## **IAEA-TECDOC-541**

# THE ROLE OF Advanced Nuclear Power Technologies in Developing Countries: criteria and design requirements

PROCEEDINGS OF TWO TECHNICAL COMMITTEE MEETINGS AND WORKSHOPS ORGANIZED BY THE INTERNATIONAL ATOMIC ENERGY AGENCY AND HELD IN VIENNA, 27–30 JUNE 1988 AND 6–9 DECEMBER 1988



A TECHNICAL DOCUMENT ISSUED BY THE INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1990

THE ROLE OF ADVANCED NUCLEAR POWER TECHNOLOGIES IN DEVELOPING COUNTRIES: CRITERIA AND DESIGN REQUIREMENTS IAEA, VIENNA, 1990 IAEA-TECDOC-541 ISSN 1011-4289

> Printed by the IAEA in Austria February 1990

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#### FOREWORD

The Agency has long been providing its Member States with a forum for the exchange of information on advanced nuclear power plants. The information was not limited to plants generating electricity, but included also plants for nuclear heat application as well as co-generation plants, the size of the plants ranging from very small to large. Developing countries investigating the use of nuclear energy in order to diversify their energy resources have been considering a variety of small nuclear power plants. In order to assist these countries, the Agency established guidelines for the introduction and planning of nuclear power.

Following the intention of a country to utilize nuclear energy, questions about the type and nature of the plants, amongst other questions, would have to be answered. On 27-30 June 1988 the Agency convened at its Headquarters in Vienna the Technical Committee and Workshop on Criteria for the Introduction of Advanced Nuclear Power Technologies for Specific Applications in developing countries.

The purpose of the meeting was to provide an opportunity to review and discuss factors necessary for selecting the type and nature of an appropriate advanced nuclear plant in developing countries. Emphasis was given to small co generating plants, because they are more versatile than plants designed only for electricity generation or heat production.

During 1988, a number of developing countries were making or planning to make feasibility studies on the application of advanced nuclear power plants and some of them have requested technical assistance from the Agency. These developments encouraged the Agency to convene on 6-9 December 1988 another Technical Committee and Workshop on Design Requirements for Applications of Advanced Concepts in developing countries.

The purpose of that meeting was to review and discuss national (advanced) nuclear power programmes in developing countries, the motivation of initiating these programmes and the specific design requirements as seen by the developing countries.

The first meeting was attended by 18 participants from 11 countries and the second meeting by 22 participants of 15 countries. The proceedings of both meetings are included in the present TECDOC.

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## TECHNICAL COMMITTEE MEETING AND WORKSHOP ON CRITERIA FOR THE INTRODUCTION OF ADVANCED NUCLEAR POWER TECHNOLOGIES FOR SPECIFIC APPLICATIONS IN DEVELOPING COUNTRIES

VIENNA, 27-30 JUNE 1988

#### SUMMARY REPORT

#### I. TECHNICAL COMMITTEE

1. Session 1 - National Programmes (Chairman, Livolant)

Session 1 was devoted to national programmes and policies in Developing Countries. Presentations were made for China and some Arabian Countries.

The Chinese presentation by Mr. Fang stated the fact that China has rich energy resources, mainly coal, which provides 70% of its primary energy production. However the sparse distribution of these resources in a large country like China, puts them far from the consumption centers, besides the difficulties associated with coal, such as transportation and pollution. So, the development of nuclear power is regarded as necessary. Projections made show an installed capacity of 4500 MWe in the year 2000 and 30,000 MWe in 2015, which should be provided by a domestic nuclear industry.

Studies are being made to use nuclear energy for district heating, based on 400 MWth light water reactors with natural circulation.

The presentation by Mr. Emara of Egypt showed that a large part of the energy resources consumed by the Developing Countries is in the form of oil, coming from the Developing Countries. Therefore, it is necessary for Arabian countries to shift to an alternative of domestic oil consumption, which could be nuclear energy, before the oil reserves are completely depleted. Moreover, the introduction of nuclear energy will be an efficient way to increase the industry level, especially when the reactors are bought with technology transfer and local participation.

The presentation stressed that an independent fuel cycle is also suitable for Arabain countries as a whole.

The two presentations show that in some countries exists a desire to develop nuclear energy, with a national contribution to the extent possible. Political considerations associated with economic reasons, however, could make the nuclear penetration very lengthy.

#### 2. Session 2 - Incentives for Advanced Nuclear Power Technology (Chairman: McDougall)

The advantages of nuclear heat supply, listed in Mr. Fang's paper, were fuel conservation, pollution reduction, and reduced atmospheric heating. Two feasibility studies on low temperature heating reactors in Northern China are underway. It is expected that the first unit will be built in the 1990s, after which these reactors may rapidly penetrate the market. With respect to future applications of nuclear energy, preliminary work on the FBR and fusion technology is carried out. The peaceful use of nuclear energy is expected to play an important role in China's development.

A paper presented by Mr. Djokolelono reported the prefeasibility study on the application of HTRs for recovery of heavy oil at the Duri oil field in Indonesia. The Duri field would require four groups of 4x200 MW(th) HTR modules for steam flood recovery. The advantages/disadvantages and economics of this application were reported. The paper represented a good opportunity for application of one of the advanced reactor technologies.

A wide variety of questions displayed the large amount of interest in this topic, reason to discuss it in more detail during the workshop.

Mr. Carvajal's presentation described Venezuela's energy situation, the possibilities for nuclear energy in the year 2000 and beyond, especially in reference to a specific application - the Orinoco oil field development. Extra heavy oil needs to be extracted and processed with high temperatures, for which the HTR seems to be appropriate. Work to-date has been very preliminary, so no economic assessment has yet been made due to large uncertainties. Advanced, high-temperature reactors may be a satisfactory solution in the development of the Orinoco Belt, besides other process heat applications with some perspective in Venezuelan industries.

Although the oil characteristics under consideration is quite different than that described in the previous paper, the potentiality for the application of advanced HTRs is similar.

Regarding the paper by Mr. Ivanov, V.A., the Soviet Union is a large user of heat and its success with nuclear generated electricity has encouraged the use of nuclear heat reactors. The paper discusses the need to emphasize safety and the manner by which it could be developed. An assessment of the applications of heat energy by temperature has been made and potential sources of these heat requirements are given. Developments in the AST-500 and the HTGR (VG-400) are discussed.

Questions by the participants focused on the development of the AST-500, resulting in an additional paper describing the AST-500 given by Mr. Ivanov in Session 3.

In his paper, Mr. Sergeev reviewed the special conditions that must be assessed when considering the installation of new power sources in Northern USSR. The conditions resulted in investigations on small nuclear power plants (SNPP) of 100 to 150 MWt, designed for electric power and heat generation.

The factors considered to be crucial to their success, consisted of two general areas: reduction in both time and cost of construction and increase of the capacity factor.

Questions followed on specific design issues and on similarities with the reactor at Bilibino. The public acceptance of these reactors was also reviewed.

The paper of Mr. Abdel-Gawad Emara reviewed the energy situation in Egypt for a number of energy demand scenarios. Key factors that reactor vendors must address in introducing advanced reactor technologies were summarized in a clear and concise way.

### 3. Session 3 - Experience of Industrialized Countries (Chairman: Carvajal-Osorio)

After presenting a summary on the long experience Belgium has in the nuclear field, Mr. Dekeyser made a survey of the strong lines of the national nuclear industry and of the research potential of Belgium with respect to

Advanced Nuclear Power Technology (ANPT) together with other non-energy applications. He also discussed a number of attractive areas for international co-operation on research and development in nuclear energy.

Mr. P. Kunsch, presented his company's experience related to ANPT, including small and research reactor studies, fuel testing and analysis of energy systems. This with respect to the choice of technology, energy mix and alternative nuclear fuel cycles. He explained the computer program MCDA (Multiple Criteria Approach in Decision Aid), which was developed by his company in order to assist in the complex decision making process.

The interesting case of France with its vast nuclear programme, was surveyed by Mr. Livolant. After presenting the origin of the energy sources for France, he explained how his country dealt with its energy needs. As much as 27% of the total energy is supplied by nuclear power, or nearly 80% in the total electricity production. Mr. Livolant also indicated the increasing availability factor of French nuclear plants. He also discussed the NP-300 nuclear power advanced plant for electricity production (300 MWe), which included extra safety features and which was specially designed for supplying the needs in countries with electric networks in the range from 2000-10000 MWe. He also briefed on the FBR technology and its success.

Mr. Moll indicated that his consulting firm has been involved in several studies on possible applications of conventional and ANPT in Europe and many other Developing Countries. He referred to a study concerning an inherently safe reactor performed for the EEC, a feasibility study done for Bangladesh, an economic evaluation of nuclear versus coal for Portugal, and several other studies (e.g. for Indonesia, Yugoslavia, Egypt and China). Further, he commented on the Build-Operate-Transfer scheme (BOT), which is now being considered as an attractive feature for Developing Countries. It implies that the nuclear plant will be constructed and operated by the manufacturer for several years, before it is transferred to the utility company.

Mr. McDougall referred to recent studies, performed by the AECL jointly with the Beijing Institute of Nuclear Energy, concerning the selection of a proper nuclear plant for supplying district heat in China, using the SLOWPOKE reactor for that purpose.

Mr. Ivanov provided information about the Soviet reactor design AST-500, which has been designed with inherent safety characteristics. The main feature of the design is that cooling of the core is provided by natural convective circulation under normal and emergency conditions. In addition, strong negative temperature, void and power reactivity coefficients will assure the self-regulating of the chain reaction and guarantee safe plant operation.

Based on the presentations, it was concluded that the developments in advanced nuclear reactor technologies were going in a direction suitable for proper application in Developing Countries.

During a general discussion, the following topics were selected for further consideration at the Workshop sessions:

- Incentives for introduction of advanced nuclear power technologies in Developing Countries;
- Advantages and disadvantages of advanced nuclear power technologies, risks associated with standardization and increased number of plants in case of smaller unit size;

- 3. Specific applications in and requirements of Developing Countries,
- 4. Criteria for selecting advanced nuclear power technologies.

#### II. WORKSHOP

1. WS Session 1 - Incentives for Introduction of Advanced Nuclear Power Technology in Developing Countries (Chairman: Hassan)

Participants discussed the main role of buyers, suppliers and the Agency during this Session. Buyers should develop a balanced energy policy and the competent authority should undertake a case study to assess and evaluate the overall energy and electricity demand and supply. The case study should include domestic energy resources, the need for nuclear power for specific applications, the urgency to change the energy supply system, economic and financial analysis, manpower requirements and organizational structures for planning and plant construction.

Public acceptance and the role of the utility and other organizations informing the public should also be addressed as well as the risks and benefits of nuclear power.

Suppliers should demonstrate the development of new advanced nuclear power technology concepts regarding (inherent) safety, reliability, minimum interference with the environment, guarantees and lowest overall cost.

The competent authorities in a supplier's country should show willingness to sign a bilateral agreement for co-operation in the use of peaceful application of nuclear energy, readiness to promote the sale of a power plant and be prepared to participate in financing the plant.

#### 2. WS Session 2 - Advantages and Disadvantages of Advanced Nuclear Power Technologies (Chairman: Kunsch)

The discussion concentrated on aspects showing the benefit as opposed to the cost of advanced nuclear power technologies from the point of view of both supplier and recipient country.

The aspects for Developing Countries can be summarized as follows:

#### Benefits

- Inherent Safety
- Environment
- Access to advanced technologies and social spin-off
- Resources and Diversification
- Real Cost.

#### Costs

- Public opinion
- External dependency
- No experience in licensing
- Not proven technologies
- Lack of standardization
- Capital drain due to small size
- Risk of splitting efforts in diversification of technologies
- Supply inflexibility specific to nuclear production.

All participants agreed that the safety issue was the most important one. One should however not misunderstand the meaning of the wording "inherent safety", which could suggest that current commercial reactors are less "safe". Rather it should be understood that inherent safety would rely exclusively on passive systems, based on the physical laws of nature. This expression should not be allowed to become a mere slogan for the manufacturer. Some participants thought that advanced systems could be associated with simple systems using passive safety mechanisms. These would be in general small size systems - even if this is not a condition, as e.g. in the case of HTRs, or FBRs.

According to Venezuela, simplicity in design in general, and in passive systems in particular, would be a very important factor for the introduction of nuclear energy in developing countries. It would reduce the problem of training, which would represent otherwise an important investment in time and personnel (Egypt). The lack of experience in licensing matters and of financial means are the major drawbacks in developing countries (Saudi-Arabia).

The USSR stressed the need for public education, as quite often the risks of nuclear power plant operations were perceived differently from those of other types of industries, e.g. chemical plants. It seems to be advisable to carry out a generic risk assessment for industrial activities and to transfer the results in an understandable form to the public.

Venezuela suggested that for an easier introduction of advanced technologies into developing countries, industralized countries should first build them domestically. However, there is always the possibility of establishing joint developments in a partner developing country even in the case that the advanced technology has not been realized before.

China indicated its commitment to the FBR technology and to the development of a national fuel cycle. It is eagerly awaiting technical assistance from other country programmes.

Belgium insisted on the long-term commitments due to the choice of a particular technology. Switching over to a different system would be a very difficult task, especially for small countries with limited resources (financial, technological and manpower).

The cost/benefit aspects for supplier countries are as follows,

#### Benefits

Costs

-	Market opening to Developing Countries	-	Absence of internal market
-	Employment of qualified persons	-	Insufficient knowledge on external Developing Countries markets

- Financial Risk.

Belgium stressed that the supplier industry which is benefit-driven, cannot commit itself to costly and lengthy development technologies for the external market without sufficient direct needs in the domestic market. Moreover, identification of the outside market would be necessary, before approaching Developing Countries, according to France. Egypt supposed that developments could be made as qualified staff would be available anyway in the Industrialized Countries.

While France has presently no sizeable domestic market for low temperature heat applications, the USSR has an important district heat application potential, and is ready for commercial contacts with other countries. The USSR stressed the environmental benefits of these projects. Participants were then exchanging opinions on how to give a boost to the development of advanced technologies and suggested:

- to make a market survey on application (low temperature, process steam, electricity, etc.) - in the interested Developing Countries (France).
- that ways of collaboration should be considered between Developing Countries and academic or research institutes in the suppliers' countries through bilateral agreements.
- that Developing Countries could be host for developing specific technologies.
- that technical visits from Developing Countries to specific nuclear facilities in Industrial Countries should be promoted.
- 3. WS Session 3 Specific Applications in and Requirements of Developing Countries (Chairman: Djokolelono)

As specific applications, desalination and enhanced oil recovery received most attention.

#### a. Desalination

In a number of countries, there is need for fresh water in a large quantity, which could be fulfilled by installing nuclear power plants (conventional or advanced), which could operate as co-generation plants (i.e. produce electricity and desalinate sea water).

The potential users should be confident that the plant is safe and that the produced water is not in contact with radioactivity.

Developing countries, as potential users, should co-operate in performing the feasibility studies of this nuclear option to fulfill their future needs. Co-ordinated research and development programmes are also required.

Information on water and electricity production and related economics from operating plants (e.g. in the USSR) should be disseminated.

#### b. Enhanced Oil Recovery

The need for nuclear power comes from the conservation measures of oil resources, since the crude oil burning way to produce steam consumes a substantial part of the lifted crude oil. Venezuela, Indonesia and possibly other countries have important oil reserves in the form of heavy crudes. In order to make these crudes useful to the rest of the world, a substantial amount of energy will be required, providing an opportunity to nuclear energy.

The choice of a suitable reactor type depends largely on the requirement of the driving steam and hot water. However, the safety characteristics of a HTR favours its application.

#### 4. WS Session 4 - Criteria for selecting Advanced Nuclear Power Technologies (Chairman: Crijns)

Criteria for selecting Advanced Nuclear Power Technologies would not be totally different from the general criteria and guidelines already developed by the Agency for Developing Countries starting a nuclear power programme in general or introducing a SMPR in particular.

The criteria according to the reports TECDOC-347 and 445, would be:

- 1. National development and energy policy
- 2. Government commitment to nuclear power
- 3. National infrastructure: legislation, organizational infrastructure, and manpower
- 4. Siting
- 5. Electrical grid
- 6. Industrial participation in nuclear power programme
- 7. Contractual approach
- 8. Economics and financing
- 9. Safety aspects
- 10. Fuel cycle
- 11. Waste management
- 12. Public acceptance
- 13. Environmental effects
- 14. Technical assistance.

It was felt that the following criteria should be added to the list:

- simplicity
- competitiveness
- international co-operation
- standardization
- proveness
- short lead times
- limited capital investment
- appropriateness of application
- decommissioning
- safeguarding.

After discussion about the contents and the meaning of the various criteria, the participants felt that safety aspects, appropriateness of application, economics and financing, national energy policy and simplicity should receive highest attention, when starting a programme.

## CHINA'S ENERGY SITUATION

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## Abstract

The paper describes China's energy situation, reasons for the development of nuclear energy, recent progress of nuclear power projects.

China has rich resources, its coal and water resources occupy the top places in the world, but not well distributed. Uranium resources also can be used for the development of nuclear energy.

Coal takes around 70% of the primary energy consumption, it will certainly make the rail congesting and the environmental pollution even worse. Systematic studies have been made in order to formulate a strategy that could solve China's long term energy demand. It has been unanimously agreed that nuclear power is the most realistic solution.

Construction of 300MW e PWR at Qinshan Nuclear Power Plant and twin 900MWe PWR at Daya Bay Nuclear Power Plant is going ahead. The feasibility of Qinshan second stage of two 600MWe PWR units has been studied for some time and is still investigated in depth.

China's target for nuclear power capacity in Qinshan and Daya Bay is 4500MWe by the year 2000. Further nuclear power plants will be constructed according to investment possibility. Nuclear power will play an important role in the next century, tentative object by 2015 is 30000 MWe.

#### CHINA'S ENERGY SITUATION

Energy industry is an important sector of the development of national economy. We have taken energy problem as strategic one to be studied and developed. The situation of the development of China's energy industry is as follows:

1. China has rich resources, its coal and water resources occupy the top places in the world, but not well distributed, 60% of coal reserves in North China and 67% of water resources in the southwest part of China. We also have uranium resources which can be used for the development of nuclear energy.

2. In 1987, 925 million tons of raw coal, 134 million tons of crude oil, 13.7 billion cubic meters of natural gas and 493 billion KWH of electricity were produced. But the speed of energy development still can not meet the needs of development of national economy and people's life.

3. In order to speed up the development of electric industry, different effective measures should be taken. From the point view of construction and distribution of the electric power plants, we should build a series of thermal electric plants nearby the coal mines, we also will construct a number of large thermal power stations on the main ports along the coasts. At the same time, a number of hydropower stations will be constructed in middle and up reaches of Yangtze and Yellow River and in the area of Hongshui River.

## WHY SHOULD CHINA DEVELOP NUCLEAR ENERGY

A recent analysis shows that in the distribution of primary energy consumptions today in China, coal takes around 70%. This figure is already too high. If, however, the coal consumption further increases, it will certainly make the rail congestion and the environmental pollution even worse. From now on, with the production centers of coal and development of hydropower resources will move further to the western part of China, the unbalanced situation will be getting even worse. Furthermore, the fossil reserves are limited, and, with the development of the national economy, more and more of them need to be converted

into chemical products. Systematic studies have been made in order to formulate a strategy that could solve China's long term energy demand. It has been unanimously agreed that nuclear power is the realistic solution.

## PROGRESS OF CURRENT NUCLEAR POWER PROJECTS

We have just began to construct the nuclear power plants, and we will build the nuclear power plants in area of east China and Guangdong Province where it is more developed in economy and lack of energy.

1. Qinshan Nuclear Power Plant

Located about 130 km southwest of Shanghai. The construction began in June 1983 under the direction of the Ministry of Nuclear Industry.

Qinshan 300MWe PWR is now in full swing. The top of the safety shell for the nuclear reactor has been sealed and construction of the generator's housing has been completed. The installation of the reactor will begin in autumn of 1988. The construction of the auxiliary buildings and the conventional buildings is going ahead. The plant is scheduled to go into operation by 1990.

2. Daya Bay Nuclear Power Plant

Daya Bay twin unit each 900MWe PWR project sited near Guangzhou and Hong Kong. The owner is Guangdong Nuclear Power Joint Venture Company, which is a joint venture of China Light & Power Company of Hong Kong and Guangdong Nuclear Power Company. The primary civil construction is being performed by several organizations. The two 900MWe PWR units are scheduled to be put into operation in 1992 and 1993 respectively.

## PROSPECTS OF NUCLEAR POWER IN CHINA

It was decided to fisrt develop in China medium sized PWR nuclear power units. Qinshan is an excellent site to construct four units of 600MWe. It is envisaged

to undertake two further stages of construction at Qinshan site with two units in each stage. The 600MWe units will be constructed by making full use of the acquired experience from the 300MWe PWR and incorporating proven international technology. The feasibility of Qinshan second stage project has been studied for some time and is still investigated in depth.

China's target for nuclear power capacity in Qinshan and Daya Bay is 4,500MWe by the year 2000. Further nuclear power plants will be constructed according to investment possibility. In this century, the main goal for us is to posses techniques in the fields of equipment manufacturing, construction, safety and operation for laying foundation for rapid development of nuclear industry in the next century.

According to the prediction, nuclear power will be 30,000MWe by the year of 2015. By the middle of the next century, nuclear power is expected to be more than 20% of total electricity generation, and will play an important role in China's electric power systems. We believe that with the development of the national economy, nuclear power will become one of the main energy resources in most areas of China in the foreseeable future.

## NUCLEAR POWER PROSPECTS IN THE ARAB WORLD

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#### Abstract

A world outlook of the share in energy supply and demand for developing and developed countries is given. With respect to nuclear energy and according to recent estimates by the World Energy Conference, the share of nuclear power in the world energy mix will be limited varying from 4% by the year 1985 to 14% by the year 2060. The Arab energy situation was also presented. It is shown that Arab countries should have a policy to ensure both the diversification of their energy sources and the acquisition of knowhow to produce the energy they need. Nuclear energy can be considered as a relatively accessible energy alternative. Social political and economic factors should be properly considered before any decision about the preference of any energy source. Studies related to estimates about future nuclear energy prospects in the Arab world appear to be overoptimistic. A regional development of nuclear energy is recommended especially in the field of uranium exploration and extraction. Any nuclear programme should be viewed as a developmental programme of a regional multinational developmental project. "A Pan Arab Nuclear Fuel Cycle" should be reconsidered. An Arab energy institute for promotion of research activities and implementation of energy planning programmes is proposed.

## I. INTRODUCTION:-

The 1973 energy crisis - the sudden decrease in oil supply and vast increase in oil prices has created, at that time, a type of overwhelming near-panic fears and much concern about energy supply and demand especially among industrialized countries. The crisis has created a powerful stimulus among these countries to review their energy situation, present attitudes and future prospects.

On worldwide basis, a number of prospective studies concerning energy supply and demand for the next 25 years have been issued by scientific groups and concerned governmental sections and institutions (1). An International Energy Agency (IEA) has been established with the objective of securing fossil fuel flow to industrialized countries.

In general and, despite of the diversing conclusions of some of the forecast studies, there is a more or less general agreement that the era of easy, cheap, abundant oil is reaching an end. In this context, the viability of alternative energy sources has been considered and their potential to fill the energy supply-demand gap has been assessed.

In the opinion of many researchers nuclear energy can be considered as one of the most technologically developed, readily accessible energy alternative to produce power at a competitive price and with extremely limited environmental impact. Although the TMI accident in the seventies and the Chernobel accident in April 1986, were used by opponents as examples of "a credible accident", to strengthen their provocation and opposition to nuclear energy, yet the world orientation and intention to use nuclear power is continuing.

For many countries particularly industrialized developped nations, nuclear energy can be considered as a reliable energy source, not only for the next few decades,

but also for a long time to come, especially if the technology of fast breeder reactors is establised and ultimately proven, a situation which renders nuclear energy as an inexhaustible energy source.

## II. A WORLD OUTLOOK :-

At present there are 397 reactors operating all over the world and 133 are being under construction (2). According to the WAES\* estimations (1), it is shown (Fig.1) that industrialized countries constituting 20% of the world population by the year 2000, are expected to consume 75% of the total world\*\* energy consumption. Developing countries constituting 80% of the world population are expected to consume only 25%. Of the total world energy production in the year 2000 amounting to 170 MBDOE\*\*\* developing countries are to supply more than 70%; a situation that indicates the role of developing countries in satisfying world energy needs at the expense of their natural resource endowment; thereby depleting these natural resources.

A more scrutinized analysis regarding the per capita commercial energy consumption in both developed and developing countries shows a per capita consumption of 6465 kg C.E. for the first group as compared to 565 kg C.E. per capita consumption for the second group (3).

<sup>\*</sup> WAES : Workshop on Alternative Energy Strategies.

**<sup>\*\*</sup> Total World :** World outside communist area (WOCA).

<sup>\*\*\*</sup> MBDOE : Millions of barrels per day of oil equivalent.



FIG.1. Shares of WOCA energy consumption (source: Ref. [1]).

The situation is, however, worsened if energy resources (oil, gas, coal) are compared in both groups of countries (developing and developed)(Fig.2). It is clear that contrary to the general belief, the developed countries' natural resources in T.C.E. exceed those for developing countries. The case of the U.S. is a clear example of disparity (3). The big oil reserves of the developing countries should not be considered as an excuse for their intensive exploitation and consequently rapid depletion of these resources.

In our opinion a national coherent policy in oil exploration and production should be adopted by oil producing countries. For the year 1985 an aggregate R/P ratio of 44 was recorded for oil producing countries. Figure 3, shows an application of two Arab crude oil production



FIG.2. World distribution of proven fossil fuel reserves (1980) (source: Ref. [3]).

policy models (4). It is shown that to keep a 20/1 R/P ratio which is considered as close to the technical limit of many countries, Arab oil production is expected to decrease with the beginning of the next century. It is worthmentioning that in 1979, the R/P ratio of each of Algeria, Bahrain, Egypt, Morocco, Dubai and Sharja was already below 20.

For oil producing developing countries in addition to rapid exhaustion of their national endowments the



FIG.3. Illustrative application of two Arab crude oil production policy models (source: Ref. [4]).

revenues they obtain are continuously eroded by inflation and price increase of industrialized commodities.

For developing countries the question is how to keep a level of economic growth at a reasonable rate, how to secure reliable energy sources for future generations, and how to create a stable cooperative world energy system.

On worldwide basis, the total world energy consumption as presented to the world energy conference amounts to 8% Gtoe increasing to about 20 Gtoe by the year 2000 (5). Table 1, shows the percentage contribution of the different types of energy sources up to the year 2060 (5). It is forseen that a trend to substitute hydrocarbon

## TABLE 1 : Percentage Contribution to World Energy Consumption According to Fuel Type. ( World 100% )

	1960	1980	1985	2000	2020	2040	2060
Hydrocarbons	45	58	54	48	38	33	27
Alternatives	55	42	46	52	62	67	73
- Coal	36	25	27	27	30	32	34
- Nuclear		2	4	7	12	13	14
- Renuables	19	15	15	18	20	22	25
Energy surpluses		6	13	17	24	33	37

(Source Reference 5 )

fuel by other alternatives (coal, nuclear,...etc) exists. The contribution of nuclear energy to the total world energy consumption varies between 4% by the year 1985 up to 14% by the year 2060. This shows that nuclear power consumption will increase by a factor of 2.5 which is much higher than that cited for coal increase (0.25).

Concerning the uranium fuel supply, the uranium supply and demand study by the uranium institute (6) shows that it is expected that if the present trend with thermal nuclear reactors is to continue, fuel requirements will approach the limits of existing production by the late 1990s (Fig.4).



FIG.4. Western world uranium production capability production and reactor requirements:1983–1995 (source: Ref. [6]).

The introduction of breader reactor types and the utilization of thorium for U-233 production will, however, drastically change the situation, thereby shifting the time margin of uranium resources depletion by a number of several decades.

As is already indicated (Table 1) hydrocarbons should be substituted by other energy alternatives. In this respect coal is expected to play an important role in meeting world energy demands. Nuclear energy can be considered as a promising energy source particularly for industrialized developed nations.

It should be noted, however, that we should not be trapped in the methodology of a short term global analysis which may lead to a false feeling of abundance of fuel resources. The problem is, however, more complicated and by a long term analysis a more real picture can be depicted.

It is expected that the worse situation will be with the developing nations having their conventional energy resources depleted and being unable to acquire and assimilate the technologies of new prospective energy sources.

## III. THE ARAB WORLD :-

As stated in the ECWA report on "Arab Energy Prospects to 2000"(4), that the Arab countries' prime concern in the face of rapidly growing demand for its depletable oil resources, "is that these resources should last as long as it would need them. In more explicit terms, oil should continue to provide the fuel, raw material, and foreign exchange required for a rapid balanced development and diversification of the Arab economies, and it must continue to provide until it can be replaced by alternatives that would be available (with appropriate technologies) at competitive prices and in sufficient quantities".

According to Shehab El-Din (7) and based on different scenario assumptions, the total world installed electrical capacity for the Middle East and North Africa amounts to 170 GWe and 730 GWe for the year 2000, 2030 respectively (Table 2). Of this amount, nuclear energy is estimated to represent 29% in the year 2000 and 50% in the year 2030 which means a power generation of 50 GWe and 365 GWe respectively. On the other hand, and according to different IIASA scenario assumptions, the most likely nuclear generating capacity is indicated in this study to be 15 GWe (7).

In our opinion, these estimates (1982) are overoptimistic. For the Arab World, only three countries have already research reactors of limited power (Egypt, Lybia and Iraq). No official commitments, involving the commissioning of nuclear power reactors of any size is known for any Arab country except for Egypt. Research activities are, however, scattered everywhere.

According to Shehab El-Din (7) and on global basis, although region (VI) indicated in the study (Middle East and North Africa) Possesses 292 X  $10^9$  KWY of ultimately recoverable resources, it is assumed that only 20% of these resources will be domestically used. The remainder is to be exported for the purpose of sustaining continuous economic growth at a reasonable rate and for the purpose of satisfying world needs of fossil fuel. On these basis, the life span of the energy sources is estimated to range from 47 year to 76 year according to the variation in basic scenario assumptions. If on the other hand, a 100% local consumption is presumed, a life span of 378 and 235 years is foreseen. These estimates clarify the situation of oil producing countries as energy suppliers and consumers.

In our opinion, the Arab Countries in addition to the adoption of an oil conservation policy, should endeavour an energy source with its technological know-how to ensure their self reliance and independence, to satisfy their future energy needs associated with high rates of population increase, and to realize their aspirations of economic growth.

In contradistinction to the aforementioned nuclear energy projections (Table 2), the present status of Arab countries is far behind. If nuclear energy is to have a share in future Arab energy mix. Intensive programmes, stern actions, development of already available manpower and technological capabilities should be put into action with a sense of urgency. It should be understood that, the main objective is to have a programme which can be secured by fuel from domestic origin with the participation of national technological capabilities, to ensure the programme's continuity, its development and long lasting viability without any disruption at any stage.

Some estimates in the Arab World presume that the demand for energy in the Arab World may reach 100 GWH a year in the year 2000 (8). The same author could predict that there will be a chance of operating 20 nuclear power stations in the Arab World by the year 1985. As it is presently clear, these estimates are far beyond the situation in the Arab World.

To realize the above mentioned objectives, any Arab country or a group of countries embarking on a nuclear programme, should put plans and develop capabilities to ensure the following:

- The capacity of financing the programme at its different stages.
- The possibility of having a national fuel supply potential.
- 3) The availability of national nuclear fuel cycle services.

World Region	1980		Year 2000					Year 2030			
	Nuclear Installed Capacity (Give)	Primary Energy Consumption (10 <sup>9</sup> kWY/Y)	Primary Electrical Energy (10 <sup>9</sup> kWY/Y)	Installed Electrical Caoacity (GWe)	Installed Nuclear Cepacity (GWe)	Alost Likely Nuclear Capacity (GWe)	Primary Energy Consumption (10 <sup>9</sup> kWY/Y)	Primary Electrical Energy (10 <sup>5</sup> kWY/)	Installed Electrical Capacity Y) (GWe)	Installed Nuclear Capacity (GWe)	
I	159.5	3.89	1.28	870	290	280	6.02	2.71	1.840	220	
ŧı	37.9	3.69	1.22	830	230	300	7.33	3.30	2.240	930	
ett –	125.7	4.29	1.42	966	345	510	7,14	3.2:	2 190	1.020	
IV	4.8	1.34	44	200	90	4C	3.68	1.66	1,130	565	
V	12.8	1,43	.47	320	100	60	4.65	2.09	1.420	710	
M	0	0.77	25	170	50	15	2 38	1.07	730	365	
MП	0	1 44	48	326	100	10	4 46	2.01	1,370	655	
WORLD	340.7	16 B	5 54	3,782	1,215	1,215	35 7	16.07	10,910	5 4 5 5	

Table 2 : World Energy and Nuclear Power Projections

(Source Reference 7)

- 4) The availability of a technological capability and industrial infrastructure to acquire and assimilate the technology and effectively participate in the commissioning of nuclear plants.
- 5) The availability of manpower and the necessary competent authorities to operate, maintain and ensure the safety of the project.

As we understand it, the Arab countries are facing a challenge of introducing a new sophisticated capital intensive technology in a weak, technologically underdeveloped social structure. In addition, the problem of financing may represent, for individual countries, a type of heavy burden unless plans are adopted for regional nuclear development programmes. In this respect, the interconnection of electrical grids is important for the acquisition of nuclear plants with high generating capacities.

With respect to insuring a domestic fuel supply, the Arab world can be viewed as being endowed with natural deposits of uranium ore. Primary uranium mineralizations in the Arab world are estimated to be in the order of 70,000 ton (8). Secondary mineralizations are estimated to be in the order of 5 x  $10^6$  ton uramium (as  $U_{3}O_8$ ).

These natural resources represent a real potential for a near term supply of a nuclear programme competent with the Arab world needs and capabilities (8). Extensive efforts should, however, be put to develop these resources and to lay the foundation of a uranium extraction industry. The experience of Morocco is worthmentioning in this respect.

For the nuclear fuel cycle services, however, the situation is not as encouraging. El-Fouli et al., have shown that for a nuclear programme as originally proposed by the Egyptian authorities (7200 MWe), a national fuel cycle is not economically justifiable (9).

For a regional fuel cycle, it is shown that at least 3 to 4 countries with nuclear power programmes, similar to that of Egypt will represent a minimum to reach the state of economic feasibility.

## IV. THE CASE OF EGYPT:-

The case of Egypt is an example of treating nuclear energy programmes as a bundle of purely technical questions. The supreme council for energy planing has developed two scenaries for energy supply and demand up to the year 2000 (low growth and high growth rates)(10). The council has proposed that 40% of the generating capacity in Egypt will be nuclear by the year 2000.

The Ministry of Electricity has had a strategy to build 8 power stations of approximately 900 MWe each in the years from 1981-2000.

The Egyptian programme was viewed as being a technical undertaking and in the view of official energy planners can be realized as any other conventional energy production programme. Social, political and economic questions were not given appropriate consideration. Preliminary estimates by the Nuclear Power Authority showed that 44 billion Egyptian Pounds are needed for commissioning 8 reactors by the year 2000, 31 billion are needed for operation and maintenance up to the year 2017 (11).

Opposition to the programme was noticed quite early in current newspapers especially among concerned groups and organizations (12). The following issues were raised by different groups:

- The economic burden the country has to incur in the execution of the project.
- The rational behind the decision of having as high as 40% of power production from nuclear sources.
- The already accepted view of having plenty of natural gas resources in the country; a question which can dramatically change the pattern of energy mix proposed to meet energy demand.
- The fact that a turn-key project, as was originally proposed by project planners, will not realize project objectives of technology transfer, manpower development and technical capability improvement.
- The lack-in the opinion of many of scientific and technical infrastructure at a level and having the capacity to actively participate in the execution of the project and to be capable of acquiring and assimilating the new technology.
- The presence or absence in the country of uranium deposits and nuclear fuel manufacture capabilities sufficiently enough to ensure continuous domestic fuel supply for the project.
- The economic feasibility of the fuel cycle back end services in the absence of any other programmes in the Arab area.
- The social status, the educational and cultural level of individuals supposed to execute, run and maintain the power plants.

A turning point on the debate was the Chernobel accident which made all programme activities to cease.

Quite recently (13) the Shoura Council has issued a new document recognizing the necessity, on national basis, of acquiring nuclear technology for both energy production and technology acquisition.

Energy planners in Egypt are now satisfied with a plan to build a nuclear power station of 2 reactors of approximately 900 MWe each.

In addition to energy conservation plans, they are stressing the possibility of introducing natural gas as a major component in the energy mix of the country. The possibility of importing coal is also stressed.

Quite recently it was announced by the Minister of Petroleum that it was possible to increase the country reserves of fossile fuel in the last few months through intensive exploration and oil extraction (14).

In the opinion of many executives, the financing of the nuclear project is the main obstacle that hinders the realization of the project.

We argue that the same problem of financing also applies to the new plans of importing coal for energy production in the country. The balance of payment is not in favour of importing any energy producing source.

In our opinion, it is the social status, the economic priorities and the values that prevail in the society that decide in the final analysis.

In the present situation and with the chance of having a loan to execute the power plant, the nuclear option seems to be the most readily available alternative.

#### V. CONCLUDING REMARKS:-

- 1) The Arab Countries should have long term plans to secure their future energy needs and should possess the know-how to control the technology for energy production. Nuclear energy can be considered as the energy alternative presently available to suffice these needs on competitive basis.
- 2) Any nuclear programme, on national or regional basis, should be viewed as a developmental programme with the objective of raising the living standards of the people and aiming at improving the scientific and technological capabilities of the country of concern.
- 3) In our opinion, the real challenge is social and political, and for some countries may be economic. The Arab Countries are at different stages of development in the general sense and at different levels of development in the nuclear field. Any prospective analysis will seem to us as being uncertain.
- 4) Projections for the year 2000 of nuclear energy share for the Middle East and North Africa proposed by IIASA and IAEA (7) and amounting to 15 GWe representing 9% of the total installed capacity of the Arab Countries appear to be overoptimistic.
- 5) If the political will, were to exist, the prospective introduction of nuclear energy in the Arab countries depends on the establishment of cooperative plans for all stages of the nuclear fuel cycle.
- 6) Technical obstacles which hinder a regional nuclear programme should be eliminated. The question of
interconnection of the grids should be settled. A minimum grid size of 2.5 GWe is necessary for any country before nuclear development could begin.

- 7) The success of any future plans is dependent on the capacity to ensure a continuous domestic fuel supply. There are bases to assume that an Arab cooperative activity in this field is highly prospective.
- 8) A "Pan Arab Fuel Cycle" should be reconsidered since as previously indicated, the economic feasibility of the fuel cycle is questionable. It is "envisaged that at least 3 to 4 countries with nuclear power programmes similar to the originally proposed by Egypt could achieve the set requirements to meet economical national integrated fuel cycle centre at or before the year 2000 (9)".
- 9) In the final analysis it is the social status, the economic system, the values and life style which decide and which are crucial questions in the choice of any energy system.
- 10) For any regional development programmes including the development of a nuclear programme, a minimum of mutual dependance, cooperation and political understanding is needed. Experience shows that this was not always the case with previous development programmes and projects.
- 11) There is a need for an Arab Energy Organization for research, development and energy planning.

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# PROSPECTS FOR FUTURE REACTOR UTILIZATION IN CHINA

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## Abstract

Referring utilization of nuclear energy as district heating describes the situation of present residential heating, advantages of nuclear heat supply, adaptation of low-temperature nuclear heat reactors. It is planned first to build one or two low-temperature nuclear heat reactors in appropriate northern cities of China in 1990s. Take the example of City-Qiqihar in Northeast China, it is estimated that if the district heat supply system installed with 400 MWth low temperature heat reactor, the consumption of around 300,000 tons coal could be saved, it would mitigrate rail congestion and reduce environmental pollution. Utilization of nuclear energy as district heating is favorable in cold regions, that is the Northeast, Northwest and North China areas. Nuclear heating cost will be lower than conventional coal fire heating by comparison.

Referring advanced reactors describes the research work on FBR, plan for building experimental FBR, and research works on fusion technology. FBR will be the next generation reactor, it is important for economic utilization of nuclear fuel due to its inherent function of breeding fissile materials. Research work of FBR is carried out for several years, 50-100 MWth experimental FBR will be built before the year of 2000. Feasibility study of HTGR is carried out in recent years. Efforts also devoted to research and development work of the advanced light water reactors and fusion technology.

China will vigorously promote the peaceful application of nuclear energy in national economy and its people's life by full utilizing the existing nuclear industry and technology. International cooperation in nuclear field can not only bring benefit to both developed and developing countries but accelerate the peaceful application of nuclear energy in the world.

## UTILIZATION OF NUCLEAR ENERGY AS DISTRICT HEATING

Advantages of nuclear heat supply are fuel conservation, improvement of the environment, and reduction of heat discharged to the atmosphere. At present, about one quarter of the total coal supply is consumed as residential fuel. When a nuclear district heating system replaces individual heating boilers, the combustion process emissions from thousands of small stacks are eliminated.

Beijing Institute of Nuclear Engineering(BTNE) and Institute of Nuclear Energy Technology(INET) of Tsinghua University are carring out the feasibility studies on low temperature nuclear heat reactors. INET is developing a prototype heating reactor, which is designed for a high degree of inherent safety with natural circulation.

Low temperature nuclear heat reactor can be used as an ideal district heat source for cities located in the Northeast, Northwest and North China areas. Take the example of Qiqihar-City in Northeast China, it is estimated that if the district heat supply system installed with 400MWth low temperature nuclear heat reactor would be adopted, then the consumption of around 300,000 tons coal could be saved, it would mitigate rail congestion and reduce environmental pollution as well. Utilization of nuclear energy as district heating is favorable in cold regions of China. Nuclear heating cost will be lower than conventional coal fire heating by comparison.

It is planned first to build one or two low temperature nuclear heating reactors of 400MWth in appropriate northern cities of China in 1990s. Afterwards, based on the acquired experience, the nuclear heating reactors may have an extensive application in China.

# DEVELOPMENTS OF ADVANCED NUCLEAR REACTORS

Great efforts should be devoted to research and development work of the advanced light water reactors, fast

breeder reactors, high temperature gas cooled reactors and fusion technology.

FBR will be the next generation reactor, it is important for economic utilization of nuclear fuel due to the inherent function of breeding fissile materials. Research work of FBR is carried out for several years, but it is still in the stage of fundamental study. We also need to promote international cooperation in order to learn the advanced technology and experience. China has paid a significant attention to develop FBR,50-100MWth experimental FBR will be built before the year of 2000.

Feasibility study and several design studies have been carried out in INET. The primary study of the modular HTR process steam application in heavy oil fired steam generators can save a great amount of oil annually.

Research work on fusion technology started early in 1958. Since then more than 20 neutron-type experimental facilities were installed. In particular, the China Tokamak No.l was put into operation in 1985. Many research works have been done also in magnetic and electric fields of high intensity, low temperature, super conductivity, super vacuum, high energy particle beams etc., and some scientific research results have been already made. Currently, the research work on nuclear fusion is continuing on the foregoing basis.

## CONCLUSION

The purpose of developing nuclear energy in China is not only to meet the present energy demand but, more importantly, to pave way for China's development in the next century. Peaceful application of nuclear energy is the fundamental policy to guide the development of China's nuclear industry. We shall vigorously promote the peaceful application of nuclear energy in all aspects of China's national economy and its people's life by fully utilizing the existing nuclear industry and technology. We are ready to cooperate widely with foreign countries in nuclear industry. International cooperation in nuclear field can not only bring benefit to both developed and developing countries but accelerate the peaceful application of nuclear energy in the world.

The peaceful use of nuclear energy in China is our fixed policy. We believe that with the efforts of the Chinese scientific and technical staff, the nuclear power will have a rapid development and play a growing role in the development of national economics and energy resources.

## THE APPLICATION OF HTR IN INDONESIA

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#### Abstract

A prefeasibility study of the application of HTR in Indonesia, namely for the heavy oil recovery in the Duri oil field, was completed and reported to the Indonesian Government. The study was performed under the cooperation between KWU, Interatom of the Federal Republic of Germany and BATAN, BPPT, MIGAS, LEMIGAS, Pertamina of Indonesia.

From the Duri steam flood requirement 4 units of 4 HTR 200-MWt-Modules were exercised and positive results were obtained. Various advantages, economic and technological, on the use of nuclear steam supply systems over conventional crude-burning ones were identified. Later an economic study of only one unit introduction, resulting in similar positive results, was also presented by KWU.

Whilst uncertainties associated with assumptions still have to be verified through a comprehensive feasibility study, some issues are hindering the decision to proceed. Among others are short term practice of oil production sharing contracts, actual low oil price, optimistic view on the availability of other energy resources, institutional and safety aspects.

## The Duri Oil Field

The Duri oil field located in the Riau province, Sumatera, is operated by the Caltex Pacific Indonesia (CPI). The size of this field is about 18 km long and 8 km wide and covers an area of 10,500 ha. The reservoirs are lain in various depths, from 67m to 244 m, called the Rindu, Bekasap Pertama, Bekasap Kedua, Baji, Jaga and Dalam.

The available crude in place is estimated in the amount of 9 7.1 X 10 barrels, of which using primary recovery method only 7,5% can be lifted. Since the crude has high average gravity and viscosity (22 API, 120 cp), its exploitation requires an advanced method (enhanced oil recovery, EOR). Since September 1975 the CPI has been using the steam flood method, i.e. by injection of high temperature and pressure steam into the injection well and by pumping the crude out 4 to 8 surrounding wells. The injected steam has 26 atm pressure and 232 ° C temperature. The required steam is prepared by purifying water from the Rokan river nearby then evaporate it using energy from burning the crude extracted from the local wells. By this method about one fifth of the extracted crude is consumed. The steam generators used are of mobile type which can be simply relocated from time to time as required.

In Indonesia it is estimated that in the coming years the amount of crude produced by the use of steam flood method will constitute larger and larger share. Therefore even if the EOR at present poses higher cost, the role is increasing in the future.

## The Prefeasibility Study

The fact that producing crude by consuming one fifth out of the product has yet crossed the conservation concept of the energy resources, it also becomes an environmental issue later as well as an economic one whenever the oil price escalates consistently. Meanwhile studies for fuel alternatives, namely coal from the same island or gas from the Natuna islands, showed higher in cost as well as in capital.

The prefeasibility study executed jointly by KWU, Interatom, BATAN, BPPT, MIGAS, Lemigas, and Pertamina was aimed to judge whether a nuclear alternative using HTR-Module is a viably economic solution and whether a further comprehensive feasibility ought to be performed.

It was assumed that several units of HTR-Module are installed for cogeneration of required injection steam and

electricity consumed for production processes as well as for the oil complex needs.

In general the cogenerating plants shall serve for the whole life-time of the oil field as the steam generator and as electricity source utilizing a back-pressure turbine. But as soon as the oil field stops producing crude, the back-pressure turbine can be converted into a full condensing one, yielding electricity for the rest of the reactor life. This idea was thought to be very sound but it means that the plant shall be optimized for this future purpose too.

Furthermore the nuclear alternative needs a substantial lead time of 6-8 years while the conventional (crude-burning) steam generation is proceeding steadily in the Duri field. This means that the later the HTR is introduced the less unit number is required.

# Conclusion of 4X4 HTR-Module study

The technical concept, based on the proven KWU LWR-technology and the AVR experiences, is well established. Its safety concept is based on inherent physical properties, which enable the reactor to shut down the nuclear reaction and to remove the decay heat without relying on active engineered safety systems.

Compared to conventional steam supply by using only oil fired steam generators, the HTR-Module application for the Duri project, will increase the total government take from the year 2001 to 2010 by 10.2 billions US\$, at 2 % p.a. real oil price increase. This corresponds to an increase of about 63 %, from 16.2 billions US\$ (for the conventional alternative) to 26.4 billions US\$ (for alternative 4X4 HTR-Module power plants). This additional "government take" will have considerable advantages for the

national economic growth. For Caltex, it will increase their revenues, too.

Another important aspects from the national economic point of view is that approx. 17 millions bbl/y of crude can be substituted by nuclear fuel. Taking 26 US\$/bbl real oil price (1987 US\$) or correspondingly 40 \$/bbl current in the year 2000, the resulting additional foreign currency amounts to about 700 millions a current US\$/y.

The alternative fuels to crude and nuclear energy could be coal or natural gas from Natuna gas field. The required equivalent amount of coal would be about 5 million tons per year. Considering appropriate environmental conditions, the HTR-Module altenative offers by far more advantages compared to coal. Natuna gas can be considered as a competitive fuel to Duri crude. But up to now, no adequate information for comparison investigation is available.

The date of possible introduction of a first HTR-Module power plant depends on the already running oil field development. Further conventional steam generators may be put on order and go into operation. This means that at a certain date the investment in conventional facilities might be so large that a decision for an HTR-Module alternative could be too late.

The minimum time for start of HTR-Module operation would be about 7 years :

- 1 year for feasibility study including determination of site data and clarifiation of financing,
- 2 years of site dependent preplanning
- 4 years of plant construction.

That means, if the first HTR-Module plant should go into operation in 1995, it would be necessary to take the decision early enough, and the year 1988 is recommended.

The maximum level of estimated production capacity of the Duri oil field, which will be reached in the coming 10-12 years, requires approx. 4000-5000 t/h steam and 140 MW electricity. Generating steam and electricity the 4x4 HTR-Module plants can save 90-100 million tons of crude oil in 40 years. Further additional crude could be substituted if in the adjacent oil industry its electricity requirement is supplied from these HTR-Module power plants. The substituted oil will be available for increasing the export capacity or for domestic demand. Producing 300 MW electricity in total, the 4x4 HTR-Module will give a surplus of 160 MW which can be fed into the nearby grid, where the demand is increasing too.

With the decision to introduce this HTR-Module technology, Indonesia has the opportunity to take the advantages of the corresponding technology transfer. In the project implementation and plant construction. it is expected that up to 40 % of the overall investment can be supplied domestically.

Additional aspect is that the Indonesian industry jointly with KWU/Interatom may get the opportunity, to have an access to the Asian market for the HTR technology. The HTR-Module offers also additional applications. such as steam and electricity cogeneration in refineries, petrochemical complexes and other industries.

Based on the statement mentioned before and the jointly elaborated results, considerations can now be taken whether or not this technology should be introduced in Indonesia. If a positive decision for a project by the respective authorities of

Indonesia is considered, thereafter a feasibility study should be initiated and the respective financing should be secured. The feasibility study should include items such as:

- determination of site data required for the HTR-Module plant;
- optimization of field development plan adjusted to HTR-Module application;
- plant concept including involvement of Indonesian suppliers;
- organizational set up of the owners/operators group;
- financial planning.

## Introduction of 1 x 4 HTR-M

The input data for the investment covers one unit of four HTR reactor modules, one unit of water treatment plant, owners cost/common facilities, steam distribution systems and various operating & maintenance costs. The difference to the former analysis is that a certain sale price of steam and of electricity has been assumed. In other words, having assumed steam and electricity prices, economic merits of 1x4 HTR-M are calculated.

The investigations are performed on the basis of current price. A 4% p.a. inflation rate and 2 DM/US \$ exchange rate are assumed. Sensitivities in oil price and currency exchange rate variations have also been performed.

For economic evaluations the most relevant criteria calculated are net cash flow project operation, internal rate of return (IRR), net present value (NPV) of the net cash flow, payout time.

The feasibility and profitability of this project, like other projects in the oil industry, depend essentially on the oil price development. It was assummed also that oil price used

is 26 US\$/bbl (real) in the year 2000 and 31 US\$/bbl in the year 2015 (upper boundary). These values corresponds with the chosen escalation in the study : 1, 2 and 3% p.a. oil price increase in real terms

The analyses show that the payout time is 12-13 years. The resulting net cash flow, including interest during construction, indicates that in the first years (including 4 construction years) equity is needed in order to assure the financiability of the project. In the period after 2000, the net cash flow curve shows very positive figure. The influence of DM/US\$ exchange rate variations is insignificant.

The internal rate of return on investment lies between 14 to 20% at 1 and 3% real oil price increases. An IRR on investment of approximately 10 % p.a. for the power plant utility business respectively the electricity supply sector, has been considered as quite reasonable and is generally accepted. Considering an IRR of 15 % p.a. as acceptable, the presented result shows promising prospects.

## Hindering Issues

The nuclear alternative, according to the study, is justified in the long term (more than 20 years) assumming an escalation of oil price. While current practice in the product sharing contract refers to period of about 15 years, the prospect in more than 15 years is beyond the company's interest. Especially when the actual low price of oil at present does not justify the usually accepted forecast of price escalation in the near term.

High capital cost of HTR modules compared to crude burning mobile steam generators is the next argument. The question is who shall invest and bear the risk additionally to the existing running project.

Another variant is that the oil company will purchase required steam and electricity to a nuclear company on site. This nuclear company will construct an NPP and apply for licenses, and will take care that all the steam and electricity produced are well absorbed by the oil production process and by the surrounding community continuously.

Furthermore the current arrangement in the production sharing contract includes incentives in which the oil company enjoys, that make nuclear alternative poses disadvantages due to the large capital investment. Among these disadvantages are the following :

- Investment credit, i.e. a substantial part of investment, is paid from the revenue to the company annually as an incentive of using new technology.
- Depreciation is accounted with the double declining balance, combined with the straight-line method
- Internal rate of return shall be in the range 20-25%.

The figures of proven oil reserves range between 6.6 X 10 9 barrels to 9.5 X 10 barrels, in both figures the Duri heavy oil is included. The rates at which the proven reserves are increased by new discoveries depend on the exploration expenditure. Exploration has to be stimulated to maintain an adequate production level. But since 1982 the actual exploration and development expenditure has declined steadily. Exploration activities in more remote and offshore areas have posed increasing cost in the past, therefore to be optimistic it is likely that only with increasing total exploration expenditure a sufficient amount of new discoveries can be obtained in the long run.

According to the MARKAL study, it was predicted that today proven reserves would be depleted about the year 2007 in case of

the high figure even with the reduced domestic oil consumption strategy. On the other hand the natural gas alternative for the Duri is sought to use the Natuna gas, being even expensive (about 5 \$/MSCF) and capital intensive (1000 km long), although from the reserve point the amount is very large (80.58 TCF). At present the gas alternative investigation is being proposed before stepping into further HTR study.

If ultimately this Duri HTR project is executed, it is the first in the world that an oil company in a developing country becomes the host for nuclear reactors. It implies that large efforts to convince various institutions shall be carried out especially in the safety aspects of the HTR.

## Concluding Remarks

The preliminary feasibility study on application of HTR for the Duri oil field was successfully performed under the cooperation between KWU, Interatom of the Federal Republic of Germany and BATAN, MIGAS, LEMIGAS, Pertamina of Indonesia. Various advantages, economic and technological, on the use of nuclear steam supply systems (operating in the cogeneration mode) over conventional crude-burning ones were identified.

Uncertainties associated with assumptions still have to be verified through a comprehensive feasibility study, as thereby some issues are yet to be deliberated. Among these are short term practice of oil production sharing contract, actual low oil price, optimistic view on the availability of other energy resources, institutional and safety aspects.

#### ACKNOWLEDGEMENTS

The authors wish to acknowledge valuable contributions of all members of the Prefeasibility Team, especially Dr. Sudibjo R. of LEMIGAS for his critical remarks during the preparation of this paper.

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## PERSPECTIVES FOR ADVANCED NUCLEAR POWER TECHNOLOGY APPLICATIONS IN THE DEVELOPMENT OF THE VENEZUELAN ORINOCO OIL BELT

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Abstract

Venezuela has the world greatest deposits of extra-heavy oil located at the Orinoco Oil Belt, estimated to contain 1-2x1012 barrels of crude oil. In addition, deposits other than the Orinoco Oil Belt, containing around 260x10<sup>9</sup> barrels of oil with less than 14  $^{O}$ API, have been identified in the country. Assuming a recovery of just 15-20 %, Venezuela could sustain a production rate of 2x10<sup>6</sup> bpd during more than two centuries. So it becomes of great importance for Venezuela to develop and to make this energy reserves available, not just in its own benefit, but for the rest of the world. Steam injection is the most promising method for heavy oil extraction. Also, it is unavoidable to process this kind of oil to obtain useful commercial products. Both, the extraction and the processing, requires substantial amounts of medium and high temperature process heat, implying a large demand of energy which might even get close to a negative net energy balance if unefficient energy production methods are employed. Recent important developments in advanced nuclear power technologies open new possibilities which demand their consideration as the main energy sources in the Orinoco Oil Belt develop-Moreover, it has been found that extra amounts of hydrogen ment. are required for obatining light synthetic crudes from heavy oil. High temperature reactors represent a valid option for the production of the required hydrogen. Additional perspectives appears for the future, as abundant hydrogen produced by nuclear plants, might be used as an energy transportation mean to supply part of the energy demand of the main populated centers of Venezuela. This paper covers a preliminary study, indicating good perspectives for the use of nuclear energy in the exploitation of the extra-heavy oil resources from the Orinoco Oil Belt.

INTRODUCTION

Venezuela has the world greatest deposits of extra-heavy oil located at the Orinoco Oil Belt (OOB). It is estimated that this vast reservoir contains between one and two billion  $(10^2)$  barrels of crude oil. In addition, other 260.000 million barrels of heavy oil, of 14 <sup>O</sup>API or less, have been identified in other deposits around the country (Ref. 1).

This huge amount of oil, with a 10 to a 20 % recovery possibility using known technology, places Venezuela as one of the largest suppliers of oil for many years ahead if it is decided to exploit this resource in sufficient magnitude. In consequence, it is very important that Venezuela devotes efforts to make this energy reserves available.

However, the extraction and processing to obtain useful products from this extra-heavy oil, constitutes a serious challenge for a developing country due to its magnitude, its requirements and the consequences.

Heat stimulation, through steam injection, appears as the most promising method for the extraction of heavy oil. This implies extraordinary amounts of heat to produce the required steam. Additional energy in the form of process heat is needed to treat this crude, an energy demand which tends to increase as more extensive and deep reserves become exploited. This situation calls for efficient and economical ways to produce and use the energy required.

A logical tendency is to try to supply all needed energy from the same crude oil being extracted. However, as the project extends and, after the easiest to recover oil portions have been extracted, the amount of oil used for energy production starts to increase considerably with strong negative economic effects. Also, strategy and environmental factors dictates consideration of different sources, including the nuclear option. At present, when still considerable uncertainties are existant, it is very risky to exclude one or more of the energy supply possibilities.

The General Electric Company of England, in a 1981 Caracas meeting, proposed the use of Magnox reactors for the extraction of the OOB heavy oil (Ref. 2). However, the supply of the energy required for the processing of such oil was not considered in such a proposal, perhaps because conventional nuclear reactor concepts present serious limitations for that purpose.

Although it is possible to establish general trends in the oil situation, history has demonstrated the rapid changes that can occur in this respect, making it very difficult to estimate what the situation would be for the next future. There is consensus in the fact that oil will continue to play a very important role for many years to come, and that light oil reserves are starting to decline rapidly.

A method recently developed in Venezuela, to make direct use of heavy oil as fuel for heat and electricity production, the so called "orimulsion" (Ref. 3), consists of the burning in the furnaces of a mixture of heavy oil with water and some surfactants. The significance of this is that probably a considerable portion of the OOB oil reserves could be used in this way in the next future. However, environmental restrictions may impose limitations to this new possibility due to the high sulfur and other impurities content of the OOB oil.

In the oil processing required to obtain the most demanded commercial products, besides other related processes, for example, to obtain petrochemicals and fertilizers, a considerable demand for hydrogen becomes present. The production of this hydrogen requires substantial amounts of energy, part or all of which could be supplied by nuclear means.

Other aspect which could be connected to hydrogen production, is the particularity that the Venezuelan population is concentrated in the northern part of the country, far from the country main energy sources. This suggests the possibility of using hydrogen as an energy transportation mean. High temperature nuclear reactors could produce abundant hydrogen to be used both for the OOB oil processing and to satisfy part of the energy demand of the main populated centers of the country. In addition, a varied industrial activity in the area present additional possibilities for high temperature process heat applications which could be provided by advanced nuclear reactors.

Another factor in favor of nuclear energy is the environmental issue. Energy produced by nuclear reactors represents the least impact on the environment with zero emission of the ordinary pollutants connected with fossil fuel burning. In particular, in the Venezuela's case, a starting pulp industry in the OOB region, based on pine tree forestry, calls for a restricted use in the area of contaminating energy sources. Also, concerns for a global warming due to high carbon dioxide concentrations in the atmosphere is calling for restrictions in the burning of ordinary fuels.

After some general related information about Venezuela, this paper presents data on the OOB especial characteristics and its oil properties. Then, reference is made to the energy needs for the development of the OOB, presenting an example of energy flow, just to give an idea of the magnitudes involved and how easily the energy demand could increase as the oil recovery project expands.

Since this is a very preliminary study and the considerations are for a period of time beyond the near future, large uncertainties in the economics are involved; so, economic aspects are not presented in this paper. Furthermore, it is recommended that initial efforts should be concentrated in the technical feasibility only. However, it is recognized that economics is one of the main decisive factors in selecting the appropriate technology.

#### INFORMATION ABOUT VENEZUELA

Venezuela is located north of the equator, in the northern part of South America, with coasts in the caribbean sea. It presents a political and economic situation which could be considered as one of the bests in the region, with a high income per capita and the highest electricity consumption per inhabitant in Latin America. Its economy is oriented to make of Venezuela a highly industrialized country based on its income from oil exports and other industrial goods. However, growth during the last years has slowed down considerably due to a heavy external debt obligation and the rapid decline of oil prices.



FIG.1. Oil world production (1986) (millions of barrels per day).

Venezuela's contribution as an primary energy supplier to the world has been of significance, as can be seen in Figure No. 1. However, its oil and gas reserves are of a such magnitude that the production to reserves ratio can be kept relatively low (see Table 1). (The oil reserves figure includes proven reserves from the Orinoco Oil Belt).

Other useful information about Venezuela has been included in Table 1. Although population density is relatively low, there are highly populated regions north of the Orinoco river, mainly in the north-central region, where the capital Caracas is located (Figure No. 2).

Most of the crude produced in Venezuela is of medium gravity, 25 <sup>O</sup>API on the average. In a growing demand scenario, not present now, the tendency is for an increased proportion of heavy components. Several enhanced oil recovery projects have been running for several years, including steam injection in fields at the Zulia's region (Figure No.2). This has provided Venezuela with a valuable experience in heavy oil handling and processing, and considerable technological research is currently going on this field.

The recently completed Raul Leoni Hydroelectric Plant (Guri), has an installed capacity of 9000 Mw, being one of the largest in the world. There are other very important hydroelectric resources located mostly to the south of the Orinoco river, at long

#### GENERAL INFORMATION ABOUT VENEZUELA

(Refs.10,11) (Data is given for 1986 unless otherwise especified)

AREA: 916,700 Km<sup>2</sup> POPULATION: 17.8 MILLION POPULATION DENSITY: 18 INH/Km<sup>2</sup> 388,000x10<sup>5</sup> Bs. (25,900x10<sup>6</sup> US\$) \* G.N.P.: OIL PARTICIPATION: 16% (Down from 30% in 1980) GNP/INH: 21,800 Bs/CAP (1450 US\$/CAP) \* TOTAL ENERGY CONSUMPTION: 479,000 BEP/day (1.12×10<sup>5</sup> TJ/yr) ENERGY CONSUMPTION PER CAPITA: 9.82 BEP/yr ELECTRICITY CONSUMPTION: 20,700x10<sup>3</sup> BEP/yr (40,000 GWh/yr) (30% Hydroelectricity) ELECTRICITY CONSUMPTION PER CAPITA: 1.16 BEP/yr (7.5 GJ/yr) INSTALLED ELECTRICITY CAPACITY : 15,250 HW MAXIMUM ELECTRICITY DEMAND: 5,905 MW (Dec 1987)  $284,600 \text{ m}^3/\text{day}$ OIL RESERVES: 8,800x10<sup>6</sup> m<sup>3</sup> OIL PRODUCTION: (1.79x10<sup>6</sup> Bb1/day)  $(55.4 \times 10^9 \text{ Bbl})$ OIL PRODUCTION TO RESERVES RATIO: 1.2% (86 yr) GAS RESERVES: 2,650×10<sup>9</sup> m<sup>3</sup> GAS PRODUCTION: 24×10<sup>9</sup> m<sup>3</sup>/yr GAS PRODUCTION TO RESERVES RATIO: 0.9% (110 yr)

\*) Official currency exchange ratio: 15 Bs./US\$



FIG.2. Venezuela's territories and sites of energy related activities.

distances from the main comsumption centers. Due to this, thermoelectricity has a considerable participation and its contribution will be even larger at the end of this century, when useful hydroelectric sites will start to get scarce. This aspect was considered in a study performed with the IAEA assistance, in 1978, about the nuclear electric power perspectives for Venezuela (Ref. 4). This study established nuclear power as a valid option for the end of this century. However the electricity demand growth has slowed considerably with respect to the predictions at that time, and now Venezuela has a large installed capacity on reserve.

Due to the Raul Leoni (Guri) hydroelectric plant important contribution in satisfying a large portion of the country's electricity demand, long high-voltage lines are needed to transmit large loads to the main consumption centers.

Venezuela's main industrial zones are located: one at the south-east part of the country by the Orinoco river side, near Ciudad Bolivar and the Guri Dam (see Figure 2), where large steel and aluminum heavy industries are located; a second extended industrial zone, with many small and medium producers, located at the north-central region, close to Caracas, and a third zone at the north-west region where most of the oil and related industries are concentrated.

## THE ORINOCO OIL BELT AND ITS OIL CHARACTERISTICS

The Orinoco Oil Belt is located at the northern side of the Orinoco River, the main river in the country, covering an extension of approximately 700 km. long by 70 km. wide, as shown on the map of Figure No. 2.

A high portion of the oil deposits are found between 200 and 1200 meters of depth, with indication of deposits deeper than the 2000 m. level. The highest portion of the oil has been discovered in the Miocene (Tertiary period) with a minor part in the Paleozoic and in the Cretaceous period (Ref. 5).

One very important characteristic of oil from the OOB is that it is present in the reservoir as a free liquid. This constrast with other extra-heavy oil deposits where the oil is present forming part of impregnated sands, which is the case in Canada.

The most recent estimates, after studies finished by the Ministry of Mines in 1984, give a figure of  $1.2 \times 10^{12}$  barrels of oil in situ at the OOB (Ref. 3).

Table II presents the most important characteristics of three types of crude oil from the region. When compared with oil residues from the Middle-East, venezuelan extra-heavy oil fits between the long and short residues of the Light Arabian crudes, although with much higher metal content (Ref. 7).

As usual in oil deposits, the OOB contains small portions of light and medium oils, which, however, could represent an important quantity due to the reservoir proportions, fractions which could be easily recovered during a preliminary exploitation. TABLE II: ORINOCO BELT CRUDE CHARACTERISTICS IN COMPARISON WITH MIDDLE-EAST

OIL RESIDUES (Ref. 7)

**************************************	API	VISC.(60°C) (cSi)	CONRADSON CARBON (1w)	с/н	VANADIUM (pnm)	SULPHUR (%w)	NITROGEN (\$w)	ASPHALTENES (1w)	LOWER/UPPER HEATING VAL. MJ/kg.
Cerro Negro Guahibo-Lache Jobo-Morichal	8,9 9,1 9,0	5000 2928 5400	15,2 14,2 11,8	8	430 409 390	3,99 3,55 3,92	0,76 0,63 0,52	10,1 1 <u>1</u> ,6 8,6	39/42
Middle-East Light Arabian Large Res. (45% v)	16,6	72	8,0		28	3	0,16	2,9	
Short Res. (13% v) Heavy Arabian	6,5	3000	23		90	4,05	0,34	10,0	
Short Res. (23,2% v)	3	10 <sup>5</sup>	27,7		269	6,0	0,48	13,5	

FRACTION	00B ZUATA	LIGHT ARABIAN
C5-190°C(%v)	1,1	28,2
Sulfur (%w)	1,44	0,03
190-343°C(%v)	15,5	27,5
Sulfur (%w)	2,27	0,82
cetane index	32,3	58*
343-510°C(%v) Sulfur (%w) Conradson Carbon	24,7 3,36	29,5+ 2,09+
(% p)	0,24	0,65+
Nitrogen (ppm)	1973	810+
510°C(% v)	58,7	14,0 **
Sulfur (% w)	4,20	4,0 **
Vanadium (ppm)	794	76 **

Fraction: 343-356°C Fraction: 265-370°C Fraction: 565°C

Under a 20% recovery of the oil in situ, the extractable reserves by known techniques could amount to  $240\times10^9$  barrels, or about 1.3 times the recoverable portion of the rest of the world heavyoil reserves.

Due to the large fraction of vacuum residues contained in the OOB oil, preprocessing is required in order to convert it to a product that could be fed to conventional refineries. The main step consists in reducing its large metal and sulfur content of the crude, besides an hydrogentation treatment necessary to increase the hydrogen to carbon ratio (Ref. 6).

Conventional technologies for heavy oil processing, when used in the OOB crude, produce a relatively low fraction of highvalued commercial products after the refinery processing, besides a very high catalizer consumption due to its high metal content. For this reason, Venezuela has been working intensively in developing new processes with better synthetic crude yields and lower costs, together with indigeneous catalizers made from local raw materials.

Processing based on hydrocracking is the most promising method, especially when using fluidized catalitic beds, such as in the LC-Finning and the H-Oil processes. Venezuela also has developed (and patented) several processes, such as the HDH one based on the hydrocracking method (Ref. 7). In all these processes, the hydrogen consumption is relatively high, which means high energy and methane consumptions, although better economy has been obtained with the HDH process when compared to conventional methods.

The main idea in the heavy-oil treatment is to start with a fraction composition reach in heavy components, as shown by the left bar in Figure No. 3) and to convert it to a synthetic crude with increased proportion of lighter fractions, as indicated by the other bars in the figure, in this case for three variants of the HDH process.



FIG.3. Oil fractions before and after processing the OOB crudes (HDH process).

The extraction process of heavy oil and its treatment to convert it to useful products, is a costly process which adds considerable value to the cost of each barrel of heavy crude. Former studies have calculated an added cost between \$12 and \$17 US dollars per barrel just for the processing (Ref. 7). So, under present oil market conditions, the perspectives for an important development of the OOB look diminished now. To this we have to add the influence of recent discoveries in Venezuela of important light oil deposits (Ref. 8), which could induce delays in the heavy oil exploitation. However, there is consensus on the fact that much better oil prices are not so far in the future in an increasing oil demand scenery. Also internal consumption in the countryl is steadly growing.

#### ENERGY DEMAND IN THE ORINOCO OIL BELT DEVELOPMENT

It is estimated that about 5% of the oil in the OOB could be extracted by primary recovery methods. This yield could increase up to a maximum of 30% by heat stimulation, first, in a steam soak process and, afterwards, by steam drive.

The steam requirements vary accordingly to the deposit characteristics and the oil properties. For the purposes of this study, it is considered that an oil production to steam consumption ratio of about 25 barrels of oil per ton of steam could be obtained during the soak phase. This means that to extract, say, 125,000 barrels of oil per day, 5,000 tons of steam per day (or 208 t/h) are required. In passing to the drive phase, a big jump in steam consumption is needed, since it is estimated that only from two to four barrels of oil are obtained per ton of injected steam; that is, an average of 42,000 t/d (or 1750 t/h) of steam in our example. In a real case, however, the amount of steam varies considerably during the project lifetime, ranging possibly from 30,000 up to 80,000 t/d to produce the 125,000 barrels of oil.

The former figures also indicate a considerable amount of water to be extracted mixed with the oil. In the soak phase it could represent up to 20 % of the total volume, and in the drive phase, up to 80%. Beyond this proportion, it becomes worthless to continue the exploitation of the field, forcing the expansion to other unexploited deposits.

The energy demand for producing the injection steam, plus the energy requirements for pumping and transfering the total amount of liquids, might become of considerable magnitude. Sased on experience obtained in other enhanced oil recovery projects, it is said that about 1/3 of the total energy content of the extracted crude has to be invested in these processes (Ref. 9). To this, we have to add the energy consumption in the preprocessing required by the OCB oil, as aiready mentioned.

Heat losses in the steam transmission lines, and along the injector well due to dissipation through the walls, adds to the energy demand, especially when extensive and deeper deposits are to be exploited. Also, steam quality drops considerably with depth, asking for higher steam temperatures at the injection point with increased energy demand. This last consideration imposes a limitation to the depth of the wells, since, otherwise, not enough driving pressures would be obtained.

As the volume of the liquids to handle increases, due to the increasing amount of water coming from the condensed steam needed to sustain the same crude output, the pumping power requirements also become larger.

Once the oil has been extracted, its processing requires large volumes of hydrogen, besides the heat for the process itself, both indicating large energy consumption. After this, the synthetic crude passing through a refinery would ask for additional amounts of energy.

Considering all these steps, much more than the mentioned one third ratio of crude would be needed to satisfy the energy demand. If unefficient ways are employed, a negative total energy balance might be reached. Assuming an ever increasing rol of oil in the future, reasonable and efficient methods become an imperative. Then, careful studies should be carried out taking into consideration all possible energy sources available if a rational OOB development is wanted.

In Figure No. 4, a simplified energy flow scheme is indicated for the case of the OOB, where only rough figures are given just to provide an idea of the magnitudes involved.



FIG.4. Energy balance in the OOB heavy oil extraction and processing (steam drive condition).

The main assumptions for these calculations were:

A production of 125,000 bpd of oil under steam drive conditions, with a 2.1 b/t oil to steam ratio; 300 injector wells 1000 meters deep; steam heat loss rate through the walls of the injector well of 400 BTU/hr-ft; a heat loss rate caused by viscosity taken here as 100% of the extraction pumping power, and, finally, an arbitrary figure of 30 Mw to cover all surface pumping requirements.

For the crude processing, the figures given in Ref. 7 for the HDH process (case B) were employed, including the requirements of methane as feedstock for hydrogen production, with an energy equivalency of 10,580 Kcal/ $m^3$ ).

It was found that 53,200 BEP/day (1 BEP =  $6.43 \times 10^{9}$  Joules) are required for the production of 125,000 bpd, that is, around 43% of the energy content of the crude produced.

These figures could give a good idea of the energy required in the OOB, especially if a large scale development is under consideration. In such a case, we could be dealing with quantities, say, ten times larger, indicating that about one half of a million of barrels of oil per day would have to be burnt if the nuclear alternative is not employed.

Finally, the atention is called to the fact that the amount of energy required varies considerably during the project life. This fact adds atractiveness to a modular concept in the installation of the energy producing units.

### NUCLEAR ENERGY PERSPECTIVES IN THE OOB DEVELOPMENT

Since the purposes of this Meeting is to discuss the criteria for the introduction of advanced nuclear energy in the solution of specific problems, like the Venezuelan case, only general comments are presented to give proper consideration to the conclusions of the Meeting, after which more detailed studies would be probably recommended.

When all possible applications of nuclear produced high temperature heat are compared with the necessities in the development of the OOB, plus other industrial activities in Venezuela, a match is observed.

Figure No. 5 is an schematic representation of some of the fundamental processes in which nuclear heat could have an important participation, either in competition with or complementing other primary energy sources.

Let us consider more in detail one of these cases, for example, the methane reforming process, shown in Figure No. 6. When considering all the main application reactions of this process, we can see that most of them have potentiality for their application in Venezuela, especially in the hydrocracking processing of heavy oil.



FIG.5. Nuclear process heat for different applications.



Ref.: H.J.Laue, et al., IAEA's Training Course at Karlsruhe Nuclear Res. Center, 1978

FIG.6. Products of nuclear process heat obtained from methane and water.

Venezuela also has an important steel industry, for which a nuclear assisted ore reduction process could be considered for the future. Ammonia is of great significance in the production of fertilizers for a fast growing agricultural industry in Venezuela. Methanol is also important, mainly in the petrochemical industry, as well as a possible energy source for transportation.

Abundant hydrogen produced by nuclear reactors, through the methanation process, could be used in chemical heat pipes to transport energy from the production site to distant and dispersed industrial centers. In this case, a mixture of water and methane is converted by heat into hydrogen and carbon monoxide. This gas is transported and recombined in the usage place, delivering heat and with water and methane as byproducts. This methane could be used too as energy source or it could be fedback, by pipe lines, to the nuclear heating plant to close the cycle.

Nuclear energy, extracted from uranium, a mineral which does not have any other use, produces heat in an efficient and concentrated way. The fact that this option is totally independent from the oil produced, creates appropriate conditions for a more intensive and extensive oil recovery, allowing a much larger fraction of the oil in situ to be extracted, and with a larger economical return. Previous studies considering nuclear power as heat source for heavy oil recovery in Venezuela (Ref. 2), indicated that this option represents economic advantages after the easiest to recover oil portion had been extracted.

The steam properties required in heavy oil extraction, preclude the use of light water reactors due to their relatively low pressure and temperature conditions. Temperatures above 300 °C and pressures between 10 and 15 MPa (1500 to 2200 psi) are necessary for deposits up to 800 meters deep (Ref. 2), similar to the ones at the OOB. Gas cooled reactors would be the only choice here, including the conventional Magnox reactor and its advanced version, the AGR. However, when considering the heat requirements for the processing of the extracted heavy oil, besides the other already mentioned applications, high temperature gas cooled reactors, the HTGR's, are the only option to consider at present.

The HTG reactor can produce heat with temperatures up to  $900^{\circ}C.$ , more than enough to cover the oil extraction requirements and most of the processing needs. HTGR's ranging from 200 to 1000 Mw(t) or more, could be implemented successively as the project expands. The HTGR modular concept, with inherently safe features added on, does not require extra development to enter commercial operation and it presents several advantages for the type of application here discussed.

The main advantage of modular HTG reactors, besides their inherent safety, which is becoming a decisive factor in the next generation of commercial nuclear plants, is that their modular concept facilitates their construction and approval processes, reducing considerably the capital investment and allowing a more economical implantation of units of sizes far more convenient. This last point become of paramount importance in a project like the OOB development because of its expansion requirements. In addition, a more flexible fuel cycle in the HTGR's opens new possibilities for the use of thorium resources which have been identified in the country.



FIG.7. Schematic flow diagram for the OOB crude extraction and processing.

Figure No. 7 presents an schematic flow diagram of the main processes involved in the OOB development using a nuclear heat source as the main energy supplier. A cogeneration option has been selected for its advantages with respect to the energy requirements in a project like this.

In this scheme, the energy generated in the reactor would go to the steam plant, from where a portion of the steam would be used for electricity generation, and the rest of it, the larger portion, would be directed to the injector wells in the oil field.

pretreatment, the crude coming from the After producing а field goes through further processing, selecting for our example, the HDH process (Ref. 7). A portion of the residues, mainly from distillitation unit could be used as heat energy the vacuum where hydrogen is the gasification plant, source for being produced; also for the hydrotreating process. This is the place At the end, the synthetic where the hydrogen would be consumed. crude comes out ready to be processed at conventional refineries.

The nuclear generated heat could be combined with other energy sources, looking for an increase in the operational versatility of the whole plant and the optimization of the production. Environmental considerations related to a fast developing forest industry for pulp production, which is taking place at the OOB region (the Uverito plantations), impose serious limitations to the burning of fossil fuels even if strict emission controls were implemented in the region. The situation gets worst when considering that the oil residues to be burnt have high concentrations of sulphur, nitrogen and some metals, like vanadium and nickel. Nuclear energy would be of great advantage in this case due to the abscence of atmospheric contaminants of such category.

Together with the advantages, the disadvantages of using nuclear energy in the OOB have to be considered in its proper perspective in a necessary cost versus risk analysis. Technological dependence, high investment costs and radiation control are some of the subjects which have to be addressed when considering the possibility of employing nuclear energy for the Orinoco Oil Belt development. So more detailed studies are recommended.

#### CONCLUSIONS

The technical feasibility for the application of advanced nuclear power technologies in the OOB development are very good and this possibility should be taken into serious consideration together with other alternatives in future studies.

Due to the magnitude of the oil reserves, this case is of very much interest not just for Venezuela but for the rest of the world, since Venezuela could continue being an important primary energy supplier.

Efficient ways to satisfy the energy requirements in the OOB development should have to be considered since, otherwise, negative energy balance levels might be approached.

Specific needs and conditions in the Venezuelan case, favor the consideration of nuclear heat as one of the main energy sources. Besides the needs for the heavy oil extraction and processing, there are other interesting opportunities for nuclear process heat applications, for instance, in large steel and aluminum industries.

A particular situation involved with environmental concerns in the development of a recent large and growing pulp industry at the OOB region, increases the possibilities for the use of nuclear energy.

An added incentive for the future is the possibility to use nuclear produced hydrogen in a long distance energy transmission system to supply part of the energy demand of the main Venezuelan populated centers which are distant from the energy producing locations.

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# TECHNO-ECONOMICAL PREREQUISITES FOR THE USE OF NUCLEAR ENERGY SOURCES FOR DISTRICT HEATING

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#### Abstract

The paper describes that at present there are technoeconomical prerequisites for extensive introduction of the nuclear power plants not only for electricity generation but also for residential district heating in USSR. More the 60% of all fuel and energy resources are being consumed for industry and residential district heating.

For economical reasons the plants for district heating must be built near the consumers, so their design should practically completely eliminate any possibility of occurence of severe accidents e,g. as low as  $10^{-6}$  to  $10^{-7}$  per year of unit operation. This enhancement in safety can be achieved by improving the factors such as reactor safety, design, equipment reliability etc.

In USSR a number of nuclear power plants with enhanced safety features and with a capacity of 50-500 MW are being developed both for district heating and heat plus electricity generation. Technoeconomical studies have been carried out at I.V. Kurchatov institute and mathematical models of nuclear district heating plants and heat net works have been developed.

In the northern and north-eastern areas of USSR the nuclear power plants with relatively low capacity may prove to be competitive. During a first stage of district heating water-water reactors can be used but the use of the HTGR is also possible at a later stage.

VG-400 and low-power modular reactors HTGR-M are being developed in the USSR. The first one is designed for cogeneration of high temperature heat, steam and electricity. A second one will be used for large industrial complexes where a high reliability of energy supply is required in the continuous processes.

#### INTRODUCTION

In the USSR the share of fuel spent for electricity generation currently amounts to about 25%. At the same time industry and residential district heating consume more than 60% of all fuel and energy resources.

The experience gained by the national power engineering in the field of power-and-heat supply and district heating as well as the tendency to heat load concentration create favourable conditions for more extensive use of nuclear energy for the needs of industry and residential district heating.

Replacement of the fossil fuel by the nuclear one will contribute essentially to improvement of the fuel-energy balance of this country, whose specific feature is long distances between the districts of intense energy consumption and large fossil fuel deposits, and will benefit in solving the ecological problems.

## SAFETY OF NUCLEAR ENERGY SOURCES

It is necessary for further stable development of nuclear power that improvement of the nuclear energy sources aiming at enhancement of their safety would be accomplished at higher rates comparing with extension of the fields of their application. New nuclear energy sources must be developed, whose design would practically completely eliminate any possibility of occurence of severe accidents with radicactive product releases to the environment.

Of special importance become the problems of nuclear safety improvement in the case of use of nuclear energy sources for district heating as for economical reasons such energy sources must be built near the heat consumers.

Since any power-intense process cannot be absolutely safe the most objective criterion of the nuclear energy source safety can be

the quantitative expression of the risk of its operation including determination of the probability of severe accidents with core degradation and radioactivity release to the environment.

In accordance with the estimates of the International Nuclear Safety Advisory Group (INSAG) the safety level of the modern lightwater reactors corresponds to the probability of core degradation within the range  $10^{-4}$ - $10^{-3}$  events per a year of the power unit operation.

Most of the experts admit that for secure expansion of nuclear power scales in perspective the probability of severe accidents with radioactive product release must not exceed  $10^{-6}-10^{-7}$  events per a year of the power unit operation.

From the technoeconomic point of view it is expedient to develop the nuclear energy sources possessing such safety characteristics in two parallel ways:

1. Radical improvement of the reactor units and <u>development of</u> <u>the power units of enhanced safety</u> on account of the inherent safety properties, improvement of the design approaches, increase in the equipment reliability, wide use of diagnostic means in the framework of the traditional types of the reactor units <u>on the basis of</u> <u>mastered reactor technologies</u> with a permissible probability of a serious core damage not exceeding  $10^{-5}$  events per a year of the reactor unit operation. The total probability of radioactivity release harmful for the population health must not exceed a level of  $10^{-6}$ .  $10^{-7}$  events per year on account of use of the protection and localization systems (containments, filters etc.).

2. Development of the reactor units and power units of maximum attainable safety on account of further reduction of the probability of the proper reactor degradation to  $10^{-7}$  events per a year of operation mainly due to the inherent reactor safety. Realization of the way may require development of quite new design approaches and may be connected with the necessity of mastering new reactor technologies, which, in turn, entails significant time and material expenditures.
However at the current level of development of the probabilistic analytical methods it is expedient, for some objective reasons, to use them only as an addition to the qualitative analysis performed by comparison of the design approaches with the commonly accepted safety indices such as:

- the neutron-physical characteristics of the core, which enable the possibility of appearance of the excess reactivity bringing the reactor to hazardous run-aways to be prevented;

- ability to effective accumulation of heat within a wide range of reactor operation conditions;

- increased stability margin of the reactor materials under the accepted design operation conditions;

- mutual chemical passivity of the core components and the environment;

- amount of the heat removed by natural convection of the coolant;

- use of passive localization systems;

- use of the gravity for termination of the nuclear reaction;

- reactor cooldown without any actions of the personnel,

etc.

USE OF THE NUCLEAR ENERGY SOURCES FOR DISTRICT HEATING PURPOSES

In determination of the places of different nuclear energy unit in the system of the national economy energy supply, the structure of heat generation must be taken into account regarding both the energy carrier type and its temperature potential.

Fig.1 presents the approximate diagram of dependence of heat consumption on the used temperature potential for the national economy and industry.

The structure of district heating for perspective is shown in Table 1 (assessment) /1/.



temperature potential in the national e (1) and industry (2)

TABLE I. Heat consumption structure forecast for perspective, %

Index	1980	Perspective
Hot water production	28	30
Steam generation	28	40
Production of high-tempera- ture process heat	44	30
Total	100	100

As the nuclear heat sources for meeting heat energy needs of residential and industrial consumers, can be used NPPs with uncontrolled steam extraction, nuclear cogeneration plants (NGoGP), nuclear one-purpose district heating plants (NDHP) and nuclear industry heating plants (NIHP).

In practice uncontrolled steam extractions from the turbine unit are used most frequently for heating the NPP itself, personnel settlement and other nearby heat consumers. The heat load from one VVER-1000 reactor unit may amount to 300-450 Gcal/hr. In the USSR a significant experience of nuclear energy source application for cogeneration of heat and electricity has been gained. The Beloyarsk, Kursk, Novo-Voronezh, Rovno, Armenian, and Kol'skaya NPPs can be named as examples. The amount of thermal power extracted from these NPPs varies from 20 Gcal/hr to 265 Gcal/hr. Use of special turbines permits the amount of thermal power extracted from one VVER-1000 reactor unit to be increased to 900 Gcal/hr.

Though the nuclear cogeneration plants have thermodynamical advantages over separate methods of heat and electricity generation, there is an extensive region where application of single-purpose energy sources is more expedient, for example, in the areas with scarce resources of cooling water or where great amounts of electricity are not needed. There can be other technoeconomical reasons for which the single-purpose nuclear energy sources are found to be more attractive.

In the USSR, basing on the approaches adopted for the reactor unit AST-500, a number of nuclear energy sources of enhanced safety and with a capacity of 50-500 MW are being developed both for district heating and heat and electricity cogeneration.

The plants are being designed taking into account the additional safety requirements adopted in the USSR for such plants /2/.

The reactor unit AST-500 is described in detail in some publications /3/.

The technical solutions realized in the AST-500 design have undergone the full cycle of calculation and experimental checks including an extensive analysis of hypothetical accidents. At the nuclear district heating plant which is under construction in the city of Gorkii installation of the equipment of the first AST-500 is near completion.

At the I.V.Kurchatov Institute of Atomic Energy the technoeconomical studies of the district heating systems based on the vesseltype water-water reactors designed for the above-mentioned power range have been carried out. For this purpose the mathematical models of NDHP and heat routes have been developed. The mathematical



1 - 50 rouble/T.F.E. 2. - 70 rouble/T.F.E. 3. -90 rouble/ T.F.E.

modelling permits the multivariant calculations of the AST parameters together with heat routes to be performed.

The curve representing the ratio between the cost of heat obtained from AST and that from the fossil-fuelled boiler systems depending on the energy source capacity is shown in Fig.2.

As seen from Fig.2 reduction in the energy source capacity decreases appreciably the AST competitiveness comparing with the fossil-fuelled boiler unit. Nevertheless in the North and North-East areas of the USSR the nuclear energy sources of relatively low capacity may prove to be competitive.

As seen from Fig.1 the nuclear energy sources based on the water-water reactors can cover, to a considerable degree, the national economy needs in heat energy. Taking into account that the water-water reactors have been mastered both in their construction and operation it is reasonable to use the energy sources with these reactors for district heating purposes at the first stage. However there is an extensive region of the national economy where the heat carriers with high temperature potential are needed.

New possibilities are opened up for nuclear district heating with mastering the high-temperature gas-cooled reactors (HTGR).

Introduction of HTGR is connected with use of efficient energy carriers which would permit to transmit heat in the most convenient and economical way.

The domestic and foreign experiences show that in the nearest future a real possibility is foreseen for development of the chemothermal systems for heat energy transmission and accumulation. Currently the possibility of developing all main elements of such systems has been confirmed. In this case the energy generated by HTGR will be spent to catalytic steam conversion of methane. The cold converted gas supplied to the consumers can be used as the starting semiproduct for petrochemical needs, hydrogen, ammonia and methanol production or as the reducer in the metallurgy.

The heat energy in the chemically-bound state can be released in the methanators producing process heat, steam or hot water.

Since this method permitsheat to be transmitted to many hundreds of kilometers practically without losses the possibility appear to spread district heating to a great number of distant consumers.

The converted gas can be stored in the gas storages (accumulators) and used in the peak-reserve devices, which permits the reliability of district heating to be increased.

Also attractive is the prospect for using the HTGR in energotechnological complexes.

As the energy source for energy-technological complexes the reactor installation VG-400 is being developed, which is designed for cogeneration of high-temperature heat, steam and electricity.

The main parameters of VG-400 are:

Reactor thermal power 1060 MW

Helium pressure in the primary circuit 4.9 MPa

Helium temperature at the core outlet 950°C.

The low-power modular reactor installation HTGR-M is also being developed. Its unit power is about 200 MW (th). For energy supply of large energy-technological complexes such modules can be combined, which also will increase the reliability of energy supply of continuous technological processes.

The works carried out in the USSR, the USA, the FRG and other countries show that it should be expected that the HTGR-based NPPs would possess high safety parameters, primarily assured by the inherent safety properties of the reactor itself.

Table 2 presents the estimates of potential scales of HTGR application for perspective and fig. 3 shows a typical Q-T diagram of petrochemical and petroleum refining plants and the corresponding diagram of the HTGR helium coolant.

Consumer	Fuel consumption (Mil.T.F.E.)	Potential volume of HTG: introduction $(GkW/T)$		
Chemical industry and fertilizer pro- duction	110	20-30		
Ferrous metallurgy	150	10-50		
Petrochemistry and synfuel production	140-160	25-50		
Oil production	60-80	15-30		
Other fields	<b>300-3</b> 50	20-30		
Total	800-900	90–160		

TABLE 2. Potential scales of HTGR introduction (perspective)



Typical Q-T diagram of petrochemical and petroleum refining plants and corresponding diagram of the HTGR helium coolant

In the USSR the works are also being carried out on developing other types of reactor units of various capacities and purposes, which can be used as the energy sources for both the residential and industry district heating.

#### CONCLUSION

The complex of investigations which have been carried out by various research and design organization<sup>S</sup> shows that at present there are technoeconomical prerequisites for extensive introduction of the nuclear energy sources not only for electricity generation but also for residential district heating, intense power processes, technological processes including, in perspective, production of synfuel. For the above purposes more than 60% of all fuel resources are being spent at present (and in the nearest future).

One of the most important problems on whose solution extensive intoduction of the nuclear energy sources depends is the one of development of the safety concept and determination of the concrete standard requirements to the nuclear energy sources on its basis;

fulfillment of these requirements would warran<sup>t</sup> the assured safety when extending the scales of these energy sources application. At present for the analysis of the nuclear energy source safety the deterministic approach in combination with the qualitatively-probabilistic one is justified. However it is evident that in the future the role of the qualitatively-probabilistic approach to the safety assurance problems in nuclear power will increase.

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# THE USE OF NUCLEAR POWER IN REMOTE ISOLATED REGIONS

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#### Abstract

Northern and north-eastern regions of USSR have many kinds of mineral raw materials, rare and precious metals, diamonds and some other valuable natural resources, but their developments are limited due to shortage of power supply. As a result, fuel cost, as well as electric power and heat cost in these regions are substantially higher than developed regions of the country, which becomes the most important prerequisite for the efficient use of nuclear fuel. However, the construction and operation of nuclear power plants in these regions is difficult due to a number of factors such as extrememely sever natural and climate conditions, poorly developed industrial and economic infrastructure, seismisity and lack of water.

Small nuclear power plants with power units of up to 100-150 MWt designed for combined electric power and heat or only heat generation have good prospects in these remote isolated regions of USSR, but the design of these plants should be based on the concept of practical elimination of a dangerous radioactive impact for the nearby areas under any accidents. There are a number of ways and approaches for increasing their efficiency e.g., reduction of cost and time of construction and putting into operation, using maximum degree of plant prefabrication, updating of current safety regulation codes, rationalization of spent fuel handling, increasing of capacity factor and utilization of heat removed from turbine condensers.

The USSR has experience of the construction and long-term trouble free operation of the Bilibino nuclear power and heat generation plant with a number of units of 12 MWe each, as well as the Kolskaya nuclear power plant with VVER-400 reactors in the Murmansk district. A number of regions over the vast North and North-East territory of the USSR have no year-round and reliable ways of communication with the central and other industrially developed regions of the country and sometimes with the neighbouring region as well. This especially applies to the means of delivery of cargoes (freights). The Asian North regions of the USSR primarily pertain to such isolated and difficult-of-access regions.

These regions are the most important source of many linds of mineral raw materials, rare and precious metals, diamonds and some other valuable natural resources. Power supply, however, being the main factors determining the possibilities and rate of economic and industrial development of any region does not often meet even the present demands of these regions and, moreover, limits their further development. The principal cause of such a situation is an extremely non-uniform distribution over the USSR North region territorv of local fuel and energy resources economically expedient for use and their high cost of delivery from other reolons.

As a result, fuel cost. as well as electric power and heat cost in some North regions are substantially higher than the values characteristic of industrially developed regions of the country.

It is obvious that such a situation being characteristic of any region for a sufficiently long-term perspective is the most important prerequisite for the efficient use of nuclear fuel.

As a counterbalance, however, extremely severe natural and climatic conditions, poorly developed industrial and economic infrastructure and remoteness characteristic of the isolated North regions substantially complicate the construction and operation of nuclear power plants (NPP) in these regions. Considerable difficulties arise due to a short construction (summer) season.

long-term frozen ground (permafrost) present almost everywhere seismicity and lack of water resources. Complicated conditions of freights delivery result in increased costs and rather rigid weight and size limitations at the transportation of equipment, building materials and structures.

A localized character, relatively small scale Øf industrialization and considerable remoteness (by hundreds of kilometers) of industrialization centers from one another remain characteristic of the isolated USSR North regions for the longterm perspective. This causes little concentration of electric and heat loads and accordingly small (not more than some tens of MWD required capacities of electric and heat generation plants within the isolated power centers and local power systems. In this case the nuclear power plant will be if not a single but the main source of power in this power center and must reliably operate under autonomous conditions, i.e. have a possibility of manoeuvrable power variation according to the electric and/or heat consumption schedule.

Therefore, the prospects of nuclear power development in remote isolated regions of the USSR are associated primarily with using small nuclear power plants (SNPP) with power units of up to 100-150 MWt designed for combined electric power and heat or only heat generation. As a result, the SNPPs should allow, according to safety conditions, their siting in the immediate vicinity to populated areas and other consumers of heat. The resulting rather rigid requirements put to SNPP safety are hardened by that in case of a severe accident an urgent evacuation of population from the regions considered may prove to be difficult or even impossible due to weather and climatic conditions. This leads to a necessity of developing the SNPPs based on the concept of practical

elimination of a dangerous radioactive impact for the nearby areas under any accidents.

By the present time considerable work has been done in the area of SNPPs in the USSR. It includes design and experimental studies, development work, engineering end economic assessments, the creation of experimental nuclear power plants TES-3, ARBUS, BK-50 and, at last, the construction of the first-in-the-country commercial nuclear heat and power generation plant in the town of Bilibino - the administrative and industrial centre of a remote. difficult-of-access district situated in Chukotsk peninsula. So, in the USSR there is experience in creation and long-term troublefree operation of the Bilibino nuclear power and heat generation plant (NPHGP) with the units of 12 MWe each, as well as of the Kolskaya NPP with the WWER-440 reactors in the Murmansk district. This experience demonstrates the efficiency of using nuclear power sources in the North regions substantially differing in their geographical location, climatic conditions and industrial development level, including the SNPPs.

However, the goal, complicated and specific conditions оf constraction and operation of the SNPPs in the North make us to continue a search for the ways of ensuring an especially high level of their safety, increased reliability and utmost simplification of their operation. In so doing the priority is being given to the use of naturally proceeding processes, inherent safety properties, passive-principle-based protection systems and other similar solutions /1/. It should be noted that such solutions, e. g., the reduced power density of fuel, natural coolant convection and some others, as a rule do not favour the achievement of favourable economic factors by the small power plants. At the same time, the feasibility of the small power plants is determined by their economic competability.

Resently, to solve the problemes of remote regions power supply on the basis of the above high-safety concept the reactor designs for various-purpose SNPPs have been developed: -PWR with heat exchangers placed within the reactor vessel and with natural convection of the primary coolant; -water cooled and graphite moderated reactors with tubular fuel pins and natural primary coolant convection (an advanced version of the Bilibino NPP reactors).

A conceptual design of the pool-type reactor with atmospheric pressure at the water surface of the pool for district heating has been completed.

Technical and economic calculations show that NPPs with such reactors should be competable in isolated remote regions of the Far North. The problems of improving their technical and economic characteristics, however, remain to be the subject of further research, design and development studies. The experience gained from the design, economic assessments and studies of SNPPs siting in the North regions allows to nominate a number of approaches the and ways of increasing their efficiency.

 The reduction of cost and time of construction and putting into operation of SNPP. Here the expedient directions of work are as follows:

1.1. The creation of SNPP with the maximum degree of plant prefabrication: the delivery of equipment as modules or complete sets including large modules, with their transportation by water ways, and as a final development of such an approach - the creation of floating SNPPs.

1.2. The use of single-reactor SNPPs.

Design studies have shown the cost of single-reactor SNPF construction to be appreciably (by 25-30 per cent) less than that of an SNPF consisting of two and more power units provided the

overall installed capacity being the same. But the solution of the question of the construction and operation of a single-reactor SNPF, however, in each concrete case depends on the extent to which the required lavel of reliability of power supply from nonnuclear power sources will prove to be expedient and economically validated. In a general case the problem of choosing an optimum number of the SNPP power units is connected with the development of a combined scheme of the electric, heat and fuel supply of the region or district for the current and perspective time aiven period. In so doing unit powers, the number, composition and type af power and heat generating equipment of the SNPP and of all power plants wich will function in combination with a SNPP other are being optimized. A possibility of using already available nonnuclear power sources, the rational degree of their use in the peak periods of power consumption schedules and some other problems associated with the determination of parameters, the structure and the time of putting into operation of all elements of the given region power generation complex are being determined as well. At the same time the length of power and heat transmission lines is determined und their parameters are optimized.

1.3. Updating of current safety regulation codes.

Amounts of radionuclides, radioactive materials and media accumulated during operation of SNPPs are considerably less than those in large-scale NPPs. The SNPP has low fuel power density, uses primary coolant natural convection under all operating and emergency conditions and is characterized by the absence of largesize primary pipes. All these and some other features of SNPP& allow to ensure increased safety requirements by using in a number of cases more efficient solutions than those suggested by currant safety codes and guidelines developed as applied to large-scale

NPPs. Owing to this, the SNPP technical and economic factors can be improved without decreasing their safety.

1.4. Rationalization of spent fuel handling.

Until recently the SNPP design practice was primarily oriented at the provision of integrated spent fuel storages designed for the total service life of the plant (25-30 years). That has called for considerable additional expenditures and would result in additional complications when removing or taking the SNPP out of service. Therefore, the SNPP design should be carried in conjunction with the optimization of spent fuel handling, bearing in mind the reduction of the number of spent fuel subassemblies stored at the plant and revealing the possibility and expediency of their transportation to centralized spent fuel storages or for reprocessing.

2. Increasing of the capacity factor.

It is obvious that striving for an increased capacity factor, i.e., for the SNPF base-load operation or close to it, contradicts the real plant operation conditions in small autonomous industrial regions where the power consumption schedules should be followed.

For an optimum solution of this problem it is necessary to study the problems of joint SNPP operation within local power supply systems with fossile-fuel or water power plants. Besides, the development studies have shown the efficiency of using the heat accumulators, as well as some other industrial processes which can be conducted during the time of main load reduction, for example, desalination of salt water, purification of industrial and general waste discharges, the production of electrolytic hydrogen.

3. The utilization of heat removed from turbine condensers.

The design studies have shown that the discharge heat can in principle be used for heat supply of greenhouses, desalination and

purification of water, as well as for getting additional electric power with the use of turbines in which low-boiling substances. e.g., freens, are used as working fluids.

To some or other extent these approaches are typical for solving the problem of more efficient use of nuclear power plants in any regions. especially if the problem is concerned with the NPP utilization in isolated small local power supply systems.

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#### **INCENTIVES FOR ADVANCED NUCLEAR POWER TECHNOLOGY**

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## Abstract

The implementation of a nuclear power programme in Egypt has been delayed as a result of problems related to financing of the project, public criticism and opposition to the source of energy itself and unfavorable national and international circumstances originating after the accident at Chernobyl.

Concerned officials and planners are unanimous about the suitability and competitiveness of nuclear power as an energy source which can have a large share in the energy mix proposed to satisfy the growing energy needs of the country.

## I. Egypt's Primary Energy Resources:

It is seldom to find any study about the energy problem in Egypt that does not survey the primary energy resources available in the country. For the purpose of this presentation I shall make use of the data available in the Joint Egypt/United States Report on Energy Assessment which seems to me to compile the most reliable information.(1) A summary of Egypt's potential of different fuel materials and primary energy resources is given in the following:

## I.1 OIL

Significant production of crude oil began in Egypt in the sixties. Published estimates of proven recoverable reserves range between 1.67 and 5.1 billion barrels. A production target of 1 million bbl/day was set for the year 1982.

#### I.2 GAS

Proven reserves of nonassociated gas are 71 to 113 billion cubic meters in the Delta Basin and the Western Desert. For associated gas current production rates in the Gulf of Suez vary from 11 to 28 cubic meters per barrel of oil.

It is worth mentioning that the actual potential of gas resources in Egypt is not well evaluated. Several studies refer to a much higher potential of gas reserves in the country.(2)

## I.3 COAL

The Maghara deposits in Sinai are considered as the most economically recoverable reserves amounting to 35.6 million tons of recoverable coal. It is assumed that the deposit is just large enough to support one electric power plant of economic size. I.4 Oil Shale and Carbonacious Shale:

Deposits of oil shale and carbonacious shale are very limited in Egypt. They can be only exploited if there were a severe lack of other alternatives.

1.5 Uranium and Thorium:

No data are available that describe any potential deposit of uranuim or thorium in Egypt. Indications show a good chance of finding such deposits but this necessitates detailed prospecting and ground surveys. The known phosphate deposits in Egypt can be considered as an economic source for uranium only if phosphate industries and fertilizer's production were developed.

I.6 Geothermal Energy:

Very limited information is available on the extent of geothermal energy sources in Egypt.

I.7 Water Supply:

The River Nile is the only significant source of surface water in Egypt. Since construction of the High Dam at Aswan, approximately 56 billion cubic meters of water per year have been available for downstream uses. The capacity of the High Dam power station reaches 1750 Mwe. Due to successive years of drought in Africa, the year 1988-1989 is a critical year and a low flood of the river may lead to a drop in the capacity of Aswan Dam power station to 700 Mwe which means a loss of 8 billion Kw.h per year.

I.8 Solar Energy:

Egypt has a very high incidence of solar radiation that could be used as an energy source. For comparison northern Egypt receives 12.5 percent more direct solar radiation than Albaquerque, New Mexico, in the USA, and southern Egypt receives 70 percent more.

I.9 Wind Energy:

Two locations in Egypt have an average daily and annual wind speeds high enough to be considered for the development of windpower generators. These are the Mersa Matruh region and the Hurghada region.

I.10 Biomass Sources:

There is some potential of biomass energy in Egypt. As in all developing countries it is difficult to assess the magnitude of the potential of non-commercial energy sources.

## II. NUCLEAR POWER AND ENERGY SUPPLY-DEMAND SCENARIOS:

The supreme council on energy planning in Egypt has carried a study on energy balances and possible alternatives up to the year 2005. Three scenarios for energy supply and demand were proposed according to different economic development rates, these are: A. Scenario of high development rate:

Where it is proposed that an average development rate of 6.3% will be sustained up to the year 2005. For this case energy demand will be satisfied by the contribution from the different alternatives as hydropower, coal and nuclear power. It is estimated in this scenario that a need of 101 billion kw.h electrical energy are needed by the year 2000. These will be satisfied as follows: 12% hydropower, 22.5% coal, 37% nuclear, 28% thermal power (oil & gas).

B. Scenario of intermediate development rate where it is proposed that an average development rate of 5% will be sustained from 1985-2005.

C. Scenario of low development rate where it is proposed that an average development rate of 4.3% is sustained.

#### III. INCENTIVES FOR ADVANCED NUCLEAR POWER TECHNOLOGY:

As already mentioned Egyptian nuclear programme was faced by a number of obstacles. These vary from purely technical to social and political issues. In the following, the different issues raised in favour of SMSRS will be discussed from a third world perspective.

#### CAPITAL COST AND FINANCING

The burden a country has to incur in case of SMSRS is much less than that with high capacity power reactors. This is a situation which is more favorable to developing countries which normally have a deficit in their balance of payment. Also with relatively limited capital investments there, is a reduced capital risk. It should be also emphasized, however; that small reactors would be only adopted if they had a specific generating cost that is not much higher than that of large reactors.

#### SAFETY CONSIDERATIONS

Inherent safety features claimed to be achievable with small reactors are important factors in their use in developing countries with lower potential of trained manpower to operate, control and maintain a nuclear power plant. Certainly some lower power reactor designs if they have low power density and/ or particular geometrical configuartions may

or particular geometrical configurations may incorporate some passive safety features not easily achievable by larger systems. Strong negative coefficients of reactivity are also regarded as important design features. Small reactors with low power densities are considered to offer the possibility of walk-away safety.

In this connection, on the other hand, it should be mentioned that passive safety features are not restricted to small reactors. It may be acknowledged that some small reactor concepts may be able to exploit such features effectively in order to produce a nuclear system which is particularly attractive to developing countries in terms of both its capability for safe operation without a fully developed engineering and regulatory infrastructure and its lower investment risk.

#### Know-how and Technology Transfer:

In any nuclear debate in a developing country the question is raised as whether it is the nuclear technology which is aimed at or is it electric power which is needed in the first place. One of the issues raised by opponents of turn key nuclear projects is that the implementation of a nuclear programme should be considered as a means for development of all sectors of the national economy in the given country. This implies that local participation should start with the early beginning of the nuclear programme and should increase with time. If this concept is applied with SMSNRS, we may be faced with a number of problems.

The concept of modularisation, factory fabrication and shipment to construction site would give no chance for costumers participation and on-the-job training of personnel.

If this is the case, there is a crucial question that should be answered. Why a developing country would prefer to buy a small power reactor rather than a small fessil fuel plant?

It is generally accepted that the current generation of LWRs are oversized for developing countries with limited grid capacities and that the latter may benefit from SMSRS in many applications such as cogeneration of electricity and heat, non electric applications and others

However; close investigation shows, that the suitability of a special type of application depends largely on the country of concern especially the degree of industralization and the population distribution in the country terretories. For Egypt, for example, district heating does not represent a citizen's need.

Process heat and combined heat and power applications necessitate the presence of already operating industrial complexes. The population distribution and urban centres in Egypt are localized along the Nile Valley. Remote area are extremely underdeveloped and the use of small power reactors in these areas may not be justifiable. Solar and biomass energies may be more appropriate.

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## POTENTIAL AREAS OF R&D ACTIVITIES IN THE FRAMEWORK OF INTERNATIONAL CO-OPERATION RELATED TO ADVANCED NUCLEAR TECHNOLOGIES

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#### Abstract

The Belgian nuclear background and its relation towards the international nuclear community are first reviewed.

A survey is then given of the strong lines of the national nuclear industry and of the research potential of Belgium with respect to advanced nuclear power technologies and some non-energy applications.

Finally, a number of attractive areas for Belgian international co-operation in research and development, related to advanced nuclear power technology, are briefly discussed.

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#### 1. THE BELGIAN NUCLEAR BACKGROUND

Belgian industry has a tradition of more than 50 years in the processing and metallurgy of uranium and its by-products, mainly radium. The countries involvement with nuclear energy dates back more than 30 years and has been marked by several important pioneering achievements among nations, dedicated to the exclusive use of nuclear energy for peaceful purposes.

To-day the total installed nuclear capacity in Belgium amounts to 5500 MWe, while nuclear generation covers about 67 % of the national production of electricity. On average electricity consumption represents roughly 5500 kWh per capito and per year. Nuclear electricity is generated exclusively by pressurized water reactors, of which 7 units are being operated on Belgian territory at the sites of Doel (4 units) and Tihange (3 units). The oldest (Doel 1) is in service since 1974. The precursor of this reactor family has been the BR3 reactor, a small 11 MWe PWR located at the site of the Nuclear Research Centre (SCK/CEN) in Mol. This was the first PWR actually built in Western Europe. It was coupled to the grid in 1962 and only last year, 1987, it was decided to take this reactor out of service.

#### 2. INTERNATIONAL POSITION OF BELGIUM

Belgian designers, industrial manufacturers and researchers have been, and still are, involved in feasibility studies, design work, engineering, fabrication and R & D activities related to different kind of reactors with advanced technology features, going from small experimental reactors to demonstration reactors and large power reactors. Among these the fast breeder reactor has received a very substantial amount of effort.

In many cases those activities are performed in the framework of international agreements with European partners and with countries outside the European Community, industrialized as well as developing countries. With respect to the latter, Belgium participates in intergovernmental agreements, concerning general co-operation in nuclear energy, with several developing countries. Bilateral collaboration contracts have also been established with several countries for specific supporting tasks. As an illustration, Belgium has installed a research reactor of the TRIGA type at the University of Kinshasa in Zaire. This work was done in the framework of an inter-university collaboration. Actually this was the first nuclear reactor installed on the African continent.

#### 3. ADVANCED NUCLEAR TECHNOLOGIES

Belgian industry, design and engineering companies and research establishments are actively involved in a diversity of peaceful applications of nuclear technology, mainly energy oriented applications. Noteworthy, because of their modern and advanced nature, are the following activities :

- state of the art in the design and construction of primary reactor circuit components, electro-mechanical equipment, civil engineering works, instrumentation and control systems;
- development and fabrication of advanced nuclear fuels (rods and fuel assemblies);
- advanced reactor core design studies and core management strategies;
- assessment of reactor safety and improvement of safety equipment and systems;
- research in reprocessing technology and its spin-off, e.g. the conditioning of highly radioactive waste;
- research on the treatment, conditioning and disposal of radioactive wastes of various nature (also the exploitation of a waste processing plant at SCK/CEN);
- irradiation testing of nuclear fuels and structural materials (mainly in the reactor BR2 at SCK/CEN);
- commercial production and conditioning of radio-isotopes for biomedical and industrial applications.

Many of these activities are still going on at a strong level, in particular the further improvement of the nuclear fuel for light water reactors (e.g. mixed oxide fuel and fuel with burnable poison) and for fast breeder reactors (high density fuel and advanced cladding made of oxide dispersion strengthened ferritic steel), the research in the geological disposal of nuclear waste, the study and experimentation of reactor accidental behaviour and the commercial production of radio-isotopes.

New subjects have been started, such as decommissioning studies and remote techniques for safe close down and clean up of end-of-life plants. Also non-energy applications of the neutron are gaining importance, e.g. diagnostics by activation analysis, condensed matter characterization, assaying of uranium and thorium in ore samples and fundamental physics research.

## 4. POTENTIAL AREAS OF R & D RELATED TO INTERNATIONAL CO-OPERATION

The items which are briefly discussed hereafter are rather illustrative and by no means exhaustive. They pertain to different R & D domains where

particular expertise has been built up in the past and by which Belgium could effectively contribute to the introduction of advanced nuclear power technology in both industrialized and developing countries. A rather complete spectrum is considered, ranging from reactor system development and design up to waste conditioning and final disposal (with the exception of fuel enrichment and reprocessing).

#### a. <u>neutronic reactor core studies</u>

These studies comprise core optimization, life time and reactor control systems, as well as transient operating modes. Core management methodology during reactor life time can also be studied. Experiments in a low power critical reactor facility can be carried out. Example : neutronic studies have been performed for the projected Swiss district heating reactor GEYSER; work performed by BELGONUCLEAIRE and SCK/CEN.

#### b. safety commissioning

Belgian experience is at hand for the safety assessment and check-out of (peaceful) nuclear installations during their construction phase and their start-up.

c. pressure vessel long-term behaviour

Expertise is available for the experimentation and assessment of test methods regarding the embrittlement and degrading of metallurgical properties of pressure vessel steel induced by neutron irradiation bombardment.

#### d. prospection techniques

Quite an interesting technique has been developed for rapid, accurate and inexpensive analysis of uranium and thorium contents in natural samples (rock or mineral ores); the assay is based on the detection of delayed neutrons after exposing the sample to a short irradiation (e.g. in the research reactor BR1).

#### e. processing of radioactive waste

- . low active waste :
  - conditioning by encapsulation in bitumen
  - conditioning by high temperature slagging incineration (produces granitic type of granular concentrates)
- . high active waste : development of vitrification process

## f. disposal of radioactive waste

A Belgian R & D programme on geological disposal of highly active waste was initiated in 1974 with the main objective to demonstrate the technical feasibility and safety of final disposal in a deep argillaceous formation. This has lead to an extensive in-situ research programme (co-sponsored by the CEC), being conducted at an underground facility constructed in a geological clay formation at the site of SCK/CEN-Mol.

Scientific international collaboration in this promising field is also being promoted through the IAEA.

#### g. in-pile testing of nuclear fuels and structural materials

The materials testing reactor BR2 at Mol is a powerful and well-known tool for efficient irradiation testing. Beside the very high neutron fluxes, unique in Europe among MTR's, it offers the capability of large experimental channels and the flexibility of neutron spectrum tailoring. Nuclear fuel pins and sub-assemblies, representing different types of power reactors (including full scale and pre-irradiated pins) can be tested in BR2 under relevant neutronic, thermophysical and cooling conditions. Nominal, off-normal and accidental reactor conditions can be simulated.

#### h. education and training

Universities, the Mol nuclear research centre and nuclear companies in Belgium have a long-standing tradition in training and specialising foreign students and researchers in different nuclear disciplines. In the framework of bilateral agreements between governments or with the IAEA, many trainees, especially from developing countries, are given the opportunity to specialise in Belgium.

#### 5. CONCLUSION

Being among the very early pioneering countries in nuclear energy deployment, Belgium has established to-day an impressive nuclear electricity generating capacity and has developed a mature nuclear infrastructure comprising a multitude of industrial and research activities.

Through its past experience, its vast expertise and present-day advances in almost all nuclear application areas, Belgium can offer attractive potentialities to contribute to the introduction of advanced nuclear power generation in the developing countries.

## MULTICRITERIA DECISION ANALYSIS FOR NUCLEAR TECHNOLOGY ASSESSMENTS

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#### Abstract

The paper presents the nuclear services of BELGATOM, an architect and consulting engineers company, in particular the use of decision tools for further studies on the introduction of advanced nuclear technologies. These decision tools are based on Multiple Criteria Decision Analysis (MCDA) and its' purpose is to assist the decision makers in either selecting "good" solutions or in ranking alternatives in presence of several decision criterias. This MCDA can be applied in any field such as personnel management, site selection, R & D, capital budgeting, energy management etc.

BELGATOM has also developed its' own tools called STRANGE (Strategy for Nuclear Generation of Electricity) for solving the strategic problems such as fuel cycle optimizations, reactors deployment etc.

1. BACKGROUND

BELGATOM is an architect and consulting engineers formed by the two main Belgian engineering companies, BELGONUCLEAIRE and TRACTEBEL. BELGATOM is the co-ordination vehicle for the overseas and domestic nuclear engineering activities of its two shareholders. Its contribution to the national programme was of upmost importance and it is covering a wide range of services in the nuclear industry. These services include :

- The complete engineering of

- . nuclear power plants ;
- . research reactors ;
- fuel fabrication plants ;
- . laboratories ;
- . testing loops ;
- . various facilities for
  - fuel handling and storage ;
  - fuel reprocessing ;
  - waste treatment and storage.
- The nuclear safety.
- The quality assurance.
- The fuel economics, management, design.
- The seismic, stress, structure analysis.
- The operation of nuclear facilities.

In the frame of the BELGATOM consulting activities, BELGONUCLEAIRE executed, in 1987, several contracts for the National Research Center at Mol (CEN/SCK), the National Agency for Radioactive Waste and Fissile Materials (ONDRAF/NIRAS), and the Commission of the European Communities (CEC).

In addition to its activities within BELGATOM, BELGONUCLEAIRE is also very active in the development to an industrial scale of the MOX technology, within the frame of a Belgian - French industrial association, and in the European R & D fast reactor programmes associating five countries and in the associated development of the European Fast Reactor.

For the TCM, it is worth mentioning the safety assessments performed in 1987 :

- Safety assessment of the nuclear facilities (for the CEN/SCK).
- Evaluation of inherent LMFBR core safety features (for the CEC).

In the field of system analysis and strategic studies performed by BELGONUCLEAIRE in the frame of BELGATOM, recent evaluations have been made in the use of alternate nuclear fuel cycles in the European Community.

In this context, decision tools based on multiple criteria analysis have been used and original approaches have been developed. These tools are of interest for further studies on the introduction of advanced nuclear technologies. We describe them shortly.

#### 2. MULTI-CRITERIA DECISION ANALYSIS (MCDA)

Multi-Criteria Decision Analysis is a fast growing field of operational research. Its purpose is to assist decision makers in either selecting "good" solutions, or in ranking alternatives in presence of several decision criteria which are quite often in conflict. Applications can be found in virtually every field from personnel management, site selection, R & D capital budgeting, energy management, etc.

MCDA analysis encompasses :

- the definition of the relevant criteria and how to describe them, like weights, priorities, preference functions, etc.;
- (2) the analysis of the system and definition of alternatives, flow sheets, etc.;
- (3) the system assessment using the criteria.

Two main types of problems are found in practice :

- (A) A finite number of alternatives have to be ordered according to preferences (Ranking problem).
- (8) One or several actions have to be selected from an often larger or even infinite set of alternatives (Choice problem).

For the problem type (A), a large asset of so-called outranking methods are available, which are most popular in Western Europe (rather than the utility function approach mostly used in USA).



FIG. 1 An example of a hierarchy of criteria for energy mix decision-making.

Fig. 1 shows a cross-section of possible criteria for a national energy mix decision. The set of criteria has a hierarchical structure. At each level, the criteria are assumed to give an exhaustive representation of the decision problem. They are broken down into simpler aspects when moving down the hierarchy. The fundamental entities at the bottom level are used to rank the candidate solutions, which are in this case the debated energy systems.

The analysis is based on pairwise comparisons. It is first applied between the criteria, to gain information on their relative weights. Then, candidate solutions are compared pairwise according to each individual criterion.

The result is a ranking relation between solutions, which might be incomplete. In fact, outranking methods do not attempt to aggregate criteria at any price to force a ranking, and they accept that some solutions may be incomparable, if too large conflicts exist between the criteria.

Software packages are available to assist the decision process. They are now well implemented instruments.

Strategic problems like fuel cycle optimizations or reactor deployment belong to problem type (B). BELGONUCLEAIRE has developed its own approach called STRANGE (STRAtegy for Nuclear Generation of Electricity). It is based on mathematical programming optimization techniques. The variables are continuous, and uncertainties can be taken into account in the form of scenarios. The method is interactive : a solution meeting all the constraints and considered as a good compromise with regard to the conflicting criteria is produced at each calculation step. The following dialogue gives to the decision-maker the opportunity to revise this solution, by changing priorities between criteria, or by imposing additional constraints.



Fig. 2 MOX quantities as a function of time and assurance of supply.

Fig. 2 shows a strategy calculation on MOX recycling in European power plants. The quantities of MOX loaded into LWR's are plotted in function of time for several compromise solutions. Each solution depends on how much priority is given to the security of natural uranium supply. The other criteria in this analysis are the fuel cycle costs and the level of employment in the nuclear industry. This calculation has been presented at the IAEA meeting mentioned in the references.

Several other real world examples of this methodology are available in the open literature.

Note that an extension of the method - called MOMIX - (Multi-Objectives with MIXed variables) allows to include integer and boolean variables in the analysis, representing for instance the investment decisions in the fuel cycle industry.

#### 3. CONCLUSIONS

Thanks to the know-how gained through the national programme in Belgium, BELGATOM can offer consulting and engineering services to the nuclear industry worldwide. Valuable instruments for the assessments of nuclear technologies can be provided by Multi-criteria Decision Analysis.

## USEFUL REFERENCES.

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On MCDA :

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## FRANCE: A NATIONAL SURVEY

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#### Abstract

The paper is a collection of tables and graphs describing the French overall energy situation such as total energy resources, energy supply, net electricity generation of all available resources from 1950 to 1987, electricity consumption, electricity balance, thermal electric capacity, primary energy consumption and primary energy demand forecasted upto year 2000. This paper also gives the French nuclear units performances and expected capacity in commercial operation on 1st January of each year from 1989 to 2000.

Treating, conditioning and disposing of, low, medium and high activity wastes are performed by France's own means and on it's own territory.

#### ENERGY RESOURCES IN FRANCE

	PROVED RESERVES 86/01/01 (Mtoe)	PRODUCTION 1986 (Mtoe)	R/P (1) (years)	INDEPENDENCE (2) (%)	
COAL	286				
LIGNITE	39	10.6	11	52.5	
OIL	30	3.5	9	4	
GAS	35	3.5	סנ	14.5	
URANIUM (3)	<b>63</b> 0 (4)	32	<b>2</b> 0	52	
TOTAL	1020	49.6	221	46 (5)	

(1) Reserves divided by the production of 1986

(2) Production divided by consumption of 1986

(3) Reasonably Assured Resources

(4) Used in present LWRs

(5) Taking account of electricity production and exports

SOURCE : WEC-86, IAEA

#### RENEWABLES IN FRANCE

Commercial Energy Mtoe/y in 1985

WOOD	33 ( <b>*</b> )
URBAN WASTES	<b>0-3</b>
INDUSTRIAL WASTES	Ø3
GEOTHERMAL	©-2
INDUSTRIAL THERMAL WASTES	80.0
SOLAR ENERGY	0.04
TOTAL	3.92

SOURCE : Observatoire de l'Energie - DGEMP 87

## FRENCH ENERGY RESOURCES



## PRIMARY ENERGY DEMAND FORECAST IN FRANCE

## Mtoe

	المتحديد والمتحديد والمتحدين والمحاد و		and the second sec	the second se	the second s
	1987 (1)	19	90	200	ю
COAL	17.8	(2) 16.8	(3) 17.8	(2) 16.0	(3) 15.7
OIL	84.7	73.9	81.3	<b>62.</b> 0	80.2
GAS	24.1	23.2	25.6	23.9	32.2
HYDROPOWER	15.9	15.9	16.0	16.4	16.5
NUCLEAR	58.0	69.4	69.1	85.2	93.2
RENEWABLE	4.0	4.1	4.2	5.2	4.5
ELECTRICITY EXCHANGE	-6.6	-8.9	-6.7	-11.1	-8.9
TOTAL PRIMARY ENERGY	197.9	194.4	207.3	197.6	233.4
(including non-energy use)	(12.7)	(11.2)	(12.4)	(10.6)	(12.9)

(1) Actual, provisional figures

(2) Scénario A : low economic growth, high energy prices

(3) Scénario B : high economic growth, low energy prices

SOURCE : Observatoire de l'Energie - DGE IP 1987

## Mtoe

1987 (Provisional figures)	COAL	OIL	GAS	ELEC- TRICITY	RENEWABLE	TOTAL
Primary energy production	9.7	3.6	3.2	74.7	4.0	95.2
Imports	9.1	95.0	22.9	1.9		128.9
Exports	- 0.8	- 11.0	- 0.4	- 8.5		- 20.7
Change in stocks		- 1.5	- 0.8			- 2.3
TOTAL	18.0	86.1	24.9	68.1	4.0	201.1
ENERGY INDEPENDENCE %	53.9	4.2	12.9	109.7		47.3

## 1986

Primary energy production	10.6	3.5	3.5	14.5 (H) 56.4 (N)	4.0	92.5
Imports	11.6	93.0	21.7	1.8		128.1
Exports	- 0.8	- 12.3	- 0.2	- 7.4		- 20.8
Change in stocks	- 1.2	1.8	- 0.7			- 0.1
TOTAL	20.2	86.0	24.2	65.3	4.0	199.7
ENERGY INDEPENDENCE %	52.5	4.0	14.5	108.6		46.3

## <u>1973</u>

Primary energy production	17.3	2.2	6.3	10.7 (H) 3.3 (N)	<b>2.</b> 0	41.8
Imports	10.4	141.2	7.6	1.0		160.2
Exports	- 1.3	- 12.8	- 0.1	- 1.7	~	- 15.9
Change in stocks	1.7	- 2.1	- 0.3			- 0.7
TOTAL	28.1	128.5	13.5	13.3	2.0	185.4
ENERGY INDEPENDENCE &	61.5	1.7	46.4	105.0		22.5

(H) Hydro-electricity - (N) Nuclear Electricity

SOURCE : Observatoire de l'Energie

## FRENCH PRIMARY ENERGY CONSUMPTION

(BY SOURCE)

## Mtoe

	COAL	OIL	GAS	HYDROPOWER (1)	NUCLEAR	ELEC.(2) EXCHANGES	RENEWABLE	TOTAL
1960	43.6	29.4	2.5	9.1	0.03			84.6
1970	34.9	94.6	8.2	12.7	1.3	- 0.1	2.0	153.6
1973	27.8	126.6	13.3	10.7	3.3	- 0.7	2.0	183.0
1975	24.9	110.1	15.6	13.5	4.1	+ 0.6	2.3	171.0
1979	31.9	118.8	21.0	14.9	8.9	+ 1.3	3.0	199.8
1980	31.0	110.9	21.2	15.4	13.6	+ 0.7	3.2	<b>196.</b> 0
1981	28.5	100.5	22.0	16.5	23.4	- 1.0	3.4	193.3
1982	29.2	94.2	21.5	16.4	24.2	- 0.9	3.6	188.2
1983	26.3	90.5	22.5	16.0	32.0	- 2.9	3.7	188.1
1984	25.2	86.9	23.5	15.2	42.5	- 5.5	3.8	191.6
1985	24.1	84.3	23.3	13.4	49.8	- 5.2	3.9	193.6
1986	20.0	84.9	23.6	14.4	56.1	- 5.7	4.0	197.3
1987 (3)	17.6	84.7	24.1	15.9	58.0	- 6.6	4.0	197.9

(1) Hydropower + climatic correction on total electricity
 (2) Exports (-) or imports (+)
 (3) Provisional figures

SOURCE : Observatoire de l'Energie

**FRENCH ENERGY SUPPLY** ( SOURCE SHARES IN PERCENT )


## NET ELECTRICITY GENERATION IN FRANCE\*

TWh

YEARS	CONVENTIONAL THERMAL	HYDRO	NUCLEAR	TOTAL
1950	17.0	16.3		33.3
1955	24.1	25.6		49.7
1960	31.6	40.5	0.13	72.2
1965	54.1	46.4	0.90	101.4
1970	78.9	56.6	5.15	140.7
1973	113.0	47.5	14.0	174.5
1975	101.2	59.9	17.4	178.5
1980	118.8	<b>69.</b> 8	57.9	246.5
1981	92.0	72.7	99.6	264.3
1982	92.2	71.0	103.1	266.3
1983	76.0	70.8	136.9	283.7
1984	60.6	67.4	181.8	309.8
1985	52.1	63.6	213.1	328.8
1986	40.5	64.4	241.4	346.3
1987 (1)	37.0	71.6	251.3	359.9

(1) Provisional figures

# SOURCE : E.D.F. Résultats Techniques d'Exploitation 1987

\* EDF is the unique utility in France and enjoys the monopoly of electricity distribution.



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YEARS	CONSUMPTION		BALANCE IMPORTS-EXPORTS
	Inland (1)	Net (2)	(3)
1925	11.4	9.85	
1930	17.2	15.05	
1935	17.8	14.95	
1 <b>94</b> 0	19.1	16.1	
1945	19.7	15.5	
1950	33.4	28.9	0.36
1955	49.6	44.1	- 0.06
1 <b>96</b> 0	72.0	65.2	- 0.1
1965	102.2	94.1	1.0
1970	140.0	130.1	- 0.5
1973	171.3	159.9	- 3.0
1975	180.7	168.3	2.5
1980	248.7	231.5	3.1
1981	258.3	240.6	- 4.8
1982	261.4	243.4	- 3.8
1984	282.3	261.0	- 24.8
1985	303.0	279.8	- 23.4
1986	317.7	294.1	- 25.5
1987 (4)	327.5	303.0	- 29.7

(1) Inland consumption equals domestic generation plus imports less energy used for pumping

(2) Net consumption equals inland consumption less transportation and distribution losses

(3) Balance: Imports (+), Exports (-)

(4) Provisional figures

SOURCE : E.D.F. Statistiques 1987, Résultats Techniques d'Exploitation 1987

## ELECTRICITY BALANCE IN FRANCE

	19	973	1986		1987 (1)	
	TWh	(8)	TWn	(8)	TWh	(%)
Hydro	47.5	27.2	64.4	18.6	71.6	19.9
Conventional Thermal	113.0	64.8	40.5	11.7	37.0	10.3
Nuclear	14.0	8.0	241.4	69.7	251.3	69.8
TOTAL GENERATION	174.5	100	346.3	100	<b>359.</b> 0	100
Balance Imports-Exports	- 3.0		- 25.5		- 29.7	
Used for pumped storage	171.5 - 0.2		320.8 - 3.1		330.2 - 2.7	
Gross Inland Consumption	171.3		317.7		327.5	
Losses	- 11.4		- 23.6		- 24.5	_
NET INLAND CONSUMPTION	159.9		294.1		303.3	
of which : High Voltage Low Voltage	114.8 45.1		175.9 118.2		179.5 123.5	

(1) Provisional figures

## PEAK LOAD DEMAND OF THE FRENCH NETWORK

1950	Thursday December 21	6.6 GWe
1955	Wednesday December 21	8.9 GWe
1 <b>96</b> 0	Thursday December 15	12.9 Gwe
1965	Thursday December 9	17.5 GWe
1970	Friday December 18	23.3 Gwe
1975	Tuesday December 16	32.0 GWe
<b>198</b> 0	Tuesday December 9	44.1 GWe
1982	Thursday January 14	45.4 GWe
1984	Tuesday January 24	47.4 Gwe
1985	Wednesday January 16	60.0 Gwe
1986	Monday February 10	58.0 Gwe
1987	Tuesday January 13	62.3 Gwe

SOURCE : E.D.F. Résultats Techniques d'Exploitation 1987

# **1987 FRENCH ELECTRICAL OUTPUT**



## (8)

	1973	<b>198</b> 0	1983	1984	1986	1987
INDUSTRY (Iron and Steel not included)	36	41	47	46	50	45.9
IRON AND STEEL	18	22	26	25	26	28.4
RESIDENTIAL AND COMMERCIAL	22	37	42	44	47	48.9
AGRICULTURE	8	10	12	13	15	12.5
TRANSPORTATION	4	4	4	4	4	4.0
TOTAL FINAL ENERGY	22	29	33	34	36	36.55

TODEVA CTIC	1 <b>99</b> 0		2000		
FUNIXABIS	Scenario A	Scenario B	Scenario A	Scenario B	
INDUSTRY (Iron and Steel not included)	55	52	60	56	
IRON AND STEEL	27	27	<b>3</b> 0	32	
RESIDENTIAL AND COMMERCIAL	53	52	63	60	
AGRICULTURE	16	14	15	15	
TRANSPORTATION	4.6	4.5	5	5.1	
TOTAL FINAL ENERGY	40.3	38.8	47	44.3	

\*(A) = high hypothesis, (B) = low hypothesis.

SOURCE : Observatoire de l'Energie - DGEMP 1987

#### HYDRO PLANTS IN FRANCE

#### Nominal Capacity

#### Mile

## AS OF 87/12/31

	E. D. F.	non E. D. F.	TOTAL
Run of river	6 700	900	7 600
Daily storage	4 700	300	5 000
Storage	7 100	400	7 500
Pumping	1 700	0	1 700
Pumping and storage	2 500	0	2 500
TOTAL	22 700	1 600	24 300

#### MEDIAN POTENTIAL OUTPUT (1)

#### Gwh

AS OF 87/12/31

an a	E. D. F.	NON E. D. F.	TOTAL
Run of the river	34 300	4 400	38 700
Daily storage Storage	16 000 11 500	1 000 600	17 000 12 100
Pumping and storage	1 200	о	1 200
	63.000	6 000	60, 200
TOTAL	63 000	6 000	69 200

(1)The annual potential output of an hydro plant is the amount of energy it would produce or store during the year without any unexpected outage or exploitation constraint. The median potential output is the average value of annual potential outputs over the maximum number of available annual data.

SOURCE : E.D.F. Résultats Techniques d'Exploitation 1987

## THERMAL ELECTRIC CAPACITY IN FRANCE

## AS OF 88/01/01

	CAPACITY Mine
ELECTRICITE DE FRANCE	
CONVENTIONAL THERMAL TOTAL	17 200
Series (125 MWe, 4 units 468 MWe (250 MWe, 28 units 7 020 MWe (600 MWe, 10 units 5 835 MWe (700 MWe, 4 units 2 740 MWe	
gas turbines, 21 units 853 MWe	
NUCLEAR TOTAL	49 113
Gas Graphite Reactors, 4 units 1 740 MWe PWRs, 47 units 45 940 MWe LMFBRs, 2 units 1 433 MWe	
TOTAL E. D. F.	66 313
CHARBONNAGES DE FRANCE (1)	, an a fair an
CONVENTIONAL THERMAL	3 300
Series (125 MWe, 8 units 936 MWe (250 MWe, 4 units 1 018 MWe (600 MWe, 2 units 1 175 MWe	
OTHERS AUTOPRODUCERS	an fan 194 yw Brennen yn Armen yn Alfren yn Africa yn yn Armen yn Africa. Yn yn Armen yn Armen yn Armen yn Afri
CONVENTIONAL THERMAL	3 700
Series 125 MWe, 2 units 234 MWe	
NUCLEAR	305
PWR	
TOTAL DOMESTIC	73 618

(1) State owned coal mining Company

SOURCE : E.D.F. Résultats Techniques d'Exploitation 1987

## As of 88/01/01

53 UNITS INSTALLED

capacity 49 693 net MWe

Reactor type	Name of the unit	Capacity net MWe	Year of commercial operation
4 units Gas-graphite 2 015 net MWe	CHINON-A-3 ST-LAURENT-A-1 ST-LAURENT-A-2 BUGEY-1	480 480 515 540	1967 1969 1971 1972
2 units LMFBR 1 433 net MWe	PHENIX CREYS-MALVILLE	233 1200	1974 1986
47 units PWR - 1 PWR- 300 305 - 34 PWR- 900 30520 - 12 PWR-1300 15420	CHOOZ A-1 FESSENHEIM-1 FESSENHEIM-2 BUGEY-2 BUGEY-3 BUGEY-4 BUGEY-5 DAMPIERRE-1 GRAVELINES-B-1 TRICASTIN-1 TRICASTIN-2 GRAVELINES-B-2 DAMPIERRE-2 TRICASTIN-3 DAMPIERRE-3 GRAVELINES-B-3 GRAVELINES-B-3 GRAVELINES-B-4 TRICASTIN-4 DAMPIERRE-4 LE BLAYAIS-1 ST-LAURENT-B-1 ST-LAURENT-B-1 ST-LAURENT-B-2 BLAYAIS-2 BLAYAIS-2 BLAYAIS-3 BLAYAIS-3 BLAYAIS-4 CHINON-B-1 CRUAS-MEYSSE-2 GRAVELINES-C-5 PALUEL-2 CRUAS-MEYSSE-2 PALUEL-2 CRUAS-MEYSSE-4 GRAVELINES-C-6	310 880 880 920 920 900 890 910 915 910 890 910 915 890 910 910 910 910 910 910 910 9	1970 1977 1978 1979 1979 1979 1980 1980 1980 1980 1980 1981 1981 198

SOURCE : CEA ELECNUC Data Bank

#### NUCLEAR POWER PLANTS IN FRANCE

#### As of 88/01/01

UNITS	INSTALLED	(continued)

در از می از می این از می داد. این هم می در از می این این می در این ا		
PALLJEL-3	1290	1986
ST-ALBAN-1	1300	1986
PALUEL-4	1290	1986
FLAMANVILLE-1	1290	1986
ST-ALBAN-2	1300	1 <b>9</b> 87
FLAMANVILLE-2	1290	1 <b>9</b> 87
CHINON-B-3	<b>87</b> 0	1987
CATTENOM-1	1265	1987
CHINON-B-4	<b>89</b> 0	1988
CATTENON-2	1265	1988
BELLEVILLE-1	1275	1988
NOGENT-1	1275	1988

10 UNITS UNDER CONSTRUCTION capacity 12 960 net MWe

Reactor type	Name of the unit	Capacity net MWe	Year of commercial operation
10 units PWR - 8 PWR-1300 10210 - 2 PWR-1450 2750	BELLEVILLE-2 NOGENT-2 CATTENOM-3 PENLY-1 GOLFECH-1 CATTENOM-4 CHOOZ-B-1 PENLY-2 GOLFECH-2 CHOOZ-B-2	1275 1275 1265 1290 1275 1265 1375 1290 1275 1375	1988 1989 1989 1990 1990 1991 1991 1992 1993 1993

(Without commitment date) 1 UNIT ANOUNCED

Reactor type	Name of the unit	Capacity net MWe	Year of commercial operation	
net Mwe				
-1 PWR-1450 1375	CIVALIX-1	1375	(1995)	

SOURCE : CEA ELECNUC Data Bank



#### REACTOR TYPE

- GRAPHITE-GAS REACTOR
- . HEAVY WATER-GAS REACTOR
- PWR.ONCE THROUGH COOLING SYSTEM
- PWR,CLOSED COOLING SYSTEM,TOWERS

### PWR STANDARDIZED SIZE

- 34 \_ 900 MW+ PWR CLASS UNITS 1
- 20\_ 1300 MWe PWR CLASS UNITS
- 3 1450 MWe PWR CLASS UNITS

8	TATUS	of the units		
1		IN OPERATION	63 UNITS	·
		in commercial operation :	48 UNITS	
		UNDER CONSTRUCTION	10 UNITS	
	۲	CLOGEDOWN UNITS	6 UNITS	

1987 COMMITMENT		
(Chooz #2)	1 UNIT	
COMMITMENT AFTER 1	86	
(Civaux 1)	1 UNIT	

#### FRENCH NUCLEAR UNITS PERFORMANCES.

	YEARLY VALUES						
NUCLEAR UNITS	1987	1986	1985	1984	1983		
		KC Kp	Kd Kp	Kd Kp	<b>КО</b> Кр		
All EDF units	75.9 \$5.1	78.8 71.3	81.0 75.1	79.9 74.4	69.4 66 2		
PWR-900 units	78.8 \$6.3	83.2 75.4	83.0 76.7	81.9 76.0	72.1 68 4		
PWR-1300 units	72.9 65.6	64.5 59.8	97.3*101.0*				
GCR units	<b>30.2</b> 40.9	50.7 39.7	48.8 47.1	56.3 56.2	44.6 45.3		
Phenix,FBR unit	74.5 75.2	73.2 74.4	56.8 56.5	70.2 68.9	55 3 54.7		

#### UNITS ON COMMERCIAL OPERATION.

values with \* : results from commercial operation less than twelve months.

#### Definition

Kd : performance indicator, energy availibility factor.

Kp : generation indicator, net load factor.





NUMBER OF UNITS IN COMMERCIAL OPERATION AT THE END OF THE YEAR

# EXPECTED CAPACITY IN COMMERCIAL OPERATION ON JANUARY 1RST OF EACH YEAR.

YEAR	1989	1990	1991	1992	1993	1994	1995	2000
Total plant	51,4	54,0	56,6	59,4	60.3	63.0	62,5	65,6
Graphite-gas	2,0	2.0	2.0	2,0	1,5	1,5	1,0	-
PWR-300	0,3	0,3	0.3	0,3	0,3	0.3	0,3	-
PWR-900	30.5	30,5	30,5	30,5	30,5	30,5	30,5	30,5
PWR-1300	17.2	19,8	22,4	23.7	25.1	26,4	26,4	26,4
PWR-1450	-	-	-	1.5	1,5	2,9	2,9	7,3
FAST BREEDER	1,4	٦,4	1,4	1.4	1,4	1,4	1.4	1,4

#### Net GWe values

\* CIVAUX-1 and 3 PWR-1450 are taken in account before 2000.



Nuclear share in the electric generation %

General survey of policy

The French fuel cycle industry covers all the steps from uranium exploration to waste management. For safety reasons as well as to ensure better exploitation of the energy content of natural uranium and to achieve competitiveness, we have choosen to close the fuel cycle and to recycle uranium and plutonium.

Uranium supply, enrichment, fabrication and reprocessing

- France has got 5 milling plants, Bessines, l'Ecarpière, Lodève, Mailhac and Langogne, totaling a production capability of 3 850 tonnes per year. Cogéma holds 87% of this capacity and C.F.P. 13%.

French known uranium resources amount to 103 000 tonnes, recoverable at less than 130 \$/kg.U and the production of 1987 reached 3 389 tonnes. For additional supply France relies upon French mining companies operating overseas and imports from the international market

- Conversion to UF<sub>6</sub> is achieved in the two Comurhex plants of Nalvési and Pierrelatte. The available capacity of 12 000 tonnes U per year and the production of 1987, 11 000 tonnes, allow domestic supply as well as exports.

- Enrichment supply of the French nuclear power plants is ensured by the Georges Besse plant of Tricastin operated by Eurodif, a multinational company shared by Cogéma (51.55%) and foreign partners. The facility has been operated last year at 3/4 of its nominal capacity of 10.8 MSWUS per year. R&D works underway within C.E.A. on the atomic vapor laser process SILVA (AVLIS) will allow the launching of an industrial facility, when needed, by the turn of the century.

- PWRs fuel fabrication is ensured by two plants, located in Romans and Pierrelatte, whose global capacity amount to 1100 t.HM/y and whose production reached 979 t.HM/y in 1987.

Fuel assemblies for Gas-graphite reactor are fabricated by SICN in Annecy.

MOX fuels for FBRs and PWRs are fabricated by CEA in the Cadarache workshop.

COGEMA has a project of an industrial facility called "MELOX", to be built in Marcoule, due to be commissioned by 1993 and to reach its nominal capacity of 100 t.HM/y in 1995.

- For reprocessing Cogéma operates two plants, UPl at Marcoule for gas cooled reactors irradiated fuel and UP2 at La Hague for LWRs fuel. In 1987, 425 t.HM were reprocessed at La Hague in the UP2 plant whose nominal capacity amount to 400 t.HM. Construction is under way to expand La Hague capacity 1 600 t.HM by the beginning of next decade, when the UP2-800 and UP3 plants will be commissioned.

Waste management

- For low- and medium-level waste, without  $\propto$ , disposal, Andra (Agence Nationale des Déchets Radioactifs) operates two surface facilities using the earth-mounted concrete bunker concept. The Centre Nanche will be closed by 1990 having received some 400 000 m<sup>3</sup> of conditioned waste. The second one, in Soulaines-Dhuys, has received the official licensing authorization in 1987, and will be commissioned by 1990.

- High-level vitrified waste and  $\propto$ -waste disposal will take place in deep geological repositories. Studies under way deal with site selection for an underground laboratory. The geologic formations investigated are salt, granite, shales and clay. The final disposal site is to be choosen by the end of the century.

France implements a policy aiming at treating, conditioning ani disposing of its low, meliim and high activity waste by its own means and on its own territory.

## CRITERIA FOR THE INTRODUCTION OF THE SLOWPOKE ENERGY SYSTEM IN THE PEOPLE'S REPUBLIC OF CHINA

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Abstract

The criteria for the success of nuclear programs remain unchanged today from those in existence for the last twenty years. The nuclear energy source must fill a <u>'need'</u> - a need first identified in the form of electricity generating capability.

Once this electricity generating need is established, the preferred solution must meet a number of technological and economic criteria. Nuclear technology, when offered for export is assessed for example, by the ability of the host country to pay for the high initial capital cost; and the host country must assess the ability of its grid to absorb a large 'one-time' addition to its power plant inventory. It must assess the ability of the existing infrastructure to cope with the newly introduced technology.

The criteria has not changed but the solutions offered are substantially different. Nuclear products now offer solutions to a variety of needs from large and small electricity units, through space and industrial heat applications to desalination. The nuclear product offering is greater because many of today's nuclear products are simpler and consequently more likely to meet the criteria of need, technological acceptance and economic viability. Public acceptance is also tremendously enhanced.

Atomic Energy of Canada Limited (A.E.C.L.) and the Beijing Institute of Nuclear Engineering have carried out a study of the applicability of A.E.C.L.'s Slowpoke Energy System (S.E.S.) in China. This study reviews the criteria for acceptance of the S.E.S. in China. With examination of specific sites, the study highlights the ability of this new nuclear technology to satisfy a newly identified need when measured against accepted criteria of technology transfer.

## 1. INTRODUCTION

The Agency has stated that the purpose of this meeting is to provide an opportunity to review and discuss factors necessary for selecting the type and nature of an appropriate advanced nuclear plant in developing countries. This is an exceedingly complex target that has to be considered within the context of the energy needs of the developing country, and the sources of supply of energy around the world.

For the first two decades of the nuclear industry, the need that lead to installation of nuclear plants was a need for electricity. In order to be competitive, nuclear plants had to be large. Consequently, only those countries with the very large demands of national grids were able to take advantage of nuclear power. To-day the nuclear industry has broadened its product line and is capable of competing competitively in the supply of electricity, high temperature heat for industrial purposes, low temperature heat for space heating, and in combinations of these needs. The new product line offers nuclear units in a size range that moves from 10 MWt through several hundred MWt up to new smaller electricity generating units that are competitive at 300 MWe.

The Beijing Institute of Nuclear Engineering (BINE) and Atomic Energy of Canada Limited (AECL) have spent a considerable amount of time reviewing the factors that must be considered in the selection of the type and nature of a nuclear plant to be used for the supply of heat in China as we prepared a feasibility study investigating the use of the SLOWPOKE Energy System in China. Mr. Hao and I appreciate this opportunity to share with you what our organizations have learned and the conclusions that we have drawn.

## 2. BASIC FACTORS

First, the demand for energy must be described - will the energy source need to supply electricity, high temperature heat for industrial purposes, low temperature heat for space heating, or some combinations of these needs?

Then a number of likely sources of energy can be determined.

At this point the three basic factors to be considered are unit size, economics and the appropriateness of the technology, factors which are likely to apply to any application of nuclear technology in developing countries

## 2.1 Unit Size

Criteria for determining unit size can be classified under:

- Substitution - replacement of aging, inefficient, and often very polluting plant,

- Growth rate the country's rate of growth of energy demand will impact on the size and rate of additions to the country's energy supply capability, and
- Size of Units available until recently the buyer had a limited size range to choose from.

## 2.2 Economics

Criteria for assessing the relative costs of the variety of options available include:

- 1. Total unit energy cost.
- 2. Initial capital cost of the unit.
- 3. Foreign exchange issues, which are very country specific. Many developing countries prefer to compensate the vendor in the form of goods (barter or countertrade) rather than with payment of convertible currencies.
- 4. Financing options A financing package, when structured to spread the required payments over an extended period of time may permit the buying country to pay for the technology using funds derived from the future savings.

AECL is committed to improving relationships between the developed and developing countries. At a recent nuclear conference in Korea, AECL President J. Donnelly talked of international cooperation and the role of nuclear energy in the economics of the future (1). Specifically he questioned the ability of technology transfers to be effective without sound relationships and economic partnerships. " Our challenge is to devise innovative mechanisms guaranteeing payback for investors in the long term. Be warned that failure to meet this challenge will bring dire consequences in the form of a backlash from the underprivileged nations, struggling with their industrial revolutions without the benefit of abundant clean energy."

As you are aware, AECL has introduced the BOOT model in the company's efforts to introduce CANDU technology to Turkey. The Build, Own, Operate and Transfer idea is a working partnership that provides guidance, eases financial burdens and provides the recipient country with access to evolving technology.

## 2.3 Appropriateness of the Technology

Although the cost of the alternatives is a critical measure of competitiveness, it may not be the major concern. The necessity of making forecasts for the prices of the variety of raw energy sources makes any assessment of cost competitiveness less than exact. If an energy source is the right size and close in cost to the lowest price option, although not the cheapest under the given assumptions, it will likely remain a viable choice subject then to the appropriateness of the nuclear technology being offered.

The criteria to be considered here are:

- Is the technology an appropriate match with the existing skill base?
- Can local industry meet the manufacturing requirements and is the supplier willing to localize much of the manufacturing?
- How much project management experience exists?
- Is the offered technology proven?
- Are their licensing issues? and finally,
- Does the developing nation share the industrialized nations well publicized concern over nuclear energy of any type?

Note that these factors and criteria are applicable to all countries regardless of their state of development. However specific local factors must still be considered in each case. In order to examine these factors in a more meaningful way, the prefeasibility study of the application of the SLOWPOKE Energy System technology in the People's Republic of China is described in detail. Some of the lessons learned in this assessment have general applicability and may offer assistance for the introduction of the newly expanded nuclear product line into developing countries around the world.

## 3. SLOWPOKE ENERGY SYSTEM

First, a brief description of the product being considered for application in the People's Republic of China.

The SLOWPOKE Energy System concept is based upon the 20 kWt research reactor called SLOWPOKE-2 (2).

## 3.1 SES-10 Heating Reactor

The very favourable experience with this reactor has provided the foundation for the development of the SLOWPOKE Energy System, a system ideally suited for supply of a community's local energy requirements at competitive prices.

The SLOWPOKE Energy System has been designed to operate at up to 10 MWt and incorporates the key technical features of the research reactor, namely: atmospheric pressure, natural convection cooling and remote monitoring instead of an on-site operator. The overall design objective is to achieve high reliability and safety at low cost.

The reactor core, coolant riser duct and heat exchangers are installed in a water-filled pool inside a steel-lined concrete casing as shown in Figure 1. The pool water serves as moderator, coolant and heat transport system.

Primary heat transport from the core is by natural circulation of the pool water through plate-type primary heat exchangers. The secondary circuit delivers heat to the building heating system by



Figure 1: Schematic Diagram of the SLOWPOKE Energy System

way of the secondary heat exchanger. Thermal power is measured in the secondary coolant circuit for purposes of metering the heat supplied to the district heating system.

One of the fundamental driving forces in the design is the safety philosophy. The primary goal is to meet all Canadian regulatory requirements in a manner that permits unattended operation for periods of a week or longer. In addition, the basic safety principles should be easily understood by interested people with limited knowledge of nuclear technology.

Since the product from the heating reactor is hot water at 85°C, this nuclear heat source is common to a variety of applications including:

- building heating and hot water,
- low grade industrial process heat, and
- electrical generation.

To meet the requirements of these applications, the commercial SLOWPOKE Energy System comprises:

- the nuclear heat source,
- the heat exchanger interface,
- a heat conversion system, and
- a building and associated civil structures.

## 3.2 The SLOWPOKE Demonstration Reactor

To prove that the design goals for the SLOWPOKE Energy System can in fact be met, a 2 MWt demonstration reactor has been constructed at the Whiteshell Nuclear Research Establishment in Manitoba. It achieved first criticality in July of 1987, within two years of the start of construction. The primary purpose of operating and testing this reactor is to validate the computer models which have been used to simulate the performance of the commercial units. It will demonstrate that the SLOWPOKE Energy System can be operated safely and reliably without an operator in the reactor building. It will provide heat to laboratory buildings, thereby illustrating the SLOWPOKE's ability to provide reliable space heating throughout the year and in the longer term it will be utilized to demonstrate electricity production.

Since the commercial units will be located in populated areas, it is important to gain realistic experience in monitoring and controlling the reactor. Such experience is critical to satisfying client wishes to purchase only proven technology.

## 3.3 Other SLOWPOKE Applications

The potential heat conversion systems include hot water distribution systems for heating communities or building complexes, absorption cooling systems and other thermal processes suited to low temperature heat such as vaporization processes for desalination, or the heat to electricity conversion system.

## 4 THE PEOPLE'S REPUBLIC OF CHINA

## 4.1 Unit Size

One of the first tasks of the feasibility study was to assess the needs of the People's Republic of China with respect to district heating, and in particular to a heat source that comes in a standard 10 MWt size.

The People's Republic of China is a developing country with a population in excess of one billion people. The rapid growth of industry and agriculture and the continuous increase of residential living standards in China requires a huge amount of energy to support the modernization drive. Coal currently plays a very large role in the energy structure, accounting for over 70% of total energy consumption and the share of coal is increasing annually. This trend is not expected to change in the near future.

Of the primary energy consumption in China, the portion used for power generation accounts for only one-fifth, while the portion used for various forms of heating (process heating, building heating, cooking, etc.) accounts for 70 %. Low temperature heat accounts for an important portion of this heating market.

In 1980, it is estimated that one third of the People's Republic of China's raw coal production was burned in the industrial and household sectors. This figure is expected to increase substantially in the future.

China has demands for very large systems in the highly industrialized areas of the country which are experiencing rapid growth in their energy requirements. In these cities, large dedicated heating plants or cogeneration facilities will be appropriate. But China is a vast country with many large cities remote from these highly industrialized centres. The heating needs of these cities could be met with several 10 MW unit SESs.

In conclusion, the demand in China is such that large and small size nuclear heating plants can be utilized effectively.

## 4.2 Economics

The cost of alternative energy sources such as the SES is initially measured using total unit energy cost and initial capital cost. The initial investment (including the capital cost and the fuel cost of the first unit) must be acceptable for the users, noting that lower capital requirements increase the size of the market as more communities are able to purchase a unit and stay within their budget. These costs are then compared to the compatible data for coal-fired boilers.

AECL and BINE believe that when the trend of coal prices is taken into account that the SLOWPOKE Energy System is a cost effective option for the People's Republic of China. In particular, its low cost allows communities to consider the nuclear option while remaining within their spending authority.

By the end of 1984, 23 cities in China had set up district heating systems supplied by either co-generation power plants or local hot water boilers. Many cities are currently making decisions on the heat source to be used in their systems. If small heating reactors can be operated safely and reliably, and can produce excellent economic results, they will be rapidly developed.

The SES also offers the opportunity to lower the foreign exchange requirements in two important ways. First, the product is less expensive as it is being provided in smaller sizes. Secondly, the new reactors are capable of offering increased localization through designs that account for the state of industrial development in the recipient country.

Under the 'ability to be competitive financially' come a myriad of issues and any business plan developed for the introduction of the SLOWPOKE Energy System into China must meet the following requirements:

- the economic and technical risks assumed by each party must be commensurate with their scope of responsibility, and be balanced by the potential economic or other benefits to be gained by each party.
- the benefits to AECL must include an appropriate recognition of the investment made by AECL in the still ongoing research, development and demonstration of the SLOWPOKE Energy System technology.
- the technical risks to China must be minimized, with responsibility assumed primarily by AECL,

- the business arrangement must be financially attractive to AECL over say a 10 year period, and meet corresponding BINE and Chinese goals,
- any transfer of AECL technology through the building of SLOWPOKE Energy Systems in China must be balanced by a long term commitment by China which is beneficial to AECL. AECL has shown a willingness to be innovative in the past, with the BOOT model, and a form of technology transfer is now under active investigation by BINE and AECL.

## 4.3 Appropriateness of the Technology

In reviewing the criteria listed, note that a small simple unit such as the SES is a welcome technology for a developing country. To reiterate the first three are:

- Is the technology an appropriate match with the existing skill base?
- Can local industry meet the manufacturing requirements and is the supplier willing to localize much of the manufacturing?
- How much project management experience exists?

AECL and BINE have concluded that the SES is a very attractive and workable option with regard to these criteria.

Proven technology and the ability to meet licensing requirements are critical criteria. The SLOWPOKE Demonstration reactor is in place to meet this need and the commitment of an urban prototype in Canada will fully satisfy this requirement. In licensing, the appropriate Chinese and Canadian authorities have held initial discussions and no major hurdles have been identified.

In addition to public opinion which is discussed later, what special criteria were identified?

As part of its detailed market survey BINE selected specific cities or areas in the Northern regions for detailed assessment. This region is geographically and meteorology diverse. The cities or regions selected are all industrial. have high population and population density, and all experience prolonged and severe cold weather.

Because the SLOWPOKE Energy System is restricted in the temperature of the heat it supplies to the central heating system, the prefeasibility study had to determine very quickly the ability of the existing Chinese systems to employ 85°C water efficiently. The study showed that at present, the practical operating temperature of small heat networks in the cities of China is about 50-63°C. Generally, the heating temperature supplied by domestic power plants ranges from 50°C to 75°C. Therefore the SLOWPOKE Energy System is appropriate and in fact particularly attractive as low temperature district heating systems are proving increasingly popular and cost effective.

BINE examined a number of communities and concluded that the 10 MWt size of the SLOWPOKE Energy System was uniquely able to meet the demands of the People's Republic of China as the country moves through the 1990's. It was particularly appropriate because it possessed a number of attractive features that enhanced its ability to access the market by solving dilemmas raised by these less frequently seen criteria:

- its small size allows a community to add heat in small increments,
- its size and construction permits flexibility in siting,
- it provides the People's Republic of China with the opportunity to educate and provide the public with confidence in the nuclear option prior to major capital commitments,
- nuclear energy of any kind represents less strain on the transportation system currently overloaded with the transportation of coal, and
- it represents an early opportunity to address the pollution issue.

Pollution is the most significant of the specific issues listed above. In the People's Republic of China, small boilers with capacity of less than 4 ton/hour represent 92 % of the number of boilers; their efficiency is less than 60 %.

At the present time the atmospheric pollution from the particle dust and sulphur dioxide produced directly by coal burning is quite serious in the more industrialized and densely-populated areas of China, especially in the northern regions. Measurements show that particle dust in most of the 50 cities surpasses the national standard Grade II (0.3 mg/cubic meter). The SO2 discharged in one fifth of 56 cities surpasses the national standard Grade II (0.15 mg/cubic meter). The quality of the environment is also related to local meteorological phenomena in which the dispersion of pollutants remains below the lowest altitude of the inversion layer. In China the lowest altitude of inversion layer is in the northern regions. The atmospheric pollution during the heating period in the Northern regions of the country goes far beyond acceptable limits and, with the development of industry and agriculture, is worsening each year.

The Chinese government believes that in order to improve the environmental quality, to raise the thermal efficiency of heating facilities, to save energy resources and to improve the living conditions of inhabitants, centralized heating must be developed. Indeed the 36th paragraph of "The Provisional Regulations on Energy Saving" dictates that "every new house and public building should be put into the uniform plan and heated by centralized heating."

Within the cities visited, the Institute investigated other factors peculiar to the nuclear industry. A key factor for a first installation anywhere is local leadership, be it a developing country or a socalled developed country. Without this factor, smaller systems such as those being developed today are unlikely to get the approval they need.

Another factor in our study that may or may not have universality is the quantity of valuable land lost to the numerous small systems which currently exist.

## 5. CONCLUSIONS

In summary, with the AECL/ BINE feasibility study, it became clear that a wide variety of factors influence the making of decisions.

Although it is essential that any mutually successful business arrangement that AECL may arrange with the People's Republic of China must meet identified needs, must work effectively in the local environment and must be cost effective, numerous other factors are very important. As specific opportunities are investigated, different advanced reactor designs will be found more or less advantageous when examined against these novel criteria.

In the study carried out by AECL and BINE, the small size of the SLOWPOKE Energy System led to a number of advantages that will improve the likelihood of successful application of this technology in the People's Republic of China. With a smaller unit, the decision making process is simplified, and the public is more comfortable with the small unit as decisions can be taken locally and are therefore more under their control.

And finally, the size of the SES has enabled AECL to build and operate a demonstration unit and thereby meet another of the criteria very important to acceptance of new technology in a developing country. AECL and BINE believe that the successful introduction of these units enhances the public acceptance of all nuclear plants.

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## A CONSULTANT'S VIEW OF THE USE OF ADVANCED NUCLEAR REACTORS IN DEVELOPING COUNTRIES

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#### Abstract

This paper outlines some of the differences between the institutional, financial and economic problems of a small nuclear steam plant and those of a large power plant. If there are no other nuclear plants in a country then very careful considerations should be given to the institutional problems. In the light of the experience of MOTOR-COLUMBUS this paper has examined the problems and difficulties of introducing advanced reactor systems such as the HTRs to provide high temperature steam to allow the extraction of oil from heavy crude oil deposits or from tar sands.

Most frequent reason for abandoning nuclear power projects in developing countries over the last decade has been the failure to obtain funds. Certain factors have been indicated in this paper which make the provision of funds easier for a small nuclear power plant as compared to large one.

#### INTRODUCTION

Over the past 15 years or so, there have been many proposals by developing countries to introduce nuclear power for electricity generating purposes, but various problems, notably finance, have prevented success.

MOTOR-COLUMBUS Consulting Engineers has been engaged in the evaluation of many of these proposals and has first hand knowledge of the difficulties that have led to the abandonment or postponement of the schemes.

This paper examines the rather different problem of introducing advanced reactor systems for other purposes in the light of that experience.

Every case has its own difficulties and needs individual analysis. To provide a focus for our attention, we will examine the problems that are likely to arise in proposals to use small High Temperature Gas-cooled Reactors (HTRs) to provide high temperature steam to allow the extraction of oil from heavy crude oil deposits or from tar sands.

The analysis will concentrate on economic, financial and institutional problems. The technical uncertainties that are inevitably associated with a new reactor type are not evaluated.

#### ECONOMICS

In the power field, large nuclear power plants are today more or less competitive with coal fired power plants, depending on the local coal price. They provide cheaper energy than oil, though at today's oil prices the margin is not very great. At smaller sizes, even 600 MW, their competitiveness is less clear. It is claimed that large HTRs, such as the proposed German ABB/HRB 500 MW HTR design, is competitive with large Light Water Reactors.

Small modular HTRs, of which at least three designs exist on paper, are of a substantially different design from this larger plant, so scaling laws are not a good tool for deriving a price. Price indications that have been received when a real opportunity for construction appears to be possible have not been very competitive.

The relative prices of oil fired and HTR plants for power generation should give a good indication of the relative prices for steam generation, since the turbine designs for the two plants are quite similar. If anything, an oil fired plant for steam production for oil extraction should have improved competitiveness since most of the oil handling and storage facilities will already exist.



The dependence of the economics on oil prices are illustrated in the figure. The product selling price, line A, is equal to the world market price, and if product is used for generating steam it should be costed at this value. The production cost of oil excluding the cost of steam will be independent of the oil selling price and is shown by line B. If steam for extraction is raised using oil, the total product cost will be given by a line of constant slope such as C. Production can, therefore, only be justified when the oil price rises above that given by point 2.

If an HTR can give low cost steam the total production cost will be given by line D. In this case, there can never be a situation where oil generated steam is economic. On the other hand, if HTR steam is more expensive, as shown in line D, there is a range of oil prices from points 2 to 3 where it is more economic to use oil to generate the steam than an HTR.

A little thought will show that the oil price at which the HTR becomes more competitive than oil for steam raising will not be very different from the oil price at which it becomes competitive for electricity generation, since the turbine designs in each plant type are quite similar. A recent modular HTR price indication suggests that the break even oil price could be as high as 30 - 40 \$/brl.

#### FINANCE

The provision of finance has been the biggest problem for recent nuclear power projects. Various schemes, such as Build, Operate, Transfer (BOT), are being examined to ease this problem but the basic difficulty remains unchanged, namely that the product, electricity and/or steam, is sold internally for local currency, but the repayment of loans and payment of interest has to be guaranteed in foreign currency.

This particular difficulty is at least partially overcome in the case of steam production for oil extraction. The product is saleable abroad and it will, therefore, be easier to show that foreign currency will be available to service the loans. Other favourable factors are that the nuclear equipment will only be a small proportion of the total finance for the project, and that there is a likelihood that an international oil company with good sources of finance can be associated with the project.

Some aspects of BOT are perhaps worth noting. The family of inter-related agreements for BOT is complex, involving many parties and is potentially an in-line program item. The agreements must deal with the problems of guarantees that the debt will be serviced over a relatively long period of 20 years. Some BOT discussions have broken down over this issue when suppliers have sought host government guarantees. The strong finance related point in favour of BOT is that the plant would be built, operated, and maintained by well experienced groups from highly industrialized countries. This aspect greatly enhances the probability of raising the necessary loan financing on good interest terms. Another issue needing analysis for each particular country is the question of safety responsibility and the legal status of the operator/owner group.

#### INSTITUTIONAL FACTORS

The construction and operation of a nuclear plant of any type brings with it safety problems which do not have an equivalent in the case of oil fired steam plant. This means that the country must either have a competent regulatory body already in existence, or must set one up. In the latter case, the cost implications, if only a single installation is to be supervised, will be significant.

The staff for a nuclear steam raising plant will also have to be more highly qualified than that for an oil fired plant, despite the inherently good safety characteristics of an HTR type nuclear plant. Licensing, the supervision of plant safety, the storage/disposal of irradiated fuel and decommissioning at the end of plant life (or after depletion of the oil reserves) will require a degree of management attention much greater than if an oil fired plant were chosen. The small size of the plant makes these factors relatively more important than they would be in the case of a large power plant. Initially, for 0 to 5 years, it is recommended that an overseas nuclear utility is responsible for plant operation in countries without nuclear power plant experience.

#### CONCLUSIONS

This paper has only been able to outline some of the differences between the institutional and economic problems of a small nuclear steam raising plant and those of a large power plant. A detailed study of each individual case is necessary to determine whether a nuclear plant is the best choice. The institutional problems will need very careful considerations if there are no other nuclear plants in a country.

On the other hand, there appear to be factors with respect to a small NPP that can make the provision of finance easier than for a large nuclear power plant. This is very important, as failure to obtain finance has been the most frequent reason for abandoning nuclear power projects in developing countries over the last decade.

The competitiveness of nuclear steam supply in small sizes remains the largest uncertainty, and this cannot be resolved until a similar plant has been built under commercial conditions. Even then, the uncertainties that surround the future movements of oil prices make the use of the HTR for oil extraction (and indeed the whole economics of the extraction of heavy oil deposits) a matter of opinion.

### **REALIZATION OF THE 'INHERENT SAFETY' PRINCIPLES** IN THE AST-500 AND AST-300 REACTOR DESIGNS

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#### Abstract

The principal decisions ensuring the AST-500 and AST-300 reactor plants safety due to iherent safety features, based on gravety and thermodynamics laws with construction the safety systems in terms of passive means application without any energy sources and mechanisms are considered.

#### INTRODUCTION

About ten years ago the design of the AST-500 reactor for district heating nuclear plant (DHNPP) was worked out in the USSR.

Development of the designs was preceded by the investigations directed on drawing up the safety requirements with allowance for necessity to approach the nuclear power source for economical reasons, to the city.

As a result, the following requirements for construction and siting of the DHNPP have been drawn up in addition to those accepted in the practice of the NPP construction and operation /1/;

measures preventing melting of the fuel elements in depressurization of the reactor vessel shoud be provided;

it is nessary to take into account the external impacts (aircraft impacts, the shock wave from the near explosions);

measures on limiting the storage period for the irradiated fuel and the radioctive wastes should be taken;

the collective dose of irradiation the city population under both normal operation and emergency conditions are to be limited.

REALIZATION OF THE "INHERENT SAFETY" PRINCIPLES IN THE AST-500 AND AST-300 REACTOR DESIGNS

Determination of the experience in development and operation of the nuclear power plants in the USSR and abroad shows that vessel-type water-water reactors are the most qualify as a heat sours for DHNPP nowadays.

The vessel-type water-water reactor is one of the most studied both with point of view the termal physics and equipment production technology. However the necessity to satisfy the additional requirements demanded to revise the traditional decisions. The main design goal was to use maximum the inherent safety principles based on the gravity and thermodynamics laws with construction safety systems in terms of passive means application without any energy sources and mechanisms.

One of the most total criterion how managed to use above principles in the design can be the evaluation of the self-regulation degree under normal operation and emergency conditions.

A high degree of the self-regulation in the AST reactors is achieved by follows:

The reactor core and the circulation circuit are designed in such way that negative temperature, void and power effects of reactivity are ensured, that in combination with the boric adjusting absens and using the burning up absorber ensures the self-regulation of the chain reaction and guarantees the nuclear safety.

Using the natural circulation of the main circuit coolant both under normal operation and emergency conditions in combination with the integrated arrangement of the main circuit equipment still more promotes the self-regulation degree raising.

Absence in the main circuit any pumps expels the accidents in connection with their damages or the DHNPP power loss and promotes more "soft" the dynamic process running.

The "rigit" heat coupling between the coolent flow and power makes it possible to bear considerable concentrated power increase in the core assembles becous of the flow self-profiling in the assebles and using the individual draught tubes.

Heat discharge from the reactor to the heat-supply system water is done via three autonomous loops of the intermediate circuit Such three-loop system assures also decay heat removel from the shut-down reactor in all emergency conditions, connected with the heat removal system failure with keeping the integrity of at least one of the three loops.

In normal reactor operation conditions water circulation in the secondary circuit loops is reacht by the use of circulation pumps. However for the emergency reactor cooling in case of the DHNPP power loss, the pumps damages or accidents caused by external effects, three indepedent emergency cooling systems are used at each reactor

Their number equals the number of the secondary circuit loops. The capacity of each system assures reliable emergency cooling of the reactor. Thus the decay heat removal from the reactor core is assured by the naturel water circulation in all circuits.

Peculiarity of the reactor AST is large water guantity presence in the reactor vessel and therefore high heat storage of the main circuit, that defines thermal lag of the emergency proceses in connection with the deterioration or absolute heat removal loss from the reactor. For example in case of the complete refusal of all reactor heat removal systems only at the expense of heat storage in the main and intermediate circuits the time of the upper limit pressure achievment in the main circuit is about two hours.

The rigit heat coupling between the main and intermediate circuits makes it possible to protect the reactor against the overpressurization without mounting the safe valves in primary circuit, that excludes the possibility of the radioactive coolant

release when the safety valve fails to sit. It also exludes the possibility of unwatering the reactor core.

Details of the AST-500 reactor construction is the presence of the second (safety) vessel, designed for the pressure arising from the basic vessel depressurization. The second vessel presence ensures both the core covered fully with water under depressurization of the basic vessel and localization of the radioactive products.

The basic treatments to ensure the reactor AST safety are:

the natural circulation of the main circuit coolant under normal operation and emergency conditions;

the integrated arrangement of the main circuit eguipment; Two-vessel design, when the basic vessel is mounted into the safety vessel with a minimum clearance;

three-circuit scheme of heat removel from the reactor with the intermediate separating circuit, wherein pressure is lower than in the heating supply system;

three-loop scheme of the heat removal system;

the double fast-responce stop valves placed inside the safetyvessel and operating in case of the depressurization of the tubes are installed on the pipelines of the systems connected with the primary circuit;

the fast-response stop values cutting off one of three loops of the heat removal systems in the emergency depressurization of the heat exchanger are installed on the pipelines of the secondary circuit;

the safety cooling system consists of three indepedent loops and is arranged with the use of the principle of the natural circulation through three circuit heat removed through the line heat exchanger at the cost of evaporation of the coolant accumulated in the tanks of the safety cooling system;

the relible protection of the reactor against the overpressurization above the permissible pressure is ensured without mounting the safety values in the primary circuit.

# SUPPLEMENTARY SAFETY ANALYSIS OF THE AST-500 REACTOR PLANT

In the design of the AST-500 reactor plant all accidents, regulated by the national rules have been considered. However in connection with the Chernobyl accident the supplementary safety analysis for the hypothetical accidents have been carried out.

The supplementary safety analysis includs:

accidents caused by deterioration or absolute loss of the heat removal from the reactor to the district heating grid under postulated failures of the safety systems;

accidents with the reactivity caused by the not sanction control rod ejection or cooling the main circuit coolant and without fuction the emergency protection systems;

accidents caused by depressurization of the main circuit with additional failures of the safety systems and the absolute heat removel loss from the reactor;

accidents caused by collection mistakes of the attending staff.

The supplementary analysis above hypothetical accidents confirmed high degree of the reactor AST-500 inherent safety. The analysis results had shown that the attending staff has enough time (from 12 hours to several days the reactor exists in safety state without the attending staff enterference) to recover the protecting systems servisibility. However to soften the process cource of the hypothetical accidents the additional measures have been worked out, wich being carried out at the DHNPP in Gorky at present.
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	Parameters	Dimension	Numerical	value
	na na mana mana mana mana mana mana man	anangenilitaan na miiga liyooza kita kazan maran katanista maringi	AST-500	AST-300
1.	Reactor thermal power	MW	500	300
2.	Parameters of the primary circuit coolant:			
	pressure	MPa	2.0	2.0
	inlet/outlet temperature	oĸ	399/477	393/473
3.	Heat exchange surface/number of sections	™ <sub>S</sub> ∖	5000/18	2000/15
4.	Parameters of the intermediate circuit;			
	pressure	MPa	1.2	1.2
5.	Parameters of the heat grid:			
	pressure	MPa	2.0	2.0
	supply header/return header temperature	°ĸ	<b>4</b> 1 <b>3/3</b> 33	393/323
6.	Core parameters:			
	specific power density	mw/m <sup>3</sup>	27.1	23
	fuel element diameter/type of fuel	mm/	13.6/U0 <sub>2</sub>	13.6/UO <sub>2</sub>
	number of fuel assemblies		121	85

TABLE 1. MAIN TECHNICAL DATA OF THE AST-500 AND AST-300 REACTORS

The main technical data of the AST-500 and AST-300 reactors are given in Table 1, the structural scheme and schematic diagram in Fig. 1 and Fig. 2.

The hypothetical accident process course with the absolute heat removal loss (Fig. 3); and the heat removal loss with simultaneous failure of the emergency safety systems (Fig, 4) are shown in Fig. 3 and Fig. 4.

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FIG.1. 1 — core; 2 — heat exchanger of the primary and secondary circuits; 3 and 4 — lower and upper parts of the reactor vessel; 5 — reactor cover; 6 — drives of the control and protection system; 7 and 8 — lower and upper parts of the safety vessel.



FIG.2. Principle diagram of the NHP heat supply system.



FIG.3. The hypothetical process course with the absolute heat removal loss from the reactor.



FIG.4. Loss of the heat removal into the heat grid with simultaneous failure of the control and emergency safety system.

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# TECHNICAL COMMITTEE MEETING AND WORKSHOP ON DESIGN REQUIREMENTS FOR THE APPLICATION OF ADVANCED CONCEPTS IN DEVELOPING COUNTRIES

VIENNA, 6-9 DECEMBER 1988

# SUMMARY REPORT

# I. TECHNICAL COMMITTEE

### Session 1 - Incentives for Advanced Nuclear Power Technology (Chairman: Vrillon)

Four papers, two from China, one from France and Mexico, were presented during this session.

The Chinese presentation by Mr. Xu Mi stated that the total generating capacity in China amounted to about 100 GWe in 1987. The known U resources are adequate for 30 years operation of some 15 GWe of thermal reactors, so such reactors can not contribute significantly to the long term energy need, based on indigenous resources. Hence, breeder reactors are of interest, this paper giving an overview of the fast reactor development programme in China. The beginning of this programme took place in the late 1960s, with the construction of several test facilities (sodium loops for technological studies) and the development of computer codes for FBR core design. An experimental fast reactor of about 25 MWe is planned for the turn of the century. The first demonstration FBR is envisaged in the period between 2015 to 2020, based on a modular type reactor (100-150 MWe).

The presentation by Mr. Zhang stated that China is now preparing the design and construction of a classical 600 MWe PWR based on the Qinshan design. After several reactors of this type have been built, a series of 600 MWe advanced PWR (APWR) might be constructed via international co-operation. The objectives of the Chinese APWR are very ambitious, but they give a good idea of how to satisfy future needs.

Mr. Zacueta of Mexico's paper presented the situation of the current and future (2010) electricity demand in Mexico. The Comisión Federal del Electricidad (CFE) envisages a nuclear programme based on the BWR technology constructed at Laguna Verde, using the concept of standardization to BWR. This approach is considered to be advantageous, as it will utilize the established engineering knowhow and industrial infrastructure. Improvements in the design will only be made when worthwhile.

According to the paper by Mr. Boher of France the success of the French Nuclear Power Programme is, amongst other things, based on standardization and a clear and simple organization between the utility (EdF), the constructor (Framatome) and the safety authorities, with the research and development support of CEA. The backfitting information received from operating plants is incorporated in the new designs. Some examples of the application of French advanced technologies, which provide enhanced safety and are based on the French safety approach, were given. In this context, it can be noted that initiating events beyond current DBE spectrum are considered in the safety analysis. For such events the analysis takes into account also non-safety-grade equipment. All these advanced technologies can be integrated in an advanced reactor able to fulfill the needs of the Developing Countries in the 1990's. A short presentation of the Turkish situation was given (no paper) by Mr. Oezmen of Turkey. Nuclear projects have been negotiated for 18 years, but due to financial problems no plant has materialized. Despite financing restraints Turkey has to go nuclear within 10 years, aiming at 4 GWe by 2005.

# 2. Session 2 - Design Requirements (Chairman: Hewitt)

Three papers from Indonesia, Sweden and Switzerland were presented in this session.

The first two papers together include requirements in terms of how they may be met by advanced PWRs, PIUS reactor types and high temperature gas-cooled reactors. The third paper provides an overview of the technical factors of the broad range of advanced reactor concepts available from industrialized countries. All three papers are attentive to the special circumstances of the Developing Countries.

A basic premise of the paper of Mr. Pedersen is that, because nuclear power technology may be identified as a unique resource which is both technically viable and ultimately necessary worldwide, special care must be exercised in all future implementations in order that increased societal opposition precipitated by severe reactor accidents will not deny future generations the cremendous environmental benefit of this technology. The global nature of both the potential benefits and the potential consequences of nuclear power would render meaningless any variance in the design requirements with regard to place, time and jurisdiction. Therefore, apart from relatively minor supplementary and user-oriented variations, the requirements applied to all nuclear power plants should be equally stringent whether applied in industrialized or developing countries. This paper proceeds to trace the shifting general licensing environment and societal attitudes with respect to nuclear power, following the significant events at TM-1 and Chernoby1. The difficulties experienced in adapting traditional LWR designs to the changing requirements are discussed. The BWR-90 and the PIUS designs are used as examples in the discussion of how the new requirements may be most effective (see also session 4, paper 5).

The high temperature gas-cooled reactor was described by Mr. Sarlos in the context of its applicability in developing countries due to its economic competitiveness and safety related advantages. Specific characteristics cited include generous operating safety margins, high fuel burnup, high fission product retention throughout the fuel cycle, conventionality of balance-of-plant, large locally manageable components in construction, and minimal logistics of fuel transportation (e.g. compared with coal). These characteristics may be considered as fulfilling many of the requirements that might be specified for a developing country.

The HTR-500, already the product of international co-operation suitable for matching to a modest-sized local grid, is portrayed as an example system for purposes of this paper.

The paper by Mr. Nurdin of Indonesia showed the overview of advanced reactors available in sizes suitable for developing countries included ALWRs, PIUS (and ISER) concepts, Candu-3, and HTRs of various origins. The receiving country's infrastructure, including professional and technical manpower with the appropriate skills and availability to support the activities during construction, operation and maintenance, were discussed as keys to the success of the nuclear project. Also within the developing country, main requirements for the introduction of advanced reactor concepts are identified as:

- 1. plan for the year 2000, and before
- 2. low risk (technical, economic and operational)
- 3. competitiveness with clean coal technology (i.e. world-wide acceptance, flexibility and low construction and operation cost).

The prospect of integrating suitable plants into the receiving country's technical and economic infrastructure, citing the particular case of Indonesia, was discussed.

# 3. Session 3 - Specific Applications (Chairman: Holm)

Three papers from Canada, China and Venezuela were presented at this session.

According to Mr. Carvajal the Venezuela has the largest reserves of extra-heavy oil in the world  $(1.2 \times 10^{12} \text{ barrels})$ . For the exploitation of this crude oil from the Orinoco Belt an enhanced oil recovery concept is proposed, through injection of steam, generated by an advanced nuclear plant. This plant would also be able to provide high grade heat for upgrading the oil. The concept presented is a modular system able to supply most of the energy required for the production of 100,000 barrels per day of refinery ready synthetic oil. Three 1200 MW(th) high-temperature gas-cooled reactors, built successively, would supply most of the injection and process steam as well as the electricity required. As the field gets depleted, unused portions of the steam would be used for electricity production.

The Chinese presentation by Mr. Zhong stated that although China is a country rich in energy resources, especially coal and hydropower, the present energy supply still cannot meet the demands of the rapid economic growth. In particular, energy demand in the Eastern part of China and the Guangdong Province in the South, where industrialization is proceeding at a brisk pace, suffers from power shortages. With the energy resources located far away, long-distance transportation becomes a serious problem. Today, coal transportation takes about 50% of the railway capacity.

To increase the energy supply, the use of nuclear power is under development and the PWR will be the basic reactor type in the near future. For the next century an advanced reactor concept, which should be adequately safe, economically viable and have highly efficient fuel, will be selected. The high-temperature gas-cooled reactor (HTR) is one of these advanced concepts, receiving support from the Government for R&D work.

The Application of the HTR in China is foreseen in two stages:

- In the first stage the HTR is intended to be used for the generation of process steam for heavy oil recovery, and in co-generation to provide process steam and electricity for petrochemical complexes and highly concentrated industrial areas. Also, small HTR power plants, of the order of 300 MWe, would be considered in certain areas.
- In the second stage of application, HTR reactors could provide high temperature process heat (up to 950° C) for coal gasification and liquefaction as well as for shale oil production.

Two HTR application studies are in progress: heavy oil recovery and petrochemical industry applications. HTR R&D work is being performed at relevant Chinese institutes and collaborated with institutions in the Federal Republic of Germany, the USA and Japan. According to Mr. Hillborn, to meet the expected growing need for nuclear heating, Atomic Energy of Canada Ltd. (AECL) has designed a 10 MW nuclear heating plant, SES-10, for large building complexes and small district heating networks. The plant, producing hot water at temperatures below  $100^{\circ}$  C, incorporates a small pool-type reactor based on the Slowpoke research reactor. A 2 MW(th) demonstration reactor is now being tested at the Whiteshell Nuclear Research Establishment. The plant is designed with a safety philosophy that should permit unattended operation with remote monitoring for periods of weeks or longer.

With estimated capital costs in the range of 5 to 7 million US dollars unit energy costs could be as low as 0,02 dollars/kwh, for a unit operating at 50 percent load factor for a 25-year period. Competition with oil-fired boilers is envisaged at oil prices exceeding 25 \$/barrel.

For the future, a possible application may be the production of heat and electricity for remote and isolated communities, provided a more efficient Rankine engine can be developed.

### 4. Session 4 - Advanced Reactor Design (Chairman: Kim)

The Canadian presentation by Mr. Hart stated that the next generation CANDU-3 and its size (450 MWe) was developed from extensive market studies. A number of the design requirements were developed in order to represent a new generation CANDU whilst utilizing proven systems and components, such as a 100-year operating life, 3-year operation between outages and a 30 month construction schedule. Further the design aimed at a reduction in the man-rem per year (i.e.40) and should be energy competitive with coal-fired stations at a coal price of 40 US\$ per tonne. Integrated design engineering techniques were used to achieve the design requirements. The CANDU-3 has an overnight cost of less than 1500 US\$ per kWe and should be an economical plant, specially for those countries with limited financial resources and small grid sizes.

The modular high temperature reactor for various applications was presented by Mr. Leuchs. The main features, leading to the high passive safety level, are its modular design and the use of the spherical graphite fuel elements. The modular design ensures a maximum fuel temperature below  $1600^{\circ}$ C in any accident and sufficient core cooling by passive heat transport mechanisms even in case of loss of coolant. The reactivity effect of withdrawn control rods is worse than that of water ingress. The reactor can be coupled to a steam generator, steam reformer or heat exchanger with only minor modifications. In the base version (for steam generator) the outlet gas temperature is about 700°C, whereas for the steam reformer or heat exchanger version the outlet gas temperature is about 950°C. The latter version will have a somewhat lower thermal power level. The thermal output of the base version is about 200 MW.

In the next century severe problems are likely to develop in the world in the following areas like environmental impacts from fossil fuel systems and radioactive waste and nuclear proliferation (terrorism) troubles, mostly coming from Trans-U elements in the U-Pu solid-fuel reactors, said Mr. Furukawa.

In order for civilian utilities to obtain more rational power-stations, not only should these problems be solved, but also problems related to resources, technological safety and simplicity, flexibility in size and siting and integral economy in fuel cycle. A new strategy "THORIMS-NES" [Thorium Molten-Salt Nuclear Energy Synergetics] is proposed and exists of three technologies: [I] Thorium, [II] Molten-Salt and [III] separation of power-stations and fissile breeding plants, refusing fission breeders. The typical 350 MWth Small Molten-Salt Power Station [FUJI-II] has significant characteristics as follows: (i) no need of graphite exchange and continuous chemical processing except fission-gas removal, (ii) fuel self-sustainability (no fissile supply), which means very low excess-reactivity and little fuel transport, and (iii) no core melt-down, easy operation and maintenance, little radioactive waste and thermal pollution, high thermal efficiency and medium-temperature heat supply, etc. This FUJI could be developed during this century at low cost and be integrated with present energy technologies (including the liquid Na reactor technology).

The modular high temperature gas-cooled reactor (MHTGR) developed by General Atomic was presented by Mr. Holm. This modular reactor differs from the one described under 4.2, mainly through the use of prismatic fuel blocks containing coated particle fuel and the somewhat higher thermal output (350MW). A US study of a four unit MHTGR plant supplying steam to one turbine plant, with an electric output of 538 MWe, shows a cost of 1150 million US dollars, and a construction schedule of about four years in a developing country.

Mr. Pedersen presented the advanced concepts developed by ABB-ATOM (included in paper 1 of session 2). On one hand, the BWR-90 concept based on the previous BWR-75 line and on the other, the PIUS/SECURE concept based on the traditional PWR technology.

The new concepts aim both at a high level of safety by application of rigorous safety design criteria, as well as at reasonable costs, flexible and reliable operation, and short construction schedules. The BWR-90 includes features to cope with a molten core accident, in accordance with design criteria in Finland and Sweden, whereas the PIUS/SECURE concept comprises built-in process features which provide safety against severe accidents - core degradation (e.g. self-protective, passive reactor shutdown, and one week inventory of water for cooling the reactor core).

### 5. Session 5 - Other Aspects (Chairman: Hewitt)

This session contained two (miscellaneous) papers from Romania and Canada on disparate subjects. The first paper develops a new variation among the lightly enriched once-through fuel cycles available to owners of current and advanced CANDU power plants.

The second describes the design and development programme of a new small-sized nuclear electric plant (AMPS) meeting various requirements related to its application in a submarine, but having potential applicability also in special land-based applications.

Mr. Lungu of Romania talked about the detailed study which is based on the concept of modifying the composition of the 37-element CANDU fuel bundle so that the currently specified natural uranium elements are replaced by 18 thorium elements forming the outer ring, and 19 enriched (4%) uranium elements in the remaining sites. A reference scenario is cited in which the Th/LEU cycle provides a long term prospect for a near breeder fuel cycle. This cycle exhibits a much smaller reprocessing requirement in the near term than would be required by the alternative natural uranium and slightly enriched uranium cycles, relative to the case of the transition to the final self-sufficient equilibrium thorium cycle. The paper by Mr. Hewitt stated that the Autonomous Marine Power Source (AMPS) was conceived in order to provide truly air-independent power in the marine environment through the application of advanced reactor concepts within small units requiring minimal operator attention. The AMPS is based primarily on proven technologies. However, certain new features, relating primarily to assured passive auxiliary cooling modes in the unstable marine environment, are undergoing early experimental proof-of-concept verification. The AMPS and its new technological developments may be adapted to special small-scale land-base applications.

### II. WORKSHOP

1. WS Session 1 - Objectives, Goals and Requirements Related to Design of Advanced Concepts and Their Application in Developing Countries (Chairman, Pedersen)

This subject involves a variety of aspects, on the one hand, the impacts on design requirements due to recent licensing developments, and the effects on licensing requirements due to design developments, and on the other hand, the owner/user points of view (specific applications, unit sizes, etc.), and public perception of safety and acceptability. The interaction between licensing and design is obviously of importance, but very difficult to foresee, and therefore, the subject was addressed from the owner/user point of view, as this approach must take into account all these items to some extent.

From a user/owner point of view the general requirements on a nuclear plant project can be summarized as follows: it must be economic and affordable, and it must have a high safety level.

The first category can be expanded by the following exemplifications and clarifications:

- . <u>economic</u> implies competition with alternative large scale energy sources
- environmentally benign less environmental impact than alternative energy sources and reduced pollution (cf. the greenhouse effect problems)
- <u>availability of small and medium size plant units</u> taken into consideration limited grid size, as well as need for keeping down first plant investment
- willingness of vendors to transfer technology (plant design knowhow) for active local participation in a plant project (for education of the user/owner personnel)
- possibility of domestic supply of components and equipment taken into consideration the local industrial infrastructure
- shop fabrication and modular design possibly but not necessarily of interest
- plant design based on proven technology and existing infrastructure e.g. LWR fuel infrastructure
- <u>simplification of plant design</u> for easier understanding and ease of operation, and to reduce investment cost

- . flexibility of operation power control, load following, etc.
- design for good maintainability good accessibility, good communication routes, etc.) and good operational reliability - reserve capacity in important systems - to ensure high functional availability
- . <u>design measures aiming at low occupational radiation exposure of</u> personnel
- high level of protection of invested capital low risk of major damages

The second category, safety-related requirements, can be expanded and clarified in a similar way:

- . <u>defined basis for safety requirements</u> e.g. the USNRC 10CFR50 with Appendices and associated CFRs in the US, the Canadian safety approach, etc.
- . <u>traditional spectrum of Design Basis Events (DBE)</u> Anticipated Operational Occurrences and Design Basis Accidents (DBA), in correspondence with current practice for LWR plants
- consideration also of events beyond the traditional DBE spectrum severe accidents, combination of low frequency events, failure to scram in DBA situations - analyzed on the basis of best estimate/engineering judgement, including possible beneficial effects of non-safety-grade equipment
- low releases of radioactivity to the environment in normal operation, as well as in accident situations
- . <u>"defence in depth" principle</u> with a shift in emphasis from mitigation to prevention and protection
- . <u>design for long lead times</u> before human action is needed following an accident
- <u>minimized reliance on operator action</u> as well as equipment performance, following an accident
- . limited releases of radioactivity to the environment for a protracted period of time, following an accident, so that emergency evacuation planning will not be needed.
- 2. WS Session 2 Developing Countries Preparedness and the IAEA's Role with Regard to Advanced Nuclear Technologies Applications (Chairman: Carvajal-Osorio)

The important role of the Agency in assisting Developing Countries (DC) to utilize nuclear power, with emphasis on future applications of advanced concepts, was recognized. At the same time it was recommended that the Agency's support should not just continue but even increase. Specific suggestions aimed to help the Agency to meet these goals were presented in conclusion of this workshop session.

The importance of obtaining more information was stressed, firstly, by the Agency, to identify the countries having specific needs which are likely to be satisfied by Advanced Nuclear Power Technologies (ANPT), secondly, by the tentative users, to have at hand for consulting a compendium of latest designs and the status of ANPT. For this purpose a questionnaire could be developed and distributed. Assurance should be given that the questionnaires are sent to the right people in order to get a realistic reply. The reports on Small and Medium Power Reactors should be updated. Consultant meetings could be arranged for assisting the Agency in these activities. Also other organizations which may have related information could provide it to the Agency.

It was made clear that caution has to be exerted with respect to the word "advanced" since there is no distinct separation between the so-called advanced concepts and the latest reactor designs based on current technologies.

There was some concern that the Agency promotion capabilities might be somehow impaired, due to its regulatory functions. However, it was clarified that, during the last few years, the Agency has been executing an important promotional campaign in favour of nuclear energy applications. It was proposed to send the Director General of the Agency a declaration of support for such activity, from the technical point of view, and in favour of the ANPT.

Efforts should concentrate on supporting those advanced concepts which offer higher levels of safety together with higher grades of simplification, as well as those employing proliferation-resistant fuel cycles and easier handling of radioactive wastes.

Agency activities related to environmental studies should be continued, since these influence positively nuclear energy development and its acceptability, especially in the future when environmental considerations will play a more important role in the energy issue.

Potential users of ANPT should be well aware of all the assistance programmes existing at the Agency. These, at least, should help Developing Countries carry out the first steps towards a rational use of nuclear energy, such as training, technical assistance programmes and co-ordinated research projects supported by the Agency. Industrialized countries could also provide direct assistance even before concrete negotiations are made for a specific application.

Developing Countries should also take immediate action toward public education, promotion of peaceful applications of nuclear energy and the establishment of necessary regulations. In the intermediate range, environmental issues and the energy-developing relation should form part of national programmes.

Those participating countries, which are currently analysing specific nuclear applications, are encouraged to establish related research coordinated projects.

Finally, attention is called to the IAEA's publications which reflect the work done by the Agency in relation to the topic (see Bibliography).

### 3. WS Session 3 - Future Energy Requirements (Chairman: Hart)

The delegates spoke in turn, making a broad spectrum of points relative to the energy demands of the future. At the request of the chairman, the focus covered the period up to 2010. The key points presented are summarized below:

Design

The nuclear power plants of the future must be simpler, and offer both reduced capital cost and reduced construction schedule. In the time frame considered the design must be evolutionary rather than revolutionary.

### Trends

There is a worldwide trend towards industry consolidation. There are increased numbers of joint ventures, collaborative efforts, and comprehensive technology transfer offerings.

There is a need for more collaboration between Developing Countries in order to make nuclear power manageable.

#### Nuclear Power Programme

Purchasing countries must consider a nuclear programme and should not make an evaluation on a single unit basis. The programme must cover all aspects of the nuclear plant including licensing, industrial infrastructure and the fuel cycle. It is important to integrate the nuclear programme with the countries' industrialization programme; in this way nuclear power can be a mechanism for acquiring and enhancing technology with a wide industrial application.

### • Demand

Energy demand in the period considered will be dominated by electricity requirements. Various process heat demands will develop, particularly for high temperature process heat in the period beyond 2000.

Data was presented that indicated a 100% increase in energy consumption over the next 30 years, if the current 2.3%/year growth is maintained. It was noted that the world is now energy poor, with development in as much as 75% of the world limited by power supply. It is therefore necessary to move quickly in placing nuclear power plants in operation.

#### Safety

The current focus on safety was expected to continue, leading to new licensing requirements. This was projected to include features that would preclude a need for evacuation for any postulated event.

### • Environmental Concerns

Many countries that have been immune to public pressures and concerns regarding nuclear power plant environmental/radiation release issues are increasingly required to respond to public challenges.

There is also a worldwide concern over the environmental impact from coal fired power plants, including acid rain and the greenhouse effect

The environmental advantages offered by nuclear power, when clearly presented, should lead towards basic acceptance.

# . Economics and Financing

The need for competitive economics during the early years of nuclear plant operation was emphasized. Also required are new and innovative financing arrangements; the BOOT (Build Own Operate Transfer) model is an example.

### FAST REACTOR TECHNOLOGY DEVELOPMENT IN CHINA

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### Abstract

It is important for all the vast countries in the world to consider the strategy of various energy resources, as is also the case for China.

The Government has decided to adopt nuclear energy as the supplementary or replacement energy for the regions lacking in general energy resources. Qin Shan 300 MWe PWR and Da Ya Bay 2 x 900 MWe PWR are under construction. They mark the beginning of the nuclear power programme of China.

However, there are not enough uranium resources in China according to recent exploration. As we know the uranium resources in China that could be used for civil applications would be sufficient to provide fuel only for about 15 GWe for 30 years in the case of thermal reactors operating on the one-through cycle (based on U-235). That is the reason why China is interested in the development of fast breeder reactors.

Since the late 1960s, a small group of people has begun the basic research work on FBR in China with only a small amount of funds and material resources. The emphasis of research work was put on the core neutronic physics, thermo-hydraulics, sodium purification and impurity analysis, compatibility of materials with sodium and some sodium facilities on a small scale. So far more than ten miniature sodium loops and facilities have been set up.

As the first step of FBR technology development in China, we plan to construct an experimental fast reactor of 25-50 MWe around 2000 to get experience in FBR design construction and operation. The main technical options and some initial design boundary conditions have been preliminarily decided for this reactor.

1. Introduction

It is important for all the vast countries in the world to consider the strategy of various energy resources, as is also the case for China.

There is plenty of coal, oil and hydro-power in China, but it is not enough with respect to the per capita energy resources; Furthermore, the coal is concentrated in the North, especially in Inner Mongolia Autonomous Region and Shan Xi Province. Hydro-power is mainly available in the southwest of China. In order to avoid the heavy transportation of coal from north to the south and to avoid the expensive transfer of electricity from the west to the east, the Government has decided to adopt nuclear energy as the supplementary or replacement energy for the regions lacking general energy resources. Qin Shan 300 MWe PWR and Da Ya Bay 2 x 900 MWe PWR are under construction. They mark the beginning of the nuclear power programme of the Country.

However, there are not enough uranium resources in China according to the recent exploration. As we know the uranium resources in China that could be used for civil applications would be sufficient to provide fuel only for about 15 GWe for 30 years in the case of thermal reactors operating on the one-through cycle (based on U-235). In other words, the economically exploitable uranium reserves would be only 100,000 tons. Of course, in the future some new mineral reserves may be found. However, an energy programme cannot be planned on the basis of assumed resources.

In 1980, the total capacity of electricity generation in China was about 60.5 GWe. In 1987 it was about 100 GWe. At the Seventh Congress on 25 March 1988 Premier Li Peng said China will set up 9 GWe capacity of electricity generation per year during the coming 5 years. Therefore if China develops only thermal reactors based on U-235, the contribution of nuclear power to the electricity generation will be insignificant. That is the reason why China is interested in the development of fast breeder reactors.

The FBR technology development in China is still at a basic and preliminary stage. This situation is due to the late development of thermal reactors for nuclear power and limited funds offered by the Government.

### 2. The History of FBR Technology Development in China

Since the late 1960s, a small group of people has begun the basic research work on FBRs in China with only a small amount of funds and material resources. The emphasis of research work was put on the core neutronic physics, thermo-hydraulics, sodium purification and impurity analysis, compatibility of materials with sodium and some sodium facilities on a small scale. So far more than ten miniature sodium loops and facilities have been set up (some have been shut down) as is shown in table 1.

#### 2.1. Reactor Neutronic Physics

At the zero power facility some reactor neutronic physics experiments have been done, which are:

- research of the criticality and of the characteristics of safety;
- measurements of reactivity coefficients of small specimens;
- measurements of the relative distribution of fission and of the ratios of fission rates;

- research of measurement methods for reactivity including oscillator technique, numerical inverse dynamic technique and source multipication technique, the time-range analysis technique of neutron noise of the zero power fast neutron facility;

- the measurements of fast neutron spectrum; and

- the research of real-time measurement on-line.

Name and Function	Parameters	Values	Commissioning Time
Fast neutron zero power facility	Load (max):	50 kg <sup>235</sup> U	July 1970
Isothermal sodium loop	Temp.(max): Sodium velocity (max):	600 <sup>0</sup> C 12 m/s 0 50ppm	July 1970
Sodium purification 1970 loop	Sodium volume: Oxygen after purification:	150 kg 20 ppm	September
Sodium heat transfer facility, for single pin	Flow rate (max): Sodium temp (max): Heat transfer power: Pump head:	20 m <sup>3</sup> /h 550 <sup>0</sup> C 50 kW 5.5 kg/cm <sup>2</sup>	October 1970
Thermal convection sodium loop	Sodium temp (max): Flow velocity: Sodium volume: Oxygen:	700 <sup>0</sup> C 6 cm/s 41 15 ppm	1972
Control rod drive mechanism component test facility	Medium: Flow rate (max): Driving range: Eccentric distance:	water 1 t/h 800 mm ±30mm	October 1979
Sodium loop for plugging meter	Temp. (max): Flow rate: Sodium volume:	450 <sup>0</sup> C 1 m <sup>3</sup> /h 281	October 1981
Sodium stress corrosion test facility	Temp. (max): Load (max): Sample deformation range:	700 <sup>0</sup> C 600 kg 0 - 10 mm	December 1981
Alternative current electromagnetic pump test sodium loop	Flow rate (max):	5 t/h	1968
Alternative current electromagnetic pump test sodium loop	Flow rate (max):	10 t/h	1969
Direct current electromagnetic pump test sodium loop			1968
Sodium valve test loop	Temp. (max): Flow rate:	450 <sup>0</sup> c 18 m <sup>3</sup> /h	1984

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Because the core of the zero power facility is too small, it can not simulate the experimental fast reactor that will be set up around 2000 in China. The facility is only used with the aim to master various experimental techniques and to analyse experimental methods.

2.2 Thermohydraulic Studies

At the sodium heat transfer facility, with flow rates of  $15 - 20 \text{ m}^3/\text{h}$ , the following experiments have been done:

- heat transfer experiments of liquid sodium turbulence flow in a circular tube;

- heat transfer experiments of liquid sodium flow with low Peclet number in a circular tube;

- heat transfer experiments of liquid sodium turbulence flow in a concentric annular with internal heating;

- heat transfer experiments of liquid sodium flow in an eccentric annular with internal heating;

- influence with two sides heating on the transfer coefficient when the sodium flows through the concentric annular.

### 2.3 Sodium Technology

As far as the sodium technology is concerned, research has mainly been carried out on sodium purification and impurity analysis. Sodium purification facilities have been installed, the maximum output of purified sodium is up to 240 kg per day. Table 2 presents the sodium quality after purifying by two purification facilities. The methods of impurities analysis adopted in our labs are presented in Table 3. Besides those, a miniature chemistry sodium loop has been set up on which the research of measurement on-line, mainly by manual plugging meter has been carried on. Expansion graphite methods have been established to extinguish sodium fire on a small scale.

#### 2.4 Materials

With the isothermal flow sodium loop, corrosion selection research has been done for more than 30 types of Chromium-nickel austenitic stainless steel and some alloys based on Nickel. We have carried out studies on corrosion aspects, mass transfer and subsequent microstructural changes of alloys of 316 stainless steel based on modified Vanadium and Titanium contents in the thermal convection sodium loop. The study of stress corrosion characteristics on 316 stainless steel has also been done in the stress corrosion test facility with high temeprature sodium.

# 2.5 Codes

About ten years ago, computer codes for FBR core design have been written in ALGOL for computers made in China. Since these computers are no longer in use and the nuclear data have become outdated, the codes are not used any more.

Currently we import and get through international cooperation some computer codes, mainly from US, for nuclear data, reactor neutronics and shielding, thermohydraulics, fuel pin design, mechanics and safety.

Purification loop	0	с	Fe	Co	Ni	Cr	Mn	Si
1	10	20						2
2	2	3	0.15	0.01	0.01	0.01	0.005	1.1

### Table 3 Analysis Methods for Impurities in Na

Element	Method
Oxygen	vacuum distillation
Carbon	<ol> <li>combustion in high temperature</li> <li>combustion of the residue after distillation</li> </ol>
Iron Cobalt Nickel Chromium Magnese	vacuum distillation-atomic absorption spectrophotometry (with flame of graphite furnace)
Silicon	Molybdenum blue spectrophotometry
Potassium Cadmium	flame spectrophotometry

### 2.6 Components

During the past years, experience in design construction and operation of miniature sodium loops has been gained. We have successfully trial produced small magnetic pumps, valves, cold traps and some sodium instruments. For control rod mechanism, some key components have been fabricated and tested, for example, bellows, buffers and grapples.

# 3. The Envisagement for the Plan of FBR Technology Development in China

According to the needs of the electricity generation in China and to the trends of FBR technology development in the world, we envisage that the first demonstration FBR would be built around 2015 - 2020 in China. Preliminary research on the long-term nuclear energy development strategy indicated that it would be reasonable to select a modular fast breeder with a size of 100 - 150 MWe (MBRC) as the first demonstration FBR. The main reasons for this selection are:

- lower technical-economical risks
- easier to match local electricity grids with providing either 60, 90 or 120 MWe.

In order to get experience, a modular breeder reactor prototype (MBRP) will be set up in about 2010.

But the purpose in the case of China to develop FBR technology aims not only at the effective application of uranium resources, but also at the needs of shorter doubling time for FBR; in other words to get the beneficial energy of uranium resources in a rather short period. It is envisaged that a first large fast breeder reactor (LFBR) with a high breeding capability will be constructed around 2035 - 2040, after which consecutive LFBRs will come on line.

As the first step of the FBR technology development in China, we plan to construct an experimental fast reactor (named FFR as it will be the first fast reactor) with 25 - 50 MWe around 2000 to get experience in the design, construction and operation of FBR for electricity generation. During the operation of the reactor, it will become a fast neutron irradiation facility for the development of fuels and materials. The FBR long-term strategy is shown graphically in Fig. 1.



FIG.1. FBR long term strategy.

For this reactor, the main technical options have been decided as follows:

We have selected (Pu, U) $O_2$  as the fuel for the FFR loading, because mixed oxide fuel has been adopted for fast reactors in the world for about 90 reactor years. The operation and irradiation results have shown that mixed oxide fuel has a high temperature stability and a good irradiation performance. However, the fast reactor with mixed oxide fuel has a weaker breeding capability due to the lower heavy atom density and a softer neutron spectrum. We are going to pay great attention to metal alloy fuel, encouraged by the excellent success of the ARGONNE National Laboratory in this domain.

As for the cladding and hexagonal tube material, 316 stainless steel (Ti-modified) has been chosen for this reactor, considering that it could meet the needs of the maximum irradiation damage for the experimental fast reactor.

Power	MWe	25 - 50
Na outlet temperature of the core	<sup>0</sup> C	530
Steam temperature	0 <sub>C</sub>	480
Steam pressure	bar	90
Maximum permitted cladding temp.	0 <sup>C</sup>	700
Maximum burnup of the fuel	MWd/t	5 x 10 <sup>4</sup>
Refuelling scheme	reactor pool inner stora	ge for spent
	fuels, two rotating plug moving fuel handling mac	s with straight hine
Safety	two independent shut down passive decay heat remove	n systems, al system

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Concepture Design	
Primary Design	
Dotail Design	
Components Fabrication	
Instattation and Adjustment	
Loading	
Start up and Test	—
Architecture and Civit Engineering	
R and D in New Labs	
New Labs Construction	
<b>Frototypes and Modules Test</b>	
Prototypes and Modules Fabrication	
Engineering Development Labs	
Construction	

FIG.2. Schedule envisaged for FFR.

For the primary system arrangement of the liquid metal fast breeder reactors two principal design concepts have been used: the loop type and pool type. Generally, the advantages (and disadvantages) of both concepts roughly balance each other <sup>1)</sup>. But it is especially emphasised that in the pool concept, leakage in the primary system components and piping does not result in leakage from the primary system, the mass of sodium in the pool is rather bigger, thus providing a larger thermal capacity. We also find the fact that the pool type is favoured for designed large scale fast reactor: SPX-2, SNR-2, CDFR and BN-800. It is attributed to the success of EBR-2, PHENX, PFR, BN-600 and SPX-1. Considering all above the pool concept has been chosen for the FFR.

So far the boundary conditions for the FFR design have been preliminarily decided as shown in Table 4.

Before 1990, the main research work on fast reactor technology includes:

- The conceptual design of the FFR
- Sodium technology
- Reactor core neutron experiments and fuel bundle thermo-hydraulic experiments and reactor hydraulic simulation.
- Trial production and tests of the fuel and materials
- Trial fabrication of prototypes of the fuel subassembly and control rod driving mechanism
- 2 -3 sodium loops for thermo-hydraulic research and some sodium devices tests.

For the coming quinquennium (1991 - 1995) it is mainly planned to continue the experimental research work and to establish the experimental conditions for the sodium components needed by the FFR. The schedule envisaged is shown in Fig. 2 for the FFR and its support activities.

### 4. International Cooperation

At present, in the domain of nuclear science and technology, our Government has established cooperative relations with developed countries for example, Germany, France, Japan and Italy. We are naturally interested in seeing the enlargement, as much as possible, of the exchange and cooperation in the domain of the fast reactor technology.

### **ACKNOWLEDGEMENTS**

Thanks are due to my colleagues of the Department of Reactor Engineering and Technology, IAE, for providing various information and data. Special thanks are due to Dr. Prof. Vuong Chau, Director of the Committee on Fast Reactor Technology, DRET, IAE, for his guidance in preparing the paper.

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# DESIGN REQUIREMENTS FOR APWR IN CHINA

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# Abstract

Prospects of the introduction, development and the required conditions of APWR have been described in this paper. It is expected that by the year 2000 and 2015 the installed nuclear capacity in China will reach to 6 GWe and 30 GWe respectively.

Problems faced by China for the development of nuclear generation are discussed such as uncertainty of plans, key techniques which are not yet mastered, higher construction cost etc. The technology of APWR must be highly credible and trusted by the investors and regulations for their introduction in China.

Fundamental philosophies and design requirements for the development of APWR in China are also discussed such as safety (core melt frequency = 1 X  $10^{-6}$ /reactor year), simplicity, increase in design margins, increase in plant availability and reliability, reduction in plant capital cost, increase in plant life time upto 60 years etc.

#### 1. Introduction

In the near future, building nuclear plant in China is only a supplementary means of electricity industry. Eventually, however, the nuclear plants are substitutes for fossil fuel stations. According to prediction of future energy demand, the amount of nuclear generation would be more increased in China in the 21st century. It is expected that 6 GWe capacity of nuclear plants will have been reached by 2000 and 30 GWe by 2015. Approval of additional fossil fuel plant construction plan will not be expected before 2000.

At present, through technology cooperation and economy, nuclear equipment (for Qinshan-1,300 MWe) and whole nuclear power plant (Daya Bay nuclear power plant) have been imported and now the design and construction of 600 MWe PWR nuclear power stations (Qinshan-2 and 3) are being prepared mainly based on self-reliance with cooperation on the introduction of foreign advanced technologies and key equipment.

At the same time, China is paying more attention to the APWR development and has the desire to incorporate into the international cooperation. After several 600 MWe nuclear plants have been built, a series of 600 MWe APWR nuclear plants will be built instead of PWR.

# 2. Maintaining Orientation of APWR

The development of PWR in China is originated from nuclear submarine PWR. Based on the 300 MWe Qinshan-l PWR nuclear power plant has been designed on self-reliance and several feasibility studies of nuclear plant, such as Jinshan, Lhasa, Hainan, have also been done. A complete set of devices for PWR engineering research has been basically established and the fair-sized ability of manufacturing PWR equipment has been already formed. Thus it can be seen that the conditions of developing APWR are better than other kinds of reactors for our country.

PWRs are used in most of the operational nuclear power plants in the world and APWR development is carried out in most countries. So if we insist on developing APWR, it will be convenient to cooperate with foreign countries.

# 3. Choosing 600 MWe Output

The 600 MWe output APWR plant is considered because it has the following advantages:

- Better match of low electric demand growth and load growth uncertainty.
- Lower absolute capital cost with small financial burden, lower financial investment and risk.
- Short total construction leadtimes and reduced construction cost.
- High degree of shop prefabrication and potential for series production and standardization.
- Easier to fit smaller and weaker grids and lower requirements on grids.
- Easier to use passive safety systems and easier to simplify the systems.
- Excellent performance records in the world.
- Have some experience with small PWR in China.

#### 4. APWR Design Requirements

### 4.1. APWR Fundamental Objectives

In the field of nuclear generation there are some problems to be resolved in China, such as uncertainty of plan for developing nuclear plants. Some key techniques are not yet mastered, the capability of nationalized equipment manufacturers is low and the construction costs are higher than expected. These reasons make the wish to build nuclear plant impossible to fulfill.

In order to make APWR a suitable choice, the following tests have to be met:

 Technical excellence; the APWR must be an outstanding power generation system in all aspects, including safety, technical performance and environmental capability.

- Economic advantage; the APWR must be economically attractive in comparison to its competitors on a life cycle cost basis.
- Investment protection; the APWR must provide very high protection of the utility investment, particularly in terms of
  - (1) extremely low risk of severe accident,
  - (2) assured licensability,
  - (3) predictable construction cost and schedule,
  - (4) predictable operating cost and plant availability.

In short, the APWR must be a highly credible nuclear power plant trusted by the investor and regulations.

4.2. APWR Design and Philosophy

The following set of fundamental philosophies has been established to guide the development of the APWR requirements and conceptual design:

Safety

Nuclear safety is of paramount importance and must play a dominant role in the development of APWR requirements.

- Simplicity

Elimination of unnecessary complexity can improve safety and reliability, as well as reduce the number of expensive systems and components included in a reactor.

# - Design Margin The APWR will be a vital plant with substantial built-in margin to provide inherent capability to deal with adverse situations.

#### - Reliance on experience

The APWR design is based on proven technology; its success is not dependent upon as yet untested technological advances. Up to now no nuclear plant has been completed in China, and we have less experience with PWR plant, so we hope to participate in international cooperation to develop APWR. Having heard our tentative plan about APWR development, the Bureau of Nuclear Science and Technology is of the opinion that advanced core, simplification of systems and using passive safety systems should be given sufficient consideration.

4.3. Design Objectives

The major design objectives are:

- increased plant availability up to 90% and improved component reliability,
- reduced plant capital cost,
- significant reduction in uranium ore requirements and separative work units,
- an expected lifetime of up to 60 years,
- reduced risk to the utility and the public (core melt frequency of l x 10<sup>-6</sup>/reactor year,
- reduced occupational radiation exposure,
- increased design margin.

# IMPROVEMENTS IN A PROTOTYPE BOILING WATER REACTOR: LAGUNA VERDE, MEXICO

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# Abstract

Laguna Verde is the first nuclear power station in Mexico. It has two GE Boiling Water Reactos which will produce 654 MWe each via Mitsubishi turbine generators. At this moment we are ready to load fuel on Unit 1 and 50% complete on Unit 2 beginning electromechanical installation.

The project has required 3,600 millions dollars including interest rate, over 1,000 full time engineers and about 3,800 direct labour workers. Additionally, QA, engineering, construction, start-up and operations prepared and are using approximately 4,400 procedures to perform their activities. Furthermore, 54 industry branches in Mexico have been qualified by quality assurance and they have been providing equipment, components and sub components for the project.

However, what we appreciate the most is our "people".

Constructing Unit 2 has given us the opportunity to realize the benefits of standardization. Once "people" become familiar with a design concept, a BWR-5 with a Mark II containment in this case, the engineering, construction and testing process improves drastically. As of this date, the average savings in man-hours required to build Unit 2 is 40.59% versus the amount needed for Unit 1.

We are not making any dramatic change in the design concept of Unit 1, what we are changing in Unit 2 are our working methods and improving when it is appropriate. For instance, large bore piping, HVAC ducts and cable trays are remaining as they are in Unit 1; however, small bore piping, conduit and tubing will be routed in a different manner to reduce as much as possible the number of supports. Supports in Unit 2 will be multidisciplinary since many interferences in Unit 1 were due to an excessive number of supports which were installed on a per discipline basis.

We have not achieved that point yet, but in general in control systems, instrumentation and computers there is plenty of room for improvements, by using fiber optics, multiplexers, etc. We will certainly try it.

The message is, a developing country does not have the luxury of changing its design concept when a new reactor is to be purchased. The idea is once you jump ahead of the learning knee, keep on using the same concept unless improvements are worthy.

#### INTRODUCTION.

In spite of the economic crisis the demand for electricity in México has been increasing at an average annual rate of 5.6% and during 1988 it is expected to exceed 7%.

During 1987 the contribution of the different technologies within Comision Federal de Electricidad to the generation of electric power was as shown in Figure 1.

A Key objective of the mexican energy program is to diversify as much as possible the use of different energy sources to generate electricity with the ultimate aim of becoming less dependent on oil. Most countries appear to be doing this by installing either nuclear or coal fired power plants, although after Chernobyl nuclear is being seriously questioned while coal is being critized due to problems such as acid rain and the green house effect.

Based on above mentioned demand rate of 5.6% per year, Comision Federal has developed the scenario shown in Figure 2 for the period up to year 2014. To meet this scenario additional power plants will have to be comissioned to reach the target of 61870 MWe. Taking into account the proven national resources and industrial infrastructure this will require 10800 MWe in hydro plants, 5300 MWe in coal plants, 1700 MWe in geothermal and an additional 21660 MWe from other sources.

The deficit of 21660 MWe can be satisfied by installing an 18200 MWe capacity provided by dual plants which can burn either imported coal or oil and a 6350 MWe capacity derived from nuclear plants. This later would require 8 units similar to Laguna Verde 1 and 2. The above share takes full cognizance of the national capability of engineering and construction.

The idea of designing, constructing and comissioning 8 additional nuclear units in the next two decades is a real chalenge for a developing country. To achieve it, although perhaps not the best way, certainly the easiest way is to built 8 identical units adopting a standard design.



1987

FIG. 1.

INSTALLED CAPACITY (MW) HIGH DEMAND- HIGH DIVERSITY



YEAR 2010 FORECAST

FIG. 2

# THE CONCEPT OF STANDARIZATION.

Should mexico desire that additional nuclear power plants be constructed both construction time and capital cost must be reduced to make nuclear energy competitive with other souces. Adopting a standard design concept has proven to be a successful strategy in countries such as France, South Korea, Canada and Japan, to reduce construction time and investment.

A standarization program should include engineering, construction, testing and operation of nuclear power plants.

The engineering and design of a single nuclear unit may be as high as 25% of its total cost. If additional units are to be built the cost per unit will be reduced since the same conceptual or basic engineering can be usedfor each unit with only certain detailed engineering required to address specific siterelated problems and to reflect the 'as built' condition of a unit.

As demand on engineering man-hours to support construction is reduced, the engineering resources can be focused onto perhaps more operations oriented matters such as performing analyses which would result in modification packages to enhance Safety features or availability of the nuclear units. Maintaining the same design concept under a standarization program allows for improvement in the detailed engineering. For instance, architect-ingeneers have a tendency to support and/or restrain plant components by engineering discipline or speciality. For this reason separate support systems are provided for piping, cable trays, HVAC ducts, conduits and instrument tubing. This represents essentially a "selfish" way to build a plant and is perhaps the number one cause of interferences during the construction first prototype is built however and all Once the process. interferences are solved, engineers may then start thinking of designing multidisciplinary supports.

Standarization also forces engineering to become more efficient by using tools such as a CAD system, standard tables, computer programs, system models, etc., that may not be cost effective if a single unit only is to be built.

As an example, experience in the French program indicates that there is a ratio of 5 to 1 between the man-hours required to design the first unit to those required ones for subsequent units.

Constructing identical or similar nuclear units instills confidence in the construction supervisors and crew. Since the same construction procedures are used, the construction sequence can be optimized based on previous experience, they become very familiar with construction drawings, etc. The end result, using again as an example the French program, is that the number of construction man-hours is reduced by more than ane half between the first unit and the following units of the same vintage.

Another important advantage of a standarization program is the capability of being able to provide to the national industry a clear perspective of future needs, since many times local industry is somewhat reluctant to adopt a quality assurance program to provide either services, equipment or materials for nuclear power plants. Some countries such as South Korea and Argentina have been quite successful increasing the participation of the national industry to levels as high as 80%.

With respect to the operation of a nuclear unit it is important to point out that the standard design concept simplifies greatly the training process, and operators can move easily from one plant to another, operating procedures are common for all plants and the operating experience both external and internal is evaluated and incorporated for a single type of plant. Further the chances of achieving high system capacity factors are enhanced since people are concentrated on solving problems whose solution will apply to several units, instead of trying to solve different matters for different units.

Last but no least, the licencing process for a standard design such as preparation and maintenance of the safety analysis reports including

the technical specifications and responses to all questions from the regulatory body, would be valid for all plants bounded by the approved design.

### EVOLUTION OF THE BOILING WATER REACTOR.

As of today 17% of the electricity demand in the world is satisfied by nuclear power plants with a total generating capacity of more than 300,000 MWe. One fourth of this generation capacity is provided by Boiling Water Reactors, BWR'S.

At the world level there are three suppliers of BWR'S with similar designs: Asea-Atom in Sweden, NWU in West Germany and General Electric in the United States. General Electric has provided the highest number of BWR'S to the nuclear industry including the two Laguna Verde reactors.

The General Electric design philosophy for the modern BWR has evolved into a simpler design. Back in 1956, Dresden 1 had 4 Joops with a double cycle system. Later, based on operating experience the design was modified as iJustrated in Figure 3. First, the steam dome was deleted, then the steam generators were eliminated, and finally 3 out of the 5 recirculation loops were removed.

Refering to the primary containment, General Electric began with a dry containment, followed by 3 generations of pressure suppression containments as shown on Figure 4. These containments have been designed with a high volume suppression pool, to retain the fission products within containment, providing a very effective protection to the health and safety of the public.

An advanced boiling water reactor is currently being developed, under the direction of Tokyo Electric Power Company to become the next generation standard BWR in Japan. The first two units are scheduled to start commercial operation in 1996 and 1998. These units will be provided by a joint venture of General Electric, Hitachi and Toshiba.



DOUBLE CYCLE - 4 LOOPS

DOUBLE CYCLE-SLOOPS











FIG. 4.

The major improvements of this reactor are:

- Improvements in nuclear fuel.
- Elimination of external recirculation loops.
- Design of emergency core cooling systems not considering core uncovery.

- Solid state/digital instrumentation and control
- Pressure suppression containment with a covered pool with improved access for maintenance and inspection.

### LAGUNA VERDE AND ITS TECHNOLOGY

Laguna Verde has been designed based on the latest comercially available technology from General Electric. Each of the two units has a direct cycle steam conversion system with two external recirculation loops and utilizes the Mark II containment concepto which is very similar to the containment for the advanced BWR.

Furthermore, Laguna Verde has implemented in its design all improvements required by the regulatory body after the TMI accident, to assure a high level of safety.

# LAGUNA VERDE UNIT 1 AS A STANDARD PLANT.

Laguna Verde is the first nuclear power station in México It consists of two GE boiling water reactors, which will each produce 654 MWe via Mitsubishi turbine generators. When the project began Unit 2 was one year behind of Unit 1; however, as of today Unit 1 has loaded fuel and is in the process of hot testing while Unit 2 is about 50% complete and beginning electromechanical installation.

The major work tasks required to construct Unit 1 are listed in Table 1. Additionally, the engineering, construction, start up, operation and other activities for Unit 1 required the preparation and implementation of the procedures indicated in Table 2.

Although it was not originally planned, the combination of the experience gained in commisioning Unit 1 and the delay in construction of Unit 2 has combined to provide a very favorable situation for improvements to Unit 2.

It is not expected to make any dramatic change in Unit 2 such as removing the two external recirculation loops connected to the reactor

таѕк	UNIT	AMOUNT
REBAR	TON	26,152
CONCRETE	M3	160,985
PAINTING	M <sup>2</sup>	415,690
PIPING	КМ	135
PIPING SUPPORT	PZA	36,353
HVAC DUCTS	TON	734
ELECTRIC CABLES	KM	2, 546

# TABLE 1. LAGUNA VERDE UNIT 1 - MAJOR WORK TASK

# TABLE 2. LAGUNA VERDE UNIT 1 PROCEDURES

BRANCH	No. OF PROCEDURES
ENGINEERING	101
CONSTRUCTION	390
START UP	1523
LICENCING	15
QUALITY ASSURANCE	55
OPERATION	6097

pressure vessel. However, what is expected is to optimize what we already have and know. As an example the savings in the civil construction of Unit 2 versus Unit 1 are evident from the information presented in Tables 3 and 4.

These savings in construction time and man-hours shown are the result of several factors:

- Confidence of the construction personnel.
- Engineering is providing to construction a "clean" design since
   approximately 39,000 design changes were processed during the construction of Unit 2.
TABLE 3. LAGUNA VERDE NUCLEAR POWER PLANT — COMPARISON BETWEENUNITS 1 AND 2 (in man-hours)

	UN	IT	DIFFERENCE
CONCEPT	ONE	тwо	DIFFERENCE
PRIMARY CONTAIMENT WALL	200, 411 .	88,391	112,020
DIAFRAGM FLOOR	69, 648	36,107	53,541
FLOOR OF FUEL POOL	54, 018	24,632	29,386
WALL OF FUEL POOL	242,444	148,124	94,320
INT. WALLS FROM ELE. 25.10 TO 33.00	1		
AND INTERMEDIATE SLABS ELE. 18.70 TO 25.10	271,639	70,178	201,461
SLAB AT ELE. 25.10	52,655	53,104	(449)
WALLS OF MAIN STEAM TUNNEL	ł		1
(REACTOR SIDE)	19,424	9,315	10,109
(EAT-WEST CORRIDOR)	10,406	7,133	3,273
EXTERIOR WALLS ELE. 25.10 - 33.00	16,664	16,120	544
SLAB AT ELE. 33.00	78,439	47,726	30,713
EXTERIOR WALLS ELE, 33.00- 39.40	142,308	87,543	54,765
INTERIOR WALLS ELE. 33.00- 39.40	153,920	101,259	52,661
SLAB AT ELE. 39.40	84,808	30,807	54,001
EXTERIOR WALL ELE. 39.40-49.90	253,766	178 ,855	79,911
SLAB AT ELE. 49.90	59,122	51,829	7,293
INTERIOR WALLS ELE. 39.40-49.90	201,353	147,893	53,460
STEEL STRUCTURE OF SLAB AT ELE.3300	34,247	43,580	( 9, 333)
STEEL STRUCTURE OF SLAB AT ELE. 39.40	49,057	21,220	27, 837
STEEL STRUCTURE OF SLAB AT ELE. 49.90	55,745	16,946	38,799
SUPERSTRUCTURE	158,340	146,228	12,112
TOTAL	2,228,414	<b>1,3</b> 26,990	901, 424
PORCENTAGE OF SAVIN	IG	40	.45 */•

TABLE 4. LAGUNA VERDE NUCLEAR POWER PLANT — TIME DURATION AND EFFICIENCY COMPARISON BETWEEN UNITS 1 AND 2

CONCEPT	TIME DURATION (MONTHS)		EFFICIENCY			
	UNIT-1	UHUT-2	UKIT-1400%	UNIT-I	UNIT- 2	UNIT-1-100%
PRIMARY CONTAINMENT	34	23	67.6%	129 MH/M <sup>3</sup>	51 MH/M <sup>3</sup>	39.5%
DIAFRAGM FLOOR	15	6	40.0%	528 MH/ M <sup>3</sup>	213 MH/M <sup>3</sup>	40. 3%
INTERIOR WALLS	27	14	51,9%	187 MH/ M <sup>3</sup>	95 MH/M <sup>3</sup>	50.8%
SLABS	18	12	66,7 %	183 MH/M <sup>3</sup>	122 MH/M <sup>3</sup>	66.7 %
EXTERIOR WALLS	17	12	70.6%	179 MH/M <sup>3</sup>	122 MH/ M <sup>3</sup>	68.2%
SUPERSTRUCTURE	18	10	55.6 %	105 MHV.T	97 MH/ <sub>T</sub>	92.4%

- Construction is getting what they need to construct only i.e.
  materials and construction documents (not conceptual engineering documents)
  - The engineering information is being provided on a plant system basis which should simplify transferring systems from construction to start-up.
  - People are already used to working under a QA program according to established procedures.

Furthermore, during the construction of Unit 1, vendors were qualified by quality assurance to provide services or materials to the Laguna Verde Project . Due to the familiarity of the engineering team with the project, the scope of supply can be defined for each potential vendor so as to attract his interest.

Standarization should not mean 'do not make any changes'. Whenever it is practical to introduce new technology, the project team must be open to accept it. For instance the usage of solid state/digital electronics should be analyzed. For signal transmission, multiplexer and fiber optics should also be considered.

Referring to training, Laguna Verde has a second generation simulator which was developed by the National Electric Research Institute (IEE). This simulator consist of 9 main panels and 13 auxiliary panels where 48 computer models interact to simulated the integrated dynamic response of the plant to the postulated transients. This simulator is one of the most complete and advanced in the world for a BWR Unit. The investment will be recovered as plant operators proctice accident mitigation, emergency procedures, special transients, etc. Needless to say that the more BWR Units are constructed the more they would benefit from the simulator as experience is gained.

Finally, Laguna Verde Final Safety Analysis Report consist of 22 main volumes and 16 additional reports. During the licencing process the

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mexican regulatory body asked approximately 3500 questions. It was necessary to consider and evaluate each question to provide satisfactory answers. Obviously, this tremendous licencing effort would not have to be repeated if similar BWR's were to be constructed

#### CONCLUSION.

Considering that BWR technology remains essentially the same and that the new models will be tested and available late in the next decade, it appers to be reasonable for Mexico to continue building, at least for the forecoming years, BWR'S similar to Laguna Verde.

It is difficult to forecast at the middle of our economic crisis if additional nuclear units would be built, however, what is worth to mention is that the biggest asset in our nuclear industry are our people; our engineer and technicians who are ready to make proper decision in the engineering, construction and operation of future units. B. BOHERDirection technique,Framatome,Paris-La Défense,France

Abstract

Through the very important nuclear program (PWRs) that FRANCE set up to cope with the electricity demand in the 1970's, and the continuous effort to improve the nuclear plants, the development of advanced concepts is explained. Some examples of the great diversity of French advanced technologies are shown. Most of them will be applied in the last French model, as also the French Safety approach which is described here. This last model (called the N4 model) is presently in the erection phase at CHOOZ, and will be in operation in few years (expected for 1991). These advanced concepts could be integrated in a nuclear power plant able to fulfill the needs of the developing countries for the 1990's, with the guarantees of proven technology which explains the success of the French Program.

### I. INTRODUCTION :

The incentives for advanced nuclear technology in FRANCE are the results of a unique situation based on the huge French Nuclear Program (Fig. 1). It is very well known that the success of such a program relies on focussing the French Nuclear Industry'skills and abilities on standardized products.

I would like to add the importance of the French organization which explains also the success, with one national utility "ELECTRICITE DE FRANCE" (EDF), one builder of the NSSS, FRAMATOME, and the support of the "COMMISSARIAT A L'ENERGIE ATOMIQUE" (CEA) which provides the means for the considerable R and D effort.

Not only the success of the program can be explained by these considerations but also the French approach relating to the incentives for advanced nuclear technology.

Through these main points, it is easy to understand that the information coming back from the different sites through EDF thermal production division or from units built abroad, from FRAMATOME maintenance teams, or from our factories, and from such organization as INPO, OCDE, IAEA, can be integrated into the general objectives of the different teams. These objectives are mainly :

- to improve high level of safety,

- to increase operating capabilities ;
- to enhance man/machine interface and provide greater assistance to the plant operators, fully integrating the lessons learned from THREE MILE ISLAND 2 :

- to continue progress in the fields of plant operational maintenance :

- to decrease the occupational exposure to radiation.



FIG.1. Electricité de France PWR series.

All the knowledge gained from experience makes possible to define R & D works and to guide the development of new concepts with the goal to fill these objectives. Either the application of these new concepts or their incorporation into the successive series of units is carefully considered. The same attention is given for the backfitting on existing units when feasible, consistently with the spirit of continuity for proven technology which is the standardization policy applied on the EDF nuclear power plants.

Let me give you :

- 1) some examples of the last advanced technologies which are, as I told you, the results of the integration of considerations and experience, at home and abroad, of the men involved in the nuclear field, and
- 2) the French Safety approach which is also the result of the experience gained, and, of the fundamental objectives more and more shared internationaly, as recently recalled by the INSAG-3 report.

## II. SOME EXAMPLES OF THE FRENCH ADVANCED TECHNOLOGIES :

# The load follow mode and the frequency control :

When nuclear power generation becomes the predominant source of electricity, plant output must be varied to meet grid demands. Adapting to changes in demand requires the use of two separate techniques :

- load follow, which is used to meet day night cyclic load variations.
  Typically, load follow is a daily cycle that can vary load between full power and 30 % of full power;
- remote frequency control, which permits rapid adjustment in load from a central location to maintain a stable grid frequency. The magnitude of variation is several percent (typically 5 %) around the load demand value.

Both techniques are being used in FRANCE, where nuclear power plants produce more than 70 % of all electricity. For example in 1987, the number of cyclings was 2213 for the 900 MWe units. The use of load follow strategies is becoming a reality in plant operation and will become more common as utilities increase the share of nuclear power in their total generation mix.

## M The AFA :

Lower-cost fuel cycle will be achieved with the use of the Advanced Fuel Assembly with less neutron absorbent grids, and with the high burn-up capabilities of the fuel (objective 45 000 MWD/ton), allowing one-quarter of core reloads.

The basic geometry of this fuel assembly (fuel rod diameter, pitch, etc...) and its technology are identical to those of a standard 17 X 17 fuel assembly, like those widely in use today. Thus it is possible to rely on the considerable operating experience built up with these standard assemblies.

The improved features of the AFA with respect to the standard model are :

o Replacement of Inconel 718 by Zircaloy 4 as the grid material,

o Removable upper and lower endpieces.

The first modification allows an enrichment gain of approximately 0.07 % in absolute value (thus reducing the necessary enrichment of the fuel reloads to 3.1 %) due to the lower neutron absorption of zircaloy. In addition, zircaloy's lower cobalt content enables significant reduction of the level of activity of the reactor coolant.

The second modification offers the possibility of replacing a defective fuel rod from either end of the assembly, when it is removed from the core.

## The advanced control room :

The advanced control room is based on ergonomic principles aiming at alleviating the operator tasks and maximizing the understanding of events under all circumstances. Each redundant operator desk is a single working area where the operator has a total access to information and control devices. The information display is elaborated in a very synthetic and comprehensive form.

This was made possible by a large scale implementation of computer-based man/ machine interface, with display screens and modern interactive dialogue techniques.

This advanced design takes advantage of the post-TMI improvements in the 900 MW and 1300 MW class series, and from the experience gained in the implementation of plant control computer systems.

## Use of low release materials :

Inconel 690 was chosen as the U-tube material after detailed reviews of its development and qualification programs. Inconel 690 features resistance to the various mechanisms of SG tube degradation and is equivalent or superior to Inconel 600. Using this improved alloy also leads to less release of radioactivity into the reactor coolant system.

### The Safety Systems :

In continuity of the 1300 MWe - series, the main important-to-safety systems are :

- o The Residual Heat Removal system (RHR), comprising two identical independant trains, each one having one pump and one heat-exchanger. It is located inside the reactor building so that, should it leak or rupture, no contaminated water would be released directly to the environment. It removes residual heat from the core in normal shutdown or after accidental conditions once low pressure is reached.
- o The Safety Injection System (SIS), comprising two identical trains having each one medium-head pump and one low-head pump, capable of injecting in the four loops and four accumulators. It removes heat from the core to the containment in case of LOCA.
- o The Containment Spray System (CSS), comprising two identical trains, each one having one pump, one heat-exchanger and one spray ring. It removes heat from the containment to the ultimate heat sink in case of LOCA.
- o The Reactor Coolant System (RCS) overpressure protection system which allows to bleed the primary circuit is composed of three lines of pilot-operated safety valves and isolation valves in series.
- o The Auxiliary Feedwater System, comprising two independent trains, each one having one motor driven pump and one turbine-driven pump, supplying emergency water to the steam generators.
- o The Atmospheric Steam Dump, which permits bleeding of the secondary system : each of the four steam lines is equipped with two power-operated relief valves (PORV) and isolation valves in series.

#### The safety systems philosophy :

- o The separation of functions is a principle which is implemented on the recent French plants as far as possible : it implies to dedicate one safety function to one system and avoid multi-function systems. Therefore no system configuration change is needed in case of accident. It also simplifies plant operation and makes the systems more understandable and predictable to the operator.
- o A general two-train system organization is adopted as in the French plants already in operation. The main advantage is that it allows an easy installation, an easy separation of redundant paths, and a low susceptibility to common cause failures.
- As regards provisions for maintenance, no extra-redundancy is required for safety systems in standby : preventive maintenance is performed at reactor shutdown when the system is no more necessary ; in case a system is found unavailable (for example as a result of a periodic test), power operation is allowed only for a limited period of time following which the plant must be placed in the fallback mode, in accordance with the operating technical

specifications. According to the French operating experience, no significant plant unavailability occurs as a result of safety systems unavailability (a fraction of day per year).

Safety systems normally operating, as they can be maintained only in service, are required either to have an extra-redundancy or to have a back-up available to perform maintenance without lowering the safety level.

In addition to the design of the redundancy within safety systems, consideration is given to the hypothetical conditions which would result in a total failure of a redundant system. Diversified back-ups are provided to face such conditions; for systems which are frequently actuated in their safety function, a short-term back-up is provided; for systems which are seldom actuated in their safety function, a long-term back-up only is provided.

Probabilistic approach gives adequate guidance in this area. As far as possible, diversity is implemented on these back-ups to prevent additional common cause failures.

## The reactor protection system

The Reactor Protection System is a fully computerized, multimicroprocessor based system. Its performs core power distribution reconstruction and in situ computation of margins with respect to physical limits, such as DNBR and linear power. Core power distribution reconstruction is based on axial-radial synthesis method, using algorithms that can be rapidly computed using a microprocessor.

Besides an accurate and automatic protection on safety limits, it provides data that facilitates the optimisation of plant operation.

## The containment :

The Containment chosen by EDF is a double-wall reactor building comprising an unlined prestressed concrete inside shell and a reinforced concrete outside shell. The annalus air space is maintained at subatmospheric pressure and filtered prior to discharge to the atmosphere.

## III. SAFETY ENHANCEMENT WITH THE FRENCH SAFETY APPROACH :

The safety approach : With the "basic design conditions" the "complementary design conditions" and the "considerations of severe accidents", this approach is determined in order to meet the French Safety objectives and through them the fundamental objectives recalled recently by the INSAG-3 report of the IAEA.

#### ENHANCED SAFETY IN PLANT DESIGN CONDITIONS :

o The design of a nuclear power plant is based on a deterministic list of events - from normal conditions to unlikely accidents - called plant design conditions. These conditions are classified (from 1 to 4) according to their estimated frequency and the allowable radiological consequences (table 1). A more severe classification is adopted, compared to previous plants, on events for which the feedback from experience indicated a higher frequency than originally estimated.

T.	А	В	L	E	1

PLANT CONDITION	ESTIHATED FRECHENCY (PER YEAR)	MLLOHABLE RADIOLOGICAL CONSEDVENCES
1	> 1 10 <sup>~2</sup> ~ 1	RAD. EFFLUENT RELEASE AUTHORIZATION
3	10 <sup>-4</sup> ~ 10 <sup>-2</sup>	<b>3 2 57 K.E.</b> 15 2 57 T.
4	10 <sup>-1</sup> ~ 10 <sup>-1</sup>	0. 15 SV N.R. 0.45 SV - 7.

o In particular, concerning the steam generator tube rupture incident, foreign experience has evidenced it was not an hypothetical accident. Consequently the rupture of one tube is considered as a class 3 incident, with rather severe allowable consequences. The rupture of two tubes is studied as a class 4 accident.

To avoid any risk of water discharge at the secondary safety valves, an improvement of the atmospheric steam dump is performed : it consists in :

- the doubling of the valves on each steam line : two PORV and two isolation valves in series,
- the qualification of the valves for water discharge.
- o The multiple tube rupture and the coincident stuck-open secondary safety valve were also investigated as beyond-design events. It is demonstrated that no fuel uncovery nor borated water tank emptying resulted from these events.

#### ENHANCED SAFETY THROUGH COMPLEMENTARY DESIGN CONDITIONS :

- o A major concern in the design is the progressivity of the safety measures which are implemented : it is necessary to ensure that no cliff-edge effect exists beyond design conditions, that no large step in the consequences exist when considering events with a slightly lower probability than the design-basis events.
- o In this sense, it was decided to add to the list of design conditions, a number of complementary design conditions corresponding to the total failure of redundant systems.

Mitigating means, termed "back-up", and procedures, termed "H", were determined in order to meet the French safety objective for these complementary conditions.

- o The safety objective is the following :
  - no unacceptable consequences should result from the operation of a PWR plant with frequency higher than 10<sup>-6</sup> per reactor year,

- applied to a particular family of events, no unacceptable consequences should result from this family with a frequency higher than  $10^{-7}$  per reactor year.

It must be pointed out that up to now this objective was a guidance of the Safety Authority, not a regulatory requirement. For the first time, on the N4 project, this objective is applied to the justification of the complementary conditions, and therefore used in the licensing.

- o Unacceptable consequences are interpreted as severe core degradation, which is very conservative, since it ignores the mitigating effect of the containment.
- o Figure 2 illustrates the risk associated to the total loss of a redundant system. H represents the benefit in terms of consequences of the migating means (if they work, core melt is prevented). H represents the benefit in terms of probability of the mitigating means (if they fail, core melt probability is lowered from P1 to P2; no credit is given for P1/P2 > 100).



FIG.2. Risk associated to the loss of a redundant system.

TABLE 2

N4 - COMPLEMENTARY DESIGN CONDITIONS				
(ATHS)				
(H1)				
(H2)				
ננאז				
- LONG TERM TOTAL LOSS OF LHSI PUMPS				
(H4)				





CSS ALLOHING LONG TERM MUTUAL BACK-UP



FIG.3. RCS overpressure protection.

- o Table 2 lists the complementary design conditions, and Table 3 gives the corresponding design improvements.
- o Among these improvements is the newly implemented overpressure protection system, which ensures bleeding of the RCS in the H2 procedure (RCS bleed/feed following total loss of feedwater in steam generator).

The system was originally designed to answer the post-TMI concern on the reliability of the safety or relief valves.

It is installed on the pressurizer and comprises three discharge lines, each one equipped with a tandem of pilot-operated safety valves (Fig. 3). Each tandem is composed of a safety valve, closed at operating pressure, which opens in case of overpressure, and an isolation valve in series, open at operating pressure, which closes if the RCS depressurizes. All three lines participate to the overpressure protection ; one line, in addition, ensures the pressure control at a lower set pressure. The main safety benefits are to :

- ensure stable operation without risk of valve chatter for any type of discharge flow condition (Steam, 2-phase, Water),
- provide capability of remote manual opening under post-accident conditions using safety-grade equipment ;
- ensure reliability of valve reclosing (avoid the stuck-open relief valve);
- improve accuracy of set point adjustment and provide capability to periodically verify setpoints and valve operability without valve dismounting;
- maintain valve leaktightness even in case of reduced margin to the trip-point.

## ENHANCED SAFETY THROUGH CONSIDERATION OF SEVERE ACCIDENTS :

o The defense-in-depth concept, which has governed the design of the safety systems, has been reinforced by the consideration of total loss of redundant systems, as described earlier.

A severe accident, i.e. an accident which would involve severe core degradation, can therefore only occur if :

- a wrong mapagement of the accident is made, for example errors of diagnosis performed by the operating team, or use of a wrong accident procedure, or :
- the accident is followed by multiple concurrent safety system failure, beyond what has been considered in the design.
- o At this point, it is still possible to perform <u>prevention</u> of severe core degradation. Two measures are implemented :
  - the development of a complete set of accident procedures based on the "physical states approach" or "symptom oriented approach", in place of the event-oriented procedures which are presently implemented on French plants.

The physical states approach implies :

- o the diagnosis of states based on a survey of the parameters used for the different systems (primary circuit, secondary circuit, containment and safeguard systems),
- o the identification of operator actions as individual objectives (residual heat removal, restoration of the water inventory, subcriticality..);
- o the prioritization, for each stage, of the objectives and immediate actions.

With such a set of procedures, the operating team should avoid diagnosis errors, and always perform actions which are appropriate to the cooling state of the reactor.

- The provision for bringing on site additional safety features, to cope with successive failures of on-site means which would occur within several days or weeks. The U3 procedure is developed to be able to connect additional mobile pumps and heat exchanger in order to restore (or increase the redundancy of) heat removal at medium term.
- o Finally, in the unlikely case where all the above-mentioned measures would have been inefficient, the <u>mitigation</u> of the consequences of a core melt is considered.

The reactor is provided with a large, dry, non-compartmentalized double containment.

The susceptibility for a short-term failure of such a containment is considered low. Present R & D efforts, in France and worldwide, on phenomena like steam explosion, direct containment heating, or hydrogen deflagration, should confirm this statement.

To cope with the random failure of a containment penetration following a severe hypothetical accident in the reactor building, or of a safeguard circuit carrying bighly contaminated water outside the containment, a special procedure designated U2 has been developed with the aim of pinpointing and sealing off the leak, and subsequently to provide for re-injection where necessary of the contaminated water recovered back towards the reactor building.

- o Concerning delayed conceinment failure, two phenomena may be involved :
  - the first one is a loss of containment leaktightness due to overpressure.
    Pressure rise would result from the formation of non-condensibles, mainly
    CO and CO<sub>2</sub> due to the attack of the raft concrete by the corium,
    accompanied by more or less extensive vaporisation of water depending on
    the acenaric envisaged.

Due to the fact that the containment integrity would be progressively reduced by the coming our of flaws and that its ultimate resistance being exceeded the time to lose the containment integrity therefore varies from one to several days depending on the assumptions made. This process gives the operator time to take action to prevent containment failure with the best control of radioactive releases. This action, formalized by the U5 procedure, involves limiting the pressure increase inside the containment by a rustic filtered venting system, composed of a sand bed filter caisson connected to the effluents discharge stack. The system is actuated manually by opening two remote manual isolation valves allowing the venting of the containment atmosphere into the filter. It permits to reduce the source term which would result from the containment failure, at least by factor 10, and therefore makes it compatible with feasible offsite emergency planning.

- The second phenomenon would involve basement melt through due to the attack of the concrete by the corium. Design features incorporated into the plant (no drains in the middle portion of the reactor building raft) will prevent early contact between the corium and the outside. Should be the raft be penetrated, after approximately six days, and allowing for radioactive decay and the ground retention factor, the immediate radiological consequences would remain small having regard to the probability of occurence of this event and the measures which have been taken to implement internal or external emergency plans.

## IV. CONCLUSION :

These examples, chosen among the great diversity of advanced technologies resulting of the French experience, and the safety approach considered, will be in operations at CHOOZ B1 on the FRAMATOME 1500 MWe model (N4 Model) at the beginning of the 1990's and later on on all other units foreseen in FRANCE until the beginning of the next century. Not dependent of the size of the PWR model, these concepts could be integrated in an advanced reactor able to fulfill the needs of the developing countries for the 1990's. T. PEDERSEN ABB Atom AB, Västerås, Sweden

### Abstract

Design requirements basically fall into three categories. The first comprises the requirements set forth by the regulatory bodies, and forms a mandatory formal base level that must be complied with. The second encompasses requirements that originate from the utilities and potential plant owners/operators, and these normally represent more detailed and practical viewpoints to be used as design goals. The third and last category consists of requirements that are generated by the designers in order to ensure that the results of the detailed design work will meet the requirements of the two first categories. Examples on such designers' requirements are practical rules for general layout arrangement, installation of components, physical separation of redundant safety-related equipment, radiation protection and selection of materials.

If these design requirements could be fixed once and for all, life would be easy for the reactor designer, but this is not the case. The established reactor concepts are steadily challenged by more stringent and supplementary requirements, a shift in emphasis from mitigation to prevention and protection, and calls for simplification and cost reductions, and so it is necessary to develop new, advanced reactor concepts.

For these reasons, ABB Atom is developing advanced reactor concepts, along two different lines: - a new generation of its BWR concept, the BWR 90, and a PWR type of reactors, the PIUS/SECURE family. The latter comprises two basic versions, - the SECURE reactor for generation of heat only (at about 150 °C), and the PIUS reactor for generation of electricity or for cogeneration of heat and electricity.

The paper discusses some important current design requirements and some trends, under discussion in industrialized countries, that would have significance also for developing countries. Design features of the ABB Atom advanced reactor concepts are also presented and discussed.

1. Introduction

Looking at the world and the energy supply on a global basis it is easily concluded that nuclear energy represents a very important option for the future. In order to keep this option open it is imperative, however, that nuclear energy remains an acceptable energy source in the industrialized countries. To this end it is absolutely necessary that severe reactor accidents be prevented, as an accident anywhere in the world would have serious impact on the public perception of the acceptability of nuclear power throughout the world.

Quite clearly, nuclear risks have a special position in the minds of the public. The memory of the TMI accident nine years ago is still on everybodys mind, even thogh nobody was killed, whereas an aircraft crash with hundreds of deaths

results in headlines in news media, but will soon be forgotten by the public. The more recent Chernobyl accident has served as a final confirmation of the unacceptability of nuclear power for doubters and opponents, and it seriously shook the confidence of those who had remained open-minded. The fear of nuclear radiation is so strong that the mere possibility of major accidents is seen by many, perhaps a majority in some countries, as a sufficient reason for abandoning the nuclear option. Large efforts are consequently needed to restore public confidence in nuclear power technology, i.e. to show that it is safe, economical and with a smaller environmental impact than any other available large scale energy source.

As a consequence, it may be assumed that design requirements for new nuclear power plants will be more stringent and address also concerns about situations that are beyond the Design Basis Events being stipulated today, independent of where the plant is to be built. A discussion on whether the design requirements that are, or will be, applied in an industrialized country should be applied also in a less developed country thus seems quite meaningless. The formal requirements should beyond any doubt be equivalent, but some supplementary user-oriented functional requirements would most probably be beneficial.

## 2. The current situation regarding safety requirements

The formal requirements on early nuclear power plants were rather lax, merely describing general principles for ensuring the nuclear safety, e.g. the USNRC General Design Criteria and the "defense in depth" principle. Later on the regulatory bodies have responded to new safety concerns by issuing new requirements and guidelines. Over the last decades these escalating formal safety requirements have resulted in a plethora of added-on safety features in current LWR plants, and this has increased the costs of nuclear power plants. Besides, it implies that nuclear safety tends to be relying on complex interactions of a multiplicity of systems and equipment, leaving ample room for human mistakes.

In the current generation of LWR plants the nuclear safety, primarily protection of the public and the environment against "serious reactor accidents", is based on a combination of inherent characteristics and active and passive engineered safety features. An example of an inherent characteristic is the self-stabilizing of the core achieved by negative reactivity coefficients for moderator temperature, void and fuel temperature, as opposed to e.g. the positive void coefficient of the Russian graphite moderated RBMK reactor (cf. Chernobyl). In order to ensure protection of the reactor core against damagw by overheating it is necessary, however, to have engineered safety systems, e.g. for core cooling and residual heat removal.

These engineered systems, which normally are active systems, are designed with utmost care, with strict quality requirements and redundancy, installation with physical separation, etc., and they should thus have a high functional reliability.

The term "serious reactor accidents" may be given different interpretations depending upon how risks from radiation are judged in comparison with other risks. Whatever the definition, however, modern power reactors which have not suffered significant damage to their cores will in practice contain fairly small amounts of radioactive matter in their coolant circuit, and most of it will furthermore be nonvolatile. Accidents involving leaks or loss of coolant therefore produce relatively trivial environmental radiological consequences as long as the core remains intact. Major releases with prompt or significant delayed health effects must be preceded by core overheating or melting (in short core degradation). Prevention of core degradation thus is the primary goal for the nuclear safety -avoiding serious reactor accidents. Current BWR and PWR plants are designed to withstand a spectrum of malfunctions, incidents and accidents, socalled Design Basis Events (DBEs), and various analysis methods are used to verify that no serious reactor accident will be the outcome of such events. One tool that is often used in evaluations of nuclear safety, is probabilistic calculations, - Probabilistic Safety Assessments (PSAs). The results from these generally show that the probability of a core damage is low, and that even if it were to occur, the containment would in most cases be able to mitigate the off-site consequences.

In some countries the design requirements from the regulatory bodies have to some extent been extended beyond the spectrum of Design Basis Events to cover also situations involving core melt. Examples in this area are the stipulations for the FILTRA and the FILTRA-MVSS installations at the Swedish Nuclear Power Plants, the new Finnish Design Criteria, and studies conducted in e.g. the Federal Republic of Germany, and France.

Engineers who are familiar with nuclear power plant design, may agree that the safety level is satisfactorily high in most of the plants that are in operation today, provided that they are operated by well educated and dedicated personnel. On the other hand, most engineers may also agree that it may be impossible to eliminate the risk of component failures or malfunctions, and perhaps even more important to prevent operator errors. The possible safety implications of such failures of components and erroneous operation cannot be neglected when the nuclear safety is evaluated.

All in all, the entire nuclear industry is convinced that the current generation of BWR and PWR plants is safe compared to both other energy sources and to other human activities. There are clouds on the sky, however. In some countries the nuclear option has more or less been ruled out because of political opposition. In other countries the licensing process has become so complicated that the question of nuclear safety is a topic which has moved from the control room of the plant into the court rooms.

Is it possible to reverse this trend? Several attempts have been made to convince the general public by means of probabilistic arguments, but more or less in vain. "If the risk of a core damage is that low, why TMI and Chernobyl?" The distinction between the design of these reactors, and especially the Chernobyl reactor, relative to the design of most current BWR and PWR plants is very clear to the nuclear industry and its associates, but it is rather difficult to explain to a member of the general public.

The question above from the general public has some significance for the nuclear industry, however, as there is one common aspect of the accident sequences at TMI and Chernobyl (in spite of the very large design differences). This common aspect is that the operators during (or before) the accident more or less advertently prevented the function of safety systems.

This aspect of operator actions is in PRA (Probabilistic Risk Analysis) terminology called "comission" errors as opposed to "omission" errors (when the operator doesn't when supposed to do so). Some types of comission errors are included in the PSAs, e.g. the closing of manual service valves in safety systems, but advertent operator actions (e.g. based on erroneous instrumentation) resulting in shut-off of safety systems are not included.

In the current generation of BWR and PWR plants two methods are generally utilized to prevent human errors, by providing interlocks against the shutting off of safety systems, and by providing autonomous automatic operation of safety systems in order to give the operators a lead time before their action is required, i.e. enabling them to review the situation and check instructions before they act. (An example on this is the 30 minutes rule adapted in Sweden, originated in the 1960s as a designer's rule, and the FRG.)

These two mitigative methods will obviously reduce the risk of comission errors. It can be argued, however, that interlocks will only provide a longer time before a comission may occur; they cannot prevent the occurrence. If the operators are conviced that a safety system in a given situation should be shut off, they will also find a way of doing so, perhaps not from the control room, but maybe from a local electrical room.

The autonomous automatic operation of safety systems is an effective means only as long as the accident is "beneath the umbrella of the Design Basis Events", however. In beyond DBE situations it may still be desirable to override the automatic sequences and shut off certain safety systems. This aspect has in fact been included at least in some emergency operating procedures.

## 3. The outlook for the current generation of LWR designs

The cost increases for nuclear power plants which are due partly to the increased number of systems and components and associated increases in building volumes, and partly to long construction periods and associated interest during construction (IDC) costs, have motivated the evolution of larger plant sizes in order to keep the specific costs down.

The long construction periods mentioned above can to some extent be blamed on the complexity of the safety systems and their interactions. The licensing and analysis work takes a long time, but still worse are the delays that in some countries have been caused by public interventions and hearings.

A larger unit does not per se require a larger operating personnel organisation than a smaller one, i.e. the personnel costs per kW decrease and make up for increases in investment (capital) costs. In other words, the effects of cost increases due to new requirements can be reduced by increasing the unit size, but also by reduction of other operation costs. The latter is the reason for the development efforts that are being made in many countries to improve the fuel utilization in high converter reactors.

The large units thus are more or less a must for the current generation of LWR plants for economical reasons. But they also imply significant disturbances on the electric grid operation if they are disconnected from the grid, i.e. they require substantial spinning reserve and backup power sources which add to the costs for the utilities if the grid size is limited. Besides, the capital costs are high, and large efforts are made to ensure an effective physical protection of the plant.

Another problem for the utilities is the large step increase in generating capacity associated with a large unit. If the demand rises slowly, the utility will have excess capacity and less coverage of costs. As a result there is a rather widespread feeling that smaller plant units - in the 600 MWe range - would be a preferable solution, provided that the specific investment costs can be reduced.

In order to re-establish the confidence of the utilities in the viability of nuclear power as an economically and environmentally sound energy source, the nuclear industry must work hard to accomplish plant designs that permit carrying out efficient construction, and which at the same time reduce significantly the safety-related uncertainties that may open up for attacks by opponents, resulting in delays in the project schedule and subsequent cost increases. Restoring of the public acceptance and confidence will require the nuclear industry to take the concerns of the public more seriously and direct development activities towards simplified and more forgiving designs with a much more transparent safety. Today, a number of such activities are also under way - design of simplified reactors, reactors with "passive" safety features, "inherently" safe reactors, a.s.o. - in addition to work on more moderate modifications of the current technology.

### 4. Advanced ABB Atom BWR plant

The nuclear power programme in Sweden has been very successful, with very good operating experience of the nuclear power plants. In spite of this the politicians have decided that all nuclear power plants shall be phased out by the year 2010, following a referendum in 1980 on the future of nuclear energy in the wake of the TMI accident.

As a result of the political decisions there are no near term prospects of new nuclear power plants in Sweden, but despite the grim current outlook for new projects ABB Atom, the only nuclear plant vendor in Sweden, is carrying out quite a lot of work on advanced designs of water-cooled reactors. One line of development is a plant design concept that is designated BWR 90. It is directly based on the present BWR product line, the BWR 75, drawing on the experience from the construction, commissioning and operation of the Forsmark 3 and Oskarshamn 3 nuclear power plants. The merits of this line is quite obviously that it represents a "proven design", but on the other hand the construction of BWR 90 plants is depending on a solution of the public acceptance problem.

These modern 1100 MWe BWR plants, which are the forerunners and models for the 1300 MWe ABWR (Advanced BWR) that has been designed by GE (USA) and Hitachi/Toshiba (Japan), are characterized by glandless internal recirculation pumps with wet motors, fine motion control rod drives, inerted prestressed concrete containment of pressure-suppression type, 4 times 50 % capacity trains of cooling systems and other safety systems, powerful computer systems for manmachine communication and core calculations, well organized layout and installation to ensure good accessibility for maintenance and service, and a modern turbine cycle with feedwater tank (aerator).



Figure 1. Annual capacity factors



Figure 2. Annual occupational radiation exposures

The operating experience with ABB Atom BWR plants, and especially the BWR 75 plants, has been good, extremely good in comparison with world average results. This is illustrated by the diagrams in Figures 1 and 2, showing annual capacity factors and annual occupational exposures.

The goal of the design review is to evaluate possible improvements and simplifications aiming at reduced building volumes and shortened construction time, decreased amount of systems, subsystems and components, as well as measures to simplify the operation, testing and maintenance. In short, an evaluation of possible reductions in the investment cost for a complete nuclear power plant, and possibilities for cutting the costs for operation of the plant, all in order to reduce the kWh cost of the generated electricity as far as possible.

A typical example on modified plant design is the control equipment which is built up around decentralized microcomputers using a multiplexing connection and interface. The control of plant processes and components is predominantly performed via the computer-based work stations, with colour Video Display Units (VDUs), keyboards and tracker balls. This concept allows substantial reduction of space for equipment and subsequently savings in terms of building volumes.

On the other hand, it has also been decided that the new concept, shall take into account even more stringent safety requirements than its predecessor, and include design measures to cope with a "core melt" accident. Recent regulatory developments indicate a need to do so; requirements in this regard are in fact codified in Finland and Sweden, as noted above.

The plant systems and buildings are laid out and designed to satisfy aspects of safety, maintenance and communication in a balanced way. Safety requirements, notably the physical separation of safety-related equipment, have indeed influenced the layout strongly.



Figure 3. BWR 90 - Building arrangement

The buildings have been arranged in such a way that the essentially "nuclear" and safety-related portions of the plant, i.e. the reactor, control and diesel buildings, are situated on one side of a wide communication area, with the "conventional" portions, the turbine and auxiliary systems buildings, on the other (cf. Figure 3). This arrangement is advantageous when building the plant, and also during plant operation, since the conventional part does not interfere with the nuclear part, but still it provides a compact layout with short piping and cabling connections.

The reactor building encloses the primary containment and forms a secondary containment, including a common bottom slab. The building also houses all primary process and service systems for the reactor, such as handling equipment for fuel and main components, fuel pools, reactor water cleanup system and hydraulic emergency systems.

The diesel buildings are located on opposite sides of the reactor building, which provides a high degree of physical protection. They contain the four standby power diesel generators with their auxiliary equipment, pumps and heat exchangers for the safety-related cooling systems, as well as safety-related auxiliary power supply and control equipment.

The indicated layout is associated with a substantial reduction of building volumes as compared with the Forsmark 3 and Oskarshamn 3 design, which in turn implies a significant cost reduction.

The engineered safety systems are characterized by their consistent separation and division into four subsystems. This configuration was adopted in the first Forsmark plants, and the concept has been reconfirmed as constituting an optimal arrangement with respect to safety, layout, testing during normal operation and maintainability.



Figure 4. BWR 90 - Emergency cooling system configuration

Each of the four ECCS divisions is located in its own bay, adjacent to the reactor containment and surrounded by thick concrete walls. This physical separation is maintained all the way to the ultimate heat sink, as illustrated schematically in Figure 4 for the emergency cooling system configuration. As in the case of the emergency cooling systems, the safety-related electrical equipment is also divided into four independent and physically separated parts or subdivisions, and the reactor protection system operates in a two-out-of-four logic.

The BWR 90 containment consists - as in all previous ABB Atom BWR plants - of a cylindrical prestressed concrete structure with an embedded steel liner, as shown in Figure 5. The containment vessel, including the pressure suppression system and other internal structural parts as well as the pools above the containment, forms a monolithic unit which is statically free from the reactor building, apart from the foundation slab.



Figure 5. BWR 90 - Reactor containment

Recent regulatory developments indicate a need to strengthen the capability of the reactor primary containment to withstand the effects of a core melt accident, and in order to achieve enhanced safety during a degraded core accident the following modifications have been made to the containment of the reference plants:

- The openings between drywell and wetwell for steam blowdown to the suppression pool are arranged horizontally
- The relief pipes from the safety/relief valves are drawn into the suppression pool via the lower drywell rather than penetrating the drywell wetwell intermediate floor
- A pit is provided at the bottom section of the lower drywell for the purpose of collecting and confining fuel melt debris. The cylindrical wall and pit form a pool which is filled with water in the event of a severe accident.

These arrangements improve the reliability of the pressure – suppression system and reduce the probability of containment leakage in the event of an accident. In addition, the containment vessel may optionally be vented through an external filter of the type now being installed in all operating Swedish nuclear power plants.

The reactor design has not been changed much. The recirculation system is based on the ABB Atom glandless internal pumps driven by wet motors of the squirrel cage asynchronous type. The motors are supplied individually with "variable frequency-variable voltage" power from frequency converters. This type of pump has been operating reliably (for more than two million operating hours) in ABB Atom plants since 1978.

The internal pumps provide means for rapid and accurate power control in the high power range, and they are advantageous for load following purposes. The plant is characterized by the capability to accept a 10 % step change in power with an equivalent time constant of 15 s with the reactor at constant pressure, or 5 s with floating pressure control. Ramp load changes of 20 % per minute are accepted and useful for all operating plants of BWR 75 model.

The internal recirculation pumps have more than 10 % excess flow rate capacity, and this built-in "redundancy" implies that the reactor can be operated at full power even if one pump should fail.

The load follow characteristics and the capability of operation at full power with one recirculation pump out of operation have been successfully demonstrated in the operating plants.

The control rod and control rod drives for the BWR 90 are of the well-proven ABB Atom design. The cruciform rod is based on a solid steel blade with drilled horizontal holes filled with the  $B_4C$  absorber. In the top the absorber consists of hafnium metal pins making the rod tip more grey and providing a long life.

The control rod drive (CRD) utilizes separate electro-mechanical and hydraulic functions, the former used for normal, continuous, fine motion of the control rod and the latter for fast insertion (scram).

The diversified means of control rod actuation and insertion (together with generous reactor pressure relief capacity), in combination with the capability of rapid reduction in the recirculation flow rate (pump run-back), has led to regulatory acceptance of the system as being a sufficient ATWS measure in all ABB Atom reactors. Thus, the CRD design is "ATWS proof".

The reactor core in Forsmark 3 is composed of 700 fuel assemblies, each consisting of a bundle of 64 fuel rods in a 8 x 8 square lattice pattern surrounded by a fuel box acting as a coolant channel. Some of the fuel rods contain a burnable absorber (Gd<sub>2</sub>O<sub>3</sub>) to suppress excess reactivity, and axial and radial grading of the burnable absorber (BA) content is used for controlling the power distribution, i.e. for minimizing the power peaking. The advanced BA design has significantly reduced the need for control rod displacements during operation, and control rods are always withdrawn or inserted in one predetermined sequence (without swapping) and at full power most rods are fully withdrawn from the core.

In the 1000 MWe BWR 90 the size of the reactor core was reduced to 676 assemblies, based on the continued tuning of the BWR fuel and fuel management. In particular, the SVEA fuel (cf. figure 6) enables a very flat internal power distribution to be achieved. This reduced core has in fact already been demonstrated in the Forsmark 1 and 2 plants, having 676 fuel assembly cores,



originally laid out for producing 940 MWe. In 1985, both units started trial operation at the uprated power level of 1008 MWe.

The standard SVEA fuel assembly design contains four  $4 \times 4$  subassemblies with a cruciform water gap between them, and this water gap significantly increases reactivity and reduces local power and burnup peaking factors. It also contributes to a mechanically favourable fuel channel structure with a very low creep deformation rate and a minimum amount of neutron absorbing Zircaloy. The standard SVEA design yields a substantially improved fuel utilization, and for plants with a great deal of load following and/or higher power densities the combination of SVEA fuel with a liner cladding has become a preferred option, not only in ABB Atom BWR plants.

Still better performance is promised by a recently introduced version having sub-assemblies with the fuel rods arranged in a  $5 \times 5$  lattice. The  $5 \times 5$  version is designed to be hydraulically compatible with other fuel, and this results in a maximum acceptable fuel rod diameter very close to that of a standard 17 x 17 PWR rod.

With as many as 100 rods in the new design the average linear heat rate is reduced by more than one third compared with the standard version. This allows for operation with high peaking factors and eliminates the need for liner cladding. Low fuel temperature and fission gas release, as a result of the low linear heat rate, contribute to a high burnup potential.

The low fission gas release makes it possible to reduce the plenum length and increase the active length of the rods. With a peak linear power of 31 kW/m, which is the currently assumed limit for operation without power ramp rate restrictions, the end of life pressure will be less than 3 MPa at an average burnup of 50 MWd/kgU with an initial helium fill pressure of 0.4 MPa.

Based on the new SVEA design the BWR 90 is now being re-evaluated, considering increased unit sizes, - an 1150 MWe plant with 700 fuel assemblies and an 800 MWe plant with 500 assemblies.

Two important features of the fuel cycle as regards flexibility and cost are spectrum shift and coast down operation.

The core coolant flow range at full power, including excess pump capacity, determines the possible variation of average core void. In the ABB Atom internal recirculation pump reactors the pumps have about 15 % excess flow capacity at full power, which provides a valuable reactivity control amplitude, allows load following operation, and provides a means of affecting the axial power distribution. This flow window at full power is routinely utilized as well as the associated spectral shift effect. This way spectrum shift operation has since long been a standard procedure in ABB Atom BWR plants, which implies significant fuel savings.

A key to modern process communication applied to the BWR 90 is the use of microcomputers for process control. Process communication from the control room is realized by means of distributed functional processors. These in turn interact via serial communication links with a number of object-oriented process interface units.

The functional processors are generally arranged as a dual system, with the two processors receiving the same process information, so that the standby unit may take over the control functions automatically in the event of a fault in the operative unit.

The arrangement satisfies the requirements of redundancy and physical separation. It includes intelligent self-monitoring of protective circuits.

The decentralized configuration, combined with the use of isolation devices, reduces the safety concern of a damaged control room. If the control room should become unavailable, the local microprocessors will take control, and the operating personnel may supervise the shutdown and decay heat removal from a separate emergency monitoring centre.



POD Plant Operation Desk ROO Reactor Operation Desk TOD Turbine Operation Desk AOD Auxiliary Systems Operation Desk OIP Overall Indicating Panel

Figure 7. BWR 90 - Control room arrangement

The man-machine communication in the control room is facilitated by the consistent use of video display units (VDU), keyboards, and display maps. The control room contains several work stations, the reactor operation desk, the Balance of plant operation desk, the turbine plant operation desk, and the supervisor's desk (cf. Figure 7).

Each work station in the control room is equippped with three VDUs. Typically, one VDU will display a total view of the process of interest, another will provide a list of alarms, and a third VDU will display a diagram with sufficient detail to facilitate operator action.

An "overview" of plant functions and status is provided by a special overview panel, which may contain conventional instruments as well as computerbased CRT displays (CRT projections or EL displays).

The electrical power systems for safety-related objects are strictly divided into four separated sub-distributions, as already implemented in the operating BWR 75 plants, but some simplification have been made.

The design of the process systems in the BWR 90 has reduced the ratings of some of the major loads, and the introduction of static power supply converters for the feedwater pumps has reduced the requirements on minimum short circuit power on the busbar system significantly. Modern switchgear components, having higher short circuit ratings, have made a significant simplification of the structure of the auxiliary power systems possible.

It can also be noted that DC distributions at several voltage levels for power supply to control equipment has been replaced by power supply from the battery-backed AC distribution, using distributed AC/DC converters for the supply to the various types of equipment.

Experience has shown that short construction periods are very important if the total plant capital cost shall be kept low and the records from the construction of the two latest BWR 75 units have been scrutinized thoroughly to evaluate the potential for savings. The construction period of Oskarshamn 3 was only 57 months, which is really very short, but a careful evaluation of project planning indicates that a further reduction to 54 months should be practicable.

The new layout saves some 150 000  $m^3$  of building volume and represents a significant cost reduction. Cost reductions should also be attainable in each of the areas of control equipment, auxiliary power supply plant and ventilation systems. The "leak-before-break" criteria imply that the number of pipe whip restraints can be reduced, i.e. the cost will decrease. The cost reductions have been estimated to be in the order of \$ 55 Million for a 1000 MWe plant.

BWR 90 represents a natural step in the series of reactor developments by ABB Atom. This series has been characterized by designs which from the beginning have contained features well in the front of the regulatory and operational requirements. The plants have also shown to be fairly easy to adjust to new requirements. The base for the BWR 90 development work has been a review and re-evaluation of the current design solutions. All well-proven design features from the operating plants have been recognized, and only small modifications to adapt to new or anticipated safety criteria, to apply modern technology, and to simplify system designs and operating procedures have been introduced. All design changes are based on consistent cost benefit considerations to comply with the aim of making the plant as economical as possible.

Thus, compared to the previous ABB Atom BWR 75 the BWR 90 has significantly reduced building volumes, shorter construction period, and decreased

amounts of systems and components, and includes measures for simplified operation, testing and maintenance, i.e. the costs will be lowered, and the plant operation more simple.

For possible new 700 or 1000 MWe nuclear plant projects in the future the BWR 90 will really represent a very advanced boiling water reactor plant, closely based on the excellently operating ABB Atom BWRs (the models for the ABWR now being introduced by GE, HITACHI, and TOSHIBA), and designed to cope even with "core melt" accidents. The latter implies that with the BWR 90 design concept the public and the environment will be protected even in severe accident situations.

## 5. Safety requirements and design goals for future plants

Recent experience has demonstrated that nuclear power, if improperly handled, can be an hazardous technology. Release of a significant fraction of the nuclide inventory in a large power reactor following a core degradation accident can have uniquely harmful consequences. In view of the deep-rooted public fear of radiation there must be correspondingly uniquely strong guarantees that accidents leading to such releases will not occur.

The current LWR technology is widely perceived as not providing sufficiently convincing such guarantees. In many countries the reactor safety issue therefore now represents a strong impediment to the future rational use of the nuclear option. Mere administrative reforms and passage of time may not be sufficient to remedy this situation. A new, more convincing approach to reactor safety may be needed to break the impasse.

One important question mark related to the future role of advanced reactor concepts such as the BWR 90, is the public acceptance problem, since they are designed in the same manner as units that are already in operation, and that are the objects of the current public distrust. The technical community may be convinced that the design and the installation represent the highest "conceivable" safety level, and that the design measures introduced to handle a molten core would effectively eliminate significant ground level releases of radioactivity, even in the very unlikely course of events leading to core degradation.

This may also be accepted by the safety authorities and the utility people, but the acceptance of the public is more questionnable. The small remaining risk, and the risk of unforeseen events outside the traditional "Design Bases Events" umbrella, may just as well be taken as a proof of an unsafe installation, as noted before. A key question here may be "Is an emergency evacuation plan involving exercises needed?" If the answer is negative, preferably for reasons understandable to the lay public, this public may possibly be made to accept the design as basically safe.

From the utility point of view there is also a question mark related to the economic risk involved with a "degraded core" accident; will they be convinced that the "Design Basis Events" umbrella covers all important conceivable events, and that they are not just the formal requirements and stipulations from a regulatory body?

The question above on emergency planning may seem puzzling, but it clearly has a significance. The discussions and problems related to the Shoreham and Seabrook plants in the US are well known, and in Italy a lot of the opposition against nuclear has focussed on the need for emergency planning. In a recent "Policy Issue" document (SECY-88-203) the USNRC Staff presents views and suggestions related to "Key Licensing Issues Associated with Advanced Reactor Designs" as regards criteria for assessing the advanced reactor concepts in the areas of:

- accident selection
- siting source term calculation and use
- adequacy of containment system
- adequacy of offsite emergency planning.

The document is restricted to assessment of the modular high-temperature Gas-cooled Reactor (MHTGR), and the small liquid sodium cooled breeder reactors (PRISM and SAFR), and does not address Advanced Water-cooled Reactors. The general discussion and suggestions may very well be applied also to water-cooled reactors, however.

The Advanced Reactor Policy, which is referenced by the Staff, states that advanced reactors must, as a minimum, provide the same degree of protection of the public and the environment that is required for current generation LWR plants. In this regard, the staff has interpreted current generation LWR plants to be those evolutionary designs currently under review as standard plant designs (ABWR/APWR). The policy further states that the Commission expects advanced designs to provide enhanced margins of safety.

The defense in depth in nuclear safety regulation is a philosophy that entails use of various layers of requirements to help ensure that safety is achieved through mutiple, diverse and complimentary means.

These layers of requirements address the different stages and aspects of plant safety which can be generally categorized as prevention, protection, mitigation, and emergency planning.

The advanced reactor designs discussed by the Staff do maintain this "four category" defense in depth, but they generally make a shift in emphasis from the mitigation features, that are so important in current LWR plants, to highly reliable protection features.

The staff states that for advanced reactors it is considered necessary to select a spectrum of accidents, which must be considered in the design, beyond the traditional LWR design basis accident (DBA) envelope,

- to ensure advanced designs comply with the Safety Goal and Severe Accident Policies
- to provide a sufficient test of the capability of the design to allow use of mechanistic source terms for siting determinations and for decisions regarding containment and emergency evacuation plans, and
- to ensure the shift in emphasis in defense in depth from accident mitigation to protection, as compared with LWR plants, doesin fact provide designs with safety at least equivalent to that of current generation LWR plants.

To this end, the Staff has proposed the following event categories to be defined and evaluated:

- Event category I, corresponding to the current Anticipated Operational Occurrences (AOOs) class of events considered in LWR plants. The frequency range for these events goes down to approximately 10<sup>-2</sup> per year. The events shall be analyzed similar to what is done for LWR plants.

- Event category II, corresponding to the current DBA category for LWR plants. It will include internal events down to a frequency of approximately 10<sup>-4</sup> per year, and a traditional selection of external events. The events shall be analyzed similar to what is done for LWR plants.
- Event category III, corresponding to those severe events, beyond the traditional DBA envelope, which should be used by the designers in establishing the design bases for their designs. This category would include internal events (less likely events plus multiple failure events) down to a frequency of approximately 10<sup>-7</sup> per year, external events beyond those included in EC-II, consistent with their application to future LWR plants, and, using engineering judgment, additional bounding events to account for plant specific uncertainties. These events could be analyzed on a best estimate basis.

There is also a fourth category – Event category IV – which is intended to be used in the assessment of the need for offsite emergency planning. It includes internal events of similar frequency to those events considered in the basis for the emergency planning zones and requirements for LWR plants (described in NUREG-0396). These events would be analyzed in a PRA (Probabilistic Risk Analysis).

In order for the NRC to accept that no traditional offsite emergency planning (other than simple notification) be needed for advanced reactor designs, the Staff proposes the following:

"While an offsite emergency plan would still be required, such a plan would not have to include early notification, detailed evacuation planning, and provisions for exercising the plan if:

- the lower level Protective Action Guidelines (PGAs) of the Environmental Protection Agency (EPA), which stipulates a site boundary whole-body dose of 1 Rem and thyroid dose of 5 Rem, are not predicted to be exceeded within the first 36 hours following any event in categories EC-I, EC-II, and EC-III, and
- a PRA for the plant, including at least all events in categories EC-I through EC-IV, indicates that the cumulative mean value frequency of exceeding the lower level PGAs within the first 36 hours does not exceed approximately 10<sup>-6</sup> per year."

In this context it may also be noted that in Italy they are contemplating a time limit of 5 days, compared to the 36 hours stipulated by the NRC Staff. Consequently, this type of requirements must be taken seriously by both vendors and utilities, as well regulatory bodies.

As noted above, Probabilistic Safety Assessments (PSAs) are often used as a tool in safety evaluations to show that the design of a nuclear plant is sound, and that the safety level will be high. One obvious shortcoming of the PSA methodology is the limitation in considering the socalled "comission" errors, which do represent a significant risk factor in todays nuclear power plants. In order to **prevent** comission errors it is necessary to design a new reactor system in such a way that it will be neither necessary nor possible to ever shut off essential safety functions. It should also be possible to demonstrate convincingly such features for interested members of the general public. This reactor system should of course be designed to withstand the whole spectrum of DBEs that are set forth for the current BWR and PWR plants, but it should furthermore preferably be able to withstand also extreme events which are beyond the DBEs of today. It would obviously be advantageous if this reactor could be based on conventional LWR technology, and use their fuel for which an enormous manufacturing knowhow and operating experience are available. Such a reactor should be able to improve the public acceptance of nuclear power in the industrialized countries, and it should also make the nuclear option more attractive for less developed countries, where energy sources are scarce, but where the technical infrastructure is not yet developed enough to ensure that a conventional LWR plant of current design can be safely built and operated.

Together with the interest in smaller economical units, expressed by many utilities, these aspects are general incentives for the development work on new reactor types, even though all the aspects are not specifically applied for all concepts.

## 6. The PIUS type reactors

Existing LWR technology can be adapted and modified to provide the requisite safety assurance in a simple, easily comprehended way, and to eliminate the need for complex plant design originating from safety considerations. An example on this is the PIUS/SECURE reactors in which the prime nuclear safety goal, protection of the core against significant damage, is based on simple, immutable natural laws, the Process Inherent Ultimate Safety (PIUS) principle. Apart from the design features that are required to meet these design goals, the PIUS/SECURE reactors are rather similar to current LWR (PWR) plant designs. ABB Atom has been working on these reactor design concepts for more than a decade.

It should be noted here that ABB Atom is responsible for the LWR program within the ABB Group of companies. For the PIUS work it can draw on its own experience which in addition to BWR knowhow also encompasses PWR fuel fabrication and design, as well as on the experience of ABB Reaktor, formerly BBR, who built the Mülheim Kährlich PWR in Germany.

In the development objectives for the PIUS reactor of ABB Atom all the abovementioned aspects have been included as design prerequisites, and the safety objectives can in short be described in the following way.

There shall be no credible paths to core degradation accidents in spite of undisputably pessimistic assumptions, and in line with this, the following design assumptions have been formulated regarding protection of the core:

- The plant design shall be forgiving, even if the operators make mistakes in emergency situations
- Function of active equipment (pumps, valves, etc.) is not credited in emergencies
- Load carrying structural members can fail at any time
- The plant may be subject to deliberate destructive intervention (obviously with some restrictions)
- Outside intervention to ensure core cooling can be provided after a post accident "grace period" of one week.

These design assumptions strongly restrict the choice of design solutions. The reactor system design and heat extraction process must clearly be such that, following any credible incident or failure, whether it involves equipment breakdowns, operator mistakes or intentional destructive acts, the system reverts by itself, without reliance on acts of humans or on activation of equipment, to a state with assured passive long term core cooling. Active control measures are needed to keep the systems in operation, but they must always by overruled by ever-present natural forces under abnormal conditions before core damage can occur.

To prevent core degradation accidents compliance with the following two criteria is required in PIUS plants:

- The core shall be submerged in water at all times
- The heat generation rate of the core must not exceed the cooling capability of the submerging water in order to avoid dryout (DNB).

The only practical way of complying with the first condition is to place the core at a low level in a large pool of water, kept in place by a multibarrier prestressed concrete vessel, the integrity of which is ensured by a large number of independent tendons. The water volume must be sufficient to remove, by its evaporation, the core decay heat for the whole of the "grace period". This water must contain a neutron poison (boric acid) to keep the reactor subcritical, since reliance on control rod insertion does not meet the design assumptions.

At the end of the one week grace period the reactor pool must be refilled with water, from internal or external sources, using on-site or off-site (mobile) power supplies. A number of pumps and water sources in the plant can be used for this purpose, but water supply by means of a fire truck is also possible.

A core placed at the bottom of a large pool of water containing boric acid is an arrangement of no interest as such. A circuit for extracting useful heat from the core is needed, but this must be done without in any way blocking the core off from the water in the pool by means of valves, etc.

How this is achieved is easily understood from figures 8A - 8D.

In figure 8A a heat source (a reactor core) is placed near the lower end of a pipe (riser) in a pool of water. The heat will cause a natural circulation flow up through the pipe.

In figure 8B the flow is by means of a pump returned back to the inlet of the pipe again, instead of emerging into the pool. If the pump flow rate matches the natural circulation flow rate up the riser, as determined by the thermal and hydraulic conditions, there will be no exchange of water with the surrounding



Figure 8. The operating principle of PIUS

pool. A separated primary circuit with permanent openings to the pool has thus been accomplished by means of the pump work in an operational equilibrium state, and the heat generated by the core now remains in the circulating water. Quite obviously, a loss of pump power will make the system revert to its natural equilibrium state with the core being cooled in a natural circulation mode via the upper and lower riser pipe openings to the pool.

In figure 8C a heat exchanger has been added to the recirculation system to keep its temperature at a constant level. The heat generated by the core is now extracted for a useful purpose. The upper part of the riser is bent downwards to permit stable layering of hot water above cold at both ends, in socalled density locks. A pressurizer steam bubble is also added so that the heat extraction can take place at elevated temperature, e.g. for steam generation.

The interface between hot and cold water in the lower density lock is kept at a rather constant position by small adjustments in the pump flow rate in correspondance with the temperature difference between the riser and pool water. The normal pump speed control range is rather limited, particularly with respect to higher than normal speeds. By design measures, the maximum pump speed is limited to a level slightly higher than that corresponding to full reactor power.

The reactivity and power output of the reactor core can be controlled by means of changes in boron content and coolant temperature, while there is always an open natural circulation path through the core and the pool.

This arrangement fulfills the design criteria given above. Keeping the system in operation requires a rather accurate flow control, which however presents no problem during normal operating conditions as the core outlet water temperature is maintained constant.

In major transients of potential safety importance the control system can no longer cope with the forces imposed on the coolant by the laws of thermohydraulics and gravity, and borated pool water enters the primary coolant system through the density locks and causes reactor shut down or stabilization of core heat output at a safe level.

In figure 8D the system response in an accident situation is illustrated. Upon loss of the heat sink and failure of control systems to reduce the reactor power, and failure of protection circuits to initiate and achieve a reactor scram, the primary circuit coolant heats up. The coolant pump which is speed controlled in accordance with temperature measurements in the lower density lock (in order to maintain constant condition), will speed up to some extent, and then peak. The primary coolant core outlet temperature increases to boiling, and the void content in the riser water yields a rapidly increasing buoyancy relative to the pool water. The result is a rapid increase in the riser flow rate, and pool water will be sucked into the primary circuit, shutting down the reactor.

This is a typical example on system response to incidents and accidents in the plant; the outcome of the analyses that have been made is the same -the reactor is shut down and cooled, or the reactor core output is stabilized at a safe level, without risk of core damage.

The PIUS/SECURE reactor family comprises two basic versions, the SECURE (previously called SECURE-H) for district heating purposes and the PIUS (previously called SECURE-P) for generation of electricity or for cogeneration of electricity and heat. The former is now offered by ABB Atom on normal commercial terms, whereas the latter is still under development.

The SECURE reactor is designed for supply of 150 °C water (or steam) to a heat distribution system, and operates at 2.0 MPa with a core coolant outlet temperature of 190 °C. The power density of the core is extremely low, only about 15 kW/kg U, which means that the risk of fuel failures will be negligible. The PIUS reactor is more similar to other power reactors, supplying 270 °C steam at 4.0 MPa to a turbine plant. The operating pressure of the reactor system is 9.0 MPa, and the core coolant outlet temperature is 290 °C. The core power density is less than 25 kW/kg U, and the average fuel rod linear heat rating is 11.9 kw/m, a very low value.

Apart from the obvious differences in reactor data, depending on the intended application, the same basic principles are generally applicable to both versions. In the following the description is limited to the PIUS reactor.

The heat exchanger and the main coolant pump of the primary circuit can in principle be located inside the concrete vessel or outside it. The latter arrangement has formed the basis for the design of the SECURE, while in-vessel location of pump and steam generator was the solution originally selected for the PIUS, as described in many publications.

A design study of a PIUS plant with external pump and steam generator loops was launched in 1987 in a joint effort by ABB Atom and Ansaldo Spa of Italy, utilizing proven components and technology as far as possible, with the aim of reducing the amount of component and equipment development and verification, i.e. to make an early commercial introduction possible.

Figure 9 shows a cross section through the reactor building of a 600 to 650 MWe plant with the concrete vessel and its internals, the out-of-vessel steam generators and coolant pumps and the reactor containment.



Figure 9. PIUS 600 - Reactor building section

The 2000 MWth core, located in the lower part of the concrete vessel is probably the simplest water reactor core ever designed. It consists of 213 fuel assemblies very similar to standard PWR assemblies but simpler, because of the absence of control rod guide thimbles (PIUS does not use control rods), and shorter (active height 2.5 m). The 316 fuel rods in each assembly, arranged in an 18 x 18 square array are of normal PWR diameter (9.5 mm).

The core data are significantly relaxed in comparison with current PWR practice in terms of linear heat load, temperatures and flow rates. Burnup reactivity compensation is mainly achieved by means of gadolinia, resulting in a low coolant boron content and consequently a strongly negative moderator temperature coefficient of reactivity throughout an operating cycle.

From the core the 13000 kg/s coolant flow passes up through the riser pipe, leaves the vessel through nozzles in the upper steel extension and continues in the hot leg coolant pipes to the upper end of the four straight tube once-through steam generators. These are of the type used in the Mülheim-Kährlich plant in Germany although much smaller.

The main coolant pumps, that are placed below the steam generators, are sized-up versions of the wet motor design used in the older ABB Atom BWR plants that have given excellent, trouble free service for many years.

The cold leg piping enters the steel extension at the same level as the hot leg nozzles and the  $260^{\circ}$ C return flow from the steam generators is sent downwards to the reactor via the annular downcomer outside the riser. On its way down it is accelerated to 15.6 m/s at a level slightly below the main coolant piping connection so that the static pressure decreases to slightly above that in the riser. At that point there is an open connection between downcomer and riser, the socalled siphon breaker, that prevents siphoning off of the pool water inventory in case of an hypothetical pipe rupture in the cold leg. The kinetic energy in the high velocity downcomer water is recovered in a diffuser.

The pressurized steam bubble is located where it should rightly be in all pressurized water systems, namely at the highest point, under the steel extension head.

The always open natural circulation path through the core goes from the pool to the lower "density lock" (thermal barrier) below the core, through the core itself, the riser, an annular passage outside the upper part of the downcomer and the upper "density lock" back to the pool.

The primary system is thermally insulated from the pool by a layer of stagnant water, held in place by thin stainless steel sheets.

The prestressed concrete vessel cavity has a diameter of 13.4 m and height of 33 m and contains totally  $3750 \text{ m}^3$  of water. To the top of the concrete monolith the steel vessel extension mentioned above is fixed by means of special tendons, the lower ends of which are anchored to the outside of the vessel.

Outside the core the concrete vessel can house spent fuel from a full service life of 40 years, if desired. Refuelling is made after removal of the components above the core in analogy with current LWR practice.

The "inherent" core safety of this design, i.e. the absence of core damage in the complete lack of intervention by surveillance and protection systems, has been verified for all the relevant transients such as loss of feedwater, inadvertent reactivity insertion, primary pipe breaks, secondary steam line breaks etc. Coolant pump trip causes immediate reactor shut down and a substitution of process water with high-boron pool water throughout the primary system.

This "selfprotective thermohydraulics" of the primary system has of course nothing to do with normal operation of the plant which is taken care of by ordinary control systems. It only serves as a "safety net", always there if everything else fails.

In the absence of core damage under any credible conditions there should be little incentive for investing in a full-fledged containment for protection against the rather trivial releases that can occur e.g. in a pipe rupture incident. However, to comply with the present regulatory requirements on extremely low releases of radioactivity in any accident situations, such a containment is necessary. A pressure suppression type of containment built to resist the impact of a crashing aircraft, is provided as shown in figure 9 with the concrete vessel and the external primary circuit in the drywell.

The absence of control rods might be assumed to result in a very sluggish manoeuverability of the reactor. In fact the opposite is the case. The combination of once-through steam generators and a strongly negative moderator temperature coefficient of reactivity makes possible simple operation with elegant respnse to load demand variations. Plant power in the short time scale is essentially controlled by means of the admission valves to the main and feedwater pump drive turbines. Longer term reactivity control is by means of boric acid concentration in the coolant.

A sudden increase in grid demand is met by a plant operating at part load simply by increasing feedwater flow and opening the main turbine throttle. The resulting cooling down of the primary system automatically adjusts the reactor power. Daily load following can be made with little or no change in coolant boron content throughout an operating cycle.

Statements regarding PIUS are frequently made to the extent that disturbances in operation must be expected due to inadvertent ingress of borated water through the "density locks". In reality this is no problem, as shown both by test rig operation and theoretical analysis. The plant will withstand grid voltage transients caused by electric faults such as short circuits that are disconnected by the first and second steps of relay protection without this occurring.

Since all the components above the core vessel are removed for refuelling, access for maintenance presents no special problems. The absence of control rods and the ensuing alignment requirements simplifies in-vessel component handling.

The construction time from first pouring of concrete to power operation has been estimated to 49 months.

The design just described has been costed and compared, principally with a standard 700 MWe ABB Atom BWR which is known to be very competitive costwise. The large concrete vessel represents a major investment without counterpart in other designs and also entails some reduction in steam data (4.0 MPa is used) and thermal efficiency due to a lower operating pressure (9.0 MPa) in comparison with a conventional PWR. However, the drastic simplification made possible by the elimination of many redundant, diverse safety systems compensates for this.

The costing analysis of the PIUS 600 (630-650  $MWe_{net}$ ) yields a specific investment cost (including owner's costs and interest during construction) that corresponds closely with that of a 700 MWe BWR.

Due to the somewhat lower thermal efficiency of PIUS the fuel cycle costs will be slightly higher. On the other hand the PIUS plant is more straightforward and simple with less equipment and components, especially safety-related equipment, which will result in a lower cost for operating and maintenance personnel. The resulting life time energy cost corresponds closely to that for the 700 MWe BWR plant.

A 1000 MWe nuclear power plant has some life time energy cost advantage compared with a 700 MWe unit, but according to recent costing analyses this cost advantage is smaller than normally assumed.

For the ABB Atom BWR plants (at a Scandinavian site) the cost difference is in the order of 10 % (8-13 %). This cost advantage for the 1000 MWe unit will not always yield a utility preference over the smaller units, however, taking financing, risk and electric consumption increase rate into account.

Although PIUS is based on conventional LWR technology in most essential respects the plant configuration is sufficiently different to make experimental verification necessary in a number of cases before lead plant final commitment.

The principal novel features are:

- the prestressed concrete vessel
- the water soaked thermal insulation between the hot primary system and the "cold" pool water
- the thermal barriers ("density locks") where hot process water is layered above "cold" pool water
- The dynamics of operation with a contiguous water mass with two different chemical compositions (boron contents) and temperatures and the function of the associated "self-protective thermohydraulics"

The first two items have been adequately covered in a previous large scale international program carried out in Sweden aiming at introducing prestressed concrete pressure vessels for BWRs. The results of this program can be directly applied to PIUS.

An extensive 4 year program has been carried out to study the function of the "density locks", i.e. their ability to prevent transport across them of dissolved material (essentially boric acid) under the influence of turbulent disturbances. Full scale tests at atmospheric pressure combined with single tube tests at full temperatures and pressures have provided convincing evidence that undesirable diffusive transport can be kept to an adequately low level.

By far the most extensive verification effort has involved demonstration of the dynamics and "self protective thermohydraulics" of a complete PIUS type system using a non-nuclear heat source in the shape of a 60 rod fuel assembly heated by up to 40000 amps of DC. The DC supply was programmed to adjust its power to the surrounding coolant in real time as a function of its composition (simulated boron content), temperature and void content just as a nuclear core would do, using point kinetics equations.

Figures 10 and 11 give typical results of the tests with a comparison of values measured and precalculated with the RIGEL code. Figure 10 illustrates load following operation and shows how reactor power adjusts to a temporary reduction in grid demand. The full line represents the computer prediction and the dotted line measured values of power output.

Figure 11 shows what happens at a reactor coolant pump trip. As can be seen the reactor is rapidly shut down and reverts to natural circulation cooling in






Figure 11. Reactor coolant pump trip in test loop Reactor power variation

a transient which in a conventional PWR requires instant control rod insertion to avoid severe consequencies.

The results of this program fully confirm the "self-protective thermohydraulics" of the system and the predictive capability of the RIGEL code specially developed to study PIUS dynamics.

Summarizing, the novel technical features of the PIUS design have been sufficiently studied to eliminate any concerns regarding technical feasibility and practicability of the concept.

The maturity of the PIUS design and verification has reached a stage which makes an order of a full scale lead plant a realistic option within a few years time.

### 7. Conclusion

As noted in the introduction, safe and reliable operation of all the nuclear power plants in the world is a prerequisite for the nuclear energy to remain an option for the future energy supply to a rapidly expanding world population.

The nuclear industry may claim that the current generation of BWR and PWR plants are economically competitive with other large scale energy sources, and also very safe compared to other human activities, but the public perception does not necessarily correspond to these factual statements from the industry side.

Recurrent serious accidents would be totally unacceptable and result in a phase-out and abandoning of nuclear power, and the first commandment to the nuclear industry therefore is to ensure accident free operation of the present fleet of nuclear power plants and those now under construction.

A second commandment is to provide, for the future, reactors that will ensure continued accident free operation, even with a vastly expanded number of nuclear power plants spread all over the world, some of them inevitably operated under more adverse conditions than hitherto considered.

The reduction in risk of "serious reactor accidents" can be used as an ultimate measure on the capability of new concepts to live up to this second commandment, together with the lead times offered for manual corrective actions. In this context, it can be noted that while some new LWR concepts aim at just a gradual improvement of the "ultimate" safety level, i.e. for a certain reduction in core degradation accident risk, other concepts, of which PIUS is one, aim at elimination of core degradation accidents as a practical concern.

As pointed out above the basis of the PIUS development effort is a firm conviction about the need to build a future nuclear expansion on a technology where safety against severe accidents is a built-in feature of the reactor configuration that cannot be compromized by malfunctioning equipment or human intervention. This need seems to be realized in several countries where an evaluation of the PIUS concept is under way or planned, and it should also beyond doubt be taken into consideration in developing countries that are contemplating the introduction of nuclear power.

# THE HIGH TEMPERATURE REACTOR AS A TECHNOLOGY FOR DEVELOPING COUNTRIES

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## Abstract

World population is growing rapidly, 80 million per year. From 5 billion in 1987 to 11 billion beyond the middle of the next century. Most of this growth is taking place in developing countries. Growth of world population increases energy demand until the year 2000 and afterwards. At the same time, ecological stresses on our environment must be limited.

The HTR 500, a power plant for electricity generation with the possibility to extract process steam and district heat, is an important step on the way to commercialization for the high-temperature reactor and is a well-suited answer to all challenges, as it can supply energy under conditions acceptable to the environment and to mankind. It meets the modified demand of the power plant market, which for reasons of grid size, scope of financing and the lower increase in electricity demand internationally, shows a trend towards power units in the size range of 300 to 600 MW.

Electricity growth rates in industrialized countries are lower after the oil price escalations, and in developing countries grid sizes are often too small for the operation of large LWRs. This indicates a potential for medium-sized power reactors such as the HTR 500. Due to the use of HTR-specific characteristics and extensive application of the technology of the THTR 300 reference plant, the HTR 500 can compete with coal-fired plants of comparable size, and is even competitive, considering the electricity generating costs of large LWRs.

In view of the steadily increasing scarcity of energy resources in parallel with a dramatic growth of world population, the HTR 500 with its excellent safety characteristics and environmental compatibility is capable of furnishing an important contribution to the solution of the energy problems in the Third World.

## **KEYWORDS**

High-temperature reactors, energy demand, medium-sized reactors, advantages of HTR 500, developing countries, power plant market.

## 1. Introduction

The development of the world energy consumption has hitherto been characterized by an extreme unbalance with regard to population and standard of living. The developing countries with a low per capita income and high population have a relatively low share in the energy consumption, while the industrialized countries with a high per capita income and comparatively low population consume the overwhelming part of the energy.

The infrastructure currently available in the developing countries requires the use of energy carriers which are easy to handle: these are oil and coal, supplemented by native fuels such as wood and manure. Particularly in the developing countries with their pent-up demand of energy, a relatively strong growth of population has to be expected, these countries will in future claim a higher share of the overall energy demand. In the long run, the industrial countries will therefore have to face a growing competition of the developing countries in the distribution of the dwindling energy resources.

In view of the dramatic rise of world population, the intensified application of nuclear energy seems indispensable. In 1987, the global population reached 5000 million people. Fig. 1 shows the development of the population and the forecasting of the energy consumption from 1900 to 2030. It is evident that a gigantic energy demand will arise.

Nuclear energy is the way of generating energy which is most compatible with environmental requirements and which is capable of covering an important fraction of this enormous energy demand. For this purpose, the HTR 500 offers ideal preconditions, especially for developing countries.

By using uranium and thorium in these countries, the latter can preserve their national fossil fuels for more valuable applications in energy generation, thus having at the same time the capacity of conserving valuable resources and having an economy effect. This is the major potential of the use of nuclear power plants, provided that capital requirements and plant sizes are within reasonable limits.

Summing up, we can say that medium-sized HTRs can offer a technically better and more economic way of solving important future tasks of the developing countries.



FIG.1. Evolution of the world energy consumption between 1900 and 2030.

# 2 Technical Concept of the HTR 500 Power Plant

The concept selected for the HTR 500 system incorporates extensively the THTR 300 technology, licensed and realized in accordance with the current state of science and technology. The experience accumulated in designing, building and operating the THTR 300 is fully exploited, allowing further simplifications and optimization. The transition to commercial power plants is therefore possible, with low risk to manufacturers and owners.

The HTR 500 nuclear power station is a dual-cycle plant with high-temperature pebble bed reactor and steam power plant possessing the steam conditions and the high efficiency of today's fossil-fired plants. Its main features are:

- Integrated design of all primary system components in a single-cavity concrete pressure vessel;
- Use of standardized components and proven materials from the THTR 300 wherever possible;
- Separation of operating and safety systems, leading to simple design;
- Accident control making use of the slow transient characteristics of the hightemperature reactor.

### TABLE 1. HTR 500 MAIN DESIGN DATA

Thermal power	MJ/s	1390
Electric power	MW	550
Plant efficiency	%	39,6
Mean load factor	%	80
Mean power density in the core	MJ/sm <sup>3</sup>	6,6
Primary gas pressure	bar	55
Hot gas temperature	°C	723
Cold gas temperature	°C	266
Main steam pressure at turbine inlet	bar	180
Main steam temperature at turbine inlet	°C	525
Feedwater temperature	۰C	190
Conderser pressure	bar	0,06



FIG.2. HTR 500 reactor pressure vessel with internals.

The HTR 500 is designed for a thermal reactor power of 1390 MJ/s and an electrical net output of 550 MW. The main design data of the power station are listed in Table 1.

A vertical section through the prestressed concrete reactor vessel, showing details, is illustrated in Fig. 2. The cold gas flows downward through the reactor core, cooling the in-core and reflector rods as well as the metal internals. The fuel elements containing low enriched uranium pass through the reactor core once only. The spent

fuel elements are then discharged through the discharge pipes. By adopting standard structural elements previously used in the THTR, such as:

- component parts of the concrete reactor vessel,
- steam generators,
- circulators,
- shut-down facilities,
- ceramic blocks of the core structure,

the nuclear licensing procedure is simplified and the effort required for design, construction and commissioning reduced.

The HTR 500 is an energy source universally applicable to the generation of power and heat, and is suitable for the following applications:

- Power stations for generation of electricity;
- Cogeneration of power and process steam, with the possibility to supply highpressure steam up to 530°C;
- Cogeneration of power and steam for district heating.



FIG.3. HTR nuclear power plant with cogeneration of process steam.

The flow diagram for a typical application of the HTR for the generation of power and process steam is illustrated in Fig. 3.

The HTR 500 is an important innovation in the field of reactor technology. Due to its universal applicability in the electricity and thermal energy market as well as to its high flexibility regarding the site and the fuel cycle, the HTR is predestinated for close-in siting and particularly suited for countries starting to build up their nuclear energy supply.

# 3. Prospects for Future Advanced Reactor Utilization

At the end of 1986, industrialized countries accounted for 93%, while developing countries accounted for 7% of the totally installed nuclear electrical capacity. The characteristics of the HTR 500 are especially suited to meet the requirements of the international market for future power reactors.

The power range between 100 and 600 MWe corresponds to the modified demand of the energy market. Because of the slower increase of electricity demand, grid size, standby power reserve and scope of financing, the electricity market shows a worldwide trend towards power units between 300 and 600 MW. This has become evident in the United States as a result of an inquiry among the American public utilities and it applies particularly well to developing and threshold countries.

For special applications in industry - when steam and electricity are required - and for special siting conditions, the HTR is competitive with conventional power generation plants of the same size, without involving the problem of environmental pollution. In developing and threshold countries, the HTR, compared to conventional plants, helps to simplify infrastructural problems, e.g. problems of fossil fuel transport.

A long-term development target of the HTR is the conversion of fossil fuels into upgraded energy carriers and chemical products, which would result in the conservation of the environment and fossil fuel resources.

The medium-sized HTR is particularly suited for developing and industrial threshold countries, because its characteristics match the local requirements of these countries:

• The high inherent safety which renders the HTR insensitive to disturbances during operation and rules out core meltdown, predestines the HTR for close-in siting and ensures simple operation;

- Continuous refuelling increases the availability and fuel handling;
- Flexibility in fuel supply, i.e. use of high, medium or low enriched uranium, allows each country to use the most favourable fuel cycle;
- Flexibility as to spent fuel management due to the suitability of the spherical fuel elements for direct ultimate storage reduces potential uncertainties at the end of the fuel cycle;
- The secondary circuit is designed in modern, conventional technology;
- The design characteristics (civil engineering) allow domestic participation in HTR projects in developing countries;
- The potential of supplying the overall thermal energy market in addition to electricity generation, i.e. the whole range of electricity and process steam application, is of particular interest to countries which are about to build up their nuclear energy supply system;
- The high thermal efficiency (40% for electricity generation) results in a low waste heat load to the environment rendering the HTR especially suitable for dry cooling, hence, for arid areas.

In developed countries, the tendency with energy generation is to improve safety and environmental compatibility, whereas in developing countries the infrastructure for a reliable energy supply often has to be created.

# 4. Incentives for Advanced Nuclear Power Technology

# 4.1 Special Features of the HTR 500

In the USA, the HTR is often called the "forgiving reactor", because it has a goodnatured reaction to disturbances or maloperations even without any automatic countermeasures. Extensive experimental investigations as well as comprehensive theoretical analyses have shown that the good-natured characteristics of the hightemperature reactor are maintained, even under the most adverse accident conditions. On the basis of the operating experience, the special characteristics of the pebble bed HTR can be summarized as follows:

- High thermal efficiency for power generation, thus releasing less waste heat to the environment. High temperatures permit the use of hot-steam turbines with steam conditions corresponding to those of the most modern conventional power stations (approx. 530°C).
- In addition to power generation, the HTR is ideally suited to the requirements of the high quality market.
- High inherent safety owing to the temperature resistance and high thermal capacity of the ceramic core structure. With the high temperature potential of the ceramic fuel elements, core methown is impossible.
- The coolant gas has a low level of activity and thus low radiation exposure for operating and maintenance personnel, and low release to the environment.
- High fuel cycle flexibility by using high, medium or low enriched uranium. A change of fuel cycle can be carried out in an existing plant, as refuelling is continuous during operation. The pebble bed reactor is the only reactor design which permits this procedure. The use of thorium opens an additional energy resource.
- The spent fuel elements are suitable for long-term storage and offer maximum flexibility with regard to waste disposal, since both direct transfer to final storage or fuel element reprocessing is possible. The decision remains simply one of economics.
- No evacuation of the population, even in the event of hypothetical accidents,
  i.e. accidents which because of the improbability of their occurrence are not taken into consideration in the design.

Thus it can be summarized that the HTR 500 is an energy source capable of providing a worldwide contribution to an energy supply which is in compliance with the important criteria "sufficient, economic, safe and environmentally compatible".

# 4.2 Features for Commercial Applications in Developing Countries

The potential success of the HTR 500 from the vendor's point of view is based on its technical and safety-related advantages, combined with its competitiveness.

During operation, little radioactive waste material is generated and the operating personnel has access to nearly all components for easy maintenance and repair procedures. Due to the high heat capacity of the graphite structures of the reactor core, there is plenty of time for countermeasures. All components are designed with large margins between their permissible loads during operation and during accident conditions.

HTR fuel burn-up is very high, therefore, reprocessing of HTR fuel is not required. The spent HTR fuel elements are suitable for long-term interim storage or final disposal without any effort for conditioning. The fuel cycle does not involve the penalty of reprocessing, which incurs the additional problems of acceptance.

The unit sizes of the HTR programme allow great flexibility in adjusting the grid extension to expected growth rates of electricity consumption. This also gives a chance to optimize the final arrangements of the utility or country.

An HTR with a power size of 550 MW has the advantage of corresponding better to the low increase in power demand and thus requires lower investments in advance.

Due to its highly conventional component requirements and simple operating principles, the decision to use an HTR plant has an impact on national industry without being a burden to the national economy. Because of its design principles, a major part of investments, mainly in civil engineering, can be taken over by domestic industries. In addition, there will be no additional burden to the national infrastructure caused by handling and transportation of heavy components. In comparison to conventionally-fired power plants, financing of the entire infrastructure for the unloading, storage and transportation of coal, is not required.

The consequent utilization of HTR-specific characteristics as well as an optimization of the design of components and circuits lead to the result that the electricity generating costs of an HTR 500 are competitive with those of larger or similar power plants.

# 5. Specific Safety and Design Requirements

The primary objective of any reactor protection system is the safe containment of the radioactive fission products. The classical active safety systems for shutdown, decay heat, removal and activity containment are also used in the HTR 500 to minimize the impact of the power station on the environment and to protect the plant itself under all operating and accident conditions.

The inherent safety characteristics, however, effectively counteract any hazard arising in case of failure of the active safety measures provided for accident control. The inherent safety characteristics combined with the passive safety systems ensure sufficient protection, even in the event of extremely improbable accidents.

The HTR's principal inherent safety characteristics are the following:

- Negative temperature coefficient of reactivity, with beneficial effects on selfstabilization and limitation of reactor power;
- Ceramic core structure and fuel elements capable of withstanding high temperatures;
- Low ratio of power density to heat capacity, resulting in a slow rise of fuel element temperature under accident conditions;
- Inert, phase-stable gaseous coolant ruling out total loss of coolant.
- The HTR 500 design has a burst-safe prestressed concrete reactor vessel.

These HTR-specific characteristics render this reactor particularly well-suited to developing countries beginning to use nuclear energy.

# 6. Market Potential for the HTR 500

The share of the world's nuclear electrical capacity installed in developing countries is small. In order to introduce nuclear power in these countries, a plant size has to be offered which will suit local requirements.

For countries in the process of development, there are some obstacles to be overcome before they can benefit from the advantages of nuclear energy. This has to do with the development of the reactor plant capacity up to 1300 MWe, which is now technically feasible, safe and economic.

The development of the LWR up to this plant capacity took place during a period of about 30 years. The reason of this development was to realize highest economy by using specific cost reduction with increasing capacity. This task has been carried out successfully for industrialized nations.



FIG.4. Nuclear power plant unit capacity development.



FIG.5. Nuclear power plant unit capacity development.



FIG.6. Nuclear power plant unit capacity development.

In industrial nations, lower energy growth rates have been experienced in recent years. This is illustrated in Fig. 4, the result of an enquiry among US utilities carried out by the Gas-Cooled Reactor Association. It shows the intentions to build future plants preferably in a power range of 400 to 700 MWe.

The development of the HTR 500 meets exactly the requested plant capacity. This strategy for the unit capacity makes it possible to meet the demand of the year 2000 as estimated for OECD countries, or areas of OECD countries under development. This is illustrated in Fig. 5, by assuming that the largest unit size in a national grid should not exceed 10% of the total grid capacity.

As can be seen, 5 countries are in a position to use a power plant unit size of between 400 and 600 MWe.

Fig. 6 shows the corresponding illustration for developing countries. In the range of 400 to 600 MWe unit capacity there are 4 countries meeting the 10% criterion up to the year 2000.

The time scale on the horizontal axis indicates the development of the HTR 500. A period of four years is required for design optimization, preparation of the licensing

documents and implementation of the licensing procedure. Starting in 1992, a construction period of less than 5-6 years is required for the HTR 500. This is in time to meet the predicted market requirements for the groups of countries or areas referred to.

Owing to the fact that the plant capacity is only 550 MWe, the total capital expenditure for such a nuclear power station is in a range which enables developing countries in particular to take a decision in favour of nuclear energy.

# 7. Conclusion

The peaceful use of nuclear energy in the Western countries shows a remarkable balance of success as to safety, economics and environmental compatibility. The future of high-temperature technology depends on several factors, such as the development of the power demand, but also on the political context of acceptance.

If we accord an increase in energy consumption to those people who live in other parts of the earth and currently consume less energy than we do, it is to be foreseen what will be the world energy consumption in the year 2030. We have to eliminate the dangerous potential of conflict between the developed and developing countries and offer to the people of the Third World the possibility of an energy supply which is sufficient, competitive, safe, and environmentally compatible.

At present we still need all energy sources in the developing countries, fossil, renewable, and nuclear energy. Nothing is an alternative, everything is additive. Unlike fossil fuels, nuclear fuels are energy resources which can practically only be used for the provision of energy, and thus directly affect the conservation of more valuable resources.

The HTR 500 is a good candidate for nuclear energy production in developing countries. The development of the HTR reactor allows the beginning of construction in the 90s. It is therefore a special challenge to convince the decision-makers to use the advantages which the high-temperature reactor system can offer.

The future of nuclear energy still lies ahead.

# ACCEPTANCE OF THE ADVANCED REACTOR CONCEPT IN INDONESIA

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## Abstract

The new comming concept of nuclear reactor that we call the advanced concept, from their size point of view can be classified into three main categories; small, medium and high power reactor of 900 MWe or more (today commercial nuclear power plant).

We try to discuss in this paper our understanding on the advanced concept in which we want to find the match between the condition we have and our needs, with the level/status of the advanced concept today.

Besides that, either bilateral or international cooperation is discussed as well in the brief historical evolution of nuclear R & D in Indonesia from 1964 until now.

I. INTRODUCTION

The exchange information on the advanced concept for application in the developing countries is really a good opportunity to discuss and to formulate together some preparations necessary leading for the utilization of the peaceful uses of nuclear energy for electricity and for the process heat as well.

The success of the first nuclear power project in a developing country will be a very good incentive for the next projects and it will be a good stimulant as well for the other developing countries.

In general, the condition pre sine-quanon for the success in managing a project are as follow  $/ 1_{-}7$ :

- availability of highly qualified (competent) people managing the project to attain the target of the project

- availability of manpower having qualification necessary at all disciplines for the project implementation
- the capability of local infra-structure to support the activities during the construction phase, and during operation & maintenance phase.

Those conditions can be prepared in a realistic program if a plan of development is clear and practically feasible (public acceptance, economicly justified). Only buy such plan, those three dominant elements can be prepared soundly. In this context, the bilateral and international agreement - cooperation in the field of nuclear science and technology including for the project management have an important role.

### II. ADVANCED REACTOR CONCEPT

II.1. Present Commercial Nuclear Power Plant

The first nuclear power plant that will be constructed in the 1990's for Indonesia is belong to the present commercial power reactor, the class of 900 MWE Pressurized Water Reactor.

This class (type) of reactor has established concept and based on the proven technology, they have no more what we call "childhood phenomena".

Advanced for these class of reactor mean that : by organizing a coordination of present technology and sinchronizing them into the project, we can achieve the construction time less than 6 years, the operational realibility more than 80%, and longer reactor life time (more than 30 years) without much backfitting. Further more, advances mean that for the next projects, briefly speaking that the new-one must be more reliable and economic than the plant before.

It is worth to show here an intersting example from French reactor series, there is an increasing improvement in the serial reactors (projects) : CP1, CP2, P4, P'4 and the last one N4.

## II.2 Small and Medium Size Nuclear Power Plant

For this class of reactors, we discuss here three interesting concepts :

- light water reactor concepts
- heavy water reactor concepts
- high temperature reactor concepts

II.2.1.1.Compact Reactor of 50 MWe and 300 MWe "NP300  $/2_{7}$ 

As shown by slight, from reactor coolant system we note 2 main advances :

- motor coolant pump is integrated into steam generator; simple and effective
- ensure the natural circulation for residual heat removal (passive safety).

This concept adapt the French Nuclear Sub-Marine design for its mechanical design and use The French Pressurized Water Reactor fuel, so it is worth to note here that operational reliability and flexibility is excellent.

Reactor	NP300	PWR900
Parameter		
Electrical output	300 MWe°	900 MWe
Mode of operation	Load follower	Load follower
	Base load	Base load
Reactor type	Compact - PWR	PWR - 3 loops
	2 SG + 2 PP	3 SG + 3 PP
Steam pression	53 bar absolute	56 bar absolute
Steam flow rate	1853 t/h	5460 t/h
T water supply	220 °C	220 °C
OPERATING CONDITION		
Pression	155 bar absolute	155 bar absolute
Inlet temperature	278 °C	286 °C
Outlet temperature	312 °C	323 ℃
Coolant flow rate	24600 M3/hour	68100 M3/hour

Principal Charateristics of NP300 compared to PWR900

° should be verified

These reactors have been licensed by Institute for Nuclear Safety and Protection (I.P.S.N.) of French.

\* This concept is proposed by Technicatome (FRENCH)\*

# II.2.1.2. Advanced Light Water Reactor with the power of 600 MWE "SBWR&AP-600"

The programme of these reactors is under coordination of EPRI and U.S. DOE of the U.S.A. Foreign participation is welcome for this program  $\sqrt{37}$ .

The basic idea of this improvement is how to get an excellent operational reliability and to have the plant more economic.

This achievement will be obtained through the following works :

- simplification of the plant

- minimize the construction time
- increase safety margin
- improve availability, operationability and maintenability
- provide the basis for standardization and assure ready licensability.

All of these guidelines are based upon the existence of proven technology.

Advanced Pressurized Water Reactor (AP-600)

The advances planned for the 600 MWe Pressurized Water Reactor are as follow :

- 1. low power density
- 2. canned motor coolant pumps integrated with steam generator channel head
- 3. passive safety system

Ad.1. Lower power density permit the reduction of linear heat rating and automatically improve the thermal safety margins. The advantages of these consequences are the reduction of the DOPPLER effect and Xenon poisonning, so they will ameliorate the uranium utilization in the core.

In this concept, reactor core is surrounded by radial consisting of 90 per cent stainless-steel and 10 per cent water. This refector can improve the neutron economy and minimize the fast neutron fluence - damage to the reactor vessel.

Ad.2. From the reactor coolant system, we note the following indications:

- less complexe installation, the cross over leg coolant piping are elimated
- using the proven canned motor having a demonstration tract record of high reliability
- inherently reduction of the potentials for small loss of coolant accidents (locas)

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Commercial reactor of ALWR in U.S.A. is predicted to be in the year of 1999.

# II.2.2. PIUS and ISER Concepts $/4_7$

Process inherent ultimate safety principle has been developing by Asea-Atom SWEDEN. In this concept, they explore maximumly the low of nature to get better plant safety features and economic.

In Pius conceptual design, utilization of natural convection for heat removal (residual) and for the shutting the reactor down has been embedded. For the plant of 500 MWe, to submerge and to cool the reactor core, including one week of decay heat removal by evaporation necessite 2300 cubic meter of water. Concequently they use prestressed concrete pressure vessel.

According to mechanical design of the pressure vessel, the conceptor believe that there is no way the water inventory can be lost; ensure the good condition of the core . From maintenability point of view, the cenceptor estimate that a good operational reliability can be obtained as well.

After extensive cost analysis, Asea conclude that electrical generating cost of 500 MWe Pius design is about the same as 1000 MWe Boiling Water Reactor.

### ISER :

The Iser Inherently safe reactor concept is being developed in Japan (industries, university and research institutions).

The main differences from Asea Pius Concept are :

- the use of of a steell vessel to contain the reactor, steamgenerators and the surrounding cold pool
- the use of external drives for the main reactor circulation pumpes
- the use of charasteristics similar to those classical PWR

From the conceptual operating condition, a vessel wall thikness of 30 cm is needed. The reduction of the water in the pool permit only a period of about one and half day to cool the core after a major fault, and after that the operator intervention is required. As in the case of PIUS design, they have to demonstrate :

- the effectiveness of the "density locks" at the interfaces between the cold pool and the reactor coolant circuit in normal operation
- that in all possible errors the density locks will permit entry of the cold borated water in time to shutt the reactor down and to prevent damage to the core
- the development of the insulation required to reduce heat transfer from the reactor circuit to the cold pool to an acceptable level.

Commercialization of the reactors using these concepts (PIUS & ISER) is predicted to demonstrate around the year of 2010.

### II.2.3. Heavy Water Reactor Concept

There are two differenct concepts of heavy water, first is called CANDU using the pressure tube concept and the second is called Pressurized Heavy Water Reactor using the pressure vessel concept.

The first one is developed by AECL (CANADA) and the second by KWU (WEST GERMANY) and ENACE (ARGENTINE).

CANDU 300 with a power of 400 MWe /57

According to the conceptor, we note 5 main advances of this reactor as shown below :

- construction time is around three years due to plant simplification, standardization (prefabrication is possible) and due to plant architecture having accessability at all directions
- high operational reliability due to on-load refueling and services
- easy maintenability and repair due to good accessibility, standardization and simplification.
- either the components or the systems are the same as the standard CANDU 600 and the components life-times are designed 40 years
- utilization of natural uranium fuel.

The reliability and the economy of the plant can decrease due to the following phenomenas :

- the creep of the pressure tube at high temperature under intensive nuclear radiation, replacement is required.
- formation of zirconium hydride in the pressure tube that can cause the brittleness of the pressure tube
- tritium content increase by the time due to irradiation that can increase the dose to the operator in case of leakage
- disponibility of heavy water.

ARGOS PHWR 380 with a power of 300 MWe /6 /

The design concept of ARGOS PHWR 300 is the same as ATUCHA reactor.

According to the conceptor of this reactor, we summarize the following advances of this concept:

- flexibility on reactor operation, an outstanding load-following capability
- high operational realibility, utilization of automatic on-load refueling.

## II.2.4. High Temperature Reactor Concepts /7,8/7

Peach Bottom and Fort St. Vrain HTR have been operated respectively since 1960's and since the 1970's in the United States of America. The pebble bed reactor, AVR 15 MWe and THTR 300 have been operated as well in Gemrnay since 1968 and 1986 respectively.

Either the HTR in the USA or in Germany, they show an excellent safety characteristics. At Peach Bottom and Fort St. Vrein HTR, the mean annual dose for person working there is about 3 man-rem, it is about 30 man-rem in Magnox reactors and it is about 400 man-rem in LWRs operated in Germany (GRS Report 1981). Another example is coming from THTR 300, where the radioactivity releases of this HTR compared to the licensed values are listed in the following table :

## THIR 300 RADIOACTIVITY RELEASES

SUBSTANCES	ACTIVITY RELEA	SE AS & OF
	LICENSEE	VALUE
	1986	1987
GASEOUS		
Noble gas	0.05	0.02
Aerosols	42.3	11.2
Iodine 131	1.6	4.1
Carbon	0.54	0.62
Tritium	15.3	56.2
TIÔND		
Fission and activation product	0.04	0.05
Tritium	3.1	9.1

Now THIR 300 becomes reference design of 500 MWe HTR, and agreement for the preparation of safety analysis report has been signed since 29 July 1988 (Swize and German institutions).

Besides their excellent safety characteristics, following advances of these HTR concepts  $\sqrt{7,8}$  are worth to summarize here :

- various applications: electricity generation, process steam for industrial processes, coal processing, district heat etc.
- lower electrical generating cost, the THIR 500 is comparable with modern 1300 MWe light water reactors
- even in the hypothetical accident, radioactive exposure is so low, evacuation of the population in the vicinity of this plant is not required (public acceptance), so it can be built near the cities.
- besides the active safety system, these concepts benefit a number of inherent safety characteristics:
  - \* negative temperature coefficient of reactivity, self stabilization and limitation of reactor power
  - \* ceramic core structure and fuel elements are capable of withstanding high temperature (up to about 3500 °C), eliminating the possibility of core melt-down
  - \* low ratio of power density to heat capacity resulting in a slow rised of fuel element temperature under accident conditions
  - \* inert gaseous coolant.

### III.1 THE PENETRATION OF NUCLEAR ENERGY IN OUR ENERGY SYSTEM PRODUCTION

Indonesia is a country with a population of more than 170 million. It consists of many island, where about 100 million live in Java. The rest of population are found non uniformly on the other islands. Most of them, including Java island are far from the source of energy (oil, gas and coal). Since the development of those islands or regions are running well, they represent the growing potential of energy demand, where in the future this demand can be possibly met by small-medium size, reliable nuclear power plants.

The following table shows the regions having the population more than 2.5 million in Indonesian: \*

Region	Population (million)	Area(km2)	P.Growth(%)
West Java/Jakarta Raya	38,718	44,826	2.47/3.52
East Java	30,973	47,922	1.23(84-86)
Central Java (Yogyakarta	30,348	37,391	1.21 (1.09
North Sumatera	9,517	71,680	2.47
South Sulawesi	6,544	62,482	2.14 (85-86)
South Sumatera	5,453	103,688	3.14 (82-83)
Lampung	5,140	35,376	5.58 (80-85)
West Sumatera	3,929	42,297	1.90
Aceh	3,987	55,390	2,68
West Nusatenggara	3.071	20,177	2.16
East Nusatenggara	3,029	47,876	2.05
West Kalimantan	2,819	146,760	2.05
Bali	2,659	5,561	1,51
Riau	2,513	95,561	2.92

\* AN OFFICIAL HANDBOOK INDONESIA 1988, Dept. of Information Republic of Indonesia.

In 1987, at the Pacific Basin Conference held in Beijing, our Director General "Dr, Djali AHIMSA" reported that the projected installed capacity by the year 2000 is around 10 000 MWe for the Java grid system, so before the end of this century a current design of 900 MWe Pressurized Water Reactor is economicly justified for the grid system of Java island. This result was coming from the upgrading of the feasibility study done in 1985-1986. The updating work was carried out by an inter-agency team (IAEA, Bechtel International Inc., Softratom and related national institutions for energy development).

As the question of the first nuclear power plant project, an invitation has been extended to five leading nuclear supplier companies to conduct a study on the possibility of their leading participation /-9 7.

Soon after that, we had also done the prefeasibility study on the utilization High Temperature Gas Cooled Reactor (high temperature steam production) for oil mining in Duri - RIAU (BPPT, BATAN, MIGAS, PERTAMINA, LEMIGAS from Indonesian side and KWU, Interatom from German side). The result of this study has been reported by Mr. Mursid Djokolelono et all at IAEA Technical Committee Meeting on Design Requirements, Operation and Maintenance for Gas Cooled Reactors, San Diego, USA, September 21-23, 1988. It is an economic study of a plan consisting of 4 HTR-modules of 200 MWt, one back pressure turbo generator, giving (76 MWe net) and process steam (260 kg/s: 75 bar and 320 °C).

It is worth to mention here the concluding remarks of this study /107:

- For a source of low pressure ( 30 ata) and low temperature
  ( 250 °C) steam, an HTR can easily comply the requirement.
  But the elcetric load variations coming from the production
  field as well as the ones coming from the oil industry, lighting
  and households shall be overcome without disturbing the steam
  continuity.
- The crude selling price represents the most dominant factor in the economic calculation of HTR application. The present oil price situation and its declining tendency do not support the profitability of this nuclear alternative. Further, if the burned crude has the price of the well-head cost and the electricity consumption is provided by the field associated gas or the crude in the same manner, it is very difficult for any other energy source to compete.
- As a large capital is involved certain premiums given to the oil contractor shall be prearranged and the crude production scenario be optimized to make the nuclear alternative interesting.

The effort to lower the capital cost and to extend the life-time of the nuclear reactor are obviously general remedy to be taken.

- The scheme of having an independent utility company to supply the nuclear heat and electicity may be sound if the burden of financing can be overcome, since along negative cashflow will involve. Furthermore the selling price of heat and of electricity will fully determine the feasibility. To incorporate the nuclear power plant inside the oil contractor company can relieve the burden of long negative cashflow. But this scheme shall involve a long term committment from the company.

According to the result of prefeasibility study on the utilization of HTR, we note that the required remedies are : longer reactor life time and reduction of construction cost where from the technological progress mentioned by the conceptors in most references; these remedies are not far from us.

From the table figuring the Indonesian population, region "Duri-RIAU" is the last one in that table. There are many regions in Indonesia as the growing industrial areas or economics activities, they can represent the new potential energy demands. The reliable-competetive small or medium nuclear plant is possibly suitable for those regions in the future. Fortunately due to the communication problems most data are not yet with me, so I can not elaborate this quantitatively.

## III.2 FACTS AND EXPECTATIONS

III.2.1. The present high power reactors having the power of 900 MWe or more (today commercial power reactors) have world-wide operating experiences. Optimizations have been done since many years in all aspects of reactor operation, reactor economy and safety. They are available and the new-one shows better operational reliability and economy. Progressive improvements of these reactors are world-wide efforts and co-operations.

They are Pressurized Water Reactors and Boiling Water Reactors.

Another type of reactors that can possibly be categorized to this class is large Heavy Water Reactor (Bruce and Candu 600), but their power are less than 900 MWe. III.2.2. The recents studies and developments in the small and medium size reactors are quite interesting, the international cooperation is open for developing the new concept, they are still in evolution and the progress can be always expected to establish the technology and to get the proven concept.

In general, due to their size, most parts of systems can be fabricated in the shop and ship to the site; so they can minimize the construction time. This methode is explored maximumly in order to make the small medium size reactors competetive to the large-one.

Candu300 is proposed from now by AEC-CANADA. The light water reactor NP300 (TECHNICATOME-FRENCH) and the high temperature reactors (from GERMANY and USA) can be available in the coming years. Around the year 2000 the SEWR and AP-600 (both from USA) can be expected.

The PIUS and ISER need much more time to be demonstrated as the commercial plants. They can be expected around the year 2010.

III.3.3 The developing countries in general are characterized by two main problems :

- non adequate infra-structure
- non adequate qualified manpower at all disciplines.

By these characteristics, it is really needed to have the plant that is easy to operate, good accessibility to the system for maintenance and repair. Besides that the plant with high reliability-safety is evidently expected. In this context the plant having the inherently passive safety features in addition to the active safety will much more suitable to guarantee the continuation of energy development in the country (public acceptance).

## IV. EVOLUTION OF NUCLEAR R & D IN INDONESIA /11,12,13/

IV.1. On October 1964, the first nuclear chaine reactor took place in Indonesia at Research Centre for Nuclear Techniques in the 250 KWt Triga research reactor.

Due to the development of our needs for the reactor physics, core thermohydraulics, isotopes production and fundamental research in 1971 this reactor was upgraded to 1 MWt. In 1979, one more Triga research reactor of 250 KWt built in Yogyakarta, another research centre of National Atomic Energy Agency.

One of interesting utilization of this reactor is to study the shielding material and its fabrication.

To support the nuclear power program in the country, a modern Material Testing Reactor with the power of 30 MWt was constructed near Jakarta. It can be used for the fundamental research as well. This MTR was coming from West Germany (Interatom GmbH).

The supporting Facilities for this reactor are as follow :

- \* waste treatment plant coming from French (Technicatome)
- \* research reactor fuel fabrication plant coming from West Germany (Nukem)
- \* pilot plant for power reactor fuel coming from Italy (Nira)
- \* engineering and safety laboratory coming from Italy )Nira ?)
- \* radiometallurgy installation for post irradiation examination "PIE", coming from West Germany
- \* nuclear mechano-electronic installation coming from Canada (AECL).

We can see here the bilateral or international cooperation characterizing the development of our program and it is worth to mention that the assistances of IAEA have important contribution.

This infra-structure are related very much to the Water reactor either light water reactor or heavy water reactor.

By this infra-structure, BATAN can contribute and participate in international research and development programs for the advanced reactor concept.

Maximum utilization of this infra-structure to support all activities for the reactor power program mean a good economization of the investment.

## V. CONCLUSION

Since Indonesia is a growing nation where the development have been doing in many sectors and levels, the need for much more energy is clear. They can be in the form of electricity, process steam for industrial purposes and possibly for coal processing as well, and even for the desalination water.

The plants prepared for electricity generation only, base load mode of operation are more relevant due to lower fueling cost, but anyhow the flexibility in reactor operation is a good option.

For small and medium reactors (multi purposes utilization) base load mode of operation is acceptable, the flexibility for adjusting the productions is required.

In addition to active safety features, the optimum inherent - passive safety are expected.

Since the construction cost and the reactor life time determine the competetivity of the plant, we expect that vigorious efforts can reduce the construction cost (time) and can design the compenents having the life time more than 40 years. These can be the best remed.

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## MAIN DESIGN PARAMETERS FOR AN ADVANCED NUCLEAR PLANT FOR THE VENEZUELAN ORINOCO OIL BELT DEVELOPMENT

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#### Abstract

The main design requirements for a steam-supply advanced nuclear plant proposed for both the extraction and the preprocessing of extra-heavy oil from the Orinoco Oil Belt are here presented. The model under consideration is a modular co-generation concept able to supply a large fraction of the energy required by an oil field producing 100.000 barrels per day of refinery ready synthetic oil. Three 1200 Mw(th) hightemperature gas cooled reactors, built successively, would supply most of the process heat, the injection steam and the electricity required, this in accordance with the main design criteria of a high variability of steam demand along the field life-time. The energy balance would be supplied by burning oil processing residues. Although design parameters depend on particular oil deposit characteristics and crude properties which may change with the field location in the OOB, the values established for this model are: process heat maximum conditions of pressure 100 bars at a temperature of 500°C, and for injection steam pressures between 120 to 170 bars to saturation temperatures. Current design studies performed elsewhere, based on high-temperature gas cooled reactors, present design characteristics which could satisfy the requirements of the model under consideration. Further more detailed studies are recommended.

### INTRODUCTION

The supply of oil in the world is depending increasingly on heavier crudes as the light oil reserves start to decline. Being oil an unsubstituible energy source for many applications, some of them of fundamental value in the developing of third world countries, especially for those with no other alternatives, it becomes of paramount importance to have a dependable supply from this important resource. The present situation of an oil glut with relatively low prices is in general believed to be temporary since several important oil producers will be runing out of oil in a not so distant future (Ashby, 1988).

Venezuela's contribution to the world oil market has been of significance and still there are considerable oil reserves in the country (8800 millions of m , end of 1986; PODE, 1987). Venezuela also has the world largest deposits of extra-heavy oil (1.2x10<sup>12</sup> barrels; Rodriguez, 1987) located mainly at the so called Orinoco Oil Belt (OOB) (Fig. No.1). So, it is of great importance and interest to expend efforts to make available this energy





FIGURE No. 2: HEAT-STIMULATED OIL RECOVERED



FIGURE No. 3: a) INTEGRATED AND b) NON-INTEGRATED MODELS FOR THE OOB DEVELOPMENT

resource in the next future. However, the exploitation of this extra-heavy oil (API gravity from 8 degrees up) represents a challenge due to the special characteristics of such a project.

Current oil market situation has caused some postponement in the OOB development (Marzin, 1988). Also, important light and medium oil reserves have been recently found in the country Tippee, 1988), causing additional delays in the plans for extraheavy oil exploitation.

A small fraction (5 to 10%) of the oil in the OOB deposits could be extracted by primary means, representing, anyhow, important amounts of oil due to the huge deposits present. However, secondary and tertiary recovery methods have to be used if higher yields are wanted. Being this the case, steam injection (Fig. No.2) is one of the most promising methods as demonstrated by experience obtained in several enhanced oil recovery projects around the world. Steam soak followed by steam drive processes could increase the extraction to from 20 to 30% of the oil in place.

Considerable amounts of steam are required in the heavy-oil extraction process, especially when going from the soak production stage to the driving stage. The energy needs could be supplied by several means. The most inmediate idea is to use oil residues obtained in the heavy oil upgrading process. This is the so called integrated model (Fig. No. 3a), where all the energy required is obtained from the same oil being extracted. In the non-integrated model (Fig. No. 3b), part of the energy is supplied by outside energy sources.

In an integrated model, additional amounts of oil, probably as high as 35% or more (Carvajal, 1988), need to be produced in order to obtain the same oil output as compared to the nonintegrated model. This would mean higher portions of the oil extracted to be burnt in order to produce the increased amount of steam, resulting in a heavy economic burden.

The non-integrated model requires the availability of abundant and not so expensive energy sources. The nuclear option presents good perspectives in cases like this, due to its low fuel costs.

The use of nuclear energy in the development of the OOB have been previously proposed (Perret, 1981). Extraction of the heavyoil only, without its processing, was suggested employing Magnox technology, with results indicating economic competitiveness at that time.

Light water reactors are practically excluded from this application due to their relative low temperatures of operation. Gas cooled nuclear reactors and other more advanced high temperature reactors are the only choice in this case, since they could produce high enough steam pressures for injection and high enough heat temperatures for the oil processing.

Once the crude has been extracted, it has to go through an upgrading process in order to obtain a crude with enough quality to be used as feedstock in conventional refineries. This upgrading process consists of treatment, conversion and decontamination, steps representing high energy consumptions.

The conversion process consists mainly of the hydrogenation of the crude to improve its hydrogen to carbon ratio, a heavy energy consumer step because of the large amounts of hydrogen needed to be produced. The HDH upgrading process, a Venezuelan development (Cavicchioli, 1986), was chosen for this study.

The possibility of employing advanced nuclear reactors concepts, adds new dimensions to the OOB development, due to a possibly higher versatility and new ecomic perspectives. When cogeneration, that is, electricity and process heat simultaneously produced, is considered, higher economic incentives could be forseen.

Nuclear energy, also, presents clear advantages with respect to fossil fuel burning when environmental effects are taken into proper consideration. This aspect gains more importance in Venezuela because of the presence of a fast growing forestry industry being established in regions inside the DOB, with the intention to satisfy all the paper needs of the country in the next future.

The main objectives sought with this work are, firstly, to present the Venezuela's OOB case for discussion here, at this Meeting, as a case with good perspectives for the employment of advanced nuclear energy, collecting impressions from the participants, as well as to make use of the IAEA's experience in the field. Secondly, to build up enough indigenous knowledge and human resources capable of following latest advances in the subject and able to continue the required studies.

The main design requirements for the heat and steam conditions for the OOB case are here presented. The model assumed in this study consists of an oil plant capable of producing 100.000 barrels per day of synthetic oil at a constant rate during 25 years. Steam soak followed by steam driving stimulation is considered for the secondary recovery. Energy would be supplied mainly by an advanced nuclear power plant based on high temperature gas cooled reactors. For the required oil treatment, the HDH upgrading process has been selected. Natural gas is used as feedstock for the production of the hydrogen needed in the crude upgrading process. Oil residue burning is allowed in this model for the energy balance, especially in process stages where fossilfueled systems are indicated as indispensable or more convenient.

Although mention to an especific type of high temperature gas cooled reactor is made, the choice of the reactor type remains as an open issue until more detailed studies could be performed.

#### THE OOB HEAVY OIL AND ITS EXTRACTION AND PROCESSING NEEDS

The OOB is located at the northern margin of the Orinoco River covering an approximated area 700 Km long by 70 Km wide, (Fig. 1). A high portion of the oil deposits are found between 200 and 1200 meters of depth, with indication of deposits deeper than 2000 meters. The highest portion of the oil has been discovered in the Miocene (Tertiary period) with a minor part in the Paleozoic and in the Cretaceous period (CADAFE, 1981).

Table I presents the crude characteristics of three different oil samples from the OOB. When compared with oil residues from the Middle-east, venezuelan extra-heavy oil fits between the long and the short residues of the Light Arabian crudes, but with much higher metal (vanadium and nickel) content. Sulfur and nitrogen are also present in relatively high concentrations, but the asphalten fraction does not exceed 12%. The hidrogen to carbon ratio is low, indicating a need for heavy hydrogenation.

Figure No. 4 summarizes the major steps required to convert the heavy oil into a high-quality synthetic crude ready to be fed to a conventional refinery.

The first step is to heat the heavy oil in the deposit to reduce its viscocity in order to be able to extract it. When steam is used as the heating medium, there is another effect produced by the steam condensation which helps in the process. What happens is that when heat is added, some distillation occurs inside the deposit, and the vaporized oil produced is transported by the steam to the heated front, where it condenses in presence of lower temperatures, mixing with the heavier fractions and, so, helping in the reduction of the oil viscocity (Prats, 1987).

In the conversion process, the main stage is the hydrogenation of the crude to increase its hydrogen to carbon ratio.
API	VISC.(60°C) (cSi)	CONRADSON CARBON (%w)	с/н	VANADĪUM (pom)	SULPHUR (%w)	NITROGEN (%w)	ASPHALTENES (%w)	LOWER/UPPER HEATING VAL. MJ/kg.
8,9	5000	15.2		430	3,99	0,76	10,1	
9,1	2928	14,2	~ 8	409	3,55	0,63	11,6	39/42
9,0	5400	11,8		390	3,92	0,52	8,6	
16,6	72	8,0		28	3	0,16	2,9	
6,5	3000	23		90	4,05	0,34	10,0	
) 3	10 <sup>5</sup>	27,7		269	6,0	0,48	13,5	
	API 8,9 9,1 9,0 16,6 6,5 ) 3	API VISC. (60°C) (cSi) 8,9 5000 9,1 2928 9,0 5400 16,6 72 6,5 3000 ) 3 10 <sup>5</sup>	API         VISC.(60°C) (cSi)         CONRADSON CARBON (%w)           8,9         5000         15,2           9,1         2928         14,2           9,0         5400         11,8           16,6         72         8,0           6,5         3000         23           )         3         10 <sup>5</sup> 27,7	API       VISC.(60°C)       CONRADSON CARBON       C/H         8,9       5000       15,2         9,1       2928       14,2       -8         9,0       5400       11,8         16,6       72       8,0         6,5       3000       23         )       3       10 <sup>5</sup> 27,7	API       VISC. (60°C) (cSi)       CONRADSON CARBON (%w)       C/H       VANADIUM (ppm)         8,9       5000       15,2       430         9,1       2928       14,2       -8       409         9,0       5400       11,8       390         16,6       72       8,0       28         6,5       3000       23       90         3       10 <sup>5</sup> 27,7       269	API       VISC. (60°C) (cSi)       CONRADSON CARBON (%w)       C/H       VANADIUM (ppm)       SULPHUR (%w)         8,9       5000       15,2       430       3,99         9,1       2928       14,2       -8       409       3,55         9,0       5400       11,8       390       3,92         16,6       72       8,0       28       3         6,5       3000       23       90       4,05         0       3       10 <sup>5</sup> 27,7       269       6,0	APIVISC. (60°C) (CS1)CONRADSON CARBON (%w)C/HVANADIUM (ppm)SULPHUR (%w)NITROGEN (%w) $8,9$ 500015,24303,990,76 $9,1$ 292814,2-84093,550,63 $9,0$ 540011,83903,920,5216,6728,02830,16 $6,5$ 300023904,050,34 $0$ 310 <sup>5</sup> 27,72696,00,48	APIVISC. (60°C) (CS1)CONRADSON CARBON (%w)C/HVANADIUM (ppm)SULPHUR (%w)NITROGEN (%w)ASPHALTENES (%w) $8,9$ 500015,2 9,14303,990,7610,1 $9,1$ 2928 9,014,2 5400-8409 3,553,550,6311,6 $9,0$ 540011,83903,920,528,616,6728,02830,162,9 $6,5$ 300023904,050,3410,0 $3$ $10^5$ 27,72696,00,4813,5

TABLE I : ORINOCO BELT CRUDE CHARACTERISTICS IN COMPARISON WITH MIDDLE-EAST OIL RESIDUES

FRACTION	00B ZUATA	LIGHT ARABIAN
C5-190°C(%v)	1,1	28,2
Sulfur (%w)	1,44	0,03
190-343°C(%v)	15,5	27,5
Sulfur (%w)	2,27	0,82
cetane index	32,3	58*
343-510°C(%v) Sulfur (%w) Conradson Carbon	24,7 3,36	29,5+ 2,09+
(% p)	0,24	0,65+
Nitrogen (ppm)	1973	810+
510°C(% v)	58,7	14,0 ##
Sulfur (% w)	4,20	4,0 ##
Vanadium (ppm)	794	76 ##

(Ref.: Cavicchioli, 1986)

Fraction: 343-356°C Fraction: 265-370°C Fraction: 565°C







FIGURE No. 5 : NON-INTEGRATED MODEL FOR THE OOB EXTRACTION AND PROCESSING USING THE HDH PROCESS

This can be accomplished either by carbon rejection methods or by hydrogen addition, the latter being preferred for its much faster rate and high conversion yields; however, it is more expensive for requiring large amounts of hydrogen. Well known processes, such as the LC-Finning or the H-oil, could be implemented, although in this case, the HDH process, based on hydrocracking, has been selected.

The HDH process presents some advantages when dealing with venezuelan heavy oils. Their metal high concentrations ruin the catalyzers used in the process, becoming a necessity to implement methods which could use cheap catalyzers obtained from indigenous materials, as in the HDH process. Also, the hydrogen economy has been improved, under moderate temperature and pressure conditions (Cavicchioli, 1986).

Finally, the decontamination stage indicated in Figure No. 4, involves hydrotreating, catalytic sulfur and nitrogen removal, and flue gas treatment for obtaining a high-quality product with reduced environmental impact.

Figure No. 5 presents an schematic view of the main processes involved in the whole operation. This corresponds to a nonintegrated model since a nuclear plant (1) is supplying a large portion of the energy demand.

The nuclear plant produces high temperature heat for the steam plant (2), the upgrading plant (3) and the gasification unit (4).

The steam plant (2) produces both the injection steam and the steam needed for electricity generation, in a cogeneration mode. Other steam requirements, at various temperatures and pressures, could also be provided by tapping the main steam lines.

The upgradind plant (3) is based on the HDH process, as already mentioned. The gasification unit, based on a steam reforming process, produces the high volumes of hydrogen required in the hydrocracking and the decontamination stages. Natural gas, of which there are abundant reserves in the country, is used as raw material in the hydrogen production.

The desulfurization units (5) and (6), which receives the distillates produced in the atmospheric and the vacuum distillation towers, (7) and (8), also consumes part of that hydrogen.

Residues obtained at the distillation stages could be used to supply part of the heat needed for steam production, as well as for the upgrading and the gasification plants. The residues could also be used as feedstock for hydrogen production instead of natural gas. Even with a nuclear plant supplying most of the heat, oil residues could still play an important role where they are either more convenient or practical to use or where fossil fuels cannot be substituted.

However, there always exists a practical limitation in employing these oil residues: In an optimized processing, a fraction of the residues have to carry most of the impurities that have been extracted from the processed oil, producing impurity concentrations too high to be able to use or sell such residuals for burning under stringent environmental regulations. One possible way for the disposal of this contaminated portion would be by reinjecting them into depleted oil fields. Large amounts of water are needed in the whole process, of which there are no limitations in the zone since it can be obtained either from the Orinoco river or from in-site wells. High-quality water is required to produce the process steam and the steam for electricity generation, so, demanding for a treatment plant (12). For the injection steam, a much lower water quality is allowed since, otherwise, the processing of the large volumes involved could become rather expensive. However, some treatment will be needed to avoid problems associated with impurity deposition in the steam generators. Water recycling for steam injection could be performed at station (9), where oil and water is separated.

Electricity is generated at station (11), satisfying the demand of the whole complex, and with an excess being exported either to the national electrical grid or to other plants.

There is an important point to be mentioned about the proposed cogeneration model: this model allows relative large variations of the steam requirements along the whole project life-time since these variations could be accomodated by diverting part of the steam for electricity production, or viceversa. Moreover, when the oil field becomes depleted, the nuclear plant would still have years of life ahead. Since this could happen at a time when the country will be more dependent in thermoelectricity to satisfy its demand, as forescasted for Venezuela at some time in the future, these nuclear plants could be fully used for electricity production.

## TABLE 11: MAIN PARAMETERS USED/ASSUMED FOR THIS STUDY

0il Produc	tion Rate (barrels per day, BPD)	100.000
Project Li	fe-Time(secondary & tertiary phases)(years)	25
Oil in Pla	ice (bbl/acre-foot)	720
(	$m^3 \text{ oil/Km}^2 - m)$	0.093

Total Oil Produced in the Zone (barrels) (during driving production)	0.73 x	10 <sup>9</sup>
Recovery of Oil in Place (%)	25	
Area of Development Zone (Km <sup>2</sup> )	100	
Total (max.) Number of Producing Wells	800	
Well Spacing Area (Km <sup>2</sup> /well)	0.125	
Distance Between injector wells (hexagon pattern)(m.)	500	
Distance Between Production Wells(hexagon pattern) (m.)	289	

# TABLE III : THE OOB PROJECT DEVELOPMENT DURING SOAK AND DRIVING PRODUCTION

PRODUCTION PHASE:		Soa	k Produc	tion —			Driv	ing Proc	luction	<u> </u>	<u>\$</u> }					
HEAT & STEAM POWER REQUIREMENTS(GWth)	1.2	1.2	1.2	1.2	1.2	   1.6 	1.6	2.0	2.0	2.3	2,3	2.3	2.3	2,3	2.3	2.3
STEAM REQUIREMENTS						1 6 1										
OIL/STEAM (B/T) STEAM (10 <sup>3</sup> TPD)	25 4	25 4	25 4	25 4	25 4	1 5 1 20	5 20	2.5 40	2 <b>.5</b> 40	2.0 50	2.0	-			2.0 50	2.0 50
FIELD DEVELOPMENT:						<i>i</i> 1										
OIL PRODUCTION PRODUCTIONS WELLS	100. 500	.000 1 500	BPD 500 250	500		500	500	600	600	800	800	800	800	800	800	800
INJECT WELLS-DRIVING	-	-	-	~	-	100	100	200	200	400	400	400	500	500	500	500
TOTAL # WELLS:	500	500	500	500	500	600	600	800	800	1200	1200	1200	1300	1300	1300	1300
PRODUCT. PER WELL: (BPD)	200	200	200	200	200	200	200	167	167	125	125	125	125	125	125	125
NUCLEAR REACTORS ON LINE	0	0	0	0	0		1	2	2	3	3		-			
TOTAL NUCLEAR POWER AVAILABLE (Gwth)	0	0	D	D	D	1 1 1 1.2	1.2	2.4	2.4	3.6	3.6		-		. (	,
L		•		<b>.</b>		ı					_1{5_		~••	•	t (ye	ears)
0		1	2			5					10	20				25

#### MODEL USED IN THIS WORK

Table II presents the main parameters assumed for the project, to be considered together with Table I, already indicated, where the oil characteristics were specified.

Table III indicates the major development characteristics of the project during its 25 years of life. During the first 5 years, the soak production stage is carried out without any nuclear heat usage. During this stage, oil residues burning is considered as more convenient due to the shortness of the period and the need for smaller and mobile plants.

The oil field in this model is assumed as producing 100.000 barrels per day (BPD) at a constant rate during the project life; of course, a goal difficult to achieve in real life. The field might have just finished a primary recovery stage, followed now by soak production during five years. After this, the steam driving phase starts, making use of nuclear power to produce the increased amount of energy now required. The first nuclear reactor construction is fisnished just on time for the starting of the steam drive. The other two reactors enter into operation every two years thereafter, along with the increased injection steam demand.

During the steam soak period, a production ratio of 25 barrels of oil per ton of injected steam has been assumed. During steam driving conditions, the oil to steam ratio goes down considerably, to between 5 and 2 barrels of oil per ton of steam. This parameter establishes the total amount of steam needed to be produced for injection, 50000 tons per day as the maximum from the tenth year of the project on.

The power required to produce the injection steam, plus the other processing heat needs, amounts to a total ranging from 1200 Mw(th) to 2300 MwT(th), as indicated in Table III.

The steam drive starts with 500 producing wells in operation, 250 of which have been used during the soak process (cyclic injection and production). Then, 100 new injection wells enter into operation, a number later increased by steps to 400 for full production with 800 producing wells, making a total of 1200 at the beginning of the tenth year after driving started. An hexagon pattern, with an injection well at the center of each hexagon and the production wells at the corners, is suggested. This arrangement presents several advantages for enhanced oil recovery according to experience (Herrera, 1977).

The steam for injection is distributed to the oil field from a centralized station, where the nuclear plant and, probably, the upgrading plant too, would be located, as Figure 6 suggests.

Figure No. 7 corresponds to the global energy flow diagram for this model, at maximum steam and heat demand. A total power requirement of 2300 MW(th) is indicated, together with a natural gas import of 13040 barrels of oil equivalent per day (BOE/d) (1.9 million normal cubic meters) used as feedstock for the hydrogen production, as required in the HDH process (option B; Cavicchioli, 1986).

The above figure for the total power demand comes from the fact that about 15580 BOE/d are needed to produce the injection



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FIGURE No. 6: OIL FIELD LAYOUT WITH CENTRALIZED NUCLEAR AND UPGRADING PLANTS
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FIGURE NO. 7: ENERGY FLOW DIAGRAM FOR THE OOB CRUDE EXTRACTION AND UPGRADING USING THE HDH PROCESS

steam, plus 10600 BDE/d for the processing heat, and 4880 BDE/d for the electricity demand (127 Mwe, assuming a generation efficiency of 35%). This electricity is used as pumping power during the extraction process, as well as for inside the plant oil transportation, and for all electricity needs in the upgrading plant.

The above energy figures correspond to 31% of the energy content of the final product, with an additional 13040 BOE/d in the form of natural gas as feedstock for the hydrogen production. This gives and idea of the energy expensiveness of the whole project.

HEAT AND STEAM CONDITIONS AND ENERGY SUPPLY

With respect to heat and steam requirements, there are two main processes to be considered: the extraction of the crude and its upgrading. The other processes, although important, require only a small energy fraction of the total.

Heat and steam conditions depend on oil properties, deposit characteristics and the processing method selected. Since all of these could vary considerably from case to case, very especific conditions have not yet been defined at this early stage of the study. The figures used here have been selected according to studies carried out previously and to reported experiences obtained in several pilot projects completed in Venezuela and elsewhere.

Injection steam pressures depend on oil deposit pressures which varies with well depth. In accordance to OOB characteristics, the steam should have pressures starting around 120 bars (12 MPa), up to probably 170 bars (17 MPa). Being such steam at saturation conditions, the correspondent temperatures are  $325 \cdot C$  and  $350 \, ^\circ$ C.

Although high-quality steam, i.e., close to 1.0 (saturated) or even overheated steam, might be considered to avoid premature condensation, the use of such a high quality could cause problems associated with impurity deposition in the steam generators. Then, some humidity is desirable to keep the impurities disolved in the liquid fraction, a quality of 0.8 being recommended.

Deeper oil deposits might require the use of reheaters in order to avoid premature condensation before the steam could reach the oil deposit. However, too high temperatures should be avoided because heat losses start to get too large and the efficiency of the driving effect starts to decrease (Prats, 1987).

In the oil upgrading process, most of the energy is required in the form of process heat with small steam demand (hydrogen production apart). Heat should be supplied at 500 °C. in the HDH process to run chemical reactors involved in the operation (Solari, 1988). Lower-grade heat might be supplied for the distillation needs by heating the crude at the distillation tower entrance employing smaller heat exchangers connected to intermediate cooling loops coming from the steam plant. Different steam pressures are required at the various steps, with a maximun around 100 bars (10MPa) (Gutierrez, 1977). In the steam reforming process employed for hydrogen production, the highest energy demand is for heating the reformer oven with temperatures of the order of 900 °C., too high value for today's nuclear reactors. In this case, conventional heating using high quality fuels is indicated. However, very-high temperature gas cooled reactors, under design studies at present, could be considered since they are capable of reaching those temperatures. Additional steam is required in the reforming process, in an amount of approximately 130 tons per hour, for the production of the hydrogen needs in the upgrading of 100.000 BPD. A steam pressure of 100 bars is indicated, and it could be supplied from the steam plant.

As energy source, a combination of different alternatives should be considered after an optimization study, economic and other factors properly taken into account. Before this could be done, any selection is somehow arbitrary.

Oil residues burning to satisfy the energy needs, at a first glance, looks as the best option. But when oil prices start to rise and the oil fraction consumed for its production increases,



FIGURE No. 8. HTG REACTOR WITH SPHERICAL FUEL ELEMENTS USED FOR SUPPLYING HEAT AND STEAM



FIGURE No. 9: NUCLEAR REACTOR MULTIPLE USES FOR PROCESS HEAT AND STEAM

other alternatives have to be sought. In high energy consumming processes, nuclear energy represents a valid option due to its particular characteristics, mainly its low fuel cost and reduced environmental impact.

Taking into consideration latest developments in advanced medium-sized nuclear reactor concepts, the high temperature gas cooled reactor, developed mainly in Europe and the USA, is considered as the most promising reactor type able to satisfy the specific OOB development requirements.

Three nuclear reactors, with 1200 Mw(th) of power each, have been chosen for this model. Figure No. 8 shows a vertical section of one of the possibilities as a high temperature gas cooled reactor, with spherical fuel elements. A prototype, similar to this one, the THTR-300, is running now at the Federal Republic of Germany.

With a total of 3600 Mw(th) available, there are around 1660 MW(th) to be employed in electricity production (580 MWe) in cogeneration operation. Since 127 Mwe are needed for the whole plant operation, this leaves around 450 Mwe to be exported. The nuclear plant supplies heat to the steam plant by having the reactor primary gas coolant circulating through several heat exchangers and steam generators. A portion of the steam goes to the gasification unit, where hydrogen is being produced. The upgrading unit consumes relatively small amounts of steam, but process heat is required in large amounts. Figure No. 9 gives an idea, among several possibilities, on how the reactor could be used for this project.

#### NUCLEAR POWER USE AND ECONOMICS

Economics is the main factor in selecting the best combination of processes and energy source alternatives. However, there are other factors which ought to be considered but are difficult to be represented by economic figures.

Factors such as social effects, environmental impact, technology transfer, industrial advancement, and infraestructure improvements, cannot be fully assessed under pure economic numbers. Nuclear energy introduction to a country, it has been demonstrated, has a lot to do with those aspects, with positive effects in most, if not all, the cases.

Recent events in nuclear power use in the world, are forcing the development of safer, simpler, smaller, standarized reactor concepts, a tendency pointing out toward higher economic competitiveness of nuclear energy. Also, an extra factor is that for nuclear energy to make a real important contribution in the world energy scenery, it has to compete not just in electricity production but in processing heat applications.

Besides the above considerations, in Venezuela there are several factors favoring a future application of nuclear power. In this study, the technological feasibility for the application of advanced nuclear energy concepts in the OOB heavy crude exploitation is presented as a valid option.

At this early stage of this study, an economical analysis would present large uncertainties due to the presence of still unknown parameters. These parameters include from oil deposit characteristics to the cost of nuclear power produced by advanced nuclear reactors, including, in between, future oil prices.

Is for those reasons that it is strongly recommended to continue studies in order to better define the factors involved and, so, to be able to perform a feasibility study including a detailed economic analysis.

Certain postponement in the development of the OOB, produced by current oil market conditions, could be better seen as an opportunity to consider in more detail the application of advanced nuclear reactor concepts. Commercial use of these new type of reactors might be current at the time when needed in this project.

Then, one of the main purposes of the related studies taking place at IVIC, is to follow the development of such advanced concepts to be able to properly consider their posible application in the OOB exploitation, as well as to suggest the proper actions required.

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High temperature gas cooled reactors have also the capability for several other applications not discussed here in any detail, and difficult to achieve with other type of reactors. Processes in the steel, the petrochemical and the aluminum industries, long distance energy transportation (through "chemical pipes"), high efficiency electricity production, and others, with good perspectives in Venezuela (Carvajal, 1988), are only possible with the reactor types here discussed.

HTG reactors also present excellent safety characteristics, something of great importance for public acceptance, opening the possibility of their installation closer to industrial centers. Their simpler designs imply much shorter licencing and construction times, reducing costs and, so, making them competitive in smaller sizes, reducing, in consequence, the associated economic risks and increasing their versatility.

However, to define a specif type of reactor, more detailed studies should follow.

Extensive literature can be found about HTGR's (NED, 1984; Arndt, 1988), with several plant design studies completed for different steam and process heat applications, including enhanced oil recovery (IAEA, 1984; TRS, 1984).

Indonesia and China are carrying out specific studies for heavy oil recovery applications using HTG reactors, but no published papers were found. Canada has considered an organic cooled Candu reactor for oil recovery from the Alberta's tar sands (Puttagunta, 1977) but with no heat production for processing.

Great Britain completed studies for the OOB case, again only for the extraction process (Perret, 1981). In this study, where Magnox technology was considered, economic competitiveness, with respect to burning at that time high-priced oil, was indicated for an oil to steam production ratio bellow 20 barrels per ton.

## CONCLUSIONS

Good perspectives are seen for nuclear power application in the ODB development, both for the oil extraction and the upgrading processes. In this case, high grade heat and steam are required, ruling out the possibility of employing conventional light water reactors. So, in case of considering the nuclear option, advanced nuclear reactor concepts, such as the hightemperature gas cooled reactors, are the only ones capable of satisfying the requirements.

Advanced nuclear power systems are still under development, some with prototypes already running. However, since the OOB development in large commercial scale is not going to be carried out in the near future, there is enough time to perform the necessary studies to get properly prepared, though the complexity of such a project calls for a prompt action.

It is recommended to establish a model as close as possible to an especific real case in order to narrow design parameter ranges. Then, particular nuclear plant design options should be included, selecting the most appropriate ones, with economic analysis following in a prefeasibility study.

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# APPLICATION STUDIES OF HTR IN THE PEOPLE'S REPUBLIC OF CHINA

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## Abstract

This paper describes the energy situation, uneven distribution of energy resources and possible applications of HTRs in China. The HTR application in China could be separated into two stages. In first stage the HTR with helium temperature of 750°C can be used for producing electricity and/or process steam for thermal recovery of heavy oil, petrochemical complexes and concentrated industrial areas. In second stage the HTR application in China, will provide high temperature of process heat (up to 950°C) for liquid and gaseous fuel productions.

It has been studied that about 200-250 MWe units of modular HTRs with a steam generating capacity in the range of 250 to 350 tons per hour can meet the requirements of steam consumption in the thermal recovery oil fields, which are at present one-sixth of the total chinese petroleum reserve, and it results in the annual savings of approximately 0.3 million tons of oil.

This paper also describes that a study has been started jointly by the China and Federal Republic of Germany, for the application of HTR in petrochemical industry.

Design requirements of the HTR for future applications in China such as excellent inherent safety, flexibility of fuel cycle, simple and domestic fabrication of fuel elements, generation of low pressure, high pressure,superheated steam and appropriate capacity (200-350 MW(th), have been also discussed in this paper.

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#### 1. THE ENERGY SITUATION AND THE POSSIBLE APPLICATIONS OF HTR

China is a developing country with rich energy resources, especially coal and hydropower. At present, China is self-sufficient in all energy products and has a small part for export. Total production of commercial energy in 1987 reached 910 x  $10^6$  tce (ton coal equivalent, 1 kgce = 7000 kcal), in which raw coal amounted to 920 x  $10^6$  ton, crude oil 134 x  $10^6$  ton, natural gas 13.7 x  $10^9$  M<sup>3</sup> and electricity production 496 x  $10^9$  kWh (including hydropower and thermal power generation).

On the other hand, the present energy supply still cannot meet the demands of the rapid national economic growth and the household consumption. The average per capita consumption of energy in China is still much lower than that of the world average. Contradiction between energy supply and demand, particularly shortage of energy supply and electricity in the most industrial area, such as northeastern and eastern part of China and Guandong province, exists because of the uneven distribution of energy resources. About 60% of the coal resources are found in North China and 70% of the hydropower resources are concentrated in the Southwest. It results in a serious problem of long-distance transportation of energy resources.

In order to supplement the energy demand, it has been decided in accordance with the strategy of developing nuclear power, that the PWR be the basic reactor type in the near future. But in view of the development of nuclear energy in the next century a choice has to be made to develop a safe, economical and highly fuel efficient type of reactor. The high temperature gas cooled reactor (HTR) is one of the advanced reactors to be decided by our Government to do the R and D work for future applications. In the HTR R and D programme, some topics are covered such as, fuel element technology, graphite development, metallic and ceramic materials, helium components, fuel element handling, instrumentation, engineering and reactor design, test facilities, HTR safety as well as fuel reprocessing technology.

In parallel an application study on the HTR module will be carried out in order to investigate the technical and economic feasibility for relevant Chinese oil and petrochemical industries, as well as power generation in special areas which have small power grids, and to investigate the potential of applications on a long-term basis. Some users are involved for example, Shengli oil field for heavy oil recovery by steam drive, Yan-shan petrochemical company for process steam and electricity generation application, as well as Zhong Qing city and Hainan province for generating electricity only or for cogeneration and so on. These application studies have been started by a special unit which involves INET, BINE and other institutes as well as supported by industry and local governments.

provided that the HTR is able to produce high temperature process heat of more than  $950^{\circ}$ C and, considering the technical development situation on the conventional and nuclear industrial field, the HTR application in China could be separated in two stages:

The first stage, the HTR with helium temperature of 750°C can be used producing process steam or/and electricity. The main application users might be:

- Providing process steam (mainly) for the thermal recovery of neavy oil, cogeneration is also possible.
- Cogeneration plant to provide process steam and electricity for petrochemical complex and concentrated industrial area.

- A small HTR power plant with a capacity of 300MWe to meet small grid capacities in special areas, where coal is not available, the transport of coal is difficult or environmental pollution is serious. A high efficiency conventional turbine generator can be used.

The second stage of HTR application will provide high temperature process heat (up to 950°C) for liquid and gaseous fuel production, such as coal gasification and liquefaction, shale oil production, but all these processes are high energy-intensive. Application of high temperature nuclear heat in this field will remarkably reduce consumption of oil and coal in such processes. Therefore, the use of nuclear energy as a substitute for oil and coal would be an important way to reduce the fossil fuel consumption and to get economic benefit in the future.

#### 2. THE STUDY OF HTR INDUSTRY APPLICATIONS

## 2.1 Application Study on Heavy Oil Recovery

The heavy oil geological reserve in China is very rich. At present the quoted figure of the heavy oil resources constitutes about one-sixth of the total Chinese petroleum reserve. From a medium and long-term point of view, the enhanced oil recovery will be increasingly important in future petroleum extraction. Conventional thermal recovery (by steam drive) needs a great quantity of steam with high temperature and high pressure. About 30-40% of the produced crude oil will be consumed to supply such an amount of injection steam.

This has fostered the idea of substituting nuclear energy for fossil fuel to generate oilfield injection steam. There are different incentives, with varying importance depending on the situation of the specific oilfield:

- economic advantages due to low steam costs,
- avoidance of air pollution like CO2, NOx, SOx,
- better yield of the oil resources,
- to solve the transportation problems for other low-cost fossil fuel like coal,
- to increase the amount of crude oil which can be posted abroad for foreign exchange.

Therefore, the units of the modular HTR with a high safety standard and steam generating capacity in the range of 250 to 350 tons per hour (reactor thermal power is about 200-250 MWe) can meet the requirement of steam consumption in the thermal recovery oilfield.

This application study has been started in 1986, the Shanjasi heavy oil section of the Shengli oilfield has been selected as a reference field for HTR application project.

The heavy-oil deposit Shanjasi is one of the Shengli oil fields which are located in the delta area of Huang-Ho River in Shangdong province. The production of all fields reached 30 million tons of crude oil in 1987. An increase of approximately 4 million tons/year is anticipated.

The proved initial OIP in Shanjasi section is 66 million tons, but exploration is not finished and 100 million tons are expected.

Shanjasi oil is a very heavy and viscous crude which does not flow at reservoir conditions. Oil has been produced by conventional steam injection (steam soak operations) since 1984. The steam (P=170 bar, t=355°C) is supplied from small air-fired boilers with capacity of 10-20 tons steam per hour. The general reservoir data are the following:

Depth, m	1110 - 1200
Original reservoir pressure, bar	110 - 120
Sp. gr. of crude oil, <sup>O</sup> API	13
Initial reservoir temp., <sup>O</sup> C	55
Viscosity of degassed oil	
Under reservoir conditions. CP.	8000 - 10000
Porosity, %	28 33
Permeability, md	500 - 3500
Initial oil saturation, %	65
Average thickness of the	
Pay zone, m	28 - 85

In this strategy the HTR delivers steam for approximately 34 years, when the main phase of oil production continues for about 20 years at a level of one million tons of oil yearly. The recovery rate was set at 55% according to the simulation results (20% by steam soak, 27% by steam drive, 8% by waterflooding).

The proposals for field development are using 2 HTR with total thermal output 400 MW for steam generation. The main data for heavy oil recovery with 2 HTR are following:

Modular HTR	2 units
Total thermal output	400 MW
Steam generation (100-170 bar, 350 <sup>0</sup> C)	520 t/h
Min. required number of injectors	35
Steam-oil-ratio	4 : 1
Produced oil	3120 t/d
Output of one producer	60 t/a
Min. required number of producers	52
Oil production per year about	$1 \times 10^{6} t/d$

Using the HTR nuclear steam system instead of the oil-fired conventional steam generators can annually save approximately 0.3 million tons of oil. It would be much more important for the national economy.

2.2 Application Study of HTR in Petrochemical Industry

Provided the HTR is able to generate steam with temperature of  $540^{\circ}$ C and pressure of 190 bar, the total range of process steam for the petrochemical industry can be supplied.

Yangshan Petrochemical Corporation (YSPC) has been chosen as a reference user for this study. The main purpose of the HTR plant is to combine the supply of steam and electricity in order to replace nuclear energy for the oil consumption in the conventional boilers. Yanshen Corporation supports this study and gives the necessary information and advices.

YSPC is located in Fangshan district in South-West of Beijing. The straight line distance from Beijing center is about 55 km. It is a petrochemical industry centre which has a refinery and different kinds of chemical plants. The refinery of Yanshan is one of the largest in China, having an annual crude oil refining capacity of 7 million tons. The annual energy consumption for supplying steam, process heat and part of electricity in this huge petrochemical complex is in the order of 1.2 million tons of oil. Saving oil consumption is an important policy at the moment and in the future in China. Therefore, the consideration to use an HTR nuclear energy source instead of conventional boilers to save oil consumption seems to be attractive and beneficial.

The total requirement of steam in different pressure and temperature ranges is approximate: 730 t/h in summer and 1650 t/h in winter. The main steam consumers are the subsidiaries of the refinery, and some chemical plants. The steam parameters are 118 bar/450°C, 47 - 50 bar/133°C, 8 - 13 bar/280°C (main requirements) and 3 bar/133°C for heating. The steam is supplied mainly by combined steam-electricity production (cogeneration). Annual consumption of electricity is about 1 billion kWh. The total electricity supply capacity in this area should be up to 120 MWe, which will increase in the near future due to the expansion of the production scope of the company. Electricity is supplied mainly from the local power system (utility), and partly supplied from own cogeneration power plants.

Part of the energy (steam and electricity) production facilities have been operated for more than 17 years. For several reasons (availability, reliability, environmental protection and fuel saving etc.) they have to be replaced within the next decade.

Considering the energy demand, there is at least the potential of a cogeneration plant consisting of four HTR modules suited for the Yanshan area. The main parameters are:

-	Total thermal power output of plant	800 MWth
-	Live-steam mass flow approx.	1000 t/h
-	Live-steam pressure/temp.	190 bar/530°C

The layout and design of a cogeneration extraction backpressure turbine could be adopted to the required steam mass flow and quality conditions of the user.

Taking a capacity factor of 80%, the oil substitution potential is about 0.5 million tons a year. In principle the two alternative uses for the substituted oil are:

- for export purposes as an excellent "foreign currency earner" to finance both the import portion and the domestic part of the investment,
- as a feedstock for the new petrochemical production unit at YSPC.

The application studies have been started jointly by the INET, Beijing, China and KFA, Jülich, FRG, supported by Chinese companies (Shengli oilfield, Yanshan Corp. and relevant manufacture companies) and German companies (KWU/Interatom and HRB, etc.).

The preliminary study on the HTR applications will be carried out and include the following topics:

- The technical proposal to use the HTR as a substitution energy source by industrial users, e.g. process schemes.
- Proposal for the domestic fabrication of partial HTR components.
- First economic assessment of the HTR application, mainly investment costs, steam and electricity production costs and financing possibility.

### 3. THE DESIGN REQUIREMENTS OF THE HTR FOR FUTURE APPLICATIONS

For an HTR nuclear steam supply system application under Chinese conditions, the following requirements might be of importance:

- 1. The appropriate capacity: in most cases the HTR plant will have a small and medium size with thermal output of 400 - 1000 MW. It can meet the energy requirements of the heavy oil fields and the typical petrochemical industry centres as well as the special local grids. Therefore, the modular HTR unit each with a capacity of 200 - 350 MW(th) might be an appropriate design.
- The steam parameters should be suited for industrial processes or power generation. It must be possible to generate low pressure steam as well as high pressure, superheated steam, e.g. of 190 bar, 540<sup>o</sup>C.
- 3. The safety properties should be suited for industrial sites. Even under hypothetical accident assumptions, a dangerous release of radioactivity should be impossible. An HTR design with excellent inherent safety features is important.
- 4. The flexibility of the fuel cycle should be considered. The HTR can not only use low enriched uranium fuel but also highly enriched uranium with thorium fuel. Investigation of uranium-thorium fuel cycle is necessary under Chinese conditions.
- 5. A simple design and domestic fabrication of fuel elements are required. The technology of spherical fuel elements is being developed in China.
- 6. Design conditions of the high process heat applications, such as for coal gasification and liquefaction, are necessary to investigate the long-term HTR applications.

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# SLOWPOKE HEATING REACTOR: TECHNICAL AND ECONOMIC REVIEW\* (Abstract)

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Since global energy requirements are expected to double over the next 40 years, nuclear heating could become as important as nuclear electricity generation. To fill that need, AECL has designed a 10 MW nuclear heating plant for large buildings. Producing hot water at temperatures below 100°C, it incorporates a small pool-type reactor based on the successful slowpoke research reactor. A 2MW prototype is now being tested at the Whiteshell Nuclear Research Establishment in Manitoba, and the design of a 10 MW commercial unit is well advanced. With capital costs in the range 5 million dollars to 7 million dollars, unit energy costs could be as low as 0.02 dollars per KWh, for a unit operating at 50% load factor over a 25-year period. By keeping the reactor power low and the water temperature below 100°C, much of the complexity of the large nuclear power plants can be avoided, thus allowing these small, safe, nuclear heating systems to be economically viable.

If we can achieve our immediate goal of licensing and gaining public acceptance for a low-power, unpressurized heating reactor, in an urban environment, an important precedent will be established for the future installation of more advanced commercial reactors, for electricity generation and other applications. Conversely, if a small, simple reactor cannot be licensed or cannot gain public acceptance, detailed technical and economic studies of more advanced concepts are irrelevant.

 $<sup>\ ^{\</sup>ast}$  The presentation was based on the following AECL publications:

District Heating with Slowpoke Energy Systems G.F. Lynch AECL-9720, Chalk River, March, 1988

<sup>2</sup> Slowpoke-Heating Reactors in the Urban Environment J.W. Hilborn and G.F. Lynch AECL-9736, Chalk River, June, 1988

# CANDU 3: TECHNICAL REVIEW AND STATUS\* (Abstract)

R.S. HART Advanced Concepts, Atomic Energy of Canada Limited, Mississauga, Ontario, Canada

CANDU-3 is the latest and smallest Canadian PHWR for electricity generation. Rated at 450 MWe, CANDU-3 offers a highly flexible plant configuration, adaptable to a wide range of different user requirements and sites. Simplified component installation and extensive use of pre-assembled modules has shortened the construction schedule to less than three years. CANDU-3 should have particular appeal to countries without hydro or economic fossil fuel resources, countries with small or subdivided grid systems, and countries wanting to develop a nuclear industry with minimum investment.

\* The presentation was based on the following AECL publications:

1.	CANDU 300 - the Next Generation R.S. Hart
	presented at: 28th Annual Conference Canadian Nuclear Association Winnipeg, Manitoba, Canada June, 1988
2.	Advances in Engineering and Construction
	presented at: ANS/ENS 1988 International Conference Washington, D.C., U.S.A. October - November, 1988
3.	Candu 300 - Technical Outline Document PPS-74-01010-003 AECL, Canada
4.	Designing for Plant Life Extension R.S. Hart
	presented at: Topical Meeting on Nuclear Power Plant Life Extension Utah, U.S.A. July - August, 1988

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## MODULAR HIGH TEMPERATURE REACTOR FOR VARIOUS APPLICATIONS\* (Abstract)

(Abstract)

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The HTR-Module is designed as an unsophisticated, safe and economic heat source for various applications. The reactor can be coupled without major modifications to a steam generator for the production of steam and power or to steam reformers or intermediate heat exchangers for the direct use of nuclear heat. Due to its special design the HTR-Module achieves an extremely high passive safety level. This allows its erection at industrial sites in populated areas.

<sup>\*</sup> The presentation was based on the publication of Kerntechnik 52 (1988) No.2; pages 96-100.

# RATIONAL RESTART OF SAFER FISSION ENERGY UTILIZATION DEPENDING ON THE SMALL THORIUM MOLTEN-SALT POWER STATIONS

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## Abstract

In the next century, (a)environmental impact of  $CO_2$ ,  $SO_x$ ,  $NO_x$  and  $CH_4$  from fossil fuel systems, and (b)radiowaste and nuclear proliferation (terrorism) troubles mostly coming from trans-U elements(including several thousand tons Pu) in the U-Pu solid-fuel nuclear power-stations would become severe problems in the world.

To get more rational world-wide applicable power-stations as civilian utility facilities, not only the above problems (a) & (b) but also the followings should be solved:

- (c) resource, (d) technological safety & simplicity,
- (e) flexibility in size and siting, and
- (f) integral economy in fuel cycle.

A new strategy named "THORIMS-NES" [Thorium Molten-Salt Nuclear Energy Synergetics] is being proposed by us, which composes of the three technologies: [I]Thorium, [II]Molten-Salt and [III]separation of power-stations and fissile breeding plants, refusing fission breeders.

The typical 155MWe Small Molten-Salt Power Station[FUJI-II] has significant characteristics as follows:(i)no needs of graphite exchange and continuous chemical processing except fission-gas removal, and (ii) fuel self-sustainability(no fissile supply) in full life after initial 500 days operation, which mean very low excess-reactivity (nearly no control-rods), few fuel transport, no core melt-down, easy operation and maintenance, few radiowaste and thermal pollution, high thermal efficiency and medium-temperature heat supply, etc. mostly achieving the targets mentioned in the aboves.

This FUJI could be developed among this century by a small cost, smoothly joining with the present energy technologies (including the molten Na reactor technology).

As stated by Mr. David Lilienthal<sup>1</sup>, we have to restart for the establishment of more rational fission energy systems worldwide applicable in the next century. These systems should be improved in solving the following problems:

- [A] "natural" safety,
- [B] nuclear proliferation resistance,
- [C] universal resource,
- [D] flexible power-size, and
- [E] excellent economy.

The nuclear energy is the nuclear chemical reaction energy, and nuclear energy systems should basically behave as nuclear chemical engineering systems suitable for accomplishing a practical fuel cycle.

However, all established fission reactors are depending on the solid uranium(plutonium) fuel technology, which is not suitable only for solving the problems of  $[A \sim E]$  but also for fuel managements including chemical processing.

Therefore, the rational nuclear energy system should depend on the new design philosophy applying such as a <u>thorium fluid-</u> <u>fuel</u> concept and introducing another principle to solve [D], which would not be satisfied by <u>Fission Breeder</u> Power Stations such as LMFBR or MSBR widely recognized as an ideal target in the present fission community.

Now, the new philosophy should be applied depending on the following three principles:

[I] Thorium utilization,

[II] Application of fluid-fuels, esp. molten-salts, and

[III]Separation of fissile-breeding and power-generating functions.

This philosophy was named as <u>"THORIMS-NES</u>": ["Thorium Molten-Salt Nuclear Energy Synergetics"]<sup>2</sup>.

Thorium molten-salt reactor concept --- New Philosophy---

The principle [I], Th utilization, has been basically chosen from the wide benefits in problems [A]  $\sim$  [E]. In contrast with highly localized U resource, Th is a non-localized(non-monopolized) and 3  $\sim$  4 times abundant resource held by almost all high-population countries except Japan, France etc.. The fissile <sup>233</sup>U coming from fertile <sup>232</sup>Th(100% in natural abundance) has a strong gamma-activity in practice, which is highly effective for safeguard and not suitable for nuclear weapons. The new technology depending on this <sup>232</sup>Th-<sup>233</sup>U fuelcycle would be able to establish nearly non-production and effective incineration of trans-Uranium elements(Pu, Am, Cm), which have too high mass-numbers for producing from <sup>232</sup>Th. It would be a crazy situation that Pu of several ten thousand tons should been distributed all over the world in the next century. The general benefits of [I] will be explained in later.

The next problem is how to realize the Th utilization, which did not succeed so long, even though disturbed by the existence of strong military interest on U-Pu cycle. Its solution should be based on the choice of the principles [II] & [III].

The principle [II] had charmed the nuclear energy community in early developmental stage, but almost all concepts of them had come to deadlocks after the long struggles. However, the molten-fluoride concept, especially the "Flibe(LiF-BeF<sub>2</sub>)" base molten-salt fuel concept developed by the brilliant long effort of Oak Ridge National Laboratory(ORNL), USA, among 1947-1976 could establish its sound technological basis. It depends on the success of MSRE (Molten-Salt Reactor Experiment, 7.5 MWth experimental reactor operated in 1965-1969) and their intense R & D works named <u>Molten-Salt Reactor Program</u> (MSRP: 1963-1976). Their final technological status is explained in their reports<sup>3</sup>, on which some improvements were added by France, USSR, India, Japan etc..

The significant technological characteristics of this concept will be explained briefly in the next section(cf.Column 1).

ORNL had continuously searched the developmental way for the "thermal breeder" named **MSBR** [Molten-Salt Breeder Reactor ]. The most elegant design was summarized as the "<u>single-fluid</u> <u>type MSBR</u>" at 1970. However, its practical establishment is not easy due to the following difficulties, at least:

- (a) core-graphite exchange in every 4 years,
- (b) development of continuous chemical process in situ, and
- (c) further improvement of doubling-time for the fissile fuel supply enough in the next century<sup>4</sup>.

If the principle [III] could be accepted, the several benefits would be expected by the separation of two functions. The fissile producing breeders are "process plants" (not generating any electric power like as LMFBR or MSBR), and might have high performance by bigger size and by concentration of siting such as 10 sites in the world, which is effective for safeguard, too. The fission power stations are one kind of "utility facilities", and should be flexible in size(small, medium and large) and simple in operation and maintenance. Now several new design concepts would be able to apply on both. Already, three types of fissile producing breeders, and Small Molten-Salt Fission Power Stations were proposed essentially guaranteeing the establishment of THORIMS-NES as explained in later.

## Some important technological aspects

Fuel salt: As mentioned in the above the most successful MSR concept is depending on molten fluoride fuel based on the Flibe('LiF-BeF<sub>2</sub>), which is a solvent salt chosen from the indication of low thermal neutron absorption cross sections of constituent elements (cf.Table 1). The molten Flibe is surprisingly similar to the molten MgO-SiO<sub>2</sub> in several physico-che-

			(natural abundance)	c I	ross-se m barn	(natural	element)
1.	80				0.19		
2.	<sup>2</sup> 1 <sup>H</sup>	[D]	(0.0148%)		0.519	( <sub>1</sub> H	332.6)
3.	6 <sup>C</sup>				3.53		
4.	<sup>11</sup> 5 <sup>B</sup>		(80.0%)		5.5	( <sub>5</sub> B 7	67,000)
5.	2 <sup>He</sup>				7.36		
6.	4 <sup>Be</sup>				7.6		
7.	9 <sup>F</sup>				9.6		
8.	83 <sup>Bi</sup>				33.8		
9.	10 <sup>Ne</sup>				39.		
10.	7 <sub>3</sub> Li		(92.5%)	4	5.4	( <sub>3</sub> تىغ	70,500)
15.	40 <sup>Zr</sup>		185.	17.	1 <sup>H</sup>	332	2.6
16.	13 <sup>AL</sup>		231.	67.	17 <sup>Cl</sup>	33500	).

Table 1. Natural elements and practically separable isotope having tiny thermal neutron cross-sections.

mical characters except the melting point. [It should be recognized that MgO-SiO2 is a major component in the earth mantle holding unique working functions.]

In conclusion, this molten-fluoride concept might be the best and idealistic among several fluid fuels due to the following characteristics (cf.Column 1):

(1) the possibility as a multi-functional medium for the fuel, blanket, coolant and chemical processing, by holding the high solubility of several useful fluorides, except the fission-gases, which will be beneficially removable considering their nature as severe neutron absorbers (poisons) and environmentally hazardous radioactivity, (2) <u>no radiation damage</u> as an idealistic "ionic liquid",

(3) chemically inert, low vapor pressure, moderate viscosity and thermal conductivity, and high heat capacity in fairly high working temperature (500  $\sim$  800°C), promising the high safety and economy assurance.

Some important physical properties are shown in Table 2, comparing with the other typical liquids: water and Na.

- Column 1. Most Significant Characteristics of "FUJI" (SMALL MOLTIN-SALT FISSION POWER STATION) [# : unique in "FUJI-II"]
- [a] THORIUM RESOURCE UTILIZATION abundant  $3 \sim 4$  times than U non localized --- non-monopolized high performance in thermal neutron reactors
- [b] NUCLEAR PROLIFERATION RESISTANCE [no high gamma-radioactivity in <sup>233</sup>U fuel unsuitable for nuclear-weapon [no Plutonium] easier safe-guard
- <u>fuel self-sustainable</u> no transport of <sup>233</sup>U (except initial stage) incineration of trans-U elements(Pu. Am, Cm) # best & practical method(no need of R&D)
- [c] HIGH SAFETY
- $\sim 0.1$  % or less very low excess-reactivity fuel self-sustaining
  fuel self-sustaining
  alf-controllable
  following self-controllable
- power-load following reactor vessel ---sımple tank normal pressure, no opening, no control-rod ? no compatibility problems: salts ←→ graphite, Hastelloy N[N1-Mo-Cr alloy] triple confinement --- same as solid-fuel reactor reactor vessel high-temp.containment } ↔ { thin clad-tube × reactor vessel reactor container
- high-temp.containment } (reactor vessel
  reactor container
  in heavy accidents --- nearly no troubles
  fuel(coolant) flow stop; core≾1100°C only
  reactor non-stop / (high heat capa.of core)
  [reactor stop ← in boiling at 1500°C]
  in reactor destruction (military, terrorist)
  no explosion, no burst
- nearly no release of radioactive gases (Kr,Xe,T) leaked fuel --- automatic to drain-tank chemically inert, insoluble in water rigid glassy-solid (freezing at 500°C) suitable for radioactivity confinement
  - no core melt-down [no China-Syndrome]
  - no re-criticality
  - no fire or easy suffocation-extinguishing
- [d] ECONOMICAL <u>SMALL</u> POWER STATION ( Civilian Utility Facility )
- reliable cheap resource : (cf.[a]) easy safe-guard : (cf.[b]) high safety : (cf.[c]) simplicity in components & subsidiary facilities (favorable for smaller reactors)
  - no solid-fuel assembly: (fabri., inspect., storage, transport, dismantle, reprocess., refabri.,--) no fuel- & graphite-exchange or shuffling in core

- simple configuration of reactor vessel easy reactor operation (easy restart) easy remote maintenance no continuous chemical processing(batch process) no U-enrichment facility (↔fissile producers) few size-indep.fac. such as fuel handling facil. high efficiency in electric generation : 44~46 % few thermal pollution & few radio-waste(no transU) [~60 % of LWR] [few maintenance works] easy utilization of medium-temp. heat(600-700°C)
- (by surface grinding, 0.5mm in depth) fuel self-sustaining(excep.initial transient stage) no fuel supply & transport (except fertile-Th)
- simple fuel cycle (idealistic in coupling with fissile breeders)

[e] RATIONAL R & D COST AND PROCEDURE

simplicity in design-principle and configuration no radiation damage in fuel(no need irradi.test) no core-graphite exchange fuel self-sustainable--no control-rod? ↔ [MSBR] no continuous chemical processing (ORNL) applicability of excellent results of MSRE [Experimental Reactor operated in 1965~69,ORNL] similar in size as miniFUJI(pilot-plant)
few additional R & D items : mechanical data of modified Hastelloy N, high-temp. containment technology, steam technology for electric generation, fuel-salt impurity monitoring, etc. applicability of <u>liquid-Na technology</u> results (especially in large components) similar molten-material technology : high-temperature and low-pressure easier than Na-technology : chemically inert, very low thermal shock, no oxidized vapor condensation, lower flow-rate (due to high heat-capacity) predictability in Molten-Salt Chemistry
fuel salt = idealistic "ionic liquid" : ionic interaction 5 electro-static, classical 

	Na (500°C)	LiF-BeF <sub>2</sub> (Flibe) (500°C)	#LiF-BeF <sub>2</sub> -ThF <sub>4</sub> (635°C)	*NaBF <sub>4</sub> -NaF (500°C)	H <sub>2</sub> O (300°C)
chemical composition		66-34	72-16-12	92-8	l
melting point (°C)	98.	459.	499.	384.	0.
vapor pressure (mmHg)	4.	<0.1	<0.1		6.4x10 <sup>4</sup>
density (kg/m <sup>3</sup> )	833.	2050.	3330.	1870.	1 712.
thermal conductivity	158.	2.39	1.23	0.84	1.29
kinematic viscosity	0.29	1 · 7.44	3.0	0.8	0.13
specific heat capacity 10 <sup>-6</sup> X(J/m <sup>3</sup> K)	1.05	4.79	4.53	2.82	, 4.11

Table 2. Some important physical properties of typical liquid materials including the fuel salt# and secondary coolant salt\* for MSR

(#) The effect of fissile component is negligible.

Graphite: As a neutron moderator and reflector, graphite is the best material for thermal reactors and is confirmed to be compatible with molten fluorides in bare state.

However, the penetration of salt into micro-pore should be protected by limiting the pore diameter less than 1  $\mu$ m, and the neutron irradiation limit should be 3 x10<sup>22</sup> nvt (>50 kev). Satisfactory materials have been developed by the efforts of USA and France<sup>5</sup>.

Containment materials: Fortunately, an easy manufacturable, weldable and high temperature resistant (till 850°C) alloy compatible with molten fluorides has been developed basically depending on the proposal of Dr.H.Inouye, ORNL. This alloy, Hastelloy N (Ni-[15 $\sim$ 18]-[6 $\sim$ 8]Cr in weight %), has been modified adding about 1% Nb for the protection of Te attack on surface. This was experimentally supported by Kurchatov Institute.

This alloy needs not to be introduced in reactor core region suffering any severe irradiation and thermal shock. [MSR core occupied by graphite and fuel salt only.] The most thin material will be one or 2/3 inch diameter tube in the heat exchanger. Therefore, the reactor design would be quite simple and easy.

The triple confinement of fuel-salt is soundly established by (1) reactor vessel,(2) high-temperature containment and (3) reactor container, in contrast with the solid-fuel reactors, which are also depending on the triple confinement, but the troublesome thin clad-tubes are placed in the high neutron flux, high flow velocity and high thermal shock resion of core.

**Reactor chemical aspects:** This problem has been beautifully examined and basically solved classifying the F.P.elements to the 4 groups concerning their chemical behavior;

group(1): rare gas elements--- really no solubility

group(3): noble(insoluble) metal elements such as Mo, Nb--group(4): unstable salt elements such as Te, O, H, D, T---

Group(3) elements behave as [i] undissolved floating materials(shifting to cover gas system) and [ii] plate-out materials on graphite or [iii] on metal surfaces. Their mass ratio were 50 : 10 : 40 % in MSRE. Some part will be filtrated out. The plate-out materials on graphite will be removed for reuse by grinding 0.5mm in depth.

In group(4) elements, Te will induce a shallow surface brittleness on Hastelloy N. However, it was protected by controlling electro-chemical redox-potential of fuel salt, and modifying the alloy composition by the addition of about 1% Nb. Tritium will be produced in a fairly big amount such as 1 Ci(=37GBq) per MWth·day, which release to environment should be minimized to  $10^{-3}$  in the order of magnitude. This problem was also solved by ORNL: T will be effectively transferred to coolant salt NaF-NaBF<sub>4</sub>(8-92 mol%) through the heat-exchanger tube-wall exchanging with H of water content (~200ppm) in coolant salt, and will be recovered in He cover gas phase.

In practice, the corrosion of Hastelloy N by introduced contaminants such as air and moisture was surprisingly small and negligible even at the small experimental reactor MSRE.

Commercial power stations would be much safer in corrosion.

General feature of MSR technology: Molten-fluoride technology in general seems to have an exotic look. However, it is much simpler and more rational than liquid Na reactor technology, although these are both low pressure and high temperature molten material technologies. The liquid Na technology has several weak points relating with (i) high chemical reactivity, (ii) high thermal conductivity inducing severe thermal shock on structural materials, and (iii)oxidized vapor condensation in cover gas space. These phenomena need not worry in MSR technology, and its high heat capacity will be effective for reducing the flow rate to one fourth in Na (cf. Table 2).

The almost all results elaborated in Na cooled reactor development could effectively contribute for molten-salt reactor development, especially concerning their large component developments.

## Small Molten-Salt Power Stations

Following to our new philosophy, a design study of Small Molten-Salt Fission Power Stations such as FUJI-II was proceeded in recent 4 years, getting the excellent results<sup>2,6</sup> than our expectation, as follows:

(i) a single-fluid type graphite-moderated Molten-Salt converter in power of 350MWth(155~161MWe),

(ii) no need of any core-graphite exchange in full life,

(iii)no need of any continuous chemical processing except removal of fission-gases and tritium, and

(iv) an excellent performance such as a fuel self-sustaining characteristic, which means no necessity for fissile-supply [except Th supply of 400g/day] after the initial transient stage of about 500days, in such a small reactor.

(v) a simple structure assuring a weld-sealed simple reactor vessel(no big flanges), and easy operation and maintenance.

These characters might be useful for minimizations of excess-reactivity (assuring nearly no control-rod design), neutron loss, operational mistakes, fuel transportation, radio-waste amount etc., and for improvements of safety, eco-



Figure 1. Conceptual figure of Small Molten-Salt Reactor "FUJI-II"(350 MWth) contained in the high-temperature containment

nomy, nuclear-proliferation resistance, and performance in coupling with fissile-producing breeders (due to nearly no need of fissile except initial stages).

More detailed characteristics of FUJI-II have been listed in **Column 1.** Its conceptual figure and some numerical data have been shown in **Figure 1** and **Table 3.** Its highly significant aspects of safety could be recognized from the several relating indications.

## Developmental program of small Molten-Salt Reactor, "FUJI"

This program consists of the three phases (cf.Figure 2). The first <u>Short Term Program</u> is the construction of supercompact Molten-Salt Reactor of 7MWe (miniFUJI-II) as a pilotplant. The second <u>Middle Term Program</u> is the construction of the standard Small Power Station such as 155MWe one (FUJI-II).

	standard power station (fuel self- sustaining) FUJI-II	pilot-plant (super- compact) <u>miniFUJI-11</u>	experimental reactor(ORNL) (operated in 1965~69) <u>M S R E</u>
heat capacity (MWth)	350.	16.7	7.3
electric power(netMWe)	161.	7.	
thermal efficiency(%)	46.	42.	
reactor size (m) (diameter x high) high temp.containment	5.5x 4.1 12. x 8.	1.8x 2.1 3.7x 3.2	1.45x 2.2
fuel: conversion ratio	1.002	0.58	
233U inventory(kg)	370.	27.	32.
$232_{\text{Th}}$ (ton)	20.1	0.65	
fuel salt: <sup>233</sup> UF (mol%)	0.22*	0.47*	0.14
total volume (m <sup>3</sup> )	13.7	0.45	2.1
flow rate (m <sup>3</sup> /min)	33.2	1.59	4.5
temperature (C)	585 ~ 725	560 ~ 700	632~654
main piping(inn.dia.cm)	25.	8.	15.
(*) $^{7}$ LiF - BeF <sub>2</sub> - ThF <sub>4</sub>	$-^{233}$ UF <sub>4</sub> = (72-3)	x) - 16 - 12 -	x [mol%]

Table 3. Main parameters of several Small Molten-Salt Reactors

The third Long Term Program is the establishment of integral fuel cycle system [THORIMS-NES] by developing the fissile producing breeders, with which several size of Molten-Salt Power Stations are coupled effectively solving the problems  $[A] \sim [E]$  until the early next century (2010-2030).

GENERAL R & D: The basic R & D of molten-fluoride reactor technology was comprehensively established by ORNL, except the data-base of modified-Hastelloy N, and the confirmation of electric power generating technology, which is soundly supported by Ni-alloy technology. Therefore, for the reconfirmation of ORNL works and for the training of basic technology, the following R & D should be performed spending three years:

oFuel-salt chemistry by an integral test loop : for the studies of impurity monitors, F.P. behavior, mass transfer phenomena, etc.

oCoolant-salt chemistry by an integral test loop: for the studies of impurity monitors, tritium behavior, mass transfer phenomena, etc.

oIntegral steam generator test loop: for material tests and confirmation of electric power generating technology.

oMaterials data: for the examination of data-bases of modified Hastelloy N, Graphite and the others.

oConfirmation of the other **basic technologies:** salt preparation, chemical analysis, and batch chemical process, etc.

**PILOT-PLANT(miniFUJI-II):** After about 3 years' works for general R&D and reactor design, the construction of pilot-plant named miniFUJI-II could be started. Its reactor performances are shown in **Table 3**, and the size of reactor vessel is shown in **Figure 1** comparing with FUJI. The flow velocity of fuel salt in core is about 0.5 m/sec. This reactor is nearly same in size as **MSRE:** and the diameter of main piping is about 8cm, half of that of MSRE as shown in **Table 3**. It has a simple structure of closed reactor-vessel not exchanging any core graphite. Therefore, its construction is much more proven works.



The operation is easy in general and only including the 99% removal of fission-gases insoluble in salts, and supplying about Th 7.5g/d (=2.7kg/y) and  $^{233}$ U 5.8g/d(=2.1kg/y). The total demand of  $^{233}$ U is only about 30kg in initial stage followed by 2.1kg in each year. Its one third can easily be replaced by  $^{239}$ Pu of LWR aiming its incineration.

The main purpose of this pilot-plant operation is an endurance test of the integral facility. The monitoring of materials such as core-graphite and structural metals will be performed using a spare space of control-rods, which will be two. Several reactor-chemical examinations will be proceeded at the pump-bowl and drain tank of fuel salt.

SMALL MOLTEN-SALT REACTOR POWER STATION (FUJI): Tentatively FUJI-II (350MWth) is proposed as a standard one. However, it might be modified between  $200 \sim 500$  MWth in size not changing any basic characters.

Even in the early developmental stage of MSR, the smaller size one could be kept significantly high economy comparing with the small solid-fuel reactors, and the technological effort could concentrate on the small one at first, not necessarily competing with any proven larger-size solid-fuel reactors matured technologically. These small MSR might also be useful for medium temperature heat supply to industries and public as mentioned by Novicov<sup>7</sup>.

Fissile <sup>233</sup>U Fuel Salt Supply(in Early Immature Stage): In our program, the total demand of <sup>233</sup>U in this century is only



Figure 3. <u>THORIMS-NES</u>(Th Molten-Salt Nuclear Energy Synergetic System) COMPOSED OF <u>MOLTEN-SALT FISSILE BREEDERS</u>[MSB] AND <u>MOLTEN-SALT</u> REACTOR POWER STATIONS[MSR]---INCLUDING SOLID FUEL REACTORS

about 450Kg, in sum of 40 and 400Kg for miniFUJI-II and FUJI-II, respectively. Its preparation will not be difficult, because USA had produced  $^{233}$ U of total 540Kg: the sum of 40 and 500Kg for MSRE and LWBR[Light Water Breeder Reactor], respectively. The preparation of  $^{233}$ U was mainly done by Thorex process from Th oxide irradiated in reactors using Savannah River and Hanford facilities.

In future, spent solid fuels will be treated directly converting to molten fluorides by the **Dry Processing Plant** for fluorination, which is now successfully being developed at Demitrovgrad, USSR in the cooperation with France and Czechoslovakia. This will also be effective for the simplification of solid fuel cycle.

# <sup>233</sup>U Fuel Cycle: Coupling with Fissile Producing Breeders

As shown in Figure 3, each "Breeding & Chemical Processing (Regional) Center" settled about 10 sites in the world will accommodate 4 10 <u>Molten-Salt Fissile Producing Breeders(MSB)</u>, two <u>Chemical Processing Plants</u> and one <u>Radio-waste Managing</u> <u>Plant</u><sup>4</sup>. These Regional Centers should and might be heavily safeguarded.

For the next century, the following three types of Molten-Salt Fissile Producing Breeders (MSB's) have been proposed: a) Accelerator Molten-Salt Breeder (AMSB)<sup>4</sup>:

- by the neutrons generated from spallation reaction of Th nuclei with 1GeV protons,
- b) <u>Impact Fusion Molten-Salt Breeder</u> (**IFMSB**)<sup>8</sup>: by the application of new ideas of axially symmetric mass driver and shaped-projectile accommodating DT-pellet, and

c)<u>Inertial-confined Fusion Hybrid Molten-Salt Breeder(IHMSB)</u><sup>9</sup>: by the adoption of the first wall of molten-salt waterfall for the elimination of radiation damage.

These systems are aiming to achieve the straight preparation of the fuel salt concentrated to  $0.5 \sim 0.7 \text{mol}$ % in  $^{233}\text{UF}_4$  content, possible to supply directly to the above Molten-Salt Fission Power Stations fueled in low concentration( $0.2 \sim 0.3$  mol%). They should be developed until 2010-2030 of next century. In our preliminary opinion, AMSB will be the most sound and promising concept. However, IFMSB would be able to become a "dark horse" in some case.

## Summary

One of the practical and rational approaching methods for establishing the idealistic **Thorium resource utilization program** has been presented, which might be surprisingly effective to solve the principal energy problems [A]  $\sim$  [E], concerning <u>safety</u>, <u>proliferation</u>, <u>resource</u>, <u>power size</u> and <u>economy</u>, for the next century.

The first step will be the development of <u>Small Molten-Salt Reactors</u> as a flexible power station, which is suitable for early commercialization of Th reactors not necessarily competing with large proven Solid-Fuel Reactors. Therefore, the more detailed design works and practical developmental programming should be performed under the international cooperations soon, soundly depending on the basic technology established already.

This reactor(MSR) seems to be idealistic not only in power-size, safety, siting, nuclear proliferation resistance and economy, but also as an effective partner of <u>Molten-Salt</u> <u>Fissile Breeders(MSB)</u> in order to establish the simplest and economical Thorium molten-salt breeding fuel cycle named **THORIMS-NES** in all over the world including the developing countries and isolated areas. This would be one of the most practical replies to Lilienthal's appeal of "A NEW START" in Nuclear Energy<sup>1</sup>.

## ACKNOWLEDGEMENTS

The author wishes to express sincere thanks to many friends of USA, France, USSR, India, Japan and the other countries for their kind cooperations and encouragements.

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# THE MODULAR HIGH TEMPERATURE GAS COOLED REACTOR (PRISMATIC FUEL)

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## Abstract

This paper describes the characteristics of modular high-temperature gas-cooled Reactor (MHTGR) such as passive and inherent safety, economic competitiveness and its' suitability in many developing countries for electrical and process heat needs.

General Atomics (GA) proposes the consideration of a reference MHTGR which consists of four identical 350 MW(t) reactor modules and two turbogenerator sets to achieve a plant output of 538 MW(e). A vertical cylindrical concrete enclosure fully embedded in the earth is used for housing each reactor module.

Important advantages of this electrical generating system for a developing country have also been described such as simplicity, minimal investment risk, modularity, minimal transportation requirements for fuel supply, insensivity to price increases and process heat capability.

The performance of MHTGR is dependent on the reliable operation of major components/systems such as reactor vessel, helium circulation, steam generator, control rod drives, fuel and power conversion system etc. The advantage is that the technologies of these components/systems are well proven since 1956.

## 1. EXECUTIVE SUMMARY

The Modular High-Temperature Gas-Cooled Reactor (MHTGR) is a passively and inherently safe, second generation nuclear power plant. MHTGR technology and components have been proven in previous High-Temperature Gas-Cooled Reactors (HTGRs). Economic competitiveness has been achieved by elimination of the need for costly, engineered safety systems, by maximizing factory production of the components and by minimizing the need for field labor and assembly. Given the conventional (fossil-type) energy conversion area, minimal investment risk, smaller size of each module, and the simplicity, the MHTGR is particularly well suited to the electrical and process heat needs of many of the developing countries of the world.

The MHTGR achieves its inherent safety by use of a TRISO-coated ceramic fuel microparticle which retains its structural integrity and radionuclides up to very high temperatures (>2000°C). The core size, power density, and primary system configuration was selected to maintain safe fuel temperatures even under extreme conditions such as complete loss of flow, loss of coolant, pressure vessel failure, and rod withdrawal without scram. In fact, the system has been designed such that it can (with no active or passive cooling systems) reject its decay heat to the surrounding earth and maintain its structural integrity without compromising the radionuclide containment in the fuel particles.

General Atomics (GA) (formerly GA Technologies Inc.), the developer of particle fuel and prismatic graphite fuel blocks, proposes consideration of the Reference MHTGR plant which consists of four identical 350-MW(t) reactor modules and two turbogenerator sets to achieve a plant output of 538 MW(e). Each reactor module is housed in a vertical cylindrical concrete enclosure that is fully embedded in the earth. A common control room is used to operate all four modules and the turbine plant (Fig. 1-1). Summary design parameters of the reference four-module MHTGR plant are:

Thermal power	1400 MW(t)
Electric output	538 MW(e)
Net efficiency	38.4%
Steam conditions	538°C (1000°F)/16.6 MPa (2415 psia)
Core exit helium temperature	687°C (1268°F)
Cold helium temperature	259°C (498°F)
Core power density	5.9 W/cm <sup>3</sup>
Equilibrium fuel burnup	92,200 MW(d)/T

Each module consists of a reactor vessel and a steam generator vessel connected by a coaxial cross duct (Fig. 1-2). The reactor vessel is 22 m high and 7.4 m in diameter and contains the core, graphite reflectors, metallic core support structure, and radial restraining devices.


Fig. 1-1. Reference four-module MHTGR plant



Fig. 1-2. MHTGR side-by-side arrangement with prismatic fuel

A helical coil steam generator and the main circulator are located in the steam generator vessel, which is 25.8 m high and 4.2 m in diameter. The main circulator is mounted vertically on the steam generator vessel. A shutdown heat exchanger and a shutdown cooling circulator are located at the bottom of the reactor vessel, providing cooling for investment protection and to ensure high availability factors and quick maintenance turnarounds.



Fig. 1-3. MHTGR fuel components

The reactor core is graphite moderated and helium cooled, and uses prismatic fuel elements in the form of hexagonal blocks as employed in the Fort St. Vrain (FSV) HTGR (Fig. 1-3). The fuel consists of coated particles of 19.8% enriched fissile uranium oxycarbide (UCO) and fertile particles containing thorium oxide (ThO<sub>2</sub>). The particles are bonded together in fuel rods and are contained within sealed vertical holes within the graphite blocks.

The active core occupies an annular region surrounded by inner, outer, top, and bottom removable unfueled reflector blocks (Fig. 1-4). An annular core geometry was selected to maximize the unit thermal rating consistent with limiting fuel temperatures during accident conditions. The active core is 10 blocks (7.93 m) high, and the average active inner/outer core diameters are 1.65 and 3.50 m, respectively.

Reactivity control is provided by control rods which are individually supported by mechanisms located in penetrations above the reactor vessel and inserted into channels in the inner and outer reflector regions. Reserve shutdown capability is provided by insertion of FSV type boron pellets into separate channels in the inner row of active fuel elements. Refueling is accomplished with the reactor shut down and depressurized. FSV experience indicates that typical outages for refueling will be less than 15 days.



Fig. 1-4. Annular reactor core and control rod layout

Significant innovations have been incorporated in the balance of plant to realize acceptable economics. Features such as modularization with increased shop fabrication, below-grade silo installation, and distinct demarcation between safety-related and nonsafety-related systems (with conventional code requirements) are expected to lead to a reduction in overall plant site labor and in critical onsite craft requirements. The result is expected to be a much-reduced construction schedule (less than 4 years).

### 1.1. PROJECT COSTS

The costs for an initial and follow-on 4 x 350 MW(t) MHTGR plant have been developed under the U.S. program by GA, Bechtel National Inc. (BNI), Combustion Engineering (C-E), and Gas-Cooled Reactor Associates (GCRA). The costs presented herein were developed assuming that an MHTGR facility in a developing country would follow an initial plant in the United States. The follow-on plant costs were modified (reduced in most cases) to reflect the purchase of many of the components such as pressure vessels, circulators, turbine/generators, etc., from the most cost-effective international supplier as well as to reflect the lower costs of labor in a developing country. Under these assumptions, the proposed 538-MW(e) facility would cost approximately \$1.15 billion (including initial core and first reload fuel) or about 10% less than if built in the United States under the same seven-year project schedule. Annual costs for fuel will be \$43 million, with operating and maintenance (O&M) costs being \$17 million, at an 80% capacity factor. The levelized energy costs at a 9% discount rate should therefore be ~49 mils/kWh, making the MHTGR cost-competitive with other sources of electricity.

#### Contractual and Financing

GA anticipates that the MHTGR would be constructed by an international joint venture involving:

- GA as nuclear steam supply designer and fuel supplier.
- An international reactor vendor as the contractor of all foreign-supplied major components.
- An international architect/engineer as the overall plant designer and construction manager.
- The customer as supplier of all domestically available items.

GA envisions a turnkey contract between the customer and the joint venture supply organization with each of the partners under fixed price contracts for their scope of supply. The joint venture which would be formed to supply the MHTGR would be based on similar arrangements in the United States involving GA, Siemens/Interatom, and a major U.S. architect/engineer.

## 1.2. THE MHTGR IN A DEVELOPING COUNTRY

The MHTGR is an excellent choice as the cornerstone for building a dependable, economic electric generating system within a developing region, province, or country. Among the more important advantages are:

- Conventional nonsafety-related fossil energy conversion area which maximizes the opportunity for domestic supply of major balance-of-plant components such as turbine/generator sets.
- Simplicity which minimizes the need for highly trained operating personnel and exotic equipment.

- Inherent safety which ensures public safety even in the event of major equipment failure and operator error.
- Minimal investment risk which nearly eliminates the possibility that the generating capacity will be lost permanently as a result of an accident.
- Modularity which allows power to be added economically in ~135 MW(e) increments to more closely match increasing electrical needs.
- Minimal transportation requirements for fuel supply heavy railroads for coal or pipelines for oil to fuel the facility are not required.
- Insensitivity to price increases the impact of oil, natural gas, or coal price increases will be greatly diminished.
  Power costs are also very insensitive to uranium prices, since ore is less than 10% of total power costs.
- Process heat capability which would allow an MHTGR to be used for applications such as heavy oil recovery, steel making, and coal gasification.

A successful MHTGR program will give a developing nation a reliable electrical and process energy source so necessary for continuing economic progress.

2. "PROVENNESS" OF PERFORMANCE AND RELIABILITY OF COMPONENTS AND SYSTEMS

While the MHTGR concept is new, the technology on which it is based is not. More than 25 years of effort and over a billion dollars have gone into the design, testing, construction, and operation of systems and components directly applicable to MHTGRs. As a result, the experience base for its implementation is extensive and reliable.

The MHTGR draws on proven technology which has been demonstrated by more than 40 carbon dioxide-cooled reactors built and operated since 1956, and by the following five high-temperature gas-cooled helium reactors: the Dragon Pioneer Plant in England; the 15 MW Electrical AVR Experimental Reactor in Germany; the 40 MW Peach Bottom Plant in Pennsylvania; the 300 MW THTR Plant in Germany; and the 330 MW FSV Plant in Colorado.

The successful performance of the MHTGR is dependent on the reliable operation of the following major components/systems:

- Reactor vessel.
- Helium circulator.
- Steam generator.
- Control rod drives.
- Fuel.
- Power conversion system.

Steel vessels for nuclear reactors have been manufactured for over 30 years. They have performed exceptionally well in over 400 commercial reactors and in every nuclear submarine and aircraft carrier afloat. MHTGR vessels are comparable in size to current, large, lightwater reactor vessels; and the Dragon, Peach Bottom 1, and AVR HTGR have all operated safely and successfully with steel vessels. Their experience base is extensive and proven.

The technology base of the helium circulator includes the precedent carbon dioxide and helium-cooled reactors, and in particular, the FSV circulators which are similar in size and power. While the FSV circulators operated well and met all design specifications, the bearing seals and the support systems for the water-lubricated bearing caused many problems that resulted in poor plant performance.

To eliminate the potential for water ingress, the circulators for the MHTGR incorporate a magnetic bearing system. Such bearings are in operation in Europe, and directly applicable systems are currently being tested by James Howden and Company in Britain, HRB in Germany, and S2M in France. Today, over 45,000 h of magnetic bearing experience has been accumulated. In Germany, a small helium circulator with magnetic bearings has over 15,000 h of trouble-free operation.

The helically-coiled steam generators are located in the separate steam generator vessel. Thirty-four helical steam generators have been fabricated and successfully operated in nuclear power plants including

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FSV, the AGR, and the THTR. The performance history of these systems has been outstanding, with tube failure rates significantly lower than for water RSs.

The design of the control rod drive system used for reactor control and shutdown is derived directly from very similar cable/drum systems which have operated successfully in other gas-cooled reactors. Regardless of excellent control rod drive performance, however, the MHTGR incorporates additional safeguards. There are diverse backup shutdown mechanisms available: (1) passive control insertion by gravity, (2) boronated graphite balls for reserve shutdown, and (3) automatic shutdown due to a large negative temperature coefficient which results from the inherent physics of the reactor core. These performance characteristics, which are truly fail-safe, have been confirmed by testing at the AVR, Peach Bottom, THTR, and FSV reactors.

The use of coated fuel particles is one of the key elements to MHTGR safety. This technology has been advanced for over 25 years in both the United States and Germany. Minute fuel particles are coated with multiple layers of pyrolytic carbon and silicon carbide to form an impervious shell, able to withstand temperatures above 2000°C. Radioactive materials created during fission are trapped within this shell and cannot escape. Since virtually all radioactive contamination is confined within the particles, the primary system activity is low and worker exposure to radiation is much less than at conventional nuclear plants.

Since 1961, more than 12 billion coated particles have been produced for use in power reactors and advanced energy and defense applications. Tests of the latest fuel manufactured revealed only two defective particles in 100,000. The power conversion system - the generation of electricity from steam - is basically identical to that of modern fossil-fuel power plants.

## A ONCE-THROUGH Th/LEU FUEL CYCLE AND ITS IMPACT ON A LONG-TERM SCENARIO BASED ON CANDU-TYPE REACTORS

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#### Abstract

Prospects for extended PHWRs' lifetime up to 100 years were presented at the recent (1986) European Nuclear conference. Even for a shorter lifetime a long-term solution for reactor fuelling is needed. Therefore, the reprocessing requirements for the transition from the once-through operation mode to a recycle operation mode is an important factor.

Feasibility studies were performed at INPR for different PHWR once-through fuel cycles, namely: natural uranium (UN), slightly enriched uranium (SEU) and thorium enriched uranium (Th/LEU), taking into account the corresponding reprocessing requirements for the transition to the recycle mode.

The report presents the result of a prefeasibility study dedicated to a Th/LEU fuel cycle using standard (37 fuel pins) bundles consisting of 19 LEU (4% enriched) pins surrounded by 18 Th pins (i.e. the outer pin ring). Both LEU and Th fuel pins are of graphite disc-oxide fuel type suitable for high power and extended burnup. LEU pin irradiation results at NRU (Canada) between 5000 and 33 000 MWd/t at linear powers between 300 and 700 W/cm presented by AECL reveal excellent behaviour, as well as the irradiations in 600 MW - Gentilly 2 PHWR up to 10 000 MWd/t. We expect similar behaviour of the Th pins. When compared with UN and SEU fuel cycles, the Th/LEU once-through cycle is singled out by its lower requirements as far as development of reprocessing capacities. If a three step development is required in the first two cases (20-100, 200-400 and 800-1600 tonnes/year reprocessing unit capacities), then two steps would suffice in the second case (20-100 and 200-400 t/y). The required reprocessing capacity would be lower by a factor of 4 to 10.

The report also presents a Data Base elaborated at INPR for this Th/LEU cycle in 600 MWe CANDU PHWR.

#### 1. Introduction

In a foreseeable future featuring moderate uranium prices the recycle mode may appear to be competing only poorly with the once-through mode of reactor fuelling, i.e. the way the once-through mode is currently appraised.

In a previous paper [1] the different once-through fuel cycles in PHWRs were analysed taking into account the associated back end problems in an extended nuclear power programme. The slightly enriched uranium (Th/LEU) fuel

cycle was found to reduce the back end costs down to half of the natural uranium (NU) fuel cycle back end costs.

Prospects for extended PHWRs' lifetime up to 100 years were presented at the recent (1986) European Nuclear Conference [2]. Even for a short lifetime a long-term solution for reactor fuelling is needed. This solution could be the recycle mode. Therefore the reprocessing requirements for the transition from the once-through operation mode to a recycle operation mode is an important factor. The Th/LEU once-through cycles are singled out by their much lower reprocessing requirements.

This paper presents the results of a prefeasibility study dedicated to a Th/LEU fuel cycle using standard (37 fuel pins) bundles consisting of 19 LEU (4% enriched) pins surrounded by 18 Th pins.

A brief report is also presented, dedicated to the impact of this fuel cycle in a long-term scenario based on CANDU PHWRs reactors.

## 2. Proposed once-through Th/LEU fuel cycle

In a comprehensive analysis of the field [3], these once-through thorium cycles are defined as those fuel cycles in which pure thorium fuel bundles can be irradiated together with uranium fuel bundles in a CANDU reactor, with parameters judiciously chosen such that its overall fuel cycle cost is competitive with that of the once-through fuel cycle costs. Some of them (as the Valuebreeder concept [4]) are characterized by considerable amounts of energy that could be obtained from burning U233 formed "in situ". A burnup as high as or higher than 60 000 MWd/t thorium is needed [5], if no credit is postulated for U233 contained in the spent fuel. The uranium fuel bundles with a low enrichment in U235 (1.8% in [4] and 4 or 5.5% in [5]) are irradiated with a burnup low enough to supply the excess of neutron needed to build up and to burn U233 in the thorium bundles.

A once-through low enriched uranium/thorium cycle could be envisaged, using a single type of bundles, as in the case of the natural uranium cycle or other cycles using homogenous fuels.

The case of a once-through uranium/thorium cycle based on the use of standard (37 fuel pins) bundles consisting of 19 LEU (4% enriched in U235) pins surrounded by 18 Th pins (i.e. the outer ring pins) is discussed in this paper.

This fuel cycle is characterized by burnups values low enough to be sustained by the already existing irradiation data. The pins are of graphite disc-oxide fuel type suitable for high power and extended burnup. LEU pin irradiation results at NRU (Canada) between 5 000 and 33 000 MWd/t at linear powers between 300 and 700 W/cm presented by AECL [6] reveal excellent behaviour, as well as the irradiations in 600 MW Gentilly 2 PHWR up to 10 000 MWd/t [7].

It is expected that the thorium pin will reveal a better irradiation behaviour.

The burnup limitation led to a higher uranium consumption than in the case of higher burnup[5], i.e. only 20% lower than in the case of the natural uranium cycle. Therefore, this cycle using rather large amounts of separative work could be applied only in the case in which this separative work will be available at the lowest cost of the actual projections.

Table	1
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Fuel Parameters

Th fuel pins		
	No of pins per bundle	18
	Pin weight per bundle (kg Th)	6.92
	Discharge burnup (MWd/kg Th)	28
	Max. fuel temperature	1400
LEU fuel pins		
	Enrichment (%)	4
	No. of pin per bundle	19
	Pin weight per bundle (kg U)	8.24
	Discharge burnup (MWd/kg U)	44
	Max. fuel temperature	1500
	Bundle average discharge burnup (MWd/kg HM)	36



FIG.1. Graphite disc-oxide fuel element.

#### 3. System design and performance data

General reactor performance data are very similar to those of the once-through natural uranium cycle [8], except for fuel parameters which are summarized in Table 1.

Fuel pin structure is presented in Fig. 1. The pellet form differs from that given in [6], being rather similar to that achieved by the fuel through the irradiation induced fuel redistribution. The calculated temperature distribution for 565 W/cm linear power is given in Fig. 2.

Reactor fuel management information is summarized in Table 2.



FIG.2. Calculated isoterms in graphite disc-oxide fuel.

Table 2	2
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Fuel management information for a 638 MWe CANDU PHWR

Nominal rate of fuelling (bundles/day)	4
Losses (at 80% load factor and 0.2% enrichment tails)(%)	
- conversion	1.2
- fabrication	0.5
U <sub>3</sub> 0 <sub>8</sub> requirements (t NU)	
- initial core	188
- annual equilibrium	73
- 30 year cumulative	2342
Separative work requirements (t SWU)	
- initial core	91
- annual equilibrium	63.3
- 30 year cumulative	1956
ThO2 requirements (t Th)	
- initial core	5.4
- annual equilibrium	8.2
- 30 year cumulative	246

#### 83.2\$/kg U U30g cost 20 \$/kg Th ThO, cost 70 - 150 Separative work cost \$/kq Conversion of U308 to UF6 6 \$/kg U NU bundle fabrication cost 48 \$/kg U Th/LEU bundle fabrication cost 80 \$/kg U Interim storage 40 + 4/y per kg HM Active bundle transport after 10y cooling NU bundle 20 \$/kg U Th/LEU bundle 40 \$/kg M Encapsulation option 200 \$/kg U Conditioning Disposal NU bundle 30 \$/kg U Th/LEU bundle 180 \$/kg M Reprocessing option Reprocessing NU bundle 290 \$/kg U Th/LEU bundle 365 \$/kg M Vitrification NU bundle 40 \$/kg U Th/LEU bundle 200 \$/kg M Disposal waste, NU bundle 30 \$/kg U Th/LEU bundle 180 \$/kg M Load factor 80% Discount rate 48 Tail assav 0.20% Base date of monetary unit 1.1.1984

#### Table 3

Base values of economic parameters [9,10]

### 4. Economics

The base case used for economic evaluations if a 638 MWe CANDU PHWR. The ground rules adopted were the same as those used in [8]. The reference values used for relevant economic parameters are give in Table 3.

In Table 4 the fuel cycle costs have been expressed as unit costs and as average product costs for different separative work unit cost.

## 5. Reference scenario

The reference scenario is defined as a NU-SSET scenario, i.e. a scenario based on the use of NU once-through cycle followed by a nearby SSET (Self

Table	4
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Unit o	cost	and	product	cost
--------	------	-----	---------	------

		Natural Uran	ium	Th/LEU	
Separative wor	k cost (\$/kg)	_	70	130	150
Unit total dis	counted				
capitalized fu	el cycle cost (\$/)	(gWe):			
Front end		334.3	365.8	476.8	513.8
Back end	(Encapsulation)	277.7	259	259	259
Total	(Encapsulation)	612.0	390.8	501.7	538.7
Back end	(Reprocessing)	303.3	49.7	49.7	49.7
Total	(Reprocessing)	637.6	415.6	526.5	563.5
Average produc	t fuel cycle cost	(m\$/kWh)			
Encapsula	tion option	5.04	3.22	4.14	4.44
Reprocess	ing option	5.26	3.42	4.34	4.65

Sufficient Equilibrium Thorium) fuel cycle. The initial fissile inventory needed to reach the SSET equilibrium is the plutonium extracted from the NU spent fuel.

Additional improvements to the CANDU-PHWR neutron economy are needed in order to achieve self-sufficiency. Cf. [10] these are: increase in  $D_2O$  moderator purity, use of enriched zirconium alloys and lowering of the specific power to reduce parasitic absorption in Pa 233.

These considerations and some other ones were taken into account in an INPR prefeasibility study dedicated to SSET fuel cycle in 638 MWe-CANDU PHWR.

Fig. 3 [11] plots the evolution of the  $k_{eff}$  of the thorium lattice during 72 years of recycle operation with a burnup equal to 9400MWd/t. A lot of similar curves, obtained at different burnups, led to the burnup dependence of the final (after 72 years of recycle)  $k_{eff}$  given in Fig. 4. The sensitivity results presented in Table 5, obtained using the same procedure, seem to prove that the results dependence of the self shielding uncertainties is within acceptable margins.



FIG.3.

For 0.5% fuel cycle losses, a burnup between 5 000 and 14 000 MWd/t could be obtained in a standard 638 MWe CANDU PHWR, depending upon the adjuster rod allowance (0 to 5mk) and the use of ThO<sub>2</sub> or metal fuel [12].

For a reactor system averaged load factor equal to 80% these burnup limits become 10 000 to 17 000 MWd/t and the use of 99.95% purity  $D_2O$  could increase this burnup with about 4000 MWd/t.

The achievement of these performances, mainly fuel cycle losses reduction and high  $D_2O$  purity management, requires operating experience with both the fuel cycle facilities and the reactor.

Fig. 5 plots the operating fissile inventory (expressed as NU spent fuel amount for a 638 MWe unit) built up in the first 18 years of operation. After year 18 a near SSET fuel cycle of constant burnup (15 000 MWd/t) operation is considered, with the following improvements achieved gradually up to the year 90:

- the fuel cycle losses reduction, from 1.5% to 0.5%
- adjuster rod allowance, from 5 mk to 0 mk
- D<sub>2</sub>O purity increase, from 99.75% to 99.95%
- out-of-core delay reduction from 2 years to 1/2 year.

The initial fissile inventory, expressed as the total NU spent fuel element reprocessing requirements, as calculated for a fuel cycle with a burnup equal to 10 000 MWd/t and 2 years out-of-core delay amounts to 680 t and 1020 t for the reactor core and the out-of-core inventory, respectively.



FIG.4.

It gradually decreases to the fissile inventory, as calculated for a burnup equal to 15 000 MWd/t and 1/2 year out-of-core delay, namely (also expressed as NU spent fuel) to 680 t and 170 t for the reactor core and out-of-core inventory, respectively.

The main problem of this reference scenario consists of the transition reprocessing requirements (minimum 1700 Mg NU spent fuel) which are as high as the reprocessing requirements of the subsequent 40 years of recycle operation of a 638 MWe CANDU PHWR.

For a reprocessing unit cost equal to \$290 per kg NU spent fuel as given in [10], the cost associated with the transition of a 638 MWe CANDU PHWR to this recycle mode amounts to about M\$500.

In the case of 10 GWe reactor system power, the transition reprocessing requirements are about 27 000 t which could represent the production of a reprocessing capacity of about 1500 t/year for a reprocessing facility life of about 20 years. The needed development will consist of three steps: a 20 - 100 t/year pilot plant, an intermediate 200 - 400 t/year pilot plant and one of two 800-1600 t/year commercial size units. This development will require much more than 25 years.

Nuclid	<sup>k</sup> eff mk	Burnup change equivalent for 500 MWd/t/mk
Th 232	- 0.89	-430
Pa 232	1.08	540
U 233	0.07	36
U 234	- 0.01	-7
U 235	0.00	0
U 236	0.14	69

k<sub>eff</sub> changes for a resonance absorption increase by 10% after 300 000 MWd/t cumulative burnup for 10 000 MWd/t burnup per cycle



FIG.5. Near SSET inventory evolution.

#### 6. Th/LEU scenario

In order to avoid the above mentioned problems associated with the reference scenario a Th/LEU once-through fuel cycle has been considered instead of the NU once-through fuel cycle.

The proposed scenario is defined as a Th/LEU - SSET scenario. The initial fissile inventory needed to reach the SSET equilibrium is the uranium contained in the Th spent fuel pins. In this case the reprocessing requirements for the transition from the once-through mode to the recycle mode could be as low as 1 kg spent fuel pin per kg SSET fuel, if only thorium pins

are reprocessed, or about 1.8 kg Th/LEU spent fuel bundle per kg SSET fuel, if both thorium and LEU pins are (for this transition) reprocessed. We consider there that the LEU pin will be encapsulated or reprocessed later, with no connection with the transition to the recycle mode.

The initial fissile inventory, expressed as the total Th spent fuel pin reprocessing requirements, as calculated for a SSET fuel cycle with a burnup equal to 10 000 MWd/t and 2 years out-of-core delay, amounts to 77 t and 120 t for the reactor core and the out-of-core inventory, respectively. It gradually decreases to the fissile inventory, as calculated for a burnup equal to 15 000 MWd/t and 1/2 year out-of-core delay, namely (also expressed as thorium spent fuel) to 77 t and 20 t for the reactor core and the out-of-core inventory, respectively.

The inventory difference (100 t for a 637 MWe CANDU PHWR) is large enough to supply more than 80 years the fissile needed for the case of an initial ratio equal to 1.6 (i.e. b = 1.035 for 15 000 MWd/t - see Fig. 4 - and 2.5% out-of-core losses) gradually decreasing to 1.

The transition reprocessing requirements are about 200 t, i.e. at least 8.5 times lower than in the case of NU spent fuel.

In the case of a 10 GWe reactor system power, the transition reprocessing requirements are about 3200 t which could represent a reprocessing capacity lower than 200 t/year. The needed development could consist of up to two steps, a 20 - 100 t/year pilot plant and a commercial facility of 200 - 400 t/year, and could require less than 20 years.

For a reprocessing unit cost equal to \$365 per kg Th spent fuel as given in [10], the cost associated with the transition of a 638 MWe CANDU PHWR to recycle mode in this scenario will be about M\$115 i.e. 4 - 5 times lower than in the reference scenario.

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## DESIGN AND SAFETY FEATURES OF THE AMPS NUCLEAR ELECTRIC PLANT\* (Abstract)

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This paper describes the key design features of the Autonomous Marine Power Source (AMPS) plant with special emphasis on safety aspects. The AMPS plant, initially conceived for commercial submarine applications, utilizes a low pressure, low temperature reactor heat source coupled to a low temperature organic Rankin cycle engine generating electric power.

The strategy employed in the design of AMPS has been to identify design features which are conducive to minimize design and operating complexity and enchancing safety and reliability. A number of innovations vital to meet the special demands of the AMPS application have been incorporated in its' design.

The Autonomous Marine Power Source plants have a number of intrinsic safety features designed to minimize the likelihood of fuel failures or core damage as a consequence of fuel overheating following postulated design basis accidents.

The AMPS development programme which is underway since 1985 has also been described in this paper.

<sup>\*</sup> The presentation was based on the paper presented at the Canadian Nuclear Society Annual Conference, Winnipeg, Manitoba, 12–15 June 1988.

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