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Integral design concepts of advanced water cooled reactors

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FOREWORD

Nuclear power has played a significant role in the supply of electricity over the past two decades. The two major nuclear accidents, namely Three Mile Island and Chernobyl, have considerably affected its further growth. Reconsideration of reactor design and safety aspects of nuclear power has been an active area of development. Elimination of some accident scenarios, simplification of systems and reliance on natural phenomena have been a major part of these activities, which led to the development of new design concepts.

Under the sub-programme on non-electrical applications of advanced reactors, the International Atomic Energy Agency has been providing a worldwide forum for exchange of information on integral reactor concepts. Two Technical Committee meetings were held in 1994 and 1995 on the subject where state-of-the-art developments were presented. Efforts are continuing for the development of advanced nuclear reactors of both evolutionary and innovative design, for electricity, co-generation and heat applications. While single purpose reactors for electricity generation may require small and medium sizes under certain conditions, reactors for heat applications and co-generation would be necessarily in the small and medium range and need to be located closer to the load centres.

The integral design approach to the development of advanced light water reactors has received special attention over the past few years. Several designs are in the detailed design stage, some are under construction, one prototype is in operation. A need has been felt for guidance on a number of issues, ranging from design objectives to the assessment methodology needed to show how integral designs can meet these objectives, and also to identify their advantages and problem areas.

The technical document addresses the current status of the design, safety and operational issues of integral reactors and recommends areas for future development.

EDITORIAL NOTE

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SUMMARY

1. INTRODUCTION

Many developing countries are experiencing a rapid growth of their economies and an increasing need for the supply of both heat and electrical energy. The present primary energy production is based predominantly on fossil fuels, adding to the CO_2 burden on the environment. Nuclear power has the potential to bring about a substantial reduction in CO_2 releases arising from both heat and electricity generation. Although there was a rapid growth in the seventies and early eighties, in the nuclear share of total electricity generation, several factors including the impact of the Three Mile Island and Chernobyl accidents have resulted in dampening further growth. For harnessing the potential benefits of nuclear energy in meeting the future needs of both heat and electricity, it is necessary to develop cogeneration nuclear plants having the following characteristics:

- Low capital investment to reduce the financial risk.
- Construction period of 5 years or less to improve the economics of nuclear heat and power generation.
- High operational reliability through simplification of the plant systems.
- Enhanced safety features enabling siting near densely populated areas.

In response to these needs nuclear power plant designers in many countries have been developing new designs including integral reactors. Integral reactors are heat and/or power generation reactors, in which the primary coolant system components including steam generators(SGs), pressurizers and pumps are contained within the reactor pressure vessel (RPV). The current loop type power reactors have these components located outside the RPV.

The purpose of this report is to provide up-to-date technical information on the current status of the design and development of integral reactors in the Member States, based on information presented in two technical committee meetings organized by the Agency on the subject. Important aspects regarding integral reactor design and development are highlighted in the summary and the presented papers are included under the following headings:

- Development programmes and conceptual design descriptions.
- Specific systems and analyses.
- Operational, manufacturing and decommissioning aspects.

2. HIGHLIGHTS OF THE MEETING

The rated power of current integral reactor designs is limited to 700 MWe due to manufacturing limitations in the size of the RPV. The maximum diameter of the RPV is limited to 7 m, based on present technological capabilities. The steam generators have to be of a special compact design with a high power density to enable their location inside the RPV.

In-service inspection(ISI), maintenance and replacement of equipment and components is recognized to be more difficult as a result of the compactness of the integral reactor designs and therefore, these aspects have received special attention in the design stage itself Some unique solutions and special tools have been developed for this purpose

The design characteristics are chosen to enhance nuclear safety and thus enable siting close to population centers. Decommissioning is also facilitated due to the availability of a fairly large RPV which can be used to store all the active components for a few decades in a safe manner

New concepts of integral designs are being developed in the Republic of Korea and the Russian Federation In other countries, some modifications to existing designs have been undertaken Other design areas receiving active attention include safety-related heat removal systems for integral reactors, design of compact steam generators and decommissioning aspects

From operational point of view, integral reactor designs do not differ in principle from loop type reactors The principal advantages of integral reactors over current generation loop type reactors include the following:

- Enhanced safety level due to the location of primary coolant circuit within the RPV, in particular, a reduction in the probability of accidents accompanied by core damage
- Use of natural convection principle in the design of the primary coolant circuit not only provides a passive system for emergency decay heat removal, but also permits design of natural circulation reactors operating at rated output
- Reduction of neutron fluence on the reactor vessel to a negligible level enhances RPV life substantially
- Significant increase in shop-fabrication of the reactor systems reduces the volume of assembly work at site, and improves conditions for implementation of quality control procedures.
- Stringent requirements of leak tightness for the outer containment shell are relaxed as a closely fitted steel containment vessel called guard vessel functions as the first containment barrier A reinforced concrete shell would be adequate as an outer containment for protection from external effects Consequently, requirements of special safety systems for primary coolant inventory control and decay heat removal are also substantially simplified
- Potential reduction in construction time improves the economics of integral reactors
- Simplification of decommissioning work enables an earlier reuse of the site

Aspects currently receiving greater attention from designers of integral reactors include the following·

- Significant increase in overall dimensions and weight of the RPV resulting in the need to employ special handling and transport facilities during construction/assembly work
- Restriction of maximum reactor power capacity to 700 MWe as a consequence of limitation in allowable overall dimensions due to constraints in production capability of the industry

- Special design characteristics of the steam generators such as compactness and high power density which have a critical impact on the RPV dimensions
- Designs to facilitate comprehensive planned maintenance for trouble-free operation and replacement of major components.
- Use of nitrogen gas for pressurization

3. STATUS OF DESIGN AND DEVELOPMENT WORK IN MEMBER STATES

3 1 Argentina

Design and development work for the Argentinean project called CAREM is continuing The project is ten years old and the initial design power level was 15 MWe The current work is on a 25 MWe reactor, with thought being given to a 100 MWe reactor. Engineering for the 25 MWe plant is scheduled for completion by the end of 1996. Financial, political and siting decisions are expected to be made in 1996, which if favorable, will lead to construction in 1997. Experimental work is under way in a high pressure loop to study critical heat flux (CHF) and dynamic response, and to make a comprehensive study of the reactor physics of the core in the RA-8 critical facility Development and testing is being carried out on the core internals, control rod drives & position indicators and the reactor protection system, especially the trip system

3.2. China

Loss-of-coolant experiments are being carried out for the Chinese nuclear heating reactor (NHR200). A test loop has been in operation since 1989, and allows tests on the three possible positions for a break to occur, (i) pipes on the upper plenum, (ii) steam generator pipe break and (iii) boron injection pipe below the vessel water level. Experiments have shown good agreement with the results of calculations on the influence of the break position on RPV water level, on discharge quality and hence on the depressurization rate. Depressurization is quite slow (thousands of seconds), due to the small size of pipe connections.

China is carrying out a study on the choice of a reactor system for a co-generation plant. The study is based on a 2x450 MWt plant and hinges on the configuration of the intermediate loop A comparison is being made between two alternatives; (i) steam generator in the RPV supplying steam to the turbine and an external steam water heat exchanger, and (ii) a system with an in-reactor high pressure water heater which provides hot water to generate steam in an inverted U-tube steam generator The steam passes to the turbine and back-pressure steam provides the heating load.

The second alternative allows a lower primary pressure and hence a saving in cost The possibility of radioactive leakage into the tertiary circuit is reduced due to the high reliability of water-water heat exchangers and by having a secondary pressure higher than the primary pressure. The overall efficiency loss from having a 10 MPa primary is offset by the higher efficiency of the second heat exchanger which is a steam generator rather than a low efficiency steam/steam heat exchanger

3 3 Indonesia

Indonesia is presently giving senous consideration to the introduction of nuclear power Strategic planning in Indonesia also envisages the utilization of an integral reactor design for the supply of heat and electricity to many of its islands. The perspective plan is to Install 12,000 MWe of nuclear power by 2019, of which the majority will be based on large reactors of existing designs. There is, however, a potential market for a 30 MWe design, suitable for several small islands in Indonesia resulting from the high cost of transportation of fossil fuel

3 4 Italy

The reactor continuing to be developed in Italy is named Inherently Safe Immersed System (ISIS). This reactor has components which are passive at category "B" in the IAEA definition which means that they need no valve movements or logic circuits for system initiation. The reactor uses some of the concepts used in the "PIUS" design but has a very small heated primary inventory with density locks to give access to highly pressurized cold water in the event of a reactor malfunction. Data on safety analysis shows the very high level of safety that can be achieved The problems of deployment are now economic rather than safety related The system could be competitive in a co-generation mode, perhaps with a pressure reduction to decrease the mass of steel needed. Further work on tackling the economic competitiveness aspects is under way.

3 5 Republic of Korea

The Republic of Korea has recently initiated design work for an integral reactor to be used for power generation and sea water desalination. The design dates from mid 1994 and the schedule points to construction around 2005. The primary vessel is contained in an outer safeguard vessel, half-filled with water, and is designed to the same pressure as the primary. Residual heat removal in emergency is through the vessel wall to the water in the safeguard vessel and from there, by heat pipe to a cooler outside the containment. The internal pressurizer uses nitrogen gas for pressurization, with pressure driven sprays and no heaters. The heat exchanger is a once through helical one, giving 30 C super heated steam There is a steam injector to drive a containment spray system. A new control rod drive mechanism(CRDM) is under development, giving finer movement than the previous Korean magnetic jack type The fuel elements are hexagonal An extensive research and development programme is envisaged

3 6 Japan

Development work at the Japanese Atomic Energy Research Institute (JAERI) has concentrated on maintenance and cost estimates of the Marine Reactor-X (MRX). This is a compact system with forced circulation of the primary, in-vessel control rods and a water filled containment, cooled by a natural circulation system The water filled containment eliminates the need for a secondary heavy shield, giving weight advantages even over a diesel system when the weight of fossil fuel to cross the Pacific Ocean is taken into account With a fleet of 20 ships, such a nuclear powered ship was shown to be economically better compared with a diesel powered ship. The suggested mode for maintenance and refueling (every four years) is to lift out the entire core with its containment and to replace it with another one. The estimated time for this operation is three weeks. The same principle would be used for decommissioning, in an appropriate facility.

3.7. Russian Federation

In the Russian Federation, a series of integral reactor designs are being actively pursued by the Experimental Machine Building Design Bureau (OKBM).

The reactor called Atom Thermal Electric Plant (ATEC 200) comes in sizes from 80 to 250 MWe. They have natural circulation systems for residual heat removal, positioned in the upper head. They are intended for use in remote locations and are designed to be sited below ground level.

The "AST 500" is a natural circulation district heating reactor which is ready for operation but due to public acceptance problems, the project has been suspended for the present. The safety of its design and operation have been reviewed by an IAEA OSART team and found to be adequate.

The Passive Safety Integral Reactor Plant (VPBER 600) is a 640 MWe reactor designed earlier A number of significant changes have been made to its design, including moving the pumps from the bottom of the vessel to a position above the core, and addition of a core catcher. A comprehensive in-service test and inspection programme has been set up and equipment design carried out.

Integral Reactor for a Floating NPP (NIKA 120) is designed by the Research and Development Institute for Power Engineering (RDIPE) as a floating power unit for use in northern Russia and is under development. The plant consists of two reactors, each of 42 MWt. The reactor vessels are enclosed in a safeguard vessel which is immersed in a bubbling tank. The fuel is 21% U-235 as UO_2 in a zirconium matrix. Control rods are designed to stay inserted even if the floating unit is inverted in the sea.

Development work is also being carried out on a small unit called the Autonomous Co-generation NPP (UNITHERM), with a power of 30 MWt or 6 MWe for use in difficult-to-reach regions. Emphasis is on maintenance-free operation between the annual visits of the maintenance team. There is an intermediate heat exchanger to isolate the final steam supply including that to the turbine, from any risk of radioactive contamination. Such an intermediate circuit is necessary for a district heating system to act as a pressure barrier Maintenance of the steam generator and turbine can be carried out using conventional methods as the steam is free from radioactive contamination. The penalty is an increased generation cost due to loss of thermal efficiency and the cost of an extra plant system.

Development work is in progress with regard to emergency heat removal systems on the "ABV-6" reactor system, planned for installation in the floating nuclear power plant, "Volonolom". This is a 38 MWt plant using a uranium-aluminum(U-Al) alloy fuel in a natural circulation reactor with nitrogen pressurization There are five emergency heat removal paths some of which are dedicated systems and others are shared systems(e.g use of the coolant purification system heat exchanger for emergency heat removal). There is a passive system using the flow of stored water to the normal steam generators and discharging the resulting steam to the atmosphere The ABV-6 containment system embodies two rupture discs The first releases steam from the reactor containment shell to the pressure suppression pool, and the second releases pressure in the suppression pool compartment to the auxiliary building The reactor and containment systems were modeled using the computer code RELAP5 MOD-3, with modification of some code modules to allow for release of dissolved nitrogen A severe accident scenario with a double ended break of the pressurizer surge line, failure of the emergency core cooling system (ECCS), failure of the shielding tank coolers and no operator action, was modeled. The results of the analysis have conclusively shown the very high level of safety achievable by this design.

For many years, OKBM has been conducting design and development work on highly efficient cassette steam generators. These cassettes are produced on an automated assembly line and have been successfully used in many reactors They are also specified for VPBER 600 design, where 216 independent sub-sections are assembled into two different shapes of boxes to fill the annular space available The design is based on straight tube steam generators with secondary flow inside the tubes to ensure that they are always in compression There is considerable practical experience with these cassette steam generators in VVER plants (500,000 hr) and they are backed by a large programme of developmental tests

OKBM is also working on radiological aspects of decommissioning integral reactors The main advantage of integral reactors results from the large water filled space between the core and the pressure vessel This results in a reduction in activation level of the RPV by a factor of up to 10^4 compared to VVER reactors, with a corresponding reduction in activity in the adjacent concrete structures The overall effect on occupational dose during decommissioning, involving breaking up and removing all active plant components, is a reduction by a factor of ten

Design work on a new concept called Integral Reactor with Inherent Safety (IRIS) based on the "PIUS" reactor was started under the leadership of IPPE It is a 600 MWe reactor with natural circulation The steam generator is within the main vessel and the upper density lock is replaced by a by-pass tube which descends below the water level in the outer tank, which in turn, is below the water level in the reactor vessel during normal operation An increase in power leads to vapour formation and flow of primary water/steam into the outer tank Borated water enters the primary through the lower density lock and shuts the reactor down The design is still in the conceptual stage

Design work at the Kurchatov Institute is continuing on the use of super critical water as the primary fluid in the "V-500 SKDI" reactor With subcritical water, power increase is limited by the possibility of departure from nucleate boiling (DNB) This possibility is eliminated in super critical conditions, since the fluid remains in a single phase through all the temperature range Furthermore, the enthalpy of super critical water approaches that of steam, giving improved heat transfer in the steam generator Density variation with temperature provides an excellent negative reactivity feedback for reactor stability The water density effect is strong enough to allow compensation for reactivity changes with burn up, without control rod movements and only a small change in primary temperature Some modern fossil-fueled stations operate with supercritical water in the boiler at 26-28 MPa and 560-580 K, giving a basis of practical experience for the use of these conditions Safety analyses have confirmed the high level of safety achievable by this concept, even in a very severe transient caused by multiple failures

4 TECHNICAL DESIGN ASPECTS

4 1 Steam generator design

In steam generator design for integral reactors, the primary objective is to develop a compact steam generator to enable locating them inside the reactor pressure vessel, eliminating the possibility of a large loss of coolant accident (LOCA) and making efficient use of the space available This implies a high power density in the tube bundle

The main candidates for tube material are titanium alloys and Inconel, mainly 690 and incaloy 800 The choice would be related to national experience Titanium alloys have the following advantages

- Low coefficient of linear elongation, about forty percent lower than stainless steel
- Less sensitivity to thermal loads, as a consequence of a low module of elasticity

Inconel has the advantage of larger thermal conductivity Experience with its use has been mainly in the present generation loop type reactors

Incoloy 800 exhibits excellent properties for heat transfer and corrosion resistance and the allowable stress is larger than stainless steel Experience with Incoloy 800 as a tube material is quite large especially in France and Germany

It could be said that all three materials show good mechanical properties and have demonstrated good corrosion resistance as a tube material for steam generators The basic arrangement in integral reactors is to locate the SG within the RPV, in the annular region above the core level Consideration, however, has to be given to the distance between the lower part of the SG and the core, to prevent secondary water activation and radiation damage to the component

With regard to tube support, straight tube SG allows a simple support systems, using simple spacers, since flow is parallel to the tube, while in helical steam generators the flow crosses the tube Tube support in the latter case is more complicated Corrosion and build-up at tube supports with the secondary working fluid on the vessel side are more pronounced Primary coolant outside the tubes has the following advantages

- a high resistance to stress corrosion cracking,
- safety advantages in case of tube failure and
- a reduced risk of crud accumulation at the tube plate connection

In all cases, a provision for the possibility of in-service inspection of the complete bundle is recommended. The design must allow tube plugging, component removal and replacement. Steam generators that are not once-through type are more suitable for load following due to the water inventory but this option does not appear to be followed in any of the Member States.

Some reactor designs prefer secondary water boiling inside the tubes to reduce the reactivity effect in the core in case of a steam line break. Some designs place the secondary outside the tubes to reduce hydraulic losses, especially if the design uses natural circulation.

Hydraulic stability of parallel tubes is one of the most important design aspects to be considered. Experience shows that instabilities can be controlled through careful design. One of the features currently adopted is the introduction of orifices at the tube inlet to increase the secondary side pressure losses in the liquid phase. To avoid operational problems, the chemistry of the secondary coolant must be maintained at a high quality level for this type of SGs, to avoid crud deposition. Even in the case of some crud deposition, adequate experience exists regarding washing it away. RPV penetration for feed water and steam outlets can be optimized in number according to the needs of diameter limitations and to the specific design requirements.

4.2. RPV manufacturing and transport

Dimensions of RPVs are dependent on reactor size. The largest integral reactor pressure vessels that are currently considered (SPWR, SIR, VPBER-600) have the following dimensions:

Diameter	6.5 - 7.2 m,
Height	20 - 25 m,
Wall thickness (cylindrical part).	265 - 280 mm

These dimensions are comparable with the largest pressurized water reactor (PWR) pressure vessels (diameters, wall thicknesses) and boiling water reactor (BWR) pressure vessels (heights, diameters). Some designers have made enquiries with potential manufacturers and they received positive answers on the possibilities of manufacturing these large pressure vessels. Existing manufacturing technologies in the following areas can be used in the manufacture of integral RPVs

- materials
- forging of semi-products
- welding and cladding
- machining
- inspection and testing

Guard vessels/containment can be manufactured at existing manufacturing facilities. No specific problems have been identified concerning the manufacturing of these large components also The feasibility of RPV transport may be an important issue to be taken into account in site selection Access by water can solve most transport problems

4 3 Primary circulation

Natural circulation of primary coolant is an inherent feature of the integral reactor arrangement due to its simple configuration and low hydraulic resistance of the pnmary circuit Reactor coolant natural circulation has reliability, simplicity and safety advantages These advantages override economic considerations at lower unit powers As the power level increases, economic considerations become more important and hence forced circulation may be preferred However, under special conditions, for example in marine reactors, forced coolant circulation is utilised even in units with a low rated power

For heat only reactors of any power level, natural circulation seems to be the most preferred solution due to a lower core power density and high reliability requirements typical of this kind of reactor

Cost considerations and technological limitations in RPV manufacture appear to limit the use of natural convection cooling At present integral reactors with reactor coolant natural circulation are limited in their power level to 1000 MWt

4 4 Operation and maintenance

The operational mode of integral reactors is determined by the specific requirements of the consumer They can follow the load or be base loaded General operational procedures do not differ from loop type PWRs Improvement of the reliability of integral reactors comes partly from the use of natural circulation in the primary circuit The possibility of a primary circuit failure is reduced by compactness of the integral reactor and absence of primary circuit branched pipe systems With regard to system pressurization, the following three options for the primary circuit pressurizing system have been presented

- steam pressurizer with heaters,
- gas - steam pressurizer and
- self - pressurizer

There is ample positive experience available on the three types of pressurization The choice between them depends on particular design features and past experience Furthermore, most designers prefer the integrated arrangement of pressurizer Hydraulic processes in integral reactors are more inertial compared to loop type reactors Therefore, a higher grace period for human intervention is available

Greater attention should be paid to the reactor vessel inner surface inspections and steam generator repair and maintenance provisions in the design phase, since these aspects strongly influence the plant availability factor In-service inspection of integral reactors does not impose any new or specific problems Particular features that must be taken into account when considering in-service inspection solutions include the following

- Long distance from the top of the RPV to the core
- Geometry of steam generator tubes

Standard techniques for in-service inspection are not directly applicable, but can be adapted to integral reactors. RPV inspections must be carried out according to the applicable codes and standards. Depending on accessibility, the inspection can be carried out from outside and/or inside the vessel.

Problems of accessibility for maintenance should be taken into account at every phase of the design, because the equipment placed in the integral reactor vessel is not easily accessible compared to loop type reactors. For example, access to the inner reactor vessel surface in the built-in heat exchanger (steam generator) zone is rather limited. This disadvantage can be compensated by the use of highly reliable components and the use of remotely controlled equipment for inspection, maintenance and repair.

A specific feature of integral reactors is the complexity of replacement of in-vessel components. Nevertheless, most of the designs have made adequate provisions for the possibility of replacement of in-vessel components. One of the most important design criteria is that the replacement of components, especially, steam generators should be possible without complex operations.

4.5. Control

4.5.1. Control mechanisms

Four types of control for integral reactor were identified in addition to soluble boron as given in the table below:

Type	Advantages	Disadvantages
Conventional external drives	Proven.	Need height for drives. Rod ejection possibility. Long rod connectors in integral designs.
Internal hydraulic	Reduced number of penetrations. Simple. Compact. No rod ejection.	Need reliability data.
Internal electro - mechanical	Reduced penetrations. Compact. No rod ejection. Requires only cable connection.	R & D needed.
No rods (not suitable for present generation loop type reactors)	Simple Eliminates all rod accidents Compact	No fast scram. Safety during design basis and beyond design accidents must be demonstrated.

4.5.2. Instrumentation

The following aspects need attention during the design phase:

- The integral water-cooled reactor complicates the in-vessel control detector arrangement. There is a need for new technical solutions regarding instrumentation design. There is no difference with other reactors in principle but there may be engineering problems.

- In-core instrumentation is needed on prototype reactors but should not be required on production reactors. This is because of the low power density and small size which gives stability to the power distribution
- There is a licensing requirement for in-core instrumentation in some countries
- Failed fuel detection can be left to detectors in the chemical and volume control system (CVCS) as in large light water reactors (LWRs)

4 5 3. Control diversity

Most integral reactors being considered have some type of control rods and also have soluble boron This provides diversity in the physical means of shut-down and in the technical means of implementing it

System-integrated PWR (SPWR) has no control rods and relies on boron systems only There are three independent systems Philosophically, this is similar to the fast breeder reactor (FBR) situation where there are only control rods for shut-down but there are diverse control rod drive mechanisms

4 6 Containment

The philosophy of containment for integral reactors is the same as that for loop type designs The containment has the following functions

- Protection from the effects of internal events, especially, retention of radioactivity released from the core
- Protection from external impacts

Single and duel containment designs are possible The low discharge rate from LOCA in integral reactors, results in pressure suppression systems being very attractive The guard vessel concept where the size of the leak tight containment is minimized to a shell which fits closely around the RPV has been developed for integral reactors It provides simplicity of design and has positive benefits in accident management

The guard vessel is usually made of steel but could also be made of pre-strressed concrete However, the emphasis on improved safety characteristics of integral reactors has led to a preference for steel, rather than steel lined concrete as it allows for ultimate heat removal by heat transfer through the steel All containment designs share the general objective of plant size reduction

The guard vessel concept can be applied to integral reactors, but not to loop type reactors since the whole of the primary circuit including CVCS is compact and can be contained in a vessel of reasonable size The advantages are

- Very effective containment of radioactive species both in normal operation and in upset conditions
- Possibility of reduced specification in terms of pressure and volume of the containment
- The additional protection the guard vessel provides, appears in some conditions to allow construction of nuclear plants closer to centers of population

For marine reactors, a water filled containment giving pressure suppression and elimination of the additional weight of a shield has been adopted for some designs.

5. SAFETY AND ACCIDENT MANAGEMENT ASPECTS

5.1. Decay heat removal

All designs use the in-vessel steam generators or dedicated in-vessel heat exchangers for decay heat removal(DHR). Steam generators require valve movements to use them for this purpose. In all cases there is an external heat exchanger to transfer the heat to the atmosphere or to water tanks. This arrangement permits the use of a passive system based on natural circulation. Use of an independent heat exchanger in a dedicated circuit is claimed to give added reliability at the expense of extra cost. Since the primary coolant feed and bleed pipes as well as the CVCS pipes are small in diameter, the possibility of a large LOCA is eliminated. If the water level drops below the steam generator/heat exchanger level or drops to a level where natural circulation within the vessel is prevented, heat continues to be removed through steam/water being released through the break and cool water returned to the vessel by various systems or by make up from the inventory maintenance system. In the low pressure NHR, such a system is unnecessary and is not provided since heat can be removed by condensation on the surface of the heat exchanger.

5.1.1. Effects of nitrogen

There is a need to improve the availability of data and the method for dealing with the effects of nitrogen originating in gas pressurizers, on heat transfer in emergency conditions.

5.1.2. Severe accident in an integral reactor

There is a significant increase in safety of the integral reactor in comparison with the loop type reactor, due to a reduction in accident initiators and the use of passive safety features and inherent characteristics.

Integral reactor accident sequences have not been fully analyzed. It would be expedient to carry out such analyses since the results could influence reactor design. It is also necessary to carry out probabilistic safety analysis.

The reliability of available codes (RELAP/SCDAP/MELCOR) should be evaluated for severe accident analysis in integral reactors and the need for modification and validation determined. International cooperation could play a key role in code validation for integral reactors.

5.1.3. Passive safety systems

There are certain safety features and advantages in integral reactors related to decay heat removal due to their power range and compactness. They are, especially, beneficial if they are implemented with passive initiation as well as passive operation.

The benefits of these systems are

- Extension of the time available for operator action
- Possible reduction in redundancy requirements
- Use of intermediate circuit eases maintenance problems in the steam generators and turbine due to the absence of radioactivity
- Cost reduction due to a reduction in the number of system components

An intermediate circuit is necessary for heating applications to eliminate the possibility of radioactive leakage to the end user This protection is particularly effective if the intermediate pressure is higher than the primary pressure

Since a water/water heat exchanger can be more efficient than a steam generator, in terms of specific transfer capacity and in certain design conditions, an *intermediate circuit can allow greater power in a given size of* RPV *or a reduction of* operating pressure for the same power The choice is an optimization between space available and the relative costs of a larger internal steam generator compared with an intermediate circuit

Passive system initiators used in integral reactors are

- Rupture discs
- Non return or check valves
- Valves which operate on changes of pressure differential
- Systems constantly in operation in normal as well as accident conditions

5 2 Primary water inventory maintenance in accidents

Integral water-cooled reactors make efficient use of the primary coolant inventory to prevent core damage under emergency conditions Under LOCA conditions, the steam generators of integral reactors assist decay heat removal for a longer time than in present generation designs since they remain covered by water, cool the primary fluid and reduce the loss of coolant vapor

The following are the basic systems for minimizing coolant inventory loss

- Use of a guard vessel or containment which fits closely around the lower part of the reactor vessel In LOCA this space rapidly fills with coolant ejected from the vessel and the core remains covered In some designs, e g SPWR, this space is connected to the outer water filled pressure suppression environment which gives an adequate supply of water to keep the core covered

- Provision of water by passive means from external tanks The feed may be by gravity at low pressure, gravity from tanks pressurized by automatic connection to the pressurizer in appropriate accidents, or by passive pumping devices such as steam injector pumps

The system using a guard vessel has the advantage of providing for a cheap and simple plant with no need for extra supplies of emergency coolant

The system using gravity feed tanks assisted by steam pressurization has the advantage of operation over the full pressure range The disadvantages are the limited volume of the tanks and the need for isolation valves to isolate pipe breaks outside the containment Besides, steam injectors are not yet proven for the duty required and are undergoing further development

5 3 Depressurization

A safety depressurization system is provided on some designs where it is simple and economic to do so Blow down into pressure suppression tanks is employed in SPWR and SIR

The integral water-cooled reactor with the pressurizer inside the pressure vessel gives a more direct and efficient connection between the pressure vessel and the emergency relief valves than in present generation reactors where the pressurizer and relief valves are separated by piping

VPBER-600 has an additional possibility of a partial depressurization which operates when the gravity boron injection tanks are discharged by connection to the pressurizer steam space

Depressurization is also achieved through the heat exchangers or steam generators used for decay heat removal There are other depressurization systems such as spray which are under operator control

5 4 Diversity/Redundancy of passive systems

Redundancy is necessary to compensate for single failures in components such as valves whether in themselves passive or not Diversity should not be strictly necessary due to the very high reliability of passive systems. However, it is prudent to provide diversity

5 5 Core melt

The probability of core melt in integral designs is expected to be lower than for the loop type reactors due to the following features

- Large water inventory above the core.
- High reliability of passive safety systems

The increased bottom diameter of the pressure vessel gives a larger contact area between core melt and the vessel surface giving increased heat transfer to the environment and a reduced possibility of molten corium penetrating the vessel

Designs where the space around the lower part of the vessel becomes flooded may claim that this cooling of the vessel ensures that the melt is retained within the reactor pressure vessel In the absence of a total understanding of the physics of core melt progression and of the need to ensure that the water does not dry out, this claim must be regarded as unproven

Many reactors make no design provision for core melt since the probability of core melt is very low It is, however, recognised that it would be prudent to make a provision to mitigate the consequences of a core melt situation

5 6 External events

- The Integral water-cooled reactor being compact provides better protection from external events due to a reduction in the number of primary pipelines
- Resistance to earthquake is a strong positive feature for integral reactors Other external events have to be treated in a way similar to other reactors
- The reduced probability of internal events increases the relative importance of external events in the safety analysis of integral reactors

5 7 Human factors

The Integral water-cooled reactor provides slower progression of thermal and hydraulic processes in the primary circuit It prolongs the grace period and decreases the human factor effect

5 8 Emergency Evacuation

The enhanced safety of integral water-cooled reactors significantly decreases the probability of accidents which require population evacuation

6 DECOMMISSIONING

Greater attention should be given to the problems of decommissioning in all phases of integral reactor design Decommissioning cost must also be taken into account in technical and economic estimates

6 1 Integral reactor decommissioning features

A thick water layer between the core and reactor vessel in integral reactors ensures low radioactivity of the reactor structures Thus, together with the absence of a branched pipe system in the primary circuit, it reduces the quantity of radioactive wastes and simplifies decommissioning

6 2 Decommissioning concept

The low radioactivity of an integral reactor vessel can lead to a preference for "immediate dismantling" (after preparatory work) The concept of "delayed dismantling" can be still used to further reduce radiation dose to workers

6 3 Decommissioning schedule

The decommissioning schedule is similar to that of existing PWR and BWR units and may include the following stages

- fuel assembly unloading,
- dismantling and breaking up of in-vessel highly radioactive structures,
- dismantling and breaking up of the RPV,
- dismantling of low radioactive and non radioactive equipment,
- radioactive equipment and spent fuel removal from plant site

Alternatively, the relatively large reactor vessel can be used for storage of all active components for about 50 years until their activity has reduced to a low level for easy handling for ultimate disposal This option is especially attractive if the vessel is below ground allowing removal of many structures above ground

7 CONCLUSIONS

In the technical committee meetings, a diversity of viewpoints and opinions was offered on the design and development of integral reactor concepts Consensus exists on many aspects of integral reactors The following are the conclusions

7 1 The presentations confirm the engineering validity and sound advancement of the integral design approach for advanced light water reactors (ALWR) Presentations have also covered new issues, areas and designs not covered previously

7 2 Integral reactor design activity is strong in many Member States Some designs have been built, some are in the detailed-engineering stage and most are in the conceptual design stage

7 3 For further development, a clear definition of user requirements is necessary, which will clarify design criteria and specifications

7 4 There are many similar designs for which realization in construction is unlikely Concentration of effort on fewer projects for the detailed design stage would be beneficial and cost effective

7 5 Integral designs cover from low to medium power range Low power reactors generally use natural circulation These designs may be more applicable than loop type designs for district heating and in remote locations

7 6 International cooperation is strongly recommended in carrying out further development of this type of reactors The efforts should include computer code validation and simulation of accidents in integral reactors, noting the specific problems of modeling natural circulation and the effect of non-condensables in accident condition

7 7 Other areas that should receive special attention include

a) Economic comparison of different integral reactor systems and identification of benefits covering the whole of the life cycle.
b) Ways to maximize the safety of integral reactors, especially to enable their siting near population centers
c) Requirements for plants in remote areas where more stringent restrictions on operation may be needed.
d) Decommissioning of integral reactors
e) Design of compact steam generators
f) Survey of market potential for integral reactors.

DEVELOPMENT PROGRAMMES AND CONCEPTUAL DESIGN DESCRIPTIONS

XA9745970

NEW GENERATION NUCLEAR POWER UNITS OF PWR TYPE INTEGRAL REACTORS

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Abstract

Design bases of new generation nuclear power units (nuclear power plants - NPP, nuclear co-generation plants - NCP), nuclear district heating plants - NDHP), using integral type PWRs, developed in OKBM, Nizhny Novgorod and trends of design decisions optimization are considered in this report.

The problems of diagnostics, servicing and repair of the integral reactor components in course of operation are discussed. The results of safety analysis, including the problems of severe accident localization with postulated core melting and keeping corium in the reactor vessel and guard vessel are presented. Information on experimental substantiation of the suggested plant design decisions is presented.

INTRODUCTION

The integral lay-out realized in boiling water reactors and BN-type reactors is a result of the search for optimum technical and economically substantiated decisions.

An analogous search process is also characteristic of the reactors of PWR type.

Investigations and developments allow the conclusion to be made that certain conditions integral reactors have considerable advantages as for mass and size in comparison with loop-type and unit-type plants.

Besides the integral lay-out, the reactor has advantages as for safety, quality of fabrication, mounting, building time and removal from operation. But the integral lay-out objectively complicates the reactor design and the problems of operational service, it causes the necessity to use highly reliable in-reactor equipment.

In the development of integral reactors especially important are specific characteristics of the heat exchanger (steam generator) built into the reactor, because the reactor vessel dimensions depend largely on heat exchange surface dimensions.

The lifetime reliability of the reactor components should be confirmed by operational experience as a part of operating reactor plants and their prototypes or by broadened complex representative tests at testing facilities in the conditions corresponding to operation conditions in the plant.

For some decades, OKBM specialists developed ship nuclear power plants and experience has been accumulated on the development of some equipment and the NPP as a whole, fabrication and experimental development of some equipment, designer supervision of the fabrication at the factories and in course of operation.

The afore-mentioned allowed the development of new generation nuclear power plants with integral reactors.

First of all is the reactor plant AST-500 for NDHP, which may be located in the vicinity of large cities.

The AST-500 reactor plant is the first in the group of the plants with integral PWRs. Its characteristics are widely known. Its main peculiarities are following: natural coolant circulation in the reactor, high safety level provided by passive means.

The high safety level of the RP AST-500 was recognized by national technical and ecological expert examination, supervision bodies and a special commission PRE-OSART IAEA.

The main fundamental decisions of the NPP, such as integral reactor design, use of guard vessel, use of passive safety systems of various principles of operation with deep redundancy and self-actuation became the basis of the whole group of the developed plants of ATETS-200, VPBER-600 type and the others.

The main advantages of the integral design in comparison with traditional loop-type designs:

- localization of radioactive coolant in one vessel (excluding purification system);
- absence of large diameter pipelines and nozzles in the primary circuit;
- keeping the core under water level at any loss-of-tightness due to the proper choice of guard vessel volume;
- decrease of neutron fluence to the reactor vessel to the level, excluding any noticeable change of the vessel material properties, radiation embrittlement (fluence $<10^{17} n/cm^2$);
- higher completeness of the reactor plant important equipment, of the guard vessel at the delivery to the site and as a result increase in the quality of mounting the power units as a whole;
- reduction of NPP building time to the reducing of the installation work and simplification of construction work;
- considerable simplification of the technology and operations at NPP decommissioning and RP change for repeated use of NPP structures.

Possible negative consequences of the integral reactor design are the following:

- delivery of off-gauge heavy cargo from the factories;
- the necessity to increase considerably the rated load of mounting cranes at the site.

Corresponding analysis and the experience of delivery of AST-500 reactors and guard vessel to the sites of Nizhny Novgorod and Voronezh confirm the feasibility of such delivery by the existing engineering means.

REACTORS PLANT FOR NUCLEAR COGENERATION SYSTEMS OF ATETS-200 TYPE

Reactor plants of ATETS-200 type are a group (ATETS-80, ATETS-150, ATETS-200) of plants of the same kind, developed on the base of an integral reactor with natural coolant circulation, they are autonomous sources of electric energy and heat.

Compactness of the integral reactor, simplicity of the primary circuit and use of a highly efficient steam generator allows natural circulation in all conditions and excludes the use of pumps when providing electric power of up to 200 MW. The ATETS-200 reactor vessel dimensions are not greater, than the dimensions of AST-500 reactor vessel, mastered by the industry.

The investigations confirm the possibility of increasing power up to 250-280 MW.

The plant is notable for the wide variety of passive channels for residual heat removal:

- to each of two heat exchange loops (Fig.1) a channel is connected, which provides

ATETS-200 reactor plant flow diagram

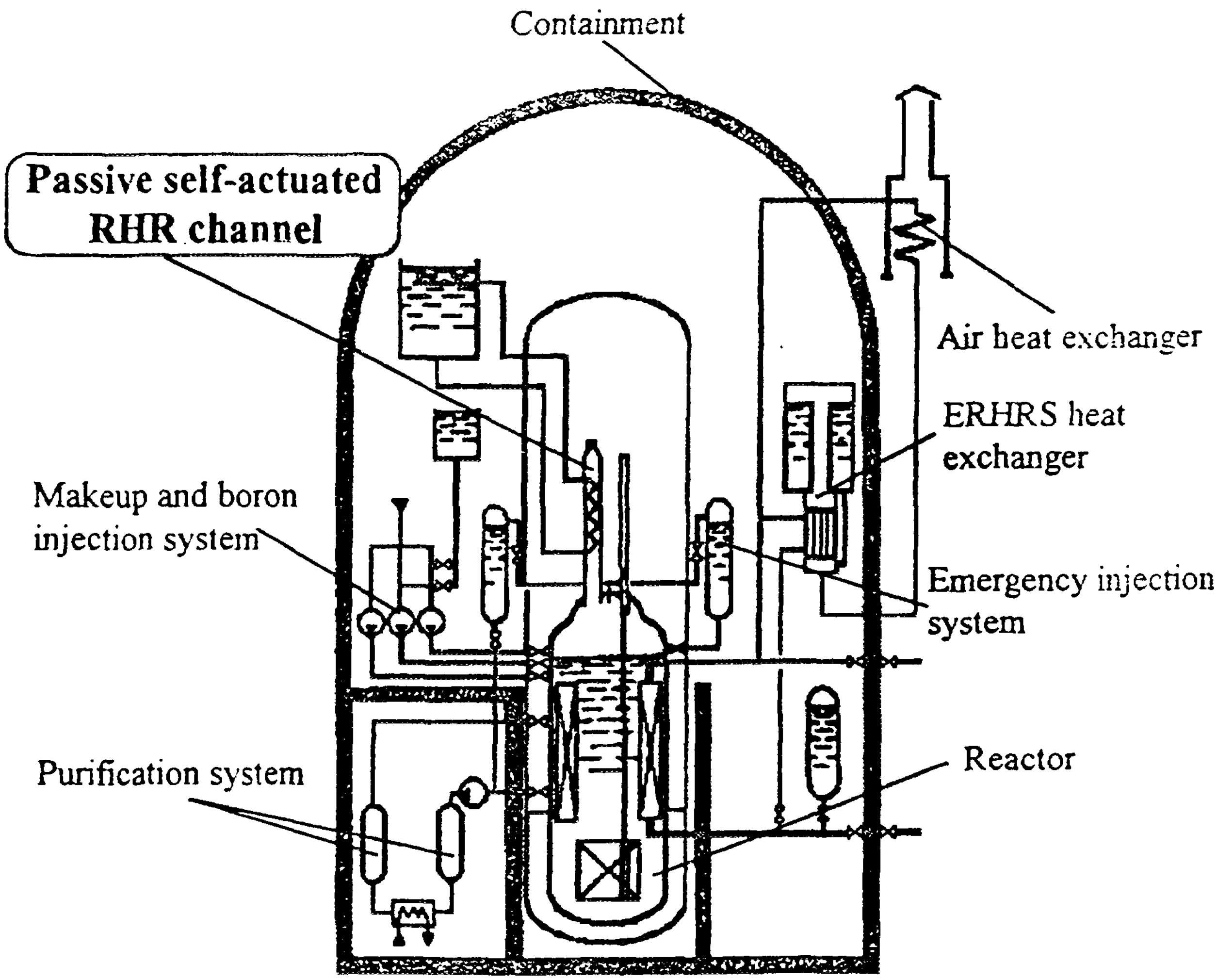
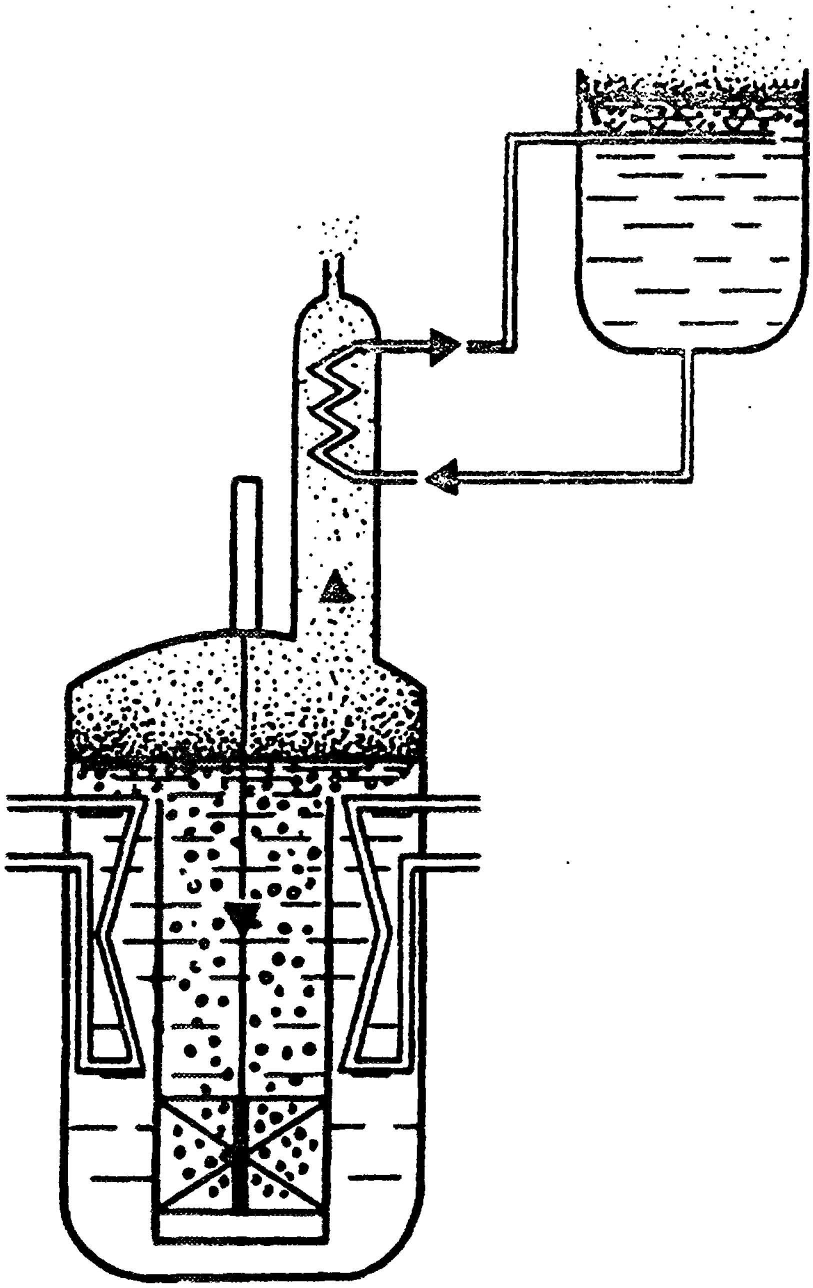


Fig.1



ERHR channel
on the Reactor Upper head

Water inventory in ERHR tank - 150 m^3

Cooldown diration (grace period) - 24 hrs

The channel self actuates at the reactor pressure increase up to 21^{+1} MPa

Fig.2

residual heat removal through SG with natural circulation with heat removal to water tanks, from where water is evaporated to atmosphere;

- independent passive channel of heat removal (Fig.2) is located on the reactor. With its help primary circuit heat is transferred through the wall of the condenser-heat exchanger by natural circulation to a water tank and then it is removed to atmosphere.

Self-actuation of ERHRS channels with emergency protection actuation and, if necessary, of actuations of location system with use of self-actuation devices is provided.

VPBER-600 REACTOR PLANT

An integral reactor with forced coolant circulation at emergency power level operation and natural circulation for residual heat removal is used in the design of the VPBER-600 reactor plant for the power unit of a new generation NPP of 640 MW(el) power.

Forced coolant circulation is provided with the help of six leak-tight circulation electric pumps, located on the bottom of the reactor vessel.

In the design of some equipment and systems the decisions have been mode which have been verified by long experience of operation of the existing nuclear power plants.

Calculation analysis of the wide range of accidents, performed on the base of both deterministic and probabilistic approaches demonstrated the high safety of the plant. Safety in the course of three days is provided by passive means without power supply from outside nor personnel intervention.

Form the point of view of a deterministic approach for severe core damage, multiple failures of safety systems elements and systems as a whole are necessary. The probability of severe damage to the core, evaluated deliberately conservatively is $< 10^{-8}$ per reactor/year.

Nevertheless the search for design decisions for optimization of the reactor design and improvement of characteristics including safety provision for severe accident-accidents with postulated melting of the core continues.

As a result circulation pumps were moved from the bottom to the cylindrical part of the reactor vessel, reactor internal heat exchangers of the system for emergency heat removal were excluded, engineering decisions for the limitation of the consequences of severe accident were proposed and the possibility of corium confinement in the reactor vessel or guard vessel was shown.

Moving the circulation pumps to the cylindrical part of the vessel simplifies the operational servicing of the reactor, excludes the possibility of coolant leakage below the core and improves the conditions for corium confinement in the core and in the reactor vessel and for creation of an in-reactor corium catcher.

The schematic diagram of the system of severe accident localization, presented in Fig.3, includes:

- heat exchangers-condensers, of the system of purification and boric reactivity compensation, total power 10 MW are located in the guard vessel;
- two safety complexes (DN 100), each consisting of a membrane-rupture device and a safety valve in series;
- two temperature-actuated devices for the reactor pressure relief;
- bubbler of approximately 300 m^3 volume;
- receivers of 600-700 m^3 volume.

The bubbler and receivers have the same strength as the guard vessel. In a severe accident with core melting when the temperature in the reactor reaches 500-700°C, safe

SYSTEM OF SEVERE ACCIDENTS LOCALIZTION

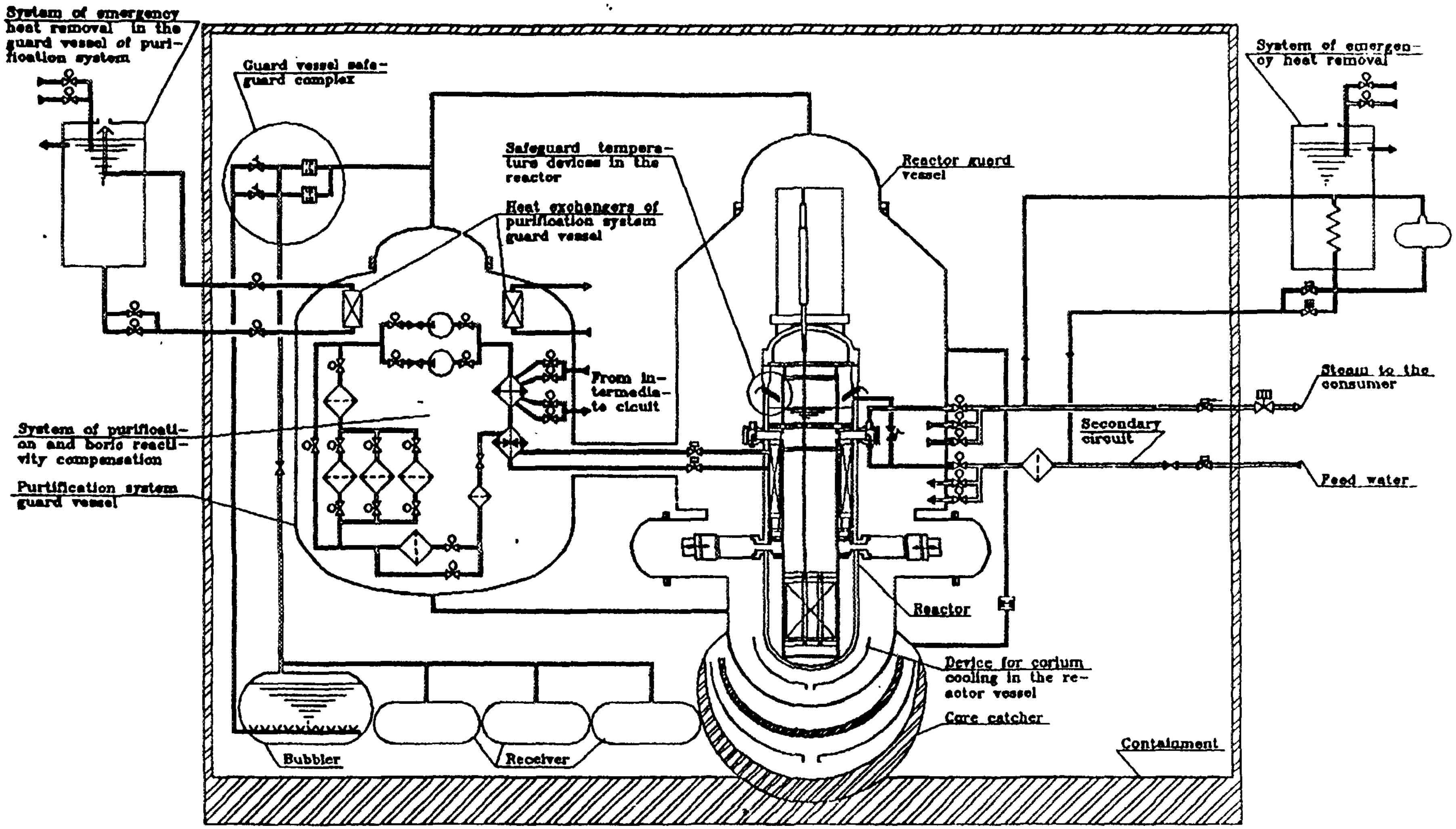


Fig. 3

Complex System of VPBER-600 Thermo-physics and Safety Investigations

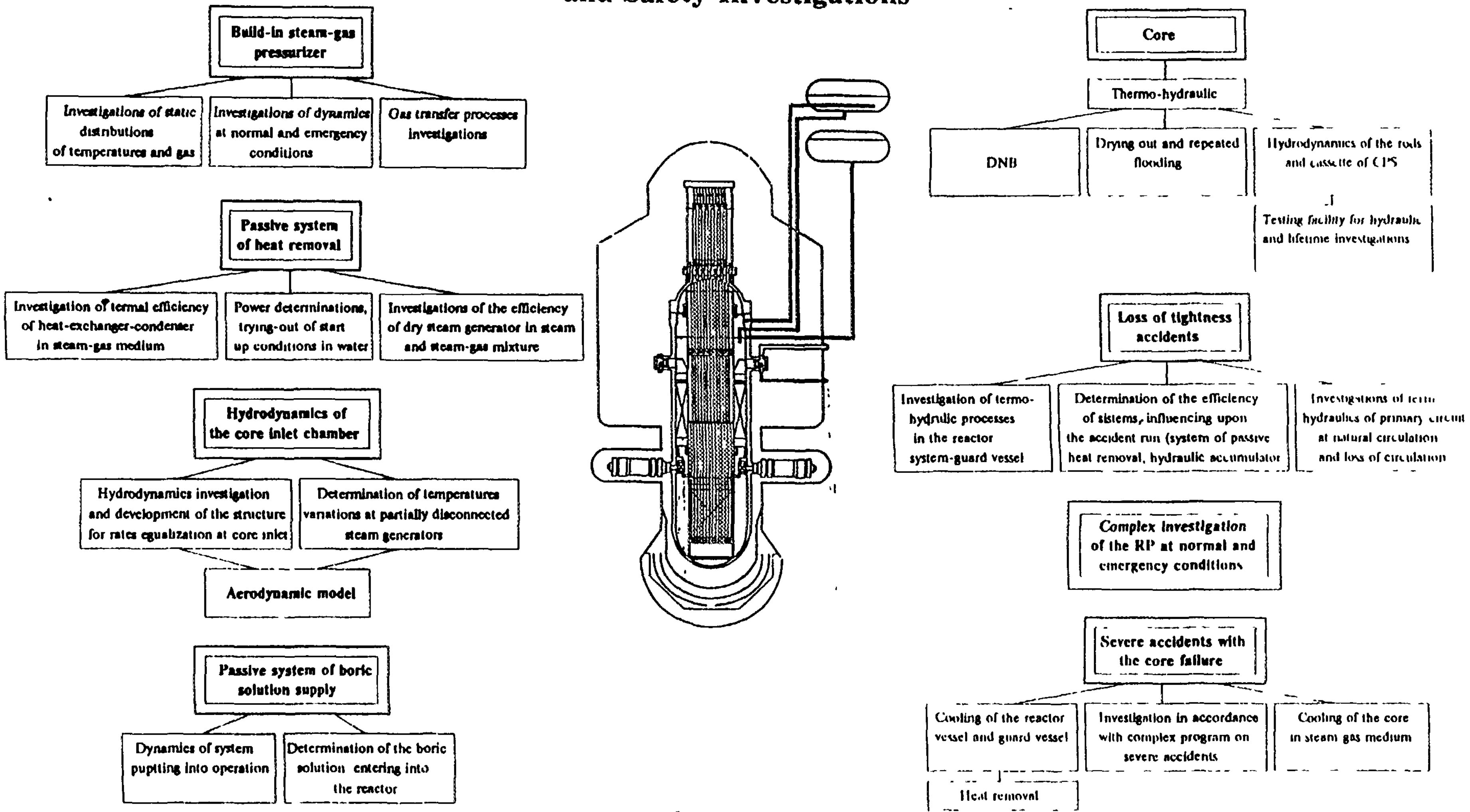


Fig.4

temperature devices of the reactor are opened and steam-gas mixture is discharged from the reactor to the guard vessel, untilpressure is equalized in the guard vessel and the reactor. When pressure in the guard vessel is 5.0 MPa a membrane-rupture device on the guard vessel is broken, a safety valve is opened and steam-gas mixture is discharged from the guard vessel to the bubbler. Gases liberated in core melting are pressed out to receivers, this passively solves the problem of provision for hydrogen safety.

Heat removal from the reactor vessel when corium is confined in the reactor or from the guard vessel false bottom when corium is confined in the guard vessel if it leaves the reactor vessel is performed with the help of heat exchangers-condensers in the guard vessel of the purification system.

In normal operation heat exchangers in the guard vessel of the purification system are disconnected from heat exchangers unit by the pipeline for water supply and are connected remotely by the operator in the event of severe accident.

CALCULATION AND EXPERIMENTAL JUSTIFICATION OF DESIGN DECISIONS OF INTEGRAL REACTORS

Integral reactors, developed in OKBM, being one of the varieties of PWR, are based on the common research and development work and on the experience in the creation, operation and development of such reactors.

But the novelty of the design decisions, connected with the integral lay-out of the reactor, the presence of a steam-gas pressurizer and guard vessel, the absence of circulation loops in the circuit and some others, demands special research work to be performed.

A lot of research work, connected with the experimental study of thermo-hydraulic processes in integral PWRs with a built-in steam-gas pressurizer have been performed in the existing experimental base.

The experimental investigations performed confirmed the main design decisions for equipment and systems and allowed substantion of the correctness of the chosen regime parameters, reliability and safety of the plant.

Together with the problems of the study of thermo-hydraulic processes, occuring in the plant in emergency conditions and of substantiation of the operability and efficiency of the provided safety systems, the most important problem for the experiments is to collect representative information for computer code verification.

The main investigations, which are being performed at present are the following:

- investigations of DNB in fuel assemblies and temperature state of fuel elements at partial and complete dry-out of the core;
- investigation of the conditions of steam condensation from steam-gas mixture in the heat exchangers-condensers of the emergency residual heat removal systems and in the built-in steam generators;
- investigations of steam-gas mixture distribution inside the pressurizer;
- investigations of water-gas and chemical conditions in the primary circuit, including gas transfer in the circuit;
- investigations at integral facilities, including a wide range of emergency conditions with primary circuit loss of tightness and heat removal disruption;
- investigations, verifying thermo-hydraulic and lifetime characteristics of steam generators.

Fig.4 shows the complex of facilities for thermo-physical investigations and safety of VPBER-600.

To verify the results on the problem of corium confinement, it is necessary to perform additional investigations into thermo-physical, physico-chemical and thermo-mechanical processes, to improve calculation modes and computer programs. Besides, the conservativeness of assumptions made considerably compensates for the lack of information and gives every reason to obtain a positive solution of the problem of corium confinement in the reactor vessel or guard vessel.

Now testing facility for an integral PWR of 200 MW power is being made ready for putting into operation.

MAINTENANCE OF INTEGRAL REACTORS (EXAMINATION, REPAIR, DIAGNOSTICS)

The scope and contents of the procedures for maintenance of integral reactors, developed in OKBM meet the requirements of national regulatory documentation. Thereby the following peculiarities of the integral reactor are taken into account:

- presence of the guard vessel;
- location of steam-generators (heat exchangers) in the reactor vessel.

A complex of special devices for scheduled servicing and, if necessary, for repair and reconditioning work which account for the peculiarities of integral reactor lay-out has been developed and tested in AST-500 reactor conditions.

As for inspection of metal and welded joints the following measures are provided in the design:

- periodic visual inspection with video recording of the part of the reactor vessel visible in the zones between heat exchangers with the help of a periscope and of the whole surface when the heat exchangers (SG) are removed;
- periodic eddy current and ultrasonic inspection of the welding and main metal of the reactor vessel in the core zone:
- periodic outside visual and ultrasonic inspction of the reactor vessel with the help of a rotational device and a universal self-propelled device;
- periodic radiographic inspection of the welds of nozzles and penetrations in the upper part of the reactor;
- periodic inspection of the main metal and welds of the reactor vessel using test sample;
- periodic visual inspection of the state of in-vessel devices on removal from the reactor.

The strength and leak-tightness of the structures is confirmed by:

- periodic hydraulic tests of the reactor and heat exchangers of the primary and secondary circuits (steam generators);
- periodic pneumatic tests of the guard vessel.

In RP power operation, constant control monitoring of the reactor and guard vessel leak-tightness is provided by measuring the GV environment parameters (pressure, activity, gas content), also acoustic-emission methods of inspection are used.

Constant monitoring of primary-secondary circuit heat exchanger (SG) leak-tightness at RP power operation is performed by measuring the activity and gas content of the secondary circuit medium.

In the event of heat exchanger (SG) loss of tightness the leaking section is looked for, the leaking part is plugged or (if necessary) the whole section is substituted with the help of special devices.

Analysis and discussion of the decisions made by operating staff experts have shown their acceptability during operation.

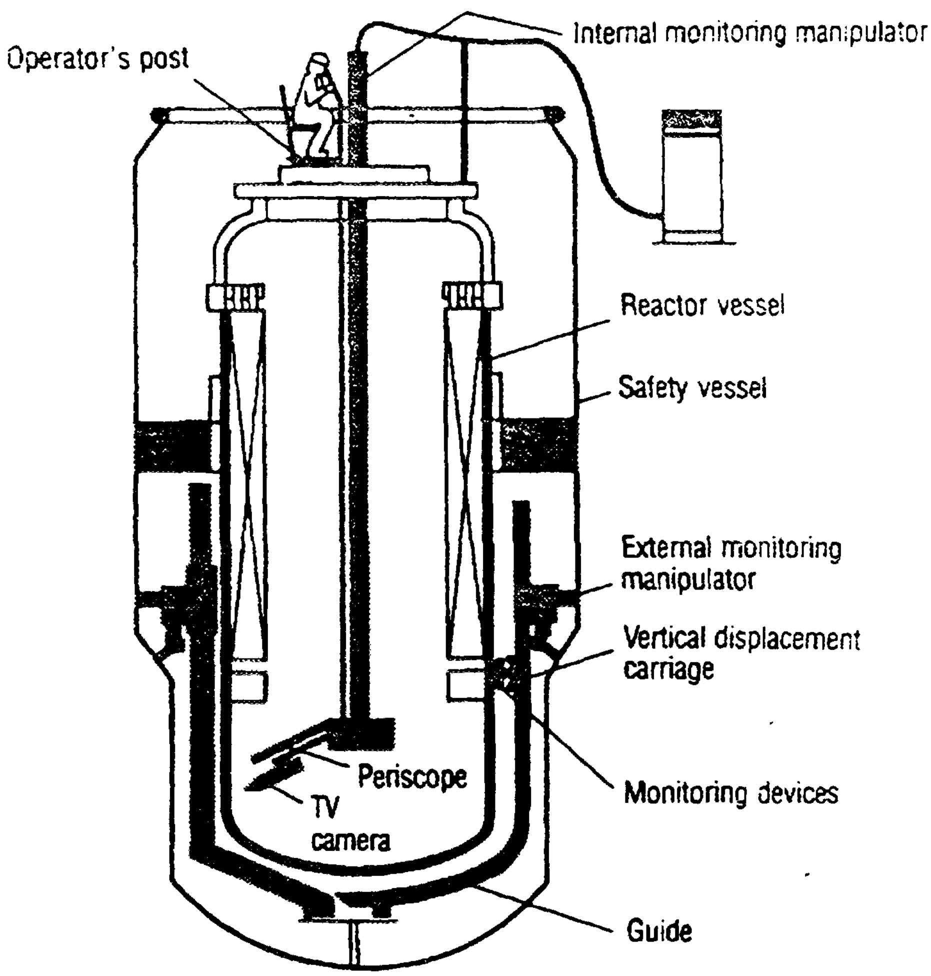


Fig.5

Arrangements for monitoring the AST-500 reactor vessel

When the problems of scheduled servicing and repair are considered, the problem of the cleating to operate the plant before the end of its lifetime should also be discussed.

One of the advantages of the integral reactor is that it is simple from the point of view of technology and radiation to remove it from operation. The presence of a thick water layer between the core and reactor provides low radiation levels from the structures. It allows performance of dismantling work in the reactor cavity using standard equipment without using special means of protection and unique mechanical arms, very soon after reactor shut down and core unloading. The main part of the equipment is low radioactive or even not radioactive and may be dismantled in the same way as at industrial plants. The mass of in-reactor equipment with high radioactivity, which is dismantled by standard means is 2% of the mass of the reactor unit.

In conclusion it should be mentioned, that the results of OKBM work on the reactor plants NDHP, NCP, VPBER show, that an integral lay-out of the reactor gives additional, new possibilities for NPP increase of safety in comparison with loop-type plants. The difficulties of operational servicing, caused by compactness are overcome when highly reliable in-reactor equipment is used.

XA9745971

REACTOR TYPE CHOICE AND CHARACTERISTICS FOR A SMALL NUCLEAR HEAT AND ELECTRICITY CO-GENERATION PLANT

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Abstract

In China, heat supply consumes more than 70 percent of the primary energy resource, which makes for heavy traffic and transportation and produces a lot of polluting materials such as NO_x, SO_x and CO_2 because of use of the fossil fuel. The utilization of nuclear power into the heat and electricity co-generation plant contributes to the global environmental protection.

The basic concept of the nuclear system is an integral type reactor with three circuits. The primary circuit equipment is enclosed in and linked up directly with reactor vessel. The third circuit produces steam for heat and electricity supply. This paper presents basic requirements, reactor type choice, design characteristics, economy for a nuclear co-generation plant and its future application.

The choice of the main parameters and the main technological process is the key problem of the nuclear plant design. To make this paper clearer, take for example a double-reactor plant of 450 × 2MW thermal power. There are two sorts of main technological processes. One is a water-water-steam process. Another is water-steam-steam process. Compared the two sorts, the design which adopted the water-water-steam technological process has much more advantage. The system is simplified, the operation reliability is increased, the primary pressure reduces a lot, the temperature difference between the secondary and the third circuits becomes larger, so the size and capacity of the main components will be smaller, the scale and the cost of the building will be cut down. In this design, the secondary circuit pressure is the highest among that of the three circuits. So the primary circuit radioactivity can not leak into the third circuit in case of accidents.

keywords

small-sized nuclear co-generation, integral, self-pressurized, forced circulation, water-water-steam technological process

1 Introduction

With the development of human society, more and more heat supply is needed . The heat supply consumes more than 70 percent primary energy resource in China, while the electric power supply consumes only about 20 percent of the resource. It not only uses up about more than 4 × 10^8 tons coal every year, but also burns a great quantity of oil, which makes for heavy traffic and transportation , and causes environmental pollution and unnecessary resource waste. It is a good idea to construct a batch of nuclear co-generation plants near to the cities where the steam and heat supply is centralized in the future. This is a new way to save energy, relax traffic and transportation and reduce environment pollution.

The development of nuclear co-generation plant is a possile of prospective economical way, but how to ensure that the small-sized plant clean, safe and cheap is the first problem to be considered by the designers. Based on previous design experience, the plant design objectives are put forward as follows:

(1) Low basic capital investment. The specific cost of the plant should be less than 750 $/KW;

(2) Construction period: A single-reactor plant needs about 5 years and a double-reactor plant needs about 6 years;

(3) Operational reliability: The availability is up to about 85%-90% and the load factor is more than 80%;

(4) No large pipe break accidents (all the plant pipe diameters are less than 100mm) and no LOCA in the primary circuit. The reactor melt probability is less than 10^{-7}/reactor-year. The reactor can be cooled using its own resources and the storage battery sources when normal onsite and offsite electric power is lost. The plant can be built in a nsely-populated area because of the reactor's passive safety and high reliability;

(5) Being simplified in system, compacted in layout and of small constructive scale;

(6) Because the high and large reactor building is cancelled and the foundation loading is lightened, the plant can be built in seabeach, soft soil, seismic areas and so on. On the plant site choice, it is similar to the fossil-fueled power plant;

(7) The heated steam for heat supply has no radioactivity and its radiation level almost equals that of natural radiological reference state;

(8) The reactor building is pressurized and airtight. The environment around the plant would not be polluted in case of radioactive leakage.

2 Reactor type choice for nuclear co-generation plant

The base of reactor type choice is whether it will meet the above-mentioned design objectives, technology reality and feasibility and users requirements' or not.

Pressurized water reactors (PWR), boiling water reactors (BWR) and high temperature gas-cooled reactors (HTGR) can all be used as the co-generation plant. Comparing with the other types of reactors, for PWR there is a lot of experience in experiment, research, design, manufacture, installation and operation in China.

The steam supply parameters are also the base of reactor type choice. According to the steam parameters which a few large Chinese technological process users need at present, the majority is steam in the middle or in low pressure. From an application of view, PWR and BWR can all meet users' requirements. The primary circuit pressure of a BWR is usually 6.86MPa. Its core outlet temperature is 285 ℃, while usually the core outlet temperature of a PWR can reach 310-325 ℃ which can produce medium pressure steam in the third circuit. The PWR has a larger application range than the BWR.

In summary, a PWR should be chosen as the nuclear co-generation plant in China, and the steam supply capacity is usually no more than 1000t/h, so the single-reactor thermal power will not be more than 600MW. A double-reactor in a plant would be better suited to our conditions in which heat and electricity are supplied at the same time and how much the electricity will be produced is based on how much the steam is consumed.

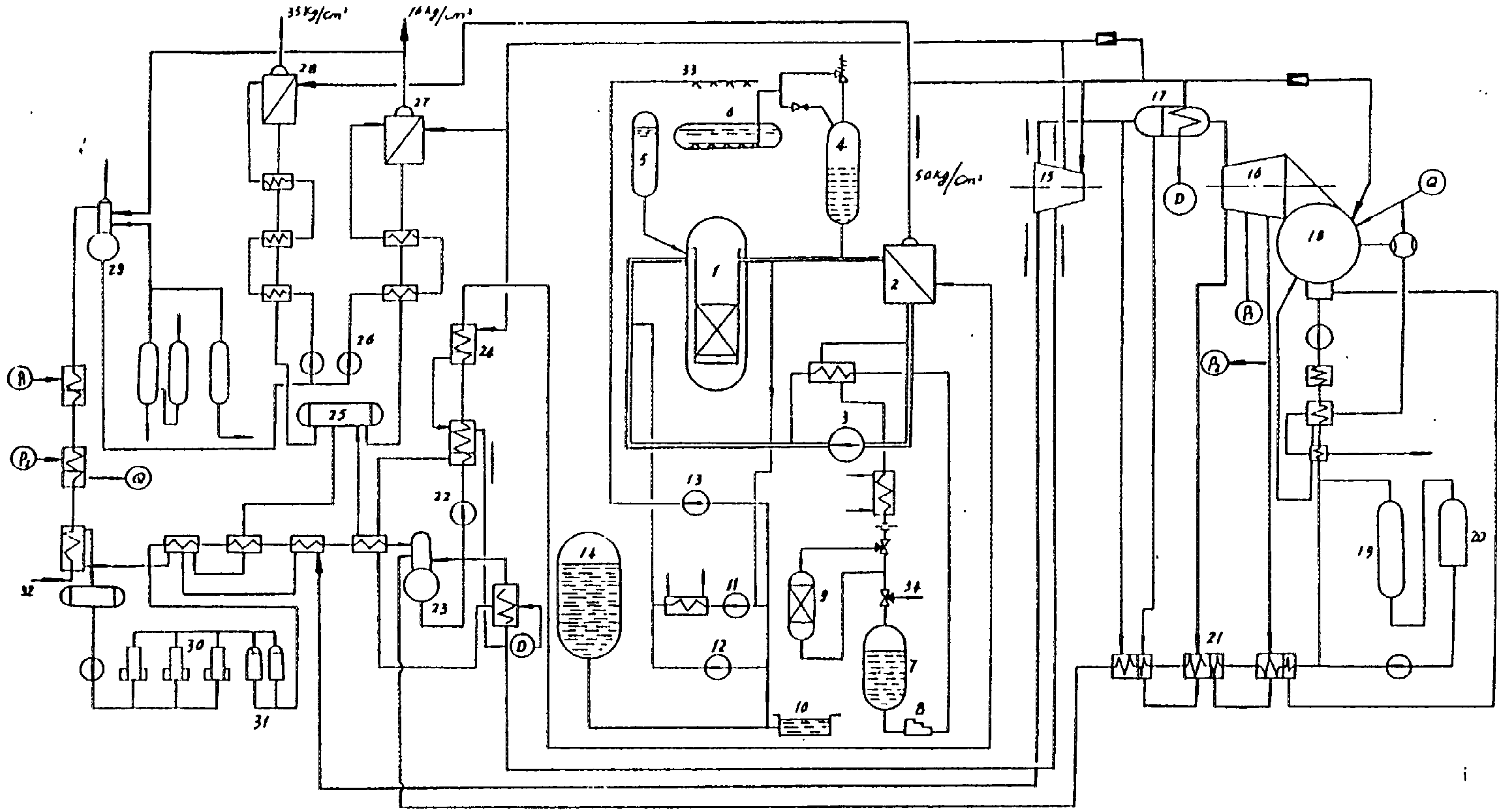


Figure 1 Main System Diagram of Nuclear Thermal Power Plant

1.reactor
2.steam generator
3.main coolant pump
4.pressurizer
5.safety injection tank
6.relief tank
7.volume control tank
8.charging pump
9.resin bed
10.containment sump
11.residual heat removal pump
12.safety injection pump
13.spray pump
14.refueling water tank
15.high pressure turbine
16.low pressure turbine
17.moisture separator and reheater
18.condenser
19.hydrogen ion exchanger
20.mixed bed
21.low pressure heater
22.main feed water pump
23.high pressure deaerator
24.high pressure heater
25.storage tank
26.pressurizing pump
27.low pressure evaprator
28.medium pressure evaprator
29.low pressure deaerator
30.electric magnetic filler
31.mixed bed
32.make up water for third circuit
33.containment spray
34.to boron recycle system

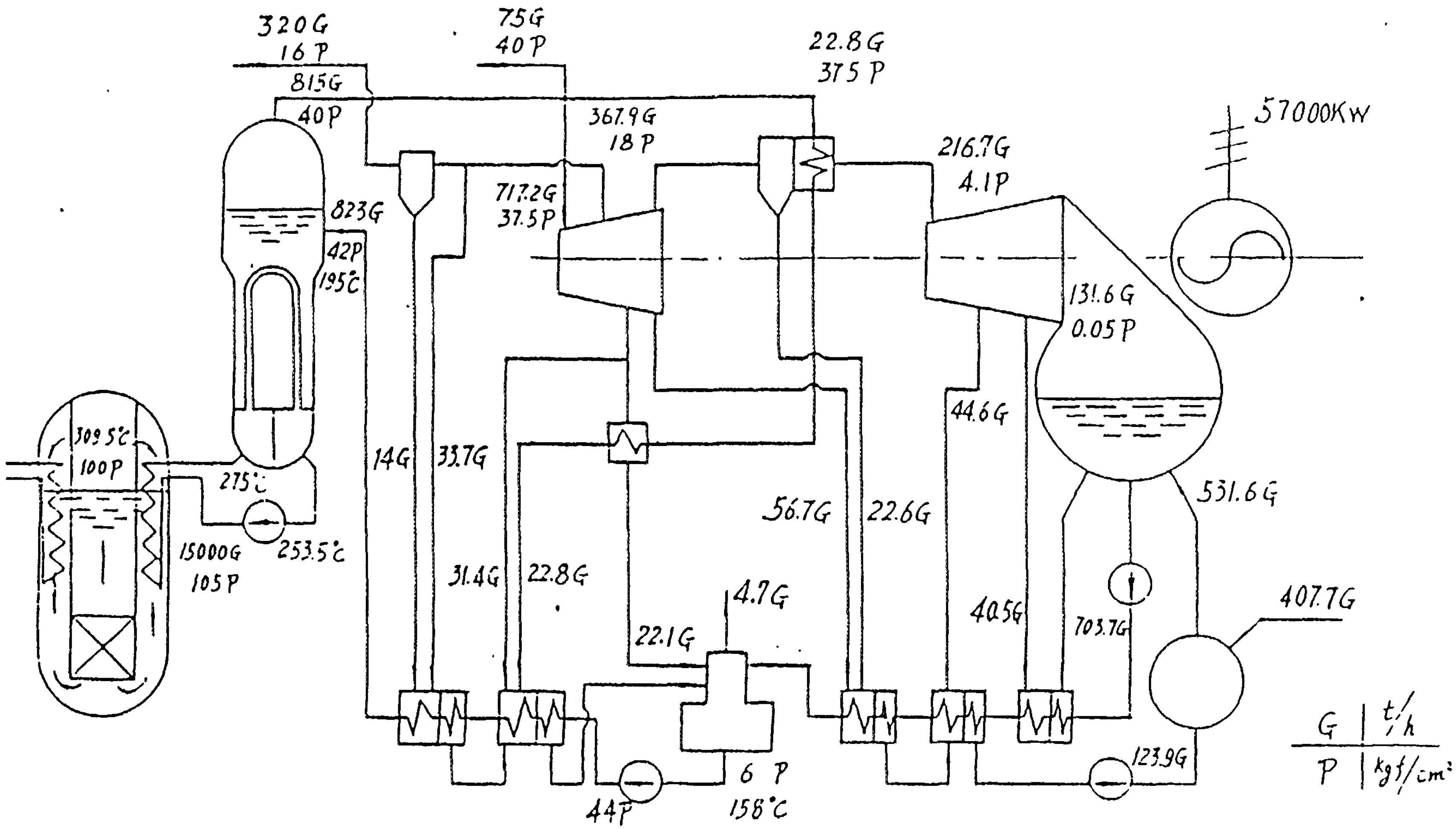
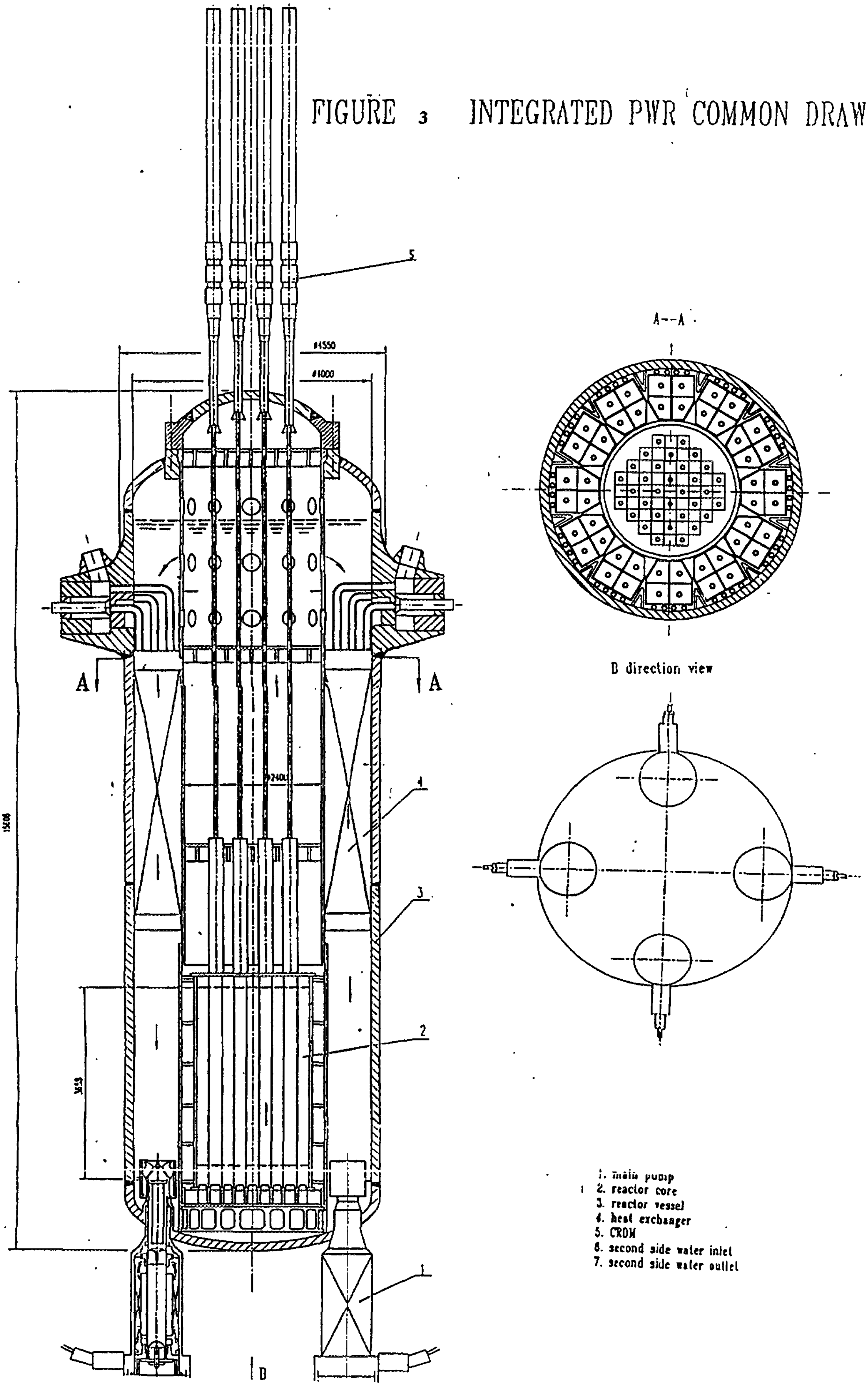


Figure 2 Small Sized Nuclear Co-generation Plant Thermal-hydraulic System Flow Diagram

FIGURE 3 INTEGRATED PWR COMMON DRAWING



1. main pump
2. reactor core
3. reactor vessel
4. heat exchanger
5. CRDM
6. second side water inlet
7. second side water outlet

3 Summary of the small-sized nuclear co-generation plant design

A 450 MW nuclear co-generation plant will be considered as an example in this paper. In order to simplify the system, reduce the construction scale and the cost of the plant, an integral self-pressure-stabalized water reactor is adopted.

3.1 Main parameters and main technological process

The Choice of the main parameters and the main technological process is the key problem of the nuclear co-generation plant design. In conventional consideration, the original design of the plant of Shanghai Jinshan General Petro-chemical Works adopted the water-steam-steam technological process. (figure 1). The normal operation pressure of the main coolant is 15MPa. The middle circuit steam pressure is 5MPa. The third circuit produces 75t/h steam at pressure 4MPa , 320t/h saturated steam at pressure 1.6MPa and electric capacity 52000KW (the above-mentioned is single-reactor).

The disadvantage of this technological process is as follows:

(1)Radioactivity would be carried into the third circuit steam in case of failure of the heat exchangers, a detection system and the necessary emergency provisions must be set up;

(2) The efficiency of the steam-steam heat exchanger is very low, so the number of the exchangers is large (9 heat exchangers needed in one reactor), and the building area is large;

(3) More makeup water pre-heat exchangers are needed in the third circuit and therefore cause a very high cost.

According to the experience in the water-steam-steam technological process and in order to ensure the non-radioactive steam to be supplied, the three circuits systems are still adopted in the nuclear co-generation plant. In order to simplify the system, reduce the equipments size, and ensure operation reliability and passive safety, the water-water-steam technological process is adopted in the steam supply, its main parameters and principle flow diagram are shown in figure 2.

The base to choose operating pressure and temperature of the three circuits is:

(1)To ensure that 3.92MPa saturated steam in the third circuit would be produced;

(2) To balance the heat exchangers size between the primary and the secondary circuits.

The main coolant pressure adopted 10MPa. Because the reactor adopts self-pressure-stabalized operation, the core outlet temperature is 309.5℃, the core inlet temperature is 289.5℃ and the coolant flow is 14,000t/h.

The middle circuit coolant (water) pressure, which is the highest among the three circuits, is 10.5MPa. So the primary circuit radioactivity cannot leak into the third circuit.

The middle circuit can adopt two, three or four loops. In this paper two loops are adopted. The total circuit flow is 15,000t/h, and its main pump is a wet-stator pump. Its hot leg temperature is 280℃ and cold leg temperature is 258.5℃ .

The third circuit steam pressure is 4 MPa. It adopts U-type SGs. The heat transfer area of each SG is 2666.5m^2 and its design area is 3066.5m^2. The margin is 15%.

The majority of pipes and valves in the secondary and the third circuit are manufactured from low alloy steel and carbon steel.

A comparison of the effects of water-water-steam and water-steam-steam technological process on the heat transfer is shown in table 1. While keeping the steam temperature in the third circuit at the same level, this design has these main advantages:

The primary pressure has been reduced a lot, the temperature difference between the secondary and the third circuits becomes larger, therefore the size and capacity of the main components will be smaller, and the scale of the building will be decreased, which will result in cost cut down.

On the same conditions, if the water-water-steam technological process is adopted, each reactor can produce 5000 KW more electric power.

In summary, it is recommended for a small-sized nuclear co-generation plant to choose the water-water-steam technological process to provide non-radioactivity in the third circuit.

Table 1 Two main technological process comparision

item	water-water-steam	water-steam-steam
reactor power MW	450	450
primary pres. MPa	10	15
primary flowrate, t/h	14000	12000
core outlet temp. ℃	309.5	307.5
core inlet temp. ℃	289.5	282.5
third loop steam supplying pres. MPa	4	4
third loop steam supplying temp. ℃	250	250
difference temp. between primary and second ℃	30.3	30.6
difference temp. between second and third ℃	22.4	16.2(low pres. SG) 17.5(middle pres. SG)
primary and second heat exchange area m^2	2832	3000
second and third heat exchange area m^2	6133	12674

3.2 Reactor design

Figure 3 is the integrated reactor general drawing, including reactor core, 32 groups of control rod drive mechanisms, reactor vessel, reactor internals and so on. The reactor is self-pressured. The average steam quality is about 0.1% in the core. The upper steam space in the reactor vessel is $12.5m^3$ which is larger than that in looped layout by about 60% (with the same power capacity).

The reactor core is made up of 57 17 × 17 Advanced Fuel Assemblies (AFA). The fuel rod diameter is 9.5mm. The average heat flux is $27.4w/cm^2$. The power density is 115.4 W/cm^3. The coolant average flow velocity is 3.84m/s. The central temperature of the UO_2 is 485 ℃ . The maximum central temperature is 1500 ℃. The burnable poison Gd_2O_3 is mixed in the UO_2 so that the reactivity need not be controlled by boric acid.

Twelve units of heat exchanger are installed in the upper side of the core. The heat transfer area of each heat exchanger unit is 255.5 m^2. The tube material is 0Cr18Ni9Ti. The straight-type tubes are linked the manifolds with inner hole welding. The distance between the bottom of the tube and the top of the core is 1m, which can prevent the secondary coolant from being activated seriously.

The four pumps with wet-stators are installed under the bottom of the reactor vessel. The pump structure is similar to that of a boiling water reactor. Its pressure head is 0.13MPa, flow rate is 3500t/h, and hot state power is 210KW. The pump motor power is selected as 300KW. An inertia wheel in the pumps is not needed because the main coolant system has enough natural circulation capacity.

The pipe with maximum diameter in the primary circuit which links the reactor to the outside is the drain pipe of the purification system, and its diameter is 50mm, so only a small break accident is considered in the design. The drain pipe passes from the reactor vessel top side through the heat exchanger and extends to the bottom of the heat exchanger. The location of other pipes which link the reactor to the outside, are all higher than that of the heat exchanger manifolds. In case of a small break accident, the steam will run off first and there is 8m-depth of water (about $100m^3$)which is above the core, so the possibility of the reactor core exposure is very low.

4 Preliminary analysis on general investment, amount of oil replaced and economy

The estimated preliminary investment of the double-reactor plant is about 211 million dollar. The plant completed investment is 441 million dollar.

The plant load factor can reach 80%. The main reasons are as follows:

(1) Simplified systems;

(2) The primary circuit pressure is reduced from 15MPa to 10MPa;

(3) The component reliability is increased obviously, especially the water-water heat exchanger is more reliable than a water-steam one in the primary circuit.

In calculating the amount of oil replaced, 7008 hours operation time is considered.

The double-reactor has 8 pumps and total consumption of power is 1600KW in the hot state. The third circuit feed water pump can save power consumption by 20%, but the middle circuit pumps need more power. So the saved offsets the loss. But the pressurizer, wastes disposal system, boron recycling system and high pressure injection system can save some electric power. So the electrical power used in the plant is estimated as 16.5MW. In our estimation, the oil replaced capacity is about 5.15×10^5t/a.

The following is a brief plant economic analysis:

(1) Electricity and heat co-generation cost

The estimated plant operation cost is 23.5 million dollar each year. It produces 0.798 billion KWhr electricity (about 0.683 billion KW.hr for users), 3558 billion Kcal heat. If the price of one million Kcal heat is considered as 2.95 dollar, the heat supply can obtain 10.5 million dollar profit. Thus, the gross cost of the electricity is 0.016 dollar/KW.hr and its net cost is 0.019 dollar/KW.hr.

(2) Retrieval funds

The production cost is about 23.5 million dollar, but the cost of the oil which would be replaced every year, is about 60.6 million dollar. The difference is 37.1 million dollar. The time needed to recover capital is about 12 years. If the oil price is 176.5 dollar/t, the time is 7 only years.

(3) Thermal efficiency

The thermal efficiency of a large nuclear power plant is about 33%, while the nuclear co-generation plant is about 80%.

In summary, the small-sized nuclear co-generation plant is more economic than the coal or oil fuel co-generation plant.

5 Design characteristics and its prospects

(1) Adopting integrated self-pressurized water reactor.

(2) The primary circuit operational state is: the primary coolant is at low pressure; the reactor core has a low power density; the fuel in low temperature; while the coolant has a high average temperature. So not only the safety but also the economical steam is ensured. The third circuit steam pressure can reach 4MPa and thermal efficiency can come up to about 80% while the primary pressure is only 10 MPa.

(3) Adopting a water-water-steam technological process improves the operation reliability of the heat exchangers in the reactor. The middle circuit pressure is higher than the primary's, so radioactivity leakage is prevented from the primary to the third. The size of the heat exchangers and related buildings is reduced.

(4) The reactor and the primary circuit have adoped an adopt integrated layout. The inner diameter of the containment is only 16m, so the reactor building is reduced to a minimum and the nuclear island covers 2518m^2, only 32% that of looped layout.

(5) Large amount of concrete and steel will be reduced and the general cost will also be reduced because of the small constructive scale. And a half of million tons oil will be replaced every year. So the time to recover capital is estimated as about 7 years.

(6) High passive safety and reliability. About 20% thermal power can be removed by the natural circulation of coolant when the main pumps stop. So a core melt accident will not occur when the main pump motor electricity is lost. It is almost impossibly difficult for radioactivity to leak into the third circuit because of the high pressure in the middle circuit. The accident probability drops down because the systems are simplified and the number of valves and pumps also decreased. The water-water heat exchanger is more reliable than a steam generator in the reactor. All the above-mentioned can reduce unexpected plant shutdowns and improve the plant availability.

(7) The plant can be built near a city or a large enterprise because of the reactor's high passive safety and two-layered containment.

(8) The plant layout is compact and the main and the auxiliary buildings take up less land, and the foundation loading is lightened. So it is favourable for the plant to be built on seabeach, soft soil or a seismic area.

(9) The main components of the reactor and the primary circuit can be manufactured and installed in the factory. On site installation and examination work become less. The nuclear island constructive scale is small. So the double-reactor constructive period can be shortened to 5 years.

(10) The shape of the main and the auxiliary buildings on the nuclear island is simple, it is easy to construct. So the construction period is shortened.

Table 2 Main Performance Comparison

No	item	integrated layout	looped layout
1	reactor power MW	450 x 2	450 x 2
2	steam supply capacity t/h	75 x 2 (middle pre.) 320 x 2(low pre.)	75 x 2(middle pre.) 320 x 2(low pre.)
3	electricity supply capacity KW	56947 x 2	52000 x 2
4	thermal efficient %	~ 80	79.15
5	plant load factor %	80	70
6	replaced oil amount 10^4 t/a	51.455	43.6
7	primary operation pre. MPa	10	15
8	containment dia. inner m outer m	16 20	29 33.30
9	nucl. island land area m^2	2518	7884
10	nucl. island constrution area m^2 volume m^3	20000 190000	23320 353185
11	conventional island land area m^2	6000	8522.7
12	conv. island constrution area m^2 volume m^3	22800 190000	31728 229808
13	general investment 10^8yuan	10	13
14	construction period year	5	6-7
15	maximum postulated accident	drain pipe(ϕ50) rupture	main pipe(ϕ150) rupture

Table 2 is the main performance comparison between the integrated and the looped pressurized water reactor. The parameters in table 2 indicate that this design meets the basic requirements of a nuclear co-generation plant.

According to the above-mentioned, the small-sized nuclear co-generation plant is economically competitive. Referring to more than twenty users' steam consumption, it is suitable to adopt 400-600MWt power for the single-reactor. For the nuclear co-generation plant, its main function is heat supply, while the amount of electricity supplied depends on how much the heat is supplied. To avoid causing a high turbine cost, the electric power proportion should not be too large. If the heat supply is the main function and the electricity supply is auxiliary, the three circuits and water-water-steam technological process should be adopted.

For the reactor choice, PWR and BWR have all been considered. But the former is put first to raise the reactor core temperature, produce medium pressure steam in the third circuit and expand the scope of steam supply. The reactor and the primary circuit adopt an integrated layout and forced circulation. It is more compact than BWR with natural circulation. It is more suitable that the operation pressure of the self-pressurized water reactor adopts 10MPa according to the steam parameter requirements of domestic chemical industry.

In accordance with the above discussion, for a small-sized nuclear co-generation plant, conclusions are made as follows:

(1) Adopt integrated, forced circulation, self-pressure-stabalized water reactor.

(2) Reactor operation pressure 10MPa.

(3) Single-reactor power 400-600MWt, maximum to 1000KWt.

(4)Adopt water-water-steam main technological process and electricity supply depends on steam supply.

(5) No boric acid and simplified system.

This design also suits a small nuclear electric power plant. But the heat exchangers should be replaced by steam generators in the reactor vessel.

The design can also be used for centralized heat supply in a city. On condition of not changing the reactor and the auxiliary system, only decreasing operation pressure and temperature, and the steam generator being replaced by the heat exchangers in the third circuit, this design would turn to be nuclear heat supply plant, while the research and experiment are not needed furthermore.

Natural circulation and forced circulation should be compared further in the future and the control rod drive mechanism type needs improving. A core with a high conversion ratio in a small reactor is worth researching further.

In summary, a small nuclear heat and electricity co-generation plant could become an economic, safe and clean energy source in a city or a large enterprise in the future. Its basic capital investment is low, the construction period is short, it has a wide range of uses, it has good prospects and it should have a proper position in the energy resource development of China.

THE CAREM PROJECT: PRESENT STATUS AND DEVELOPMENT ACTIVITIES

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Abstract

The CAREM Project is a low power NPP of 25 Mwe, with an integrated self pressurized primary system The cooling of the primary system is of the natural circulation type and several passive safety systems are included The owner of the Project is Argentina s CNEA (Comision Nacional de Energia Atomica) and its associated company, INVAP, is the main contractor

The present status of the CAREM Project is presented The possible evolution of the CAREM project is mentioned in relation with a new containment design A short description of the Experimental Facilities, listed below, already in operation and under construction are also included

- CAPCN High Pressure Loop Natural convection loop to verify dynamic response and critical heat flux
- RA-8 Critical Facility, designed and constructed for the CAREM Project (that may be used as a general uses facility)
- RPV Internals The whole assembly of absorbent rods, connecting rods and the rode guides are being constructed in a 1 1 scale The aims of this experimental facility are vibration analysis and manufacturing parameters definitions
- Control Drive Mechanisms A serie of verification and tests are being carried out on these within RPV hydraulically driven mechanisms

Other development activities are mentioned in relation with the thermalhydraulics, Steam Generators and Control

1. CAREM Project. Present status.

1.1. Introduction

The CAREM, a small NPP Project owned by the CNEA /1/ has been developed jointly by CNEA and INVAP /2/ The concept was born by the 80,s when the idea was presented at LIMA, Peru, during an IAEA conference back in 1984 At that time, the Argentinean nuclear experience was several RR built (RA-0, RA-1, RA-2, RA-3, RA-6, RP-0) several Projects on RR, an enrichment plant and the experience gained in the follow up of two NPP in operation (CNA-1 and CNE) It was thought that the next step to reach the nuclear maturity should be to work in the design and construction of a NPP This development in a medium size economy and restricted financial resources should have a limited risk and so the focus was addressed to small NPP In addition the NPP should be able to operate in isolated cities, a common situation in Argentina That meant reduced ability to obtain help in operational incidents/accidents by the operating personnel, reduced industrial/ technical resources in the site, long distances from fully developed areas, reduced roads and transportation capacities And being the first domestic NPP, it was also considered convenient to work in an electrical plant with a small output when connected to the national electric network Similar conditions could be find in a good number of countries

The above mentioned criterium developed in the CAREM technical characteristics i e a LWR, with Integrated Primary System and extensive use of the so called passive systems (e g natural circulation cooling)

Fixed arbitrariously the power in 15 Mwe, a R&D program was started and so the construction of a thermalhydraulic lab and a critical facility

At present the work is addressed to the 25 Mwe but a new version of 100 Mwe is also foreseen

1.2. Organisation

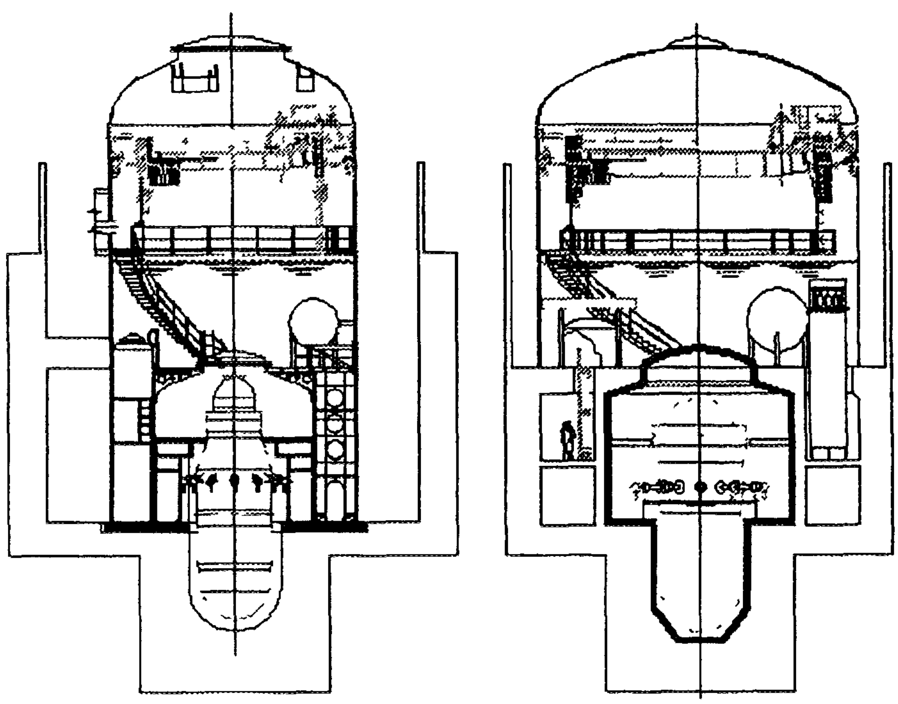
CNEA has subcontracted the engineering of the CAREM to INVAP However by itself is working on the development of the Fuel Elements and in part of the nuclear instrumentation The follow up activities on the CAREM Project are made by CNEA with a Coordination Group which receives all the engineering from INVAP Afterwards the engineering enter in a revision process inside the CNEA by groups specially appointed by the CAREM Coordinator Every three months the Coordination Group have a meeting with their counterparts in INVAP were technical discussions are held Also visit and revisions of the experiment works are done In some especial topics as RPV design and manufacturing, foreign experts are invited to assist and make special revisions Experts from other companies are also sometime required

1.3. Engineering Stage

The CAREM Project is under the development of the Detail Engineering needed to start the construction A technical revision was made by the owner of the project (CNEA) in 1992 and a cost estimation following TRF 269 was finished in 1993

During the present year the main tasks were devoted to work on those subjects with lack of experience and were research and development were needed as RPV Containment, Internals, Thermalhydraulics of different Systems (RPV, Secondary, Primary, Containment etc) lay-out This in depth studies encourage the Project Personnel to study different solutions for the containment and RPV From the left to the right in the next picture we have the previous design and the one in which studies are done at present The "old" design has a containment made totally with steel and the RPV with a conical shape The design was originally aimed to have a modular containment which could be transported in big pieces in order to reduce the on site works However, the solution suggested for the fixation of the containment was a difficult one from the point of view of civil engineers Because of the reduced containment size it was necessary to have the conical shape (small space for the RPV head) The latest design made from concrete in its lower part allow to have bigger available spaces so the RPV is of the straight type This solution increases the maintainability of the NPP, mainly the SG are more available for repairs, one of the weak points in the "old" The more available space allow us to change the SG feeding pipes from a downwards type to a horizontal which permits an easier manufacturing solution Up to today the only drawback of the latest design is the corresponding to the drive mechanisms support which now is positioned in a bigger cylinder

In the following sections a short description of the activities in process except for those related to the SG in the CAREM Project are mentioned In relation with the SG, a Special Program is been conducted That Program has started with the construction of a mini SG to be tested in a high pressure loop Studies are underway to define the whole scope of data that should be obtained from a 1 1 model The studies include the qualification process and the search of a facility for testing the SG

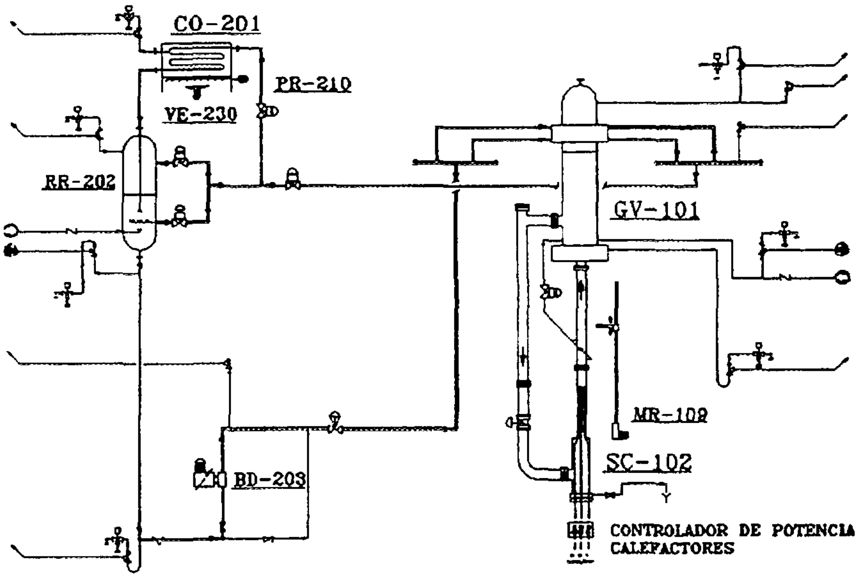


2. CAPCN (Circuito de Alta Presión y Convección Natural): High Pressure Loop.

2.1. Description and Similarities with CAREM-25

The High Pressure Natural Convection Loop is part of the Thermal Hydraulics Laboratory Designed, Constructed and operated by INVAP for the CNEA Its purposes are to verify the thermal hydraulic engineering of CAREM NPP mainly on two subjects dynamic response and critical heat flux These are accomplished by the validation of the calculation procedures and codes for the rig working in states corresponding (by similarity criteria) to the operating states of CAREM reactor

CAPCN resembles CAREM in the primary loop, while the secondary loop is designed just to produce adequate boundary conditions for the heat exchanger Operational parameters are reproduced approximately for intensive magnitudes (Pressures, temperatures, void fractions heat flux etc) and scaled for extensive magnitudes (flow, heating power, size, etc) Height was kept in a 1 1 scale



CAPCN was constructed according ASME for the following parameters 150 bar, 340°C for the primary and 60 bar 340°C for the secondary (pump exception T<240°C) The level difference between center points of core and steam generator is ~5 7 m Primary loop may operate in saturated regime (self-pressurized), or subcooled (dome pressure increased with nitrogen injection), with heating power up to 300 kW and different hydraulic resistance

CAPCN states corresponding to full power of CAREM

PRIMARY		*SECONDARY*	
Pressure	122 5 bar	*Steam pressure*	*47. bar*
Hot leg temperature	326 °C (saturation)	*Steam temperature*	*290. °C (superheated)*
Cold leg temperature	280 °C	*Cold leg temperature*	*200. °C*
Natural circulation flow	1 08 kg/seg	*Flow*	*.128 kg/seg*
Heating power	263 kW		
Riser quality	~ 1 % .		
Heating control	feedback loop of pressure + core dynamic		

The nuclear core is reproduced by electric heaters operated by a feedback control loop of dome pressure (primary self pressurized) For the dynamic tests the heaters are 1 2 meters long

The heaters bundle that will be used for the CHF tests differs from the one used for the dynamic ones, in order to have a configuration that allows to reach heat fluxes high enough to be sure to obtain departure from nucleate boiling for the complete range of pressures, flows and subcoolings. The heated length is 400 mm. The seven rods bundle has two of the rods which allow an overpower of 20% These rods have six thermocouples each in order to ensure the measurement of the exact location of CHF The rest of the rods only have two thermocouples that guarantee CHF detection if it takes place on one of these rods. All the thermocouples are located near the upper end of the heated zone because the axial power distribution is uniform in the CHF heaters bundle

The group of states foreseen for these tests include the following range of main parameters·

	Upper value	Lower value
Mass flow [kg/m^{2}s]	750	200
Pressure [bar]	130	115
Heaters outlet quality [%]	5	- 30

Natural circulation flow may be regulated by a valve in the cold leg and a by pass to the bottom of riser. A gamma densitometer is available for void fraction measurements The heat exchanger (SG) has two coils, once through, secondary inside For the CHF tests the steam generator is useful only as a cold source, so the secondary loop operating parameters are not relevant as long as they can be controlled

The secondary loop pressures and cold leg temperatures are controlled through feedback loops operating valves. The pump allows the regulation the flow The condenser is an air cooled type with flow control

Both loops allow automatic control and can be pressurized by nitrogen injection

The present configuration of CAPCN allows the study of a stationary state similar to the CAREM conditions of pressure, specific flow and enthalpy

The combination of primary and secondary states is limited only by conditions attainable with the heat transfer capacity of the heat exchanger As a consequence this loop will permit to validate the calculation tools used on the CAREM Project (RETRAN and ESCAREM) in those conditions.

The inclusion of a SG (15 tubes) with a design similar to the CAREM will allow to have a full 1-D thermalhydraulics analogy, allowing the extrapolation of results directly to the CAREM-25

The thermal-hydraulic design of CAREM reactor core has been performed using a version of 3-D, two fluid model THERMIT code In order to take into account the strong coupling of the thermal-hydraulics and neutronics of the core, THERMIT was linked with an improved version of CITATION code (developed in INVAP, and called CITVAP) This coupled model allows to obtain a 3-D map of power and thermal-hydraulic parameters at any stage of the burnup cycle.

The thermal limits were calculated using the 1986 AECL-UO Critical Heat Flux Lookup Table, validated till now with all the available measurements in the operational range in order to ensure a 95/95 reliability/confidence in the thermal limit. The CAPCN CHF test results will be used to improve these calculations by increasing the experimental data in the operating range and fuel element geometry of the CAREM core.

2.2. Experimental Program

The Program is divided in two main subjects

Dynamic Tests /3/

In order to reduce the range of experiments it was defined that the studies will be limited to the regimes, states and perturbations foreseen for the CAREM 25. The studies will be limited to those parameters included in the modelling by computer codes.

Several conditions were established:

- The primary subcooled state is relevant only during the start up of the CAREM
- The existence of secondary subcooled and saturated states will depend on the method adopted for the start up. The situation can be avoided if there is steam available.
- Nitrogen in the dome will be limited up to 30 Kg/cm2.
- Possible perturbations in the Primary Side are produced by neutronic changes and power extraction form the secondary side.

Parameters of interest are:thermal balances; water flow; thermal transference coefficients in SG, dome, and void fractions in the riser; Primary and secondary pressure and temperature; dome volume; hydraulic drag; neutron kinetics.
This stage of experiments is at present underway and will be finished by march of next year.

CHF Tests /4/, /5/ and /6/

It is known that to perform reliable thermalhydraulic calculations it is necessary to perform CHF experiments for the start up and nominal power states. In this case the CAPCN will be used with some modifications in the control system in order to obtain stable values for the pressure, water flow, water quality, with a positive power ramp. Part of the experimental work to be started next year will be devoted to determine limits and operational conditions for the CHF experiments. A detailed study was already done with the corresponding Program. For the CHF experiments themselves it was necessary to make carefull studies because it was not possible to install a section of the same dimensions as for CAREM Fuel Element. Variables kept as in CAREM are rod diameter, pitch and ratio total section/water flow section. Preliminary experiments for temperature oscillations due to slug flow patterns will be made before starting the CHF experiments.

2.3. Start-up and experimental results

The phases accomplished for the Start-Up Program are:

- Hydraulics characterisation, cold state.
- Primary isolated: selfpresurization test, control loops calibration. Degasification of primary side
- Whole system operation. Secondary side in liquid phase. Initial calibration of condensed tank control loops in temperature and flow. Thermal balance.
- Whole system operation. Operating regime at low power with overheated steam and automatic control.

At present the following results has been obtained:

Self pressurization in the operating range of CAREM-25 with natural convection was confirmed.
The Thermalhydraulic Process was controlled during the different power conditions without major problems.
The production of over-heated steam is compatible with the CAREM operating conditions.
Considering the loop as an experimental machine the following experience has been obtained:
The Data Acquisition and Control System has fulfilled its required performance. However, some problems related with hardware characteristics aroused: temperature were measured with an accuracy lower than required. An interface between the DACS and the sensors was frequently in needs of recalibration. They are already replaced and a safety controller was added to the Supervision System. Some improvements were made: New Fuel Assembly spacer (CAREM design) in order to make a durability test of this component; new measurements in the heat exchanger and new improvements in the valves used for control. Some modification in the condenser design were also needed to reach the full power state (300 Kw of electrical heat). In order to reduced the experimental errors several data will be also recorded through a new interfase, with a lower error and frequency of calibration needs. Improvements in the thermal isolation were also done.

3. RA-8.Critical Facility.

3.1 Description

The RA-8 critical assembly has been designed and constructed as an experimental facility to measure neutronic parameters of the CAREM NPP, under contract and supervision of the National Atomic Energy Commission (CNEA) of Argentina. It may be used, with relatively minor changes, as a facility to perform experiments for other light water reactors. It provides a reactor shielding block and reactor tanks that can be adapted to hold custom designed reactor cores.

The RA-8 critical facility is located in the PILCA IV Sector of the PILCANIYEU TECHNOLOGICAL COMPLEX, in the Province of Rio Negro, Argentina, at approximately 30 km East of San Carlos de Bariloche. It occupies the main hall of a building shared with the Laboratory for Thermalhydraulic Tests (LET), described in Section 2 of this report, and other special facilities for CAREM Project. Geometry and location of core shielding inside the main hall are such that radiation dose levels are acceptable in adjacent rooms, for all operational conditions.

General Characteristics of the RA-8, are:
Low operating power, which makes cooling systems unnecessary.
Extinction systems.

Rapid insertion of control rods
Dumping of the moderator

Regulation and safety rods. There are at present 13 mechanisms to drive control rods in and out of the core. The Control System allows the definition and use of some of the control rods as Regulation Rods, and some as Safety Rods. The number of rods assigned to each function depends on the specific core being tested.

Argentine Regulatory Authority (ENREN) imposes the following requirements for the design of critical facilities:

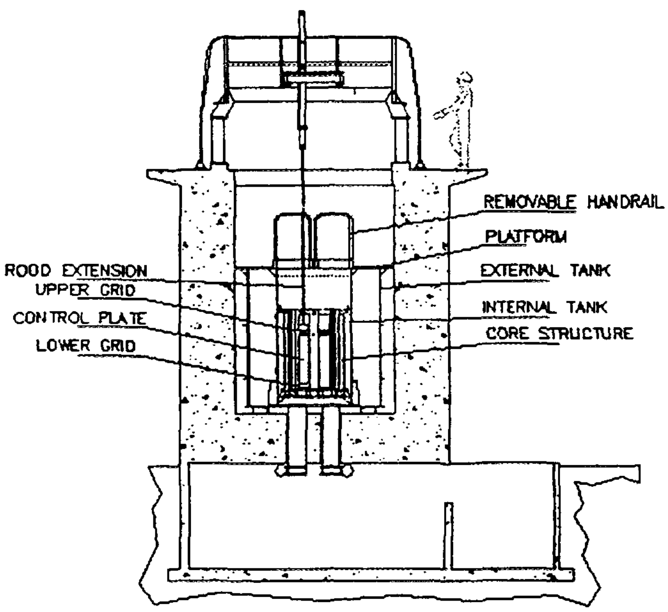
- Negative reactivity introduced by control rods must be higher than 50% of the critical assembly reactivity excess.
- Core reactivity with control rods must be negative and higher than 3000 pcm.
- Core must remain subcritical in at least 500 pcm after extraction of the control rod of maximum negative reactivity
- Reactivity worth of control rods defined as regulation rods must be such that their insertion makes the core remain subcritic in at least 500 pcm.
- Movement of any control rod mechanism must not produce a reactivity insertion higher than 20 pcm/sec.

Operating modes. There are two possible ways of operation:

Operation by critical height. (Reactivity is determined by the moderator level surrounding the core)
Operation with control rods.(Reactivity is regulated by the amount of absorbing material introduced in the core)

Water System: water level is controlled by the RA-8 Control System. During operation, water fills simultaneously two concentric and connected tanks. The inner tank is designed to hold the core, its structural components, and nuclear instrumentation. The water in the outer tank serves the purposes of shielding and reflector. Filling of the tanks is performed in two stages: a first stage of fast pumping, followed by a second stage of slow pumping to approach operating level. Safety Logic takes into account the position of safety and control rods to allow pumping of the moderator into the reactor tanks. Tanks are emptied by the opening of two butterfly valves, centred in the inner tank, which dumps water into the hold-up cistern, below the reactor block. It takes no more than 4 seconds to empty the inner tank.

The water system also has provisions to add boron and to clean, drain and recirculate water. Water temperature can be varied in up to approximately 75 °C. The hold-up cistern has its own water recirculation system.



Data acquisition is done simultaneously and independently by two means: the "hard logic" Instrumentation and Control system needed to operate the facility, and the Control and Data Acquisition System, a microprocessor based system, by means of which the reactor operator will be informed of reactor and experiment related parameters.

At present the facility is being completed and the cold initial start-up is programmed for the end of the present year (1995). The core elements (Fuel Rods and absorbers) for the RA-8 are in manufacturing process and expected to be finished in the first half of next year (1996). The experimental program is foreseen to last for about one year and a half.

3.2. Experimental Program

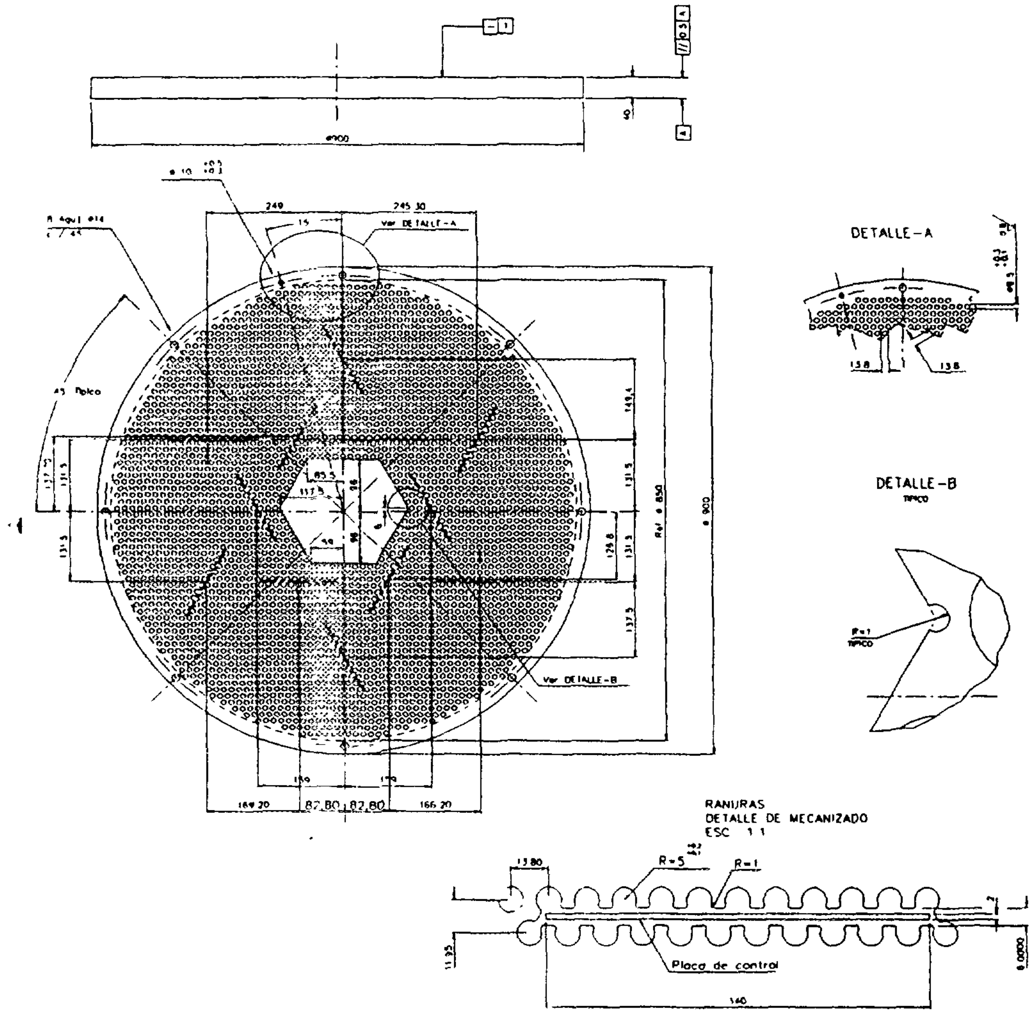
The cores to be used for CAREM related experiments are made from fuel rods with the same radial geometry of the ones for CAREM, but shorters with a length of 80 cm. The pitch of the core was studied by calculation and is in principle the same as for the CAREM. The core calculations are made with a Diffusion Code (CITVAP), so the size of the core has to be enough to have good results. A central homogeneous zone is needed to study perturbations as: rods loaded with different concentrations of burnable poisons, absorbing rods, guide tubes, structural materials, etc. The maximum core reactivity covering all the experiments is around 7500 pcm. To meet the requirements given by the ENREN and with the experiments, plate type absorbers made from bare Ag-In-Cd are used. The distribution of the absorber elements is given in the following figure. The central absorber is not shown.

The studies will be conducted over different cores using two enrichments (E= 1.8 % and E= 3.4 %), some of the defined cores are:

A. One region of E= 1.8 %.
B. Two regions: inner of E= 3.4%, outer of E= 1.8%.
C. One region of E= 1.8% , perturbed with non fuel rods (guide tubes, control rods, burnable poisons) homogeneously distributed in the core.
D. Two regions, the inner with E= 1.8% and perturbed with non fuel rods, the outer region with E= 3.4%.
E. Two region, the inner with fuel rods resembling CAREM fuel elements, the outer with the needed fuel rods to reach enough reactivity to perform experiments with different configurations for the CAREM FE.

A detailed experimental program defines the experiments to be conducted with each core type. Some of the already defined measurements are:

Influence of different boron concentrations at different temperatures. Extensive properties as : critical height, critical buckling and reflector saving. Intensive properties as disadvantage factors, fission ratio (U235 & U 238), epithermal to thermal fissions ratio, epithermal to thermal absorption ratio, average spectra in fuel and moderator. Fluxes and spectra in non fuel rods (control rod guides); similar in macrocells (assembly of non fuel rods with neighbouring fuel rods). Power distribution. Mutual influence in CAREM Control Rods positioning Reactivity changes for different Boron concentrations, different temperatures and different void fractions. Determination of control rods reactivity and fuel rods with different concentrations of burnable poisons.



NOTAS

1) Cantidad total de agujeros diametro $9.5^{+0.3}_{+0.1}$: 3500 (tres mil quinientos) sobre ⌀850 .

4. Benchmarking

The calculation line in INVAP for the CAREM Project and the validations already done are presented.

Nuclear Data

ESIN library originated from the WIMS (1976) updated with data for Ag, In, Cd and Gd from ENDF/B-4 and Nb from WIMCAL-88 (COREA). The WIMS has been in used by INVAP with validated good results in Plate Type FE (for RR) of 20% enrichment and fresh cores. The Gd isotopes were tested using the CONDOR in two numerical benchmarks with good results; however the dispersion of results given by participants are quite wide.

Cell Code CONDOR 1.3 /9/

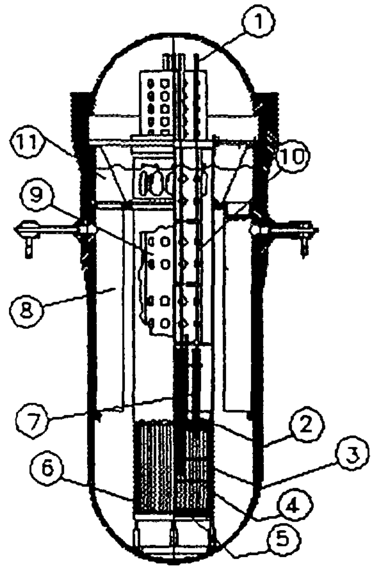
A preliminary validation with 91 typical PWR cells (fresh) was done /10/ and /11/, with a difference of 400±800 pcm Two more benchmarks /12/ and /13/ with mini FE of PWR using burnable poisons were done with satisfactory results Recently more comparisons (49 cases) were performed with results obtained for cores similar to that of the CAREM /14/ and /15/

Core Code CITVAP

The code was validated for plate type fuel elements of 20% enrichment with good results For 90% enriched FE the results are not as good

The calculations to validate the line are underway in order to reduce the experimental works with the RA-8

5. RPV Internals.



STRAIGHT PRESSSURE VESSEL

1 CONTROL DRIVE
2 CORE UPPER GRID
3 CORE PRESSURE SEAL
4 FUEL ELEMENTS
5 CORE LOWER GRID
6 CORE SUPPORT STRUCTURE
7 ABSORBING ELEMENT
8 VAPOUR GENERATOR
9 CONTROL DRIVE ROD STRUCTURE
10 CONTROL DRIVE ROD
11 POSSIBLE PRESSURE VESSEL SUPPORT ELEMENTS

Up to this stage of CAREM Project, several design aspects in the internals are recognised that need experimental verifications The aim of these experiments is to verify the behaviour under normal and abnormal conditions and to define the manufacturing and assembling allowances as well as handling procedures and auxiliary tools Following is a general description of the arrays under construction and the foreseen experiments for each of them

A dummy of a sector of the core containing the following items:

Core support, three FE, upper structures with control rod guides The experiments will be done with water at room temperature The aim is to make fine adjustments in the design and manufacturing and the influences of the different variables in the behaviour of the assembly Also to verify the design of couplings and auxiliary tools This stage will be started by the end of the present year

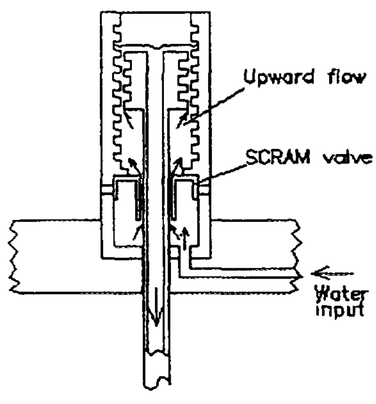
A 1:1 in length of a Sector of the Control Rod Drive Structure (for one Control Rod) with the Connecting Rod attached to the dummy core mentioned above and a Drive Mechanism.

The experiments will be carried out in air and in water at room temperature The objectives are to obtain the manufacturing and operational allowances

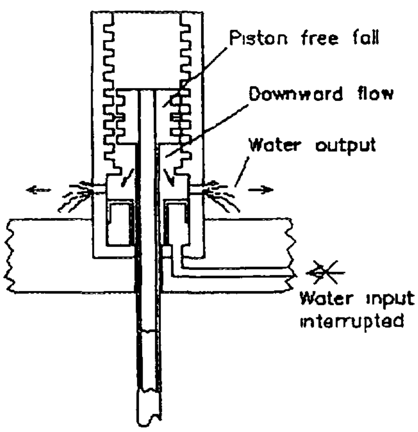
Examples of the experiments to be conducted during this stage are definitions related with alignment, clearances in linear bearings Dynamic Analysis to determine natural frequencies and mode shapes and responses of the system under various external excitations

6. Control Drive Mechanisms.

Reactor normal operation



Reactor SCRAM



The CAREM mechanisms are hydraulic type lodged inside the RPV The driving circuit of water provides a constant water flow (base) over which positive or negative pulses produce the movement of the connecting rod The development of the mechanisms involved different stages

Preliminary conceptual verification done to verify that theoretical approach and numerical results were in agreement with experimental results

"Cold Prototype' To determine experimentally the minimum base flow and its operational limits minimum flow to support the column and maximum without extracting the control rods Characteristics of pulse improvements in the SCRAM valve were part of this stage Also manufacturing hints in order to simplify the design, to improve reliability and if possible to reduce costs and design of auxiliary tools The objectives of these experiments were almost totally achieved At present the works are conducted in order to reduce the SCRAM time Once finished, a "final prototype will be manufactured to perform the characterisation The experiments are expected to be finished during 1995

"Warm " Experiment. T= 80 °C, atmospheric pressure. Characterisation of the mechanism and the driving water circuit at different temperatures. Study of abnormal situations: increase in drag forces; pump failure; Primary level influence; SCRAM valve failure; uncontrolled water flow and temperature; two phases water injection; suspended particles influence; air bubbles influences; drainage blockage.
"Hot" Experiment. A simple loop is under design to reach CAREM nominal operational values in normal and abnormal conditions. The objective are the characterisation of the mechanisms, durability tests, and behaviour of systems under abnormal conditions:breakage of feeding pipes; LOCA, behaviour under relief valves actuation.

7. Instrumentation and Control.

7.1. Drive Mechanisms Instrumentation

The position of the piston in the Drive Mechanism gives the position of the neutron absorber in the core. The length of the piston is 300 mm and the length of the cylindrical cover is 1850 mm. The method used to measure the piston position is called Magnetically Variable Inductance.

The piston made from magnetic Stainless Steel is moving inside the cylinder covered with an electrical coil with high concentration of rings in one end and decreasing to the other. All other pieces of the Drive Mechanism are non-magnetic. A fine measurement of the inductance gives a measurement of the piston position and consequently of absorber position. The resolution measured was better than 1%.

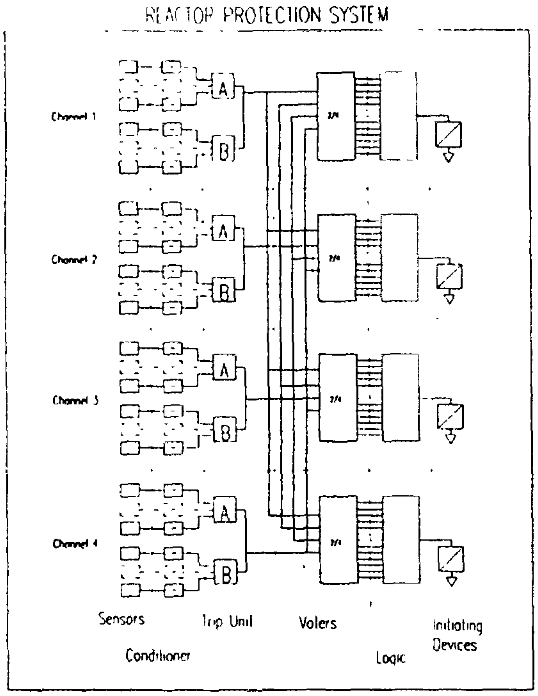
The test were made at room temperature and non immersed in water and the inductance was measured. Interference and very low frequencies are a challenging situation and due to that, lab instruments of high complexity and cost were used. The design of a specifically designed instrument to measure inductance was finished and a prototype to operate at CAREM conditions will be tested in the "warm" and "hot" test mentioned in Section 6. of this work.

7.2. *Reactor Protection System (RPS)*

Description

The RPS is based in solid state intelligent processing units and hard-wired multiplexed voting and protective logic units. It has four redundant, independent channels, whose main features are:

- High reliability and availability as a result of design criteria and technology
- Fault-tolerance with on line auto-verification routines and auto-announcing capability
- Compactness and robustness
- High simplicity



Current developments

The current developments are originated from the Safety Requirement Specifications of the RPS.

Trip Unit:

The trip unit performs the data acquisition of the safety variables and compares them against the Safety System Settings to initiate the protective actions in case of anticipated operational occurrences or accident conditions.The development of the Trip Unit is subdivided in the following stages:

- Software Requirements Specifications & prototype
- Software ***design***, code and implementation
- Hardware Requirements Specifications & prototype
- Hardware ***design*** and implementation
- Integration Requirements Specifications
- Integration Hardware/Software
- Validation
- Installation & Commissioning
- Operation & Maintenance

At present, the status of the development is at ***design stage*** in both hardware and software

Voting and Protective Logic Unit

The voting and protective logic unit performs the voting of the redundant safety trip signals coming from the Trip Units in a logic arrangement of 2 out of 4 and then, according to the logic relations of the trip signals and initiation criteria, triggers the protective actions.
The development of the Trip Unit is subdivided in the following stages:

- Hardware Requirements Specifications & prototype
- Hardware Requirements Specifications
- Hardware *design* and implementation

7.3. Supervision and Control System

Description

The highly automated digital Supervision & Control System, has an architecture of 5-level hierarchy with distributed processing and modern control technology It is conformed by different types of processing units

- Supervision Units (SU)
- Information Units (IU)
- Control Units (CU)
- Field Units (FU)

The Supervision & Control System is totally independent of the RPS High system reliability and availability are achieved by the use of redundancy and fault-tolerance in communications and processing unit

Operator interface is based on digital visual display units for safety, alarms, logics, processes and documentation presentation in the reactor main control room and others supervision and control centers Modern technologies as touch-screens, track-balls, custom keyboards, etc are used

Current Developments

The Supervision & Control System includes the development of a Control Operating System that implements all the low level functions as

Real-Time Data Base
Communication System
Control Functions
System Management
Historical Data Base
Man-Machine Interface
Signal Acquisition and Actuation

This Control Operating System acts as a software platform on which the Supervision & Control System application is built on The development process is divided in the following phases

Software Requirements Specifications & Prototype
Software Coding, Implementation & Integration
Instalation & Operation
Software Design Specifications
Validation & Verification

The ***Ward & Mellor Methodology*** is applied in every phase of the development process At present, the status of the development is at Coding phase

8. Fuel Elements

The activities in this subject are being carried out by CNEA itself At present the detailed engineering for the CAREM 25 Fuel Elements and absorbers are under execution

Development of equipment for components and FE manufacturing.

The following tasks have already been carried out

- Development and construction of equipment for caps welding by TIG method
- Development and construction of dies for stamping and cutting elastic spacers components
- Development and construction of FE assembly and final control boards
- Construction of different manufacturing and metrological control devices for FE manufacturing
- Prototype of elastic spacer for the FE
- Dummy FE to define handling tools

Current developments

The following tests are under definition stages

- Elastic spacers mechanical and stress tests
- Fuel element seismic behaviour test
- Thermalhydraulic behaviour in a low pressure loop
- Thermalhydraulic behaviour in a high pressure loop

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AN INTEGRAL REACTOR DESIGN CONCEPT FOR A NUCLEAR CO-GENERATION PLANT

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Abstract

An integral reactor concept for nuclear cogeneration plant is being developed at KAERI as an attempt to expand the peaceful utilization of well established commercial nuclear technology and related industrial infrastructure such as desalination technology in Korea. Advanced technologies such as intrinsic and passive safety features are implemented in establishing the design concepts to enhance the safety and performance. Research and development including laboratory-scale tests are concurrently underway to evaluate the characteristics of various passive safety concepts and provide the proper technical data for the conceptual design. This paper describes the preliminary safety and design concepts of the advanced integral reactor. Salient features of the design are hexagonal core geometry, once-through helical steam generator, self-pressurizer, and seismic resistant fine control CEDMs, passive residual heat removal system, steam injector driven passive containment cooling system.

1.0 INTRODUCTION

The drought experienced due to the climatic anomalies and the worsening level of pollution have reduced inland water resources significantly for a number of years. A nuclear co-generation plant which can be used for sea water desalination as well as electricity generation can provide a solution in some coastal countries such as Korea and middle east nations. In this regard, Korea Atomic Energy Research Institute (KAERI) has undertaken a study for the development of advanced integral reactor for the application to these purposes as an attempt to expand the peaceful utilization of nuclear energy.

Most of power reactors that are currently in operation and under development have loop type configurations which enable large-scale power output and thus provide economical power generation. On the other hand, integral reactors receive a wide and strong attention due to its inherent characteristics of enhancing the reactor safety and performance through the removal of pipes connecting major primary components. Small and medium reactors with integral configurations of major primary components are actively being developed in many countries. The design concepts of those reactors vary with the purposes of application.

KAERI has been putting efforts to research and develop new and elemental technologies for the implementation into the advanced reactors. In parallel with those efforts, an advanced integral PWR with implementation of those technologies as well as passive safety features is under conceptual development. The reactor power of 300 MWt is considered as suitable size for energy supply to the industrial complexes, remotely located islands, and especially isolated area. The reactor core is conceptually designed with no soluble boron and hexagonal fuel assemblies to enhance the opertional flexibility and to improve the fuel utilization. The reactor safety system primarily functions in a passive manner when required.

This paper describes the conceptual design features of the advanced integral reactor under development at KAERI, and also important R&D subjects concurrently in progress in order to prove and confirm the technical feasibility of the design concepts.

2.0 System Description

2.1 Reactor Core and Fuel Design

Table 1 gives the basic reactor parameters. The fuel design is based on existing KOFA (Korean Optimized Fuel Assembly) design technology. Most design parameters of fuel rods are the same as those of the KOFA except geometrical arrangement which is changed from the square array to hexagonal array. The hexagonal fuel assembly yields the lower moderator to fuel volume ratio(V_m/V_f) and the hardened neutron spectrum which results in stronger

TABLE 1 BASIC REACTOR PARAMETERS

MIRERO - PLANT DATA			
Design Lifetime	60 years	Reactor Type	PWR
Thermal Power	300 MWt	Plant Style	Integral Primary Circuit

Primary Circuit		**Pressurizer**	
Design Pressure	17 MPa	Type	Integral with RV
Operating Pressure	12.5 MPa		Self-Pressure Control
Coolant Flow	1.8×10^3 Kg/sec		
Core Inlet Temp.	285 °C	**Main Coolant Pump**	
Core Outlet Temp.	315 °C	Number	4
		Type	Glandless, Wet Winding
Reactor Core			
Moderator	Light Water		
Fuel	Low Enriched UO_2	**Containment**	
Fuel Assembly	Hexagonal	Type	Passive, Steam Injector Driven
Reactivity Control	Fuel Loading, Burnable Poison, Control Element Assemblies, No Soluble Boron	**Safety Systems**	
		Decay Heat Removal	Passive, Natural Convection - Safe Guard Vessel, Heat Pipe, Hydraulic Valve
Clad Material	Zircaloy-4		
Power Density	66.7 KW/liter		
Avg. Linear Heat Generation Rate	8.4 Kw/m		
Refuel Cycle	24 months	Emergency Core Cooling	Not Necessary
Active Core Height	1.8 m		
Active Core Diameter	2.0 m		
Steam Generator			
Type	Helical - Once Through		
Steam Temp.	290 °C		
Steam Pressure	4.7 MPa		
Superheat	30 °C		
Feedwater Temp.	240 °C		
Feedwater Flow	174 Kg/sec		
Tube Material	I690 T/T		

moderator temperature coefficients and higher plutonium conversion ratio. The effective fuel rod length is reduced to 180cm. Fuel utilizes low enrichment, uranium dioxide, which is operated at a low specific power density(17 kW/kgUO_2). The uranium enrichment of the fuel is selected to achieve the 18 months(or longer) operating cycle. The fuel assembly section is a 22.9cm hexagon and the geometry is provided to accommodate control element assembly in each fuel assembly. The fuel assembly consists of 360 fuel rods and 36 guide tubes for control absorbers and/or insertable burnable absorbers and 1 guide tube for central in-core instrument.

The core is rated at 300MWt and consists of 55 fuel assemblies. The average linear heat generation rate is 8.4kW/m which is much lower than that of conventional PWRs. The low power density and increased thermal margins with regard to critical heat flux ensure the core thermal reliability under normal operation and accident conditions. The core is designed to operate without the need for reactivity control using soluble boron over the whole power range. The elimination of soluble boron from the primary coolant is a major potential simplification for the advanced light water reactor. From the point of the view of the reactor control and safety, soluble boron free operation offers potential benefits through the presence of a strong negative moderator temperature coefficient over the entire fuel cycle and therefore improves reactor transients and load follow performance. Control rods provide the means of core reactivity control except for long term reactivity compensation for fuel depletion provided by the burnable poison and have enough shut down margin at any time under cold clean conditions including refueling conditions.

2.2 Primary Circuit

Internal Arrangements and Flow Paths

Figure 2 shows the general arrangement of the reactor pressure vessel and its internal structures. The core is low down in the vessel. Helically coiled once through steam generator is located between the core support barrel(CSB) and reactor vessel. Thermal shields are provided below the steam generator surrounding the core to reduce the neutron fluence level on the reactor vessel. There are four main coolant pumps installed on the reactor vessel above the steam generators. The upper plenum of the vessel forms a pressurizer to maintain the operating pressure.

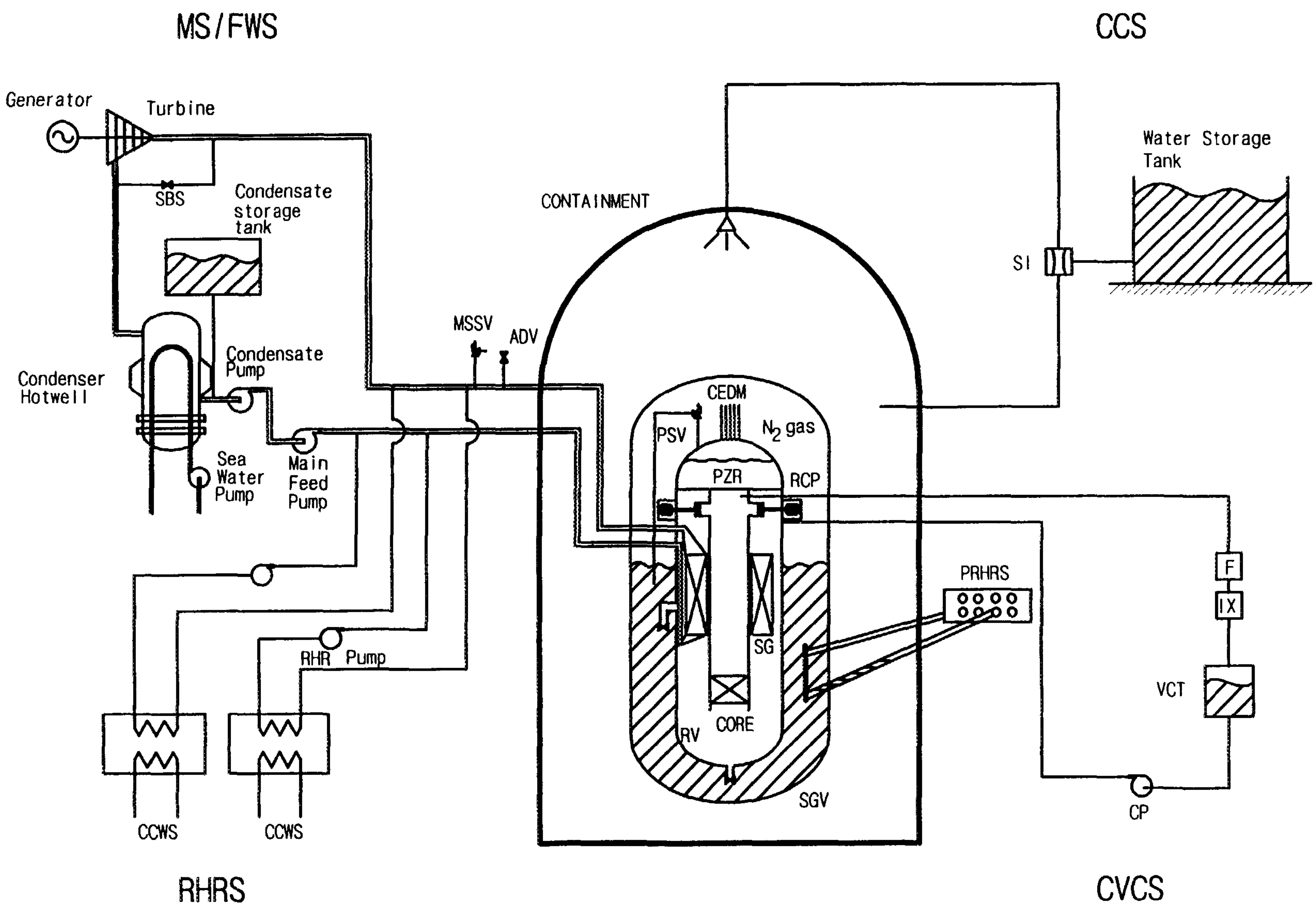

Figure 1. Schematic Diagram of Advanced Integral Reactor Systems

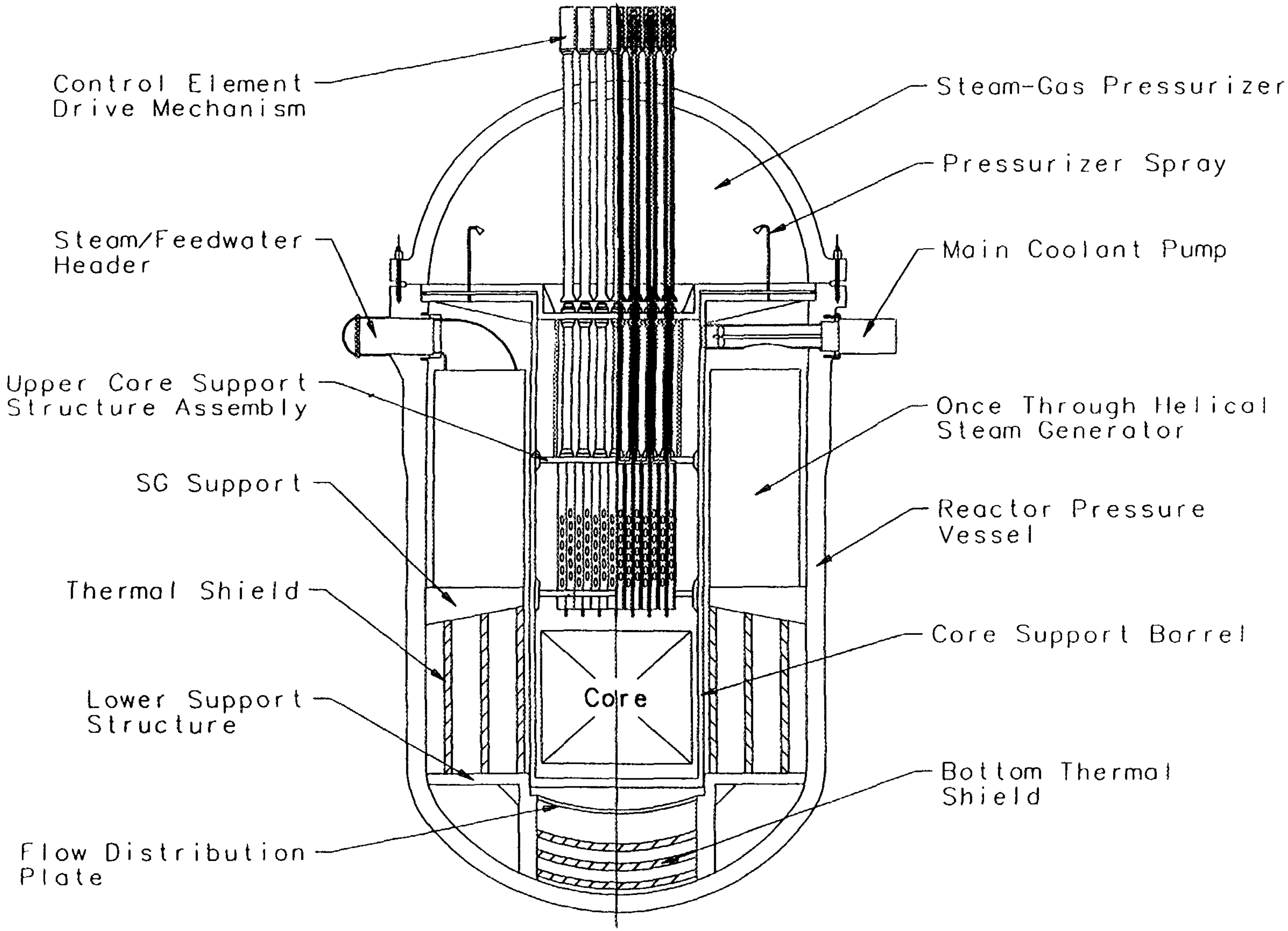

Figure 2. General Arrangement of MIRERO

The primary coolant flows up through the core and CSB riser, through the pumps, down through the steam generator and back to the lower plenum under the core. The current design is capable of 50% of full power operation only with natural circulation.

Steam Generators

The once-through steam generator is located within the reactor vessel in the annular space between the core support barrel and the reactor vessel inner wall. The secondary coolant is completely evaporated in a single pass through the steam generator. The steam generator tubes are divided into two groups that can be operated independently and the tubes in each group will be cross wound to reduce thermal twisting.

The steam generator consists of tube bundle, downcommer, two feed water and steam headers, shrouds to guide primary flow, and tube supporting structures. Approximately 1100 tubes are helically coiled in the effective heat transfer region and the effective coiling height is 3m. To assure equal steam quality from individual tubes, the length of tubes in the effective tube bundle is maintained as nearly equal as possible by varying the number of tube starts and helix angles in each tube column. The tube material is inconel 690 with 19 mm outer diameter through the whole tube length. The tube bundle is supported by eight perforated radial support plates, which transfer the load to the bottom support structure located on the supporting lug. Each tube can be accessed easily through the feed water and steam headers which are attached to the reactor vessel for in-service inspection and maintenance.

Pressurizer

The upper part of the RPV is filled with the mixture of nitrogen gas and steam providing a surface in the primary circuit where liquid and vapor are maintained in equilibrium under saturated condition. The pressure of the primary system is equal to the nitrogen partial pressure plus the saturated steam pressure corresponding to the core outlet temperature. Thus the reactor operates at its own operating pressure matched with the system status.

The nitrogen gas partial pressure of 2 Mpa is chosen to maintain subcooling at the core exit in order to avoid boiling in the hot channel during transients. The volume of gas space is large enough to prevent safety valves from opening during most severe design basis transient.

Reactor Coolant Pumps

The pumps are sealed type (i.e. glandless) canned motor pumps with added inertia to increase pump rundown time. With no shaft seals the small LOCA associated with seal failure in standard commercial designs is eliminated. With the primary water level lowered, they can be removed radially for servicing or replacement without having to remove the vessel closure head.

Control Element Drive Mechanism (CEDM)

The requirements for the CEDM are fine position control, lubrication by primary coolant, easy access to electrical parts, and high seismic resistivity. Soluble boron free reactivity control and load following operation over its full power range require a fine positioning capability of the control rod. The investigation of the failure frequency of operating CEDMs has revealed that the major source of trouble was in the electrical components requiring an easy access for maintenance. The long protruded part for the extension shaft stroke outside the reactor pressure vessel leads to high level of seismic excitation and reduces the margin to design stress limit. The current magnetic jack type CEDM used in the Korean Standard PWR is considered to be inadequate to meet the fine control requirements because it has only a step wise positioning. In addition the adoption of a self-pressurizer which occupies the upper plenum of the reactor vessel introduces difficulties in lubricating the moving part with the primary coolant since the latch mechanism would be located in the steam-gas region of the self pressurizer. Therefore, new concept of CEDM was proposed. The proposed design consists of position encoder, brushless DC servo motor, lift magnet coil, rare earth permanent magnet rotor, driving tube and split ball nut assembly. The rotor, driving tube and ball nut assembly are all connected into a single piece and lodged within the pressure housing which forms the pressure boundary. The encoder, DC servo motor, and lift magnetic coil are installed outside the pressure housing for easy maintenance. The use of a brushless DC servo motor with rare earth permanent magnet rotor allows a maintenance free operation of the motor. The fine control capability of the

CEDM is assured by the use of ball nut - lead screw mechanism and its lubrication with primary coolant is provided by placing this part below the water steam interface surface in the pressurizer. The ball nut assembly is of three pieces split type. Lift magnet located below the DC servo motor engages the ball nut to the lead screw by lifting the driving tube and the rotation of the rotor induces linear motion of the control element assembly up and down. The lead screw is part of extension shaft at the bottom of which a control element assembly is attached. When the scram signal is issued, the current supply to the lift magnet coil is cut off and the split ball nut releases the lead screw while dropping down by gravity and spring forces.

2.3 Engineered Safety Features

The safety concept is taking advantage from the intrinsic safety characteristics of integral reactor and pursuing passive safety principles common to most small and medium reactors. The fundamental safety characteristics are:

- Low ratio of power density to heat capacity resulting in a slow rise of fuel element temperature under accident conditions;
- A substantially negative moderator temperature coefficient resulting from no soluble boron usage generates beneficial effects on self-stabilization and limitation of reactor power;
- Integral Reactor Vessel eliminates large primary coolant pipes and thus large break loss of coolant accidents;
- Large Passive pressurizer significantly reduces pressure increase for decreased heat removal events;
- Large volume of primary coolant provides more thermal inertia and makes plant more forgiving;
- No RCP seals eliminates potential for seal failures, a concern during station blackout;
- The use of passive safety systems leads to directly to simplification in design since it eliminates the need for multiple redundant safety systems with their redundant safety grade power supplies.

2.3.1 Reactor Shut-Down System

The reactor trip in emergency is done by simultaneous insertion of the control rods into the core by gravity following the drive mechanism de-energization, which is actuated by trip signals from the automatic control system. In the case of failure to actuate the electromechanical protection system, the reactor shutdown is accomplished by the emergency boron injection system. Activation of the system is done by manually opening valves in the pipelines connecting the system to the reactor. Both shutdown systems ensures the reactor shutdown and its shutdown margin is sufficient enough to keep the cold clean reactor in a subcritical state.

2.3.2 Residual Heat Removal System

Normal decay heat removal when cooling down for maintenance and refueling, the steam generators with turbine bypass system are used and heat is rejected through the condensers. This can be achieved by natural circulation on the primary side but requires feed pumps and other equipments on the secondary system. If the secondary system is not available, active decay heat removal system with steam generators are used to remove decay heat and heat is rejected through the component cooling system.

Should there be no ac power available, decay heat is removed by natural convection system which only requires battery power to operate the initiation valves and passive decay heat removal system which is composed of heat pipe and heat exchangers. The heat is ultimately rejected to atmosphere by natural convection of air flow. Thus, there is theoretically infinite time of heat removal without operator intervention. One of the advantages of heat pipe passive decay heat removal system is that this system is continuously operating during normal plant operation to remove the heat loss from reactor vessel through the wet thermal insulation.

2.3.3 Emergency Core Cooling System (ECCS)

The integrated primary system concept eliminates all large primary circuit pipe work, thus intrinsically eliminates large loss of coolant accidents. The largest pipe break in the primary circuit is the break of the connection pipe supplying chemical and volume control system(CVCS). To prevent siphoning off the reactor water inventory in the hypothetical event of a CVCS line break, open connections are made between the steam generators and pressurizer. Thus there is no possibility of rapid emptying of the reactor vessel requiring massive and early injection of ECCS water. Since reactor vessel is flooded all the time by the water in the safe guard vessel, there is no need for external emergency core make up. The safe guard vessel is sized to provide a minimum of 72 hours' heat removal without operator intervention.

2.3.4 Containment Overpressure Protection System

Since the maximum pipe break is small due to the integral nature, the pressurization rate of containment is slow. Energy released to the containment through the break point is removed using the steam injector driven containment spray system to prevent exceeding the containment design pressure. The steam injector is a simple, compact passive pump that is driven by supersonic steam jet condensation. The steam injector can operate even by atmospheric pressure steam. The steam from break point is supplied to the steam injector. The steam injector pumps up the water from a water storage tank on the ground to the spray nozzle located at the top of the containment.

3.0 Research and Development(R&D) Activities

To evaluate the characteristics of various passive safety concepts and provide the proper technical data for the conceptual design of the advanced integral reactor, the following R&D activities are being performed.

<u>Hexagonal Semi-Tight Lattice Fuel Assembly</u>

Numerical Analysis Technology is being development to analyze the hexagonal fuel assembly core with tight lattice. Some thermal-hydraulic experiments will be performed to understand the phenomena in the semi tight hexagonal lattice. The suitable thermal-hydraulic analytic model will be developed, especially thermal-hydraulic correlations, which are vital for the semi-tight hexagonal geometry. The developed model is incorporated into computer codes for the design and safety analysis.

<u>No Boron Core Concept</u>

The use of no soluble boron in the core design causes to utilize large amount of burnable absorbers to properly hold down the excess reactivity at the beginning of cycle and to install considerable number of control rods for the reactor control and operation. The optimization in the number of burnable absorbers and control rods is required with respect to the reactivity compensation with fuel burnup and reactor control through the cycle, and this study in conjuntion with the core design with hexagonal fuel assemblies are thus investigated in this R&D subject.

Natural Circulation Phenomena for Integral Reactor

To investigate thermo-hydraulic characteristics of primary circuit in natural circulation operation mode, an experimental test loop is being designed. An computer code is being developed to model the thermo-hydraulic behavior of the primary circuit.

Hydraulic Valve Application for PRHRS

To investigate the possibility of passive initiation of the isolation valve located inside Safe Guard Vessel, an experimental study is being performed.

Heat Pipe Application for PRHRS

A separate type heat pipe is experimentally studied for PRHRS. A computer code is developed to model the thermo-hydraulic behavior of the heat pipe.

Helically Coiled Tube Once Through Steam Generator

The thermal hydraulic design and performance analysis computer code, ONCESG, for a once through steam generator is developed. An experimental study is being performed to generate the heat transfer correlation and pressure drop correlation of the helically coiled tube once through steam generator.

Critical Heat Flux Test

An experimental test facility is being constructed to study critical heat flux and pressure drop correlation for the tight latticed hexagonal fuel assembly.

Steam Injector Application for PCCS

An experimental study is being conducted on a steam injector driven passive containment cooling system. A computer code is developed to model the thermo-hydraulic behavior of the steam injector.

Wet Thermal Insulation

An experimental investigation is being conducted for material selection and performance test for the wet thermal insulation system.

Fluidic Diode Application for Passive Pressurizer Spray System

An experimental study is being conducted on the fluidic diode device for passive pressurizer spray system. A computer code is developed to model the thermo-hydraulic behavior of the fluidic diode.

4.0 Summary

An advanced integral reactor is currently under conceptual development at KAERI. Main features of the reactor are summarized as follows;

Integrated Primary System

This feature implies that all primary components - core, steam generators, pumps, pressurizer - are contained in a single pressure vessel. This arrangement allows the complete removal of all primary circuit pipework, thereby the elimination of the large break LOCA.

Enhanced Core Design

The core is designed to operate without the need for reactivity control using soluble boron during the whole power range. The core is designed to enhance reactor operating safety margin by specifying a very low core power density and linear heat generation rate. The core has long refueling cycle(18 months or longer) to improve plant availability.

Self-Pressurized Primary System

The free volume above the reactor water level is used as steam/gas pressurizer. The operating pressure is determined by initial gas pressure and the steam pressure corresponding to the core exit temperature. Thus the reactor operates at its own operating pressure matched with the system status.

Enhanced Primary Circuit Natural Circulation Capability

The primary circuit natural circulation capability is enhanced by the very low head loss of the integrated primary system. The difference in temperature between cold and hot water together with the difference in height between steam generator and the core produces driving force to circulate primary coolant. The required pump capacity to circulate primary coolant is reduced by proper selection of these parameters.

Passive Engineered Safety Features

The design provides for all the required safety functions following plant transient or accidents through passive safety features. The residual heat removal, inventory control for the reactor coolant system, emergency core cooling, and containment heat removal are all passive and integrated within the containment system.

AN INTEGRAL DESIGN OF NHR-200

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Abstract

Nuclear heating application has received a wide attention in China due to the favourable economic and environmental aspects. The Nuclear Heating Plant NHR-200 is seen to provide the required energy for district heating, industrial processes and seawater desalination for many sites in China and possibly abroad. The paper summarizes the technical description of the plant and give its main characteristics related to the integral design approach.

1. The development of NHR in China

Because the energy consumption as sensible heat at temperature less than 150°C accounts for about 25% of the total energy consumption in China, therefore, in order to mitigate the energy shortage, environmental pollution caused by coal burning and overburden on coal transportation, great attention has been given to the R& D of the nuclear heating reactor (NHR), which has been as one of the national key projects in science & technology in China since 1980s The NHR could be used in the district heating, air conditioning, sea water desalination and other industrial processes Therefore the NHR could substitute the nuclear energy for fossil fuel and change the energy composition in China This will be of significance in socio-economic development and environmental protection

Research work on possible application of nuclear heat was initiated in early eighties During 1983-1984, the INET used its existing pool type test reactor to provide space heat for the nearby buildings Meanwhile, two types of NHR, i e deep pool type NHR and vessel type NHR, have been developed by INET Based on the heating grid conditions in China and the comparison among various design concepts of the NHR, the vessel type NHR has been selected as a main development direction As a result, construction of a 5MWt experimental NHR (NHR-5) started in 1986 at INET The reactor was completed in 1989 and has been operated successfully for space heating since then In the meantime, a number of experiments have been carried out to demonstrate the operating and safety features of NHR

In China it has been decided to construct a 200MWt NHR demonstration plant to realize the NHR commercialization The feasibility study of the first DaQing demonstration plant has been completed and approved by the respective authorities

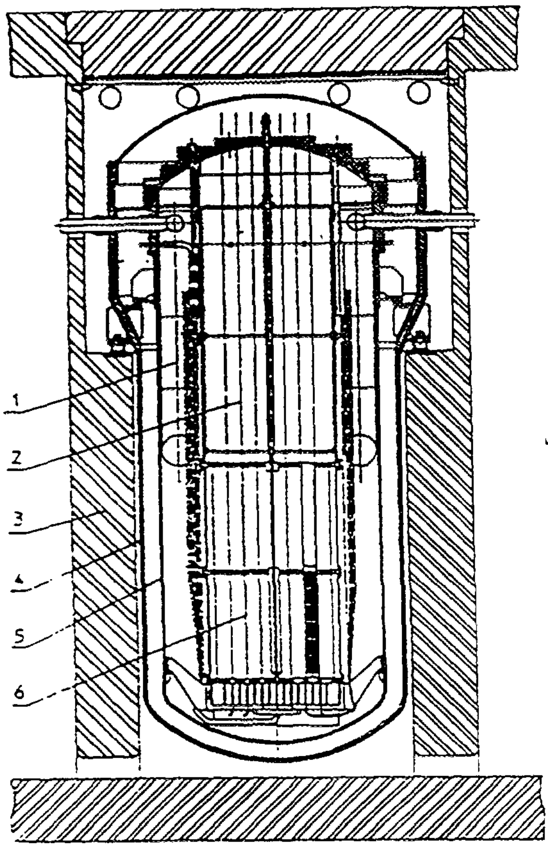
It is worth to mention that during the last few years the INET has closely cooperated with Siemens-KWU, German and the former EIR, Switzerland on the R&D of the NHR, and recently the INET also has information and personnel exchange on the matter with the respective institutions in Russia and other countries

2. Technical description of the NHR-200.

For a nuclear district heating reactor it has to be located near the user due to the medium of heat transmission (hot water or low pressure steam). It means a NHR is surrounded by a populous area Using emergency actions as an essential element in the ultimate protection of the public can thus become impractical Therefore, in all credible accidents the radioactive release from heating reactor has to be reduced to such low levels that off-site emergency actions, including sheltering, evacuation, relocation and field decontamination will not be necessary In the other hand there is a serious challenge to the economy for a NHR The capacity of a NHR can not be as big as that of NPP due to the limitation of heat transmission

Moreover the load factor is also much lower than that of NPP. It is obvious that to meet the safety requirement and lower the capital investment are the major concern in the design of a NHR. The only solution is to have a design with inherent safety characteristics and passive safety as much as possible instead of the complex engineering safety features.

The reactor structure and core cross section of NHR-200 are shown in Fig.1 and Fig.2 respectively. The simplified schematic diagram is given in Fig.3. The main design data of NHR-200 are listed in Table 1.



1. primary heat exchanger
2. riser
3. biological shield
4. containment
5. pressure vessel
6. core

Fig. 1. The NHR-200 reactor.

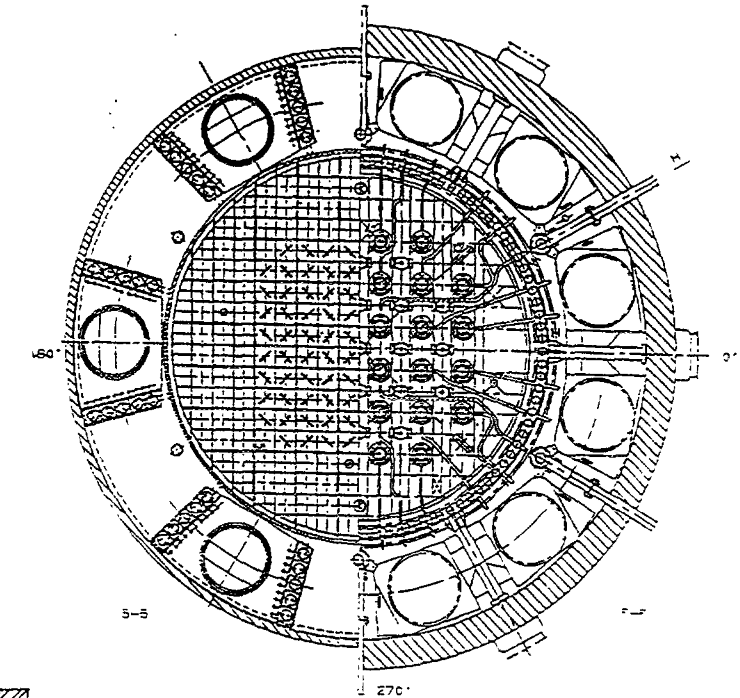


Fig. 2. Core arrangement.

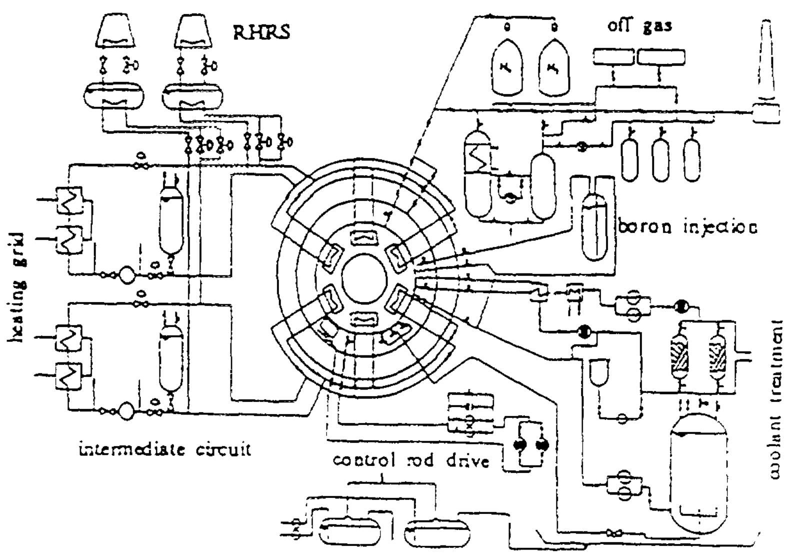


Fig. 3 Schematic system diagram

Table 1 Main design parameters of NHR-200

Name	Unit	Value
Rated thermal power	MW	200
Pressure of the primary coolant circuit	MPa	2.5
Core inlet/outlet temperature	°C	145/210
Core coolant flow rate	t/h	2376
Intermediate circuit pressure	MPa	3.0
Intermediate circuit temperature	°C	95/145
Intermediate circuit flow rate	t/h	3400
Heat grid temperature	°C	130/80
Fuel assembly type		12×12-2
Fuel assembly number (initial core)		96
Enrichment of fuel (initial core)	%	1 8/2 4/3 0
Average core power density	kW/l	36 23
Average fuel linear power density	W/cm	77

The major features of the NHR-200 design are

(1) Integrated arrangement, self-pressurized performance and dual vessel structure

(2) Natural circulation for primary loop

(3) Passive safety systems including Residual Heat Removal System and Boron Injection System

(4) Low operating parameters with large safety margin including temperature, pressure and power density

With these features, the probability of pressure boundary break is much lower than that of conventional NPP The mitigation of a LOCA is much easier and there is no ECCS in the NHR The supporting systems such as on-site diesel generators, component cooling system, service water system, instrument air system and ventilation system etc do not provide with safety functions, namely they are non safety-related in the NHR design While in a NPP these systems are safety-related and sometimes loss of them are dominant initiators resulting in core damage

3. The characteristic related to integral design

3 1 Integrated arrangement

As showing in Fig 1 the reactor core is located at the bottom of the reactor pressure vessel (RPV) 6 primary heat exchangers divided into two groups are arranged on the periphery in upper part of the RPV The system pressure is maintained by N_2 and steam All penetrations of RPV are therefore of small size (the largest one sized D50) and located at the upper part of RPV No pipe goes inside and down to the lower part of the RPV so that should a pipe break occurs outside the RPV the water ejection will last not so long, then turn to steam ejection This arrangement reduces the amount of losing inventory to a great extent Meanwhile a guard vessel fits tightly around the RPV so that the core will not become uncovered under any break of primary pressure boundary even at the bottom of RPV This vessel can also function as a containment

The reactor vessel is cylindrical, with a weld hemispherical bottom head and a removable, flanged and gasketed hemispherical upper head The RPV is 4 8m in diameter, 14m in height and 197 Tons in weight The cylindrical portion of the vessel is welded by 65 mm steel plate then lined with 6 mm stainless steel layer The RPV will be manufactured and fabricated in factory then shipped to the site The guard vessel consists of a cylindrical portion with a diameter of 5m and an upper cone portion with maximum 7m in diameter The guard vessel is 15 1m in height and 223 Tons in weight The upper and lower portions are fabricated in factory then shipped to the site separately, then welded together on site

3 2 Natural circulation

The reactor core is cooled by natural circulation in the range from full power operation to residual heat removal The hydraulic resistance along the primary circuit is dominant by the primary heat exchangers The "U" type tube bundles are adopted for PHE_S in order to give facilities for repairing The pitch of fuel elements is chosen to 13mm, a little tight than usual The flow resistance does not increase too much, but it is good for negative reactivity feedback There is a long riser on the core outlet to enhance the natural circulation capacity The high of the riser is about 6m Hence the average coolant velocity in core is 0 57m/s about 2 times larger than that in Although the core power density increases comparing with NHR-5 the MDNBR is still larger than the limitation with a sufficient margin

A number of measurements and experiments have been carried out in NHR-5 and demonstrated that the capacity of natural circulation is sufficient to carry out the heat in the power range from full power operation down to residual heat removal Even in case of interruption of natural circulation in the primary circuit due to LOCA the residual heat of the core can be transmitted by vapor condensed at the uncovered tube surface of the primary heat exchangers [1]

3 3 Self-pressurized performance

At the upper part of RPV there is a space filled with the mixture of N_2 and steam providing a surface in the primary circuit where liquid and vapor can be maintained in equilibrium under saturated condition for pressure control purpose during steady state operations and during transients The pressure of the primary system is equal to the N_2 partial pressure plus the saturated steam pressure corresponding to the core outlet temperature A hydraulic instability way take place in some conditions in steam-water two phase flow under lower pressure and in natural circulating loop A N_2 partial pressure of 0 6Mpa is chosen to maintain sub-cooling in the core outlet with large margin in order to avoid two-phase flow even in the hot channel during normal operation and transient The volume of upper space is large enough to prevent the safety valve from opening during most of the transients, such as loss of off-site power In the present design the safety valve is only opened in case of ATWS induced by loss of main heat sink The operational experiences gained from NHR-5 shows that the N_2 as a blanket gas does not cause a problem with water chemistry The concentrate of nitrate and nitrite in the coolant is around 5 ppb

3 4 Spent fuel storage

There are 96 fuel assemblies in the active core The expected region discharge burn-up is 30000 Mwd/T One fourth of the fuel assemblies are scheduled to be refueled after each 5 winter seasons operation Therefore, the total discharged spent fuel assemblies at the end of lifetime of NHR-200 are only about the amount of another two cores of fuel assemblies It makes possible to storage the whole spent fuel inside the RPV There are in total 200 cells divided into two layers for spent fuel assembly storage surround the active core A stand-by small pit for defective fuel assemblies is reserved in the reactor building This arrangement makes refueling and spent fuel storage equipment and facilities quite simple In many countries the concept of spent fuel storage inside RPV is under developing to increase the burn-up of fuel For NHR-200 this arrangement also could increase the burn-up about 15%

4. The accident analysis

The accident analysis shows that the design features of NHR-200 provides an excellent safety characteristics not only with a low probability of initiators, but also with a unserious consequences of accidents. The results of LOCA analysis of NHR-200 are listed in Table 2[2].

Table 2. Results of Loss of coolant accidents analysis for NHR-200

	D50 pipe break inside guard vessel	D50 pipe break outside guard vessel followed by failure of isolation	Safety valve stuck open	Small crack at the bottom of RV	ATWS initiated by loss of off-site power followed by safety valve stuck open
Accidental lasting time (sec)	~1000	~72000	~11000	~10000	~3000
Ultimate pressure in the PRV (MPa)	~1.65	0.12	~0.6	~0.9	~2.37
Total loss of inventory (Ton)	~14	25.7	~10.1	18.2	6.8
Water amount remained above the core (Ton)	124	112.3	128	~120	131.2

5. Conclusion

In light of the requirement on safety and economy a NHR should be designed with inherent safe features instead of complex safety systems. An integrated arrangement is one of the key characteristics The experiences gained from NHR-5 four winter seasons operation and the practice of NHR-200 design show that an integrated arrangement with self-pressurized performance, natural circulation and in-vessel spent fuel storage is realistic and has many advantages.

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AN EVOLUTIONARY APPROACH TO ADVANCED WATER COOLED REACTORS

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Abstract

Based on the result of the Feasibility Study undertaken since 1991, Indonesia may enter in the new nuclear era by introduction of several Nuclear Power Plants in our energy supply system. Requirements for the future NPP's are developed in two step approach. First step is for the immediate future that is the next 50 years where the system will be dominated by A-LWR's/A-PHWR's and the second step is for the time period beyond 50 years in which new reactor systems may start to dominate. The integral reactor concept provides a revolutionary improvements in terms of conceptual and safety. However, it creates a new set of complexe machinery and operational problems of its own. The paper concerns with a brief description of nuclear technology status in Indonesia and a qualitative asssessment of integral reactor concept.

INTRODUCTION

In Indonesia, a developing country with low per capita gross national product, energy consumption since 1970 has been continually increasing in support of the development in all sectors. In case of electricity, the rate of increase of electricity consumption during the last two years has been more than 15% per year. This poses serious challenges to both natural and financial resources of the country. The fossil energy is not unlimited. Hence, it is necessary to diversify our energy resources. The integration of Nuclear Power Plants (NPPs) in energy supply development has been considered based on the national energy policy which is stipulates among others : diversification, environmental concerns and conservation in support of sustainable development.

A comprehensive and in-depth Feasibility Study has been under taken since November 1991 to give strong justification to Nuclear Power Plants introduction in our electricity system. This study covers both techno-economic and safety aspects as well as site and environmental aspects.

Based on the Feasibility Study Report, also called Non-Site Study Report, which is submitted on December 30, 1993, coal fired plants will dominate the electricity generation system. The optimization shows that the first NPP is feasible in the year 2004. The contribution of NPP will increase in accordance with the demand.

Regarding the reactor nuclear technology and safety requirements, two steps approach has been considered : for the next 50 years period and beyond 50 years. In the first period, the NPPs will be dominated by present proven system with some evolutionary improvements in technology and safety. Beyond 50 years, new evolutionary and revolutionary type of reactors may emerge, with very special characteristics in terms of conceptual, safety, fuel and performances aspects. In this period, the integrated type of reactor, which is a complete departure from current LWR design, might have a great chance to enter in the market.

STATUS OF NUCLEAR TECHNOLOGY IN INDONESIA : FEASIBILITY STUDY FOR FIRST NPP

In September 1989 the Indonesian government decided to perform a new the NPP's feasibility study including a comprehensive investigation of the Muria site. The study itself should be carried out by the National Atomic Energy Agency / BATAN under the directives of the Energy Committee (PTE) of the Department of Mines and Energy, and with the cooperation of other institution such as State Electricity Enterprise (PLN).

On August 23, 1991, an agreement was signed in Jakarta between the Indonesian Ministry of Finance and BATAN on behalf of Indonesian government, and the consultancy company NEWJEC Inc. (Japan). This agreement contracts NEWJEC Inc. for a four and a half year period to perform a site selection and evaluation, as well as a comprehensive NPP's Feasibility Study.

The scope of the feasibility study includes two main components :

- The non-site studies, covering energy demand and supply , energy economy and financing, technology and safety aspects, the fuel cycle and waste management, and general management aspects, among other things.
- Site and environmental studies, covering field investigation and assessment of candidate sites, site qualification and evaluation, and environmental, socio-economic and socio-cultural impacts.

On December 30, 1993, the Feasibility Study Report (FSR), also called Non-site Study Report (FSR), was submitted. A final report including a site and environmental report, and a preliminary safety analysis report will be provided at the end of the four and a half year contract. These documents will provide the information necessary for site permit application, for the design engineering basis and other project and industrial infrastructure preparations.

Safety aspects are of utmost concern in the FSR, which cover and assess not only the proven designs available in the market at present, but also advanced systems expected to enter the market in the near future.

The main results of the FSR are among others Primary Energy Supply Scenario shown on Table 1.

Table 1. Energy Supply Scenario Share of Primary Energy Supply (%)

PRIMARY ENERGY	1990	2000	2010	2019
Oil	60.21	60.90	51.14	34.34
Gas	32.52	18.60	7.01	3.41
Coal	5.72	18.21	35.55	54.29
Nuclear	0	0	3.92	6.18
Others (hydro, geothermal)	1.55	2.40	2.38	1.79

Coal fired plants will dominate the electricity generation system. Nuclear Power Plants will increase in accordance with the demand. The data from this scope of work are used for optimization studies in the development of the Java-Bali electric system. The optimization shows that the first NPP's operation is feasible in the year 2004. In the year 2019 the share of NPPs will give a contribution of 10% of the electricity supply, an amount equal to about 12,600 MW.

APPROACH TO FUTURE REACTOR

The approach for nuclear reactor technology and safety is developed in two steps. The first step is the next 50 years where the system will be dominated by the present proven system notably LWR's and to a lesser extent PHWR's with some evolutionary improvements in technology and safety of the system. The main reasons for this is : the present NPP's are **quite competitive** with respect to coal plant, they are **cleaner** than fossil plants and they have **good operational and safety performance.**

The evolutionary improvements in the safety and technology must be performed to reach better performance, better economics, better protection for the investors and wider acceptance by the public.

The so-called "advanced reactor", such as ABWR or AP-600, offer improved safety systems features by simplification and introduction of passive safety features. In other point of view, the improvement of man-machine interface in the system is also a good point for these advanced reactors in order to resolve the complexity of the machine and make the operator's life easier.

In longer term future, despite efforts in conservation and efficiency programs in energy production and use, the tendency of energy demand will keep increasing well into the long term future. This is due to increasing of living standard and the population growth especially in developing countries.

Nuclear energy can have a very important role in energy supply system, if it succeeds in its development and rises to challenges it is facing. If not, the role of nuclear energy will be replaced by coal or gaz.

The challenges that should be resolved are :

- possible weapons proliferation and diversion of nuclear materials
- sabotage against nuclear facilities
- safety concerns
- radioactive wastes and environmental issues
- standardization of future plants

From the challenges mentioned above, the possible requirements for next generation of NPP's would be in essence based on revolutionary concepts which are among others :

- design NPP's that provides inherent resistance to sabotage and man-induced events through physical, security, and institutional design and arrangement
- concerns over safety should be eliminated through non catastrophe reactor and fuel design (no core melt accident)
- standardization of future NPP's is of great development in order to reduce and keep the development cost at reasonable level, on the other hand this will improve the economy of the NPP's

In this context, the **integral reactor concepts** provide a remarkable inherent safety features.

QUALITATIVE ASSESSMENT OF INTEGRATED REACTOR CONCEPTS

Inherent Safety Concepts

After Three Mille Island-2 and Chernobyl-4 accidents within the nuclear reactor society in the world there are groups who are developing ideas of a new generation of reactors which is characterized by a much higher standard of safety incorporating forgiving reactor system and inherent safety design. This set of inherent characteristics by definition shall tolerate any mistake occured during operation. It can mean that in an accident condition the inherent characteristics could allow enough time for the operator to correct the mistakes, could respond to neutralize the hazard, and could be left safely without any human interference.

At present it has been recognized that inherent safety characteristics are the dominant advantage of High Temperature Gaz Cooled Reactors which incorporates multicoated ceramic particle fuel, inert helium coolant, high thermal capacity and good convective heat transfer whereas, PIUS design concept are manifesting the solution for light water reactor versions.

It is generaly accepted among the nuclear reactor designers that the technology of "next generation" reactors must provide a guarantee that the core degradation accident and consequences risk of serious radiation releases will not occur. A core degradation accident can be defined as one where there is widespread break of the cladding and large scale release of fission product inventory to the coolant. The upper end of this accident is of course a complete core melt.

Many ways can be undertaken to prevent a core degradation accident. The following two conditions must be fulfilled :

- keep the core submerged in water at all time to maintain integrity of fuel
- make sure that heat production does not exceed cooling capability of water

This two conditions could be achieved by placing the core within the neutron poisonned coolant pool. The last will submerges the core when needed by passive nature. Core cooling is effected by evaporation this water.

In the other point of view, in order to eliminate one the most important initiating event of the core degradation accident, i.e. loss of primary coolant accident, the integration of Steam Generators and pressurizer in the reactor pressure vessel has been considered. In such reactor design, due to the absence of large diameter external piping associated to primary system, no large break LOCA has to be handled by the safety system.

Moreover, the simplification design is also considered in some integral type reactors by eliminating control rods. The power output controlling is replaced by adjusting the flow rate and the concentration of boric acid solution. In this case, no reactivity initiated accident should be considered any more.

This new design, which is a complete departure from current Water Cooled Reactor plants, exhibits a remarkable inherent safety characteristics for all accident sequences. However, it creates a new set of complex machinery and operational problems of its own.

Table 2 shows the features of four integral reactor design concepts.

Table 2. Features of Some Integrated Type Reactors

NAME OF REACTOR			PIUS (SECURE-P)	ISER - CV	SPWR H-H	CAREM
DEVELOPMENT ORGANIZATION			ASEA-ATOM (SWEDEN)	Univ. of Tokyo (JAPAN)	JAERI (JAPAN)	CNEA-INVAP (ARGENTINA)
PLANT	THERMAL OUTPUT	MWt	1616	645	1100	100
	ELECTRIC OUTPUT	MWe	500 ·	210	350	27
PRIMARY CIRCUIT	CORE OUTLET/INLET TEMPERATURE	°C	294/263	323/289	310/280	326/284
	CORE OUTLET PRESSURE	MPa	9.2	15.5	13	12.25
	NUMBER OF LOOPS		4	1	4	
	NUMBER OF SGs/PUMPs	UNIT	4/4	1/4	4/1	12/0
	MASS FLOW	kg/s	9975	3254	6667	410
CORE	EQUIVALENT DIAMETER/LENGTH	m	3.84/1.97	2.6/1.97	2.89/2.0	/1.4
	NUMBER OF ASSEMBLIES		193	89	120	61
	OUTPUT DENSITY	MW/m^3	71	63	84	
	URANIUM LOADING	tons	67.5	27.0	35.5	
SECONDARY CIRCUIT	OUTLET PRESSURE	MPa	4.0	5.7	5.0	4.7
	STEAM TEMPERATURE	°C	263	300	285	290
	MASS FLOW	ton/hr	2,990	1,280	2,000	
	FEED TEMPERATURE	°C	210	26	210	200
PRESSURE VESSEL	MATERIAL		PS Concrete	STEEL	STEEL	STEEL
	INNER DIAMETER/HEIGHT	m	13/34.5	6/26.4	6.6/25.9	2.84/11
	THICKNESS	m	7.8 - 8.5	0.3		0.12
	VOLUME	m^3	4,300	600		
	WEIGHT	tons	135,000	1,400		120

Table 2. (cont.)

NAME OF REACTOR	PIUS (SECURE-P)	ISER - CV	SPWR H-H	CAREM
SYSTEM COMPOSITION	Integrated reactor with SG	same as left	same as left	same as left
	Immersion of the primary system in boric acid solution pool	same s left	Main circulating pump at RPV top dome	Natural circulation
	No control rod	same s left	same as left	
SAFETY FEATURES	Passive shutdown of reactor by density lock	same as left	Passive shutdown by hydraulic pressure valves	Fast shutdown by neutron absorbing element backed up by passive boron injection

Safety System Features

Integral design concept represents a complete departure from existing light water reactor plants. The safety systems are designed on the basis of simplicity and reliability. They are not depend on active engineered safety system, but ensured by the passive nature. In case PIUS/ISER, the reactor shutdown is effected by allowing the injection of boric water to the primary coolant. The interface between primary coolant and boric water pool is assured by **density lock**, in case of PIUS/ISER and by **hydraulic pressure valves** in case of SPWR. CAREM design uses control rods for fast shutdown. In all cases no active engineered equipments are needed.

As mentionned above, the integration of primary system into RPV and elimination of control rods (in case of PIUS/ISER), the initiating events to accident conditions such as LOCA and RIA can not be hypothesized. Then, this concept could obviously provide a higher safety level against core degradation accident than current LWR design.

Reactor Design

In an integrated reactor the whole high energy primary system such as core, SG, primary coolant and vapor dome is contained inside a single pressure vessel. The core is placed at the bottom above which the SGs are locatefd. Self pressurization of the primary system is the result of the liquid-vapor equilibrium in the upper part of RPV.

In such configuration, the dimension of RPV becomes relatively large; 13.4 m diameter and 36 m height for PIUS (500 MWe) and 6 m diameter and 26.4 m height for ISER (210 MWe). Such design will challenge the plants maintainability, especially for the refuelling system and surveillance of Steam Generator tubes and the machinery.

Standardization

In order to be more economical and better licensability, the standard size of the plant must be determined. The existing conceptual designs show various sizes ranged from 10 MWe up to 600 MWe per unit.

Although standardization is primarily an economic and licensing stability issue, the achievement of high degrees of standardization also has benefits from the standpoint of operations and maintenance, training, spare part management, and other simplifications that could have indirect safety benefits.

Mini size plants, 25 - 30 MWe, might be an interesting option for electricity generation in many isolated islands in Indonesia, indeed, the economic competitiveness of such plant is a necessary conditions.

Steam Generator Design

Steam Generator (SG) is one of the most important components of the reactor system. It is important as the core itself, because reliability with a minimum amount of maintenances is mandatory for a long time period of time. Indeed, the availability of the plant will be determined by the SG reliability. Specifically, in integral type reactor, enhancement of the thermal effectiveness and resistance against corrosion is of utmost concern to increase the plant performance.

Seismic Design Criteria

The internal reactor pressure vessel structure of integral type is completely different from the existing design, i.e. by introduction of SGs in RPV just above the core. Such configuration requires certainly higher requirements to seismic design.

Needs in the Future

In order to support the development of integral concept extensive experimental program have been undertaken. However, the following research and development (R&D) are considered necessary to develop the reactor more efficient :

- An efficient and reliable SG including the material development related to corrosion protection.
- Reactor dynamics including the accidental conditions using large scale thermal-hydraulic test facility.
- A demonstration/experimental plant might be necessary to be constructed to demonstrate the operability and maintainability.

Beside this, to reach proveness of integral design technology the international promotion must be supported by creation of industrial consortium. The work must be developed together between designer, vendor, utility and the government.

SUMMARY

In the immediate future that is the next 50 years, the nuclear energy system will be dominated by evolutionary improvements in technology and safety of the present proven design of LWR and PHWR. Beyond 50 years, a new revolutionary concepts may emerge. Among this, the integrated reactor concepts provide a remarkably safety inherent with enhancement of passive features. The proveness of this concept must be followed by demonstration of experimental reactor and supported by an industrial consortium.

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THE INHERENTLY SAFE IMMERSED SYSTEM (ISIS): SAFETY AND ECONOMIC ASPECTS

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Abstract

ANSALDO has conceived a reactor called ISIS (Inherently Safe Immersed System), an innovative light water reactor with easily understandable safety characteristics.

The main targets are: passively safe behaviour, no pressurization of the Reactor Containment under any accident condition, control of plant capital cost and construction schedule by virtue of the modular concept and the compact layout.

The ISIS concept, described in general terms in the paper, builds up on the Density Lock concept originally proposed by ABB ATOM for the PIUS plant (ref. /1/), featuring innovative ideas derived from ANSALDO experience and based on proven technology from both LWR and LMR.

1. MAIN TARGETS OF THE ISIS CONCEPT

1.1 Safety targets

Significant progress has been made in the last years towards nuclear reactors that rely to the smallest possible extent on safety-related active systems, which, even using up-to-date technology, are felt by the public as prone-to-fail, no matter how low the frequency target for their loss is set.

The ISIS concept, under development in ANSALDO, largely embodies this progress. The main safety targets may be summarized as follows:

- No core melt-down and negligible release of radioactivity in any accident condition, by virtue of the reactor concept itself.
- Prompt reactor shut down occurring naturally after any abnormal condition.
- Reactor cooling in natural circulation for unlimited time.
- Self-depressurization of the reactor after a postulated failure of the pressure boundary.

1.2 Economic targets

The economic target aims at a viable industrial power plant based on specific overnight-capital cost and construction time competitive with those of the Light-Water Reactors under development. This is achievable by means of following features:

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

ISIS overall design Parameters

Thermal power	650	MWth
Net electric power	200	MWe
Core inlet temp.	271	°C
Core outlet temp.	310	°C
Operating pressure	14	MPa
Feedwater temp.	120	°C
Steam pressure	4,6	MPa
Steam outlet temp.	290	°C

3. THE ISIS PRIMARY SYSTEM

The Primary System of the ISIS reactor is of the integrated type (fig. 2), with the Steam Generator Unit (SGU) housed in the Reactor Vessel, to which feedwater and steam piping are connected.

Within the Reactor Vessel, an Inner Vessel provided with wet metallic insulation separates the circulating low-boron primary water from the surrounding highly borated cold water.

Hot and cold plena are hydraulically connected at the bottom and at the top of the Inner Vessel by means of open-ended tube bundles, referred to in the following as Lower and Upper Density Locks. The Inner Vessel houses the Core, the Steam Generator Unit and the Primary Pumps.

Outstanding feature is the complete immersion of the Pressure Boundary, made up, for each module, of a Reactor Vessel and of a separated Pressurizer with interconnecting Pipe Ducts, in a large pool of cold water.

During normal operation, the heat generated in the core is transferred to the SGU via the water circulated by the Primary Pumps, which are located at the top of the Inner Vessel. In case of unavailability of this heat transfer route, the cold and highly borated water of the Intermediate Plenum enters the Primary Circuit from the bottom, mixes up with the hot primary water, shuts down the reactor and cools the core in natural circulation. The same process, by heating the intermediate plenum water and the Pressure Boundary metal, activates the natural heat transfer route towards the Reactor Pool, which contains approximately 6.000 cubic meters of cold water.

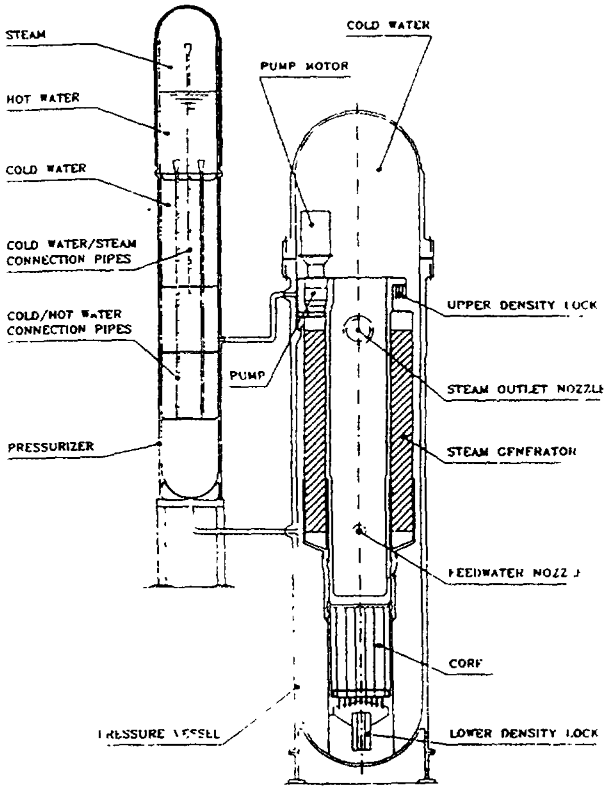

Fig. 2 - ISIS Reactor Module

The water inventory in the Reactor Pool is large enough to allow the water itself to remain below the boiling point after removal of the decay heat for about a week.

Cooling down of the plant pool is guaranteed, anyway, for an unlimited time, by virtue of two loops provided with water-air heat exchangers in natural circulation, sized to reject to the atmosphere, at steady state, approximately 2 MW and thereby capable to prevent the pool water from boiling.

Similarly to the PIUS reactor concept, the shut down and cooling functions of the core are carried out, in any condition, by the highly borated cold water of a plenum, which is hydraulically connected to the primary system by means of density locks.

However, unlike the PIUS, the intermediate plenum of ISIS contains a relatively small inventory of cold water (approximately 300 cubic meters per reactor module) at primary system pressure.

4. MAIN COMPONENTS

Reactor Vessel

The Reactor Vessel is of cylindrical shape with hemispherical heads.

The construction material is low-alloy carbon steel, internally lined with austenitic stainless steel.

The main openings of the Reactor Vessel are the water/steam nozzles and the two connections to the Pressurizer.

Core

The reactor core consists of 69 typical (17 X 17) PWR fuel assemblies with a reduced length to limit pressure losses.

Steam Generator Unit (SGU)

The SG features an annular tube bundle with helicoidal tubing.

The steam is generated tube-side. The feed water piping is connected to feed water headers, located symmetrically inside the reactor vessel within a calm zone, provided each with two tubeplates laid out vertically. The tubes depart circumferentially from the tubeplates.

A similar arrangement is provided at the top for the two steam headers connections.

The vertical arrangement of the tubeplates aims at preventing crud deposition at the tube-to-tubeplate connections, where the corrosion is likely to occur.

The higher outer rather than inner tube pressure, a reversed situation with respect to a conventional SGU, reduces the risk of flaw growth in the tubes.

Primary Circulation Pumps

The two Primary Pumps of the variable speed, glandless, wet winding type (like the pumps manufactured by Hayward Tyler Fluid Dynamics) are fully enclosed within the Reactor Vessel. The pump motor is cooled by the water of the Intermediate Plenum.

Above Core Structure (ACS)

The ACS, shaped like a flat-bottom cylindrical glass, provides the support for the core instrumentation and forms the inner wall of the annular riser of the primary water. The ACS is open at the top. The water within it is part of the intermediate plenum and this helps to limit the primary water inventory in the reactor module to a minimum. The ACS is flanged to and suspended from the top of the Inner Vessel for easy removal to allow standard fuel handling.

Pressurizer

The Pressurizer is of a slim cylindrical shape with hemispherical heads.

The pressure control function is carried out in the upper part, which is externally insulated to limit heat losses from the steam and hot water plena.

The remaining bottom part contains a cold water plenum, hydraulically connected to the upper hot water plenum by means of a number of pipes.

The function of the pipes is to enhance mixing of the hot water with the cold water, in case of water flow towards the reactor vessel during transients.

Interconnecting Pipe Ducts

The two Pipe Ducts between Pressurizer and Reactor Vessel connect hydraulically the top and the bottom of the respective cold water plena in order to create a common cold water plenum.

The choice of two connection levels makes natural circulation possible in case of temperature difference between cold plena. If the normal decay heat removal route (i.e. the active steam/water system) is lost, the uninsulated wall portion of the Pressurizer would thus help removing by conduction the decay heat towards the Plant Pool.

Conveyed water to and from each vessel, belonging to a common cold water plenum, does not significantly contribute to the thermal loadings on the pressure boundary during transients.

Air Coolers

Two finned-tube Air Coolers are arranged in loops in natural circulation.

Each Air Cooler is rated 1 MWth at 30 °C ambient air and 95 °C pool water inlet temperature.

The onset of natural circulation occurs every time the pool water temperature becomes higher than the ambient air temperature.

Operation of the air coolers would prevent, for unlimited time, the pool water from boiling, in case of long-term loss of the operational decay heat removal system.

The technology of the air coolers in natural circulation is derived from the design and operating experience of ANSALDO in the field of the LMFBRs.

5. FULL-POWER OPERATION OF THE ISIS REACTOR

During normal operation the hot/cold water interface level in the Lower Density Lock is maintained constant by varying the speed of the Primary Pumps.

Any rising of the interface level is counteracted by an increase of the pump motor speed. Any lowering of the interface level is counteracted by slowing down the pump speed.

Dynamic Analysis of ISIS Control System is in progress. Preliminary results, not yet published, confirm that the reactor power is controlled by the concentration of boron in the primary water and by the intrinsic negative feedback of the core.

6. NATURAL BEHAVIOUR OF THE ISIS REACTOR UNDER ACCIDENT CONDITIONS

In the design of ISIS emphasis has been put in the prevention of core damaging accidents.

The two main safety functions, reactor shutdown and decay heat removal, are performed without recourse to the usual sensor-logic-actuator chain, i.e. with no inputs of "intelligence", nor external power sources or moving mechanical parts, according to the definition of Category B Passive Components (ref. /2/).

An active Reactor Protection System, aimed at anticipating passive system interventions, is included in the design, but is not credited in the safety analysis.

As anticipated in the Reactor System Description, mixing of the Primary Water with the Intermediate Water and the consequent natural heat transfer toward the Reactor Pool is the basic feature to assure safety under Design Basis Accidents such as Loss Of the Station Service Power and Loss of Heat Sink (ref. /3/).

During these DB Accidents the pressure boundary integrity assures the availability of water to cool the core and to transfer the decay heat to the Reactor Pool.

In case of LOCA Accidents, the Core shutdown and cooling functions are possible only if a sufficient inventory of water remains available.

The design features of ISIS guarantee the availability of this water because of the prompt self-depressurization of the system which is the consequence of the same hot-cold water mixing process.

To illustrate the effectiveness of this self-depressurization capability, the two following DB Accidents are presented :

- double ended break of the lower pipe connection between RPV and Pressurizer;
- Steam Generator tube rupture.

Additionally, the extremely fast transient following an hypothetical break at the bottom of the RPV is reported as an exercise to better understand the thermalhydraulic phenomena linked to the self-depressurization.

All transient analyses have been carried out using the RELAP5 computer code, with a nodalization made up of 256 control volumes 262 flow junctions and 78 heat structures; neutronic point kinetics has been used to evaluate the power in the core.

Loss Of Coolant Accident

This accident consists in a double ended break of the lower, 150 mm nominal diameter line connecting the RPV and the Pressurizer. This accident scenario has been chosen because this is the largest line of the pressure boundary and also because the break location is far from both Density Locks, thus worsening the loss of cold water from the vessels (ref. /4/).

Considering that, the break location is 25 m below the Reactor Pool water level, the absolute pressure at the break outside the RPV is 3.5 bar. No action is credited of any active protection or control system.

When the accident starts, interconnected thermalhydraulic phenomena occur simultaneously within both RPV and Pressurizer. Cold water outflows from both RPV and Pressurizer; hot primary water replaces the losses in the Intermediate Plenum through both Density Locks. This phase lasts about 2-3 seconds. Then flashing hot water causes Primary Pumps cavitation which, in turn, allows the inlet of intermediate water into the primary system and the Core via the Lower Density Lock with a quick decrease of generated power.

The Reactor behaviour can now be explained considering that the Primary Pumps remain cavitating all over the transient and the primary system behaves like two channels hydraulically connected in parallel.

Both channels, the one made up by the Core and the Riser, and the second by the Downcomer and the SGU, are alternatively flooded by intermediate water entering the primary system through the Lower Density Lock. Self-depressurization of the system takes place mainly because of the following two water mixing effects (fig. 3):

In the RPV, hot primary water flowing from the Upper Density Lock mixes up with the large volume of cold intermediate water of the RPV Head, purposely provided for this function [1].

In the Pressurizer, hot water flowing down through the vertical pipes mixes up with the large volume of cold intermediate water underneath.

The system pressure at the break equals the external pressure in about 450 seconds.

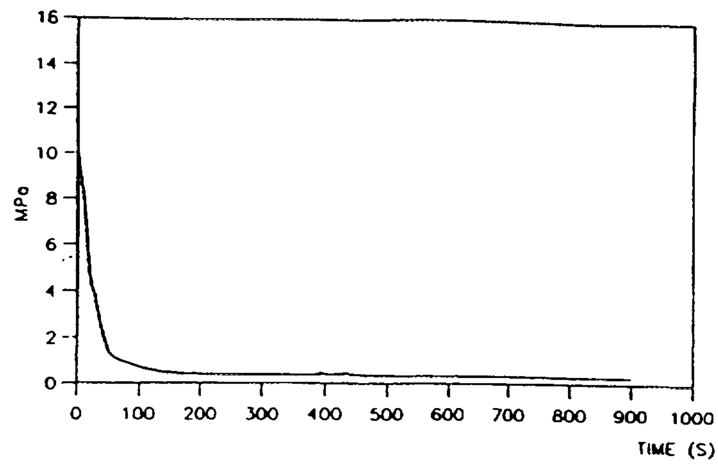

Fig. 3 - LOCA
Core pressure

At this moment the RPV water stops flowing out and reversal flow of cold, high-boron water from the Reactor Pool sets on.
The core is shutdown (fig. 4) by intermediate water entering through the Lower Density Lock.

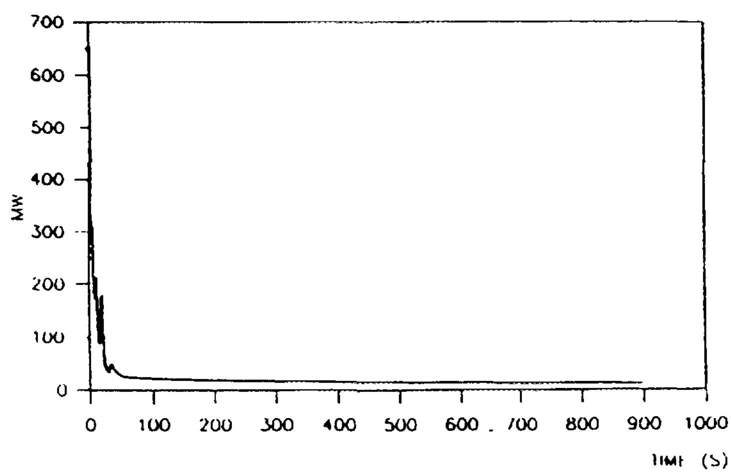

Fig. 4 - LOCA
Nuclear power

Figure 5 shows that the maximum cumulated amount of water loss is less than 120t (approximately 25% of the initial inventory) and only the following regions of the Reactor Module remain temporary uncovered:

- the Head of the RPV (with the water level always remaining above the Upper Density Lock);
- the Pumps, the upper part of the Riser and the SGU;
- the hot region of the Pressurizer.

[1] Thermalhydraulic phenomena in the large plenum of the RPV Head would be better predicted by a 3D code which is, in any case, needed to optimize the design of the internal structures to enhance water mixing in this region.

Later on in the transient, reversal flow from the Reactor Pool starts recovering the water level in the RPV; at the end of the computer run (i.e. after 900 seconds) about 40 t of water have already entered the RPV from the Reactor Pool.

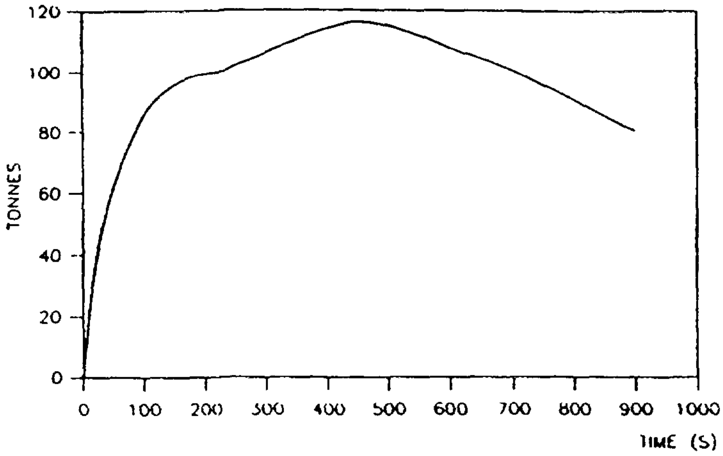

Fig. 5 - LOCA
Cumulated water loss

During the transient the Core never uncovers or heats up as shown in Figure 6. The maximum temperature of the "average" fuel rod has remained lower than at nominal conditions. A similar behaviour is shown for the clad surface temperature.

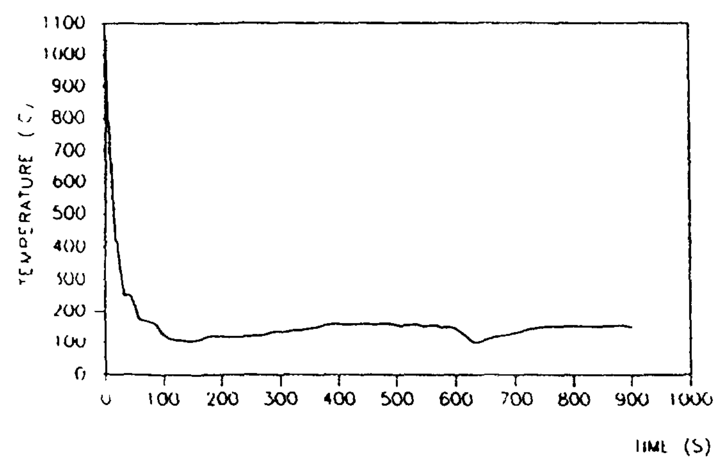

Fig. 6 - LOCA
Maximum temperature of average fuel rod

Steam Generator Tube Rupture

In this accident a break of 10 cm^2 cross section located at the connection between SGU tubes and steam headers is simulated; the break size is approximately equivalent to the cumulated cross sections of 8 SGU tubes.

No credit has been taken for action of active systems that can mitigate the consequence of the accident, but for the Primary Pumps Speed Control System which delays the inlet of highly borated water through the Lower Density Lock. The steam pressure and the feedwater flow rate are assumed accordingly to remain constant during the transient. [2]

When the accident occurs, water from the primary system enters the SGU ruptured tubes at a max mass flow rate of 96.5 Kg/s.

An equal amount of intermediate water enters the primary system through the Upper Density Lock as long as the Primary Pump Control System is capable to control the hot-cold interface level in the Lower Density Lock.

Primary water with increasing boron concentration enters the core and reduces the generated power (fig. 7).

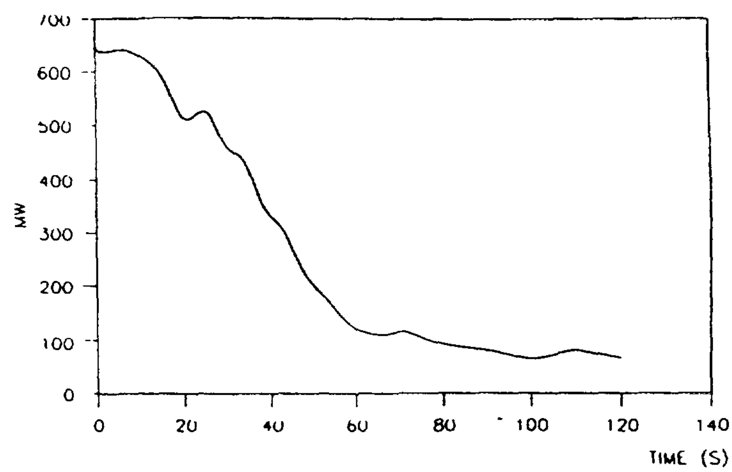

Fig. 7 - Steam Generator Tube Rupture
Nuclear power

The amount of intermediate water entering the Upper Density Lock is replaced in the RPV by water leaving the Pressurizer. In the Pressurizer itself fast depressurization takes place because of the hot-cold water mixing process already explained above for the LOCA

[2] Crediting the SGU isolation, the transient would behave very similar to the transient of loss of heat sink which has been shown to cause a fast reactor shutdown (ref. /4/)

transient.

Both effects of reduced core power with associated lower primary water temperature and Pressurizer self-depressurization reduce the overall primary system pressure (fig. 8) down to the secondary system pressure (tube-side SGU pressure) which has been assumed to remain at its nominal value.

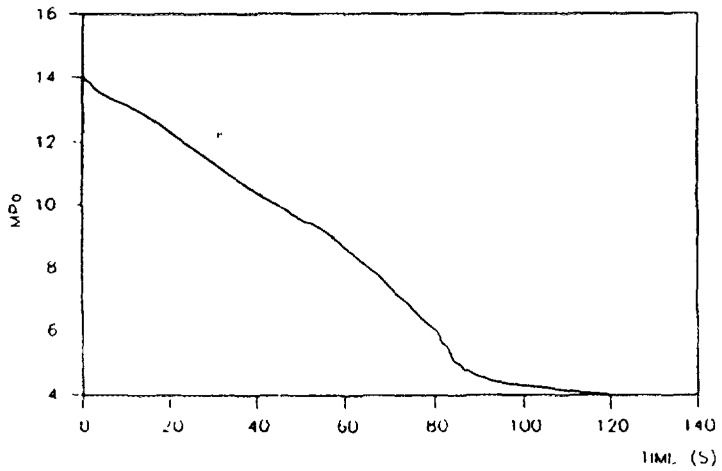

Fig. 8 - Steam Generator Tube Rupture
Core pressure

At this time the primary water stops flowing into the SGU tubes. Figure 9 shows that the cumulated amount of water loss is less than 8 tonnes which corresponds to the inventory of the hot water in the pressurizer.

The curve of the fuel temperature shows a steadely decreasing pattern, fig. 10.

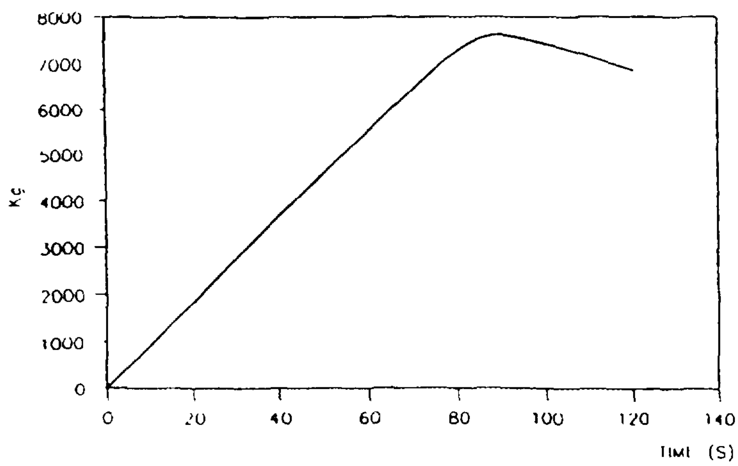

Fig. 9 - Steam Generator Tube Rupture
Cumulated water loss

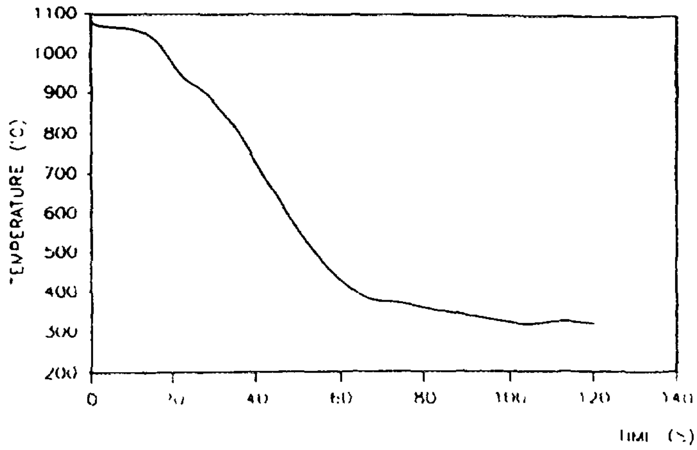

Fig. 10 - Steam Generator Tube Rupture
Maximum temperature of average fuel rod

Break at the bottom of the Pressure Vessel

In this exercise an hypothetical break of 500 cm^2 cross section has been assumed to occur at the bottom of reactor pressure vessel; this accident scenario is arbitrary and imagined to generate a very severe thermalhydraulic transient; in fact the break is positioned at the lowest location of pressure boundary and therefore has the potential of completely emptying the RPV. This exercise is intended to demonstrate that the self-depressurization process can avoid the uncovering of the core even in this case. No protection or control systems, no any other active system was credited during the accident analysis.

When the transient starts, there is a large blowdown of intermediate water from the RPV and Pressurizer into the Reactor Pool (the initial mass flow rate through the break is about 7000 kg/s). The escaping flow rate is fed by displaced primary water which is mostly contained in the SGU. Primary water leaves the SGU from the bottom via the Downcomer and, after few seconds, also from the top via the Primary Pumps and Upper Density Lock.

At the very begining of the transient the water flowing down through the Downcamer splits in two streams: the one leaves the Inner Vessel through the Lower Density Lock and the second flows up through the Core, the Riser and leaves the Inner Vessel through the Upper Density Lock. The reactor core is continously fed by primary water flowing upwards and its temperature is continuously decreasing because it is kept cooled since the beginning of the transient.

At the time of about 7 seconds, with Primary Pumps in cavitation, the primary water stops leaving the Inner Vessel through the Lower Density Lock and a reversal flow of intermediate water sets on flooding the core.

At this moment the usual way of natural circulation of ISIS reactor is recovered and the primary system fed with cold and borated water.

The mixing of cold and hot water initiates the self-depressurization of the system in the way described before for the case of LOCA (fig. 11).

The system continues its depressurization up to the time of about 200 seconds when its pressure drops below the Reactor Pool pressure at the break location.

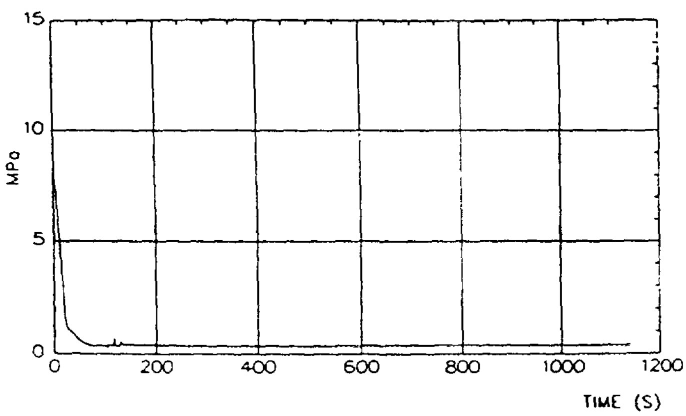

Fig. 11 - Break at the bottom of the RPV
Core pressure

At this moment, the total mass of displaced water (figure 12) is less than 200 tonnes (approximately 50% of the total inventory of one module) and the RPV has been emptied only down to about the center line of the SGU.

After about 1000 seconds, the initial water inventory is completely recovered and the reactor is in the state of stable cold shutdown.

The evolution of the generated power is shown in fig. 13; the power reduction during the first 7 seconds is caused by the void effect associated to the depressurization and the following shutdown is assured by the borated water.

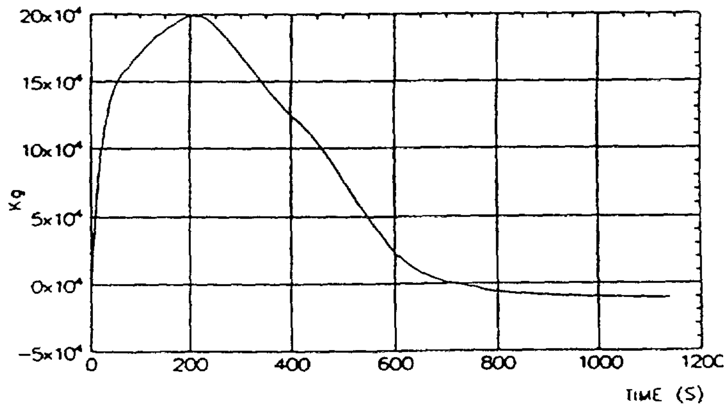

Fig. 12 - Break at the bottom of the RPV
Cumulated water loss

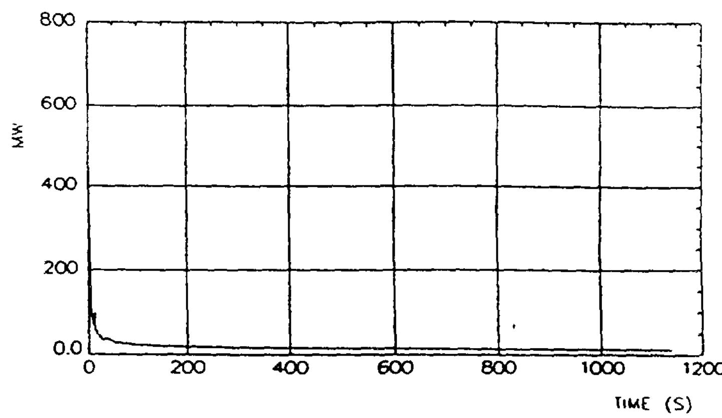

Fig. 13 - Break at the bottom of the RPV
Nuclear power

The fuel temperature has steadily decreased as shown in fig. 14 and 15.

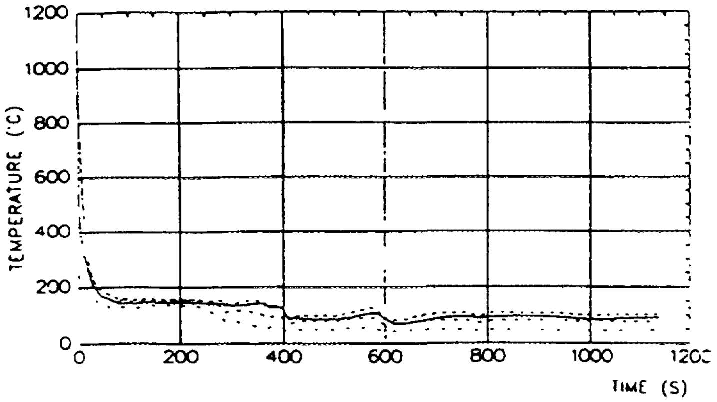

Fig. 14 - Break at the bottom of the RPV
Maximum temperature of average fuel rod

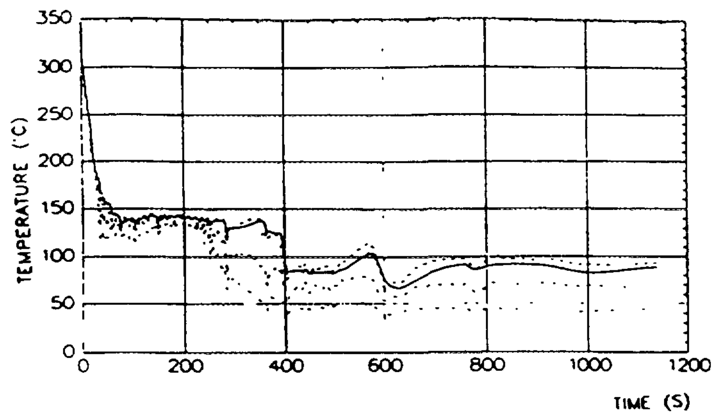

Fig. 15 - Break at the bottom of the RPV
Clad surface temperature at different elevations

7. Modular And Compact Plant

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

The selected unit power of the ISIS Reactor Module (200 MWe) is consistent with this design target.

Layout studies of the ISIS power plant are in progress in ANSALDO, to optimize component arrangement and reduce erection time of the Reactor Modules and of the Balance of the Plant.

The compact reactor layout is made possible by the integrated design of the primary circuit.

8. Economic aspects

At the beginning of the development of the ISIS concept (about seven years ago) it seemed reasonable to foreseen a moderate increase of the cost of the fossil fuels in the near future which would have improved the economic competitiveness of nuclear energy. Today, instead, two facts worsen this competitiveness:

- the fossil fuels price has remained low and stable,
- the efficiency of the modern electric energy generating conventional power plants is continuously increasing.

The importance of the second fact is such that it will drastically affect the energy market, in particular it will impact the market of nuclear energy.

In the past, the efficiency of electricity production of the nuclear power plants was similar to that of the conventional power plants. Under that condition it was profitable to generate electricity by the large-size nuclear power plants that dominate the nuclear panorama.

Today, the efficiency of the modern Combined Cycle Turbo-Gas (CCTG) Power Plants has exceeded 50% and in the near future (before the year 2000) will reach and perhaps trespass 60%, while the one of the nuclear water reactors stagnates at about 33%.

This new fact will have two main consequences:

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

Qualitatively it can be affirmed that today an efficient use of energy favours the fossil fuels for electricity and of the nuclear fuel for heat production, because of the lower electric efficiency of the nuclear power plants.

Quantitatively, a preliminary economic evaluation carried out comparing 60% efficient CCTG, co-generative CCTG, conventional boilers and nuclear power plants, has shown that nuclear power plants could recover part of their lost economic attractiveness only if exploited as co-generating or as thermal power plants.

The co-generative use appears attractive from 3000 hours/yr. and the thermal use from 5000 hours/yr. upwards (increase of the value of the plant in the order of more then 50 %)

The increase of the value of the plant is more important for co- generative reactors and can largely exceed 100% for specific site conditions where heat can be used during the most part of the year.

A trivial condition for any interest of a prospective utility in a co-generative nuclear plant is that an adequate co-generating reactor exists. For this the reactor designers, besides providing the reactor with convincing characteristics of radiological safety, have to overcome the unfavourable scale-effect of downsizing, because the thermal power needed is mostly in the order of hundreds of megawatts against the thousands of megawatts available from the today large nuclear reactor conceived for electricity generation only.

In the view of a reactor designer, the smaller reactor can be competitive, in spite of downsizing, provided that:

- the number of systems of the larger plants is strongly reduced,
- the specific mass of steel of the NSS it is not significantly increased (scale effect neutralised),
- the specific operation & maintenance costs become not excessive.

The co-generating version of the ISIS reactor is being designed to cope with these requirements.

The technical features illustrated in this article and the results of preliminary analyses for an use of ISIS as co-generating reactor can be summarised as follows:

- no active safety system is necessary to assure safety. All active safety systems can be eliminated.
- no adverse scale effect on the specific mass of steel of the ISIS NSSS with respect to the larger modern PWRs. This is possible also thanks to the milder operating conditions of a reactor designed for co-generation (e.g., lower operating pressure).

Ongoing studies explore furthermore the possibility of reducing operating & maintenance costs, taking profit of the predicted simple operation of ISIS, of the reduced number of systems, of the modular approach that makes possible to share facilities, such as fuel handling and component handling equipment, between the reactor modules installed in the same reactor building.

9. CONCLUSION

The ISIS is an innovative Nuclear Power Plant under development in ANSALDO. It is based on original ideas derived by ANSALDO experience on proven LWR and LMR technologies.

The main features of ISIS are as follows:

- Outstanding passive safe behaviour of the Reactor, which means core shut down and cooling functions ensured in all accident conditions and no release of primary coolant outside the Reactor Building.
- Compact reactor layout, associated with modular fabrication and erection, made possible by the integrated design of the primary circuit.
- Reactor concept flexible for combined generation of heat and electricity made possible by the modular solution and the low sensitivity to downsizing.

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AN INTEGRATED NUCLEAR REACTOR UNIT FOR A FLOATING LOW CAPACITY NUCLEAR POWER PLANT DESIGNED FOR POWER SUPPLY IN REMOTE AREAS WITH DIFFICULT ACCESS

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Abstract

The paper describes the conceptual design of an integrated advanced safety nuclear reactor unit for a low capacity floating NPP designed for power supply in areas which are remote with difficult access.

The paper describes the major structural and lay-out components of the steam generator and reactor units with main technical characteristics.

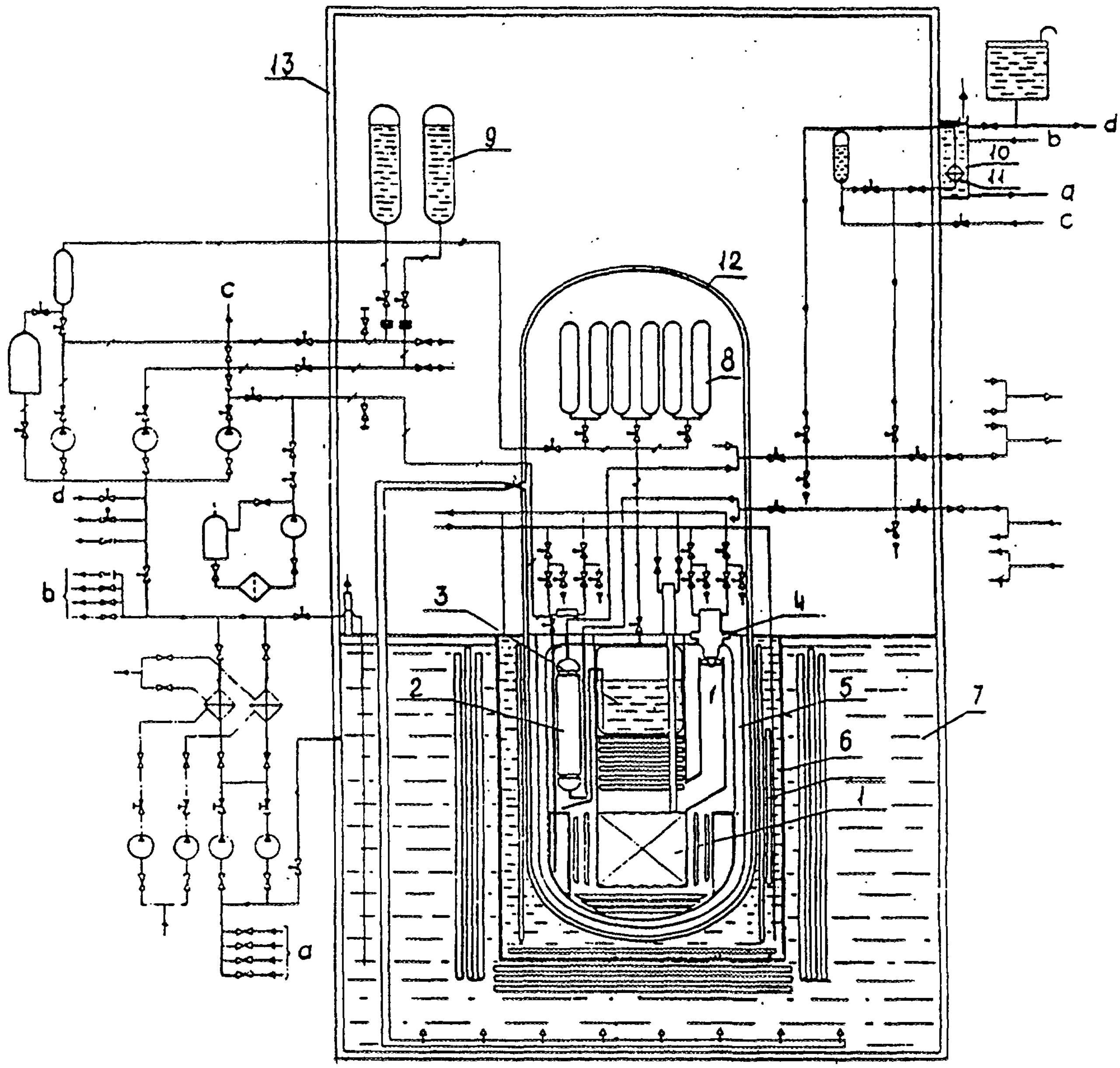
Conceptual design of a reactor facility with enhanced safety for a low capacity floating and environment friendly NPP has been developed in Research and Development Institute of Power Engineering to provide electricity supply to areas which are remote with difficult access.

The most advanced technical solutions well mastered during recent years in designing of navy nuclear power facilities have been used.

The following technical solutions provide high safety level of the reactor facility:

- application of well mastered technology of water-cooled reactors with developed inherent safety features;
- location of the total primary circuit equipment into one vessel of an integrated nuclear steam supply system (NSSS);
- provision of defence - in-depth barriers to prevent ionizing radiation and radioactive fission products release into environment, realization of technical measures to protect confining barriers and maintain their effectiveness;
- application of safety systems, mainly based on passive operation principle;
- independence from external power sources.

The Development of the reactor facility (RF) for floating NPP was carried out in compliance with current Russian rules and requirements to provide safety of stationary and floating nuclear power plants and ship nuclear power facilities and in accordance with the modern notion of prospect enhanced safety NPPs, elaborated so far by the world community. It was considered that RF met safety requirements, if its radiation effect on personnel, population and environment in normal operation and design-basis accidents did not lead to excess of specified dose rates, and limits this effect in beyond design-basis accidents. Technical and organizational measures were assumed to ensure safety with any design-basis initial event with superposition of one failure independent of the initial event of any of the following safety system components: active or passive component having mechanical movable parts or one personnel error independent of the initial event. Besides one failure independent of the initial event, a nondetectable failure of components, not monitored during operation, which influences accident propagation, and results in violation of safe operation limits is taken into account.



· Fig.1. Reactor facility flow diagram:

1 - core; 2 - steam generator; 3 - pressurizer; 4 - primary circuit electric circulation pump; 5 - steam-generating system vessel; 6 - iron-water shielding tank; 7 - bubbling tank; 8 - high pressure gas cylinder; 9 - emergency flooding cylinder; 10 - emergency cooldown tank; 11 - heat-echanger-condenser; 12 - safeguard housing; 13 - containment.

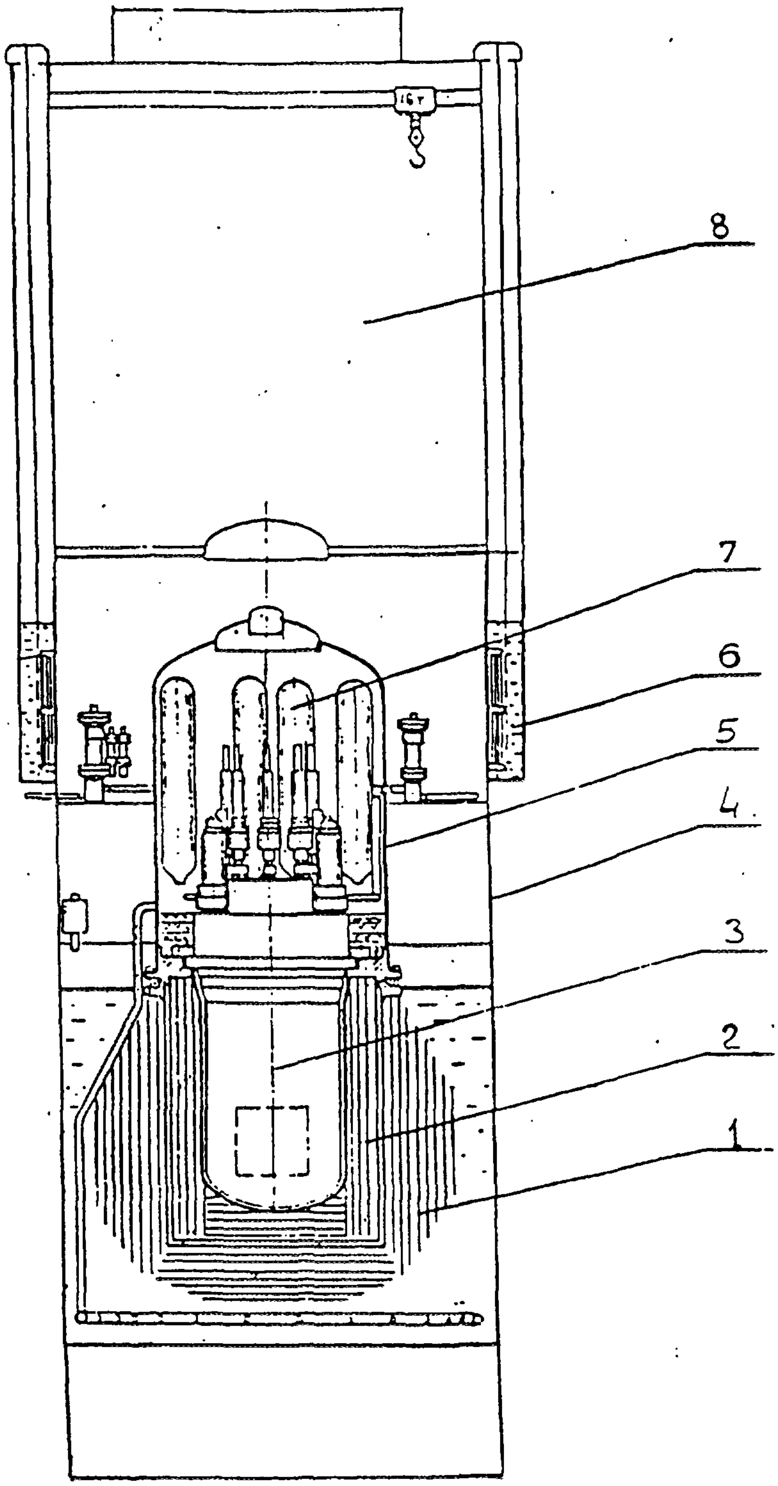


Fig.2. Reactor facility general arrangement. Elevation:
1 - bubbling tank; 2 - iron-water shielding tank; 3 - steam-generating system; 4 - containment; 5 - safeguard housing; 6 - emergency cooldown heat-echange - condenser; 7 - high pressure gas cylinder; 8 - refuelling and repair room.

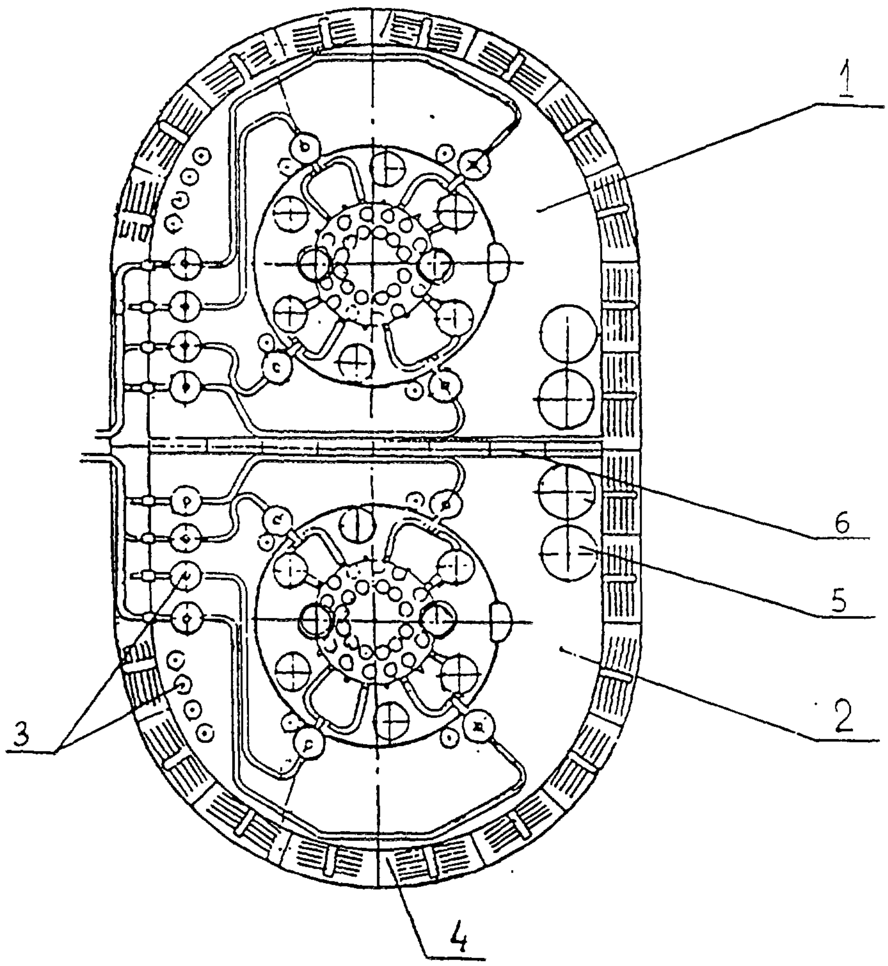


Fig.3. Reactor facility general arrangement. Plan:

1 - steam-generating system No.1; 2 - steam-generating system No.2; 3 - isolating valves for steam and secondary circuit feedwater; 4 - emergency cooldown tank; 5 - emergency flooding cylinder; 6 - strong leaktight partition.

Structurally the RF is divided into two separate and independently operating steam generating facilities with identical principal flow charts and equipment (see Figs. 1-3).

The RF employs water-moderated, water-cooled reactors with inherent safety and control features due to negative power and temperature reactivity coefficients. The core physical characteristics are so selected that the above coefficients be negative in the entire range of temperatures during core life. This eliminates spontaneous reactor power excursion in normal startup and heatup and stabilizes operation in steady-state conditions and transients.

The adopted core structure rules out the possibility of forming local critical masses both in normal and emergency modes. Formation of secondary critical mass in hypothetical accidents, resulting in partial or full core melting is also excluded.

All equipment of the primary circuit (the core with reactivity compensation components, steam generator, electric circulation pumps, pressurizer) except gas cylinders are located in cylindric vessel of the integrated nuclear steam supply system (NSSS) (see Fig. 4).

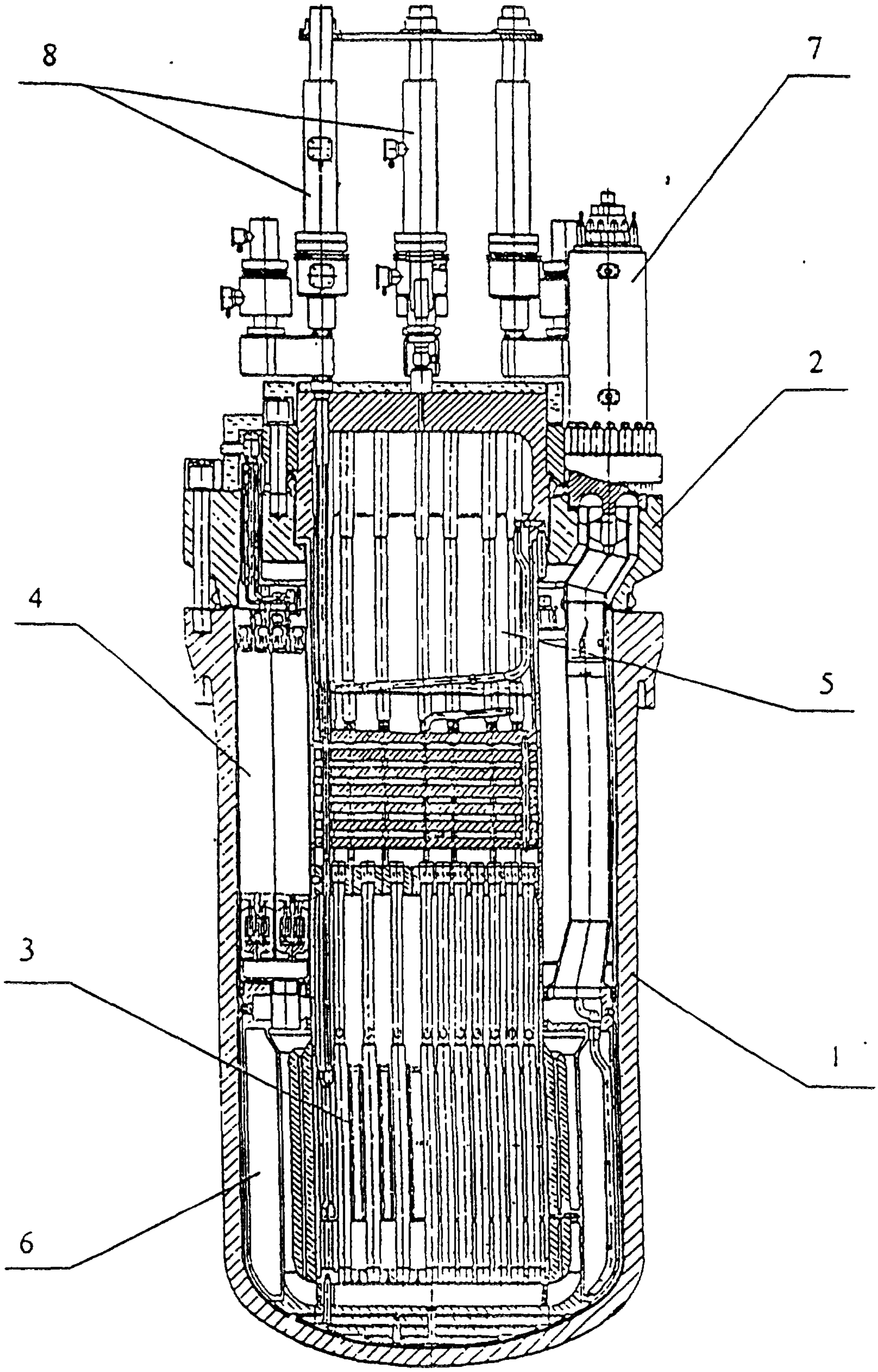


Fig.4 Steam-generating system:

1 - vessel; 2 - ring-type cover; 3 - core; 4 - steam generator; 5 - pressurizer; 6 - intermediate capacity; 7 - electric circulation pump; 8 - control and protection members drives.

The pipeline between gas cylinders with pressurizer and make p system pipelines is connected to the NSSS cover and ends at this point.

Such NSSS structure allows:

- to reduce the number and extent of external primary circuit lines down to minimum and reduce probability of its depressurization;
- to provide high level of natural circulation for the primary circuit coolant;
- to increase water volume above the core and to improve its cooling conditions in a hypothetical accident related to primary circuit depressurization and water cut-off from the NSSS.

In order to decrease the leak and mitigate accident consequences in case of pipeline rupture which connects pressurizer with receiving cylinders, a constricting device is installed on it. Check valves are installed on makeup pipelines (in places of its attachement to NSSS cover).

RF employs defence-in-depth principle, based on barrier system to prevent ionizing radiation and radioactive substances release into environment supported by technical procedures to protect these barriers and maintain their efficiency.

In accordance with this the RF safety systems provide for:

- reactor emergency trip and keeping it subcritical;
- emergency heat removal from the core;
- confining of radioactive products within specified boundaries.

Flow chart and design of these systems are based on their passive action as much as possible. In order to ensure reactor emergency trip and keeping it subcritical the following systems have been designed:

- main emergency protection;
- additional emergency protection;
- liquid absorber injection system.

The main and additional emergency protections have different activation principles. They are activated automatically by CPS and complex system for technical means control (CSTMC) signals or by power cut-off.

Liquid absorber is injected by remote control.

Design of all compensating group (CG) activation devices is based on insertion of shim rods into the core using dump springs or the rods weight during power cut-off from actuators and CPS; the design also ensures keeping shim rods in inserted position in case of the unit turning-over.

The shim rods ensure reactor tripping and keeping it subcritical in all operational modes in case of one (any) CG failure (stuck in the extreme position) by means of CGs remaining in operation.

Additional emergency protection is activated automatically by a signal from CPS in response to failure of two or more CGs.

Besides activation by CPS signals the additional emergency protection becomes operational without any external signal under increase of primary circuit coolant temperature up to specified ultimate value. This is achieved by the additional emergency protection design.

Liquid absorber is injected into integrated NSSS only when the reactor has not been transferred to subcritical state due to failures of the main and additional emergency protections, but such situation is hardly probable.

Availability of the listed means for reactor emergency protection and keeping it subcritical can provide safety reactor trip not only in case of design-basis accidents, but also in hypothetical accidents, which are not related to core or NSSS damage.

In order to avoid spontaneous chain reaction during scheduled maintainance or repair works, CG actuator design envisages clutches, which are installed below "cold" startup position. Electromagnetic clutches are opened only by signals, which allow the core start-up.

To provide heat removal from the core in different emergency situations the RF has the following safety systems:

- emergency cooling system;
- emergency core flooding and cooling system.

RF can be cooled down by feedwater pumping into the steam generator by the steam-turbine means in both operational and emergency modes. In case of primary circuit depressurization the makeup system is also involved in heat removal from the core.

In emergency situations, which are not related to steam turbine failure, the RF is usually cooled by feedwater pumped into the steam generator by steam turbine. In case of the steam turbine failure and during power cut-off the integrated NSSS is automatically disconnected from the steam turbine (from steam and feedwater supply) and emergency cooling system (ECS) starts operating. Heat removal from primary coolant is provided for both operating and non-operating primary circuit electric circulation pumps (due to natural coolant circulation). In case of RF power supply loss the disconnection of NSSS from steam turbine and ECS activation are ensured by isolating and cut-off valves, which perform this function in "normal" position.

Emergency cooling system, consisting of 4 independent sections, can provide NSSS cooling during 24 hours in case of failure of any 2 ECS sections even with the plant blackout. Heat removal from primary coolant and its transfer to water in ECS tanks results from coolant natural circulation in this system. With electric supply availiable the water in ECS is cooled by pumping it through heat exchangers, which are cooled by outboard water. In case of NPP power blackout heat from ECS heat exchangers is removed due to heating followed by evaporation of water from ECS tanks.

Core makeup and core emergency flooding systems provide removal of residual heat from the core and eliminate its drying and further melting in case of primary circuit depressurization and coolant circulation failure. Each of these systems has redundances with respect to equipment and water supply channels to NSSS. The makeup system operates automatically by pressure drop in primary circuit signal from CSTMS and activation of the emergency core flooding system does not require any operator actions or CSTMC activation and occures due to membrane rupture under primary circuit pressure decrease down to specified level.

Taking into account that the makeup system besides RF normal operation assurance is supposed to provide safety, three independent NSSS water supply channels with high pressure pumps in each channel are envisaged in the makeup system. Water for primary circuit makeup is taken from condensate-feedwater system of steam turbine or from water storage tanks, and after water level in bubbling tank reaches specified level the makeup is performed in closed cycle, i.e. water for makeup is taken from bubbling tank, cooled in heat exchangers and delivered into NSSS; the coolant, flowing from NSSS again comes into the bubbling tank.

The emergency core flooding system is based on passive operation principle and comprises two independent channels. Water is supplied into NSSS through these channels due to gas pressure from pressure cylinders.

In case of primary circuit depressurization the feedwater is delivered into steam generator from steam turbine in parallel with operation of makeup and emergency core floodibg systems. This ensures additional removal of heat, released in the core and increases time needed for the core drying in hypothetical accidents due to natural convection of gas and steam in NSSS and partial steam condensation on steam generator surface. Steam turbine failure to supply water into the steam generator activates the ECS.

The availability of cooled iron-water shield around NSSS vessel and favourable conditions for natural circulation of steam-gas medium inside NSSS thereby reduce probability of core melting in hypothetical accidents and eliminate possibility of NSSS vessel melting down in such accidents.

In case of an accident, all actions to localize it are automatic. Control systems (CPS and CSTMC) have three-channel scheme and the control and emergency signals are generated using major principle (2 out of 3) that ensures sufficient operational reliability.

The following safety barriers are envisaged to reduce radioactivity release into environment in design - basis and hypothetical accidents:

- corrosion resistant matrix of fuel elements;
- claddings of fuel elements;
- leaktight primary circuit;
- safeguard vessel;
- containment.

Small number of the equipment, its lifetime characteristics and high automation degree ensure RF operation during one year without maintenance within the containment. This allows to adopt additional technical measures to enhance RF safety. Namely, ventilation system equipment located outside the containment is isolated with isolation valves designed for maximum emergency pressure in safeguard vessel, and is made operational only when it is necessary to make repairs inside the containment (gas is discharged from containment into environment via special filters under control). Conditioning system operates in closed cycle, its total equipment is located in safeguard vessel. Such technical solutions eliminate radionuclides release from containment in normal operation of floating NPP.

Leaktightness of the containment and safeguard vessel and no inside repairs allow to maintain decreased oxygen content (11-13 %) in equipment rooms, thus enhancing their fire safety.

The RF design solutions eliminate a possibility of safeguard vessel and primary circuit damage in case of earthquakes, aircraft crash or other external impacts.

In conclusion, the basic technical characteristics of the reactor facility are as follows:

Thermal power, MWt	2 * 42
Steam output, t/h	2 * 60
Superheated steam parameters:	
temperature, °C	290
pressure, MPa	3.53
Core life as evaluated for nominal power, h	20000
Specific core power rating, kW/l	68.0
Fuel composition	uranium dioxide dispersed in zirconium matrix
Enrichment with U^{235}, %	21
Time period of RF continuous operation without maintenance, h	8000
Mass, including safeguard vessel and containment, t	750

Due to high safety level (probability of severe beyond design-basis accidents, which can result in serious core damage or melting and following radionuclides release into environment does not exceed 2x10я5-8 я0per reactor per year) the RF can be recommended to be used for low capacity floating NPP, and its unique mass-dimension characteristics enable to construct a plant with such a draught (estimated as 2.6 m), which will make it possible to ship it along Nothern rivers to regions which are far away from the coast.

XA9745978

AN AUTONOMOUS NUCLEAR POWER PLANT WITH INTEGRATED NUCLEAR STEAM SUPPLY SYSTEM DESIGNED FOR ELECTRIC POWER AND HEAT SUPPLY IN REMOTE AREAS WITH DIFFICULT ACCESS

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Abstract

The paper contains basic conceptual principles used to develop the technical assignment for an autonomous nuclear power plant with integrated nuclear steam supply system (NSSS) designed to provide heat and electricity for areas *which are remote with difficult access*. The paper also describes technical procedures and equipment, NPP thermal hydraulic flow chart, steam generator design, safety aspects as well as operational and maintenance procedures.

1. *Introduction*

In areas of Russia which are remote with difficult access, for instance in the Extreme North, Far East, or Siberia a possible reasonable alternative to fossil-fuel energy sources, mainly hydrocarbon, is autonomous nuclear power plants (NPP) of relatively low capacity, shipped to the site in large modules and completely withdrawable from the site upon decommissioning. For the areas in question, small settlements and enterprises with low power demand are typical. The complexity of constructing electric transmission lines, gas pipelines, liquid fuel pipelines, and high cost requires to use local self-contained energy sources. Application of NPPs for power and heat supply may be cost efficient and promising from social and ecological point of view.

A necessity arises to develop new approaches to designing these power sources, naturally, taking into account the experience gained in reactor construction based on modern safety level. Such approaches are to be based on:

- use of well proven technology of water-cooled reactors, in particular, those used at transport facilities with adequate optimisation of their characteristics;
- maximum use of equipment which has operational field-checked prototypes;
- use of designs with inherent safety features;
- maximum ecological safety of NPP considering nature specifics in the Extreme North and Far East of Russia extraordinarily sensitive to production activities
- preservation of the natural landscape, flora and fauna due to NPP delivery and removal in large plant-manufactured modules which minimize erection and dismantling jobs at the site;
- minimal capital investments and operating costs to make such NPP competitive with conventional fossil power plants.

The paper briefly describes possible implementation of these approaches using a NPP equipped with the integrated nuclear steam supply system.

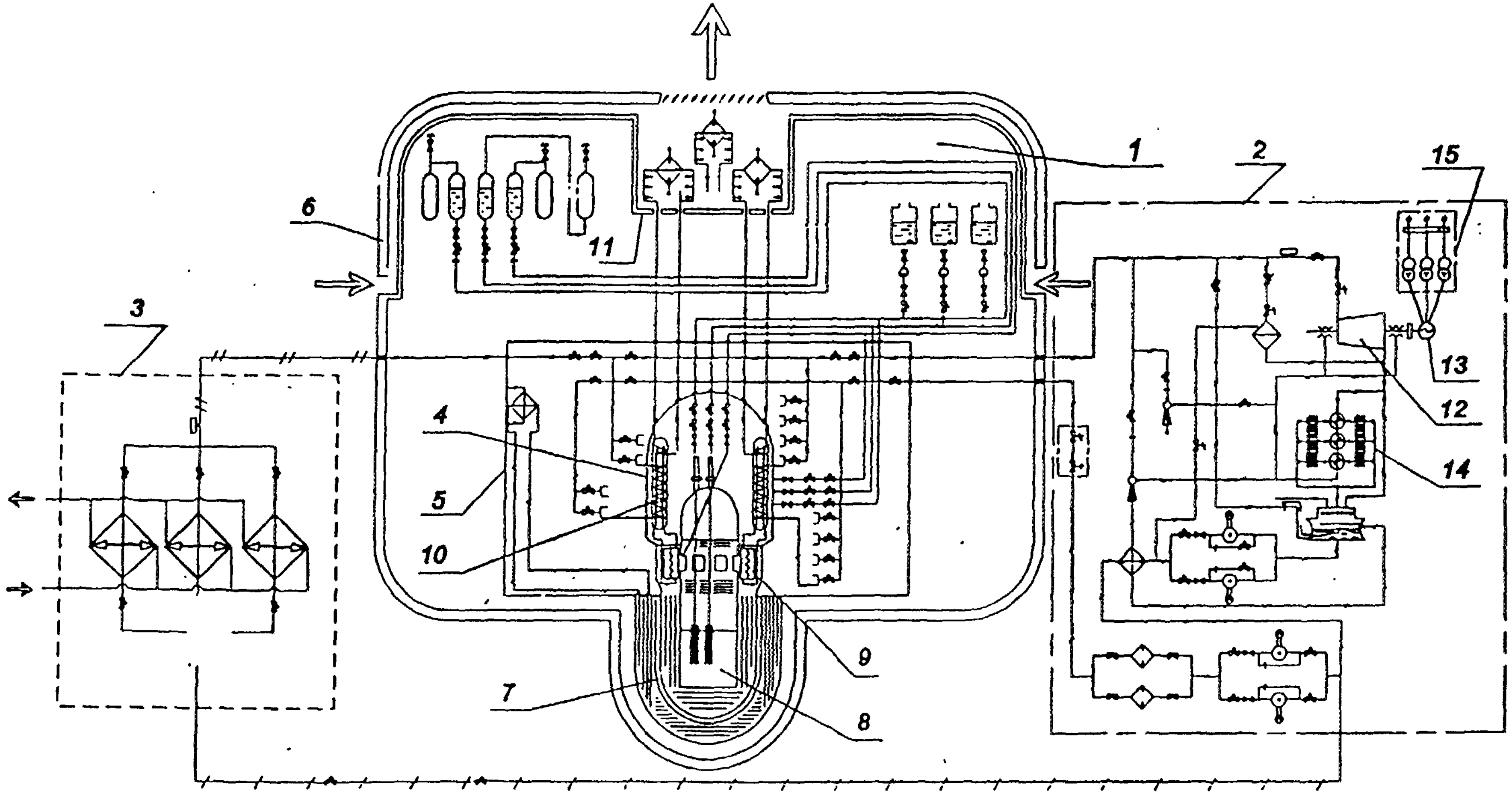

Fig.1. UNITHERM NPP Diagram
1 - Reactor unit; 2 - turbine unit; 3 - heat supply unit; 4 - safeguard housing; 5 - contaiment ; 6 - shock-proof casing; 7 - steam generating unit; 8 - core; 9 - intermediate circuit heat exchanger; 10 - steam generator; 11 - emergency cooling system heat exchanger; 12 - turbine; 13 - generator; 14 - air cooled condenser; 15 - transformer unit

2. Selection of thermal and technical parameters of a nuclear power plant (NPP). Integrated nuclear steam generating unit design and its main components

The UNITHERM NPP thermal-hydraulic cycle (Fig.1) includes three interrelated process loops, the last of which accommodates all heat consumers (turbine-generator unit and heat system or process steam boilers).

Selection of coolant parameters for the above-mentioned process loops was based on the proven range of operating pressures and temperatures typical for NPP water-cooled reactor primary circuits, and on the experience with mobile NPP operating with primary coolant natural circulation Besides, variation limits of the primary coolant parameters without reactivity compensation of nuclear fuel burnup by control rods were taken into account On the other hand, selection of coolant operating temperatures and pressures for the intermediate and steam-turbine loops was greatly influenced by the conditions which could provide acceptable efficiency and orientation on the development and operating experience with steam turbine units that might be considered as existing prototypes. The analysis of thermal characteristics of steam turbine units of such type enabled selection, taking into account the above reasons, of coolant parameters for the process loop of heat consumers

Transport of heat from the primary circuit to heat consumers during the phase change in the intermediate loops reduces the required coolant flows and increases pressure in the natural circulation system, while the desire of optimal distribution of the available temperature difference between the coolants of the primary and third circuits determines the intermediate loop coolant parameters Considering the above reasons, the parameters of the UNITHERM NPP process loop coolants are as in Table 1 below.

Table 1

The UNITHERM NPP process loop coolant parameters

Parameter	Value
Primary coolant parameters (high-purity water of NPP primary coolant quality)	
pressure, MPa	16.0 16 5
core inlet temperature, °C	245 225
core outlet temperature, °C,	325 305
Intermediate loop coolant parameters (water of NPP secondary coolant quality).	
pressure, MPa	3 0
temperature, °C	234
Heat consumer loop coolant parameters (water of NPP secondary coolant quality)	
pressure, MPa	1.0 . 1 2
steam temperature, °C	207 210
feedwater temperature, °C	45 60

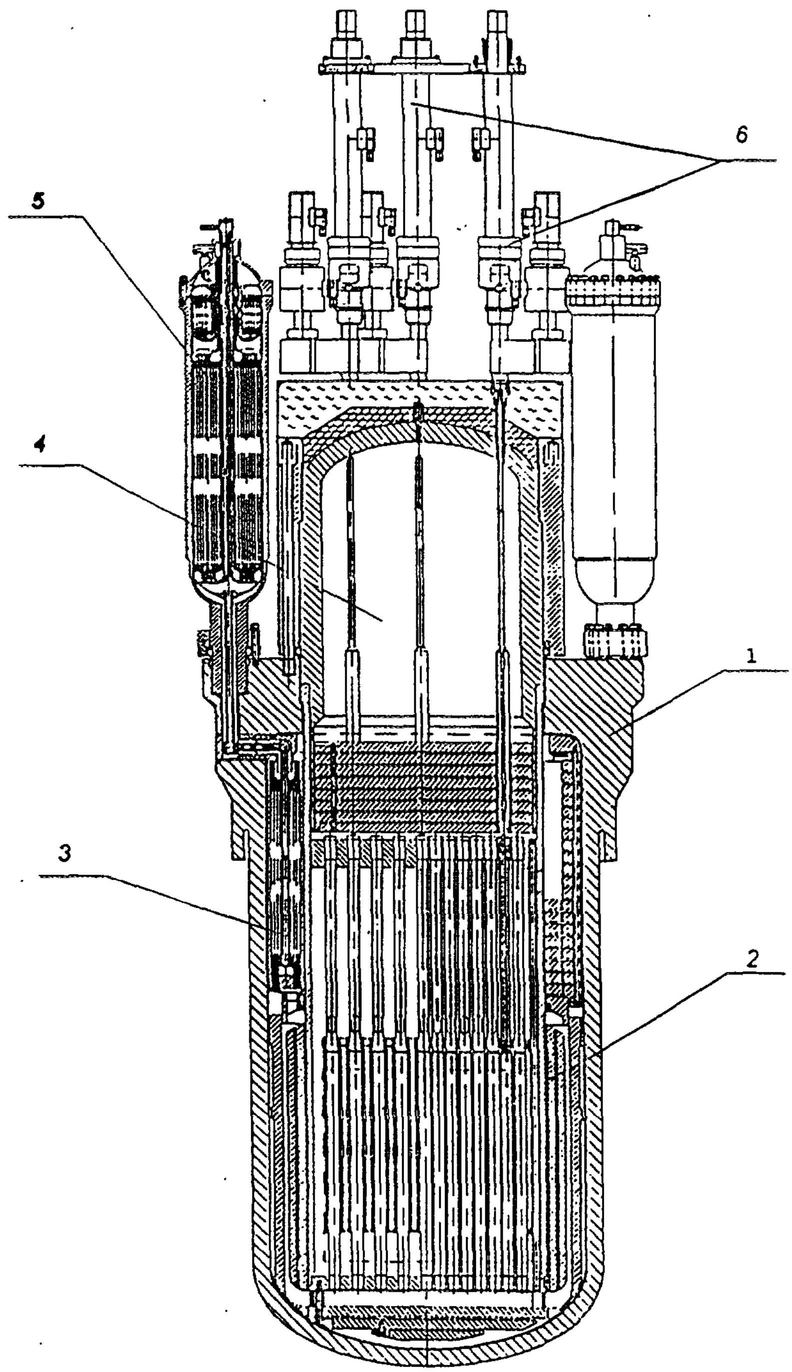

Fig.2. Nuclear Steam Supply System
1 - Vessel; 2 - core; 3 - intermediate circuit heat exchanger;
4 - pressurizer; 5 - steam generator; 6 - control rods drive

The proposed UNITHERM NPP has been designed to employ integrated water-cooled NSSS as a heat source (Fig. 2). NSSS combines in one vessel the main primary circuit components - core, steam generator (SG), pressurizer, control and protection elements. This allows to avoid primary circuit pipework, reach extremely compact location of ionizing radiation sources and potentially dangerous working fluid - primary coolant. The NSSS design ensures core cooling and heat supply to steam generator by convection of primary coolant and thereby allows to eliminate forced circulation. Such approach to the design of the NPP main component - nuclear reactor - is necessary to reach maximum possible reliability and simplicity due to absence of active elements with continuously moving mechanical parts. The group of absorbers with associated drives, which perform the function of emergency protection and compensation for reactivity variation, is a single movable element. The absorbers are displaced once during NPP continuous operation when starting NSSS unit in normal operation. In emergency situations it is possible to drop the absorbers which perform the function of emergency protection.

The integrated NSSS vessel made of 15X2MFA-A steel with corrosion-resistant cladding consists of the shell with welded elliptical bottom and flange.

The central part of the vessel accommodates the removable shield with the core, lattices of absorber rods and devices of additional emergency protection. The thermal shield which also functions as core reflector and radiation and thermal shield, is arranged around the removable shield in the bottom section of the vessel.

Inside the removable shield above the core there is a cavity with pressurizer equipped with a set of flat screens serving as upper radiation shield.

In the annular space between vessel and removable shield the steam generator tube system banks are located. Intermediate circuit heat exchanger is heat exchanger with coil-type heat transfer surface, where primary coolant moves in the tubes while the intermediate circuit fluid moves in the intertube space. The heat transfer surface is made up by 24 banks of the same type, the shells of which are designed to withstand the primary circuit pressure. The banks are pairwise combined into 12 assemblies which are connected to 12 sections of the intermediate circuit heat exchanger.

Steam collecting headers and water distribution chambers of intermediate circuit heat exchanger sections are located in the vessel flange area.The chambers are also connection points of SG sections to the NSSS vessel. The shell of each SG section is a cylindrical vessel with welded spherical bottom with a coil-type tube system inside cooled by the coolant from the heat consumer circuit. In each section of the SG above the tube system the coil-type system of independent cooldown circuit is arranged. From the top the shell is closed with a spherical head.

In the choice of characteristics and structure of the core and its control elements the following priority concepts have been adopted:

- maximum possible reduction of operative reactivity margin, in particular, the fraction of total efficiency of shim elements, per group of absorber rods with individual drive;
- optimal power, coolant temperature and fuel feedback factors;
- specific power density (about 15 kW/l) which guarantees specified long-term operation without fuel element clad leaking, and minimum specific levels of residual heat release for reliable heat removal in severe accidents;
- increased reliability of emergency core chain reaction suppression system by using in the control system of additional passive emergency protection channels with operation mechanism differing from that of main functional components.

To reduce core overall reactivity the adopted design philosophy excludes withdrawal of control elements from the core in power generation mode during continuous NPP operation. In this period, the core operates in self-control mode due to variation of coolant average temperature and absorber burnup.

The additional emergency protection actuator is a structure with leaktight vessel, which by its lower part enters the core instead of one fuel assembly, while the upper part is flange connected to the NSSS vessel head. Inside the actuator vessel, actuating element consisting of the accumulator with absorber and interconnected receiving chamber made of two elastic membranes. The space between the vessel and the actuating element is filled with nitrogen, control fluid. Gas (He-3 or boron trifluoride) is used as absorber.

When there is no emergency signal, the membranes of the receiving chamber are under control fluid pressure and the absorber is displaced into the accumulator. The emergency signal activates the electromagnetic switch and control fluid is discharged to the NPP containment volume. By its pressure in the accumulator the absorber is displaced to the receiving chamber spacing the membranes apart. To return additional emergency protection to initial position, the control fluid from the tank outside NSSS vessel is fed to the actuator which compresses the membranes and displaces the absorber to the accumulator.

3. *Design Safety Aspects*

In accordance with modern NPP safety approaches, radiation exposure on personnel, population and environment in normal operation and design-basis accidents should not lead to excess dose rates for people, and in beyond design-basis accidents, this effect should reasonably limited. To this end, technical and organisational measures are taken to ensure safety with any initiating event envisaged by the design with superposition of one failure independent of the initiating event of any of the following safety system element: active or passive element with mechanical movable parts or one personnel error independent of the initiating event. Besides one failure independent of the initiating event, it is necessary to take into account the nondetectable failure of elements affecting safe operation, which are uncontrolled in operation, and influencing accident propagation.

Safety of the UNITHERM NPP is achieved by a complex of technical solutions among which the following are worth mentioning.

The NPP employs water-moderated, water-cooled reactor with inherent safety features which reflect its capability of keeping safety on the basis of internal feedbacks, natural physical processes applying passive residual heat removal systems and automatic protection devices which ensure chain fission reaction suppresion without intervention of operator. The UNITHERM NPP is also capable of self-controlling chain fission reaction due to negative temperature, power and void coefficients of reactivity. The core physical characteristics are so selected that the above coefficients are negative in the entire range of temperatures during the core life both in normal and emergency operation modes. This eliminates spontaneous core power excursion in normal startup and heatup and stabilizes operation in steady-state and transient conditions when heat consumer circuit modes of operation change.

After NSSS is started up and brought to a preset load, moving of shim groups upwards is mechanically blocked thereby eliminating possibility of unauthorized introduction of additional positive reactivity.

The NSSS design is such that all potential leak initiation locations are in the top part of the vessel with limiting equivalent leak diameter sufficiently small and not exceeding DN 20. The integral layout of NSSS unit with rather efficient iron-water radiation shield between the core and wall of NSSS vessel excludes vessel brittle fracture because of metal neutron irradiation. All this allowed to exclude accidents associated with large and medium leaks, and prevent dangerous propagation of accidents due to core dryout. To this end, the containment (Fig. 3) is used designed for localization of primary leaks within the inner volume. The use of three-section liquid

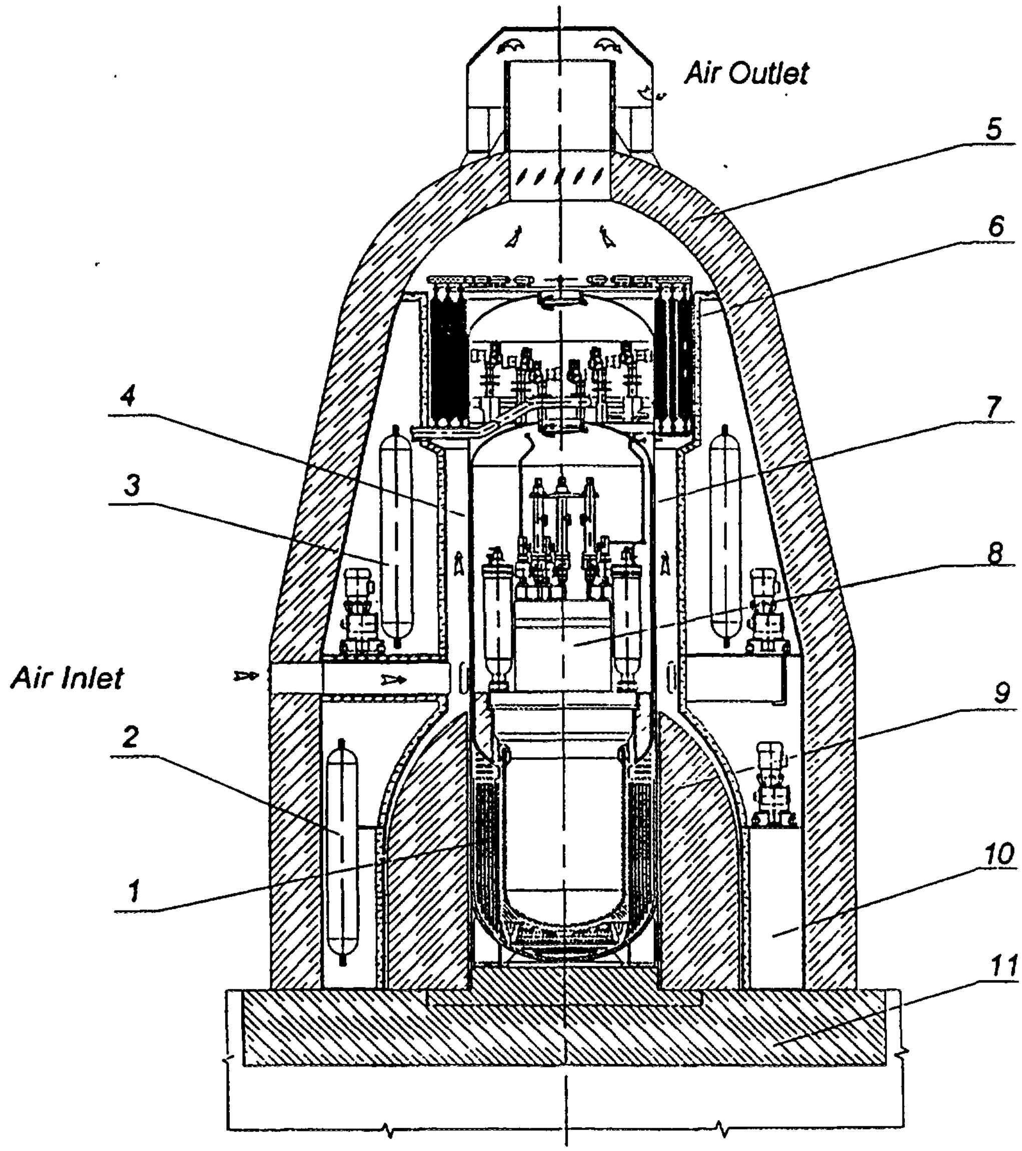

Fig.3. Reactor Plant

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

absorber feed system ensures flooding this part of the containment in the emergency situation under consideration with liquid medium up to the level above potential primary circuit depressurization points which completely eliminates core drying in any design initiating events or accident scenarios. Thus, as a maximum design-basis accident it is possible to consider the primary circuit leak of conventional equivalent size of DN 20 max. The estimates showed that propagation of such an accident follows the scenario typical for the small leaks in containment without deterioration of its leaktightness and core damage. This is contributed by reasonable emergency cooling core system (ECCS) redundancy and its passive principle of operation using no forced circulation means.

Incorporation into the UNITHERM NPP of additional localizing safety barrier - safeguard housing - enables even in the case of beyond design-basis accidents due to containment depressurization practicale to eliminate radioactivity release to the environment and risk of core drying. Beyond design-basis accident due to containment depressurization and postulated damage of 10 % of fuel elements and primary coolant and radionuclides discharge to the safeguard housing does not lead to significant radiation damage for population and individual exposure will not exceed 0.11 rem/y.

The UNITHERM NPP three-loop thermal-hydraulic design when consumers even with two consecutive interloop leaks can be reliably protected by reasonably redundant shut-off and cut-off localizing valves against radionuclides discharge to heat consumer circuit, and against harmful effect of ionizing radiation on personnel.Thanks to this, NPP personnel is beyond the area of ionizing radiation and the ionizing radiation rate on the NPP protection surfaces does not exceed the background values in normal operation. The dose rate of ionizing radiation in maximum design-basis accident 100 m away from NSSS is only by 10% above the background level.

Special attention has been paid to NPP UNITHERM emergency core cooling system (ECCS) which plays the important role in safety assurance. This was mostly due to the fact that because of inapplicability of traditional technical solutions we had to search for new ones taking into account not only general approach to NPP design, but climatic conditions of the NPP site as well.

The ECCS is designed as independent process loop associated with the intermediate loop. In emergency situations, the heat removed from the core via steam generators arranged within the NSSS vessel is fed to the intermediate loop,and further, through heat exchangers of the loop, is removed, via independent ECCS, to its heat exchange surfaces cooled by atmospheric air. The low winter temperature level in UNITHERM NPP application areas demands the selection of low-boiling coolant of the aforementioned independent loop of ECCS. To this end ammonia may be used.

Specific features of ECCS is that it does not have isolating and cut-off devices, i.e., the system is in continuous operation. Therefore, marked seasonal ambient air temperature fluctuations may greatly influence the amount of heat discharged through the system to atmosphere. So, in summer the capacity of heat removed through the system is 3-4 % of the nominal NPP heat capacity, the respective figures in winter may increase as high as 1.5-2 times. To reduce heat losses, the system of shudder-type gate valves is envisaged in the air duct. Switching of the gate valves from summer to winter operation is made during NPP preventive maintenance. Apart from its main functions, ECCS provides the possibility to keep NPP in hot stanby, i.e., at minimum possible core power level when power take-off is stopped.

Another improvement in reliability and safety of the UNITHERM NPP is the passive nature of core protection system. During NPP operation under load variations, core power is self-controlled, whereas variation of reactivity during continuous operation practically compensated for by burnable absorber and temperature effect, and only once a year reactivity is adjusted by remote relocation of absorber rods.

Emergency scram of NSSS and the core subcriticality is achieved by insertion of absorber rods in the core by motor-operated drives or by gravity and compressed spring energy in case of de-energizing of drives. Shutdown of NSSS with malfunction of the above absorber rods is ensured by using additional emergency protection based on alternative design philosophy. To prevent unauthorized withdrawal of control and protection system elements in commissioning the electromagnetic "arrestors" are provided in the drives limiting movement of absorber rods.

For quantitative evaluation of the UNITHERM NPP safety the following possible scenarios of severe accidents have been considered (unauthorized introduction of positive reactivity in the core, loss of preferred power, and primary circuit depressurization)and probabilities of final states of the following categories have been determined:

- the first category – accident localized without violation of safety limits;
- the second category – accident localized with partial deviations from safety limits and without core damage;
- the third category – accident localized with significant deviations from safety limits and accompanied by transition to core steam cooling which in the case of long-term accident can lead to partial core damage.

Predictions showed that core damage probability in any of the above-mentioned accident situations will not exceed 10^{-5} 1/y. In this case, probability of core damage with primary circuit depressurization is $6.7 \cdot 10^{-12}$, and with blackout is $5.4 \cdot 10^{-8}$ 1/y.

4. *Operation and maintenance*

The UNITHERM NNP design allows complete shop manufacture, assembly and adjustment with further shipment of 10-15 blocks of 100-175 t to the site where a small-scale erection and commissioning is required. The blocks may be shiped by barges or other vessels, and when unloaded - further transported by trucks and trailers. Upon life expiration, NPP is completely removed from the site to specialized enterprises for disassembling and utilization. The waste nuclear fuel may be either reprocessed or buried. Also buried are the reactor monoblock vessel and lower steel-water shielding components together with the containment lower fragment. Remaining metal may be reused after adequite treatment.

The erection works for the UNITHERM NPP will be carried out at the beforehand prepared site in minimum scope because transportable shop-fabricated blocks can be directly installed on the foundations designed according to future local plant operation conditions. Upon installation and fastening of the blocks on the foundations works are carried out to locate connecting pipes and cable, consumers are connected to district heating system and electrical networks, process loops are filled with working fluids and commissioning tests of the systems and components are made. The planned duration of the UNITHERM NPP erection works is 4-5 months since NPP blocks shipment to the site.

Location of the UNITHERM NPP on the site envisages combination of the entire complex into the following free-standing blocks:

- reactor plant;
- turbine-generator;
- air-cooled condenser unit;
- main control room and communication station;
- transformer substation, if necessary;
- auxiliary eqipment and plant systems rooms;
- NPP emergency auxiliary power supply room.

NPP is planned to operate continuously during 1-year periods. At that time the service personnel performs constant surveilance making no maintenance operations. The NPP staff is beyond the irradiation area both in normal operation and in the case of ultimate design-basis accident. Two persons work in each shift. The total number of persons required for current supervision over the UNITHERM NPP operation is 12-15 men for 5 shifts one of which is reserve to substitute operator on vocations, desease, etc. The staff qualification rerquirements depend on functional duties which in the process of operation are reduced to sufficiently simple operation of the equipment control and keeping constant communication with regional service center. All repair and preventive maintenance works are made by qualified specialists of part-time service team of 5-10 men, once a year during 1-2 weeks.

Analysis of the performed calculations for steady and dynamic NPP operation modes shows that accepted flow chart and design solutions for implementation of conceptual issues are right in principal. Parameters of circuit coolants are within permissible limits for the entire range of NPP load and outer conditions variation. Transition from one operation mode to another, including transition from full load shedding and restoration, are carried out due to reactor self-control and do not require personnel or reactor plant automatic equipment intervention.

5. *Conclusion*

Table 2 presents some consumer characteristics of the UNITHERM NPP designed for power supply of settlements and industrial enterprises. In case of necessity, heat may be extracted for district heating connecting the consumers to the mains,interconnected to the NPP intermediate loop. When the UNITHERM NPP is used for combined electricity and heat generation, the most optimal heat removal is within 20-30 % of the nominal thermal capacity of the NPP.

The estimates of required demand for power plants of the suggested design showed the figure of some tens in Russian Federation. This fact allows to plan a centralized regional service of such NPPs to ensure transport, erection and commissioning, preventive on-site maintenance and repair, and withdrawal from the site upon life expiration.

Thus, designing and evaluation activities carried out at RDIPE make it possible to suggest the concept of self-contained transportable NPP of small capacity to supply power to areas of Extreme North and Far East of Russia. Modifications of UNITHERM NPP are possible to suit operating conditions in other areas of the world. We expect that not only developed but also developing countries may become interested in such power plants.

Table 2

Basic UNITHERM NPP consumer characteristics

Characteristic	Value
Thermal capacity, MWt	from 7 up to 30
Electric capacity, MWe	from 1,5 up to 6 5
Number of integrated steam supply system	1 2
Refueling-free life, y	up to 20
Safety assurance	In accordance with the Russian Federation normative documentation
Number of protection barriers against radioactivity releases	5
Type of cooling of condensers of the turbine-generator unit and safety systems	Air No local water sources are required
Seismic resistance	8 points against MSK-64 scale Protection against radioactivity releases to the environments is ensured at 9 points
Service of NPP in operation	Supervision during continuous operation Preventive maintenance by a mission team once per year Personnel is beyond area of ionizing radiation (category B)

The authors are very thankful to RDIPE colleagues O G Gladkov E N Gol'tsev, A M Evdokimov, A I Loseva for their active and fruitful cooperation in developing the conceptual approaches which greatly contributed to the preparation of this paper

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IRIS: MINIMIZING INTERNAL ENERGY ACCUMULATED IN THE PRIMARY CIRCUIT OF AN INTEGRAL PIUS TYPE PWR WITH NATURAL CIRCULATION

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Abstract

This reactor concept is a development of the well known PIUS reactor.

The main IRIS (Integral Reactor with Inherent Safety) features consist in an integration of all the primary equipment into a pre-stressed concrete reactor vessel (PCRV), natural circulation in the primary circuit and a reactor design with free levels of the coolant and the cold borated water (linked through the gas pressurizer) instead of the upper density lock.

The large volume of the cold borated water in the PCRV provides not only passive shutdown the reactor in emergency (like it is in PIUS), but condensation of the vapor during accidents with the primary coolant boiling. The containment and safety systems may be considerably simplified.

The large scale PCRV makes possible to store inside the vessel all the burnt-up during the reactor lifetime fuel assemblies.

A thick borated water layer between the core and PCRV- walls allows decrease the residual PCRV activity upto an environmentally- acceptable level and to simplify reactor decomissioning.

The high level of safety makes possible siting of this reactor near population centres.

Introduction. The objective of this report is to illustrate conceptual advantages of the integral type PWR, called IRIS (Integral Reactor with Inherent Safety). The reactor concept is a result of development of well-known PIUS project ideas. This innovation establishes some new important properties.

Design features. IRIS design features are: integration of all the primary equipment into a pre-stressed concrete reactor vessel (PCRV); natural circulation in primary circuit; absence of the upper density lock. Principial scheme of the reactor is shown in fig.1.

Instead of an upper density lock there are communicating vessels with free levels linked through the gas pressurizer and single lower density lock. One of these vessels is the total primary circuit, and another one is a tank with cold borated water. Hence, despite natural circulation in them, this communicating vessels system will support the pure and borated water interface in the density lock inherently.

Only a primary leakage can destroy this interface and cause a proportional flow of borated water into the primary circuit through the density lock.

IRIS REACTOR SCHEME

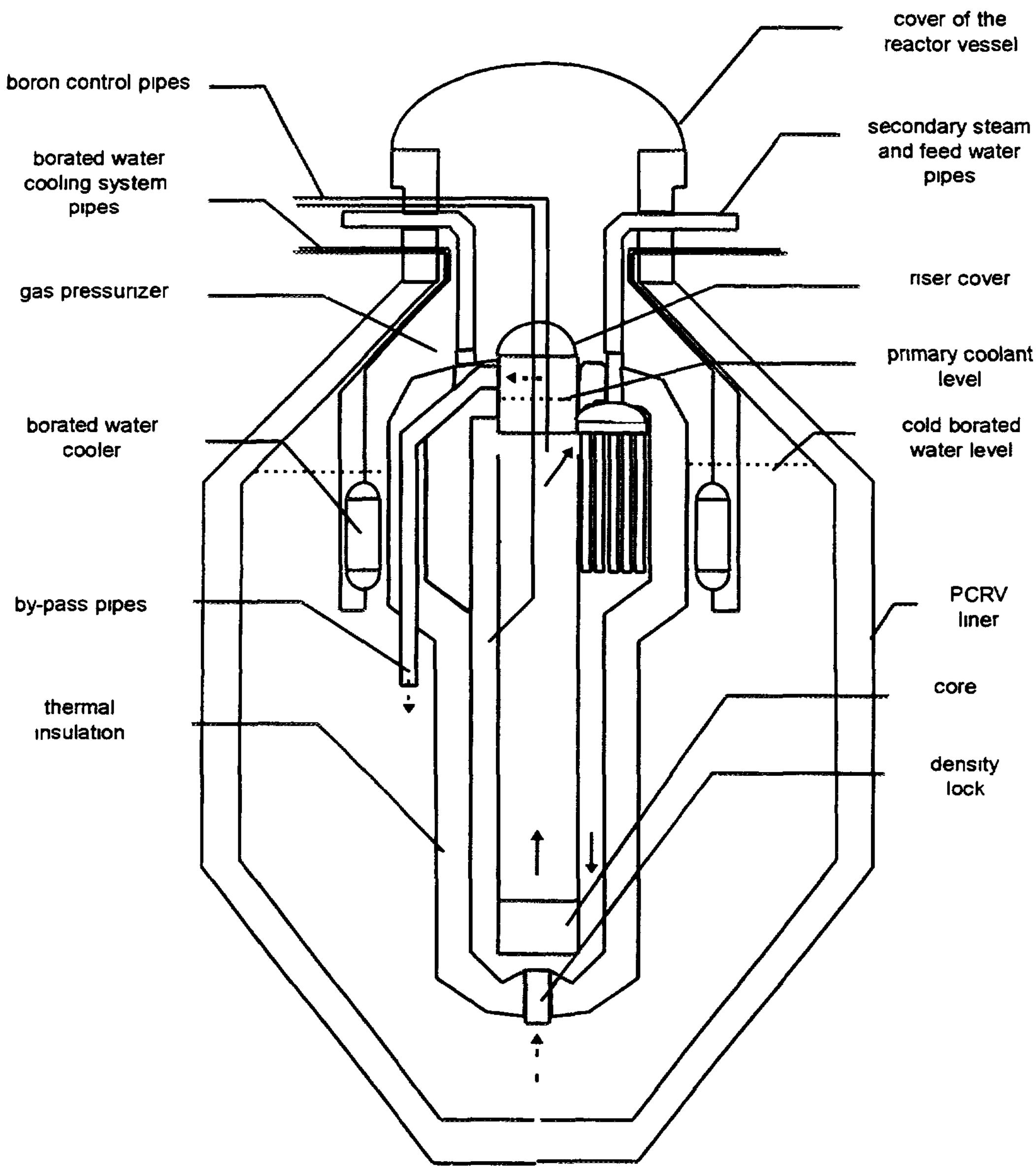

Fig 1.

To exclude the core overheating there are by-pass pipes from the primary circuit volume into the borated water tank. They are arranged above the level of primary coolant. If a coolant overheat was to happen, the level will rise up to the by-pass pipe's location and come to the borated water tank. Same portion of cold borated water will enter into the core through the density lock.

Therefore, due to the large scale reactor vessel, we have an inherent passive safety system, which is always ready for action, does not impede the normal operation, independent of anybody.

Minimizing of accumulated energy.

The large amount of cold borated water in the PCRV may be used for condensation of the vapor during accidents with primary coolant boiling due to overheating or depressurization by leakage. For this purpose, the following design modifications were performed:

-the riser was locked by a cover;

-the outlets of by-pass pipes were located several meters below the level of cold borated water;

-the cross-section of by-pass pipes was increased up to 1 m^2.

If depressurization or overheating the coolant occurs, these innovations cause the steam-water mixture to pass through the layer of cold borated water.

There are three types of probable leakage:

- destruction of the boron control pipe (∅50mm) outside the reactor vessel;
- gas pressurizer leakage due to destruction of PCRV cover design or double (inside and outside the vessel) destruction of steam generator section leg pipe (∅200mm);
- double (inside and outside the vessel) destruction of the cooling system pipe (∅50mm) under the cold borated water level.

Therefore, large primary leakage is a leakage from the gas pressurizer only. If this leakage occurs, the reactor design will keep the coolant inside PCRV. The steam-water mixture will be squeezed through by-pass pipes into the volume of cold borated water and condensed there.

In the case of very large leakage, when the depressurization speed is very high, there is a probability of squeezing the steam-water mixture through the density lock too (according to a hydraulic resistance of the each channel). During this process full reverse flow in the core is possible. It may cause overheat of the cladding. However an extended cross-section of by-pass pipes allow to pass all the bulk of boiling coolant during several seconds.

Large primary leakage. The scenario of this accident is the following:

-depressurizing of the cold borated water and degasing;

-propagation of the depressurization wave into the primary coolant via the by-pass pipes, penetrations in the riser cover (from the top) and the density lock (from the bottom);

-boiling of the primary coolant;

-raising of the primary coolant level up to the by-pass pipes' position due to coolant density reduction;

-probable stopping of the primary circulation;

-squeezing the steam-water mixture through the by-pass pipes and density lock (according to a hydraulic resistance of the each channel) into the volume of cold borated water and it's condensation;

-reactor shut-down due to negative fuel and coolant temperature reactivity feedback;

-restoration of the primary circulation;

-compensation of coolant losses by cold borated water;

-pressure reduction down to atmospheric;

-reactor's heat removal by steam generators.

Small primary leakage. The most probable reason for small primary leakage is a external destruction of the boron control pipe (∅50mm), which opens into the primary coolant. There is slow depressurization of coolant during this process. The primary coolant level in the riser comes up to the by-pass pipes location, going into the cold borated water volume and condensing there. This process accelerates the depressurization and decreases the coolant outflow. Coolant losses are compensating by cold borated water flow through the density lock. It results in shutdown of the reactor. When the coolant pressure decreases to atmoshperic, the coolant outflow will be stopped. The residual amount of water is enough for the heat removal process.

The probability of leakage via pipes of the borated water cooling system is very negligible: it can be caused only by simultaneous breakage of the pipe inside- and outside the vessel. Breakage in a single place will be detected: there is a checking of pressure in the cooling system. Large losses of cold borated water are excluded by the design: the borated water coolers are located in the upper part of the volume. Therefore, loss of borated water will be limited by the volume above the damage position. When the level of borated water reduces below the rupture location, outflow of borated water will be stopped. The accident will be continued with the same scenario, described for the large leakage. The rate of depressurization will be slower. There is no stoppage of the primary circulation during this accident.

It is apparent that the containment and safety systems may be considerably simplified due to innovation in question.

Decomissioning. Another IRIS feature is a thick (about 3m) borated water layer between the core and PCRV- walls. It allows decrease in PCRV activation significantly. It makes possible decrease of the residual PCRV radio-activity up to an environmentally acceptable level and to simplify reactor decomissioning.

Radioactive waste storage. The large scale PCRV may be used as a long-time storage facility for burnt-up fuel assemblies. Hence, there is no radioactive waste with high activity outside the reactor vessel.

Siting near population sentres. The high level of safety (core melt probability about 10^{-8} per reactor*year for PIUS) makes possible siting of this reactor near population centres.

THE LIGHT WATER INTEGRAL REACTOR WITH NATURAL CIRCULATION OF THE COOLANT AT SUPERCRITICAL PRESSURE V-500 SKDI

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Abstract

Pressure increase in the primary circuit to above the critical value gives the possibility of constucting the V-500 SKDI (500 MWe) lightwater integral reactor with natural circulation of the coolant in a vessel with a diameter less then 5 m. The proposed reactor has a high safety level, simple operability, its specific capital cost and fuel expenditure being lower than a conventional PWR. The development of the V-500 SKDI reactor is carried out taking into consideration verified technical decisions of current NPPs on the basis of Russian LWR technology.

1. Introduction

Nowadays together with improvement in the conventional NPP designs some organizations are developing reactors characterized by fewer potential accidents initiators and higher level of inherent safety. Integral PWRs may be considered as an example of such new generation reactors. These reactors can satisfy the safety principles formulated in [1], especially in the case of coolant natural circulation.

However their high specific capital cost may be the reason for their noncompetitiveness with other power plants. It is connected, first of all, with low specific power in such reactors, because of the difficulty of placing the steam generator (SG) large heat transfer surfaces within a steel pressure vessel and providing necessary coolant flow.

Under the constant heat transfer surface of SG and the secondary circuit parameters the amount of heat power of the reactor transferred from the primary circuit to the secondary one can be increased due to the temperature increase of the coolant in the primary circuit. However at subcritical water pressure the possibility of the departure from nuclear boiling hampers the increase of the reactor heat power.

The problems connected with the departure from nuclear boiling are eliminated at supercritical pressure as at supercritical pressure (SCP) the liquid is of one phase through all the range of temperatures. This property also simplifies the provision of the stable coolant circulation.

Increasing the temperature of the coolant at supercritical pressure and corresponding increasing of the temperature difference between the primary and secondary sides gives the possibility of increasing by several times the reactor power.

2. Choice of NPP parameters

The application of supercritical pressure in both thermal and fast pressurized water power reactors with thermodynamic efficiency up to 44% was considered in many designs in the 60's. The main difficulties in the construction of such reactors were connected with the development of reactor materials required for work under high temperature conditions, for example, for the efficiency of 44% it was necessary to have a primary coolant temperature of about 550°C. The SCP LWR construction was postponed because of this problem and good perspectives for the development of LWRs with subcritical pressure.

The RSC Kurchatov Institute returned to the idea of the NPP development with SCP at the beginning of the 80's. After severe accidents at NPPs the accent was transferred from the increase of NPPs efficiency to enhancing the safety while keeping economic competitiveness. The RSC Kurchatov Institute has carried out studies which show that in the case of pressure increase over the critical value in an integral LWR all safety requirements for the future reactors can be met with economic characteristics at least on the level of NPPs of traditional layout.

The following four main requirements formed the basis of the development of the reactors at SCP carried out jointly by the RSC Kurchatov Institute and EDO Hydropress. They are:

- its safety level must meet safety requirements for the reactors of future generation,
- its economic characteristics must have advantage over other reactors,
- it must have potentially lower environmental impact,
- it should not depart significantly from existing LWR technology and take into consideration the utilization experience of the water at SCP in heat power stations.

It must be noted, that nowadays water at SCPs is widely used in fossil plants. The operation of these power plants has shown a high reliability of the equipment under these conditions.

Fig.1 shows water enthalpy and density at SCP 23.6 MPa (P_{cr}=22.1 MPa) versus temperature. The temperature at which the derivative of enthalpy versus temperature is maximum is denoted as T_m. For comparison physical properties of water at 15.7 MPa are given in Fig.1.

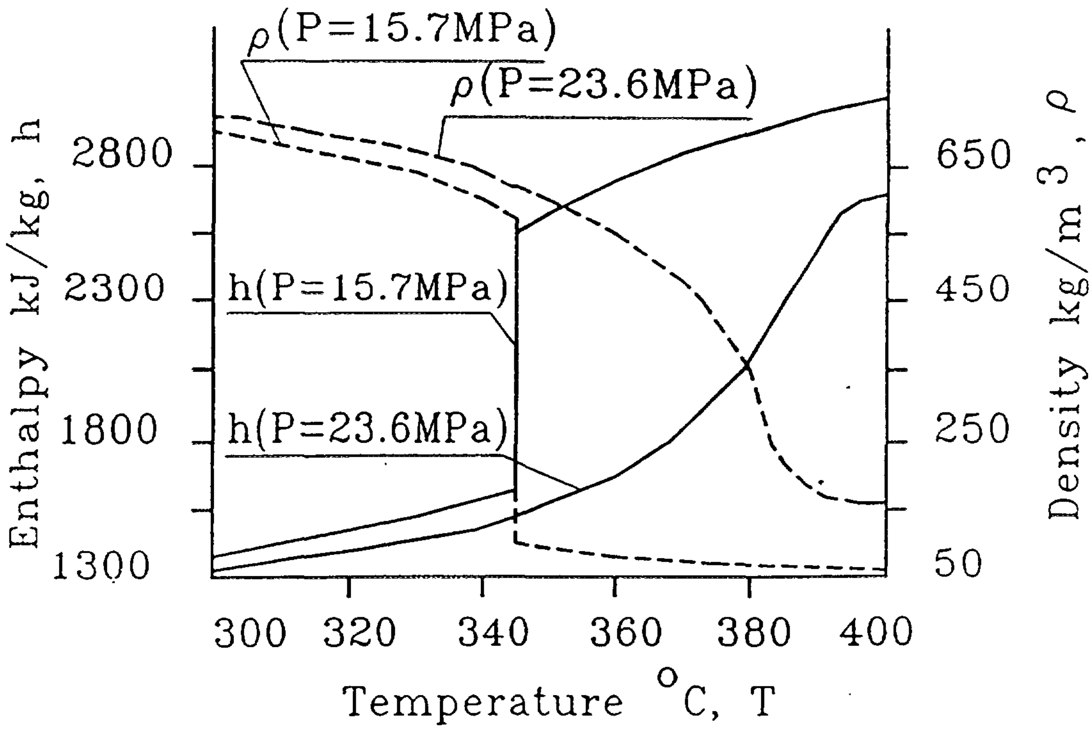

Fig. 1. Water enthalpy and density vs. temperature.

There are some correlations, e. g. V.Protopopov, V.Silin [2,3] for the calculation of heat transfer of SCP water flow inside the tubes and correlations for the conditions at which heat transfer deterioration was observed.

The large amount of heat transfer experimental data of SCP water flow in large bundles was obtained in the RSC Kurchatov Institute.

The experimental ranges covered are:

Pressure: 23.5 and 29.4 MPa,

Mass velocity: 350 to 5000 kg/(m^2/c),

Bulk water enthalpy: 1.0 to 3.0 MJ/kg,

Heat flux: 0.18 to 4.5 MW/m^2.

Experimental heat transfer was satisfactorily described by correlations obtained at SCP water flow in tubes as a resulting. Note that in the experiments conducted in the RSC Kurchatov Institute there was no heat transfer deterioration in the experiments with multirod bundles within the same test parameter range at which heat transfer deterioration occurred in tubes. Available information on supercritical heat transfer and hydrodynamics allowed reliable estimation of thermohydraulics characteristics of the core, SG and primary loop.

The value of primary operation pressure (P_0) was determined from the requirement to maintain the supercritical pressure during transients taking into account pressure support system features. The minimal margin relative to critical pressure P_{cr} was adopted equal to 1.5 MPa which defined the P_0 value equal to 23.6 Mpa. A greater pressure margin would cause increase in vessel mass and consequently would worse the technical and economic characteristics.

The coolant temperature was chosen on the basis of the following ideas:

- thermodynamic efficiency increases with the temperature but if it is above the T_m value it is more difficult to supply the NPP with necessary materials and the water properties of as a coolant become worse,
- to improve the fuel cycle features and safety level a decision was made to use the sharp change of the coolant density in the vicinity of T_m temperature to maintain the core criticality during fuel lifetime.

Taking into account these considerations the core inlet temperature was chosen below T_m and the core outlet temperature was chosen close to T_m that is approximately 380°C.

To maintain the core criticality between refuelings the coolant density increases smoothly during fuel lifetime. This is achieved by primary coolant temperature decrease in the SG which increases the feedwater flow at the given reactor thermal power. The growth of feedwater flow decreases the steam overheating in the SG and increases the heat transfer to the secondary side. As a result the primary coolant temperature decreases at SG outlet leading to the decrease of core inlet and core average coolant temperatures.

The chosen coolant parameters provide:

- growth of NPP efficiency up to 38%,
- several times decrease of SG heat transfer specific surface due to the increase of temperature difference between the primary and secondary sides as compared to an NPP with subcritical pressure,
- decrease of coolant mass flow due to the high coolant heat capacity in the region of T_m,
- significant difference of coolant densities at core inlet and outlet,
- use of a fuel cycle with high conversion coefficient and core criticality maintenance by varying the neutron spectrum during fuel lifetime due to the sharp density increase in the region of Tm temperature,
- high heat transfer coefficients in the core and SG,
- high average coolant heat capacity.

Taking these factors into account it was decided to develop the design of reactor of maximum unit power with natural circulation with regard to the capabilities of the existing reactor technology. The parameters of steam fed to the turbine at the chosen coolant temperature were adopted on the basis of optimization calculations. In these calculations the NPP efficiency increase with pressure and steam overheating and simultaneous decrease of average specific heat flux in the SG were taken into account. The decrease of specific heat flux in the SG requires the increase of SG heat transfer surface at given thermal reactor power.

3. Main technical characteristis of V-500 SKDI

V-500 SKID is an integral PWR in which the core and steam steam generators (SG) are contained within the steel pressure vessel, Fig.2. The

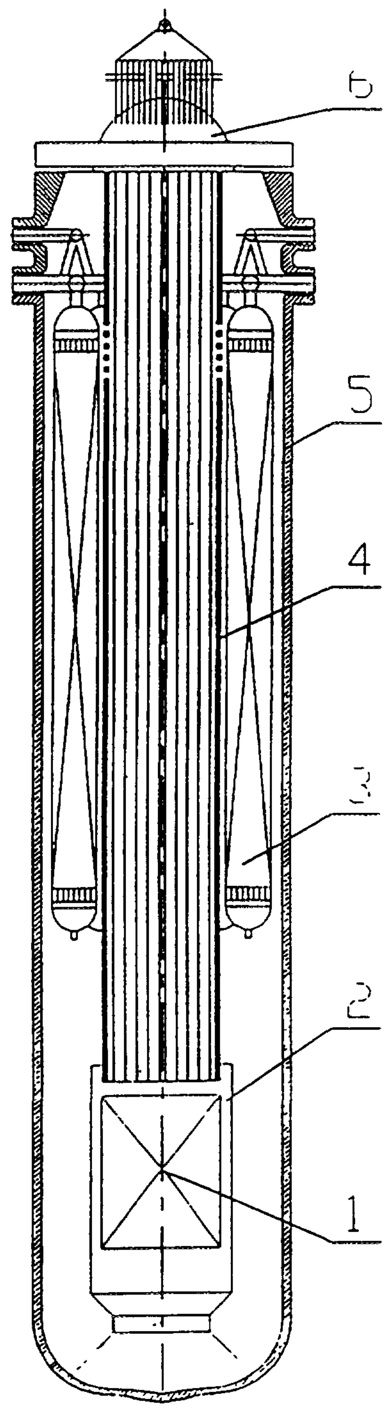

Fig. 2 V-500 SKDI reactor; (1) reactor core, (2) core barrel, (3) steam generators, (4) guard tube block shroud, (5) reactor pressure vessel, (6) reactor closure head

pressurizer is located apart from the pressure vessel. The guard tube block shroud is of 2.8 m in diameter and it separates riser and downcomer parts of the coolant circulation path. The hot coolant moves from the core through the riser and upper shroud windows into the steam generators located in the downcomer. The coolant moves due to the difference in coolant densities in the downcomer and riser. The pressurizer is connected, by two pipelines, to the reactor pressure vessel and the water clean up system. Injection of cold water into the pressurizer improves the pressurizer parameters.

3.1. Reactor pressure vessel

The possibilities of contemporary reactor pressure vessel (RPV) fabrication techniques were taken into account while choosing the reactor capacity. Due to the extreme weight of RPV ingots and the technique of their fabrication, the external RPV diameter should be less than 5 m. The RPV's height is 23500 mm, its external diameter is 4780 mm and its wall thickness is 330 mm. The electrical power of the reactor was found to be 515 MW.

The RPV is made of 15X2MΦA-A steel and the RPV flanges are made of 25X3MΦA-A steel. The upper part of the RPV has 6 nozzles of 350 mm diameter for the steam outlet, 6 nozzles of 200 mm diameter for the feedwater inlet and other nozzles are of 150 mm diameter.

3.2. Core

The core has 121 shroudless fuel assemblies, being on a spacing of 226 mm. 85 fuel assemblies have 18 control absorbing rod clusters and 36 fuel assemblies have burnable poison rods. Fuel assembly has 252 fuel rods arranged on a triangular lattice with 13.5 mm pitch. The fuel rod design is based on the VVER-1000 fuel rod design. Stainless steel is expected to be used as the fuel cladding material. The fuel height is 4200 mm.

The results of the V-500 SKDI core neutron calculations are listed in the table 1.

Main characteristics of V-500 SKDI core

Table 1

Name, size	Value
1. Core sizes (m)	
- equivalent diameter	2.610
- fuel length in cold state	4.200
2. Fuel rod cladding material	SS
3. Mean fuel burnup (MW day/kg U)	40
4. Mean volume power density in the core (MW/m^3)	68.2
5. Makeup fuel enrichment (%)	3.5
6. Assembly number in the core	121
7. Assembly number with absorber rod	85
8. Mean fertile coefficient	0.78

3.3. Steam generator

The steam generator (SG) is a once-through vertical heat exchanging apparatus arranged in the annular space between the RPV and guard tube block shroud. The SG consists of 18 modules which are joined into 6 sections. Each of the sections has an individual steam header and feedwater header, inserted through the RPV nozzles. The SG modules consist of titanium alloy tubes of 10.8 m in length, 12 mm in outer diameter and 1.3 mm in wall thickness, surrounded by a stainless-steel shroud. The guard tube shroud is freely installed on the core support barrel and after removing it from the RPV there is an opportunity to repair or to remove the SG section.

The main technical parameters of V-500 SKDI are listed in Table 2.

Main characteristics of B-500SKDI

Table 2

Characteristic	Beginning/ end of fuel lifetime
1. Thermal power (MW)	1350 / 1350
2. Electric power (MW)	515 / 515
3. Operation pressure at the core outlet (MPa)	23.6 / 23.6
4. Coolant temperature (°C)	
- core inlet	365 / 345
- core outlet	381.1 / 378.8
5. Core coolant flow, kg/s	2470 / 2880
6. Time period between refuelings (rated power) (year)	2
7. Fuel lifetime (year)	6
8. SG steam pressure (MPa)	10.0

4. Equipment accommodation

The reactor with the concrete well equipment, pressurizer system and emergency core cooling system (ECCS) is installed inside the hermetic guard vessel (GV), designed for 0.5 MPa internal pressure, Fig. 3. A part of the ECCS and the reactor water clean-up systems are arranged in the compartments, joined to the GV by pipelines.

The reactor, the refuelling water pool, the emergency feedwater storage tanks and other equipment is installed inside the containment.

5. Safety assurance

The inherent safety features of integral supercritical coolant pressure LWRs and the large margin of the main parameters permits enhancement of the safety level of such an NPP in comparison not only with the usual loop-type

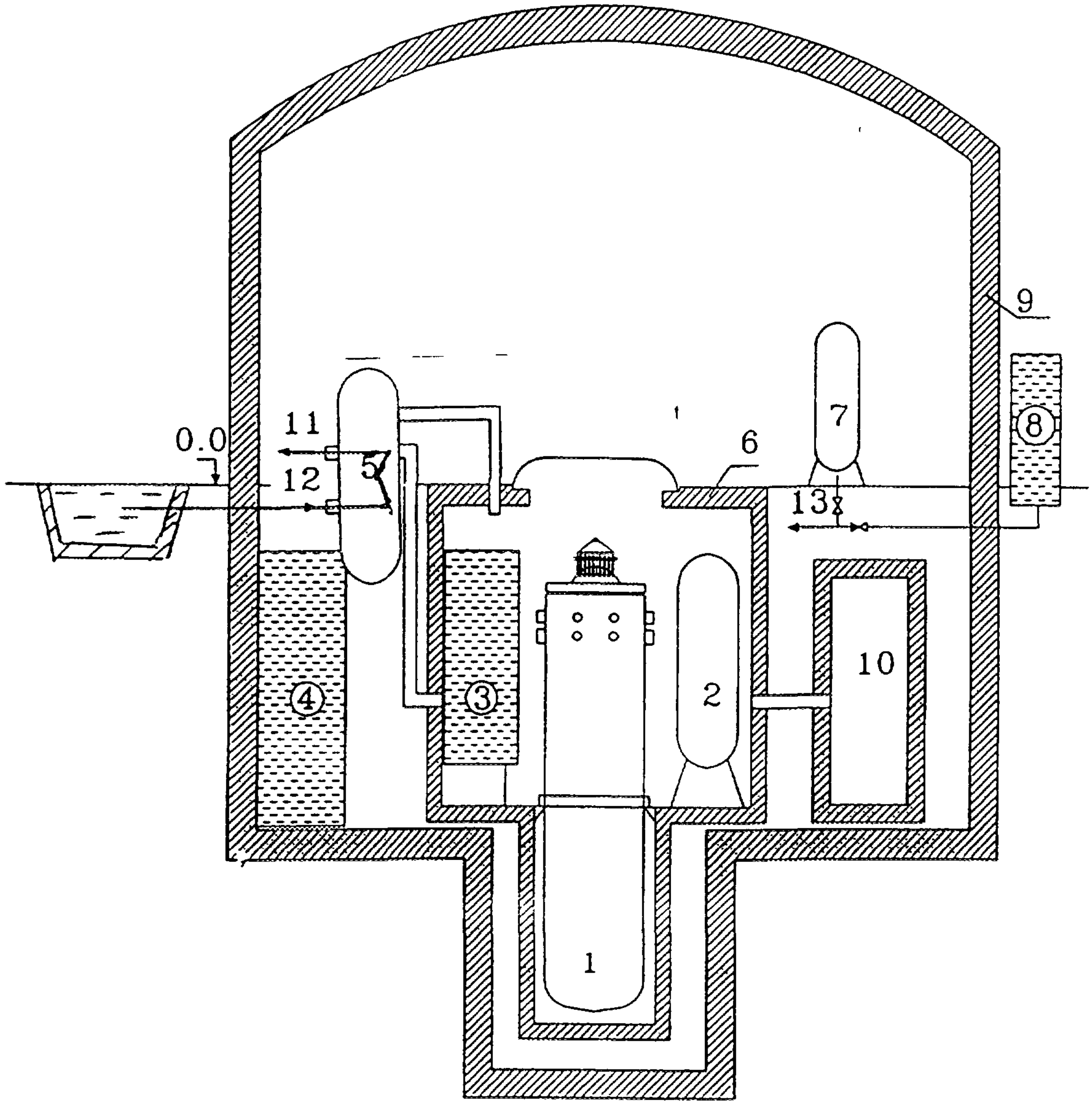

Fig. 3. General layout of the containment arrangements for the: (1) reactor, (2) pressurizer, (3) hydrotank, (4) spent fuel pool, (5) bubble-condenser, (6) guard vessel, (7) hydro-accumulator, (8) water storage tank, (9) containment, (10) water clean-up system compartment, (11,12) cooling water, and (13) into feedwater pipelines.

PWR, but even with a integral subcritical coolant pressure LWR. In comparison with the loop-type PWR, the integral-LWR with natural circulation of coolant has the following advantages:

- reduction of primary circuil length,
- elimination of the probability of rapid core dryout in case of primary circuit depressurization,
- elimination of losts of coolant flow accidents,
- reduction of the probability of coolant leak are from the primary circuit to the secondary one (all SG tubes and welds are under compressive loads),

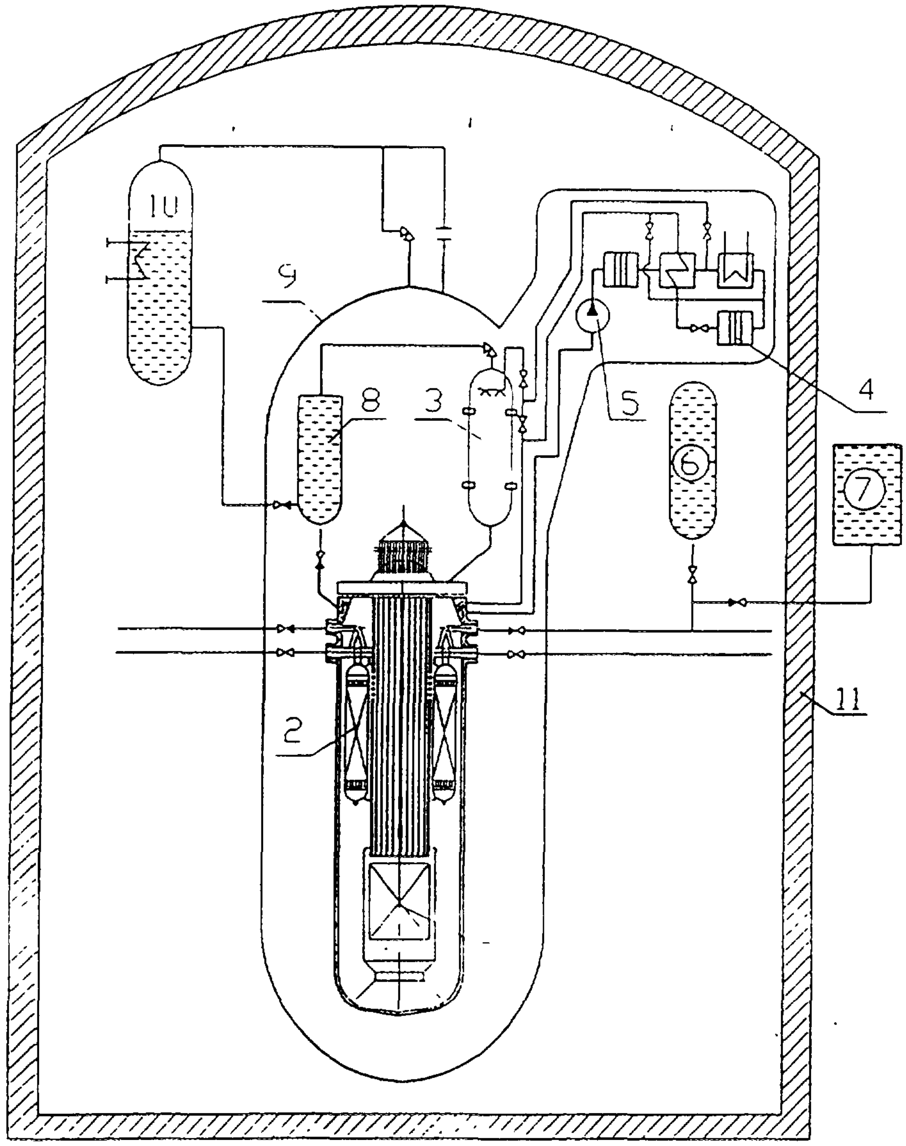

Fig. 4 Passiv heat removal systems: (1) reactor, (2) steam generator, (3) pressurizer, (4) water clean-up system, (5) pump, (6) hydro-accumulator, (7) water storage tank, (8) hydrotank, (9) guard vessel, (10) bubble-condenser, (11) compartment.

- self-regulaion of coolant flow through the fuel assemblies,
- localization of the reactor within a small volume,
- decrease of the neutron fluence to the RV due to a large gap between the core and the RV wall.

In addition to the above mentioned the supercritical coolant pressure gives the following inherent safety features:

- absence of the departure from nucleate boiling on the fuel cladding,
- highly reliable coolant natural circulation,
- reduced reactor specific thermal power due to an increase in efficiency,

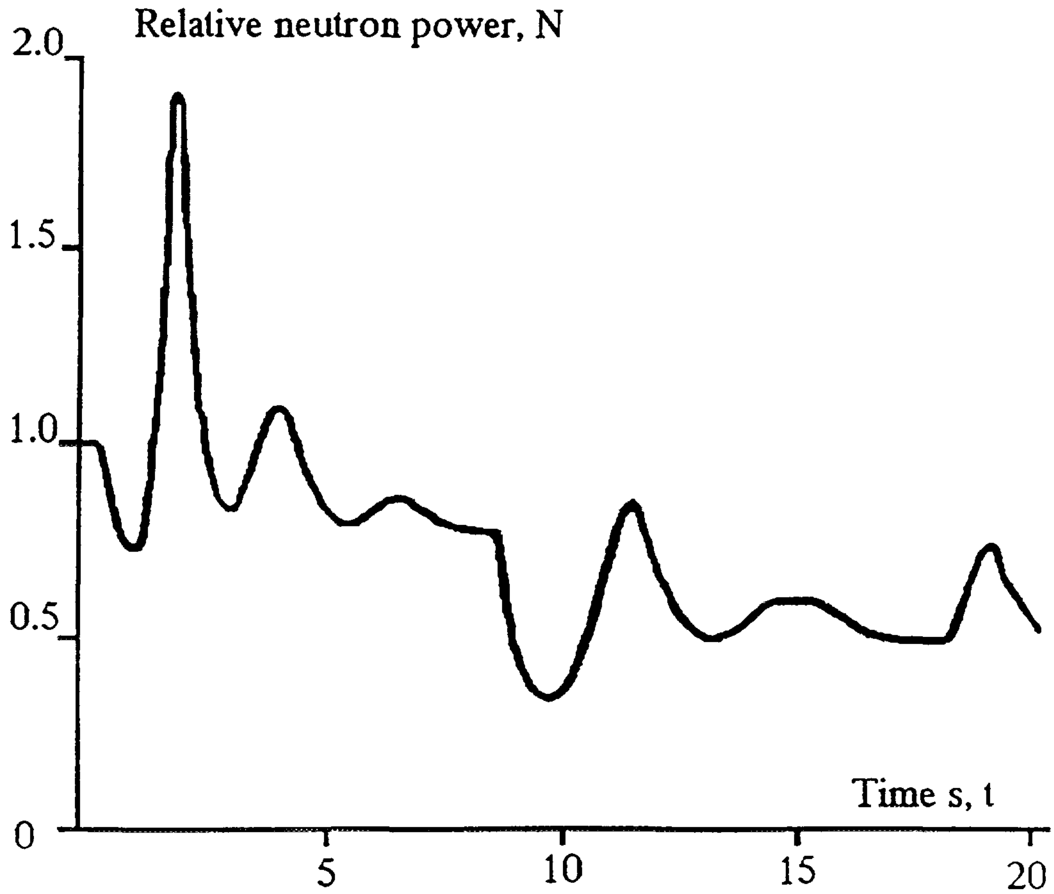

Fig 5 Neutron power change during the initial period of an accident caused by blackout without scram in conjunction with simultaneous main steam pipeline rupture.

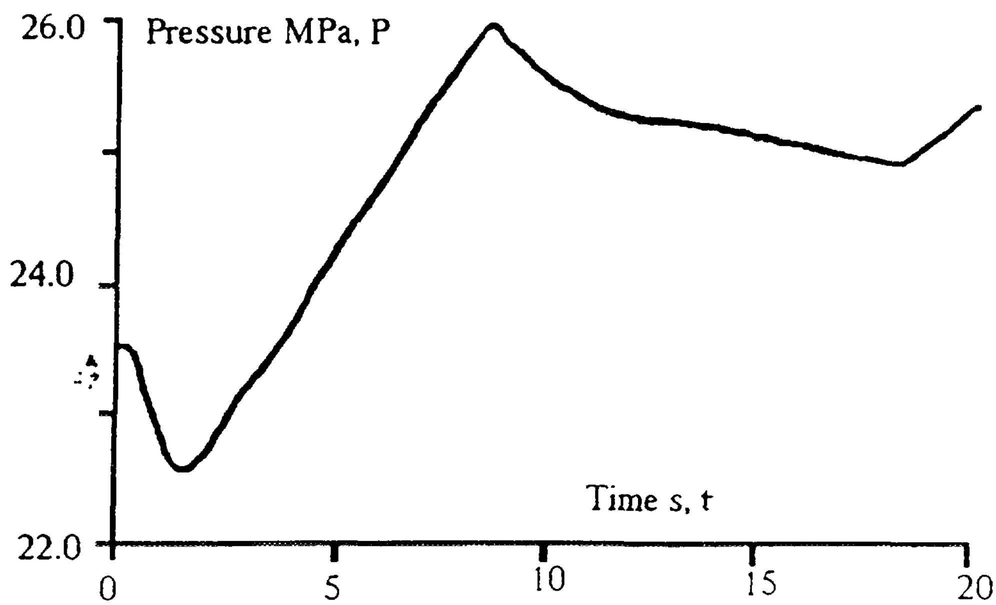

Fig. 6. Primary pressure change during the initial period of an accident caused by blackout without scram in conjunction with simultaneous main steam pipeline rupture.

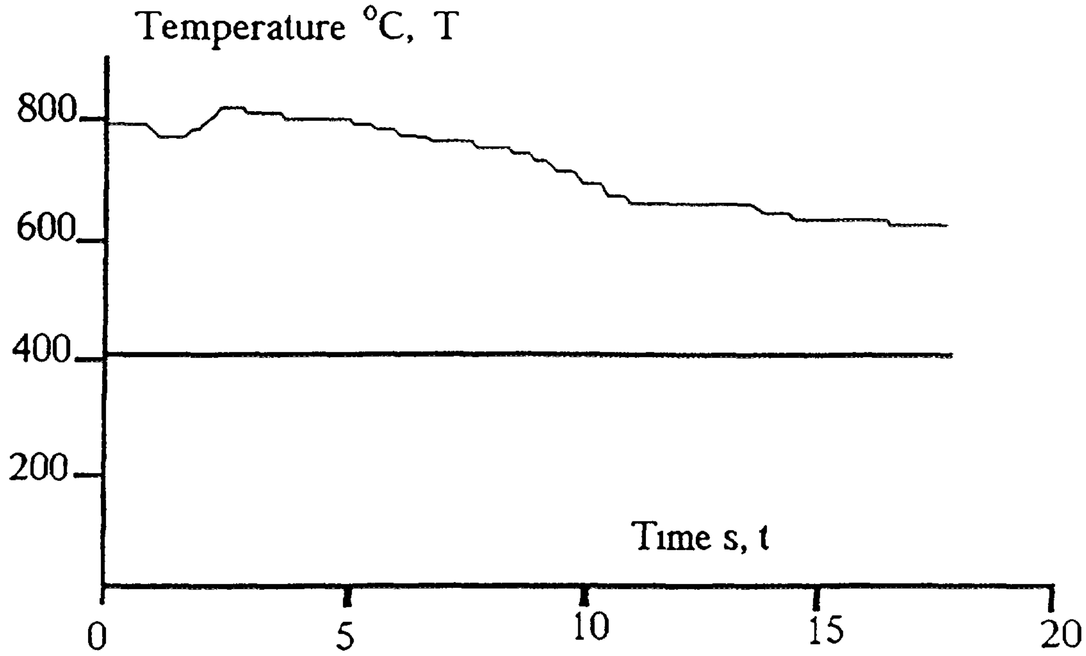

Fig. 7. Change of the fuel and cladding temperatures during the initial period of an accident caused by blackout without scram in conjunction with simultaneous main steam pipeline rupture.

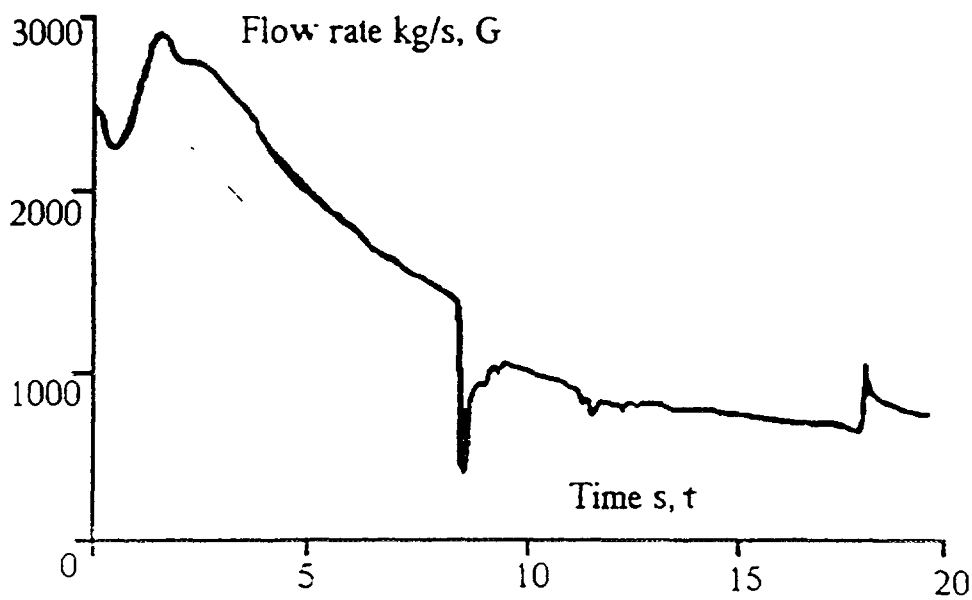

Fig. 8. Change of the coolant flow through the core during the initial period of an accident caused by blackout without scram in conjunction with simultaneous main steam pipeline rupture.

- reduced probability of coolant leak are from the primary circuit to the secondary one due to a large reduction of SG heat transfer surface,
- self-annealing of radiation defects in the RV wall due to its higher temperature level (350-380^{o}C).

The neutron spectrum shift core reactivity control gives the possibility to eliminate the following accidents:
- anticipated withdrawal of control rods under nominal conditions,
- unborated water ingression into the core.

Besides, the neutron spectrum shift reactivity control excludes a large increase of reactor power under all operational conditions.

The reactor has high a stability both to the NPP power and to core coolant flow deviations, which simplifies reactor control.

V-500 SKDI has passive and active safety systems which provide both separately and together planned emergency reactor cooling and decay heat removal during 48 hours in case of complete blackout. The core cooling passive systems consist of the heat removal system which removes heat through the steam generators and secondary circuit (PHRS2) and the primary circuit system (PHRS1), Fig. 4.

Estimations of severe core damage probability showed that for the V-500 SKDI it is at least four order of magnitude lower than for current NPPs under operation.

As an example let's consider a hypothetical accident of the VVER-500 SKDI reactor caused by a blackout without scram in conjunction with a simultaneous main steam pipeline rupture.

The calculation results, obtained with SKDI code developed by RSC Kurchatov Institute show, that reliable cooling of the core occurs even at this low probability accident (Figs. 5, 6, 7, 8).

The reactor power will tend to the level of residual power density in according with the water flow through the steam generator, which is determined by the performance of PHRS2.

6. Technical and economical characteristics of NPP with V-500 SKDI

At the chosen parameters the gross thermodynamic efficiency for V-500 SKDI equals 38.1% and the net efficiency equals 37.0%. The absence of main circulation pumps (MCP) reduces the auxiliary power requirement. The main equipment masses for VVER-1000 and V-500 SKDI are presented in Table 3.

Main equipment mass

Table 3

	V-500 SKDI	VVER-1000
1. Vessel (t)	930	330
2. Upper block (t)	150	158
3. In-vessel equipment (t)	175	170
4. Steamgenerators (t)	55	1288
5. Pressurizer (t)	260	214
6. Main circulation pumps (t)	-	520
7. Main circulation pipelines (t)	-	232
8. Safety tanks (t)	-	320
9. Total mass (t)	1570	3250
10. Specific metal expenditures per MWt(e) (t/MW)	3.25	3.45

V-500 SKDI capital costs relative to those of VVER-1000 will decrease by the reduction of the necessary expensive equipment (absence of primary piping, MCP, accumulators and outside SG). The use of spectral reactivity control gives the possibility of improving the fuel cycle in comparison with that in current LWRs. According to estimations the natural uranium requirements in V-500 SKDI will be 1.1 times less than in VVER-1000. The reduction of the number of equipment units and the plant layout simplification will lead to a decrease of concrete specific expenditures and construction costs. The proposed safety systems and the guard vessel prevent steam and fission products from being released into the containment volume. Consequently the main goal of the V-500 SKDI containment is the defense of the plant against external effects. This simplifies the containment design as compared with an LWR containment. Accordingly in estimates made for beyond design basis accidents the values of population exposure dose limits are not exceeded at a distance of 500-600 m from the NPP. This permits us to consider the opportunity of an NPP location close to large cities and its use for nuclear co-generation.

The factors enumerated above should lead to the improvement of V-500 SKDI technical and economical characteristics as compared with those of current middle and large sized nuclear power plants.

Alongside with development of the V-500 reactor design possible other ways of using water of supercritical pressure in nuclear power were considered. It seems promising to develop integral plants of small power for which the problems of reactor vessel manufacturing are simplified with the acceptable decrease in costs. For example, for a reactor of 250 MW (el) power the vessel OD is 3,5 m with the wall thickness being 240 mm.

From the ecological point of view it seems interesting to use the economically justified possibility, in the case of development of reactors with water of supercritical pressure, of the application of dry and wet water cooling towers to remove heat from the condensers. It will be especially important in the case of using reactors with water of supercritical pressure a for nuclear power-and-heating plants and its location in the vicinity of towns. With the thermodynamic efficiency of the cycle being of the order of 38% the specific amount of steam coming to the turbine is approximately 25% less than this value for WWER-1000. During operation of NPPs with reactors using water of supercritical pressure under the conditions of nuclear power-and-heating plant the steam flow rate into the condenser will reduce by 20% more.

7. Conclusion

Integral NPP with an electrical capacity of 500MW with natural circulating coolant may be created in a vessel with a diameter of less than 5 m with the use to a supercritical primary pressure. The V-500 SKDI safety level satisfies the requirements of the new generation to NPPs. The V-500 SKDI economic characteristics are not inferior to those of current NPPs due to the reduction of capital costs, improved fuel cycle characteristics, plant layout simplification and simpler operation. The NPP may be created on the existing Russian industry base without any significant change of modern LWR technology. It is necessary to construct a V-500 SKDI prototype reactor for testing of the accepted design and to make new decisions. The possible ways of using water of supercritical pressure in nuclear power have been considered.

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THE MRX INTEGRAL REACTOR: MAINTENANCE AND COST EVALUATION FOR SHIP APPLICATION

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Abstract

An advanced marine reactor, MRX, has been designed to be more compact and lightweight with enhanced safety. The reactor is an integral PWR with a water-filled containment vessel, in-vessel type control rod drive mechanisms and an emergency decay heat removal system using natural convection. These are adopted to satisfy the essential requirements for the next generation of marine reactors, namely, compact, light, highly safe and easily operated. The engineered safety is accomplished through a simplified system using water in the containment vessel. The LOCA analysis shows that the core flooding is maintained even when taking into account the ship inclination. To shorten the time of the maintenance and refueling works, a one-piece removal method is proposed. This method involves removing the containment vessel with its internals and replacing it in another one whose maintenance and refueling have already been completed. The economic evaluation of nuclear ships equipped with MRXs shows that some types of nuclear container ships will hold an economically dominant position over diesel ships in the near future, because of the environmental costs of diesel ships. R&D works have been making progress in the safety study of the thermal hydraulic phenomena, in-vessel type control rod drive mechanisms, the automatic control system, the nuclear ship simulator, etc.

1. Introduction

Compared to ordinary ships, nuclear ships are capable of long and continuous periods of voyage using high power without refueling. This advantage would contribute greatly to the diversification and expansion of sea transportation and ocean development in the future, as well as contribute to the survey and research activities on a global scale, especially in the Polar region. Another advantage is that no discharge of CO_2, NOx and SOx occurs during navigation, which helps to prevent environmental disruption due to NOx, SOx and the greenhouse effect due to CO_2. Especially in regards to NOx and SOx, the discharge quantity will be severely restricted even for ships in the future. The above mentioned characteristics strongly motivate us to develop marine reactors as an economical power source gentle to the natural environment.

On the other hand, the marine reactor should be compact and lightweight since it has to be installed in a narrow and limited space on a ship. Furthermore, a smaller marine reactor is more economical for a ship[1]. It must have high safety characteristics as well as easy operation and maintenance. It should also be possible to operate automatically to a great extent, since the operation must occur in the ocean environment without aid from land.

The Japan Atomic Energy Research Institute (JAERI) is conducting a design study on an advanced marine reactor, MRX, to obtain a more compact and lightweight reactor with enhanced safety[2]-[4].

The MRX is designed to use 100MWth for a reactor plant of an icebreaker scientific observation ship, but the concept could be applied to those of general commercial nuclear ships with wide output ranges of 50 through 300MWth

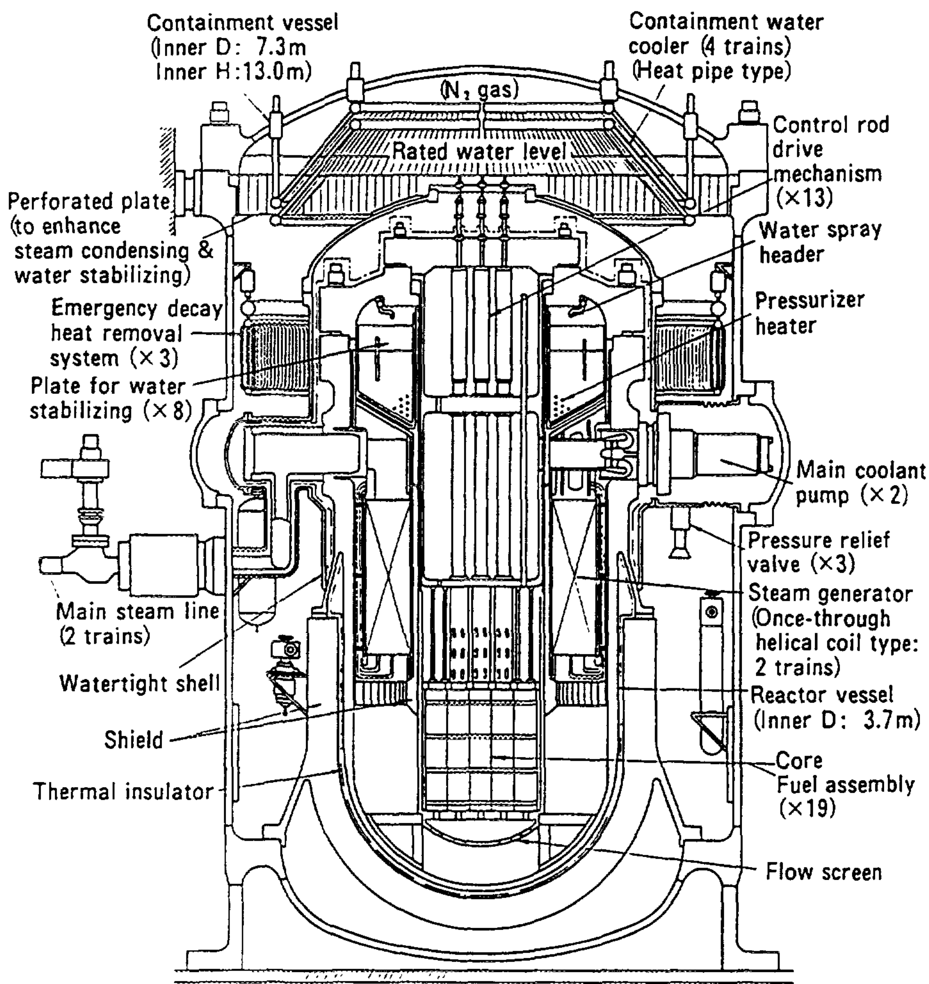

Fig.1 Concept of MRX plant

Table 1 MRX basic design description

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

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

depending on the type, size and velocity of ships. In addition to the icebreaker, the MRX series reactors will be favorably applied to high speed merchant ships, very large container carriers and super high-speed container ships, which need high power and long distance voyage. A view of MRX is shown in Fig. 1.

2. Design Features of the MRX

The improvement of the economy of reactor system depends strongly on the reduction of construction and operation costs. The construction cost can be reduced by means of making the compact, light and simple reactor system. The MRX adopts the following design features to be realized the above mentioned reactor system with highly passive safety : (a) Integral PWR. (b) In-vessel type control rod drive mechanisms. (c) Water-filled containment vessel. (d) Emergency decay heat removal system using natural convection. Table 1 shows the major specifications of the MRX.

(1) Integral PWR

Integral PWR could eliminate the possibility of large scale pipe rupture accidents, simplify the safety systems and reduce the dimensions of the reactor plant.

(2) Reactor core and reactor pressure vessel (RPV)

The core consists of 19 hexagonal fuel assemblies. The hexagonal assemblies, rather than rectangular ones, have been selected for reducing neutron leakage from the core and to operate with a small number of control rod clusters. The design conditions of the core specific to the marine reactor are as follows: (a) To maintain non-criticality ($k_{eff} < 0.99$) under the condition of normal temperature without use of a soluble poison even if one of the control rod clusters which has the largest reactivity worth is withdrawn from the core. (b) To operate the reactor with a sufficient power level for steerageway (≧30% of full power) even in the case that one of the control rod clusters which has the largest worth cannot be withdrawn from the core. (c) To keep enough residual reactivity (≧2%) for overriding Xe poisoning at the EOL. (d) The life time of the fuel assembly is 8 years with the plant factor of 50% (~23,000MWd/t). (e) The refueling frequency of 4 years with 52.6% of the fraction of the core withdrawn.

The fuel handling system is installed in land facilities. The average power density is sufficiently low (41kW/l) which shows that the core has enough margin for thermal reliability.

The RPV is relatively larger in size because of an integral PWR. This provides a larger primary water inventory with increasing the distance between the reactor core and the RPV, and reduces the neutron fluence at the RPV. The calculated value of the irradiation of fast neutrons is below $8x10^{15}n/cm^2$

($E \geq 1.11MeV$) at the inner-surface of the RPV for full power reactor operation of 20 continuous years.

(3) Control rod drive mechanisms (CRDMs)

The CRDMs are placed in the upper region inside the RPV to enhance the reactor safety with eliminating the "Rod Ejection Accident" and to achieve a compact reactor plant.

(4) Steam generator and primary circuit

The steam generator of once-through helical coil type is placed in the RPV diagonally upper region of the core. Two trains are adopted for the main steam and feed water lines. The vertical distance between the upper surface of the core and the bottom of the steam generator is selected to be 36cm to obtain a compact RPV, and an iron shield is installed between them to satisfy the design condition of the dose rate equivalent in the engine room.

The whole primary circuit is almost incorporated in the RPV except main coolant pumps, a volume control system and a residual heat removal system. The two main coolant pumps are placed in the hot leg at the upper cylindrical region of the RPV. The pressurizer is installed in the upper part of the RPV. For the maintenance and inspection, it is so designed that the reactor components in the RPV and the primary coolant pumps can be removed remotely and the steam generator tubes can be inspected from outside the RPV.

3. Safety

3.1 Safety concepts and safety systems

A large LOCA cannot occur in the MRX, since only small size pipes ($\leq 50mm$) exist in the primary system. The emergency water injection systems are not provided in the MRX. The engineered safety system consists of the water-filled containment vessel (CV) system, the emergency decay heat removal system (EDRS) and the containment water cooling system (CWCS). During a small LOCA, the engineered safety system keeps the core flooding and removes the decay heat without emergency water injection. Figure 2 shows a basic idea of the engineered safety system. The decay heat in a LOCA is transferred to the atmosphere by natural convection of the primary water in the EDRS, the CV water and the CWCS working fluid. According to the PSA, this engineered safety system has a high reliability. The probability of the functionally disordered trouble is $2x10^{-6}$ at first one month after starting the operation of this system. The residual heat removal system (RHRS) is not necessarily essential in the emergency core cooling system of the MRX. The RHRS is used for controlling the temperatures of water in the RPV and the CV for long term cooling after a LOCA and is also used for long term decay heat removal during a scheduled reactor shutdown.

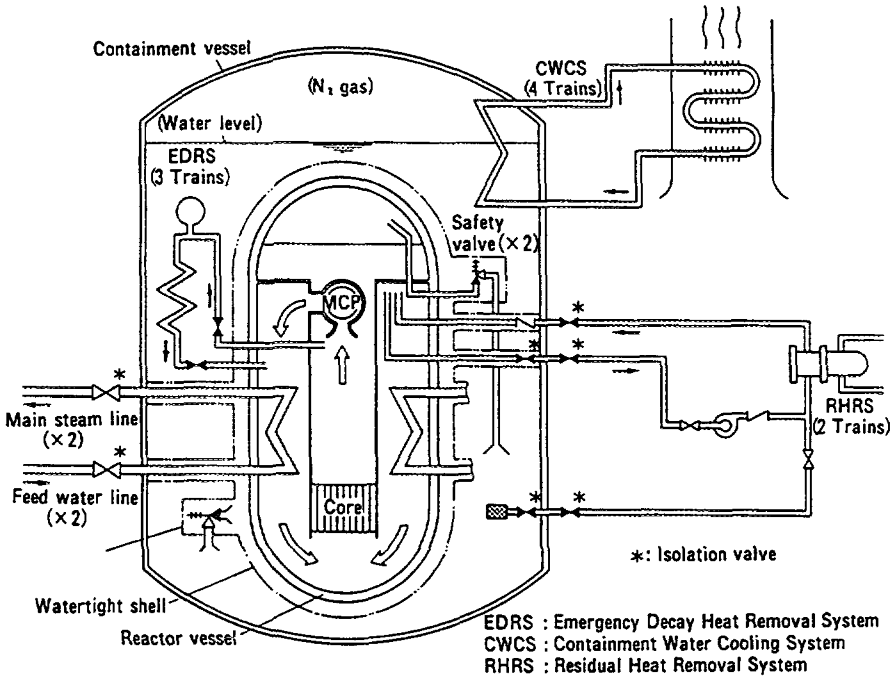

Fig.2 Engineered safety system of MRX

(1) Containment vessel (CV)

The design pressure of the water-filled CV is 4MPa to withstand a high pressure at LOCA. The size of the CV is designed to satisfy the requirements of radiation shielding and the maintenance of instruments installed in the CV. The RPV is surrounded with a watertight shell for thermal insulation. The shell can stand against the static pressure at anticipated transients. Pressure relief valves are installed to mitigate the rise of pressure in the space between the RPV and the watertight shell due to pipe breakage accidents in this area.

(2) Emergency decay heat removal system (EDRS)

The EDRS is a closed system which transfers decay heat from the core to the CV water. It includes three trains, each of which has a capacity to remove the core decay heat. Each train consists of a hydrogen reservoir tank, a cooler, two valves and piping. In the case of accidents, the valves of each train are opened actively by the signal of battery, then the primary coolant circulates by natural convection removing decay heat from the core and is cooled in the cooler placed in the CV water.

(3) Containment water cooling system (CWCS)

The CWCS is a heat pipe system for long term decay heat removal transferring the heat in the CV water to the atmosphere. It includes four trains. In the event of an accident, the water temperature in the CV will be kept lower than the design value by the arbitrary three trains operated using natural convection. For its working gas in the CWCS, anti-freezing gas such as

R22 ($CHCIF_2$) will be used taking into account of low temperature condition in ice-sea atmosphere.

3.2 Safety evaluation

Accidents of primary coolant loss, steam generator tube rupture, steam line break, feed water line break and total loss of electricity have been analyzed for safety evaluation of the MRX. Figure 3 shows typical results during LOCA obtained RELAP5/Mod2 calculation assuming the double ended guillotine break of 50mm dia. pipe occurred in the CV. Cooling by the EDRS, the CV water and the CWCS is taken into account, but the function of the RHRS is neglected. The maximum design values of pressure and water temperature in the CV are 4MPa and 200°C as mentioned in Table 1, and the allowable minimum water level in the RPV is 0.5m above the upper edge of the core, which is determined to keep core flooding taking into account of ship inclination and oscillation. Figure 3 shows that the maximum values of pressure and water temperature in the CV and the minimum water level in the RPV are 1.3MPa, 140°C and 0.8m, respectively. These satisfy the design conditions sufficiently. Through these analyses, it has been proved that the passive safety features applied to the MRX have sufficient functions in the safety point of view.

4. Decommissioning

For nuclear ships, it is essential from the economical point of view to shorten the time of the maintenance and refueling works. In operation of nuclear ships, the maintenance and refueling facilities as supporting systems are very important to make these works shortly, simply and safety. From this standpoint, the design study of a one-piece removal method is being carried out. This method is that the CV with its internals is removed for the maintenance and refueling, and is replaced to another one whose maintenance

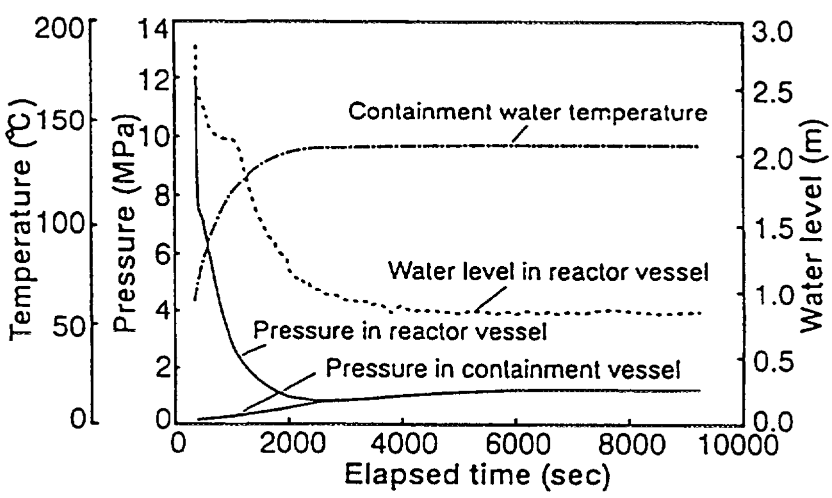

Fig.3 Typical transient of LOCA in MRX
(Double ended guillotine break of 50mm dia. pipe)

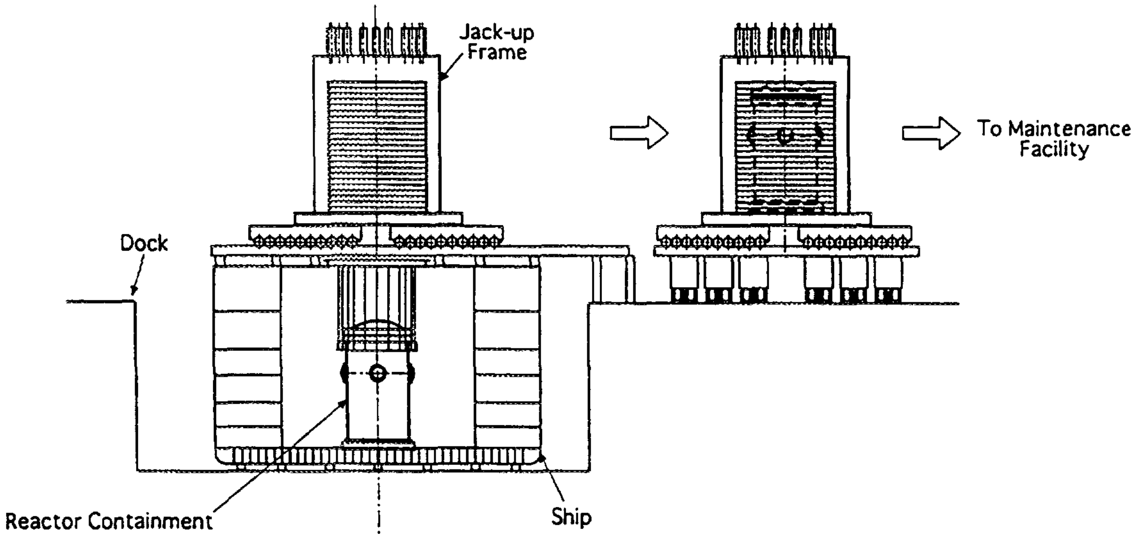

Fig.4 Concept of one-piece removal of CV with its internals

and refueling have already been completed. Figure 4 shows the concept of the one-piece removal of the CV with its internals. It is thought that this method is promising because the integral type marine reactor is relatively small and light. The merits of this system are: (a) To shorten the days required for the maintenance and refueling. Ships are required to stay in dock about 3 weeks for these works. (b) To carry out the maintenance and refueling in a large space of land facility with highly safe. (c) To reuse the reactor after the ship's life. (d) To make the decommissioning of the ship easy.

5. Economics and potential market

Whether or not nuclear ship has economic benefit is a fundamental matter to judge the social incentive to realize the practical use of nuclear ships. To evaluate the economics of nuclear ships, the construction and operation costs of nuclear and conventional ships have been studied and compared as parameters of the ship's speed, containers' capacity in terms of TEU (Twenty-feet Equivalent Unit), etc. Figure 5 shows the comparison of RFR (Required Freight Rate, $/TEU : Operation cost to transport one container) between two container ships of 6,000 TEU and 30 knots (one uses reactors equipped with two MRXs of total power 348MWth and the other a diesel engine), in case of Asia-North America route, commissioned for 20 years from the year 2015. The quantity of container transport through this route is the largest among the the world's three biggest sea routes (Routes of Asia-North America, Europe-North America and Asia-Europe). The crude oil price is assumed to be $36/bbl. averaged within the service period. This figure shows, (a) The capital cost of the nuclear ship is about 2 times larger than that of the diesel ship. (b) On the other hand, the fuel cost of the nuclear ship is about 1/2 of that of the

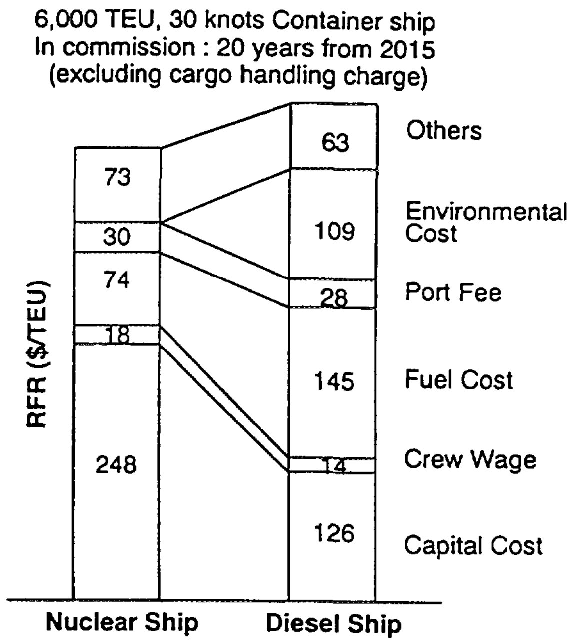

Fig.5 RFR of 6,000 TEU container ship

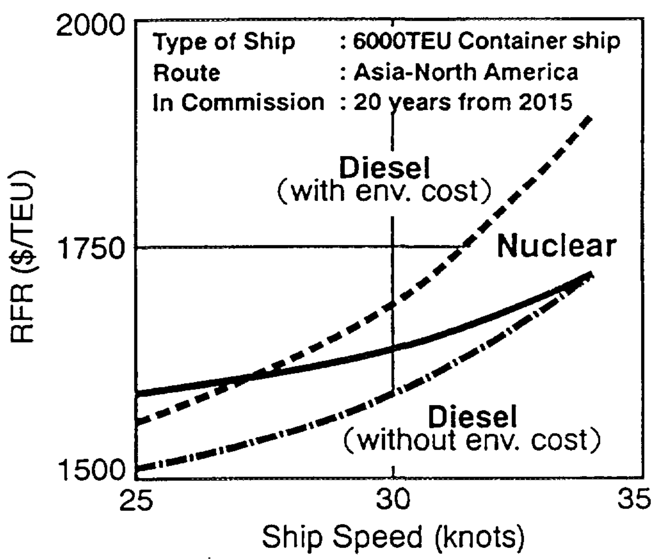

Fig.6 RFR as a function of ship speed

diesel ship. (c) The environmental cost of the diesel ship accounts for 22% of the RFR. The cost study of nuclear and conventional ships shows that the situation becomes more favorable for nuclear ships as increasing of the speed and the load. Figure 6 indicates the RFR as a function of ship speed. Over 28 knots, the nuclear ship holds an economically dominant position over the diesel ships, because the diesel ships must have higher environmental costs.

6. R&D work

To realize nuclear ships with commercial use, it is necessary to obtain an economical, safe and reliable reactor system. In addition, the supplementary items such as the international agreement on safety, the preparation of the maintenance yard, etc., should be solved. R&D programs on nuclear ships have been proposed and being performed to solve technical subjects so that the nuclear ships will be put into commercial use in the future.

(1) Experimental study on thermal-hydraulics

To study the hydrothermal behavior in the water-filled CV, especially steam condensation characteristics, a small scale test facility (volume ratio : about 1/300 of MRX) has been fabricated and fundamental experiments on the following behavior are in progress(5), (a) thermal and hydraulic responses in both the RPV and the water-filled CV under LOCAs, (b) evaluation of mechanical loads generated by LOCAs, and (c) capability of natural circulation and decay heat removal. Furthermore, to confirm the function of the safety features such as an integral PWR with a water filled CV and passive safety systems, a large scale synthetic test facility is planned. The thermal power of the facility is 5MW (1/20 of the MRX), however the height is same as that of the MRX because it is most important to simulate accurately the natural circulation condition. Experiments on board are planned to obtain the behavior under the ship inclination and oscillation.

(2) Development of components

The components of in-vessel type CRDMs such as the motor, the latch magnet, etc., have been developed (6), and the function and reliability tests using the full mock-up CRDMs are planned. The water-proofed design is being performed for the components and the thermal insulator placed in the water-filled CV.

(3) Automatic control system

It is important to reduce the number of reactor operators from the view of the ship's economy. From this standpoint and to enhance the ship's safety, highly automated control systems have been studied which will be adopted and will cover the whole operations during normal, abnormal and accident conditions. This system consists of control and diagnostic systems as shown in Fig. 7. The control systems generate control signals for control equipments, for example, control rods, pressure control valves, flow control valves, etc., in accordance with the reference signals (demand signal) and signals of each measured parameter. If a difference exists between the signals of the reference and the parameter, a control signal is generated based on the operational procedure and changes the parameter to the demand conditions. The diagnostic systems are provided to monitor the malfunction of the systems and

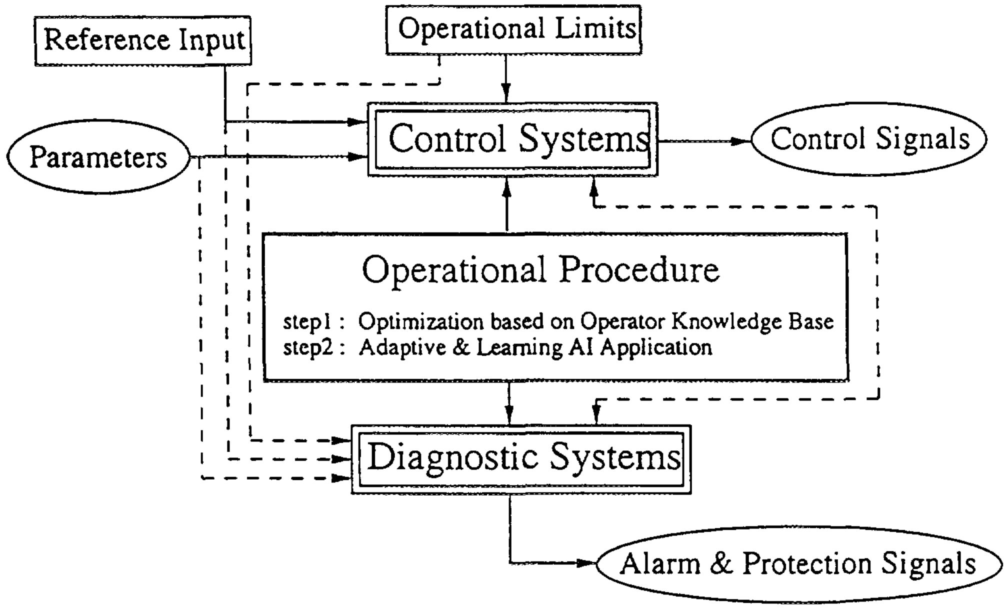

Fig.7 Advanced automatic control system for MRX

the plant operating conditions. For the operational procedures, the optimization is made based on the operator's knowledge and the learning AI.

(4) Development of nuclear ship simulator

The nuclear ship simulator NESSY (Nuclear Ship Engineering Simulation System) has been developed in JAERI and used for the simulation of the nuclear ship "Mutsu" so far[7]. It can simulate both the behaviors of the reactor systems and the ship motions. Mutual interactions can be analyzed for the situations such as changes of the reactor power, the steam generator water level, etc., due to the ship motions caused by waves or the navigator's maneuvering.

The accuracy of the system has been verified with the operation data of the "Mutsu" and it is proved that this system is a very useful tool for developing advanced marine reactors[8]. Modifications of the models and the parameters are being made for the MRX reactors since 1995.

7. Conclusion

An advanced marine reactor, MRX, has been designed to be more compact and lightweight with enhanced safety. The engineered safety is accomplished through a simplified system which is suitable in particular for a marine reactor, since it must be operated by limited number of crews. The LOCA analysis shows that the core flooding is maintained passively even taking into account the ship inclination. The one-piece removal method is proposed to keep the maintenance and refueling works short and safe. This method also makes the ship's decommissioning easy and enables us to reuse the reactor after the ship's life. The economic evaluation shows that for container ships of 6000 TEU traveling over 28 knots, the nuclear ships will hold an economically dominant position after 20 years from the present time, because the diesel ships must have higher environmental costs. In addition to the design study, extensive R&D activities are being performed. These can contribute largely to the realization of the nuclear ship in commercial use in the future. Considering that the MRXs are small size reactors with highly safe capabilities and transferable ones, they have a wide variety of uses in the energy supply system.

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CONCEPT, EXPERIMENTAL AND CALCULATIONAL INVESTIGATIONS OF A MICROMODULE REACTOR

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Abstract

In addition to the evelutionary improvements and development of new additional devices and safety systems, much prominence is now given to the search of technical principals providing the enhanced safety for a NPP using new reactor design concepts as a whole. One of such concepts refers to the experience of the Institute of Physics and Power Engineering (IPPE), accumulated in the area of the exploitation of reactor channel loops for research purposes. A long-term experience of using such loops reveals that they feature a high reliability, safety and high technical and engineering parameters.

Introduction

In addition to the evelutionary improvements and development of new additional devices and safety systems, much prominence is now given to the search of technical principals providing the enhanced safety for a NPP using new reactor design concepts as a whole. One of such concepts refers to the experience of the Institute of Physics and Power Engineering (IPPE), accumulated in the area of the exploitation of reactor channel loops for research purposes. A long-term experience of using such loops reveals that they feature a high reliability, safety and high technical and engineering parameters.

1.Reactor Concept

The concept suggested consists in subdividing the primary circuit of a reactor in small volumes with minimal hydraulic connections between each other. This concept is realized by employing devices of special construction, the so-called micromodules (MM). A micromodular channel is a tubular pressure vessel (Fig.1) 100-200 mm in diameter, where a fuel assembly is positioned in its lower part and a heat exchanger (HE) (1-2 circuits) in the upper part. Using flow splitters, the natural circulation of coolant is maintained in the primary circuit of the MM. Each micromodule is provided with inlet and outlet pipelines for secondary circuit coolant and with a pipeline connecting the MM primary circuit with the pressurizer (P). Thus, from the viewpoint of the construction and thermal-hydraulic scheme, the MM is a miniature integral reactor. To provide a prescribed reactor power, the core is composed of the necessary number of micromodules which are inserted in the moderator (e.g., graphite, heavy water, etc.).

Qualitative analysis and quantitative estimations obtained in late 1970's and early 1980's revealed that based on the above mentioned concept, the creation of a reactor of enhanced safety is possible at the expense of the following factors:

- The subdivision of the primary circuit in a number of small independent volumes (up to 100 l), resulting in reduction of the scales and accident consequences due to the depressurization in the primary circuit. In the event of the MM vessel rupture, the heat removal from fuel assemblies is accomblished due to water inflow from the pressurizer and then due to water supplied by feed pumps.

- The absence of circulation lines of large cross-sections in the primary circuit: in the case of the rupture of the small diameter (about 10 mm) pipeline connecting the MM with the pressurizer, the reliable cooling of fuel assemblies will be provided during a long-period of time at the cost of possible steam condensation in the-micromodule heat exchanger.

- The limited velocity and amount of coolant discharge while disrupting the primary circuit, makes the problem of the accident loca.lization within a plant easier.

- The compactness of the MM primar.y circuit creates ideal conditions for the natural circulation (NC) in the circuit in each flow regime.

- In the case of channel reactors, it is easier to solve the problem of organising independent (autonomic) cooling system for control rods in the MM-based reactors. This system can operate as an additional channel for heat removal in case of failure of major heat removal systems.

- In view of the fact that the MM power and correspondingly, its cross-sectional dimentions are dozens and hundreds times lower than the reactor power, there exists a real potentiality to conduct experiments on different operation conditions, accident conditions included, using electrically heated modules of the full scale hight.

The above given considerations stimulated the development of a reactor based on the proposed concept.

2. project of Micromodule-Based Reactor

In the 1980's, the Research and Development Institute of Power Engineering (RDIPE) together with the Institute of Physics and Power Engineering (IPPE) has developed a number of reactor designs (with a capacity from 20 to 300 MW) for the Nuclear District Heat Plant. Fig.2 shows the schematic diagram of a reactor. and Table 1 incorporates the main parameters of the RKM-150 unit, for which the technical design and extensive calculational and experimental investigations were performed. Alongside with the concept consisting in sectioning the primary circuit into small volumes, the MM- based reactor safety is guaranteed by a set of other factors:

- Self-protecting features of the reactor: negative power, temperature and steam reactivity coefficient: the stability of the power generation field; - high heat capacity of reactor; -natural circulation of water in the primary circuit.

- The availability of safety systems: - reactor shutdown by introducing absorbers under the action of the force of gravity; - the micromodule replenishment with water during the vessel rupture from the pressurizer (passive) and then by pumps; the circulation of the secondary circuit with water from hydroaccumulators (passive) and then by pumps; heat removal from the MM by the evaporation of water from tanks (passive) via the water cooling tower (active) or to the network circulation loop (active).

TABLE 1. MAJOR CHARACTERISTCS OF THE RKM-150 REACTOR

CHARACTERISTICS, UNITS	VALUE
Thermal power, MW	150
Reactor Core -moderator -MM number -number of controls -height, m	 graphite 220 44 3
Micromodule - power, kW average maximum -number of fuel elements -in a fuel assembly, -heat removal surface of the fuel assembly, m^2 -vessel diameter in the core, mm × mm in the heat exchanger region -full height, mm	 682 1070 36 3 85 118 5 [illegible] 180 × 10 10890
Primary circuit water parameters -pressure, MPa -temperature, °C at the fuel assembly inlet at the outlet of the fuel assembly of maximum power -flow rate in the MM at maximum power -the number of pressurizers	 7 85 136 265 1 6 4
Heat exchanger 1-2 of the circuit -the number of Field tubes height, m -heat transfer surface, m^2	 37 3 6.3
Secondary circuit water parameters -pressure, MPa -inlet and outlet temperature, °C -flow rate, t/hr	 1.18 80/140 2130
Network circuit parameters -pressure, MPa -inlet and outlet temperature, °C -flow rate, t/hr	 1.6 70/130 2135

- By additional technical solutions: moderate parameters (the heat fluxes of fuel elements, pressure, temperature); independent cooling system for absorbing rods; a good thermal contact between the MM vessel and graphite: pressure relation in circulating loops $P_1 > P_2 < P_3$, etc.

3.ComputationalStudy

The experimental investigations using MM models were preceded by numerical ones mainly aimed at the selection of geometric and operating regime parameters for the MM and reactor unit as well as the prediction of their behaviour under accident conditions. In the process of calculations for normal operating conditions, the limiting MM power value was obtained and the dynamics of natural circulation development at varying power, etc. were determined.

Calculations were also performed for typical accidents such as the disrupture of the pipeline between the MM and the pressurize, the MM vessel failure and others. In view of the complexity of hydrodynamical and heat transfer processes under such condition some problems should be experimentally verified. These questions will be touched upon later when discussing experimental data in Section 4 and now we shall consider some calculational results for beyond-design accidents in more detail.

In a beyond-design accident with postulated instantaneous MM uncovery, under the reactor emergency shutdown conditions, the maximum fuel temperature at a fuel assembly power of 1070 kW was 1070 C. The fuel melting takes place when the pipeline connecting the MM with the pressurize breaks just near the MM vessle with the simultaneous failure of the reactor emergency protection or when the MM vessel breaks simultaneously with the emmergency protection system failure and the disrupture of the pipeline between the MM and pressurizer. In both cases, the fuel melts only in one MM subjected to the failure.

In beyond-design accidents accompanied by the failure of secondary circuit heat removal systems, the absorber cooling system ensures that the maximum fuel element cladding (12OO C) was not exceeded. If this system fails also, the element temperature of 1200 C is riched for ~ 12 hours because of the large heat capacity of the reactor. During this time it is necessary to take adequate measures to monitor the accident process. Table 2 gives an idea on the radiation exposure as a consequence of different normal and accident conditions.

TABLE 2. YEAR EFFECTIVE EQUIVALENT DOSE VALUES ON THE BOUNDARY OF THE SANITARY-PROTECTION ZONE (2 Km)

SITUATION	Design values, Zv/year	Tolerable limits, Zv/year
Normal operation	7.6×10^{-8}	2×10^{-4}
Design accidents	6.6×10^{-8}	0.1
Beyond design accidents	4.1×10^{-3}	0.1

4.Experimental Study

To study the MM thermal-hydraulics, two electrically heated full scale models were designed and manufactured in the Institute of Physics and Power Engineering: of high (up to 10 MPa) and low (up to 1 MPa) pressure. Both models practically are entirely analogous to the standard micromodule design (Fig.1). The first model was used to validate working design parameters and investigate a number of accident senarios. The second one was employed to study the temperature behaviour of fuel elements at the final phase of accidents. In the latter case, the heat flux simulation from the graphite moderator was also provided.

4.1.Steady-State Conditions

At the first stage of investigations, the effort was mainly concentrated on the validation of the design of the MM parameters, among which the coolant flow rate in the primary circuit under natural circulation conditions is most important.

In Fig.3 the flow rate in the primary circuit, G, versus the fuel element power is presented at high pressures. In the single phase flow region ($N < 800$ kW, $P_1 > 5.0$ MPa), the flow rate G_1 does not depend on the pressure in the primary circuit. With further increasing of power, the appearance of a steam phase raises the rate of circulation earlier the lower the pressure. With a design pressure value of 7.85 MPa and a maximum power-rating channel capacity of 1070 kW, the coolant is subcooled and boiling (even surface boiling) is actually absent. In this case, the flow rate over the primary circuit reaches Gi = 1.5 kg/s which is in fair agreement with the design value of the standard micromodule flow rate (1.6 kgls). It should be noted that for the MM design parameters, the flow rate is found to be stable. Only with reduced pressures (~ 2.5 MPa) in the primary circuit, minor flow rate variations appeared within a narrow range.

4.2.Dynamics of Micromodule

With the natural circulation of water in the primary circuit, the flow rate increases as the power rises. There was a danger that this circumstance will restrict the rate of the MM power increase due to a slow increase of the flow rate to a new (increased) value or initiate the circulation instability.

Experiments have eliminated this possibility. In the case of a momentary raise of the power from initial values of 100, 300 and 550 kW by 200 kW, the set-up of a new flow rate value occures smoothly and readily (for < 1 min). Thus, the natural circulation is not a factor which could hinder providing the reactor with the necessary manoeuvre characteristics in the process of its operation as a part of a District Heat Plant.

4.3.Accident Regimes

4.3.1.Depressurization of Primary Circuit

While studying accident situations, the major attention was focused on accidents associated with the depressurization of primary circuit components and the disturbance of heat removal over the secondary circuit.

4.3.1.1.Rupture of the pipeline between MM and pressurizer

Consider the behaviour of the MM channel in the case of the disrupture of the pipeline connecting the micromodule with the pressurizer. The peculiarity of this situation is that it is impossible to feed water to the MM from the pressurizer. The calculations obtained for this accident did not predict severe consequences; however, this fact needed an experimental verification. The problems to be studied were as follows:

- The MM depressurization results in ceasing or reversing of the circulation in the MM or degrading the heat removal from fuel elements in the first phase of the accident;

- the amount of water remaining in the MM compared to the amount at the moment the pressure reduces up to the atmospheric value in it;

- the maximum permissible power of the fuel assembly filled with water under the flooding conditions (zero flowrate);

- the rate of coolant losses at the final accident phase at a pressure near to the atmospheric.

Fig.4 shows the results of one of the experiments investigating the break-down of the pipeline connecting the MM with the pressurizer. The rupture is located just in the vicinity (100 mm) of the MM vessel and is simulated using a fast-responce device. The initial power of the channel amounted to 1070 kW and then reduced according to the law of residual heat variation. The delay time was accepted to be 10 s, which is considerably heigher than the design-basis value (i.e below 1 *s)*.

The experiments indicated that the pressure in the MM reduces to the atmospheric pressure for 1-2 min. The water flow rate through the fuel assembly slitly increases at the beginning of the process (to ~ 1.9 kg/s) and then for 2-2.5 min is practically equal to the nominal one (1.5 kg/s), providing the reliable cooling of fuel elements. No negative outcomes were found in the temperature behaviour of fuel elements over this time period.

The amount of water remained in the MM to the moment of reducing the pressure to the atmospheric value amounted to 54 kg., the water level being ~ 2.6 m above the upper edge of the fuel assembly.

To define the limiting fuel assembly power under sarbotage conditions occurring after reducing the pressure, some preliminary tests were carried out using a 7-rod bundle with the inlet cross-section blocked, (i.e. under more severe conditions as compared to the MM). They revealed, that for a 30-rod fuel assembly MM the limiting power should be 120-130 kW, thich is higher than the net power of the residual heat in fuel elements and the heat influx from graphite (for the present 100 kW design). In the full scale MM without blockage at the fuel assembly inlet (i.e. for real conditions) at the atmospheric pressure, a power of 150 kW was reached without any indications of any critical phenomena. Thus, the condition of the fuel assembly filled with water is sufficient to remove the residual heat under no-crisis conditions.

After the MM pressure is reduced to the atmospheric value, the further development of the process is characterized by the ongoing loss of coolant. The presence of the heat exchanger in the MM results in that the overwhelming portion of steam generated in the fuel assembly condenses on the surface of this heat exchanger and returns through the downcomming line of the circuit to the fuel assembly inlet. As a result, the rate of steam losses through the rupture is considerably lower than the rate of steam generation in the fuel assembly. The measurement of the rate of mass losses in this phase of the accident is taken at different values of the fuel assembly power, and different levels of water in the MM. The obtained data allowed the time interval to be defined, over which the fuel assembly has reliable cooling, and the fuel element cladding temperature is close to the saturation temperature of water at the atmospheric pressure This time interval ranges from 36 to 44 hrs, and only then the fuel element heat-up will occur, but no higher than 610 C in accordance with the calculations

4.3.1.2.The Microinodule vessel rupture

While breakage of the MM vessel, and the link between the MM and the pressurizer remains intakt, the MM is provided with water from the pressurizer. However, with a large break of the MM vessel, the flow rate of the loss of coolant exceeds the flow rate of the supply; therefore the pressure and the amount of the coolant in the MM decreases until the amounts of losses and supplied liquid are equal

To study such the situation, the MM vessel was provided with pipe connections of which the breaks at different elevations were simulated (i.e 5190 mm and 1140 mm above the upper edge of the fuel-assembly) All experiments were conducted at an initial power of 1070 kW reducing in accordance with the residual heat expected.

It is experimentally established that if the disrupture of the MM vessel occurs at these elevations in the initial phase of the accident, the heat removal from the fuel assembly is effected in crisis-free conditions with the fuel assembly filled with water Therefore the subsequent cooling of the fuel assembly is beyond question even if only the feed is present in these cases. The contrary is the case when the MM vessel ruptures below the fuel assembly where the MM is incapable to retain the supplied water and the flow rate entering it from the pressurizer is insufficient to provide a continuous flow over the entire fuel assembly cross-section. It is possible under these conditions that water will not flow around a portion of the fuel elements, which will lead to their overheating.

The experiments were conducted as follows: A preset flow rate of water of room temperature (15-20OC) was fed to the upper part of the micromodule, then the power of a predetermined value was supplied to the fuel assembly and the micromodule vessel; the fuel element temperature being recorded at four points over the fuel assembly cross-section and at ten points of the entire elevation. Considerable attention was paid to the investigation of the influence of the feed water conditions (e.g. below or above the "flow-over" wiridoiqs of the heat exchanger; from one or two sides of the MM vessel) Tables 3 and 4 present some results of these experiments It is obvious from Table 3 that there is a significant scatter in fuel assembly temperature (65-400°C) This shows a non-uniform distribution of inlet water over the fuel assembly cross-section. Of great interest is the fact associated with reduction of the fuel element surface temperature as the power increases from 10 to 15 kW in the present case. To our mind, this is due to the boiling-up of water on those simulators where water is available, and its more uniform distribution over the fuel assembly cross-section At a certain stage, further increase in power logically results in a rise of the rod temperature

With increasing the flow rate of water being fed to the MM low fuel assembly temperatures are observed at sufficiently high powers (Table 4) In the RKM-150 reactor, the parameters of water flowing around the secondary circuit and the net MM power (together, with the heat influx from graphite, amounting to ~ 40 kW) meet the conditions of the test the results, of which are shown in the second column of temperatures, Table 4. However in the RKM-150 reactor the flow rate of water flowing to the MM from the pressurizer is significantly higher (no less than 4000 kg/hr) than in the above mentioned case (700 kg/hr). Thus, the reliable cooling of the fuel elements without their overheating will be provided in each phase of accidents accompanied by the MM vessel rupture at any elevation.

In addition to the above consideration, experiments were conducted. simulating accidents with MM vessel rupture at the elevated pressure and a power level of ~ 600 kW. These experiments confirmed the absence of the fuel element overheating under conditions of this accident even at flow rates of feeding water considerably 'ower than in the RKM-150 MM

Table 3

THE THERMOCOUPLE INDICATIONS AT WATER CIRCULATION ROUND THE MM AT A FLOW RATE OF 100 KG/HR AND ONE-SIDED FEED.

(The MM vessel is not heated and the water flow rate over the secondary circuit is equal to zero)

FUEL ASSEMBLY POWER, KW	10	15	20
Thermocouples	Temperature, °C		
T3	358	98	399
T6	346	99	370
T7	.320	98	367
T8	275	97	160
T9	306	98	159
T11	135	113	111
T15	90	98	98
T35	65	105	98

4.3.2.Secondary Circuit Zero Flowrate ,

Experimental study has been carried out of an accident with ceasing the second circuit water flow rate through the heat exchanger of one MM, which can be caused by blocking the flow area by an outside subject. The.initial parameters of the micromodule were set up to correspond- to its operation at a maximum design-based power of 1070 kW. After terminating the flow rate of the secondary circuit water, the MM power reduced according to the law of residual heat variation with a delay of 10 *s*.

Experiments showed that in such an accident. the pressure over the primary circuit slightly exceeds the nominal value (by 0.2 MPa) during a short period of time 15s) The circulation over the primary circuit provides a crisis-free cooling for the fuel assembly As for the whole of the reactor. the pressure in the emergency MM reduces after decreasing the power The duration.of the experiment was 10 min By that time about 32 kg of water retained in the MM whereas to fully cover the fuel assembly, 18 kg is sufficient (Note, that according

Table 4.

FUEL ELEMENT TEMPERATURE AT WATER CIRCULATION AROUND THE MM WITH A FLOW RATE OF 720 KG/HR (ONE-SIDED FEED.
(The water flow rate in the secondary circuit is 11 t/hr and inlet temperature is - 65 C)

Fuel Assembly Power, kW	102	60	60
Vessel Power, kW	0	40	20
Total Power, kW	102	100	80
Thermocouples	Temperature, °C		
T3	107	103	105
T5	93	67	55
T6	107	104	103
T11	104	103	96
T24	109	106	104
T35	105	103	102

to calculations, the time required to decrease the water amount in the MM to 18 kg is 13 min). Thus the experiment reasonably well justified the predicted time of starting the fuel assembly uncover, which is assumed in the fuel element temperature behaviour evaluations.

4.4.Thermal-Hydraulic Characteristics

One of the possible trends in further improvement of the reactor of the proposed design is to use a micromodule with boiling water in the primary circuit. Fig.5 illustrates the experimental results of an experiment in which the micromodule is filled with water to a certain level (V= 90, 80 and 65% of the total free micromodule volume) and cut off the pressurizer. Thus, the pressure balance under heating-up conditions is due to the gas (steam-gas) volume being in the upper part of the micromodule. It is evident from the figure that a micromodule power of 1070 kW can be reached with V_0= 80% at a pressure of 5.3 MPa. With filling V_o = 65 %, such a power is reached at a pressure of 3.3MPa; the coolant flow rate of the primary circuit with boiling water in both cases being higher than under the non-boilin conditions. It is necessary, however, to note that flow rat fluctuations were observed in a restricted range of powers (e.g. with a amplitude of 1.5 % at V 80 % within the range of power from 250 to 350 kW and 15 % with V = 65 % at a power up to 150 kW.

4.5. In-Pile and Other Tests

one of the conditions of the reactor reliability and safety is the favourable water-chemical conditions for the construction materials. The traditional measures of water purification and the introduction of correcting agents in the MM reactor being the "blind" part of the loop are practically impossible.

The study of the water-chemical conditions was performed using MM models under in-pile conditions. All in all 5 micromodules were tested in the reactor of the First Nuclear Power Station. In total, the time of operation during testing amounted to about 150 000 days and nights. The duration of the operation of four micromodules amounted to about 10 years for each of them. The main results of experiments are as follows:

- while filling the gas space of the pressurizer with nitrogen, the ammonia synthesis takes place in the water of the primary circuit and the pH value is set up at a level of 9-11;

- the content of corrosion products in stainless steel is on the average - 0.15 mg/kg for iron, for nickel and chrome it was 20 mg/kg;

- the amount of chlorides does not exceed 0 05 mg/kg. The amount of gasses amounts to 30-150 H.cm^3 /kg;

After 5 years of testing, one of the micromodules was withdrawn out of the reactor to assess the corrosion condition. The visual inspection showed that the fuel elements and the MM vessel are in a satisfactory state. No mechanical damages of fuel elements and spacing grids were detected and no visible depositions were present on the fuel element surfaces. After examination, this MM was placed again into the reactor, where it operated for about 5 years more.

In addition to the above mentioned studies, the work was performed on a number of directions of supporting the RKM-150 reactor design. They included comparative experiments on enhancing the absorbing materials control rods system. The best results were obtained with usinig D1/2 Ti O5 which was adopted in the technical design of the reactor

Also, the problems of the fuel element cladding tightness control system were studied. It was found that such a control, which does not require sampling the coolant can be implemented using gamma-ray detectors being moved between the micromodules in the above-reactor space, when the reactor is shut down.

Conclusion

1. The concept is proposed to enhance the reactor safety at the cost of subdividing the primary circuit into small parts. This is achieved by means of micromodules incorporating a fuel assembly, a heat exchanger and a natural circulation of coolant.

2. Wide experimental and computational investigations confirmed the design characteristics of such a reactor (RKM-150) regarding:

The therrnohydraulics involving the temperature regimes;
flow rate and natural circulation stability;
thermal core safety under the natural circulation;
the water-chemical condition;
the manoeuvrability in var.-Ing the power during load variation;
and the correspondence to the nuclear and radiative safety requirements.

SPECIFIC SYSTEMS AND ANALYSIS

XA9745983

EMERGENCY HEAT REMOVAL IN THE INTEGRAL WATER COOLED ABV-6 REACTOR FOR THE VOLNOLOM FLOATING NUCLEAR POWER PLANT

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Abstract

Several independent active and passive safety systems are employed in the design of integral WCR in order to provide together with the reactor inherent safety features realization of the emergency heat removal function:
- active primary coolant clean-up system transmitting heat to the tertiary and then to the forth circuit,
- passive emergency heat removal through steam generator to the atmosphere,
- reactor cavity flooding system providing heat transfer from the reactor vessel to the reactor metal-water shielding tank,
- high and low pressure make-up system using water from the special storage tank.

Operation of the systems under LOCAs and accidents with normal heat removal systems failure is considered in the paper. Special emphasis is done to the description of systems characteristics, systems interaction and modes of operation which are influenced by the reactor integral design.

Emergency heat removal (EHR) is one of the major safety functions that should be provided by the design of reactor safety systems.

Under the design of the ABV-6 reactor designated for the floating NPP "Volnolom" the following specific features of the reactor were taken into account:
- Integral reactor design (Fig.1)
- Low reactor capacity (38 MWt)
- High reactor heat accumulation capability (~1.8 s per °K)
- Natural convection in the reactor primary circuit

- High margin in the strength characteristics of the primary circuit equipment (tolerable pressure of 31 MPa is ~2 times higher than the nominal design pressure)
- Employment of metal-water shielding tank containing 26 m^3 of water
- High thermal conductivity of reactor fuel made of uranium-aluminum alloy.

These features of the ABV-6 reactor strongly influence the solution of the EHR issues.

1. The integral arrangement of the primary circuit results in exclusion of big primary pipelines as well as a larger primary coolant inventory in the reactor pressure vessel (RPV). These result in long period before it is necessary to start coolant supply to the RPV under LOCAs, since reactor core uncovery in the case of reactor make-up system failure would take place only several hours after the beginning of the accident. Besides, there is a possibility for the EHR through steam-generator in integral reactor design and this also enlarges the grace period.

2. The relative portion of heat dispersed to the surroundings is higher at small reactor capacity. This allows decrease in the time required for the EHR systems to operate.

3. The high reactor heat capacity allows reduction of the EHR system capacity compared with the residual heat rate. The excess of extracted heat is accumulated in the reactor primary coolant and structures in the first stage of the accident without unacceptable reactor temperature and pressure rise.

4. Natural convection ensures reliable heat removal from the reactor core with sufficient departure from nucleate boiling margin during transient and accident conditions.

5. Large margins in the strength of the primary circuit together with high heat capacity of the reactor ensure tightness of the primary circuit under accidents with failure of all EHR trains for a long period of time. For some beyond the design basis sequences this time is unlimited.

6. The metal-water shielding tank is an effective heat sink. The amount of water in the tank is sufficient to provide the EHR for 3 days.

The ABV-6 reactor plant is equipped with the following systems that can be used for the EHR:

- Active two trains system supplying water to the steam generator (SG) from the feed water storage tank (48 m^3) by the emergency feed water pumps with a flow rate up to 15 m^3/h each (Fig.2).
- Passive two trains system supplying water to the steam generator from pressurized storage tanks containing 4 m^3 of water (Fig.3).
- Active primary coolant purification system with pump and heat exchanger designed to remove 1370 kW (3.3% of reactor rated power) of heat directly from the primary water (Fig.4).
- Active two trains emergency core cooling system (ECCS) supplying water to the reactor vessel by 3 high-pressure (head 17 MPa, flow rate 1.2 m^3/h each) and 2 low-pressure (head 3.5 MPa, flow rate 20 m^3/h each) pumps (Fig.5).

ABV-6 reactor

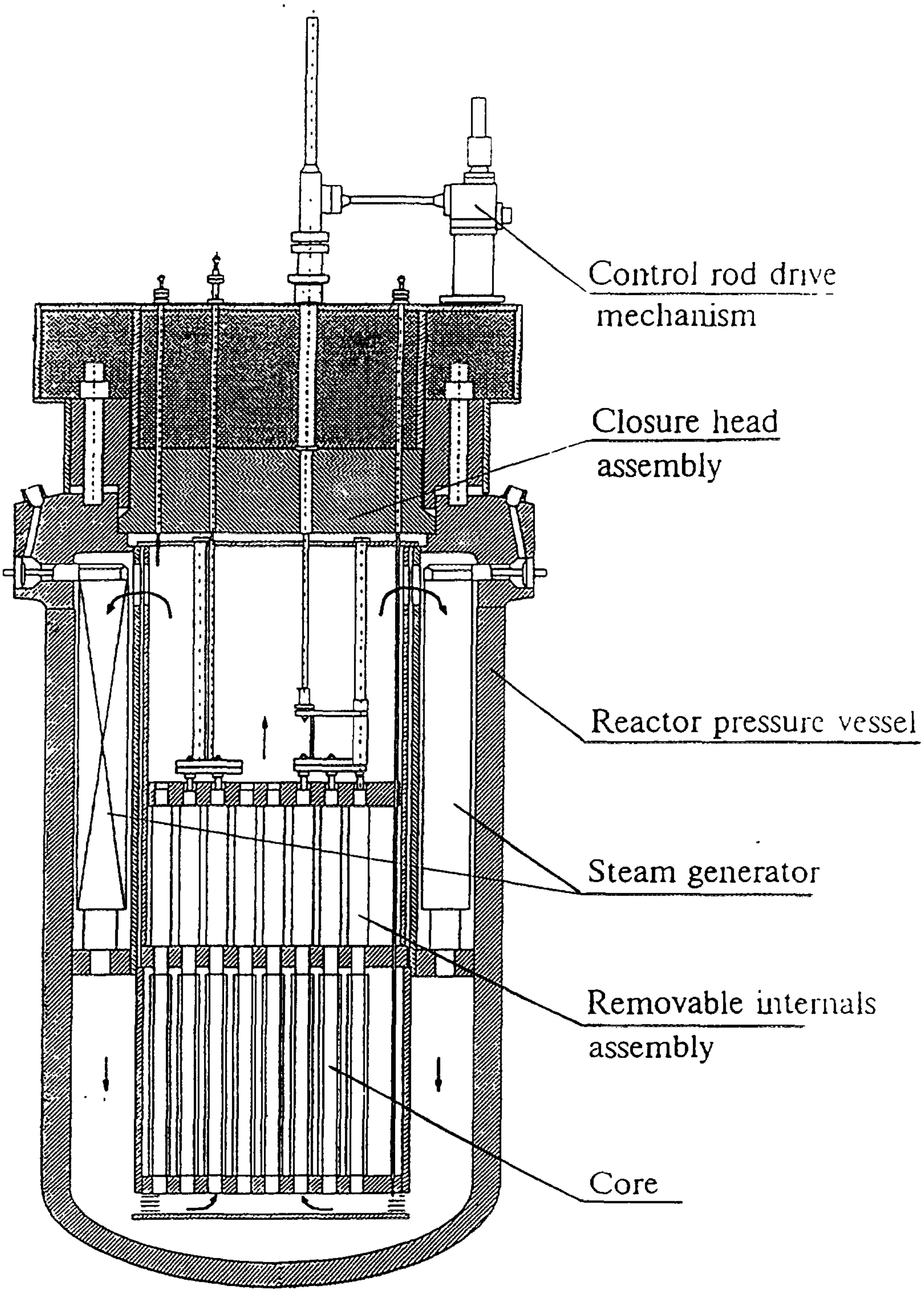

Fig. 1.

Engineered emergency residual heat removal system

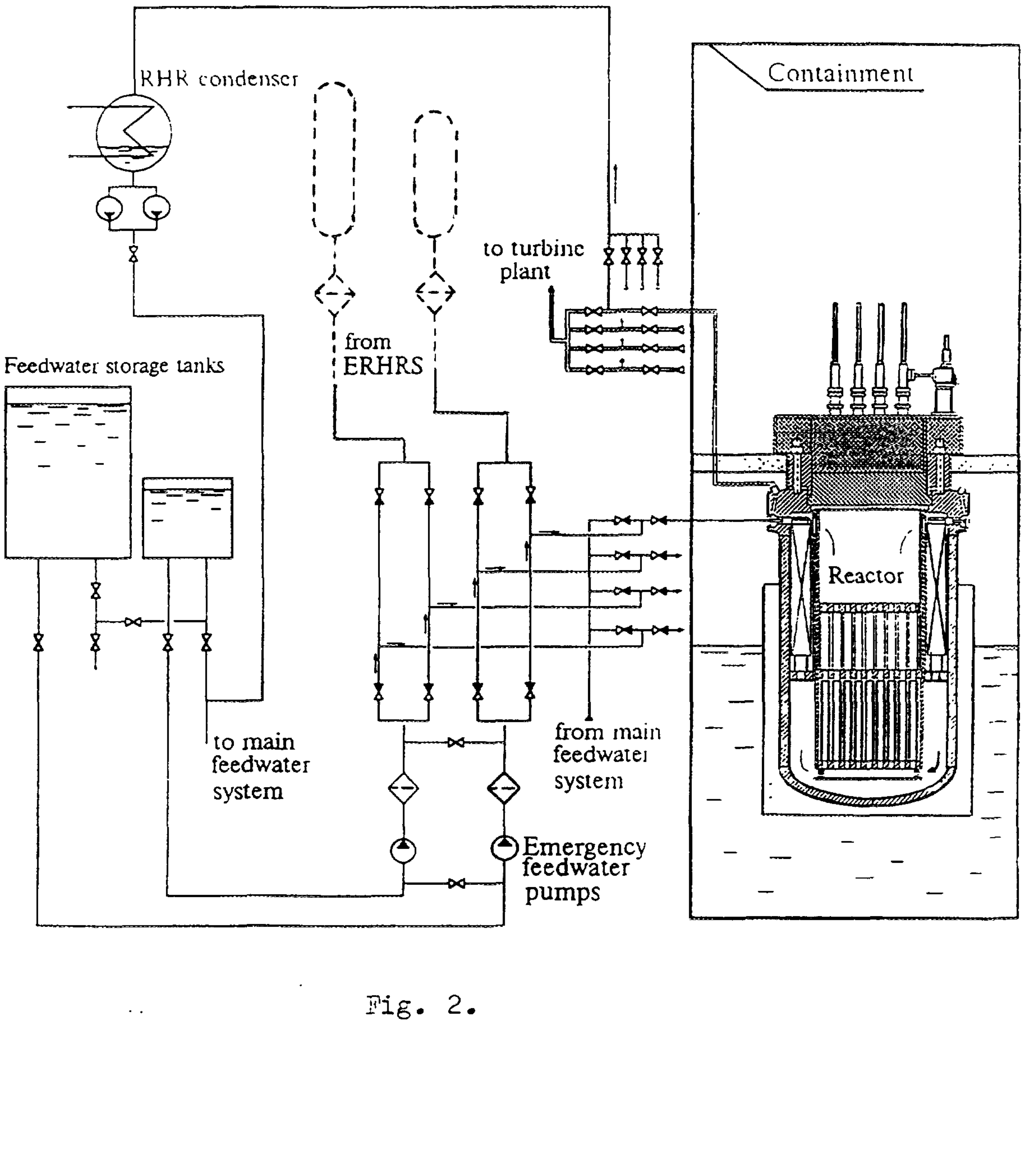

Fig. 2.

Passive emergency residual heat removal system

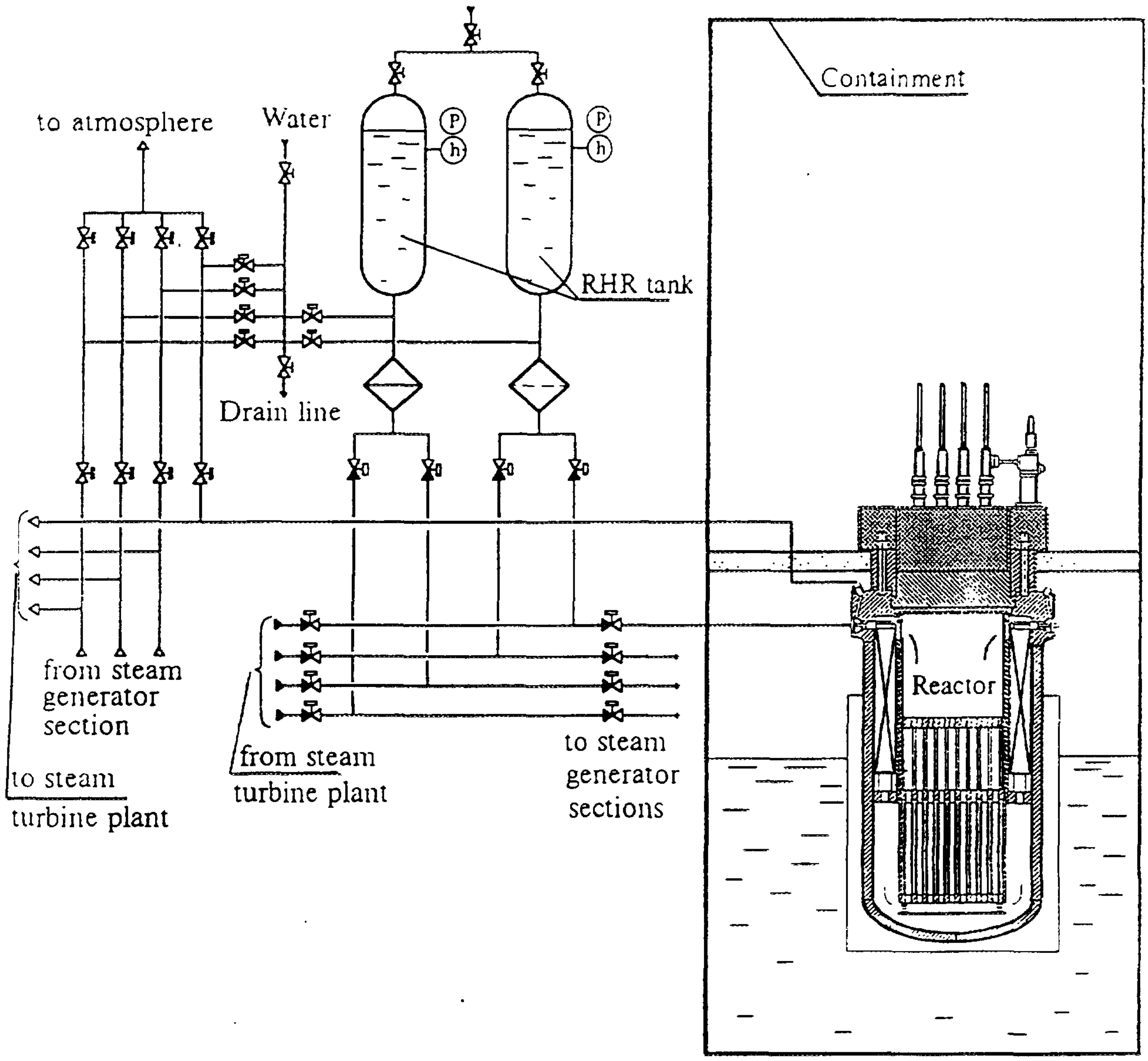

Fig. 3.

Channel for heat removal throught purification system cooler

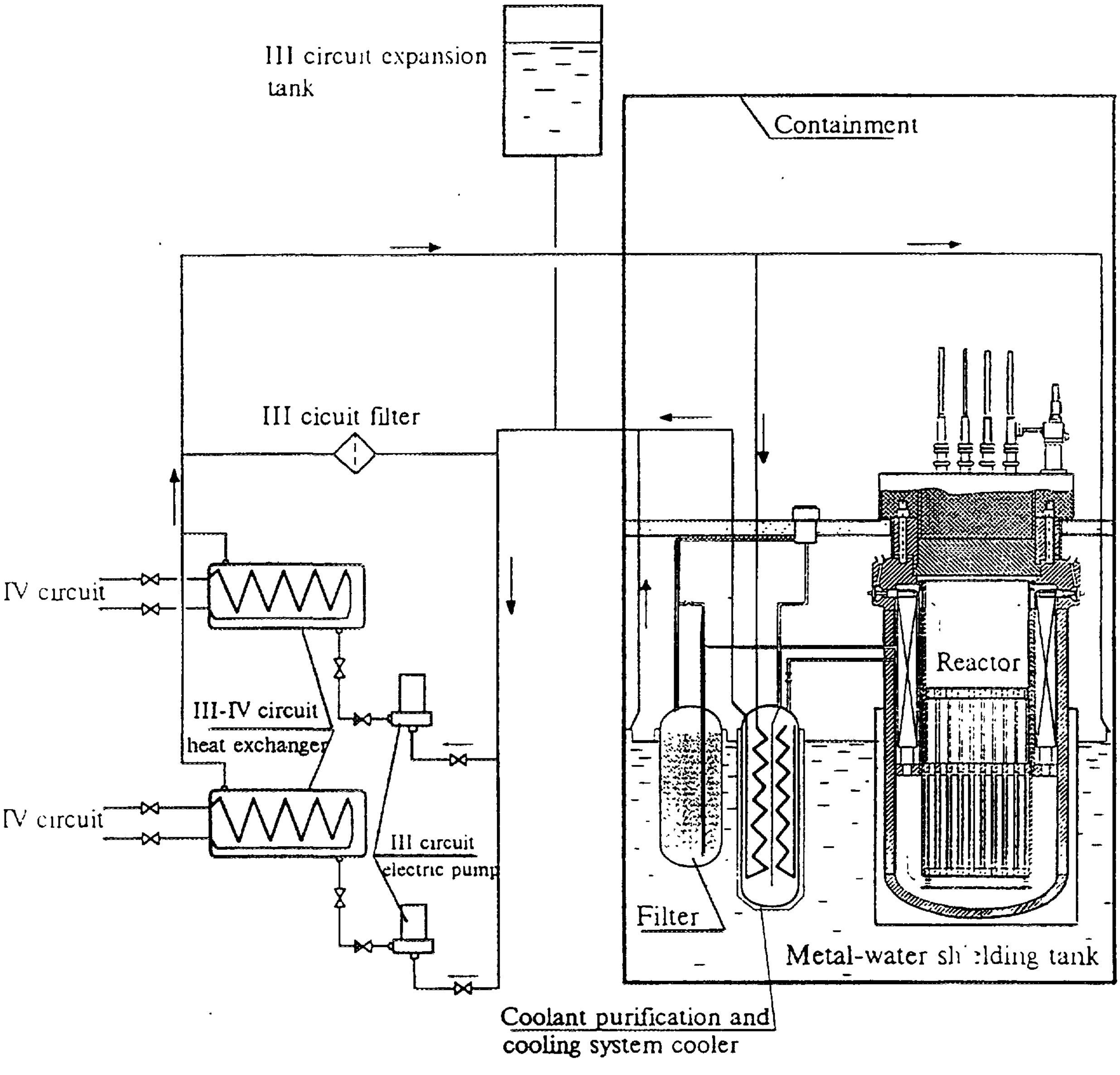

Fig. 4.

Emergency core cooling system

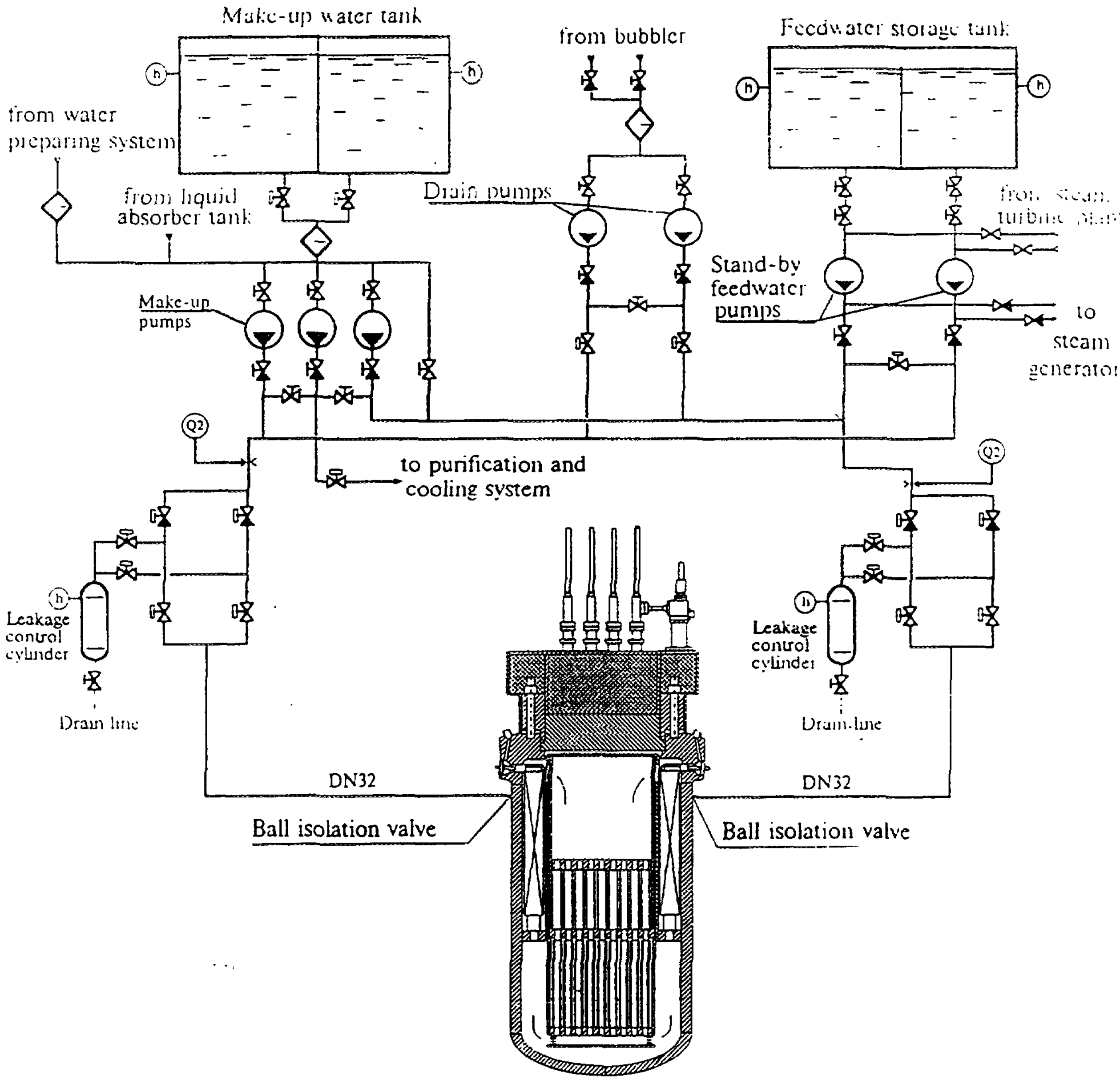

Fig. 5.

Reactor vessel emergency cooling system

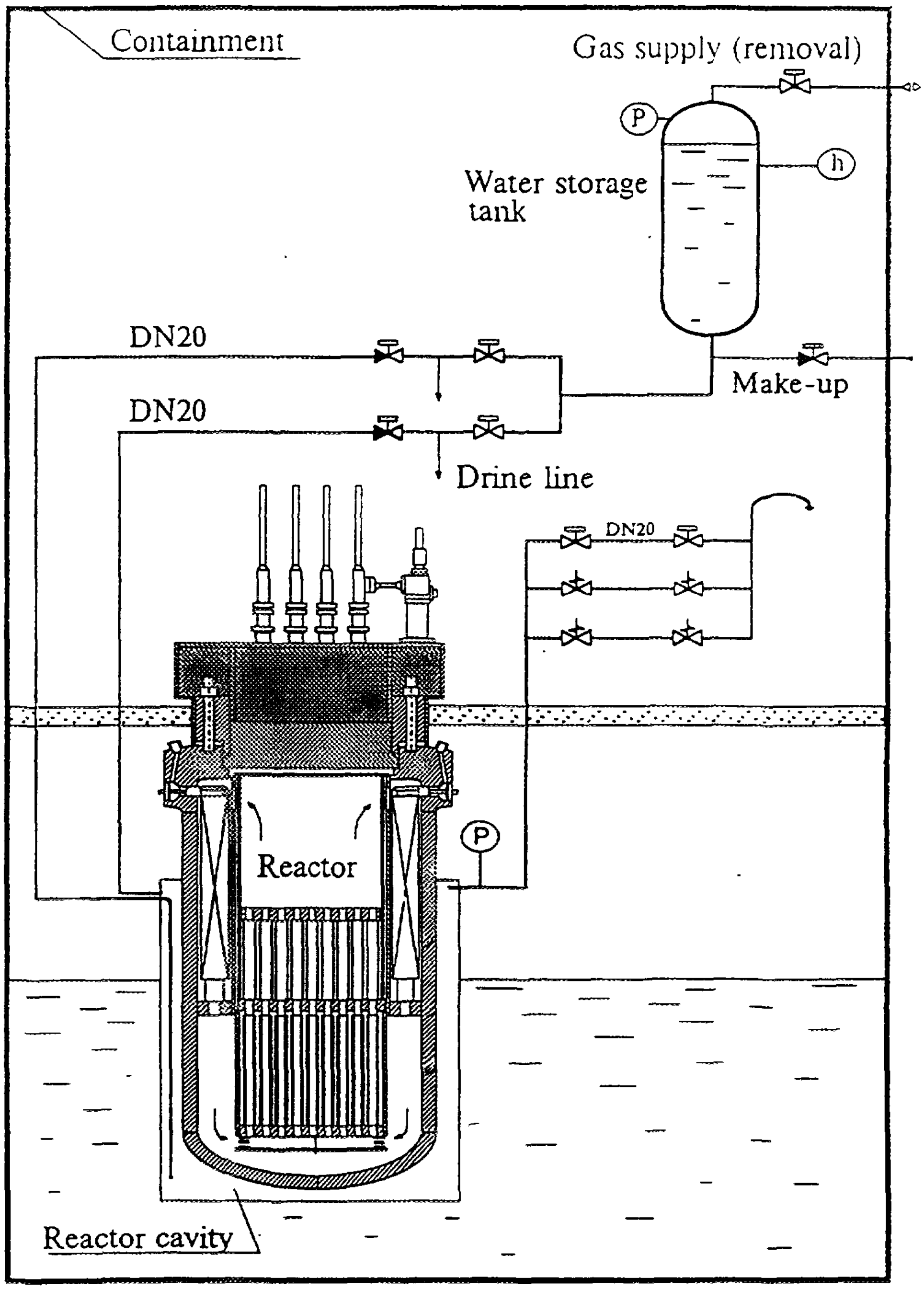

Fig. 6.

Table

ABV-6 REACTOR SAFETY CHARACTERISTICS UNDER BDBA

Initial event	Safety systems failure	Accident characteristics
Black-out	One train of passive EHR failure	Reactor pressure drops from 15.4 MPa to 13.5 MPa by the moment (3.5 h) of depletion of hydroaccumulator. Further on the pressure increases up to 25 MPa in 36 h and then it goes down thanks to dispersion of heat to surroundings. The tightness of primary circuit is kept.
	Two trains of passive EHR failure	Reactor pressure goes up reaching the primary circuit strength limit (31 MPa) in 5 h. Without operator actions control rod drive sealing failure takes place.
	Scram failure	Negative feedback shut the reactor down. Passive EHR system starts to operate in 2 min at the pressure of 18.7 MPa. By the moment of the pressurized water tanks depletion (3.5h) the primary pressure drops to 15.7 MPa and then increases up to 29.2 MPa in 27 h. The tightness of primary circuit is kept.
Pressurizer pipeline break	One train of the ECCS failure	One high pressure pump is sufficient to prevent core uncovery. Minimal water volume above the core is 1 m^3 even in case of only high pressure pump operation and is 2 m^3 if low pressure pump operates.
	Two trains of the ECCS failure + failure of heat removal through the SG	Core uncovery in 3 h. Beginning of FP release from the fuel in 4.5 h. Core melt in 7.2 h. Maximal RPV temperature does not exceed 710 °C if the reactor cavity flooding is provided during the core melt. If not, the RPV temperature reach 1200-1300 °C. The RPV melt through is prevented anyway. If the EHR through the SG takes place core uncovery starts in 10 h and all further processes go slower.

- Passive reactor cavity flooding system with a pressurized water tank up to 7 MPa containing 1.2 m^3 of water (Fig.6).

Safety analysis for all initial events considered has demonstrated a high level of reactor safety system effectiveness and sufficiency under design and beyond the design basis sequences. Some results of the safety analysis are shown in the Table.

It is reasonable to add that according to the PSA, the frequency of black-out with simultaneous failure of two passive EHR system trains does not exceed 10^{-10} per reactor-year. Cumulative frequency of core melt was assessed as $2.5*10^{-7}$ per reactor-year.

The above information allows the conclusion that the integral design concept provides broad range of opportunities to enhance reactor safety with reliance on clear and simple design solutions. This concept looks rather promising at least for small water-cooled reactors.

XA9745984

HIGHLY EFFICIENT CASSETTE STEAM GENERATOR FOR INTEGRAL REACTORS

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Abstract

A once-through cassette steam generator for an integral reactor is considered. The steam generator consists of separate cassettes and is arranged in the annulus between the reactor vessel and the core barrel. Specific developments of integral reactor designs show that the technical feasibility of their implementation is determined to a substantial degree by the characteristics of steam generator used. This paper describes a steam generator with improved overall and thermotechnical characteristics compared to existing steam generators. The high specific characteristics of steam generator allow development of an integral reactor for 600 MWe NPP.

1. INTRODUCTION

Integral reactors characteristics depend greately on the characteristics and structure of the steam generator.

Power rise of integral reactors is limited due to the low specific power of the existing steam generators, in particular those made of helical tubes.

The necessity to locate steam generator inside the reactor vessel causes some additional requirements on its structure, the most important of which are high compactness of the heat exchange surface and high specific power of the steam generator.

OKBM has developed various types of once-through SGs being operated in marine reactor plants PWRs. These SG types, in particular, include:

1) SG made of helical tubes;

2) Cassette-type SG made of straight tube steam-generating elements.

It has been proven by numerous experiments that the specific design of cassette-type SGs with straight steam-generating elements has better specific characteristics compared to SGs with helical tubes other conditions being equal.

For example in the operated cassette steam generators, developed by OKBM specific power in the effective heat-exchange zone reaches 60 MW/m^3.

The design of cassette steam generator for VPBER-600 reactor is considered in this report.

2. STEAM GENERATOR CONCEPT

When developing the SG for the VPBER-600 reactor special attention was paid to design decisions, which ensure:

1) high reliability requirements;

2) operation and repair safety;

3) possibility for inspection during fabrication, including 100% non-destructive testing of materials and welds;

4) minimum weight and dimensions, which are especially important for SGs of integral reactors;

5) maximum unification of SG units;

6) organization of parallel process flows during fabrication;

7) maximum automation of fabrication process;

8) block-section-by-block-section assembling and replacement of the SG after lifetime exhaustion, that allows a decrease in the joint diameter in reactor vessel;

9) the SG on the secondary side is made of some independent block-sections, that allows isolation of non-leaktight block-sections for both feed water and steam;

10) diagnostics during operation;

11) isolation of any non-leaktight subsection during repair;

12) possibility of using structural units that show good performance in operated SGs.;

13) guaranteed approval of SG performance by representative testing of full-scale cassettes in test facilities.

3. TECHNICAL CHARACTERISTICS

Table 1 gives calculated technical characteristics of the steam generator during nominal operation mode.

Table 1

Parameter	Value
Primary circuit	
Coolant flowrate, kg/s	10140
Pressure, MPa	15.7
Temperature, °C	
at the inlet	325
at the outlet	294
Pressure loss, MPa	0.343
Secondary circuit	
Steam-generating capacity, kg/s	950
Steam pressure, MPa	6.38
Temperature, °C	
feed water	230
steam	305
Pressure loss, MPa	1.47

Table 2 gives the main structural data of the steam generator.

Table 2

Parameter	Value
Number of block-sections, including:	12
rectangular	6
trapezoidal	6

Number of independent subsections	216
Number of steam generating elements	66396
Heat exchange surface, m2	14260
Compactness of heat exchange surface, m^2/m^3	342
Thermal power density in the zone of effective heat exchange, MW/m^3	45
Mass in dry state, t	180

4. DESIGN DESCRIPTION

The steam generator is an once-through vertical cassette surface-type heat exchanger consisting of straight-tube steam generating elements. where steam with the required parameters is generated due to the heat from primary circuit coolant. The flow of media in the steam generator is counter-current: primary coolant flows downwards between tubes inside block-section shrouds, secondary medium (water-steam) flows upwards inside the steam generating elements.

The design of the steam generator is based on know-how decisions which include:

1) chemical composition of tube system materials;

2) design of steam generating elements;

3) method of spacing of the steam generating elements;

4) design decisions for assuring SG hydrodynamic stability;

5) the method of assembling the steam generating elements into the block-section;

6) the device for fixing and sealing the SG block-section into the reactor vessel internals;

7) the technique for obtaining a strong, leak-tight joint of titanium alloy and steel.

The SG consists of 12 identical block-sections 1 (see Fig.1), located uniformly in the annular space between the reactor vessel and core barrel.

Each block-section is individually isolated by valves for feed water and steam.

The block-section consists of steam generating modules (cassettes) 2, header 3, feedwater tubes 4, steam tubes 5, shroud 6, nozzle with the seal 7, strong and leak-tight joints 8 and 9.

Steam generating elements are assembled into steam generating modules 2. Module groups are united into independent subsections, supplied with feed water individually.

The block-section header 3 is intended to organize feed water supply, heat removal and block-section fixing in the reactor vessel. 18 holes for welding of feedwater tubes are located over the header centre and 18 holes for the attachment of steam tubes are located over the periphery.

Feed tubes 4 are in the compartment 10, and intended for the supply of feed water from the header 3 to subsections.

Steam tubes 5 serve for steam removal from the subsection to header 3.

The shroud 6 embraces steam generating elements of the block-section and serves for:

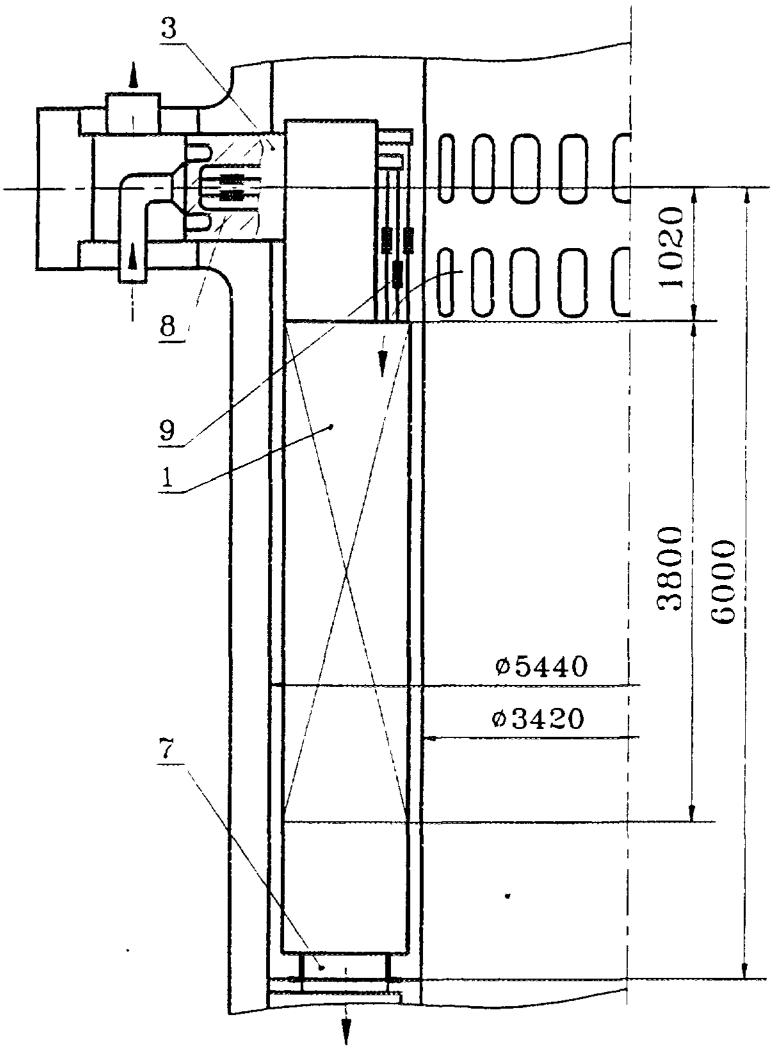
1) the organization of primary coolant flow;

2) forming of specified geometry dimensions of the block-section;

3) tightening of the steam generating elements to exclude vibration.

The shroud is fixed to the block-section header.

The nozzle with sealing 7 is located in the lower part of block-section shroud and intended for prevention of coolant leakage bypassing the steam generating elements and



1. Block-section
2. Module
3. Header
4. Feedwater tube
5. Steam tube
6. Shroud
7. Nozzle with the seal
8,9. Strong and leak-tight joints
10. Compartment

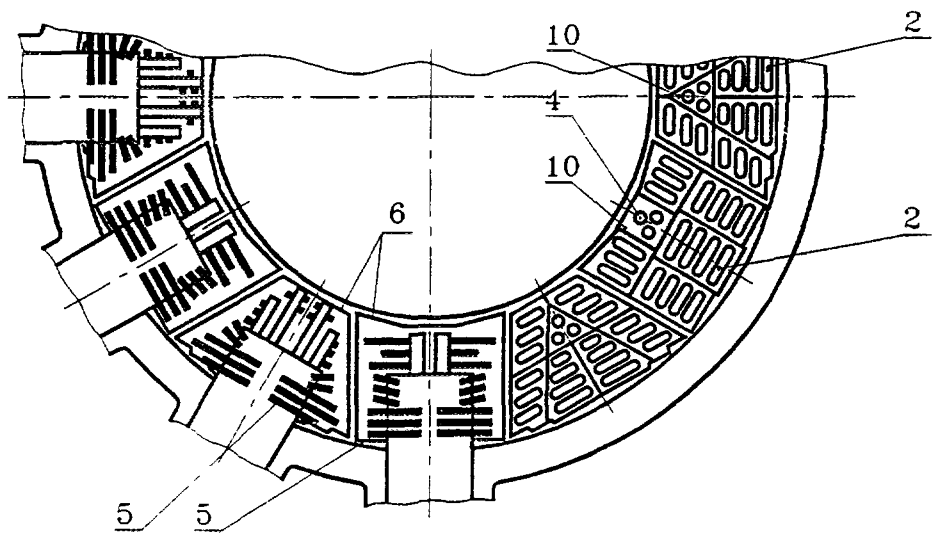


Fig.1

for restraint of the lower block-section part from latteral displacements. The seal design assures longitudinal temperature displacement of the block-section.

Strong and leak-tight joints 8 and 9 ensure the connection of steam generator items, made of titanium alloys to the stainless steel items and are located on each feedwater tube 4 and steam tube 5.

The block-section is installed into the reactor vessel and headers 3 are welded to the vessel from outside.

The steam generator design permits inspection by non-destructive methods during fabrication, including 100% radiographic inspection of all welded joints subjected to primary and secondary pressures.

Unification of items and assembly components by block-section, modules and steam generating elements, application in steam generator design of a minimum quantity of items of standard sizes and assembly of components of different types and dimensions allows organization of parallel process flows for automated manufacturing. This assures high quality of a steam generator and reduces the fabrication cycle.

The steam generator operates as follows. Primary coolant from the pressure chamber enters the shrouds of block-section 1, flows downward between steam generating elements transferring heat to the secondary medium. In the lower part of the shroud the coolant leaves the block-section through the nozzle with a seal 7 and goes further to the core inlet.

Feedwater enters feed tubes 4, then it enters modules 2 and is distributed between steam generating elements. Then water goes upward through the steam generating elements and is converted to superheated steam. Steam from the modules passes through steam tubes 5 and holes in the block-section headers and is supplied to steam chambers and removed from the steam generator.

The steam generator design gives the possibility of diagnostic control in operation and during sheduled shut downs of the reactor by the methods adopted in Russia. It allows estimation of the real state of the steam generator and to ensure its operational safety.

The steam generator design is maintainable and allows to leakage wear in and to isolate any non-leaktight module in case of intercircuit leakage. If one module is isolated, the heat exchange surface is reduced by 1/216 part.

The modular structure of the steam generator allows performance of complex representative testing of steam generator including the confirmation of lifetime characteristics in the rigs of relatively low power.

In the steam generator project design decisions are realized which have resulted in limitations of cross-sections, through which primary coolant leaks in case of some structural failures, including the rupture of a feed tube sheet or of the nozzle in the vessel.

Examination of various SG element failures has shown that the maximum equivalent diameter of the leak is between 5-40 mm.

5. STRUCTURAL MATERIALS

The following principal requirements were taken into account during choice of structural materials for a cassette type straight-tube steam generator:

1) sufficient level of mechanical properties at operating temperatures taking account of long term operation;

2) corrosion resistance in primary and secondary working media;

3) good weldability and provision of required properties of the welded joints.

Titanium alloys are used as structural materials for steam generating elements, tube sheets and module covers, feedwater tubes, block-section shrouds. Titanium alloys developed in Russia for once-through steam generators possess the abovementioned properties and have lower temperature stresses in comparison with steels under equal operating conditions due to lower values of linear elongation coefficient and modulus of elasticity.

The validity of utilization of titanium alloy as structural materials for cassette steam generator has been confirmed by positive operation experience of a large number of numerous steam generators made of titanium alloys in Russia ship plants with PWRs.

Stainless steel (type 18% Cr, 10% Ni) is used as the structural material for block-section headers and section of feed water tubes and steam tubes welded to them.

Titanium alloy tubes are connected to stainless steel tubes using special strong and leak-tight adapters.

6. RESEARCH AND DEVELOPMENT WORK

The steam generator for the VPBER-600 reactor has been developed using experience of development, experimental investigations, fabrication and operation of prototype cassette steam generators.

Beginning from the middle of the 1970s, OKBM has carried out investigations to create cassette-type one-through SGs from straight tube steam generating elements. Work has been performed on the following:

1) theoretical investigations and design developments;
2) experimental studies (development work);
3) technological development of SG units and elements;
4) creation of an up-to-date base for SG fabrication;
5) account of SG operation experience.

When designing cassette SGs all normal and emergency modes of operation were analyzed, the factors exerting the most damaging effect on the SG structure were determined, and the most loaded parts of the SG were revealed.

So, to verify SG characteristics an R&D package has been defined and performed. It includes the following experimental investigations:

1) thermotechnical and hydraulic tests;
2) hydrodynamic investigations;
3) aerodynamic investigations;
4) mechanical tests;
5) thermocyclic tests;
6) hydrocyclic tests;
7) wear resistance tests;
8) vibration tests;
9) shock resistance tests;
10) studies of stability of throttle hydraulic characteristics;
11) studies of SG operability on emergency degradation of feedwater quality;
12) SG field inspections;
13) verification of SG maintainability properties.

All the scope of the R&D work has been performed mainly on OKBM experimental facilities. OKBM has reports with test results for each test type.

OKBM has the universal experimental base allowing it to solve the entire complex of tasks arising when designing SGs. It is evident that each specific SG design should be subjected to acceptance tests aiming at demonstration of the SG characteristics to the Customer. The OKBM experimental base includes test facilities of various powers including those of 45 MW.

Mass production of the cassette-type SGs is established in Russian enterprises which have modern technological equipment including automated complexes.

7. OPERATION EXPERIENCE OF CASSETTE STEAM GENERATORS

Now, cassette SGs are operated as a part of transport VVER reactor plants.

More than 212000 straight-tube steam generating elements operate as part of cassette SGs.

Total operating experience of all active steam generators is more that 500000 hours. Simulating a standard operation model in the laboratory verified lifetime of more than 100000 hours.

Cassette SGs have never failed during operation.

8. CONCLUSION

Positive experience of theoretical and experimental investigations, accumulated in OKBM, the results of cassette steam generator operations in operating PWRs, wide layout potentialities of cassette steam generators in combination with high specific characteristics (compactness and specific power) allowed development of integral type reactor plants, of various powers (including 600 MW electric power) and purposes.

LOCA FEATURES PECULIAR TO AN INTEGRAL WATER COOLED PWR

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Abstract

LOCA initiated by a guillotine break of the pressurizer surge line has been considered in the paper. The failure of two emergency core cooling system (ECCS) trains was also postulated, that turns the considered accident sequence into a beyond the design basis (BDB) class. Basic design characteristics of the ABV reactor and the containment system are presented as well as the factors of much importance to the accident progression.

SCDAP/RELAP5/MOD3.1 was used as the computer code for the simulation of reactor and containment system behavior in the course of the accident. Since a noncondensable driven pressurizing system was employed in the reactor design, the presence of dissolved nitrogen in the primary water was taken into account in calculations.

The important feature of the simulated accident is the primary system refilling with the water of pressure suppression pool driven by the pressure difference between containment system compartments.

Main Design Characteristics of Reactor Plant and Containment System

A beyond the design basis loss-of-coolant accident (LOCA) for the floating nuclear power plant (FNPP) with two water-cooled integral ABV reactors (Fig. 1) has been considered. Primary coolant natural circulation is ensured in the ABV reactor. The multi-sectional once-through steam generator is built inside the reactor pressure vessel. A noncondensible driven pressurizer is connected to the reactor by a pipeline. The design characteristics of the ABV reactor plant are presented in Table 1.

The emergency core cooling system of the FNPP includes two active (high and low pressure) subsystems designed to make up the primary coolant losses even at the guillotine break of any primary pipeline. Besides the ECCS, the active residual heat removal (RHR) system with emergency feedwater pumps (EFP) or the passive RHR system with pressurized accumulators can be used to provide emergency heat removal through the steam generator.

All primary system pipelines, including the pressurizer surge line, have 15 mm dia. orifices to reduce the leak flow rate during a LOCA.

The FNPP accident localization system includes the following leak-tight compartments (Fig. 2):

- Reactor containment shell.
- Pressure suppression pool (PSP) compartment.
- Auxiliary equipment compartment (not shown on Fig.2).

Table 1 Design characteristics of ABV reactor plant

Characteristic	Value
Reactor thermal power, MW	38
Primary coolant	
pressure, MPa	15.4
core inlet/outlet temperature, C	245/327
natural circulation mass flow rate, kg/s	85.3
inventory, kg (m^3)	5500 (7.6)
Secondary coolant	
steam pressure, MPa	3.14
steam temperature, C	290
feed water temperature, C	106
feed water mass flow rate, kg/s	14.7
Pressurizer	
type	gas
total volume, m^3	3.0
water inventory, m^3	2.4

Safety membranes are installed:
(1) in the connection between the containment shell and the pressure suppression pool compartment,
(2) in the connection between PSP and auxiliary equipment compartments (not shown on Fig.2), so that all plant compartments are separated at normal operation. The rupture of the first safety membrane occurs when the containment shell absolute pressure exceeds 0.2 MPa. After that, the mixture of water, steam, and noncondensible gases starts to flow through the gap around the metal-water shielding tank (Fig. 2) to the pressure suppression pool. In this way, a major portion of the steam mixture condenses on the outer surface of shielding tank and the remaining steam condenses in the pressure suppression pool. Cooled noncondensible gases accumulate in the free volume of the PSP compartment. The rupture of the second safety membrane occurs when the absolute pressure inside this compartment exceeds 0.3 MPa.

The above-mentioned metal-water shielding tank surrounds the reactor and contains around 26 tons of water. Under normal operation this water is cooled by special heat exchangers.

The following specific features of the ABV reactor plant design can be highlighted as the factors of high importance to the LOCA progression:

- elimination of large break LOCA thanks to exclusion of large-diameter primary pipelines;

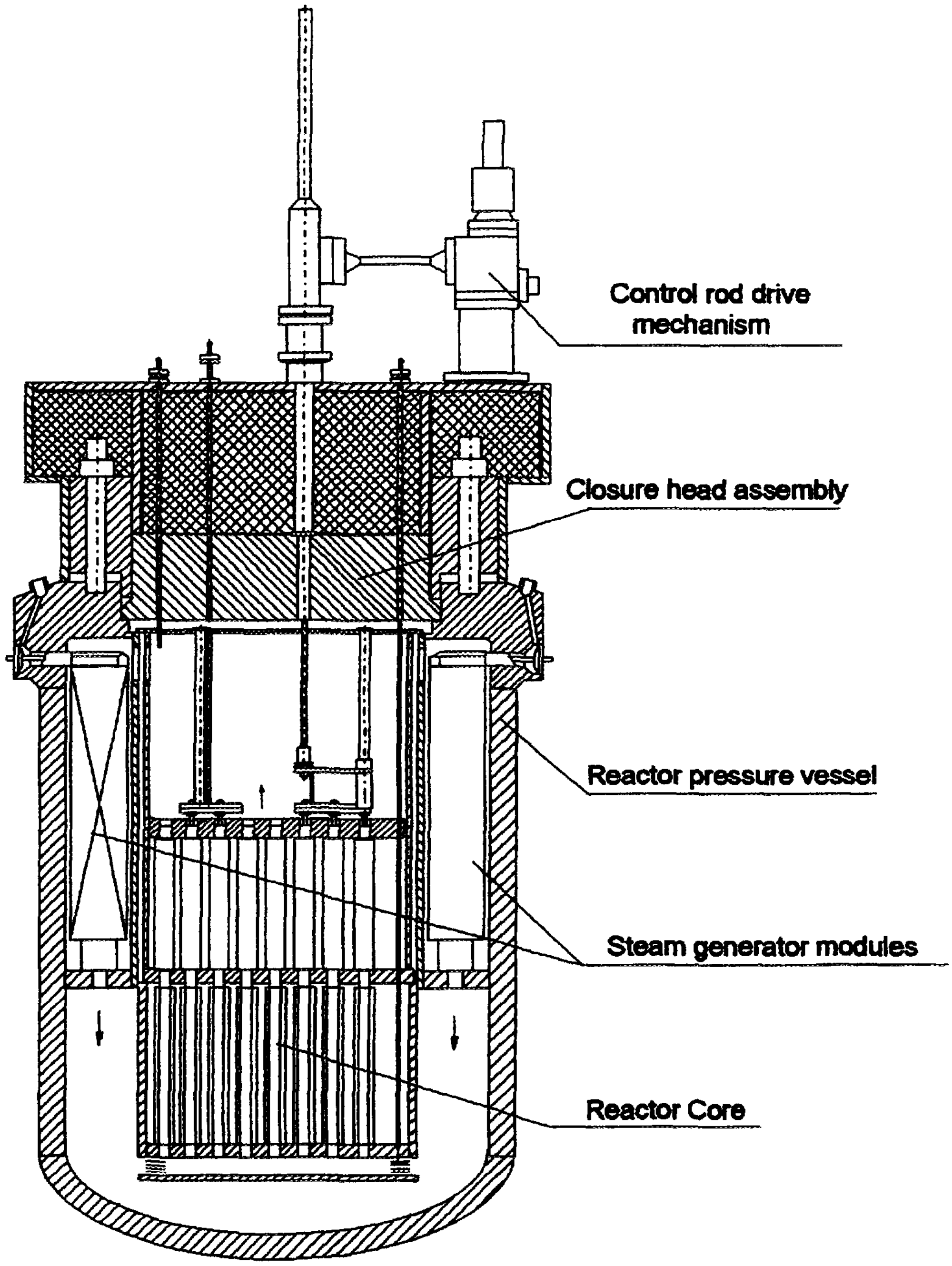

Fig. 1 ABV REACTOR

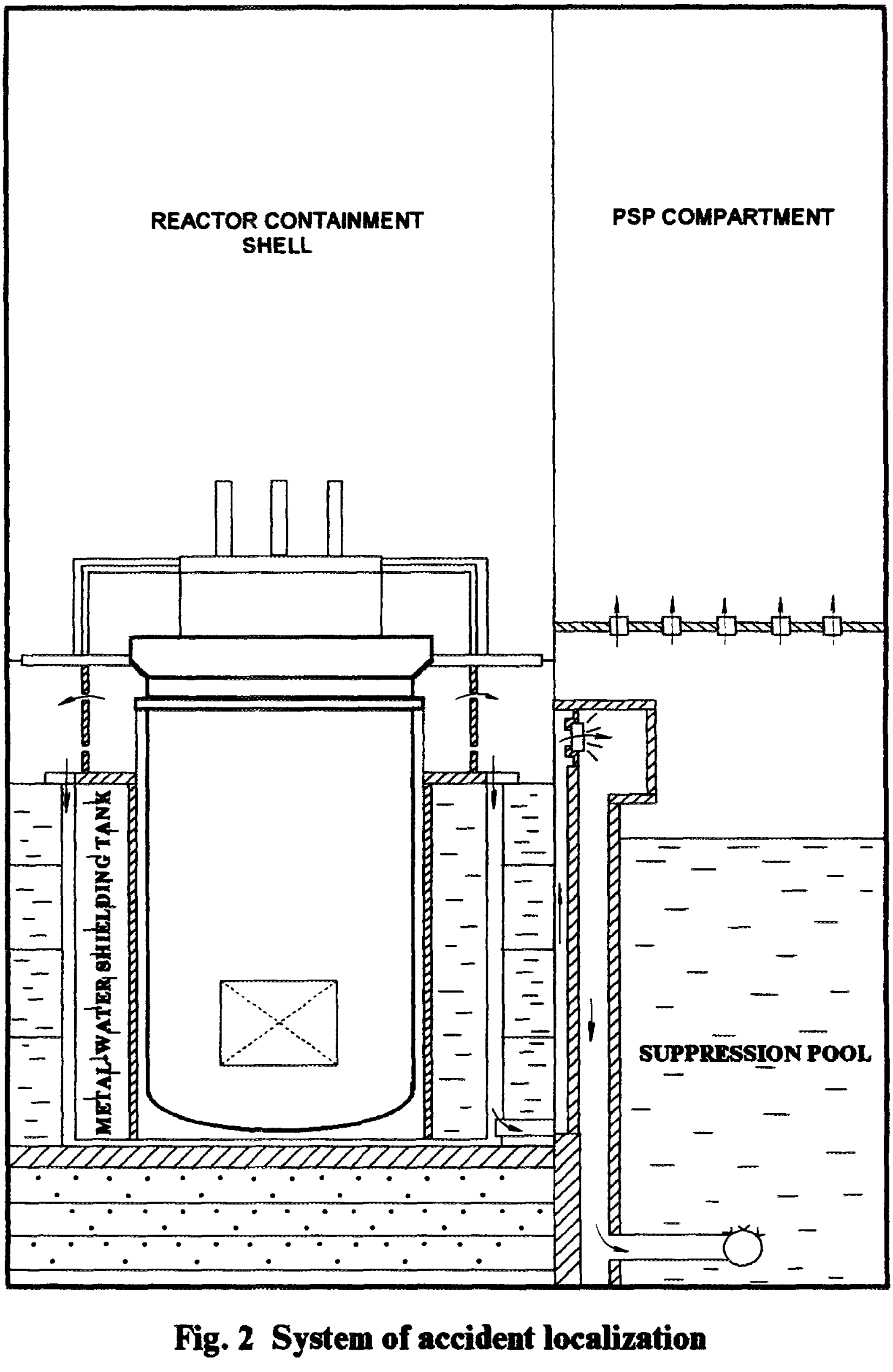

Fig. 2 System of accident localization

The main design characteristics of these compartments are presented in Table 2.

Table 2 Containment system design characteristics

Plant premises / Parameter	Containment shell	PSP compartment	Auxiliary equipment compartment
Initial free volume, m^3	172	40	70
Water inventory, m^3	-	25	-
Total heat exchange surface of compartment walls and equipment, m^2	228	85	22
Heat exchange surface of the walls cooled with external air, m^2	128	38	22
Total mass of compartment walls and equipment, kg	35600	13300	20600
Maximum permissible pressure in compartments, MPa	0.6	0.6	0.6

- employment of the outflow restrictors to decrease the primary coolant loss during a LOCA;
- upper location of the penetrations of the primary pipelines through the reactor pressure vessel;
- large water inventory above the core;
- favourable conditions for the emergency heat removal through the steam generator thanks to the primary system integral design;
- nitrogen presence in the primary system, including nitrogen dissolved in the primary coolant during reactor plant operation in transient regimes. The analysis of this factor is one of the main objectives of the study.

Accident Scenario

The postulated LOCA scenario is based on the following assumptions and events:

1. The accident is initiated by the double-ended break of the pressurizer surge line near the reactor pressure vessel.
2. Both trains of the ECCS fail to operate.
3. Additional failure of the shielding tank cooling system is supposed.

4. Full power operation of the reactor plant is assumed before the start of the accident.

5. Reactor scram occurs when the reactor pressure reduces to 12.5 MPa.

6. The EFP starts to operate after reactor shutdown and provides around 15 m^3/h of feedwater at a temperature of 40 C.

7. The total amount of nitrogen dissolved in the primary water inside the reactor vessel is assumed to be 9.4 kg. The assessment of initial nitrogen concentration in the reactor coolant was based on the Henry law.

8. No operator action is performed to mitigate the accident.

Simulation Results

The SCDAP/RELAP5/MOD3.1 computer code was used to determine both the reactor and containment system behavior in the course of the accident. Several modules of the code were modified for this study to simulate the process of dissolved nitrogen release from the primary water to the vapor phase within the reactor vessel.

The computer simulation results showed that the reactor coolant leak rate through the pipe break decreases rapidly during the initial 30 seconds of the accident. Its value reduces from 16.5 kg per s at the accident beginning to about 3 kg per s, when the reactor end of the pressurizer surge line becomes uncovered and the leak flow changes from liquid to vapour. The subsequent discharge rate of steam follows the primary pressure (Fig. 3).

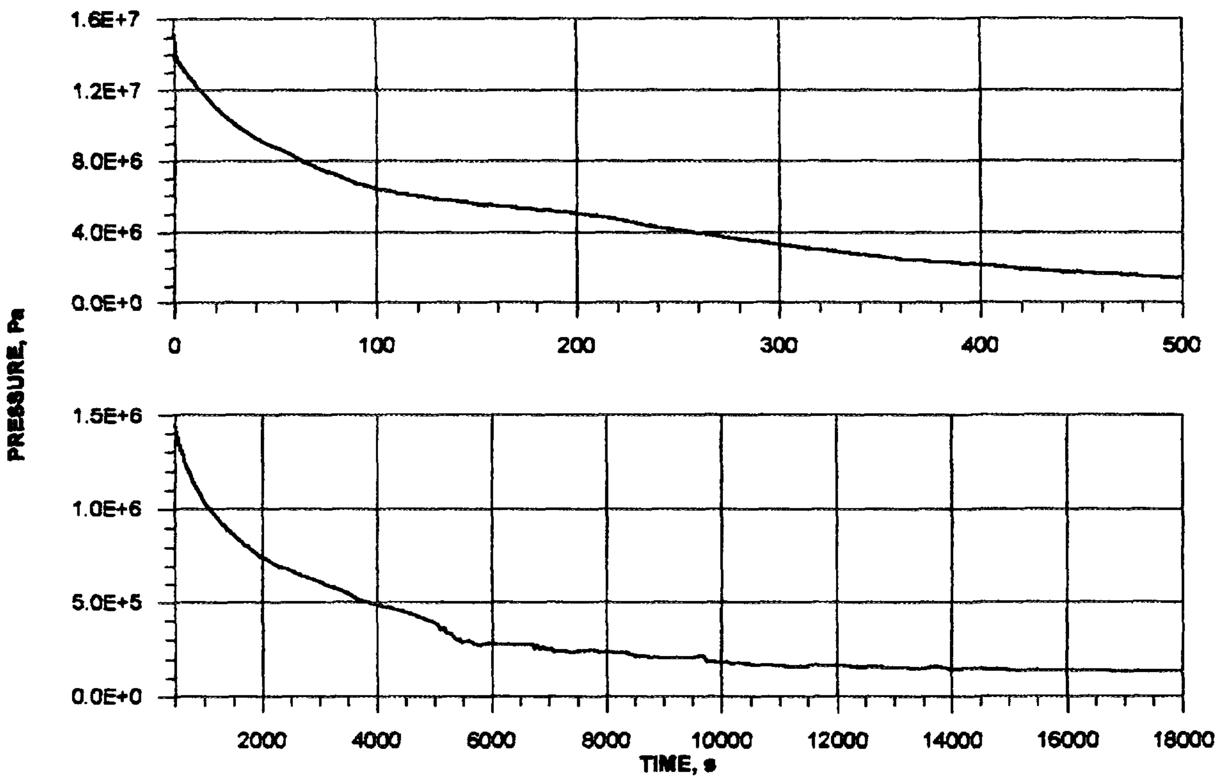

Fig. 3 Reactor pressure change during the accident

The heat removal from the primary system through the steam generator has an essential influence on the reactor coolant discharge rate and the rate of reactor pressure reduction. Heat transferred to the steam generator secondary exceeds the energy transferred to the coolant from the core for almost the whole duration of the accident, causing a decrease in the primary system pressure and temperature. However, the process of nitrogen release from the primary water (Fig. 4) leads to a reduction in the steam condensation rate at the primary side of the uncovered steam generator tubes. The break flow reduces at a lower rate as the result of this process. A significant part (about 20 %) of the released nitrogen mass remains in the vapor phase inside the reactor pressure vessel and localizes mostly above the water level inside the steam generator.

The containment shell pressure continues to rise even after the first safety membrane rupture at 16 s and reaches its maximum value just before the second membrane ruptures at approximately 2650 s (Fig. 5). The values of containment atmosphere pressure and temperature do not exceed permissible design limits of 0.6 MPa and 150 C during the accident progression. Their maximum simulated values are 0.34 MPa and 130 C, respectively.

A steady pressure decrease in all containment system compartments begins at ~5000 s of the accident. At the same time, a positive pressure difference between the containment shell and the PSP compartment starts to reduce to zero and becomes negative at ~7000 s. This result is explained by decreasing discharge rate of the primary coolant and continuing heat removal from the containment atmosphere to the containment vessel walls and interior equipment, including the metal-water shielding tank.

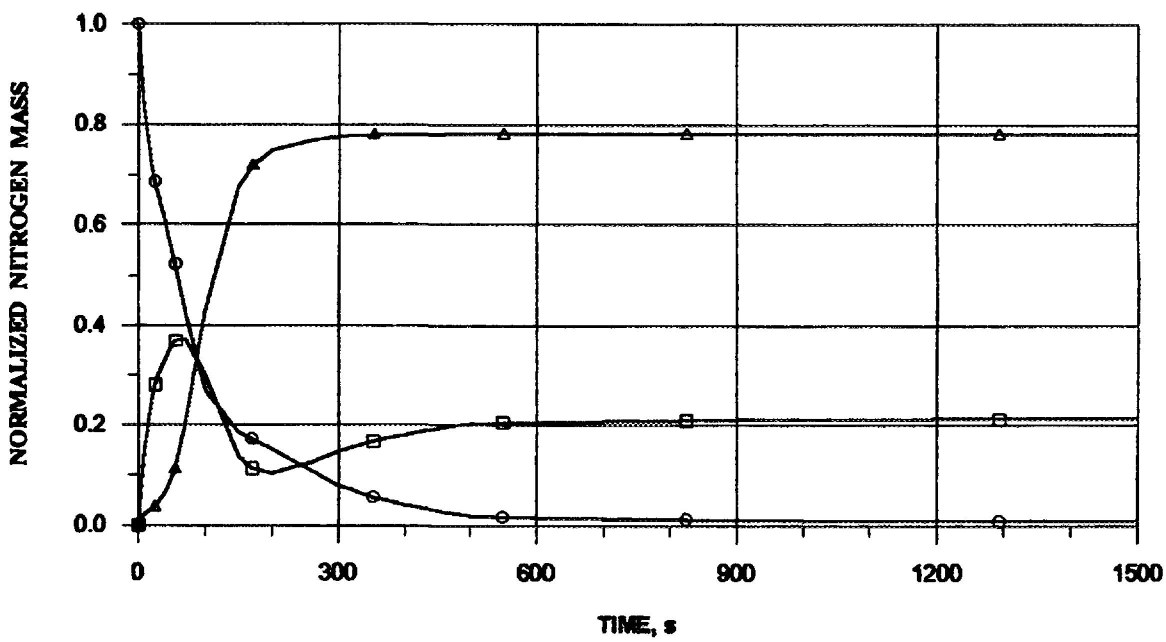

Fig. 4 1. Nitrogen content in coolant inside reactor vessel (O – primary coolant liquid phase, □ – primary coolant vapor phase) 2. Nitrogen discharged from reactor vessel (▲)

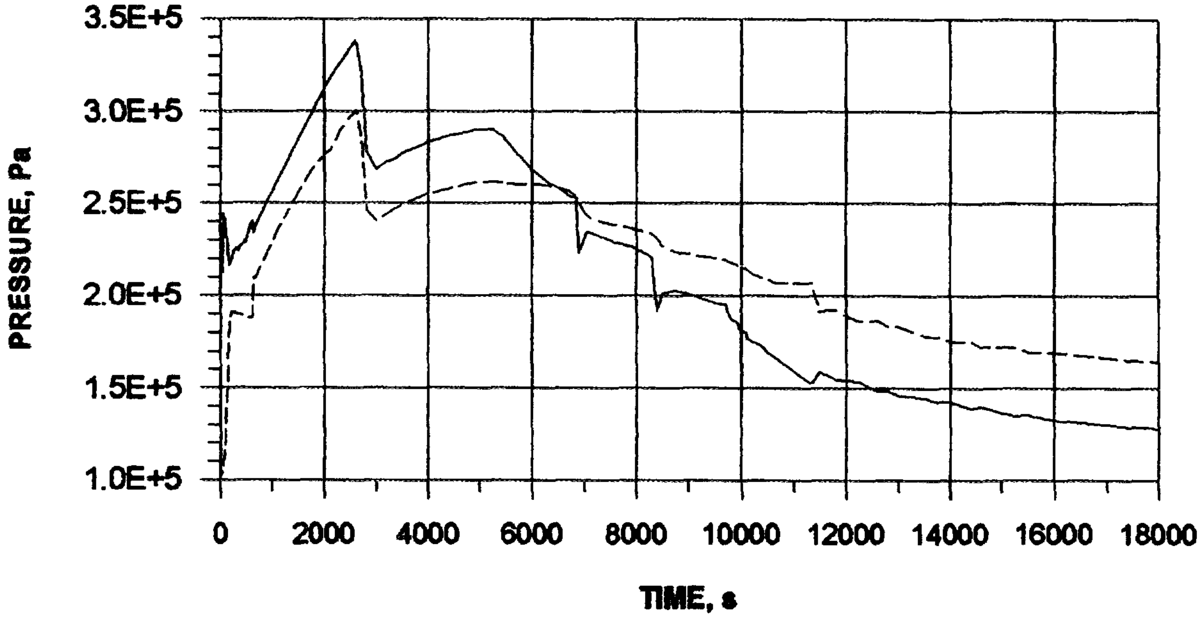

Fig. 5 Pressure inside containment shell (——) and PSP compartment (- - - -)

During the next phase of the accident, a gradual displacement of water from the PSP compartment to the containment shell is observed as the result of increasing pressure difference between these compartments. At ~10000 s, the rising water level inside the containment shell covers the pressurizer surge line break located at the elevation of ~3.5 m above the shielding tank bottom level. After this event, the reactor begins to refill with suppression pool water (Fig. 6). The mass of coolant inside the reactor pressure vessel increases up to the value of ~6000 kg and stabilizes beginning at ~13000 s. During the last 5000 s of the transient, the calculated leak flow rate has a small value. The direction of the flow alters irregularly, so that the coolant mass within the reactor remains almost invariable for the rest of the accident simulation.

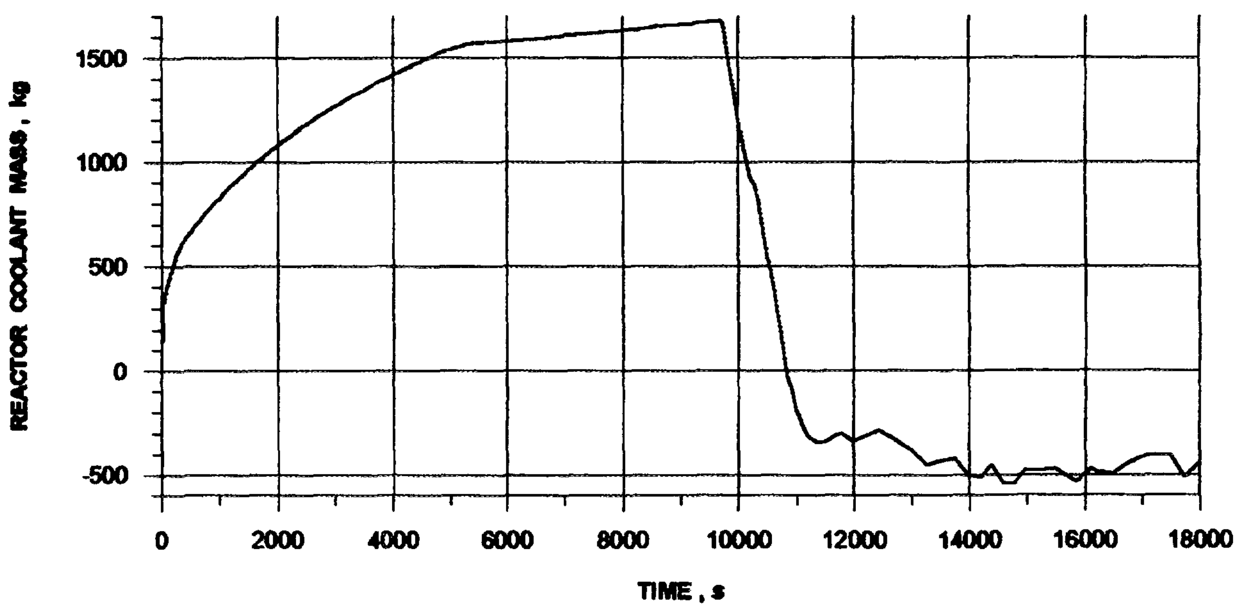

Fig. 6 Total loss of reactor coolant

The results of the accident simulation indicate that a sufficient inventory of the primary coolant remains within reactor vessel to maintain reliable core cooling. The critical heat flux is not reached in the core during the accident progression.

The further state of the reactor plant will depend on assumptions concerning the degree of leak-tightness of the containment system compartments, RHR system working conditions, and operating staff efforts to activate failed safety systems. The available grace period is long enough to undertake measures for accident control.

Conclusions

In the course of performing the simulations, it was found that:

1. Neither core uncovering nor the critical heat flux in the core was reached without any operator action during the accident progression.

2. The process of nitrogen release from liquid to vapour phase within the reactor pressure vessel led to increased total loss of the primary coolant, but the values of containment shell atmosphere pressure and temperature did not exceed their permissible design limits.

3. Reactor refilling with the water of pressure suppression pool driven by the pressure difference between containment system compartments occurred during the last phase of the accident.

XA9745986

STUDY OF THE KINETICS OF THE STEAM-ZIRCONIUM REACTION USING A ONE ROD ASSEMBLY MODEL

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Abstract

The results of the investigation of steam-zirconium reaction kinetics at the HPE simulator surface are presented in the paper. The dynamic characteristics of the hydrogen production resulted from the heated surface dryout are determined.

INTRODUCTION

The necessity of the knowledge of the laws of hydrogen generation during the interaction between the HPE shells made of the zirconium alloy and water coolant is highly increased when the problem of severe out-of-project accident was initiated, so their evolution is determined by the hidrogen discharge.

In [1] on the base of publications available the analisis was made of the hidrogen generation processes under severe accidents resulted from the circulation break off. From the estimations presented there regarding the amount of generated hydrogen during some first hours after the accident, it follows that the general source of the hydrogen is the reaction of zirconium with steam-water mixture. For example, the velocity of hydrogen generation may reach the value of 726 kg/h for the ordinery commercial APP USA.

Among the latter publications dealing with this problem the paper of the Japan authors [2] is remarcable. In this paper the results of the in-pile experiments are presented devoted to the study of hydrogen generation during the steam-zirconium reaction when the accidant situation was initiated by reactivity. The HPE was cooled by subsaturated water in the film boiling regime. The amount of the hydrogen generated was determined by the void fraction gauge at the channel exit and from the metallografic examinations also. In spite of the significant discrepancy between the results obtained by these two methods, the data allow to make conclusions about the strong temperature influence upon the hydrogen production.

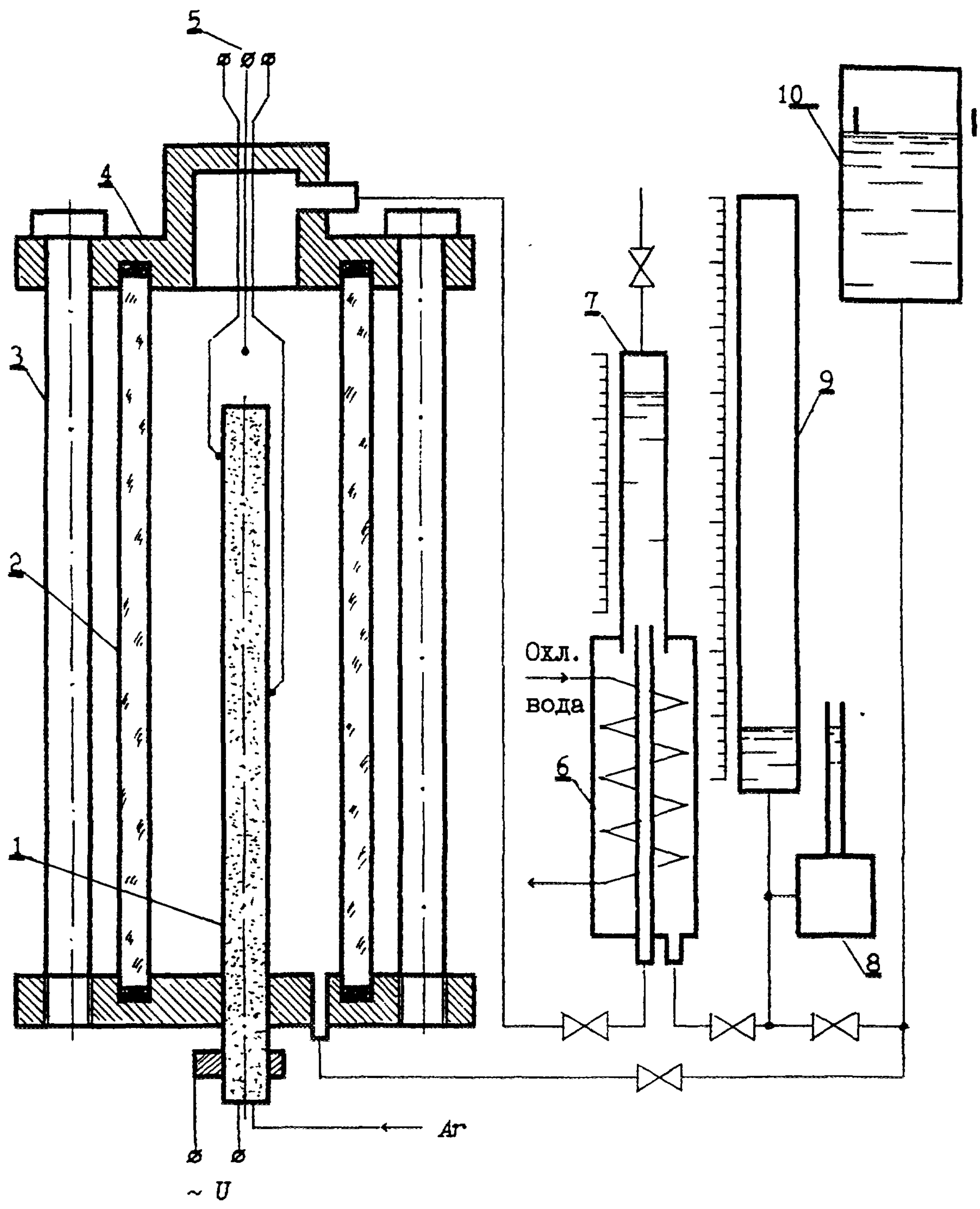

Fig. 1. Scheme of the Facility for Steam-Zirconium Reaction Investigation.

1 - Heating Element (HPE Simulator); *2* - Transparent Body; *3* - Bars; *4* - Фланец; *5* - Thermocouples; *6* - Refrigerator; *7* - Hydrogen Accumulator; *8* - Level Gauge SAPPHIRE-22DD Type; *9* - Replacing Tank; *10* - Water Feed Tank.

The aims of the present paper are as follows: experimental study of the hydrogen generation process as a result of the HPE zirconium shell oxigation in steam-water atmosphere under conditions of assembly dryout due to the circulation failure; using of the data obtained when developing the corresponding codes.

Presented in the paper are the results of first experiments having, in general, methodological purposes. At the one-rod assembly model the peculiarities of the hydrodynamic processes of its dry-out was visualised and the reliable method of hydrogen production measurement was veryfied.

EXPERIMENTAL FACILITY

The experimental facility is presented schematically in Fig. 1. Its basic elementa are the next:

- heating element: spiral or with internal heater *1*;
- assembly body made from the transparent quartz-glass tube for assembly dre-out process visualization *2*;
- bars *3*;
- flange *4*;
- thermocouples *5*;
- refrigerator *6*;
- calibrated measuring glass for hydrogen accumulation *7*;
- level gauge (volume) in *8*;
- expansion tank for collection of water replaced by steam and hydrogen *9*;
- water-provided tank *10*.

The collection and treatment of the information is fullfilled by the computer-controlled system.

During the process of the model dry-out a steam-water mixture is directed to the refrigerator where it is condenced, while hydrogen is led to the accumulator. Hydrogen replaces water to the calibrated replacing tank. The pressure difference gauge registers the change of water level in replacing tank vs. time what enables to identify the moment of the steam-zirconium reacton beginning and to measure the amount og the hydrogen produced. The thermocouples are used for measuring the temperatures of the simulator surface and that of steam. For the simulator to be vacuumed and and after that filled with argon serves a special system.

EXPERIMENTS

In the experiments carried out two types of heater were used: one of spiral type and another as a rod with internal heating.

The spiral type heater was fabricated of stainless steel capillary tube 4 mm in diameter. Inside this heater a small 60 × 5 mm plate zirconium was arranged. The chromel-alumel thermocouple was aranged at the plate centre.

The rod (HPE simulator) was made of the zirconium tube ∅9.1 × 0.5 mm. It was heated by molybdenum-made electric heater arranged inside which was insulated from the rod wall by pressed magnesium oxide. The total length of the simulator was 550 mm and that of the heated zone was 300 mm. The AC voltage was supplied to the simulator shell ahd heater from the transfomer. The internal part of the simulator was filled with argon.

The technique of the experiments carrying out was as follows. After the facility being filled with distilled deairated water one turned on simultaneously the simulator heating and the system recording in time the temperatures of the simulator surfase and exitingd steam and water level in accumulation tank. Once the electric power was turned on, the simulator surface reached the saturation temperature and the boiling took place. The beginning of boiling was accompanied by intensive dry-out of the model and consecutive increasing of the rod surface temperature. The front of temperature increase was spread down the rod following the water level in assembly. The experiment was finished when the volume gauge signal reached its extreme value.

RESULTS OF THE EXPERIMENTS

The experiments have shown that the noticeable hydrogen production due to the steam-zirconium reaction took place at the temperature about 650 °C. It was observed both visually as hydrogen bubbles in the accumulator and by record of the water level changes in the tank *9*. In the teats with spiral heater where the temperature didn't exceed 800 °C, it was marked also that the amount of hydrogen produced diminished with every consecutive turning on of the heater. It apparently may be explained by the

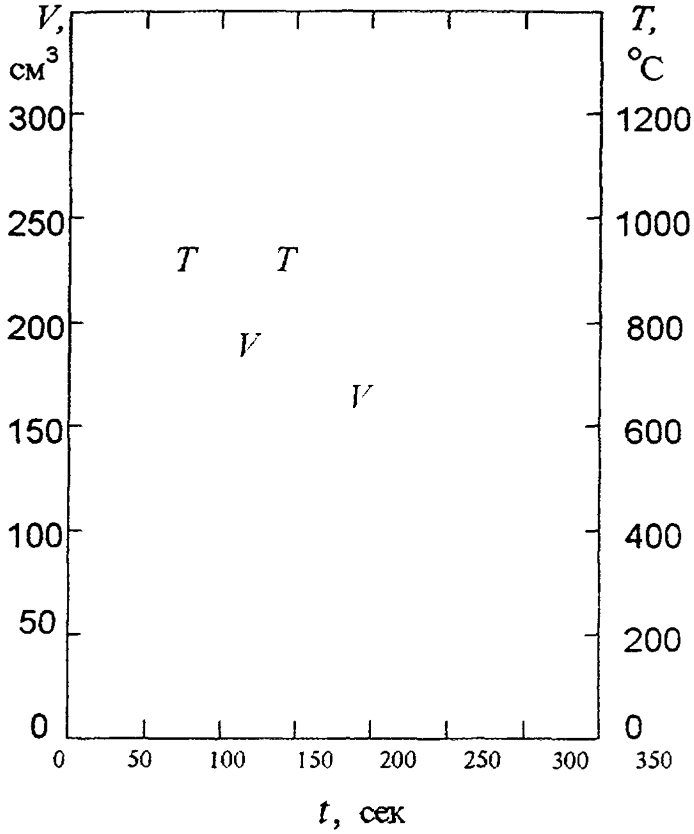

Fig. 2. Change of the water volume in accumulator tank and simulator surface temperature vs. time.
———— - Test N.1; – – – – - Test N. 2.

influence of the zirconium oxide layer at the rod surface on the steam access to the clean ziconium surface and, finally on the hydrogen production. At the test with the rod simulator, however, at the temperature of 1000 °C there was a local break-off in oxide film. It may be resulted from both by the rod distorsion and by the hydrodynamic exerting of the steam-droplet mixture on its surface.

In Fig. 2 the results of two tests carried out at the identical initial conditions with the rod HPE simulator are shown. The electric power in both tests was 1.5 kW. Fig.2 illustrates evidently the dynamics of the assembly dryout and the hydrogen discharge during steam-zirconium reaction. The curves *V(t)* and *T(t)* are correlated with each other qwite well. So, the signal from the level gauge being initially constant begins to increase with the beginning of the steam-zirconium reaction. It's the moment when the rod temperature reaches its maximum value. At the meam temperature of 1000 °C the amount of the hydrogen produced during 170 s was as much as 143 cm³ in the test No.1 and 110 cm³ in the test No.2. The considerable rod deformation over the length was marked.

In [3] the measuring of the amount of hydrogen produced in the steam-zirconium reaction was carried out at the temperature of 1000 °C. The measuring technique was in determination the mass increase of small zirconium specimens in steam atmosphere during the definite time and consecutive calculation the amount of hydrogen. The mean value of the hydrogen volume produced by 1 cm^3 of the specimen surface per 1 s was about 0.0136 cm^3. In our tests this value was as much as 0.0125 cm^3. The agreement is quite satisfactory. It may be consider as a confirmation of the reliability of the dilatometric method used for the measuring the amount of hydrogen produced.

CONCLUSONS

Experimentally it has been shown that the dilatometric method of the amount of hydrogen produced during steam-zirconium reaction is a perspective one for carefull investigation of this reaction kynetics.

The visualization of the process of the one-rod assembly dry-out has been carried out.

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LOSS OF COOLANT EXPERIMENTS FOR THE TEST NUCLEAR HEATING REACTOR

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Abstract

A Series of tests has been done for the three cases (breaking pipe opened in the gas plenum, near the liquid level and submerged in the water) in the Test Heating Reactor.

Experiments show that the three cases of LOCA (Loss of Coolant Accident) have different patterns. In the case of a breaking pipe connected into the gas plenum the quantity of lost water is independent of the diameter of the breaking pipe. In the case of a breaking pipe located near the liquid level, the quantity of lost water depends on the location of the pipe. In the case of breaking pipe submerged in the water, all water above the break will be discharged.

The patterns of the discharge for the three cases are given in the paper.

Key words: Loss of coolant; Nuclear heat reactor

1. Introduction

District heating reactors must be built near the heating load (heat grid) . That means they must be built near cities. So the safety requirements for DHR are very strict.

For integral reactors protection against loss of coolant is one of the most important considerations.

On the pressure vessel of a integral reactor, there is no penetration pipe with big size, LOCA is possible only through pipes with small sizes. The possible ways of LOCA are as follows:

① Water discharge through the safety valve.

② Water discharge caused by a pipe breaking (The pipe is opening in the steam plenum) .

③ Water discharge caused by a breaking pipe (the pipe is opening under the water level.

For checking the design safety of the 5MW Test Heating Reactor (NHR-5) and investigating the characteristics of LOCA during the discharging process, a series of tests have been done on the model thermohydraulic loop in INET.

The test model loop was built on the base of similarity theory for the NHR-5.

The loop consisted of two circuits. The first circuit included two heating test sections (to model fuel assemblies) , a rising pipe (to model the chimney) , steam-water separator, steam condenser, cooler, down pipe, valves (to model resistances) and measurement apparatus.

The Condenser and cooler are cooled by the water of the secondary circuit. Secondary circuit water transfer heat to the environmental air by an aircooler.

In order to study the effects of drain position on the drain process, three positions of the drain orifices have been selected. The first drain orifice is located at the top of the loop. It is used to model the case of " safety valve opened and can't be reset" . The second orifice is located in the steam plenum of the separator. It is used to model the case of a breaking pipe opened in the steam plenum. In the NHR-5, most pipes penetrating the pressure vessel are connected to the steam plenum. The third drain orifice is located under the water level. It is a pipe, opening under the water level at about 3m. It is used to model the case of a boron-injection pipe breaking.

The diameter of the heating rods, heating length, height of the chimney, resistance coefficient at the inlet of the heating section, total resistance of the loop, density of the heating power and parameters are the same as those in the 5MW Test Heating Reactor, in order to keep the geometric and hydraulic similarity conditions.

In the water discharge test through the safety valve, according to the geometric similarity, the diameter of the drain orifice should be selected at 4. 3mm. To study the relationship between the orifice

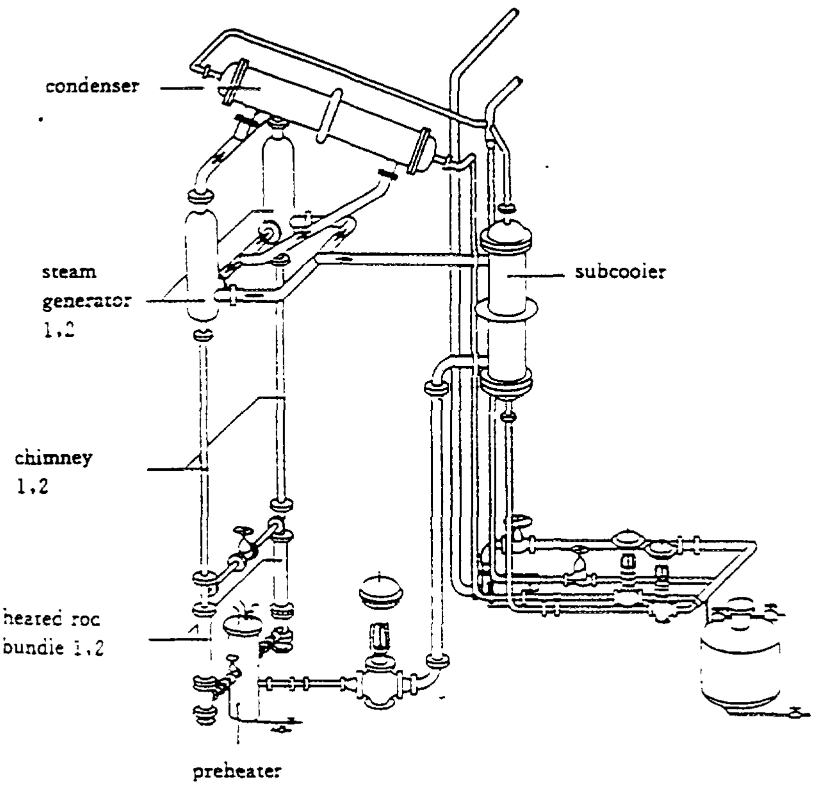

Fig.1. Schematic diagram of the test loop

diameter and quantity of drain water, besides 4.3, orifice diameters 6.0 and 8.5 m m are also selected in the tests.

During the test, the pressure of the steam plenum, water temperatures at the inlet and outlet of the heating section, drain water flow rate, and surface temperature of the heating elements are measured. Pressure and pressure defferences are measured through transducer type 1151. Drain water quantity is measured by weighting the condensate and measuring the water level.

2. Discharge test through the safety valve

After opening the safety valve and cutting the heating, the pressure of the test loop decreased fast. Part of water vaporized. The drain rate decreased with decreasing of the pressure, until the loop pressure was balanced with the back pressure (drain stoped).

A typical test curve of the pressure decreasing with time is shown in Fig.2, as well as a curve calculated according to gas critical flow method. The two curves fit well enough. It means that the drain

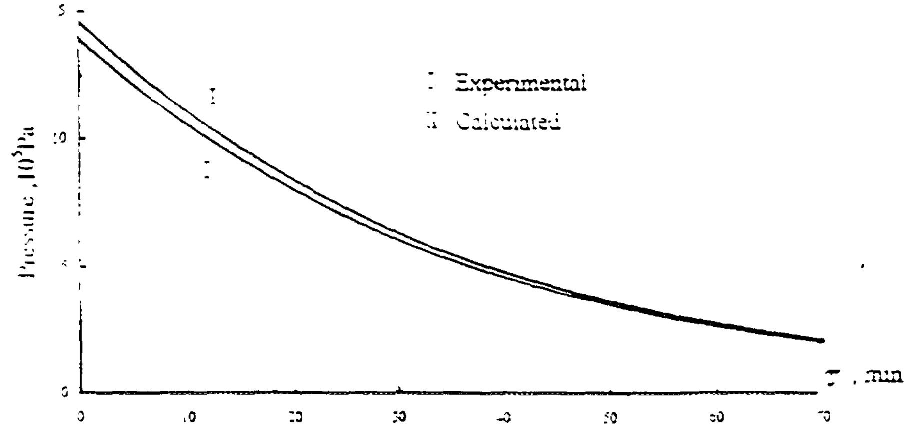

Fig.2 Pressure change in the upper plenum

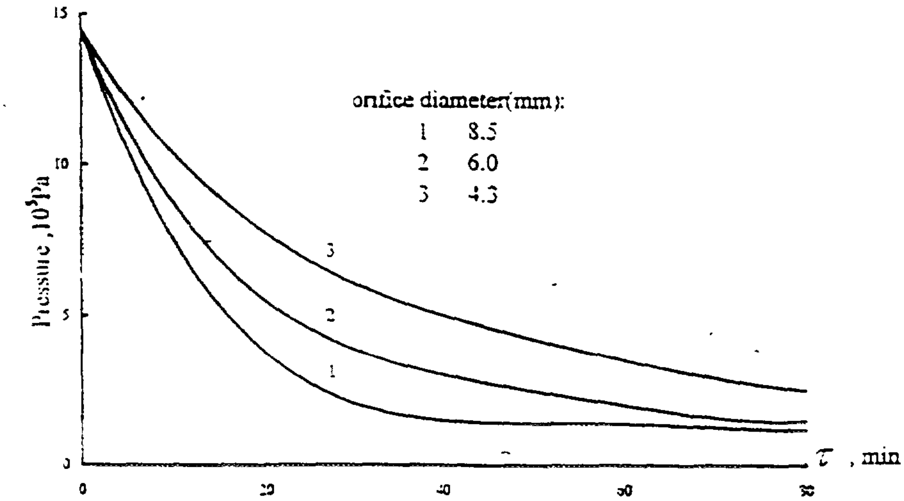

Fig.3 Effect of the orifice diameter on the pressure in the upper plenum

through the safety valve essentially is a gas critical flow. It is easy to understand, because during the drain process, the ratio of back pressure to the loop pressure most of the time is lower than the critical pressure ratio. In the drain test through the safety valve the discharge orifice is located at the top of the loop. So almost is only vaporized steam is discharged.

That the drain water quantity can be calculated by the critical flow method is an important conclusion for the design.

Experiment shows another important fact that is for the same initial pressure and temperature the drain water quantity will be the same, no matter the orifice diameter is big or small. Only the balance time will be shorter or longer (longer balance time refers to the smaller orifice diameter) . (See Fig. 3) . This means that, at the initial conditions of the design parameters (pressure and water temperature) , for the Test Nuclear Heating Reactor, during the process of drain through the safety valve the total quantity of drain water is about 18 ~ 20 % of initial water inventory, no matter whether the safety valve is fully opened or partly closed.

For the Test Nuclear Heating Reactor (NHR-5) to cover the reactor core and to ensure necessary cooling only about 40 % of initial water inventory is needed. So the NHR-5 is enough safe during a LOCA through the safety valve. That the total quantity of drained water no more than 18 ~ 20 % of initial water inventory is confirmed by the calculation of heat balance. So it is creditable.

3. Discharge test of a pipe break from the steam plenum

The main difference between discharges through the safety valve and a pipe break from the steam plenum is the position of the discharge orifice. The pipe from the steam plenum is located lower than the safety valve. In normal regimes it opens into the steam volume. But during the discharge process, it may some times be covered by water or two phase liquid. In this case the total drain quantity of water depends on the opening is covered or not.

Discharge from different heights (distance to the initial water level) has been tested.

In the case of small heights at the beginning of the discharge process the level of two phase liquid (cause by the vaporization) may be higher than the discharge orifice. It causes a mixed discharge of water and steam, and increased the total drain quantity. Specially for the bigger orifice diameter the discharge process is stronger, the total drain quantity is increased notable. This can be seen in Fig. 4. For the case of mixed discharge, the total drain quantity may be increased from 19 % to 25 % of initial inventory (orifice diameters are increased from 4. 3mm to 8. 5mm respectively) . The less the height difference between the initial level and discharge orifice, the more the total drain water quantity.

Tests showed that when the water inventory increased from 71% to 84% of total volume the total drain water quantity increased from 17.1 % to 20.3% of the total water inventory.

4. Discharge test of breaking pipe from the water volume

In the NHR-5 some pipes are connected under the water level, such as the boron injection system pipe, the pipe for the hydraulic driving facility for control rods etc. A pipe break of this type is more dangerous. In the case all the water above the inlet of the pipe will be discharged as well as some water lower than the inlet because of siphonage.

A discharge test has been made to model a pipe breaking in the boron injection system. The boron injection pipe is opened under water at a level equivalent to the middle of the chimney.

In the test steam content, drain water quantity, liquid level, pressure and their transient characters have been measured.

The typical pressure change during the test is shown in Fig. 5. The change of drain water quantity is shown in Fig. 6.

From these figures it can be seen that at the beginning of the drain process discreasing of the pressure wasn't very fast, but increasing of the drain water quantity was quite considerable. This can be understood, because at the beginning water was discharged, vaporized steam compensated for the change of water volume, so the pressure changed slowly. When the liquid level was lower than the opening of the discharge pipe, steam and gas began to drain. The pressure decreased faster, but the drain water quantity changed reasonably slowly. The decrease of the liquid level from linear changed to slower. Eventually draining stopped because of losing heat and pressure.

When the liquid level was lower than the opening of the pipe, some water was drained because of siphonage. The greater the distance between the pipe opening and the liquid level, the less the drain water quantity. Consequently there was a smaller the decrease of the level.

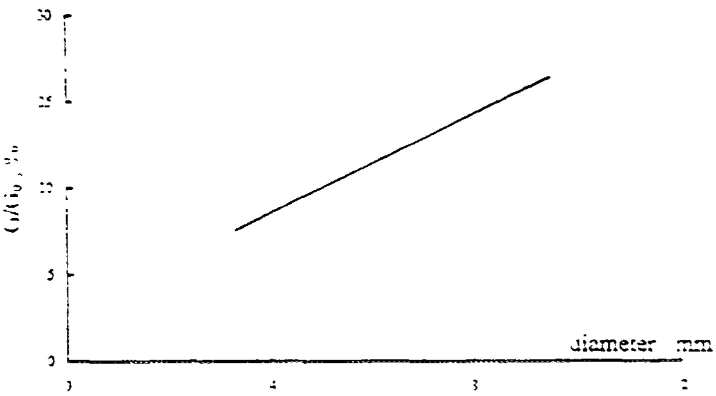

Fig.4 Effect of the orifice diameter on the discharged water quantity

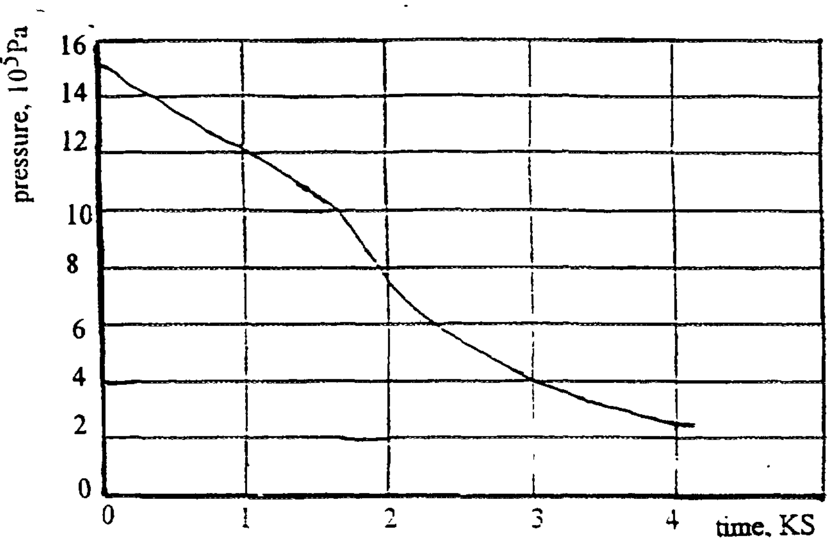

Fig.5. Pressure change during the discharging process

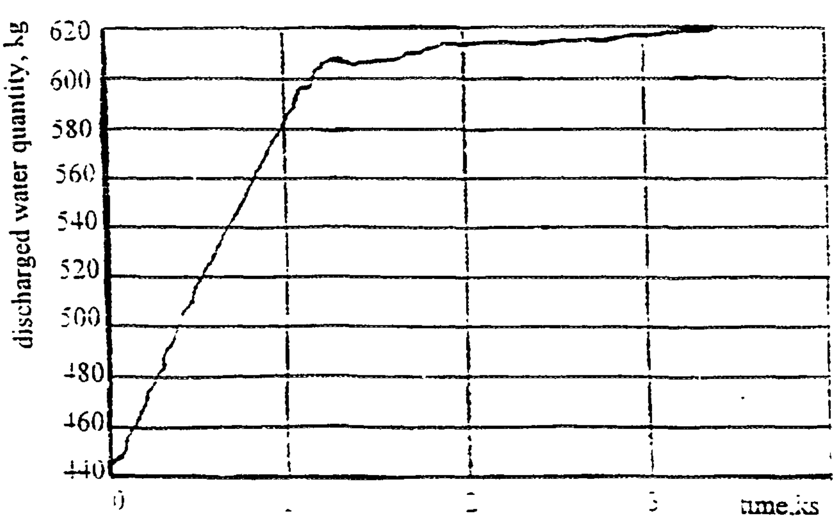

Fig.6. Change of the discharged water quantity with the time

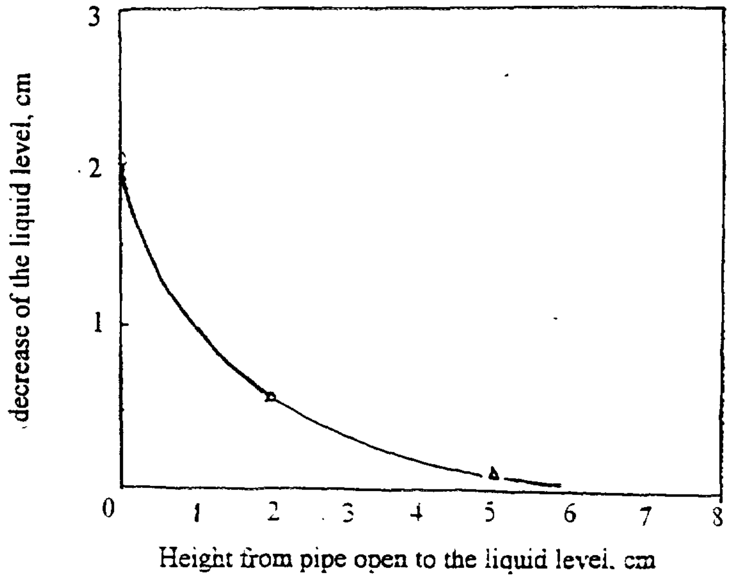

Fig.7. Decrease of the Liquid level caused by the Siphon Effect (experimental reseelts)

The relationship between the drain water quantity and the height (to the liquid level) of the opening is shown in Fig. 7. From this can be seen that when the height is more than 5cm, the height almost doesn't effect on the drain water quantity.

5. Preliminary conclusions

From the above experimental study the following conclusions can be made.

(1) The total drain water quantity during the safety valve discharge is no more than 20 % of the initial water inventory, no matter whether the safety valve is full open or not. (this is confirmed by the heat balance) .

The drain velocity and flow rate can be calculated by the critical flow theory.

(2) For the discharge caused by breaking of a pipe from the steam plenum, the drain quantity depends on the position of the pipe in the steam volume. During the discharge process water may be drained discontinuously. The bigger the pipe diameter, the more the discharged quantity. The total drain water quantity is more than for the opening of the safety valve at the same discharge orifice diameter.

(3) For the discharge from the water volume, the drain quantity is more than that of a drain from the steam volume. All water above the pipe opening will be drained. When the level is lower than the opening of the pipe, the drain quantity depends on the height difference between the liquid level and the opening of the pipe. In a certain range the bigger the height difference, the smaller the drain quantity. Under the conditions of our tests, when the height difference is more than 5cm, its effect on the quantity can be negligible.

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OPERATIONAL, MANUFACTURING AND DECOMMISSIONING ASPECTS

XA9745988

CAREM: OPERATIONAL ASPECTS, MAJOR COMPONENTS AND MAINTAINABILITY

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Abstract

The paper presents the design related aspects of operation and maintenance of the CAREM reactor and the principal features of its main components.
The paper covers three main topics: operational aspects, major components and maintainability

Operational aspects
A strong negative thermal coefficient, the use of burnable poisons to compensate burnup and no use of soluble boron for reactivity control characterized reactor control.

Hydraulically driven control rod drives are fully contained in the pressure vessel.
The following research and development activities are being carried on:

A critical facility for testing main core characteristics is in final construction stage.

A full scale model of control rod drives is currently under test.

A full scale model of one steam generator will be constructed and tested.

Major components
Pressure vessel: The empty pressure vessel weights 100 tons. This fact facilitates both its manufacturing and transport. Internals and steam generators will be mounted on site. Being the reactor self-pressurized, no pressurizer is included.

Steam Generators: Twelve once-through steam generators are symmetrically placed inside the pressure vessel. Specific design aspects are discussed in the paper.

Containment: The containment is of pressure suppression type. Second shutdown system, pressure relief tank, and equipment and installation for manual reactor refueling and for handling of RPV internals, are all placed inside the containment. Provisions are also made for accommodating RPV internals during refueling and maintenance operations.

Maintainability
Lay-out: The balance of plant lay-out is conventional. For nuclear island layout, attention has been paid to the fact that the containment will not be accessible during reactor operation. This fact imposes special demands on equipment reliability, that will assure high plant availability.

In service inspection: it is currently under study. Inspection of pressure vessel welds will follow standard practices. Inspection of steam generators will be performed by conventional eddy-current techniques adapted to tube geometry. Other in-service inspections required by ASME XI, SS-50-SG-O2, and local regulatory authorities, are being evaluated, most of them being similar to the standard for non-integrated reactors.

Fuel and waste handling: manual refueling of the reactor implies changing 31 fuel elements (approx. 70 kg each) per year. Spent-fuel-pool capacity covers seven years of operation; afterwards, dry spent fuel element storage is considered.

Decommissioning: The CAREM concept does not impose specific conditions on plant decommissioning.

1. INTRODUCTION

CAREM reactor features have been described elsewhere /1/. This paper deals with detailed engineering aspects of its design, that point to important differences between the CAREM, and conventional non-integrated PWRs.

The main aspects to be discussed are:
- Operational aspects, including reactor control and control devices.
- Major components engineering: reactor vessel, steam generators and containment vessel
- Maintainability, as related to lay-out, in-service inspection, fuel and waste handling, and decommissioning.

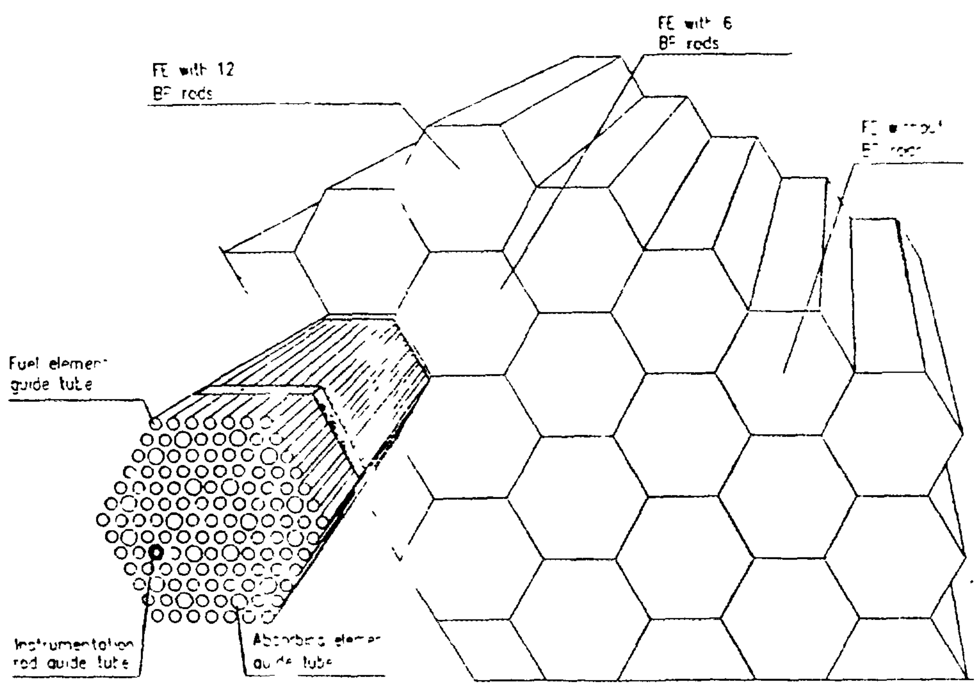
The R&D status corresponding to each item is mentioned when convenient.

2. OPERATIONAL ASPECTS

2.1 Concept of reactor control

Two characteristics of CAREM reactor core design will be discussed:
- strong negative temperature coefficient
- no soluble boron for burnup compensation

The stron negative temperature coefficient enhances the self-controlling features of the PWR: the reactor is practically self-controlled and need for control rod movement is minimal. In order to keep a strong negative temperature coefficient during the



FE - Fuel element
BP - Burnable poison

/1/ Chapter 6 of IAEA's TECDOC on "Status of Small and Medium Reactors" In print.

whole operational cycle, it is necessary to do without soluble boron for burnup compensation. Burnup reactivity compensation is obtained with burnable poisons, i.e. gadolinium oxide dispersed in the uranium oxide fuel. Nonetheless, soluble boron is used to compensate cold-hot reactivity difference (the strong negative temperature coefficient means that a large difference in reactivity must be compensated between cold and hot states). Soluble boron is also used as a back-up of the safety shutdown rods: if the protection system of the plant orders to shut reactor down, and safety rods fail to do so, the reactor is shutdown by boron injection.

The effect the two core design features have on burnup, tend to cancel each other. A strong negative temperature coefficient means that fuel arrangement is not optimum from the reactivity point of view, but this charecteristic is compensated by the fact that power density in the core is lower than normal for PWRs. As a consequence, fuel temperature is lower, and the reactivity at hot state is higher.

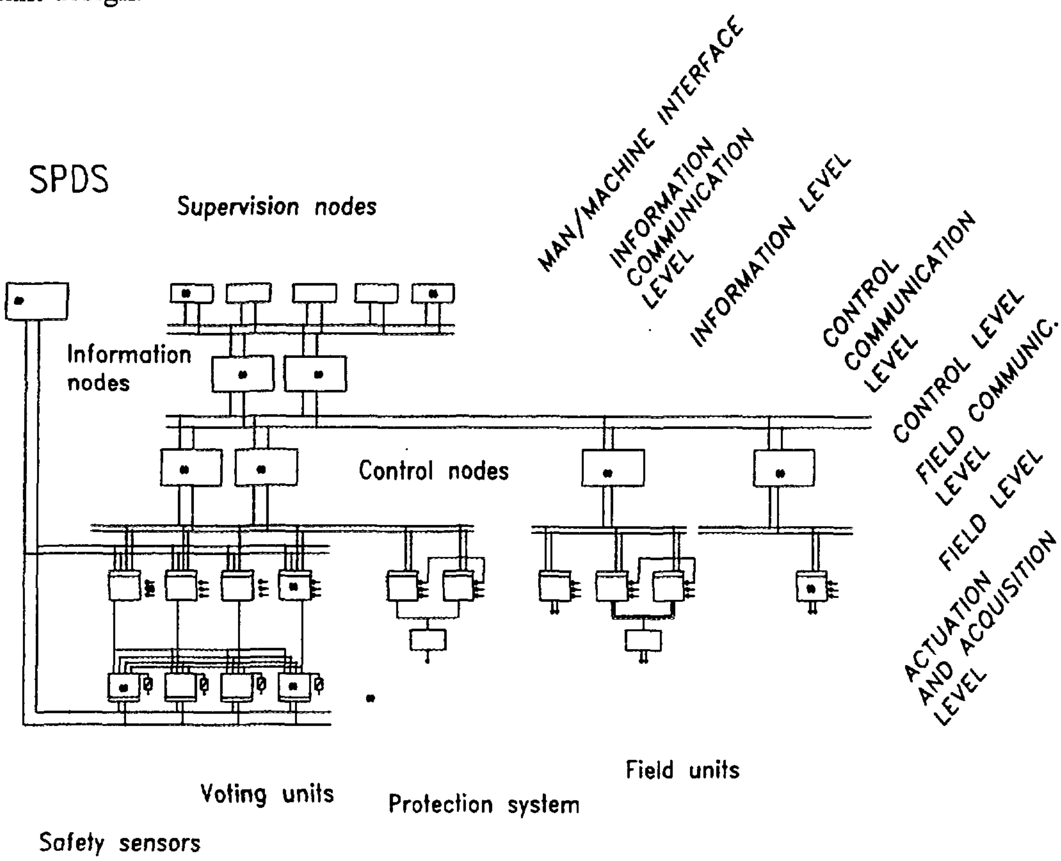
2. 2 Control systems and devices

2.2.1 Control and protection system

The plant is controlled by a microprocessor based distributed system with a redundant communication network. The critical functions of the system required for the plant to operate, are also duplicated.

The protection system, also based in microprocessors, uses a minimum of programmable actions. The software is written directly in machine language (assembler), with a strong effort made to keep it extremely simple. Apart from these small pieces of software, the protection system is "hard".

These features are similar to most recent designs of non-integrated PWRs: they are not a particular characteristic of integrated reactors but a new trend in nuclear power plant design.

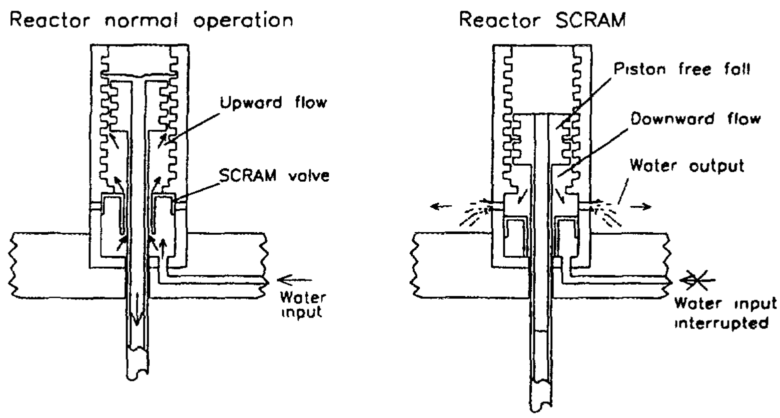


2.2.2 Control devices

The CAREM reactor uses hydraulic control rod drives. This type of rods are wholly contained in the pressure vessel, carrying the concept of integration one step further.

A sketch of its functioning is included. Rods are kept in position by water flow. A positive flow pulse causes the rods to climb one step, while a negative flow pulse causes the rods to go down one step. If water flow is suddenly interrupted, the rods, in a fail safe action, are dropped into the core. The water, after passing through the drives, is dumped into the primary system, inside the pressure vessel. The main advantage of this type of drive is that it can fit in the volume available inside the pressure vessel of an integrated reactor. The use of conventional electromagnetic control rod drives would require a higher containment.

The protection system is able to trigger a scram using a built in valve, that causes the rods to fall by gravity.



However, the drives are designed in a fail safe way:

rods will fall automatically in the event of a plant black-out or in a loss of coolant accident, due to the interruption of the water flow through the drives, without requiring a trip from the protection system.

2.3 R & D activities related to operational aspects

There are on-going research and development activities in every one of the aspects mentioned above:

- A critical facility is being finished, and neutronic tests on the core are scheduled to begin as soon as fuel for the facility is available. These tests are needed to validate the codes used for neutronic calculations.
- A mock-up of the protection system is being designed and will soon be under test.
- A prototype of control rod drive was tested last year. Results of these tests were used to design and built a full scale prototype, which is now being tested at atmospheric pressure. Follow-up tests at full pressure and temperature are scheduled to begin this year.

3. MAJOR COMPONENTS

3.1 Pressure vessel

The pressure vessel, without head, steam generators and internal components, weights 100 tons.

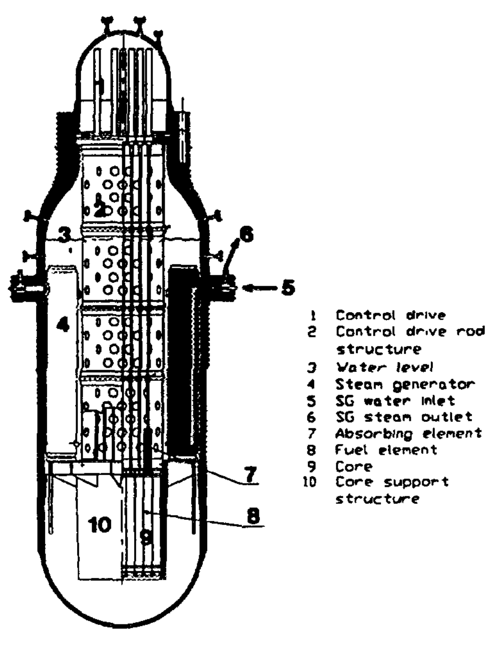
The internal arrangement of the vessel is dominated by the in-vessel disposition of steam generators. Hydraulic control rod drives are also included inside the vessel. For refueling, control rod drives must be dismantled. Steam generators can be replaced during a longer than normal maintenance outage.

The reactor is self pressurized, so no prressurizer is needed. A steam volume, in the upper part of the vessel, accommodates
volume changes during operation.

The following "in-vessel" variables must be monitored during operation:
- Water level
- Pressure
- Core inlet and outlet temperatures
- Neutron flux
- Control rod position
- Coolant flow

In a natural convection reactor, coolant flow measurement is not straightforward. Different methods are being evaluated; correlation of flow with pressure differences across steam generators is the most promising one.

The first four variables can be measured without difficulties specific to this type of reactor. Control rod drive position is more difficult to measure as a consequence of drive design. Ultrasonic methods are being studied for solving this problem.

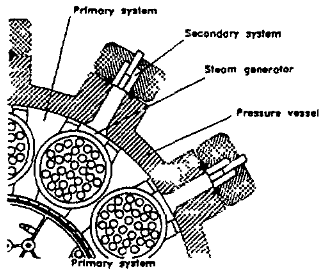


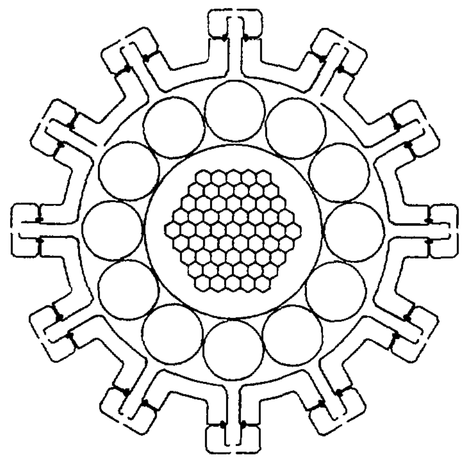
3.2 Steam generators

CAREM uses once through, straight tube steam generators. Twelve steam generators are disposed in an annular way inside the pressure vessel, above the core. The primary side flows through the inside of the tubes, the secondary side flows through the outside. A shell and two tube plates form the barrier between primary and secondary circuits.

This type of steam generators has some advantages:

- Flow of primary and secondary circuits, is countercurrent, which produces a highly efficient heat exchange.

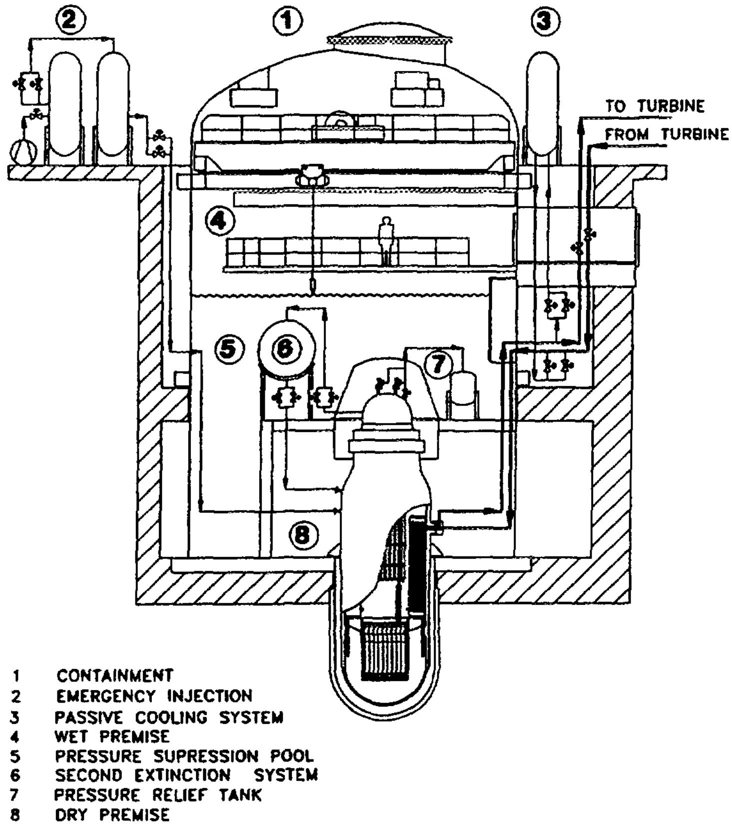




- Pressure drop in both primary and secondary circuits is low. Pressure drop in the primary is of fundamental importance in a natural circulation cooled reactor.
- The operation of replacing a steam generator is not a difficult
- Steam is superheated, eliminating the need for moisture separators and steam dryers.
- There are no specific problems related to the manufacturing of this type of steam generators.

However, this type of arrangement poses some problems that must be addressed by design:
- There is a feedwater inlet and a steam outlet penetration for each steam generator in the pressure vessel: In CAREM, these penetrations are concentric; in this way twelve penetrations are avoided.
- Different thermal expansion coefficients between shell and tubes: prestressed tubes are used, to compensate this difference.
- The quality of the water in the secondary circuit must be extremely good: the full condensate flow passes through a polisher.
- The in-service inspection of tubes and shell can be quite difficult: This aspect is discussed below.

3.3 Containment vessel



The containment is of the pressure suppression type. This type of containment is appropriate for integrated reactors, because reactors of this type don't need large containment volumes for accommodating primary circuit equipment. The height and volume of the containment are determined by the requirements posed by refueling operations and not by the need to accommodate operating equipment.

Containment wall acts as heat exchanger for removing decay heat to the ultimate heat sink, which is the atmosphere: after a LOCA, pressure in the containment rises up, till it reaches design pressure, more than three days after the accident. At this pressure, heat transfer through the containment wall to the atmosphere is enough for preventing further pressure increases.

No specific manufacturing, transport or erection problems related to this component have been identified.

It is not necessary to access the containment during reactor operation: maintenance tasks can be carried out during refueling, as all equipment in the containment is static (no movable parts, apart of a few safety system valves)

3.4 R & D activities related to major components

Research and development activities on major component design are concentrated in the steam generators. The heat transfer features of the tubes have been tested at CAREM operational conditions in a thermohydraulic loop. A full scale model of one steam generator is scheduled to be tested next year.

4. MAINTAINABILITY

4.1 Lay-out of the reactor and plant

There are no special features related to plant lay-out caused by the integrated design of the reactors. However, some lay-out related features are worth of mentioning:

- As the protection system works with a two out of four logic, four independent instrumentation rooms are placed near the containment. Independently routed cable trays connect these rooms with the main control room and the secondary control room.
- The containment is not accessible during operation, so equipment placed inside the containment must be highly reliable, capable of operating continuously for one year without maintenance.

4.2 In service inspection of major components

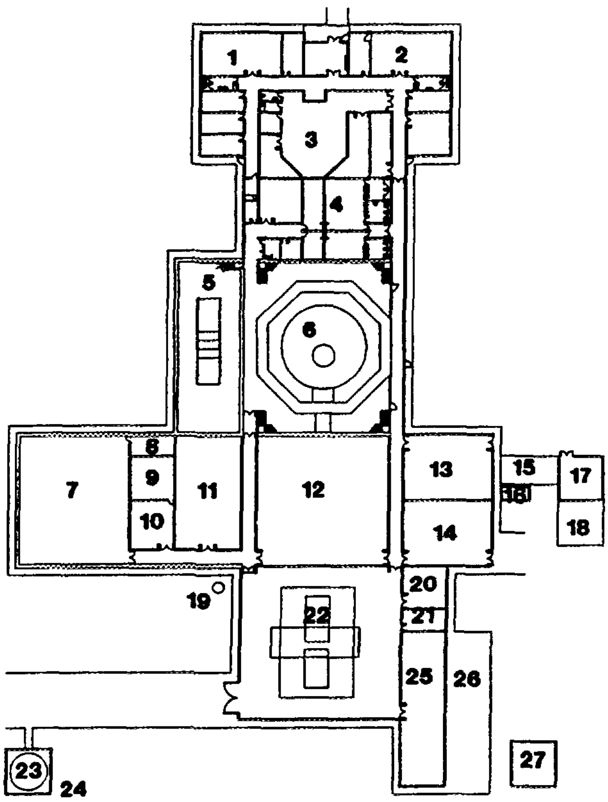
For an integrated reactor, in service inspection is somewhat more complicated than for non-integrated ones. This problem arises from the fact that integrated reactors have major components included in the pressure vessel; hence, it is difficult to access to these components for inspection. The components included in the pressure vessel also make more complicated the inspection of the pressure vessel itself, because they can render inaccessible some areas of the pressure vessel.

The standards that were studied regarding in service inspection are ASME Code, Section X1 and IAEA Safety Series 50-SG-O2. Most of the requirements of these standards can be fulfilled, taking in account that most in service inspection activities can be carried out during annual reactor refueling.

The problems identified to this date are:

- Inspection of welds of the pressure vessel at core level: The plate that separates the hot leg from the cold leg prevents accessibility to these welds from the inside of the vessel. They will be inspected from the outside of the vessel.
- Inspection of longitudinal welds of the shell of steam generators: This weld can be avoided using a complete forge for the shell. If engineering and/or manufacturing reasons call for the use of a rolled plate, there will be only one longitudinal weld, and 0steam generators will be designed so as to have this weld accessible for inspection from the inner part of the pressure vessel.
- Inspection of circunferential welds between steam generator shells and plate tubes: Parts of these welds are inaccessible. A sampling inspection is permitted for this type of welds.

- Inspection of steam generator tubes: The top of straight tubes is readily accessible when the pressure vessel is open, thus no problem with the inspection of tubes is foreseen. To plug a tube, it is necessary to plug both ends of the tube, the lower end being accessible only through the tube. There are technics developed for plugging tubes in this geometry: it is easier than the sleeving techniques that are currently being used. A complete tube rupture could turn repair impossible: in this highly improbable event, the full steam generator can be replaced, not a very complicated operation in a CAREM.

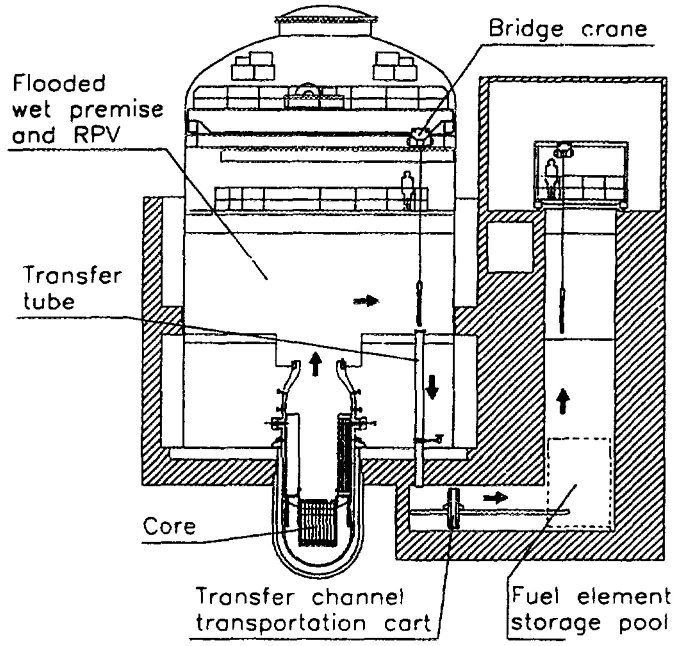


1 Offices/administration
2 Secondary Control Room
3 Main Control Room
4 Cold/hot Dressing Room
5 Spent Fuel Pool
6 Containment
7 Waste Treatment Plant
8 Hot Laboratory
9 Decontamination
10 Hot Workshop
11 Ventilation Extraction
12 Reactor Auxiliary Systems
13 Demineralized Water System
14 Ventilation Injection
15 Fuel And Chemical Store Area
16 Neutralization Pool
17 Auxiliary boiler
18 Chillers
19 Chimney
20 Mechanical Workshop
21 Electrical Workshop
22 Turbine Hall
23 Fire And Plant Water Tank
24 Water Treatment And Pump Hall
25 Batteries/Electric Board Room
26 Transformers
27 Diesel Generators

4.3 Fuel and waste handling

The reactor is refueled annually. Refueling is manual. Half of the core is replaced in every refueling. Spent fuel is placed in a pool. The pool has capacity for accommodating the spent fuel produced by seven years of reactor operation. After seven years, the older spent fuel elements will be removed to a dry storage.

Waste handling facilities are similar to the ones for conventional PWR. Solid waste management must be tailored to the site and country characteristics.



4.4 Decommissioning

No specific features related to decommissioning of integrated reactors has been identified. Actually, dismantling of the plant will probably be easier than the corresponding to a conventional PWR: dismantling of the primary system is achieved by just removing the pressure vessel.

5. CONCLUSIONS

The paper gives and overview of the status of CAREM design. The description makes emphasis in those topics that can imply important differences with respect to traditional PWR technologies.

Although some areas in which R&D activities are neccssary have been identified, the problems specific to integrated type reactors are well within the present technological capabilities of the industry.

PROBLEMS IN MANUFACTURING AND TRANSPORT OF PRESSURE VESSELS OF INTEGRAL REACTORS

XA9745989

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Abstract

Integral water-cooled reactors are typical with eliminating large-diameter primary pipes and placing primary components, i.e. steam generators and pressurizers in reactor vessels. This arrangement leads to reactor pressure vessels of large dimensions: diameters, heights and thick walls and subsequently to great weights. Thus, even medium power units have pressure vessels which are on the very limit of present manufacturing capabilities. Principal manufacturing and inspection operations as well as pertinent equipment are concerned: welding, cladding, heat treatment, machining, shop-handling, non-destructive testing, hydraulic pressure tests etc.

The transport of such a large and heavy component makes a problem which effects its design as well as the selection of the plant site. Railway, road and ship are possible ways of transport , each of them having its advantages and limitations.

Specific features and limits of the manufacture and transport of large pressure vessels are discussed in the paper.

INTRODUCTION

Though the Czech Republic at present time does not intend to develop and construct integral reactors, a contribution on problems of manufacturing and transportation of large pressure vessels which is based on the experience from the manufacture of VVER reactors can be inspiring. In addition to that, the situation in the Czech Republic can be representative to other countries which could potentially utilize integral reactors. The situation is typical with following features:

- inland position,
- complicated terrain,
- geological situation limiting the choice of construction sites,
- lack of water transport ways,
- complicated network of railways and roads having limited transporting profiles as well as limited bearing capacities.

In addition to that, heavy machinery factories which are qualified to produce nuclear components are located out of the direct reach of water-transport.

Integral water-cooled reactors are typical by the elimination of large -diameter primary components i.e. steam generators and pressurizers in reactor vessels. This arrangement leads to reactor pressure vessels of large dimensions: dimeters, heights and thick walls and subsequently to great weights. Thus, even medium-size power reactors have pressure vessels which are on the limit of present manufacturing capabilities.

REVIEW OF LARGE REACTOR PRESSURE VESSELS

There are 424 reactors in operation in the world, from which are 239 PWRs, 89 BWRs, 2 PHWRs (with a pressure vessel). Under construction there are 48 PWRs and 6 BWRs, for which pressure vessels have been mostly manufactured. Totally, 384 RPVs have been manufactured.

Tab 1: Large reactor pressure vessels

Unit	Inner diameter [m]	Height [m]	Wall thickness [m]	Pressure [MPa]	Weight [t]
PWR 1)					
Atucha 2 (PHWR)	7.37	11.5	0.28	11.5	975
Emsland (Konvoi)	5.00	9.75	0.25.	15.7	385
Chooz B	4.5	10.9	0.225	15.7	385
Sizewell B	4.39	10.8	0.22	15.7	335
ABB 80 +	4.62	12.5	0.229	15.7	410
VVER - 1000	4.136	10.9	0.192	15.7	320
BWR 1)					
Kruemmel	6.7	19.1	0.163	7.06	590
Fukushima	6.42	19.6	0.157	7.17	550
Oskarshamn	6.4	17.8	0.16	7	490
Nine Mile Pt. 2	6.38	18.7	0.165	7.17	540
Kashiwazaki 6	7.1	17.3	0.174	7.31	642
Integral reactors 2)					
SPWR	6.6	25	0.285	13	1256
SIR	5.8	19.9	0.28	15.5	502
VPBER - 600	5.97	20.15	0.265	15.7	850
ISER	6	23.2	0.3	15.5	1150

Notice:
1) Dimensions were taken from available literature. Heights and weights of vessels were determined approximately after subtracting the ones of covers.
2) Dimensions and weights have been determined from [2], [3],[4] by approximation after subtracting covers.

From the comparison of pressure vessels of PWRs, BWRs and those of integral reactors it follows, that diameters and lengths of last ones are comparable with those of BWRs. Due to the higher operating pressure which is equal to operating pressure of PWRs, walls of integral reactor vessels are more thick (in the range of 250 - 300 mm) and their weights exceed the weight of the to date heaviest RPV of ATUCHA 2 (975 t).

MANUFACTURING ASPECTS

In the manufacture of RPVs of integral reactors, existing manufacturing methods and materials can be used. The design of integral reactors which is noted for a large water layer between the cover and the vessel wall, substantially decreases the radiation damage of the vessel material during the operation.

Materials

Materials will be used which have proven to be convenient for PWRs and properties of which are well-known: weldability, uniformity of chemical composition and mechanical and technological properties, even for very thick-walled components and extremely heavy forgings.

Following materials should be considered:

Ni-Mo-Cr type: SA 508 Cl.2, 22NiMoCr37, SQV2A,
Mn-Mo-Cr type: SA 508 Cl.3, 16MND5, SA 533 Gr.B,
Cr-Mo(Ni)type: 15Ch2MFA,15Ch2MNFA,
and corresponding Japan steels, eventually improved modifications.

The technology of casting extremely large ingots (up to 600t) from these steels have been mastered by leading firms assuring the required homogenity of chemical composition and high purity.

Semiproducts

Principal semiproducts for pressure vessels are forged rings and plates. Plates are subsequently hot-stamped to the form of *spherical or elliptical dishes for bottoms and covers.*

Rings for integral RPVs will be about 4m long, 6-7m of outer diameter and 250 - 300 mm thick, flange-rings will be shorter with the wall thicknes about 500 mm. There is not such need of complicated thick-walled nozzle rings as at PWR pressure vessel. Plates for dishes will have to have diametres about 10m. It is not excluded to make the part by welding.

Existing procedures of heat- treatment of forgings for quality (quenching, tempering) give high certainty in reaching homogenous structure and mechanical properties even in very thick walls.

Welding

Vessels are welded from individual rings and dishes by circumferential welds. Automated submerged-arc welding to a narrow gap is generally applied. A promissing method could be electron beam technique which is applicable to large material thicknesses and allows the performance of high-quality welds in short time spans.

The simultaneous heating by gas flame or by electricity is applied during the welding.

The number of circumferential welds in an integral RPV will be 4-8.

The necessary welding equipment consists of a welding automat, rotating positioners of an appropriate loading capacity and of a heating device (electric heaters or gas torch). Monitoring and registration of welding parametres is required.

Cladding

Cladding is performed by automated submerged-arc method by using strip or wire (multiple wire) electrodes; plasma technique is a pespective method.

Following requirements have to be fulfilled when cladding:

- the cladded surface must be clean , without impurities and sulphur inclusions which is assured by piercing ingots by the mandrel which removes the central part of the ingot containing segregates , or by using hollow ingots.

- proper laying of individual beads and their mutual overlapping.

- an optimized heat input which is important for the depth and properties of the heat-affected zone under the cladding

All the measures are aimed at eliminating the formation of underclad cracks.

The cladding equipment consists of a cladding automat, positioners (rotating ones for cylindric surfaces, tilting and rotating ones for spherical eliptical and face surfaces) and a heating device.

<u>Stress-relief annealing</u>

Each veld and cladding have to undergo the stress relief annealing by a prescribed procedure: heating-up - dwell - cooling-down, so as to remove residual stresses in the material which attain up to the yield stress of the material. Each section welded from two or more parts must be annealed: it must be kept in view that each annealing impairs material properties.

For the annealing, a furnace must be at disposal which enables placing in even a whole pressure vessel Dimensions of integral RPVs require very deep furnaces (more than 20m). (E.g. the annealing furnace at ŠKODA Reactor Shop has dimensions 7.5x7.5x15m which makes possible to anneal parts up to 13m long: pressure vessels VVER 440 and VVER 1000 are 11.8 and 10.8 long, respectively) In the last resort, a local annealing can be applied.

<u>Machining</u>

Individual semiproducts and sections are machined on large numeric- and programme controlled machine tools: vertical lathes and horizontal milling and boring machines. The machines have to enable rigid clamping of heavy workpieces and precise control of the machining

Main machining operations are as follows·

- turning of cylindrical surfaces of rings and welding edges,
- turning of spherical and/or eliptical surfaces of bottoms and covers including welding edges.
- boring holes and cutting threads in vessel flanges for flange bolts.
- boring holes in vessel covers for control-rod-drive nozzles, instrumentation nozzles and flange bolts.
- machining of main nozzles and emergency-cooling nozzles including welding edges.
- machine or manual grinding of outer and inner surfaces of vessels and covers, etc

Inspections and tests

A quality assurance programme must be followed during all phases of the manufacturing process a part of which is a programme of inspections and tests.

Main groups of inspections and tests:

- material properties (chemical composition, structure, mechanical properties, transition temperatuure curve etc.)

- geometry (dimensions, shape, surface quality)

- integrity (surface integrity, volumetric integrity, leak-tightness, strength).

Current techniques, apparatusses and devices which have been used at the manufacture of classic PVR and BVR pressure vessels are applicable for integral RPVs.

E.g.,· radiography of thick velds is performed by linear accelerators in special shielded cells. A heavy piece is transported to such a cell by a special truck vith rotating positioners which have to have an appropriate loading capacity. At Škoda, the linear accelerator NEPTUN 10 MeV is used which is capable to test thicknesses up to 450 mm. A trolley rail-vagon vith rotating positioners of 400 t loading capacity attends the testing cell.

Pressure test

The hydraulic pressure test is an important testing operation which is required by all relevant codes and standards. The test gives the final confirmation of the integrity and strength of the pressure vessel. The arrangement of of a large integral RPV about 20 m heigh vill be similar to one of a BVR RPV having similar outer dimensions. The test pressure the value of vhich is different in individual national codes is in the range of 1.1 - 1.6 of the operational pressure, vill be comparable vith one of PVR RPVs, i.e. 17.3 - 25 MPa.

Check assembly

Good experience has been made at Škoda vith application of the final check assembly of reactor internals, which vas recommended by the original Russian technical project. This operation consisting in assembly of all reactor internals at the manufacturing shop made possible to find an optimal position of individual parts and remove all problems in the shop. As a result, assemblies of reactors and installation if internals into pressure vessels on sites vere smooth and fast.

There are more components in integral reactors: in addition to classic internals there are steam-generating elements, vater-stream guiding structures etc. A specific feature is a long distance betveen control rod drives and the core vhere control rods are placed. For such a complex lay-out,the check assembly in the shop could be a good confirmation of mutual fitting of individual components.

Shop transport

Large and heavy parts of integral RPVs require appropriate shop transporting and handling means.

Principal handling an transporting operations include:

- transport of semiproduts to the shop,
- transport of workpieces including the whole RPV between individual workplaces and machines,
- tilting of workpieces from the horizontal position to vertical one and back; finally, the whole pressure vessel have to be erected to the vertical position for the pressure test and back,
- rotating workpieces during fabrication (welding, cladding) and inpections (ultrasonic tests, radiography),
- loading workpieces to the annealing furnace, radiographic cells etc.,
- installation of components during the check assembly,
- final expedition of the RPV from the shop (loading on a truck or ship).

The necessary transporting and handling means are rotating and/or tilting positioners, trucks, vagons and cranes of corresponding loading capacity. Cranes have to have an appropriate lifting travel and - for a precise setting - a microdrive.

TRANSPORT OF LARGE PRESSURE VESSELS

The transport from the manufacturing plant to the construction site is most probably the main limiting aspect of pressure vessels of integral reactors. Though there were several pressure vessels the manufacture of which was completed on the site, the present interest both of plant owners and manufacturers is to have shop-fabricated pressure vessels.

There are three possible ways of transporting such a large and heavy component: railway, road, ship or a combination of them.

Railway transport

Advantages:

- accessibility: both manufacturing plants and construction sites are usually served by railway.

Disadvantages:

- limitations in the diameter (transporting profile) and in the weight,
- a special railway truck is needed

Road transport

Advantages:

- accessibility: both manufacturing plants and construction sites have an access by road.
- variability: there is a variety in choosing a most convenient route of transport.

Disadvantages:

- limitations in diameter (not so strict as at railways: there is more possibilities in the selection of the route).
- a special truck is needed

Water transport

Advantages:

- no limitations by dimensions and weight
- low price

Disadvantages:

- accessibility: navigable water way is needed both at the manufacturing plant and on the site

Combined transport:

For the final part of the route, both railway and water transport are combined with the road one for the transport from the port or railway station to the reactor building. A combination: railway - ship - road - sship was used at the transport of the VVER-1000 vessel from Plzen to Belene in Bulgaria.

An example of the transport of VVER pressure vessels from Skoda

A principal requirement laid on the VVER reactors which were manufactured in the Czech Republic under the Russian licence was the one of transportability of all components by rail. The requirement was based on conditions of countries in which the units were constructed, i.e. the choice of construction sites and their availability and localization of manufacturing factories out of the reach of water-transport. Škoda Nuclear Machinery Plzeň is one of three plants in which VVER reactors were produced (*besides Izhora Plant near St.Petersbourgh and ATOMMASH at* Volgodonsk, both in Russia). 21 VVER-440 reactors and 3 VVER-1000 were produced at Škoda. A special railway truck KRUPP was used for the transport of large components including pressure vessels. In several, cases the railway transport was combined with road and water one. E.g., pressure vessels for NPPs Nord in Germany and Zharnowiec in Poland, were transported by rail to the river port on the Danube near Bratislava and then by the ship to the Black Sea and around the whole Europe to ports and sites in Germany and Poland.

Tab.2: Sites of VVER - type NPPs and ways of RPVs transport

Site, country	Way of the transport
VVER-440	
4x Paks, Hungary	Railway to Bratislava, river-boat to Paks
2x Bohunice, Slovakia	Railway
3x Nord, Germany	Railway to Bratislava, ship to Greifsvald
4x Dukovany	Railway
4x Mochovce	Railway
2x Zharnowiec,Poland	Railway to Bratislava, ship to Gdansk
VVER-1000	
1x Belene,Bulgaria	Railway to Jihlava, road-truck to Bratislava, river-boat to Belene
2x Temelín	Railway

CONCLUSIONS

1.Pressure vessels of integral reactor of medium output have outer dimensions which are comparable with ones of BWRs. Walls of them are more thick and their weights exceed to date heaviest RPV.

2.No substantial differences are expected between existing manufacturing technologies of reactor pressure vessels of PWRs and BWRs and one of future integral reactors. Most significant there will be the size factor. Integral RPVs will be preferrably completely manufactured at shops.

2.Inspection and test techniques and devices for integral RPVs will correspond to ones currently used at PWR and BWR technology. The shop check assembly of reactor internals should be considered as convenient preparation for the on-site assembly.

3.The transport of large pressure vessels is the limiting factor of deployment of integral reactors. Integral units of a medium output should be constructed on sites in the reach of water transport. For the local generation of electricity and heat, small modular units transportable by railway or by road seem to be more prospective.

4.The introduction of integral reactors would require additional investments to productional base, mainly in increasing the loading capacity of manufacturing equipment: welding positioners, supports of machine tools, means of the shop transport etc. The scope of such investments will undoubtedly depend on the number of units required by customers. An appropriate ad-hoc solution would be certainly found for the manufacture of small number of RPVs.

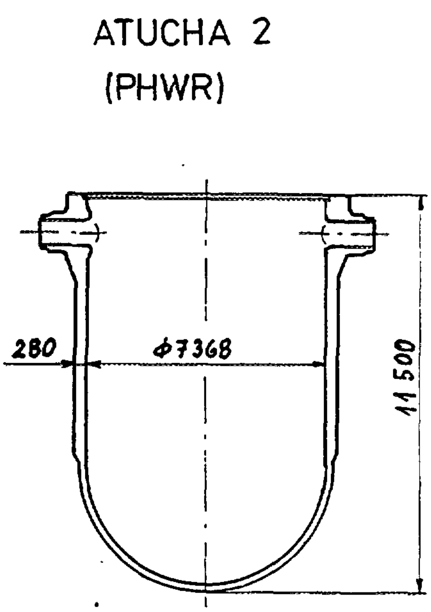

FIG.1: PWR PRESURE VESSELS

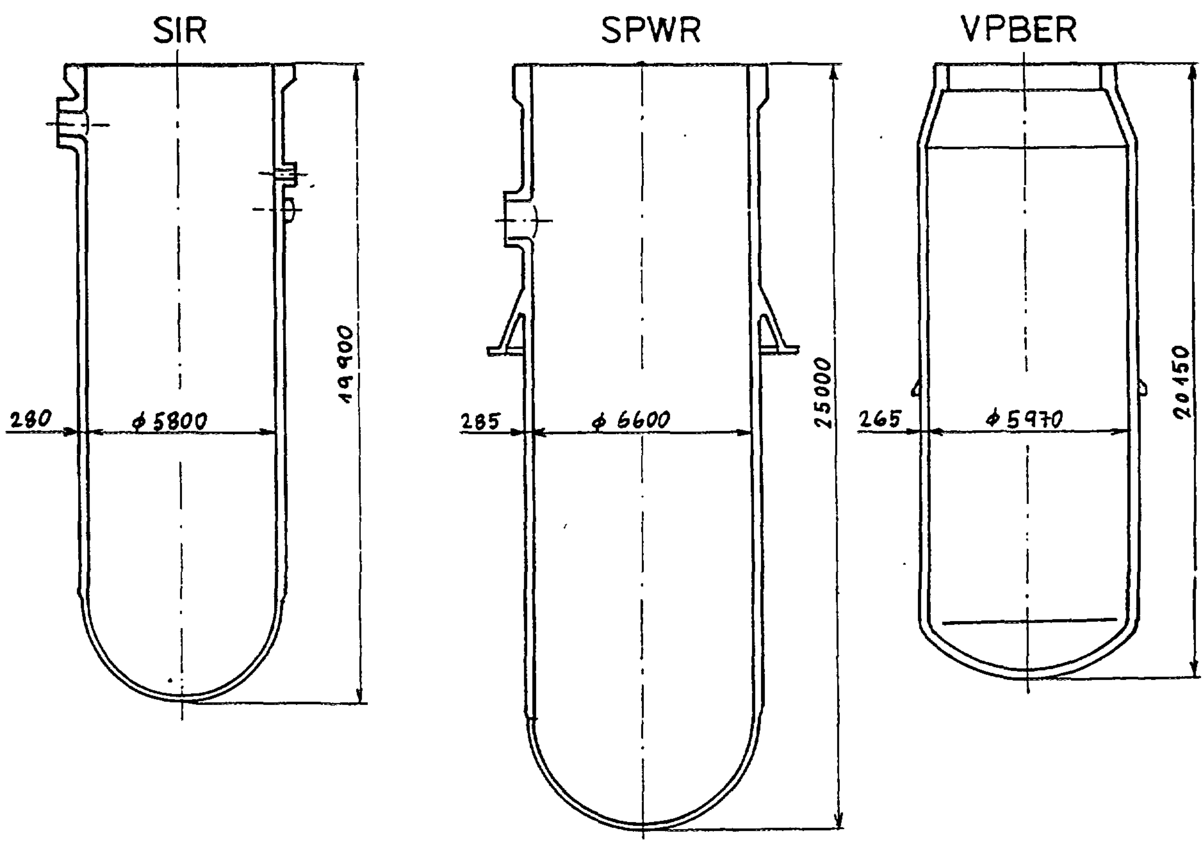

FIG.2: PRESSURE VESSELS OF INTEGRAL REACTORS

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RADIOLOGICAL ASPECTS OF INTEGRAL NUCLEAR REACTOR DECOMMISSIONING

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Abstract

World nuclear power tends constantly to decrease the permitted ultimate dose of NPP personnel and population during normal operation and accidents and NPP decommissioning.

1. ISSUES AND HOW TO DECIDE THEM WHEN DECOMMISSIONING NPP

The conditions and radiological safety of reactor decommissioning project are determined by:

1) decommissioning concept ("immediate" or "delayed" dismounting),
2) structure and equipment radioactivity;
3) radioactive waste amount (activity, mass, volume);
4) technology of removable and non-removable equipment;
5) doses of personnel and population.

It is known, when decommissioning NPP with BWR or PWR, that radioactive waste forms an activity of more than 10 MCi. The total mass of waste at NPP decommissioning is several hundreds thousands of tons, and about 1-2% of the waste has high and medium radioactivity, which should be disposed off. The potential danger of radioactivity requires protective measures, eliminating release of radioactive materials into the environment.

Irradiation doses of the personnel involved in NPP decommissioning range from several hundreds to tens of thousands of man.rem depending on the decommissioning strategy adopted.

One of the main problems of NPP decommissioning is handling of hot large equipment. NPP reactors accumulate substantial radioactivity (up to 10 5 Ci), which causes high radiation levels from 1 to 10 Sv/hr

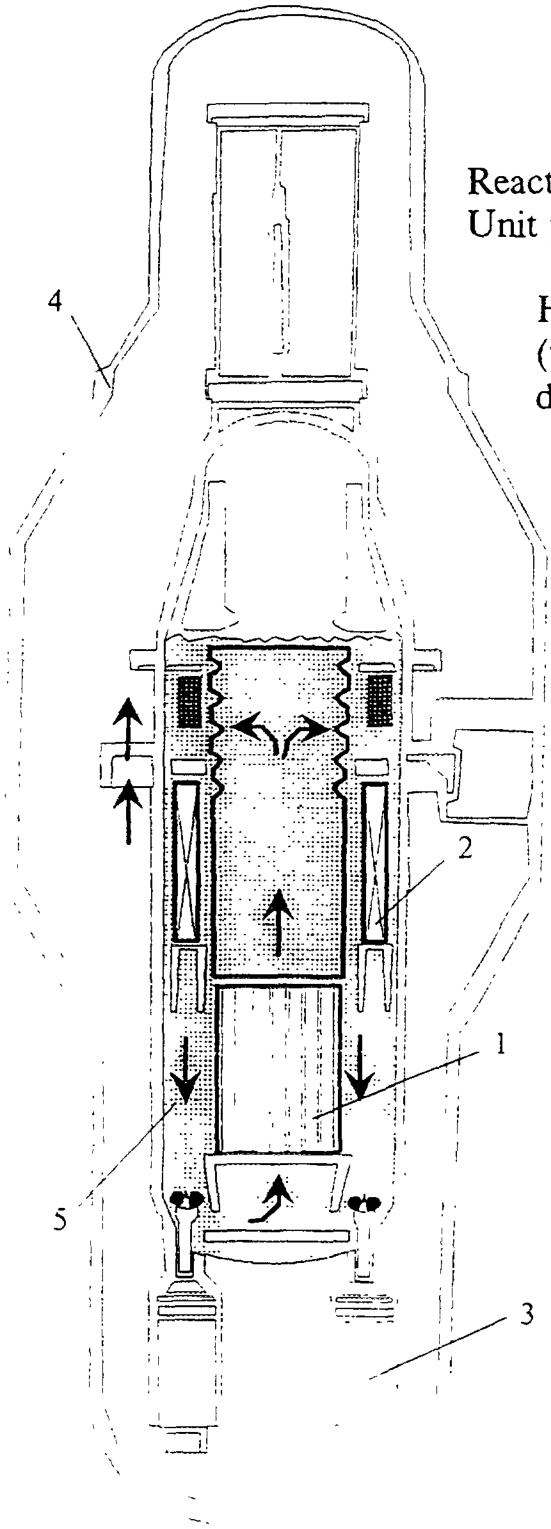
Dismounting and disassembly of vessel and structures requires development of special remotely controlled equipment, heavy protective flasks and, hence, large costs or prolonged plant preservation that reduces irradiation doses, potential releases and environment contamination.

But, from the point of view of economic, quick rehabilitation of the site is not evidently advantageous as compared with immediate dismantling.

2. INTEGRAL REACTOR ADVANTAGES

The reactor structure, eliminating a large number of issues during its disassembly, has obvious advantage This property is peculiar to AST-500 and VPBER-600 integral reactors developed in OKBM, N.Novgorod (Fig-s 1, 2).

VPBER-600 REACTOR



Reactor thermal capacity - 1800 MW
Unit power - up to 630 MW(e)

Heat output
(when electrical power is
decreased down to 430 MW)- up to 645 Gcal/hr

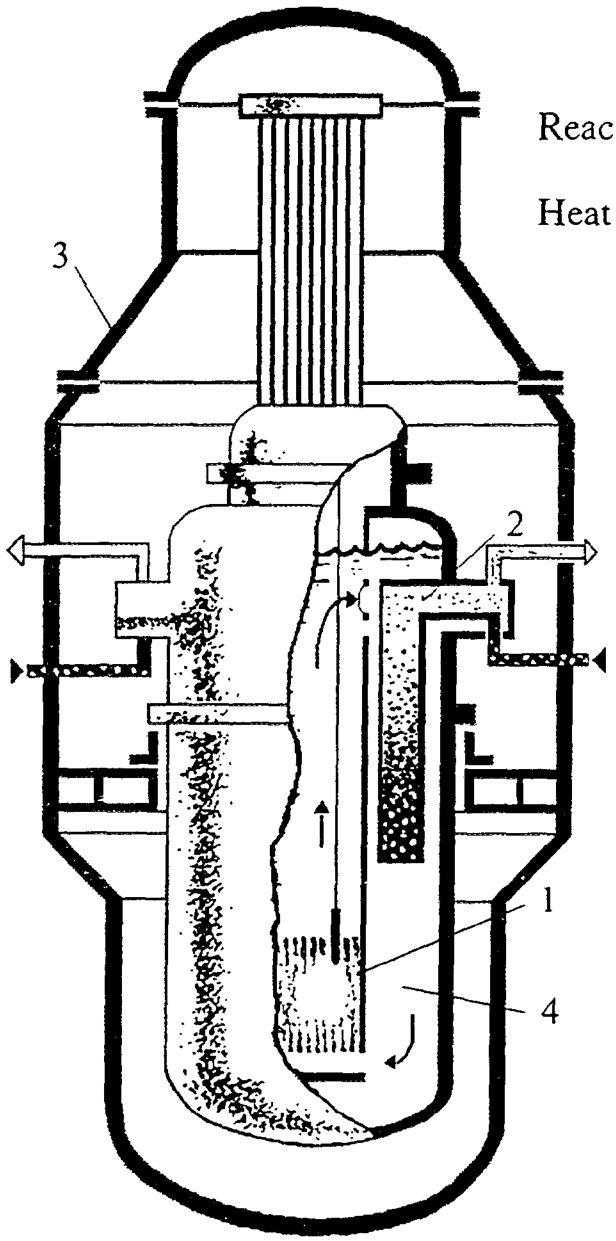
Creation time - 8 years
Lifetime - 60 years

VPBER-600 is a qualitatively new level of PWRs development, further safety enhancement due to consistent implementation and intensification of inherent self-protection properties, wide use of passive safety systems.

1. Reactor core
2. Steam generator
3. Motor pump
4. Guard vessel
5. Water gap between reactor core and vessel

Fig.1

AST-500 REACTOR



Reactor thermal capacity - 500 MW

Heat output - up to 430 Gcal/hr

1. Reactor core
2. Heat exchanger
3. Guard vessel
4. Water gap between reactor core and vessel

Fig. 2

IRRADIATION LEVELS FROM THE EQUIPMENT OF AST, VPBER INTEGRAL REACTORS AND PWR-TYPE REACTOR

Equipment	Dose rate, nSv/s		
	AST-500	VPBER-600	VVER-1000
Reactor vessel:			
inner side	160	330	~3000000
outer side	10	17	~60000
Guard vessel	10	14	-
Reactor concrete silo lining	20	14	~25000
Reactor silo concrete	0,2	0,15	~200

Fig. 3

A characteristic feature of these reactors, the integral layout, which allows the arrangement of the main reactor equipment in one vessel separated from the core (neutron source) by a thick water layer, ensures low neutron fluence to the reactor vessel and low induced activity of the main metal-intensive equipment - reactor vessel, guard vessel and structures in the concrete reactor silo, and, hence, low irradiation levels after shutdown (Fig.3).

Already 1 year after plant decommissioning personnel can work in the reactor silo without protective measures during the whole working day (including dismounting of such metal-intensive equipment as the reactor and guard vessels). In this case the maximum irradiation dose rate in the reactor silo in the area of the core is 0.05 mSv/hr (Fig-s 4, 5).

The maximum induced activity in the main and guard vessel and concrete structures of the VPBER-600 reactor does not exceed 50 Ci and of the AST-500 reactor - 15 Ci.

So, this AST and VPBER reactor design property defines their decommissioning concept - "immediate" disassembly in 1-3 years after shutdown (following termination of preparatory work).

IRRADIATION LEVELS AT VPBER-600 DECOMMISSIONING

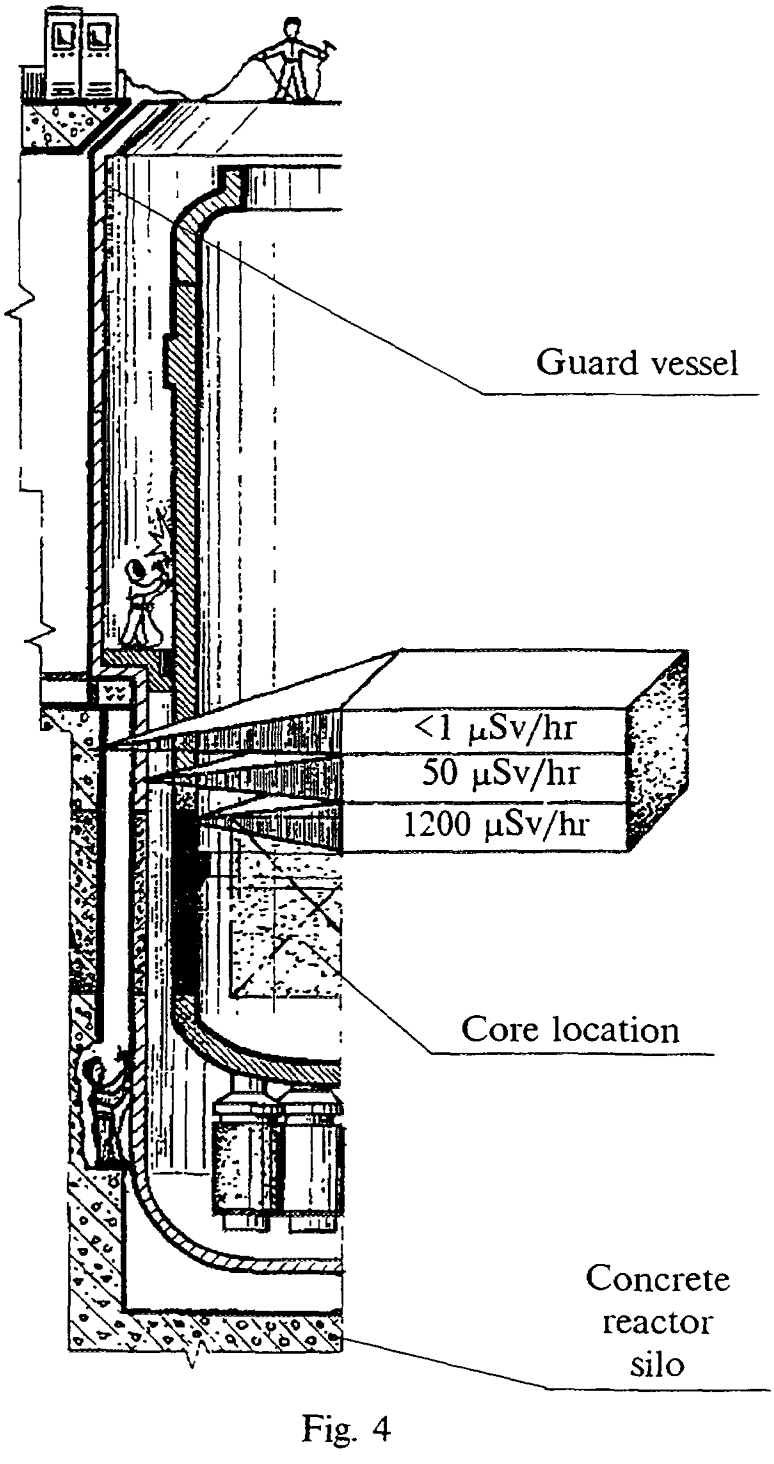


Fig. 4

IRRADIATION LEVELS AT AST-500 DECOMMISSIONING

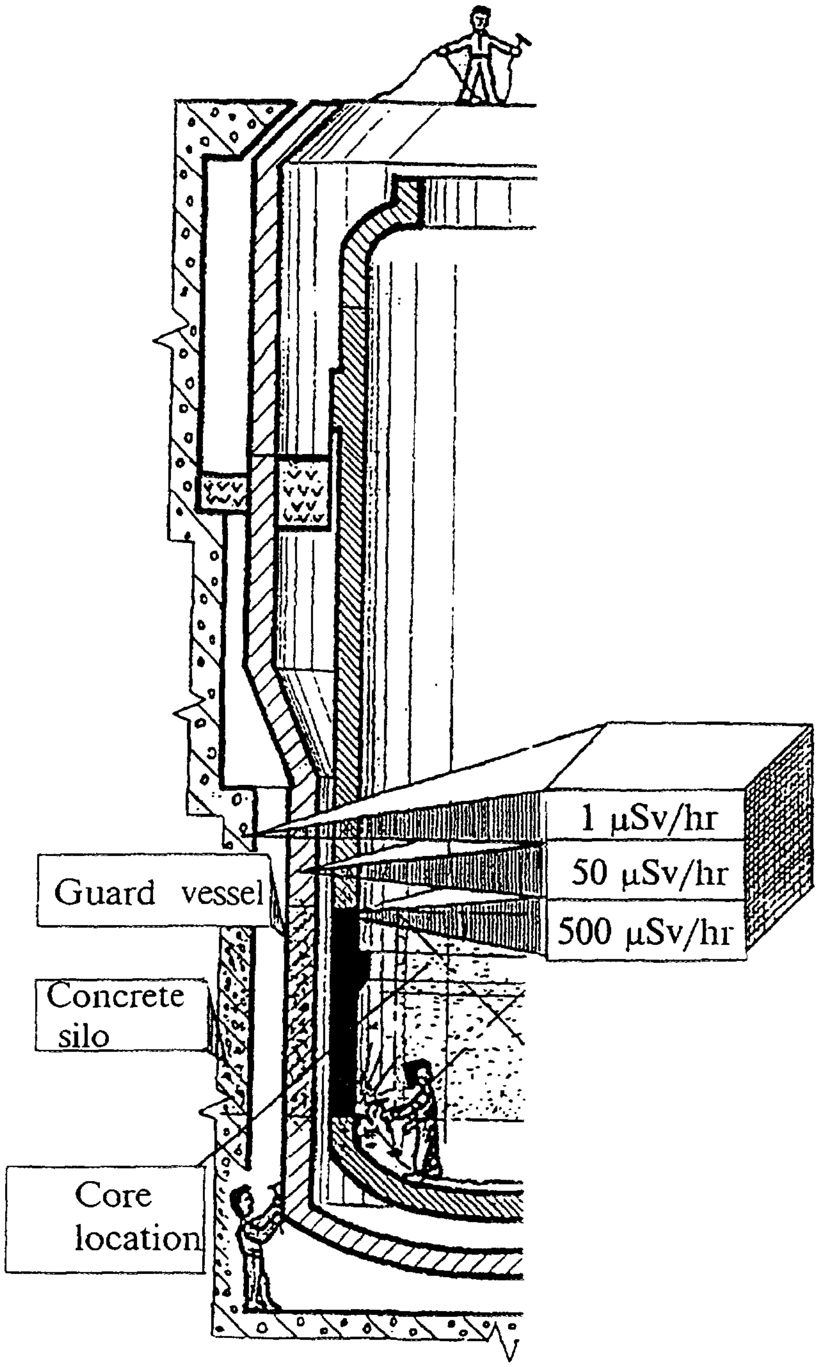


Fig. 5

3. REDUCED AMOUNT OF RADIOACTIVE CORROSION DEPOSITS

The AST and VPBER plants also show a low activity in the primary coolant caused by corrosion products and, inturn, low surface contamination of equipment because of the application of slightly activated materials corrosion-resistant for core and heat exchange equipment.

4. INTEGRAL REACTOR DECOMMISSIONING

Reactor plant decommissioning starts at unloading of spent fuel and internals. Unloading of fuel assemblies and dismounting of radioactive internals do not require special devices and do not limit methods and the time needed for decommissioning work, because equipment from the reactor is dismounted and unloaded by means of standard plant facilities used during operation (Fig 6).

At the same stage (calculated for 1-3 years) slightly activated and clean equipment is dismounted and disassembled, if it is not needed for the following operation stages.

The secondary stage of work (1-2 years duration) is associated with dismounting, disassembly and removal of reactor plant hot equipment from the plant site.

Slightly activated equipment is disassembled by conventional general industrial methods under control of the radiological safety service, and special facilities and mechanisms are not required.

Hot equipment is broken into fragments by means of remotely controlled manipulators in the concrete silos of the plant

Fragments are packed into protective transport fasks and removed to regional radwaste store.

Spent fuel assemblies in special van-containers are transported to fuel reprocessing plants.

The last stage (1-3 years) is decommissioning of the whole plant including building structures.

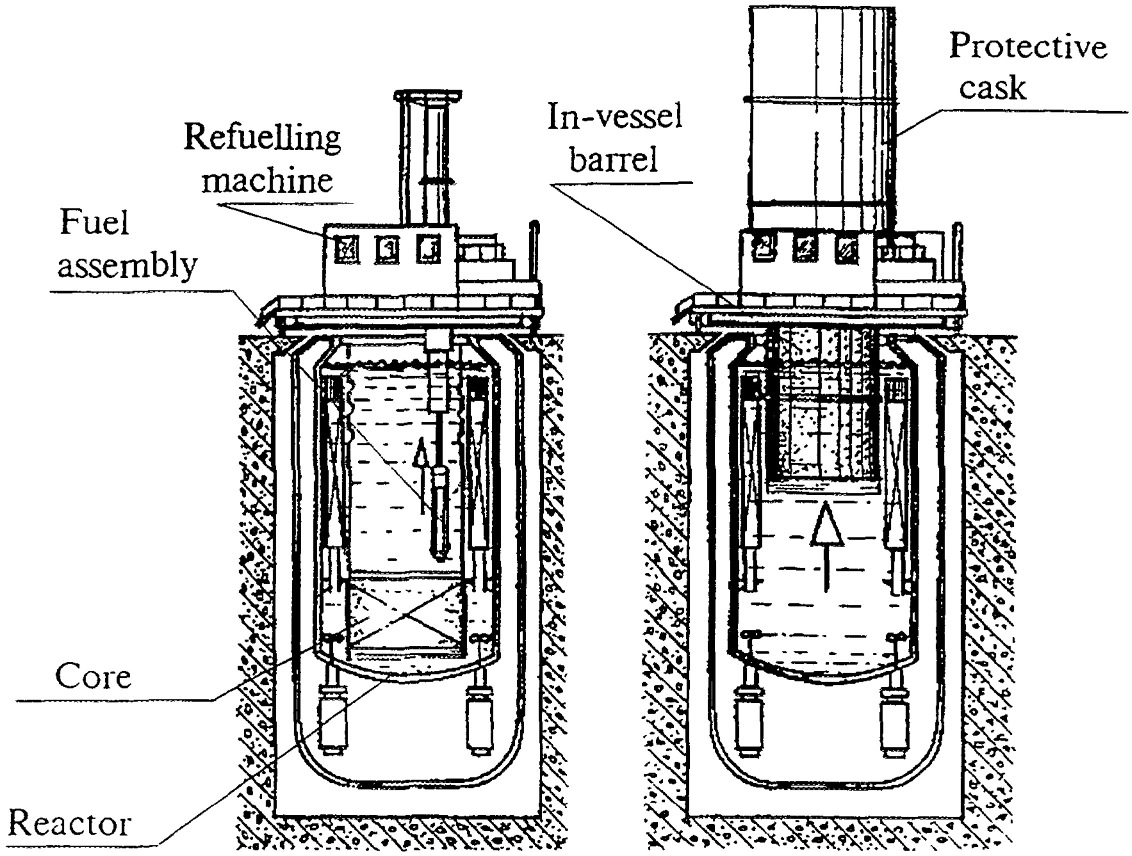
5. EVALUATION OF THE AMOUNT OF RADWASTE AT THE TIME OF INTEGRAL REACTOR DECOMMISSIONING

Radioactive equipment and structures of AST and VPBER plants can be divided in contaminations groups according to their accumulated activity (in correspondence with national radiological safety norms, that allows evaluation of mass, volume and methods of handling (Fig-s 7, 8).

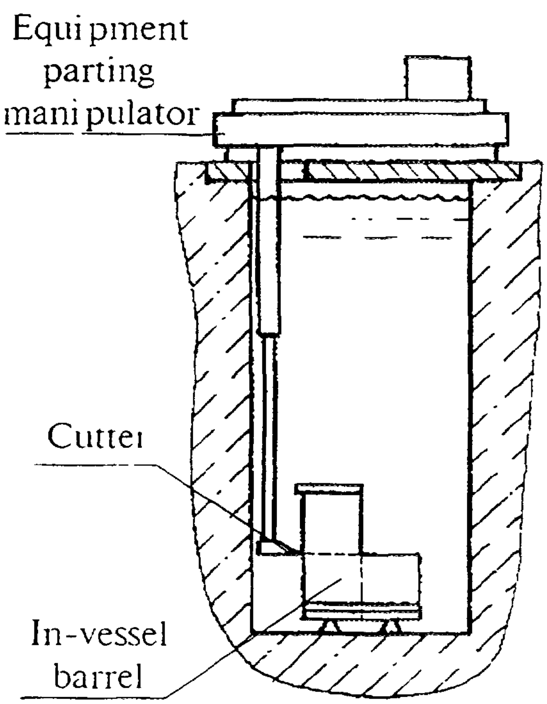
The hot equipment and structure mass (in-vessel barrel, guide tubes-connecting device unit, etc.) is ~60 t for AST and ~90 t for VPBER, the medium equipment mass (reactor vessel, heat exchange equipment, etc.) is ~270 t for AST and ~1850 t for VPBER, the cold and clean equipment mass (guard vessel, concrete reactor silo structures, etc.) is ~1000 t for AST and ~3750 t for VPBER.

Liquid radwaste formed during cleaning of the circuit water, the reactor and equipment decontamination (liquid radwaste activity does not exceed 1000 Ci) should be reprocessed - it is evaporated, concentrated and bituminized (AST-500) or cementated (VPBER-600) to confine the activity.

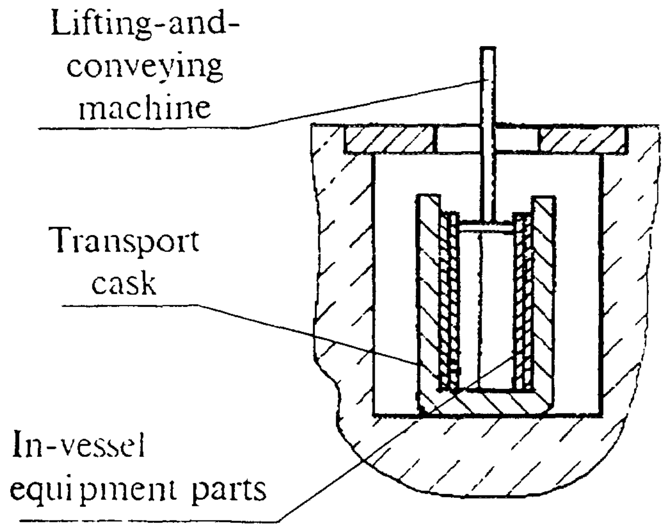
Solid radwaste is also reprocessed it is ground, burnt, pressed and sintered It substantially decreases the total volume of radwaste



Fuel assembly and internals reloading



Parting of in-vessel equipment



Packing of in-vessel equipment parts into transport cask for removal and disposal

Fig. 6

VPBER-600 MAIN EQUIPMENT MASS AND RADIOACTIVITY

Equipment characteristics	Total radioactivity, Ci	Mass, t	Disassembly and transport conditions
Hot, group 3 (in-vessel barrel, guide tube-connecting devices unit, etc.)	$\sim 1*10^7$	100	Removed from reactor by standard means used during operation. Transported in protective casks
Medium, group 2 (reactor vessel, steam generator piping, pumps, etc.	170	1850	The equipment is dismounted and parted by conventional engineering means under the control of radiological safety service. Protective casks are required to transport some equipment
Cold, group 1 and clean (guard vessel, structures in concrete silo of reactor, etc.	<4	3750	Dismounted as at general industrial plants and does not require special protective measures. Transported without protective casks, organizational measures are provided

Fig. 7

AST-500 MAIN EQUIPMENT MASS AND RADIOACTIVITY

Equipment characteristics	Total radioactivity, Ci	Mass, t	Disassembly and transport conditions
Hot, group 3 (in-vessel barrel, guide tube-connecting devices unit, etc.)	~$1*10^6$	60	Removed from reactor by standard means used during operation. Transported in protective casks
Medium, group 2 (reactor vessel, heat exchanger piping, etc.	80	270	The equipment is dismounted and parted by conventional engineering means under the control of radiological safety service. Protective casks are required to transport some equipment
Cold, group 1 and clean (guard vessel, structures in concrete silo of reactor, etc.	1-2	980	Dismounted as at general industrial plants and does not require special protective measures. Transported without protective casks, organizational measures are provided

Fig. 8

VPBER-600 AND VVER-440 (NPP"LOVISA") REACTOR UNIT DECOMMISSIONING CONDITIONS

Parameters	VPBER-600	VVER-440 (Atomnaya energiya, v.67, is.2 Aug., 1989)
Power, MW(e)	630	470
Service time, years	60	30
Hold-up time following reactor shutdown, years	1	2
Reactor vessel radioactivity, Ci	50	7000
Total radioactivity, Ci	$\sim 10^5$ (without internals)	$\sim 1.3*10^6$ (without internals and steel cassette-screens)
Irradiation dose rate in reactor silo, mSv/hr	0.05	130
Radioactive waste amount, t	2200	8460 (for 2 units)
Collective irradiation dose for personnel, man.Sv	2,5	23 (for 2 units)

Fig. 9

6. IRRADIATION DOSE EVALUATION DURING INTEGRAL REACTOR DECOMMISSIONING

Because the activity and mass of radioactive equipment of the integral reactors considered are substantially lower than those ones of conventional NPP reactors, this influences irradiation doses.

According to calculations personnel irradiation dose during VPBER-600 and AST-500 unit decommissioning is 2.5-3.5 man.Sv taking account of plant-level systems).

The radiological effect on the inhabitants during decommissioning was evaluated proceeding from the amount of radioactive product releases which can enter the atmosphere. The result obtained was that during AST-500 equipment and reactor vessel disassembly the reactivity release to the atmosphere including cleaning in wet scrubbers and aerosol filters gives the value ~3 mCi for Co 60, that is ~20% of the specified limit.

7. COMPARISON OF DECOMMISSIONING CONDITIONS FOR INTEGRAL AND VVER-TYPE REACTORS

The advantages of integral reactor decommissioning emerge when comparing decommissioning conditions for VPBER-600 and VVER-440 "Lovisa" (Fig.9).

The concept of "immediate" dismounting in 2 years following termination of preparatory work is suggested both for "Lovisa" and VPBER-600 reactors. The main problem of NPP "Lovisa" decommissioning is the handling of the hot reactor vessel which must be broken up in hot chambers by remotely controlled equipment and be removed in large thick-walled casks to stores or the reactor vessel and equipment must be removed and disposed off as a whole. One should note that special manipulators and other expensive facilities are not required for breaking up the VPBER-600 reactor vessel.

Comparison results show that in spite of higher power a more prolonged operation period and shorter hold-up time following shutdown, safety characteristics (activity value, radwaste amount, irradiation doses) during VPBER-600 reactor decommissioning are tens and hundreds times smaller than those for VVER-440 reactor.

8. CONCLUSION

So, the structural features of AST and VPBER reactors (including the integral layout of their main primary equipment and location of a leaktight reactor in an additional guard vessel) ultimately simplify the technologically difficult, radiologically dangerous and expensive decommissioning of PWR (VVER)-type NPP reduce the amount of radwaste and the costs and make integral reactor decommissioning safe for both personnel and inhabitants.

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