Session in IAEA General Conference held on September 18, 1995, Japanese Minister of STA gave the statement, whose concluding remarks are as follows: Explosive population growth and rising living standards are likely to raise the world's energy consumption sharply in the 21st century. There is also concern that energy-related environmental problems such as global warming will become more serious. The international community thus urgently needs to resolve energy issues. As a source of energy which enjoys an excellent stability of supply and places a light burden on the environment, nuclear power will increasingly play a bigger role.

Japanese Long-Term Program for Research, Development and Utilization of Nuclear Energy mentions about international cooperation and try to promote cooperation with the neighboring regions of Asia and developing countries[1]. It mentions furthermore that in development and utilization of nuclear energy assurance of safety is an absolute premise, and accidents or even incidents at nuclear facilities in one country make people in other countries, too, uneasy and could even have a negative impact on development and utilization of nuclear energy in different countries. That being the case, in cooperation with the neighboring region of Asia and developing countries it is important that emphasis be placed on assurance of safety as a problem common to all countries.

Small and medium power reactors seem proper to be utilized in these areas since they usually attain much more excellent safety features than large power reactors. However, Japanese Long-Term Program does not mention anything about small power reactors for these areas. In Japan the electric power network system is almost completed and the small and medium power reactors are not required. From these reasons the activity on small and medium power reactors in Japan is very low.

2. INTERNATIONAL SPECIALISTS' MEETING ON POTENTIAL OF SMALL NUCLEAR REACTORS FOR FUTURE CLEAN AND SAFE ENERGY SOURCES

However in 1991, International Specialists' Meeting on Potential of Small Nuclear Reactors for Future Clean and Safe Energy Sources, SR/TIT, was held at Tokyo Institute of Technology[2]. This is the almost only one international meeting ever held in Japan. The contents of this meeting show almost all Japanese activities on small reactors. The Japanese contributed papers in this meeting are shown in Table 1. Water reactors(JAERI), HTGR (Tokai Univ., Fuji, Kawasaki), sodium-cooled fast reactor controlled by graphite reflector (CRIEPI, Toshiba), sodium-cooled and lead-bismuth-cooled long life fast reactors(TIT) and molten-salt reactor(Tokai Univ.) were studied. However, all these activities were in design study stages. All of the works have been proceeded since then to improve their performances, but the situation is almost the same even at present.

SPWR(System-integrated PWR) has been designed at JAERI since 1986 as a next generation power reactor realizing highly passive safety, satisfactory maintainability and economical competitiveness at the same time. It is a new type of integrated pressurized water reactor which installs a poison tank in its reactor pressure vessel for reactor shutdown instead of control rod drive system. This concept is considered to be applicable to wide range power sources from small scale to large scale and is expected to be useful for world wide nuclear reactor utilization in near future.

JAERI has been also carrying out design studies on the advanced marine reactors since 1983 in order to develop attractive one for the next generation. Two reactor concepts are formulated; the one is MRX(Marine Reactor X) for ships navigating on sea surface, the other is DRX(Deep-sea Reactor X) for a power source used in deep sea. They are characterized by an integral type PWR, built-in type control rod drive mechanisms, a water-filled container and a passive decay heat removal system, which realize highly passive safe and compact reactors. MRX was designed as 100 MWt for an ice-breaking scientific observation ship, but it could be applied to wide output range as 50 through 300 MWt according to a variety of required output due to type, size and velocity of ships, and applied also to very safe small power stations on land.

Research Association on High Temperature Gas-Cooled Reactor Plant has been studying the Modular High Temperature Gas-Cooled Reactors. The studies cover both pebble bed core(200Mwt) and block fuel core(350Mwt).

CRIEPI(Central Research Institute of Electric Power Industry) proposes a graphite-reflector controlled small-size sodium-cooled reactor called 4S-LMR to fulfill the future energy demand to encourage the human beings and the global environment. Specific design policy is as follows; The void reactivity and all reactivity temperature coefficients are negative, no refueling for 10 years, simple core burnup control without a control rod and its driving mechanism, safety system independent of emergency power and active decay heat removal systems, elimination of control and adjustment components from the reactor system, load following without operation of reactor control system, minimum maintenance and inspection of reactor components, quality assurance and short construction period based on shop fabrication, no core damage in any conceivable transient events without scram, complete containment of nuclear materials for a long period. It is expected that 4S-LMR would be utilized for some special purposes such as for urban siting for decentralized electric units, and for sea water desalination, although the future work is needed to resolve several key issues.

Conceptual design study for small size long life fast reactor has been performed in TIT(Tokyo Institute of Technology). The reactor was mainly consisted of three parts, inner blanket which was located in the center, peripheral region which was filled with coolant material, and core which was located between the peripheral region and the inner blanket region. Both sodium-cooled and lead-bismuth-cooled fast reactors were studied at their power rating up to 100Mwt. Safety aspect was implemented by reducing excess reactivity during burnup smaller than or equal to 0.1%dk and by minimizing coolant void coefficient. Calculated results showed that 30 year reactor life time without refueling with maximum excess reactivity no more than 0.1%dk could be satisfied by both sodium cooled and lead-bismuth cooled fast reactors. In case that the coolant in the core and blanket regions was completely voided, sodium cooled reactors gave positive value of coolant void coefficient while the lead bismuth cooled reactors up to 75 MWt gave negative value over whole burnup period. However in case that coolant in all regions including reflector region was voided both sodium cooled and lead-bismuth cooled reactors gave strongly negative value of reactivity.

Because of the superior characteristics of lead-bismuth cooled reactor, both lead-bismuth-cooled and lead-cooled reactors have been later investigated intensively. Safety analysis of lead or lead-bismuth cooled small safe long life fast reactor has been performed. The reactor is proposed to be used in relatively isolated area and operated up to the end of life without refueling or fuel shuffling. In the present paper the reactor power and life are set to be 150MWt and 12 years, respectively. In order to assure its safety performance against accidents, the following accidents without scram are simulated with neutronic-thermal-hydraulic analysis: unprotected loss of flow(ULOF), unprotected rod run out transient over power(UTOP), simultaneous ULOF and UTOP accident, and simultaneous ULOF, UTOP and unprotected loss of heat sink(ULOHS) accidents. For each type of accidents, four cases of long-life small reactors (lead-cooled metallic-fueled, lead-cooled nitride-fueled, lead-bismuth-cooled metallic-fueled, and lead-bismuth-cooled nitride-fueled reactors) have been analyzed. It is shown that all the proposed designs can survive these accidents without any help of operator or active devices.

K. Furukawa, et al., proposes the Thorium Molten-Solt Nuclear Energy Synergetics depended on the isolation of fissile-producers and power-stations, allowing the smaller stations. In this system the Flibe base molten-fluoride is applied and fissile-producing facilities are separated from the power stations. For the power stations, several types of small molten-salt power stations are proposed as FUJI series reactor designs(350MwtFUJI-II, 250MwtFUJI-V). For the fissile-producing facilities, accelerator molten-salt breeder, impact fusion molten-salt breeder and inertial-confined fusion hybrid molten-salt breeder are proposed.

TABLE I. Japanese Contributed Papers in International Specialists' Meeting on Potential of Small Nuclear Reactors for Future Clean and Safe Energy Sources[2]

1. General Topics
   1-1) H. Sekimoto (TIT): Several features and applications of small reactors
   1-2) K. Taketani (Chuo Univ.): Expected characteristics of future reactors for human beings
   1-3) Y. Tachi (Hitachi): Design and construction of small power reactors
   1-4) Y. Fujii-e, Hiroto Kawakami (TIT, Toshiba): A self-consistent nuclear energy supply system, a friendly FBR to fuel cycle and environment
2. Small reactor deployment plans
   2-1) S. Hattori (TIT/CRIEPI): Technical and economical potential of small reactors
3. Water cooled small reactors
○ 3-1) M. Higuchi (JAPC): An overview of the simplified LWRs
○ 3-2) K. Sako and J. Oda (JAERI): Concept of highly passive safe reactor SPWR
○ 3-3) H. Kobayashi, K. Sako, H. Iida, and K. Ishizuka (JAERI): Application of MRX (Advanced Marine Reactor) to ships
○ 3-4) H. Iida, Y. Ishizuka, H. Kobayashi, and K. Sako (JAERI): Application of the DRX (Deep-See Reactor X) to a deep-see power source
   3-5) H. Akie and Dr. Y. Ishiguro (JAERI): Water moderated Th/U-233 breeder
4. Liquid metal cooled reactors
○ 4-1) S. Hattori and Handa (CRIEPI, Toshiba): Present design features of the super safe small and simple reactor
○ 4-2) S. Zaki and H. Sekimoto (TIT): A concept of long-life small safe reactor
   4-3) A. Otsubo and K. Haga (PNC): Concepts of a high temperature fast reactor
   4-4) M. Kawashima, H. Endo, and A. Shimizu (Toshiba, TIT): Conceptual study on the liquid metallic fueled core
   4-5) M. Uotani, I. Kinoshita, A. Ohto, K. Yoshida, and N. Ueda (CRIEPI): Conceptual design of modular double pool reactor
   4-6) H. Endo, M. Kawasaki, and A. Shimizu (Toshiba, TIT): Safety features of liquid metallic fueled core
   4-7) K. Haga, H. Seino, and A. Otsubo (PNC): Natural circulation liquid-metal-cooled fast reactor
   4-8) K. Hida (Mitsubishi Heavy Industries): Characteristics of small LMFBR for future energy resource
   4-9) F. Kasahara, S. Ohta, H. Endo, and A. Shimizu (Toshiba, TIT): Geometrical effect of reactor vessel reflected on the energy mitigation for CDA energetics
5. Gas cooled small reactors
○ 5-1) T. Hayashi, H. Hayakawa, and T. Nakata (Tokai Univ., Fuji, Kawasaki): Behavior of small HTGR core under reactivity accident
6. Molten-salt reactors
○ 6-1) K. Furukawa (Tokai U): Flexible Thorium Molten-Salt Nuclear Synergetics
7. Special uses
   7-1) H. Yasuda (JAERI): Conceptual study of small reactors for space use
   7-2) M. Sasaki, J. Hirota, S. Tamao, K. Kanda, and Y. Mishima (MAPI): Design study of a medical reactor for BNCT

○ Small and medium power reactors which are referred in the present paper.

3 CONCLUDING REMARKS

It can be expected that in Japan the bigger role of nuclear power in the future and importance of cooperation with the neighboring region of Asia and developing countries are well understood The small and medium nuclear reactors are suited well to these regions The design studies of small and medium reactors have been performed intensively in Japan Nevertheless Japanese nuclear vender companies do not have any strong interests to small and medium nuclear reactors, and both industries and government have not any concrete projects to promote and build some of these reactors

REFERENCES

[1] Long-Term Program for Research, Development and Utilization of Nuclear Energy, Atomic Energy Commission, Japan (1994)

[2] SEKIMOTO, H (Ed), Potential of Small Nuclear Reactors for Future Clean and Safe Energy Sources, Elsevier, Amsterdam (1992)

XA9846705

STATUS OF DEVELOPMENT — AN INTEGRAL TYPE SMALL REACTOR MRX IN JAERI

T. HOSHI, M. OCHIAI, J. SHIMAZAKI
Japan Atomic Energy Research Institute,
Ibaraki, Japan

Abstract

JAERI is conducting a design study on an integral type small reactor MRX for the use of nuclear ships.

The basic concept of the reactor system is the integral type reactor with in-vessel steam generators and control rod drive systems, however, such new technologies as the water-filled containment, the passive decay heat removal system, the advanced automatic system, etc., are adopted to satisfy the essential requirements for the next generation ship reactors, i.e. compact, light, highly safe and easy operation.

Research and development (R&D) works have being progressed on the peculiar components, the advanced automatic operation systems and the safety systems. Feasibility study and the economical evaluation of nuclear merchant ships have also being performed.

The experiments and analysis of the safety carried out so far are proving that the passive safety features applied into the MRX are sufficient functions in the safety point of view.

The MRX is a typical small type reactor realizing the easy operation by simplifying the reactor systems adopting the passive safety systems, therefore, it has wide variety of use as energy supply systems.

This paper summarizes the present status on the design study of the MRX and the research and development activities as well as the some results of feasibility study.

1. INTRODUCTION

A small type reactor might be of great advantage to the energy supply for such the limited area as a ship, an island, a remote place, etc. Wide variety of energy usage, for examples, an nuclear propulsion, an electric power supply, a district heating, a sea-water desalination, etc., is considered.

Nuclear ships have outstanding advantages so as to enable long-period navigation with high power and long-period underwater navigation. It is therefore, thought that the nuclear ships will contribute largely to the advancement and diversification of marine transportation, scientific activities in the ocean, etc., in the future. Because of no production of such polluting materials as NOX, SOX and CO2 produced in the use of fossil fuel, the utilization of nuclear reactor into ships contributes to the global environmental protection.

At present nuclear ships are not expected to be applied immediately for practical use mainly from economic aspect. Looking into future, however, there is a lot of potentiality that the needs of nuclear ships would be actualized according to to the change in economic and social circumstances and owing to the increase of demands for their outstanding advantages.

The Japan Atomic Energy Research Institute (JAERI) is carrying out the design studies on advanced nuclear ship reactors of which features are compact in size, light in weight, highly safe and easy to operate. The Marine Reactor X (MRX) is aimed for the use of a bigger merchant ship or an ice-breaker.

In parallel with the design study, research and development works are being conducted to realize the systems in use and to prove the operational and safety functions of the systems.

2. CONCEPT OF ADVANCED MARINE REACTOR MRX [1]

2.1 Basic Concept

To put nuclear merchant ship into the commercial service, the improvement of economy is very essential as well as increasing reliability and operability of the reactor systems.

The improvement of economy should be emphasized in developing the reduction of construction and operation cost as follows;

-The construction cost of reactor systems can be reduced by means of making the reactor systems compact, light and simple. Adoption of passive safety systems contributes for the simplification of systems.

-The operation cost of reactor systems can be reduced by means of easy operation and maintenance and improving the reliability of both systems and components. Simplification of systems and the advanced automatic operating and supporting systems will be effective. Reduction of nuclear fuel cost is also essential.

-Simplification of plant systems increases the reliability so that reduction of malfunctions and defects of components is expected. Adoption of the advanced automatic operation and supporting system is also effective to prevent human errors.

In order to satisfy these requirements, following typical design features have been adopted (Fig.1).

- Integrated type PWR
- In-vessel type control drive mechanism
- Water-filled containment
- Passive decay heat removal systems
- Advanced automated control systems
- One-piece removal of the reactor systems

Fig.2 and Table 1 give a conceptual drawing and major design parameters of the MRX. A basic idea of engineered systems is shown in Fig. 3.

2.2 Description of Reactor and Primary Systems

(1) Integrated type PWR

Integrated type PWR could eliminate possibility of large scale pipe rupture accidents and then simplifies the safety systems. It also reduces the dimensions of reactor plant system. Because of compact dimensions of the integrated reactor system, it should remind the capability of maintenance and inspection of components. In the MRX, it is so designed that the reactor components and the primary coolant pumps can be removed remotely and the steam generator tubes can be inspected from out side of the reactor pressure vessel.

TABLE 1. MRX BASIC DESIGN DESCRIPTION

Reactor type	: Integral PWR
Thermal power (MWt)	: 100
1. Core and reactivity control	
Fuel/moderator material	: UO_2/H_2O
Fuel inventory (tons of heavy metal)	: 6.326
Average core power density (kW/liter)	: 41
Average/maximum linear power (kW/m)	: 7.626/30
Average discharge burnup (MWd/t)	: 22,600
Enrichment (initial and reloaded)	: 4.3/2.5% (without/with Gd)
Life of fuel assembly (year)	: 8
Refueling frequency (year)	: 4
Fraction of core withdrawn (%)	: 52.6
Active core height (cm)	: 140
Equivalent core diameter (cm)	: 149.2
Number of fuel assemblies	: 19
Number of fuel rods per assembly	: 493
Rod array in assembly	: Triangle
Pitch of assemblies/fuel rods (mm)	: 326/13.9
Clad material	: Zircalloy 4
Clad thickness (mm)	: 0.57
Type of control rod	: Cluster
Number of rod clusters	: 13
Number of control rods per assembly	: 54
Neutron absorber material	: 90 % enriched B_4C
Additional shutdown system	: Boron injection
Burnable poison material	: Fuel rod with Gd_2O_3 and burnable poison rod of borosilicate glass

2. Reactor coolant system	
(1) Coolant	
Coolant medium and inventory	: H_2O (41 t)
Coolant mass flow through core (kg/s)	: 1,250
Cooling mode	: Forced
Operating coolant pressure(MPa)	: 12
Core inlet/outlet temperature(℃)	: 282.5/297.5
(2) Reactor pressure vessel	
Inside diameter/Overall length (m)	: 3.7/10.1
Average vessel thickness (mm)	: 150
Design Pressure (MPa)	: 13.7
(3) Steam generator	
Number of SG	: 1 (2 trains)
Type	: Once-through helical coil
Configuration	: Vertical
Tube material	: Incoloy 800
Heat transfer surface per SG (m^2)	: 754
Steam/feed water temperature (℃)	: 289/185
Steam/feed water pressure (MPa)	: 4/5.8
(4) Main coolant pumps	
Number of cooling pumps	: 2
Type	: Horizontal axial flow canned motor
Pump mass flow rate (kg/s)	: 640
Pump design rated head (m)	: 12
Pump nominal power (kW)	: 145
3. Containment	
Type	: Water filled (simple wall)
Inner diameter/height (m)	: 7.3/13
Design pressure (MPa)	: 4
Design temperature (℃)	: 200

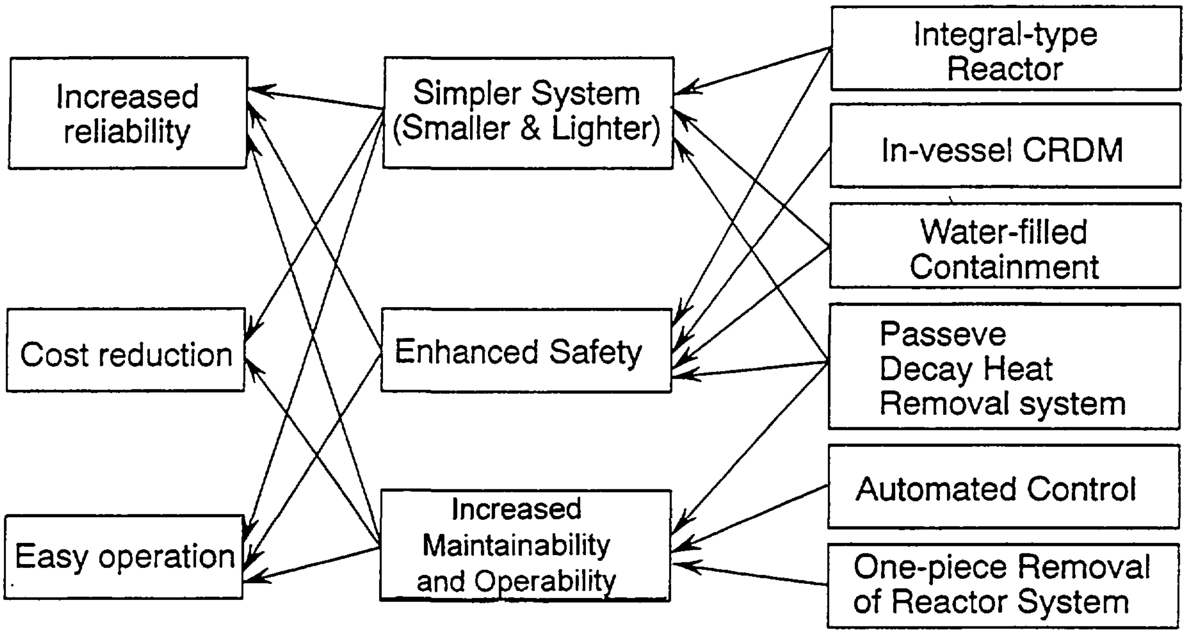


Fig.1 Requirements and New Technologies adopted in MRX

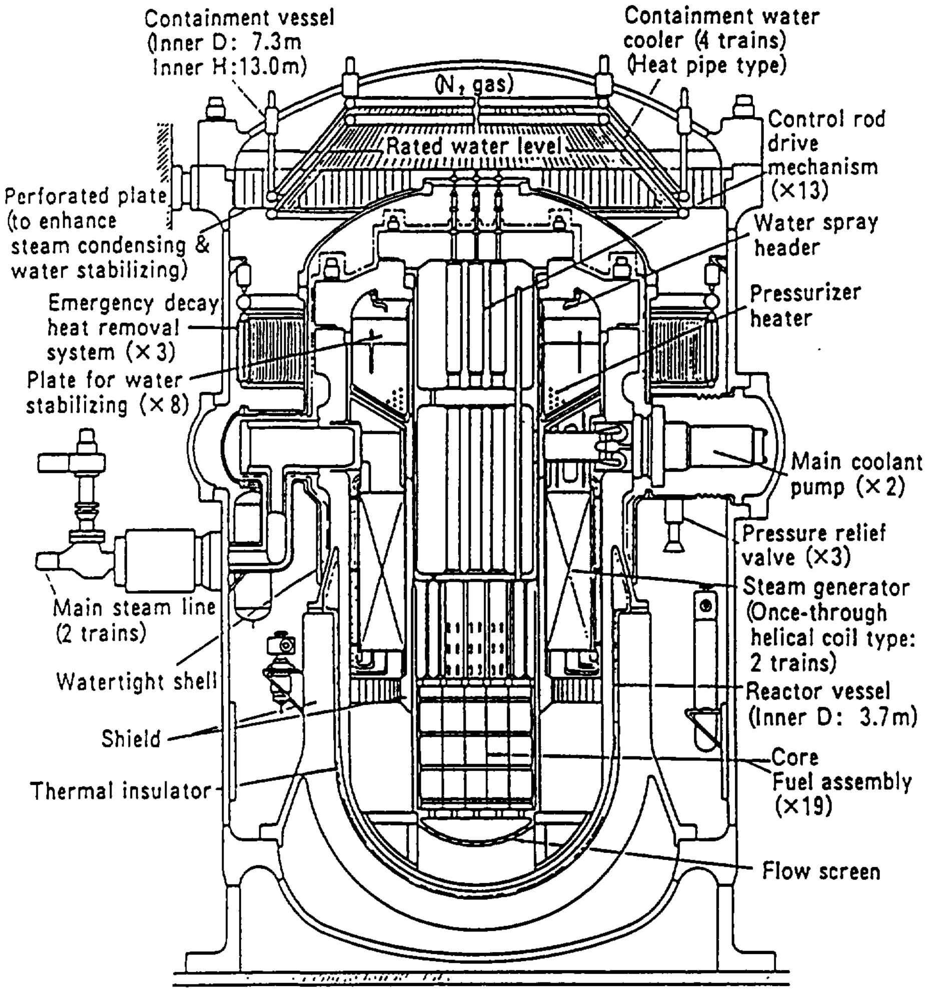
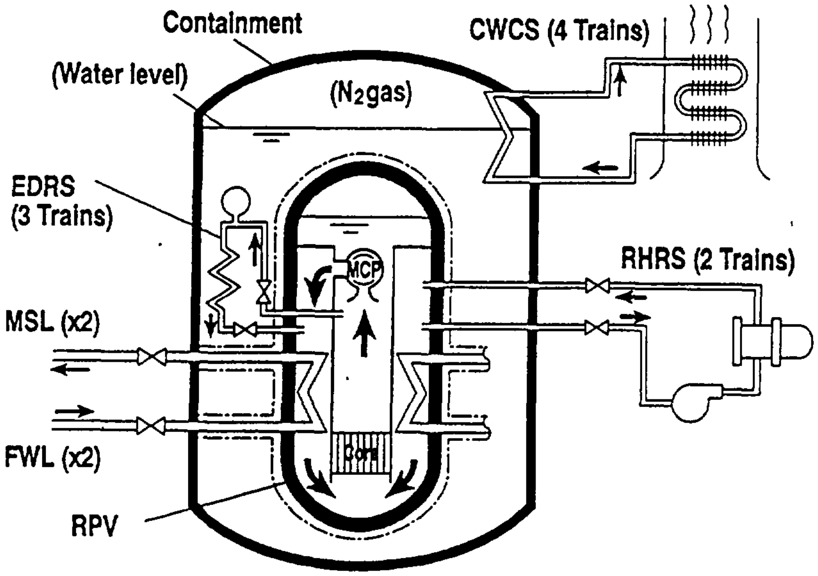


Fig.2 Conceptual Design of MRX



EDRS : Emergency Decay Heat Removal System
CWCS : Containment Water Cooling System
RHRS : Residual Heat Removal System
MSL : Main Steam Line
FWL : Feed Water Line

Fig.3 MRX Reactor Cooling Systems

(2) Reactor core and reactor pressure vessel

The core consists of 19 fuel assemblies and of 13 control rod clusters. Conventional PWR type fuel rod (9.5 mm O.D.) is employed. The reactor pressure vessel (RPV) of 6.8 m I.D. and 9.3 m H. is relatively larger in size. This provides a larger primary water inventory with increasing the distance between the reactor core and the RPV, and reduces the neutron fluence at the RPV. The average power density of 42 kW/l is sufficient low so as to have an enough margin for larger load change of the ship operation.

(3) Control rod drive mechanism

In-vessel type control rod drive mechanisms (CRDMs) are placed in the upper region inside the RPV. Employment of the in-vessel type CRDM could eliminate the possibility of rod ejection accident and enable the reactor plant compact.

(4) Steam generator and primary circuit

Steam generator of once-through helical type is positioned in the RPV. Two trains are adopted for the main steam and feed water lines. The whole primary circuit is almost incorporated within the RPV. The pressurizer is installed in the upper part of the RPV. Two main coolant pumps are placed in the hot leg at the upper cylindrical region of the RPV as shown in Fig.2.

2.3 Engineered Safety Systems

(1) Water-filled containment

In the MRX, water injection systems are not provided for LOCAs and core flooding during LOCAs is maintained by the water-filled containment system which could maintain core flooding passively by limiting the blowdown of the primary coolant into the containment. The design pressure of the containment vessel is 4MPa to withstand a high pressure at LOCAs.

A compact reactor plant is realized by the water-filled containment because water in the containment acts as the radiation shield so that installation of concrete shield could be avoided.

(2) Emergency decay heat removal system(EDRS)

This system transfers decay heat of the core to the containment water at the event of isolation of reactor containment. It includes three trains and one of trains has ability to remove the core decay heat of 50%. Each train consists of a water reservoir tank, a cooler and two valves. In any case of accidents, coolant is circulated by natural convection.

(3) Containment water cooling system (CWCS)

This is a heat pipe system for a long term heat removal from the containment water to the atmosphere. For its working gas, anti-freezing gas such as R22 (CHClF2) will be used taking into account of low temperature conditions in the ice-sea atmosphere.

2.4 Automatic Control System

It is very important to reduce the number of reactor operators in the economy and safety points of view. To reduce operator actions, highly automated control systems will be adopted and will cover whole operations, i.e., start-up check-out, start-up, power operation and shut-down in normal operations as well as safety actions during abnormal and accident conditions.

The system consists of control systems and diagnostic systems as shown in Fig. 4. Control systems generate control signals for control equipments, for examples, control rods, pressure control valve, flow control valve, etc., in accordance with the reference signals (demand signal) and each parameters. If difference in

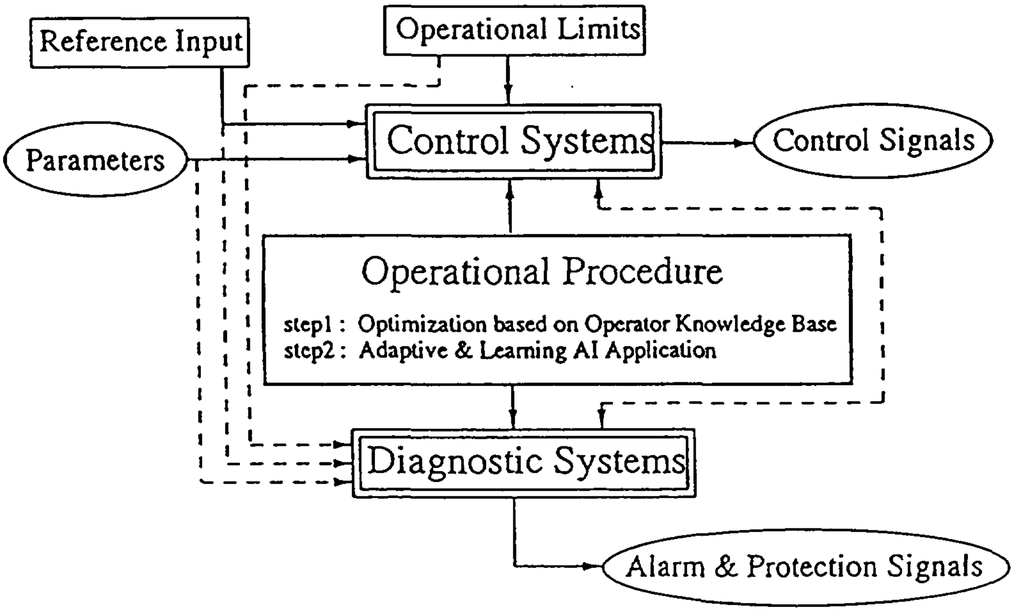


Fig.4 Advanced Automatic Control and Diagnostic Systems

each parameter is existed, control signals are generated by a given operational procedures to bring parameters to demand conditions. Giving demand signals as reference signals before starting operation, no operator actions would not be requested. To monitor the malfunction of systems and plant operating conditions, diagnostic systems are also provided.

Although concepts of the systems are similar as the conventional plant, adaptive and learning AI systems are being adopted.

2.5 Maintenance

For nuclear ships, it is very important to shorten the period of the maintenance and refueling from the economical point of view. From this standpoint, the design study of one-piece removal method is being carried out. This method is that the reactor containment is removed itself with the RPV and the auxiliary systems and then is transferred to the maintenance facilities as shown in Fig 5. After the removal, the new reactor containment of which maintenance is already completed is replaced.

It is thought that this method is promising because the integral type reactor is relatively small and light. The merits of this method are ;

(a) to shorten the period in the dock required for maintenance and refueling,
(b) to carry out the maintenance and refueling in the large space of land facilities safely,
(c) to reduce the cost of the maintenance and refueling by using them commonly,
(d) to re-use the reactor system after the ship's life, and
(e) to make the decommissioning of the ship easily.

2.6 Safety Evaluation

In the evaluation of safety characteristics of the MRX, analysis of LOCA, steam line break accident, feed-water line break accident, and total loss of electricity have been made.[2]

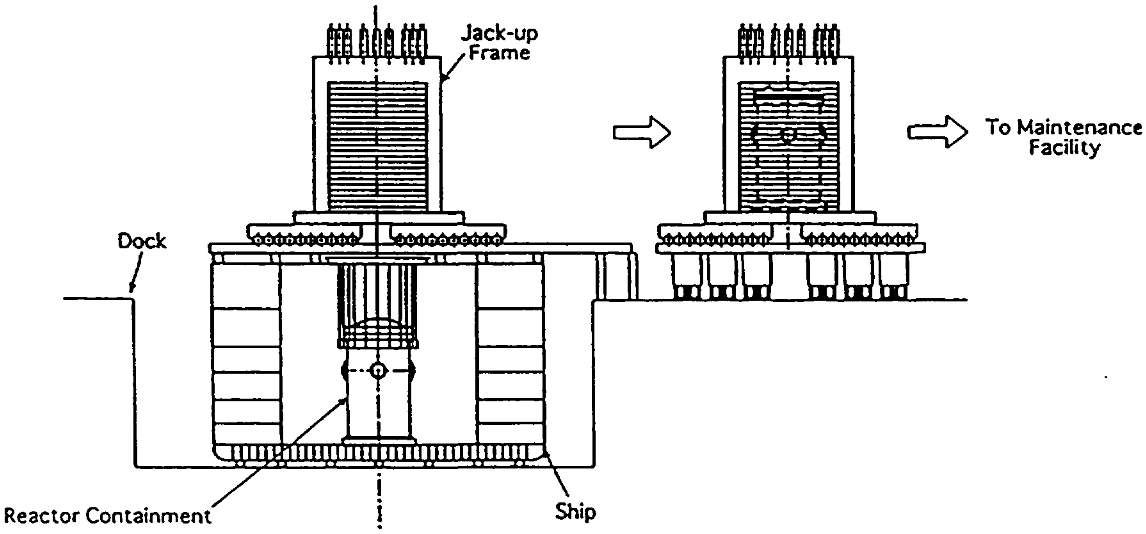


Fig.5 Concept of One-piece Removal of Reactor Containment

Fig. 6 shows a typical result of LOCAs obtained by the RELAP 5/Mod2 calculation assuming the double ended guillotine break of a 50 mm dia. pipe of the Residual Heat Removal System of which case is to be the most severe one in the design basis accidents of the MRX. The reactor goes shutdown at 119 seconds after the pipe break and the pressure in the containment reaches to the maximum at 1250 seconds after the break and then the escape of water from the RPV stops resulting that the drop of water level stops. The maximum pressure of the containment vessel during LOCA is 1.8 MPa which is lower than the design pressure of containment vessel (4 MPa). Water level in the RPV is higher (2m) than top of the core even in the ship inclines 30 degrees.

Through these analysis, it is being proved that the passive safety features applied into the MRX are sufficient functions in the safety point of view.

3. RESEARCH AND DEVELOPMENT PROGRAM

Research and development program on a nuclear ship has been proposed and being performed to solve technical subjects so that the nuclear ship will be put into commercial uses in the future.

The subjects are assorted into two groups, one is the research and development on the reactor system and the other is those on the nuclear ship systems. Because of importance to show the reliability of system, integral system tests using a large scale synthetic test rig is planned. A prototype integral test reactor program is also under discussion in JAERI.

R & D time table is shown in Fig. 7.

3.1 Research and Development of Reactor Systems

(1) Experimental study on thermal-hydraulics

Since such new and unique technologies as the integral type reactor concept, passive safety systems, etc., are being adopted, overall thermal-hydraulic characteristics of reactor systems are studied through the following experiments:

(i) Small scale thermal hydraulic test

To study the thermal-hydraulic behavior in the water-filled containment during the LOCAs, the Small Scale Test Rig (volume ratio: about 1/300 of MRX) has been fabricated and fundamental experiments are in progress.[3] In the experiments, following behaviors are evaluated.

- thermal and hydraulic responses in both the reactor vessel and the water-filled containment under LOCAs
- evaluation of mechanical loads generated by LOCAs
- capability of natural circulation and passive decay heat removal

(ii) Large scale synthetic test

To confirm the function of the safety features such as an integral type PWR with a water filled containment and passive safety systems, installation of Large Scale Synthetic Test Rig is planned. The conceptual view of the rig is shown in Fig. 8. Thermal power of the test section is 5 MWt (about 1/20 of the MRX), however, the height of the rig is same as that of the MRX because it is most important to simulate accurately the natural circulation condition. To obtain the behavior under the ship motions, inclinations and vibration, tests on a boat, where the test rig is loaded on a boat, is planned.

These experimental results will be very useful for understanding the phenomena related to the passive safety systems and for verification of the system.

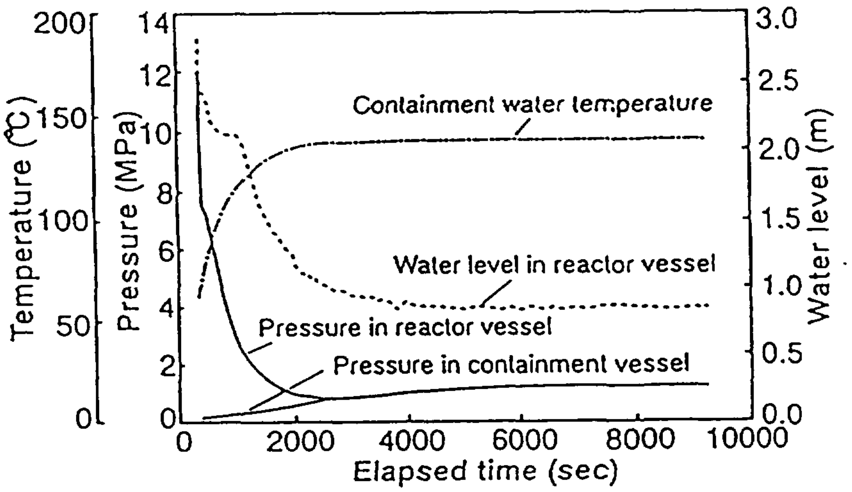


Fig.6 Typical Transient of LOCA in MRX
(Double ended guillotine break of 50mm dia. pipe)

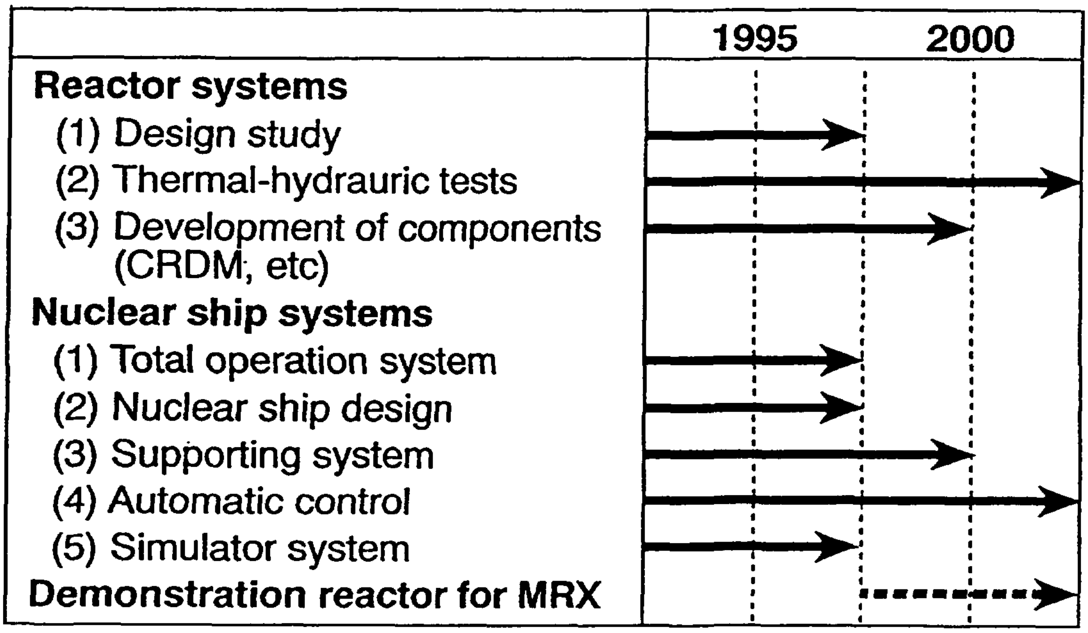
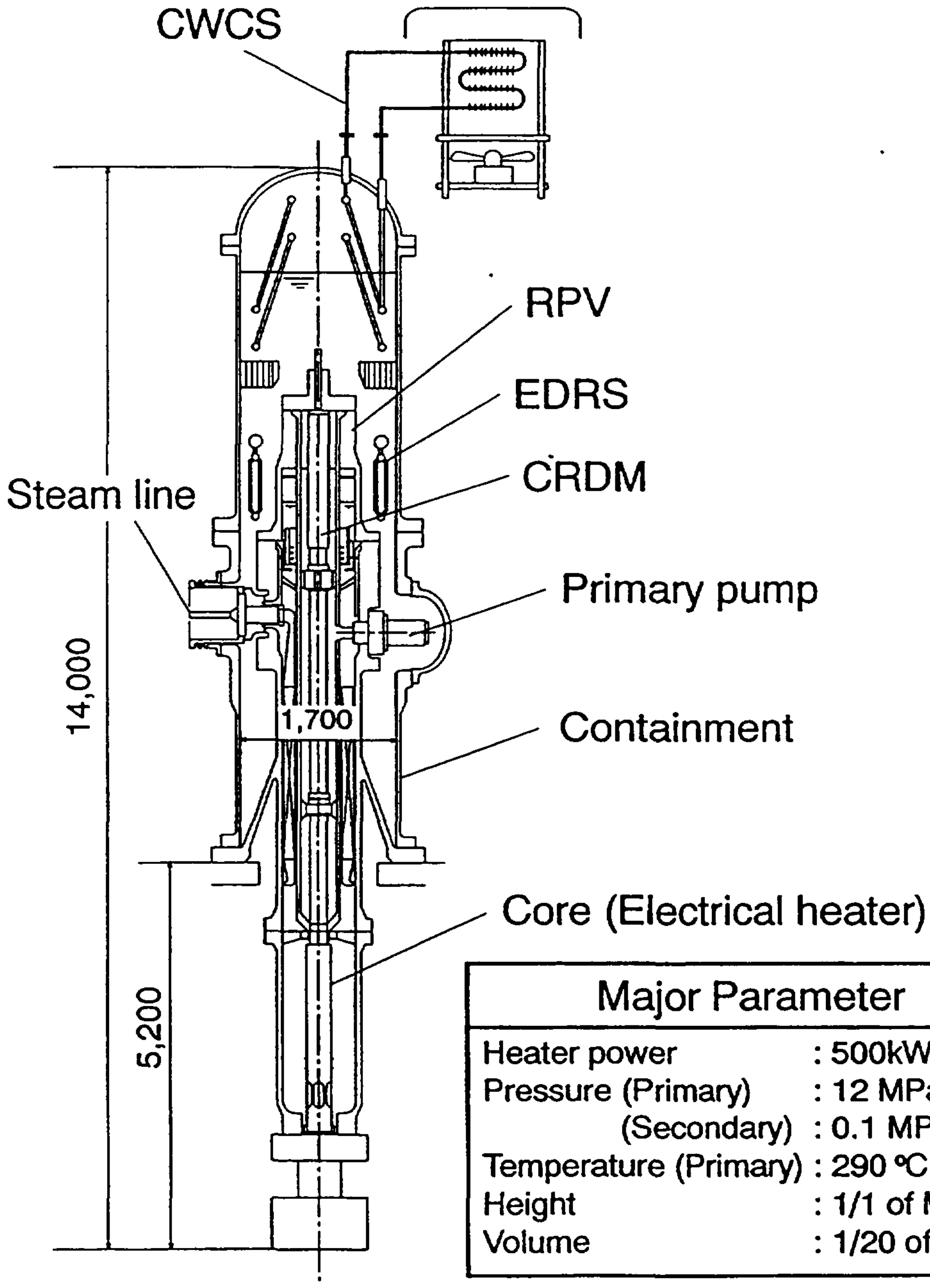


Fig.7 R&D Schedule on Advanced Nuclear Ship in JAERI



Major Parameter	
Heater power	: 500kW (5MW)
Pressure (Primary)	: 12 MPa
(Secondary)	: 0.1 MPa
Temperature (Primary)	: 290 ℃
Height	: 1/1 of MRX
Volume	: 1/20 of MRX

Fig.8 Conceptual View of Large Scale Synthetic Test Rig

(2) Development of components

(i) In-vessel type control rod drive mechanism (CRDM)

In order to avoid a control rod ejection accident and to make the reactor system compact, the in-vessel type CRDM which is operable at the PWR operation conditions is applied. High temperature and pressure water-proof components such as a motor, a latch magnet, etc., have been developed.[4] Function and reliability tests using the full mock-up CRDM is being planned.

(ii) Water-proof components and insulator

Components in the containment are submerged in water, therefore development of water-proof components and thermal insulator is requested. Easy maintenance is also essential. Conceptual design is under way.

3.2 Research and Development on Nuclear Ship Systems

In order to put nuclear ships into the commercial use, it is indispensable to realize an economical, safe and reliable reactor system. In addition, such supplementary items as international agreement on safety, sea rescue, preparation of maintenance yards, etc., should be solved.

Following activities are being progressed.

(i) Conceptual design of nuclear ships and supporting system

In order to determine the design conditions of reactors and to study total operation systems, conceptual design of many kinds of nuclear ships such as a general cargo ship, a container ship, an ice breaker, a deep-sea submersible, etc., are being performed.

(ii) Evaluation of economy

Cost evaluations not only on a rector system but also on a ship system is being made to clarify the most cost sensitive items. A typical result of cost evaluation for RFR (Required Freight Rate: operation cost to transport one container) of an 6,000 TEU container ship is given in Fig. 9 and Fig. 10.

(iii) Development of nuclear ship simulator

The nuclear ship simulator NESSY (Nuclear Ship Engineering Simulation System) has been developed in JAERI and used for the simulation of the nuclear ship "Mutsu" so far.[5] It can simulate both behaviors of the reactor system and the ship motions. Mutual interactions such as changes of the reactor power, the steam generator water level, etc., due to the ship motions by wave or maneuvering can be analyzed.

The accuracy of the system has been verified with the operation data of the "Mutsu" and it is proved that this system is very useful to analyze the plant behaviors in normal and accident conditions.[6] Modifications of models and parameters are being made for the MRX reactors.

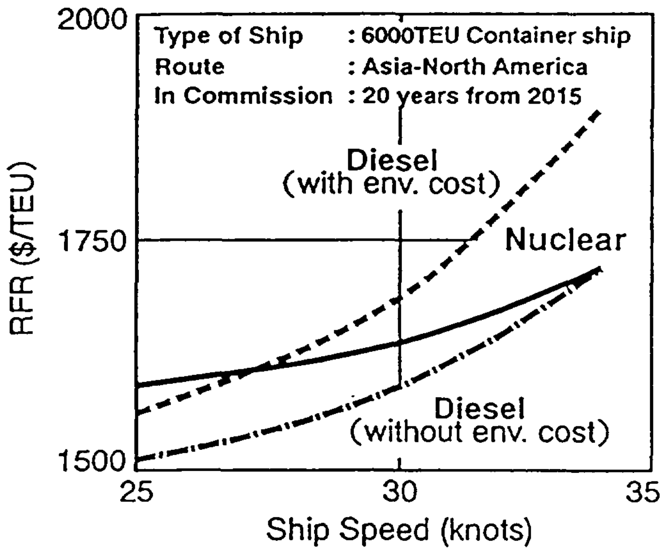


Fig.9 RFR as a function of ship speed

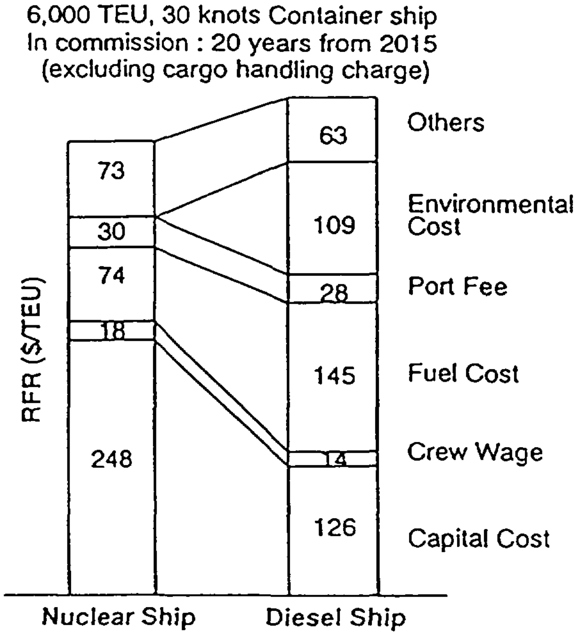


Fig.10 Comparison of RFR

4. Conclusion

A concept of the advanced nuclear ship reactors MRX has been established. The reactor systems adopt the passive safety systems. The safety capability of the reactor system has been proved by analyses. In addition to the design study, extensive research and development activities are being performed and these activities can contribute largely for the realization of the nuclear ships in the commercial use in future.

Considering that the MRX are small size reactors with highly safe capability and a transferable ones, they have wide variety of use as the energy supply system.

REFERENCES

[1] K. Sako, et al.,"Advanced Marine Reactor MRX", International Conference on Design and Safety of Advanced Nuclear Power Plants-ANP'92, Oct. 25-29, 1992, Tokyo, Japan

[2] A. Yamaji, et al., "Core Design and Safety System of Advanced Marine Reactor MRX", ICONE-3, Apr. 23-27, 1995, Kyoto, Japan

[3] T. kusunoki, "Steam Condensation Behavior of High Pressure Water's Blowdown Directory into Water in Containment under LOCA", ibid.

[4] Y. Ishizaka, et al., "Development of A Built-in Type Control Rod Drive Mechanism (CRDM) for Advanced Marine Reactor MRX", International Conference on Design and Safety of Advanced Nuclear Power Plants -ANP'92, Oct. 25-29, 1992, Tokyo, Japan

[5] T. Kusunoki, et al., "Development NESSY (Nuclear Ship Engineering Simulation System) and Its Application to Dynamic Analysis", ibid.

[6] M. Ochiai, et al., "Present Status of Nuclear Ship Engineering Simulation System", The 1994 SCS Simulation Conference, Apr. 11-14, 1994, San Diego, U.S.A.

REALIZATION OF SAFETY CULTURE INTO A REACTOR PLANT-4S (SUPER SAFE, SMALL AND SIMPLE) LMR

S. HATTORI, I. IKEMOTO, A. MINATO
Central Research Institute of Electric Power Industry,
Japan

Abstract

International Nuclear Safety Advisory Group(INSAG) defines Safety Culture as the following;

Safety Culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance.

While such Safety Culture is certainly a critical element of nuclear safety assurance, it is important to design nuclear power facilities as friendly to operators as possible with minimum dependence on human factors.

From the viewpoint of ensuring supply in our global society, it will be necessary to have multiple approaches to further promote the use of nuclear energy worldwide despite various social and cultural restrictions. It should then be considered, as prospective options, to disperse small nuclear power plants throughout the world under technical, social and cultural conditions.

Under this circumstance, we have quested for and now propose a scheme of assuring sheer safety of nuclear power plants by implementing operator-friendly nuclear reactors virtually free from human errors. The scheme specifically includes the measures for improving reliability through fabricating more compact reactors with a continuous in-factory production system, simplifying maintenance and inspection of the reactor system using passive systems and further relieving operators of burden of labors.

I. INTRODUCTION

In the 21st century, we will be confronted with how to solve some very serious problems which have not been faced by mankind in the past That is the so called 'dilemma problems' that is energy security, environmental protection and socio-economic development which conflict against each other.

Nuclear power appears to have the potential to help to solve these problems However, to make it a reality, it is necessary to improve nuclear power technology to make it more applicable to a greater variety of utilization and locations

To achieve this target, the following key items are essential

1. Enhancing the efficiency of nuclear power utilization
2. Increasing the flexibility of nuclear power siting
3. Assurance and improvement of safety and reliability
4. Improving the economics
5. Promoting public acceptance of nuclear power
6. Nuclear power utilization in developing countries
7. Improving the fuel cycle and waste management
8. Providing the high proliferation resistance

To satisfy the specific key items 3, 7 and 8, the following safety features have been focused to carry out the plant design

1. Elimination the melting of fuel and the voiding of coolant at all anticipated events
2. Confinement of nuclear material in a reactor for a long time.
3. Using the recycle process with high proliferation resistance and confinement of plutonium inside reactor plant
4. Unattended reactor with easy operation

II. BASIC CONCEPT

Based on the requirements, one of the most promising nuclear reactor designs is a small or medium size modular type nuclear reactor with high inherent safety and passive characteristics. It is preferable that the following features are taken into considerations; greater simplicity, easy to maintain, inspect and operate, less influence of human factors, high reliability, improved availability and capacity, design standardization, easier to construct, quicker to construct, more flexibility in siting, lower initial investment and better adaptability to electrical grid management. In addition, the need to improve the fuel cycle, waste management and nuclear proliferation resistance, requires special design characteristics and fuel management provisions.

Our design efforts to satisfy these conditions have resulted in the development of the Super Safe, Small and Simple (4S) fast reactor with an electric power output of 50 MWe.[1],[2],[3],[4],[5] The 4S has the following safety features:

(1) Higher safety can be achieved by designing all reactivity feedback coefficients including coolant void reactivity to be negative and by controlling neutron leakage from the core by an annular reflector. The potential for super prompt criticality, particularly during start up, is completely excluded by using metallic fuel. A fully passive heat removal system is employed in the 4S so that the auxiliary support system for the safety system can be eliminated, thus improving the reliability of the safety system

(2) In order to keep strict control over the plutonium used, the 4S incorporates a new concept by using metallic fuel which significantly helps to achieve the non-proliferation goal; a large amount of fuel can be confined for a long time in the reactor vessel without refueling During the initial start up, the reactor is sealed in the presence of IAEA authorities and the IAEA maintain long-term control over operations to ensure non-proliferation

(3) Improving the fuel cycle and waste management is most important for future nuclear systems. A fast reactor technology using a metallic fuel cycle(pyroprocess of spent fuel appears to be a most promising approach.[6] The technology is valuable because it has the potential to simplify reprocessing, fuel fabrication process and nuclear waste disposal, and it includes actinide recycling which is important from the viewpoint of resource utilization. It also reduces the fuel cycle cost dramatically.

III. SAFETY FEATURES

(1) Core and Power Control System

4S employs a burnup control system with annular reflector in place of the control rod and its driving mechanism, which requires frequent maintenance service. Replacement of the reflector is not required for the entire plant life. Burnup control by vertical movement of the annular reflector eliminates necessity for complicated control rod operations.

Figure 1 shows 4S reactor assembly. The diameter and length of the reactor vessel are 2.5m and 23 m, respectively.

When the length of sub-assembly is restricted to 7m at maximum from view point of the present manufacturing process of it, thermal power of the core should be less than about 125 MWth. Under these restrictions, the core of 4S is specified to have thermal output 125 MWth, active length 4m and equivalent diameter 83 cm.

Table 1 shows the breakdown of reactivity of the 125 MWth core with metallic fuel at 350°C of the primary coolant temperature, accompanying changes in the coolant inlet temperature and the inserted reactivity required for 100% change in output. With metallic fuel which has small Doppler coefficient, the core output reaches 100% with the core ΔT of 130°C. The output can be controlled through control of core inlet temperature only in case of the metallic fuel. Taking this advantage, it was decided that the start-up of 4S's core is done by increase of water flow, which caused changes in core ΔT, after rising system temperature.

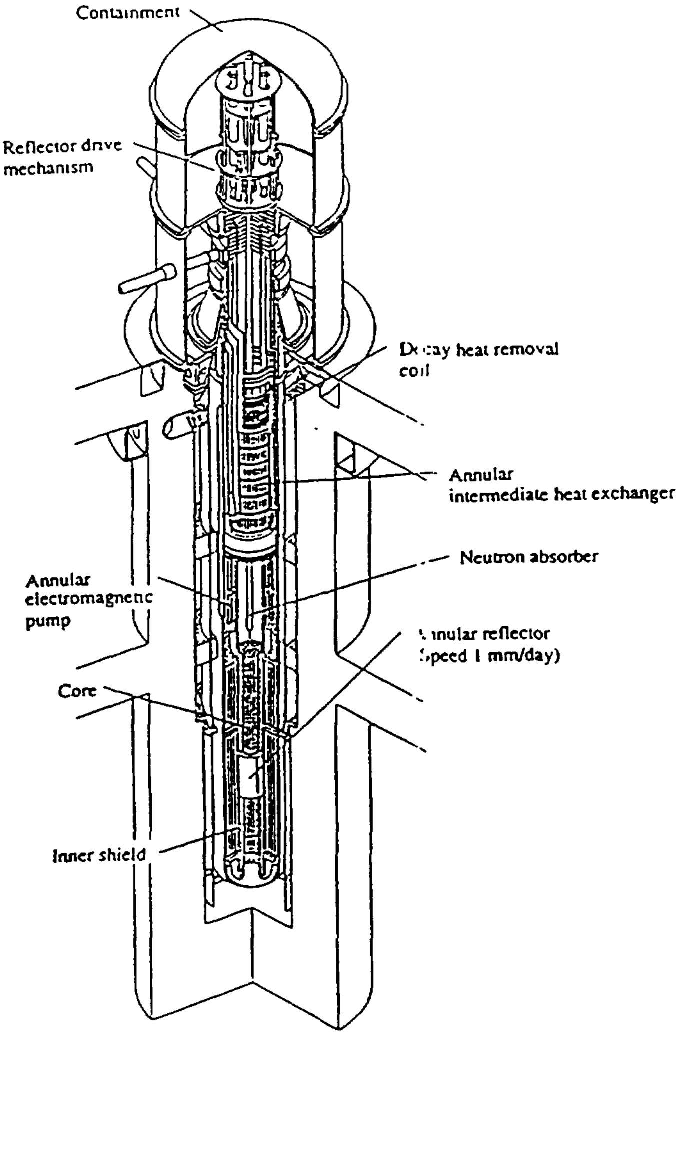


Fig. 1 4S Reactor Assembly

Table 1 Feedback Temperature Coefficients

	BOL (BARE) (0 YRS.)	BOL (REFLECTOR) (0 YRS.)	MOL (4~6 yrs.)	EOL (10 yrs.)
FUEL $\left(\frac{K/KK'}{°C}\right)$	-8.87×10^{-6}	-8.23×10^{-6}	-7.37×10^{-6}	-7.29×10^{-6}
STRUCTURE $\left(\frac{K/KK'}{°C}\right)$	-1.62×10^{-6}	-1.30×10^{-6}	-0.42×10^{-6}	-0.50×10^{-6}
COOLANT $\left(\frac{K/KK'}{°C}\right)$	-6.03×10^{-6}	-5.22×10^{-6}	-2.87×10^{-6}	-3.23×10^{-6}
CORE SUPPORT $\left(\frac{K/KK'}{°C}\right)$	-8.34×10^{-6}	-7.87×10^{-6}	-6.84×10^{-6}	-6.70×10^{-6}
DOPPLER $\left(T\frac{dk}{dT}\right)$	-1.83×10^{-3}	-2.22×10^{-3}	-2.79×10^{-3}	-2.80×10^{-3}

(2) Inherent Core Safety

There are two generic approaches to reducing void reactivity. Reducing the core height is one popular approach and a core with a small diameter is another effective method. In 4S, making the core diameter small is the preferred approach because this reduces the vessel diameter and enhances the value of the reflector reactivity. By reducing the core diameter, neutron leakage is enhanced in the radial direction so that negative void reactivity is maintained during the entire core life time. The void reactivity depends on the diameter and fuel volume fraction. For the selected core, the void reactivity of the total core is -1$ at the end of life based on the transport calculation. Other temperature feedback coefficients are all negative as shown in Table 1.

It is essential for the safety of the reactor to exclude the possibility of super prompt critical state at all time. This requires that the inserted reactivity at potential events should be below 1$ under conservative conditions, neglecting reactivity feedback coefficients.

The largest reactivity change occurs during plant start-up. The reactivity decrease from criticality at zero power under cold temperature conditions to full power is generally above 1$. The worst case is reactivity insertion under cold temperature conditions.

At plant start-up, the system temperature is raised to 350°C by heat input from the electromagnetic pump before lifting the reflector. This procedure greatly reduces the reactivity temperature swing. The reactivity to be inserted to increase the power is about 86¢ shown in Table 2, which causes the following reactivity effects; thermal expansion of fuel, structure, coolant, core support grid and Doppler reactivity. Because metallic fuel is employed in 4S, the reactivity is small compared with 150°C for MOX fuel in the same size of 4S core, mainly due to its small Doppler coefficient.

The basic dynamic characteristics of the core under various reactivity insertion conditions are shown in Fig. 2. The power transient reflects the super prompt critical condition when a large reactivity insertion occurs. On the other hand, the power transient is small for 4S during potential reactivity insertion at the plant start-up

In 4S core, all reactivity change is controlled by the reflector. This neutron leakage control system has a decisive advantage compared with control rod system from the safety point of view.

The active length of the core is 4m, which is surrounded by a 1.5m long reflector. The reflector is separated into 6 azimuthal parts, each of which can move from the bottom to the top of the core along with the core burn-up. If an uncontrolled lift of each part of the reflector occurs, the core criticality cannot be sustained. The new geometry of the reflected region causes negative reactivity insertion because of the enhanced neutron leakage.

Figure 3 shows that lifting up parts of the reflector gives strong negative reactivity except lifting up all parts of the reflector. Although the figure shows negative insertion, a small positive insertion up to ten cents may be possible if each parts of the reflector moves up a small distance from the original position. The maximum is -4$ when three parts are lifted up. This geometry gives the minimum criticality which maximizes neutron leakage.

Thus, the inherent core safety against partial movement of the reactivity control system is assured for the 4S core.

Table 2 Reactivity Swing from Cold to Hot Temperature (350°C Heat Balance at Full Power)

Fuel	-52 ¢
Structure	- 3 ¢
Coolant	-11 ¢
Core Support	- 1 ¢
Doppler	19 ¢
T o t a l	-86 ¢

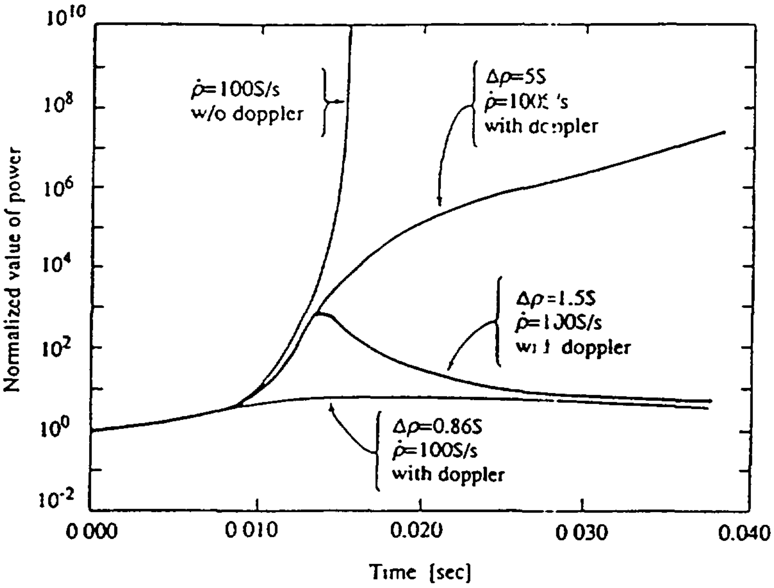


Fig 2 Reactor Power Transient for Various Reactivity Insertion with $\dot{\rho}$ =100$/s

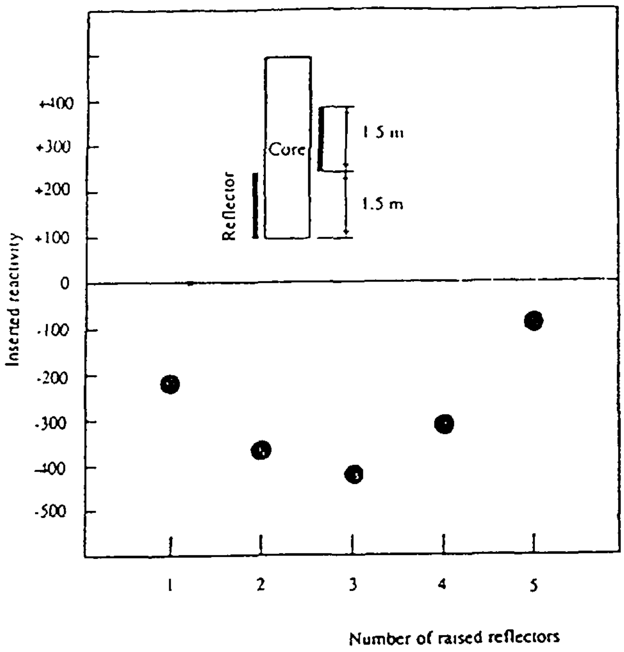


Fig. 3 Reactivity Insertion when Lifting Partial Segments of Reflector up to 1.5 m

(3) Passive Safety Features

The reactor concept of 4S is designed to enhance passive features shown in Table 3, which include passive safety system. In addition to these features, special attention is focused on excluding the reactivity insertion during plant start up.

At plant start up, uncontrolled withdrawal of control rods causes severe reactivity insertion accident in a conventional reactor. In 4S, the system temperature is uniformly raised before start-up. By this procedure, negative reactivity is inserted and rapid insertion of the reflector does not cause the reactivity insertion accident. As the magnitude of the reactivity insertion does not depend on the function of the active system, other than the pre-heated system temperature, this design feature can be classified to passive safety feature.

Table 3 Passive Design Features of 4S

Item	Specification
· Core Control System	Annular Reflector Movement (Nearly Passive)
· Primary Pump	Electromagnetic Pump
· Primary Flow after Shutdown	Natural Circulation
· Cavity Cooling	Natural Circulation
· Containment Cooling	Natural Circulation
· Secondary Pump	Electromagnetic Pump
· Emergency Room Cooling	Heat Storage System
· Safety Features	
Reactor Shutdown	Inherent Core Safety
Shutdown Heat Removal	Natural Circulation

The reactor shutdown system consists of a neutron absorber and reflector. The reactor is shutdown by inserting a neutron absorber into the core or moving down of the annular reflector which surrounds the core. Only two types of plant protection system are provided to work these shutdown system, the neutron detector installed at the outside of reactor vessel and the core outlet temperature detector in the reactor vessel.

In addition to two diverse active systems, 4S has inherent safe core characteristics. Temperature reactivity feedback coefficients and all coefficients are negative as shown in Table 1. This characteristics effectively mitigates hypothetical accidents such that both active shutdown systems would fail to function.

The shutdown decay heat removal system consists of PRACS(Primary Reactor Auxiliary Cooling System) and RVACS(Reactor Vessel Auxiliary Cooling System). The PRACS system has decay heat removal coil in the upper part of the intermediate heat exchanger(IHX). The RVACS system is natural air cooling system which removes the decay heat from the guard vessel. 4S does not have pony motor. When the reactor shutdown, natural circulation flow of the primary coolant is expected to 10% of the rated flow.

(4) Safety at Plant Start-Up

The accident of uncontrolled withdrawal of reflector at start up is especially important. The cold to hot reactivity swing of 4S in shown in Table 2. Rapid withdrawal of the reflector at cold shutdown leads to reactivity accident. In order to moderate this condition before start-up, at first the system is raised to 350°C by the primary pump heat entering. At the system temperature below 350°C, the neutron absorber cannot be withdrawn by the self-connected mechanism[4] based on the difference in thermal expansion of the stainless and Cr-Mo steels.

The reflector is lifted by the hydraulic system to reach the critical state at 350°C. Then, the reflector is periodically lifted, which motion is pre-programmed in the hydraulic system. The reactor power gradually increases by this motion and the increase of feed water flow in the turbine system.

Failure of hydraulic mechanism that causes the reflector move upward, includes failure of the speed adjustment valve. Should this failure takes place, reactivity of 5 ¢/sec at maximum is inserted. The low power trip level of neutron detector is not set in 4S for simplicity, but the trip signal is generated by 116% high power trip level only. When the trip signal is generated, the neutron absorber drops. Should the neutron absorber does not work, the hydraulic scram circuit causes the reflector to drop in a delay of 1sec, then the reactor shuts down. Should both systems do not work, the core inherent characteristics in Table 2 leads the reactor inherent shutdown state as shown in Fig. 4.

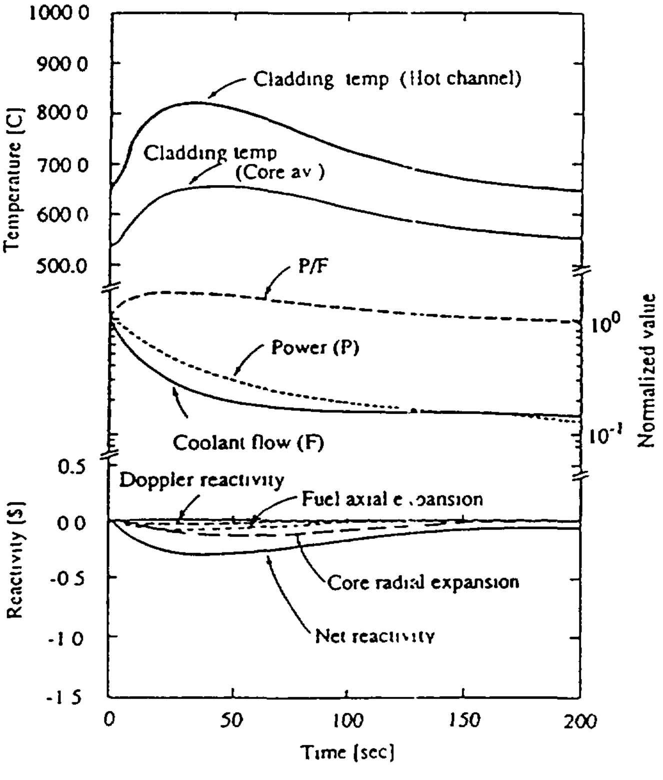


Fig. 4 Core Response for Loss of all Station Power without Scram

(5) Safety During Power Operation

Severe accidents for conventional reactor design during power operation are events without scram and steam generator accidents with total break of heat transfer tubes. The safety policy of 4S is to remove the safety concerns for this two kinds of events.

All the negative feedback coefficients are taken into consideration for the analysis of events without scram.

Radial expansion of core caused by the increase of temperature is the most effective negative feedback effect in the range of short period of the safety analysis[7].

The following initial events with assumption of failure of reactor shutdown system are analyzed.

- Reactivity insertion and loss of offsite power under horizontal earthquake
- Reactivity insertion by sudden temperature change of core Inlet due to flow increase of secondary loop or water system
- Loss of load
- Sudden loss of EMP(Electro Magnetic Pump) function
- Total loss of electric power

The largest temperature rise is caused by a horizontal earthquake accompanied by simultaneous insertion of reactivity and loss of flow under assumption of without scram. Temperature of the fuel cladding rises up to 850 °C but it drops in a short period of time without causing cladding damage by eutectic reaction and fuel is not melted.

Extreme assumption of the steam generator accidents is

- Break of all heat transfer tubes without credit of water dump

We do not expect for the protective action of water dump under accidental condition of the water steam system as it is a non-safety grade.

Therefore, we expect only the sodium-water production release system by broken rupture disk for the safety action. In this system, two 24B (0.6 mø) piping are installed into the steam generator to release the pressure and sodium-water reaction products. The maximum pressure remains lower than 0.8 MPa and the boundary integrity of IHX is maintained.

(6) Safety During Shutdown

For the total loss of all station power, a fast reactor is known to have large safety margin because of its natural circulation heat removal capability. More severe and extremely unlikely accidents respected to decay heat removal capability are examined.

Accidents with extreme loss of decay heat removal capability include

- Total loss of all station power in a very long time
- Destroy of PRACS by large aircraft falling and destruction of RVACS stack

When the offsite power does not recover for a long time, sodium in the secondary loop of PRACS is frozen at two days after the reactor shutdown and decay heat is removed only by RVACS.

After ten years, the sodium in the reactor vessel is frozen and the decay heat is removed by thermal conductivity. No specific heat entering is required.

Assumption of the destroy of PRACS by fallen large aircraft and the destruction of RVACS stack is the most severe condition. We assume the stack length of 10m of RVACS is with function under such accident, because this part is contained in the concrete building. After forty hours, the peak temperature becomes 600°C, but dose not affect the structural integrity of the reactor vessel.

IV OPERATION

One of the excellent features of the 4S is that it is simple to operate There are no feedback control systems and no human intervention is required All reactivity control is performed by the automatic movement of the reflector

Regular power operation is attained by moving the reflector upward at a constant speed of 1 mm/day to compensate for the reactivity decrease due to the burn-up of the core Since no feedback system or control system are used, the reflector speed remains constant and the electric output is adjusted by varying the feed water flow rate to control the core inlet temperature The controllable range of the power level by the water flow is ±10% at the rated power, which is limited by the steam generator heat balance Beyond this range, a back-up control mechanism to adjust the reflector position is installed in the driving mechanism

To follow the load, the core inlet temperature is changed by controlling the water flow so that the generator output coincides with the load-following control, thus causing the reactor output to follow

As mentioned above, elimination of all feedback control systems from the reactor and secondary heat transport systems makes the 4S plant control system very simple and economic

V PROLIFERATION RESISTANCE

(1) Long Confirmation of Nuclear material in Reactor

4S is one wherein a long core with small diameter is surrounding by annular reflector to control the burning and enhance the safety of the core Its lifetime is set at ten years to eliminate the need of complicated refueling work. The fuel being a nuclear material is also sealed for ten years and subjected to a rigorous control by IAEA The reactor is highly resistant to nuclear proliferation since no one is allowed to make any access to the fuel

(2) Integral Nuclear Power Plant Concept

In order to confine the plutonium and minor actinide in reactor plant, we would propose a integrated nuclear power plant concept with a high nuclear safety based on the 4S (50MWe) plant and a metallic fuel cycle facility with pyroprocessing of spent fuel.

Figure 5 shows the nuclear power plant with fuel cycle facility. The total electrical output is 1000MWe with 10 modules of two 4S units(50MWe x 2). and one turbine system The capacity of the fuel cycle facility is 20 ton/year and is able to reprocess the spent fuel from 4S for ten years. In this plant concept, plutonium and minor actinide will be confined in the facility

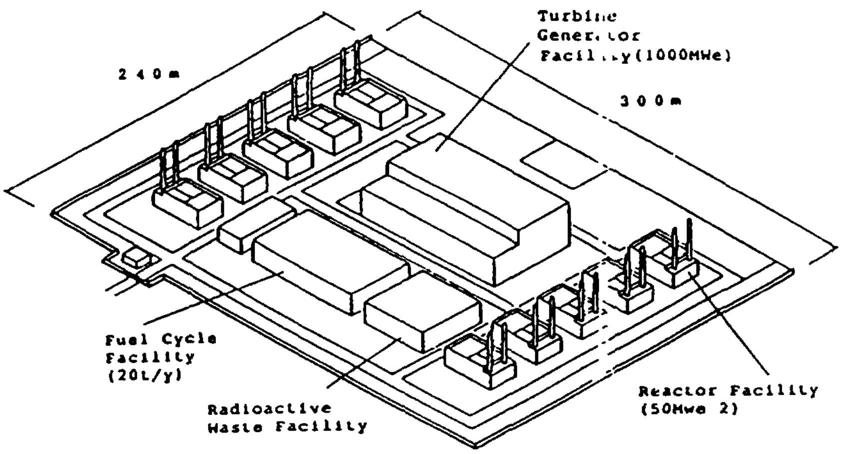


Fig 5 Nuclear Power Plant Concept with 4S-Units

The system still retains several advantages of the 4S as follows,

- Nuclear proliferation resistance
- Inherent safety of the core
- Easy operation and less maintenance

The spent fuel discharged from the reactor after ten years of operation will be treated by IFR[6] type reprocessing as advocated by ANL, so that it can be re-used as reactor fuel, while the long half-life wastes will be confined within the fuel cycle

Thus, we could establish a self-supporting system in which plutonium is safely contained for a long time until more energy is needed, while covering the management cost with by revenues from power generation

VI CONCLUSIONS

Nuclear energy is considered to be the most useful source of energy to address the demand for energy resources growing across the whole world including developing countries in the coming 21st century Besides, with a credibly forecast shortage of natural uranium, development of fast reactors will be a key factor for efficient use of this material

We have thus developed a scheme of specially designed small fast reactors which is to be implemented by improving reliability with a continuous in-factory production system, simplifying maintenance work and inspections of the reactor system using passive systems, relieving operators of burden of labors and enhancing their working safety, and ensuring operator-friendliness with minimized human errors

For the worldwide introduction of nuclear reactors in future, we believe firmly that a number of actual reactor designs incorporating the safety philosophy herein suggested should be proposed by other scientists and engineers, with growing aggressiveness toward the Safety Culture which could encourage a wider use of such reactor

ACKNOWLEDGEMENTS

The authors wish to thank Prof S Kondo, University of Tokyo, chairman of the steering committee for the evaluation of 4S plant design, for his valuable comments We would also like to thank S An, Research Advisor to CRIEPI, for his encouragement to this design study

REFERENCES

[1] Hattori S., Handa N ,"Use of Super-Safe, Small and Simple LMRs to Create Green Belts in Desertification Area", Trans. ANS., Vol.60(1989)
[2] IAEA,"Use of Nuclear Reactors for Seawater Desalination", IAEA-TECDOC-574(1990)
[3] Hattori S., Minato A., Handa N ,"Present Design Features of the Super Safe Small and Simple Reactor", Int. Specialist Meeting on Potential of Small Nuclear Reactors for Future Clean and Safe Energy Sources, Oct 23-25, Tokyo Institute of Technology (1991)
[4] Hattori S , Minato A.,"Current Status of 4S Plant Design", 2nd ASME-JSME Int Conf on Nuclear Engineering-1993, S F , California, March 21-24, 1993
[5] An S , Minato A ,"Study on the Application of Nuclear Energy to Human Welfare and Safety", 4th Anuual Scientific & Technical Conference of Nuclear Society, NE'93, Nizhni Novgorod, Russia, June 28-July 2, 1993
[6] Till C E , Chang Y I ,"Progress and Status of the Integral Fast Reactor(IFR) Fuel Cycle Development", Proc of Int Conf on Fast Reactors and Related Fuel Cycles(1991)
[7] Hattori S , Minato A ,"Passive Safety Features in 4S Plant", 2nd ASME-JSME Int Conf on Nuclear Engineering-1993, S F , California, March 21-24, 1993

GENERAL OVERVIEW OF NUCLEAR ACTIVITIES IN MOROCCO

K. KAROUANI
CNESTEN, Agdal,
Rabat, Morocco

XA9846707

Abstract

Nuclear activities have been introduced in Morocco since the early seventies. These activities concern the utilization of nuclear techniques in medicine, food and agriculture as well as training and research in nuclear physics. In 1984, Morocco decided to undertake a technical and economic feasibility study as well as the site study of the first nuclear power plant. Two years after, he decided to create the "Centre National de l'Energie des Sciences et des Techniques Nucléaires" as a technical and research support for the nuclear power program and as a promoting institute of nuclear techniques. Obviously, he also decided to set up a regulatory framework.

1. Training and research in nuclear physics

Morocco, being conscious of the long lead-times involved in developing qualified manpower, started teaching basic and applied nuclear physics for two years during the undergraduate studies in the faculty of Sciences, Rabat, since 1968 and later on in other faculties.

To improve the graduate students background, post graduate nuclear studies have been set up in Morocco and/or students sent abroad. Included in these post graduate studies, there are some research works in fundamental nuclear physics, reactor physics, radiochemistry, etc. which were published in scientific journals.

2. Medical applications

For the benefit of his citizens, Morocco introduced the nuclear techniques for the medical applications, especially for radio-diagnosis using Tc-99m and I-131, radioimmunoassay (RIA) using I-125, radiotherapy by the use of Co-60 sources or linear accelerator, curitherapy using Cs-137 and Ir-192, and for some biology investigations.

These techniques are now utilized in about eight hospitals and medical units, located in the majority in Rabat and Casablanca cities.

3. Food and agriculture

The nuclear techniques are also applied in the field of food and agriculture. These concern:

* radiation induced mutation with the aim of improving the field, quality and resistance
  of cereals and other plants to drought,

* investigations on optimal conditions of applying, fertilizers to sugar beet and to study

soil nitrogen supply,

* improvement livestock breeding through hormone dosing and monitoring the major
diseases affecting livestock,
* assessment of nitrogen fixation in various species of trees in order to develop methods for improving fertility of marginal soil,
* food irradiation processing (a pilot food irradiation facility is being built in Tangier city).

4. Industry application

The industry in Morocco has been using some nuclear techniques in:

* non destructive testing using X Ray and gammaradiography,
* sugar factories, cement factories, phosphate industry, mines, hydrology using radiometric gauges.

5. Nuclear power plant project

In 1984, Morocco undertook a site and technical and economic studies of the first nuclear power plant which were completed in 1994. The site study allowed to qualify one site among seven potential sites that were considered at the beginning. The technical and economical study covered the three commercially approved, reactor systems (PWR, BWR and CANDU). This study concluded that the first commercial nuclear power plant in Morocco could be built starting from year 2010.

6. Promotion of nuclear techniques and support for nuclear power program

Although nuclear techniques were introduced in the country since the early seventies, their utilization remained limited to a few applications. To widen their applications, Morocco created in 1986 the "Centre National de I'Energie, des Sciences et des Techniques Nucléaires (CNESTEN)" which has the responsibility to promote the nuclear techniques in the divers social and economic sectors of the country. Besides this, CNESTEN has also the role of technical assistance to the national nuclear power program in the matter of site selection, manpower training, technology transfer, nuclear safety and choice of reactor system. To cope with his missions, CNESTEN is building its first nuclear research center which includes a TRIGA Mark II reactor of 2 MW thermal power and some laboratories such as radioisotope production laboratory, radioactive waste processing facility, nuclear techniques laboratory, safety and radiation protection laboratory, and electronic and mechanical workshops.

7. Regulatory framework

Nuclear regulation constitutes an essential component of a national nuclear program. Thus Morocco designated a regulatory body which set up a regulatory framework for nuclear activities. Hence some decrees have been or are being promulgated. These concern:

* setting up of a National Council for Nuclear Energy (CNEN),
* authorization and control of nuclear facilities,
* protection against ionizing radiation,
* radioactive material transport,
* physical protection of nuclear materials,

* civil responsibility in case of nuclear damage,
* emergency planning.

8. Seawater desalination feasibility study

Being conscious of the water shortage he will face in the coming years, Morocco took part of the feasibility study of seawater desalination using nuclear energy was carried out from 1991 to 1995 for the north African countries in collaboration with the IAEA's experts, which covered the following aspects.

* geography and demography,
* water resources and demand,
* energy resources and demand analysis,
* site selection,
* overview of desalination processes,
* desalination units and nuclear reactors coupling,
* local participation.

The results of this study are summarized in the following reference "Nuclear desalination as a source of low cost potable water in North Africa. SP. (IAEA Report) Draft, January 1995, Regional Meetings: Egypt, Morocco, Algeria, Tunisia, Austria.

NEXT PAGE(S) left BLANK

STATUS AND POTENTIAL OF SMALL & MEDIUM POWER REACTORS IN PAKISTAN

XA9846708

P. BUTT, M. AHMAD
Pakistan Atomic Energy Commission,
Islamabad,
Pakistan

Abstract

Pakistan's interest in nuclear technology dates back to the late 1950s, when Pakistan Atomic Energy Commission (PAEC) was established with the responsibility of promoting peaceful uses of nuclear technology for the development of national economy. A beginning was made in the field of nuclear power generation by commissioning the 137 MW Karachi Nuclear Power Plant (KANUPP) in 1971. In addition to KANUPP, the other activities during this period included studies for nuclear power & sea water desalination in the coastal areas. This was followed by a series of feasibility and long term planning studies (1969-1981) to study and firm up the prospects of nuclear power in the country, using the ever-improving analytical tools and data base. An Energy & Nuclear Power Planning Study for Pakistan has just been completed as a technical cooperation project of the IAEA to provide a sound basis for the formulation of the strategy for future development of nuclear power. The successful functioning of KANUPP and the 300 MW NPP (CHASNUPP) presently being built by China have given the country great confidence and a sense of direction to plan more nuclear units in future, in the S&MR range, in a manner that would progressively lead to a high degree of self-reliance. A large number of studies have been carried out in the past all over the world to investigate the possibility of using nuclear generated heat for the purpose of desalting seawater. It is now proposed that IAEA may initiate a programme to install an International Demonstration Nuclear Desalination Plant (IDNDP) in some developing country to provide hands on experience in this field.

1. INTRODUCTION

Although the energy and electricity demand in Pakistan has been growing quite rapidly in recent decades yet the present levels of per capita energy consumption (0.28 TOE) and per capita electricity consumption (300 kWh) are low compared to the world norms. Electricity is available to only about 57% of the country's population and due to shortage of installed generation capacity load shedding has been a frequent practice in recent years.

In order to eliminate load shedding the Government of Pakistan has been encouraging private sector to invest in power generation and for this purpose a package of incentives was announced in 1994 [1]. As a consequence of interest shown by the private sector it is expected that by year 1998 the installed capacity in the country would be at a level of about 18,000 MW (of which some 4,300 MW will be from the private sector) and the menace of load shedding will be eliminated. However, most of the capacity additions from private sector will be based on oil. As some 80% of the oil used in Pakistan is imported so the private sector power generation projects will enhance the vulnerability of the country to changes in the international price of oil. It is therefore imperative that in addition to exploiting indigenous energy resources for the power sector to the maximum feasible extent a gradual development of nuclear power should also be undertaken so as to reduce reliance on imported fuels and to contain increase in electricity generation costs in the long run.

Nuclear power reactors can also be used as a source of heat for desalination, district heating or process heat for various industries. As most of the areas of Pakistan are either semi-arid or arid and there is scarcity of water the potential for application of nuclear technology for desalination purposes is significant.

This paper gives an estimate of electricity generation capacity requirements upto the year 2020, reviews the water requirements and availability prospects in the coming years, discusses the market potential of small and medium power reactors (SMRs), and gives an overview of nuclear infrastructure in the country. At the end it is proposed that IAEA should launch a small dual purpose nuclear power plant project to be setup with international cooperation.

2. LONG TERM ELECTRICITY REQUIREMENTS & NEED FOR NUCLEAR POWER

During the recent ten years (1983-1993) electricity consumption in Pakistan increased at an average of about 9.9% p.a, while the economic growth rate was 5.5% p.a. Energy and Nuclear Power Planning Study for Pakistan, which has been completed recently with the technical assistance of IAEA [2] and background studies carried out by the Energy Wing of the Planning Commission for the Eighth Five Year Plan (1993-98) have shown that for 6.5% to 7.0% p.a growth rate envisaged for the economy in the next 2-3 decades the electricity requirements will grow at about 8-9% p.a. The projected power generation capacity requirements in the Reference scenario of Ref. 2 are: 16,265 MW in the year 2000; 35,150 MW in the year 2010; and 78,525 MW in the year 2020.

The proven fossil fuel reserves of Pakistan are: oil 27 million tons of oil equivalent (MTOE), gas 407 MTOE and coal 481 MTOE (1.1 billion tons), and the 1993-94 annual production levels were: oil 2.8 MTOE, gas 13.1 MTOE and coal 1.6 MTOE. Oil is the most scarce of all domestic energy resources and nearly 80% of the present oil requirements are met from imports and oil import bill in recent years has been about US $ 1.5 billion per year. Gas resources though significant, have to be shared between power and other sectors i.e. fertilizer, general industries, households and commercial sectors. At present some 36% of gas production is being used by the power sector and it is believed that this share can not be increased significantly in future from domestic supplies alone. Coal reserves are significant in quantity but are generally poor in quality. At present nearly all of the coal produced in the country is being used in brick kilns but through appropriate combustion or control technologies it can be used in power generation. In addition to the above mentioned proven reserves considerable additional potential resources of fossil fuels are also present. In the case of oil and gas it is expected that by gradually enhancing the petroleum exploration level from 18-20 exploration wells per year in recent years to about 60 wells per year in the next three decades, significant new reserves may be discovered which ma\y be able to support about four times the current production levels of oil and gas. The recently discovered Thar coal field in the southern part of the country has very large resources. These account for 95% coal resources of the country (185 billion tons) and have the potential to support some 10,000 MW of coal-fired power generation capacity in the next 2-3 decades. However, this would imply large investments in determining the quality of coal, determining the extent of economically minable coal resources and development of mines.

Pakistan has some 30,000 MW of hydro power potential and essentially all of it is located in the north of the country. At present some 4,826 MW hydro capacity is installed and 1,634 MW (Ghazi Barotha 1450 MW and Chashma 184 MW) is under construction. At present development of hydro power-cum irrigation water storage projects is an attractive option for capacity expansion but due to various socio-political difficulties and site related constraints there has been no progress in the development of these projects in recent years. Nevertheless, it is expected that it may be possible to overcome these difficulties in the coming years and to construct two additional large hydro plants (possibly Kalabagh and Basha) with some 6,000 MW installed capacity in the next two to three decades.

Even with these above mentioned, favourable assumptions for the development of indigenous fossil and hydro resources there will still be a large gap between capacity which can be based on indigenous energy sources and the capacity required. It is envisaged that indigenous energy resources may, at best, be stretched to support some 35,000 MW capacity (hydro: 14,500 MW, domestic gas: 12,000 MW and domestic coal 8,500 MW) by the year 2020 against the requirements of 78,525 MW. So nearly 43,500 MW capacity will have to be based on imported oil, imported coal, imported gas, and nuclear power. Table 1 gives the electricity generation mix in 1995 and that for the years 2000, 2010 and 2020 developed on the basis of a least-cost expansion plan [2]. It may be noted that even with the addition of nuclear power plants there will be a heavy reliance on imported energy sources for power generation.

TABLE-I PRESENT & PROJECTED SHARES OF GENERATING CAPACITY, MW

	1995	**2000**	**2010**	**2020**
HYDRO	4826	5068	11150	14500
GAS	3973	4600	11900	27520
OIL	3484	5273	5100	11630
COAL	115	862	2950	13750
NUCLEAR	137*	462*	4050	11125
TOTAL	12535	16265	35150	78525

* *Existing/under Construction*

3. FUTURE WATER REQUIREMENT & AVAILABILITY

3.1. Problems Associated with Irrigated Agriculture and Projected Water Shortfall

In arid and semi-arid regions in which Pakistan lies, water is a scarce commodity. Its optimal utilization is essential to ensure public welfare and prosperity.

Agriculture is an important sector of economy in Pakistan contributing about 25% to GDP [3]. The bulk of Pakistan's agricultural production (about 78%) comes from irrigated land. These irrigated planes of the Indus basin are underlain by an extensive ground water aquifer of varying water quality.

The present irrigation system comprises the Indus river and its major tributaries, three major reservoirs of about 18.5 billion cubic metre (Bm^3) of conservation storage, 23 barrages, headworks & syphons, 12 inter-river links and 48 canal commands. The total length of canals is about 57200 km, with water courses, field channels and field ditches running another 1.6 million km [4,5].

On average about 169 Bm^3 of river water flow into Pakistan per annum [5], annual flow variation is in the range of about 124-230 Bm^3 and more than 80% of this flow occurs during summer season. Approximately 125 Bm^3 of surface water is diverted annually into the canal system. In addition, about 25000 public and about 350000 private tube wells pump annually 65 Bm^3 of ground water for irrigation [5]. This system supplies water to about 17 million hectares [5].

The main purpose of irrigation is to benefit, in the long term, the economy, the society and the environment. However at present, a serious lack of balance in water and salt exists because, nearly two-thirds of the total annual withdrawal of irrigation water from the rivers by the canal systems are estimated to be lost during conveyance. This results in the rise of water table, water-logging and salinity. Ever since the introduction of irrigation in 1800s, the salinisation of soil and of ground water in the Indus basin has been continuously on the increase. If the disposal of salt remains inadequate the salt contents of both soil and ground water could eventually increase to an intolerable level.

The future water requirement and availability have been estimated by the Planning Commission [5]. According to this study, the estimated overall requirement for all uses would be about 184 Bm^3 and 266 Bm^3 by the years 2000 and 2013, whilst water availability for the same years would be about 134.2 Bm^3 and 132.5 Bm^3 respectively. There would therefore, be a substantial shortfall of supply in the future.

Stated otherwise, the per capita annual water availability at the time of independence (1947) was about 5000 m^3. Pakistan has now reached the critical value of 1000 m^3 per person, below which a country is regarded as a water scarcity zone. The available water resource will go down further to about 900 m^3 /capita by the year 2000 and to less than 700 m^3 /capita by 2013, with increasing competition between irrigated agriculture, industrial and domestic water users [5].

3.2. Possibility of Desalination

Several preliminary studies were carried out by Pakistan Atomic Energy Commission (PAEC) in the 1965-74 period which had indicated that the arid coastal belt of Pakistan spanning about 700 km from Karachi to the Iranian border offered bright prospects for industrial and agricultural development, *provided* water, power and communication facilities were made available. Several missions from the IAEA and USA had also visited Pakistan and recommended setting up of dual purpose power-cum-desalination plants at some of the coastal settlements.

Since the arid areas are sparsely populated for places along the Makran coast other than Karachi, the water and power requirements were essentially small, which could be adequately met by means of conventional power plants, or even a floating-ship borne plant, which was also considered. PAEC had set up a solar desalination plant of about 33 m^3 per day average capacity in 1972 at Gwadar which has had a considerable impact on the area.

These earlier studies had also emphasized large dual purpose plants to supply desalted sea water to Karachi. A US group of experts in the field of desalination again visited Pakistan in 1974 and alongwith their Pakistani counterparts further investigated the scope of desalination. Some of the major recommendations are summarized hereunder [6]:

- For the development of Gwadar on the Makran coast as a major fishing centre, a near term opportunity for using desalted water exists. A diesel plant alongwith the waste heat recovery boilers and a 900 m^3 per day multistage flash (MSF) process desalination plant was recommended for immediate implementation.

- A second major action recommendation was to develop a technology in Pakistan to reclaim salted overlands and to prevent salt build-up in good lands. This would be done by pumping and concentration of saline ground water and reinjection of the concentrate into deep aquifers where it would remain.

Need for desalination plants exists in the country and is described in section 4.1.

4. SMR MARKET POTENTIAL

4.1. Domestic Market Potential

The Energy and Nuclear Power Planning Study for Pakistan [2] has not considered a unit size larger than 600 MW for candidate fossil fuel and nuclear power plants. The least cost solution for expansion of electricity generation system predicts the installation of 18 x 600 MW nuclear power units upto 2020, both with the electricity demand projected as per reference demand scenario and when certain feasible conservation measures were considered as per the energy efficiency scenario.

The geographic distribution of this nuclear capacity addition will require specific engineering studies in due course; however at this time the following capacity allocation would seem in order:

- A 600 MW unit every alternate year in the North at CHASNUPP site during 2003-2019, with a unit coming up in 2020 as well (C_2 to C_{11} i.e. 10 units).

- A similar schedule in the coastal area/southern zone during 2008-2018, followed by a unit in each of the next two years (S_1 to S_8 i.e. 8 units).

Nuclear power-cum-seawater-desalination plants could be considered for Karachi Metropolitan Region and for establishing new townships along the coast or for building new seaports alongwith industrial/export zones.

A reasonable schedule for developing the desalted water supply in the coastal/southern region would be to couple a desalination plant of 0.45 million m^3 (100 million Imperial gallons) per day size with each nuclear power plant of 600 MW size assigned to that region (total 800 MIGD). Generally speaking, plants for Karachi area could be located on the sea coast west of Karachi. Water produced would be conveyed by pipelines to Karachi & then blended with river water for distribution. However, if in the near future a smaller size (50-100 MWe) power-cum-desalination nuclear plant is available from the well known suppliers then Pakistan would like to consider setting up such a plant at one of its upcoming coastal towns like Gwadar, Pasni or Ormara.

4.2. Future of Nuclear Energy & World-wide Market Potential of SMRs

Nuclear energy has come of age with its share in total electricity generation of 17% and 7200 reactor-years of accumulated experience by end 1994 [8]. As of December 1995, there were 431 nuclear power plants in operation with total cumulative net power output of about 342.6 GW, and 495 plants with about 394.8 GW after including units under construction or on order [9]. The rate of growth at present is 4-6 GW/annum and is expected to grow at about 5-10 GW/annum or higher beyond 2000 AD [8].

The projected nuclear growth in nuclear electricity generation capacity in typical regions of the world is as follows [10]:

	CHANGE IN NUCLEAR CAPACITY BETWEEN 1990 AND 2010	
USA	- 3500	MW
EUROPE	+ 28900	MW
CHINA	+ 25000	MW
RUSSIAN FEDERATION	+ 4600	MW
INDIA	+ 2800	MW
REST OF THE WORLD	+ 50400	MW

The future potential of SMRs world-wide can be very roughly predicted from their share in the past and after recourse to the future programme of the developing economies. As of 1980, about fifty percent of all power reactors in operation were in the SMR range (upto 600 MWe), whilst out of those coming on line during 1981-90, 1991-2000 & beyond 2000, the respective proportion is or will be 20%, 30% & 44%. Assuming a conservative figure of 25% and based on potential growth of nuclear power beyond 2000 stated above, the expected share of SMRs in the developing region would be about 1250-2500 MW/annum or higher. However, as SMRs in the range of 300-600 MWe are not very attractive for countries with relatively large grids, the only parts of the world where SMRs and multi-purpose nuclear power plants could be attractive are arid or coastal regions of countries of North Africa, Asia, and single purpose nuclear power plants for the countries with smaller grids in South & East Asia.

The need for building water desalination plants using the nuclear generated heat, for developing countries, has been discussed and recognised for a long time but no such plant has been built so far. Now it will be appropriate if IAEA considers a proposal to undertake a project for the establishment of an International Demonstration Nuclear Desalination Plant (IDNDP) in some developing country. The Member States may contribute by funding or in kind by providing equipment, components and material etc. on a subsidised cost basis in addition to extending their support for necessary R & D and other related work.

5. NATIONAL PARTICIPATION

5.1. Policy

Pakistan aims at gradual indigenization of its nuclear power programme to the optimum level in order to reduce over-dependence on imported plant and fuel, conserve the precious foreign exchange component and lower overall plant cost, while raising the level of nation's industrial and technological base.

5.2. Strategy

KANUPP was established under a tri-partite agreement amongst the governments of Pakistan and Canada and the IAEA. But the Canadian government unilaterally withdrew its support in 1976. To cope with this sudden break in the supplies of nuclear fuel, heavy water, spare parts and technical information for KANUPP, PAEC had to strive hard to meet the corresponding requirements through indigenous effort. In this way the plant has all along been kept in operation, though at reduced output, and a lot of valuable technical and industrial experience has been gained in the process.

Being basically short of conventional energy resources, Pakistan is keen to make increasingly large use of nuclear power to meet its future electricity requirements. In order to achieve this objective in a manner that would gradually lead to a high degree of self-reliance, PAEC is pursuing simultaneously two plans encompassing short term and long term time horizons respectively.

The *short term* plan envisages construction of nuclear power plants with foreign assistance as quickly as possible with a view to alleviate power shortages. It is planned to purchase, when the national economy allows, proven type of commercially available plants of standard design at reasonable financing terms, ensuring full participation of PAEC and local industry for maximizing transfer of technology. With increase in local capability for design & engineering, construction and manufacturing it is intended to shift gradually from a turn-key or two-package approach to multiple package contracts for subsequent plants.

The *long term* plan aims at systematically developing local capability, in close co-operation with supplier countries, leading progressively to increasing indigenous design, engineering and manufacture of nuclear power plants together with their components and fuel.

5.3. Achievements

5.3.1. *Nuclear Power Planning*

Nuclear power planning activities have been pursued in the PAEC since early 1960s. These activities have steadily evolved in keeping with the PAEC's programme for nuclear power. Briefly speaking, the activities in the 1960s (Phase Zero) pertained to very preliminary studies, which were carried out with the help of internationally published data; whilst the objectives during the 1970-76 period (Phase one) had been to obtain know-how by way of analytical techniques covering such aspects as nuclear power planning, pre-project analysis, etc. In the consolidation phase (Phase Two, 1976-82) the efforts were directed towards consolidating the in-house capability and expanding it into highly detailed analyses.

5.3.2. *Design & Engineering*

The design and engineering phase of nuclear power project development was initiated at CHASNUPP with the hiring of SENER of Spain as Architect Engineer in 1980. Later in 1984, the efforts were expanded with the assistance of Belgian architect engineering company, BELGATOM. A formal Design & Engineering Department was set up within CHASNUPP in September 1985, by regrouping already existing divisions.

Existence of such a group is necessary for design review, PSAR preparation & review and in developing architect engineering capability. It can also assist in pre-project activities. Over 100 engineers already trained in Europe and China are participating in the design and construction of CHASNUPP.

5.3.3. *Safety & Quality Assurance*

PAEC engineers have gained useful experience from KANUPP which has been operating safely for more than two decades, whilst Directorate of Nuclear Safety and Radiation Protection (DNSRP) has already been established as an independent regulatory body to supervise all aspects of nuclear safety. Further, PAEC has also established a Centre for Non-Destructive Testing and Pakistan Welding Institute at Islamabad. These organisations are meeting the critical requirement of trained and certified manpower in these areas. More than 500 experts from the public as well as the private sector have been trained at these facilities.

5.3.4. *Trained Manpower*

National universities as yet do not have a full-fledged program in the field of nuclear engineering. PAEC has established special institutes at PINSTECH near Islamabad and KANUPP near Karachi for producing trained manpower to support its nuclear power programme and to operate and maintain the plant.

5.3.5. *Construction*

Local industry is carrying out nearly all the civil works at Chashma excluding the nuclear island and conventional island. In future plants, civil works will be mostly carried out by local industry with sizable contribution in installation activities as well.

5.3.6. *Equipement Manufacture*

PAEC has already gained extensive experience in the local development and manufacture of spares and replacement of equipment for KANUPP. It is supplying some simpler vessels, heat exchangers, etc. to Chinese main contractor for CHASNUPP.

5.4. National Infrastructure

Some local manufacturing capability exists in the public and private sectors for the manufacture of thermal power plant boiler components, heat exchangers and electrical equipment.

In the PAEC, modest efforts have been made to look after instrumentation and control, materials, nuclear fuel cycle facilities and development of infrastructure through interaction with public and private sectors.

5.5. Transfer of Technology

The 300 MW CHASNUPP plant is being constructed on turnkey basis. The contract covers some aspects of transfer of technology, which include, design information (i.e. methodology and relevant R&D test information); design participation and training; equipment manufacturing within the scope of prime supplier; civil engineering and installation; participation and training in construction and installation at plant site; and commissioning and technical information exchange.

As experienced elsewhere, transfer of technology has to be promoted through a centralized organization for best results. Its success also depends directly on adequate & specialized manpower and financial resources. International political climate is also an influencing factor.

5.6. Nuclear Fuel Cycle

A nuclear power programme requires an assured supply of nuclear fuel (uranium) to run the reactors and facilities to refine and fabricate this fuel. PAEC is engaged in R&D covering different aspects of the nuclear fuel cycle and initiated a modest prospecting programme in early 1960s. This involved radiometric surveys, geochemical measurements, geological/structural mapping and geomorphological investigations in various parts of the country. A number of promising areas were located some of which are presently being explored.

Uranium ore has been mined in the D.G. Khan area and the first ore processing plant using this indigenous ore has been in operation for some time. Essential laboratory facilities have also been installed to support the exploration and ore process development work.

Under an agreement the first core and a few reloads for CHASNUPP will be purchased from China, the manufacturing of subsequent reloads can be done in Pakistan under license.

5.7. Radioactive Waste Management

Appropriate radioactive waste management systems have been designed for KANUPP and CHASNUPP to remove radioactive liquid, gaseous and solid wastes arising from the plant. These radwaste management systems collect, store, allow sufficient radioactive decay and process the wastes through filtration, ion-exchange, evaporation, solidification, vitrification and drumming.

As the liquid wastes are discharged after dilution & monitoring, the chances of sea/river water contamination from liquid wastes are practically nil. The releases during normal operation are/will be orders of magnitude below the permissible limits prescribed by IAEA and will not cause any harm to public health and safety. There is no possibility of liquid activity to be released outside the Plant in case of emergency as it is retained in the various safety barriers designed to cope with the accidents. Thus river water shall not be contaminated from any waste discharge from the plant.

Sufficient storage capacity is provided for the entire fuel discharged during the life of the plant. High level liquid wastes will be suitably located and converted into solid wastes. These solid wastes will be packaged in standard drums for storage and subsequent removal to offsite disposal facilities for permanent burial.

7. CONCLUSIONS/RECOMMENDATIONS

As a result of experience gained in the operation and maintenance of KANUPP, planning, contracting and participation in the design of the under construction nuclear plant at Chashma (CHASNUPP) and development of associated infrastructure over the years, Pakistan is now at stage where it is capable of sustaining a modest nuclear power programme.

Electricity generation capacity requirements in Pakistan are projected to increase from a level of 12,500 MW in 1995 to about 78,500 MW by the year 2020. Some 14% of the capacity in the year 2020 can be based on nuclear power plants i.e. SMRs (18 units of 600 MW each). In the near future if power-cum-desalination plants are available with a suitable financial package and not denied due to international embargoes, Pakistan would like to consider setting up a small 50-100 MWe plant at one of its up coming coastal cities like Gwadar, Pasni or Ormara.

Except for countries of Asia very limited power capacity is currently being planned worldwide. However, most of the Asian countries have relatively large grids and therefore, would not be very keen in the building of SMRs. Nevertheless, for countries with smaller grids or for remote or coastal areas of Asia and Africa the SMRs and multipurpose nuclear power plants would certainly be attractive. In this respect, it would be worth while that IAEA considers a proposal for a standard high temperature, high pressure test loop which should be made available to the Member states.

In the past, we have been discussing quite often about the possibility of constructing water desalination plants using nuclear generated heat. Now it will be appropriate if IAEA considers a proposal to undertake a project for establishment of an "International Demonstration Nuclear Desalination Plant (IDNDP)" in some developing country. This proposal is similar to the international cooperation projects already underway such as the International Thermonuclear Experimental Reactor (ITER) being undertaken as a four party venture-comprising the European Economic Community, Japan, Russia and the US-under the auspices of the IAEA [11]. A similar Russian proposal for the Asian Foundation for Fusion Research is another example of regional cooperative effort.

REFERENCES

[1] Policy Framework and Package of Incentives for Private Sector Power Generation Projects in Pakistan, Government of Pakistan (1994).

[2] Energy and Nuclear Power Planning Study for Pakistan, Final Draft Report, ASAG, PAEC, IAEA TC Project No. PAK/0/006 (1996).

[3] Economic Survey, 1995-96, Government of Pakistan.

[4] Masihuddin, M., "Technology in the Service of Mankind", National Seminar on Water Logging & Drainage Control, Tando Jam, Pakistan (1987).

[5] Mohtadullah, K., et al, "Water Management is Key to the Sustainability of Irrigated Agriculture in the Indus Basin", Annual Sustainable Development Conference, Islamabad (1996).

[6] Report of the U.S. Team of Consultants on "Desalination Programme for Pakistan", a mission sponsored by USAid, Islamabad, Pakistan, May 1974.
[7] Water & Power Development Authority, Pakistan, Annual Report, 1993-94.
[8] Juhn, P.E., "Nuclear Power Status & Prospects in the World", 20th International Nathiagali Summer College, 1995.
[9] Nuclear News, World List of Nuclear Power Plants, March 1996.
[10] Nucleonic Week, May 11, 1995.
[11] J. Sheffield and J. Galambos, "Prospects for Tokamak Fusion Reactors", in PS/SRD 2.3, Non-Fossil Energy Technologies. 16th Congress of the World Energy Council, Tokyo, Japan, Oct. 8-13, 1995.

PART II

LESSONS LEARNED AND TECHNOLOGY TRANSFER

XA9846709

THE CAREM REACTOR: BRIDGING THE GAP TO NUCLEAR POWER GENERATION

J.P. ORDOÑEZ
Carem Project, INVAP,
Argentina

Abstract

An idea is presented as an alternative for the introduction of nuclear power in presently non-nuclear countries. This idea involves going through an intermediate step between the traditional research reactor and the first commercial nuclear power plant. This intermediate step would consist of a very small nuclear power plant, with the principal goal of gaining in experience in the country on all the processes involved in introducing commercial nuclear generation.

1. THE NEED FOR NUCLEAR POWER

All estimates of the future world-wide energy need indicate that the demand for electric power will continue to increase strongly, especially in the developing nations.

Looking at the means to satisfy this demand, it becomes quite clear that while in the near future, in most developing countries, requirements might be met by conventional methods (mainly fossil fuels and hydroelectric power) in a not so far future the use of nuclear power to generate electricity will become necessary. According to the projections, the rate of increase of the demand for electric power in developing countries will make the availability of nuclear plants mandatory around 2010.

At present, in many countries, public opinion is still quite strongly set against nuclear power projects. However, it is increasingly realised that the use of nuclear power cannot be avoided, and is indeed desirable. Nuclear power is in fact one of the least contaminating ways of generating electricity. It can be assumed that before 2010, public opinion will have recognised that the generation of electricity by nuclear means is the one form of generation of electric power that takes the best possible care of the environment. The final disposal of nuclear waste is the real pending technological problem which motivates public resistance to nuclear power, and it is most likely that by 2010 satisfactory solutions will have been implemented for dealing with radioactive waste.

The increase both in energy demand and in public acceptance of nuclear technology will lead to a revival of nuclear power in the second decade of next century, in developing countries as well as in industrialised nations.

The latter are not unaware of this situation, and their governments sustain a nuclear policy by which they can guarantee a prompt response to the challenges that this second nuclear era might pose.

Developing countries, where the main demand for nuclear power for the generation of electricity will occur, should prepare in time for this future. The concretion of any energy plan requires long time spans. The complexity of nuclear plants only emphasises this need for foresight. Clearly it would not be prudent to wait till the year 2010 to broach the subject.

2. INTRODUCING NUCLEAR POWER

In the "first nuclear era", most developing countries followed a standard path to nuclear energy. This method involved three broad steps:

- At first the applications of radiation in industry and in medicine were approached partly for themselves and also to gain experience in the handling of radioactive materials.
- The second step involved the installation of a nuclear research centre, frequently around a research reactor facility.
- Finally, a commercial nuclear power plant was purchased on a turnkey basis, from one of the suppliers from the larger industrialised countries.

This process was frequently unsuccessful. Many countries covered the first two steps, and then faltered or failed along the way to the third stage. Many countries have been trying unsuccessfully for years to obtain a nuclear power reactor. Other countries managed to purchase their first turnkey power plant, but subsequently suffered a truly disastrous experience, putting the country's nuclear development into jeopardy.

It is our belief that such failure to achieve the third step stems from the fact that the technological and organisational distance between steps two and three is far greater than is commonly realised. It is almost impossible to jump over the chasm separating endeavours of altogether different orders of magnitude. The conditions to be met before a nuclear power reactor can be envisaged successfully are of many kinds and quite exacting:

- *Economical effort*: while a research reactor can be built using funds coming from country's budget, a commercial nuclear reactor necessarily entails the securing of financial assistance in the way of loans, which in many cases are difficult to obtain.
- *Industrial infrastructure:* most components for a research reactor can be built in practically any country. This is not true in the case of even minor components for a power reactor, which may require substantial industrial and technological infrastructure.
- *Human resources:* a research reactor requires qualified staff numbering a few dozens. A power plant needs hundreds, even thousands of qualified staff members involved in the project.
- *Licensing:* a research reactor involves practically no risk to the public. A commercial power plant implies safety studies of enormously greater scope and depth. Therefore the licensing of a research reactor does not by itself constitute a valid precedent for that of a power plant, nor does it necessarily provide the background significant enough to qualify a local regulatory authority for the licensing of a commercial nuclear power plant.

- *General Cultural Attitude:* while a research reactor generates an environment of study and research, a power plant requires an attitude centred on production.
- *Time scale:* As was mentioned above, the time for the new "nuclear age" will not be ripe for the next 15 years. Many countries have already possessed research reactors for many years, and many of the research centres and reactors are not very active because they are conceived as intermediates towards the construction of full-size power plants which are not to be built in the next future. In many cases the experience and expertise gained with these nuclear research laboratories and research reactors will have become lost, obsolete or irredeemable by the time they become necessary to meet the demand of nuclear power plants that is bound to occur.

3. BRIDGING THE GAP

This analysis leads us to propose an intermediate step between the second and the third of the classical stages towards full command of nuclear power as the best means to avoid many of the pitfalls and frustrations which are otherwise quite likely to occur. We call it the "bridge project". It consists of a small power reactor, which is able to generate electric power for local consumption, and which features all characteristics and also all difficulties of a full-size nuclear power plant.

Such a proposal is not such a novelty as it might seem at first glance, because it follows the same path designed by the developed countries on their own way toward the commercially mature nuclear generation of electricity. There is hardly a case in which a small reactor, similar in size to modem research reactors, was not followed by a small power reactor, in so-called "Nuclear Demonstration" facilities. Only after successful operation of such a plant, designed to provide electric power by nuclear generation, albeit on a small scale, did developed countries advance toward commercial power plants.

In the late sixties this intermediate stage came to be regarded as an unnecessary, and it was thought that countries that were newcomers to nuclear technology would be able to benefit directly from foreign experience. It seems now fair to say that this approach produced far more failures than successes. Profiting from the experience of others, it turns out to be most preferable to follow a slow and secure path toward success, rather than to attempt to reach a difficult achievement by means of short cuts. We consider this intermediate step to be mandatory. Today a relatively modest project can make a substantial contribution for any country to be in a position to cope successfully with the nuclear generation of electricity when the real need arises. The investment required for such an installation will be amply repaid in electric energy generated, and also through later savings on the full-size commercial power plants made possible by the experience gained operating the demonstration plant for some years.

To fulfil its purpose, the Bridge Project must feature the following characteristics:

- The electric power generated should be in the range of 100 to 500 MWth. This is a project of moderate size, to be faced with modest resources in all aspects: financial, technological, manpower and industrial infrastructure.
- Technologically it must be more complex than a research reactor but less complex that full-size commercial power station.
- The safety of the reactor must be based on simple principles in order to keep licensing, operation and maintenance at a minimum degree of complexity.
- The reactor must produce saleable services: heat, steam and especially electric energy.
- Drinkable water, produced from a seawater desalination plant, is also a potential output of the plant. This service has attracted interest from many countries.

The CAREM reactor, at present under development by the Argentine Atomic Energy Authority, CNEA, and INVAP SE., is designed to fill this gap between a research reactor and a full-scale commercial nuclear power reactor. It can be used for the domestic development of nuclear energy, and will be offered on the international market as a product fulfilling all of the conditions described above.

In order to asses the market for this type of reactor, a study on the introduction of nuclear energy in new countries was carried out. The idea is that those countries which are planning to do not use at present nuclear energy, but plan to introduce it at the beginning (first two decades) of the next century, are clear candidates for installing a Bridge Nuclear Power Plant before installing their first commercially competitive nuclear power plant.

For this study, the list of the two hundred and eight countries in the world was screened, and eighty six of them were selected as potential candidates for the introducing of nuclear energy. In a second step, each of these eighty six countries was studied with more detail, taking in account social, economical, energy sector and electricity sector information. The result of this study shows that there are not less than ten but no more than thirty countries which have a high probability of introducing nuclear power before the year 2015.

XA9846710

TECHNOLOGY TRANSFER: THE CANDU APPROACH

R.S. HART
AECL-CANDU,
Mississauga, Ontario, Canada

Abstract

The many and diverse technologies necessary for the design, construction licensing and operation of a nuclear power plant can be efficiently assimilated by a recipient country through an effective technology transfer program supported by the firm long term commitment of both the recipient country orgnizations and the supplier. AECL's experience with nuclear related technology transfer spans four decades and includes the construction and operation of CANDU plants in five countries and four continents. A sixth country will be added to this list with the start of construction of two CANDU 6 plants in China in early 1997. This background provides the basis for addressing the key factors in the successful transfer of nuclear technology, providing insights into the lessons learned and introducing a framework for success. This paper provides an overview of AECL experience relative to the important factors influencing technology transfer, and reviews specific country experiences.

1.0 OVERVIEW

The diverse technologies encompassed by the construction and operation of a nuclear power plant can be assimilated by a country through an effective technology transfer program supported by the firm commitment of both recipient and supplier. AECL's experience with nuclear-related technology transfer spans four decades and includes the construction of CANDU plants in five countries. This background provides the foundation for addressing the key factors in nuclear technology transfer; providing insights into the lessons learned and introducing a framework for success.

Key Factors in Successful Nuclear Technology Transfer

While the receiving nation, the owner of the power plants and the supplying nation, are all instrumental in creating a successful environment for technology transfer, each has different objectives. The technology that is transferred encompasses a wide range of skills and disciplines. To transfer the applications oriented technology ("know-how") along with the intellectual skills ("know-why") requires motivated people supported by a well-developed educational infrastructure that addresses their specific needs. A nuclear R&D organization is an asset since it provides good experience to staff, preparing them for the nuclear power program and helping them keep abreast of new technology. Acquiring nuclear technology is a long-term process. It is, therefore, advantageous to concentrate on those technologies that offer either significant business opportunity or have application in other sectors of the economy.

Lessons Learned

A number of factors contribute to successful technology transfer including

- the availability of well-educated and skilled workforce;
- an effective training program that is coupled with an opportunity to put learned skills into practice before they are forgotten;
- the selection of appropriate technology suited to the local manufacturing environment;
- establishing the importance of nuclear safety and quality within the infrastructure of the receiving nation;
- effectively managing conflicting objectives and being able to recognize the costs associated with on-the-job training.

Finally, it is important to precisely define the scope of the technology and the transfer process and establish an organization to coordinate the program.

A Framework for Success

A framework comprised of five essential components will facilitate the demanding task of transferring nuclear technology. First, there must be comprehensive planning to address objectives, realistically examine capabilities and most importantly identify any gaps. Second, experience shows that establishing a lead agency to coordinate the technology transfer enhances the chance of success. Third, a firm commitment of human and financial resources must be made within the scope of the technology transfer program. Fourth, due to its large scope and diversity the program requires firm management. Fifth, success largely depends on establishing a relationship between the parties that is based on trust and mutual respect.

2.0 Introduction

Not only is the transfer of technology a major feature of contemporary international trade but it is also a fact of life in the sale and acquisition of nuclear power plants. Because of the many facets of nuclear technology, planning and managing its acquisition is of great importance.

From the early beginnings of the development of the peaceful uses of nuclear power by only a few nations in the mid-1940's, there has been a considerable transfer of technology. Today, 34 countries have nuclear programs at various stages of development. Canada, one of the early leaders in the development of nuclear power, has experience with a wide range of technology transfer programs spanning four decades.

Canada, itself a country without a well-established, large-scale industrialized base at the start of its nuclear power program, has developed a autonomous nuclear industry. This has been accomplished through a consistent transfer of appropriate technology to industry and utilities.

In Canada, Atomic Energy of Canada Limited, Ontario Hydro, a provincial utility, and Canadian General Electric, a private sector manufacturing company, shared in the early development of the CANDU nuclear power system. The technology established, enhanced where necessary through cooperative exchanges and programs with other countries working in the field, was disseminated to existing Canadian industry. This "internal" technology transfer was essential to the establishment of a competent and self-sufficient nuclear industry.

As the nuclear program developed, the CANDU technology was passed to other utilities. Canada has participated or is participating in the construction of nuclear power plants in five countries including India, Pakistan, Korea, Argentina and Romania, and has built high-powered research reactors in India, Taiwan and Korea.

The extent and nature of the technology transferred has varied from country to country. Through agreements with many industrialized countries, for example, the UK, USA, Sweden, France, Italy and Japan, information and technology have been exchanged with mutual benefits. It is from this experience that AECL offers some comments on the factors which are important to the success of such programs.

3.0 Major Factors In Technology Transfer

3.1 Why Transfer Technology?

Figure 1 illustrates the major factors influencing the transfer of nuclear technology in terms of scope, process, needs and resources. The environment within which the process must operate and succeed is created by the three major participants: the receiving nation, the owner of the power plants which are to be built and the supplying nation.

The national government, particularly in a developing country, looks to technology transfer as a mechanism for industrial and economic development. Eventually, it becomes a means of assuring a high degree of autonomy and security in the provision of a major energy source. The owner of the plant is primarily interested in the supply of electricity from generating plants built to a short schedule and at as low a cost as is consistent with safe, economic and reliable operation. The supplier nation is usually prepared to transfer its technology. This will enhance its market opportunities and the sale of both nuclear power plants and associated technology and provide some return on its investment in research and development.

3.2 What Technology Is To Be Transferred?

Technology can be defined as the ability to do something. Technology, therefore, encompasses the technical and managerial "know-how" embodied in both physical and human resources. Nuclear technology encompasses a wide range of skills and disciplines and involves many sectors of the economy.

A broad-based nuclear industry would cover, for example,

- regulatory licensing;
- the complete fuel cycle, from uranium exploration through fuel fabrication to eventual disposal;
- engineering design and development;
- heavy water production, if required;

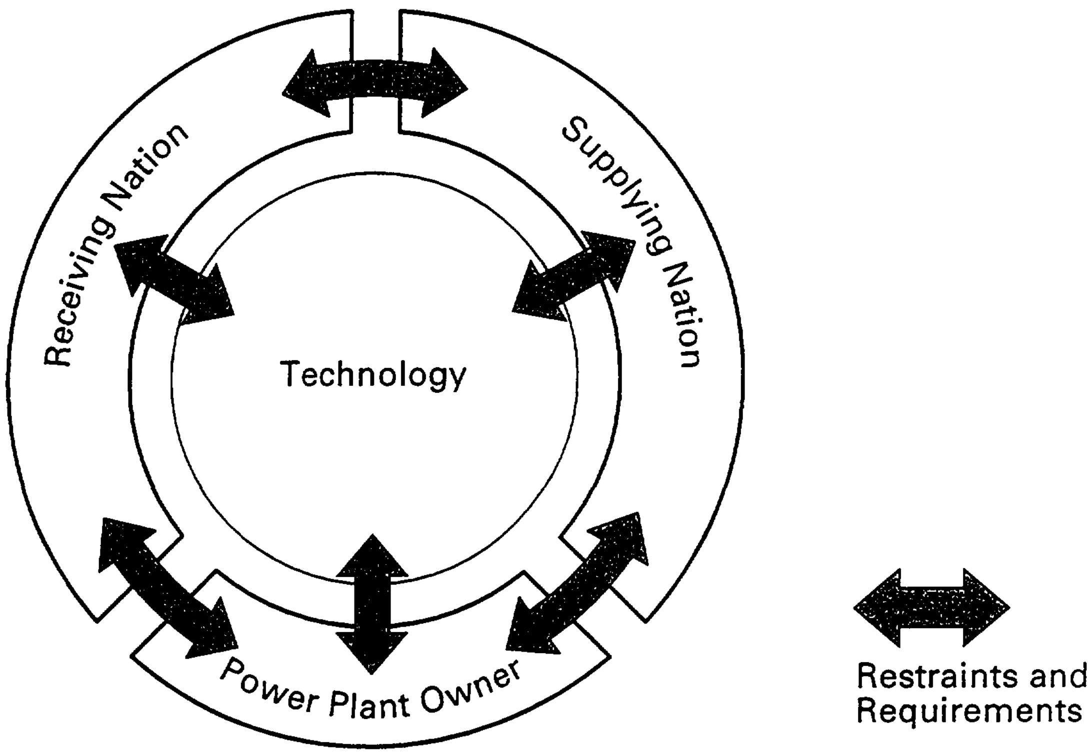


Figure 1 The Technology Transfer Environment

- construction technology;
- project and construction management;
- supporting research and development;
- component or equipment manufacture; and
- commissioning and operation of nuclear power plants.

Figure 2 illustrates the essential range of technology involved. The more application-oriented technology often termed "know-how" is associated with physical training in terms of manipulative skills. Where analytical capability related to decision-making and innovative thinking is involved and understanding of the reasoning behind the technology is necessary, the "know-why" is as essential as the "know-how". Here, the training is more intellectual in nature and is in effect an educational process.

The type of training necessary to assimilate the technology also determines the educational background required. The more physical training programs require less prior formal education whereas those individuals who will acquire the "know-why" will most likely have received university or technical college training.

3.3 What Are The Priorities?

Each nuclear power plant requires an investment of over a billion dollars and the human resources employed in the design, manufacture, construction and operation of the plant may well exceed 5000 at the peak.

The introduction of nuclear technology in any country requires the development of a sufficient number of trained personnel covering a wide range of disciplines at various levels. In addition, depending on the extent of the technology transfer envisaged, sufficient funds will also be necessary for investment in the required facilities.

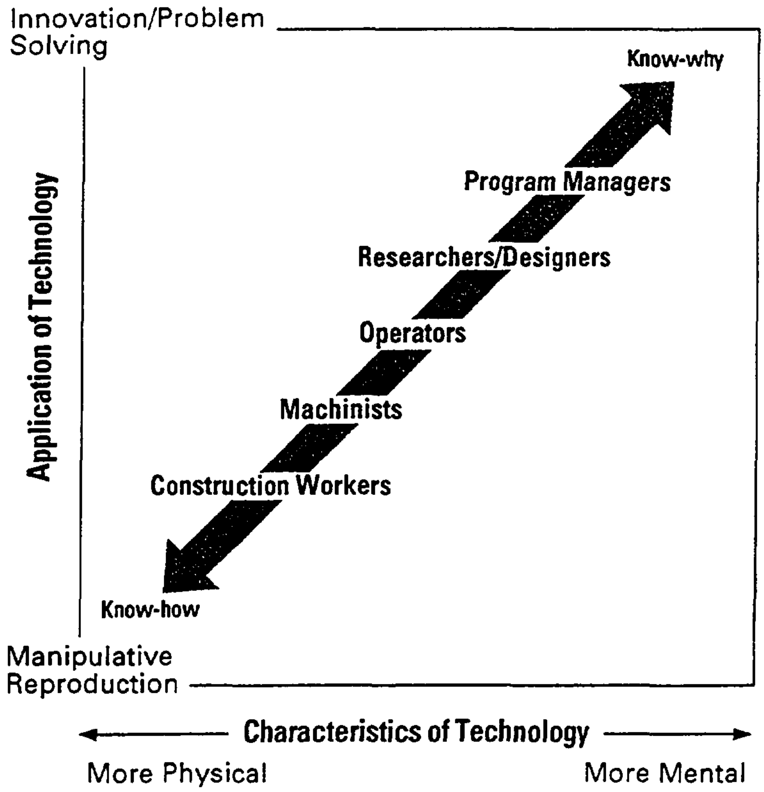


Figure 2 The Scope of Technology Transfer

The acquisition of the diverse nuclear technologies is a long term process spanning, in most cases, the construction of several nuclear plants. Clearly, it is advantageous to concentrate on those technologies for which there is significant business, even if only one or two plants are being built, or on those which also have application in other sectors of the economy.

There is some argument about the amount of technology which is essential to a nation for the introduction of its first nuclear plant. Clearly, some ability to license and regulate the industry is essential as is the ability to successfully operate and maintain the completed plant. Other areas of technology can then be introduced and developed at a pace and timing consistent with the nation's economic and industrial goals.

3.4 What Processes Are Involved?

It is essential that an appropriate infrastructure be in place to assimilate the technology. As a minimum an educational infrastructure will be required since the acquisition of even one plant requires high calibre professionals, technicians and skilled tradesmen to regulate, operate and maintain the plant. Even more trained people will be required as technology transfer programs are put in place because skilled people are the prime ingredient to their success.

In addition, it is Canada's experience that the prior existence of a nuclear research and development organization is an asset to the technology transfer process. Such an organization is most useful in providing preliminary experience to professional and technical staff for future participation in the nuclear power program. This was, in fact, one of the key factors in Canada's own nuclear power development.

Furthermore, nuclear technology, as with any high technology, is continually evolving. One of the most important means of keeping pace with future advances in the supplier's country and throughout the world is an effective and competent research and development staff.

The actual process of transferring the technology will involve a number of activities such as the:

- provision of documents, drawings and computer codes;

These represent the physical embodiment of any technology.

- provision of training, in class-rooms and on-the-job; and

Acquisition of documentation alone is analogous to buying engineering textbooks and expecting to be an engineer without attending college or completing the course work. It is more likely that only key individuals will be fully trained and those will then be expected to pass on the knowledge acquired to their compatriots.

- undertaking of cooperative programs in areas of mutual interest.

It is an effective way of providing the basic understanding behind the technology ("know-why") and keeping abreast of technological developments.

3.5 How Are The Needs Of The Individuals Met?

The effective transfer of technology is highly dependent on people and their needs, shaped by different backgrounds, different cultures and various career objectives, have to be considered.

Most of those involved, however, will have already acquired skills and capabilities which will influence their attitude towards training. Hence, technology transfer is basically an adult education process.

Adults generally look for the fastest and easiest way of learning new skills and quickly tire of processes that are not well matched to their individual needs.

3.6 How Are The Resource Needs Met?

The technology transfer program should be realistic. However, it is likely that the parties engaged in technology transfer have never gone through the process before. Therefore, it will be difficult to precisely identify the resources at the start of the program. Proper monitoring and assessment checkpoints must be built-in to allow for the reallocation of resources to meet changing program needs.

AECL has found that receiving nations assign high quality staff at the start of the technology transfer program who, on return to their country, are key figures in their nation's nuclear program. Inevitably, they are then promoted and have progressively less time to directly apply the technology they have acquired. This "turnover" of staff creates a need for an ongoing program with new human resources entering each year. The process, once started, becomes ongoing.

4.0 Some Specifics About AECL's Experience

4.1 CANDU Technologies

Figure 3 summarizes the basic technologies involved in the design, construction and operation of a CANDU nuclear power plant. It also serves to illustrate the different interests of the national government, the utility and local industry.

The government has overall interest in all facets of the technology and, in most cases, will determine the expected degree and timing of the technology transfer. In particular, the government will be responsible for regulation and safety licensing. The utility, on the other hand, needs no other technology than that required to successfully maintain and operate the plant, that is, the upper half of Figure 3. Local industry, whether engineering design companies, constructors or component manufacturers, is likely to be only concerned with the technology needed to do the job. For the most part they are interested only in the results of R&D and not in the basic R&D itself.

4.2 Country Specific Examples

CANDU In Korea

The development of Korea's nuclear power program provides a clear example of the time taken to develop a nuclear industry with a high degree of national participation or self-sufficiency. The Korean program, Figure 4, is extensive. Already 14 nuclear power plants, with a combined capacity of 13 GWe, are in operation or under construction.

The first plants, KORI-1 and Wolsong-1, were purchased on an essentially turn-key basis. In the case of Wolsong, the Korean regulatory authority began to develop its skills in licensing and regulation with the assistance of Canada's Atomic Energy Control Board. In addition, the Korea Electric Power Company (KEPCO) operations and maintenance staff were trained at AECL and Ontario Hydro facilities and received further training and a deeper understanding of the system through active participation in the commissioning of the plant.

The initial plants acquired under turnkey contracts, provided little opportunity for local participation beyond construction and the areas of regulation and operation mentioned earlier. However, through technology transfer programs, each succeeding group of nuclear power plants has provided and will continue to provide opportunities for increasing levels of participation. Even so, it is expected that some three decades will have passed before the target of near self-sufficiency can be achieved. The learning curve for each activity sector will be similar to those illustrated in Figure 4.

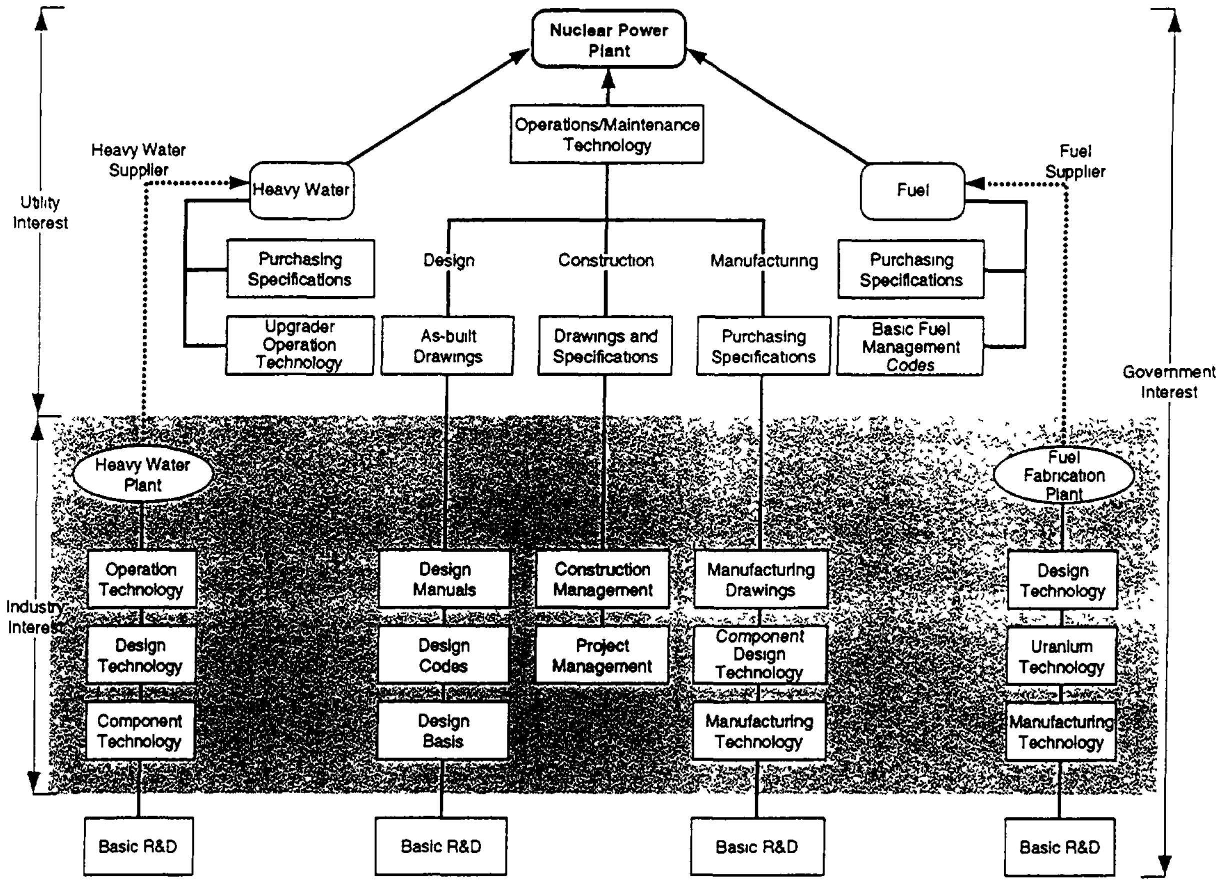


Figure 3 Summary of CANDU Nuclear Power Plant Technology

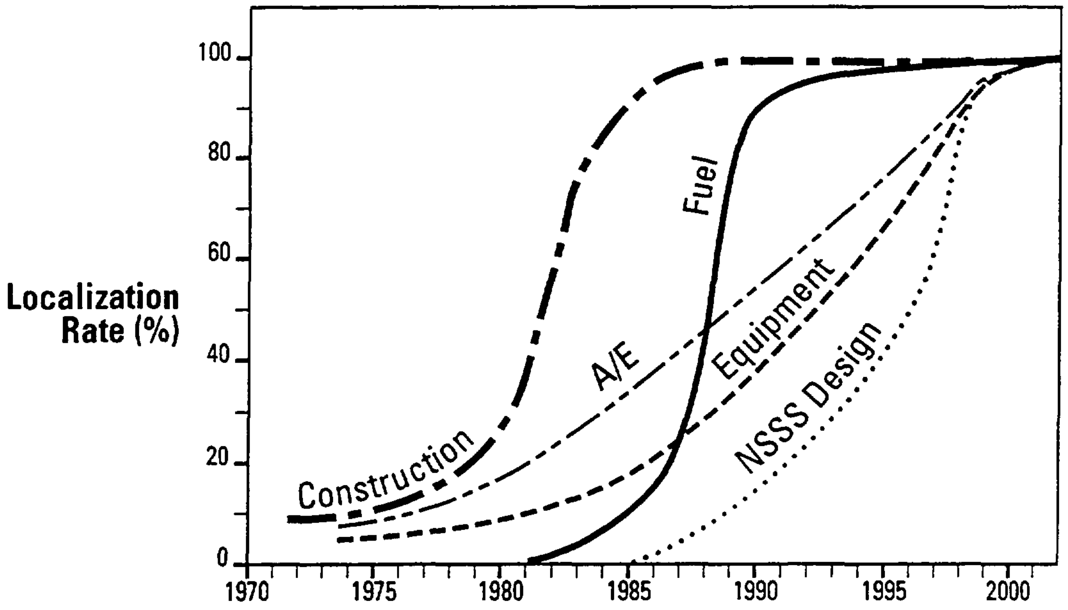


Figure 4 Korean Nuclear Program: Increasing Local Participation Over Time

Again using the Korean nuclear program as an example, the Korea Atomic Energy Research Institute (KAERI) has played a significant role in the Korean nuclear program with the acquisition of technology through cooperative agreements and its further development.

The following table identifies the increasing manufacturing localization for the Wolsong Units 3 and 4 project over what was achieved for the Wolsong Unit 2 project.

Major Nuclear Equipment Localization In Korea

Wolsong Unit 2	Wolsong Units 3 &4
• Major Heat Exchangers	• Major Heat Exchangers
• Major Tanks	• Major Tanks
• Pressurizer	• Pressurizer
• Degasser Condenser	• Degasser Condenser
• Reactivity Mechanism Deck	• Reactivity Mechanism Deck
• Steam Generator (partial)	• Steam Generator (more)
	• Calandria (W4)
	• Feeder Header Frame

CANDU In Romania

The Cernavoda site in Romania was developed for a five-unit CANDU station. Unit 1 is now in the final stages of commissioning. Figure 5 summarizes the progress in nuclear technology transfer in accordance with Romania/Canada contractual arrangements.

Many CANDU components were designed and manufactured in Romania by suppliers with no previous nuclear experience. Thus all Romanian components had to be certified for nuclear plant service in a stringent process of testing and verification.

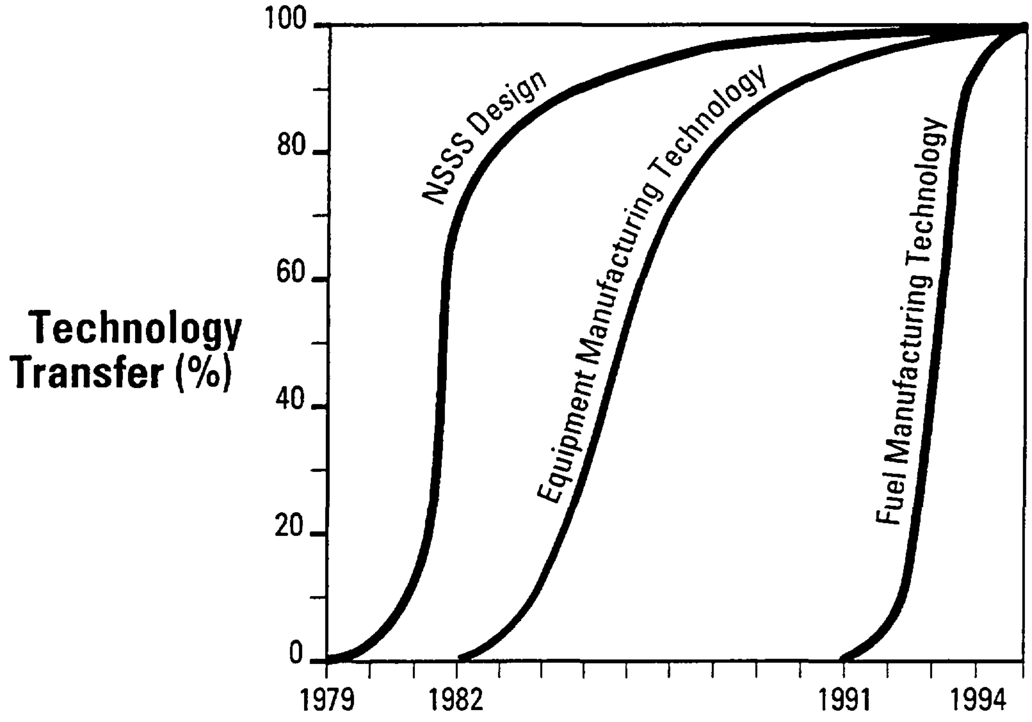


Figure 5 AECL Transfer of Technology to Romania

Small process equipment such as pumps, heat exchangers and valves went through certification in supplier facilities or in national laboratories. Other components such as 200 electric motors received final certification at the plant site.

The qualification of major equipment such as the Main Circulating Water Cooling Pumps and the Class III Stand-By Diesel Generators offered special challenges. Components with their characteristics and size had never been tested in Romania.

4.3 Lessons Learned

To date, AECL's experience in nuclear technology transfer has provided some insight into the factors that contribute to the success of technology transfer. Some of the lessons learned are:

- people are important;

Without the availability of trained personnel to interpret the documentation and implement the technology, the technology is of little practical value.

- training is essential;

Training is a necessary ingredient in technology transfer for not all the technology resides in documented form. Much of it can only be transferred through personal communication.

- practice is essential;

It is well recognized that on-the-job experience is one of the most effective ways of transferring technology. This, therefore, has been a cornerstone of all our technology transfer programs. However, if the technology once transferred is not put into practice but is put to one side to be addressed at a later date, the expertise that has been created will quickly dissipate and the technology will have to be relearned. This reinforces the argument that only those technologies of immediate interest should be considered.

- technology must be appropriate;

Experience indicates that adjustments in manufacturing techniques and equipment or in the skills demanded of labour will often be essential.

- there will be environmental differences;

The broader socio-economic-technological differences which influence the ways in which people behave and how work is achieved must be recognized. In nuclear technology transfer, the concepts of nuclear safety and quality have to be properly communicated and their importance clearly established within the nation's infrastructure.

- potential conflicts must be recognized;

It is likely that more than one objective will exist. For example, the use of local suppliers of equipment and services may appear to conflict with maintaining project schedules and overall costs. It is essential that both parties recognize the various objectives and reach an understanding regarding their relative priority.

- there may be cost impacts;

On-the-job training is an effective way to learn. However, this often extends the time taken to complete the job.

- misconceptions and misinterpretations will occur; and

Despite the best intentions of both parties, there will be misconceptions and misinterpretations about what is involved in or expected of the technology transfer. A very precise definition of the scope of technology to be supplied and the processes to be followed could minimize these problems.

- there is a need for a coordinating organization.

The reactor supplier does not control all the potential technology involved although it has access to it. Several companies, therefore, will be involved in the transfer of technology. In the receiving nation, many companies will be engaged each with its own interests and goals. The coordinating organization in the receiving nation will:

- ensure that the necessary infrastructure for providing adequately trained personnel is in place;
- determine priorities for the areas of technology to be transferred and ensure adequate allocation of funds and human resources;
- determine the most effective recipients who will receive and eventually develop the technology; and
- monitor and coordinate the actual technology transfer process.

5.0 A Framework For Success

Well over 100 billion US dollars have been spent in developing nuclear technology to its present state. Therefore, careful consideration should be given by all to building on the experience and expertise which exists in the world's nuclear community.

Experience has shown that the transfer of nuclear technology is a very demanding task requiring large commitments of both financial and human resources. Success in such a large undertaking can only be achieved if the right framework or environment is put into place. This framework will have several essential components:

<u>Comprehensive National Planning.</u> The receiver of the technology must carry out a thorough review of the objectives it sees for the technology transfer program, the present capabilities which can be applied to the development of a nuclear power program and more importantly the gaps which exist.

<u>Organize To Develop Infrastructure.</u> Experience shows that establishing a lead agency to coordinate the technology transfer enhances the chance of success. This agency will eventually put in place the complete infrastructure to support a nuclear power program.

<u>Commit Resources.</u> Within the scope of the technology transfer program decided upon, it is essential to make a firm commitment of both financial and human resources.

<u>Firm Management.</u> Because of the large scope and diversity technology transfer program in terms of resources, the program is a project in its own right which may equal the nuclear power plant project in terms of complexity and importance. Dedicated, effective and efficient project management by both receiver and supplier is imperative for program success.

<u>Develop Relationships.</u> The success of the program will depend on the will of both parties to succeed. In this respect, the development of relationships between parties which are based on mutual trust and respect is most important.

SETTING-UP OF SMR IN A DEVELOPING COUNTRY - INDIAN EXPERIENCE

C.N. BAPAT, P.D. SHARMA
Nuclear Power Corporation of India,
Anushaktinagar, India

Abstract

India envisaged a 3 stage Nuclear Power Programme on long term basis with a view to make use of large deposits of Thorium. The first stage involved Pressurized Heavy Water Reactors (PHWRs) based on natural uranium. First two PHWRs in 220 MWe range were set-up under collaboration with Canadians and subsequent were set-up and operated with totally developed indigenous technology. The designs for 220 MWe & 500 MWe PHWRs are ready. The size of reactors, coming under Small & Medium Reactor Categories, is ideal for a developing country from indigenisation of technology, synchronizing with grid and financing point of view. The paper gives Indian approach and experience gained in setting-up series of PHWRs in SMR range in India.

1. INTRODUCTION

The setting-up of Nuclear Power Plants (NPPs) in India is entrusted to Nuclear Power Corporation of India Ltd. (NPCIL). This is Govt. of India owned company set-up in Sept., 1987 to give impetus to Nuclear Power Programme in India. The corporation is sole organization that is responsible for design, engineering, procurement of equipment/components, site construction, commissioning, operation and maintenance of Nuclear Power Plants. The Nuclear Power Production in India started way back in mid 60's when two Boiling Water Reactor type Nuclear Power Plants of 200 MWe capacity were set-up at Tarapur on 'Turn-key' basis from USA. Around that time, Dr. Homi Bhabha, along with energy planners, assessed the potential of nuclear power vis-a-vis available resources in natural uranium and abundant thorium reserves. As a result, long range nuclear power programme comprising three distinct and basic stages was evolved. The three stages were :

I Setting-up of small/medium range Pressurized Heavy Water Reactors (PHWRs) based on natural uranium and heavy water. This stage yields plutonium which is to be used for next stage.

II Setting-up of Fast Breeder Reactors (FBRs) based on natural uranium - plutonium fuel with depleted uranium and thorium used as blanket that gets converted into plutonium and Uranium-233 which can be used in third stage i.e.

III Setting-up of Uranium-233 thorium based Fast Breeder Reactors. The reactor generates more fissile material from thorium than is consumed in terms of Uranium-233. This stage thus provides abundant and continual source for power production.

2. APPROACH

Thus it was very clear in the beginning that future of nuclear power in India lies in PHWR technology and therefore after setting-up two BWRs at Tarapur, the focus was shifted to PHWRs and accordingly all the efforts were directed towards acquiring and absorbing PHWR technology from the leader viz. Atomic Energy of Canada Limited (AECL). Accordingly collaboration agreement was reached between AECL and Department of Atomic Energy (DAE). As a result, two units at Rajasthan viz. RAPS-1 and RAPS-2, each 220 MWe PHWR, were set-up in the joint collaboration. Indians were trained in Canada on Douglas Point Generating Station and were also involved in certain design and engineering activities. For the first unit of Rajasthan Atomic Power Plant, the design, engineering and supply of major equipment was from Canada, while site construction, commissioning and operation was done by Indians under supervision of AECL. The second unit though constructed under supervision of Canadians, was commissioned, operated and maintained by only Indians. Some of the equipment were also manufactured indigenously. Then on, there was never looking back and Indians took on responsibility of design, engineering, operations and maintenance for series of PHWRs. As of now there are 8 PHWRs of 220 MWe capacity which are being operated and maintained at various locations in the country. Four more units of 220 MWe PHWRs are under construction. Four units of 220 MWe are in future plan at identified sites. Two units of 500 MWe PHWR have been designed and long delivery items have already been procured. The units are likely to be launched shortly. Thus it can be seen that India has fully developed technology to design, construct, operate and maintain PHWRs in SMR range. The two PHWR types - PHWR-220 and PHWR-500 - have already been shortlisted by IAEA in 'detailed design stage' category.

3. PROSPECTS FOR FUTURE

In addition, a 100 MWth reactor set-up mainly for isotope production and R&D work has been designed, constructed, operated and maintained by Bhabha Atomic Research Centre (BARC), Mumbai. One Fast Breeder Test Reactor (FBTR) at Kalpakkam is under commissioning trials. This is 14 MWe test reactor being connected to the grid in very near future. Work is in progress for design of 500 MWe Prototype Fast Breeder Reactor (PFBR) on the basis of experience gained from FBTR. Design of Advanced Heavy Water Reactor (AHWR) is also in progress at BARC and these reactors would also be in a state of readiness within next five years. Thus a beginning is already made in second stage of Nuclear Power Programme after successfully acquiring technology for implementing first stage of the Programme.

4. RESEARCH AND DEVELOPMENT (R&D)

Self reliance being one of the prominent ingredient of the Nuclear Power Programme, efforts were simultaneously made to establish strong R&D base and training programme for which BARC and other sister organizations under DAE were set-up. The various DAE units engaged in various aspects of supporting Nuclear Power Programme are given below :

4.1. Indian Rare Earths (IRE)

Mining of Rare Earth elements.

4.2. Uranium Corporation of India Ltd. (UCIL)

Mining of Natural Uranium.

4.3. Nuclear Fuel Complex (NFC)

Fabrication of fuel elements and fuel bundles, manufacturing of coolant tubes (pressure tubes), calandria tubes etc.

4.4. Heavy Water Board (HWB)

Production of Heavy Water.

4.5. Electronics Corporation of India Ltd.(ECIL)

Reactor control and instrumentation, consoles,software development & simulators.

4.5. Bhabha Atomic Research Centre (BARC)

This is basic R&D establishment which, interalia, develops basic engineering concepts and technologies required for Nuclear Power Programme implementation. The basic research and development work is then transferred, as part of Technology Transfer, to either above mentioned sister organizations or industries for large scale production on commercially competitive manner.

The BARC is also providing one year Nuclear Orientation Training Course to fresh engineering & science graduates since 1957. Total capabilities exist in complete nuclear fuel cycle including waste management and reprocessing.

5. MANUFACTURING CAPABILITIES

The equipment manufacturing in the country started right from second PHWR (Viz.RAPS-2) itself and as of now, following organizations in Public & Private sector have developed capabilities to manufacture nuclear power plant equipment and components as given below :

5.1. Bharat Heavy Electricals Limited (BHEL)

Balance Of Plant (BOP) i.e. secondary side equipment including Turbine - Generator sets, condensers etc., Steam Generators, Pressure Vessels/Tanks, Heavy Water Heat Exchangers, Electrical Motors.

5.2. Bharat Heavy Plate & Vessels Ltd. (BHPV)

Pressure Vessels/Tanks, Main Air-Locks, Heavy Water Heat Exchangers, Condensers, etc.

5.3. Bharat Pumps & Compressors Ltd. (BPCL)

Centrifugal and recipocating pumps and compressors in medium range.

5.4. National Government Electrical Factory (NGEF)

Motors in all size ranges including for Primary Coolant Pumps.

5.5. Larsen & Toubro Ltd. (L&T)

Steam Generators, Calandria, End Shields, Pressure Vessels/Tanks, Heavy Water Heat Exchangers, etc.

5.6. Walchandnagar Industries Limited (WIL)

Calandria, End Shields, Heavy Water Heat Exchangers, Pressure Vessels/Tanks,etc.

5.7. Kirloskar Brothers Limited (KBL)

Electrical Motors in medium range centrifugal pumps-mech. seal type and canned type.

5.8. KSB Pumps Ltd.

Manufacturing Primary Coolant Pumps.

5.9. Machine Tools Aids and Reconditioning (MTAR)

Precision Machinery and Manufacturing of intricate components.

5.10. Hindustan Construction Company Ltd.(HCCL), Presteel & Elecon Construction Company (ECC)

Civil construction including containment building.

Many of above manufactures have acquired or in process of acquiring ISO-9000 certification.

6. CONSULTANCY

In addition to above mentioned equipment/ component manufacturers, there are number of consultancy organizations in the country which render assistance in design and detailed analysis work required to validate the design or provide Safety Analysis for Licensing purposes. Some of the major organizations engaged are within DAE while others are in educational and industrial sectors.

7. TRAINING

The plant personnel operating and maintaining the Nuclear Power Stations are given thorough training at Nuclear Training Centres located near Rajasthan Atomic Power Station (RAPS). They are selected on the basis of minimum required educational qualification. First operators have minimum graduation (degree) qualifications. The training consists of lectures, plant walk through, check lists, emergency procedures and simulators before they are given independent responsibility to operate and maintain the nuclear power station. The training includes retraining at periodic intervals.

The design, manufacturing, operation and maintenance of nuclear power station by highly skilled and professional personnel is demonstrated by the fact that in approx. 121 reactor years of operation, no accident of radiological nature and above level 3 of the International Nuclear Event Scale (INES) has taken place. Nevertheless accidents of non radiological consequences have taken place but they have been handled quite effectively; without allowing them to escalate into serious ones. Engineering challenges posed due to equipment mal-function, mechanical fault, operator error or natural events have been successfully tackled within the resources and infrastructure facilities available indigenously. Some of such situations have been : inadvertent dousing, end shield brittleness, coolant channel inspection, fire in turbine building, flooding of plant site etc.

8. DESIGN FEATURES

The standard designs now available in India are 220 MWe PHWRs and 500 MWe PHWRs. This size range is ideally suited for Indian conditions viz. strength of the power distribution grid, evolving manufacturing facilities and limited availability of funds and could be prevalent in many developing countries. The salient features of the two nuclear power plants are given below :

Parameters	220 MWe PHWR	500 MWe PHWR
(1) Core : Horizontal Pressure Tubes	306 nos.	392 nos.
Pitch	229 mm	286 mm
Fuel Bundle	19 Elements	37 Elements
No. of Fuel Bundles/Channel	12	13
Thermal Power	770 MW	1736 MW
Av. no. of fuel Bundles replaced per full power day	8	14

Parameters	220 MWe PHWR	500 MWe PHWR
Weight of all fuel bundles Tonnes	60	121
(2) Primary Coolant Circuit	Single Loop	Two Loops
Primary Coolant Pumps	4	4
Steam Generators	4	4
Reactor Headers	4	8
(3) Total Heavy Water Requirement	250 Te	500 Te
(4) Containment : RCC double with suppression pool	Yes	Yes
(5) Shutdown provisions PSS - Primary Shutdown System SSS - Secondary Shutdown System	PSS + SSS	PSS + SSS
(6) Engineered Safety Feature	Provided	Provided
(7) Capability to cope with Station Black Out	Provided	Provided
(8) Spent Fuel Storage Bay	10 Years + 1 Core Unload	10 Years + 1 Core Unload
(9) Waste Management	at Site	at Site
(10) Construction Period	< 7 Years	approx. 7.5 Years

9. CONSTRUCTION SCHEDULE

The construction experience with regard to various 220 MWe PHWRs in terms of construction period is given below :

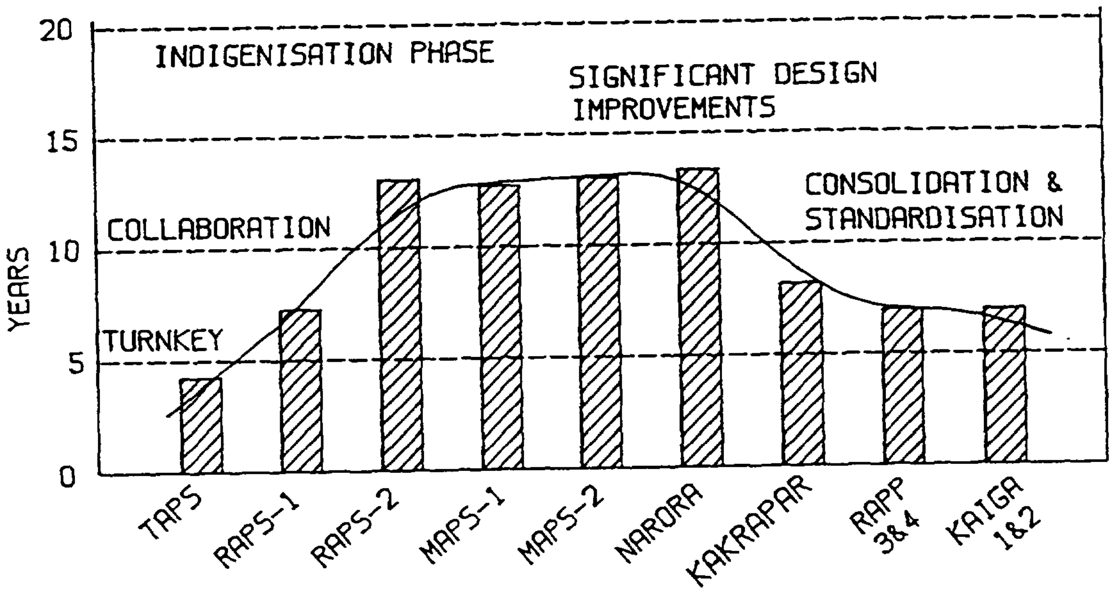


FIG. 1. Trend in schedules Indian nuclear power projects

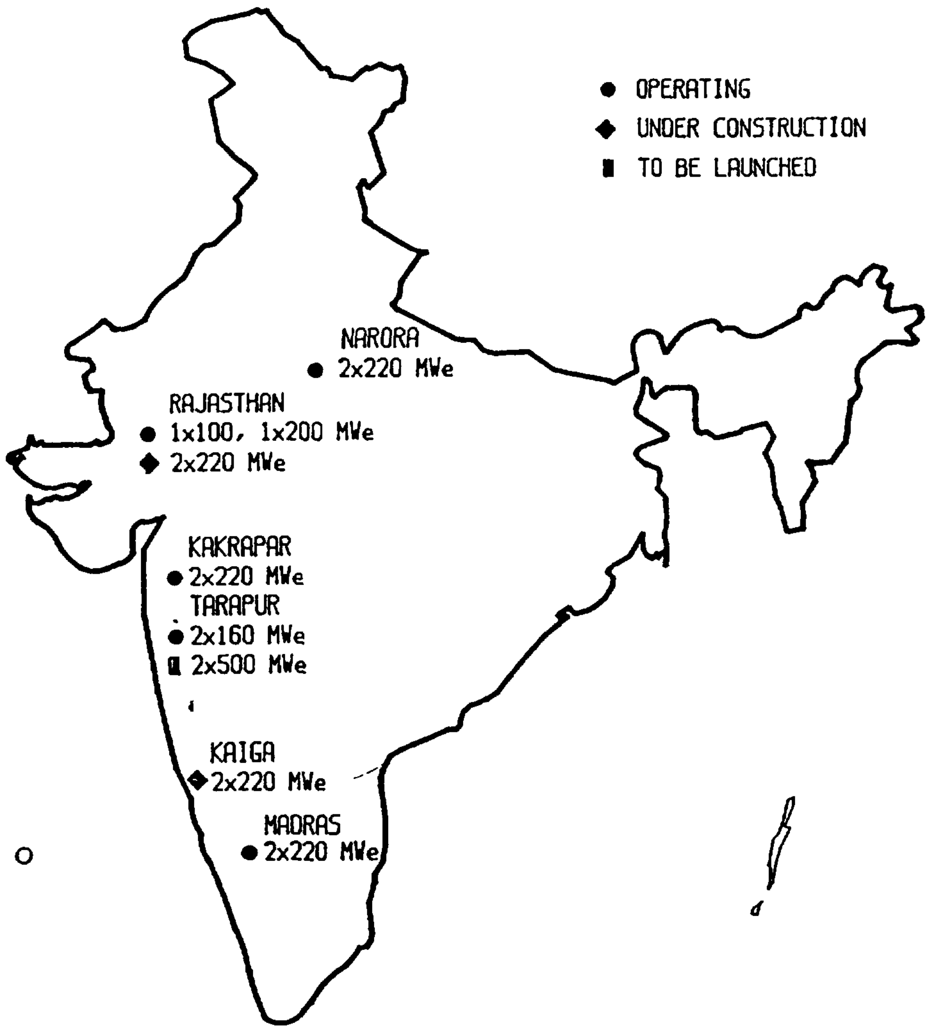
It is seen that the project schedules have extended beyond expectations, however, for each case there were valid reasons in terms of either component availability or manufacturing troubles faced during the process of learning. The manufacturer's capability and technical requirements have to be continuously assessed and alternative solution to be found out which help manufacturer to overcome his difficulty but do not comprise on technical and safety requirements. The NPPs could also not be standardized in order to progressively accommodate evolutionary designs and evolving safety requirements. Thus each subsequent project itself presented challenges in newer areas; however, now there is semblance of standardization with respect to 220 MWe PHWRs. Manufacturers have also developed and acquired 'state-of-the-art' technology that is commercially competitive. In most of the cases there are more than one manufacturers and that encourages competition. On the basis of this strength a construction schedule of less than 7 years is very much realistic for 220 MWe PHWR. Extrapolation of this and experience gained in procurement and manufacturing of some of capital equipment for 500 MWe PHWR also indicate schedule of construction around 7.5 years provided unhindered cash flow requirements are met and lead time to complete site specific design details is available.

10. FINANCING AND CONTRACTUAL AGREEMENTS

At a time when technological capabilities were fully demonstrated, the resource crunch-typical to a developing country has started affecting schedules. Initially, i.e., upto September, 1987, the funding of Nuclear Power Plants was done by the Government and funds required on yearly basis were made available from National Budget. However, subsequent to formation of NPCIL in '87, the Governmental support started receding with every passing year. The rationale behind this was to give impetus to the Nuclear Power Programme Implementation in India and also to allow NPCIL to effectively generate its own resources from operation of existing Nuclear Power Stations in competitive manner. Initially the Government proposed 1:1 Debt. Equity ratio for funds, but later on this ratio is also gradually changing. This forces NPCIL to borrow funds from free market at market rate to meet costs of new projects under construction. Thus the financing principles and assumptions very much dictate Unit Energy Cost and cost per MW of installed capacity. The utilization of generation surplus and market borrowing is flexible option in changing market economy.

India has not yet offered Nuclear Power Plant package to any other country and therefore, contractual agreements are not existing. The previous contractual agreements in nuclear field were in sixties but those were as buyer country and not as supplier country.

The pricing of nuclear power plant within home country and for export model is not identical. This is so because the taxation and duties levied are different. There are concessions available for export items and these change to



NUCLEAR POWER PROGRAMME IN INDIA
PRESENT STATUS

some extent with time. It is, therefore, not quite appropriate to project the figures but can be at best taken as indicative. These figures will emerge only at the time of negotiations taking place and the prevailing economic scenario.

As on date NPCIL is not able to finance any Nuclear Power Plant outside India. Various financial options which could be available at the time of negotiations are :

- Financing by buyer country within its resources.

- Financial assistance from world financing organizations.

- Supplier's credit for the equipments/components manufactured by the supplier (i.e. manufacturer in this case).

MILESTONES

APRIL 15, 1948	:	ATOMIC ENERGY ACT IS PASSED
AUGUST 10, 1948	:	ATOMIC ENERGY COMMISSION IS SET-UP
AUGUST 03, 1954	:	DEPARTMENT OF ATOMIC ENERGY IS CREATED
AUGUST 01, 1955	:	THORIUM PLANT AT TROMBAY GOES INTO PRODUCTION
AUGUST 04, 1956	:	APSARA REACTOR - THE FIRST IN ASIA - GOES CRITICAL
JAN. 20, 1957	:	ATOMIC ENERGY ESTABLISHMENT, TROMBAY NOW (BARC) INAUGURATED
JAN. 30, 1959	:	URANIUM METAL PLANT AT TROMBAY PRODUCES NUCLEAR GRADE URANIUM
OCT., 1969	:	TARAPUR ATOMIC POWER STATION (BWR) COMMERCIAL OPERATION
DEC., 1973	:	RAJASTHAN ATOMIC POWER STATION (PHWR) COMMERCIAL OPERATION
NOV. 19, 1982	:	POWER REACTOR FUEL REPROCESSING PLANT AT TARAPUR REPROCESSES URANIUM OXIDE FUEL
MARCH 04, 1985	:	WASTE IMMOBILISATION PLANT AT TARAPUR IS COMMISSIONED.

DESIGN FEATURES - 220 MWe PHWR

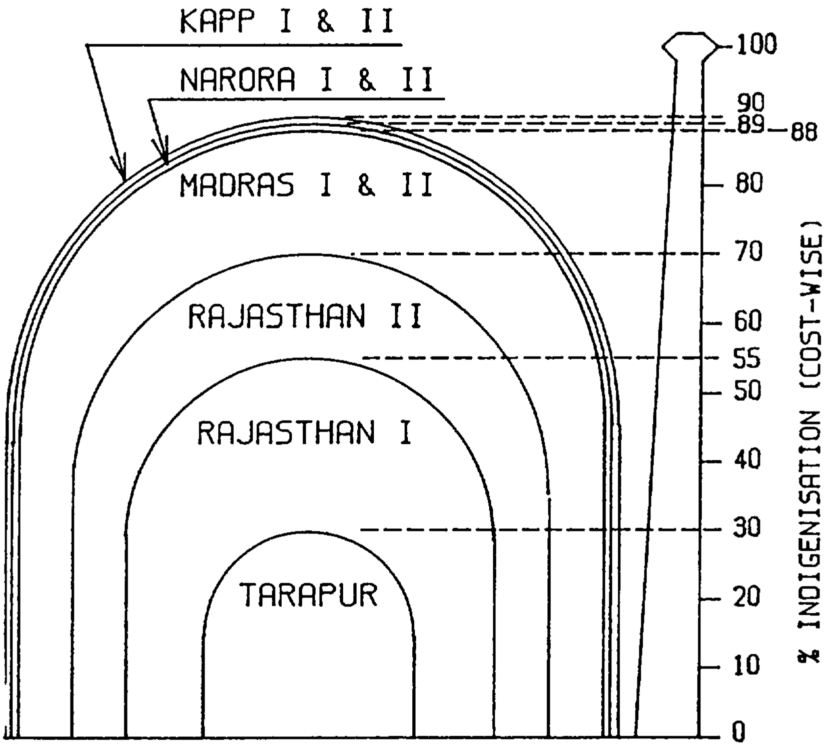
1.	CORE : HORIZONTAL PRESSURE TUBES	306 Nos.
	PITCH	229 mm
	No. FUEL BUNDLE	19 ELEMENTS
	No. OF FUEL BUNDLES/CHANNEL	12
	THERMAL POWER	770 MW
	Av. No. OF FUEL BUNDLES REPLACED PER FULL POWER DAY	8
	WEIGHT OF ALL FUEL BUNDLES TONNES	60
2.	PRIMARY COOLANT CIRCUIT	SINGLE LOOP
	PRIAMRY COOLANT PUMPS	4
	STEAM GENERATORS	4
	REACTOR HEADERS	4
3.	TOTAL HEAVY WATER REQUIREMENT	250 Te
4.	CONTAINMENT : RCC DOUBLE WITH SUPPRESSION POOL	YES
5.	SHUTDOWN PROVISIONS PSS - PRIMARY SHUTDOWN SYSTEM SSS - SECONDARY SHUTDOWN SYSTEM	PSS + SSS
6.	ENGINEERED SAFETY FEATURE	PROVIDED
7.	CAPABILITY TO COPE WITH STATION BLACK OUT	PROVIDED
8.	SPENT FUEL STORAGE BAY	10 YEARS + 1 CORE UNLOAD
9.	WASTE MANAGEMENT	AT SITE
10.	CONSTRUCTION PERIOD	< 7 YEARS

DESIGN FEATURES - 500 MWe PHWR

1.	CORE : HORIZONTAL PRESSURE TUBES	392 Nos.
	PITCH	286 mm
	No. FUEL BUNDLE	37 ELEMENTS
	No.OF FUEL BUNDLES/CHANNEL	13
	THERMAL POWER	1736 MW
	Av.No.OF FUEL BUNDLES REPLACED PER FULL POWER DAY	14
	WEIGHT OF ALL FUEL BUNDLES TONNES	12
2.	PRIMARY COOLANT CIRCUIT	TWO LOOPS
	PRIAMRY COOLANT PUMPS	4
	STEAM GENERATORS	4
	REACTOR HEADERS	8
3.	TOTAL HEAVY WATER REQUIREMENT	500 Te
4.	CONTAINMENT : RCC DOUBLE WITH SUPPRESSION POOL	YES
5.	SHUTDOWN PROVISIONS PSS - PRIMARY SHUTDOWN SYSTEM SSS - SECONDARY SHUTDOWN SYSTEM	PSS + SSS
6.	ENGINEERED SAFETY FEATURE	PROVIDED
7.	CAPABILITY TO COPE WITH STATION BLACK OUT	PROVIDED
8.	SPENT FUEL STORAGE BAY	10 YEARS + 1 CORE UNLOAD
9.	WASTE MANAGEMENT	AT SITE
10.	CONSTRUCTION PERIOD	APPROX.7.5YEARS

INDIAN NUCLEAR POWER PROGRAMME

UNIT	CAPACITY	REMARKS
TARAPUR-1&2	2 x 160	COMMERCIAL OCT'69
RAJASTHAN-1	100	—"— DEC'73
RAJASTHAN-2	200	—"— APR'81
MADRAS-1&2	2 x 220	—"— JAN'84&MAR'86
NARORA-1&2	2 x 220	—"— JAN'91&JUL'92
KAKRAPAR-1&2	2 x 220	—"— MAY'93&FEB'95
RAJASTHAN-3&4	2 x 220	CRITICALITY BY
KAIGA-1&2	2 x 220	CRITICALITY BY
TARAPUR-3&4	2 x 500	LAUNCH BY



DEGREE OF INDIGENISATION IN NUCLEAR POWER PROJECTS

POTENTIAL DESIGN

- DHRUVA 100 MWe
- FAST BREEDER TEST REACTOR 14 MWe
- PROTOTYPE FAST BREEDER REACTOR 500 MWe
- ADVANCED HEAVY WATER REACTOR 220 MWe

- Deferred payments towards expenses incurred by NPCIL and its sister organizations.
- Market borrowing from financial institutions.

These options very much depend on credit rating of and soundness of economy in the buyer country.

11. LICENCING AND PUBLIC ACCEPTANCE

The need to separate the organization engaged in Nuclear Power Programme and the organization that is custodian of nuclear safety and has Licencing Authority was recognized and accordingly Atomic Energy Regulatory Board (AERB) was constituted in 1983. This body is totally independent from all other sister organizations of DAE and NPCIL. With growing concern for nuclear safety world over and in post TMI and Chernobyl accidents, the responsibility and accountability to Public (of AERB) has tremendously increased in recent past and emphasis on thorough safety analysis and documentation has increased. Some of the delays are attributable to preparation and approval of Safety Reports and documents.

Public acceptance of nuclear power is dependent on accident free operation of nuclear power plants and right kind of communication between NPCIL and general public. In recognition of this fact, NPCIL formed separate directorate for Environment and Public Awareness in 1988. This directorate liaise with educational institution and organizes regular exhibition, seminars, press briefings and topical discussions on environment friendly and benign generation of nuclear power.

12. SUMMARY

To sum-up, India has acquired total capabilities in setting-up SMR in developing country. The size of reactor very well matches with connecting grids and indigenous manufacturing capabilities. The NPPs are capital cost intensive units and require matching financial resources. The energy needs of the country on long term basis vis-a-vis available resources have to be properly assessed. Induction of Nuclear Power Programme with a view of indigenization needs proper evaluation in terms of setting-up of large infrastructural facilities and available financial resources. Once committed it is most expensive to abandon a Nuclear Power Programme for any reason whatsoever.

XA9846712

KARACHI NUCLEAR POWER PLANT — A REVIEW OF PERFORMANCE, PROBLEMS AND UPGRADES

S.B. HUSSAIN
Karachi Nuclear Power Complex,
Karachi, Pakistan

Abstract

The Karachi Nuclear Power Plant (KANUPP), a 137 MWe CANDU Unit is located 30 Km west of the city of Karachi, Pakistan It is the first commercial CANDU PHWR, built on turn-key basis by the Canadian General Electric Company for the Pakistan Atomic Energy Commission It was declared in-service on 4 October 1972 and since then operated with a life time average availability factor of 55 9%

KANUPP during its 23 years of operation has experienced multiple challenges in keeping the plant operating and supplying safe and economical power to the Karachi grid The biggest challenge was faced in 1976, when the original vendor imposed unilateral embargo leading to the stoppage of supplies of essential spare-parts, nuclear fuel, heavy water and technical support This forced KANUPP to a new way of operating the plant which necessarily had to be based on indigenous support

Obsolescence of C&I components became evident soon after plant went into commercial operation because of explosive development and advancement in the electronic and computer technology KANUPP was, however, able to cope with the normal maintenance and improvement of its process, mechanical and electrical equipment till 80 s However, many of the critical components are now reaching the end of their designed life and developing chronic problems due to ageing The only technically suitable and commercially viable alternative is the complete replacement of CC&I components KANUPP has already undertaken this job alongwith other related work under "Technological Upgradation Project"

In order to manage ageing related degradation and carry out full scale assessment of the health of reactor fuel channels KANUPP prepared an "Integrated Safety Master Action Plan" and submitted it to IAEA for arranging international assistance After intense negotiations and with the IAEA's cooperation in May 1990, Canadian policy towards KANUPP was revised allowing it to provide assistance for Safe Operation of KANUPP (SOK) through the IAEA and only for the IAEA suggested remedial actions Work under SOK is being carried out to

- Combat ageing and obsolescence problems
- Modernize Operational Safety practices
- Improve safety systems design

This paper describes KANUPP s efforts in overcoming different problems mentioned above

1 Introduction

The Karachi Nuclear Power Plant (KANUPP) consists of a single CANDU PHWR unit with a total gross generation capacity of 137,000 kilowatts It is a natural Uranium, Heavy Water cooled and Moderated Nuclear generating station located at Paradise Point on the arid Arabian Sea Coast, about 15 miles to the west of Karachi It is the oldest operating CANDU-PHWR

Civil construction began in September 1965, following a turn-key contract with the Canadian General Electric Company (CGE) The reactor attained criticality on 1 August 1971 and subsequent full power operation, on 4 October 1972 The Plant is now operating as integral part of Karachi Electric Supply Corporation (KESC) system, contributing approximately 10% of the total demand of power in Karachi During its two decades of operation, the plant has generated about 7 9 billion units of electricity with an average life time availability factor of 55 9% KANUPP which has a design life of 30 years has now completed nearly 23 years of its successful commercial operation

2 0 **Plant Main Features**

• Location	Paradise Point, Karachi
• Owner	Pakistan Atomic Energy Commission
• Prime Contractor and Designer	Canadian General Electric Company
• Civil Consultant	Montreal Engineering Company
• Reactor Type	CANadian Deuterium Uranium (CANDU) Pressurized Heavy Water (PHWR)
• Fuel	Natural Uranium
• Moderator	Heavy Water
• Coolant	Heavy Water
• Thermal Output	432 8 MWth
• Electrical Output Gross	137 MWe
• Electrical Output Net	125 MWe

3 0 **Operation Objectives**

The operation of KANUPP is optimized to meet the following two prime objectives

- Public, plant workers and environmental safety shall be ensured
- Continuous efforts shall be made to produce economic and reliable electricity

The management ensures that the designed features, the procedures and the workers are developed to the best possible standards and that the necessary environment is created and maintained to achieve the above objectives The safety features of the plant to achieve these objectives are

- Defence-in-Depth
  A defence-in-depth concept, in the form of several successive barriers preventing the release of radioactive material to the environment, has been implemented

- Automatic Safety System
  Automatic systems safely shut down the reactor and maintain it in a safe and cooled state

- Normal and Emergency Heat Removal
  Heat transport systems are designed for reliable and efficient heat removal in normal and abnormal operation. Provision is also made for alternative means to restore and maintain fuel cooling under accident conditions

- Conduct of Operation
  The Plant is operated by well qualified, trained and licensed personnel

- Training
  Standard programmes are followed for training and re-training of operating personnel Training is particularly intensive for control room staff

- Emergency Operating Procedures
  Emergency Operating Procedure have been established, documented and approved to provide a basis for suitable operator response to any abnormal event

- Maintenance, Testing and Inspection
  Safety related structures, components, and system are subjected to regular preventive maintenance, inspection and testing to ensure that they meet their design intent

- Quality Assurance in Operation
  An Operation Quality Assurance Programme (OQAP) provides for ensuring satisfactory performance of all plant activities related to safety

- Containment of Radioactive Material
  The Plant is capable of retaining the bulk of radioactive material that could be released from the fuel during accident conditions

- Emergency Preparedness
  A "Karachi Emergency Relief Plan (KERP)" for radiological hazards has been chalked out outlining procedures for protection of public and plant personnel in the event of accident

4.0 Organization

Pakistan Atomic Energy Commission (PAEC) exerts full responsibilities for the safe operation of the plant through a strong organizational structure as defined in Figure-1 under the line authority of General Manager (KANUPP) The General Manager ensures that all elements for safe plant operation are in place, including an adequate number of qualified and experienced personnel Consequent to the challenges of embargoes and commitment to self-reliance, the following divisions and units were either established or upgraded

• Computer Development Division (CDD)	For long term solution of real time Computer Control and other related problems
• Mechanical Design & Development Division (D&D)	For local design and Manufacture of precision and custom made mechanical components
• Technical Division (TD)	For providing effective technical support in the field of design changes, modifications, plant chemistry, operating experience feedback, planning and nuclear material control
• Quality Assurance Division (QAD)	For ensuring effective establishment and execution of quality assurance programme in accordance with international standards and guidelines
• Karachi Nuclear Power Training Centre (KNPTC)	For imparting specific training to engineers and technicians in basic nuclear technology
• In-Plant Training Centre (IPTC)	For providing advanced training to engineers & technicians leading to operating license for KANUPP
• In-Service Inspection (ISI) & NDT Laboratory	For non-destructive testing, evaluation and inspection of plant components
• Control & Instrumentation Application Laboratory (CIAL)	For long term solutions of C&I problem including in-house static calibration and dynamic verification of pressure, temperature, flow and level instruments under plant operating conditions
• Health Physics Division (HPD)	For ensuring effective implementation and control of radiological protection measures and for minimizing personnel radiation dose
• Maintenance Division	For safe and efficient conductance of preventive and predictive maintenance on plant systems and equipments

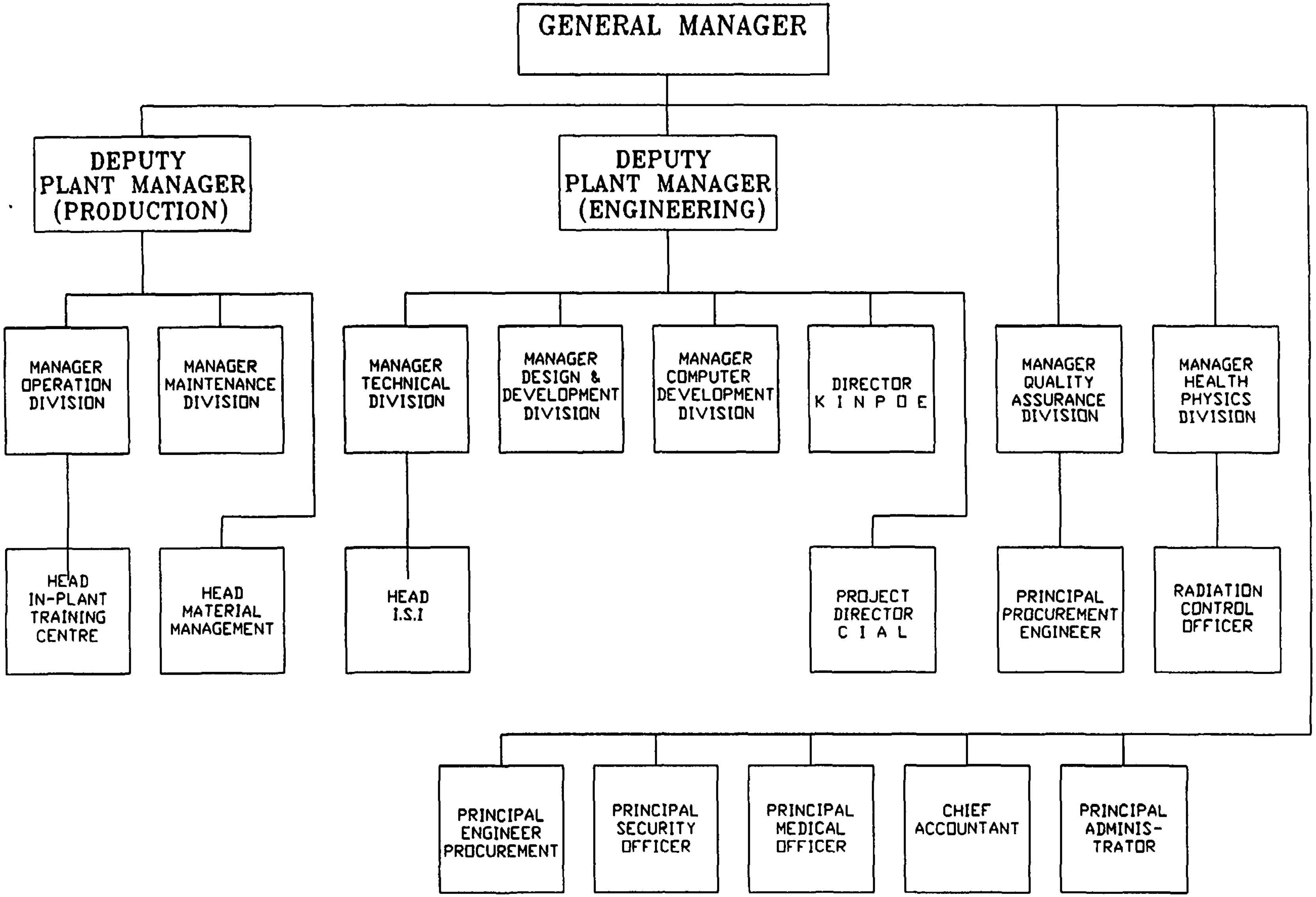


FIG. 1. KANUPP functional organization chart.

5.0 Interface with Regulatory Authority

Nuclear power is regulated in Pakistan by Directorate of Nuclear Safety and Radiation Protection (DNS&RP).

The distribution of responsibilities between regulatory body and KANUPP is such that:

- Primary responsibility for safe plant operation lies with KANUPP.
- Regulatory body sets achievable performance requirements and subsequently monitors these to ensure its compliance.
- Regulatory body has also the authority to accept or reject or modify any proposal submitted by KANUPP on system/equipment design changes, modifications and backfits.
- Regulatory body carries out annual performance review of the plant which includes quantitative measurement of safety and safety related system performance.
- Regulatory body is also responsible for arranging independent safety review of the plant after every five years in close co-operation with experts and consultants who have international experience in plant safety reviews, such as OSART.
- The regulatory body, whenever, finds its necessary and in consultation with KANUPP, provides an independent international assessment through IAEA ASSET missions for identifying areas which require potential improvements, so as to prevent incidents and also to attain an international standard of excellence.

KANUPP 'Operating Policies & Principle (OP&P)' is the key document which acts as an interface between KANUPP and DNS&RP. The OP&P clearly identifies and differentiates between actions where discretion may be applied by KANUPP and where jurisdictional authorization is required by DNS&RP.

On the basis of regulatory and operating policies and principles it is mandatory for KANUPP to report unusual events to DNS&RP within a specified time interval. These are

- Events with major safety significance are communicated promptly (within 24 hrs. of the recognition that event occurred).
- Events with lesser safety significance are communicated within a few days (usually 7 days) of the recognition that the event occurred.

6.0 Plant Performance

Since its connection with the KESC grid in 1972, plant has been operating as a base load Station. Inspite of the early post commissioning phase problems, KANUPP operated with relatively high availability factors upto the year 1977 with an annual average of about 70% (1973-1979). The plant achieved relatively low availability during the period 1978 to 1980 mainly due to non-availability of fuel and essential spare parts from the vendor country. In the wake of the Indian nuclear explosion on May 18, 1974, Canada cut off all technical assistance to Pakistan including operating and design information essential to the operation of KANUPP. The unilateral decision of imposing an embargo on the supplies of fuel, heavy water and spare parts for KANUPP resulted in the curtailment of power production and ultimate shutdown of the plant for most of the period in 1979.

The Pakistan Atomic Energy Commission started fabricating its own fuel in 1979 and since September 1980, KANUPP is operating on indigenously produced fuel. The plant availability started increasing in 1981 and it achieved the highest availability factor of 85.81% in 1994.

In 1982 the plant had to be shutdown for a period of nearly six months to carry out essential maintenance of the plant critical equipments. The major maintenance job undertaken during 1982 was the repair and overhaul of moderator system valves.

The plant was shutdown for a period of about 4 months in 1985 to carryout complete overhauling of turbine, modification on Main Generator, and maintenance jobs related to conventional, electrical, reactor, and protective system.

In 1993 the plant was shutdown for about three months to carryout the In-service inspection of reactor and to remove one sagged reactor fuel channe's

The Plant has so far generated over 7 9 billion units of electricity with an average life time availability factor of 55 9% and an average load factor of 28 4% Its performance during the year 1994 was exceptionally good when it achieved the highest availability factor of 85 81% On ten different occasions, the plant operated continuously for over two months including the longest continuous run of over 113 days in 1995 The performance of the plant over the last three years (1992-95) improved considerably when it produced on the average 530 million units of electricity per year with an availability factor of 74% and capacity factor of 44%

Feeding a relatively small grid, KANUPP during its initial 15 years of operation was required to operate at lower than full power output mainly because of limitations of the load demand and stability of the Karachi grid Load variations over a period of 24 hours were quite large and hence the utility could not apportion to KANUPP a larger base load KANUPP is not in a position to load-follow as the excess reactivity available has not been designed to cater for such large load variations as are experienced in the Karachi grid During the period of 1989 to 1993, the regulatory authority restricted the operation at about 60% capacity due to problems associated with one of its reactor fuel channels

During the 23 years of its operation the plant had, therefore, to operate at an average of 60-70% of its net capacity This resulted in relatively low load factors the maximum being 48 8% The performance factors of KANUPP are shown in Figure-2 & 3

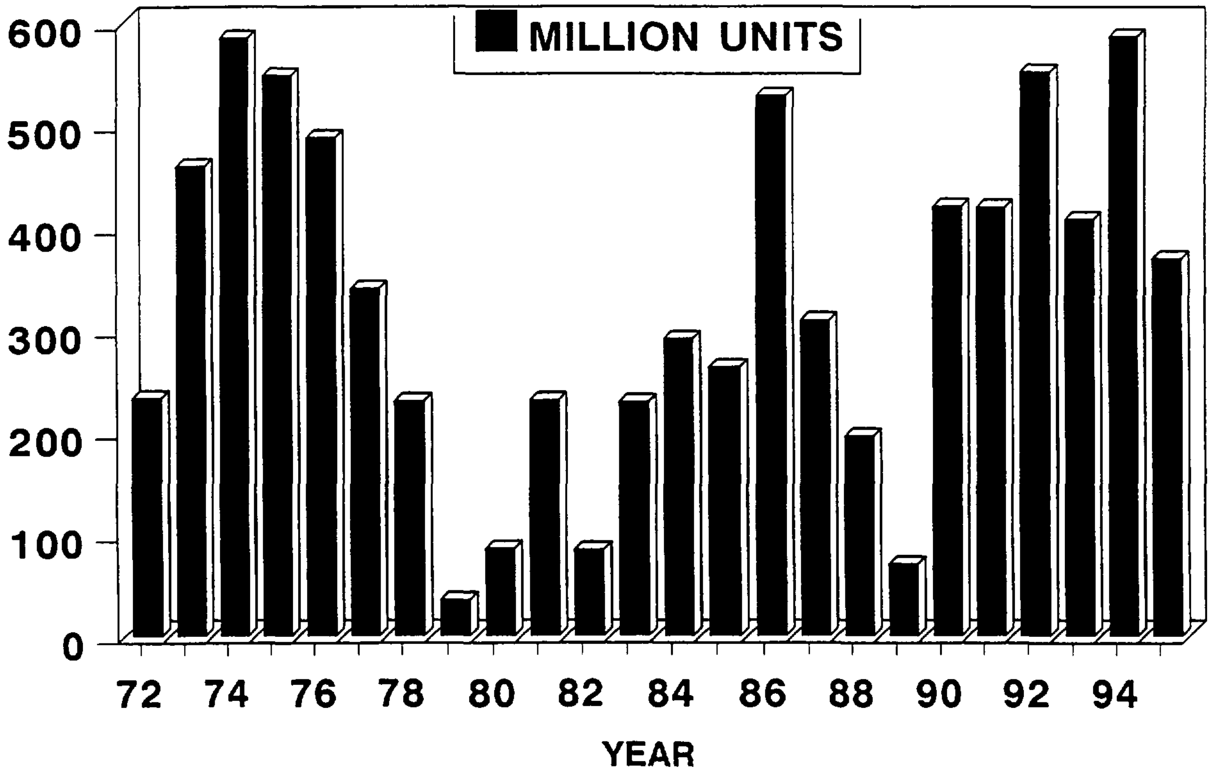


FIG 2 Generation 1972-1995 (up to 19-09-1995)

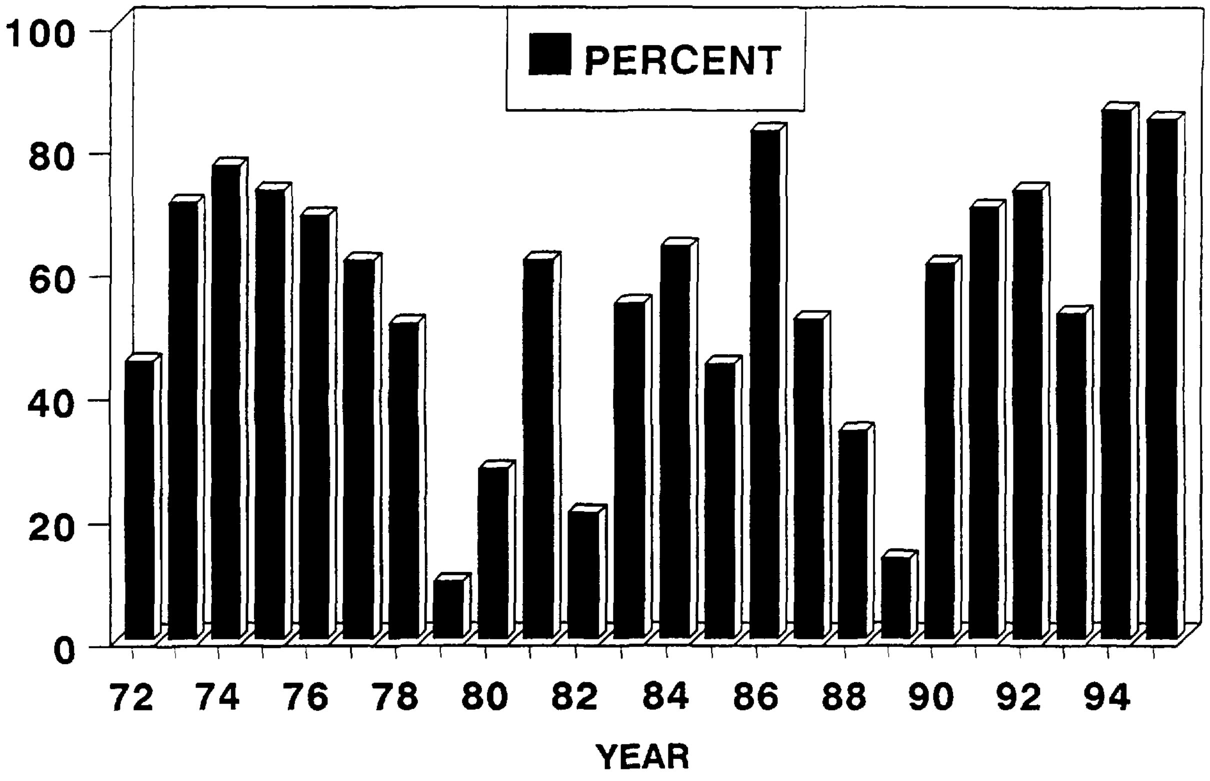


FIG. 3. Availability 1972 (up to 19-09-1995).

7.0 Plant Outages

Plant Outages during its 23 years of operation, have been high as compared to other Canadian plants. A total of 280 outages have been experienced, the average being 12 outages/year. The outage causes and their contributions have been classified as:

Outages Causes	Percentage Contribution
◆ Equipment	25.36
◆ Regulation	20.36
◆ Grid Transient.	12.86
◆ Human Error	7.86
◆ Forced	20.71
◆ Planned	7.5
◆ Safety	5.36

Operating KANUPP without vendor's support can be attributed as the major reason for such high outage rate. Many essential equipment and components which were supposed to be replaced on routine basis could not be attended in time causing unplanned plant outages. The other contributing reasons are

- Unstable Grid.
- Heavy Water Leaks (End Fitting, Valves & Pump Gland etc.)
- Fault in Controlling Computers
- Sea Weed in rush.
- Condenser Tube leak etc.

Inspite of problems of sorts, KANUPP faithfully adhered to its original safety and public risk targets The radiation control safety record has been extremely satisfactory as testified by regular testing and reliability analysis Average personnel radiation exposure has been well within the prescribed limits of International Commission on Radiological Protection (ICRP) Refer Figure-4 Release of radioactive material through gaseous and liquid effluent has remained within 3 % of the Derived Release Limits Refer Figure-5 & 6

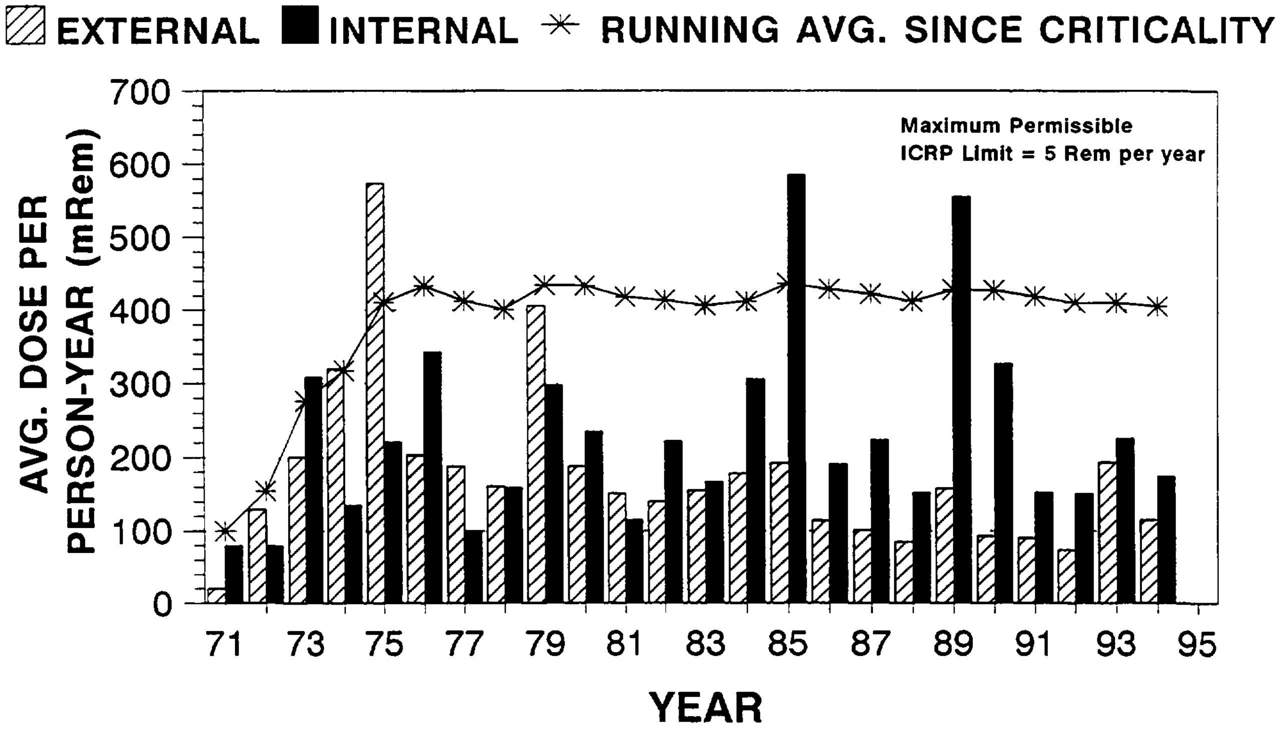


FIG. 4 Average dose per person and running average since criticality.

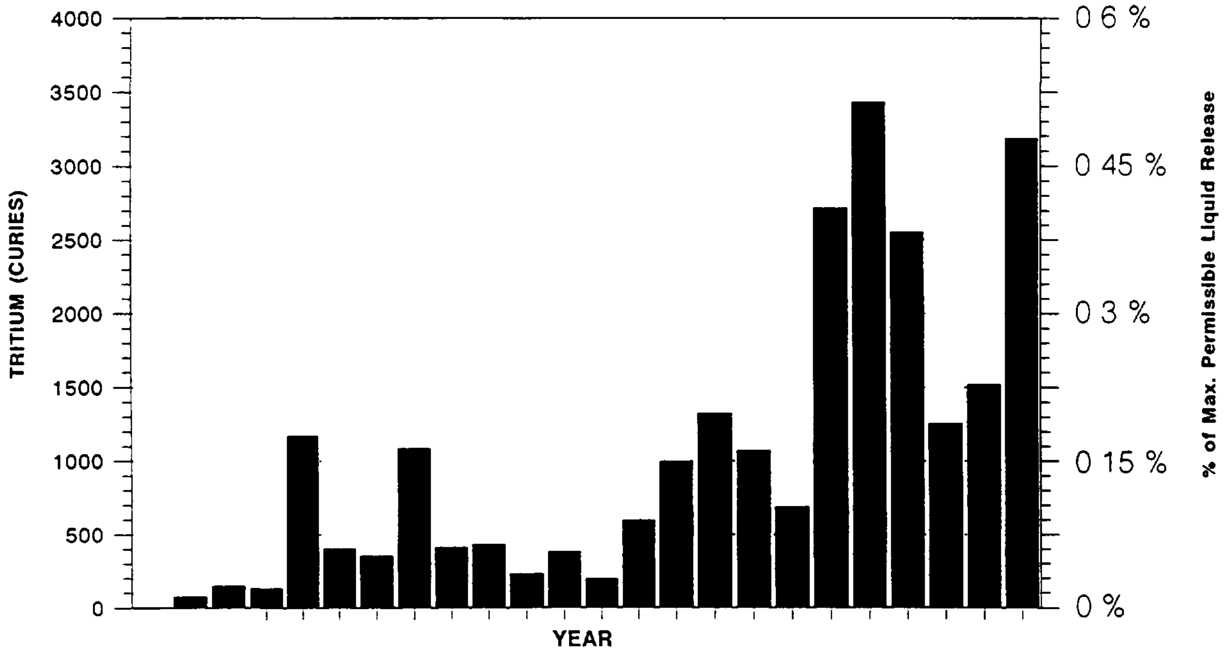


FIG 5 Release of radioactivity to the environments via liquid effluent

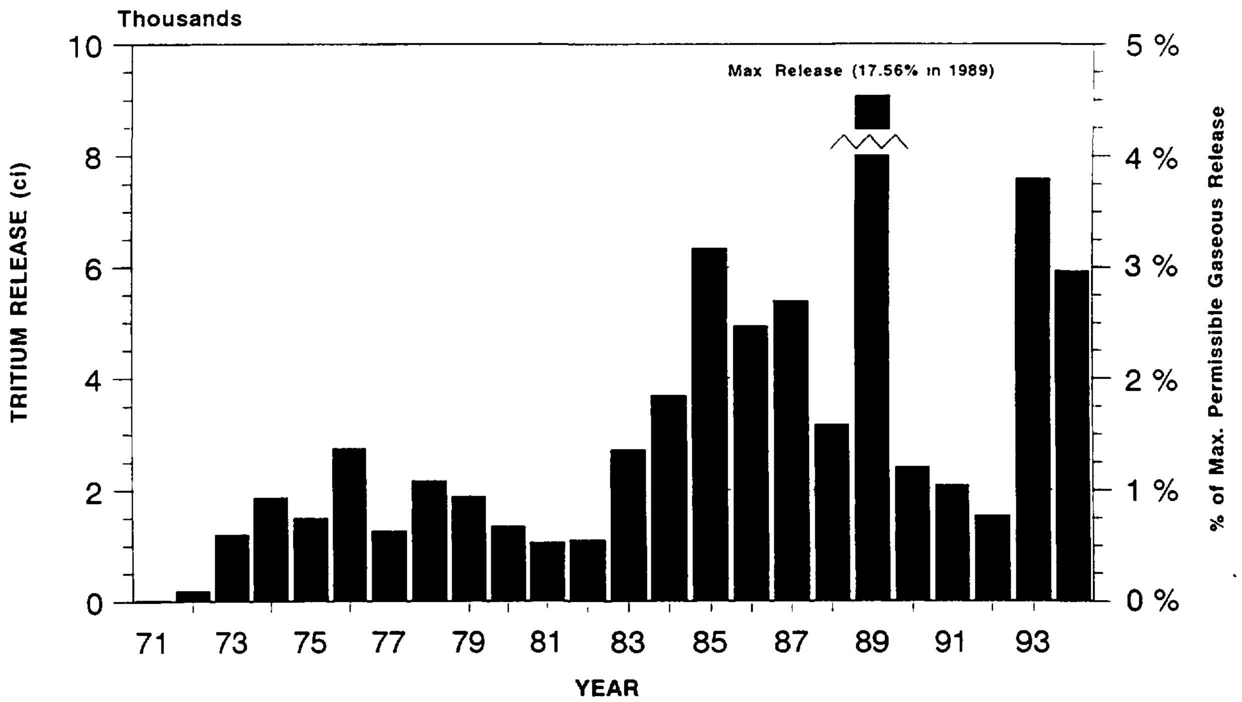


FIG. 6. Release of radioactivity to the environments via gaseou effluent.

9.0 Major Operating Problems and Remedial Actions

Some of the major operating problems experienced during its 23 years of operation and actions taken to resolve them are described below.

9.1 Problems Unique to KANUPP - Embargo and Commitment to Self-Reliance

Two decades of Plant operation since October 1971, when the first nuclear power unit was produced by KANUPP, have proved quite eventful. Following the explosion of a nuclear device by India in 1974, all sorts of nuclear assistance to KANUPP was suspended and a unilateral embargo was imposed by the vendor country on supply of technical assistance, spare parts and fuel in 1976

Operating KANUPP in the environment of complete embargo was a difficult task, especially in the absence of technical infrastructure required to fulfill the station needs. However, a determined effort on the part of its owner, Pakistan Atomic Energy Commission (PAEC) in general and KANUPP in particular kept it operating safely.

The embargoes, however, proved a blessing in disguise A self-reliance programme launched by Pakistan Atomic Energy Commission (PAEC) began to yield results. in 1980, PAEC successfully produced nuclear fuel for KANUPP while it made all-out efforts to create the technical infrastructures, Industrial resources and personnel expertise necessary to support station operation. The Design & Development Division (Mechanical), Computer Development Division, In-service Inspection Labs, Control and Instrumentation Application Laboratory and Quality Assurance Division were subsequently established at KANUPP. At about the same time, the Technical and Health Physics Divisions were strengthened to provide necessary backup for technical and radiation control support. The Karachi Nuclear Power Training Centre (KNPTC) and In-Plant Training Centre (IPTC) were established for imparting basic and advanced training in nuclear power technology to engineers and technicians engaged in the operation and maintenance of the plant.

Such technical support does not form part of nuclear power plant operation in developed countries but in the case of KANUPP, there was no other choice. Incidentally, KANUPP is the only nuclear power plant in the world which has been operating without an active technical and material support from the vendor which is vividly indicative of PAEC's commitment to self-reliance.

9 2 Normal Operating Problems.

9 2 1 Nuclear Island

9 2 1 1 Standby Heat Exchangers

Tubes of one of the two heavy water standby heat exchangers failed due to fretting caused by process water flow induced vibration only after three years of plant operation All tubes of heat exchangers were checked by ECT and the tubes showing thinning in excess of acceptable limit were plugged The tube bundles were also strengthened by installing more fasteners Later inspections showed no or slight ageing effect However, the tube bundles of one of the standby heavy water heat exchangers was fabricated locally and installed It is planned to fabricate and replace the tube bundle of all the three remaining standby and charging heavy water heat exchangers

9 2 1 2 Moderator System valve Gasket

As against the current practice of providing double gasket arrangement, valves in the Moderator System at KANUPP are provided with conventional single gasket arrangement using neoprene gasket at the bonnet This neoprene gasket in one of the moderator pump discharge valves failed (ageing degradation due to embrittlement) and caused a large D_2O spill

The frequency of inspection/replacement of the gasket of all such valves has been increased

9.2.1.3 Differential Pressure Transmitter Casing Bolts.

A differential pressure transmitter between south outlet and north inlet-header failed due to breaking of two out of four chrome plated steel bolts used for joining the two metal casings enclosing the bellows assembly, resulting in D_2O leakage

The bolts on all such transmitters have been replaced

9 2.1 4 Steam Generators

KANUPP has six steam generators with Monel 400 tubes Tube failure with ageing is a well known and expected phenomenon In fact the performance of KANUPP has been better than expected in this area

Steam Generator # 3 developed a small leak in 1989, which developed upto 4 Kg/hr over two weeks at the end of 1990 The plant had to be shutdown for more than a month and a special procedure was developed to isolate the leaky boilers and operate the plant with four instead of six steam generators in service

After performing necessary modifications the plant was operated with four out of the normal six boilers This was a unique operation as no such previous experience was available in any CANDU nuclear power plant The regulatory body gave short-term permission to operate in this condition at 50% power. The leaky tube (only one) was later plugged after acquiring necessary training and confidence on locally assembled steam generator test mock-up

9.2.1.5 Dump Valves

Six dump valves, connected in series parallel arrangement, are provided in the helium balance line to trip the reactor. The valves are 10 inch butterfly valves with diaphragms and elastomer seals connected to the upstream pressure to minimize leakage when the valves are closed.

The dump valves, provided by the vendor, were modified and special elastomer seals were installed by Canadian General Electric. No spare seals, their drawings and other parts are available. Only once a dump valve was opened for inspection. Due to daily trip tests and relatively frequent plant trips experienced, the dump valves have been subjected to rather severe duty It is planned to replace the valves with new ones, and efforts are underway for their procurement.

9 2 1 6 **Fuelling Machines/Fuel Handling.**

Fuelling Machines and end fitting components are a critical part of the primary pressure boundary when in use They are highly complex, custom built, and subject to very high radiation levels We have experienced the following ageing-induced problems so far Some have been solved for now, but the basic issue of fabricating replacements for custom-built parts remains and becomes more serious with time

- A D_2O hose ruptured leading to a major heavy water spill
- D_2O head circulating pump canned motor developed micro-crack due to excessive rubbing D_2O seeped into stator winding deforming the stator can
- Snout jaws developed cracks over a period of time within & along the groove, being the maximum stressed areas The snout jaws cracks propagated in 1988 as revealed by ISI Plant operation was not allowed by the regulatory body till replacement with locally manufactured and extensively tested snout jaws
- *Failure of mechanical seal 'o' ring installed in charge tube axial drive due to ageing*
- Closure plug locking problem The shield plug was found damaged and the charge tube latch finger had broken inside the end fitting
- Shield plugs sticking on rotation Investigation revealed that tail end was causing restriction both in axial and rotary motion due to bulging/deformation Machined to original size before replacement
- Two fuel bundles entangled due to damaged end plate of one, entered Fuelling machine magazine together and prevented rotation Dislodged by special procedures The incident is attributed to hammering received by the bundle against the channel sealing surface

9 2 1 7 **Reactor Fuel Channels**

In 1989, fuelling problems led to detection of two sagged reactor fuel channels in cold shutdown condition Subsequent inspection revealed reactor fuel channel G-12 to be sagged by 49 mm and F-15 by 12 mm The problem is not unique to KANUPP as similar problems had been discovered in other Canadian plants and rectified successfully In 1993, the Canadians under the IAEA assistance carried out an assessment of the fuel channel integrity and removed G-12 to identify the root cause of its retraction The problem was found to be specific with only G-12, whereas all other reactor channels inspected were found to be in perfectly good condition The reactor resumed operation with one channel removed

9 2 2 **Plant Computers**

9 2 2 1 **Control Computer**

Plant Control computers consist of a dual redundant digital computers, GE-PAC-4020, for reactor power regulation These computers were designed in mid sixties and were installed as part of reactor control system by the original vendor

The computers were maintained through indigenous modifications until 1989 The modification include recoding of the real time computer control software, a 15 man year effort which considerably improved the performance of main plant control system. Performance of these computers have now deteriorated to the extent that their replacement is essential for reliability and continued safe operation of the plant The replacement of regulating computers is being undertaken as part of Technical Upgradation Project (TUP)

9 2 2 2 **Fuel Handling Computers**

The existing PDP-8 computer system for Fuel Handling Control has been replaced by an Industrial IBM PC-AT Computer system, utilizing the expertise available in PAEC A new software package written in Microsoft-86 assembly language has been provided to execute the same functions as the PDP-8 fuel handling control computer

9 2 3 **Control and Instrumentation (C&I)**

During the last decade extensive advancement and innovation has taken place in computers and high technology electronics and informatics This has completely changed the design and operational philosophy of modern nuclear plants Nowadays, the modern plants are being controlled by a very sophisticated software based technology The Control & Instrumentations systems of KANUPP were designed in mid sixties and have now become totally outdated and obsolete Due to their obsolescence and ageing extreme difficulty is being encountered in their maintenance and upkeep In fact, at a certain stage, it was felt that due to ageing and obsolescence their maintainability may aggravate to a level where it will become very difficult to ensure reliability and safety of plant operation Therefore, a systematic and comprehensive programme has been prepared for functional replacement of obsolete C&I equipment A Technological Upgradation Project (TUP) has been initiated and a contract has been signed with a foreign firm for the supply of CC&I equipment the design of which has been completed by KANUPP engineers New CC&I equipment is expected to be installed and made operational by end of 1996 In addition, the following C&I jobs have already been completed either indigenously or in association with some other foreign vendors

- Replacement of T/G Instrumentation
- Upgradation of Plant Communication System
- Installation of a close circuit TV system for monitoring various areas of Reactor Building
- Radiation monitoring system
- Plant Switch Yard Extension etc

9 2 4 Conventional Systems

Following steps have been taken to resolve problems due to proximity to sea (corrosion due to airborne sea-salts) and use of sea water for condensing steam and for Cooling Process Water, causing corrosion, erosion, carry-over of silt, sea weeds, in-rush of sardines and barnacles growth on pump house equipment and associated systems

- Traveling intake salt water screens replaced with corrosion resistant material such as stainless steel
- Condenser tubes kept clean by reversing flow, and ferrous sulfate addition at frequent intervals to create protective layer on tubes
- Process cooling water pump casing and main headers piping replaced
- Radiator air cooling installed for diesel generators, replacing the sea water cooling
- Other equipment i e meteorological tower, station transformer radiators and high voltage transmission towers, which were badly affected by corrosion, were replaced
- Frequency of painting of all the plant equipment which is exposed to corrosive atmosphere/environment increased
- Complete retubing of Process salt water Heat exchangers with titanium tubes
- Replacement of chlorinating plant
- Salt Water pumps and casings
- Boiler Blowdown line replacement.
- TLO separater replacement
- Complete replacement of Fire Water Ring, etc

10 0 International Co-Operation and Safety Updgradation

10 1 Role of IAEA

The Three Mile Island and Chernobyl accidents have greatly increased the role of IAEA in ensuring safe plant operation all over the world KANUPP has been inspected by 'IAEA Operation Safety & Review Team (OSART)' in 1985 and in 1989 and on both occasions the plant was found to conform to IAEA operational standards An IAEA 'Assessment of Safety Significant Evaluation Team (ASSET)' mission also visited KANUPP in 1989 and performed in-depth analysis of plant operational safety practices The ASSET mission made several recommendations for improving the safety of the plant An overall plan identifying, prioritizing and scheduling the activities important to plant safety was developed based on the experience of KANUPP and the recommendations of ASSET missions An 'Integrated Safety Review Master Plan (ISARMAP)' was established in 1991 Based on ISARMAP, the IAEA approved initially a four year (now revised till 1997) technical assistance project namely 'Safe Operation of KANUPP (SOK)' and approached the Canadian Government which agreed to provide initial consultancy for expert assessment and planning for the required safety improvement through CANDU Owner's Group (COG) of which KANUPP was a member The objective of this IAEA project (PAK/9/010 'Safe Operation of KANUPP) was to arrange the necessary international technical support for the tasks listed in ISARMAP The support under PAK/9/010 was envisaged in following three ways

10.3 **Integrated Safety Review Master Plan (ISARMAP) Task and Status**

SR. NO.	ISARMAP TASK	CURRENT STATUS
01.	**Project Management**	
	a) Establish an Integrated Master Plan for SOK.	Established in 1991. Reviewed by Steering Committee five times.
	b) Umbrella agreement with COG for Canadian Technical Assistance.	Only safety investigations, assessment diagnostics tasks and expert and services have been allowed.
02.	**Equipment Ageing Affecting Safety**	
	a) Fuel Channel Integrity Assessment (FCIA).	Sagged pressure tube G-12 was removed and ISI of this and other 7 selective reactor channels, done with Canadian technical support. A local, non-generic problem with only G-12 due to a leaky rolled joint was identified.
	b) Improve CO_2 Annuals Gas System	The CO_2 system at KANUPP is once through and needs review for its adequacy to provide leak-before-break detection. This is important as a creak in pressure tube could be detected well before it reaches the critical crack length (~15hrs.).
	c) Repair of Steam Generator	One leaky tube was detected and plugged by KANUPP. Canadian Technical support was proposed but ultimately not required.
	d) Fuelling Machine Ageing	The fuelling machine forms a part of the coolant pressure boundary during on-power fuelling. Replacement of its aged parts is essential, however, Canadian government has so far not considered it as a safety issue.
03.	**Equipment Obsolescence Affecting Safety**	
	a) Replacement of Radiation Instrumentation	Obsolete radiation monitoring instrument has been replaced.
	b) Replacement of obsolete Computers Control & Instrumentation.	- Fuelling Machine PDP-8 computers have been replaced with Industrial Grade PC's. - Replacement of plant regulating computers GEPAC-4020 and instrumentation is planned in 1996.
04.	**Improve Operational Safety Practices**	
	a) ISI of Steam Generator and BOP Piping.	Primary system pressure boundary and critical BOP pipings were inspected in 1992 by Canadian specialists. The Systems were found in excellent condition. Eddy current testing of two steam generators was done in May 1993 by B&W Canada and tube conditions were found to be very good. Only one tube in steam generator # 3 with 26% 'Through Wall Thickness (TWT)' detected & plugged.

SR. NO.	ISARMAP TASK	CURRENT STATUS
	b) Review of modern Maintenance Technique	Techniques followed in modern CANDU plants such as infra-red thermography & laser shaft alignment are being implemented
	c) Improvement of Safety Culture	Courses to improve safety culture at KANUPP were arranged in 1991 by IAEA. Two IAEA experts delivered lecture in 'Basic Safety Principles for Nuclear Power Plants' and 'Analysis and Prevention of Safety Significant Events Root cause analysis of safety significant events have been established after these course
	d) Review of Modern Emergency Preparedness	Emergency preparedness arrangements in modern Candu were reviewed by a KANUPP specialist in 1992 Required improvements and practices are being adopted in KANUPP
	e) QA Programme	Operations Quality Assurance programme was reviewed by a Canadian specialist in July 1994 and found in line with Canadian practices
	f) Operating Experience Feedback	International Operating experience feedback has been computerized on an internal LAN, connected to CANDU OWNERS GROUP (COG) and WANO, INPO Networks
05	<u>Improve Design Safety</u>	
	a) Update of Final Safety Analysis Report (KFSAR)	Work on Phase-I (Analysis of special safety systems against limiting large break LOCA) has been completed Work on Phase-2, i e complete updating of final safety report is planned to be undertaken in 1996
	b) Probabilistic Safety Analysis Level-1	The work on KANUPP PSA Level-1 is in progress and expected to be completed by end of 1996 Reviews are done by IAEA experts at required intervals
	c) Equipment Qualifications (EQ) Review Against LOCA	Expert review was done in 1993 by two Canadians No major problem identified, however, they recommended establishment of a systematic EQ programme Few modifications were implemented as part of EQ programme
	d) Booster Cooling	The use of booster has been discontinued because of the safety concerns with respect to its cooling Canadian plants do not use boosters any more
	e) Emergency Boiler Feed Water Supply	In order to provide an un-interrupted heat sink to reactor, an independent supply of emergency feed water to steam generator is being planned Independent review by a Canadian expert of the design has been done New system is expected to be installed by early next year

SR. NO.	ISARMAP TASK	CURRENT STATUS
	f) Adequacy of Emergency Power	KANUPP has two standby Diesel Generators A third Diesel generator is being added to improve the emergency power supply needs and decrease the allowed outage times
	g) Containment Testing at Higher Pressure	The containment Pressure test was done at full pressure (27 psig) during commissioning and afterwards it is being done at 2 psig In order to assess the leak rate at higher boiler room pressure it is now planned to perform the test at elevated pressure Test at 5 psig has already been done successfully KANUPP is now approaching COG to provide expert services from Point Lepreau NGS for developing necessary procedures, to test containment at a reasonably high pressure
	h) Seismic Walkthrough of the Plant	KANUPP is designed for 0 1 g earthquake A walkthrough by IAEA experts in 1993 concluded that KANUPP can withstand twice the design basis earthquake with minor modifications Necessary steps being taken to implement the recommendations

- Foreign Experts visit to KANUPP
- Some Assistance in purchase of equipment
- Some fellowship assistance for KANUPP personnel

A steering committee was also constituted by IAEA to review the ISARMAP tasks, their scope and priorities, the results of the activities for the enhancement of KANUPP safety system and recommendations for further actions

The ISARMAP could be broadly classified into the following five areas with a number of tasks in each area

- Project Management
- Ageing
- Obsolescence
- Operational Safety
- Design Safety Improvements

A detailed description of the tasks alongwith their current status is given in section 10 3

10 2 Role of Other International Organizations

The complete isolation of KANUPP from international channels of communication partially ended in 1989 following the Three Mile Island and Chernobyl incidents which aroused an instant realization among the nuclear community to promote global safety in nuclear power plant operation. The CANDU Owners Group (COG) and, later, the World Association of Nuclear Operators (WANO) were formed to provide a forum for promoting closer co-operation among nuclear utilities in matters relating to operational experience feedback, human performance and plant safety KANUPP joined COG and WANO in 1989 and has now access to public domain informations from nuclear utilities around the world

COG is playing the leading role in accomplishment of jobs under the 'Safe Operation of KANUPP (SOK)' Project. An agreement to this context exists between KANUPP and COG

WANO has done a two week 'Peer Review' of KANUPP in November 1994 and 20 experts from all over the world conducted in-depth safety review of the plant The identified weak areas are being strengthened Under the aegis of WANO-TC, a number of technical exchange visits to other nuclear power plants has also been arranged, providing opportunity for operational experience exchange

11 0 Conclusion

It is an accepted fact that nuclear power plant must operate with the technical support of the vendor country and other international help, since the quantum of R&D, design and operational & safety experience of the vendor country can not be matched by NPP operating country alone This aspect of nuclear power plant operation has been recognized well by the international community and as a consequence institutions such as INPO, WANO & COG have been created These institutions alongwith IAEA are playing vital role in providing necessary technical services and support to the operating power plants in assessing the extent to which they are complying with their original safety standards ana also suggesting the improvements and modifications required to achieve an acceptable level of current international safety standards

KANUPP has amply demonstrated that nuclear power is feasible in a developing country The plant can operate despite heavy odds and numerous challenges It has also established that nuclear power is environmentally clean and safety remains paramount in the operation of a nuclear power station

In more than 20 years of operation, not a single KANUPP employee has lost a day's work due to radiation exposure - a testimony to the good design and safe operation of the plant

Radioactive emission to the environment throughout the operational history of KANUPP has remained below 3% of the Derived Release Limits

Various safety reviews and investigations done in the past by IAEA and other international organizations confirmed that KANUPP has maintained an excellent safety record and that its critical components such as Reactor, Boilers and Turbine Generator are in perfect condition

KANUPP is striving hard to resolve its current ageing-induced equipment problems to satisfy the original safety requirements and public risk targets which are still internationally acceptable However, as a policy the management is committed to upgrade the safety as far as possible , towards current standards and criteria

It is envisaged that the economical life of the plant would be extended 10 years beyond its design life of 30 years after providing all the required replacements of its obsolete informatics and refurbishing of nuclear island as well as conventional equipments

REFERENCES

1 Javeed Iqleem
Computer Control and Instrumentation Backfitting in a PHWR Plant
Paper presented at, IAEA Technical Committee and work shop progress in Heavy Water Reactor Design and Technology
Montreal, Canada 6-9 December, 1988

2 M A Ghafoor, J A Hashmi, Zia Siddiqui,
Identification of Efforts Required for Continued Safe Operation of KANUPP
Paper presented at, Second PHWR Operating Experience Meeting, held at Embalse, Argentina from April 3-5, 1991

3 Syed Badshah Hussain
Plant Performance, Maintenance and Development Activities
KANUPP-STR-88-2

4 Reactor Safety Re-evaluation, a Common Basis for Judgment
Safety of All Operating Nuclear Power Plants Built to Earlier Standards
Report of a Consultants' Meeting International Atomic Energy Agency 8-12 November 1993 Vienna, Austria

5 Javed Iqleem, Zia Siddiqui, Parvez Butt
Experience with Operational Safety Improvements at the Karachi Nuclear Power Plant
International Conference on Advance in the Operational Safety of Nuclear Power Plants, Vienna, Austria, 4-8 September 1995

6 Syed Badshah Hussain
KANUPP Operating Experience
Paper presented at, IAEA Technical Committee Meeting on Exchange of Operational Safety Experience of Pressurized Heavy Water Reactors 20-24 February 1989 Vienna, Austria

7 M A Ghafoor, Syed Mazhar Hasan, M Qamar-ul-Hoda, Safdar Habib
Meeting the Challenge of KANUPP Operation

8 Babar Ghias
KANUPP Outage Record
KANUPP-STR-95-02

9 Syed Badshah Hussain
Regulatory Practices Relating to Monitoring Assessment of Ageing at Karachi Nuclear Power Plant

THE ROMANIAN EXPERIENCE ON INTRODUCTION OF CANDU-600 REACTOR AT THE CERNAVODA NPP

S.N. RAPEANU, A. BUJOR
National Agency for Atomic Energy

O. COMSA
Center of Technology and Engineering for Nuclear Projects

Bucharest, Romania

Abstract

The Cernavoda Nuclear Power Plant (NPP) Project is a key component of the Romanian nuclear development program. Selection of the CANDU design represents a major contribution to this development, due to the technological feasibility for manufacturing of parts, components and the nuclear fuel based on the uranium resources in Romania. The Romanian nuclear development program also involves a nuclear fuel manufacturing plant, a heavy water production plant and organizations specialized in research, engineering, manufacturing and completion for systems and components. The agreement on technological transfer between Canada and Romania is supporting the Romanian involvement to the achievement of the Project, with a degree of participation that is gradually increasing from the first to the last NPP Unit.

1. INTRODUCTION

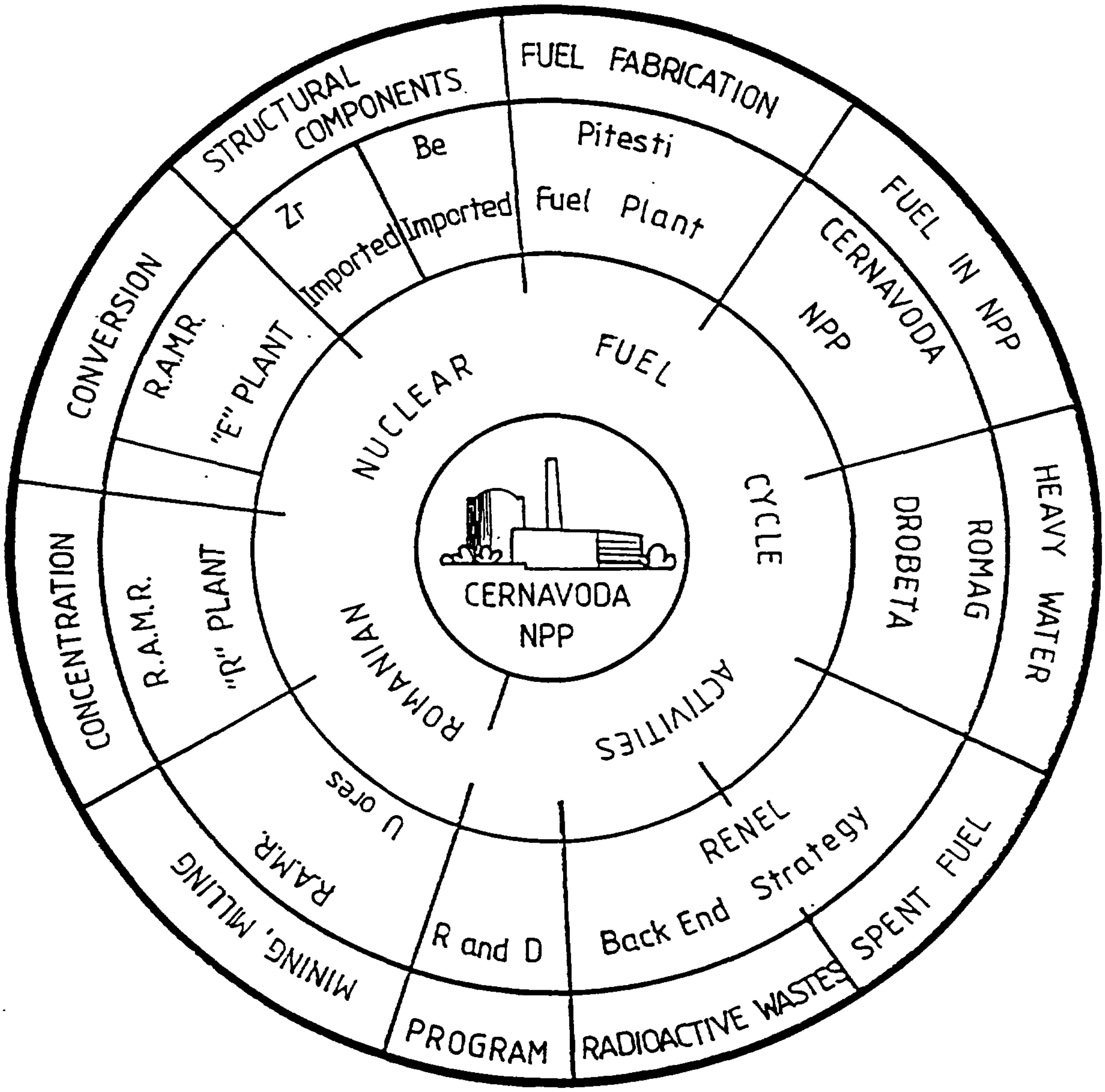
The need of diversification of energy sources, independence from foreign supplies and modernization of economy, has been the most important reason that led Romania's authorities to decide the implementation in Romania of the nuclear energy. In this respect, the initial thinking has highly regarded the Canadian heavy water CANDU reactors for their advanced safety features and use of natural uranium as fuel and heavy water as moderator and coolant, both being materials possible to be manufactured by the domestic industry.

A first nuclear power plant with four units CANDU-600 has been decided to be built at the site of Cernavoda, in the eastern part of the country, with the aim to further extend the role of atomic energy by construction in the future of other units [1].

With that in mind, a contract with Atomic Energy of Canada Limited (AECL) has been negotiated and concluded in 1979, covering licensing of generic CANDU-600 design, the nuclear steam plant design, supply of equipment and technical assistance in detailed local engineering from Canada, quality assurance and construction [2] and [3].

At the same time, contracts were negotiated for the supply of equipment and technical assistance related to the Balance of Plant and for the turbine-generator group for the first two units, with Ansaldo Impianti (Italy) and General Electric. As a result, several licenses and technical cooperation agreements were signed between Canadian companies, Italian companies, G.E. and Romanian manufacturers, with the aim to allow manufacturing in Romania of most of the equipment, with an increasing domestic participation from the first to the last planned NPP units at Cernavoda [4] and [5].

Simultaneously, at the national level, steps have been made for insuring appropriate advances in other related areas (Figure 1): research and development, fuel cycle, heavy water manufacturing, personnel training and implementation of new standards for design, quality assurance and safety regulation [6].



RENEL - Romanian Electricity Autority

RAMR - Rare and Radioactive Metals Board

Figure 1 Romanian nuclear fuel cycle - related activities

2. POLITICAL WILLINGNESS AND PUBLIC ACCEPTANCE

The Romanian nuclear program is based on the principle of peaceful use of the nuclear energy. The system of nuclear guaranties and measures for physical protection and nuclear materials accounting are implemented at all of the facilities related to fuel cycle activities, research reactors, research and development institutes, and at the Cernavoda NPP.

At the political level, the program for introducing the nuclear energy, that is the CANDU option, has received and receives now, an appropriate degree of priority and support. Despite that, until 1990, the development of the project suffered from disadvantages of a highly centralized economy, and the decisions were taken from political rather than technical and economical reasons.

Project management was not recognized and structured as an autonomous activity. The decisions were taken at different levels of central and local administration and the responsibility was diluted between various ministries. The severe limitations of imports and hard currency expenditures, imposed after 1982, made the continuation and completion of the project impossible in the planned time schedule because the volume of both the technical assistance and the equipment to be imported was minimal. Also, the training of the Romanian personnel in the field of nuclear plant design, operation and maintenance was limited. These were the main reasons that contributed to the delays in the completion of this project.

The year of 1990 has brought important changes in the economical, social and political life of the country. The difficulties of the transition period toward free market economy have led, between others, to the decrease of the industrial activity and implicitly, to a reduction of the demand of electricity, and to hardships in the possibilities of financing. These factors, together with the existence of others important and complex problems to be solved, have caused the postponement of the decision regarding the continuation of the national nuclear program. The present situation is under evaluation, for establishing a realistic program based on the assessment for cost analyses, environmental impact and evolution of the electricity demand [8].

The acute crisis of energy suffered by the Romanian society and economy in the '80's and, on the other hand, the advanced nuclear safety features of the CANDU design, fully at the western level, have insured from the beginning a very good level of public acceptance for the nuclear option. The history of safe operation and good economic performance of the existing CANDU stations as well as the introduction of the licensing standards and procedures inspired from the american practice, have also contributed to the acceptability of CANDU design. These reasons were completed by safeguards and safety requirements which increased the trust of the population in nuclear field and influenced positively the public opinion.

Still, at present, there is no significant opposition to the nuclear energy, despite the existence in Romania of the means and rights for doing so, specific to the opening democracies.

3. CERNAVODA NPP STATUS

As it is well-known the reactor CANDU-600 is based on natural uranium fuel and heavy water as moderator and coolant. The reactor has 380 horizontal pressure channels placed in a cylindrical horizontal reactor vessel (called calandria), each channel having 12 fuel bundles with 37 fuel rods each, Figure 2. The fuel is made by sinterized UO_2 powder pellets, enclosed in zircalloy sheath [2], [3] Figure 3.

The moderator is heavy water with 99.8% Deuterium at normal pressure and working temperature of 69°C, placed in the calandria vessel between the fuel channels. The coolant is pressurized heavy water, flowing between the fuel elements inside the pressure tubes.

According to the contract between the Romanian authorities and AECL, in 1979 has started the work at the Unit 1 of the Cernavoda NPP. Subsequent, the civil engineering work was started for the units 2-4 and in 1982 the fifth unit has been added. Further on, the work has been carried out simultaneously for all the five units of the plant.

The important delays accumulated until 1990 made necessary an overall review of the project, realized with the support of AIEA. A stopwork has been ordered and measures meant to reorganize and accelerate the project have been taken. A new strategy has been adopted having the following key features:

- concentration of all managerial, human, financial and organizational resources for completing as soon as possible the first unit of the Cernavoda NPP
- stop of work, except conservation activities, for the units U3-U5, having a status of completeness between 5% and 25%
- enhancement of the expert assistance in construction, management commissioning and early operation of U1 together with AECL and Ansaldo Impianti.

In this respect in December 1990, AECL and Ansaldo formed the AECL-Ansaldo Consortium (AAC). The newly formed Romanian Electricity Authority RENEL and AAC concluded a new contract in 1991, through which the Consortium performs the project management with the aim of commissioning Unit 1 by March 1995. AAC will also operate the plant for the initial period and provide both formal and on the job training for the Romanian personnel who will operate the Unit 1 when AAC lives the site.

At present, an important part of the commissioning process is completed, such that the Unit 1 is now at the stage of solving the problems for allowing the first criticality [8].

The National Commission for Control of Nuclear Activities (CNCAN), which is the Romanian regulatory body, has already authorized the following:

- loading of D_2O in the moderator system
- manual loading of the nuclear fuel
- loading the coolant (heavy water) in the primary heat transport system
- performance of high temperature tests for the primary heat transport system
- leak test for the containment.

The next steps to be achieved are:

- first criticality
- physics tests at zero power

- gradually increase of the reactor power and test operation
- operation at nominal power (authorization for guaranties)
- normal operation.

Following the present schedule, the reactor will become critical to the end of 1995, with the other steps of the commissioning program being scheduled to be performed in the first half of 1996.

In this situation, it can be stated that a large part of the problems related to the reactor have been solved; among them, it can be mentioned:

- the final safety report phase 1
- documentation support for nuclear safety
- a large part of the reliability and stress analyses
- a large part of the documentation for normal operation
- a large part of the safety and process systems has been tested and commissioned and most of the objectives of the commissioning process have been fulfilled
- an important part of the operation personnel has been authorized

It is also important to note that the commissioning process is closely watched by the utility owner and by the regulatory authorities.

4. NATIONAL PARTICIPATION

The possibility of participation of national industries in the Nuclear Power Program was one of the important reasons in the decision of implementing CANDU reactors in Romania.

Our own participation was established in the following fields.

4.1. Nuclear Fuel

An important factor supporting the decision of implementing CANDU reactors was to manufacture the nuclear fuel in Romania using the existing resources of uranium ores. Decisions have been taken for developing the domestic facilities at various stages of processing: mining, milling, concentration and conversion to UO_2 powder [7].

Early in 1979 the activities for preparing the Romanian capabilities for manufacturing of the CANDU fuel assemblies have been started. In a first stage, a pilot plant has been organized on at the site of Pitesti for implementing domestic technology following the Canadian specifications [3] and [6].

Simultaneously, theoretical activities have been carried out for design and simulation of the structural behaviour of fuel under both normal operation and accident condition. Fuel elements and CANDU fuel bundles have been manufactured and tested in irradiation facilities in the TRIGA reactor at Pitesti and abroad (Belgium, Germany and Canada).

In 1987, as the capability for manufacturing CANDU fuel has been proven, a decision was taken to extend the pilot station into a fuel manufacturing plant with the capacity to provide the nuclear fuel for all the reactors of the Cernavoda NPP [6].

In 1990 the fuel manufacturing was stopped. In order to update the technological process, the production was reorganized and the production flow was modernized and improved. The production has been resumed in 1994 following the licensing by the Canadian company Zircatec Production Ind. (ZPI), an important supplier of Canadian fuel. Based on this licence, the first load of Unit 1 contains a number of 200 fuel bundles manufactured in Romania. All the fuel required for the future operation of the plant will be produced in Romania.

4.2. Heavy water

Heavy water is one of the major components in the technical and economical assessment of CANDU NPP.

In order to reduce the costs associated with production and supply of heavy water it was decided to produce in Romania by our own forces the whole quantity of heavy water necessary for the national nuclear programme.

Based on this decision, a research and development program was carried out.

A domestical technology was developed and tested through a heavy water pilot plant and an industrial dimension was later created.

It is noteworthy that there are now two out of four modules in operation which can produce the entire necessary heavy water quantities at a high purity level (99.9%). Cernavoda 1 uses an important part of the initial supply of heavy water from domestical production.

According to a national heavy water production program, all necessary heavy water for the next units and for Unit 1 annual consumption will be made in Romania [6].

4.3. Research, development and engineering

The national nuclear program has also included research and development activities for:

- design, development of technology and testing of fuel
- technology for manufacturing of heavy water
- behaviour of materials in neutron flux
- design of systems and components for the nuclear island and the balance of plant
- testing of the nuclear equipment assimilated by the national industry
- reactor physics, fuel management and nuclear safety
- risk assessment and evaluation of the radiological impact on the environment
- design of equipment for dosimetry and radiological protection

For these purposes a new research institute, the Institute for Nuclear Power Reactors, has been established in 1977, with departments specialized for reactor calculations and nuclear safety, design of nuclear systems and components, technological developments for nuclear fuel, in-pile and out-of-pile testing facilities. In addition to that, the other tasks of research, development and engineering have been distributed among the other existing institutions with experience in nuclear physics, power engineering, chemistry, metallurgy and mechanics.

The resources and efforts invested in this direction have finally allowed the following items to be performed in Romania [6]:

- development of the technologies for manufacturing of heavy water and of CANDU fuel

- preliminary and final safety reports
- assimilation of the methodology for fuel management calculations

- execution projects for all the five units of the Cernavoda NPP
- testing of the equipment manufactured in the country for the Units 1 and 2.

At the same time, the performance in Romania of these activities has created opportunities for the development of a highly qualified personnel and of a national competence in fields of major importance for the further progress of the nuclear energy program.

4.4. Technology transfer

The agreement on technology transfer between Canada and Romania is supporting the involvement of the Romanian industry to the achievement of the project, with an increasing degree of participation through [2],[3]:

- assimilation of new materials
- assimilation and implementation of technologies for manufacturing of systems and components
- introduction and implementation of advanced design codes and quality assurance procedures at all the levels related with the construction of the nuclear power plant.

Based on the licenses purchased from the foreign partner, it was developed a national industrial infrastructure, through the endowment with modern equipment of industrial companies in the fields of metallurgy, electrotechnics and electronics and mechanical tools.

The creation of this infrastructure required major financial efforts, with the result of generating a modern and performant nuclear industry.

Similar efforts have been required for the achievement of the transfer of technology that involved:

- technical assistance from AECL for project management
- technical assistance for personnel training
- acquisition of licenses for fabrication of most of equipment including the calandria vessel, pressurizer, degaser vessel, steam generators, control devices, pressure tubes, pumps, turbine, fueling machines, etc.
- implementation of the QA system in all the companies which provide equipment and services for Cernavoda NPP
- assimilation and implementation in design, execution and montage of the ASME code
- implementation in the nuclear industry of the nuclear safety culture
- implementation of modern regulations for the nuclear activities, and their adaptation to the specific components of the project

The start of the work at the Cernavoda NPP has found the technology transfer program under implementation with many of its components not fully completed. The further progress of this program and of the on site work were not synchronized, so leading to the introduction of distortions that have negatively influenced the evolution of the project.

4.5. Personnel training

A well trained competent management and technical staff is essential to the safe and a reliable operation of a nuclear power station and other nuclear facilities.

The qualification and training of personnel, able to operate the station and to satisfy nuclear facilities' requirements, is achieved through a program, whose objective is to ensure the training at an appropriate level of quality.

In this respect in Romania has been developed a specific training program which includes, at different levels:

- post secondary school specialized in nuclear power sciences
- departments for nuclear engineering and nuclear physics at technical and science universities
- on the job training at Cernavoda N.P.P., including use of full-scope simulators
- specific training in Canada under the agreement with AECL for the future operation of the plant
- final training during commissioning
- specific training in Canada and Romania for personnel for other research and nuclear facilities.

The training program has been developed in the '80's. The main purpose, the training of Romanian personnel for nuclear plant design, operation and maintenance, was limited and the opportunities offered by the existing agreements or by the IAEA were not entirely used.

Under these circumstances, the utility RENEL and AAC have concluded a new contract in 1991 through which the Consortium will perform the management of the project, will operate the plant for an initial period of 18 months and will provide formal and on-the-job training for the Romanian personnel who will later operate the plant. In addition, a complement personnel is working on the Project Management Team (PMT), under the direction of the Canadian and Italian Managers, in order to achieve a transfer of management skills.

For other specific activities (i.e fuel fabrication, heavy water production, nuclear industries) the necessary personnel was already trained through a particular arrangement with IAEA, PNUD, ONUDI and Canadian partners.

5. CONCLUSIONS

From the results mentioned above, the approach to introducing nuclear energy in Romania is characterized by specific features, generated by strategy and objectives, as:

- buying of a licence for a nuclear power plant based on the CANDU-600 reactor with advanced safety features and suitable for the domestic conditions and resources

- buying of a licence for manufacturing of equipment for the plant

- preparing the Romanian industry for the modern requirements of the nuclear technologies

- assimilation of the Canadian concept of management

- training of qualified personnel

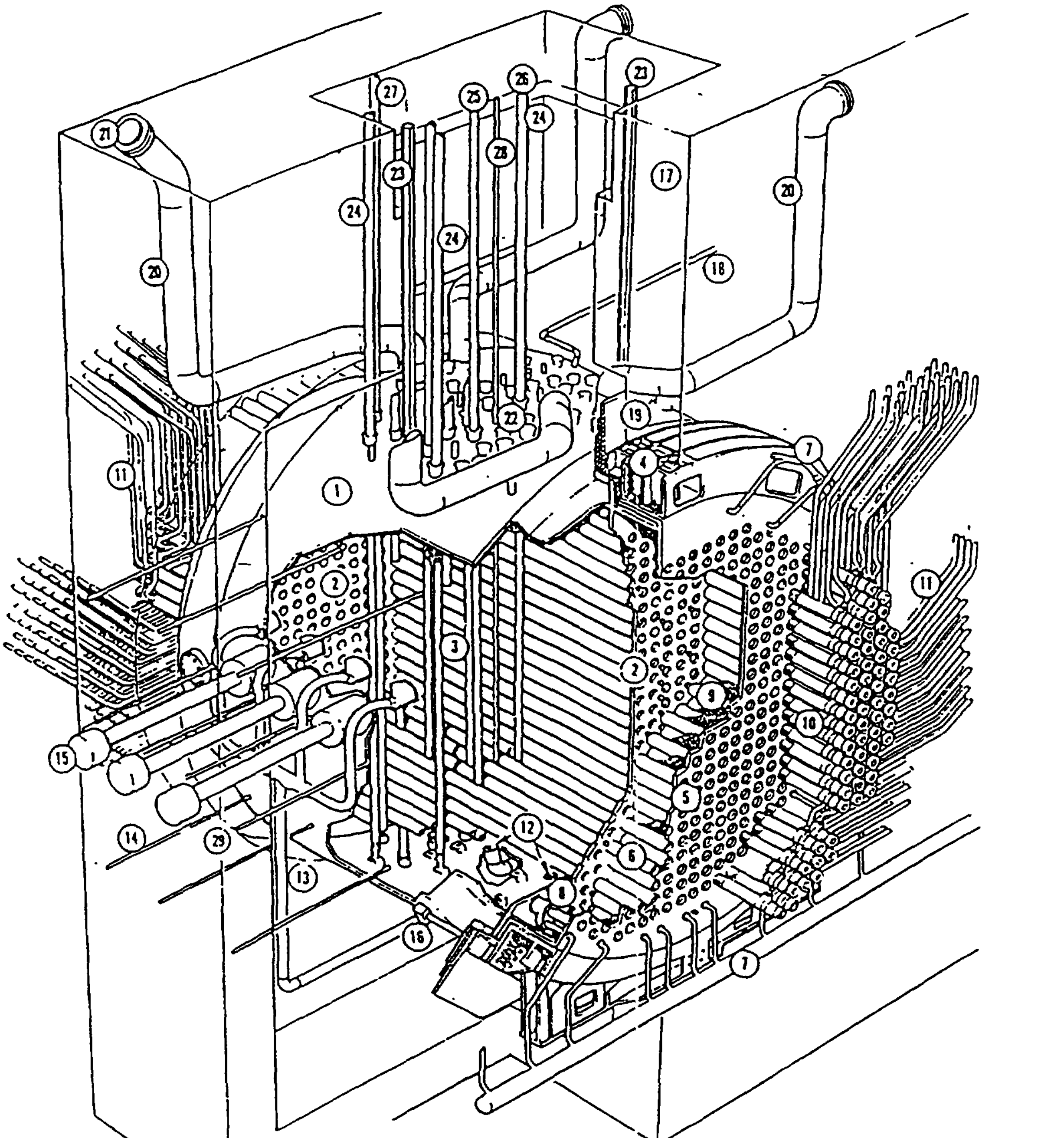


Figure 2 CANDU-600 reactor assembly

- implementation of modern nuclear regulations and of the safety culture in all the stages of completion of the project

- development of research and development capabilities for a permanent domestic support of the project.

The process of implementation and achievement of these objectives have required great efforts from the national economy. In this respect, the main conclusions and lessons learned are:

- It is important to select a partner with high scientific and technological potential and disponibility for transfer of this knowledge.

- It is essential to ensure a good management of the project, eventually with the participation of the partner, using the local resources that must be fully used in the project.

- It is necessary to insure the continuity of the work at the five units of the plant, for allowing a smooth transfer of the teams of specialists from one unit to another. This requirement is essential for the economicity of the work.

- The requirements for construction and manufacturing of nuclear components must be respected. This will lead to the overall improvement of the level of quality of the production with beneficial effects on the economicity of the work, stability of the personnel, while maintaining the opening toward high technologies. A direct consequence is the necessity of establishing a regulatory and control body, which is independent from the plant owner.

- The existence of a national competence in research, development and engineering is very important for the management of the project, plant operation and for further improvements and updating. Related to that, emphasis should be put on personnel training and an educational system based on a curriculum which is applicable to the particular design of the reactor and of the nuclear power plant. This is important due to the differences that do exist between the design and operation procedures of different NPP's, even if they have the same type of reactor.

- It is necessary that the licensing and authorization process will not introduce distortions and will not make derogations regarding:
    - completitude of the safety documentation
    - reliability of the systems and equipment
    - performance in the commissioning tests
    - training and licensing of the personnel
    - management of the project changes.

We trust that the conclusions derived from the existing experience will be put in practice for the future development of the Romanian nuclear power program, particularly in the work to complete the other units of Cernavoda NPP and in the development of related activities.

REFERENCES

[1] "Energy and Nuclear Power Planning Study for Romania (1989-2010)", IAEA, TECDOC-820

[2] "Engineering Services Agreement (ESA)", AECL, 1978

[3] "Licensing Services Agreement (LSA)", AECL, 1981

[4] Mingiuc, C., Panait, A., et al, "Cernavoda Nuclear Power Plant - Status and Prospects", IAEA, TECDOC-738

[5] Mingiuc, C., "ISPE contribution to NPP development in Romania", Bucharest, 1991

[6] Comsa, O., "Cernavoda Nuclear Power Plant Fuel Supply Aternatives", Bucharest 1992

[7] Georgescu, D., Vasile, C., "Uranium Resources Evaluation for Nuclear Power Program", Bucharest 1995

[8] "Electric Power Production, Transport and Distribution in Romania; Tendencies and Concerns", RENEL, Bucharest 1993

XA9846714

POTENTIAL ROLE OF THE ROMANIAN RESEARCH AND INDUSTRY ON THE SMALL AND MEDIUM REACTORS MARKET

S.N. RAPEANU, A. BUJOR
National Agency for Atomic Energy

O. COMSA
Center of Technology and Engineering for Nuclear Projects

Bucharest, Romania

Abstract

The need of diversifying the energy sources, independence from foreign supplies and modernization of economy have constituted the major factors in implementation of nuclear energy in Romania. The choice of the heavy water reactor CANDU-600 was made on grounds of advanced safety features, proven efficient economic operation as well as on the technologic feasibility for manufacturing of components, equipment, instrumentation, heavy water and natural uranium fuel in Romania.

Unlike turn-key acquisition approaches, the Romanian option provided an active national participation in construction the Cernavoda NPP. As consequence, important support was being given to development of the industries involved in the nuclear fuel cycle, manufacturing of equipment and nuclear materials, construction-montage, engineering, consulting, services, etc. This was done based on technology transfer, implementation of advanced design and execution standards, quality assurance procedures and modern nuclear safety requirements at international level.

The efforts materialized in an important national participation in the construction of the Cernavoda NPP and all related programs are successful. Now, Romanian firms are also involved in supplying components, equipment and services to NPP's in other eastern and central Europeans countries.

The paper presents the achievements of the Romanian economy in this field and the effort of the Romanian companies on the small and medium power reactors market. Lists with main R&D institutes, nuclear fuel cycle facilities as well as potential equipment suppliers are attached.

1. Introduction

The need of diversifying the energy sources, independence from foreign suppliers and modernization of economy have constituted the major factors in implementation of nuclear energy in Romania. The choice of the heavy water reactor CANDU-600 was made on grounds of advanced safety features, proven efficient economic operation as well as on the technologic feasibility for manufacturing of components, equipment, instrumentation, heavy water and natural uranium fuel in Romania [1].

As result of the national energy policy, in 1979 a contract was concluded between the Romanian authorities and Atomic Energy of Canada Limited (AECL) for the license of building CANDU power plants in Romania. This contract had the following main features:

- Romania buys the license for building of CANDU-600 reactors in Romania;
- AECL provides technical assistance for design of CANDU NPP, Romania having the quality of general designer of the plant;

Romania buys the reference project including the execution documentation, the technical specifications for all equipment of NPP and the internal reports regarding the research made at AECL-Chalk River Research Laboratories.

In 1982 a similar contract was signed for the Balance Of Plant (BOP) with Ansaldo Impianti (Italy) and General Electric, and in 1984-1985 the contracts with equipment main suppliers were concluded.

Romanian option provided an active national participation in construction the Cernavoda NPP (fig. 1). As consequence, important support was being given to development of the industries involved in the nuclear fuel cycle, manufacturing of equipment and nuclear materials, construction-montage, engineering, consulting, services, etc. This was done based on technology transfer, implementation of advanced design and execution standards, quality assurance procedures and modern nuclear safety requirements at international level [2].

The efforts materialized in an important national participation in the construction of the Cernavoda NPP and all related programs are successful. Now, Romanian firms are also involved in supplying components, equipment and services to NPP's in other eastern and central Europeans countries [3].

2. Nuclear fuel cycle

2.1. CANDU fuel fabrication

An important factor supporting the decision of implementing CANDU-PHWR was the capability of manufacturing the nuclear fuel in Romania using the existing resources of uranium ores and accessible technologies. As a result, measures have been taken for developing domestic facilities in all the processing stages of nuclear fuel cycle (fig. 2).

Domestical uranium resources are estimated to be sufficient for Romanian Nuclear Power Program, which involved five units at Cernavoda NPP [4].

Extraction and processing of uranium ores is made by the Rare Metals Company (RAMR). The technological processes cover the following stages :

- mining and milling of uranium ores;
- concentration up to 60% uranium in Natrium Uranate (UNa) and Natrium di-Uranate (DUNa);
- conversion, purification in U_3O_8 (yellow cake);
- reduction of U_3O_8 into nuclear grade syntherizable UO_2 powder.

Following AECL specifications, a Romanian technology for manufacturing CANDU fuel was developed. In a first stage a pilot station was built on a site near Pitesti in 1980. In 1987, as the capability of manufacturing CANDU fuel was proven, the pilot station was extended into a fuel fabrication plant. More than 31000 CANDU fuel bundles (fig. 3) were fabricated before 1990 [5].

In parallel to that, theoretical activities for design and simulation of the structural behavior of the fuel were carried out. Fuel elements and CANDU fuel bundles were tested in the TRIGA reactor at Pitesti and abroad (Belgium, Germany, Canada).

Out of pile testing of fuel bundles and experimental investigations were also performed.

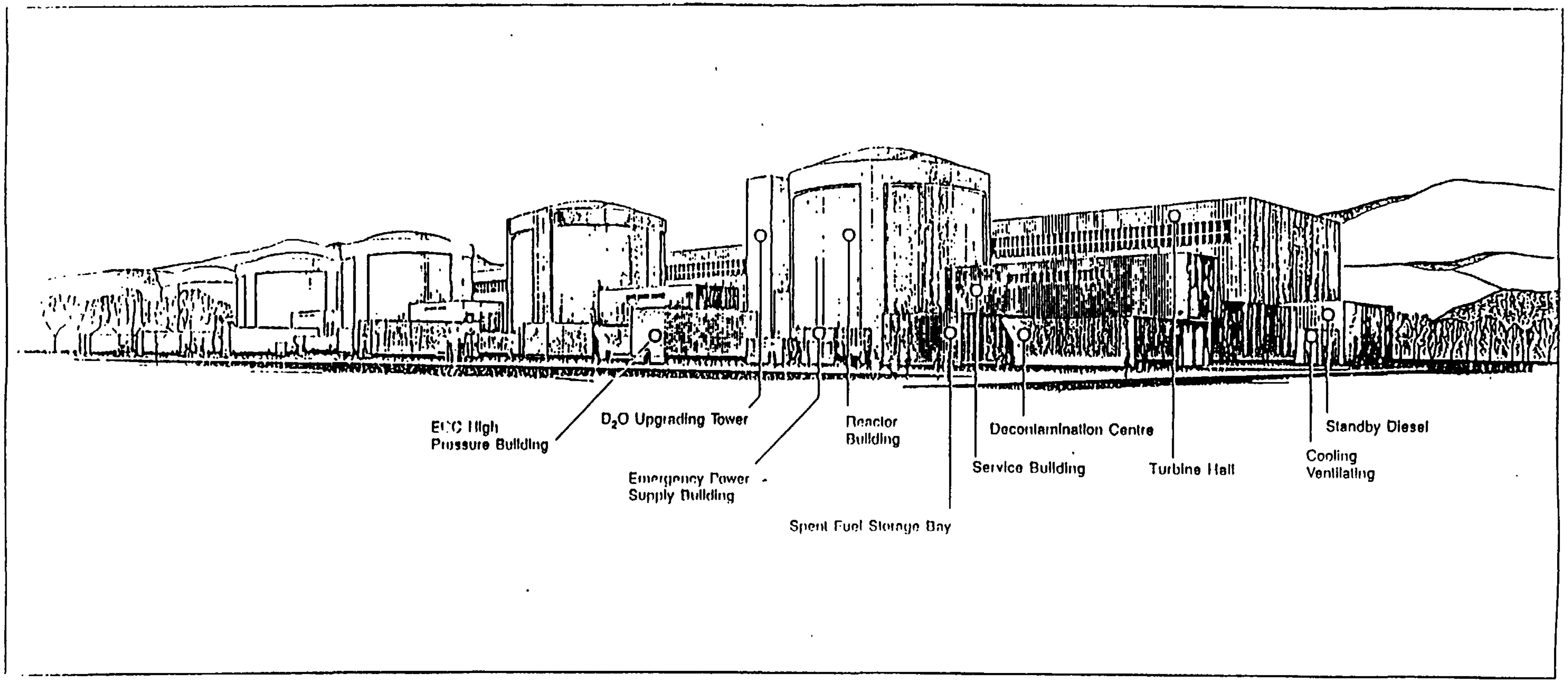
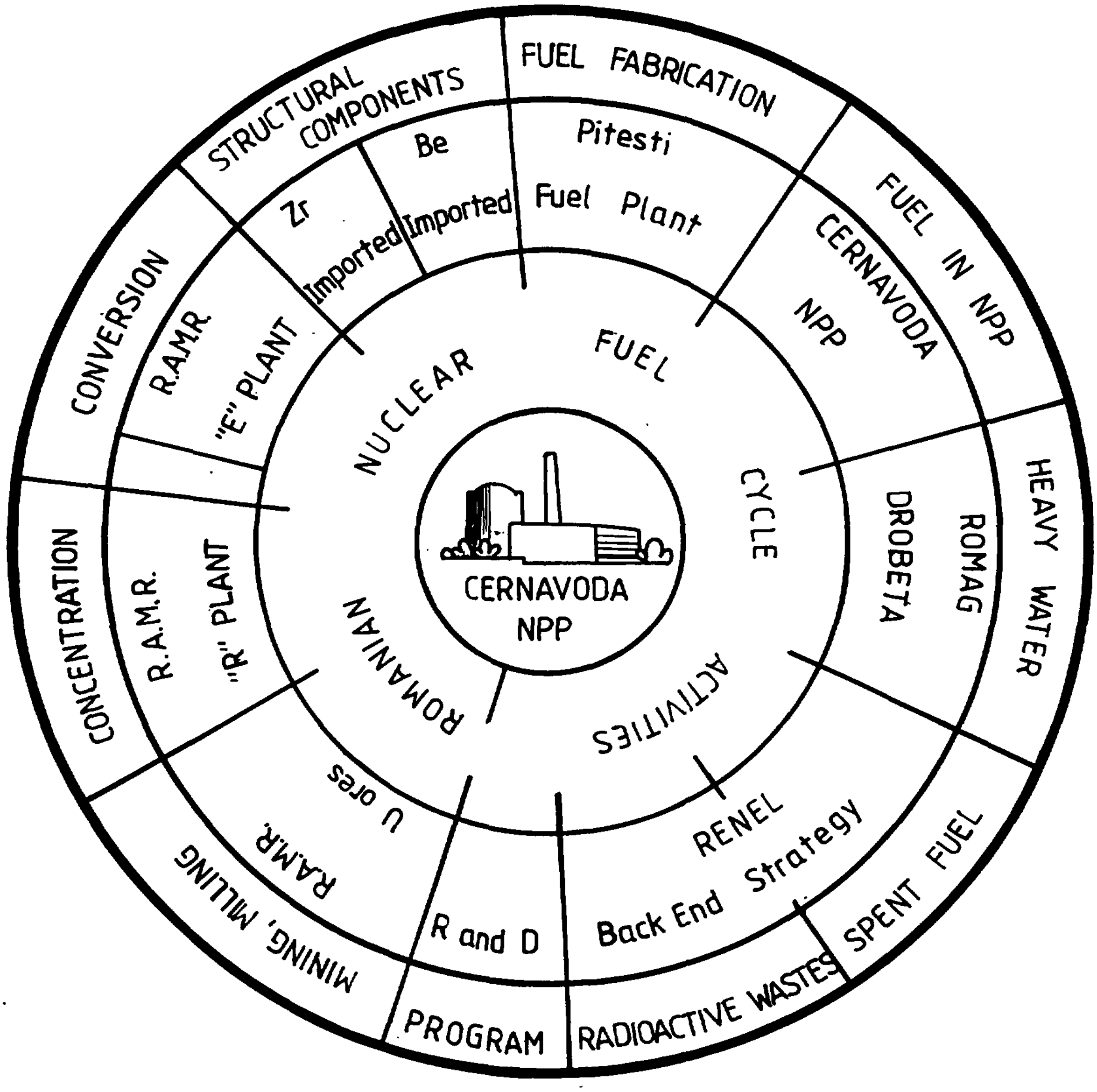


Fig. 1. 5 x 700 MWe Cernavoda NPP

After 1990 the production was reorganized and the process flow was modernized. In 1994-1995 the fuel plant was licensed as qualified CANDU nuclear fuel supplier by the Canadian company Zircatec Precision Ind. (ZPI) and AECL. Based on this license the first fuel load of Cernavoda NPP - Unit 1 contains a number of fuel bundles manufactured in Romania. The overall production capacity of the plant is 110 t/year , covering the annual consumption of Unit 1. However, this capacity will be extended so that all the fuel required for the future operation of the five units of Cernavoda NPP will be produced in Romania [6].



RENEL - Romanian Electricity Autority
RAMR - Rare and Radioactive Metals Board

Fig. 2. Nuclear Fuel Cycle - Related Activities

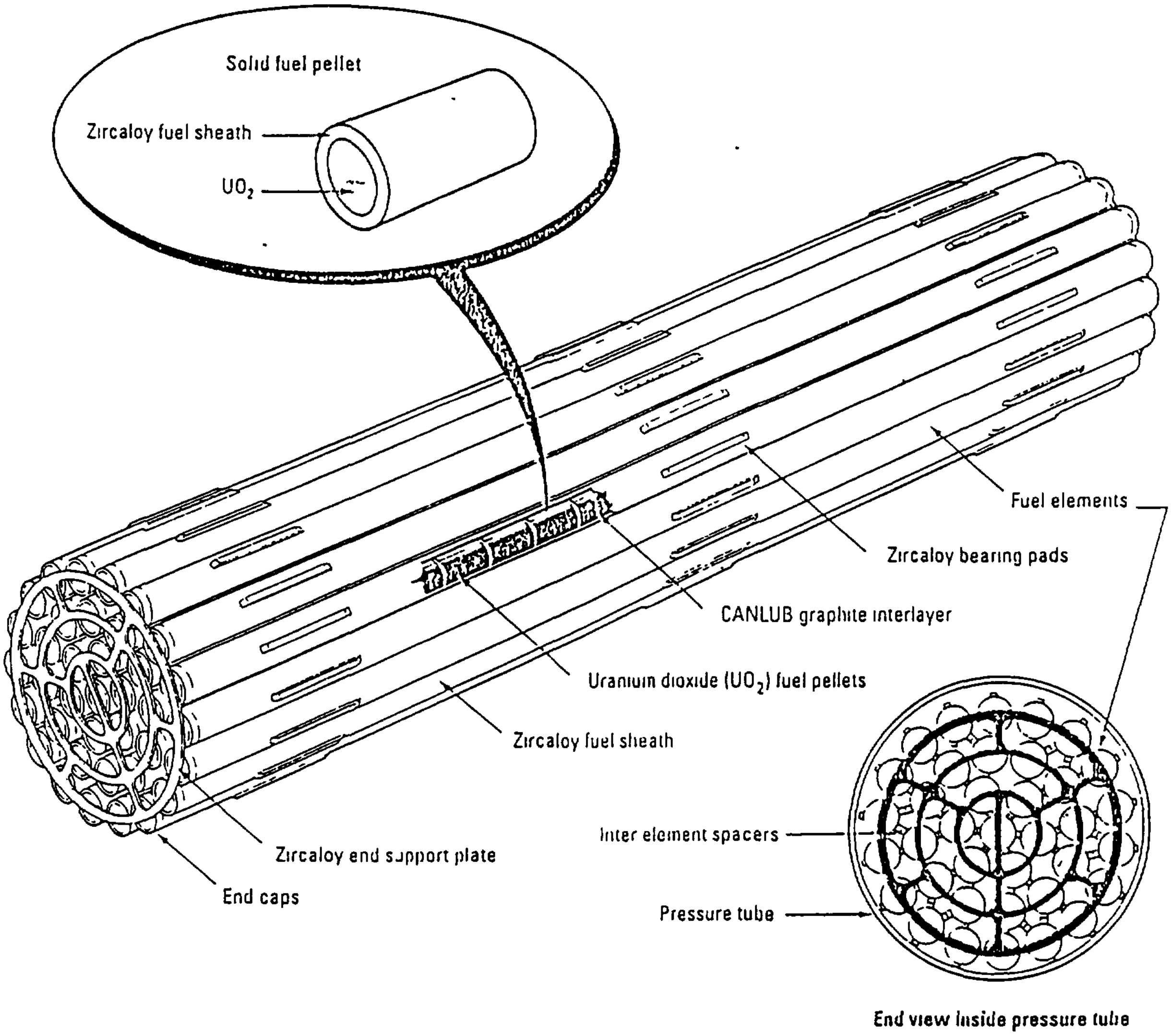


Fig. 3. CANDU - Fuel Bundle

The main future objectives in fuel fabrication field are:

- to supply the necessary fuel for Cernavoda;
- to recover the fuel fabricated before the qualification
- to assure fuel performance at international level;
- to introduce advanced fuel in CANDU for the next units.

2.2. Heavy water manufacturing

The possibility of Domestical heavy water production was another reason in the decision of the introduction of CANDU PHW reactors in Romania.

For reducing the costs and the dependencies associated with the supply of heavy water, it was decided to produce in Romania the entire quantity of heavy water necessary for the nuclear power program.

The manufacturing technology developed by Romanian specialists is based on the process of isotopic exchange between water and hydrogen sulfide, followed by a stage of vacuum distillation. The technology was tested on a pilot station developed by Institute of Cryogenics and Isotopic Separation (ICSI) Rm. Valcea and then an industrial facility was built.

At ROMAG, the Romanian heavy water plant, there are now in operation two out of four modules with a production capacity of 90 t/year each. It produces heavy water with a high purity level (99.90% Wt.), higher than CANDU technology requires. This opens important improvement opportunities since studies have shown that an 1% increase of deuterium concentration can lead to a 6-7 % increase of fuel burnup; the decrease of the fuel consumption results in changes of annual operation costs around 900,000 US dollars. Also, the heavy water produced at ROMAG has a high neutral ph (7.5±0.5) with favorable effects in reducing corrosion.

For the Unit 1 of Cernavoda NPP more than 180 t of heavy water were produced in Romania and 335 t were leased from Canada (75 t of leased heavy water was already returned in Canada). It is provided that all the necessary heavy water for the operation of the next units at Cernavoda NPP will be fabricated in Romania.

ROMAG is now the largest heavy water supplier in Europe with a design capacity of 360 t/year [7].

Romania has today specialized teams for each stage of heavy water production: design, construction, commissioning, operation, process control, special surface treatment for packing and G.S. Columns

An important aim for future developments is to assure a preventive maintenance and to reduce impact on population and environment.

An other objective in heavy water processing is to realize a D_2O detritiation facility to process tritiated heavy water from Cernavoda NPP. Such a project will to be completed at Institute of Cryogenics and Isotopic Separation (ICSI) Rm. Valcea.

2.3. CANDU reactor physics and fuel management

The CANDU reactors feature the unique characteristic of continuous refuelling during operation. The two refuelling machines and the associated systems are able to automatically transfer the fresh fuel bundles from store to the fuelling machine head, select the channel to be refuelled, perform the replacement of the irradiated fuel with the fresh one and transfer the irradiated fuel to the spent fuel storage pool in the plant. Since these operations do not require reactor shutdown, it opens the possibility of achieving high disponibility factors and thus a good economic efficiency of the plant.

However, this requires that physics calculation of the reactor core are performed on a routine basis (daily). The calculations take into account all the current operation events (real flux distribution, power levels, shutdowns) reflected in the irradiation history of each of the 4560 fuel bundles in the core and, based on these results, the plant physicist selects the fuel channels to be refuelled. The objectives and criteria considered when choosing the channels to be refuelled involve, between others, the maximization of fuel burnup and the uniformity of neutron flux distribution in the core, while maintaining the maximum channel power and the maximum fuel bundle power inside the allowed limits.

The contracts with the Canadian partner allowed for the transfer of the computer programs and the associated know-how. They also provided the specialization of Romanian physicists through tenures with foreign CANDU operators (Argentina and Canada) and through on the job training at the Cernavoda NPP during the first 18 full power months at Unit 1. These activities, together with the experience acquired in R & D applications have contributed to the creation of a strong national competence in this field in Cernavoda NPP and specialized institute in Pitesti and Bucharest.

3. Research and development

An ambitious program for research & development was developed in Romania to sustain the high level of national participation in national nuclear program.

As main components of this program, great support was given to research and development activities in areas, as:

- reactor physics and fuel management;
- nuclear safety;
- CANDU reactor improvements;
- risk assessment and evaluation of the radiological impact on the environment;
- development of technologies for fuel cycle, including manufacturing of the heavy water and of the CANDU nuclear fuel;
- design of equipment for dosimetry and radiological protection;
- behavior of materials in reactor flux;
- design of systems and equipment for the nuclear island and for the balance of plant;
- testing of the nuclear equipment assimilated by the national industry.

For these purposes in 1977, the Institute for Nuclear Power Reactors (IRNE), Pitesti, was established, with specialized divisions for reactor calculations and reactor safety, design of nuclear systems and components, technological developments for nuclear fuel, in-core and out-of-core testing facilities. In addition to that, the other tasks of research and development were distributed to other existing institutions with experience in nuclear physics, dosimetry, radiological protection, power engineering, electronics, electrotechnics, automation, metallurgy and mechanics, as the Institute for Atomic Physics (IFA) and the Center of Technology and Engineering for Nuclear Projects (CITON), both near Bucharest.

Important investments have also been made for commissioning and operation of major research and testing facilities. In this respect, we can mention the VVR-S and TRIGA reactors used for researches materials and nuclear fuel testing, the Hot Cells facilities, the testing facility for the CANDU Fuelling Machine and the EUROTEST facilities for environmental qualification (tests at vibrations, seismical and LOCA conditions) of electrical equipment used in nuclear power plants.

The resources and efforts invested in research and development have also contributed to creation of major opportunities for the development of a national professional competence in fields of major importance for the further progress of the nuclear program.

The National Agency for Atomic Energy (NAAE), the governmental body which ensues the promotion of activities and develops national strategy for peaceful use of atomic and nuclear processes and phenomena, plays an important role in the co-ordination of the research and development activities. NAAE has initiated various programs for analysis of the research projects, identification of the optimization possibilities of the technological processes in the nuclear fuel cycle and for increasing the degree of participation of the national economy to the Cernavoda NPP project. This proved necessary for recovering the disadvantages arisen from the lack of co-ordination of the research in the nuclear field in the period 1990-1994.

The main topics included in the 1995 and 1996 NAAE R&D contracts are:

1. Evaluation of the nuclear fuel cycle for the CANDU reactors and of the possible alternatives for installation of the nuclear groups at the Cernavoda NPP.

2. Evaluation of the possibilities for implementation of solutions for improving the performance and nuclear safety at NPP with CANDU reactors.

3. Evaluation of the opportunities for participation of the national industry at the completion of U2 - U5 of Cernavoda NPP.

The results will contribute to improvements identification in nuclear fuel cycle stages, heavy water manufacturing included, with cost benefits.

Another governmental institutions involved in nuclear power research and development are Romanian Electricity Authority (RENEL), Rare Metals Company (RAMR) and Institute of Atomic Physics (IFA), which carried out research & development program for HWRs through their main research & development institutes:

- Institute for Nuclear Research - ICN Pitesti;
- Center of Technology and Engineering for Nuclear Objectives - CITON, Bucharest;
- Research and Engineering Group of RENEL - GSCI;
- Institute of Research and Design for Rare and Radioactive Metals - ICPMRR, Bucharest;
- Institute of Cryogenics and Isotopic Separation - ICSI, Rm. Valcea.

RENEL research & development programs refer to:

- nuclear safety;
- nuclear fuel;
- CANDU technologies;
- radiation protection, decontamination;
- radwaste management and decommissioning ;
- improvement of operation and safety in heavy water production;
- radioisotopes, irradiation techniques and conversion of TRIGA - INR Reactor;
- computer assisting of activities.

All this components of research & development are co-ordinate by NAAE which bring it together in the national strategy of nuclear activities.

A list with the main research & development institutes and facilities is presented in Appendix 3.

4. Manufacturing and installation of components and systems for NPP

The increasing degree of participation of the national economy in the program of implementation of CANDU-600 in Romania was one of the main objectives of the agreements concluded with AECL. It envisaged the assimilation of new materials, assimilation and implementation of technologies for manufacturing of systems and components and introduction and implementation of advanced design codes and quality assurances procedures.

The implementation of this program required important financial efforts, with result of generating a modern and performant nuclear industry. As consequence, some 50 firms are now qualified as manufacturers of components, equipment and systems for nuclear plants. These units cover the production of :

- mechanical components (pipes, fittings, tanks, heat exchanger, CANDU fuel channel components and tools, reactivity control devices, filters, sub-systems for fuel handling and for fuelling machine, etc.)
- electrical installations and devices (low and medium voltage equipment, high voltage stations and networks, cables, etc.)
- technological equipment (pumps, valves, compressors, pre-heaters, Diesel groups, overhead cranes, etc.)

- instrumentation and control devices (dosimetry and radiological protection equipment, gamma monitoring system, tritium monitoring system, system for detection of D_2O leaks in H_2O, failed fuel location system, gas and liquid monitors, etc.).

Two other major component - nuclear fuel and heavy water - complete the list of industrial capability of Romania in nuclear field.

Now the Romanian industry is already an active presence in the reactors components market. In this respect we can mention here FECNE SA (the Nuclear Power Plant Equipment Manufacturing Company) Bucharest which produced emergency tanks for VVER-440 and for VVER-1000 power plants in the Czech Republic (Mohovce NPP and Duchovany NPP). The Romanian industry has also brought an important contribution, outlined in the attached promotional materials, in providing equipment and components for the Cernavoda NPP.

Lists with the most important supplier for nuclear equipment are presented in Appendix 1, 2.

A national competence in installation field of nuclear and conventional equipments was also developed through specialized companies, working under the quality assurance system in compliance with the specific national and international standards.

5. Engineering, technical assistance and general services

5.1. Technical assistance and project management

Starting from the opening of the work on the Cernavoda NPP site and until 1992 the technical assistance was insured by Romanian experts provided by specialized institutes as the Institute for Nuclear Power Reactors-Design and the Institute for Power Studies and Design, Bucharest. The size and composition of this group corresponded to the size and scope of the works and problems encountered on the NPP site. The immediate management and coordination of the work was insured in this period by the plant owner together with appropriate bodies in the Ministry for Electric Energy.

In 1992, according to the contract between the newly formed Romanian Electricity Authority (RENEL) and the AAC (AECL-Ansaldo Consortium), the whole process of completion, commissioning, grid connection and initial commercial operation for 18 months of the Cernavoda NPP - Unit 1 came under the responsibility of AAC. As result, a Project Management Team (PMT) was constituted, containing a Design Authority Representative (DAR) with specialists representing the Romanian design authority - CITON (Center for Technological Engineering for Nuclear Objectives). The responsibility of DAR span over the support systems of the plant.

The plant owner participates as secondant of AAC in the PMT regarding quality assurance program, nuclear safety, commissioning and operation activities; this position will continue for the duration of AAC involvement, according to the present contract. However, the Romanian personnel will gradually take over the responsibilities at both execution and management levels, so that after the first 18 months all the activities related to the operation of the Unit 1 will be performed only by Romanian staff.

Due to the great importance of this objective, the size of investment, the most important in Romania, the large number of employees, Cernavoda NPP has always been in the attention, and received great support from the political leadership of Romania. As it was underlined with the occasion of the achievement of the first criticality of Unit 1, April 17, this year, the government is giving a high appreciation to the efforts and results obtained at Cernavoda NPP, and full support for development of the nuclear power program in Romania as well, which involve the completion of the 5 x 700 CANDU-PHWR at Cernavoda NPP.

5.2. Quality assurance

In 1981 the first QA program in the nuclear field was elaborated for management of activities related to erection of nuclear objectives and installations. This program was structured based on Canadian and IAEA standards. IAEA subsequently, it was reviewed and updated with new Romanian standards, at international level, developed after 1983, for activities in design, construction, commissioning and operation of nuclear facilities. Also, areas as personnel training, metrology control, provisioning, as well as audit, corrective actions, document control and quality records were put under QA requirements. Later, in 1992, the law of introduction of QA standards at nuclear level for suppliers of products and services and for entrepreneurs was adopted, so that now all the participants to the construction of NPP have implemented QA programs.

The established QA programs were confirmed and are continuously monitored through technical examinations performed by experts from AECL and from the Institute for Research and Modernization in Power Engineering, Bucharest. This ensures that the level of quality of the products and services provided by Romanian firms in the nuclear field are fully at the level required by the technical specifications ordered by the customer.

The national competence in QA programs, standards and procedures is represented by National Commission for Control of Nuclear Activities (CNCAN)

5.3. Nuclear safety

In Romania original legislation and authorization procedures are in place, based on approaches developed in other advanced countries (USA, Canada) and able to respond to the IAEA reccomendations. These ensure the compliance of all the activities performed on Cernavoda site and at other nuclear objectives, with safety requirements at international level. In 1995 Romania has signed the Nuclear Safety Convention which stipulate a continuous improvement and update of safety requirements in the pace of international evolutions.

The Romanian regulatory body is the National Commission for Control of Nuclear Activities (CNCAN). Although CNCAN operates as department of the Ministry of the Waters, Forests and Environment Protection, it enjoys a great independence from politic, economic and other interests. CNCAN displays a strong disponibility for safety analyses, inspections and authorizations release activities.

In parallel to that, since the original agreement between Romania and Canada did not include co-operation in detailed safety aspects of the CANDU-600 reactor, a national competence in nuclear safety was developed in Romania. As result, the Preliminary Safety Analysis, the Preliminary Safety Report and the Final Safety Report - Phase A, as well as safety assessments for design changes and updating at Cernavoda NPP are performed in Romania. Romanian staff is now successfully completing activities related to the licensing of the Cernavoda NPP Unit 1.

Romania disposes, through specialized institutes as the Institute for Nuclear Researches (ICN), Pitesti, and the Center for Technology and Engineering for Nuclear Projects (CITON), Bucharest, of a computer programs and data bank and the capability to perform deterministic and probabilistic safety analyses for a CANDU NPP, for study and authorization purposes.

The main safety topics CITON and INR are involved in, are: operation and licensing, probabilistic safety analyses, accident analyses, sever accident management, pressure tube stress analyses, fuel behavior, CANDU improvement.

5.4. Waste management and ENVIRONMENTAL PROTECTION

The main purpose of waste management and environmental protection program is to design and commissioning processing and storage facilities for all types of wastes resulted from the operation of NPP and from other nuclear units as well as for spent nuclear fuel elements with respect of environmental protection and safety rules [8].

For storage of both the average and low level active wastes, the results of a series of studies were elaborated, lead to the possibility of building a Final Storage Facility, close to Cernavoda NPP.

Along 1995, a pre-feasibility study for the Final Storage Facility of both low and medium active wastes has been issued, this building following to be put into operation in 2002, as the latest date.

Referring to the management of the spent fuel, the studies elaborated at present by CITON have shown the necessity of performing a Spent Fuel Intermediate Storage which has to be put into operation during 2002 ÷ 2003, considering that the optimum storage period of the spent fuel at NPP is about 7 ÷ 10 years since NPP commissioning, depending also on the depositing type which can be either dry or wet. Under such circumstances elaboration of a pre-feasibility study should be advisable starting with 1996, when the optimum technical solution should be settled.

Another major project of this program represents the building of the Final Store for Spent Fuel, forecasting its commissioning somewhere towards 2025 years.

All these studies will finalize the first stage of radioactive wastes and spent fuel management obtaining thus, all the design data for opening up, in 1997, the investments of the following projects:

- Final Store for Low and Medium Active Wastes (DFDSMA), having its commissioning date: 2002;
- Spent Fuel Intermediate Store (DICA) having its commissioning date: 2002-2003.

In the next stage, the period 1997-2002, the program will be in progress by supporting the performance of all the proposed investments in the field of developing the techniques, methodologies, testing procedures of radioactive wastes storage containers and casks, radioprotection calculations, evaluation of performance etc.

We would like to underline here the permanent support which was granted by the IAEA through technical co-operation programs for our research and development works.

5.5. Personnel training

A well trained competent management and technical staff is essential to the safe and reliable operation of nuclear power station as well as of other nuclear facilities.

The qualification and training of personnel, able to operate the station and satisfy nuclear facilities requirements, is achieved through a program, whose objective is to ensure the training at an appropriate level of quality.

In this respect in Romania has developed a specific training program which include, at different levels:

- post secondary school specialized in nuclear power sciences
- departments for nuclear engineering and nuclear physics at technical and science universities
- on the job training at Cernavoda N.P.P., including use of full-scope simulators
- specific training in Canada and Romania for personnel for other research and nuclear facilities.

The training program has been developed in the '80's. The main purpose, the training of Romanian personnel for nuclear plant design, operation and maintenance, was limited and the opportunities offered by the existing agreements or by the IAEA were not fully used before 1990.

A new training program for all types of personnel involved in nuclear power program and R & D activities is in train to be defined and developed to assure the necessary level of training required by the international legislation and practices.

6. Training simulator for PHWR

6.1. General considerations

International experience of the NPPs operating demonstrated that around 50% from the major NPP accidents are due to the human factors and an important objective for a safe work of NPPs consists in the control room operators training using sophisticated tools like full-scope simulators, located in the frame of the NPP Training Center, replica of the NPP control room. This full-scope simulator is used to train the team who work in the control room and to validate this team, but it is improper for individual training. To offer the individual training possibility, the NPP Training Centers are equipped with simplified training simulator like MicroSimulators. These tools consists in a computers network on which screens the imagine of the control rooms panels is reproduced to show the evolution of the process variables and to take over the operator manoeuvre.

To assure a safe operating of the Cernavoda NPP, all these aspects of the control room operators training were kept in the Cernavoda Training Center. To realize this important objective, the works were started ten years ago, involving two Romanian institutions: Nuclear Research Center Pitesti (ICN), and Center of Technology and Engineering for Nuclear Projects Bucharest (CITON). In the first stage, the Reactor Panel Simulator was realized using a Romanian computer named CORAL. After December 1989, the modern and powerful computers became available for simulations purposes, and international collaboration programs were initiated through the IAEA.

In 1993 RENEL decided to buy a full-scope simulator from the Canadian company: CAE Electronics Ltd., for the Cernavoda Training Center. Also, in September 1992 was started the IAEA Project ROM/0/004 "Support of Cernavoda Training Center", and this project included four work packages (WP) developed by the Spanish company Technatom SA:

1.WP1 defined the project content and schedule.

2. WP2 included the phase of the training, and 5 sections were included: WP2.1 Training for elaboration and use of the acceptance test procedures (ATP); WP2.2 Local area network training; WP2.3 UNIX and simulation architectures course; WP2.4 Simulation models training; WP2.5 Instructor station and interactive graphic simulator features.

3. WP3 Elaboration phase included two sections: WP3.1 Microsimulator improvement; WP3.2 Simulator configuration management.

4. WP4 Training implementation phase included six sections: WP4.1 Evaluation and design review of Training System Development; WP4.2 Methodology to elaborate training manuals; WP4.3 Training programs for instructors; WP4.4 Analysis of Romanian training material; WP4.5 Evaluation of training program using Simulator; WP4.6 Continuing Training.

As deliverables from the project were included: Project Specifications Report, Comments to the Full Scope Simulator ATP's sample, Courses Documentation, Courses Reports, Certificates of training Attendance. The computer platform (hardware and software configuration) to support the Microsimulator development, The Tecnatom Microsimulator Environment and associated licenses. This project was completed in September 1995.

Now, in Romania exists the possibilities to realize Training Simulator for Nuclear Training Centers, and we can split these simulators into two main categories: Full-Scope Simulators (FSS) and MicroSimulators (MS). As for the beginning, it is preferable to be involved in a program for MicroSimulators fabrication, and this type of simulator is considered in the followings.

6.2. Romanian concept

Four main sections concur to Microsimulator (MS) fabrication: (a) Hardware Fabrication, (b) Real Time Execution Software, (c) Simulation Models, (d) Graphic Screen Programs. In the first and second section, not specific activities for nuclear field are included, because the powerful computers fabrication with an adequate operating system is an objective for all simulators. The third and fourth sections are particular for nuclear fields, and these activities can be developed by specialists in NPPs.

Hardware Fabrication

In this section all equipment fabrication it is included. The Microsimulator (MS) hardware includes a small PC-Network with 2 or more Personal Computers (PC), and one PC must be equipped with a powerful processor like Intel-Pentium, with 130 MHz or more. Also, the powerful computer can be substituted by an workstation equipped with RISC processor and UNIX operating system. In this case the cost of the Microsimulator gets up.

Real Time Execution Software (RTES)

RTES is the main task which assure : (1) simulation program control, (2) execution control of models program, (3) debugger for simulation program, (4) database program for data area, in a real time execution.

Simulation program control (SPC)

SPC includes the following functions: simulation load generation, simulation execution management, simulation control management, data interchange, snapshot.

Execution control of models program (ECMP)

ECMP includes models ordered execution and execution time measurement.

Debugger for simulation program (DSP)

DSP allows the simulation status monitoring, simulation variables values monitoring, simulation variables values setting, dynamic monitoring and to save values to files.

Database program for data area (DBP)

DBP is an auxiliary executive used to create or to update the variables databases.

Simulation Models (SM)

Simulation Models are executives to simulate, through a mathematical model, the dynamic operating of the NPP's systems. Some tens of executives are used to complete all the NPP's systems, and all executives use a common memory area for their common variables.

Graphic Screen Programs (GSP)

Graphic Screen Programs (GSP) are interfaces between the users, instructor or students, and the simulation program. Usually GSPs are installed on the non-powerful PCs of the Microsimulator Network. These PCs are used as a instructor or/and student station. GSP assure a friendly interface between user and the values of the variables simulated by the Simulation Models. Also, GSP allows the instructor or students to change dynamically the value of one or more variable and to initialize a malfunction or a manoeuvre

6.3. Romanian possibilities

Romanian Institutions, has the possibility to realize all these sections of the Microsimulator fabrication.

Hardware Fabrication

Using abroad-made components, it is possible to be made in Romania the Microsimulator Network Pcs. Also, it is possible to realize the communication interfaces between these Pcs, because the computer technique is now accessible.

Real Time Execution Software

This software is the Romanian main achievement. Remarkable is that this software run on the low cost Pcs, under Windows'95 operating system. Also, for the Hewlett Packard workstations, it is available the UNIX Real Time Execution System developed by Spanish company Tecnatom SA and delivered to a Romanian company in the frame of the above mentioned IAEA project.

Simulation Models

It is available and active all the standard algorithms to simulate the NPP processes like: nuclear reactor physics and thermodynamics, heat transfer, hydraulics of the nuclear and conventional circuits and equipment, electrodynamics of the electric systems, and the logic of the control systems. For all these types of processes, algorithms and simulation methods are available and Romanian institution involved in nuclear program work now to realize model builders.

Graphic Screen Programs

It is available a graphic library to write quickly software for graphic screen programs and interfaces between users and models, including indicators, buttons, and other equipment of the control room panels. Also, flow chart elements are available to quickly draw a flow diagram for the NPP system. In a short time will be available a graphic screen builder and an interface builder.

Romanian Nuclear Authorities paid a special attention to the Cernavoda NPP operators training: the Cernavoda Training Center was built; a full-scope simulator was bought from the CAE Electronics Ltd., the main company for CANDU simulators fabrication; a special program was developed with the Tecnatom SA, under the IAEA umbrella. Romanian companies involved in the nuclear program are now able to complete a training simulator program and to realize this important tool including the documentation. Romania is interested to maintain and improve these capabilities through its participation in international program developed by IAEA.

7. Conclusions

The key feature of the Romanian nuclear program was the implementation of the CANDU - PHWR, with a comprehensive participation of the domestic economy, in an increasing degree from the first to the last NPP unit. Thus, the introduction of the nuclear energy was regarded as a major opportunity for modernization of economy through endowment with modern equipment of industrial companies, assimilation of new technologies and implementation of modern design codes and QA procedures. It also allowed for the creation of a national competence in the field based on theoretical and experimental research programs in nuclear physics pursued in various universities and research institutions. this is a required for solving current operation problems, introduction of technological improvements and updating and for the further operation of the plant.

For insuring an optimum correlation the activities in different areas, unitary co-ordination proved to be necessary. For this reason, in order to cope with the situation arisen after 1990, characterized by the lack of a complete legislation system and appropriate organizational infrastructure, the National Agency for Atomic Energy was created, with responsibilities in defining the strategies and development programs in the field and in co-ordination of the horizontal industry, research and development institutions efforts, and of the international co-operation, in the accomplishment of the objectives proposed. In this respect, the National Nuclear Program, which is now under the elaboration and the co-ordination of NAAE will define, in an integrated unitary approach the objectives, and directions of work in all the related areas of activity until the year 2010.

A reference moment in the development of the nuclear power in Romania was the successful achievement of the commissioning of Cernavoda NPP - Unit 1. It also provided the opportunity for renewing the political support given for the completion of this project. We would like to underline here our co-operation with the foreign partners, AECL-Ansaldo Consortium, as well as by the IAEA in research, development, personnel training, as well as in all the other activities related to introduction of nuclear power in Romania and completion of the Cernavoda NPP project.

We trust further progress in the finalization and the good expected operational performance of Unit 1 in terms of economic efficiency and safety, will be able to provide a boost in the development of the activities mentioned in the National Nuclear Program, and in the public acceptance of the nuclear power as well.

Main lessons learned from Romanian Nuclear Development Program are:
- to develop a governmental body to ensure national strategy and nuclear activities promotion;
- to select a partner with high scientific and technological potential and disponibility to transfer it;
- to ensure a good management of the project with involvement of the partner and of the local resources;
- to ensure clear defining at all the levels of competencies and responsibilities;

- existence of a regulatory body independent from the plant owner;
- to ensure implementation and compliance with the technical standards, QA system and safety requirements in all stages of the program;
- to establish a national educational and training system specific to the particular design;
- to develop a national competence in research development, engineering and industry;
- to define the optimum level, the categories of equipments, and the area of national participation;
- to correlate national participation and nuclear industry development with nuclear power program dimensions
- to manage the transfer of technology to improve industrial level.

REFERENCES

[1] RAPEANU, S.N., BUJOR, A., CRISTIAN, I., "Nuclear Energy in Romania, Workshop World Watch 1996 (WWW '96) on Sustainable Development and International Co-operation", Bucharest - Romania (march 1996).

[2] RAPEANU, S.N., BUJOR, A., COMSA, O., "The Romanian Experience on Introduction of CANDU-600 Reactor at the Cernavoda NPP", AGM on Status and Introduction of Small and Medium Power Reactors in Developing Countries, Rabat, Morocco (October 1995).

[3] RENEL-Romanian Electricity Authority, "Annual Report - 1995", Bucharest, Romania (June 1996).

[4] GEORGESCU, D., VASILE, C., "Uranium Resources Evaluation for Nuclear Power Program", Bucharest, Romania, (October 1990).

[5] GALERIU, A.C., PASCU, A., "Nuclear Fuel Fabrication in Romania", Third International Conference on CANDU Fuel, Chalk River, Canada (October 1992).

[6] GALERIU, A.C., ANDREI, Gh., BAILESCU, A., PASCU, A., ILIESCU, M.L., "Romanian - Canadian Joint Program for Qualification of FCN as CANDU Fuel Supplier", Fourth International Conference CANDU Fuel, Pembroke, Canada (October 1993).

[7] PAVELESCU, M., "Romanian Heavy Water Production Features", International Nuclear Industry Exhibition, Beijing, China (March 1996).

[8] MINGIUC, C., PANAIT, A., RAPEANU, S.N., CHIRICA, M.T., "Main Objectives of Romanian HWR Program", Technical Committee Meeting on Advances in Heavy Water Reactors, Bombay, India (February 1996).

[9] RAPEANU, S.N., BUJOR, A., COMSA, O., "Romanian Nuclear Development Program", EAES Combined Meeting 1996, Aghia Pelagia, Crete, Greece, (May 1996).

Appendix 1

NUCLEAR FUEL CYCLE FACILITIES

Institution	Contact person
Rare Metals Company (RAMR) Bucharest	**Manager**: Constantin Bejenaru **Phone**: 401-659.49.88 **Fax**: 401-659.20.35 **Profile**: Exploration, exploitation and processing of U, T_i, M_o, B_i, C_o, Z_n, N_i
"R", "E" Plant Feldioara,	**Manager**: Constantin Bejenaru **Phone**: 401-659.49.88 **Fax**: 401-659.20.35 **Profile**: Uranium concentrates production
Fuel Bundle ManufacturingPlant (FCN)-Pitesti	**Manager**: Constantin GALERIU **Phone**: 40-048.68.10.80 40-048.61.26.50 **Fax**: 401-613.12.58 **Profile**: Fuel bundle fabrication
Cernavoda Nuclear Power Plant	**Manager**: Viorel MARCULESCU **Phone**: 40-041.23.96.46. **Fax**: 40-041.23.96.79 **Profile**: Electricity production
"ROMAG" Drobeta- Tr. Severin	**Manager**: Mihai POPA **Phone**: 40-052.22.23.97 **Fax**: 40-052.31.79.08 **Profile**: Heavy water production
STDR Pitesti	**Manager**: Serban VALECA **Phone**: 401-312.58.96 **Fax**: 401-312.58.96 **Profile**: Fuel cycle activities LLW processing and storage
STDR Bucharest	**MANAGER**: Gheorghe MATEESCU **PHONE**: 401-780.70.40 **FAX**: 401-420.91.50 **PROFILE**: Research activities LLW processing and storage
"Baita" Repository	**Manager**: Constantin Bejenaru **Phone**: 401-659.49.88 **Fax**: 401-659.20.35 **Profile**: LLW repository

Appendix 2

ROMANIAN NUCLEAR INDUSTRY

Institution	Contact Person	
FECNE Bucuresti S.A	**Manager:**	Virgiliu Craciunescu
	Phone:	401-330.84.60
	Fax:	401-330.34.04
GENERAL TURBO S.A.	**Manager:**	Dumitru CONSTANTIN
	Phone:	401-684.03.20
	Fax:	401-311.03.58
UMGB S.A.	**Manager:**	GIURGIU
	Phone:	401-410.15.00
	Fax:	401-312.39.28
TITAN Echipamente Nucleare	**Manager:**	Mihail CIOABA
	Phone:	401-628.64.80
	Fax:	401-312.81.00
IMGB S.A.	**Manager:**	Constantin COJOCARU
	Phone:	401-683.39.95
	Fax:	401-684.69.30
UZUC Ploiesti	**Manager:**	Vasile DASCALESCU
	Phone:	40-044.14.36.51
	Fax.:	40-044.11.03.29
VENTILATORUL S.A. Bucuresti	**Manager:**	Pompiliu IONESCU
	Phone:	401-410.22.76
	Fax:	401-312.34.87
UPET Targoviste	**Manager:**	Oprea BANGHEA
	Phone:	40-044.63.16.00
	Fax:	40-045.61.71.05
AIT Electroputere Craiova	**Manager:**	Cornel MONDREA
	Phone:	40-051.14.44.41
	Fax.:	40-051.19.08.29
Uzinele Electroputere Craiova	**Manager:**	Ioan LUPULESCU
	Phone:	40-051.14.77.53
	Fax:	40-051.19.98.97
FCN Pitesti	**Manager:**	Constantin GALERIU
	Phone:	40-048.61.41.60
	Fax:	40-048.61.26.50
ROMAG Drobeta Tr. Severin	**Manager:**	Mihai POPA
	Phone:	40-052.22.23.97
	Fax:	40-052.31.70.08
Uzina "R", "E" Feldioara	**Manager:**	Constantin BEJENARU
	Phone:	401-659.49.88
	Fax:	401-659.20.35
Nuclear Montaj Cernavoda	**Manager:**	Aurel OVEZEA
	Phone:	40-041.23.76.30
	Fax:	40-041.23.89.04
ICPE Electrostatica S.A.	**Manager:**	Radu CRAMARIUC
	PHONE:	401-312.61.99
	FAX:	401-312.61.98

MICROELECTRONICA	**Manager:**	Emilia GHIATA
	Phone:	401-633.44.45
	Fax:	401- 679.55.75
AUTOMATICA S.A.	**Manager:**	Liviu MUNTEANU
	Phone:	401-212.28.04
		401-633.30.81
	Fax:	401-212.28.41
ICME S.A.	**Manager:**	Stan TEODOR
	Phone:	401-312.81.45
	Fax:	401-312.81.45
FEA S.A.	**Manager:**	Grigore NEPELCU
	Phone::	401-312.76.83
	Fax:	401-212.15.51
AVERSA S.A. Bucuresti	**Manager:**	Andrei PANOIU
	Phone:	401-635.19.63
	Fax:	401-642.35.93
CASTUMAG S.A. Bucuresti	**Manager:**	Nicolae SIANU
	Phone:	401-652.10.20
	Fax:	401-337.35.04
ENERGOMONTAJ S.A. Bucuresti	**Manager:**	Viorel PETRESCU
	Phone:	401-622.50.54
	Fax:	401-322.04.13
		401-321.13.60
ELECTROTEHNICA S.A. Bucuresti	**Manager:**	Adrian ROTARU
	Phone:	401-745.73.55
	Fax:	401-312.13.60
MECANICA FINA S.A. Bucuresti	**Manager:**	Mircea RUSU
	Phone:	401-635.00.00
	Fax:	401-250.31.16
SC RETROM S.A. Pascani	**Manager:**	Radu NEMTEANU
	Phone:	40-032.76.24.59
	Fax:	40-032.76.41.05
S.C. PEROM S.A. Bacau	**Manager:**	Gheorge Gabriel OLTUL
	Fax.:	40-034.17.35.48
APARATAJ ELECTRIC S.A. Titu	**Manager:**	Ion DAIA
	Phone:	40-045.65.04.65
	Fax:	40-045.65.04.12
TRAFO Electroputere Craiova	**Manager:**	Ionel BALAN
	Phone:	40-051.14.43.95
	Fax:	40-051.19.08.51
ELBA Timisoara	**Manager:**	Gheorghe COCEAN
	Phone:	40-056.19.03.14
	Fax:	40-056.19.00.58
BETA Buzau	**Manager:**	Cornel MARICA
	Phone:	40-038.43.53.32
	Fax:	40-038.71.07.79

Appendix 3

NUCLEAR RESEARCH AND DEVELOPMENT INSTITUTIONS

Institution	Contact person
Center of Technology and Engineering for Nuclear Projects (CITON), Bucharest	**Manager**: Adrian PANAIT **Phone**: 401-780.69.25 **Fax**: 401-420.88.16 **Adress**: Bucharest - Magurele, P.O.Box: 5204-MG4 **Profile**: Design and engineering services for nuclear and conventional projects
Nuclear Research Institute (ICN), Pitesti	**Manager**: Serban VALECA **Phone:** 401-312.58.96 **Fax:** 401-312.58.96 **Adress**: P.O.Box 78 Code:0300Pitesti **Profile**: Research and development in nuclear power and material testing
Research and Engineering Group of RENEL (GSCI), Bucharest	**Manager**: Mihail PETRESCU **Phone:** 401-321.69.66 401.323.67.30 **Fax**: 401-321.10.10 **Adress**: 8, Energeticienilor St. Code: 79619 Bucharest **Profile**: Power plant general research and engineering
Research and Design Institute for Rare and Radioactive Metals (ICPMRR), Bucharest	**Manager**: Ion IONESCU **Phone**: 401-613.52.54 401-615.78.94 **Fax:** 401-613.12.58 **Adress:** 78, Republicii Bd. Code: 70132 Bucharest **Profile:** Research and design of mining, milling and uranium processing technology
Institute of Atomic Physics (IFA), Bucharest	**Manager:** Teodor NECSOIU **Phone:** 401-420.14.47. **Fax:** 401-420.91.50 **Adress:** P.O.Box MG-6 **Profile**: Theoretical and experimental analysis in nuclear and atomic physics

Institute of Cryogenics and Isotope Separation (ICSI), Rm. Valcea	**Manager**: Ioan STEFANESCU **Phone**: 40-050.71.27.44 **Fax:** 40-050.71.27.46 **Adress**: P.O.Box 10 Code: 1000 Rm. Valcea **Profile**: Cryogenic and separation processes for hydrogen isotopes (deuterium and tritium) research and development
Institute of Physics and Engineering (IFIN), Bucharest	**Manager**: Gheorghe MATEESCU **Phone**: 401-780.70.40 **Fax:** 401-420.91.50 **Adress**: 1 Atomistilor St. Code: 76900 Bucharest - Magurele **Profile**: Research and development for nuclear technology
National Institute of Materials Physics (IFTM), Bucharest	**Manager**: Mircea MORARIU **Phone**: 401-780.34.69 **Fax:** 401-420.37.00 **Adress**: P.O.Box MG-6 **Profile**: Physics and Technology of Materials
Metallurgical Research Institute (ICEM), Bucharest	**Manager**: Vasile URSU **Phone:** 401-220.55.06 **Fax:** 401-220.42.95 **Adress**: 39, Mehadia St. Code: 77769 Bucharest **Profile**: Research and development for special Alloys
EUROTEST S.A., Bucharest	**Manager**: Ovidiu DUMITRESCU **Phone**: 401-321.72.42 **Fax:** 401-323.26.28 **Adress**: 313, Splaiul Unirii Code: 73204, Bucharest P.O.Box 4-77 **Profile**: Research, equipment testing engineering and scientific services
Research and Development Institute for Electrical Engineering (ICPE), Bucharest	**Manager**: Vasile NICOLAE **Phone**: 401-322.28.13 401-322.01.10 **Fax**: 401-321.37.69 **Adress**: 313, Splaiul Unirii Code: 74204, Bucharest **Profile**: Research and development for electrical engineering

NEXT PAGE(S) left BLANK

XA9846715

THE MAIN STEPS OF THE ROMANIAN NUCLEAR POWER PROGRAM DEVELOPMENT – ACCUMULATED EXPERIENCE

T. CHIRICA, D. POPESCU, M. CONDU, M. VATAMANU
Romanian Electricity Authority,
Romania

Abstract

The paper presents a historical summary of the Romanian Nuclear Power Program development, providing details for: the main criteria and principles the Program was based upon, the contracts signed with the foreign partners to implement it, and the national participation (site contractors, suppliers and design organizations). The effect of the equipment assimilation program on the NPP Cernavoda (5x700 MWe) and especially on Unit 1 schedule and performance is analyzed.

Further on the impact of the transition from centralized to a market economy over the Romanian Nuclear Power Program development is analyzed, providing information's on its actual status and perspectives for the next 20 years. A description of the NPP Cernavoda Unit 1 actual progress and of the main steps performed by RENEL to get finance to complete NPP Cernavoda Unit 2 is included.

Finally there is summarized the accumulated experience, and its feed back on RENEL strategy to complete NPP Cernavoda Unit 2.

1.0 INTRODUCTION

The Romanian nuclear power program has been developed around the first Romanian nuclear power plant (NPP) sited at Cernavoda, in the south-east area of Romania, in Dobrogea region on the right side of the Danube River, about 160 km east of Bucharest.

Cernavoda NPP will have, at the final capacity, 5 nuclear reactor CANDU-type, turbine-generator units, each of them with a 700 MW nameplate power.

2.0 THE HISTORY

2.1 Before December 1989

The story of our nuclear power program goes back to the first contact with suppliers in 1960's. In 1977 the Romanian and the Canadian Governments formally agreed to co-operate in the field of peaceful use of atomic energy. A joint team prepared a feasibility study which led to the decision that CANDU-6 was the basic plant upon the which the Romanian build its nuclear program. The option for a Western technology was based on several reasons:

-the CANDU-6 plant was a proven one with good experience in construction and operation in developing countries;

-the CANDU-6 reactors have excellent operating and safety records;

-the CANDU design placed a singular emphasis on safety matters (containment, seismic design) which were in advance of approaches widely accepted at that time in Eastern Europe;

-the use of natural uranium as fuel and heavy water as coolant and moderator enables Romania to be self sufficient in nuclear power;

-the process equipment does not require as large an investment in manufacturing plants as that for other types of nuclear power stations.

In December 1978 Romania concluded CANDU licence contract with AECL as well as other contracts by which, the Canadian party provided engineering and technical assistance services, equipment and materials procurement from import, necessary for the nuclear part of the unit 1. The services, engineering and procurement contracts were extended for unit 2, in 1981. In February 1981 it was signed the contract for the conventional part of the unit 1 and 2 (turbogenerator set, electric generator and their auxiliaries) with General Electric (USA) and Ansaldo Spa (Italy) companies.

Increased efforts were made in Romania to manufacture many components in the country. In some areas this meant that new industrial technologies had to be introduced in many areas with the inevitable delays and problems associated with the learning period. In general many of the basic hardware technology problems have been solved, but there have been difficulties regarding aspects of quality and quality assurance documentation.

Limited work was also started on the Cernavoda units 3, 4 and 5 under the licensing agreement with AECL.

2.2 From December 1989 to present

The new political order installed in Romania after December 1989 recognised that the construction and operation of a nuclear power station requires transparency and the awareness of the necessity to perform all the work and tasks meeting all the requirements.

In October 1990, the IAEA was asked by the Romanian nuclear regulatory body to review the project. A Pre-OSART team confirmed the stop of work and repair programme recommended by the Romanian management and AECL in January 1990. The key IAEA recommendations were:

- to give more responsibility and financial control to the owner of the station;
- to reduce interference from various government ministries, in order to promote quality;
- to implement a proper management and to ensure adherence to established procedures;
- to seek and enhance expert assistance from outside Romania, especially in site construction management.

The Romanian Electricity Authority (RENEL) and the consortium formed by AECL-Canada and ANSALDO-Italy (AAC), signed a new contract, in August 1991, which enabled the consortium to perform the project management with the aim of commissioning Cernavoda unit 1 by the end of 1995. The Consortium will operate the plant for the first 18 months, and provides both formal and on-the-job training for the Romanian personnel who will operate the unit when the AAC leaves the site. A group of Romanian specialists is working on the project management team (PMT), under the direction of the Canadian and Italian managers, in order to acquire the necessary management skills.

To reduce hard currency expenditures for imported fuels, the new strategy focus on completed unit 1 as quickly as possible. The strategy included:

- Resolving the complex government approach. Now, one customer is responsible for the project - RENEL the newly formed Romanian utility.
- Improving the site and living conditions at Cernavoda. The government has implemented a large package of infrastructure improvements, including new housing for the workers, a hospital and shops.
- Limiting work on other units to preservation activities, in order to focus on completing unit 1, including the materials and resources.

2.3 The project management team (PMT) and its activities

The signature of the Project Management Contract (PMC) between RENEL and AAC represented the start of the first significant co-operation between western organisations and a utility of Central Europe for the completion of an NPP.

The Cernavoda plant is based on the CANDU technology, and this makes it different from any other nuclear installation in Central and Eastern Europe. However, from many other viewpoints the conditions of the site when it was taken over by RENEL and AAC in 1989 were similar to those of other nuclear sites in the former Communist world where NPP construction was interrupted. Most of the mechanical and electrical equipment already existed for the first two units; this equipment had been supplied under previous contracts by Canadian, Italians, United States (GE) and Romanian suppliers, but its condition after a long period of storage had to be checked; civil works were in different phases of progress for the five units; construction work had started on the first unit, but its quality did not appeared to be satisfactory; documentation had to be checked. Major changes were needed in the site organization and in particular in the overall project management.

The AAC PMT has been given full authority to manage the project on behalf of and in the interest of RENEL. Nevertheless, every effort was made to utilize to the maximum extent possible the personnel of RENEL and the Romanian contractors who already worked on site. As station owner, RENEL has the ultimate authority with regard to all aspects of construction, commissioning and operation. AAC has the responsibility and authority to determine expenditure allocations within the established annual budget. The total funds required for project completion have been estimated jointly, and the annual budget expenditures have to be kept within this limit.

After the signature of the PMC and the establishment of the integrated AAC - RENEL PMT on the site, several actions were started in parallel:

- establishment of a new QA and quality control (QC) organization;
- review of the overall plant design;
- assessment of all construction work already performed and definition of the necessary corrective actions;
- verification of the status of all existing material and equipment;
- establishment of an integrated procurement unit;
- negotiation of new contracts with the Romanian contractors based on western practice;
- procurement of adequate tools and erection equipment;
- on-the-job training of Romanian manpower;

- start of a social programme aimed at improving the living conditions for Romanian personnel working at Cernavoda;
- provision of a town site for foreign staff.

Today all the working practices and systems of AAC are fully operational, including computerized systems for material and document control and computerized scanning of the drawings. In addition the site work-force has been reduced to a more manageable 5.000 persons.

The Cernavoda unit 1 reactor has been loaded with nuclear fuel in June 1995 while the reactor criticality and first synchronisation to grid are planned for the end of 1995.

The completion of Cernavoda unit 2, being a replica of unit 1, will benefit from the overall progress of the work so far done (approx. 25%) and presently kept in preservation, of a qualified organized and trained personnel, domestic and expatriate, and of existing infrastructures and technical facilities.

4.0 KEY LESSONS

When initiating a nuclear power programme, a realistic assessment of the skills and capabilities available in the country must be made in order to define the optimum role and degree of localization. The extended schedule of the Cernavoda project was to a large part due to the strong emphasis on localization for a first nuclear unit and the assignment of the management of this complex programme to organizations that did not have sufficient experience. A contract for a first nuclear unit should essentially be of turnkey type, with subcontracts and training provided to develop the basis for increased future localization.

In the area of manufacturing, technology transfer agreements were made with experienced foreign vendors. However, while the production of local components was successful (40% local participation for unit 1), delays in manufacturing and the time taken to develop an effective QA programme led to major delays. Localizations of manufacturing and decisions on the sources and scope of supply should start well in advance of construction.

Regarding plant construction, the suppliers should organize a familiarization programme for the local contractors to inform them of the requirements and quality needs of a nuclear construction programme before a project contract is concluded. The necessity of rework, low productivity and inability of people to adapt to new work practices and requirements have delayed construction of Cernavoda. A large amount of civil and mechanical rework was successfully carried out, but at the expense of major delays.

Project management proved to be the weakest area in the initial phase of Cernavoda unit 1. Unfamiliarity with project management systems (document control, material control, critical path scheduling), and lack of detailed planning for the execution of the work led to significant delays. Excess focus on "hard' areas, such as materials, concrete, equipment and welds, without the necessary attention to "soft" areas, as QA, procedures, project planning and project systems, resulted in much rework and delays.

Training is an essential part of a first nuclear project. Because the political circumstances in Romania, a large part of the training programme provided for in the original contracts was not carried on. This resulted in both lack of understanding by Romanian organizations of the work and the programme and a lack of appreciation by the foreign suppliers of local customs and work practices. This contribute to misunderstandings and delays.

After the PMC signing the training of Romanian operating staff at a sister CANDU 6 power station, Point Lepreau in Canada was a success. Also, the co-operation of the foreign specialists with the Romanian contractors led to increases in productivity, ranging from factors of to 10, due to better tools, better planning and better understanding of the detailed work requirements.

An additional contribution to delays of the work were delays in local currency financing. The cost impact of interest during construction was often neglected.

A key factor in project construction and quality of the work was the difficulty of introducing and implementing a QA programme according to international standards. When Canadian and Romanian Authorities agreed on the Cernavoda project in 1979, the western concept of the owner did not fit into the old Romanian system. The contractors at the site reported to various ministries and also to a group that was designated as the owner, Intreprinderea Nucleara Cernavoda (INC). These groups did not worked together to build the station in a logical and controlled fashion. Instead of integrating QA into the quality system existing within the companies, it was imposed on them and therefore perceived by them as another bureaucratic measure without much meaning. Also the importance and priority of the role of QA was not adequately recognised by the management of the plant.

It was intended that INC should develop and implement a QA programme to cover all phases of the project activities, but this implementation was not successful. INC was not conceived or sized as a project manager in the Western sense but rather was an owner's representative. The level of effort and skills needed to manage the completion of a complex large project was not fully appreciated or understood. What emerged as a QA site organization generally met western standards but it did not have the expertise and authority to enforce application of the QA programme to the project. This was largely due to interference by outside organizations.

The following points illustrate some of the difficulties experienced during the old system:

- the QA programme of INC focused on construction/installation activities and did not address other project activities;
- the QA programme of INC and of site contractors were difficult to understand and use by the Romanian staff;
- audits INC and site contractors were ineffective because lack of experience and training and also lack management support. The observations were vague and it was only tried to resolve specific events or symptoms, not to review the system to establish the root cause of problems.

With the installation of the new Romanian government and the consolidation of the AAC, well defined lines of responsibility and functions were developed for all phases of the project. AAC and RENEL have developed a QA programme to cover all project activities. It complies with the applicable norms, codes and standards as formulated in the design definition.

AAC has implemented a QA organization that is capable of reviewing and reporting on all QA activities for the project. The organization is led by a QA manager and is comprised of quality specialists, seconded by quality auditors from Canada, Italy and Romania. Its mission is to perform QA engineering and project audit activities.

The QA manager of AAC reports directly to the project director, but he has also to report to the home offices (AECL for NSP related activities, Ansaldo for BOP activities and the Romanian design organization (CITON) for support systems engineered in Romania).

AAC has also executed a verification programme to review and assess all work performed to date at Cemavoda unit 1. The programme determines conformance with the design requirement and ensures that all deviations and changes are reviewed for acceptance.

Related to QA but different in focus and scope is the issue of the safety culture which needs to be addressed early in a nuclear programme. The IAEA Pre-OSART report refers to the need for a cultural changes to facilitate the safe operation of the Cemavoda station. These cultural changes must be introduced at least in the station working environment, since broader political changes may take time.

The successful performance of Cemavoda during the last few years makes us confident that unit 1 of Romania's first NPP will be commissioned and operated with observance of the strongest safety requirements and will provide a reliable source of electricity for the Romanian economy.

4.0 UNIT 2 THE CHALLENGE FOR AN ADVANCED FINANCING SCHEME

4.1 Unit 2 completion part of RENEL'S power system development program

At the and of 1994, the nameplate power installed in the Romanian national power system was 21.808 MWe, with an overall electricity production of 53.507 GWh, of which 39,5% generated in lignite and coal fired plants, 28,6 % generated in hydrocarbon fired plants and 26,5 % in hydropower plants. The rest of 5,4 % is generated by independent producers. Considering the age, the nameplate power installed may be ranked as follow: 36 % up to 15 years, 23 % among 15 and 20 years, and the rest (41%) over 20 years.

The actual forecast, based on a minimum economic growth scenario, shows an increase of the electricity demand, which in the year 2000 will be about 62.500 GWh.
The available maximum power output is evaluated to be, at present, 9.000 MWe, less than 50 % of the nameplate power. Maximum power demand, in 1994, was about 8.500 MWe, evidencing the low margin of the system.
The national grid least cost development studies performed by Ewbank Preece Ltd. and Romanian Institute of Power Studies and Design (ISPE) demonstrate the opportunity to complete and put in commercial operation, up to year 2000, units 1 and 2 of Cemavoda NPP.

After the commissioning, Cemavoda unit 1 will provide, to the grid, about 4.200 GWh, about 8 % of the electricity generated in Romania, concurrent with a significant reduction of the hydrocarbon imports (about 1,4 millions' tones) and of the pollutant emissions from fossil fuel firing.

The electricity generated in Cemavoda unit 2 will be purchased integrally by RENEL. It is not excluded the possibility of some exports.

4.2 The actual status of the unit 2

4.2.1 Activities already performed

The completion of unit 2, being a replica of unit 1, will benefit from the overall progress of the work so far done (about 662 millions USD) and presently kept in preservation. Based on a RENEL analysis, the unit 2 status at the end of 1994 was:

- equipment/material procurement: about 68 % of the required quantities are already purchased (about 545 millions USD, of which 290 millions USD local supplies);
- construction works on site: the overall progress is about 24 % (67 % civil construction, 5 % mechanical construction and 1 % electrical and I&C construction).

4.2.1 Activities to be performed

Continuing the partnership with his traditional partners, AECL and ANSALDO, a joint RENEL-AECL-ANSALDO performed, during the last half of 1994 and first half of 1995, a detailed evaluation of the activities to be performed to complete unit 2, including the cost associated to. These are summerized in Table 1.

The time schedule, from the starting of the Project up to the first grid synchronisation of unit 2, foresees 40 months for construction and 16 months for commissioning. However, optimisation studies in progress might allow to reach the target of 34 months for the construction phase and 14 months for the commissioning phase.

Table 1 Activities required to complete unit 2 and their costs

	Total (mill. US doll)	Of which import (millions US. doll)
Equipment/ materials procurement	256,6	92,7
Constructions-erection works	67,4	-
Engineering, technical assistance, commissioning, staff training, social costs	221,7	170,6
Insurances	12	12
Subtotal 1	557,7	275,3
Contingencies (5%)	27,9	13,8
Subtotal 2 (fuel and heavy water exclusive)	585,6	289,1
Fuel (initial loading)	6,1	
Heavy water	116	
TOTAL	**708**	

4.3 Main features of the project

4.3.1 Nuclear fuel

Romania devoted very large capital resources to develop a nuclear fuel bundle factory at Pitesti, with a capacity of approx. 10.000 CANDU type fuel bundles per year (sufficient for 2 CANDU 6 units), qualified for use in CANDU 6 reactors. Recently the equipment and the manufacturing technology have been up-graded in collaboration with the Canadian supplier ZIRCATEC PRECISION INDUSTRY, and the system and procedures have been qualified by AECL and ZIRCATEC.

4.3.2 Heavy water

The heavy water requirement for each 700 MWe Cernavoda NPP unit is 500 t for the initial inventory and approx. 7 t/y for yearly losses. Romania has already built a heavy water production plant (ROMAG), at Turnu Severin, having an installed capacity of 360 t/y over four modules of 90 t/y capacity each. For the next two years only two, and after three of the four modules will be in operation. ROMAG has already delivered 150 t of D2O for the initial inventory of Cernavoda unit 1.

4.3.6 Safety issues

The safety standards applied for the construction and operation of Cernavoda Unit 1 and 2 are in line with all principles set out by IAEA regulations and guides.

CNCAN has already granted the Site Permit for Unit 2, based on the submission of the Initial Safety Analysis. Partial construction permits for civil works and some process systems were granted on the basis of the submission of a Preliminary Safety Analysis Report and other specific documents. The work so far executed at site, for Unit 2, has been based on the above partial construction permits.

The licensing process of Unit 1 is in a very advanced stage focusing on the evaluation of the Criticality License Application. The licensing process for Unit 2 may fully benefit of the experience already acquired on Unit 1. CNCAN position is that the licensing process and requirements for Unit 2 will be similar to the Unit 1.

4.3.7 National participation

The completion of unit 2, being a replica of unit 1, will benefit of qualified organised and trained personnel, and of existing infrastructures and technical facilities, set up during unit 1 completion.

There are:

- local site contractors, qualified to work in nuclear/conventional islands, state owned or privatised, existing the pre-requisites of a competition framework;
- domestic manufacturers, part of them having manufacturing licenses. It is to be noted that Romanian suppliers for the unit 2 were qualified, with respect to the technical capabilities, quality assurance program, and manufacturing procedures, similar as AECL traditional suppliers;

- design and research organisations with personnel trained in developed countries and within unit 1 Project management Team, and with detail knowledge of the CANDU 6 design.

4.3.7 Construction and operation personnel

To day unit 1 is 98 % completed, so there is an experience to overcome program difficulties inherited from the past, and is presumed to be easily and profitably transferred to the unit 2, avoiding major organisational and technical problems.

The project maagement tam, on the basis of its extensive experience in the construction, commissioning and operating of CANDU nuclear power plants, has considerably improved working methodologies and made operative the concepts of quality, efficiency and team approach, clearly defining responsibilities and objectives of all Cernavoda plant departments and section involved in the various activities. unit 2 completion will fully benefit of o local personnel (owner, site contractors, suppliers, designers) qualified and organised.

NPP Cernavoda has a training centre with a full scope simulator, which together with unit 1 in operation provide the conditions to select and train the unit 2 operation personnel.

4.3.8 The challange for an advanced financing scheme

The opportunity of works completion and commissioning of NPP Cernavoda Unit 2 was demonstrated within the study "Least Cost Capacity Development between 1994-2010" performed by EWBANK PREECE in collaboration with ISPE, in 1994. This unit completion was also included in RENEL development Strategy, providing the year 2000 as commissioning term.

The value of the investment carried out, up to now, was estimated at 662 mill. USD and the difference remained for the works completion, at about 708 mill. USD. The execution period up to Unit 2 commercial putting into operation, was appraised at 58 months, starting from the actual stage of the already performed works and assuming the experience accumulated for Unit 1 achievement.

The electricity generated in Cernavoda Unit 2 will be purchased integrally by RENEL at a price that will allow to reimburse the investors and the loans, and from the potential participants to the physical completion of the project will be preferred those participating to its financing.

The economic analysis performed by RENEL has emphasised the economic efficiency of the Cernavoda NPP unit 2 completion, under a joint venture financing scenario which should provide the financial resources to cover the import part as well as a portion of the local part of the project.

PART III

SMALL AND MEDIUM REACTORS POTENTIAL MARKET AND APPLICATIONS

XA9846716

ASSESSMENT OF THE WORLD MARKET FOR SMALL AND MEDIUM REACTORS

B.J. CSIK
International Atomic Energy Agency,
Vienna

Abstract

The market for SMRs until 2015 was assessed by individual countries, taking into account energy demand and supply patterns, growth rates, energy resources, economic and financial resources, electric grids, industrial and technical development, infrastructure availability, environmental and nuclear safety concerns and other policy issues. The market assessment includes all applications of these reactors, that is electricity generation as well as the supply of process head and district heating.

It is expected that SMRs will be deployed primarily in countries which have already started nuclear projects, in particular in countries which have developed SMR designs themselves. Thus, projects would be supplied predominantly by domestic sources in the years ahead; later, the export market is expected to attain more importance. It is further expected that over two thirds of the SMR units would be in the medium size range, i.e. from 300 to 700 MW(e), the rest would be smaller. About one third of the SMRs to be implemented are expected to supply heat and/or electricity to integrated seawater desalination plants. More than half of these reactors would be below 300 MW(e) or 1000 MW(th).

The overall market is estimated at about 60 to 100 SMR units to be implemented up to the year 2015. It is recognized that forecasts, just like national development plans, tend to err on the optimistic side. Therefore, an overall market estimate of 70 to 80 units seems reasonable.

1. Introduction

Nuclear power has been used over the last four decades and has been one of the fastest growing energy options. By the end of 1995, there were 437 power reactors in operation worldwide, with a total installed capacity of 344 GW(e). There were also 34 reactors under construction, with a total capacity of 33 GW(e). At present, about 17% of electricity is generated by nuclear power. Though the rate at which nuclear power has penetrated the world energy market has declined, it has retained a subtantial share, and is expected to continue as a viable option well into the future.

The present generation of nuclear power plants has been developed to satisfy primarily the need of the largest market for these plants, which corresponds to the industrialized countries with electric grids that admit the introduction of large units. Currently, the largest power reactors are rated at about 1400-1500 MW(e). There seem to be no incentives for achieving any further increase in size, and in fact, there are no efforts directed to this end by designers. The worldwide market for nuclear power is, however, by no means limited to large reactors; SMRs always had a share of this market and this situation is expected to prevail in the foreseeable future.

A substantial number of nuclear reactor designs have been developed worldwide within the small and medium power range. Some of these have been built or are under construction, others are still in the conceptual, basic or detailed design stage. Few of the design stage reactors are under active development, most are "on hold", waiting for potential customers to express their interest All of the currently pursued advanced designs share the common goals of achieving improvements in safety, reliability and economics, with different levels of emphasis placed on these aspects. It is also recognized that there are technology thresholds which allow some technical solutions within limits of size and which contribute to the above goals, but which cannot be used in larger size reactors.

What is being offered on the market is relatively easy to assess. Designers and vendors are willing and interested in providing information on their concepts, especially regarding technical aspects. They are, however, somewhat reluctant to provide cost estimates. The recently published TECDOC-881 of the Agency (Design and development status of small and medium reactor systems 1995) contains a review of most of the designs currently pursued. Complementing the information contained in the TECDOC-881, some additional designs are described in the present document It can be concluded that there is certainly no lack of potential vendors, who offer a large variety of reactors from which potential buyers may choose.

While the assessment of what is being offered on the market is basically a "status review", the assessment of the demand is directed to the future, that is, it has the character of a "forecast" This is a more difficult task, and the results obtained will strongly depend on the assumptions, criteria and methodology adopted and applied.

Forecasts are necessarily based on past experience and current knowledge, but they are directed to predicting the future. To achieve reasonably reliable results, it is essential to base the predictions on what is realistically and objectively expected to happen, and not on what one would wish or would like to happen. Even if reality consistently refuses to follow predictions, as shown by experience, forecasts are needed because they form the basis of planning and decision making.

Regarding the overall nuclear power market, the IAEA performs annually a "forecast". The latest version (July 1996 edition), was published under the title *Energy, Electricity and Nuclear Power Estimates for the Period up to 2015* (Reference Data Series No. 1). The nuclear generating capacity estimates were derived from a country by country bottom-up approach, and include reactors pertaining to all size ranges. The low and high estimates reflect contrasting but not extreme underlying assumptions on the different driving factors that have an impact on nuclear power development. These factors, and the ways they might evolve, vary from country to country.

The present market assessment is intended to cover only SMRs, without including large reactors It is recognized that SMRs as well as large reactors constitute an integral part of the overall nuclear power market, however, they may address different specific needs. Most of the factors which affect the evolution of the overall nuclear market are equally relevant to any nuclear reactor whatever size range it belongs to. There are, however, differences too, and these have to be taken into account.

2. Reactor size ranges

The choice of ranges is somewhat arbitrary but there has been the usual practice to take the upper limit of the SMR range as approximately half of the power of the largest reactors in operation. Accordingly, reactors up to 700 MW(e) are currently considered as SMRs Other limits are defined by continuing to take reduction by a factor of two. The ranges adopted therefore are:

Very small reactors	<150 MW(e)
Small reactors	150-300 MW(e)
Medium reactors	300-700 MW(e)
Large reactors	>700 MW(e)

For heat-only or co-generation reactors, the range limits are applied to the electrical equivalencies of the thermal power. For very small heat-only reactors, for example, the upper limit adopted is 500 MW(th).

It is understood that very small, small, medium or large are relative concepts, related to the power level of the largest reactors in operation. That is, at the time when the largest reactors in operation were of the order of 200 MW(e), the corresponding upper limit of the SMR range was 100 MW(e), when 600 MW(e) units came into operation, the SMR range increased to 300 MW(e), and so on. As there are no ongoing efforts to further increase the power level of the largest units, the currently accepted SMR range is assumed to prevail for a considerable period.

Applying the current definition of the SMR range, more than a third of the operating nuclear power reactors would qualify as SMRs However, it should be noted that at the time when most of these plants were designed and built, they were considered large reactors according to the then-prevailing definition of the term.

The above defined ranges for medium, small and very small reactors expressed in power levels (MW(e)), are to be interpreted more as orders of magnitude and less as precise numbers. The large variety of reactors with different characteristics which are included in each of these ranges, are intended to respond to different requirements and uses, which need to be taken into account in order to facilitate the assessment of the potential market.

Medium size reactors are eminently power reactors whose objective is electricity generation. They can also be applied as cogeneration plants supplying both electricity and heat, but the main product remains electricity. As such, they are intended for introduction into interconnected electric grid systems of suitable size (at least 6 to 10 times the unit power) and operated as base load plants. If operated in the cogeneration mode, the heat supply would be up to about 20% of the energy produced. Economic competitiveness with equivalent alternative fossil-fueled plants is expected to be achievable under most conditions.

Small reactors are either power or cogeneration reactors which may have a substantial share of heat supply. Due to the size effect, small reactors for electricity generation only, or operated in the cogeneration mode, are not expected to be economically competitive with medium or large size nuclear power plants. They are therefore intended for special situations where the interconnected grid size does not admit larger (medium or large size) units and where alternative energy options are relatively expensive.

Very small reactors are not intended for electricity production under commercially competitive conditions as base load units integrated into interconnected electrical systems. Clearly, very small reactors of current designs are not to be regarded as competitors of large, medium or even small power reactors, of which they are not scaled-down versions. Very small reactors address specific objectives such as the supply of heat and electricity or heat only (at either high or low temperature) for industrial processes, oil extraction, desalination, district heating, etc., propulsion of vessels or energy supply of concentrated loads in remote locations They could also serve as focal projects and very effective stimulus for the development of nuclear infrastructures in countries starting a nuclear power programme.

The consideration of the specific objectives of the reactors included in each power range has major relevance for the assessment of the respective markets.

3. Basic assumptions, criteria, scope and methodology

Availability of SMRs. The market assessment is based on the assumption that suitable nuclear reactors will be available both for domestic implementation and for export, when required by interested buyers. Suitability is interpreted by meeting the technical and economic conditions as defined by potential buyers, which are often called user requirements. The user requirements must be reasonable and not be a wish-list containing a collection of desirable goals impossible to achieve simultaneously. The nuclear reactors must be licensable; the technical features must not require further research to demonstrate their viability and reliability; the costs must be within an acceptable range. Understanding the costs and benefits in the wider sense instead of only in monetary terms, the buyers must find a favorable cost/benefit ratio.

Currently, six countries have 14 SMRs under construction, 5 units in the small and 9 in the medium size ranges. These reactors should start operation before the year 2000 or soon thereafter (Table 1). The IAEA-TECDOC-881 contains descriptions of 29 SMRs designs, of which 10 are classified as being in the detailed design stage. Including additional concepts on which information is available, the overall number of SMRs in different design stages is of the order of 50 reactors. The assessment of the current situation shows that there is considerable activity in the field of SMRs, which can be interpreted as a positive sign for further development.

TABLE 1

SMRs CURRENTLY UNDER CONSTRUCTIONB

COUNTY	REACTOR	NET CAPACITY MW(E)	CONSTRUCTION START
ARGENTINA	ATUCHA-2	692	1981-6
INDIA	KAIGA-1	202	1989-9
INDIA	KAIGA-2	202	1989-12
INDIA	RAJASTHAN-3	202	1990-2
INDIA	RAJASTHAN-4	202	1990-10
KOREA, REP. OF	WOLSONG-2	650	1992-9
KOREA, REP. OF	WOLSONG-3	650	1994-3
KOREA, REP. OF	WOLSONG-4	650	1994-7
PAKISTAN	CHASNUPP-1	300	1993-8
ROMANIA	CERNAVODA-2	650	1983-7
SLOVAKIA	MOCHOVCE-1	388	1983-10
SLOVAKIA	MOCHOVCE-2	388	1983-10
SLOVAKIA	MOCHOVCE-3	388	1985-1
SLOVAKIA	MOCHOVCE-4	388	1985-1

Source: IAEA - PRIS

Note In addition, construction has been suspended but is expected to proceed on the following reactors:

Cuba -	Juragua-1 - 408 MW(e)	1983-10
Cuba -	Juragua-2 - 408 MW(e)	1985-02
Romania -	Cernavoda-3 - 625 MW(e)	1984
Romania -	Cernavoda-4 - 625 MW(e)	1985
Romania -	Cernavoda-5 - 625 MW(e)	1986

The information and data provided by designers and vendors, as well as studies and plans of various countries regarding the launching of power reactor projects in the SMR range, support the above-mentioned assumption.

Governmental role and national policy. Possibly the most decisive factor which promotes nuclear development is Governmental commitment and active support of nuclear power as part of medium to long-term national energy development and supply policy. In fact, this is considered a "necessary" condition to be fulfilled for any country expecting to proceed with a nuclear programme. In the absence of active Governmental support, neither publicly owned utilities which directly respond to national policies, nor privately owned utilities which function in a highly regulated environment, can be expected to initiate new nuclear projects. Public acceptance is understood to be a factor which affects Governmental policies and actions. Its significance depends on the influence of public opinion on political power.

Several countries have adopted and have in force medium to long-term energy development and supply policies which exclude consideration of the nuclear option, or which only provide passive support or a somewhat reluctant acceptance of nuclear power as a last resort. Governmental policies, however, do not necessarily last forever. As shown by experience, Governments as well as policies may change in time.

Infrastructure availability. For any country, the viability of starting a nuclear power programme will depend on the availability of adequate infrastructures. These infrastructures are technological, manpower, industrial, economic, financial and institutional. In principle, any country can develop its infrastructures to an adequate level, but this requires time and effort, both of which may be substantial. Countries which are not in a position to proceed with investing the effort required for developing their infrastructures on an appropriate timescale, or where such efforts are not reasonably justified by medium to long-term prospects of using nuclear power for their energy supply, are not likely to start a nuclear power programme. They are even less likely to start with the acquisition of a large size nuclear reactor, even if their interconnected electrical systems could admit such units, which usually is not the case.

Very small or small reactors present a more attractive option to start a nuclear power programme. Though these also need the availability of adequate infrastructures, relatively less development effort is required, which is therefore more easily achievable.

Programmes with large units. Several countries with operating nuclear power plants or with ongoing nuclear power programmes have interconnected electrical grid systems which readily accept large size units. Unless there are special situations which might require or which would promote the use of SMRs, these countries are expected to add further similar large size units when required to satisfy the growing electrical demand.

Economic and financial constraints. Countries with chronic economic problems, high indebtedness and scarce financial resources are not expected to invest in capital intensive projects, such as nuclear reactors. In some countries which have effectively initiated nuclear power projects, persistent economic and financial constraints have led to long and costly delays or even to interruption of construction. Countries which are expected to remain in such a situation at least in the near future, are not considered as potential market for new nuclear projects.

Scope. The assessment includes all countries, and is not limited to those which already have ongoing nuclear power programmes, or which expressed intentions of launching SMR projects. All countries are considered on an individual basis. All uses are included, electricity generation, heat only, and cogeneration. Military applications are excluded from the scope of the assessment. Reactors on which construction has been started are not considered in the market assessment, which is for new projects only.

It is considered that membership in the IAEA shows at least a certain interest in the use of nuclear energy, even though some Member States do have anti-nuclear policies in force, at least for the present. The reverse, i.e., no current interest at all in nuclear energy is assumed to be in general applicable to all non-Members, with a few exceptions such as Taiwan (China) or the Democratic People's Republic of Korea. It is noted that according to general practice within the IAEA, the use of particular designations of countries or territories does not imply any judgement by the IAEA as to the legal status of such countries and territories, or their authorities and institutions, or of the delimitation of their boundaries.

Time frames. The market assessment is for reactors coming on line up to the year 2015. This is also the period covered by current IAEA forecasts of overall nuclear power development. Beyond this date, that is beyond a period of about 20 years, forecasts become very speculative. They would be based more on postulated scenarios and general statistical analysis, than on country by country and project by project considerations, which is be basic approach adopted for the present assessment.

Construction time (measured form the first pouring of concrete to grid connection) has been on average about 8 years for reactors coming on line during the last decade. Some reactors have been built in half this time but others have taken 10 years or more. While construction schedules usually quoted by designers-vendors are much shorter, experience shows that delays do occur. For the purpose of the market assessment, the construction times assumed for medium, small and very small reactors are 6, 5 and 4 years respectively, as average values.

Up to the year 2000, only reactors currently under construction can be expected to start operation New projects could only come on-line after this date Lead-times for preparatory activities (planning, site qualification, feasibility studies, acquisition, infrastructure development or upgrading, institutional arrangements, etc) will vary according to specific projects and the particular conditions and situation prevailing in each individual country They are therefore assessed on a case by case basis

Methodology. A two-phase procedure is applied In a first phase, individual countries are assessed applying the above-mentioned assumptions and criteria in a general and global manner As a result, a short list of countries is obtained (Table 2), which contains those countries that are assessed as having a potential demand for SMRs within the period considered, and which therefore deserve a more thorough consideration

TABLE 2

LIST OF COUNTRIES FOR FURTHER CONSIDERATION

FIRST PROJECTS HAVE BEEN STARTED	NO NUCLEAR POWER PROJECT STARTED
Argentina	Algeria
Canada	Belarus
China	Chile
Hungary	Croatia
India	Egypt
Iran, Islamic Republic of	Indonesia
Italy	Israel
Korea, Republic of	Libyan Arab Jamahiriya
Mexico	Malaysia
Pakistan	Morocco
Poland	Portugal
Russian Federation	Saudi Arabia
South Africa	Syria
United States of America	Thailand
	Tunisia
	Turkey

It is noted that some countries not included in the short list might initiate and implement SMRs within the time frame adopted Conversely, not all countries selected might fulfill the expectations regarding the implementation of new projects This, however, should not alter substantially the overall results of the market assessment Neither exclusion from nor inclusion in the short list, are to be interpreted as recommendations regarding individual countries or projects

The countries selected for further consideration have been grouped according to their nuclear power development status, i e , those which have already started implementation of their first nuclear power project, and those which have not yet done so The countries cover a very wide range of different characteristics and conditions The common feature of those included in the first group is that in each of these countries, at least construction of a nuclear power plant has been started, while those included in the second group are still at the planning and study stage, some of them since several decades Having started construction of a first reactor implies basic infrastructure availability as well as experience in launching nuclear projects

In a second phase, the market for SMRs is assessed for each selected country, taking into account in addition to the plans or intentions of the individual countries, the previously mentioned criteria and a series of factors affecting the market, which are identified and discussed in the following section

The market for medium, small and very small reactors is assessed separately, and also the market for reactors expected to be imported or of domestic supply.

It has been attempted to be objective, practical and realistic. High and low estimates are obtained, which do not correspond to a too optimistic theoretical maximum, nor to an overly pessimistic absolute minimum. The estimates are rather the result of expectations under more or under less favourable conditions and scenarios, as applicable to both overall nuclear power development and to the role and market share of SMRs. In general, the high and low estimates reflect contrasting but not extreme underlying assumptions

4. Factors affecting the market

Among the many relevant factors with affect the market either promoting or opposing the implementation of nuclear power programmes and of SMRs in particular, the most important ones are identified and briefly discussed, complementing the criteria previously established.

Energy resources and supply diversification. High dependence on imported fossil energy sources (oil, gas, coal) and little or no diversification in the pattern of energy supply, are promotional conditions for nuclear power development. Countries which present these characteristics are more likely to proceed with nuclear power, then those which have abundant and cheap conventional (fossil-fuel or hydraulic) energy resources. The availability of uranium resources can be considered as promotional for nuclear power

Economic and financial resources. Nuclear power is capital-intensive and requires substantial investments. Countries with strong economies and good access to financial resources are in a better position to launch nuclear projects than those with struggling economies, high indebtedness, and a general lack of capital Privatization and deregulation of the electricity market are characterized by short-term objectives Privatization and deregulation often discourage the implementation of nuclear projects because of the high capital requirements and long-term return of investment. Within this negative context for nuclear implementation, SMRs are favored versus large plants due to less capital requirements, easier financing and shorter construction times

Interconnected electrical systems. Due to the relatively large share or the fixed cost component in the energy production costs of nuclear power, base load operation is required to achieve favorable economic conditions. Therefore, nuclear power plants intended for electricity generation as the only or the main product, are required to be integrated into the interconnected electrical grid systems. The total interconnected generating capacity limits the maximum unit size that can be added. The optimal unit size, which is determined through generation system expansion planning, however, is often smaller than the acceptable maximum unit size.

Growth rates. Countries characterized with sustained high GDP, large industrial production, high energy and electricity consumption growth rates are more likely to proceed with nuclear power programs than those with stagnant economies or in recession. High population growth rates not accompanied by economic and industrial development are not favorable to nuclear power.

Energy demand pattern. Increasing share of electricity in overall energy consumption, high share of industrial demand in overall energy demand, and high base load to peak load ratio are favorable characteristics promoting nuclear power. Concentrated large demand for energy in the form of heat favors cogeneration or heat-only reactors. Remote and isolated areas with relatively large energy (electricity and heat) demand and with lack of local energy resources present favorable conditions for small and very small reactors

Electricity supply structure. Large utilities with solid economic and financial structures supported by a rentable tariff system are more likely to possess the investment capability and credit rating required for nuclear projects, than small and weak utilities and subsidized tariff systems (governmental monetary support). Experiences of power plant operation with high availability factors, reliable transmission and distribution systems with low break down rate and low supply interruption rate and in particular experience in nuclear power, provide a favorable background for new nuclear projects.

Industrial and technical development. If adopted as priority national development goals, they act in favour of nuclear programmes. As shown by experience, nuclear projects promote quality improvements, transfer of technology, and in general the development of domestic capabilities. The side effects of nuclear power programmes are recognized as important contributions, even though they are difficult to identify and measure.

Environmental and nuclear safety concerns. Worldwide concerns about environmental pollution from conventional energy sources show an increasing trend and should in principle promote nuclear power development. On the other hand, concerns regarding nuclear safety and radioactive waste disposal tend to discourage decision makers from turning to this option. How these concerns balance and which will have a more dominant role has a substantial influence on national policies and therefore affects the market. Concerns about nuclear safety do have a positive influence on the development and market potential of SMRs, which are perceived as offering improved safety features and safety levels.

5. Market estimates

According to the methodology adopted, the market for SMRs is assessed for each country selected for further consideration, as listed in Table 2. The overall results obtained (high and low estimates) are presented in Table 3, discriminated by geographical regions and by reactor size range (medium, small and very small). A summary is presented in Table 4, which also contains an estimate of the market shares corresponding to domestic and foreign supply sources. The following are brief comments which refer to the various countries with a potential market for SMRs.

Among the countries which have ongoing nuclear power programmes, China, India and the Russian Federation represent a substantial market for SMRs. In China, there is an ambitious nuclear power programme firmly supported by the Government. In addition to large reactors and some imported medium size units, a series of domestic design medium size, some small and also several very small units (including heat-only reactors) are expected to be implemented. There is continuing firm Governmental support to the nuclear power programme in India, and a large demand for new generation capacity. It is expected that the country will proceed with its programme based on domestic design small size reactors followed by a series of medium size units. In the Russian Federation there is an ongoing nuclear power programme based mainly on large size units, but limited by economic and financial constraints. Several designs are under development, in particular in the medium and the very small reactor ranges. It is expected that a series of units in these ranges will be implemented in addition to larger plants. It is estimated that the market for SMRs in the above-mentioned three countries is of the order of 30 to 40 units (low and high estimates), of which more than half correspond to medium size reactors.

Argentina, the Islamic Republic of Iran, the Republic of Korea and Pakistan have ongoing nuclear power programmes which include reactors under construction. In Argentina, follow-up nuclear power plants are expected to be in the medium size range. The development of a very small domestic design reactor has been pursued, and it is intended to build a first unit. In the Islamic Republic of Iran, the construction of two large power reactors has been recently reinitiated, and it is planned to acquire two small units. In Pakistan, a further small reactor is expected to be followed by a series of medium size units. Though large power reactors constitute the basis of the ongoing substantial nuclear power programme of the Republic of Korea, more units in the medium range are expected to be added. Also, implementation of a domestic-design very small reactor is expected. The overall estimate for the four countries is 10 to 15 units within the period considered.

TABLE 3

SMR MARKET ASSESSMENT BY GEOGRAPHIC AREAS

		HIGH ESTIMATE				LOW ESTIMATE			
Region	**Size**	**2001-2005**	**2006-2010**	**2011-2015**	**Total (2001-15)**	**2001-2005**	**2006-2010**	**2011-2015**	**Total (2001-15)**
North America	M S VS	0 0 0	2 0 0	1 0 0	3 0 0	0 0 0	0 0 0	0 0 0	0 0 0
South and Central America	M S VS	0 0 1	0 0 0	4 1 0	4 1 1	0 0 0	0 0 0	2 1 0	2 1 0
European Union	M S VS	0 0 0	2 0 0	7 0 0	9 0 0	0 0 0	1 0 0	3 0 0	4 0 0
Eastern Europe	M S VS	1 0 1	4 0 2	7 0 2	12 0 5	0 0 0	2 0 2	6 0 1	8 0 3
Africa	M S VS	0 0 1	2 0 2	3 3 1	5 3 4	0 0 0	0 0 0	5 0 1	5 0 1
Middle East and South & Middle Asia	M S VS	1 2 0	5 5 0	10 1 1	16 8 1	1 1 0	3 4 0	5 0 1	9 5 1
Southeast Asia and the Pacific	M S VS	0 0 0	2 0 0	4 0 1	6 0 1	0 0 0	1 0 0	2 0 0	3 0 0
Far East	M S VS	5 0 2	4 2 2	2 1 3	11 3 7	3 0 1	4 1 2	4 1 2	11 2 5
World Total	M S VS	7 2 5	21 7 6	38 6 8	66 15 19	4 1 1	11 5 4	27 2 5	42 8 10

Size definition M: 300-700 MW(e), S: 150-300 MW(e), VS: < 150 Mwe or equivalent

Canada might install some additional units to replace older plants when taken out of service; Hungary would require follow-up nuclear reactors to satisfy the growing electricity demand without unduely increasing its import dependence; Italy has shut down all its nuclear power reactors following a political decision, a change in attitude and in policy could lead to the reinitiation of a nuclear programme with advanced reactors; Mexico could follow-up its operating reactors with new projects; Poland canceled its nuclear reactors which were under construction, but it could very well reconsider its attitude toward nuclear power and implement new projects; South Africa is interested in implementing some very small reactor projects for remote locations; and finally, the USA which has about 100,000 MW(e) operating nuclear capacity, could terminate its de-facto nuclear moratorium and implement some advanced reactor projects it has been developing with considerable effort.

TABLE 4

SMR MARKET ASSESSMENT SUMMARY

HIGH ESTIMATE	2001-2005	2006-2010	2011-2015	TOTAL
MEDIUM	7 (5+2)	21 (11+10)	38 (8+30)	66 (24+42)
SMALL	2 (0+2)	7 (6+1)	6 (1+5)	15 (7+8)
VERY SMALL	5 (4+1))	6 (4+2)	8 (6+2)	19 (14+5)
TOTAL	14 (9+5)	34 (21+13)	52 (15+37)	100 (45+55)

LOW ESTIMATE	2001-2005	2006-2010	2011-2015	TOTAL
MEDIUM	4 (2+2)	11 (8+3)	27 (9+18)	42 (19+23)
SMALL	1 (0+1)	5 (3+2)	2 (1+1)	8 (4+4)
VERY SMALL	1 (1+0)	4 (4+0)	5 (4+1)	10 (9+1)
TOTAL	6 (3+3)	20 (15+5)	34 (14+20)	60 (32+28)

Note: Numbers in brackets refer to units of domestic supply + units of foreign supply.

None of the above mentioned countries have nuclear reactors under construction, but all possess adequate infrastructures for launching nuclear projects. Any or all of them could reinitiate their nuclear power programmes and implement projects within the period considered. It is expected that prevalent conditions and existing situations will gradually change in favour of nuclear development. Assuming this, the expected market for SMRs would be 5 to 15 units, most of them in the medium power range.

Among the countries that have not yet initiated any nuclear power projects, Turkey and Indonesia are in the acquisition stage of their first units. Both have been intending to go nuclear for a long time. Turkey has invited bids and Indonesia is expected to do so shortly. Malaysia and Thailand have performed various studies which have indicated the convenience of the nuclear option, and the launching of a nuclear power programme is expected within the market assessment period. All four countries present a potential market for medium size reactors, in addition, Indonesia might implement a very small cogeneration unit for a remotely located site. The implementation of 5 to 10 SMR units is expected for this group of countries.

The North African countries: Algeria, Egypt, the Libyan Arab Jamahiriya, Morocco and Tunisia, show a high degree of interest in initiating nuclear power programmes. All have performed relevant studies and preparatory activities, including in some cases attempts to acquire nuclear power reactors. It is expected that new attempts will finally achieve success, leading to the implementation of 5 to 10 SMRs, including very small, small and medium size units.

Several countries which have not yet initiated nuclear power projects have performed studies and indicated interest in launching nuclear programmes. Belarus has persistent energy supply constraints and might acquire some medium size units from the Russian Federation. In Chile, the implementation of a small reactor could effectively contribute to energy supply diversification, in a situation characterized by a fast growing economy and corresponding energy and electricity demand, in particular in the center-north region. In Croatia, a follow-up unit to the 600 MW(e) plant built in Slovenia was planned. A new attempt could lead to implementing a medium size unit by the end of the period considered. Israel has consistently indicated interest in nuclear power, it has a solid nuclear technology infrastructure and could achieve implementation of a nuclear project, subject to successful conclusion of the middle east peace process. This applies also to Syria, which has intentions to proceed with medium size units. Portugal was on the verge of launching a nuclear power programme in the past, but has then desisted. A new attempt to implement medium size units could achieve success. Saudi Arabia has very large low cost oil and gas resources, but energy supply diversification seems advisable. A nuclear power programme starting with a very small or small reactor might be lounched. In addition, some other countries, have indicated interest in nuclear power and in SMRs in particular, performing studies and proceeding with building-up relevant infrastructures. Peru, Uruguay, Bangladesh are examples. There are also others, such as Cuba or the Philippines, where the construction of SMRs was suspended. In these countries, finishing the projects would have priority over the initiation of new plants. The market for SMRs estimated in the above mentioned countries is 5 to 10 units alltogether.

6. Results

The market estimates are summarized in Table 3 and Table 4, discriminated by geographic areas, for medium, small and very small units, five year intervals, high and low estimates, and also according to sources of supply, i.e., domestic or foreign.

Overall market. The results obtained indicate a market consisting of 60 to 100 units to be implemented up to the year 2015, which is a rather wide range. Probably not all countries will evolve according to either the high or to the low estimates. It seems reasonable to assume that there will be a certain compensatory effect. Also, it is recognized that forecasts, just like national development plans, tend to err on the optimistic side. Therefore, an overall market estimate of <u>70 to 80 units</u> seems reasonable.

Evolution of the market. There is a sustained gradually increasing trend of the overall market. At first, countries which have already started nuclear projects, have a predominant role. Throughout the assessment period, these countries have a share of about 70% of the market. Also, at first, projects implemented will be predominantly supplied by domestic sources, only during the latter part is the import market expected to attain major importance. Considering the overall period, domestic and foreign market shares are similar.

Market for medium size reactors. About 70% of the SMR units expected to be implemented are in the medium size range. An initially predominant position of domestic supplies is expected to gradually shift to a larger share of the foreign market. Three countries, China, India and the Russian Federation, represent together about 40% of the overall market for this size of reactors.

Market for small and very small reactors. Together, they represent about 30% of the overall SMR market, expressed in numbers of units. Very small reactors have a somewhat larger share of the market then small reactors. They are expected to be supplied predominantly by domestic sources, with the Russian Federation, China and the Republic of Korea accounting for about 60% of the units expected to be implemented. For small reactors, domestic and foreign market shares appear similar.

7. Market assessment of nuclear desalination

The market assessment of SMRs includes all applications of these reactors, that is electricity generation as well as the supply of heat for industrial processess or district heating. In view of the increasing interest in the use of nuclear energy for seawater desalination, an assessment of the market for this application in particular, is undertaken.

Seawater desalination requires energy, which can be electricity only, or heat and some electricity. When desalination plants are supplied with electricity from an electric grid which contains nuclear power generating capacity, a corresponding part of their energy consumption is effectively supplied by nuclear power. This applies to any type of industrial installation supplied with electricity from a grid, and is not considered "nuclear desalination", which has been defined as follows:

- The notion of nuclear desalination is taken to mean an integrated facility in which both the nuclear reactor and the desalination system are located on a common site and energy is produced on-site for use in the desalination system. It also involves at least some degree of common or shared facilities, services, staff,operating strategies, outage planning, and possibly control facilities and seawater intake and outage structures.

The criteria, assumptions, scope and methodology applied in the preceding market assessment of SMRs and the results obtained are used for the present assessment. In addition, the results of the market assessment of seawater desalination, which was performed within the framework of the "Options Identification Programme" (IAEA TECDOC-898), are also used. These provide an indication of which are the water-poor countries and regions, and what will be their demand for desalinated seawater.

Only the market for nuclear desalination as defined above is considered, though there are already and there will be more situations, where seawater desalination plants are supplied by electricity from a grid that contains nuclear power generating capacity. Nuclear desalination plants, in which the desalinated water produced is only used for the supply of the internal requirements of the facility, are not taken into account in the market assessment. Such installations create their own market for their product, and are not intended for the supply of outside consumers.

Based on the cost estimates provided by potential supplyers and the results of comparative economic assessments that were performed, it is assumed that economic competitivity is achievable under most conditions. It is also assumed that there will be an adequate level of confidence to implement nuclear desalination projects in general.

According to the procedure adopted, in a first phase, screening criteria are applied which lead to a list of countries assessed as having a potential market for nuclear desalination. For a country to be included in this list, i.e., to be selected for further consideration, two necessary conditions have to be fulfilled simultaneously. There has to be first of all a potential market for nuclear reactors and second, there has to be a demand of seawater desalination within the period of the market assessment.

In principle, any size of nuclear reactor (large, medium, small or very small) can be combined with a desalination system to constitute a nuclear desalination plant. Large reactors, however, appear less attractive for this application than SMRs. Large reactors are used essentially for electricity generation as base load plants integrated into large interconnected systems. When combined with seawater desalination, even for very large, 500,000 m^3 /d potable water production capacity, the electricity supplied to the grid would only be reduced by about 10 to 15%. They are therefore optimized for the conditions pertaining to the electricity market.

In the market assessment of SMRs, the application of the screening process resulted in the identification of a group of countries for further consideration. These countries were then individually assessed, and their markets for SMRs were estimated. They all comply with the first necessary condition for having a potential market for nuclear desalination. Considering the second condition, the above identified group of countries are assessed from the point of view of their expected demand for seawater desalination up to the year 2015. As a result, 20 countries are identified for further consideration. They are listed in Table 5. Each of these countries has a potential market for SMRs as well as for seawater desalination.

In addition to the countries with a potential market for SMRs, there are others with a market for large reactors only. These are assessed from the point of view of their potential demand for seawater desalination, and it is found that none of them show conditions which could justify their further consideration for nuclear desalination within the period considered.

The selected countries are assessed individually regarding their potential market for nuclear desalination. The market for SMRs has been assessed in all these countries, and the high and low estimates correspond to all potential applications, including seawater desalination. These estimates are taken as a basis for the assessment of the nuclear desalination market, which is an integral part of the SMR market. It is noted that some of the countries in the short list have a market for large reactors as well as for SMRs. It is assumed that in these countries, SMRs will be the preferred choices for nuclear desalination projects.

Within the group of countries which have experience in nuclear power projects, Argentina is not a water-poor country in general, but some regions on the sea shore have scarce fresh water resources. China is proceeding with demonstration activities in the field of nuclear desalination. Some of the very small and small reactor projects are expected to supply potable water to satisfy the demand in coastal locations with inadequate fresh water resources. Though India is not a water-poor country in general, it has substantial potable water supply problems in several regions. There are ongoing demonstration activities in nuclear desalination, and some of the small and medium size reactor projects might to be combined with desalination. The Islamic Republic of Iran has a substantial market for seawater desalination. Its nuclear reactors to be implemented are expected to be combined with desalination. In Italy, regional potable water supply problems have led to an increasing use of seawater desalination, which is expected to grow from the current overall installed capacity of about 100,000 m^3/d, to more than ten times that amount by the year 2015. Nuclear desalination is an option. There is interest in the Republic of Korea in desalination, and there are ongoing activities concerning the design and implementation of a very small reactor expected to be combined with seawater desalination. Some regions in Mexico with inadequate fresh water resources increasingly turn to seawater desalination, which by the year 2015 is expected to reach more than 700,000 m^3/d overall installed capacity. Combining a nuclear plant with desalination might be considered. Pakistan has some water-poor regions where seawater desalination combined with one of the nuclear power projects of the country could offer a viable and convenient solution. In the Russian Federation, there is experience in nuclear desalination and interest in the export market. For the supply of energy and potable water in remote locations of the country, some very small reactor projects might be combined with seawater desalination. In South Africa, very small reactors coupled to desalination are expected to be implemented, to supply energy and potable water in remote coastal locations. In the above considered group of countries, it is estimated that 16 to 23 nuclear desalination plants will be implemented within the period considered for the market assessment.

Among the group of countries which have yet to start their first nuclear reactor project, Algeria is a water-poor country and interested in nuclear desalination. The central-northern region of Chile lacks adequate fresh water resources and has fast growing energy demand. The nuclear power reactors to be implemented in Egypt are expected to include seawater desalination to supply regional demand for potable water. The site selected for the first nuclear project in Indonesia has practically no fresh water resources, and is planned to supply both internal water requirements of the plant and the demand in the adjacent region by sewater desalination. Israel is a water-poor country, and its reactor projects are expected to be combined with desalination. The Libyan Arab Jamahiriya has substantial demand for potable water and is very interested in nuclear desalination. In Morocco, there are on going activities directed to the implementation of a demonstration project of nuclear desalination with a very small reactor. Saudi Arabia has the largest installed seawater desalination capacity (about 4,000.000 m^3/d) of the world, which is expected to triple by the year 2015. Combining the generation of electricity with desalination is normal practice in the country. Syria and Tunisia are water-poor countries interested in nuclear desalination. The market estimate for nuclear desalination in the above mentioned group of countries is of 9 to 17 units.

The results of the market assessment of nuclear desalination are summarized in Table 6, discriminated by size of reactor, five-year periods, and according to domestic or foreign sources of supply of the reactors.

TABLE 5

NUCLEAR DESALINATION MARKET ASSESSMENT LIST OF COUNTRIES FOR FURTHER CONSIDERATION

First projects have been started	No nuclear power project started
ARGENTINA	ALGERIA
CHINA	CHILE
INDIA	EGYPT
IRAN, ISAMIC REPUBLIC OF	INDONESIA
ITALY	ISRAEL
KOREA, REPUBLIC OF	LIBYAN ARAB JAMAHIRIYA
MEXICO	MOROCCO
PAKISTAN	SAUDI ARABIA
RUSSIAN FEDERATION	SYRIAN ARAB REPUBLIC
SOUTH AFRICA	TUNISIA

TABLE 6

NUCLEAR DESALINATION MARKET ASSESSMENT SUMMARY

HIGH ESTIMATE	2001-2005	2006-2010	2011-2015	TOTAL
MEDIUM	-	5 (0+5)	12 (2+10)	17 (2+15)
SMALL	2 (0+2)	3 (3+0)	5 (0+5)	10 (3+7)
VERY SMALL	3 (2+1)	4 (3+1)	6 (4+2)	13 (9+4)
TOTAL	5 (2+3)	12 (6+6)	23 (6+17)	40 (14+26)

LOW ESTIMATE	2001-2005	2006-2010	2011-2015	TOTAL
MEDIUM	-	1 (0+1)	8 (0+8)	9 (0+9)
SMALL	1 (0+1)	4 (2+2)	2 (1+1)	7 (3+4)
VERY SMALL	-	4 (3+1)	5 (4+1)	9 (7+2)
TOTAL	1 (0+1)	8 (5+3)	15 (5+10)	25 (10+15)

Nuclear desalination. The results obtained indicate a market for nuclear desalination consisting of 25 to 40 units expected to be implemented up to the year 2015. All nuclear desalination plants are expected to be in the SMR size range. The high and low estimates of the nuclear desalination market assessment have been based on the respective high and low estimates of the SMR market. This implies combined expectations under more, or under less favourable conditions and scenarios, as applicable to overall nuclear

power development, the market of SMRs, and finally the market share of nuclear desalination. The overall effect of the combined expectations may result in too optimistic estimates. Therefore, the low estimate of about 25 units seems to be more reasonable. This corresponds to about a third of the SMR market.

The evolution of the market. There is a sustained gradually increasing trend of the market. The projects expected to be implemented are distributed among the countries identified as having a potential market, with no particular country or group of countries showing a predominant role. There is, however, a clear concentration of the market in the Middle East, South Asia and in North Africa. These regions account for more then half of the overall market. Regarding supplies from domestic or from foreign sources, the overall market to be supplied from foreign sources appears to be nearly twice as large as the one to be supplied by domestic sources.

Nuclear desalination with medium size reactors. About 40% of the units expected to be implemented are in this size range, practically all of them to be provided by foreign sources of supply. As a rule, it is expected that the nuclear desalination projects will consist of twin-unit stations, with both units capable of supplying energy to the desalination system. All medium size units are expected to be integrated into the interconnected electric grid system, to which they supply most of the energy they produce.

Nuclear desalination with small and very small reactors. Together, they represent about 60% of the units expected to be implemented, wit a somewhat larger share corresponding to the very small reactors. While most of the very small units are expected to be of domestic supply, a larger share of the small units is expected to be imported from foreign sources.

XA9846717

POTENTIAL ROLE OF NUCLEAR POWER IN THE MOROCCAN ENERGY PROGRAMME

A.A. HADDOU, M. TABET
Centre National de l'Energie des Sciences
et des Techniques Nucleaires,
Rabat, Morocco

Abstract

Morocco has very limited energy resources; it imports over 90% of its commercial energy. Rainfalls are irregular; seawater desalination is increasing in the southern arid zone. In the context of the national energy and water plans, small and medium-size nuclear power plants are considered both for electricity generation and seawater desalination.

1. INTRODUCTION

Morocco is a north African country with a coast length of 3446 km extended along the Atlantic Ocean (2934 km) and the Mediterranean Sea.

Nearly 50% of the total area which is 710 850 km^2 is located in the semi-arid and arid zones. The total population in 1995 was 26 910 000 with an average growth of 2.5%. Urban population is about 51% located essentially in the cities between the Atlas mountains and the coast line.

Considering its liberal and galloping economy, and due to its very limited conventional energy resources, Morocco gives a great importance to nuclear energy option with a view to electricity generation and seawater desalination.

To reach this important goal, many preliminary actions have been undertaken to prepare the country for nuclear energy introduction on the horizon of the year 2010-2015.

Among these actions, continuous attention is paid to new developments of nuclear technology taking a particular interest in small and medium power reactors.

2. OVERVIEW OF ENERGY SECTOR IN MOROCCO

Morocco has very limited fossil resources including coal reserves. Hydropower potential fluctuates between 2% and 7% depending strongly on hydrological conditions. Furthermore, it is important to give priority to the irrigation rather than electricity generation.

In 1995, the total energy consumption was 8 285 000 TOE. It was covered by oil (74.1%), Coal (23.8%), Hydro (1.9%) and Gas (0.2%). The consumption per capita is estimated at 0.31 TOE. More than 90% of the commercial energy demand is imported and about 3 million TOE is firewood consumed by rural populations.

3. SITUATION OF THE ELECTRICAL ENERGY SECTOR AT THE END OF 1995

In 1995, electricity consumption totalled 10 829 GWh produced by the National Office of Electricity (10 523 GWh), independent production (65 GWh) and 241 GWh supplied by Algeria.

The total installed capacity of 3451 MW(e) at the end of 1995 comes from 23 hydropower plants (920 MW(e)), 5 fossil power plants and 8 gas turbines (2531 MW(e)).

The projects in progress are the following:

Coal	2 x 330 MW(e)
Combined Cycle	300 to 450 MW(e)
Combined Cycle	350 to 450 MW(e)
Aerogenerator	50 MW(e)

Based on 5 to 7% growth rate, forecasts for the year 2000 and 2005 are 5300 MW(e) and 7000 MW(e) installed capacity and a consumption of 19000 GWh and 23000 GWh respectively.

4. NEW ENERGETIC ORIENTATIONS AND STRATEGY

The major changes occurred recently in the energy sector of Morocco are as follows:

- Deregulation concerning concessional production of electricity and private investment encouraging,
- Restructure and institutional development of energy sector,
- Diversification of consumed energetic products, mainly gas and coal,
- Good energy husbandry and efficiency taking into account the protection of the environment,
- Decentralization and regional development giving priority to the rural electrification,
- Interconnection with Algeria and Spain,
- Maghreb-Europe pipeline installation through Moroccan territory,
- Oil prospection strengthening,
- National nuclear power programme follow-up.

5. ACTIONS UNDERTAKEN TOWARD SEAWATER DESALINATION

The key data concerning the water resources and water status in Morocco are given below:

- Irregularity of rainfails,
- 125 Bm^3 / y of average rainfall,
- 70 large dams with a total storage capacity of 11 Bm^3,
- 5 Bm^3/ y of ground water, 60% of which are presently utilized,
- Tens of small cities in the south and arid zones have or will have recourse during the next 25 years, to desalination with capacities of 3000 to 20000 m^3/ day per city,
- The total needs of desalted water could reach then 150 000 to 200 000 m^3/day,
- Big desalination units are not suitable for Moroccan case.

For these last three reasons, Morocco is considering the desalination using small RO or MED units.

Recently, a feasibility study for nuclear desalination using a 10 MW heating reactor has been launched.

6. NATIONAL NUCLEAR POWER PROGRAMME STATUS

Until now, the preliminary steps to introduce nuclear energy in Morocco have been accomplished. The most important of them are:

- The establishment of the National Council of Nuclear Energy (CNEN) with promotional missions aiming at nuclear activities orientation and coordination, and regulation aspects,
- The establishment of the National Center for Nuclear Energy Sciences and Techniques (CNESTEN) as a technical and scientific institution with missions of research and development, training and promotion of nuclear technology and techniques in the country. The first realization of CNESTEN is a Nuclear Research Center around 2 MW Triga Mark II Reactor and many modern laboratories. The construction of the center will start at the beginning of 1997.
- The establishment of the National Center of Radiation Protection (CNRP) as a control body for all radiation protection aspects.
- The adoption of regulations concerning radiation protection, nuclear installations licensing and safety committee set-up.
- The siting and feasibility studies have been conducted successfully by the National Office of Electricity for the implementation of the first nuclear power plant but, the decision to embark has not yet been taken.
- As it has been said before, nuclear desalination is considered using a small nuclear heating reactor coupled with a MED desalination plant. A feasibility study is undertaken in collaboration with China.

7. CONCLUSION: MOST LIKELY SCENARIOS FOR A NUCLEAR POWER PROGRAMME IN MOROCCO

A nuclear power programme could be brought to a successful issue only if it is derived from a national energy plan and a long-term electric system expansion plan.

An average time span of 20 years and at least 5 units to be installed during this period seem to be optimum conditions for Morocco.

If we take into account the growth rate of electricity demand and the size of conventional units already installed or to be installed in the near future, the most suitable sizes of the foreseen nuclear power plants would be in the range of 300 to 600 MW(e), which is the range of small and medium power reactors.

Concerning the reactor type(s), it will depend strongly on technology development, safety level, competitiveness and technology transfer willingness.

MARKET POTENTIAL OF SMALL AND MEDIUM POWER REACTORS IN SYRIA

I. KHAMIS, A. HAINOUN
Atomic Energy Commission of Syria,
Damascus,
Syrian Arab Republic

Abstract

Analysis of the Syrian energy demand and forecasting was briefly introduced. The need to install an average of 500 MW annually the year 2003 was established. Moreover, short introduction of the main energy resources in the country was made. A primitive criteria for the selection of SMPR was emphasized. An emphasis for the process of introducing the first nuclear power in the country was also recognized.

1. INTRODUCTION

Syria has witnessed an important development in the field of electricity demand and generation over the last twenty five years. Such development corresponded to other developments in many other sectors of life such as social and economical ones. As a result, huge investment was made in the field of electricity generation where the total planned to-be-installed capacity will reach about 4000 MW by the 1995. In addition, another project to install 600 MW thermal plant is being considered.

Recent studies indicate that such capacities as well as already installed ones will meet the demand at both base and peak power till the years 2003 to 2004. Other statistics indicate that despite the large increase of population in Syria, the average rate of increase in energy production has reached almost 13% which is one of the highest rate in the world (see figure 1). Energy forecasting studies indicated that energy demand for the year 2000 will be about 25 billion KW.h. That means that the rate of increase in energy production will be around 9% . In addition, for the periods 2000 to 2005 and 2005 to 2020, the rate of increase will be 6% and 5% respectively. In this case, demand on energy will be 42.7 and 70 billion KW.h for the years 2010 and 2020. Considering the rate of increase of population about 3.31% , the expected population for the years 2000, 2010, and 2020 will be 16.79, 23.25, and 32.2 million inhabitants respectively. The person share of produced energy will be 1489, 1836, and 2174 KW.h annually (see figure 2). However, despite the fact that this share is lower than the current average worldwide rate which is 2200 KW.h, it is much higher than that in many developing countries. Keeping the same rate of population increase, the peak power demand will reach 4200 MW in the year 2000 and it will be 7172 and 11682 MW for 2010 and 2020 respectively.

Taking all consideration into accounts, including the already installed and contracted to be installed capacities, as well as the effective life time of such generating plants, the total available capacity in the year 2000 will be about 6900 MW. This shows the need for additional 8000 MW to cover the demand during the period 2003 and 2020. It means that, starting from the year 2003, on the average a 500 MW is annually required, with an approximate cost of 600 million of the 1996-US dollar is needed to cover both the capital cost of generating and transforming stations. The total cost covering all 8000 MW plants with their required transforming stations will be around 12 billion 1996-US dollar.

2. ENERGY RESOURCES

Electricity generation in Syria depends on three main resources, namely, water, natural gas, and fuel oil. The current annual estimated water-generated energy using all possible resources is about 2500 million KW.h which makes less than 6% of the total production rate. The rest of production should be made out of thermal resources. This totals to about 40 000 million KW.h in the year 2010, and almost 67 500 million KW.h in 2020. And, based on current average specific fuel consumption, a total of 10 and 17 millions equivalent tons of fuel oil are required for the years 2010 and 2020 ,respectively. Hence, if one considers the current price of one equivalent ton of fuel oil to be 100 US dollars, an estimated amount of 1000 million US dollars is required annually in the year 2010, and 1700 million US dollars in 2020 to cover thermal energy production using gas and fuel oil only.

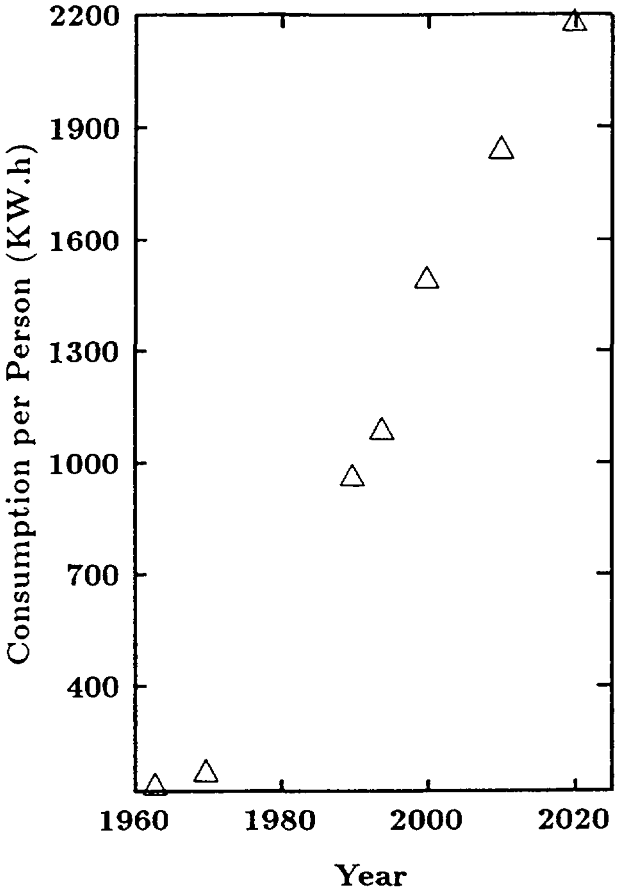


Fig. 1. CONSUMPTION PER PERSON vs. YEAR

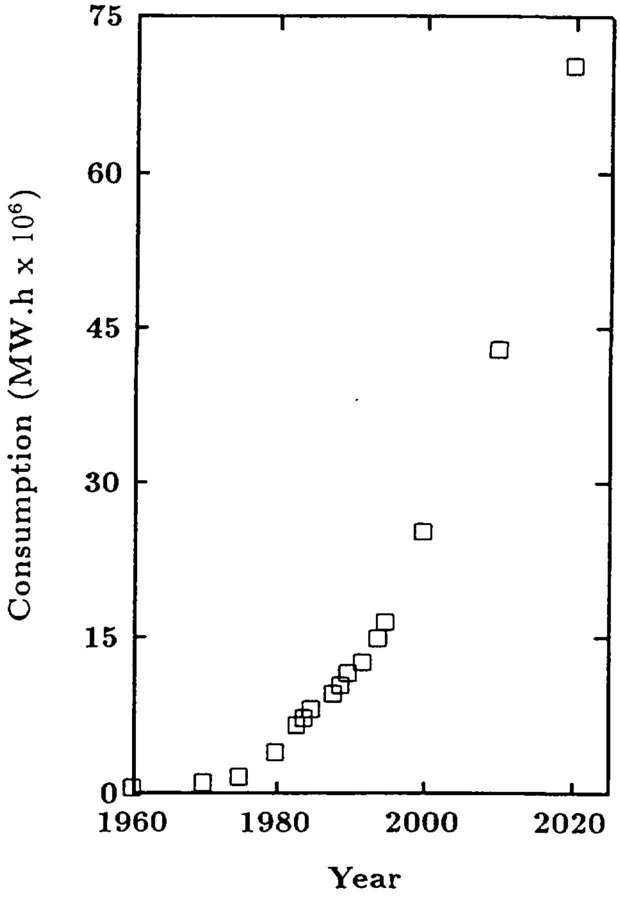


Fig. 2. TOTAL CONSUMPTION vs. YEAR

In addition to water resources, natural gas production is expected to make an important contribution to power production. The total estimated quantity that is expected to be produced in the year 2000 is about 15.85 million m^3 daily. The energy designated share out of this total is 13.5 million m^3, and the rest is diverted to be used in industrial and refinery applications. This share is expected to enable the production of 13 600 million KW.h in accordance with current strategy, where Syria is currently implementing a program to modify all gas driven turbine plants from fuel fired only to combined cycle i.e. fuel oil plus gas fired plants. The rate of electricity production is expected to reach around 17000 million KW.h after modification of all plants is over.

Fuel oil is produced locally in Syria at an annual rate of 4.3 million tons. The designated share of this rate for energy production is estimated to be 2.8 million tons for the year 2010. However, the estimated required quantity of fuel oil for the same year is approximately 6.580 million ton. In this case, the predicted deficit in fuel oil is 3.78 million tons. To substitute for such additional deficit, more refineries are required with total output roughly about 9.5 million tons for the year 2010. Whereas for the year 2020, and even in cases of keeping the current rate of natural gas production constant, and modification of all gas driven turbines is finished, the need for fuel oil will be around 12.5 million tons annually. According to the current capacities of available refineries, the 2020-deficit will be about 10 million tons annually. As a result, Syria should either import an equivalent of this deficit or erect additional refineries with total capacities equal to 25 million tons, taking into consideration that fuel oil production percentage is 40%.

Based on the aforementioned forecasting, the required 12 billion US dollars estimated cost , the additional cost of fuel resources which will be about 1.7 billion US dollars, and taking into consideration the latest studies published by IAEA in regard to the increasing demand on electricity in such a way that such demand will be overcome by the sharp increase in its prices which will lead finally to a global increase on energy which in turns, might reach 70% at the year 2020, the nuclear option seems very realistic and much promising one for a country like Syria.

3.SMPR CONSIDERATIONS

Despite all efforts being invested in the field of energy production in Syria, the situation is still not very prosperous. Especially if one looks at other disadvantages of installation of such large number of thermal plants, mainly, the environmental aspects of such sole choice of affordable energy. Therefor, other considerations are also being investigated. In fact, decision makers tried to go nuclear in the 80's and still consider such option ever since. However, the least one could say that financing of such option seems always formidable. Simultaneously, diversification of energy resources seems to be inevitable and the nuclear option seems to be the last resort available so long as urgency for such option is increasing with decreasing resources of hydro, oil, and natural gas.

Recognizing that a single unit should not exceed a given fraction of the grid size, and the annually expected required energy is about 500 MW, favor of small to medium power reactor is recognized over larger units. The early choice during the 80's was 440-VVER type power plant. Among other consideration at that time was the size of such relatively medium reactor. Still, the same size option or even smaller one is favored.

Other requirements for the introduction of first nuclear power plant in the country are being investigated including are : manpower development, organizational structures such as regulations, program planning, project implementation, plant operation and maintenance, and industrial support.
It is widely known and became apparent that SMPR has many advantages over large nuclear power plants for the followings :

- Modularization and/or standardization of SMPR in small units tends to lower economic risk through even distribution among plants,
- SMPR plants are more suitable for small and medium size electric grids,
- SMPR offers lower absolute capital cost and shorter planning and construction time,
- SMPR are more convenient for low load growth as is the case of most developing countries, and
- SMPR plants may offer better reliability in case of loss of load probability.

It should also be mentioned that, standardization feature of SMPR will be a very persuasive advantage for decision makers in Syria. Among many other reasons, it fits the predicted annual required energy on one side and requires lower total capital investment for a standardized plant in series on the other side. However, the first foreseen nuclear power plant project in the country is not expected before 2003. Later on, at least two to four units are widely perceived.

The main factor that influences the quick implementation of the nuclear option will continue to be the financing and economics of such project. In other words, the more easy to facilitate financing of nuclear power plant, the more quickly the decision will be made to embark and implement such project. This might also be very true for many countries worldwide. Of course urgency for energy is much more recognized in the developing countries where more than two billion people are without sufficient source of energy or their energy consumption is much less than the average share of person in many countries. Still, the immediate prospect of SMPR project in the developing countries is not seen in the near future, say till the end of 2000. However, the potential market for SMPRs in the range of 300 MW is well established. The estimated number of such units is expected to exceed 20 over a period of 10 to 20 years starting at the beginning of the 21st century. Of course, there are considerable number of uncertainties affecting this estimate mainly due to economics and characteristics of nuclear power plant project implementation.

4. FACTORS FACILITATING SMPR PROJECT IMPLEMENTATION

One of the key factors -influencing decision making is the availability of qualified manpower. In general, availability of competent manpower continues to pose a major problem that always affect the launching of nuclear power program and forms a constraint that might hinder the early introduction of such program. It is well recognized that nuclear technology transfer plays an essential rule in development of adequate manpower through education and training. The number and qualification of staff will depend strongly on the two sides of the contract on one hand, and on the national policy of the country embarking on nuclear power program on the other hand. In all cases, a manpower development program should clearly be established and many years before the first nuclear power plant is expected to operate.

Assistance in manpower development before launching of the first nuclear power plant project is considered the vital rule of IAEA. Such assistance should be stressed not only in manpower training development but also in many other activities related to the setup of national infrastructure dedicated to the introduction of nuclear power programmed such as : required organizational infrastructure, national regulating and licensing structure, survey of national industrial sector and its participation and development within a nuclear power program, electrical grid analysis, financing...etc. In this regard, computer software and computer simulation should be very strongly emphasized. It should also be stressed that real time or full scale simulation is not required as a first step for training. Instead, general purpose and theoretical oriented application could easily fulfill this task. Later on, the more the national nuclear program is enhanced the more the full scale simulation and on-job-training is required. In addition, since so many technical papers and documents have been published regarding SMPR, it is advisable that a yearly updated version about this subject should be done in order to keep buyers informed on what is available of SMPR in the market. Accompanied with this updated version, a small scale PC oriented simulator is suggested. Furthermore, sharing of successful and unsuccessful experience on establishment, execution, and implementation of nuclear power program among member states through IAEA will enable many countries to study the idea of launching such program very carefully. Hence, assistance from IAEA will be minimized. Simultaneously, it should be acknowledged that various publications by IAEA give solid guidelines and advises in this area. Finally, It might be advisable that IAEA could lead the way by establishing contact with many SMPR producing companies with large share of experience in designing and manufacturing such nuclear power plant simulators to make it available for member states especially from developing countries.

REFERENCES

[1] ALLAO, S., Electricity demand in Syria and the future perspectives, Al shabakat Journal, Damascus (1996).
[2] IAEA-TECDOC-739, Case study on the Feasibility of Small and Medium Power Plants in Egypt, Vienna (1994).
[3] IAEA-TECDOC-347, Small and Medium Power Reactors: Project Initiation Study, Phase I, Vienna (1985).
[4] IAEA-TECDOC-445 Small and Medium Power Reactors 1987, Vienna (1987).
[5] IAEA-TECDOC-376, Small and medium Power Reactors 1985, Vienna (1986).

XA9846719

SUMMARY OF SURVEY ON SMR MARKET POTENTIAL IN JAPAN

T. HOSHI, M. OCHIAI
Japan Atomic Energy Research Institute,
Ibaraki, Japan

Abstract

The nuclear power generation in Japan has grown to more than 30 % of the total electricity generation as of the end of 1995. Considering the increase of energy demand in the future, the steadily energy supply is requested.

The paper presents the outlook of energy supply and consumption in the future, the status of nuclear power generation and market potentials of the small and medium nuclear power plants in Japan.

1. Demands of Energy in Japan

Records and outlook of the primary energy supply and of the end-energy consumption in Japan are shown in Table 1 and 2 [1]. In the tables, estimates were made for amounts of primary energy demand and end-energy consumption for the target years of 2000 and 2010, for which two scenarios were created: one only using existing measures and the other using energy saving measures.

In Fig. 1 and Fig 2, the percentage of electricity generation in the primary energy consumption and the energy sources for electricity production are shown, respectively.

2. Nuclear Power Generation

(1) Nuclear Power Plant

The first nuclear electric power was generated in 1963 at the JAERI (Japan Atomic Energy Research Institute) by the JPDR (Japan Power Demonstration Reactor(12.5 MWe BWR) which was designed and constructed by the GE, U.S.A. and had been operated by the JAERI for the experiment and research as well as operator training.

In 1966, the first commercial nuclear power plant began operation and the nuclear power generation has steadily grown to about 41 GWe by 50 NPPs (26 BWRs, 22 PWRs, 1 GCR and 1 ATR) as of the end of 1995, which accounts for 31 % of the total electric power consumption in Japan. In addition, 4 NPPs including the prototype FBR "MONJU" are under construction and more than 6 NPPs in planning.

The map of NPP sites and the specification of plants are given in Fig. 3 and Table 3.

(2) Operating Records and Cost of Electricity Generation

Average capacity factor of total NPPs in FY 1993 was 75.4 % (BWR: 76.7%, PWR: 74.7%). Recent operating record of each NPPs is given in Table 4 [2].

The construction cost of a power plant and the generation cost in FY 1992 is shown in Table 5 [3]. Trend of the operation and maintenance cost is given in Fig. 4 [4].

Medium size NPPs are in operation in which NPPs less than a power output of 600 MWe are as follows;

Size (MWe)	BWR	PWR	GCR	ATR
< 200	0	0	1	1
200 - 400	1	1	0	0
400 - 600	4	6	0	0

(3) Decommissioning

Decommissioning of the first experimental power plant, JPDR was completed in March 1996 in the condition of IAEA Stage 3 (Dismantlement), of which work was performed by the JAERI.

Regard with commercial nuclear plants, the first commercial power plant, Tokai NPP, 166 MWe GCR, is planned to shutdown in 1998 for decommissioning.

In Japan, the electric power company is requested to reserve the decommissioning cost of NPPs during their operation. Decommissioning cost of a 1,100 MWe NPP has been estimated of about 30 billion yen which was evaluated in 1985 by the Advisory Committee for Energy, an advisory organ to the MITI (Ministry of International Trade and Industry).

3. Organizations Related to Nuclear Power Generation

Outlook of the related organizations on the nuclear energy usage in Japan is shown in Table 6.

(1) Administrative Organs

In Japan, the development and utilization of atomic energy are subjected to the" Atomic Energy Basic Law" established in 1955.

In order to accomplish the national policy and administration intentionally and democratically, the Atomic Energy Commission and the Nuclear Safety Commission have been formed in the Prime Minister's Office according to the Atomic Energy Basic Law. The Atomic Energy Commission has a responsibility for planning, deliberation and decision regarding the promotion for the development and utilization of atomic energy. On the other hand, the Nuclear Safety Commission has a responsibility for planning, deliberation and decision regarding the safety assurance on the development and utilization of atomic energy.

Safety regulation on the usage of atomic energy are strictly governed by the "Law for Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors" established in 1957.

Under the above laws, the actual regulation for construction, operation and decommissioning are made by the Competent Ministers in accordance with the kind of usage as follows;

STA (Science and Technology Agency):
- research reactor and proto-type reactor
- refining business, fabricating business and reprocessing

- waste disposal business
- use of nuclear fuel materials, etc.

MITI (Ministry of International Trade and Industry)
- commercial power reactor

MOT (Ministry of Transport)
- marine reactor
- transportation of nuclear fuel materials

(2) Research Organs

Many organs, administration-related and private, are in Japan for the research and development of atomic energy usage. Typical research organs are;

Administration-related:
- JAERI (Japan Atomic Energy Research Institute)
- PNC (Power Reactor and Nucl. Fuel Development Corp.)
- NUPEC (Nuclear Power Engineering Center)
- SRI (Ship Research Institute)

Private:
- CRIEPI (Central Res.Institute of Electric Power Industry)

(3) Electricity Utilities

The electric power generation and supply are made by nine electric power companies (utilities) who ought to supply the electricity for the prescribed districts in Japan.

For the education and training of operators and maintenance engineers for the NPPs, the training centers are being established for the BWR NPP and the PWR NPP.

To decide the basic direction in the nuclear power as well as other fields in the electric companies, the Federation of Electric Power Companies (FEPCO) plays a role.

(4) Manufacturers

Typical manufacturers related to the NPPs and their components are listed in Table 6.

4. SMR Market Potentials

(1) Questionnaires to Suppliers

Taking into consideration of the research and development activities carried out for SMRs in Japan, the IAEA questionnaires were forwarded to the research organizations and manufacturers.

Replies were made by following organizations.

Research organization;

JAERI	:	MRX, SPWR, JPSR
CRIEPI	:	MDP, 4S
PNC	:	DSFR, SATR

Manufacturer;

Hitachi	:	HSBWR, SBWR, BWR-5
Toshiba	:	SBWR, BWR-5
Mitsubishi	:	MS-600

Typical design features of each reactor are summarized in Table 7.

(2) Questionnaires to Buyers

At present the electric utilities are interesting in the large NPPs for power supply because the large increase of electricity demand is expected in the future and the sites of installation of the NPPs are limited. No answer was, therefore, made by the electric utilities.

Table 1 Primary Energy Supply

Total Supply of Primary Energy

(Units Million Crude-Oil-Equivalent Kiloliters)

FY		73	79	85	86	87	88	89	90	91	92	93
Total Primary Energy Supply		414	442	438 (0 6)	435 (s0 8)	457 (5 0)	481 (5 4)	499 (3 7)	526 (5.3)	531 (1 0)	541 (2 0)	548 (1 2)
Component Ratio (%)	Oil	77 4	71 5	56 3	56 6	56 9	57 3	57 9	58 3	56 7	58 2	56 5
	Coal	15 5	13 8	19 4	18 2	18 0	18 1	17 3	16 6	16 9	16 1	18 1
	Natural Gas	1 5	5 2	9 4	9 8	9 7	9 6	10 0	10 1	10 6	10 6	10 7
	Nuclear	0 6	3 9	8 9	9 4	10 0	9 0	8 9	9 4	9 8	10 0	11.1
	Hydro	4 1	4 6	4 7	4 6	4 1	4 6	4 6	4 2	4 6	3 8	4.3
	Geothermal	0 0	0 1	0 1	0 1	0 1	0 1	0 1	0 1	0 1	0 1	0 1
	New Energy	0 9	1 0	1 2	1.2	1.3	1.3	1.3	1 3	1.3	1 2	1.2

(Source: MITI)

Note Figures in parentheses show percentage changes from previous FY (s means decrease)

Outlook for Primary Energy Supply

FY	1992	2000		2010	
		Existing measures scenario	Additional measures scenario	Existing measures scenario	Additional measures scenario
Total primary energy supply (10^6 kl)	541	589	582	660	635
Oil (10^6 kl)	315 58 2%	315 53 5%	309 53 1%	331 50 1%	302 47 7%
Coal (10^4 tons)	11,630 16 1%	13,400 16 6%	13,000 16 5%	14 000 15 4%	13,400 15.3%
Natural gas (10^4 tons)	4 070 10 8%	5,400 12 8%	5,320 12 8%	6,000 12 8%	5,800 12.8%
Nuclear (10^8 kWh)	2,230 10 0%	3,100 12 1%	3,100 12 3%	4,800 16 2%	4,800 16 9%
Hydro (general) (10^8 kWh)	790 3 8%	860 3 1%	860 3 1%	1,050 3 2%	1,050 3.3%
Geothermal (10^4 kl)	55 0 1%	100 0 2%	100 0 2%	380 0 6%	380 0 6%
Renewable (10^4 kl)	670 1 2%	880 1 5%	1,140 2 0%	1 090 1 6%	2,080 3 3%

Note: Figures in the lower line of each matrix show ratio of persector energy supply to the total primary energy supply

Table 2 End-Energy Consumption

(Units: Million Crude-Oil-Equivalent Kiloliters)

FY	73	79	85	86	87	88	89	90	91	92	93
Final Energy Consumption	285	301	292 (1.2)	294 (0.4)	308 (4.8)	325 (5.6)	336 (3.5)	349 (3.8)	358 (2.6)	360 (0.4)	362 (0.7)
Industry	187	178	158 (▲0.4)	156 (▲1.2)	163 (4.8)	173 (5.9)	178 (2.8)	183 (3.2)	185 (0.7)	181 (▲2.0)	181 (0.4)
House-holds	52	63	71 (3.7)	72 (1.0)	76 (5.2)	80 (5.4)	82 (2.2)	85 (4.6)	89 (4.9)	93 (3.9)	94 (1.1)
Transportation	47	60	64 (2.4)	66 (3.5)	69 (4.1)	72 (5.1)	77 (6.8)	80 (4.5)	84 (4.5)	86 (2.2)	87 (0.9)

(Source: MITI)

Notes 1) Figures in parentheses show percentage changes from previous FY (▲ means decrease)
2) In "Industry" category, non-energy consumption is included.

Outlook for End-Energy Consumption

(Unit: million kl.)

FY	1992	2000		2010	
		Existing measures scenario	Additional measures scenario	Existing measures scenario	Additional measures scenario
End-energy consumption	360	395 (1.2)	390 (1.0)	446 (1.2)	425 (0.9)
Industry	181	187 (0.4)	187 (0.4)	205 (0.9)	200 (0.7)
Civil	93	113 (2.4)	109 (2.0)	136 (1.9)	128 (1.6)
Transportation	86	95 (1.3)	93 (1.0)	105 (0.9)	97 (0.4)

Note: Figures in parentheses show annual average growth rates for fiscal 1992 to 2000 and fiscal 2000 to 2010, respectively.

Table 3 Specifications of NPPs in Japan (BWR)

Utility	Unit name		Electric output (MW)	Thermal output (MW)	Type	Fuel layout	Power density (kW/ℓ)	Forced recirculation system: Recirculation system	Forced recirculation system: number of pumps	Forced recirculation system: control system	PCV model	Main contractors	Commercial operation
Tohoku Electric Power	Onagawa	1	524	1593	BWR−4	8×8	50	pump + jet pump	2	MG set	MARK-I	Toshiba	Jun. 1984
		2	825	2436	BWR−5	8×8	51	pump + jet pump	2	Thyristor	Improved MARK-I	Toshiba	Under construction
Tokyo Electric Power	Fukushima Daiichi	1	460	1380	BWR−3	8×8	41	pump + jet pump	2	MG set	MARK-I	GE	Mar 1971
		2	784	2381	BWR−4	8×8	50	pump + jet pump	2	MG set	MARK-I	GE/Toshiba	Jul 1974
		3	784	2381	BWR−4	8×8	50	pump + jet pump	2	MG set	MARK-I	Toshiba	Mar 1976
		4	784	2381	BWR−4	8×8	50	pump + jet pump	2	MG set	MARK-I	Hitachi	Oct 1978
		5	784	2381	BWR−4	8×8	50	pump + jet pump	2	MG set	MARK-I	Toshiba	Apr. 1978
		6	1100	3293	BWR−5	8×8	50	pump + jet pump	2	MG set	MARK-II	GE/Toshiba	Oct 1979
	Fukushima Daini	1	1100	3293	BWR−5	8×8	50	pump + jet pump	2	MG set	MARK-II	Toshiba	Apr. 1982
		2	1100	3293	BWR−5	8×8	50	pump + jet pump	2	MG set	Improved MARK-II	Hitachi	Feb 1984
		3	1100	3293	BWR−5	8×8	50	pump + jet pump	2	MG set	Improved MARK-II	Toshiba	Jun 1985
		4	1100	3293	BWR−5	8×8	50	pump + jet pump	2	MG set	Improved MARK-II	Hitachi	Aug 1987
	Kashiwazaki Kariwa	1	1100	3293	BWR−5	8×8	50	pump + jet pump	2	MG set	MARK-II	Toshiba	Sep 1985
		2	1100	3293	BWR−5	8×8	50	pump + jet pump	2	MG set	Improved MARK-II	Toshiba	Sep 1990
		3	1100	3293	BWR−5	8×8	50	pump + jet pump	2	Thyristor	Improved MARK-II	Toshiba	Under construction
		4	1100	3293	BWR−5	8×8	50	pump + jet pump	2	Thyristor	Improved MARK-II	Hitachi	Under construction
		5	1100	3293	BWR−5	8×8	50	pump + jet pump	2	MG set	Improved MARK-II	Hitachi	Apr 1990
		6	1356	3926	ABWR	8×8	51	reactor internal pump	10	Thyristor	Reinforced concrete	Toshiba/GE/Hitachi	Under construction
		7	1356	3926	ABWR	8×8	51	reactor internal pump	10	Thyristor	Reinforced concrete	Hitachi/GE/Toshiba	Under construction
Chubu Electric Power	Hamaoka	1	540	1593	BWR−4	8×8	51	pump + jet pump	2	MG set	MARK-I	Toshiba	Mar 1976
		2	840	2436	BWR−4	8×8	50.5	pump + jet pump	2	MG set	MARK-I	Toshiba/Hitachi	Nov 1978
		3	1100	3293	BWR−5	8×8	50	pump + jet pump	2	MG set	Improved MARK-I	Toshiba/Hitachi	Aug 1987
		4	1137	3293	BWR−5	8×8	50	pump + jet pump	2	Thyristor	Improved MARK-I	Toshiba/Hitachi	Under construction
Hokuriku Electric Power	Shika	1	540	1593	BWR−5	8×8	50.3	pump + jet pump	2	Thyristor	Improved MARK-I	Hitachi	Under construction
Chugoku Electric Power	Shimane	1	460	1380	BWR−3	8×8	41	pump + jet pump	2	MG set	MARK-I	Hitachi	Mar 1974
		1	820	2436	BWR−5	8×8	50	pump + jet pump	2	MG set	Improved MARK-I	Hitachi	Feb 1989
Japan Atomic Power	Tsuruga-1		357	1064	BWR−2	8×8	41	pump	3	MG set	MARK-I	GE	Mar 1970
	Tokai Daini		1100	3293	BWR−5	8×8	50	pump + jet pump	2	flow control valve	MARK-II	GE/Hitachi/Shimizu	Nov 1978

Table 3 (continued) Specifications of NPPs in Japan (PWR & GCR)

PWR Plants

Utility	Unit name		Electric output (MW)	Thermal output (MW)	No. of loops	Fuel layout	Power density (kW/ℓ)	Steam generator		RCP flow rate (m³/h)	Containment vessel type	Main contractor	Commercial operation
								Model	Heating surface area (m²)				
Hokkaido Electric Power	Tomari	1	579	1650	2	14×14	95	51F	4,780	20,100	Steel double	MHI	Jun. 1989
		2	579	1650	2	14×14	95	51F	4,780	20,100	Steel double	MHI	Apr. 1991
Kansai Electric Power	Mihama	1	340	1031	2	14×14	71	CE	3,381	15,900	Steel semi-double	WH/MAPI	Nov. 1970
		2	500	1456	2	14×14	84	44	4,130	20,200	Steel semi-double	MAPI	Jul. 1972
		3	826	2440	3	15×15	92	51	4,785	20,100	Steel double	Mitsubishi Corp.	Dec. 1976
	Takahama	1	826	2440	3	15×15	92	51	4,785	20,100	Steel semi-double	WH/Mitsubishi Corp.	Nov. 1974
		2	826	2440	3	15×15	92	51	4,785	20,100	Steel semi-double	Mitsubishi Corp.	Nov. 1975
		3	870	2660	3	17×17	100	51F	4,780	20,100	Steel double	Mitsubishi Corp.	Jan. 1985
		4	870	2660	3	17×17	100	51F	4,780	20,100	Steel double	Mitsubishi Corp.	Jun. 1985
	Ohi	1	1175	3423	4	17×17	104	51A	4,785	20,100	Ice condenser	WH/Mitsubishi Corp.	Mar. 1979
		2	1175	3423	4	17×17	104	51A	4,785	20,100	Ice condenser	WH/Mitsubishi Corp.	Dec. 1979
		3	1180	3423	4	17×17	105	52FA	4,870	20,100	PCCV	Mitsubishi Corp.	Dec. 1991
		4	1180	3423	4	17×17	105	52FA	4,870	20,100	PCCV	Mitsubishi Corp.	Feb. 1993
Shikoku Electric Power	Ikata	1	566	1650	2	14×14	95	51	4,785	20,200	Steel semi-double	MHI	Sep. 1977
		2	566	1650	2	14×14	95	51M	4,780	20,200	Steel semi-double	MHI	Mar. 1982
		3	890	2660	3	17×17	100	52F	4,870	20,100	Steel semi-double	MHI	Under construction
Kyushu Electric Power	Genkai	1	559	1650	2	14×14	95	51	4,785	20,200	Steel semi-double	MHI	Oct. 1975
		2	559	1650	2	14×14	95	51M	4,780	20,200	Steel semi-double	MHI	Mar. 1981
		3	1180	3423	4	17×17	105	52FA	4,870	20,100	PCCV	MHI	Under construction
		4	1180	3423	4	17×17	105	52FA	4,870	20,100	PCCV	MHI	Under construction
	Sendai	1	890	2660	3	17×17	100	51M	4,780	20,100	Steel double	MHI	Jul. 1984
		2	890	2660	3	17×17	100	51F	4,780	20,100	Steel double	MHI	Nov. 1985
Japan Atomic Power	Tsuruga	2	1160	3423	4	17×17	105	51F	4,780	20,100	PCCV	MHI	Feb. 1987

GCR Plant

Utility	Unit name	Electric output (MW)	Thermal output (MW)	Type	No. of generators	Power density (kW/ℓ)	Kind of fuel	Heat exchanger		Main contractor	Commercial operation
								Type	Heating surface area (m²)		
Japan Atomic Power	Tokai	166	587	Calder Hall type	2	0.81	Metallic natural uranium	Vertical forced circulation double steam water tube type	121,980	GEC/SC	Jul. 1966

Table 4 Operating Records of Nuclear Power Plant (May 1996)

No.	Power Plant	Reactor Type	Gross Capacity (MWe)	Operating hours (h)	Availability Factor (%) May. 96	Availability Factor (%) Apr. 96	Availability Factor (%) May. 95	Generated Output (MWh)	Capacity Factor (%) May. 96	Capacity Factor (%) Apr. 95	Capacity Factor (%) May 95	Remarks
1	Tokai	GCR	166	744	100	100	100	104,640	84.7	86.7	84.3	
2	Tokai-Daini (II)	BWR	1100	744	100	100	0	818,400	100	100	0	
3	Tsuruga-1	BWR	357	744	100	100	100	265,608	100	100	99.4	
4	Tsuruga-2	PWR	1160	744	100	100	100	862,907	100	100	100	
5	Tomari-1	PWR	579	744	100	100	32.3	430,729	100	100	31.7	
6	Tomari-2	PWR	579	0	0	53.3	100	0	0	52.7	100	(1)
7	Onagawa-1	BWR	524	703	94.5	78.2	100	364,442	93.5	77.9	99.2	(2)
8	Onagawa-2	BWR	825	408	54.8	100	—	322,471	52.5	100	—	(3)
9	Fukushima I-1	BWR	460	744	100	68.6	0	342,240	100	66.3	0	
10	Fukushima I-2	BWR	784	744	100	100	100	583,296	100	100	100	
11	Fukushima I-3	BWR	784	744	100	89.6	100	583,296	100	85.5	100	
12	Fukushima I-4	BWR	784	0	0	66.7	100	0	0	66.1	100	(4)
13	Fukushima I-5	BWR	784	744	100	100	100	583,296	100	100	100	
14	Fukushima I-6	BWR	1100	519	69.8	100	0	549,200	67.1	100	0	(5)
15	Fukushima II-1	BWR	1100	0	0	70.0	100	0	0	68.8	100	(6)
16	Fukushima II-2	BWR	1100	744	100	100	100	818,400	100	100	100	
17	Fukushima II-3	BWR	1100	744	100	55.4	100	818,400	100	52.1	100	
18	Fukushima II-4	BWR	1100	504	67.7	100	64.5	540,900	66.1	99.4	63.9	(7)
19	Kashiwazaki Kariwa-1	BWR	1100	744	100	1.5	100	809,790	98.9	0.2	100	
20	Kashiwazaki Kariwa-2	BWR	1100	744	100	100	100	818,400	100	100	100	
21	Kashiwazaki Kariwa-3	BWR	1100	744	100	100	100	818,400	100	100	100	(8)
22	Kashiwazaki Kariwa-4	BWR	1100	192	25.8	100	87.5	206,240	25.2	99.9	84.8	
23	Kashiwazaki Kariwa-5	BWR	1100	744	100	100	25.8	818,400	100	100	25.1	
24	Hamaoka-1	BWR	540	744	100	100	74.8	401,758	100	100	72.7	
25	Hamaoka-2	BWR	840	744	100	31.2	100	624,956	100	28.7	100	(9)
26	Hamaoka-3	BWR	1100	603	81.0	87.6	62.0	651,301	79.6	87.2	56.5	
27	Hamaoka-4	BWR	1137	744	100	100	100	845,923	100	100	100	(10)
28	Shika-1	BWR	540	482	64.8	100	100	245,847	61.2	100	100	
29	Mihama-1	PWR	340	744	100	100	0	252,716	99.9	99.9	0	
30	Mihama-2	PWR	500	744	100	100	100	371,720	99.9	99.9	99.9	
31	Mihama-3	PWR	826	744	100	100	0	614,454	100	100	0	
32	Takahama-1	PWR	826	0	0	0	100	0	0	0	100	(11)
33	Takahama-2	PWR	826	744	100	100	100	614,458	100	100	100	
34	Takahama-3	PWR	870	0	0	0	100	0	0	0	100	(12)
35	Takahama-4	PWR	870	744	100	100	0	647,255	100	100	0	
36	Ohi-1	PWR	1175	706	94.9	100	100	823,585	94.2	100	85.0	
37	Ohi-2	PWR	1175	744	100	100	0	873,885	100	100	0	
38	Ohi-3	PWR	1180	744	100	100	100	877,800	100	100	100	
39	Ohi-4	PWR	1180	744	100	100	100	877,800	100	100	100	
40	Shimane-1	BWR	460	0	0	0	100	0	0	0	100	(13)
41	Shimane-2	BWR	820	627	84.3	100	0	492,231	80.7	100	0	(14)
42	Ikata-1	PWR	566	744	100	100	0	420,566	99.9	99.9	0	
43	Ikata-2	PWR	566	0	0	53.4	100	0	0	53.1	100	(15)
44	Ikata-3	PWR	890	744	100	56.3	100	662,118	100	43.2	100	
45	Genkai-1	PWR	559	744	100	100	100	415,746	100	100	100	
46	Genkai-2	PWR	559	181	24.3	0	100	44,143	10.6	0	100	(16)
47	Genkai-3	PWR	1180	0	0	40.1	100	0	0	39.7	100	(17)
48	Sendai-1	PWR	890	744	100	100	0	662,067	100	100	0	
49	Sendai-2	PWR	890	744	100	100	100	662,070	100	100	100	
	Commercial Reactor Total/Average (previous Month)		41191 (41191)	27,989	76.8	80.6	73.9	23,541,854	76.8	80.9	73.3	
50	Fugen	ATR	165	0	0	71.7	100	0	0	71.5	100	(18)
	Total/Average (previous Month)		41356 (41356)	27,989	75.2	80.5	74.4	23,541,854	76.5	80.9	73.4	

[Remarks]

(1)Shutdown due to periodic inspection. (16 Apr.~)
(2)Shutdown due to increase of pressure in containment vessel (24 Apr ~ 2. May)
(3)Shutdown due to interim inspection. (3~17. May ~)
(4)Shutdown due to periodic inspection (21 Apr ~)
(5)Shutdown due to interim inspection. (11~20 May ~)
(6)Shutdown due to periodic inspection. (22 Apr. ~)
(7)Shutdown due to interim inspection. (1~10. May ~)
(8)Shutdown due to periodic inspection (9 May ~)
(9)Shutdown due to interim inspection. (27. Apr ~ 6 Mar)
(10)Shutdown due to malfunction of shaft seal part in recirculation pump B (15~25 May)
(11)Shutdown due to periodic inspection (16. Jan. ~)
(12)Shutdown due to periodic inspection. (27. Mar. ~)
(13)Shutdown due to periodic inspection. (8 Feb. ~)
(14)Shutdown due to interim inspection (13~18 May ~)
(15)Shutdown due to periodic inspection. (17. Apr. ~)
(16)Shutdown due to periodic inspection (10 Mar ~) (Back to line on 24 May)
(17)Shutdown due to periodic inspection. (13 Apr ~)
(18)Shutdown due to periodic inspection (22 Apr ~)

Table 5 Cost of Construction and Electricity Generation (FY 1992)

Type	Construction		Generation (% of fuel cost)		
Nuclear	310,000	¥/kW	9	¥/kWh	(about 20 %)
Hydro	600,000		13		(-)
Oil	190,000		10		(about 60)
LNG	200,000		9		(about 50)
Coal	300,000		10		(about 30)

Remarks:

(1) Method of evaluation used is that of the OECD used.

(2) Conditions for evaluation:

Type	Power(MWe)	Unit/site	Durable years	Capacity factor
Nuclear	1,100	4	16	70
Hydro	10 - 40	-	40	45
Oil	600	4	15	70
LNG	600	4	15	70
Coal	600	4	15	70

(3) Costs for fuel cycle, decommissioning and waste treatment and disposal are included for the evaluation of NPP.

Table 6 Organizations Related to Nuclear Energy Usage

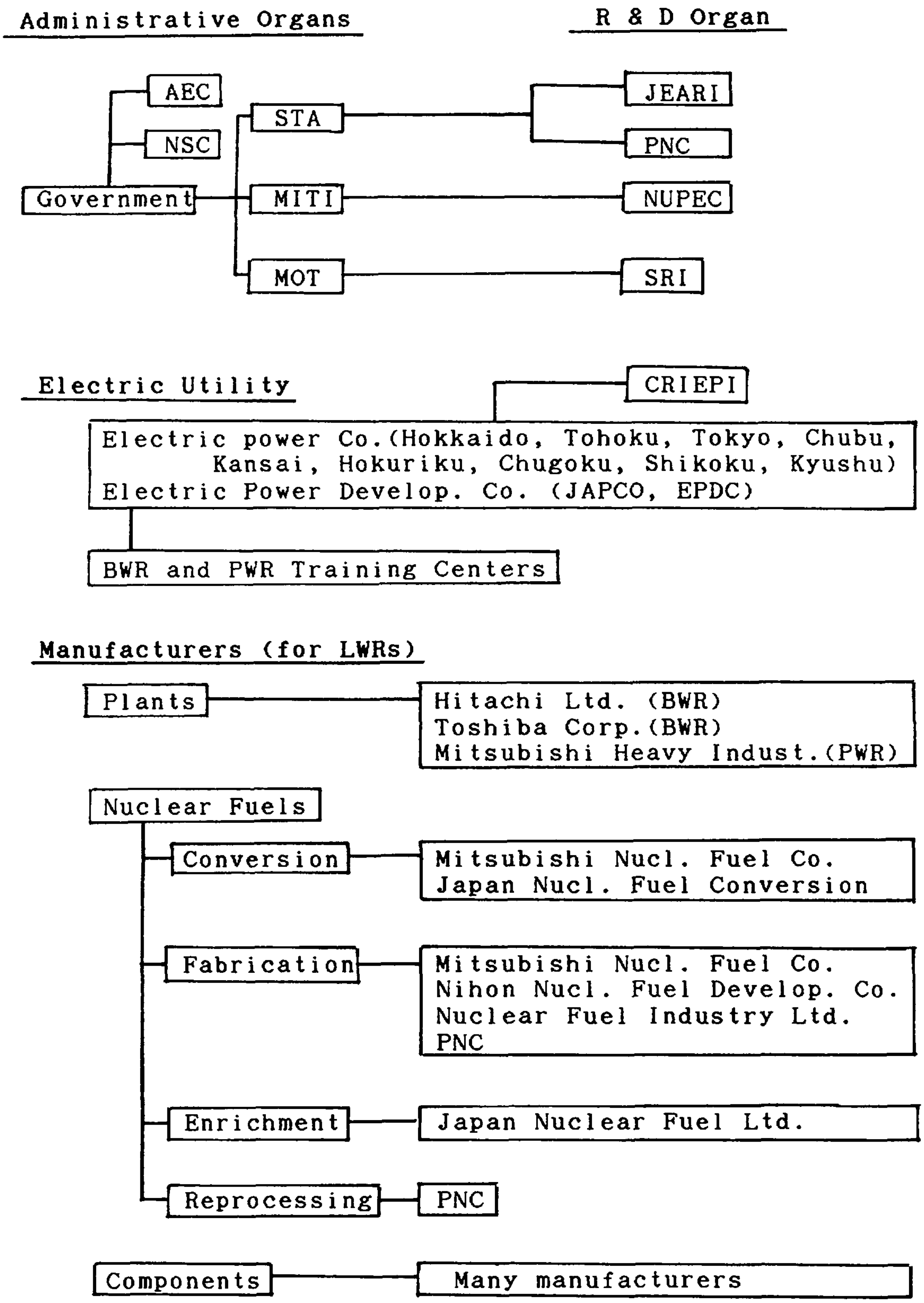


Table 7 Design Features of The SMRs in Japen

[LWR]

Model of Reac.	MRX	SPWR	JPSR
Design/ Supplier	JAERI	JAERI	JAERI
Reac. Type	Integral-PWR	Integral-PWR	Loop-PWR
Power: MWt	100	1,800	1,853
MWe	-	600	660
Coolant	L Wtr.	L Wtr.	L Wtr.
Out Temp.(C)	298	314	298
Press. (MPa)	12	13.5	12
Fuel/Clad.	UO_2Zry	UO_2/Zry	UO_2Zry
Application	Ship, etc.	Elect.	Elect.
Status (*)	BD	CD	CD

[LWR]

Model of Reac.	HSBWR	SBWR	BWR-5	MS-600
Design/ Supplier	Hitachi	GE, Toshiba Hitachi	GE, Toshiba Hitachi	Mitsubishi
Reac. Type	Loop-BWR	Loop-BWR	Loop-BWR	Loop-PWR
Power: MWt	1,800	2,000	1,600-3,300	1,820
MWe	600	600	540-1,100	630
Coolant	L Wtr.	L Wtr.	L Wtr.	L Wtr.
Out Temp.(C)	286	287	287	325
Press. (MPa)	7	7.2	7.2	15.5
Fuel/Clad.	UO_2/Zry	UO_2/Zry	UO_2/Zry	UO_2/Zry
Application	Elect.	Elect.	Elect.	Elect.
Status (*)	CD	DD	CN	BD

[HWR and LMRs]

Model of Reac.	SATR	MDP	4S	DFBR
Design/ Supplier	PNC	CRIEPI	CRIEPI	PNC
Reac. Type	Channel-ATR	Modular FBR	Int.-FBR	Loop-FBR
Power: MWt	500 x 2	840	125	-
MWe	330	325	50	0.01-0.04
Coolant	H Wtr.	Sodium	Sodium	Sodium
Out Temp.(C)	284	530	510	550
Press. (MPa)	6.9	-	-	-
Fuel/Clad.	MOX/Zry	U-Pu-Zr/ Oxide-disp. Ferritic Steel	U-Pu-Zr/SUS SUS	UN or MN/ Hastelloy
Application	Elect.	Elect.	Multi-pur.	Deep-sea
Status (*)	CD	CD	CD	CD

(*) Status
- CD: Conceptual design
- BD: Basic design
- DD: Detailed design
- CN: Construction

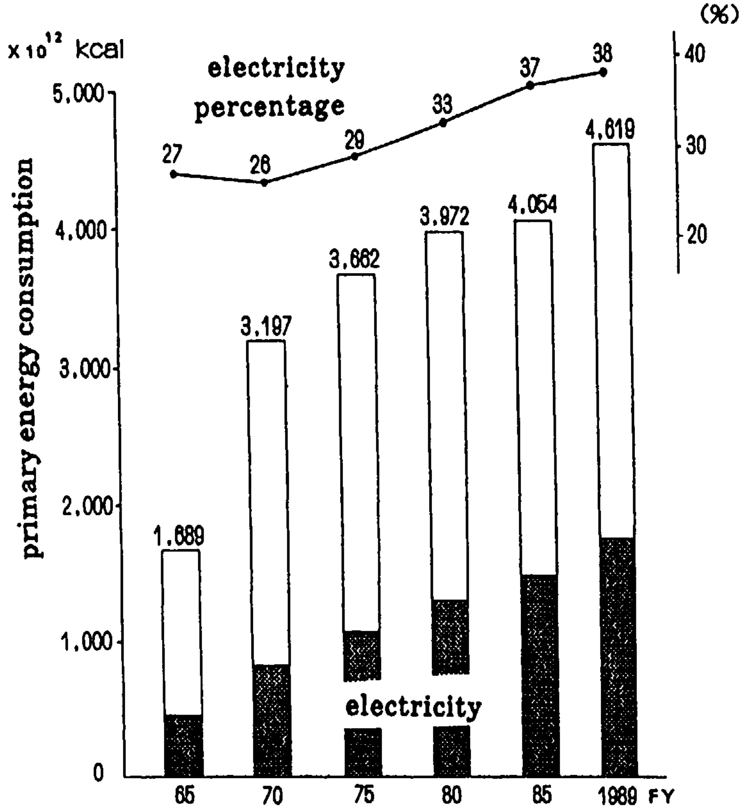


Fig. 1 Primary Energy Consumption for Electricity Generation

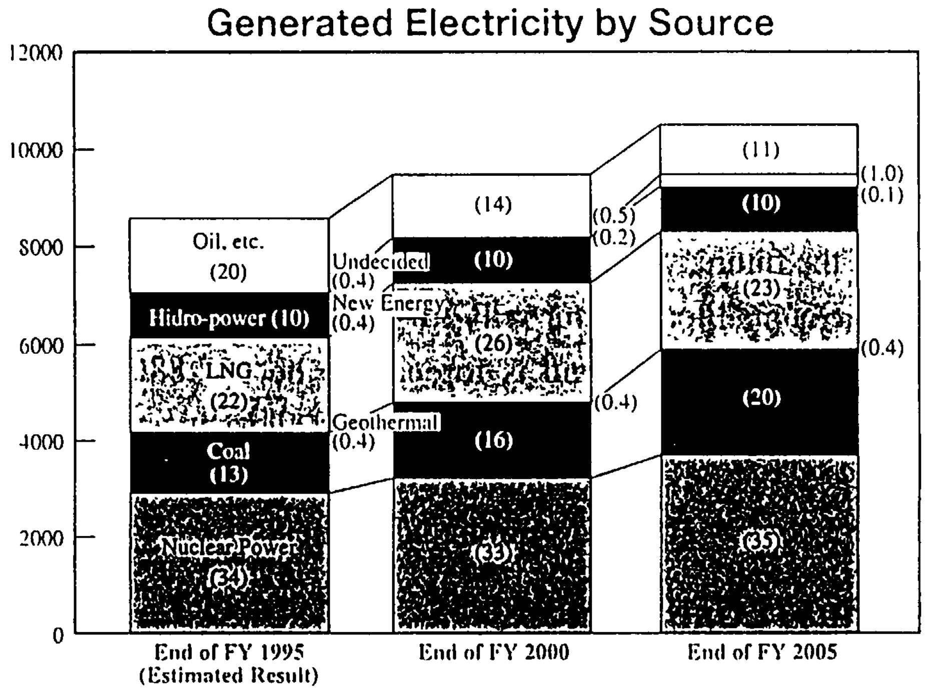
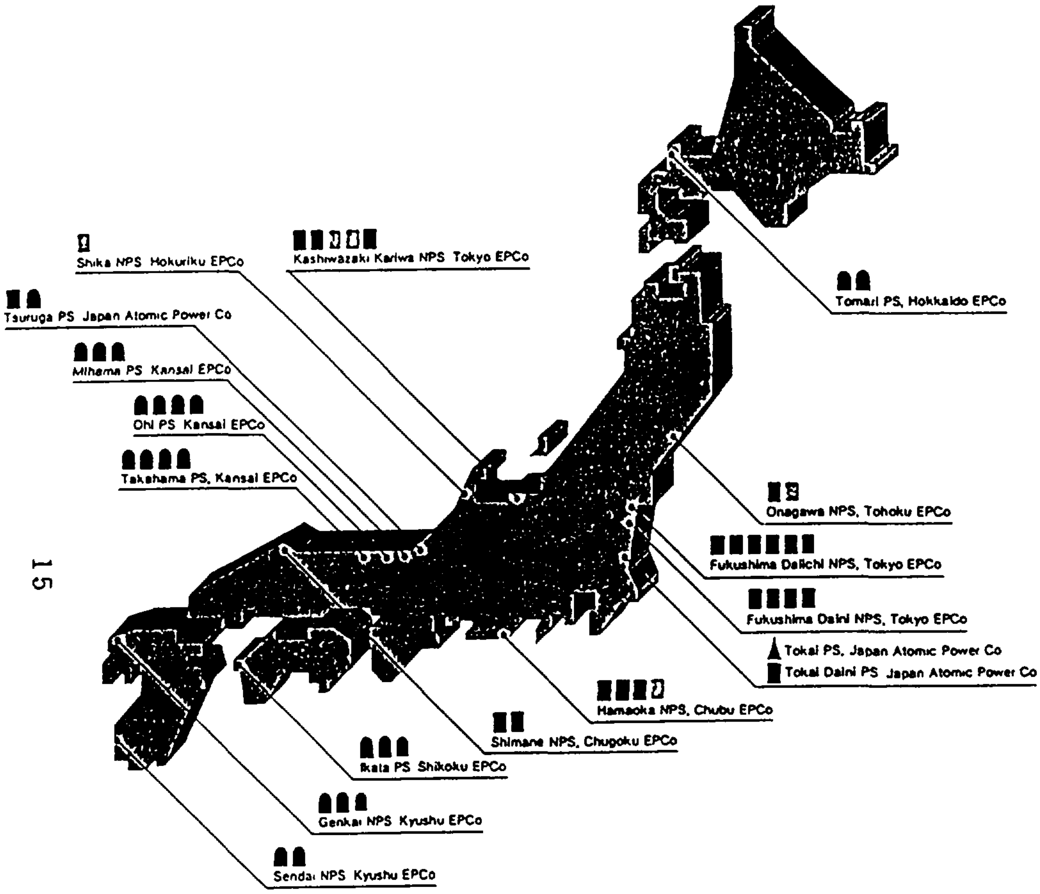


Fig. 2 Ratio of Electricity Generation by Energy Sources



BWR: ■ PWR: ● GCR: ▲

Fig. 3 Nuclear Power Plants in Japan
(In operation 1995)

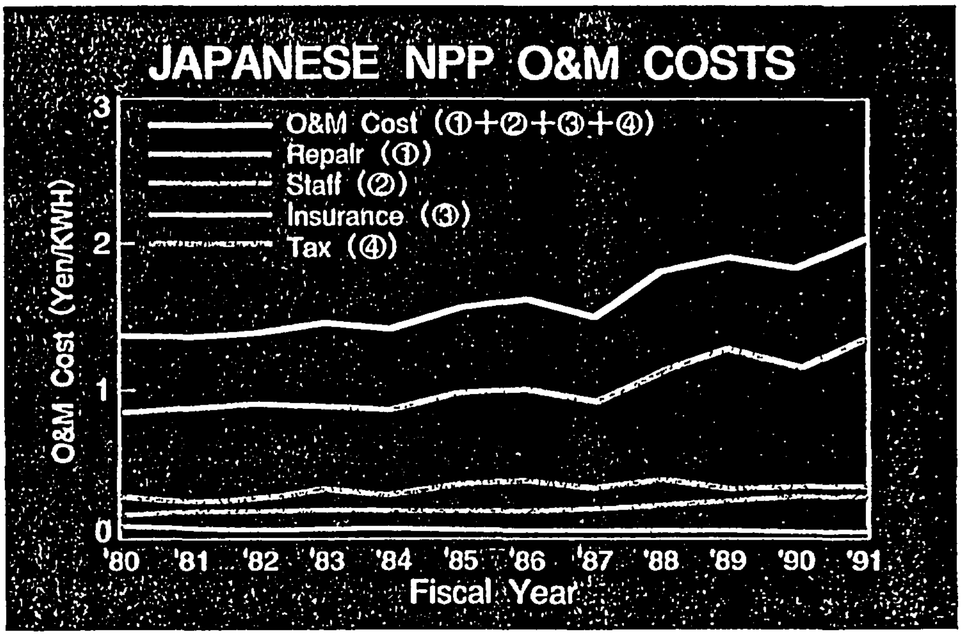


Fig. 4 Operation and Maintenance Cost of A NPP

15

5. Concluding Remarks

Steadily energy supply is requested in the future in Japan and the nuclear energy will be one of the important resources.

The large increase of electricity demand is expected and sites for the installation of NPPs is limited, therefore large size NPPs are interested in the electric utilities at present. However, design studies on SMRs are progressing both in the research organizations and in the manufacturers.

REFERENCES

[1] MITI; Draft Report on the Long-term Supply and Demand Outlook for Energy, 1994

[2] Atoms in Japan, June 1996

[3] ATOMIC ENERGY COMMISSION; White Paper on Atomic Energy, 1995

[4] UEBAYASHI, T.; "The Operation and Maintenance of Japanese Nuclear Power Plant - Today and Tomorrow", The 2nd International Conference on Nuclear Engineering (ICONE-2), San Francisco, U.S.A., March 1993

EXPERIENCE AND PROSPECTS OF DESALINATION IN MOROCCO

O. BOUCIF
Direction des Etudes et de l'Assainissement,
Rabat, Morocco

XA9846720

Abstract

At the current stage of studies of Guiding Schemes for Integrated Planning of hydrographic basins, and all the various schemes considered for the development of water resources and the transfer possibilities provided for, some hydrographic basins remain in a deficit situation in the prospect of 2020, especially Oum-Er-Rbia (240 Mm^3)[1], Tensift (60 Mm^3) and the Souss-Massa (140 Mm^3). Besides, it is worth mentioning that the drought periods recently experienced have also shown that the available hydraulic potential is very vulnerable to rainfall deficits, which could lead to deficits, well before 2020. In this respect, to preserve the future in the area of production and mobilization of water resources, it is more judicious to reconsider the planning of conventional water resources within the framework of a global vision that also integrates the use of unconventional water resources, especially desalination of sea water for Drinking Water Supply (DWS), mainly in the basins of Tensift, Souss-Massa and the South Atlas. Already, the future sites for the establishment of sea water desalination plants for DWS are to be located at the level of cities situated in the hydrographic basins, especially in the zones overlooking the Atlantic coast. It is the case mainly of Agadir, Tan-Tan and Essaouira. In 1977, a first desalination plant using the technique of mechanical vapor compression was set up in Boujdour. Its production capacity stood at 250 m^3/day. This plant was renovated in 1990 and operates normally. Seeking to face the demand of water in the Southern Provinces and to fill the water deficit amplified by the region's population and industrial development, ONEP strengthened drinking water production by setting up two RO (Reverse osmosis) plants in Boujdour (800 m^3/d) and Laayoune (7,000 m^3/d).

1. EXPERIENCE

Desalination constitutes the most widely used exploitation means of non-conventional resources in Morocco. ONEP and OCP (Office Chérifien des Phosphates) are the main operators in this respect. As for OCP, desalination was the means adopted for the production of water for industrial purposes, at the level of the Phos-Boucraa company in Laayoune, owing to the TDS (Total Dissolved Solids) sought of 25 ppm. The experience of ONEP is rather more diversified, considering the variety of techniques used.

[1] Million of cubic meters

1.1. Processes implemented by ONEP

As of 1973, the national guiding scheme of drinking water supply has shown the need for using the desalination of brackish water and sea water as a source of drinking water supply. In this paper, we will only deal with the experience of ONEP relating to sea water desalination.

In 1977, a first desalination plant using the technique of mechanical vapor compression was set up in Boujdour. Its production capacity stood at 250 m^3/day. This plant was renovated in 1990 and operates normally (See features in Appendix I Table II annexed).

1.2. Actions undertaken by ONEP

Seeking to face the demand of water in the Southern Provinces and to fill the water deficit amplified by the region's population and industrial development, ONEP strengthened drinking water production by setting up two RO (Reverse osmosis) plants in Boujdour and Laayoune(See description in Appendices II and III).

With the establishment of these two projects, the production of fresh water by ONEP will make it possible to bring the allocation per inhabitant (LPC, liter per capita) in the Saharan Provinces to a satisfactory level.

2. DESALINATION PROSPECTS IN MOROCCO

At the current stage of studies of Guiding Schemes for Integrated Planning of hydrographic basins, and all the various schemes considered for the development of water resources and the transfer possibilities provided for, some hydrographic basins remain in a deficit situation in the prospect of 2020, especially Oum-Er-Rbia (240 Mm^3), Tensift (60 Mm^3) and the Souss-Massa (140 Mm^3).

Besides, it is worth mentioning that the drought periods recently experienced have also shown that the available hydraulic potential is very vulnerable to rainfall deficits, which could lead to deficits, well before 2020.

In this respect, to preserve the future in the area of production and mobilization of water resources, it is more judicious to reconsider the planning of conventional water resources within the framework of a global vision that also integrates the use of unconventional water resources, especially desalination of sea water for Drinking Water Supply (DWS), mainly in the basins of Tensift, Souss-Massa and the South Atlas.

The future sites for the establishment of sea water desalination plants for DWS are to be located at the level of cities situated in the hydrographic basins, especially in the zones overlooking the Atlantic coast. It is the case mainly of Agadir, Tan-Tan and Essaouira.

2.1. The city of Agadir

The water resources currently mobilized or on the verge of mobilization, for DWS of Agadir will be saturated before 2020. After this date, other resources must be generated to cope with water demand. The transfer possibilities are not considered, owing to the fact that the neighboring basins are not in a surplus situation. Consequently, the only option with respect to mobilization of water resources consists in desalinating sea water.

The desalination plants will be dimensioned to produce a capacity of 45,000 m^3/day to meet the demand until 2030. After this date, the needs will be met through extension of such units. Owing to the necessary production capacity (relatively significant), the idea of using nuclear power might prove interesting and requires the carrying out of a specific thorough study before deciding on this variant, although the results of the regional study conducted by the IAEA show that the cost of 1 m^3 of desalinated water produced by using nuclear power is the same as that of the one produced by means of fossil energy.

It is worth mentioning that desalination for the strengthening of drinking water supply of this town may be envisaged before the year 2000, owing to the importance of the risk of lack of water during drought cycles.

2.2. The city of Tan-Tan

The water resources currently available for the city of Tan-Tan are expected to deplete by 2000. After that year, two variants for DWS are under identification and assessment within the framework of the Guiding Plan for Ziz, Guir, Rhis and Draâ, namely:

* DWS from the Guelta Zerga dam,
* Reinforcement of DWS from the Guelmim water expanse.

Owing to the vulnerability of volumes in the Southern dams caused by pluviometry deficits (arid region), as well as to the fact that the potential of the Guelmim expanse is not controlled at the present moment, in addition to reasons of DWS security for the city of Tan-Tan, another variant should be taken into account, namely the desalination of sea water.

This solution would make it possible to preserve the summonable potential at the level of the expanse for DWS of Guelmim. With this option, the desalination plant will be designed for a production capacity of 7,000 m^3/day to meet the needs up until 2020. After this date, the needs will be met through the extension of the desalination plants. The possibility of sea water desalination for the DWS of the city of Tan-Tan through the use of thermal power resulting from a solar pond and/or the energy of exhaust gas from the turbines of the ONE (Office National de l'Electricité) station, located at the port of Tan-Tan, is under study.

2.3. The city of Essaouira

The DWS of the city of Essaouira is ensured exclusively from the underground sources. Following the persistence of drought which prevails in the region, a significant decrease was registered at the level of resources. In the mid-term, the DWS of Essaouira is considered from the underground resources of the Meskala or Ounara region in order to meet the needs until 2006. After that year, the DWS of the city will be ensured from the waters of the future Zerrar dam.

Considering the fragility of underground water resources and for reasons of securing the drinking water supply of Essaouira, it would be timely to consider an DWS option through sea water desalination. The production plants will be designed for a global production capacity of 13,000 m^3/day to meet the needs until 2020.

2.4. Overview

The following table reviews the data relating to the desalination sites examined in the previous paragraphs.

TABLE I. OVERVIEW OF DATA RELATING TO THE POTENTIAL SITES FOR SEA WATER DESALINATION

CITY	EXISTING		FUTURE		NOTES
	Capacity (m^3/day)	**Saturation**	**Capacity (m^3/day)**	**Saturation**	
Laayoune	7,000	2000	14,000	2020	Extension of present plants to enable the coverage of mid- and long-term water needs of the city.
Boujdour	800	2020	---	---	The present plant will meet the city's long-term water needs.
Agadir			45,000	2030	Desalination may be envisaged for the year 2000, owing to the importance of the risk of lack of water during drought cycles.
Tan-Tan			7,000	2020	Desalination of sea water can be considered as of 2000.
Essaouira			13,000	2020	Resort to desalination might be considered as of 2006.

APPENDIX I. TABLE II. SEA WATER DISTILLATION PLANT OF BOUJDOUR
CAPACITY: 250 m^3/d

Population	3,000 inhabitants
Capacity	250 m^3/d
User	100% for human consumption
Desalination process	Mechanical vapor compression (MED/VCM, Multi-effect distillation/mechanical vapor compression) 1 effect
Total capital cost	Dhs (Dirhams) 7,000,000 all tax inclusive (1977)
Commissioned in	1977
Operating cost: * Energy * Chemicals * Manpower	Dhs 18.723/m^3 Dhs 16.875/m^3 Dhs 0.128/m^3 Dhs 1.72/ m^3
Energy consumption (kW (e).h/m^3)	27
Total water cost	Dhs 50/m^3 (The investment required by the rehabilitation of the plant is included in this price).
Chemicals dosage: * Sodium chlorite * Sodium bicarbonate * Sodium hexametaphosphate	 15.33 g/m^3 18.33 g/m^3 13 g/m^3

APPENDIX II. TABLE III. SEA WATER REVERSE OSMOSIS PLANT OF BOUJDOUR.CAPACITY: 800 m^3/d

Population	15,000 inhabitants
Capacity	800 m^3/d
User	100% for human consumption
Desalination process	Reverse osmosis (R.O.)
Total capital cost	Dhs 60,000,000 all tax inclusive (1992)
Commissioned in	November 95
Operating cost: * Energy * Chemicals * Manpower * Spare parts	Dhs 5.86/m^3 Dhs 4.13/m^3 Dhs 0.20/m^3 Dhs 0.73/m^3 Dhs 0.80/m^3
Specific energy consumption	5.11 kW(e)h/m^3
Total water cost	Dhs 42/m^3
Chemicals dosage: * Chlorine (Cl_2) * Coagulant ($FeCl_3$) * Sulfuric acid (H_2SO_4) * Sequestering agent (flocon 100) * Dechlorination ($NaHSO_3$) * Correction of pH	 7.5 g/m^3 25 g/m^3 49 g/m^3 10 g/m^3 10 g/m^3 20 g/m^3
Conversion factor	40 %
Salt content in water product	500 ppm

APPENDIX III. TABLE IV. SEA WATER REVERSE OSMOSIS PLANT OF LAAYOUNE. CAPACITY: 7,000 m^3/d

Population	132,000 inhabitants
Capacity	7000 m^3/d
User	100% for human consumption
Desalination process	Reverse osmosis (R.O.)
Total capital cost	Dhs 223,000,000 all tax inclusive (1992)
Commissioned in	November 95
Operating cost: * Energy * Chemicals * Manpower * Spare parts	Dhs 8.85/m^3 Dhs 6.07/m^3 Dhs 1.64/m^3 Dhs 0.72/m^3 Dhs 0.42/m^3
Specific energy consumption (kW(e)h/m^3)	5.07
Total water cost	Dhs 21/m^3
Chemicals dosage: * Chlorine (Cl_2) * Coagulant ($FeCl_3$) * Sulfuric acid (H_2SO_4) * Sequestering agent (flocon 100) * Dechlorination ($NaHSO_3$) * Correction of pH	 2.5 g/m^3 10 g/m^3 26.5 g/m^3 6.5 g/m^3 3 g/m^3 24 g/m^3
Conversion factor	45 %
Salt content in water product	500 ppm

PART IV

DESIGN DESCRIPTIONS

CAREM PROJECT: 1995 STATUS OF ENGINEERING AND DEVELOPMENT

H.J. BOADO, J.P. ORDOÑEZ
Carem Project, INVAP
Bariloche

A. HEY
CNEA, Buenos Aires

Argentina

Abstract

The CAREM Project is a low power NPP of 25 MWe, with an integrated self pressurized primary system The cooling of the primary system is of the natural circulation type, featuring several passive safety systems The project is owned by Argentina's CNEA (Comisión Nacional de Energía Atomica) and its associated company, INVAP, which in turn is its main contractor

In this work the present status of the CAREM Project is presented The possible evolution of the CAREM project is mentioned in relation with a new containment design Brief descriptions of the Experimental Facilities, listed below, already in operation or under construction are included

- CAPCN High Pressure Loop A natural convection loop to verify dynamic response and critical heat flux
- RA-8 Critical Facility, designed and constructed for the CAREM Project (that can be used as a general use facility)
- RPV Internals The whole assembly of absorbent rods, connecting rods and the rod guides is being constructed on a 1 1 scale The aims of this experimental facility are vibration analysis and manufacturing parameter definitions
- Control Drive Mechanisms A series of verifications and tests are being carried out on these hydraulically driven mechanisms
- Other development activities are mentioned in relation with the Thermohydraulics, Steam Generators and Control

1 CAREM PROJECT PRESENT STATUS

1.1. Introduction

The CAREM, a small NPP Project owned by the CNEA [/1/] is being developed jointly by CNEA and INVAP [/2/] The concept was born in the 80's when the idea was presented in LIMA, Perú, during an IAEA conference back in 1984 At that time, the Argentinean nuclear experience was several Research Reactor (RR) built (RA-0, RA-1, RA-2, RA-3, RA-6, RP-0) several projects on RR, an enrichment plant and the experience gained in the follow-up of two NPP in operation (CNA-1 and CNE) It was thought that the next step to reach nuclear maturity should be to work on the design and construction of an NPP This development in a medium size economy and restricted financial resources should carry limited risk and so the focus was set on a small NPP In addition the NPP should be able to operate in isolated cities, a common situation in Argentina This means reduced ability to obtain help in operational incidents/accidents by the operating personnel, reduced industrial/ technical resources on site, long distances from fully developed areas, reduced roads and transportation capacities And being the first domestic NPP, it was also considered convenient to work on an electrical plant with a low output Similar conditions can be found in a good number of countries

The above mentioned criteria lead to the CAREM technical characteristics i e a LWR with Integrated Primary System and extensive use of the so called passive systems (e g natural circulation cooling)

The power was initially fixed at 15 MWe, a R&D program was set up and the construction of a thermohydraulic lab and a critical facility was undertaken

At present the ongoing work is for 25 MWe but a version of 100 MWe is also foreseen

1.2. Organization

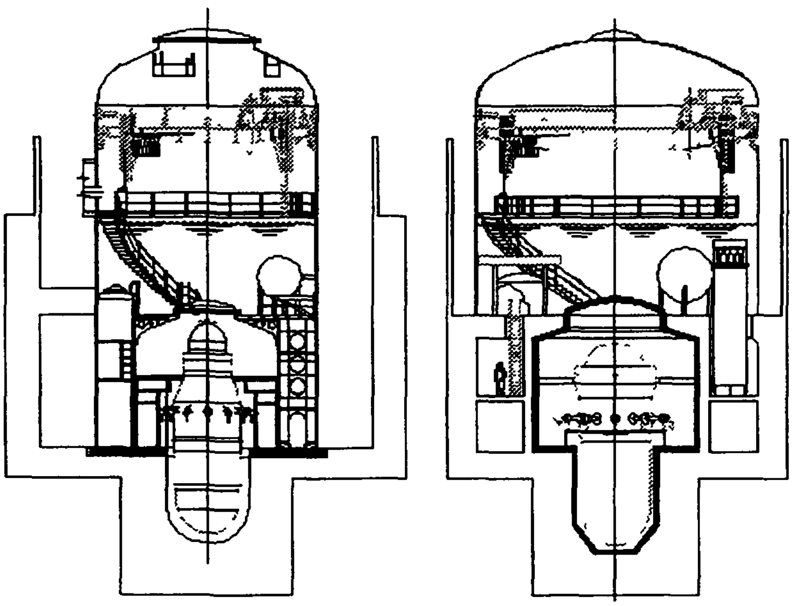
CNEA has subcontracted the engineering of the CAREM to INVAP However CNEA itself is working on the development of the Fuel Elements and on part of the nuclear instrumentation The follow-up activities on the CAREM Project are being carried out by CNEA with a Coordination Group which receives all the engineering from INVAP Afterwards the engineering enters into a revision process inside CNEA under groups specially appointed by the CAREM Coordinator Every three months the Coordination Group has a meeting with its counterpart in INVAP where technical discussions are held Visits to and revisions of the experimental work are conducted For some special topics such as RPV design and manufacturing, foreign experts are invited to assist and make special revisions Experts from other companies are also sometimes required

1.3. Engineering Stage

The CAREM Project is undergoing the development of the Detail Engineering needed to start its construction A technical revision was made by the owner of the project (CNEA) in 1992 and a cost estimation following TRS 269 was finished in 1993

During the present year the main tasks have been devoted to work on subjects in which we lacked experience and in which research and development were needed such as RPV, Containment, Internals, Thermohydraulics of different Systems (RPV, Secondary, Primary, Containment, etc) and lay-out These in-depth studies encouraged the Project Personnel to seek different solutions for the containment and RPV From left to right in the following picture we have the original design and the one on which studies are being conducted at present are showed The containment of the original design is totally encompassed in steel, the RPV is conical However, the solution proposed for the fixing of the containment was difficult from the point of view of civil engineering The second design, with its lower part made from concrete, features larger available space This solution increases the maintainability of the NPP, (mainly the SG are more accessible for repairs), improving one of the weaker points in the original design The increased available space allows also the change of the SG feeding pipes from vertical to horizontal type, which in turn permits an easier manufacturing solution

In the following sections brief descriptions of the activities in progress for the CAREM Project, except for those related with the SG, are given Regarding the SG, a Special Program is being conducted This Program has called for the construction of a mini SG to be tested in a high pressure loop Studies are under way to define the whole scope of data that should be obtained from a 1 1 model These studies include the qualification process and the search of a facility for testing the SG



2 CAPCN HIGH PRESSURE LOOP

2.1. Description the Loop

The High Pressure Natural Convection Loop is a part of the Thermal Hydraulics Laboratory designed, constructed and operated by INVAP for the CNEA Its purposes are to verify the thermal hydraulic engineering of CAREM NPP mainly in two areas dynamic response and critical heat flux These verifications are accomplished by the validation of the calculation procedures and of the codes for the rig working in states corresponding (by similarity criteria) to the operating states of the CAREM reactor

The CAPCN resembles CAREM in the primary loop, while the secondary loop is designed only to produce adequate boundary conditions for the steam generator Operational parameters are reproduced approximately for intensive magnitudes (Pressures, temperatures, void fractions, heat flux, etc) and scaled for extensive magnitudes (flow, heating power, size, etc) Height was kept on a 1 1 scale

The CAPCN was constructed according to ASME, with the following parameters 150 bar, 340°C for the primary and 60 bar 340°C for the secondary The level difference between center points of core and steam generator is ~ 5 7 m The primary loop can operate in saturated regime (self-pressurized), or subcooled (dome pressure increased by nitrogen injection), with heating power up to 300 kW and different hydraulic resistance

CAPCN states corresponding to full power of CAREM

PRIMARY		SECONDARY	
Pressure	122 5 bar	Steam pressure	47 bar
Hot leg temperature	326 °C (saturation)	Steam temperature	290 °C (superheated)
Cold leg temperature	280 °C	Cold leg temperature	200 °C
Natural circulation flow	1 08 kg/sec	Flow	0 128 kg/sec
Heating power	263 kW		
Riser quality	~ 1 %		
Heating control	feedback loop of pressure + core dynamic		

The nuclear core is reproduced by electric heaters operated by a feedback control loop on dome pressure For the dynamic tests the heaters are 1 2 meters long

The heater bundle that will be used for the CHF tests differs from the one used for the dynamic ones, in order to have a configuration that allows heat fluxes high enough to ensure the departure from nucleate boiling for the complete range of pressures, flows and subcoolings The heated length is 400 mm The seven rod bundle features two rods which allow an overpower of 20% These rods have six thermocouples each in order to ensure the measurement of the exact location of CHF The rest of the rods have only two thermocouples each that guarantee CHF detection should it occur on one of these rods All the thermocouples are located near the upper end of the heated zone because the axial power distribution is uniform in the CHF heaters bundle

The group of states foreseen for these tests include the following range of main parameters

	Upper value	Lower value
Mass flow [kg/m^{2}s]	750	200
Pressure [bar]	130	115
Heaters outlet quality [%]	5	- 30

The natural circulation flow can be regulated either by a valve in the cold leg or a by pass to the bottom of the riser A gamma densitometer is available for void fraction measurements The heat exchanger (SG) is of coiled, once through type For the CHF tests the steam generator is useful only as a cold source, so the secondary loop operating parameters are not relevant

The secondary loop pressures and cold leg temperatures are controlled through valves The pump regulates the flow The condenser is an air cooled type with flow control

Both loops allow automatic control and can be pressurized by nitrogen injection

The combination of primary and secondary states is limited only by conditions attainable with the heat transfer capacity of the heat exchanger As a consequence the loop will permit the validation of the calculational tools used on the CAREM Project (RETRAN and ESCAREM) in those conditions

The inclusion of a SG (15 tubes) with a design similar to that of the CAREM will allow a full 1-D thermohydraulics analogy, allowing the extrapolation of results directly to the CAREM-25

The thermal-hydraulic design of CAREM reactor core was carried out using a version of 3-D, two fluid model THERMIT code In order to take into account the strong coupling of the thermal-hydraulics and neutronics of the core, THERMIT was linked with an improved version of CITATION code (developed in INVAP, and called CITVAP) This coupled model allows the drawing of a 3-D map of power and thermal-hydraulic parameters at any stage of the burnup cycle

The thermal limits were calculated using the 1986 AECL-UO Critical Heat Flux Lookup Table, validated with all the available measurements in the operational range in order to ensure a 95/95 reliability/confidence in the thermal limit The CAPCN CHF test results will be used to improve these calculations by increasing the experimental data in the operating range and fuel element geometry of the CAREM core

2.2. Experimental Program

The Program is divided in two main subjects

Dynamic Tests [/3/]

In order to reduce the range of experiments it was defined that the studies would be limited to the regimes, states and perturbations expected for the CAREM 25 These studies would be limited to those parameters included in the modeling by computer codes

Several boundary conditions were established

- The existence of secondary subcooled and saturated states depends on the method adopted for the start up These situation can be avoided if vapor is available
  - The primary subcooled state is relevant only during the start up of the CAREM
  - Nitrogen in the dome was limited up to 30 Kg/cm2
- Possible perturbations in the Primary Side are produced by neutronic changes and power extraction from the secondary side

Parameters of interest are thermal balances, water flow, thermal transference coefficients in SG, dome, and void fractions in the riser, primary and secondary pressure and temperature, dome volume, hydraulic drag, neutron kinetics

This stage of experiments is under way at present and will be finished by March next year

CHF Tests /4/ /5/ /6/

It is known that to perform reliable thermohydraulic calculations it is necessary to perform CHF experiments for the start up and nominal power states The CAPCN will be used, with some modifications in the control system, in order to obtain stable values for the pressure, water flow, and water quality, with a positive power ramp Part of the experimental work to be started next year will be devoted to determine limits and operational conditions for the CHF experiments A detailed study has already been conducted with the corresponding Program For the CHF experiments themselves it was necessary to make careful studies because it was not possible to install a section of the same dimensions as for CAREM Fuel Element Variables kept the same as in CAREM are rod diameter, pitch and ratio of total section to water flow section Preliminary experiments for temperature oscillations due to slug flow patterns will be made before starting the CHF experiments

2.3. Start-up and experimental results

The start-up program was carried out during the Spring of 1993 The accomplished phases are

- Hydraulics characterization, cold state
- Primary isolated self pressurization test, control loops calibration Degasification of primary side
- Whole system operation Secondary side in liquid phase Initial calibration of condensed tank control loops in temperature and flow Thermal balance
- Whole system operation Operating regime at low power with overheated vapor and automatic control

To date the following results have been obtained

- Self pressurization in the operating range of CAREM-25 with natural convection was confirmed
- The Thermohydraulic Process was controlled during the different power conditions with no major problems
- The production of over heated vapor is compatible with the CAREM operating conditions

Considering the loop as an experimental machine the following experience has been obtained

- The Data Acquisition and Control System has fulfilled its required performance However some problems related with hardware characteristics arose temperatures were measured with an accuracy lower than required An interface between the DACS and the sensors was frequently in need of recalibration And finally during a test conducted to improve Power control and due to a human error the electrical heaters were damaged They were replaced and a safety controller was added to the Supervision System During the repair time of the electrical heaters some improvements were made a new fuel assembly separator (CAREM design) was installed in order to conduct a durability test of this component, new measurements in the heat exchanger and new improvements in the valves used for control Some modifications in the condenser design were also needed to reach the full power state (300 kW of electrical heat) In order to reduce experimental errors several data will also be recorded through a new interface, with lower error and frequency of calibration needs Improvements in the thermal isolation were made during the repair time

3 RA-8 CRITICAL FACILITY

3.1 Description

The RA-8 critical assembly has been designed and constructed as an experimental facility to measure neutronic parameters of the CAREM NPP It can be used, with relatively minor changes, as a facility to perform experiments for other light water reactors It provides a reactor shielding block and reactor tanks that can be adapted to hold custom designed reactor cores

The RA-8 critical facility is located in the PILCA IV Sector of the PILCANIYEU INDUSTRIAL COMPLEX, in the Province of Rio Negro, Argentina, approximately 30 km East of San Carlos de Bariloche It occupies the main hall of a building shared with the Laboratory for Thermohydraulic Tests (LET), described in Section 2 of this report, and other special facilities for CAREM Project Geometry and location of core shielding inside the main hall are such that radiation dose levels are acceptable in adjacent rooms, for all operational conditions

General Characteristics of the RA-8.

Design features are

- Low operating power, which makes cooling systems unnecessary
- Extinction systems Rapid insertion of control rods
  Dumping of the moderator
- Regulation and safety rods There are at present 13 mechanisms to drive control rods in and out of the core The Control System allows the definition and use of some of the control rods as Regulation Rods, and some as Safety Rods The number of rods assigned to each function depends on the specific core being tested

The Argentine Regulatory Authority imposes the following requirements for the design of control rods for critical facilities

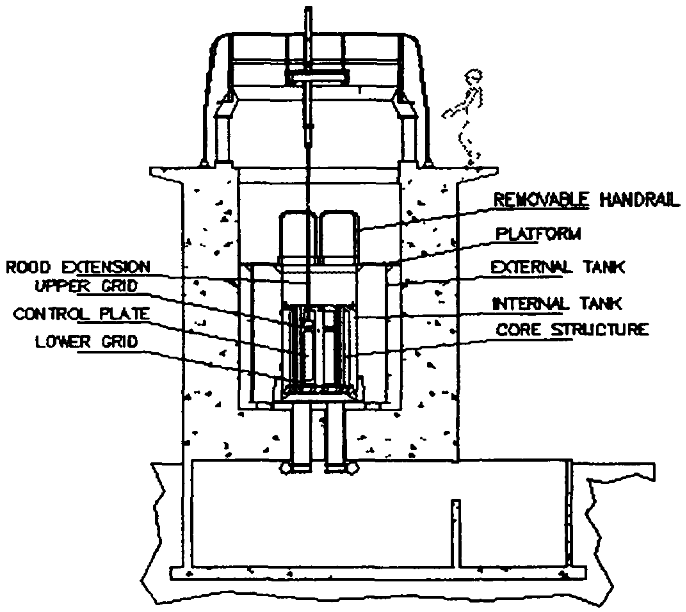
- Negative reactivity introduced by control rods must be higher than 50% of the critical assembly reactivity excess
- Core reactivity with control rods must be negative and higher than 3000 pcm
- Core must remain subcritical in at least 500 pcm after extraction of the control rod of maximum negative reactivity
- Reactivity value of control rods defined as regulation rods must be such that their insertion makes the core remain subcritical in at least 500 pcm
- Movement of any control rod mechanism must not produce a reactivity insertion higher than 20 pcm/sec

There are two possible ways of operation

- Operation by critical height Reactivity is determined by the moderator level surrounding the core
- Operation with control rods Reactivity is regulated by the amount of absorbing material introduced in the core

Water level is controlled by the Control System During operation water simultaneously fills two concentric and connected tanks The inner tank is designed to hold the core, its structural components, and nuclear instrumentation The water in the outer tank serves the purposes of shielding and reflecting The tanks are filled in two stages a first stage of fast pumping, followed by a second stage of slow pumping to approach operating level Safety Logic takes into account the position of safety and control rods to allow pumping of the moderator into the reactor tanks The tanks are emptied by means of two butterfly valves, located in the center of the inner tank, which drains water into the storage cistern, below the reactor block It takes no more than 4 seconds to empty the inner tank

The water system also features a provision for boron to be added and for cleaning, draining and recirculating water Water temperature can be varied up to approximately 75 °C



Data acquisition is made simultaneously and independently by two means the "hard logic" Instrumentation and Control system needed to operate the facility, and the Control and Data Acquisition System, a microprocessor based system, by means of which the reactor operator is informed of reactor and experiment related parameters

At present the facility is being finished and the cold initial start-up is programmed for the end of the present year (1995) The fuel elements are in the process of being manufactured and are expected to be finished in the first half of next year (1996) The experimental program is expected to last for about one year and a half

3.2. Experimental Program

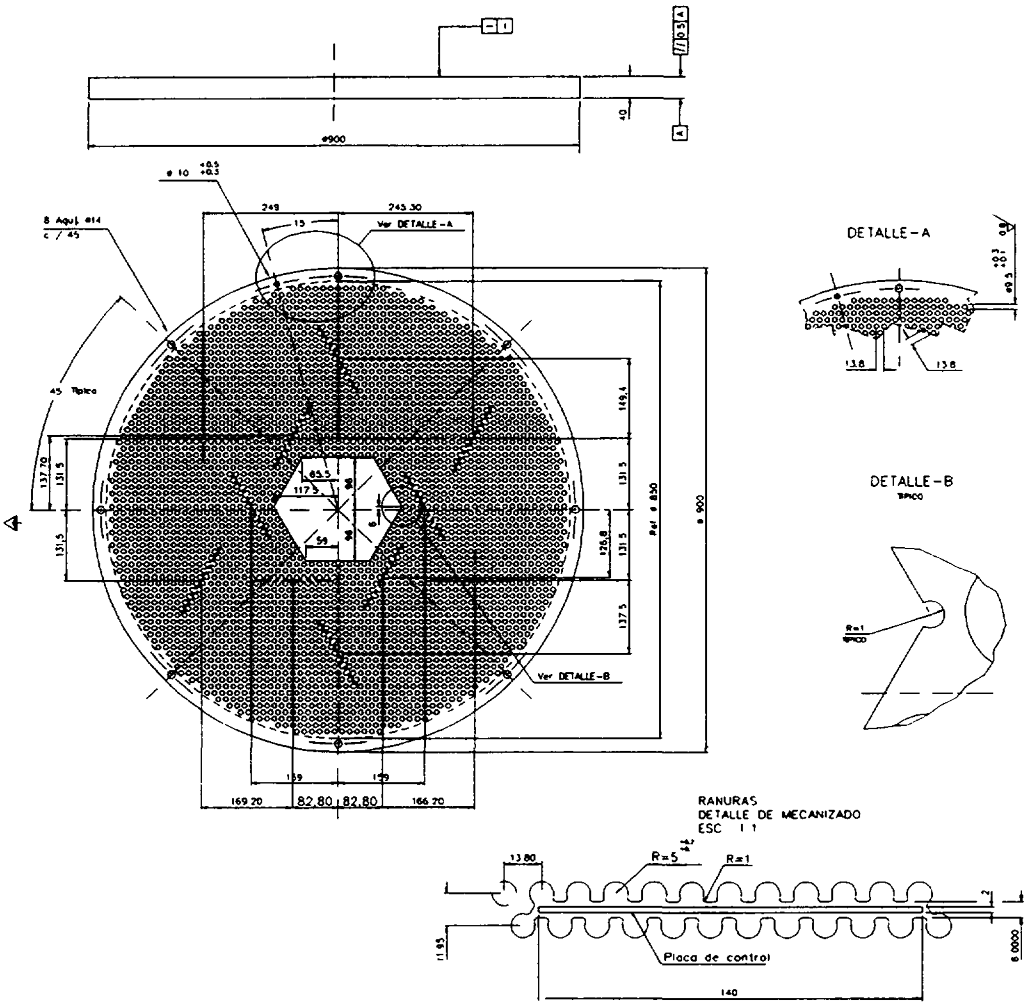
The cores to be used for CAREM related experiments are made from fuel rods with the same radial geometry as the ones for CAREM, but shorter, (80 cm) The pitch of the core was studied by calculation and is the same as for the CAREM The core calculations are made with a Diffusion Code (CITVAP), so the size of the core has to be such as to have good results A central homogeneous zone is needed to study perturbations such as rods loaded with different concentrations of burnable poisons, absorbing rods, guide tubes, structural materials, etc The maximum core reactivity covering all the experiments is around 7500 pcm To meet the requirements of the Regulatory Authority and to comply with the experimental conditions, plate type absorbers made from bare Ag-In-Cd are used The distribution of the absorber elements is given in the following figure The central absorber is not shown

Studies will be conducted on different cores using two enrichments (E= 1 8 % and E= 3 4 %) Some of the defined cores are

A One region of E= 1 8 %
B Two regions inner of E= 3 4%, outer of E= 1 8%
C One region of E= 1 8% , perturbed with non fuel rods (guide tubes, control rods, burnable poisons) homogeneously distributed in the core
D Two regions, the inner with E= 1 8% and perturbed with non fuel rods, the outer region with E= 3 4%
E Two regions, the inner with fuel rods resembling CAREM fuel elements, the outer with the fuel rods necessary to reach enough criticality to perform experiments with different configurations for the CAREM FE

A detailed experimental program defines the experiments to be conducted with each core type Some of the already defined measurements to be carried out are

- Influence of different Boron concentrations at different temperatures
- Extensive properties such as critical height, critical buckling and reflector saving
- Intensive properties such as disadvantage factors, fission ratio (U235 & U 238), epithermal to thermal fission ratio, epithermal to thermal absorption ratio, average spectra in fuel and moderator
- Fluxes and spectra in non fuel rods (control rod guides)
- Fluxes and spectra in macrocells (assembly of non fuel rods with neighboring fuel rods)
- Power distribution
- Mutual influence in CAREM Control Rod positioning
- Reactivity changes for different Boron concentrations, different temperatures and different void fractions
- Determination of control rod reactivity
- Determination of fuel rod
- Reactivity with different concentrations of burnable poisons



NOTAS

1) Cantidad total de agujeros diametro 9 5 $^{+0.3}_{+0.1}$ 3500 (tres mil quinientos) sobre ø850

4 NEUTRONIC CODE VALIDATIONS

4.1 Nuclear Data

The ESIN library originated from the WIMS (1976) updated with data for Ag, In, Cd and Gd from ENDF/B-4 and Nb from WIMCAL-88 (COREA) The WIMS has been in use by INVAP with validated good results in Plate Type FE (for RR) of 20% enrichment and fresh cores The Gd isotopes were tested using the CONDOR in two numerical benchmarks with good results, however the results given by participants vary widely

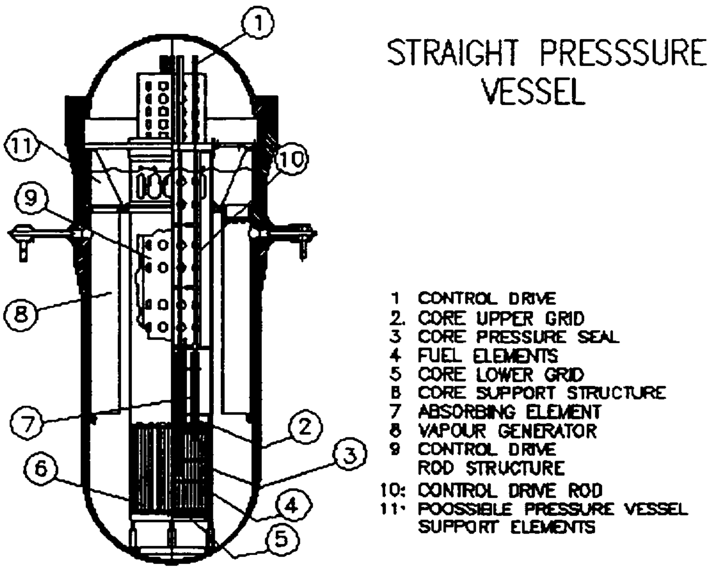
4.2 Cell Code CONDOR 1.3 [9]

A preliminary validation with 91 typical PWR cells (fresh) was made [10, 11], with a difference of 400±800 pcm Two more benchmarks [12] [13] with mini FE of PWR using burnable poisons were made with satisfactory results Recently more comparisons (49 cases) were made with results obtained for cores similar to that of the CAREM [14, 15]

4.3 Core Code CITVAP.

The code was validated for plate type fuel elements of 20% enrichment with good results For 90% enriched FE the results are not as good Further validations to the calculation line are under way in order to reduce experimental work with the RA-8

5 REACTOR PRESSURE VESSEL INTERNALS



At this stage of the CAREM Project, it has become apparent that several design aspects in the internals require experimental verifications The aim of these experiments is to verify their behavior under normal and abnormal conditions and to define manufacturing and assembling allowances as well as handling procedures and auxiliary tools Following is a brief description of the arrays under construction and the foreseen experiments plans for each of them

A dummy of a sector of the core containing the following items:

Core support, three FE, upper structures with control rod guides The experiments will be carried out with water at room temperature The aim is to make fine adjustments in the design and manufacturing and to determine the influence of the different variables on the behavior of the assembly Also to verify the design of couplings and auxiliary tools This stage will be started by the end of the present year

A 1:1 length of a Sector of the Control Rod Drive Structure (for one Control Rod)

The Connecting Rod attached to the dummy core mentioned above and a Drive Mechanism The experiments will be carried out with air and water at room temperature The objectives are to obtain manufacturing and operational allowances

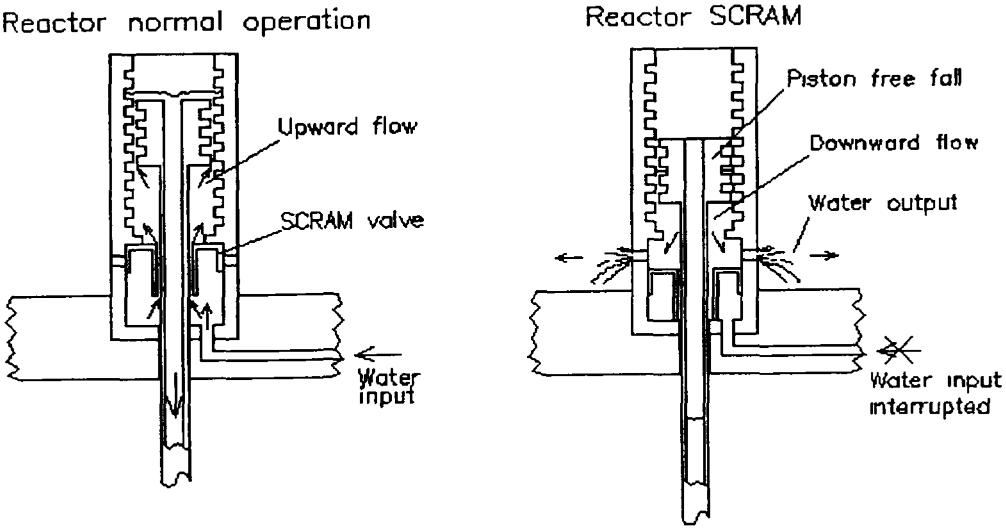
Examples of the experiments to be conducted during this stage are definitions related with alignment, clearances in linear bearings, dynamic analysis to determine natural frequencies and mode shapes and responses of the system under various external stimuli

6 CONTROL DRIVE MECHANISM

The CAREM mechanisms are hydraulic type, lodged inside the RPV The driving circuit of water provides a constant water flow over which positive or negative pulses produce the movement of the rod The development of the mechanisms involves four different stages

Preliminary conceptual verification: made to verify that theoretical approach and numerical results were in agreement with experimental results

"Cold" Prototype. To determine experimentally the minimum base flow and its operational limits the minimum flow to support the column and the maximum without extracting the rod Characteristics of pulse and improvements in the SCRAM valve were part of this stage We also found manufacturing hints that simplify the design, improve reliability and reduce costs The objectives of these experiments were almost totally achieved At present work is being conducted carried out to reduce SCRAM time Once finished, a "final" prototype will be manufactured to perform the characterization These experiments are expected to be finished during 1995



"Warm" Experiment T= 80 °C, atmospheric pressure To characterize the mechanism and the driving water circuit at different temperatures Study of abnormal situations increase in drag forces, pump failure, primary level influence, SCRAM valve failure, uncontrolled water flow and temperature, two phase water injection, suspended particle influence, air bubble influence, drainage blockage

"Hot" Experiment A simple loop is under design to reach CAREM nominal operational values in normal and abnormal conditions The objectives are the characterization of the mechanisms, durability tests, and behavior of systems under abnormal conditions breakage of feeding pipes, LOCA, behavior during operation of relief valves

7 INSTRUMENTATION AND CONTROL

7.1. Drive Mechanism Instrumentation

The position of the piston in the Drive Mechanism gives the position of the neutron absorber in the core The length of the piston is 300 mm and the length of the cylindrical cover is 1850 mm The method used to measure the piston position is called Variable Magnetic Inductance

The piston, made from magnetic stainless steel, moves inside the cylinder provided with an electrical coil with high concentration of rings on one end, decreasing toward the other All other parts of the Drive Mechanism are non-magnetic A fine measurement of the inductance gives the measurement of the piston position and consequently of the absorber position The resolution obtained was better than 1%

The tests during which the inductance was measured were carried out at room conditions Interference and very low frequencies were a challenge that had to be surmounted so instruments of high complexity and cost were used The design of a specially dedicated instrument to measure inductance was finished and a prototype to operate at CAREM conditions will be tested in the "warm" and "hot" test mentioned in Section 6

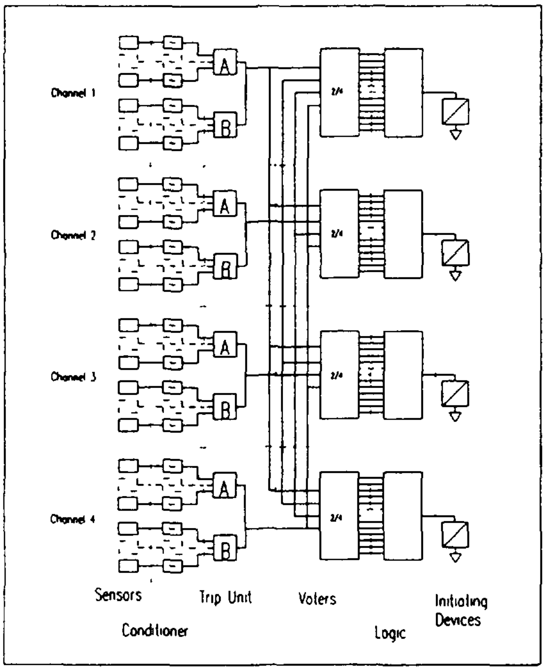
7.2. Reactor Protection System (RPS)

Description

The RPS is based on solid state intelligent processing units and hard-wired multiplexed voting and protective logic units It has four redundant, independent channels, whose main features are

- High reliability and availability as a result of design criteria and technology
- Fault-tolerance with on line auto-verification routines and auto-announcing capability
- Compactness and robustness
- High simplicity

REACTOR PROTECTION SYSTEM



Current development: The current developments originate in the Safety Requirement Specifications of the RPS.

Trip Unit The trip unit performs the data acquisition of the safety variables and compares them against the Safety System Settings to initiate protective actions in case of anticipated operational occurrences or accident conditions. The development of the Trip Unit is subdivided into the following stages

- Software Requirements Specifications & prototype
- Software ***design***, code and implementation
- Hardware Requirements Specifications & prototype
- Hardware ***design*** and implementation
- Integration Requirements Specifications
- Integration Hardware/Software
- Validation
- Installation & Commissioning
- Operation & Maintenance

At present, the status of the development is at ***design stage*** both in hardware and software.

Voting and Protective Logic Unit

The voting and protective logic unit performs the voting of the redundant safety trip signals coming from the Trip Units in a logic arrangement of 2 out of 4 and then, according to the logic relations of the trip signals and initiation criteria, triggers the protective actions

The development of the Trip Unit is subdivided into the following stages

- Hardware Requirements Specifications & prototype
- Hardware Requirements Specifications
- Hardware ***design*** and implementation

7.3. Supervision and Control System

*7.3.1 **Description***

The highly automated digital Supervision & Control System, has an architecture of 5-level hierarchical with distributed processing and modern control technology. It is conformed by different types of processing units·

- Supervision Units (SU)
- Information Units (IU)
- Control Units (CU)
- Field Units (FU)

The Supervision & Control System is totally independent of the RPS. High system reliability and availability are achieved by the use of redundancy and fault-tolerance in the communications and processing unit

Operator interface is based on digital visual display units for safety, alarms, logic's, processes and documentation presentation in the reactor main control room and at other supervision and control centers. Modern technologies such as touch-screens, track-balls, custom made keyboards, etc. are used.

*7.3.2 **Current Developments***

The Supervision & Control System includes the development of a Control Operating System that implements all the low level functions such as

Real-Time Data Base
Communication System
Control Functions
System Management

Historical Data Base
Man-Machine Interface
Signal Acquisition and Actuation

This Control Operating System acts as a software platform on which the Supervision & Control System application is built The development process is divided into the following phases

Software Requirements Specifications & Prototype
Software Coding, Implementation & Integration
Installation & Operation

Software Design Specifications
Validation & Verification

The ***Ward & Mellor Methodology*** is applied in every phase of the development process At present, the status of the development is at Coding phase

8 FUEL ELEMENTS

The activities in this subject are being carried out by the CNEA itself At present the detail engineering for the Fuel element and absorbers are under execution

Development of equipment for components and FE manufacturing The following tasks have already been carried out

- Development and construction of equipment for cap welding by TIG method
- Development and construction of dies for stamping and cutting elastic separator components
- Development and construction of FE assembly and final control boards
- Construction of different manufacturing and metrological control devices for FE manufacturing
- Prototype of elastic separator for the FE
- Dummy FE to define handling tools

Current developments The following tests are at a definition stage

- Elastic separators mechanical and stress tests
- Fuel element seismic behavior test
- Thermohydraulic behavior in a low pressure loop
- Thermohydraulic behavior in a high pressure test

REFERENCES

/1/ Deminins, M , Memoria descriptiva del CAPCN, INVAP 0758 5302 2IASS 315 1O (1994)

/2/ Carrica, Pablo, Análisis de Analogías entre el reactor CAREM-25 y el Circuito CAPCN, INVAP, 0758-8700-2TAIN-003-1O (1994)

/3/ Masriera, Nestor, Ensayos Dinamicos previstos para el CAPCN, INVAP 0758 8680 3IAIN 012 1O (1994)

/4/ Carrica, P & Baliña J , Plan de Ensayos de FCC, Parte I, caracterización de la instalacion, INVAP 0758-8720-3IAIN-001-1A (1994)

/5/ Carrica, P & Baliña J , Plan de Ensayos de FCC, Parte II, Ensayos Preliminares, INVAP 0758-8720-3IAIN-002-1AO(1994)

/6/ Carrica, P & Baliña J , Plan de Ensayos de FCC, Parte III, Ensayos Finales, INVAP 0758-8720-3IAIN-003-1O (1994)

/7/ RA-8, Preliminary Safety Report,

/9/ CONDOR 1 3, Villarino Eduardo, INVAP (1995)

/10/ Strawbidge and Barry, Criticality Calculations for Uniform Water Moderated Lattices, NSE, **23**, 58 (1965)

/11/ Raslog, Multigroup Methods in Thermal Reactors Lattice Calculation, Lecture, Bogotá, Colombia

/12/ Maeder and Wydler, International Comparison Calculations for a BWR Lattice with Adjacent Gadolinium Pins, EIR-Bericht 532, NEACRP-L-271 (1984)

/13/ Arkuszewsky, MCNP Analysis of the Nine-Cell LWR Gadolinium Benchmark, PSI-Bericht 13 (Aug 1988)

/14/ Szatmary, Experimental Investigation of the Physical Properties of WWER-Type Uranium Water Lattices, Final Report of TIC, Vol I, Akademiai Kiadó, Budapest (1985)

/15/ Id /14/ Vol III

XA9846722

THE CANDU 80

R.S. HART
AECL-CANDU,
Mississauga, Ontario, Canada

Abstract

AECL has completed the conceptual design of a small CANDU plant with an output in the range of 300 MWth, suitable for a variety of electrical and co-generation applications including desalination, oil sands oil extraction and processing, and the provision of electricity and heat to areas with low demand.

The design of this plant, called the CANDU 80, builds on AECL's extensive experience with small nuclear power plants, including NPD (22 MW(e)), KANUPP (125 MW(e)), Douglas Point (220 MW(e)) and Gentilly-1 (250 MW(e)), while taking advantage of the technological advances made by the latest large CANDU plants in the areas of design technology, construction methods, control and instrumentation, materials, and chemistry.

The "economy of scale" disadvantage is partly overcome through simplification, exploitation of the inherent characteristics of small reactor size, and incorporation of relatively low peak bundle powers (about 60% of those for large CANDU plants). For example the small core is stable, requiring no spatial control, and the limiting transients are relatively slow.

The station and plant layouts for the a CANDU 80, dedicated to electricity production and located at a coastal site are shown in Figures 1 and 2 respectively. These layouts are readily adapted to suit various co-generation and site conditions. Section and plan views of the containment system, showing the location of major components, are presented in Figures 3 and 4 respectively.

The CANDU 80 makes extensive use of component designs available from larger CANDU plants; these include two Pickering steam generators, two Pickering HTS pumps, the Gentilly-1 fuelling machine, and the same pressure tubes as Wolsong 3 and 4 (length reduced from 6 m to 5 m). Proven designs and operating parameters are used throughout (for example, heat transport system conditions are essentially the same as Pickering), while a limited number of new features serve to simplify operation, increase operating margins, and to enhance safety are incorporated.

Safety features include: low power density; stability and slow transients provided by the small core; two independent shutdown systems; double containment; fully capable emergency core cooling; low man-rem maintenance capability; passive decay heat removal via steam generators; passive moderator heat removal; passive shield cooling; and passive primary containment cooling.

This paper provides a brief overview of the CANDU 80, and discusses key features contributing to safety and operational margins.

1. INTRODUCTION

1.1 Background

The CANDU 80 is ideally suited to a variety of electricity and co-generation applications, including desalination, oil sands oil extraction and processing, and the provision of electricity and heat to areas with a relatively small demand. CANDU 80 can also fill a valuable role in the economic development of infrastructure in countries planning a large nuclear program, effectively bridging the gap between research reactors and the large (500 MW(e) to 1500 MW(e)) nuclear power plants currently available.

The CANDU 80 features low power density and large operating margins, thereby facilitating many simplifications and providing highly tolerant operating characteristics relative to large nuclear power plants. CANDU 80 takes advantage of the small CANDU reactor experience gained with early plants such as NPD, KANUPP, and Douglas Point, and the equipment, materials and chemistry advances made in the latest large CANDU reactors, while introducing a limited number of advanced features that enhance safety and reduce operation and maintenance costs.

Proven technology is used throughout the CANDU 80, updated with relevant features resulting from ongoing Canadian research and development. The CANDU 80 builds on the reactor and process system designs of the established CANDU plants, and incorporates advanced construction methods and operational features. The fuel and fuel channel technology and thermohydraulic and neutronic operating characteristics are the same as those of operating CANDU plants, and have been confirmed by extensive materials and full-scale fuel channel tests.

A high level of standardization has always been a feature of CANDU reactors. This theme is emphasized in the CANDU 80; all key components (for example, steam generators, coolant pumps, and pressure tubes) are of the same design as those proven in service in operating CANDU power stations.

1.2 CANDU 80 Accomplishments

A number of significant accomplishments are evident in the CANDU 80 design. These include:

1) The development of an economic small nuclear power plant (in the range of 300 MW(th)/100 MW(e)) thereby providing a nuclear power option for meeting many relatively small energy demands.
2) The maintenance of all traditional CANDU features including horizontal fuel channels, heavy water moderator and coolant, on power refuelling, and the ability to operate on a variety of low fissile content fuels including natural uranium.
3) The use of many proven component designs from previous CANDU plants without any significant design changes. These include steam generators and heat transport pumps from Pickering, pressure tubes from Wolsong 3 and 4, and the Gentilly 1 fuelling machine.
4) The use of proven system concepts and operating conditions. For example, the heat transport system conditions are essentially the same as those in Pickering.
5) The incorporation of large operating and safety margins, and ease of operation. The peak fuel bundle power for example is only 60% of that of CANDU 6.
6) The inclusion of a number of passive heat rejection systems to enhance safety, increase simplicity of operation, and reduce testing requirements.
7) A relatively short 24 month construction schedule.
8) The allocation of space within the containment structures for isotope production and/or test loop facilities.

1.3 CANDU 80 Safety

The CANDU 80 provides an extremely high level of safety through the provision of large operating margins and the incorporation of both traditional and advanced safety and safety support features. These include:

1) Two independent, fully capable safety shutdown systems. Both systems are passive, using gravity to inject neutron absorbing liquid into incore tubes.
2) A fully capable (pumped) emergency core cooling system to assure fuel cooling following a loss of coolant accident (LOCA). This system is backed up by two independent, passive decay heat removal systems (the moderator system and the shield cooling system).
3) A double containment system including a steel primary containment and a reinforced concrete secondary containment. The primary containment is passive, with passive post accident (LOCA or steam line failure) cooling.
4) Reject condensers to remove decay heat from the steam generators in the event of a loss of feedwater; this system is completely passive, requiring no valve or operator action to initiate operation.
5) A passive backup moderator cooling system capable of removing decay heat from the fuel via the moderator immediately after reactor shutdown.
6) The Reserve Water System, which can remove decay heat via the steam generators, the moderator system or the shield cooling system, and provide cooling of the primary containment for a minimum of 3 days without external cooling or power, or water makeup.

2. DESIGN SUMMARY

2.1 Layout

The principal structures of the CANDU 80 Nuclear Generating Station include the secondary containment building, the reactor auxiliary building, the maintenance building and a heat utilization building. The turbine building shown in the station layout, Figure 1, is typical of a heat utilization building for a CANDU 80 dedicated to electricity production and located at a coastal site; this facility can be modified as required to comply with specific co-generation or heat application requirements. Auxiliary structures include the administration building, and depending on site conditions and application, a pumphouse and/or cooling tower.

The distribution of equipment and services among the buildings is primarily by function. To the maximum extent possible, the structures are self-contained units with a minimum number of connections to other structures. The plant layout is presemted in Figure 2.

The layout provides for a short construction schedule by simplifying, minimizing and localizing interfaces, by allowing the parallel fabrication of equipment modules and civil construction, by reducing construction congestion, by the provision of construction access to all areas, by providing flexible equipment installation sequences, and by reducing material handling requirements. The layout also benefits from the application of modern human factors design practices, including a plant-wide "Link Analysis"; this serves to improve operations and maintenance efficiency, and to minimize the potential for human error.

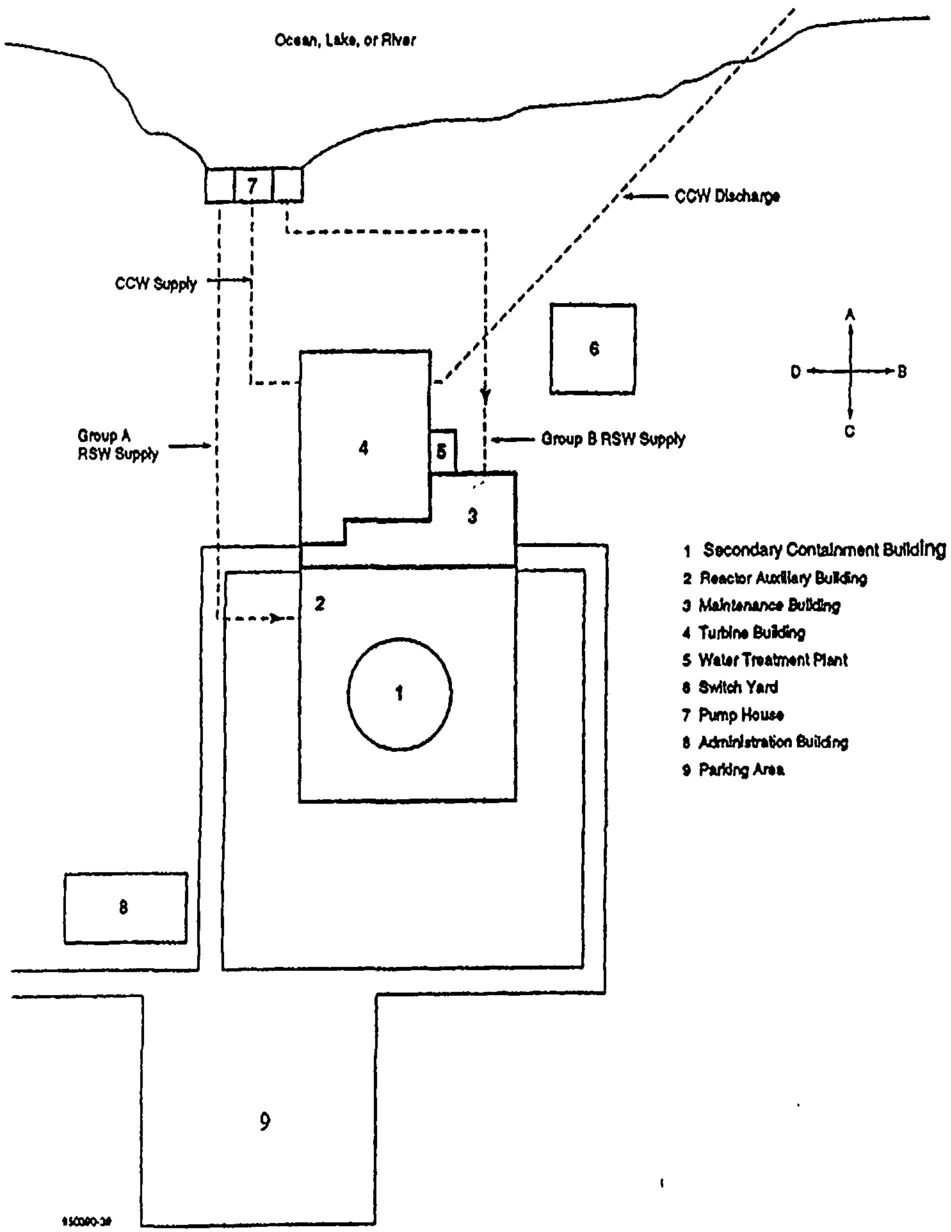


Figure 1 Station Layout

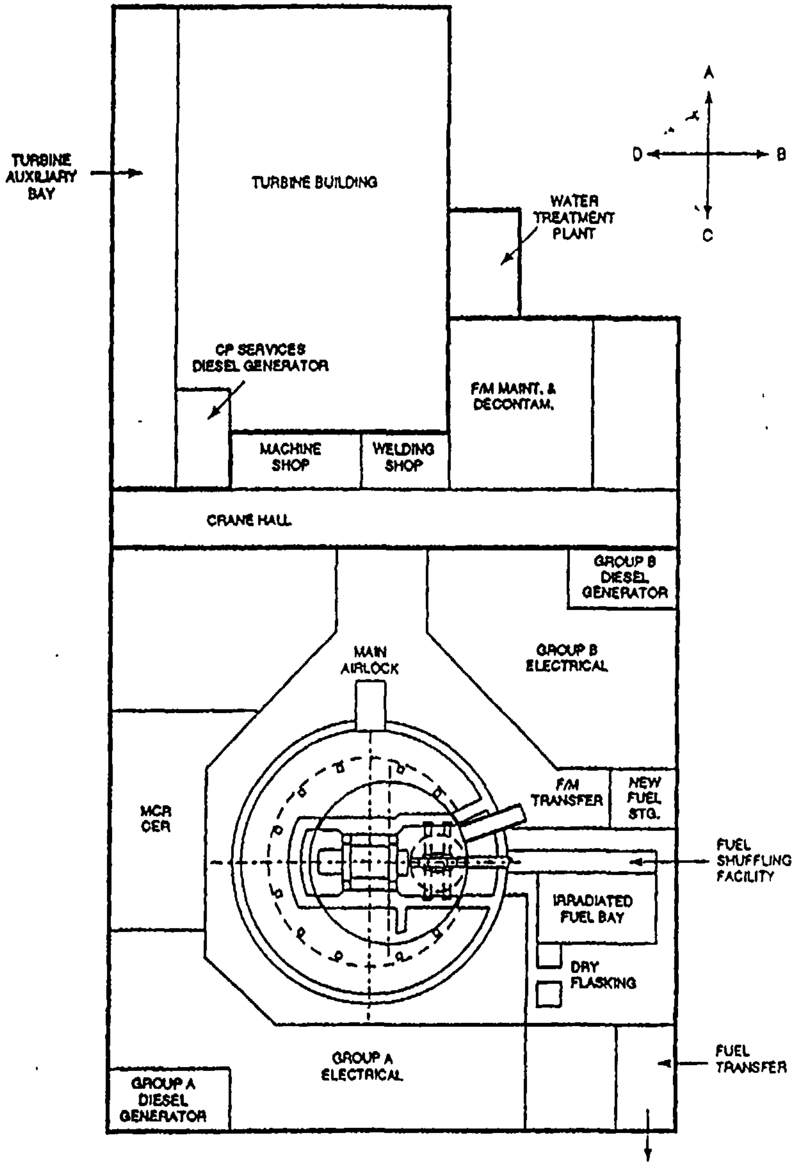


Figure 2 Plant Layout

The CANDU 80 incorporates a double containment system that builds on the operational features proven in-service at the Ontario Hydro Bruce and Darlington stations. Specifically, the high energy nuclear systems are housed within a compact Primary Containment, while key nuclear steam plant services are accommodated within confinements adjacent to the primary containment. In CANDU 80, the primary containment and the confinements are housed within a robust Secondary Containment. The CANDU 80 containment system is illustrated in Figures 3 and 4. The principal advantages of the CANDU 80 containment system include enhanced safety, reduced exclusion radius, ease of maintenance, and reduced capital and operating costs.

2.1.1 Grouping and Separation

All process systems and services in CANDU 80 are assigned to one of three groups (Group A, Group B, or Conventional Plant (CP) Services). Group A and Group B systems are primarily located in the Nuclear Steam Plant portion of the station, while CP Services are primarily located in the Conventional Plant portion of the station.

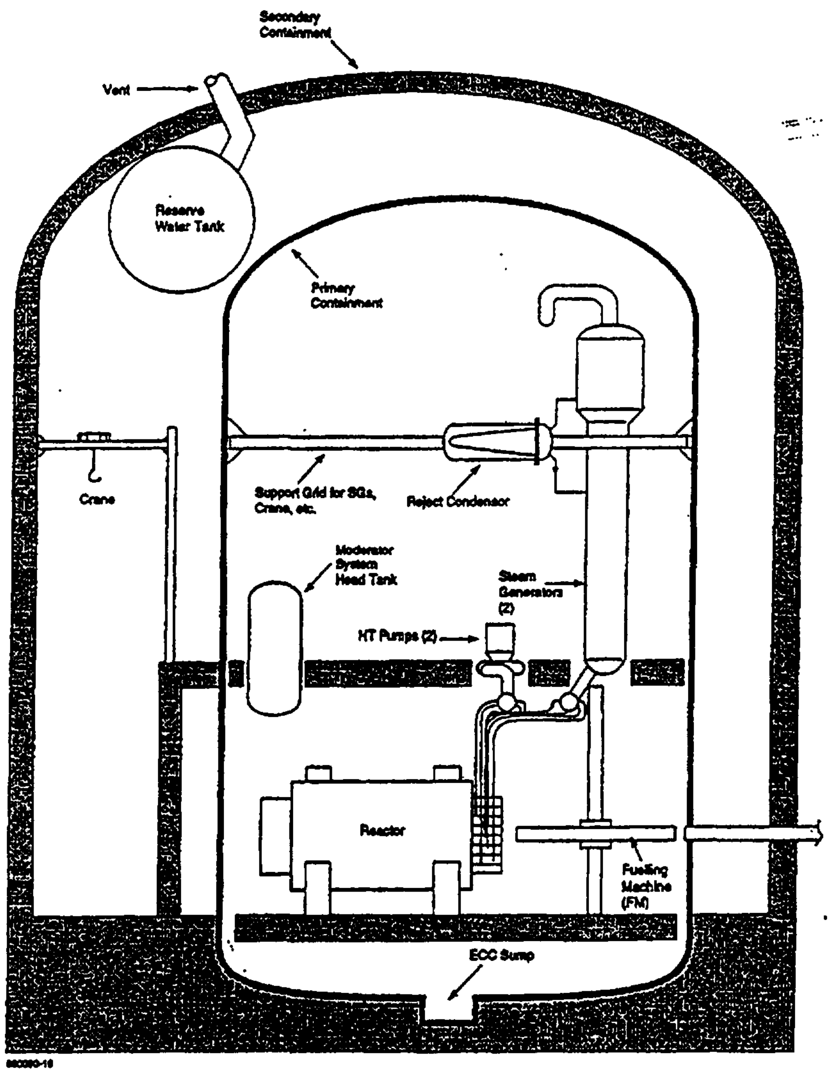


Figure 3 Containment System, Section View

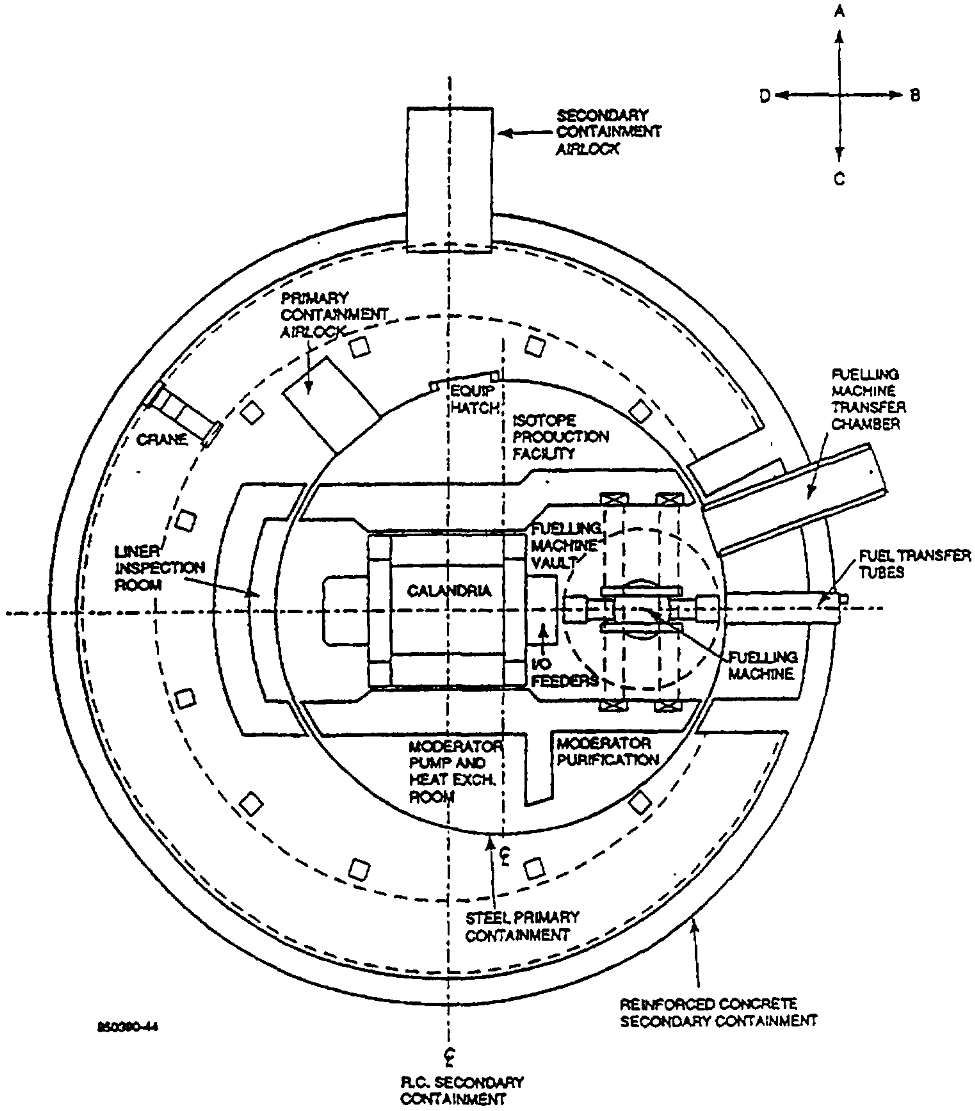


Figure 4 Containment System, Plan View

Group A and Group B systems each serve about half of the Nuclear Steam Plant (NSP) loads, and each include two of the four special safety systems; shutdown system No. 1 and the emergency core cooling system are assigned to Group A, while shutdown system No. 2 and the containment system are assigned to Group B. The control room is located in Group A; the secondary shutdown area is located in Group B. Group A and Group B services are provided to most of the principal nuclear steam plant systems; for example, the equipment served by the Group A recirculated cooling water system includes one moderator heat exchanger, one emergency core cooling system heat exchanger, and one primary containment cooling system heat exchanger, while the second heat exchanger in each system is served by the Group B recirculated cooling water system. In general, both Group A and Group B services are required for plant operation at full power, while safety requirements can be supplied by either Group A or Group B services, or in some cases, without the need for either Group A or Group B services. Group A and Group B systems are seismically and environmentally qualified, and tornado protected consistent with site requirements.

The CP Services are generally dedicated to normal power production, and are seismically qualified to local building code requirements.

To guard against cross-linked and common mode events, the Group A systems, Group B systems, and CP Services are, to the greatest extent possible, located in separate areas of the station, as shown in Figure 5. This approach to the grouping and separation of systems, an extension of current CANDU practice, results from studies that considered safety, operability, human factors, and cost.

3. SAFETY AND OPERATIONAL MARGINS

3.1 Overview

The CANDU 80 incorporates many features that enhance safety and increase operational margins. These include the approach to grouping and separation and the double containment system discussed in the previous section. Other features, including low power density and the incorporation of several passive heat removal systems, are discussed in the following sub-sections.

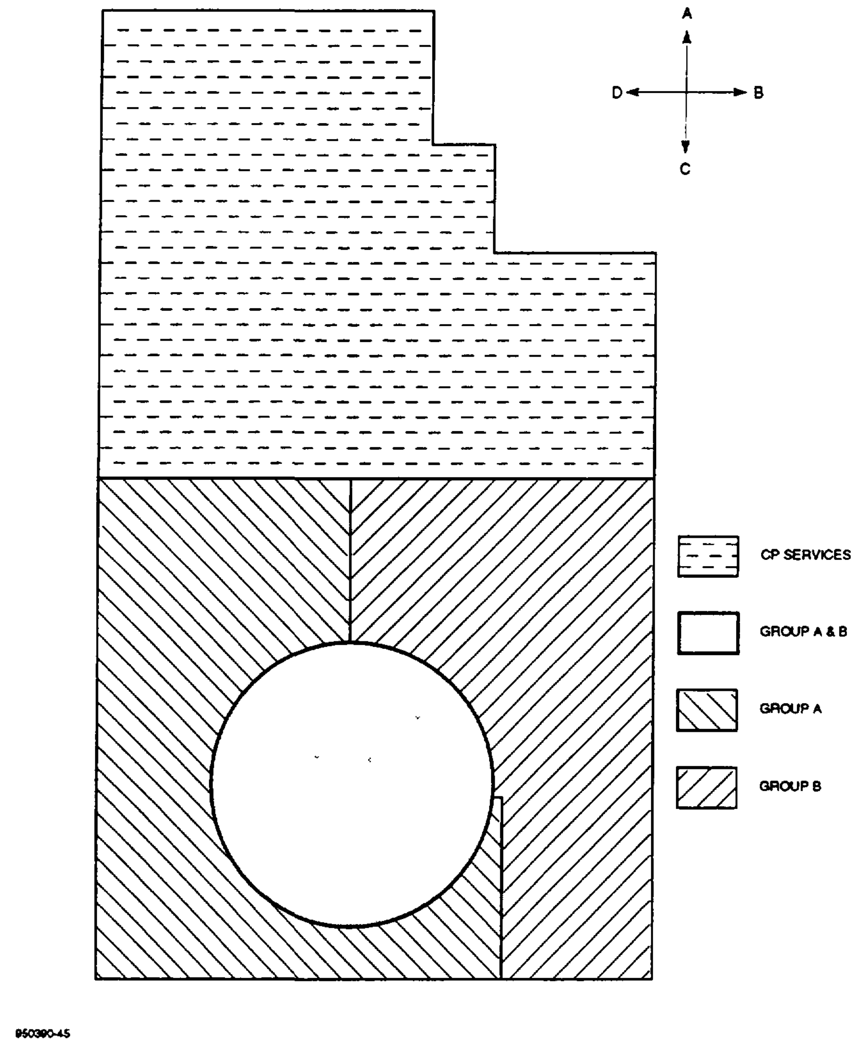


Figure 5 Area Allocation by Group

3.2 Power Density

CANDU 80 features peak fuel bundle powers that are about 60% of those of current larger CANDU plants. At the same time, heat transport system coolant conditions are about the same as those in the Pickering plants. This is accomplished by a unique fuel channel arrangement, in which the reactor coolant passes through two fuel channels, each containing 10 fuel bundles (each channel contains 12 bundles in large CANDU plants) connected in series, as shown in Figure 6. This arrangement, while making use of standard CANDU components and technology, provides the desired combination of low bundle power and Pickering HTS conditions.

3.3 Moderator System Passive Cooling

In all CANDU plants, including CANDU 80, the centre element of the fuel bundle is located less than 50 mm from the cool D_2O moderator surrounding the fuel channel. Hence, in the unlikely event of a loss of coolant from the heat transport system coincident with the failure of the emergency core cooling system, fuel can be cooled by heat rejection to the moderator. In CANDU 80 passive cooling of the moderator D_2O is provided for events that include the loss of active cooling systems.

The moderator system shown in Figure 7, circulates heavy water through the calandria to remove the nuclear heat generated in the moderator, and the heat transferred to the moderator from the fuel channels. The moderator system includes two 50% moderator pumps, two 50% plate type heat exchanger, and a head tank. The moderator pumps and heat exchangers are located in separate confinement areas, located in the interspace between the primary containment and the secondary containment. The moderator system head tank is located inside the primary containment.

During normal operation the moderator pumps draw D_2O from the top of the calandria via the moderator heat exchanger. The moderator heat exchangers are cooled by feedwater provided by the condensate extraction pumps during normal plant operation (see Section 5.4). Should feedwater flow be unavailable recirculated cooling water is directed to the secondary side of one moderator heat exchanger, to remove shutdown heat; Group A recirculating cooling water is available to one moderator heat exchanger while Group B recirculated cooling water is available to the other moderator heat exchanger.

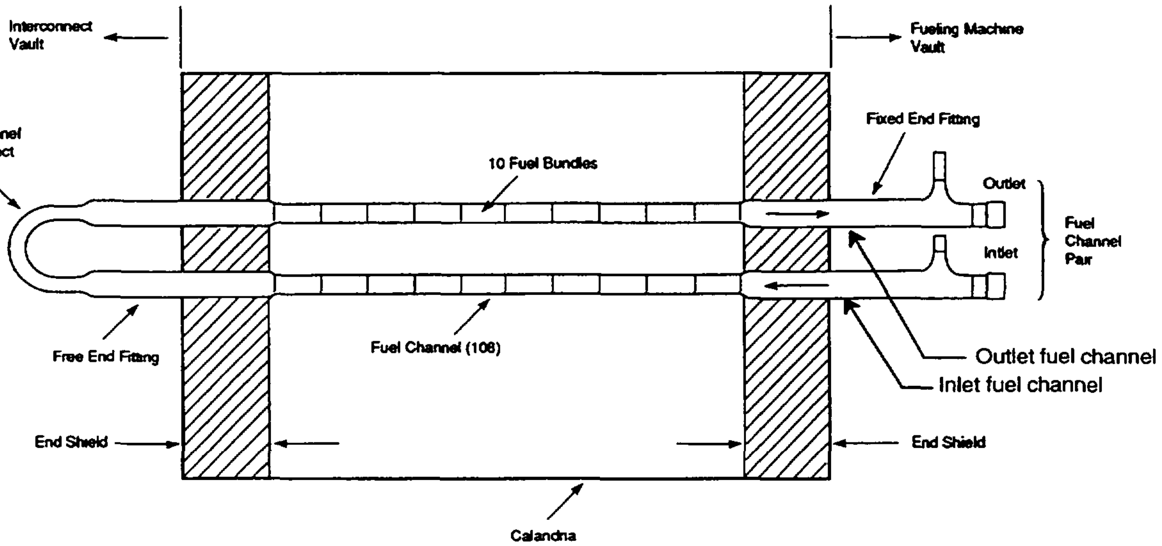


Figure 6 Arrangement of Fuel Channel Pairs

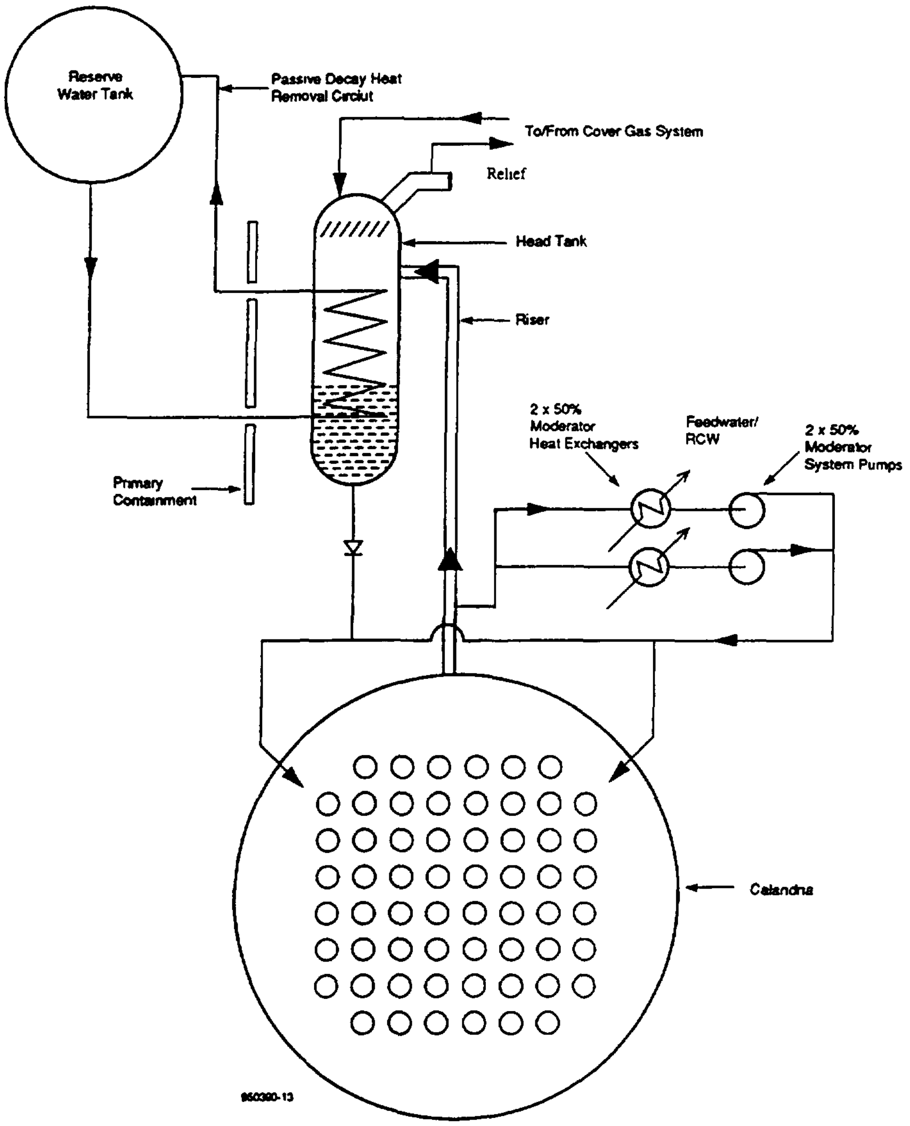


Figure 7 Moderator System Flowsheet

In the event that the moderator system pumps and/or heat exchangers are lost, the reactor is shut down, and flashing in the riser initiates natural convection circulation of the moderator. In this mode, heat is rejected to the water in the reserve water tank via a cooling coil located in the moderator system head tank, utilizing natural convection. The capacity of this mode of moderator cooling is sufficient to remove decay power under conditions of coincident loss of coolant and loss of emergency core cooling, without makeup water or cooling being provided to the reserve water tank, for a period of 72 hours. The reserve water tank includes four sub-sections; one sub-section is devoted to moderator system cooling (see Section 3.4).

Initiation of moderator cooling via the head tank cooling coil/reserve water tank is totally passive. No operator or control action is required.

3.4 The Reserve Water System

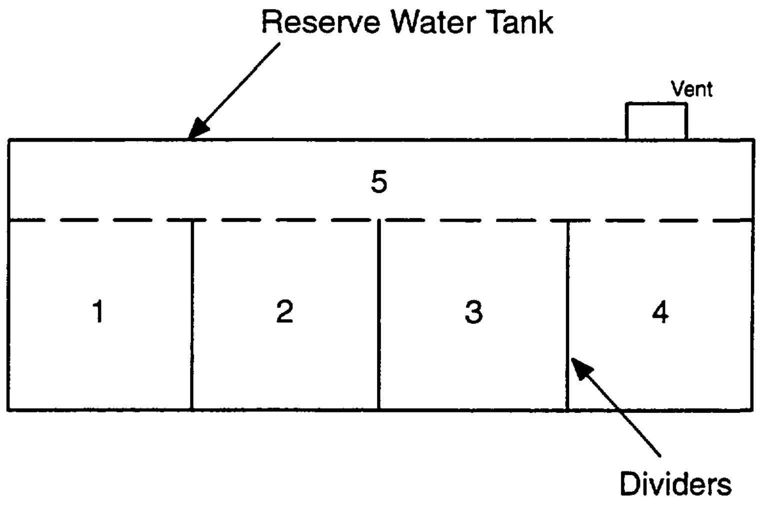
The Reserve Water System provides passive decay heat removal from the steam generators (via the reject condenser), the moderator system, the shield cooling system, and the primary containment internal environment. The principal component of the reserve water system is a large water storage tank (the reserve water tank) located at a high elevation in the interspace between the secondary and primary containment structures (see Figure 8). The lower portion of the reserve water tank is divided four compartments, each of which is open to the common upper portion of the reserve water tank. This arrangement is conceptually illustrated in Figure 8. Each compartment is dedicated to heat removal from one of the four associated systems (reject condensers, moderator system, shield cooling system, and primary containment). Each compartment contains sufficient water to remove decay heat for 24 hours following reactor shutdown. The common (upper) portion of the tank contains water for an additional 48 hours of decay heat removal. Hence, 72 hours of decay heat removal are available via any one of the associated systems. The section of the reserve water tank dedicated to primary containment cooling contains two cooling coils, one cooled by the Group A recirculated cooling water system and one cooled by the Group B recirculated cooling water system, each capable of removing decay heat. The water from the shield cooling system passes through the shield cooling system heat exchanger, cooled by Group B recirculated cooling water, before returning to the reserve water tank.

A vent line from the reserve water tank allows any steam produced in the reserve water tank to discharge to the atmosphere.

A purification system, consisting of a small pump, filter, ion exchange column, and sampling and chemical addition facilities maintains the chemistry and purity of the reserve water tank water within the design limits.

3.5 The Steam Reject System

The steam reject system, shown in Figure 9, includes a reject condenser, connected to each steam generator.



Water Allocation

Volume 1 - Reject Condensers
Volume 2 - Primary Containment
Volume 3 - Moderator
Volume 4 - Shield Cooling
Volume 5 - Common

Figure 8 Reserve Water Tank Water Allocation

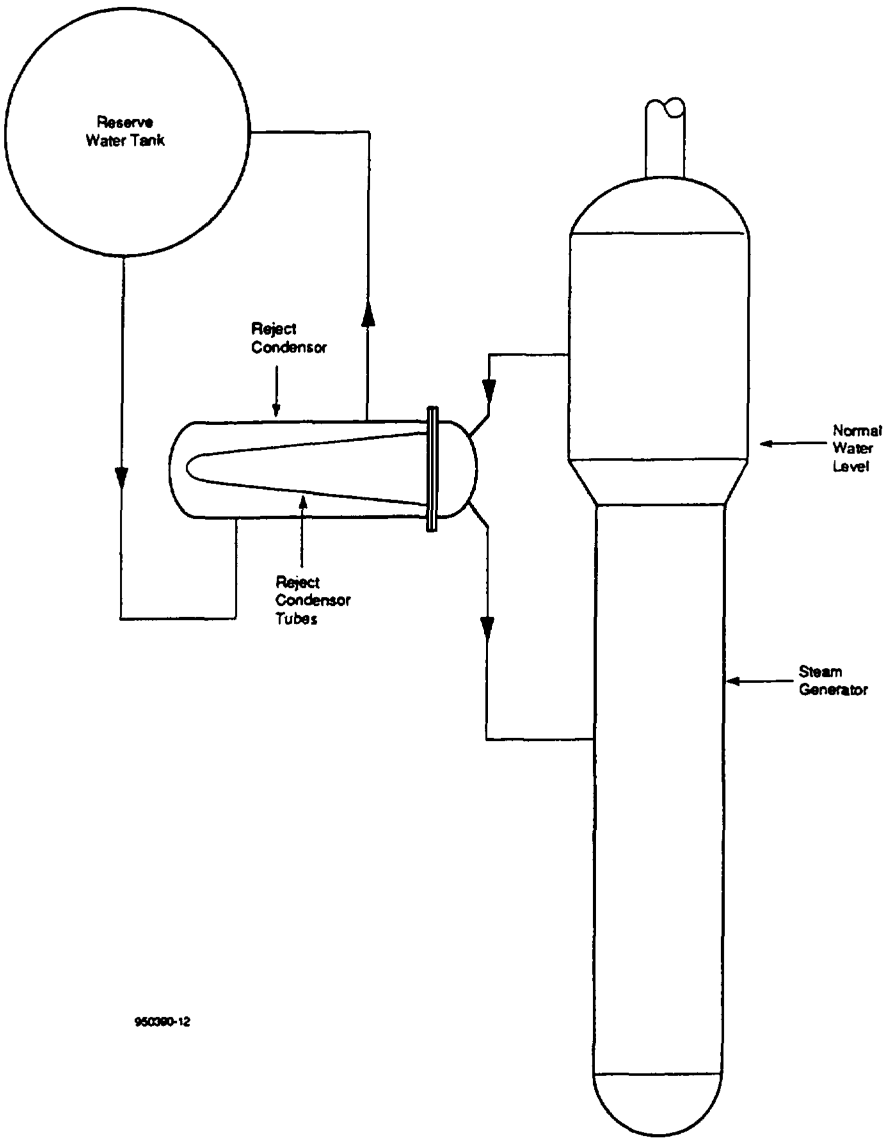


Figure 9 Reject Condenser System

When water level in the steam generator is within the specified operating range, the condensing coil in the reject condenser is filled with water, and circulation is prevented by the vapour (steam) lock in the pipe connecting the reject condenser inlet to the steam generator. In the event that feedwater flow is lost, and the water level in the steam generator drops significantly below the top reject condenser coils, steam is condensed in the reject condenser coils, and the condensate is returned to the steam generator downcomer.

The secondary side of the reject condenser is cooled by natural convection, via flow from and to the reserve water tank. The water available to the reject condensers from the reserve water tank (water in the reject condenser compartment plus the water in the common portion of the reserve water tank) is sufficient to remove decay heat, via evaporation, for a period of 72 hours without makeup or cooling to the reserve water tank.

The operation of the reject condenser system is fully passive; no valve operation or operator action is needed to initiate operation.

3.6 Shield Cooling System

The shield cooling system, shown in Figure 10 removes the nuclear heat generated in the shield water and structures and the heat transferred to the shield water, via natural circulation through the reserve water tank. During normal operation heat is removed from the shield water before returning to the reserve water tank by the shield cooling sytstem heat exchanger, which is cooled by the Group A recirculated cooling water system. The flow of cool water into the reserve water tank helps maintains the temperature of the water in the reserve water tank within the specified range. In the event that cooling water to the heat exchanger is lost, the water available to the shield cooling system (the water in the shield cooling system compartment of the reserve water tank plus the water in the common position of the reserve water tank) is sufficient to provide cooling to the end shields and shield tank for 72 hours, without makeup or cooling to the reserve water tank.

3.7 Primary Containment Cooling

The Primary Containment is inaccessible with the reactor at power. Humidity and temperature are therefore maintained at levels sufficiently low to protect the equipment located in the Primary Containment.

The Primary Containment is cooled via a water circuit, with coils located within a cooling duct inside the primary containment, which connect to the reserve water tank (see Figure 11). Two cooling coils in the reserve water tank, one provided with Group A recirculated

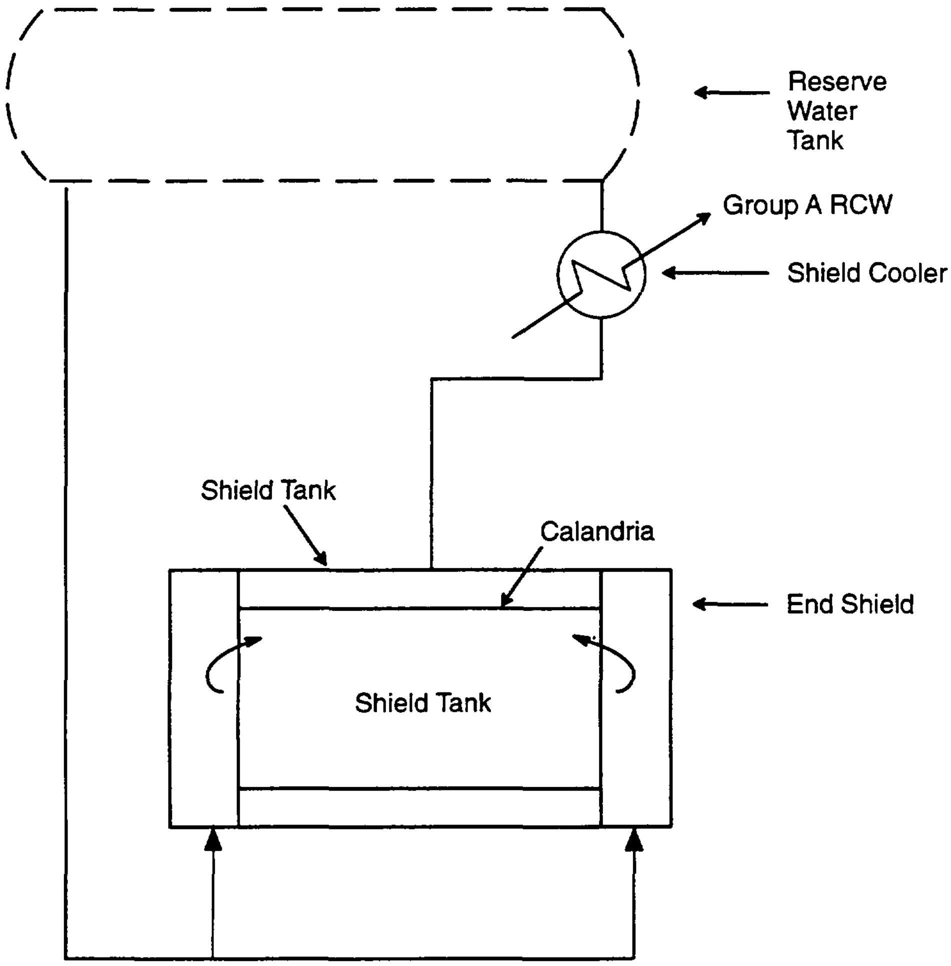


Figure 10 Shield Cooling System

cooling water and one provided with Group B recirculated cooling water maintain the temperature of the water in the reserve water tank within the design range during normal plant operations. During normal operation, 2 x 50% circulating fans located in the cooling duct and 2 x 50% pumps in the water circuit assure that temperatures in the primary containment do not exceed 50°C. One fan and one pump are provided with power from the Group A electrical distribution system while the other fan and pump are powered by the Group B electrical distribution system.

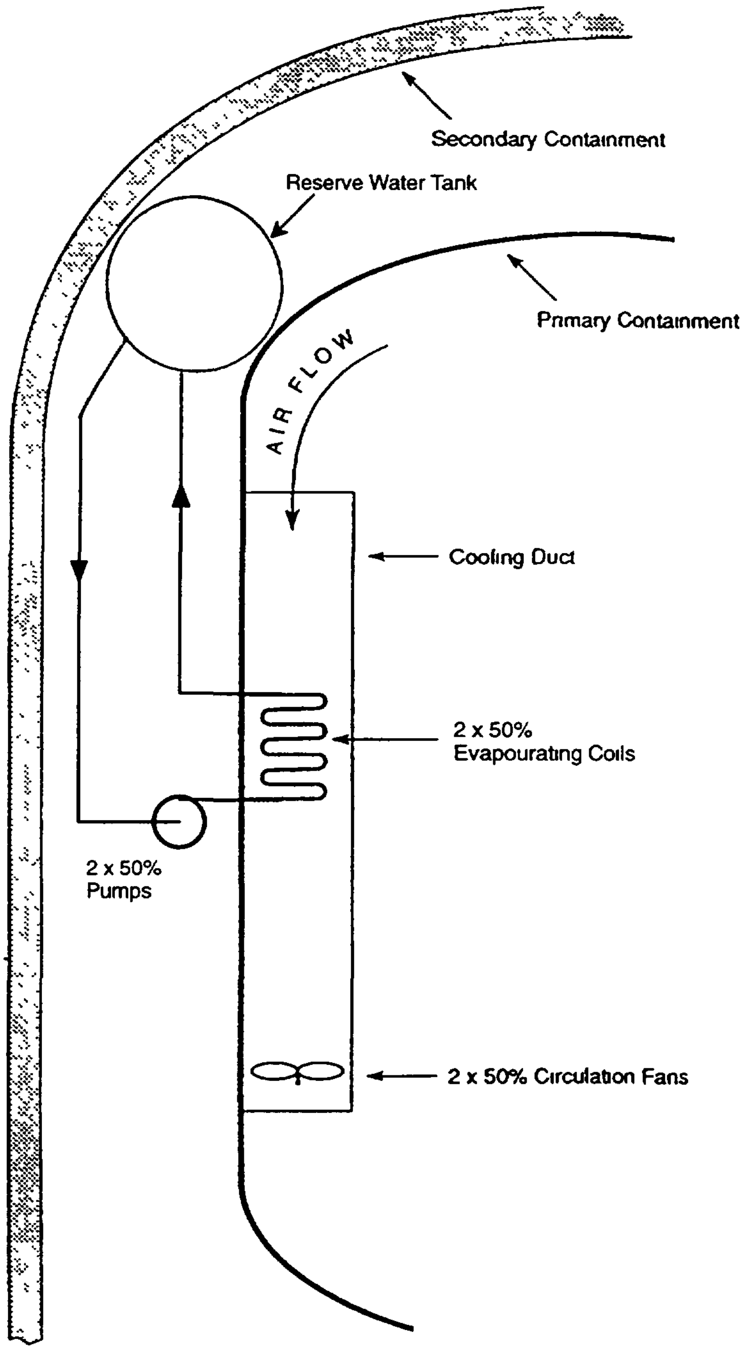


Figure 11 Primary Containment Cooling System

Following a postulated accident (Loss of Coolant Accident or a steam line failure within the primary containment), all circulation fans and pumps may be lost. Under these conditions, natural convection in the water circuit and the cooling duct maintain the primary containment temperature below 125°C (except for an initial transient period).

4. SUMMARY

The CANDU 80 is an economic nuclear plant, ideally suited for a variety of electrical production and co-generation applications, including desalination. This paper provides a brief overview of key CANDU 80 features. A CANDU 80 Technical Outline is available from the author on request.

XA9846723

ISIS, SAFETY AND ECONOMIC ASPECTS IN VIEW OF CO-GENERATION OF HEAT AND ELECTRICITY

L. CINOTTI
ANSALDO Nuclear Division,
Genoa, Italy

Abstract

ANSALDO has conceived a reactor called ISIS (Inherently Safe Immersed System), an innovative light water reactor with easily understandable safety characteristics.

The main targets are: passively safe behaviour, no pressurization of the Reactor Containment under any accident condition, control of plant capital cost and construction schedule by virtue of the modular concept and the compact layout.

The ISIS concept, described in general terms in the paper, builds up on the Density Lock concept originally proposed by ABB ATOM for the PIUS plant (ref. /1/), featuring innovative ideas derived from ANSALDO experience and based on proven technology from both LWR and LMR.

1. MAIN TARGETS OF THE ISIS CONCEPT

1.1 Safety targets

Significant progress has been made in the last years towards nuclear reactors that rely to the smallest possible extent on safety-related active systems, which, even using up-to-date technology, are felt by the public as prone-to-fail, no matter how low the frequency target for their loss is set.

The ISIS concept, under development in ANSALDO, largely embodies this progress. The main safety targets may be summarized as follows:

- No core melt-down and negligible release of radioactivity in any accident condition, by virtue of the reactor concept itself.
- Prompt reactor shut down occurring naturally after any abnormal condition.
- Reactor cooling in natural circulation for unlimited time.
- Self-depressurization of the reactor after a postulated failure of the pressure boundary.

1.2 Economic targets

The economic target aims at a viable industrial power plant based on specific overnight-capital cost and construction time competitive with those of the Light-Water Reactors under development. This is achievable by means of following features:

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

2. BASIC CHARACTERISTICS OF THE ISIS CONCEPT

The **Inner Vessel** (fig. 1), which encloses the circulating, low boron concentration,pressurized hot water of the Primary System, is immersed in the highly borated pressurized cold water of the **Intermediate Plenum**. The Inner Vessel is provided with Wet Insulation to limit heat losses towards the Intermediate Plenum in normal operation.

The **Reactor Vessel**, which is the essential part of the pressure boundary, encloses the Intermediate Plenum and contains the Integrated Components of the Reactor Module.

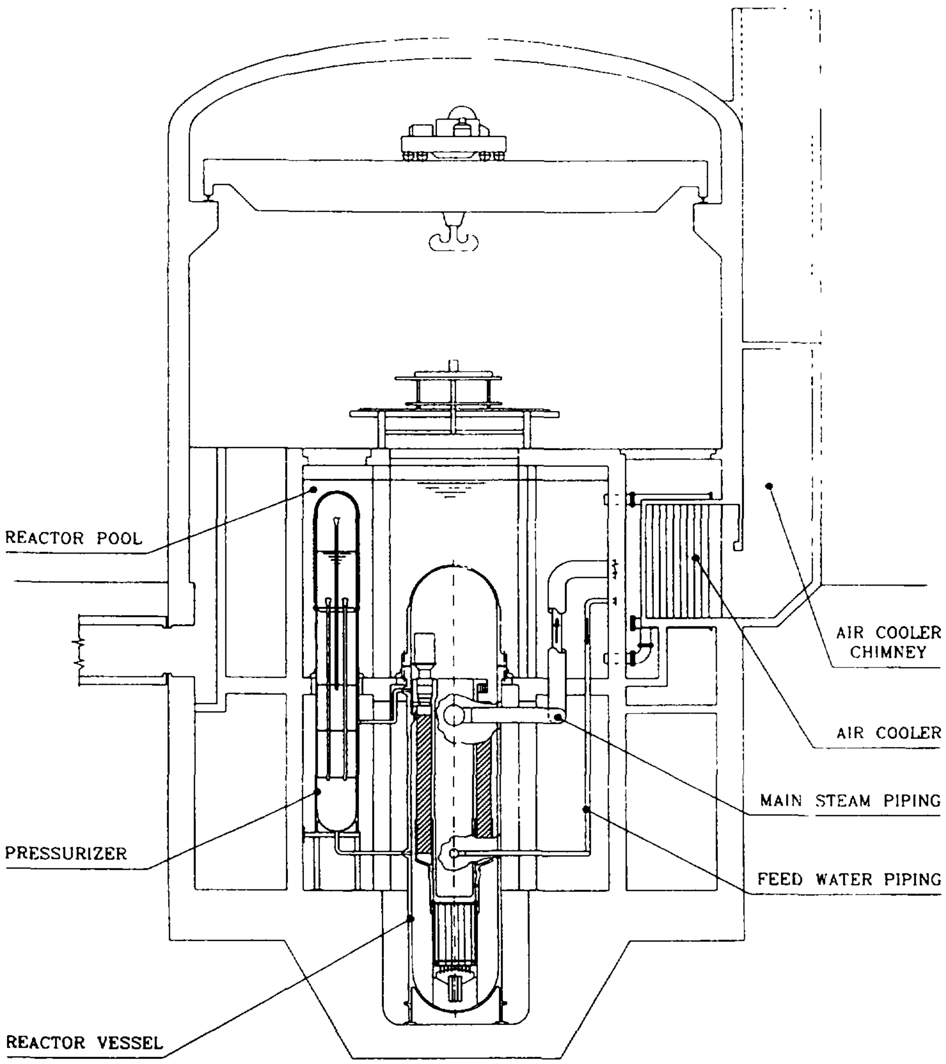


Fig. 1 - ISIS Reactor Building

- The Reactor Vessel is immersed in the cold borated water of a large **Reactor Pool** at atmospheric pressure. The Reactor Vessel is not insulated; this allows heat transfer to the surrouding water of the Reactor Pool under accident conditions.
- The **Pressurizer** upper portion performs the pressure control function; the lower portion contains cold water and provides additional heat transfer surface to the Reactor Pool under accident condition.

3. THE ISIS PRIMARY SYSTEM

The Primary System of the ISIS reactor is of the integrated type (fig. 2), with the Steam Generator Unit (SGU) housed in the Reactor Vessel, to which feedwater and steam piping are connected.

Within the Reactor Vessel, an Inner Vessel provided with wet metallic insulation separates the circulating low-boron primary water from the surrounding highly borated cold water.

Hot and cold plena are hydraulically connected at the bottom and at the top of the Inner Vessel by means of open-ended tube bundles, referred to in the following as Lower and Upper **Density Locks**. The Inner Vessel houses the Core, the Steam Generator Unit and the Primary Pumps.

Outstanding feature is the complete immersion of the Pressure Boundary, made up, for each module, of a Reactor Vessel and of a separated Pressurizer with interconnecting Pipe Ducts, in a large pool of cold water.

ISIS overall design Parameters

Thermal power	650	MWth
Net electric power	200	MWe
Core inlet temp.	271	°C
Core outlet temp.	310	°C
Operating pressure	14	MPa
Feedwater temp.	120	°C
Steam pressure	4,6	MPa
Steam outlet temp.	290	°C

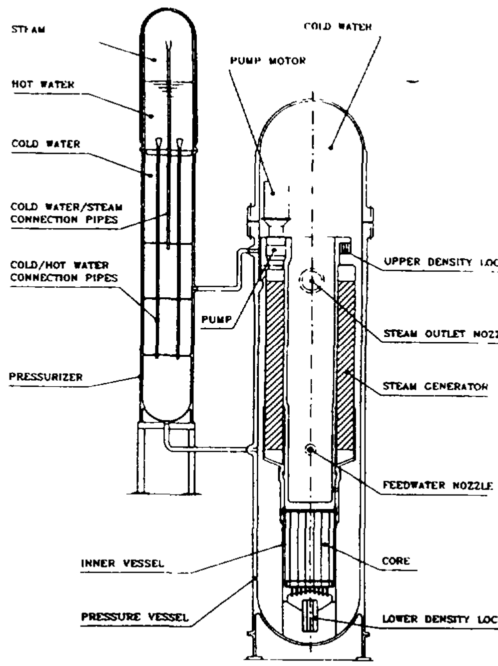


Fig. 2 - ISIS Reactor Module

During normal operation, the heat generated in the core is transferred to the SGU via the water circulated by the Primary Pumps, which are located at the top of the Inner Vessel. In case of unavailability of this heat transfer route, the cold and highly borated water of the Intermediate Plenum enters the Primary Circuit from the bottom, mixes up with the hot primary water, shuts down the reactor and cools the core in natural circulation. The same process, by heating the intermediate plenum water and the Pressure Boundary metal, activates the natural heat transfer route towards the Reactor Pool, which contains approximately 6.000 cubic meters of cold water.

The water inventory in the Reactor Pool is large enough to allow the water itself to remain below the boiling point after removal of the decay heat for about a week.

Cooling down of the plant pool is guaranteed, anyway, for an unlimited time, by virtue of two loops provided with water-air heat exchangers in natural circulation, sized to reject to the atmosphere, at steady state, approximately 2 MW and thereby capable to prevent the pool water from boiling.

Similarly to the PIUS reactor concept, the shut down and cooling functions of the core are carried out, in any condition, by the highly borated cold water of a plenum, which is hydraulically connected to the primary system by means of density locks.

However, unlike the PIUS, the intermediate plenum of ISIS contains a relatively small inventory of cold water (approximately 300 cubic meters per reactor module) at primary system pressure.

4. MAIN COMPONENTS

Reactor Vessel

The Reactor Vessel is of cylindrical shape with hemispherical heads.

The construction material is low-alloy carbon steel, internally lined with austenitic stainless steel.

The main openings of the Reactor Vessel are the water/steam nozzles and the two connections to the Pressurizer.

Core

The reactor core consists of 69 typical (17 X 17) PWR fuel assemblies with a reduced length to limit pressure losses.

Steam Generator Unit (SGU)

The SG features an annular tube bundle with helicoidal tubing.

The steam is generated tube-side. The feed water piping is connected to feed water headers, located symmetrically inside the reactor vessel within a calm zone, provided each with two tubeplates laid out vertically. The tubes depart circumferentially from the tubeplates.

A similar arrangement is provided at the top for the two steam headers connections.

The vertical arrangement of the tubeplates aims at preventing crud deposition at the tube-to-tubeplate connections, where the corrosion is likely to occur.

The higher outer rather than inner tube pressure, a reversed situation with respect to a conventional SGU, reduces the risk of flaw growth in the tubes.

Primary Circulation Pumps

The two Primary Pumps of the variable speed, glandless, wet winding type (like the pumps manufactured by Hayward Tyler Fluid Dynamics) are fully enclosed within the Reactor Vessel. The pump motor is cooled by the water of the Intermediate Plenum.

Above Core Structure (ACS)

The ACS, shaped like a flat-bottom cylindrical glass, provides the support for the core instrumentation and forms the inner wall of the annular riser of the primary water. The ACS is open at the top. The water within it is part of the intermediate plenum and this helps to limit the primary water inventory in the reactor module to a minimum. The ACS is flanged to and suspended from the top of the Inner Vessel for easy removal to allow standard fuel handling.

Pressurizer

The Pressurizer is of a slim cylindrical shape with hemispherical heads.

The pressure control function is carried out in the upper part, which is externally insulated to limit heat losses from the steam and hot water plena.

The remaining bottom part contains a cold water plenum, hydraulically connected to the upper hot water plenum by means of a number of pipes.

The function of the pipes is to enhance mixing of the hot water with the cold water, in case of water flow towards the reactor vessel during transients.

Interconnecting Pipe Ducts

The two Pipe Ducts between Pressurizer and Reactor Vessel connect hydraulically the top and the bottom of the respective cold water plena in order to create a common cold water plenum.

The choice of two connection levels makes natural circulation possible in case of temperature difference between cold plena. If the normal decay heat removal route (i.e. the active steam/water system) is lost, the uninsulated wall portion of the Pressurizer would thus help removing by conduction the decay heat towards the Plant Pool.

Conveyed water to and from each vessel, belonging to a common cold water plenum, does not significantly contribute to the thermal loadings on the pressure boundary during transients.

Air Coolers

Two finned-tube Air Coolers are arranged in loops in natural circulation.

Each Air Cooler is rated 1 MWth at 30 °C ambient air and 95 °C pool water inlet temperature.

The onset of natural circulation occurs every time the pool water temperature becomes higher than the ambient air temperature.

Operation of the air coolers would prevent, for unlimited time, the pool water from boiling, in case of long-term loss of the operational decay heat removal system.

The technology of the air coolers in natural circulation is derived from the design and operating experience of ANSALDO in the field of the LMFBRs.

5. FULL-POWER OPERATION OF THE ISIS REACTOR

During normal operation the hot/cold water interface level in the Lower Density Lock is maintained constant by varying the speed of the Primary Pumps.

Any rising of the interface level is counteracted by an increase of the pump motor speed. Any lowering of the interface level is counteracted by slowing down the pump speed.

Dynamic Analysis of ISIS Control System is in progress. Preliminary results, not yet published, confirm that the reactor power is controlled by the concentration of boron in the primary water and by the intrinsic negative feedback of the core.

6. NATURAL BEHAVIOUR OF THE ISIS REACTOR UNDER ACCIDENT CONDITIONS

In the design of ISIS emphasis has been put in the prevention of core damaging accidents.

The two main safety functions, reactor shutdown and decay heat removal, are performed without recourse to the usual sensor-logic-actuator chain, i.e. with no inputs of "intelligence", nor external power sources or moving mechanical parts, according to the definition of Category B Passive Components (ref. /2/).

An active Reactor Protection System, aimed at anticipating passive system interventions, is included in the design, but is not credited in the safety analysis.

As anticipated in the Reactor System Description, mixing of the Primary Water with the Intermediate Water and the consequent natural heat transfer toward the Reactor Pool is the basic feature to assure safety under Design Basis Accidents such as Loss Of the Station Service Power and Loss of Heat Sink (ref. /3/).

During these DB Accidents the pressure boundary integrity assures the availability of water to cool the core and to transfer the decay heat to the Reactor Pool.

In case of LOCA Accidents, the Core shutdown and cooling functions are possible only if a sufficient inventory of water remains available.

The design features of ISIS guarantee the availability of this water because of the prompt self-depressurization of the system which is the consequence of the same hot-cold water mixing process.

To illustrate the effectiveness of this self-depressurization capability, the two following DB Accidents are presented :

- double ended break of the lower pipe connection between RPV and Pressurizer;
- Steam Generator tube rupture.

Additionally, the extremely fast transient following an hypothetical break at the bottom of the RPV is reported as an exercise to better understand the thermalhydraulic phenomena linked to the self-depressurization.

All transient analyses have been carried out using the RELAP5 computer code, with a nodalization made up of 256 control volumes 262 flow junctions and 78 heat structures; neutronic point kinetics has been used to evaluate the power in the core.

Loss Of Coolant Accident

This accident consists in a double ended break of the lower, 150 mm nominal diameter line connecting the RPV and the Pressurizer. This accident scenario has been chosen because this is the largest line of the pressure boundary and also because the break location is far from both Density Locks, thus worsening the loss of cold water from the vessels (ref. /4/).

Considering that, the break location is 25 m below the Reactor Pool water level, the absolute pressure at the break outside the RPV is 3.5 bar. No action is credited of any active protection or control system.

When the accident starts, interconnected thermalhydraulic phenomena occur simultaneously within both RPV and Pressurizer. Cold water outflows from both RPV and Pressurizer; hot primary water replaces the losses in the Intermediate Plenum through both Density Locks. This phase lasts about 2-3 seconds. Then flashing hot water causes Primary Pumps cavitation which, in turn, allows the inlet of intermediate water into the primary system and the Core via the Lower Density Lock with a quick decrease of generated power.

The Reactor behaviour can now be explained considering that the Primary Pumps remain cavitating all over the transient and the primary system behaves like two channels hydraulically connected in parallel.

Both channels, the one made up by the Core and the Riser, and the second by the Downcomer and the SGU, are alternatively flooded by intermediate water entering the primary system through the Lower Density Lock.

Self-depressurization of the system takes place mainly because of the following two water mixing effects (fig. 3):

In the RPV, hot primary water flowing from the Upper Density Lock mixes up with the large volume of cold intermediate water of the RPV Head, purposely provided for this function.

In the Pressurizer, hot water flowing down through the vertical pipes mixes up with the large volume of cold intermediate water underneath.

The system pressure at the break equals the external pressure in about 450 seconds.

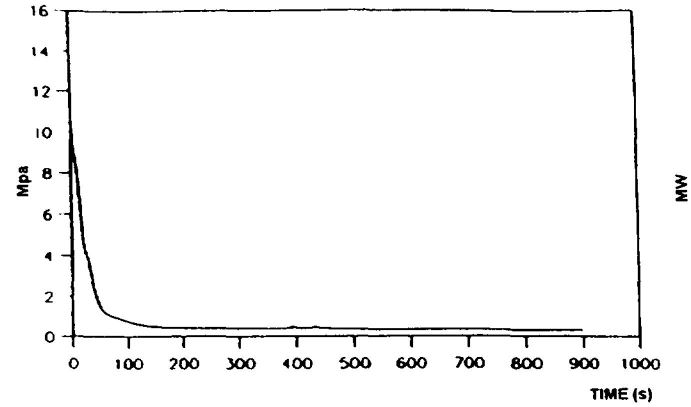


Fig. 3 - LOCA
Core pressure

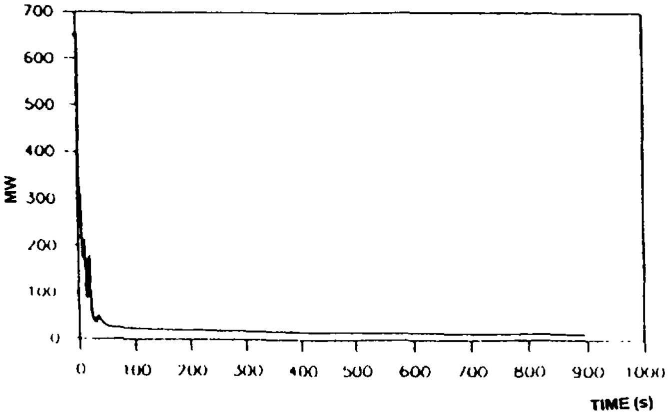


Fig. 4 - LOCA
Nuclear power

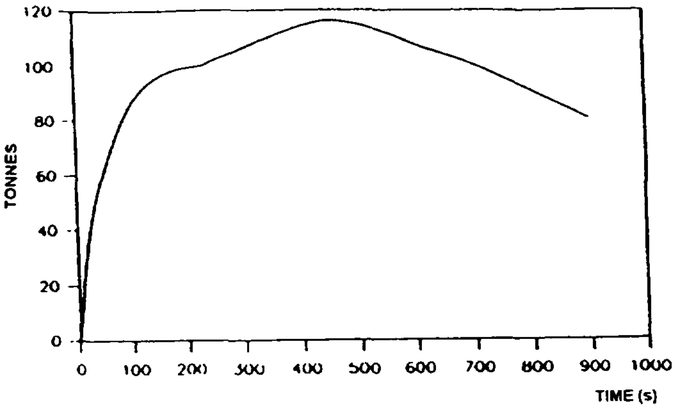


Fig. 5 - LOCA
Cumulated water loss

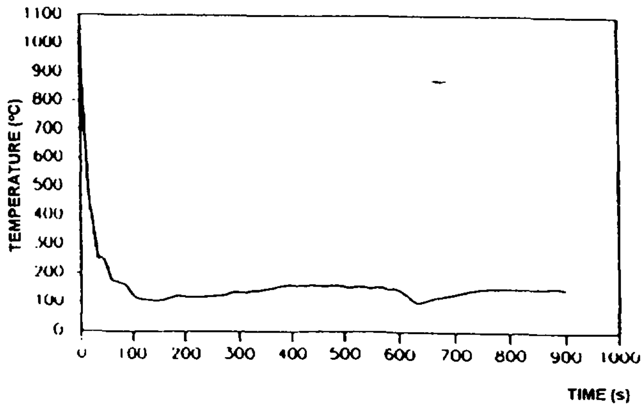


Fig. 6 - LOCA
Maximum temperature of average fuel rod

At this moment the RPV water stops flowing out and reversal flow of cold, high-boron water from the Reactor Pool sets on.

The core is shutdown (fig. 4) by intermediate water entering through the Lower Density Lock.

Figure 5 shows that the maximum cumulated amount of water loss is less than 120t (approximately 25% of the initial inventory) and only the following regions of the Reactor Module remain temporary uncovered:

- the Head of the RPV (with the water level always remaining above the Upper Density Lock);
- the Pumps, the upper part of the Riser and the SGU;
- the hot region of the Pressurizer.

Later on in the transient, reversal flow from the Reactor Pool starts recovering the water level in the RPV; at the end of the computer run (i.e. after 900 seconds) about 40 t of water have already entered the RPV from the Reactor Pool.

During the transient the Core never uncovers or heats up as shown in Figure 6. The maximum temperature of the "average" fuel rod has remained lower than at nominal conditions. A similar behaviour is shown for the clad surface temperature.

Steam Generator Tube Rupture

In this accident a break of 10 cm^2 cross section located at the connection between SGU tubes and steam headers is simulated; the break size is approximately equivalent to the cumulated cross sections of 8 SGU tubes.

No credit has been taken for action of active systems that can mitigate the consequence of the [1] accident, but for the Primary Pumps Speed Control System which delays the inlet of highly borated water through the Lower Density Lock. The steam pressure and the feedwater flow rate are assumed accordingly to remain constant during the transient[1]

When the accident occurs, water from the primary system enters the SGU ruptured tubes at a max mass flow rate of 96.5 Kg/s.

An equal amount of intermediate water enters the primary system through the Upper Density Lock as long as the Primary Pump Control System is capable to control the hot-cold interface level in the Lower Density Lock. Primary water with increasing boron concentration enters the core and reduces the generated power (fig. 7).

The amount of intermediate water entering the Upper Density Lock is replaced in the RPV by water leaving the Pressurizer. In the Pressurizer itself fast depressurization takes place because of the hot-cold water mixing process already explained above for the LOCA transient.

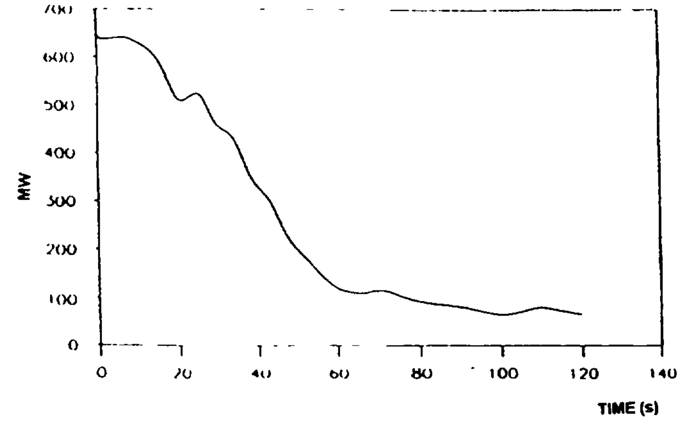


Fig. 7 - Steam Generator Tube Rupture
Nuclear power

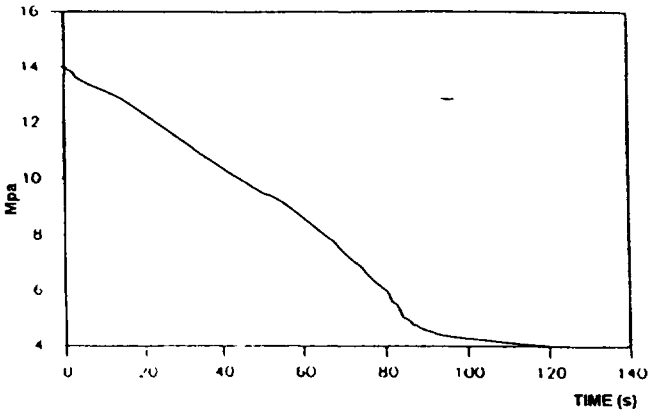


Fig. 8 - Steam Generator Tube Rupture
Core pressure

Both effects of reduced core power with associated lower primary water temperature and Pressurizer self-depressurization reduce the overall primary system pressure (fig. 8) down to the secondary system pressure (tube-side SGU pressure) which has been assumed to remain at its nominal value.

At this time the primary water stops flowing into the SGU tubes. Figure 9 shows that the cumulated amount of water loss is less than 8 tonnes which corresponds to the inventory of the hot water in the pressurizer.

The curve of the fuel temperature shows a steadely decreasing pattern, fig. 10.

---

[1] Crediting the SGU isolation, the transient would behave very similar to the transient of loss of heat sink which has been shown to cause a fast reactor shutdown (ref. /4/)

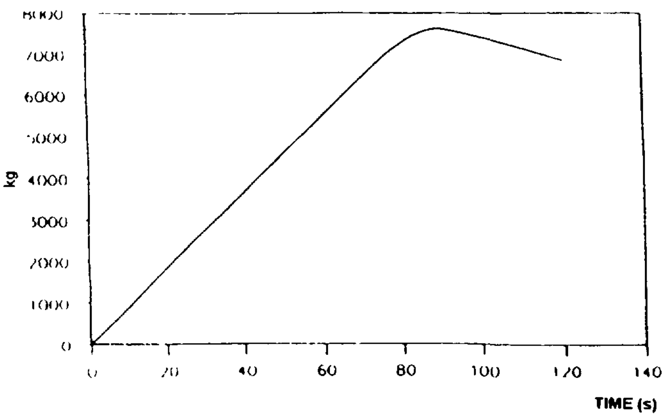


Fig. 9 - Steam Generator Tube Rupture
Cumulated water loss

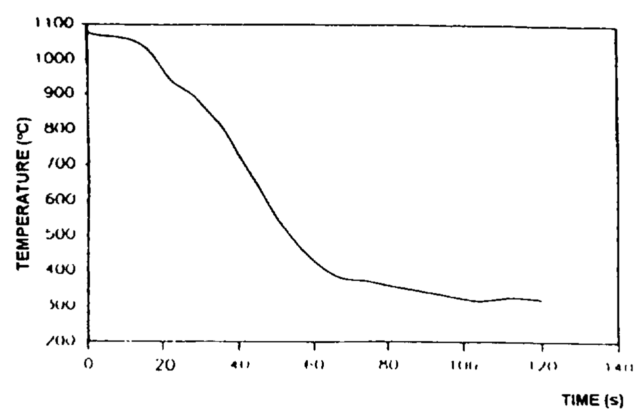


Fig. 10 - Steam Generator Tube Rupture
Maximum temperature of average fuel rod

Break at the bottom of the Pressure Vessel

In this exercise an hypothetical break of 500 cm^2 cross section has been assumed to occur at the bottom of reactor pressure vessel; this accident scenario is arbitrary and imagined to generate a very severe thermalhydraulic transient; in fact the break is positioned at the lowest location of pressure boundary and therefore has the potential of completely emptying the RPV. This exercise is intended to demonstrate that the self-depressurization process can avoid the uncovering of the core even in this case. No protection or control systems, no any other active system was credited during the accident analysis.

When the transient starts, there is a large blowdown of intermediate water from the RPV and Pressurizer into the Reactor Pool (the initial mass flow rate through the break is about 7000 kg/s). The escaping flow rate is fed by displaced primary water which is mostly contained in the SGU. Primary water leaves the SGU from the bottom via the Downcomer and, after few seconds, also from the top via the Primary Pumps and Upper Density Lock.

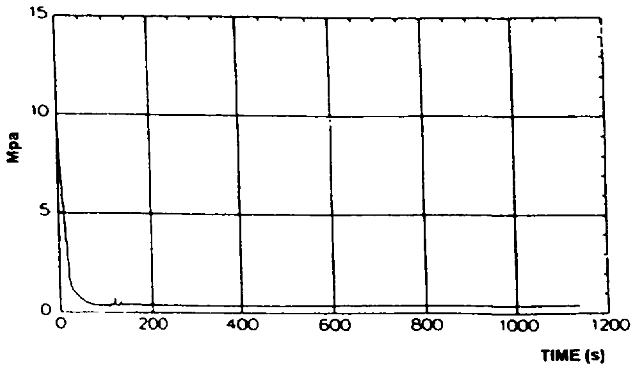


Fig. 11 - Break at the bottom of the RPV
Core pressure

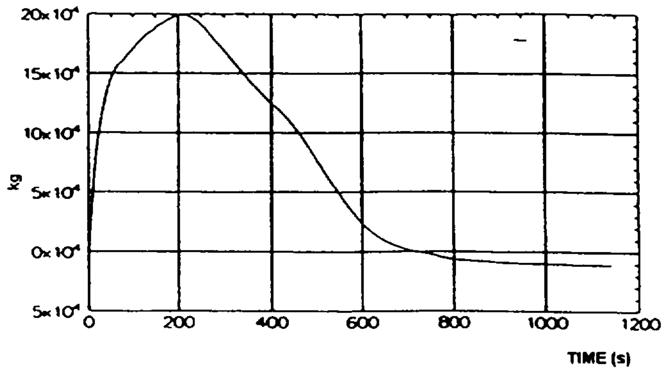


Fig. 12 - Break at the bottom of the RPV
Cumulated water loss

At the very begining of the transient the water flowing down through the Downcamer splits in two streams: the one leaves the Inner Vessel through the Lower Density Lock and the second flows up through the Core, the Riser and leaves the Inner Vessel through the Upper Density Lock. The reactor core is continously fed by primary water flowing upwards and its temperature is continuously decreasing because it is kept cooled since the beginning of the transient.

At the time of about 7 seconds, with Primary Pumps in cavitation, the primary water stops leaving the Inner Vessel through the Lower Density Lock and a reversal flow of intermediate water sets on flooding the core.

At this moment the usual way of natural circulation of ISIS reactor is recovered and the primary system fed with cold and borated water.

The mixing of cold and hot water initiates the self-depressurization of the system in the way described before for the case of LOCA (fig. 11).

The system continues its depressurization up to the time of about 200 seconds when its pressure drops below the Reactor Pool pressure at the break location.

At this moment, the total mass of displaced water (figure 12) is less than 200 tonnes (approximately 50% of the total inventory of one module) and the RPV has been emptied only down to about the center line of the SGU.

After about 1000 seconds, the initial water inventory is completely recovered and the reactor is in the state of stable cold shutdown.

The evolution of the generated power is shown in fig 13; the power reduction during the first 7 seconds is caused by the void effect associated to the depressurization and the following shutdown is assured by the borated water.

The fuel temperature steadily decreases as shown in fig. 14 and 15.

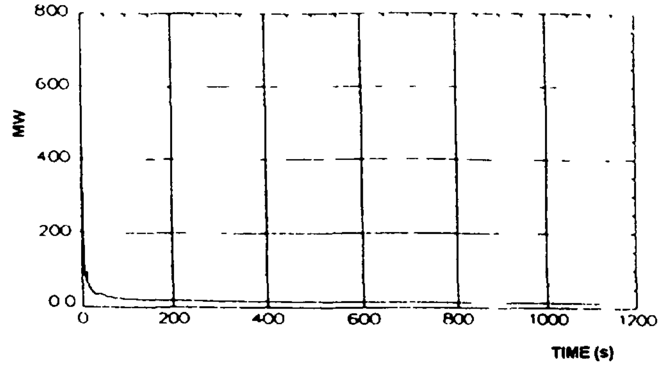


Fig 13 - Break at the bottom of the RPV
Nuclear power

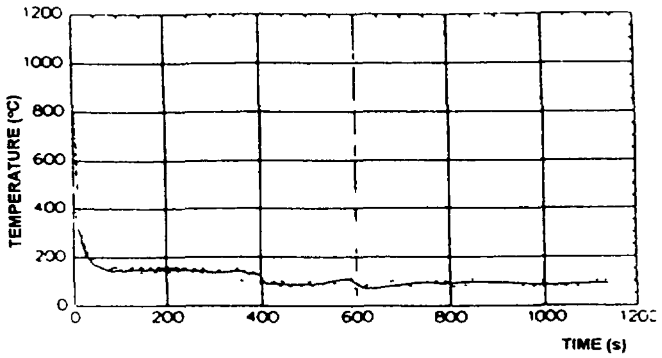


Fig. 14 - Break at the bottom of the RPV
Maximum temperature of average fuel rod

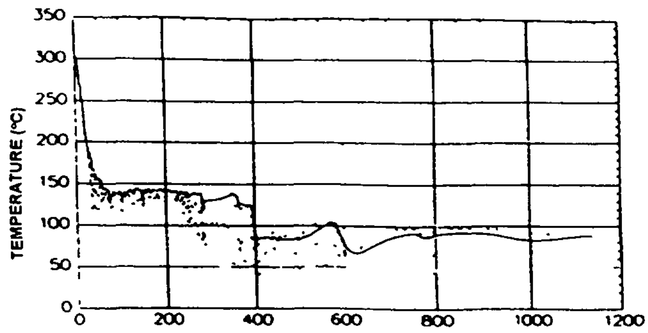


Fig. 15 - Break at the bottom of the RPV
Clad surface temperature at different elevations

7. MODULAR PLANT

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

The selected unit power of the ISIS Reactor Module (200 MWe) is consistent with this design target.

Layout studies of the ISIS power plant are in progress in ANSALDO, to optimize component arrangement and reduce the erection time of the Reactor Modules and of the Balance of the Plant.

8. ECONOMIC ASPECTS

At the beginning of the development of the ISIS concept (about seven years ago) it seemed reasonable to foreseen a moderate increase of the cost of the fossil fuels in the near future which would have improved the economic competitiveness of nuclear energy. Today, instead, two facts worsen this competitiveness:

- the fossil fuels price has remained low and stable,
- the efficiency of the modern electric energy generating fossil fuelled power plants has importantly increased.

The importance of the second fact is such that it will drastically affect the energy market, in particular the market of nuclear energy.

In the past, the efficiency of electricity production of the nuclear power plants was similar to that of the conventional power plants. Under that condition it was profitable to generate electricity by the large-size nuclear power plants that dominate the nuclear panorama.

Today, the efficiency of the modern Combined Cycle Turbo-Gas (CCTG) Power Plants has exceeded 50% and in the near future (before the year 2000) will reach and perhaps trespass 60%, while the efficiency of the nuclear water reactors stagnates at about 33%.

The higher efficiency of the CCTG will have two main consequences in the energy market.

The first consequence is that, at stable fossil fuel cost, the cost of electricity will be reduced while the cost of heat will remain substantially stable.

The capital cost of the nuclear power plants, at stable O&M and nuclear fuel costs, should be reduced to maintain the same level of competitiveness.

Fig. 16 shows how much the capital cost of a nuclear power plant would have to be reduced in the range of 50 to 60 % electrical generation efficiency of CCTG considered at stable capital cost.

The second consequence is that the fraction of power that can be extracted at low cost as useful heat for district heating or industrial use from a modern fossil fuelled cogenerative power plant reduces with the increasing efficiency in electricity generation.

A balanced mix of nuclear and fossil-fuelled plants can help to achieve the optimum ratio of thermal to electric energy generation for sites with high heat demand.

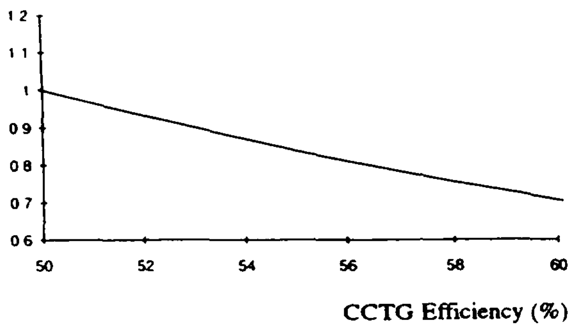


Fig 16 - Reduction of the capital cost of a nuclear power plant needed to maintain the competitiveness with the modern CCTG plants

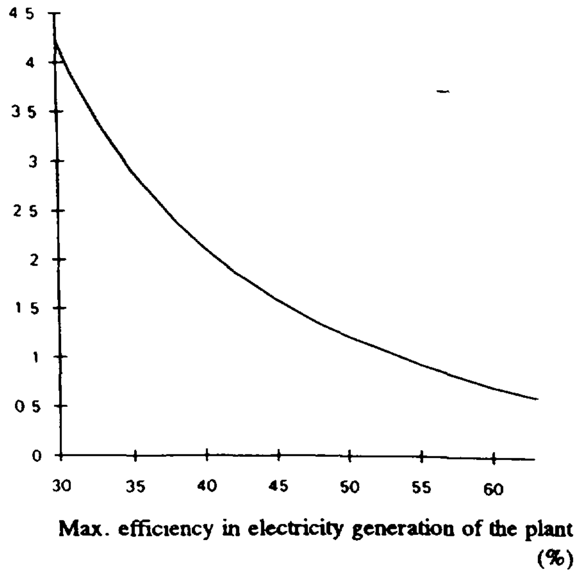


Fig. 17 - Ratio of the low cost heat to electric output of a cogenerative fossil fuelled plant

Fig.17 gives the fraction of the thermal energy that can be extracted at low cost from a fossil-fuelled plant vs the max. efficiency in electricity generation of the same plant when used only for electricity generation.

A preliminary economic evaluation carried out comparing 60% efficient CCTG, co-generative CCTG, conventional boilers and nuclear power plants, has shown that nuclear power plants could recover part of their economic attractiveness if used as co-generating or as thermal power plants.

The co-generative use appears attractive from 3000 hours/yr. upwards. That means an increase of capital cost of the economically viable nuclear plant of more than 50% over the capital cost needed for competitiveness with the CCTG plants in case of electric energy generation only, in other words an increase of the value of the cogenerative plant in the order of more than 50 %.

The increase of the value of the nuclear plant can even exceed 100% for specific site conditions where heat can be used during the most part of the year.

An obvious condition for interest of a prospective utility in a co-generative nuclear plant is that an adequate reactor design exists, that, besides featuring public-acceptable characteristics of radiological safety, be designedto overcome the unfavourable scale-effect on cost of downsizing, because the thermal power needed is in the order of hundreds of megawatts against the thousands of megawatts available from the today large nuclear reactors.

In the view of a designer, the smaller reactor can be competitive, in spite of downsizing, provided that:

- the number of the active safety related systems of the larger plants is strongly reduced,
- the mass of steel to installed power ratio is not significantly increased,
- the operation & maintenance costs do not become excessive.

The ISIS reactor has been designed to cope with these requirements.

The technical features and the results of preliminary analyses for an use of ISIS as co-generating reactor can be summarised as follows:

- no active systems are necessary to assure safety. All active safety systems can be eliminated.
- the specific mass of steel of the ISIS NSSS is comparable to that of the large modern PWRs. This is possible also because of the milder operating conditions of a reactor designed for co-generation (e.g., lower operating pressure).

Ongoing studies explore furthermore the possibility of reducing operating & maintenance costs, taking profit of the predicted simple operation of ISIS and of the modular approach that makes possible to share facilities, such as the fuel and component handling equipment, for servicing identical reactor modules of a multi-module ISIS NPP.

9. CONCLUSION

The ISIS is an innovative Nuclear Power Plant under development in ANSALDO. It is based on original ideas derived by ANSALDO experience on proven LWR and LMR technologies.

The main features of ISIS are as follows:

- Outstanding passively safe behaviour of the Reactor, which means core shutdown and cooling functions ensured in all accident conditions and no release of primary coolant outside the Reactor Building.
- Compact reactor layout and modular fabrication, made possible by the integrated design of the primary circuit.
- Flexible reactor concept for electricity generation or combined generation of heat and electricity, made possible by its modular solution and low cost sensitivity to downsizing.

REFERENCES

/1/ *PIUS, the forgiving reactor by ASEA-ATOM,* Modern Power System (October 1985)

/2/ INTERNATIONAL ATOMIC ENERGY AGENCY, 1991 - *Safety related terms for advanced nuclear plants* - IAEA - TECDOC - 626, Vienna.

/3/ AMATO S., MONASTEROLO U., MONTI R., ORAZI A., 1991 - *Response of the Inherently Safe Immersed System (ISIS) Reactor to Accident Conditions* - Third Int. Sem. on SMNRs, New Delhi.

/4/ AMATO S., ORAZI A., 1993 - *Advanced Safety Features in Conception of New Passive Reactors: the Inherently Safe Immersed System (ISIS) Reactor* - IAEA Technical Committee Meeting on Thermohydraulic of Cooling Systems in Advanced Water Cooled Reactor, Villigen, Switzerland.

/5/ CINOTTI L., RIZZO F.L., 1993 - *The Inherently Safe Immersed System (ISIS) reactor* - Nucl. Eng. Des. Vol. 143; pp. 295-300.

/6/ AMATO S., CINOTTI L., 1994 - *Using prompt self-depressurization as a key feature to assure safety* - Int. Conf. on New Trends in Nuclear System Termohydraulics, Pisa, Italy

XA9846724

DESIGN AND SAFETY ASPECT OF SMALL LEAD-/LEAD-BISMUTH-COOLED FAST REACTORS

H. SEKIMOTO
Research Laboratory for Nuclear Reactors, Tokyo Institute of Technology,
Tokyo, Japan

Abstract

A conceptual design study and accident analysis of small long-life nuclear power reactors used for remote or isolated area has been performed. Lead as well as lead-bismuth is employed as its coolant, and both metallic and nitride fuels are investigated. There are some severe requirements on these reactors for operability, maintainability, safety and proliferation resistance. Some important characteristics of the proposed designs (150MWt) are: transportability between reactor factory and operation site; capability of long-life operation (12 years) without refueling or fuel shuffling while maintaining burnup reactivity swing less than 0.1%Δk; omission of intermediate heat exchanger; relatively large contribution of natural circulation; negative total core coolant void coefficient of reactivity over all burnup period.

These two coolants require quite different operating temperature, and the lead-bismuth works in a wider range of operating temperature. The problem caused by ^{210}Po produced from lead-bismuth is not significant, but may be even desirable from nuclear-proliferation resistance point. For the lead-bismuth coolant, the amount of resources as well as price may become problem.

All of the reactors proposed in this study can survive the UTOP, ULOF, UTOP-ULOF and UTOP-ULOF-ULOHS accidents without any help of operator or active devices. All reactivity feedback components are negative for the above accidents. For the nitride fuel reactors, the Doppler effect plays important role during the unprotected accidents. On the other hands, for the metallic fuel reactors the axial fuel expansion plays important role. The radial core expansion is important for both metallic and nitride fuel reactors. The most severe criterion for the accidents is the maximum permissible temperature of cladding for metallic fuel. The margin of this temperature is small for the UTOP-ULOF-ULOHS accident. The lead-bismuth cooled reactors give better performance for the severe accidents compared to the lead cooled reactors due to their lower operating temperature.

1. INTRODUCTION

In the 21st century, the energy demand in developing countries and local areas is supposed to increase drastically. We can not expect in such places either many skillful operators and technicians or good infrastructures. In such cases small reactors are usually more desirable than large reactors. They are considered as power reactors providing better fit in terms of grid size and demand growth rate, or other industrial projects.

The small nuclear power plants proposed for the use in relatively isolated area, however, should satisfy some severe important requirements such as easy operation, easy maintenance, easy construction and decommissioning, inherent/passive safety, and nuclear-proliferation resistance. Such kind of nuclear power plant may be built in a well-developed country, transported to the target area, operated there up to its end of life (EOL) without refueling or fuel shuffling, and then transported back to the country of its maker.

The transportation requires the reactor compact. The capability of long life operation with neither refueling nor fuel shuffling means easy operation and maintenance. It means also that refueling equipment is not required in reactor site.

The inherent/passive safety feature means that during any kind of abnormal conditions the reactor can change its own power level to the decay heat level without any help of active devices or operator and without causing coolant boiling or fuel melting. In order to improve safety performance of the liquid metal cooled fast power reactor, among important options are reducing excess reactivity during burnup to be smaller than β_{eff} and reducing positive coolant dilatation and void coefficients of reactivity or even making them negative.

In the recent few years there are many efforts for satisfying these requirements. Included in these categories is minimizing excess reactivity during burnup up to few tens % of β_{eff} for about 1 year operation of sodium cooled LMFBR such as PRISM and SAFR. However these reactors still have positive coolant dilatation and void coefficients, though their overall reactivity coefficients are negative. There are several efforts to reduce or make negative coolant dilatation and void coefficients of sodium cooled fast reactor by increasing core surface or by inserting moderator or absorber material. However, in such a case generally its reactivity swing becomes larger. This excess reactivity is controlled usually using conventional control rod. Designing the long life fast power reactor, which can be operated until EOL without refueling and fuel shuffling with very small excess reactivity change (below β_{eff}) during burnup and negative coolant dilatation and void coefficients, is much more difficult to be attained especially for sodium cooled fast reactors.

In the present study some conceptual designs of long life small power reactor are proposed, which does not require either refueling or fuel shuffling during operation and can overcome the above problems of excess reactivity and coolant dilatation and void coefficients. Lead and lead-bismuth are used as a reactor coolant instead of sodium, and both metallic and nitride fuels are investigated. General overview of the proposed design and main design parameters are shown in Fig.1 and Table I, respectively.

In this design the intermediate heat exchanger is eliminated, so that heat from the primary coolant system is directly transferred to the water-steam loop through steam generator. The reactor core, pump, and steam generator are arranged to achieve a compact design. The coolant flows from the cool pool into the pump, enters the core through orifice block, and flows to the hot pool after removing heat from the core. From the hot pool it flows to the steam generator, transferring the heat into water/vapor side and goes back into the cool pool. From the cool pool the coolant is pumped back to the core.

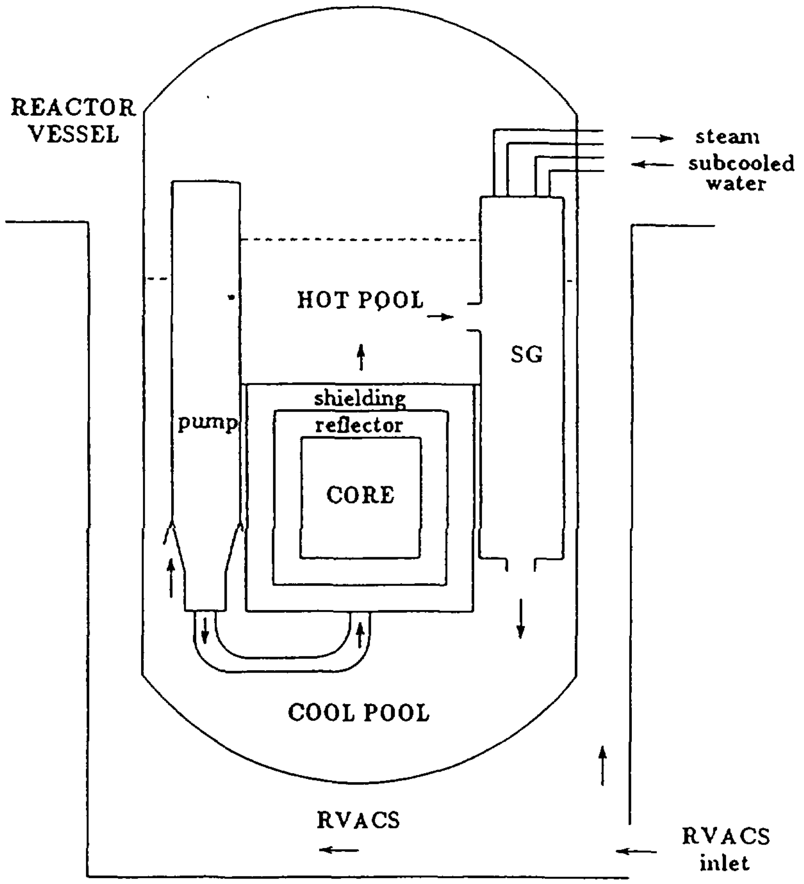


FIG. 1. Design overview of the proposed small long-life power reactor, showing aspects of the steam generator (SG) and reactor vessel auxiliary cooling system (RVACS).

TABLE I. Main Reactor Design Parameters

Reactor power (MWt)	150
Lifetime (years)	12
Volume of internal blanket (m^3)	1.5
Fuel	U-Pu-10%Zr metallic fuel, or UN-PuN nitride fuel
Shielding material	B_4C
Structural material	HT9/SS316
Pin pitch/diameter	1.2
Pin diameter (cm)	1.0
Cladding thickness (mm)	0.8
Steam generator	
Inlet water temperature (°C)	225
System pressure (MPa)	7
Secondary flow rate (kg/s)	60-70
Height (m)	4.0
Pipe diameter (cm)	2.5-2.2
Reactivity swing (%*dk/k*)	<1.2

2. DESIGN CONCEPT

In the present design excess reactivity along burnup is minimized by the following method[1-3]. The core is divided into three regions (central, middle and outer cores). Enough amount of fertile material is inserted in the central core in order to compensate the reactivity decrease along burnup. The middle and outer cores work as drivers filled with fuels of different plutonium enrichments. The plutonium enrichment is higher in the outer core. Relatively high fuel volume fraction (45-50%) is employed in order to get high internal conversion ratio and also compact core design. In radial direction, a coolant region is put between the outer core and shielding. This region is also prepared for insertion of control rod. The width of each region was adjusted to minimize the excess reactivity and coolant void coefficient during burnup. As burnup proceeds, accumulation of plutonium in the central core raises power density in this region, but the power peak still remains in the middle core.

Charging the central core with fertile material also improves the coolant dilatation and void coefficients, since the most active component locates in the outer core resulting in the higher neutron leakage for coolant dilatation or voiding. The use of lead and lead-bismuth as coolant results in relatively hard neutron spectrum which also facilitates negative reactivity coefficient of coolant dilatation and void.

Operation of the reactor up to the end of life (EOL) without refueling has some consequence on the thermal hydraulic system design. Since the power density pattern changes during burnup, the coolant outlet temperature distribution also changes with burnup. However, in the present design, by using small adjustment in the orificing system, the coolant flow-rate distribution is adjusted to make the coolant outlet temperature peaking factor as small as possible for both BOL and EOL conditions.

The melting points of lead and lead-bismuth are 327 and 125℃, respectively, then their operation temperatures are chosen to be around 400-600℃ and 250-500℃, respectively. The lower operation temperature for the lead-bismuth coolant means not only easier operability but superior corrosion resistance for the lead-bismuth cooled reactor.

The use of lead-bismuth as coolant may result in production of ^{210}Po which is a strong alpha emitter. Study on this problem[4] showed that radiation level due to gamma activity in the core is more significant and tends to impose greater restrictions on accessibility to the core. However the present reactors are designed so that no access to the core is required until the EOL and in case of troubles they are brought back to the maker. It may be even desirable from nuclear-proliferation resistance point.

For the lead-bismuth coolant, the amount of resources as well as price may become problem.

3. CALCULATION RESULTS FOR STATIC ANALYSIS

TABLE II. Sample Design Parameters

Case	Coolant[a]	Fuel[b]	Width[c] (cm)				Burnup (%HM)	
			Inner core		Middle+outer core		Average	Peak
			Radial	Axial	Radial	Axial		
A	Pb	Mt	36.3	44.4	44.4	29.9	6.0	9.45
B	Pb	N	33.5	45.0	44.2	28.2	6.27	10.3
C	Pb-Bi	Mt	34.1	45.2	44.8	29.9	5.87	9.33
D	Pb-Bi	N	34.0	45.4	43.8	28.1	6.26	10.1

[a] Pb: lead, Pb-Bi: lead (44.5%)-bismuth (55.5%).
[b] Mt: metallic, N: nitride.
[c] Axial reflector width 6.0 cm, radial reflector width 10 cm.

In the present study, eight group diffusion calculation and one group burnup calculations were performed for two dimensional *r-z* geometry. Their group constants were obtained from cell calculations with SLAROM code. Cross sections for burnup calculation were obtained by collapsing the group constants using neutron spectrum in each spatial mesh. Diffusion calculation was performed every year, and renormalization of the neutron flux by reactor power every month. In this paper four cases are discussed, whose design parameters are shown in Table II.

It has been shown[1] that this concept enables negative total core coolant void coefficient over reactor life with less than 0.1%Δk of reactivity swing. These results show highly passive safety feature of the proposed nuclear plant.

The temperature reactivity feedback coefficients of Doppler effect, coolant dilatation, core radial expansion and fuel axial expansion for each type of core at the BOL and EOL are shown in Table III. The nitride fuel gives more negative Doppler coefficient than the metallic fuel. On the other hand the coolant density coefficient of metallic fueled cores is more negative than the nitride fueled cores. The metallic fuel gives more negative reactivity feedback coefficient of the axial fuel expansion. The feedback coefficients of the radial expansion are the most negative for all designs.

Comparing the lead cooled and lead-bismuth cooled designs, the lead-bismuth cooled designs in general give slightly more negative Doppler coefficient and more negative coolant density coefficient, but they give almost similar results for fuel axial expansion coefficient and core radial expansion coefficient. Average enrichment of the fuel is slightly different for each design, and makes the Doppler coefficient slightly different. The difference of coolant density coefficients is mainly attributed to the difference in the scattering cross sections. More detail explanations are found in Ref. [3].

TABLE III Reactivity Feedback Coefficient at the BOL and EOL

Time	Case[a]	Reactivity coefficient (10^{-6} ℃$^{-1}$)				neutron generation time (10^{-7} s)	β_{eff} (10^{-3})
		Doppler	Coolant density	Radial Expansion	Axial expansion		
BOL	A	-2 26	-1 18	-9 69	-3 55	1 83	4 46
BOL	B	-5 42	-0 73	-8 57	-1 35	2 25	4 37
BOL	C	-2 67	-1 77	-9 50	-3 47	1 82	4 47
BOL	D	-5 66	-0 94	-8 69	-1 40	2 23	4 37
EOL	A	-2 13	-0 77	-9 02	-3 01	1 88	4 24
EOL	B	-5 02	-0 20	-7 69	-1 05	2 33	4 28
EOL	C	-2 64	-1 35	-8 79	-2 98	1 87	4 25
EOL	D	-5 26	-0 41	-7 74	-1 08	2 23	4 14

[a] See Table II

Coolant void reactivity coefficient was calculated for the case that all coolant in the core regions is completely voided and the coolant in the other regions is in normal condition (not voided) The obtained coefficient is negative over whole burnup period for all designs The metallic fuel gives better coolant void reactivity coefficient than the nitride fuel, and lead-bismuth cooled reactors give slightly better coefficient than the lead cooled reactors

Effective delayed neutron fraction is larger for these designs than the conventional sodium cooled FBR due to lower enrichment and harder spectrum so that contribution from ^{238}U fast fission becomes more important

The power density of the proposed designs is much lower than conventional LMFBR, and fuel pin pitch is taken to be large Therefore the contribution of the natural convection is about 30-40% of the total circulation

4 CALCULATION RESULTS FOR ACCIDENT ANALYSIS

In this chapter we discuss the safety performance of these reactors during hypothetical accidents, ULOF, UTOP and ULOHS accidents and their simultaneous accidents The accident conditions are given in Table IV Mathematical models of both thermal-hydraulic and neutron kinetic calculation are given in Ref [7] with some discussions

4 1 ULOF accident at the BOL

Fig 2 shows the hot spot temperature change during the ULOF accident at the BOL for the design cases A through D The maximum allowable temperature is different between the nitride and metallic fuels The melting temperature for nitride fuel pellet is about 2500℃ while for metallic fuel is about 1000℃ The coolant boiling temperature is 1740℃ for lead

TABLE IV. Accident Conditions

ULOF	Loss of primary loop pumping power at t=0, all primary loop pumps start to coast down, pump at coast-down half-time of 12 s
ULOHS	Loss of secondary loop pumping power at t=0, all secondary loop pumps start to coast down, pump at coast-down half-time of 12 s
UTOP	Complete withdrawal of all control rods from normal operation at t=0, the withdrawal finishes in about 15 s
UTOP+ULOF	Simultaneous UTOP and ULOF accident
UTOP+ULOF +ULOHS	Simultaneous UTOP, ULOF and ULOHS accident

and 1670°C for lead-bismuth. The maximum allowable temperature for cladding is the most critical. For the metallic fuel, it is desirable that cladding temperature does not exceed its eutectic temperature so that there will be no interaction between fuel and cladding. Previously the allowable cladding temperature was conservatively set to be about 725°C. However the recent results [8,9] show higher than 800°C for U-Pu-10%Zr metallic fuel. For the nitride fuel

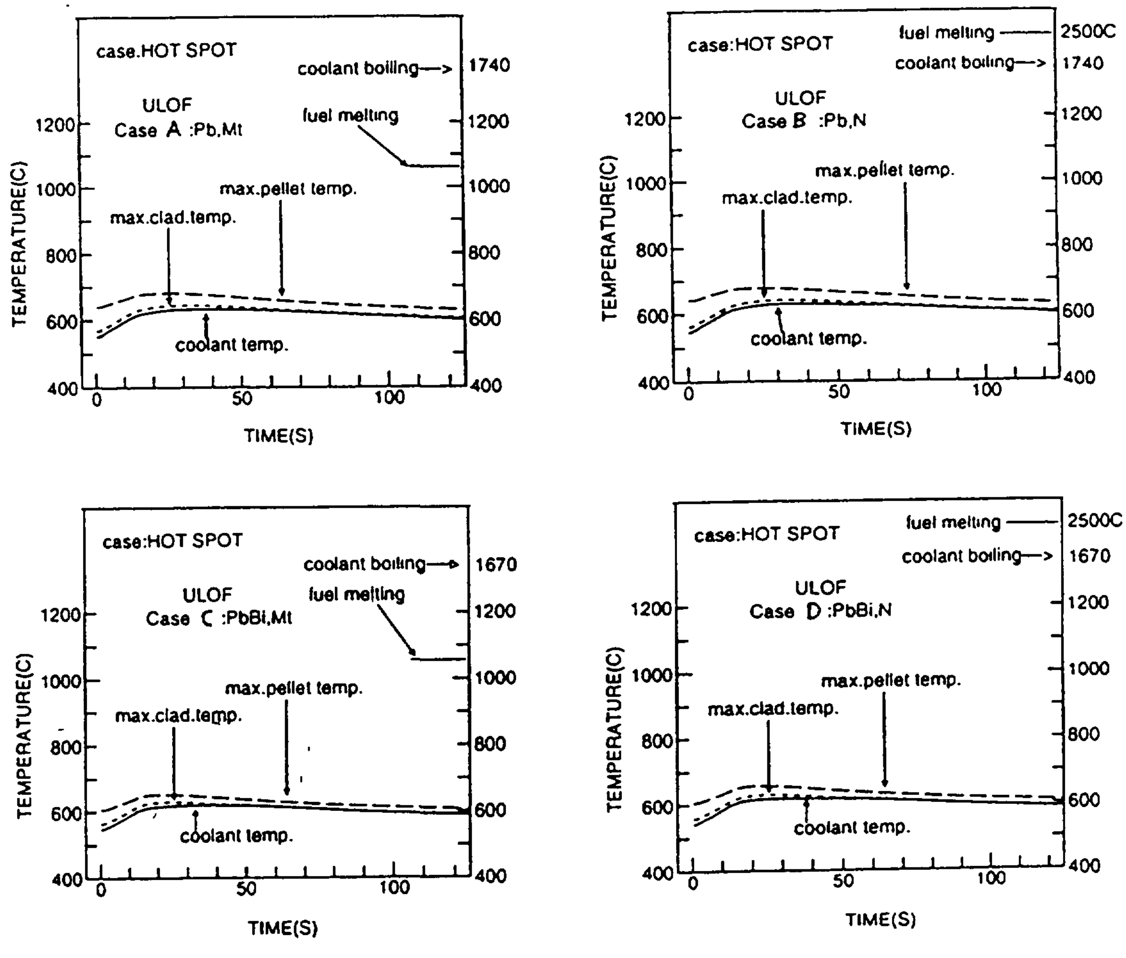


FIG. 2. Hot spot temperature during ULOF accident at BOL for design cases A to D.

this limit can be expected to be higher because of better compatibility between nitride fuel and cladding material [9]. Under the ULOF accident caused by the failure of primary pump system, the reactor can reduce its own power level into natural circulation level without exceeding any temperature limits for cladding, fuel pellet and coolant. The coolant temperature margin to the boiling temperature is large, and also the fuel pellet temperature margin from melting is large especially for nitride fuel.

Fig. 3 shows the changes of peak and average coolant temperatures, hot pool temperature, and cool pool temperature during the ULOF accident for the design case D. In general, the maximum temperature of coolant occurs after about 40s, the temperature increase in the hot pool is much faster than the cool pool. This behavior is about the same for other design cases.

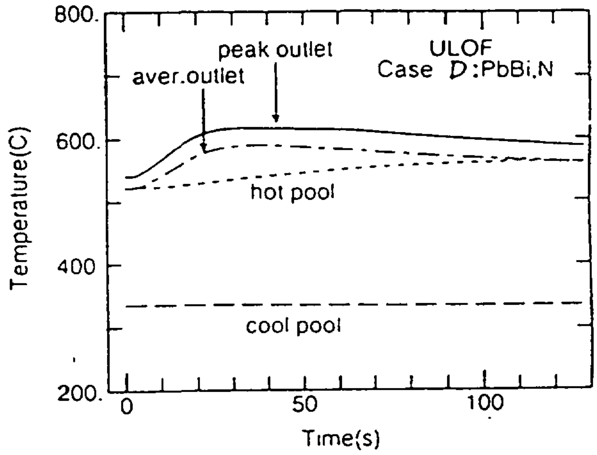


FIG. 3. Coolant temperature at the core outlet, hot pool and cool pool during ULOF accident at BOL for design case D.

4.2. UTOP accident at the BOL

Fig. 4 shows that the temperature limits for cladding, fuel pellet and coolant are not exceeded under the UTOP accident at the BOL. And as in the ULOF accident, the coolant temperature margin to the boiling temperature is large, and also the fuel pellet temperature margin from melting is large especially for the nitride fuel.

Fig. 5 shows the change of peak and average coolant temperatures, hot pool temperature, and cool pool temperature for the design case D during the accident. The coolant temperature increases rapidly until all control rods are withdrawn. After then the rate of temperature increase becomes lower. This pattern can be understood from the pattern of the reactor power change during the accident. On the other hands, the temperatures of hot pool and cool pool increase slowly. This behavior is about the same for other design cases.

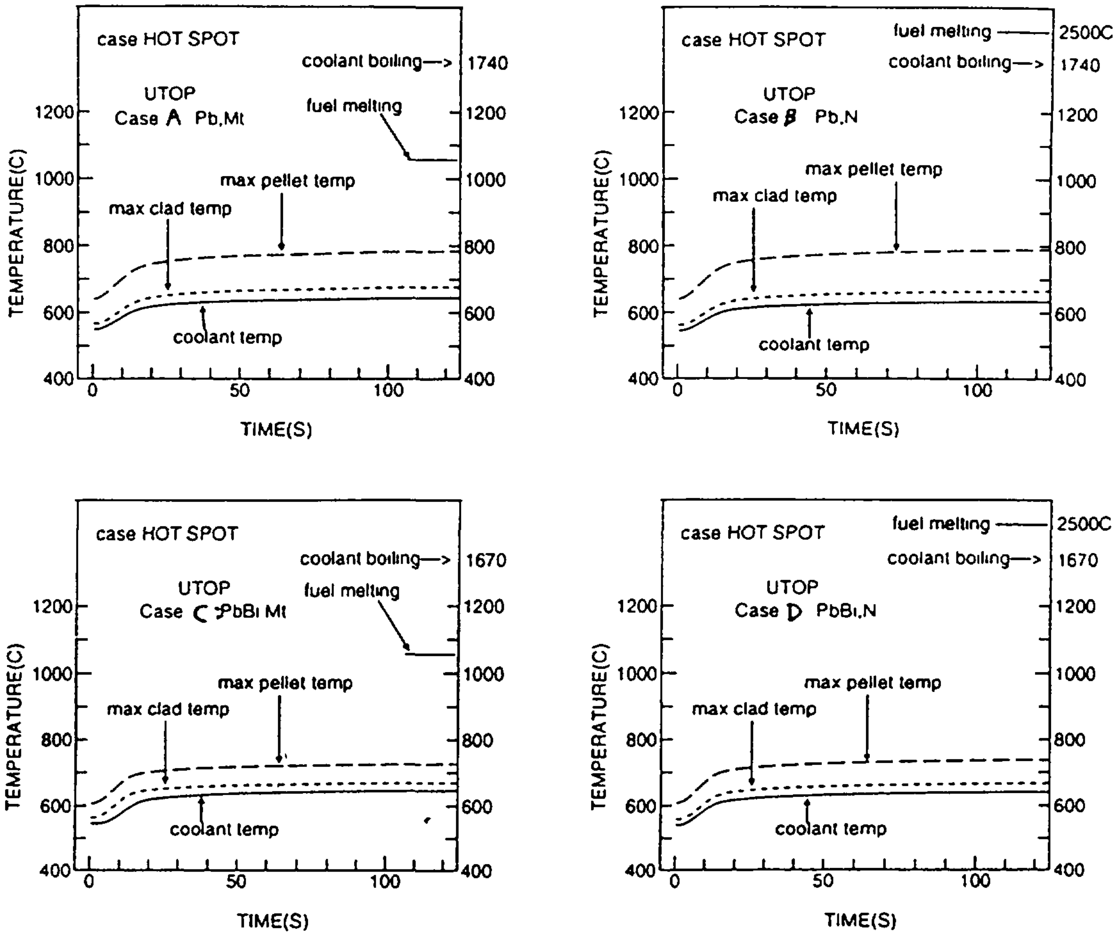


FIG. 4. Hot spot temperature during UTOP accident at BOL for design cases A to D

4.3. Simultaneous UTOP and ULOF accident at the BOL

Fig. 6 shows that the maximum fuel pellet and coolant temperature are not exceeded under the simultaneous UTOP and ULOF accident at the BOL And as in the two previous accidents, the coolant temperature margin to the boiling temperature is large, and also the fuel pellet temperature margin from melting is large especially for the nitride fuel

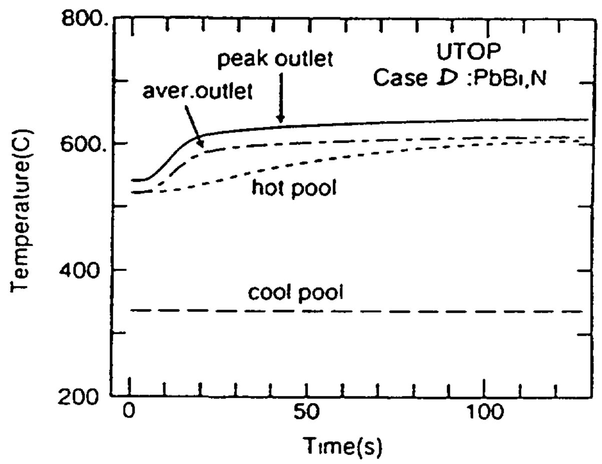


FIG. 5 Coolant temperature at the core outlet, hot pool and cool pool during UTOP accident at BOL for design case D.

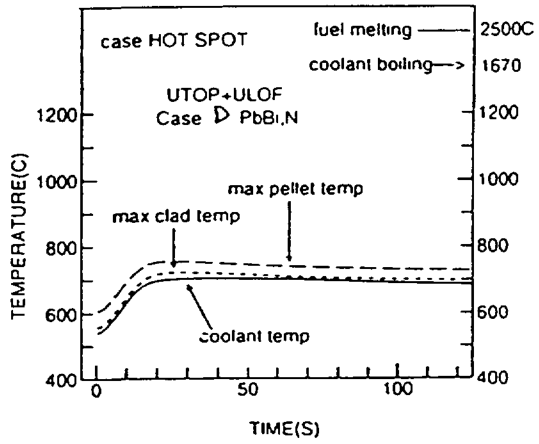


FIG. 6. Hot spot temperature during UTOP+ULOF accident at BOL for design case D

The changes of peak and average coolant temperatures, hot pool temperature, and cool pool temperature during the accident are similar to those of Fig 3 The coolant temperature increases until about 40s, and then slowly decreases. And as in the two previous accidents the temperatures of hot and cool pools increase slowly

4.4 Simultaneous UTOP, ULOF and ULOHS accident at the BOL

Fig. 7 shows the coolant peak outlet temperature, maximum cladding temperature, and maximum fuel pellet temperature under the simultaneous UTOP, ULOF and ULOHS accident at the BOL. Compared with the UTOP-ULOF accident, the temperatures are about 40-50 ℃ higher However, these values are still lower than the respective limit temperatures for cladding, fuel pellet and coolant. And as in the three previous accidents, the coolant temperature margin to the boiling temperature is large, and also fuel pellet temperature margin from melting is large especially for the nitride fuel.

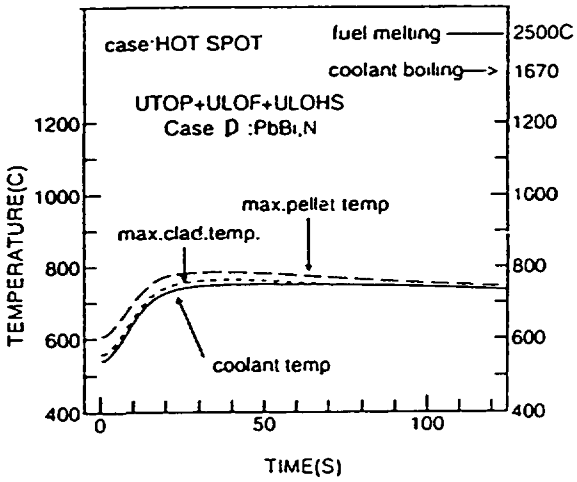


FIG. 7. Hot spot temperature during UTOP+ULOF+ULOHS accident at BOL for design case D.

Total flow-rate of coolant in the core and in the primary side of steam generator during this accident drops to the value smaller than the UTOP-ULOF accident, and after 125 seconds it becomes 1200-1300 kg/s and still decreases slowly.

Fig. 8 shows the changes of peak and average coolant temperatures, hot pool temperature, and cool pool temperature during this accident. The pattern of change is similar to the UTOP-ULOF accident. The hot pool and cool pool temperatures are almost same for both types of accident, but the coolant outlet temperature for this accident is slightly higher than the UTOP-ULOF accident.

4.5. Simultaneous ULOF, UTOP and ULOHS at the EOL.

The hot spot temperature during the UTOP-ULOF-ULOHS accident at the EOL for the design case D is shown in Fig. 9. In this simulation the value of the inserted external excess reactivity is assumed to be the same value for the simulation at BOL. However, this value is an overestimate of the actual value for this simulation. But the obtained results are

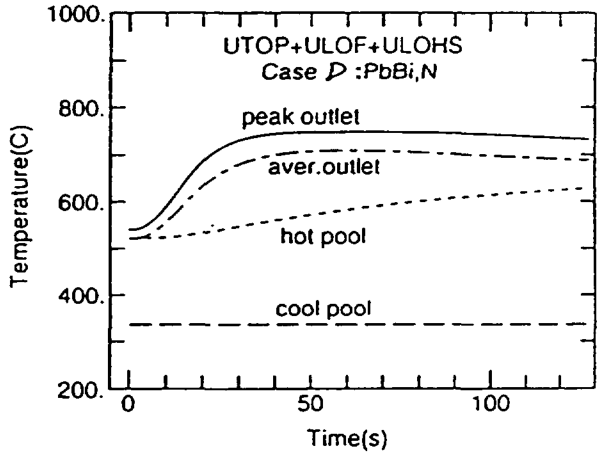


FIG. 8. Coolant temperature at the core outlet, hot pool and cool pool during UTOP+ULOF+ULOHD accident at BOL for design case D.

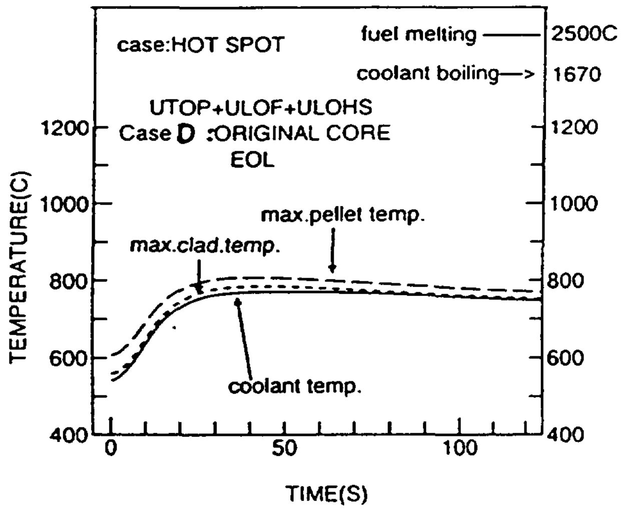


FIG. 9. Hot spot temperature during UTOP+ULOF+ULOHS accident at EOL for design case D.

acceptable from the safety point of view. The obtained temperatures are slightly higher than those at the BOL. It can be expected by the smaller feedback coefficients at the EOL shown in Table III. Anyway, all the temperature margins are large enough at the EOL also

5. CONCLUSIONS

The conclusions are summarized as follows:

(1) Long-life small safe reactor used for remote or isolated areas can be designed with metallic or nitride fuel using lead or lead-bismuth as coolant, which can be operated with very low excess reactivity during burnup and negative coolant void coefficient over whole burnup period (12 years).

(2) Chemical inertness of the coolant eliminates the intermediate heat exchanger and a compact design is employed by containing the steam generator in the core. The relatively large contribution of natural circulation (30-40%) can be attained by lower power density and larger fuel pin pitch compared to the conventional design.

(3) The metallic fuel gives slightly better results for coolant void coefficient. The lead and lead-bismuth as coolant for long-life small reactor give almost the same physics results, though the lead-bismuth coolant gives slightly better void reactivity coefficient.

(4) These two coolants require quite different operating temperature, and the lead-bismuth works in a wider range of operating temperature.

(5) The problem caused by ^{210}Po produced from lead-bismuth is not significant, but may be even desirable from nuclear-proliferation resistance point.

(6) All of the small long-life fast reactors proposed in this study can survive the UTOP, ULOF, simultaneous UTOP and ULOF and UTOP-ULOF-ULOHS accidents without any help of operator or active devices.

(7) All reactivity feedback components are negative for the above accidents. The contribution of the coolant density change to the reactivity is generally small especially for the UTOP accident. For the nitride fuel reactors, the Doppler effect plays important role during the unprotected accidents. On the other hands, for the metallic fuel reactors the axial fuel expansion plays important role. The radial core expansion is important for both metallic and nitride fuel reactors.

(8) The most severe criterion for the accidents is the maximum permissible temperature of cladding for metallic fuel. The margin of this temperature is small for the UTOP-ULOF-ULOHS accident.

(9) The lead-bismuth cooled reactors give better performance for the severe accidents compared to the lead cooled reactors due to their lower operating temperature.

(10) For the lead-bismuth coolant, the amount of resources as well as price may become problem.

REFERENCES

[1] ZAKI S., SEKIMOTO, H., "A concept of long-life small safe reactor", Potential of Small Nuclear Reactors for Future Clean and Safe Energy Sources (Proc. Int. Symp. Tokyo, Japan, 1991) Elsevier, Amsterdam (1992) 225-234.

[2] ZAKI S., SEKIMOTO, H., "Reactor physics characteristics of lead cooled and lead-bismuth cooled fast reactors", Design and Safety of Advanced Nuclear Power Plants (Proc. Int. Symp. Tokyo, 1992) P9.7-16.

[3] SEKIMOTO, H., ZAKI S., Design study of lead- and lead-bismuth-cooled small long-life nuclear power reactors using metallic and nitride fuel, Nucl. Technol., **109** (1995) 307-313.

[4] TUPPER, R. B., et al., "Polonium Hazards Associated with Lead Bismuth used as a Reactor Coolant", Fast Reactor and Related Fuel Cycles (Proc. Int. Conf., Kyoto, 1991) P5.6-1.

[5] ZAKI S., SEKIMOTO, H., Safety aspect of long-life small safe power reactors, Ann. Nucl. Energy, **22** (1995) 711-722.

[6] ZAKI S., SEKIMOTO, H., Design and safety aspect of lead and lead-bismuth cooled long-life small safe fast reactors for various core configurations, J. Nucl. Sci. Technol., **32** (1995) 834-845.

[7] ZAKI S., SEKIMOTO, H., Accident analysis of lead or lead-bismuth cooled small safe long-life fast reactor using metallic and nitride fuel, Nucl. Engin. and Design, **162** (1996) 205-222.

[8] WALTAR, A.E., REYNOLDS, A. B., Fast Breeder Reactors, Pergamon Press, New York (1981) 411 pp.

[9] HANDA, M., et al., The characteristics and irradiation behaviours of U-Pu mixed oxide, carbide, nitride and metal fuels for FBRs, Journal of Atomic Energy Society of Japan(AESJ), **31** (1989) 10 pp. (in Japanese).

XA9846725

PRELIMINARY DESIGN CONCEPT OF AN ADVANCED INTEGRAL REACTOR

K.S. MOON, D.J. LEE, K.K. KIM, M.H. CHANG, S.H. KIM
Korea Atomic Research Institute,
Taejon, Republic of Korea

Abstract

An integral reactor on the basis of PWR technology is being conceptually developed at KAERI. Advanced technologies such as intrinsic and passive safety features are implemented in establising the design concepts of the reactor to enhance the safety and performance. Research and development including laboratory-scale tests are concurrently underway for confirming the technical adoption of those concepts to the reactor design. The power output of the reactor will be in the range of 100MWe to 600MWe which is relatively small compared to the existing loop type reactors. The detailed analysis to assure the design concepts is in progress.

1. Introduction

The nuclear reactors currently under development in the worldwide nuclear societies are largely categorized into two different concepts with respect to the configurations of major primary components ; namely, loop type and integral type. Most of power reactors that are currently in operation and under development have loop type configurations which enable large-scale power output and thus provide economical power generation. On the other hand, integral reactors receive a wide and strong attention due to its characteristics capable of enhancing the reactor safety and performance through the removal of pipes connecting major primary components, even for a certain power limit due to the limited reactor vessel size which can be manufactured and transportable. The relatively small scale in the power output of integral reactors compared to the loop type reactors, however, draws a special concern for the various utilization of the reactor as an energy source, as well as power generation especially for the small-sized grid system.

Small and medium reactors with integral configurations of major primary components are actively being developed in many countries. The design concepts of those reactor vary with the purposes of application. Since the second half of 1995, Korea Atomic Energy Research Institute (KAERI) has been putting efforts to research and develop new and elemental technologies for the implementation to the advanced reactors. In parallel with those efforts, an advanced integral PWR by implementing those technologies and also passive safety features is under conceptual development. The electrial power output of the reactor will be in the range of 100MWe to 600MWe depending on the purpose of utilization such as power generation, energy supply for the seawater desalination and others. As far as the electricity generation concerned, this range of power output is considered as suitable for energy supply to the industrial complexes, remotely located islands, and specially isolated areas. The reactor core is conceptually designed with no soluble boron and hexagonal fuel assemblies to enhance the operational flexibility and to improve the fuel utilization. The reactor safety systems primarily function in a passive manner when required.

This paper describes the conceptual design features of the advanced integral reactor under development at KAERI, and also important R&D subjects concurrently in progress in order to prove and confirm the technical feasibility of design concepts.

2. Reactor Design Concepts

In general, an integral type of reactor contains all major primary components such as core, steam generator, pressurizer, and reactor coolant pumps in a single pressurized reactor vessel, which mainly differs in concept from the loop type reactor. KAERI's advanced integral reactor also applies the same general definition of integral reactors.

2.1. Reactor Core and Fuel

The achievement of intrinsic safety and operational reliability is a concern of most importance in the core design. To this end, the low core power density and soluble boron free operation are implemented as major design features of the core. The low core power density and thus increased thermal margins with regard to the critical heat flux ensure the core thermal reliability under normal operation and accident conditions. This feature, furthermore, provides passive safety benefits with respect to the enhanced negative feedback for lower operating fuel temperatures and inherent power distribution stability. The elimination of soluble boron from the primary coolant becomes a major potential simplification for the advanced reactors. From the point of the view of the reactor control and safety, soluble boron free operation offers potential benefits through the presence of a strong negative moderator temperature coefficient over the entire fuel cycle. This design feature thus provides much improved passive response for a variety of performance transients and load changes. As a result of the above two important design features, the core is more stable and resistant to transients, and therefore provides improved operational flexibility. The longer refueling cycle such as 18months or longer is adopted for the purpose of improving the plant availabilty.

Fuel assembly adapts a semi-tight hexagonal geometry to improve the fuel utilization through a relatively high plutonium conversion ratio compared to the conventional LWRs. The fuel design is based on the existing Korean Optimized Fuel Assembly (KOFA) design technology. The hexagonal fuel assembly yields the lower moderator to fuel volume ratio(V_m/V_f) and the hardened neutron spectrum which result in stronger moderator temperature coefficients and higher plutonium conversion ratio. The fuel rods are the same as those of the KOFA except geometrical arrangement which is changed from the square array to the hexagonal array. Fuel utilizes low enrichment, uranium dioxide fuel, which is operated at a low specific power density(19.6kW/kgUO_2). The uranium enrichment of the fuel will be selected to achieve the 18 months(or longer) operating cycle. As shown in Fig.1, the fuel assembly is a hexagon with 22.9cm in lattice pitch and is provided to accommodate the control assembly in each fuel assembly. The fuel assembly consists of 360 fuel rods and 36 guide tubes for control absorbers and/or insertable burnable absorbers and 1 guide tube for central in-core instrument. The same fuel assembly is utilized in the core design regardless of the reactor power output.

For 100MWe and 600MWe power output as examples, the reactor core is rated at 300 MWt with 55 fuel assemblies and 1933MWt with 151 fuel assemblies, respectively. The corresponding average linear heat generation rates are 8.4 kW/m and 9.7 KW/m which are much lower that of conventional PWRs. Table 1 shows major design parameters of the conceptual designs for the core and fuel.

2.2. Primary Circuit

Fig. 2 shows the general arrangement of the primary components and internal structures of the reactor pressure vessel. Above the reactor core, helically coiled once-through steam generator is located between the core support barrel and reactor vessel. Thermal shields are provided around the core to reduce the neutron fluences on the reactor vessel. The canned motor pumps are horizontally installed on the reactor vessel above the steam generators. The upper plenum of the vessel forms a pressurizer to maintain the operating pressure of the reactor. Since all the primary system components are installed in a single pressure vessel, there is no primary pipings between major primary components and thus it completely eliminates the large break LOCA. The primary circuit is designed to provide the enhanced natural circulation capability through the sufficient temperature difference between cold and hot water along with the sufficient difference in height between the core and steam generator to produce the driving force to circulate the primary coolant. The reactor vessel is surrounded, as shown in Fig. 3, with another vessel called as safe guard vessel which contains water up to the level of the top of steam generator. The water in the safe guard vessel is pressurized with the nitrogen gas at approximately the atomospheric pressure, and is served as an interim heat sink for the emergency decay heat removal system that will be described in the next section. This section describes the design concepts of major primary components, and Table 1 summarizes some of basic design parameters of the reactor systems.

■ <u>Steam Generator</u> : The helically coiled once-through steam generator(SG) is located within the reactor vessel in the annular space between the core support barrel and the reactor vessel inner wall. The SG is designed to completely evaporate the secondary coolant in a single pass through the S/G tube side. Since the current design concept adopts primary circuit natural circulation operation to produce approximately 50% of full power for a relatively small power output reactor design, the SG will be located high above the core considering the current manufacturing capability of a single pressure vessel. The SG consists of groups of tube bundles, downcomer, feed water and steam headers, shrouds to guide the primary flow, and tube supporting structures. The design utilizes Inconel 690 tubing and the tube bundles are supported by perforated radial support plates so that the load can be transferred to the bottom support structure located on the supporting lug.

TABLE 1. BASIC PARAMETERS OF ADVANCED INTEGRAL REACTOR

Reactor Core and Fuel

Nominal Core Power, MWt	1933(a)	300(b)
Power Density, KW/l	77.3(a)	66.7(b)
Avg Linear Heat Rate, KW/m	9.7(a)	8.4(b)
Active Core Height, m	3.66(a)	1.8(b)
Effective Core Diameter, m	3.12(a)	2.0(b)
Number of FAs	151(a)	55(b)
Fuel Rod Descriptions		
Fuel Type	UO_2	
Enrichment(Equil.), w/o	~ 3.5	
Clad Material	Zircaloy-4	
Fuel Pellet OD, cm	0.784	
Clad OD, cm	0.91	

Primary Circuit

Design Pressure, MPa	17
Operating Pressure, MPa	12.5
Coolant Inlet Temperature, °C	285
Coolant Outlet Temperature, °C	315
Coolant Flow, Kg/sec	1.2×10^4(a) 1.8×10^3(b)

Pressurizer

Type	Gas/Steam Self-Pressurizer

Steam Generator

Steam Temperature, °C	290
Steam Pressure, MPa	4.7
Superheat, °C	30
Feedwater Temperature, °C	240
Tube Material	I690 T/T
Tube Diameter, mm	19

Reactor Coolant Pump

Type	Glandless, Wet Winding Canned Motor
Number	4

Containment Overpressure Protection

Type	Passive, Steam Driven Injector

Reactor Safety Systems

Decay Heat Removal	Passive, Natural Convection Hydraulic Valve/Heat Pipe
Reactor Shutdown	Control Rods/Boron Injection
Emergency Core Cooling	Not required

Note : (a) for 600MWe, and (b) for 100MWe Power Output

The size of the SG will be selected depending on the sclae of power output with consideration of simplifying many of operational concerns including the access for in-service inspection and maintenance.

▮ Pressurizer : The large free volume above the primary coolant level is designed as a self-pressurizing pressurizer. This upper part of the reactor vessel is thus filled with the mixture of nitrogen gas and steam providing a surface in the primary circuit where liquid and vapor are maintained in equilibrium at saturated condition. The pressure of the primary system is equal to the gas partial pressure plus the saturated steam pressure corresponding to the core outlet temperature. The reactor therefore operates at its own operating pressure matched with the system status. The nitrogen gas partial pressure is chosen to maintain subcooling at the core exit to avoid boiling in the hot channel during transients. The volume of gas space is large enough to prevent the safety valves from opening during the most severe design basis transients.

■ <u>Reactor Coolant Pump</u> : The reactor coolant pumps are sealed type canned rotor pumps with added inertia to increase the pump rundown time. With no shaft seals in the pump, the small LOCA associated with seal failure of the pump as in the conventional standard design is eliminated. The required number of pumps and pump capacity to circulate the primary coolant can be reduced by the design characteristics of the primary circuit natural circulation capability.

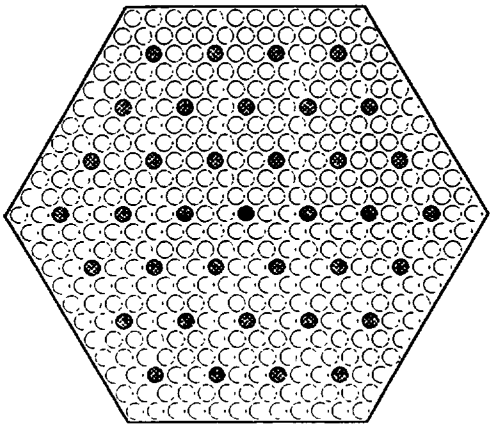
■ <u>Control Element Drive Mechanism(CEDM)</u> : The design of solüble boron free core results in the only use of control rods for the reactivity control and load change operation and thus requires a fine positioning control capability of the control rod. In addition, the adoption of a self-pressurizer in the upper plenum of the reactor vessel introduces difficulties in lubricating the moving parts with the primary coolant since the latch mechanism of control rods will be located in the steam-gas region of the pressurizer. These reasons yield the useless of the existing magnetic jack type CEDM. Consequently, a new concept of CEDM is developed and adopted. The design of CEDM consists of position encoder, brushless DC servo motor, lift magnet coil, rare earth permanent magnet rotor, driving tube, and split ball nut assembly. The fine control capability of CEDM is assured by the use of ball nut-lead screw mechanism. When the scram of the reactor is required, the current supply to the lift magnet coil is cut off once the signal is issued , and then the split ball nut releases the lead screw to drop down the control rods by gravity and spring forces. The worth of control rods provides sufficient shutdown margin at any conditions of reactor operation.

2.3. Engineered Safety Features

The safety concepts of the advanced integral reactor under currently conceptual development are basically taking advantages from the characteristics of intrinsic and passive safety principles on which most of small and medium reactors rely. The passive safety concept applies to the major engineered safety features as shown in Fig. 3 and described below.

■ <u>Passive Decay Heat Removal System</u> : When the normal decay heat removal is required, the steam generators with turbine bypass system are used to reject the heat to the condenser. This can be achieved by natural circulation on the primary side but requires feed pumps and other equipments on the secondary system. If the secondary system is not available, active decay heat removal systems with steam generators are used and the heat is removed through the component cooling system. Should there be no ac power available, the core decay heat is removed to the water contained in the safe guard vessel through the natural convection system, as shown in Fig. 3, with passive actuation of initiation valves installed on the side and bottom of the reactor vessel. The heat is then passively removed through the heat pipes to the outside of the containment. Therefore, there provides theoretically infinite time of heat removal without any intervention by operator. One of the advantages of the passive decay heat removal system usinh heat pipes is that the system can be continuously operating during normal operation to remove the heat transferred from the reactor vessel to the water in the safe guard vessel through the wet thermal insulation.

■ Passive Emergency Core Cooling System : Since all large primary circuit pipes are eliminated, the large LOCA is intrinsically not considered and thus no conventional emergency core cooling system is required. However, the break in the connection pipe from the chemical and volume control system(CVCS) may cause the loss of the primary inventory through the siphoning effect. To prevent the siphoning loss of the reactor water inventory in the hypothetical event of a CVCS line break, the installation of a siphon breaker is conceptually considered. Since the reactor vessel is always externally flooded with the water in the safe guard vessel, there is no need for the external emergency core make-up. The safe guard vessel is sized to provide a minimum of 72 hours heat removal without the operator intervention.



Lattice Pitch: 22.9cm
Rod Pitch: 1.142cm
Flow Area: 196cm^2
No of Rods: 397
No of Fuel Rod: 360
No of GT for CR: 36
No of GT for Instrument: 1

Figure 1. Hexagonal Fuel Assembly

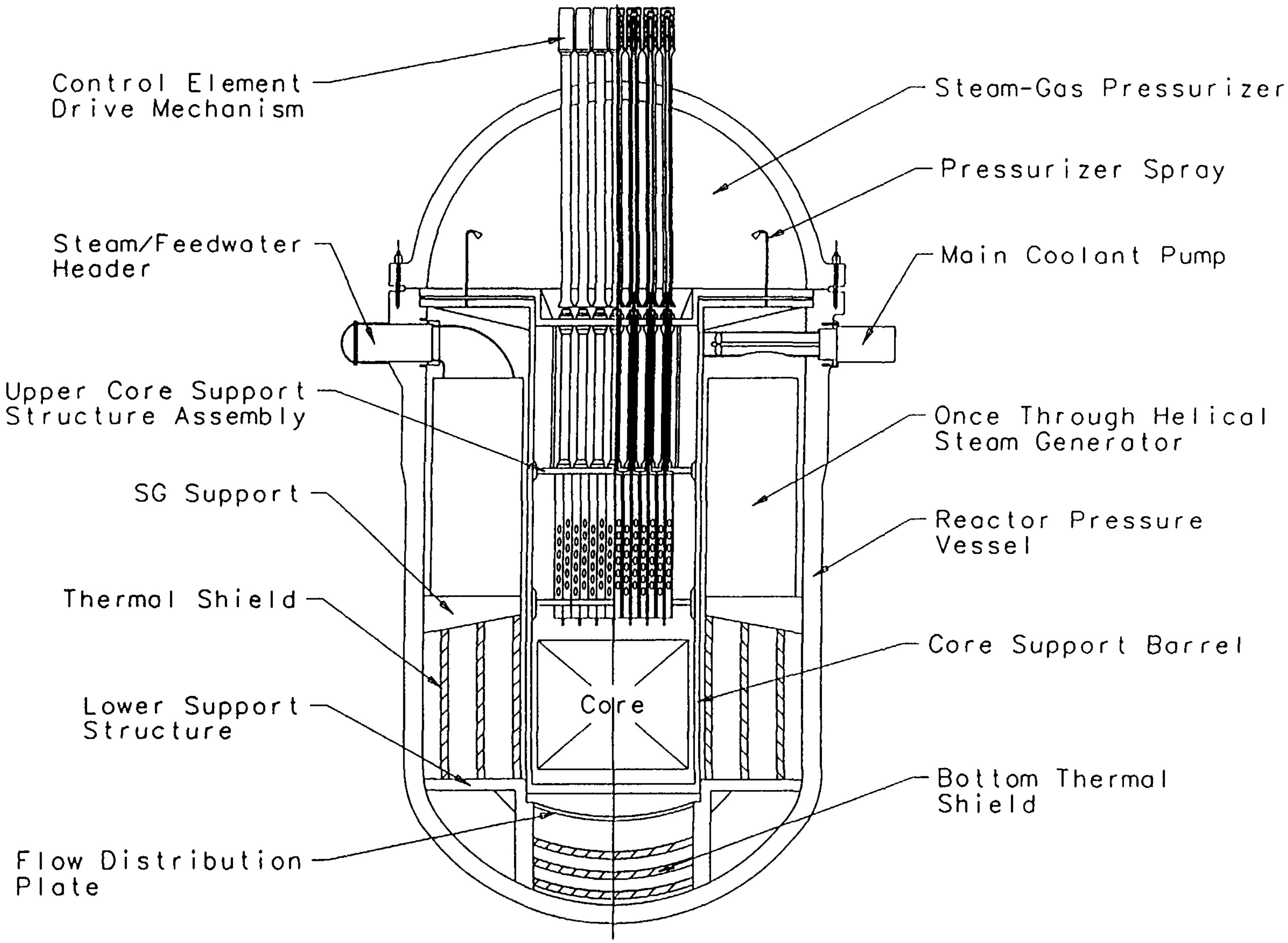


Figure 2. General Arrangement of Primary Components and Reactor Internals

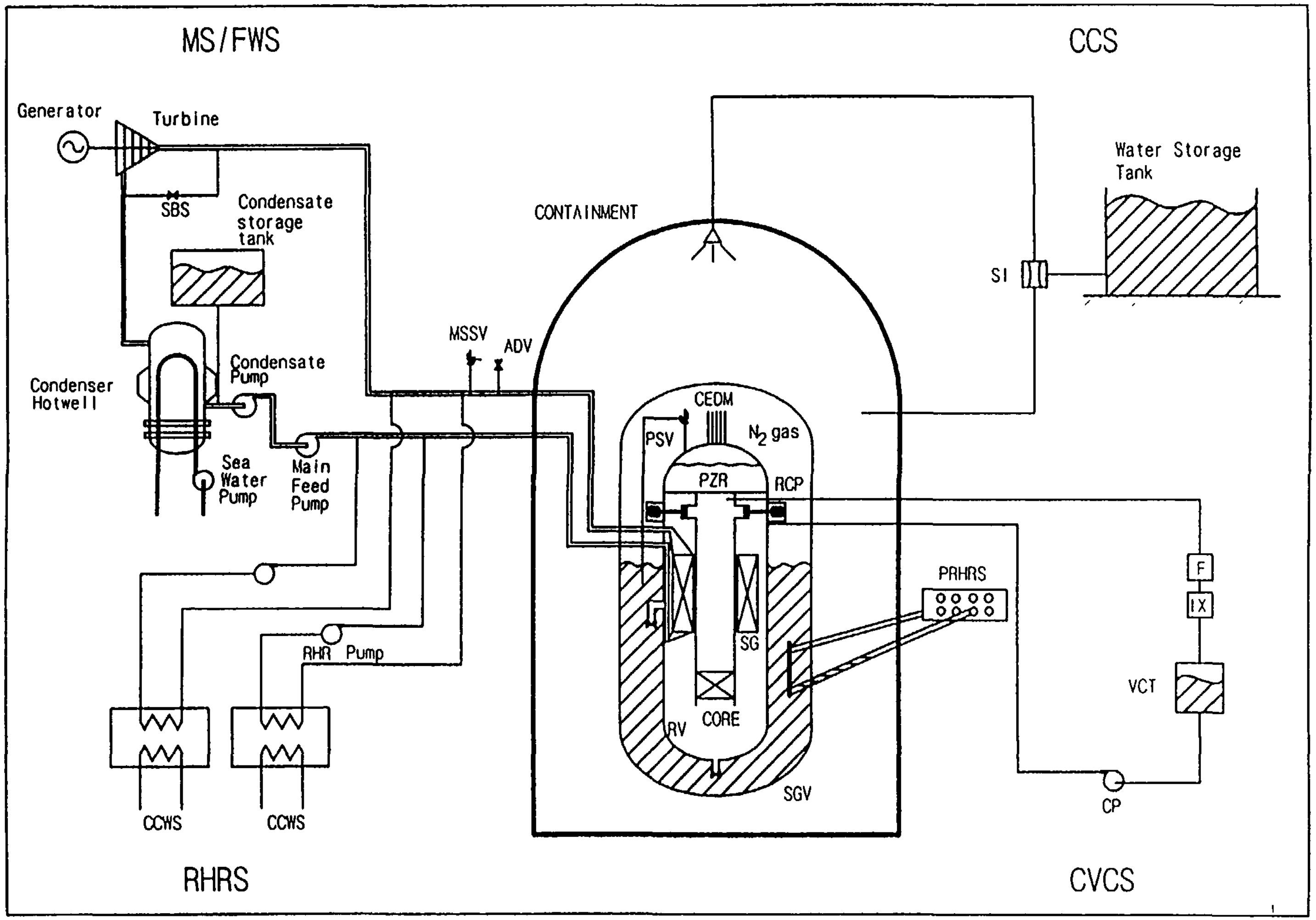


Figure 3. Schematic Diagram of Advanced Integral Reactor Systems

■ Reactor Shut-Down System : The reactor shut-down system is consisted of the control rods and the emergency boron injection system. The reactor trip at emergency is accomplished by simutaneous insertion of control rods into the reactor core by gravity following the control element drive mechanism de-energization which is actuated by trip signals from the automatic control system. In case of failure to actuate the eletromechanical protection system, the borated water from the emergency boron injection system shutdowns the reactor. The individual system is fully capable of shutdowning the reactor and provides sufficient shutdown margin to keep the reactor in a subcritical condition.

■ Passive Containment Cooling System : The containment overpressure protection is provided by a passive containment spray system. Since the hypothetical pipe break is small-sized, the pressurization rate of the containment is much slow compared to that of the conventional loop type reactors. When the energy removal from the containment is required to prevent the containment pressure from exceeding the design pressure, the steam injector driven containment spray system passively actuates as the containment energy released from the break is supplied to the system. The steam injector is a simple and compact passive pump that is driven by supersonic steam jet condition. The steam injector pumps up the water from a water storage tank to the spray nozzles located at the top of the containment.

3. Research and Development Activities

In parallel with preliminarily constructing the design concepts of an advanced integral reactor, various R&D subjects are concurrently under study. The purposes of those R&D activities are two folds : to provide the proper technical data for the design features, and to evaluate the technical feasibility and characteristics of those design concepts. Major R&D activities are as follows :

■ Hexagonal Semi-Tight Lattice Fuel Assemby : Neutronic Design and analysis methodology is under development for analyzing the reactor core with hehxagonal semi-tight lattice fuel assmblies. Thermal-hydraulic tests such as critical heat flux and pressure drop tests will be conducted to evaluate the

T/H phenomena and behavior of the fuel assembly. The suitable T/H analytical models including T/H correlations will also be developed.

■ <u>No Soluble Boron Core Concept</u> : The use of no soluble boron in the core design causes to utilize large amount of lumped burnable absorbers to properly hold down the excess reactivity at the beginning of cycle and to install considerable number of control rods for the reactor control and operation. The optimization in the number of burnable absorbers and control rods is required with respect to the reactivity compensation with fuel burnup and reactor control through the cycle, and this study in conjunction with the core design with hexagonal fuel assemblies are thus investigated in this R&D subject.

■ <u>Natural Circulation for Integral Reactor</u> : The natural circulation is an important design feature of the reactor. The thermal-hydraulic characteristics of the primary circuit is thus being investigated to prove and confirm the design concept through experimental tests and the analysis using computer codes.

■ <u>Helically Coiled Once-Through Steam Generator</u> : A thermal-hydraulic design and performance anlaysis code - ONCESG for a once-through SG has been developed and tested against available design data of similar types of SG which are designed for other integral reactors. Further improvements of the code are under progress for the application to more complicated geometrical design and analysis. Experimental investigations are also being performed to generate the proper heat transfer and pressure drop correlation applicalble to the current design concept.

■ <u>Passive Equipments for Residual Heat Removal System</u> : The characteristics of the two important passive installations, hydraulic valve and heat pipe, is currently investigated regarding their performance and reliability. A small scale of those equipments will be experimentally tested. Analytical models of those installations are also being developed for the use in the analysis of the thermal-hydraulic behaviors.

■ <u>Steam Injector Application to Passive Containment Cooling System</u> : In order to investigate the performance and technical application of a steam injector concept, theoretical and experimental study is being conducted through this R&D acticity. A computer code is also under development for the analysis of thermal-hydraulic behaviors of the steam injector.

▮ Wet Thermal Insulation : This concept is implemented to properly protect the unnecssary heat transfer from the reactor vessel to the water contained in the safe guard vessel. An experimenatal investigation is underway for the proper material selection and performance tests for the wet thermal insulation concept.

▮ Fluidic Diode Application to Passive Pressurizer Spray System : A study on the fluidic diode device is experimentally being conducted for it's use in the passive pressurizer spray system. The study also includes the development of analytical models and computer codes for the analysis of the thermal-hydraulic behavior of the device.

▮ Other R&D Activities : Besides the above major R&D activities, several elemental technologies are currently being studied at KAERI to seek for their possible application to the advanced reactor design.

4. Summary and Remarks

A small and medium advanced integral reactor under currently conceptual development at KAERI based on PWR technology fundamentally utilizes the intrinsic and passive safety features to enhance the safety and reliability of the reactor. The fundamental safety charateristics of the reactor are summarized as follow :

- Low core power density that results in the increase in thermal margins provides much improved passive response for a variety of performance transients.
- Subtantially large negative MTC resulting from no use of soluble boron offers potential benefits on the inherent power stability and resistance to transients.
- Integral configuration of primary components in a single pressure vessel basically eliminates the large-size pipings and thus large break of loss of coolant accident.
- Large volume of primary coolant provides more thermal inertia and thus much enhanced resistance to transients.
- Large passive pressurizer significantly reduces the pressure increase for the decreased heat removal events.

- No reactor coolant pump seals eliminates a potential of small LOCA associated with the seal failure.
- Adoption of various passive safety systems enhances the reactor safety and reliability which are the key concerns in advanced reactor development.

The preliminarily established design concepts of the reactor require more detailed evaluation and analysis for both the integrated concept and individual design features to technically prove and confirm its concepts. The overall evaluation and analysis is now in progress. Advanced technologies adopted in constructing the design concepts are also independently being studied to assure its technical feasibility and to generate necessary basic data for the analysis and evaluation of integrated reactor design concepts. The further evaluation and analysis may possibly result in some changes and modifications in design concepts.

APPLICATION OF NUCLEAR STEAM SUPPLY SYSTEM OF NIKA SERIES FOR SEAWATER DESALINATION

XA9846726

L.A. ADAMOVICH, A.N. ACHKASOV, G.I. GRECHKO,
V.L. PAVLOV, V.A. SHISHKIN
Research and Development Institute of Power Engineering,
Moscow, Russian Federation

Abstract

The nuclear steam supply system (NSSS) NIKA has been developed on the basis of experience available in Russia in designing, construction and operation of similar systems for ship propulsion reactors. Major systems and equipment of the NSSS are designed to take advantage of the proven engineering features and to meet Russian regulations, standards, practices and up-to-date safety philosophy. NSSS NIKA-75 has been designed for arrangement on barge. This permits to manufacture all NSSS equipment at the factory and to deliver it to the exploitation area ready for operation. NSSS NIKA-300 is designed for erection on land. It seems very interesting to use those NSSS types for seawater desalination.
The main technical solutions, concept statements, technical and economical evaluations of NIKA series nuclear steam supply systems for seawater desalination are described.

Reactor design

The NSSS of the NIKA series use the integral type reactor (see Fig. 1,2) that provides space-saving arrangement of equipment and media characterized of their own or induced radioactivity and enhances reliability of the plant as a whole due to minimization of pipes operated under primary coolant pressure. Materials, parameters and media characteristics chosen for NSSS are broadly used in Russian and worldwide practice of reactor designing. In combination with proven engineering features for major equipment (e.g., core, steam generator etc.) such an approach enables to make use of extensive research experience in thermal hydraulics, properties of structural materials, corrosion, water chemistry and so on, thus eliminating the need for any further research studies and only focusing on the minimum scope of R&D activities required for development of the pilot plant.

The core has a negative reactivity coefficient in the whole range of coolant parameters variation. This feature ensures core self-control capability and is beneficial in terms of safety.

To compensate for reactivity change, burnable poison rods and control rods assembled in groups are provided. Each group is equipped with an individual drive mechanism based on linear step motor for NIKA-300 and on rotary step motor for NIKA-75. Drive mechanisms of such design have shown high-reliability performance during long-term operation at the power and ship reactors.

Primary coolant circulation is provided by means of main circulation pumps (MCP) installed on the reactor cover and fitted with sealed electrical drive mechanisms (2 MCP for NIKA-75 and 4 MCP for NIKA-300). The pump prototypes underwent long-term operation under similar conditions and demonstrated a high reliability. Simplicity of the design of the primary coolant path ensures high flow rate of natural circulation sufficient for trouble-free core cooldown in case of loss of power to MCP.

In-vessel once-through helical steam generator made of titanium alloys is incorporated in NSSS. The steam generator consists of cassettes (16 for NIKA-75 and 12 for NIKA-300), each of them comprising 6 modules. From the secondary side, the steam generator is divided into 4 sections. In case of leaks in heat transfer surface, these sections can be isolated on power operation using special isolating valves. Steam generators of similar design have been in operation for many years and demonstrated a high reliability of their performance.

All primary equipment do not require on-load maintenance and therefore can be placed in the strong leaktight safeguard vessel which is non-attended while on power operation. Under the design basis accidents radioactivity release from the primary circuit will be mitigated in the safeguard vessel.

This paper was prepared as a follow-up contribution to the meeting in Tunis, Tunisia, 3-6 September 1996.

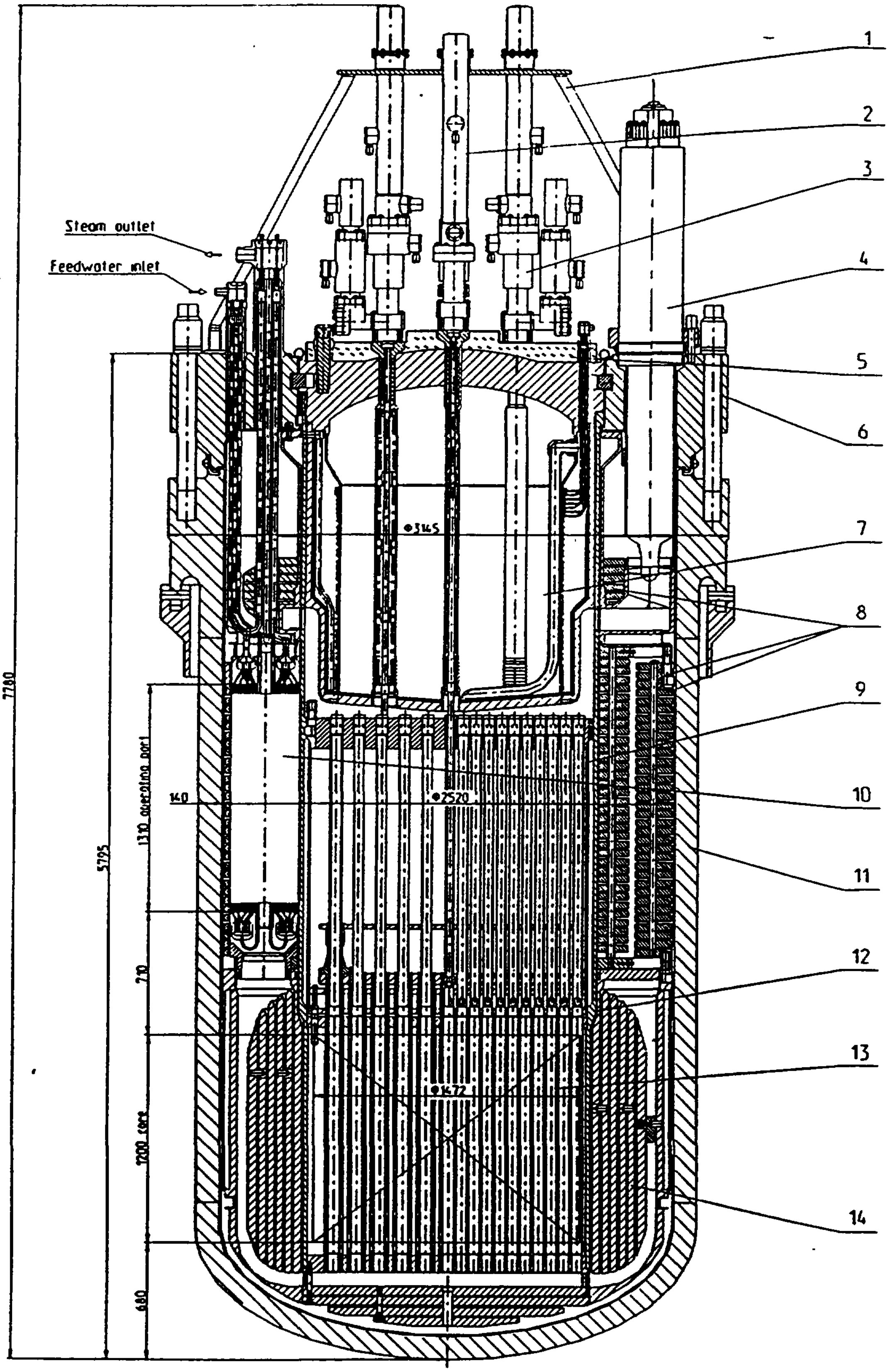


Fig. 1. General view of the reactor principle for NIKA-75.

1 - drive fastening frame; 2 - shim rod group (SG) drive (7 pieces); 3 - SG-EP drive (9 pieces); 4 - MCP (2 pieces); 5 - thermal insulation; 6 - annular cover; 7 - pressurizer; 8 - displacers; 9 - metalwork with control rod clusters; 10 - SG; 11 - vessel; 12 - core barrel; 13 - fuel assembly (379 pieces); 14 - side screen.

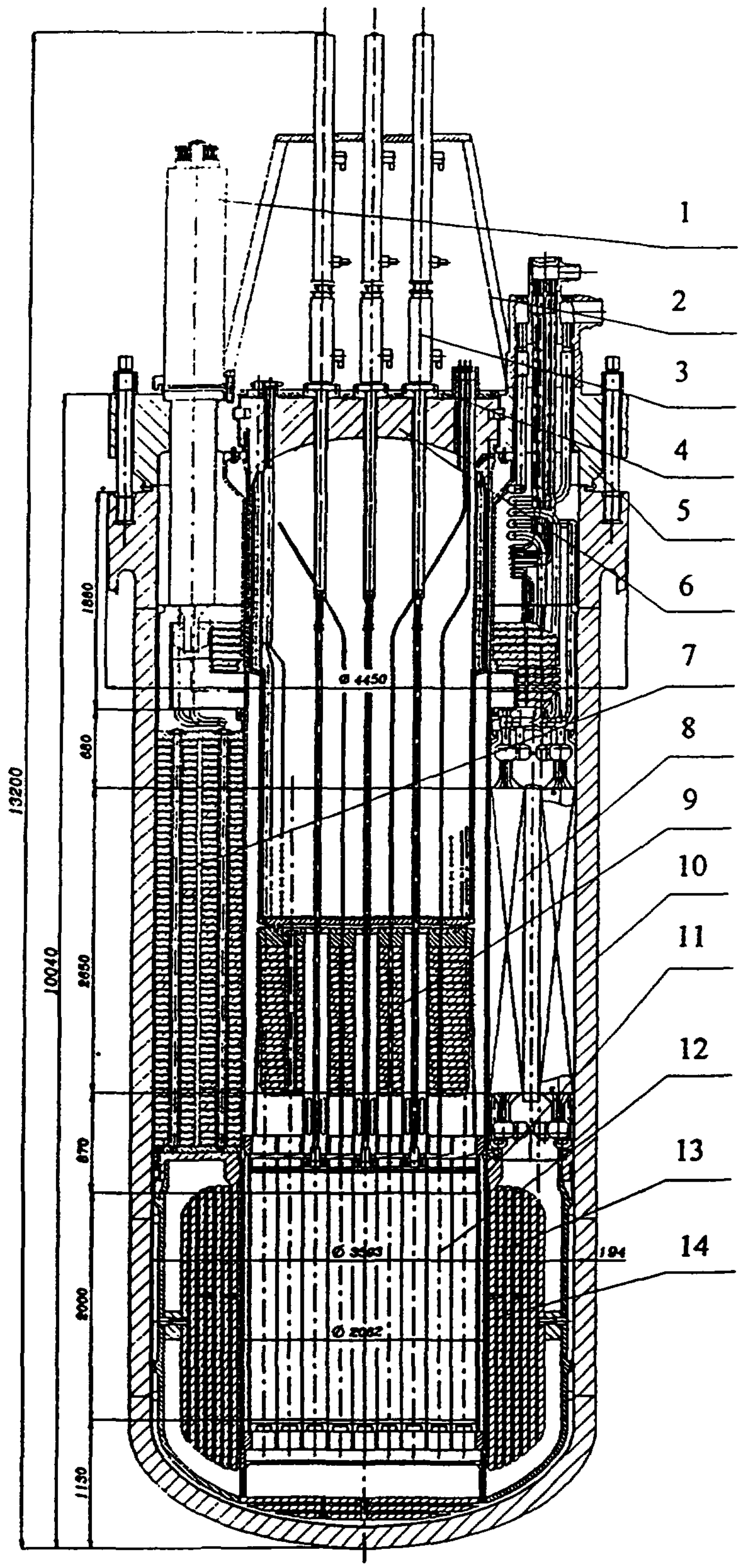


Fig. 2. General view of the reactor NIKA-300.

1 - MSP (4 pieces); 2 - drive fastening frame; 3 - shim rods drive (25 pieces); 4 - thermal insulation; 5 - annular cover; 6 - pressurizer; 7 - displacers; 8 - steam generator; 9 - protection tubes unit; 10 - vessel; 11 - core barrel; 12 - fuel assembly (57 pieces); 13 - side screen; 14 - partition.

To provide capabilities for retention of radioactive substances in case of beyond design basis events, NSSS is housed in the strong leaktight containment accessible for the purposes of equipment repair and maintenance.

NSSS will provide long-term and steady operation in the power range from 20 to 100% of full power irrespective of the number of power changes.

To prevent progression of the emergency situations into the accidents and to minimize their possible consequences, NSSS is fitted with a number of engineered safety features, namely: systems for emergency cooldown; emergency core cooling system (ECCS); reactor ,safeguard vessel and containment overpressure protection system; independent equipment cooling system and severe accident mitigation system. All the above systems are passive, i.e. they will come into action without the intervention of an operator and control systems.

In developing NSSS design concepts for medium size CNPP, primary consideration was given to ensuring operational reliability and safety of such a plant at all stages of its life cycle. It was assumed that the CNPP would meet safety requirements if its radiation effects on personnel, public and environment under normal operation and during design-basis accidents are significantly lower or at least are found within the specified limits of personnel and public exposure and standards for permissible releases and content of radioactive substances in the environment. In case of beyond design-basis accidents, such effects should be limited as much as possible.

The main design features adopted for the NSSS are aimed at ruling out any core damage beyond the specified limits of safe operation during all design-basis accidents without personnel's intervention or external assistance for no less than 72 hours. This problem should be also solved for the beyond design-basis accidents caused by any initiating events considered credited and accompanied by postulated failures of electrical control systems and active systems which rely on power supply for their operation.

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

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

- negative power and temperature coefficients of reactivity throughout the operating range of parameters;
- high flow 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.

2. **Defense-in-depth provided as a system of barriers** to off-site release 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;
- leaktight primary circuit;
- safeguard vessel;
- isolating valves;
- containment.

3. **Use of passive systems and safety features** whose operation is based on natural processes with no need for external power supply.

Such systems include:

- CPS drives design which provides assured insertion of control rods into the core by gravity and drop springs;
- interlocks in CPS drives which prevent unauthorized withdrawal of control rods from the core during commissioning, maintenance and repair;
- 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 in case of the safeguard vessel opening and under beyond design-basis accidents;
- iron and water biological shielding which apart from its direct functions, serves as bubbler tanks to hold cooling water supply as well as to remove heat from the reactor vessel to avoid its melt through under a postulated beyond design-basis accident with core dryout.
- molten core catcher(only for NIKA-300).

4. Safety systems reliability

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

- the systems are passive, i.e. they need as few as possible special actuators to initiate them, if any at all:
- the safety systems and features are diverse which are based on the 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 shutdown system is $2 \times 100\%$, of ECCS - $4 \times 50\%$, etc.)
- systems and equipment are subjected to periodic in-service inspection or continuous monitoring.

5. Protection against human errors

The design safety philosophy pays much 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 the safeguard vessel or containment);
- use of systems satisfying as far as possible the safe failure principle (the system component failures transfer the system to safety function performance or 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 cooldown systems, ECCS, system for reducing overpressure in the safeguard vessel and containment);
- reliable control systems are used, which minimize or disable erroneous operator's actions, with personnel given no access to interlocks and setpoints;
- operator support systems are provided, which rapidly assess the plant state and suggest optimum control actions;
- special hardware is used for training and maintaining the skills and knowledge of the operating and maintenance personnel, in particular, a simulator is used to drill operating personnel in various situations, including emergencies.

6. Protection against external impacts

Building structures of the power plant shall guarantee undamaged state of the NSSS containment and safeguard vessel under such external impacts as typhoon, hurricane, heavy snow and icing as well as in case of helicopter or airplane crash onto the CNPP.

Technical and economic estimation of seawater desalination

The technical and economic estimation has been carried out with the use of spreadsheets IAEA COGENERATION/DESALINATION COST MODEL [1], which, using the technical and economic indexes of nuclear plants, allows to calculate basic performances of nuclear water desalination plants. As initial data it was accepted specific cost of power plant construction 2000 $/kWe for NIKA-3000 and 3500 $/kWe for NIKA-75. The net electical power is 15 MW(e) for NIKA-75 and 100 MW(e) for

NIKA-300. The other initial data were taken from the spreadsheet [1], for example, specific water plant cost 1440 \$/(m^3/day) for multi-effect distillation, 1125 \$/(m^3/day) for reverse osmosis; interest rate 8 %. Some results of calculations are submitted in Table 2 and Fig.3-5.

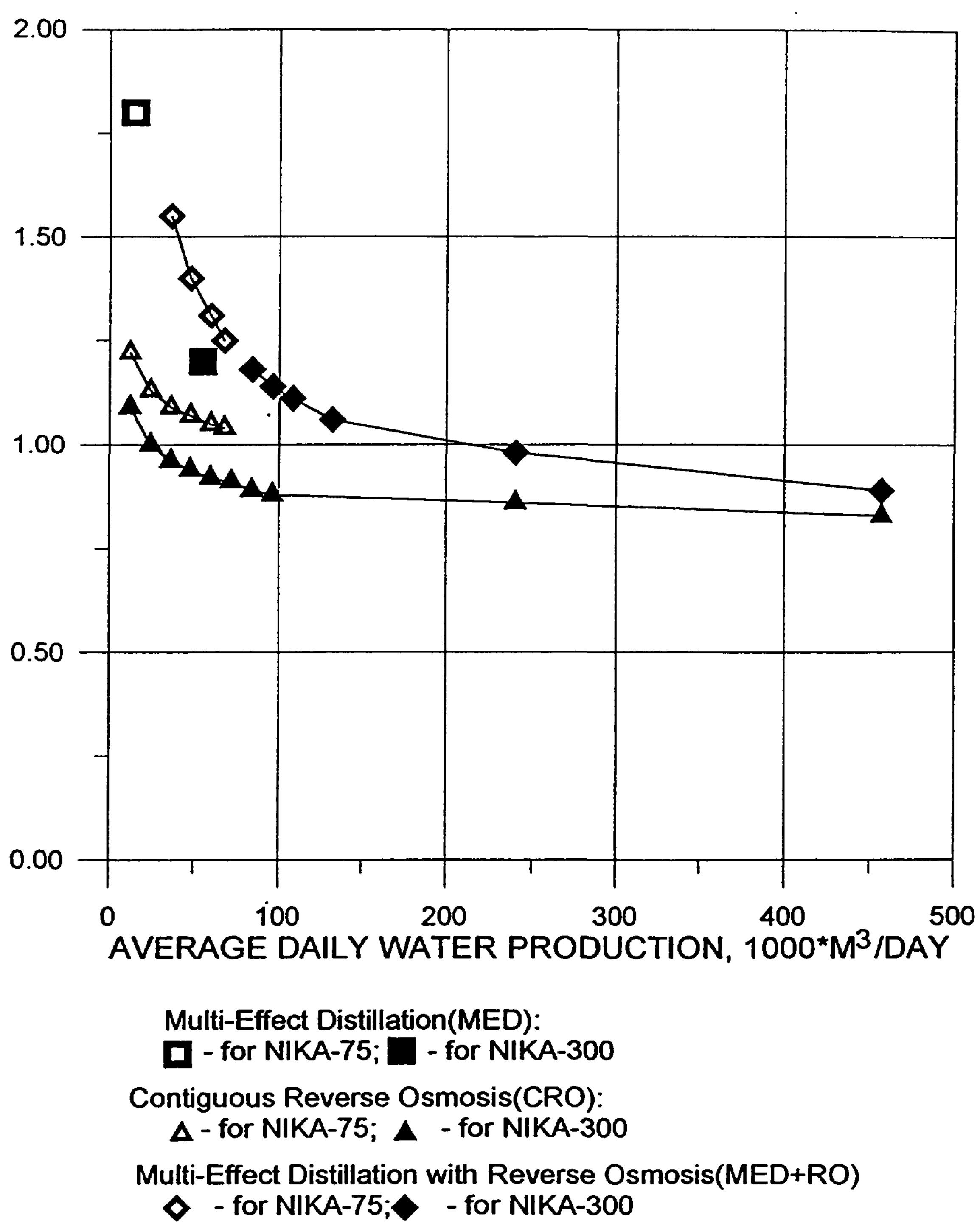


Fig. 3. Water cost versus water production.

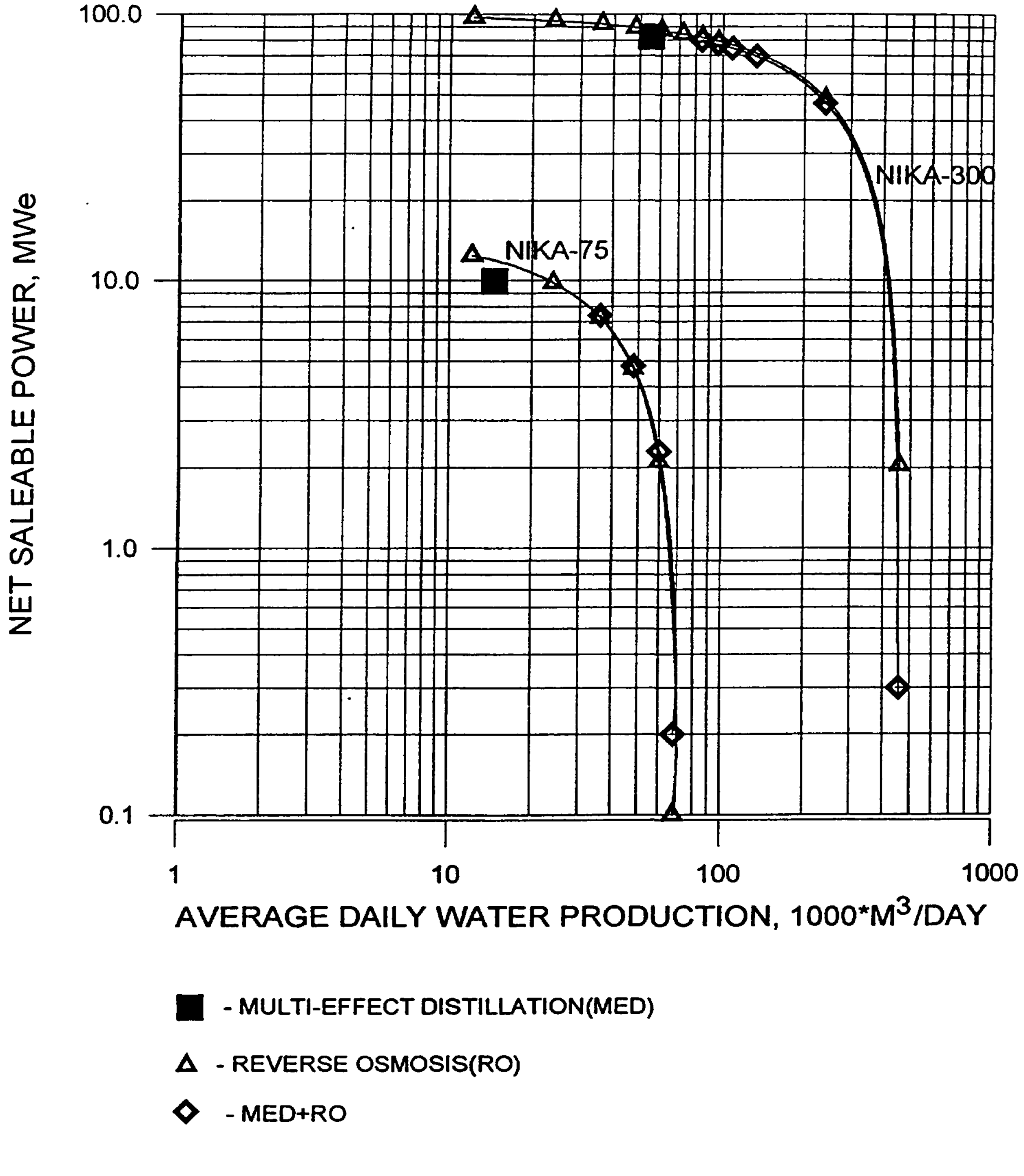
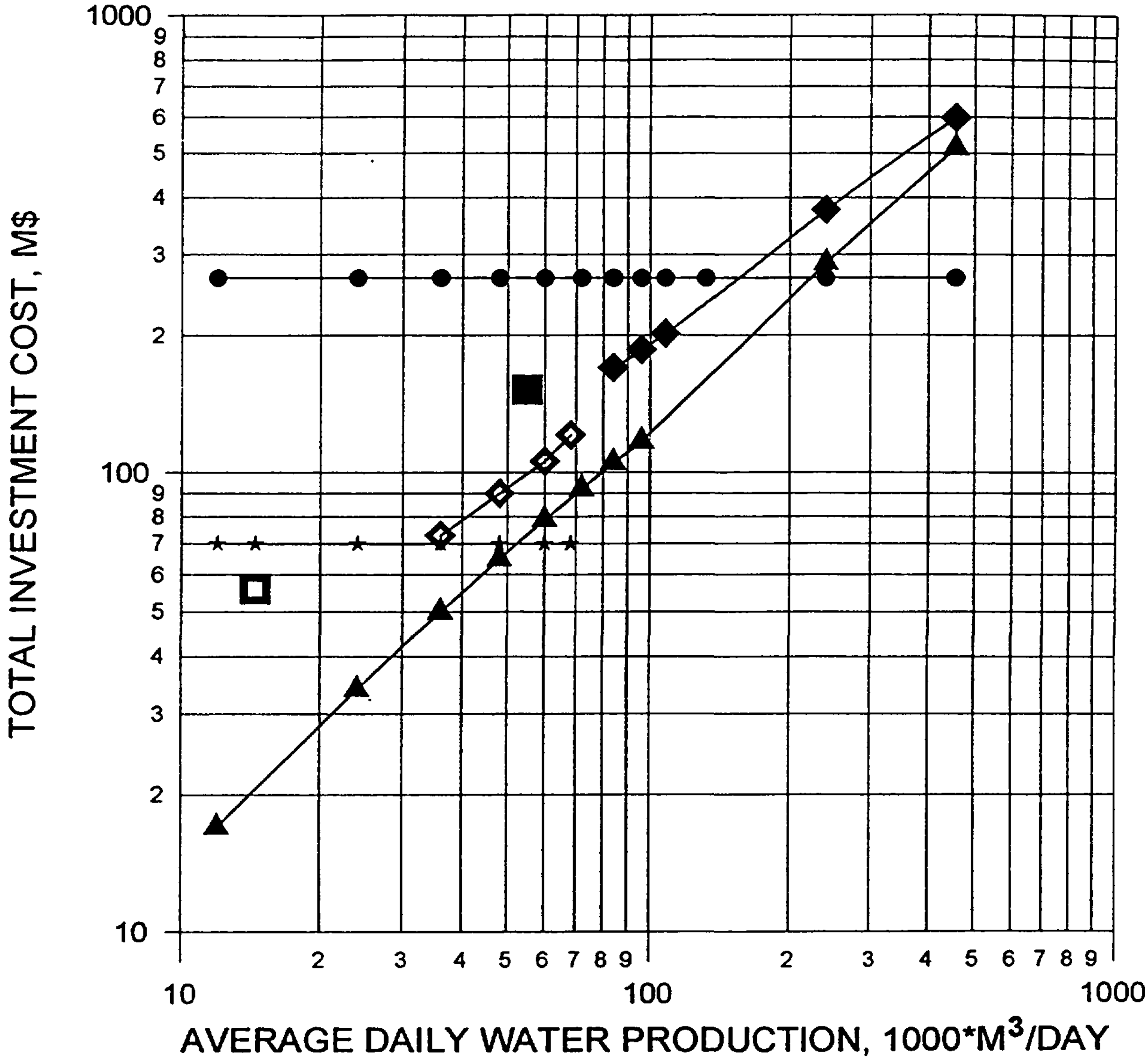


Fig. 4. Net saleable power versus water production



Multi-Effect Distillation(MED):
□ - for NIKA-75; ■ - for NIKA-300

Contiguous Reverse Osmosis(CRO):
△ - for NIKA-75; ▲ - for NIKA-300

Multi-Effect Distillation with Reverse Osmosis(MED+RO)
◇ - for NIKA-75; ◆ - for NIKA-300

Power Plant:
★ - NIKA-75; ● - NIKA-300

Fig. 5. Total investment cost versus water production.

TABLE 1 DESIGN CHARACTERISTICS OF NSSS

No	Characteristic	Unit	NIKA-75	NIKA-300
1	Thermal power of the core	MWth	75	330
2	Net electrical power	MWe	15	100
3	Steam generating capacity	kg/s	27	152 7
4	Superheated steam pressure	MPa	3 0	3 0
5	Superheated steam temperature, at least	°C	274	274
6	Feed water temperature	°C	60	180
7	Nominal pressure in primary circuit	MPa	15	15
8	Primary coolant temperature while operating at nominal power at core inlet at core outlet	°C	 260 300	 270 310
9	Operating range of power change	% N_{nom}	20 ÷ 100	20 ÷ 100
10	Effective campaign of core	years	5	4
11	Core – water-water type equivalent diameter height Fuel U^{235} enrichment U^{235} load specific power rating	 mm mm % kg kW/l	 1500 1200 19 7 260 36 7	 1800 2000 5 681 62 6
12	Service life	years	30	60

In Fig.3 dependence of fresh water cost on average daily water production is presented. The water cost decreases with increase of water production and can compete to cost of water produced by fossil cogeneration plants.

In Fig.4 dependence of net saleable electrical power on fresh water production is presented. It is visible, that electrical capacity of power plant is enough for water desalination over a wide range of fresh water flowrates. The net saleable electrical energy can be used for supply of other consumers.

In Fig.5 dependence of total investment cost on water production is presented. For water plants the size of cost increases with increase of water production.

TABLE 2

Characteristic and water plant type	Unit	NIKA-75	NIKA-300
Average daily fresh water production			
Multi-Effect Distillation(MED)	1000*m^3/day	14.5	54.95
Stand-Alone Reverse Osmoses(SARO)		12...60	12 ...456
Contiguous Reverse Osmoses(CRO)		12.09...60.42	12.09...459
Hybrid (MED + RO)		36.04...72	84...457
Fresh water cost			
MED	\$/m^3	1.8	1.2
SARO		1.12...1.38	0.86...1.25
CRO		1.05...1.22	0.83...1.09
MED+RO		1.25...1.55	0.89...1.18
Total investment cost			
Power Plant	M$	70	267
MED		54	152
SARO		24...93	24...553
CRO		17...79	17...514
MED+RO		73...121	170...598

Conclusion

New generation NSSS of NIKA series has been developed in accordance with modern safety requirements. One possible use of those NSSS is seawater desalination. Power plants based on NSSS of NIKA series can be coupled with water production plants of various types for potable water production from 12000 to 72000 m^3/day for NIKA-75 and from 12000 to 459000 m^3/day for NIKA-300. The water production costs could compete economically.

REFERENCE

[1] Technical and Economic Evaluation of Potable Water Production Through Desalination of Seawater by Using Nuclear Energy and Other Means, International Atomic Energy Agency, IAEA-TECDOC-666, Vienna, September 1992.

PART V

SIMULATION OF NUCLEAR REACTORS

XA9846727

CAE ADVANCED REACTOR DEMONSTRATORS FOR CANDU, PWR AND BWR NUCLEAR POWER PLANTS

R.S. HART
AECL-CANDU,
Mississauga, Ontario,
Canada

Abstract

CAE, a private Canadian company specializing in full scope flight, industrial, and nuclear plant simulators, will provide a license to IAEA for a suite of nuclear power plant demonstrators. This suite will consist of CANDU, PWR and BWR demonstrators, and will operate on a 486 or higher level PC. The suite of demonstrators will be provided to IAEA at no cost to IAEA.

The IAEA has agreed to make the CAE suite of nuclear power plant demonstrators available to all member states at no charge under a sub-license agreement, and to sponsor training courses that will provide basic training on the reactor types covered, and on the operation of the demonstrator suite, to all those who obtain the demonstrator suite.

The suite of demonstrators will be available to the IAEA by March 1997.

1. INTRODUCTION

CAE, a private Canadian company with considerable experience in the design of nuclear power plant simulators, has agreed to provide IAEA with a suite of nuclear power plant demonstrators at no charge to IAEA. The suite of nuclear power plant demonstrators will consist of CANDU, PWR and BWR demonstrators and will operate on a 486 or higher level PC.

IAEA has agreed to make the CAE suite of nuclear power plant demonstrators available to member states at no cost to the recipients, and to provide training on the reactor types covered by the suite of demonstrators and on the operation of the demonstrators, to all organizations receiving the CAE suite of demonstrators.

2. CAE

2.1 Background

CAE Inc., with headquarters in Toronto, Canada, and facilities throughout Canada, the United States, Europe, Asia and Australia, is the world leader in the design and production of commercial flight simulators and visual simulation systems. The company, with over 6,200 highly skilled employees, is a leading supplier of military simulation systems, electronic control systems, and maintenance, repair, modification and overhaul services for military aircraft and offers a complete range of technical information development and delivery services. CAE is also a leading supplier of nuclear power plant simulators, and is the only simulator supplier to provide full scope simulators for CANDU, PWR and BWR plants. CAE's Industrial Technologies Group is a leading supplier of environmentally friendly aqueous-based cleaning equipment, sophisticated separation technologies for various industries,

• ATTAC™ (Advanced Two-Phase Thermal Hydraulic Code Generator), which is used to generate models for key systems that may experience two-phase flow, including the heat transport system (RCS), steam generators, pressurizer and pressure relief tank; and

• TIGERS™ (The Interactive Graphics Environment for Real-time Systems), which allows the operator of the simulator to generate graphics for data and control panel mimics.

The CAE simulators incorporate integrated test and validation tools, and include an integrated instructor/operator station.

3.2 Simulator Operation

The CAE simulators replicate the performance of the Nuclear Steam Supply System (NSSS) and the impact on the plant performance of key Balance of Plant (BOP) systems The basic graphic displays for CANDU, PWR and BWR are presented in Figures 1, 2, 3 and 4 respectively; actual displays are in colour.

The simulators include a menu of typical plant malfunctions and animated pages that are used by the simulator operator to control and monitor the simulations.

The CAE simulators make extensive use of colour graphics to display data (for example, core flux, core temperature, HTS/RCS void, HTS/RCS temperature, and steam generator secondary side level), and to display the status of devices such as pumps and valves.

Animation features include the ability to zoom, resize and reposition displays; the ability to overlay data (from current or previous simulations). These features are enhanced by the "Windows" environment.

The simulations can be conducted in slow motion, real time, or accelerated time (for relatively slow transients).

The suite of nuclear power plant demonstrators (CANDU, PWR, BWR) operate in much the same manner and incorporate similar features and operational characteristics for each of the nuclear plant types. The instructor can therefore conduct simulations of the different reactor types with ease, utilizing a common approach and format.

The CAE suite of demonstrators are extremely capable and very user friendly; they are therefore of interest to both novices and experienced workers in the nuclear energy field.

The suite of nuclear power plant demonstrators will represent generic plants (CANDU, PWR, BWR) in the 600 MW(e) or larger size range. However, to assure precise simulation, detailed models of plant features/characteristics will be incorporated. This requires that the simulations be plant specific in many cases; however, the demonstrators will not identify the specific basis of the simulation, and will be made to appear as generic as possible.

4. USES OF THE CAE "PC BASED" DEMONSTRATORS

The suite of "PC based" CANDU, PWR and BWR nuclear power plant demonstrators is intended to provide basic information and training to a variety of people/organizations in the field of nuclear power, providing a level of knowledge regarding the operation of nuclear power plants.

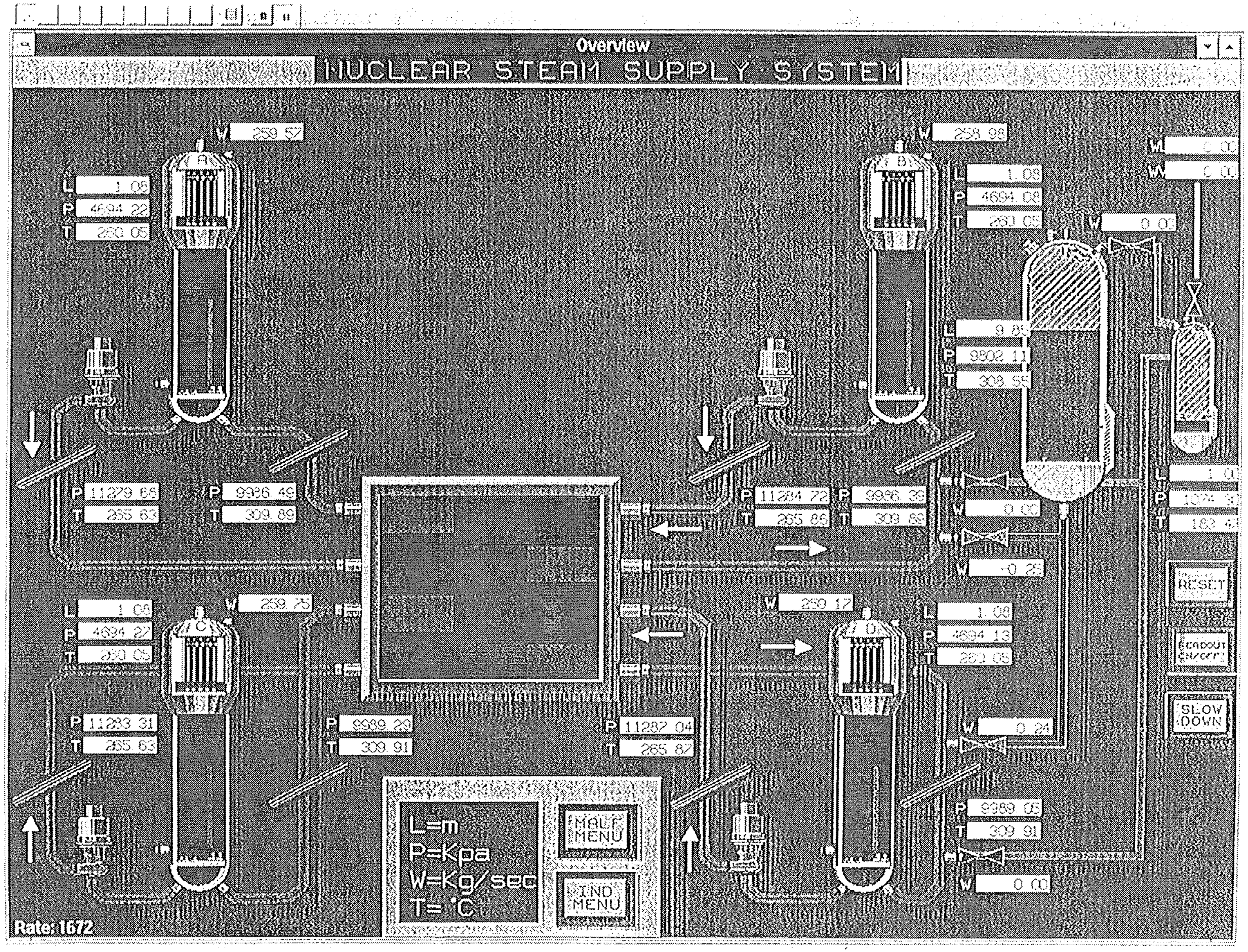

Overview
NUCLEAR STEAM SUPPLY SYSTEM
L=m
P=Kpa
W=Kg/sec
T=°C
MALF MENU
IND MENU
RESET
SLOW DOWN
Rate: 1672

FIG. 1.

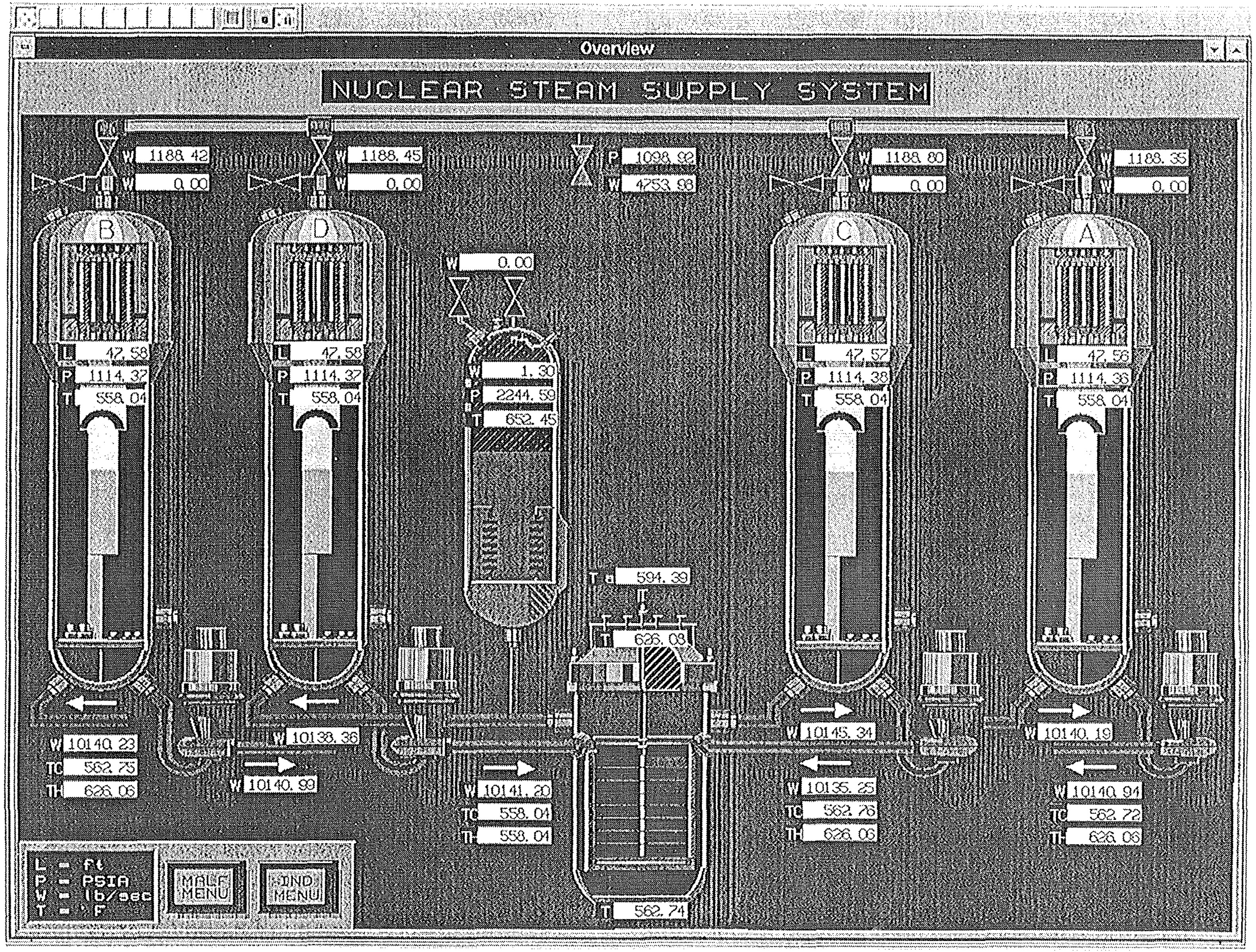

Overview
NUCLEAR STEAM SUPPLY SYSTEM
W 1188.42
W 0.00
W 1188.45
W 0.00
P 1098.92
W 4253.98
W 1188.80
W 0.00
W 1188.35
W 0.00
B
D
C
A
W 0.00
L 47.58
P 1114.37
T 558.04
L 47.58
P 1114.37
T 558.04
W 1.30
P 2244.59
T 652.45
L 47.57
P 1114.38
T 558.04
L 47.56
P 1114.36
T 558.04
T 594.39
T 626.08
W 10140.23
TC 562.75
TH 626.06
W 10138.36
W 10140.99
W 10141.20
TC 558.04
TH 558.04
W 10145.34
W 10135.25
TC 562.76
TH 626.06
W 10140.19
W 10140.94
TC 562.72
TH 626.06
T 562.74
L = ft
P = PSIA
W = lb/sec
T = °F
MALF MENU
IND MENU

FIG. 2.

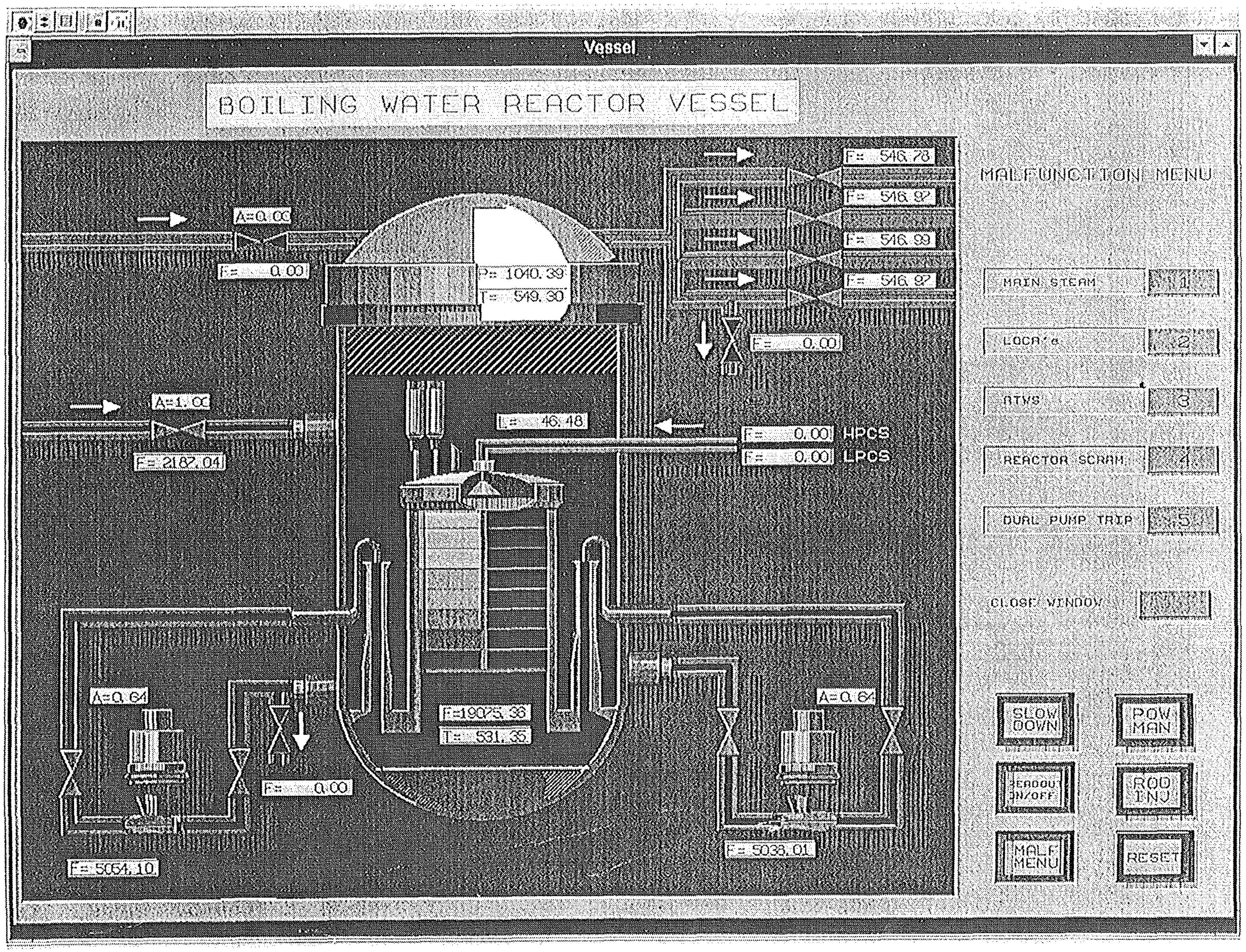

Vessel
BOILING WATER REACTOR VESSEL
MALFUNCTION MENU
A=0.00
F= 0.00
P= 1040.39
T= 549.30
F= 546.78
F= 546.97
F= 546.99
F= 546.97
F= 0.00
A=1.00
F= 2187.04
L= 46.48
F= 0.00 HPCS
F= 0.00 LPCS
MAIN STEAM 1
LOCA's 2
ATWS 3
REACTOR SCRAM 4
DUAL PUMP TRIP 5
CLOSE WINDOW
A=0.64
F=19075.36
T= 531.35
F= 0.00
A=0.64
F= 5064.10
F= 5038.01
SLOW DOWN
POW MAN
READOUT ON/OFF
ROD INJ
MALF MENU
RESET

FIG. 3.

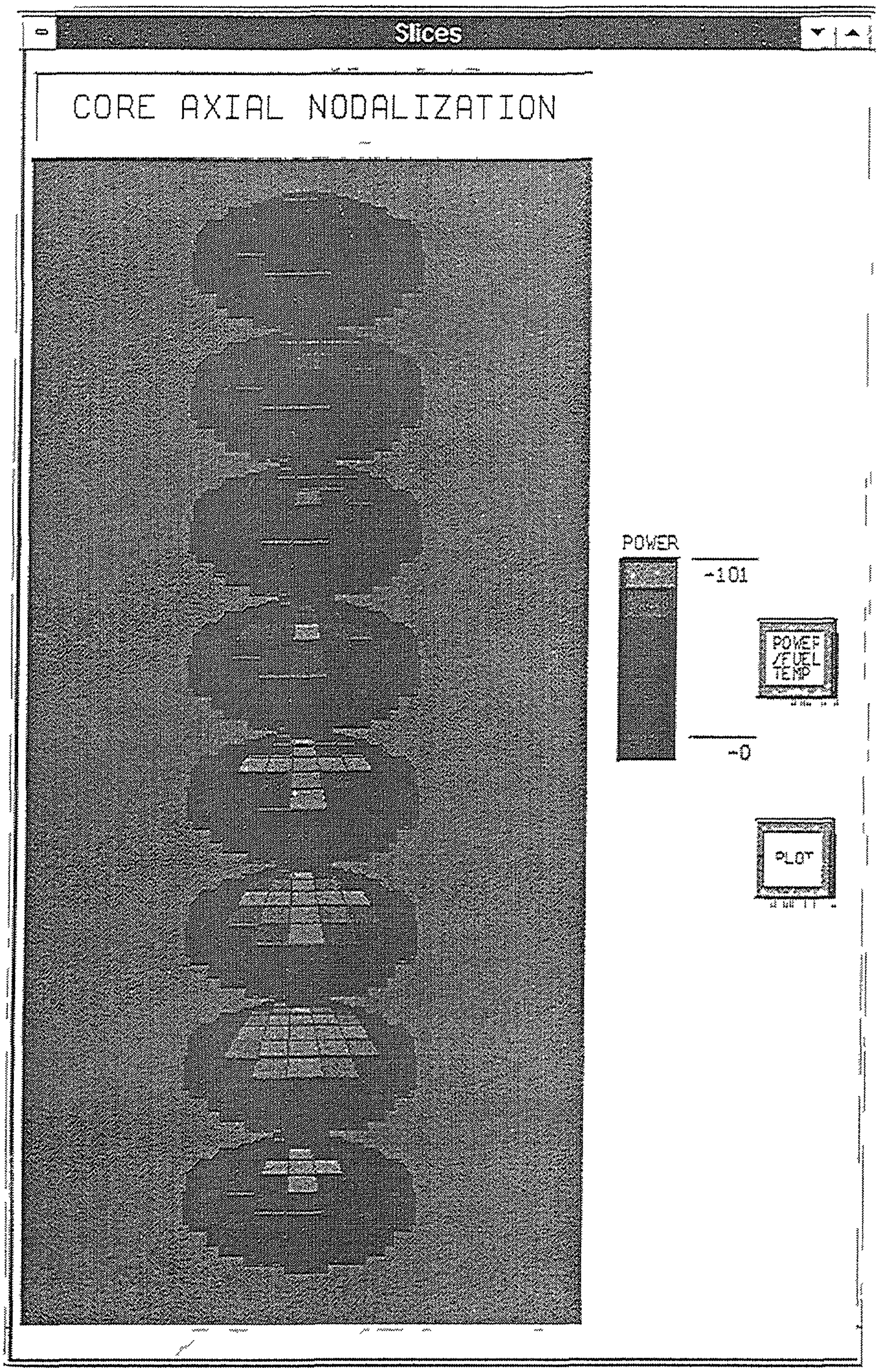


FIG 4

The intended audiences include research organizations, universities and utilities that anticipate a future involvement with nuclear power, and which do not have a basic understanding of nuclear power plant operation

The suite of demonstrators is not intended for plant operator training or to form the basis of the specific evaluations of reactor types

5. TRAINING

The individuals/organizations receiving the CAE suite of demonstrators must be familiar with both the basic characteristics of the reactor types (CANDU, PWR and BWR), and the operation of the simulators in order to derive real benefit from operation of the demonstrators. Hence, completion of a training program covering reactor characteristics and simulator operation is a prerequisite to obtaining the suite of demonstrators.

The IAEA has agreed to sponsor the training courses. The training courses will include experts in the reactor types (likely to be provided by reactor vendors) and experts in operation of the demonstrators (to be provided by CAE).

6. SUMMARY

The suite of nuclear power plant demonstrators (CANDU, PWR, BWR) provided by CAE will make very capable user friendly PC based simulators available to a wide variety of organizations world-wide, without cost, via the IAEA. These simulators will substantially enhance the nuclear power knowledge base, particularly in developing countries, and promote the understanding and acceptance of nuclear power world-wide.

The CAE suite of demonstrators would normally sell for several hundred thousand dollars for each seat (user). Making the suite of simulators available world-wide at no cost is a very generous offering by CAE, and of substantial benefit to users.

XA9846728

UTILIZATION OF EL DABAA BASIC SIMULATOR FOR MANPOWER DEVELOPMENT

W.A. WAHAB, S.B. ABDEL HAMID
Nuclear Power Plants Authority,
Cairo, Egypt

Abstract

Training with basic simulators is considered as an essential tool for training and retraining of the nuclear power plant staff. To achieve that objective ; Nuclear Power Plants Authority (NPPA) has installed a basic training simulator for PWR and PHWR simulation at El Dabaa site . The basic simulator simulates a 3-loops PWR-900 MW(e) and 2-loops PHWR-600 MW(e) . The simulator has passed in-plant acceptance tests in CEA/CENG, Grenoble, France and passed successfully on-site completion tests at El Dabaa site in Novmber 1991 . This paper presents the main features of the simulator ; training capabilities, hardware configuration and software architectures . Also training methodology by NPPA and the training experience gained by using the simulator are presented .

1. TRAINING CAPABILITIES OF EL DABAA SIMULATOR

Utilization of El Dabaa simulator for training can be achieved in two stages of training ; basic training and retraining stages .

1.1 Basic training

Training with basic simulator would meet several needs for candidate operators , among these needs :

(1) basic simulator is a valuable complete to theoritical training,

(2) team-work during various states of plant operation ; start-up, full power operation, go back to zero power, and during abnormal situations ,

(3) application of several required manual manouvers during start-up sequences ,

(4) decreasing of starting time ,

(5) achieving economic aspects ; the operators are educated and trained in such a way that they can take care of the plant as the unit is completed and ready for commercial production .

1.2 Retraining

As a complement to basic training , retraining is implemented every year or two years .The need for retraining will be concluded in :

(1)theoritical and practical training education , and
(2)analyzing malfunctions and abnormal situations .

1.3 Training capabilities

El Dabaa basic simulator has the capability to simulate three separate modules ; PWR basic principle simulation (BPS-module), high level incidents simulation for PWR, (ACCID-module) and PHWR basic principle simulation (CANDU-module) .

The simulator has several training capabilities ;

(1) planning of the excercises ,
(2) changing the time scale ; accelaration factor up to 10000 and deaccelaration down to 0.01 ,
(3) stop or freez the exercise to study and analyze the situation
(4) change easly from one state to another ,
(5) demonstration of both low level and high level incidents shown in tables (1) thru (3) ,
(6) taking a snap shot at any instant during exercise ,
(7) replay of the exercise ,
(8) demonstration Xenon poisoning simulation .

5. TRAINING EXPERIENCES GAINED BY USING EL DABAA SIMULATOR

(1) An introductory theoritical training course is essential to the trainees having different backgrounds ,
(2) active participation of trainees during practical training assist them to be acquainted with the simulator and the operational procedures ,
(3) the trainees gained a comprehensive understanding of nuclear power plant operation principles either in normal or abnormal conditions ,
(4) it recommended to analyze the status of the plant during the exercises ,
(5) well planned training program will upgrade training benefits.

TABLE 1. MALFUNCTIONS OF THE BASIC PRINCIPLES MODULE

type	nature of the incident	parameter	adjustment range
CORE	Control rod failure	reactivity	0-1000 pcm
CORE	Control rods drive failure		0/1
CORE	Control rod cluster ejection	reactivity	0- 200 pcm
CORE	Nuclear instrumentation failure		0/1
CORE	Fuel cladding		0/1
CORE	Reactor trip		0/1
LOSS	Automatic reactor trip failure		0/1
LOSS	Loss of components cooling		0/1
CS	Average temperature control system freeze		0/1
PUMPS	Primary pump #1 failure		0/1
PUMPS	Primary pump #2 failure		0/1
PUMPS	Primary pump #3 failure		0/1
LEAKS	Small primary leak	flowrate	0 - 30 m^3/h
LEAKS	Small primary leak CVCS 1	flowrate	0 - 30 m^3/h
LEAKS	Small primary leak CVCS 2	flowrate	0 - 30 m^3/h
PUMPS	RHRS failure		0/1
CS	Pressurizer pressure control system freeze		0/1
CS	Pressurizer control and CVCS failure		0/1
CS	CVCS failure		0/1
CS	CVCS opening		0/1
LOSS	Loss of protection system		0/1
LOSS	Loss of service water		0/1
SG	Steam generator discharge opening		0/1
CS	Atmospheric bypass discharge opening	opening	0 -100 %
CS	Condenser bypass discharge opening	opening	0 -100 %
LOSS	Loss of condenser vacuum		0/1
CS	Condenser level control system freeze		0/1
TG	Active power positive step	power	0 -100 %
TG	Active power negative step	power	0 -100 %
TG	Turbine trip		0/1
TG	Generator trip		0/1

TABLE 2. MALFUNCTIONS OF THE "HIGH-LEVEL INCIDENT" MODULE

The incidents, or accidents, available in the "High-Level Incident" module are the following :

nature of the incident	parameter	adjustment range
Steam Generator Tube Rupture	nb of tubes	0 - 10
Cold Leg LOCA	diameter	0 - 203.2 mm
Hot Leg LOCA	diameter	0 - 203.2 mm
Main Steam Line Break	diameter	0 - 753 mm
Relief Valve failure		0/1
Safety Valve failure		0/1
Loss of Electrical power		0/1
One coolant Pump failure		0/1
Two coolant Pumps failure		0/1
Three coolant Pumps failure		0/1
Loss of Normal Feedwater		0/1
Loss of NFWS and AFWS		0/1
Main Feed Line Break	diameter	0 - 400 mm

"0/1" refers to binary incidents, for which failure is defined.

TABLE 3. THE INCIDENTS AVAILABLE IN THE CANDU MODULE

nature of the incident	parameter	adjustment range
Control Rod failure	reactivity	0-100 pcm
Reactor Trip		0/1
Small Coolant Circuit break	flowrate	0-100 kg/s
Small Moderator break	flowrate	0-900 kg/s
Loss of Primary Pump #1		0/1
Loss of Primary Pump #2		0/1
Loss of both Primary Pumps		0/1
SG1 Feedwater opening	opening	0 - 100 %
SG1 Safety Valve opening		0/1
SG2 Feedwater opening	opening	0 - 100 %
SG2 Safety Valve opening		0/1
Loss of Condenser Vacuum		0/1
TurbineTrip		0/1

"0/1" refers to binary incidents, for which failure is defined.

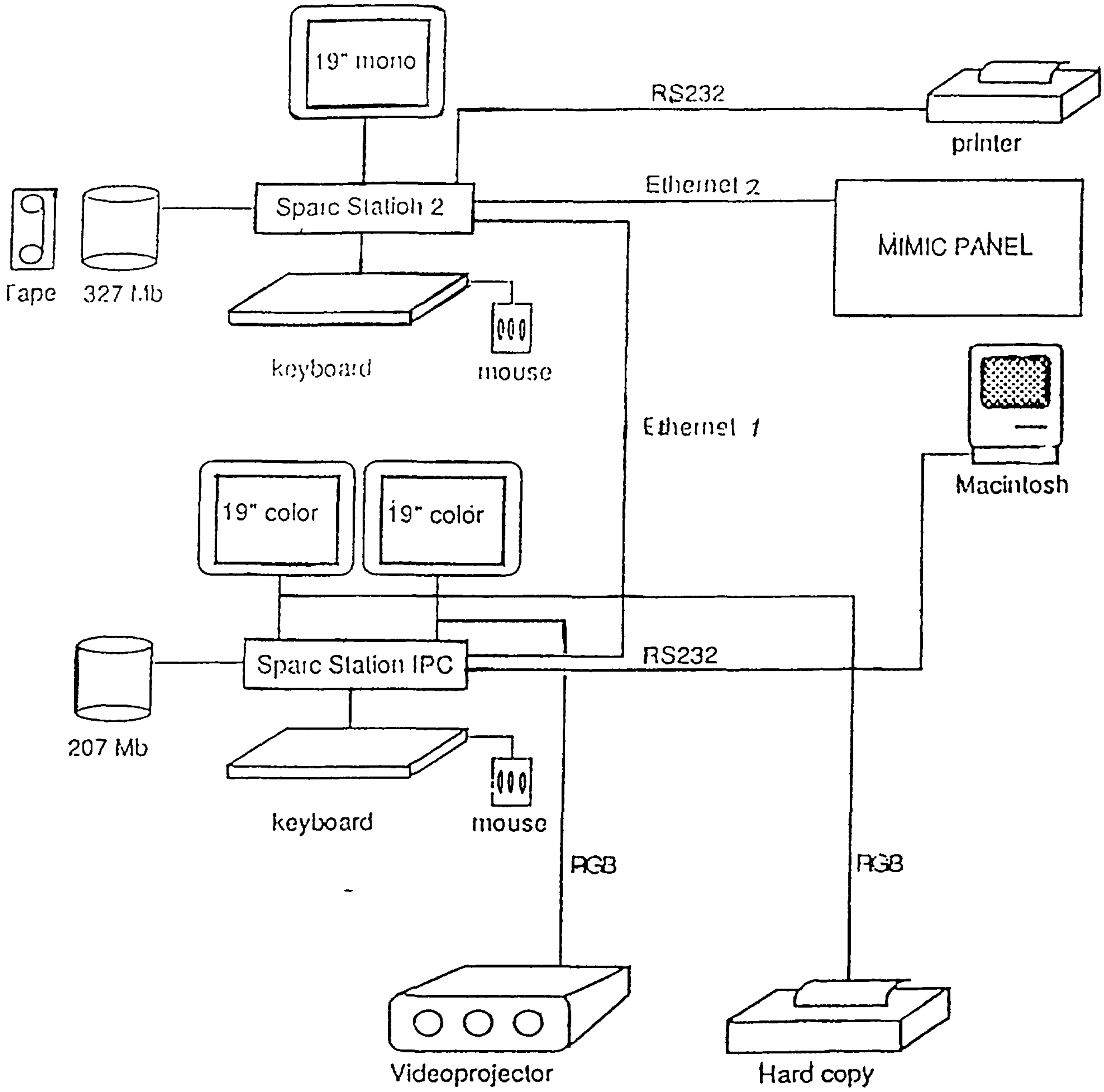


Figure 1. Hardware architecture

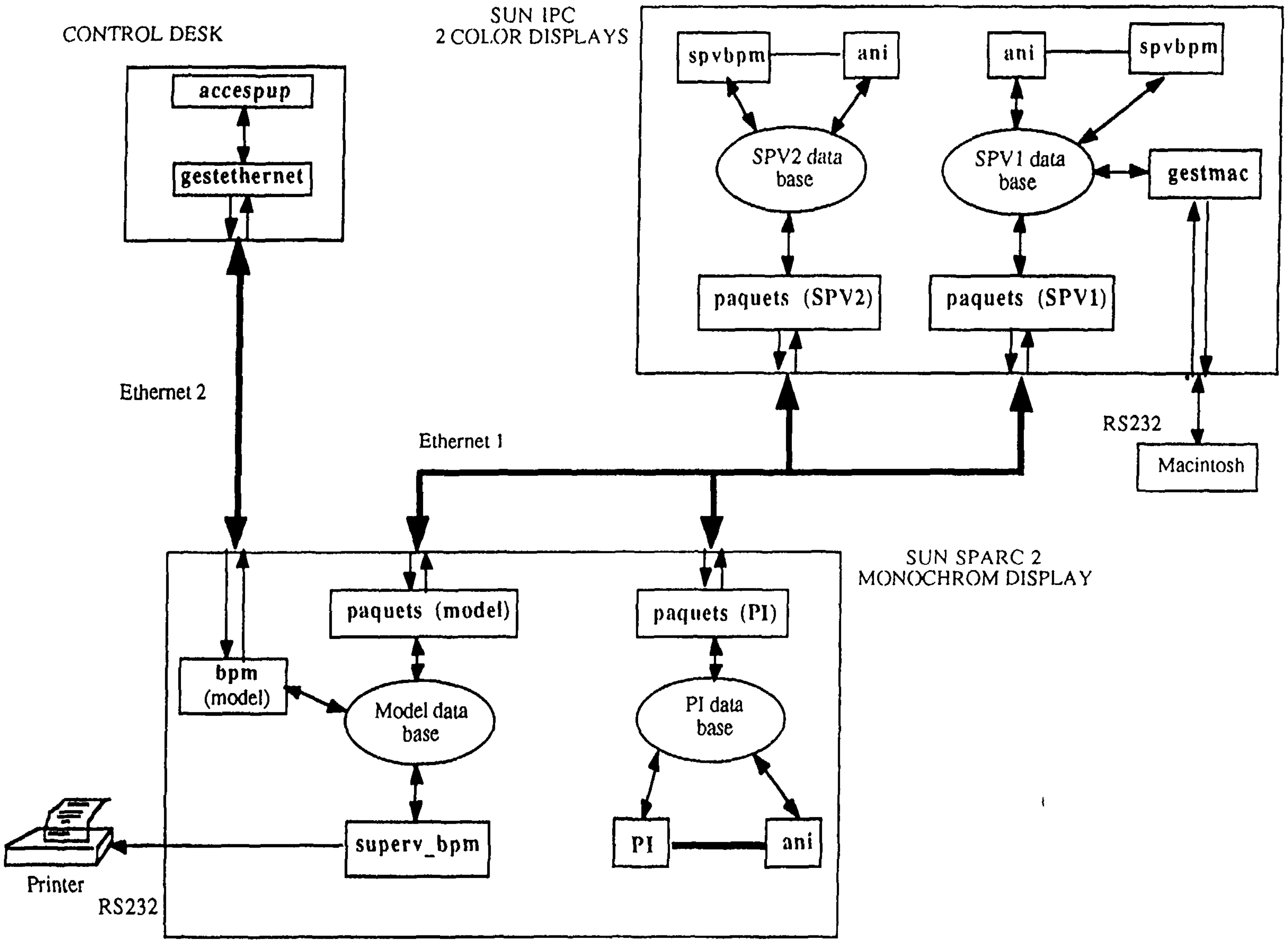


FIGURE 2 . PWR BASIC PRINCIPLES SIMULATION ARCHITECTURE

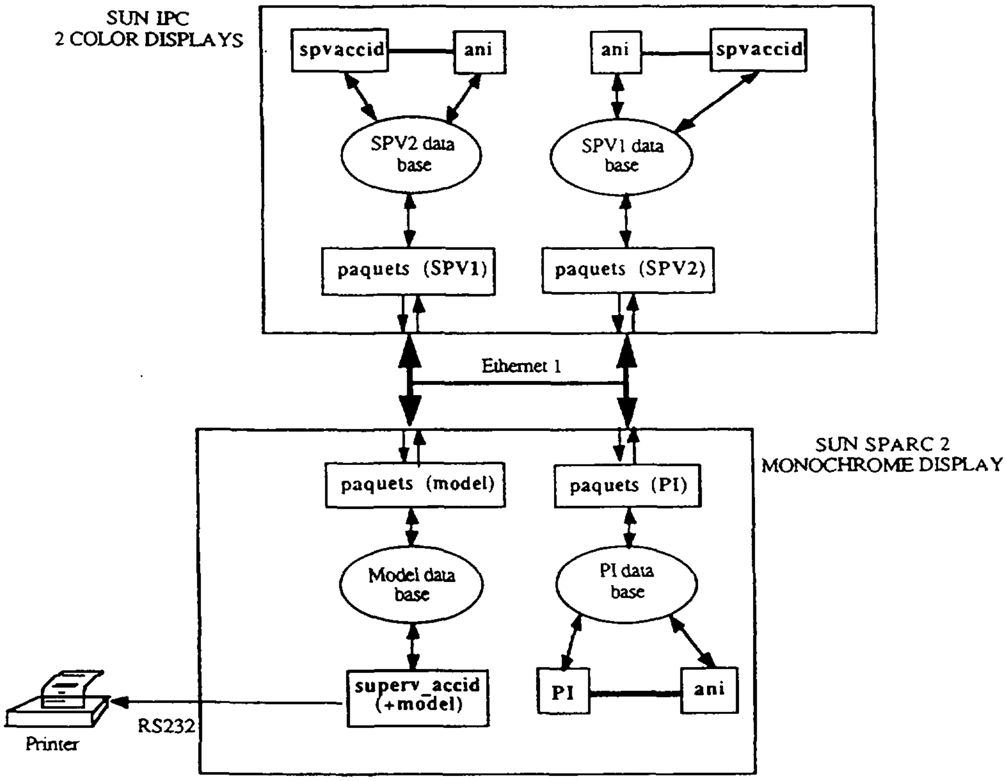


FIGURE 3 . HIGH LEVEL INCIDENT SIMULATION ARCHITECTURE

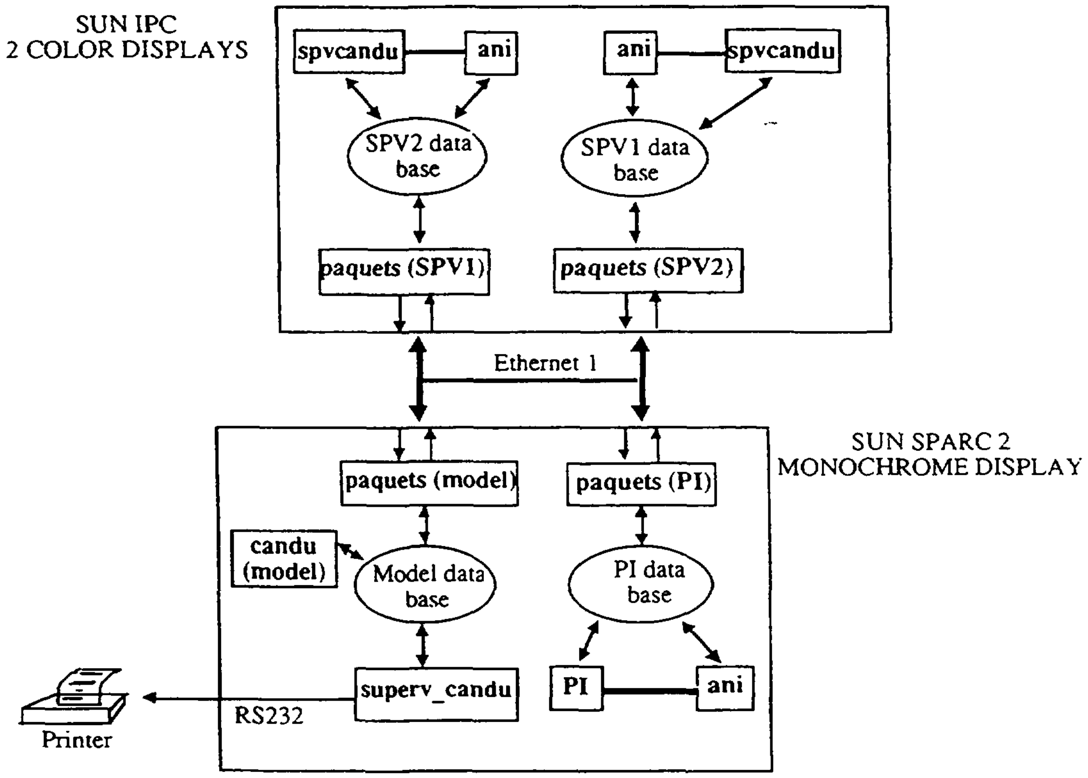


FIGURE 4 . CANDU SIMULATION ARCHITECTURE

6. CONCLUSION

The basic simulator have been used successfully as a main tool to develop the capabilities of the NPPA staff. It is planned to upgrade the capabilities simulator staff .

REFERENCES

(1) Bengt Ahlamann ; Utilities Views of the Need of Operation Training on a Simulator , IAEA Specialists Meeting on Simulator for Training of Nuclear Power Plant Operators and Technical Staff , Studusvik , Sweden , Oct. 1979 .

(2) El Dabaa Simulator Software Manual , CEA/CENG Nov. 1991 .

(3) El Dabaa Simulator Hatdware Manual , CEA/CENG, Nov. 1991 .

(4) S. B. Abdel Hamid and Wasfy Abdel Wahab, Basic Simulator for PWR and PHWR plants, Conference of Nuclear Science and Application , Cairo , Egypt , Feb. 1992 .

DEVELOPMENT OF SIMULATORS FOR SMRs

XA9846729

M.N. JAFRI, P. BUTT
Pakistan Atomic Energy Commission,
Islamabad, Pakistan

Abstract

The first step towards the introduction of simulator culture in Pakistan Atomic Energy Commission (PAEC) was taken in 1976 when the work on the development of analog computer based Basic Principles Simulator of KANUPP was initiated to test the Modified Real Time Control software. The project was revitalized in 1988 to develop a digital computer model of major KANUPP systems along with real-time simulation executive software and man-machine interface software in FORTRAN-77 on VAX-11/780. This simulator was later ported on microcomputers using C-language with four display units, entitled as KANUPP Test Simulator (KTS), and is presently being employed for training and teaching at KANUPP Inplant Plant Training Center(INPTC) and Institute of Nuclear Power Engineering (KINPOE) respectively. The acquisition of Advanced Process Simulator Software (APROS) in 1991 laid the foundation for establishing an enhanced simulator environment to meet the present day requirements and scope of simulators. The development of APROS based Engineering Analyzer for KANUPP was initiated in 1992. With the contract for 300 MWe two loop PWR nuclear power plant from China the development of Full Scope Training Simulator for CHASNUPP-1 was initiated in 1993, which is scheduled to be completed in end 1997. The process of development of simulators for SMRs provided the opportunities to achieve indigenous capabilities for the design and development of control room with real time I/O interface, real time data communication using RTPs and a general purpose security guarded real-time graphics display system, as well as considerable experience on the design and development of SMRs simulators. This paper presents information on the present state of SMRs simulator development and the achievements made in PAEC.

1.0 Introduction

Simulators by definition are the presentation of physical or technological systems without their being actually present. These representations depict the dynamic behavior of actual systems. Simulators are widely being used for design, development, modifications, analysis, design of control strategies and safety aspects, and to impart training for maintaining competence to ensure safe and reliable operation of nuclear power plants. The present trends are to set up training of plant personnel in the utilities planning for their first nuclear power plants or upgrading training facilities for existing plants by introducing the use of simulators.

The identification of operator errors in the Three Mile Island and Chernobyl accidents have demonstrated that human error has been a large contributing factor and that inadequate training or knowledge of the plant personnel has contributed to this aspect. This realization has promoted the evolution of safety culture resulting in the development of simulators which could be capable of providing an in-depth knowledge of normal and abnormal plant operations, as well as the consequences of ensuing events related to abnormal behavior. This requires the need for the adoption of a systematic approach to simulator training to establish and maintain the qualification and competence of nuclear power plant operations personnel for all operating organizations. Furthermore, the revolution in the field of computer technology has led to a major breakthrough and it has now become possible to develop plant analyzers on desktop computers under real time operation. Simulators are now being used advantageously in strengthening operator skills and proficiency under a variety of simulated plant conditions in probabilistic risk assessment, confirmation and modification of operating procedures, development of accident management strategies and severe accident studies.

For a country like Pakistan, which is in the process of implementing nuclear power program, operation and maintenance is a major challenge which could only be met by establishment of simulator facilities. In realization of the above, PAEC decided to initiate a programme for the development of SMRs simulators. This program gradually evolved and ultimately prompted the development of engineering analyzers and Full scope training simulators for PHWRs and PWRs.

The IAEA's proposal for the development of desktop simulators reinforces the PC based simulator programme launched at PAEC. Such simulators can effectively be used to facilitate understanding of dynamical behavior of systems and sub-systems of SMRs. The desktop multifunctional simulators are cost economic and can effectively be utilized for post graduate students, engineers and technical personnel for safe and efficient operation of nuclear power plants.

The indigenous efforts made towards the development of simulators for SMRs in Pakistan are presented in the proceeding sections.

2.0 Development of Basic Principles Simulator of KANUPP

The initial work on a simulator for KANUPP was initiated in 1976. This was done by developing an Analog-Hybrid computer based Basic Principles Simulator of KANUPP for testing the Modified Real Time Control Software of KANUPP before implementing on plant control computers.

The Computer Control System at KANUPP originally developed by GEC, Canada was reliability oriented rather than being availability oriented. Therefore, the control algorithms were modified to make the operation of the plant more efficient and safe. The changes in the control software were quite substantial and the possibility of wrong coding and bugs could not be ruled out. In case, the new software package was loaded and employed for plant control without prior testing, it could result in certain incident. Therefore, the new software package had to be tested under simulated plant conditions.

In order to carry out the testing of the modified software package, the following equipment was used: a digital computer similar to the computer used for plant control, a medium sized analog-hybrid computer to simulate plant systems, analog and digital check out panels, control room mock up and reactor regulation console, peripheral devices like teleprinter, teletypes, recorders, etc.

The following systems were simulated on the analog computer: Reactor kinetics model, Primary heat transport system, Turbine steam and boiler feedwater system, Moderator and Helium gas system, Blow-off and turbine unloader. In addition to the above systems, the digitally simulated Rod control system and Plant load control system were integrated with the analog model.

The reactor kinetics was represented by assuming a point reactor with six delayed neutron groups. The process system behavior was presented by linear and non-linear differential equations, look up tables and algebraic relations. This simplified simulated system was interfaced with simplified console and the control computers. The control software was loaded on the digital computers (GEPAC 4020) and the usual dynamic tests on the plant model were carried out [1]. The project was completed in 1979. The organization of the simulator is shown in Fig.1

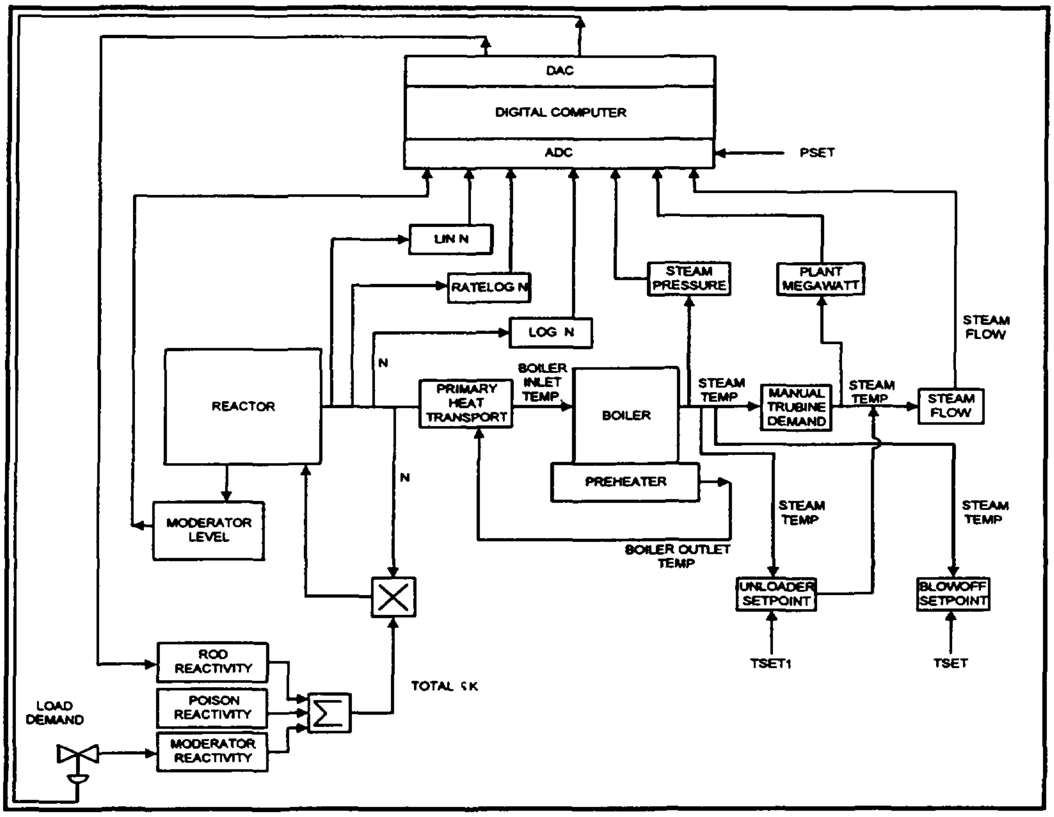


Fig. 1 Organization of Basic Principles Simulator of KANUPP

This project was revitalized again in 1988 with the objective to develop the digital computer model. An enhanced version with additional systems was developed in FORTRAN-77 on VAX 11/780 along with indigenously developed real time management software and man-machine interface software for concurrent processing and interactive graphic display system for presentation of mimics, alarms and critical parameters. This Basic Principles, Reduced Scope Simulator model of KANUPP entitled "KANUPP Test Simulator" (KTS) was validated against the recorded KANUPP transients and visual experience of senior KANUPP operators and managers.

However, in view of the non-existence of a VAX 11/780 computer at KANUPP/INPTC, the above simulator was ported on microcomputers using C language with four display units. It is being employed at KANUPP Institute of Nuclear Power Engineering (KINPOE) for teaching and at In-Plant Training Center for training of engineers and study of plant dynamics under normal and abnormal conditions.

The simulator software is integrated in form of four software modules:

- plant model software
- control programs software
- display management software
- simulation executive software

KTS comprises of the following hardware:

- Simulation Computer (IBM PC or Compatible)
  - Pentium 90 MHz, 16 MB RAM, 2 GB Hard Disk
  - SVGA display card 1 MB display RAM and 17" color monitor
  - 2 serial 1 parallel port, keyboard and mouse
- Display computer (IBM PC or Compatible)
  - Intel 486 DX4, 16 MB RAM, 2 GB Hard Disk
  - 4 port MVP card with 1 MB Display RAM for each port and 4 monitors
  - 2 serial 1 parallel port, keyboard and mouse)
- Text printer
- HP Inkjet Color Printer.

3.0 Establishment Of Simulation Environment

Advance Process Simulator Software (APROS) [2] was acquired from VTT, Technical Research Center of Finland in 1991 for VAX/VMS computer system. A six months joint engineering of ICCC and VTT for developing simulator using APROS was carried out at VTT. As a result, a basic Design and Development simulator of CHASNUPP-1 NPP covering all the major systems was developed.

In 1992, two VAX 3600 computers each with 16 MB RAM, two 550 MB hard disks and VMS operating system, interconnected on DECNET LAN were installed. On the network, six PCs were connected with X-terminal emulation. In order to accommodate more users, four text terminals were interfaced with the computers on RS232 serial link. One heavy duty printer was also made available. APROS was loaded on the aforementioned computers and simulation environment was made available to the application developers.

In parallel to the APROS environment on VAX/VMS, a graphics display system of APROS was installed on PCs. This system communicates with APROS application on VAX/VMS over network for the collection of simulated data on regular intervals, presents the dynamic data in plant mimic diagrams, generates off-line/on-line trends of preselected variables and creates history, etc.

Considering the emerging technical needs of displays versus constraints of the available graphics display system and plant process computer requirements for Full Scope Training Simulator (FSTS), indigenous development of a graphics display system package commenced in 1992. The indigenously developed package has now all the necessary features of commercially available packages [3]. The package is being continuously upgraded to accommodate all the emerging requirements and being used for the plant process computer of FSTS for CHASNUPP-1 (C1).

In 1993, it was decided to switch over to RSIC from CISC for more and economical computation power. Two SPARC computers, TATUNG Super COMPStation 7/30 computers with SOLARIS operating system were acquired. The two computers were installed in a network environment with two X-terminals and more than sixteen PCs as X-terminals. The printer and text terminals were also added to environment. Once the environment was established, APROS was made available on these computers.

For the real time I/O interface of control room and simulation computer, an indigenous effort of using micro controller 8031 has been made. As a result, a complete real time I/O interface based on 8031 micro controller and mini control room for the KANUPP Analyzer is developed [4]. The interface is capable of handling 444 digital inputs, 963 digital outputs and 9 analog outputs.

For the real time I/O interface of FSTS for C1, Real Time Products (RTPs) were chosen. In order to have a complete command to use RTPs, a representative panel of C1 control room has been designed and developed [5]. The representative panel is interfaced using RTPs with simulation computer and it can handle a maximum of 112 digital inputs, 1836 digital outputs, 32 analog inputs and 32 analog outputs. The existing simulation environment is shown in Fig.2.

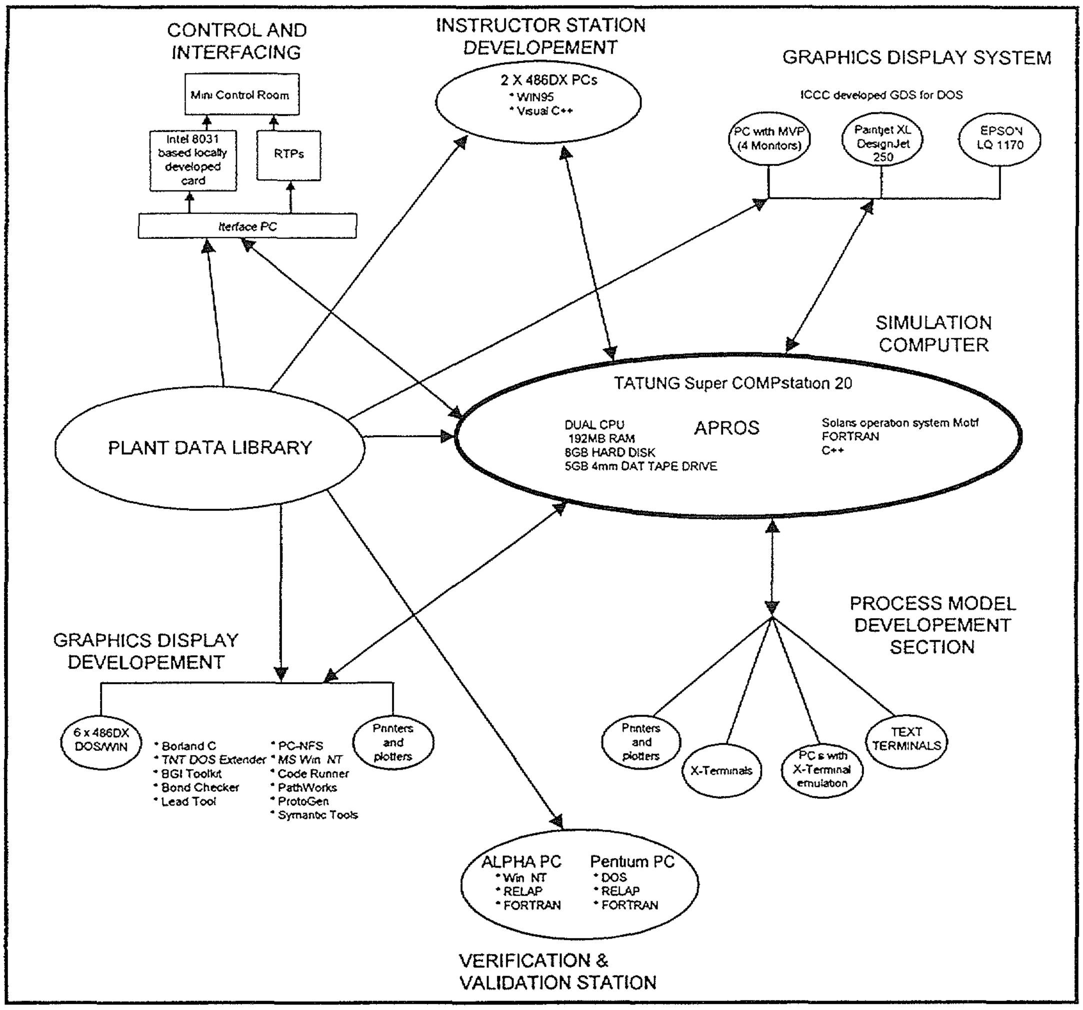


Fig. 2 Simulation Environment at PAEC

4.0 Engineering Analyzer of KANUPP

In 1992, PAEC decided to develop an Engineering Analyzer for KANUPP using Advanced Process Simulator Software (APROS). This engineering analyzer is to be used for training, understanding the dynamic behavior of the plant including normal and abnormal transients, study of plant modifications, procedure development, alarm setpoint studies and for testing the control loops after backfitting of C&I systems.

Keeping in view the plant personnel's requirements, the depth of simulation for each of the systems considered was discussed in detail and the scope of the Analyzer was defined. Work on plant data collection started in Dec. 1992 and was completed by the end of 1993. Thereafter, the development of individual models of systems was initiated and all systems contributing to the plant dynamics were included. The standalone models were prepared and tested in steady state. The individual models were finally integrated in Dec. 1994. This completed the first phase of development and the Analyzer operated at 100% steady state power.

The second phase shall cover the inclusion of moderator level control and properties of heavy water in APROS software. This shall be completed under Technical Co-operation Program of IAEA, Project PAK/4/039 "Development of a Simulator for Nuclear Power Plants". Thereafter, PAEC engineers shall implement the interlocks and operating sequence logic and validate the simulator against operational transients of KANUPP.

The summary for the present status of KANUPP Analyzer is as follows:

- The dual core concept represented by utilizing the 1-dim dynamic reactor core modules of APROS.
- A total of 33 systems have been simulated, including the complete PHT system with six boilers, three per loop.
- A total of 498 nodes with 280 nodes for two-phase flow and 218 nodes for homogeneous flow
- A total of 570 branches with 314 branches for two-phase flow and 256 for homogeneous flow
- A total of 364 heat nodes and 270 heat branches
- 6 turbine sections, 19 pumps and 158 valves
- The system meets ANSI-ANS/3.5 requirements at 100% steady state power
- The Analyzer is interfaced with an indigenously developed mini control room, Fig.3, and real time I/O interface using 8031 micro-controllers with dual port RAM concept [4].
- An indigenously developed security guarded real-time Graphics Display System (GDS) provides an on-line facility for plant diagnostics and analyses by presenting process system displays, flexible trend diagrams, alarm surveillance and log-books [3].

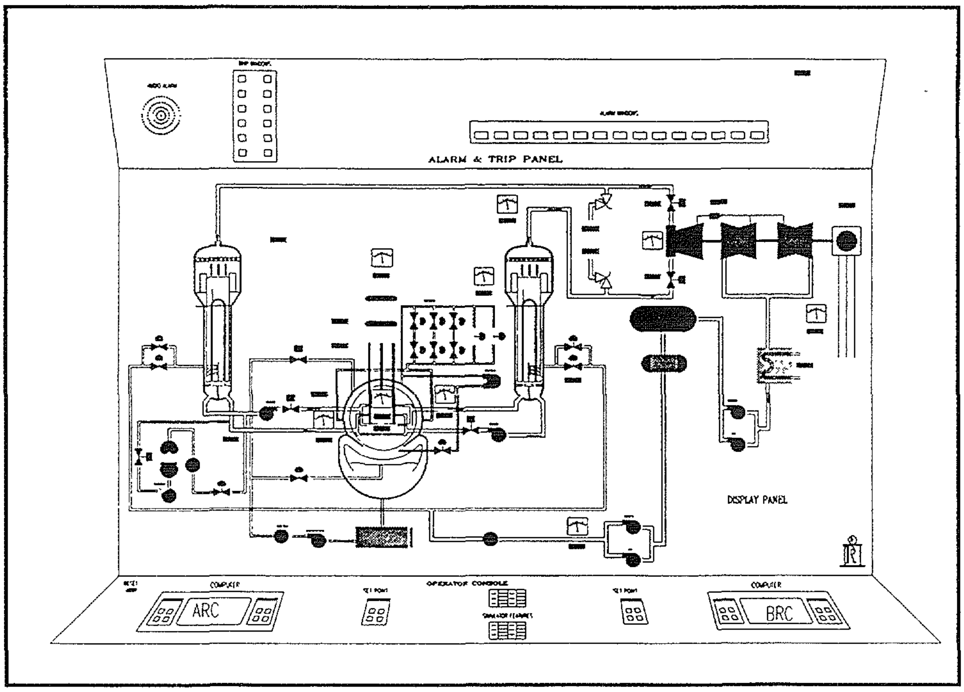


Fig. 3 Mini Control Room of KANUPP Analyzer

KANUPP Analyzer comprises of the following hardware:

- simulation computer
  - TATUNG Super COMPStation 7/30 with
    - 128 MB memory and 4 GB Hard Disk
    - CD ROM and 525 MB Wangteck Tape
    - GX Graphics Card and 17" color monitor
    - 2 serial 1 parallel port, keyboard and mouse
    - Ethernet card
- Display computer
  - IBM PC or Compatible
    - Intel 486 DX4, 16 MB RAM, 2 GB Hard Disk
    - 4 port MVP card with 1 MB Display RAM for each port
    - 4 SVGA color monitors
    - 2 serial 1 parallel port, keyboard and mouse
    - Ethernet Card)
- Text printer
- HP Inkjet Color Printer

The hardware organization is shown in Fig.4.

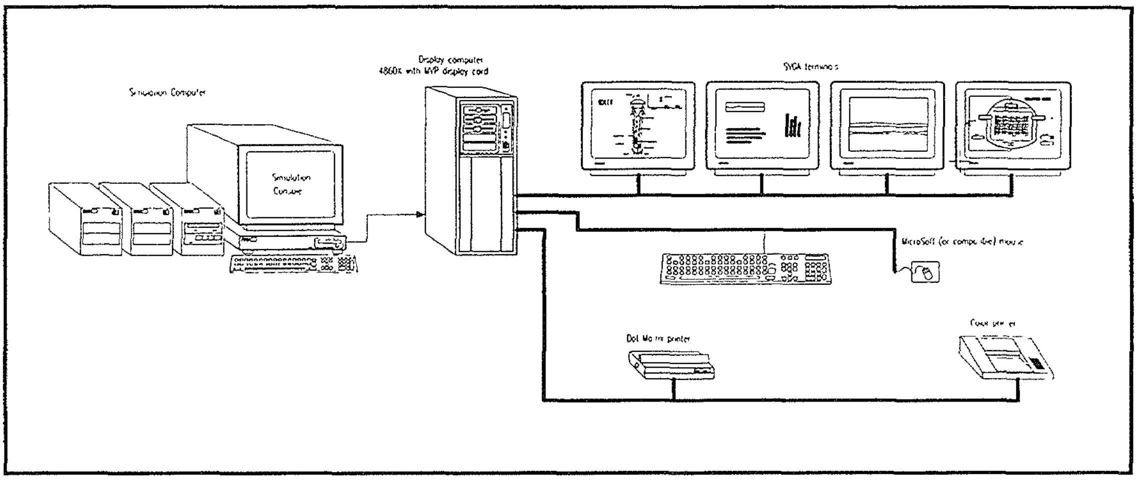


Fig. 4 Hardware Organization of KANUPP Analyzer

5.0 The Full Scope Training Simulator of CHASNUPP-1

5.1 Background

The 300 MWe two loop PWR nuclear power plant contracted from China is expected to come in operation by the end of 1998. Training of plant operators through simulators is now accepted as standard practice. It was, therefore, decided in 1994 that PAEC should develop a Full Scope Training Simulator for CHASNUPP-1 based on APROS simulation environment. The replica control room of CHASNUPP-1 FSTS is being acquired from Shanghai Nuclear Engineering Research and Design Institute (SNERDI). The elements of Full Scope Training Simulator for CHASNUPP-1 are shown in Fig.5.

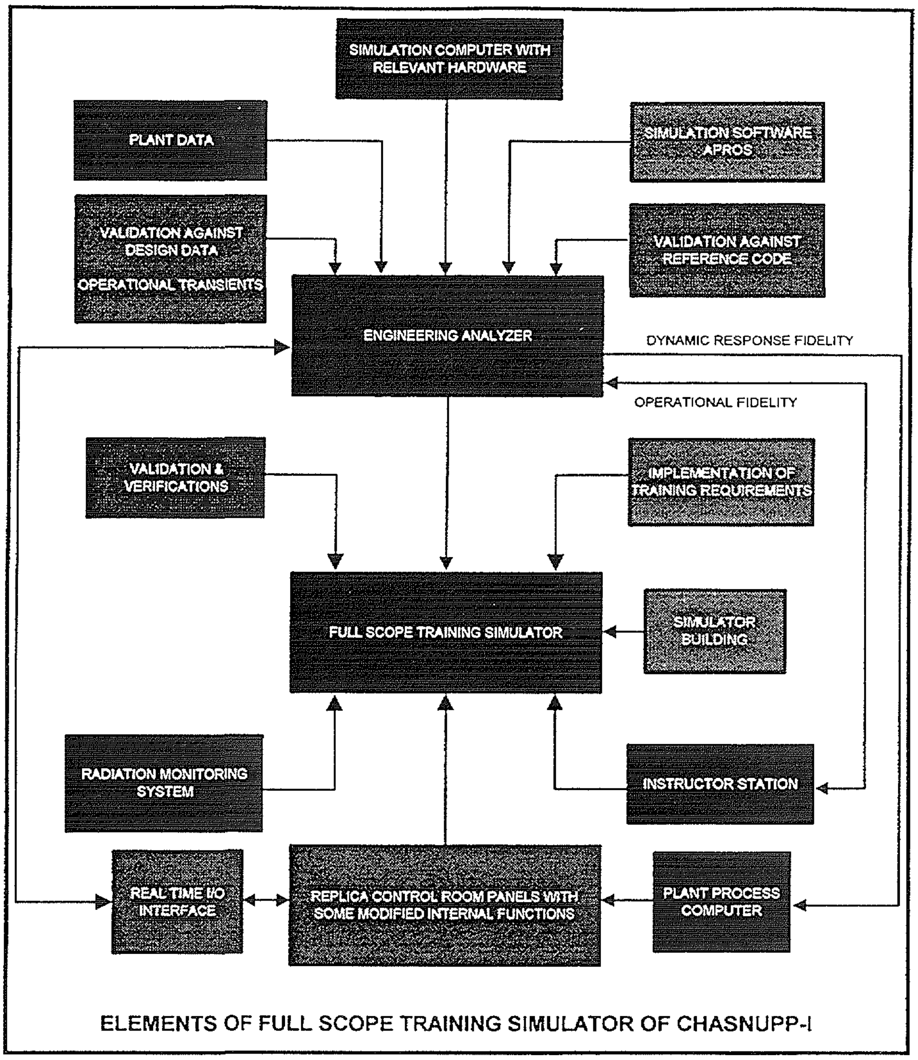


Fig. 5 Elements of Full Scope Training Simulator for CHASNUPP-1

5.2 Salient Features and Scope of Simulation

The Full Scope Training Simulator is to be used for initial and continued training of CHASNUPP-1 plant personnel to enable them to perform their tasks and functions safely and efficiently. It shall provide a valuable opportunity to develop and assess operating team skills, demonstrate and practice operator response, achieve high fidelity in training and permit observations of trainee's performance. The simulator is also intended to be used for the confirmation of plant control strategies and operating procedures and study of plant modifications.

The simulator and the training programme shall meet the ANSI/ANS-3.5 and 3.1 requirements in terms of physical, dynamic and operational fidelities. The scope of process simulation covers the main energy generation and conversion cycle with associated auxiliary systems, instrumentation and control systems, protection systems, engineered safety features and electrical power distribution systems.

The modelled reactor core is 1-dim dynamic (axial neutron flux distribution) and 3-dim static (radial neutron flux distribution). For 1-dim dynamic representation, the core has been axially divided into 20 sections and for 3d static flux into 37 representative channels each centered around a rod cluster assembly. The neutronic computations are based on 2-energy group neutron diffusion with six groups of delayed neutron precursors.

Physical models adapted to represent plant dynamics are based on 2-phase flow (5-equation drift flux model) for the Reactor Coolant system including the pressurizer and its relief system as well as the Main Steam and Feedwater sub-systems of the Nuclear Island. Whereas, homogeneous flow (3-equation model) has been adopted for the nuclear auxiliary systems and the systems of the conventional island. Systems that are not important for the dynamics of the total plant are either represented in a simplified form or presented by logical simulation.

Simulation models are currently based on the available design data supplemented by accumulated experience of PAEC engineers involved in design of the reference plant. The simulation models will ultimately be tuned by use of data obtained in pre-operational tests and the initial start-up tests of CHASNUPP-1.

5.3 Simulator Development Strategy

A proposal for the design, development and building of the Full Scope Training Simulator for CHASNUPP-1 NPP was prepared in 1993 [6]. The proposal contained the details for the training objectives, functional requirements and performance criteria for the development of FSTS. It also contained details of the software being employed for the development of FSTS and the proposed scope of simulator. The proposal was discussed and mutually agreed between the supplier and the end user which are basically sister organizations of PAEC. As a follow up the scope and extent of simulation was prepared and mutually agreed upon [7]. Keeping in view the local training needs, a systematic methodology for the development and implementation of initial and continued training programme in a cost effective manner was prepared [8].

The plant data was extracted from design documents and other relevant resource documents related to the CHASNUPP-1, and a database in Micro Soft Excel is being generated. Thereafter, the development of individual models for the systems and sub-systems was initiated. The standalone models were prepared and tested for steady state performance.

The individual models have been integrated and are being tested in integrated state. This will result in an Engineering Analyzer, which will be validated and verified for normal operations, operational transients and abnormal transients against the benchmarks generated by reference codes. The analyzer shall then be integrated with the replicated control room of FSTS.

The validation and verification is planned to be carried out jointly by PAEC, SNERDI, Operations experts from QINSHAN NPP and APROS designers from VTT, Finland. A systematic methodology to perform and document the simulator verification and validation has been prepared [9].

5.4 Present Status

- A total of 87 systems have been simulated
  - This includes 34 systems for the process and 53 systems for the EI&C
  - Seven of the process systems have been logically simulated
- Process simulation is currently based on
  - a total of 650 nodes with 150 nodes for two phase flow and 500 nodes for homogeneous flow
  - a total of 750 branches with 180 branches for two phase flow and 570 for homogeneous flow
  - A total of 264 heat nodes and 172 heat branches
  - 8 turbine sections, 24 pumps and 297 valves

- Nodes and branches may be modified during the final integration phase, if required
- For the replicated control room a total of 3970 digital inputs, 4318 digital outputs, and 1219 analog outputs are to be generated
- Plant Process Computer and Instructor Station are being developed indigenously

5.5 Hardware Organization

The hardware organization of FSTS for CHASNUPP-1 comprises of:

- Simulation Computer
  - TATUNG Super COMPStation 20
    - Super SPARC CPU 80 Mhz
    - 128 MB memory and 8 GB Hard Disk
    - CD ROM and 8 GB DAT Tape
    - TGX Graphics Card and 17" color monitor,
    - 2 serial, 1 parallel port, keyboard and mouse
    - Ethernet card
- Plant Computer, Instructor Station And RMS
  - DEC Venturis 575 (7Pc's)
    - Intel Pentium CPU 75 MHz, 16 MB RAM, 540 MB Hard Disk
    - SVGA Display card with 1 MB Display RAM and 19" color monitor
    - 2 serial 1 parallel port, keyboard and mouse
    - Ethernet Card
- Interface PC's with RTPs for I/O with Control Room
- Replica of Plant Control Room

The hardware organization is shown in Fig.6.

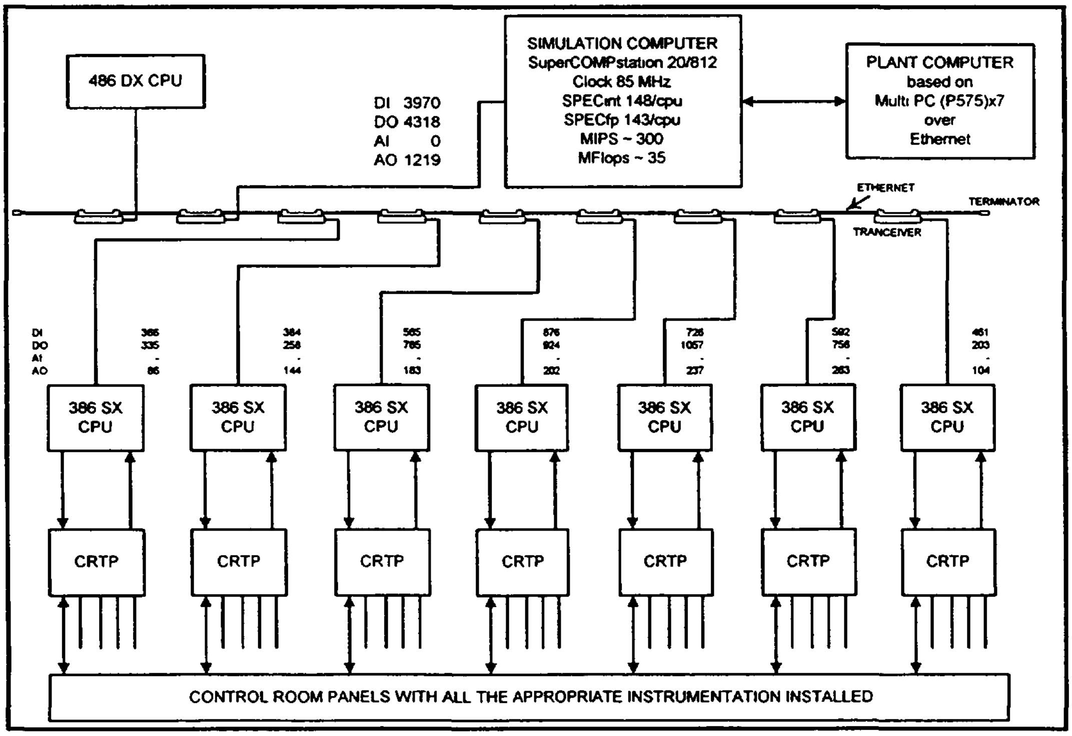


Fig. 6. Hardware organization of full scope training simulator for CHANUPP-1

6.0 Technical Fallouts of the Simulator Development Programme

The capabilities achieved/expected to be achieved by PAEC during the indigenous efforts in developing SMRs simulators are the following:

- Design, develop and build training simulators including development of microcomputer based Compact Simulators for nuclear power plants for real time decision making, development of engineering analyzers using soft panel approach and development of full scope training simulators based on APROS simulation environment.
- Development of MMI software as an aid to NPP operators
- Control, Instrumentation and protection system studies to modify, design and develop Protection and Regulation systems for NPP
- Development of software for Instructor Stations
- Design and development of Control Rooms and Real Time I/O Interface
- Development of general purpose security guarded real-time Graphics Display Systems (GDS), for providing on-line facility for plant diagnostics and analysis by presenting process system displays, flexible trend diagrams, alarm surveillance and log-books, etc.

7.0 Conclusion

The PAEC has achieved a vast experience in the field of training simulators, operator aids, real time data communication and high speed graphical display systems. It is engaged in the development of engineering analyzers, soft panel based training simulators and SCADA systems. It has the capability to modify C&I systems and Control room design for SMRs. PAEC can take up similar projects for other countries in collaboration with IAEA.

REFERENCES

1. Jafri M.N., "KANUPP Simulation For Testing Of New Computer Control Software And Studying Plant Performance Under Transient Conditions, KANUPP-STR-80-1, March, 1980."

2. Silvennoinen E., Juslin K., Hanninen M., Tiihonen O., Kurki J. Porkholm K., "The APROS Software For Process Simulation And Model Development, Technical Research Centre of Finland, Espoo, 1989."

3. Jafri M.N., Ali S., Mustafa G., "Graphics Display System (GDS), TR-ICCC-15, Nov. 1994"

4. Jafri M.N., Ali S., Ahmed Z., "Mini Control Room For KANUPP Analyzer, TR-ICCC-25, 1995"

5. Jafri M.N., Ali S., Ahmed Z., "Representative Panel for the FSTS of C-1, TR-ICCC-33, 1996"

6. Jafri M.N., "A Proposal For Indigenous Development Of 300 MWe CHASNUPP-1 NPP Simulator, TD-ICCC-4, June, 1993"

7. Jafri M.N., Ali S., Hassan M.W., "Detailed Specifications Of CHASNUPP-1 NPP Simulator, TD-ICCC-5, Feb. 1994"

8. Jafri M.N., Ali S., Hassan M.W., "A Proposal For Initial And Continuing Training Of CHASNUPP Plant Personnel On FSTS Of CHASNUPP-1, TD-ICCC-6, Sept. 1995"

9. Jafri M.N., Ali S., Hassan M.W., "A proposed Methodology For Qualification of FSTS For CHASNUPP-1, TD-ICCC-7, Sept. 1996"

LIST OF PARTICIPANTS

Abderrahmane, K.	Centre National de l'Energie, des Sciences et des Techniques Nucléaires (CNESTEN) 65, Rue Tansift Aguedal Rabat, Morocco
Al-Rushudi, S.	Institute of Atomic Energy Research King Abdulaziz City of Science and Technology P.O. Box 6086, Riyadh 11442 Saudi Arabia
Alter, J.	Israel Atomic Energy Research Commission (Permanent Mission of Israel to the IAEA) Anton Frankgasse 10 A-1180 Vienna, Austria
Aissa, M.	Société Tunisienne d'Electricité et du Gaz (STEG) 38, Rue Kamel Attaturk 1080 Tunis Tunisia
Ali, Hamza M.	Centre National des Sciences et Technologies Nucléaires (CNESTN) BP. 204, 1080 Tunis Cedex Tunisia
Al-Mugrabi, M.A. *(Scientific Secretary)*	International Atomic Energy Agency (IAEA) Wagramerstrasse 5 P.O. Box 100 A-1400 Vienna, Austria
Balz, W.	European Commission Directorate General XII Rue de la Loi 200 B-1049 Bruxelles Belgium
Bapat, C,N.	Nuclear Power Corporation of India Ltd. Vikram, Sarabhai Bhavan Anushakti Nagar, Mumbai - 400094 India
Barakat, M.F.	Arab Atomic Energy Agency, AAEA P.O. Box 402 Elmanzah, 1004 Tunis Tunisia

Baranaev, Youry D.	Institute of Physics and Power Engineering Bondarenko Sq. 1 249 020 Obninsk, Kaluga Region Russian Federation
Bittermann, D.	Siemens, KWU Koldastrasse 16, Postfach 3220 91058 Erlangen, Germany
Butt, P.	Pakistan Atomic Energy Commission P.O. Box 1114 Islamabad, Pakistan
Chermanne, J.	Consulting in nuclear and renewable energies 40 avenue des bécasinnes B-1160 Brussels Belgium
Cinotti, L.	ANSALDO, Nuclear Division Corso Perrone, 25, 16161 Genoa Italy
Fiorini, G.L.	CEA, Direction des réacteurs nucléaires Département d'études des réacteurs Centre d'études de Cadarache DER/SIS Bat 211, B.P. No. 1 F-13108 Saint-paul-lez Durance Cedex France
Ghurbal, S.	TNRC, Ministry of Scientific Research P.O. Box 30878 (Tajoura) Tripoli, Libyan Arab Jamahiriya
Gibson, I.	Cat Bells, 3 Hardy Close Martinstown, Dorchester Dorset, DT2 9JS United Kingdom
Giora, A.	Israel Atomic Energy Commission 1 Brazil Street Tel Aviv 69460 Israel
Haddou, Ait A.	Centre National de l'Energie, des Sciences et des Techniques Nucléaires (CNESTEN) 65, Rue Tansift Aguedal Rabat, Morocco

Hainoun, A.	Atomic Energy Commission of Syria P.O. Box 6091 Damascus Syrian Arab Republic
Hamroun, A.	Société Tunisienne d'Electricité et du Gaz (STEG) 38, Rue Kamel Attaturk 1080 Tunis Tunisia
Hart, R.	AECL-CANDU, Advanced Products 2251 Speakman Drive Mississauga, Ontario, L5K 1B2 Canada
Hoshi, T.	Office of Nuclear Ship Research and Development Japan Atomic Energy Research Institute (JAERI) 2-4, Shirakata-shirane, Tokai-mura, Ibaraki-ken 319-11 Japan
Hussain, B.	Karachi Nuclear Power Complex (KNPC) P. O. Box No. 3183 Karachi-75400 Pakistan
Issa, M.	Société Tunisienne d'Electricité et du Gaz (STEG) 38, Rue Kamel Attaturk 1080 Tunis Tunisia
Jerbi, H.	Centre National des Sciences et Technologies Nucléaires (CNESTN) BP. 204, 1080 Tunis Cedex Tunisia
Karouani,K.	Centre National de l'Energie, des Sciences et des Techniques Nucléaires (CNESTEN) 65, Rue Tansift Aguedal Rabat, Morocco
Kataoka, K. *(Co-Scientific Secretary)*	International Atomic Energy Agency (IAEA) Wagramerstrasse 5 P.O. Box 100 A-1400 Vienna, Austria
Kerris, M. A.	Centre de Développement de Systèmes Energétiques B.P. 180, Aïn Duccera Wilaya de Djelfa Algeria

Khamis, I.	Atomic Energy Commission of Syria P.O. Box 6091 Damascus Syrian Arab Republic
Kim, Hwa-Sup.	Korea Atomic Energy Research Institute P.O. Box 105, Yuseong Taejon, Korea 305-600 Republic of Korea
Kraiem, Hédi B.	Centre National des Sciences et Technologies Nucléaires (CNESTN) BP. 204, 1080 Tunis Cedex Tunisia
Ksouri, A.	Société Tunisienne d'Electricité et du Gaz (STEG) 38, Rue Kamel Attaturk 1080 Tunis Tunisia
Kupitz, J.	International Atomic Energy Agency (IAEA) Division of Nuclear Power and the Fuel Cycle Wagramerstrasse -5 P.O. Box 100 A-1400 Vienna
Marzouk, A.A.	SONEDE 23, rue Jawaher B.P. 1300, El-Nehu, Montfleury Tunisia
Minato, A.	CRIEPI, Ohtemachi Building 1-6-1 Ohtemachi Chiyoda-Ku, Tokyo 100 Japan
Moon, K.-S.	Korea Atomic Energy Research Institute P.O. Box 105, Yusong, Taejon 305-600 Republic of Korea
Omar, Abdel W.	Nuclear Power Plants Authority - NPPA 4 El - Nasr Avenue, Nasr City P.O. Box 108, Code No: 11381 Abbasia Cairo, Egypt
Ordoñez, Juan P.	INVAP S.E. Moreno 1089, C.C. 961 8400 San Carlos de Bariloche Río Negro, Argentina

Oukili, B.	National Office of Potable Water (ONEP) 6, rue Patrice Lumumba B.P. Rabat Shella, Rabat Morocco
Raguigui, N.	Centre National des Sciences et Technologies Nucléaires (CNESTN) BP. 204, 1080 Tunis Cedex Tunisia
Rahal, L. B.	Société Tunisienne d'Electricité et du Gaz (STEG) 38, Rue Kamel Attaturk 1080 Tunis Tunisia
Raouan, M.	Ministry of Industries 17 Av. K. Pacha 1002 Tunis Tunisia
Rapeanu, S.	National Agency for Atomic Energy 21-25 Mendeleev Street 70 168 Bucharest, Sector 1 Romania
Razek, Ibrahim D.A.	Atomic Energy Authority 101 Kasr El-Aini Street Cairo, Egypt
Sekimoto, H.	Research Laboratory for Nuclear Reactors Tokyo Institute of Technology O-okayama, Meguro-ku, Tokyo 152 Japan
Tabet, M.	Centre National de l'Energie, des Sciences et des Techniques Nucléaires (CNESTEN) 65, Rue Tansift Aguedal Rabat, Morocco
Tajeddine, B.K.	Office National d'Electricité (ONE) 65, rue Othmane Ben Affane Ex, aspirant lafuente, Casablanca Morocco
Titescu, G.	Institute of Cryogenics and Isotope Separation 1000, Ramnicu Valcea P.O. Box 10, Romania

Varga, P.	Soc. Com. Nuclearmontaj S.A. Bucharest Str. Berceni No: 104, Sector 4 Bucharest, Romania
Xue, D.	Institute of Nuclear Energy Technology Tsinghua University 100084 Beijing China
Yahia, B.	Centre National de l'Energie, des Sciences et des Techniques Nucléaires (CNESTEN) 65, Rue Tansift Aguedal Rabat, Morocco
Yasso, Khalil A.F.	Nuclear Power Plants Authority (NPPA) P.O. Box 8191, Nasr City 11371 Cairo, Egypt
Yaara, M.	SONEDE 23, rue Jawaher B.P. 1300, El-Nehu, Montfleury Tunisia

97-05334