NUCLEAR STEAM SUPPLY SYSTEM OF ULTIMATE SAFETY TR-1000 PB (TR-1000 PB NSSS)

TR9800040

G. Kiselev
ITEP, Reactor Department

ÖZET

TR-1000 PB reaktörünün nükleer buhar sağlama sistemi, 1000 MW(e) gücünde, gaz (CO_2) soğutmalı, ağır sulu bir santraldır ve reaktör ön-gerilmeli beton kap içerisinde bulunmaktadır. TR-1000 PB nükleer buhar sağlama sistemi projesinde olabildiğince kendiliğinden ve pasif güvenlik prensipleri kullanılmıştır ve bu bildiride ön-gerilmeli beton kap içerisindeki özel ergimiş yakıt tutma sisteminin, uzun süreli pasif ısı çekimi ile, ergimiş yakıtı nasıl lokalize edebileceği anlatılmıştır.

ABSTRACT

NSSS of TR-1000 PB is a 1000 MW (e) plant with gas-cooled (CO_2) heavy water reactor the main equipment of which is contained within the vessel of prestressed reinforced concrete. In project of TR-1000 PB NSSS the principle of inherent and passive safety is used as much as possible and the version is given how to localize the molten fuel in the special catcher within the prestressed reinforced concrete vessel with long-term passive heat removal.

INTRODUCTION

Crisis in nuclear power development program created after Chernobly-4 NPP accident cant't be coped with on the basis of "reasonable" safety that allows for probability of severe accident with catastrophic consequences for environment and public. However small is the risk of such consequences the opponents would always be right to follow a simple wordly wisdom: "If something may happen it would happen sooner or later". Thus, we should not only enhance NPP equipment reliability and personnel qualification but also develop the power unit whose design features and inherent properties would exclude the possibility of beyond-design-basis radiation consequences of emergency situations. One of such trends is further development of gas-cooled (CO_2) heavy water reactor.

TR-1000 PB REACTOR DESIGN

Reactor TR-1000 PB (Figs. 1-2) of channel-vessel type has heavy water as the moderator, dioxide carbon as the coolant and low-alloyed metallic uranium with natural content of U-235 as the fuel. Mixed uranium-plutonium composition may be also used as the fuel.

Heat generation in the reactor core is transferred to the secondary side by gaseous coolant through the SGs and by heavy-water coolant (~6 %) through the intermadiate circuits to the condensate heaters.

Reactor represents itself a heavy-water tank of cylindrical form with vertical channels 218 x 4 made of zirconium alloy E-125. The channels house the fuel assemblies (342 pcs) and working control rods (19 pcs.). The channels are installed in the triangle array with a pitch of 410 mm. The core diameter is 8200 mm, height - 5000 mm.

Primary parameters

Coolant	CO_2
Pressure, MPa	10
Temperature, °C	
at the core inlet	430
at the SG outlet	250
Thermal power, MW	3000

Secondary parameters

Thermal power, MW	3200
Steam capacity, t/h	4310
Steam temperature, °C	410
Steam pressure, MPa	7,0
Feedwater temperature, °C	120

Moderator parameters

Moderator	D20
Pressure, MPa	10
Temperature, °C	
at the reactor inlet	90
at the reactor outlet	140
Thermal power, MW	200

Fuel assemblies of TR-1000 PB reactor include the bundle of 126 fuel rods being spaced either by the couplings or the grids. Each fuel rod is fixed to the fuel assembly cap by means of end plug, bundle assembly is placed into the outher sheat of thinwalled zirconium tube with inner diameter 200 mm and wall thickness 1,5 mm. From below the fuel assembly sheat is provided with a tailpiece. Two types of fuel assemblies differing in fuel loading are supposed to be used in the core for radial power distribution equalization. Fuel assemblies of the central area are provided with fuel rods having fuel stack diameter equal to 6,9 mm whereas the outermost ones - 7,5 mm.

There are two alternatives of the fuel rod design:

- alloyed metallic fuel (for example, uranium) within the cladding of magnesium-beryllium pseeudoalloy PMB;

- alloyed uranium-zirconium alloy UTsN within the cladding of zirconium alloy Ts2M.

Fuel rods with cladding of PMB passed the in-service inspection on KS-150 reactor of NPP A-1, CSFR.

The main shortcomings of this alternative are:

- low temperature of cladding melting (~ 650 °C);

- low strenght properties of PMB alloy;

- low corrosion stability of fuel stack.

Fuel rod alternative is being developed to eliminate the said shortcomings, but it requires to be substantiated by the post-irradiation examinations at maximum fuel burnup ~7000 MW. day/t.

Heavy-water circuit that together with a tank incorporates a heat exchanger with the pressurizer and connecting pipelines is located wholly within the vessel being pressurized by the gaseous coolant. Moderator moves by the natural circulation. The primary circuit is provided with forced coolant circulation by means of gas blowers being rotated by the steam turbine drive.

The NSSS main equipment has an integral layout within the vessel of prestressed reinforced concrete. The vessel is a multi-cavity cylinder central part of which is housed with the reactor itself whereas the peripheral parts are occupied by the main equipment of the heavy-water and primary circuits.

To make equipment erection within the vessel convenient and to provide the successive acces to it during operation the cavities are sealed with the detachable floors. Prestressed system tendons are passing through the vessel vertical channels between cavities and through the horizontal slots of vessel inner cylindrical surface. Inner concrete surface in the cavities are covered with tight steel liner. The concrete vessel is provided with steel reinforcement and system of tubes ensuring the vessel cooling by circulation water. Diameter and height of vessel is about 40 m.

The lower part of reinforced concrete vessel under the core is provided with corium catcher. The catcher is a constituent part of vessel used for catching, long-term retention and reliable cooling of corium in the course of any core damage.

The catcher is a monolithic structure consisting of vertical cylindrical graphite crucibles placed into the steel thimbles (420 pcs) to supply and remove the cooling water of headers and connecting pipelines. The crucible inner surface is covered with protective coating of Al_2O_3 and ZrO_2. Inside the crucible there is flux that provides protection of corium accumulated in the catcher against its interaction with coolant (CO_2), moderator (D_2O) and oxygen. Above the catcher surface there is a screen that prevents the catcher from water ingress from the heavy-water tank. From outside the catcher thimble is provided with a coil wherein the cooling water circulates. Catcher cooling pattern provides for heat removal from corium into the cooling pond through the loop intermediate circuit by natural circulation.

Main performances of the catcher

Parameter	Value
1. Cooling water temperature in intermediate circuit at the catcher inlet, °C	40
2. Cooling water temperature in intermediate circuit at the catcher outlet:	
during NSSS normal operation, not more than, °C	50
under emergency conditions (corium cooling in the catcher), not more than, °C	100
3. Cooling water pressure in the intermediate circuit, MPa	0,5-10
4. Volume of the catcher cooling part (corium inventory), m^3	50
5. Number of thimbles in the catcher	421
6. Design capacity of the heat exchangers for the catcher cooling system, MW	60
7. Catcher lifetime (under waiting conditions), years	60

NUCLEAR SAFETY NEUTRONICS BACKROUND

Core neutronics, reactivity control system structure of TR-1000 PB reactor are such that under any operating conditions and during the most severe accidents the reactor uncontrolled runaway is impossible. This is reached because:

a) fuel used has low concentration of fission isotopes: their average concentration in the reactor charge is less than in natural uranium;

b) under all operating conditions the power coefficient of reactivity is negative;

c) temperature and void reactivity effects are small (0,5 V eff);

d) compensation for reactivity being slowly changed by boric acid concentration in the moderator allows the control rod reactivity margin to be less than 0,5 V eff.

Compensation for reactivity change due to poisoning and unauthorized boron removal from the moderator at the shutdown reactor is ensured by insertion of the locking absorber rods that are able to compensate for the total reactivity margin for the cold unpoisoned reactor with pure moderator.

Usage of fuel with low concentration of-fision isotopes in the heavy-water cell being optimal as for reactivity excludes the possibility of positive reactivity insertion with core cell geometry varied and when other media and materials penetrate the moderator and the coolant.

Reaching the criticality with fuel somewhere except for the reactor core is absolutely excluded, therefore the fuel being failed or melted, transported or subjected to reprocessing is absolutely safe from the viewpoint of uncontrelled chain reaction occurrence.

NSSS FEATURES PROVIDING THE OPERATIONAL SAFETY

These features are refered to as:

a) system of bringing the reactor to subcriticality;

b) residual heat removal systems for fuel;

c) system for radioactivity localization within the unit protective barriers.

Prerequisites to implement the above systems in NSSS with TR-1000 PB reactor are as follows:

a) heavy-water moderator available permits to design a double-system to provide subcriticality by injection of liquid absorber into heavy water or heavy water discharge from the core;

b) steam turbine drives available in the gas blowers fed from its own steam permit to develop a coolant forced circulation system based on the fuel residual heat removal followed by the natural circulation;

c) prestressed reinforced concrete vessel is a main barrier localizing radioactivity within NSSS because it confines the coolant (at pressure of 10 MPa), moderator and nuclear fuel thereby ensuring protection against ionization radiation.

The main feature of such vessel that distinguishes it from the metallic monolithic vessel is impossibility of its sudden brittle fracture during normal operation and any emergency situations. It is provided by the great number of prestressed reinforcement components (vertical bundles and peripheral winding), reinforced and anti-shrinkage bafs, great deformability of the reinforced concrete structure. Here, it should be taken into account that the prestressed system used to maintain all the vessel components in compressed state is allocated beyond neutron irradiation having essential effect on its strenght properties and is accessible for in-service inspection as well as for repair and replacement of the separate components with reactor being shut down.

Strong-tight enclosure shall contain the coolant at a gauge pressure of 0,5-0,6 MPa. The last barrier for gaseous coolant discharge into the environment is the containment which shall retain the gauge pressure of 0,1 MPa and withstand the external damaging effects.

ACCIDENT CONSEQUENCES

Accident analyses performed in substantiation of the project show that consequences of so-called design-basis accidents (RIA, LOCA, accidents with loss of coolant circulation and primary cooling) do not go beyond the possible damage to the separate fuel rods and the total activity release into the coolant of not more than 20 Ci.

Probalistic safety analysis has shown that the hypothetical situation with core melting is possible with probability of not more than 10^{-9} 1/reactor year that is attained due to inherent reactor neutronics, considerable redundancy and safety system defence-indepth as well as by various physical backround being applied to the safety systems.

In case of core-melt progression the main part of its structural materials is localized within the coolable in-vessel catcher whereas gaseous, dust-like and aerosol products may be propogated within the primary circuit and after its depressurization -within the strong- tight reactor enclosure.

Design study has shown that the processes of media-material chemical interaction are of the explosive nature; when mixing with air the conditions in the rooms of strongtight enclosure and containment cannot be created to initiate ignition and detonation of the explosive mixture.

In case of core melting the specified catcher structure is used for long-term in-vessel retention of molten fuel; hereat, catcher cooling by natural circulation water shall be provided within the first three years after the catcher is filled with corium. After theree years cooling is possible by heat removal through the catcher and reactor components as well as by heat removal to gas in the reactor cavity.

FUEL CYCLE PATTERN AND ECONOMY

Two alternatives of fuel cycle is under consideration:

-- open-type cycle with usage of U-235 natural content;

-- closed-type cycle with usage of depleted uranium with U-235 (0,28 %) and energy-grade plutonium (0,365 %).

Total spesific core loading with uranium is 160 t/GW;

Average fuel burnup in the reactors is 10000 MW day/t, spesific fuel consumption during reactor operation in open and closed cycles with capacity factor 0,8 is 90 t/GW year;

breending ratio:	
open cycle	0,80
closed cycle	0,84
ultimate concentration of U-235 %:	
open cycle	0,14
closed cycle	0,08

ultimate concentration of Pu %:

open cycle	0,303
closed cycle	0,370

Annual consumption of natural uranium, t:

open cycle	100
closed cycle (depending of initial U-235 content in depleted uranium)	0-30

Refuelling with the help of fuel handling machine is performed on shutdown reactor, hereat, 1/6 part of spent fuel assemblies are replaced by the fresh ones. The spent fuel assemblies are placed into the interim cooling pond wherefrom after about one-year period they are removed into the long-term cooling pond.

Feasibility studies performed for NPP with TP-1000 reactor during the period of 1985-1986 and comparison of the results and the economical indices of TP-1000 plants have shown that cost of the first fuel charge for VVER-1000 plants are close to the cost of heavy water plus the cost of fuel for TP-1000 plants. Investments, operation and energy costs are also close for the above reactor types.

ASPECTS INTERFACING THE TRADITIONAL TRENDS OF REACTOR BUILDING

It's the authors opinion that the separate aspects studies in TR-1000 PB NSSS project may be of interest for the designers of traditional reactor concepts, for example, VVER as regards the next generation plants. Such aspects are referred to as:

a) creation of prestressed reinforced concrete structures. The design experience being gained may be used to develop and construct the VVER prestressed reinforced concrete pit that is able to withstand effects during vessel failure;

b) development of tight and heat conductive fuel of "matrix" type;

c) design of corium catcher;

d) development of pipelines for VVER reactor coolant pumps.

CONCLUSION

Based on the above-mentioned aspects it is advisable to go on with development of TR-1000 PB NSSS and necessary substantiation on including the plants of such type into the nuclear power in Russia.

NPP TR-1000 US in the vessel of prestressed reinforced concrete.

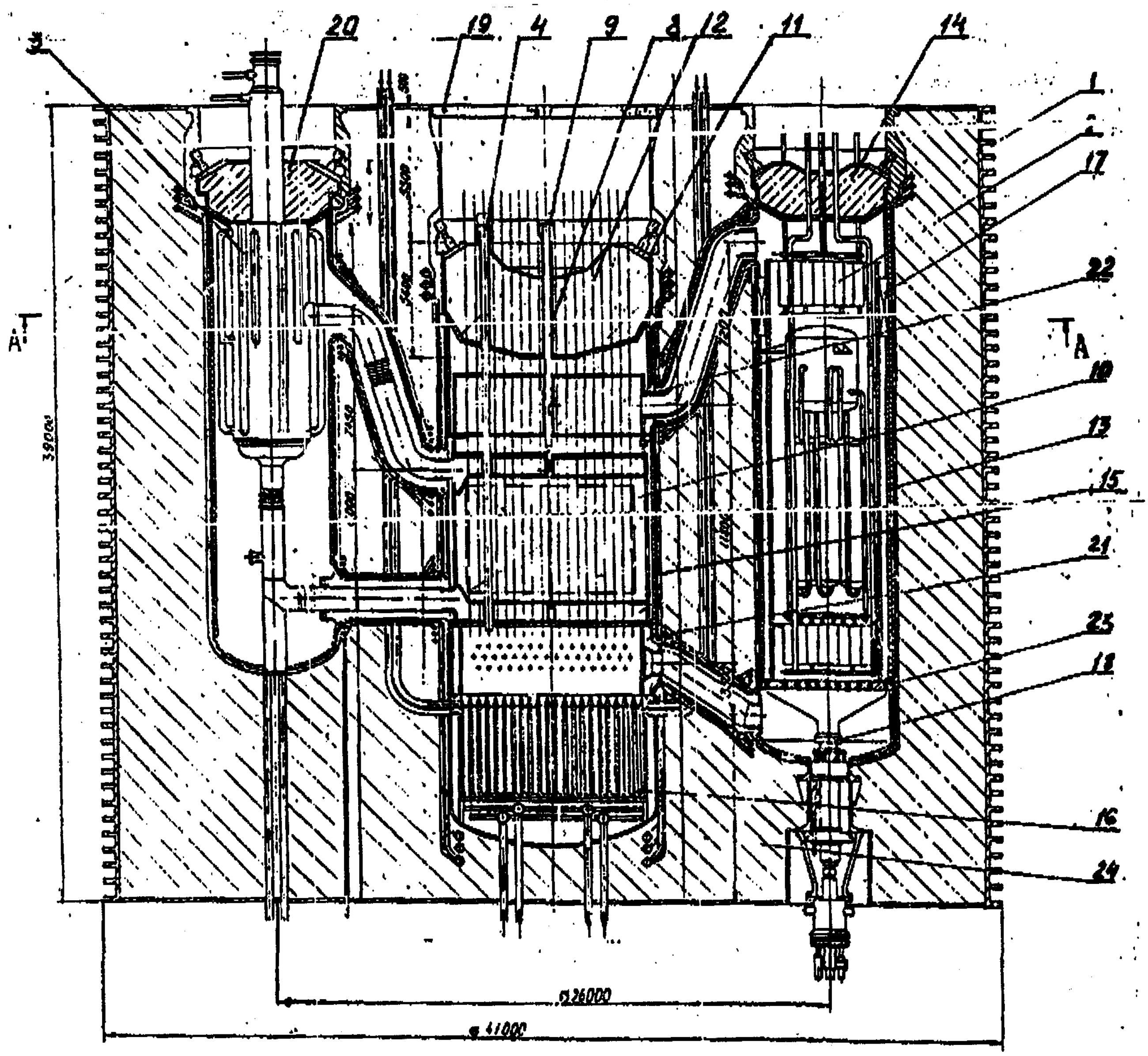

Fig.1- Longitudinal Section of Reactor

1- Vessel, 2- steamgenerator, 3- coolor of moderator, 4- technological channel, 8- channel of emergency system,10- tank of heavy water, 11- main sealing, 12- upper cover, 13- internal heat isolation, 14- cover of steamgenerator hole, 15- liner of internal vessel, 16- trap, 17- exchanger for removing of decay heat, 18- gas circulator, 19- turning circle, 20- cover of moderator cooling hole, 21- inlet chambers, 22- exit chamber, 23- check valve

A-A

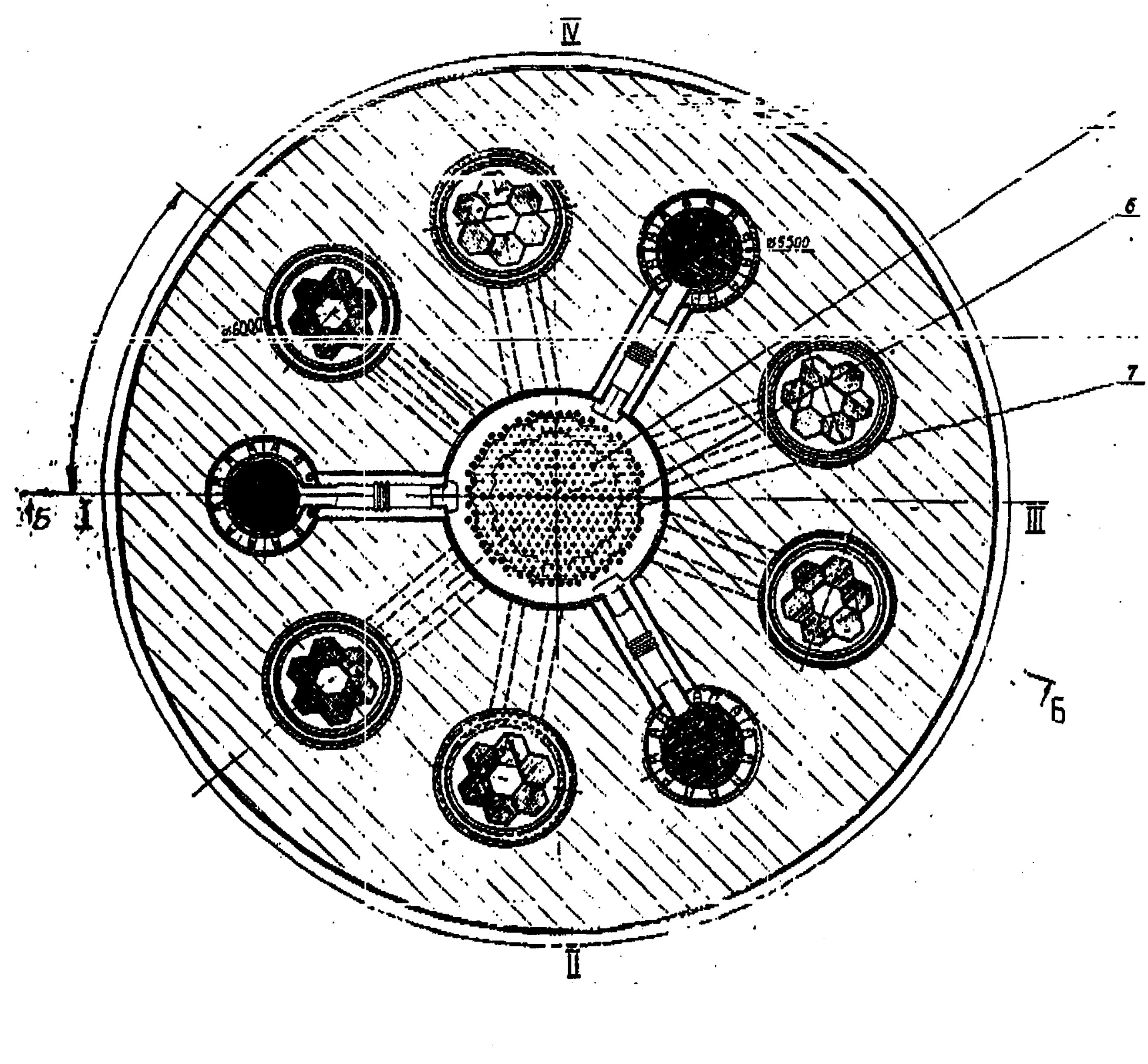

Fig.2- Acrossing Section of Reactor, 5, 6, 7 - Fuel Bundle