The evaporator would be installed in quench pool, the condenser be installed at same level as quench pool. (That is H=25m)

By above results, we realized as follows.

- The containment vessel is smoothly cooled at LOCA.
- Cooling capacity is stepped up by increasing the height(H).

CONCLUSION

The *preliminary analysis of MITSUBISHI SIMPLIFIED Small PWR(MS SERIES)* for decay heat removal system using quench pool and heat pipes is introduced as follows.

a. Short Term Decay Heat Removal System

Small-size containment peak pressure may be able to reasonably hold down under 3 kg/cm^2 (43 psig).

b. Long Term Decay Heat Removal System

Long term cooling may be able to achieve by heat pipe systems.

Therefore, we realized that this basic concept may be adequate.

DESIGN IMPLEMENTATION OF AST-500 PASSSIVE SAFETY PRINCIPLES

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Abstract

In the report are considered design decisions related to realization of passive system principles in AST-500 (nuclear district heating plant) for nuclear district heating plants being in construction and under design in the USSR. High safety-related requirements predetermined a wide application of passive means for accident prevention, protection and restrictions of accidents consequences. Inherent reactor safety, passive protective and isolating systems and devices are referred to the means of this type.

In AST-500 plant an improved safety level was shown to be achieved on account of these means and there are possibilities for further improvement.

1. INTRODUCTION

The concept of promising reactor units of improved and maximum achieved safety developed in the USSR is a complex of engineering measures preventing yield of radioactive products beyond protective and isolating boundaries not only at normal operation and in design accidents but in severe out-of-design accidents. Hereat, a set of emergency situations and accidents is determined with regard for probabilistic approach and severity of potential consequences.

The most important trend of safety improvement is use of passive principles based on application of
properties of inherent safety,
safety-related passive and isolating systems and devices,
natural, separated physical boundaries on the way of radioactivity release.

Passive means are expected to be independent from power sources, personnel action; they do not require switching on and execute action on the basis of natural phenomena.
The indications specified are available in the decisions considered further both in a set and in separate manifestations.

Specific character of AST siting requires additional safety-related requirements reflected in national regulations. They involve requirements for prevention against fuel element melting in case of loss-of-sealing of any vessel of the reactor plant with regard for external effects such as airplane crash and shock wave; requirements for radiation exposure become more strict.

In connection with it maximum use of natural physical phenomena and passive systems, application of direct-action devices, designing according to the fail-safe principle are laid down into design basis.

Safety analysis of AST-500 plant confirms meeting of safety requirements with engineered and economically substantiated margins; the requirements to prevent core destruction are also included. This is related to the full extent to AST being in construction.

Hereat, safety-related investigations have been performed also for out-of-design accidents with a great number of additional failures as well as for emergency situations and accidents with probability of origination of less than 10^{-8} - 10^{-10}. For promising plants are found out possibilities to intensify quality and properties of "forgiving" reactor.

2. MAIN DESIGN DECISIONS AND PROPERTIES ENSURING AST-500 PLANT SAFETY

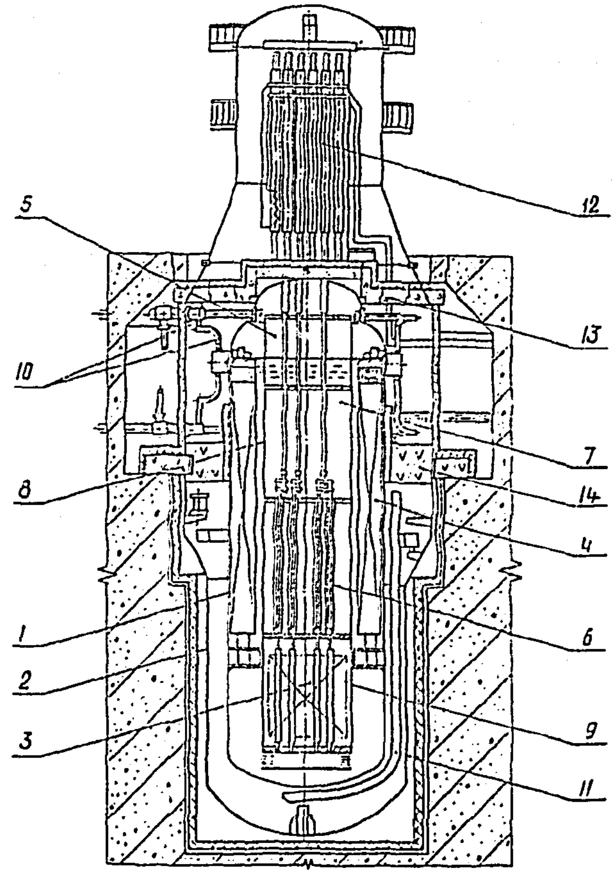
2.1. Main design decisions (see Fig.1,2)

Water-cooled water-moderated reactor:

integral lay-out
great water inventories
natural circulation for all conditions
low parameters
low specific power intensity of the core and fuel
low fast neutron fluence to the reactor vessel
heat removal principle of the reactor protection against pressure increase
minimum branching of the primary circuit

Protective vessel

tightness
operating pressure is held by the accident with reactor loss-of-sealing
value of inner volume prevents against core drying



1 - reactor vessel
2 - protective vessel
3 - core
4 - primary and secondary heat exchanger
5 - steam/gas pressurizer
6 - tie-rod tubes
7 - rising section
8 - unit of tubes and devices
9 - core barrel
10- secondary pipelines
11- turning device
12- CPS drives
13- biological shielding units
14- supporting ring

FIG. 1. Reactor unit.

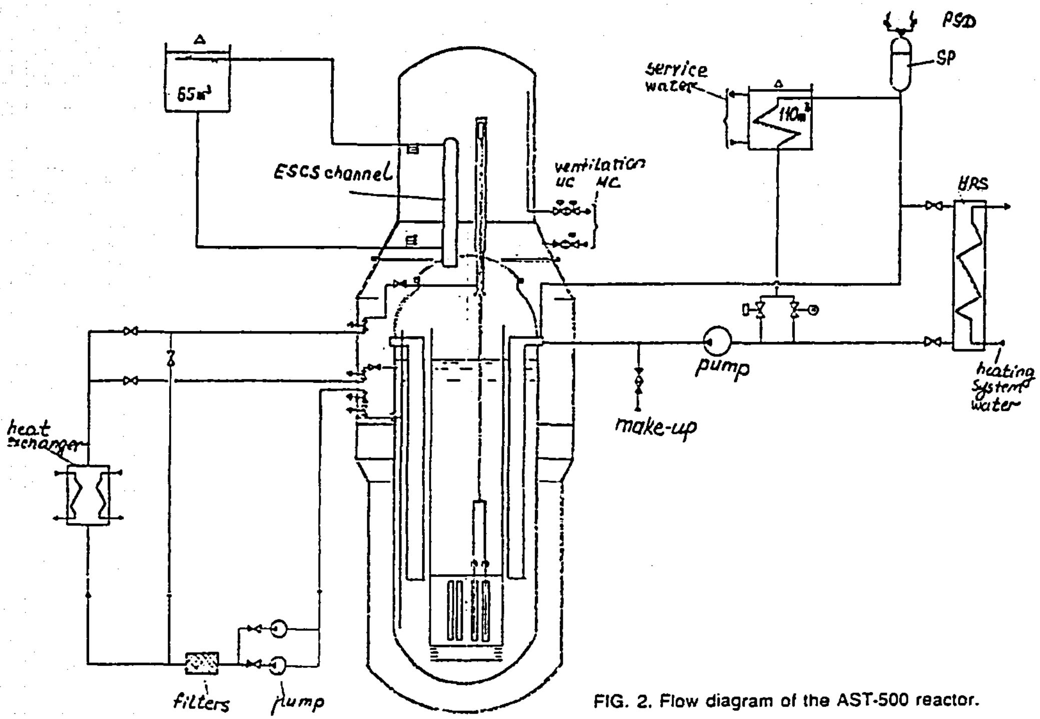


FIG. 2. Flow diagram of the AST-500 reactor.

Emergency heat removal system

redundancy of channels in the system
various types of the reactor and secondary circuit channels
combination of passive and active principles
great water inventories

Three-circuit scheme with pressure boundary excluding activity yield into the heating system

2.2. Properties of inherent safety

Self-control and self-restriction of reactor power on account of negative power, fuel temperature and moderator void coefficients of reactivity.

Self-profiling of natural circulation flow rate through the core is proportional to power.

Sluggishness of transient and emergency conditions.

Increased heat-storage capacity.

Effective primary and secondary thermal bond.

Low level of activity in fuel elements, low yield of fission products from "cold" fuel.

Low rates of water outflowing from the reactor in accidents.

Slow flaw development (leak before break criterion).

3. SEPARATION OF PROTECTIVE MEANS

3.1. Means for the reactor safe shutdown and its keeping in safe state:

1) insertion of absorbing rods into the core under the conditions of electric movement on the signals of different physical character;

2) gravity insertion of absorbing rods on signals of different physical character or as a result of operation of direct-action splitters (immediately due to growth of P_I), see Fig.3.

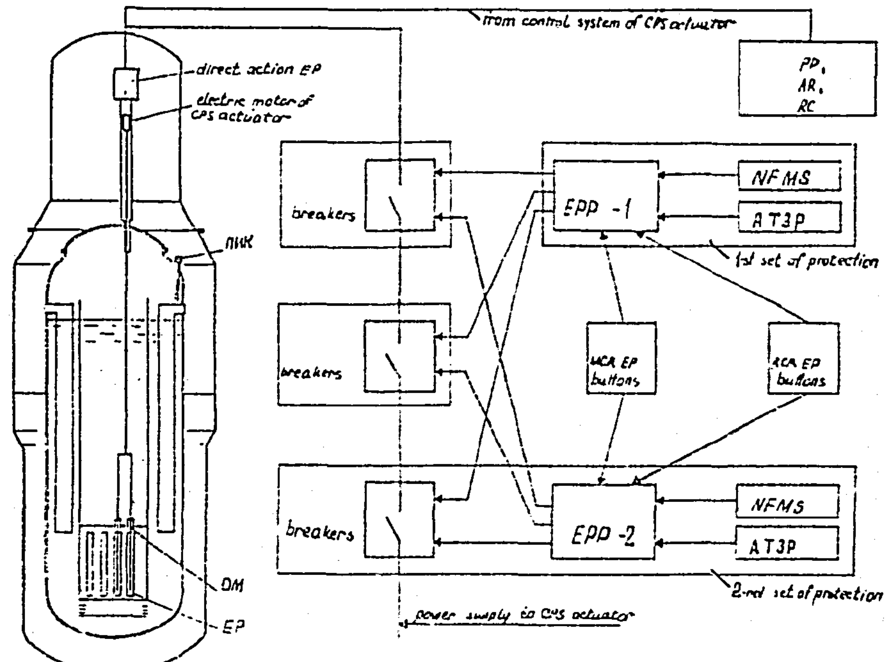


FIG. 3. Means affecting reactivity (CPS electrical and mechanical systems).

3) active injection of fluid absorber (boric acid);

4) passive injection of boric acid, see Fig.4 ;

5) self-restriction of power in the tight "hot" reactor;

6) self-shutdown of the unsealed reactor with steam generation in the core.

3.2. Fuel cooling:

1) long-term accumulation of residual heat by the coolant and metalworks of circuits;

2) active heat removal into the emergency shutdown cooling system in the secondary loops at active system connection;

3) passive heat removal into the emergency shutdown cooling system (ESCS) at constantly introduced channel in the secondary loop as well as at usage of direct-action devices for connection of other ESCS channels, Fig.7.8

4) passive heat removal via secondary pulse safety devices of direct action (Fig.5);

5) passive heat removal through the reactor emergency shutdown cooling channel at its connection with the help of direct-action device;

6) long-term passive heat removal at water evaporation from the reactor;

7) active heat removal through heat-exchangers of primary purification system.

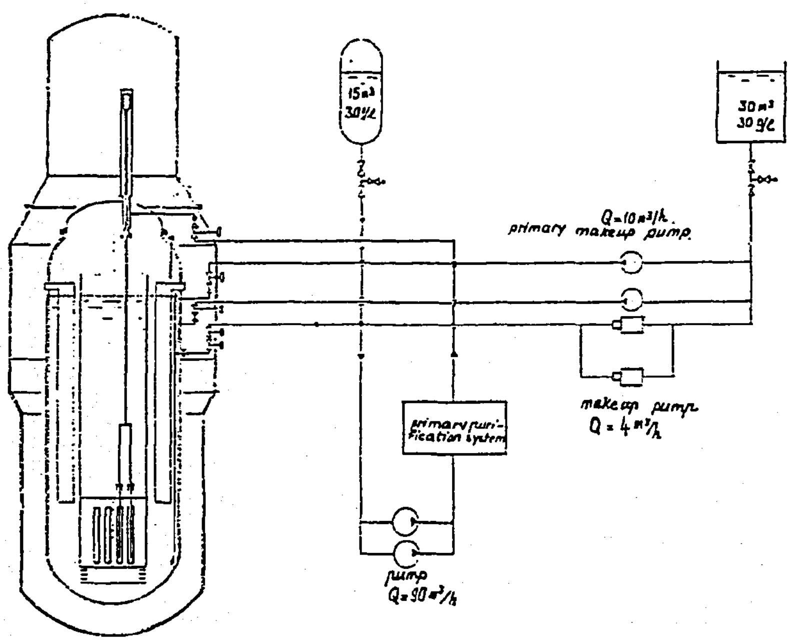


FIG. 4. Means affecting reactivity (CPS fluid system).

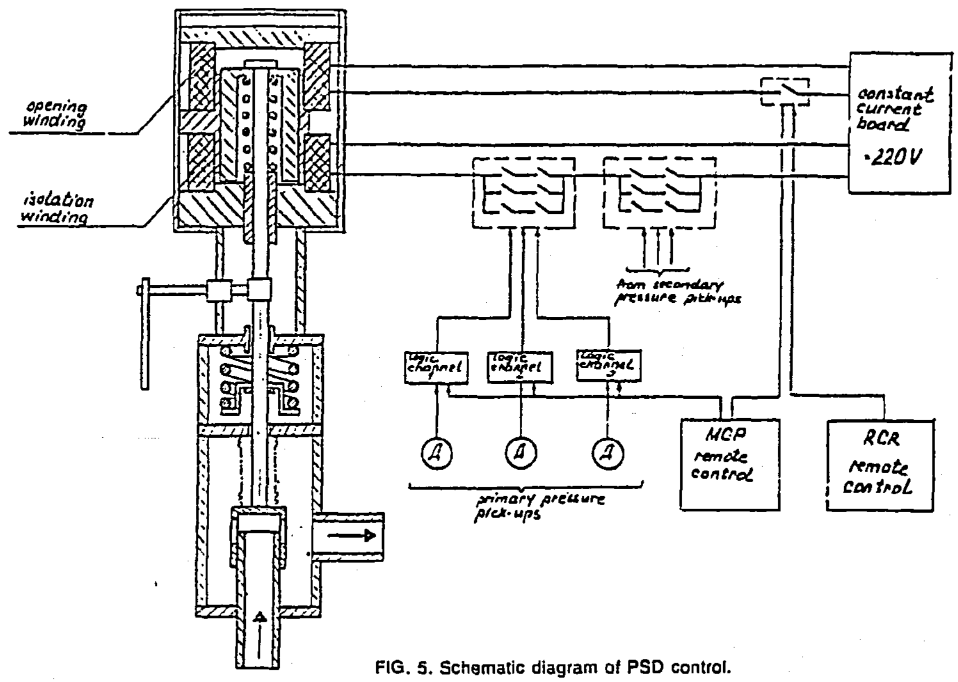


FIG. 5. Schematic diagram of PSD control.

3.3. Isolation of radioactive products and decrease of emergency radioactivity excursion (see Fig.6):

1) "cold" fuel on the basis of UO_2;

2) fuel element claddings;

3) tight reactor with the great safety factor;

4) protective vessel;

5) intermediate circuits designed for primary pressure;

6) tight spaces; reactor concrete pit;

7) bubbling of emergency effluents prior to their release into the containment volume;

8) containment with filters;

9) reduced activity of accident effluents on account of the core keeping under the water level.

Thus, safeguards envisaged by the design are based on preferable use of passive principles of protection and inherent properties of the plant, separation of protective and isolation means.

Application of direct-action devices and passive components improve the reactor plant protection against common-mode failures, including inside and outside effects and personnel errors.

4. PASSIVE AND DIRECT-ACTION DEVICES (self-operation)

4.1. Floating of the drive rack with CPS control member at loss-of-sealing of the primary circuit is prevented in passive by exceeding the control member weight over buoyancy force arizing at loss-of-sealing of CPS drive casing within the range of pressure restricted by protective systems.

Besides, overrunning clutch operating by angle wedging principle is provided in the design of the drive.

4.2. Direct-action emergency protection

This device provides a reliable connection of EP at direct effect of mode parameter - emergency primary pressure excess. Principle of bellows movement is used in the design. This device is a redundant one operating under out-of-design emergency situation of EP non-operation on control electric signals and adjusted against them by operation setting (Fig.3).

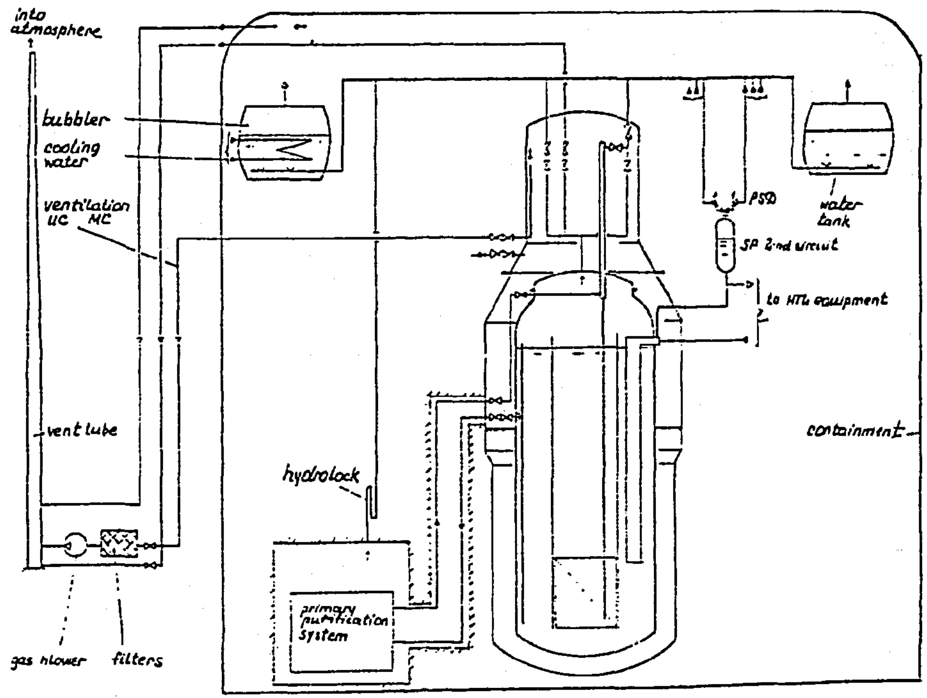


FIG. 6. Means of radioactivity blow-offs attenuation.

4.3. Solenoid valve in the emergency heat removal system

Solenoid valves are used to increase reliability of putting ESCS into operation by the initial cause of emergency situation-de-energization. Loss of electric supply bringing about cessation of heat removal into the system, simultaneously it results in de-energization of ESCS solenoid valves and their opening on account of force of compressed spring (see Fig.7).

4.4. Direct-action device for ESCS connection

This device provides a reliable operation of pneumatic valves of safety systems at direct action of mode parameter-variation of pressure causing bellows movement. This device is a redundant one and operates at failures of electric system for valves control and adjusted by operation setting (Fig.8).

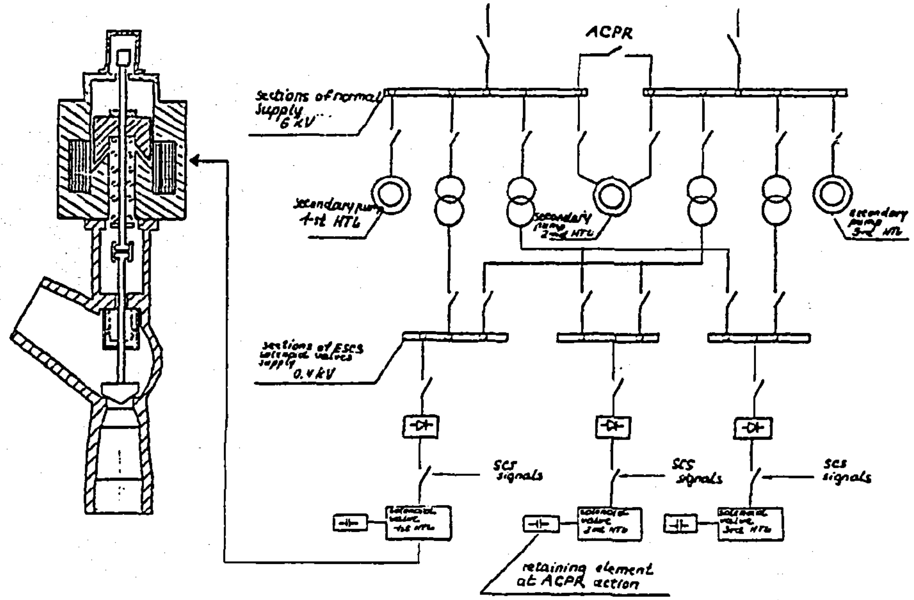


FIG. 7. Schematic diagram of ESCS valve supply and control with electromagnetic drive.

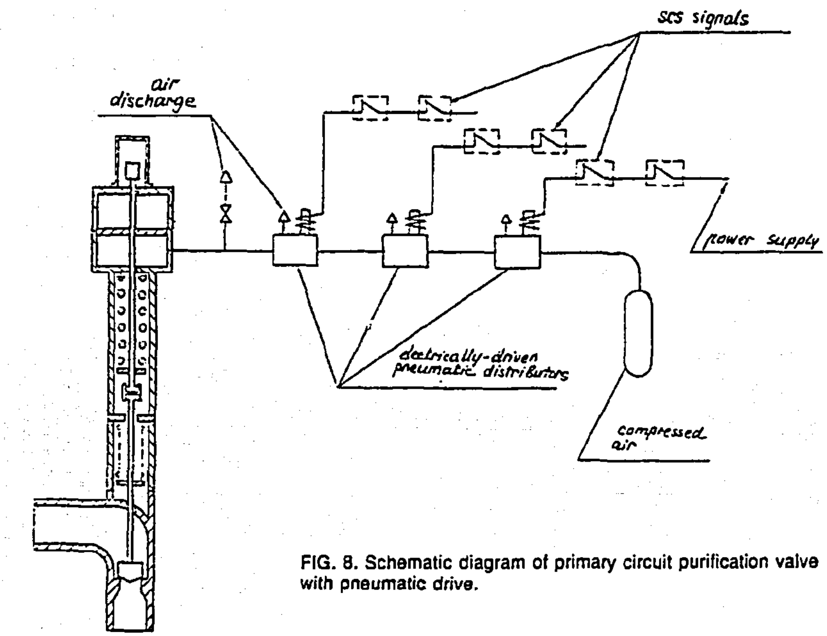


FIG. 8. Schematic diagram of primary circuit purification valve with pneumatic drive.

4.5. ESCS passive channel in the secondary loop

In addition to ESCS channels connected to the secondary loops at valves operation, analogous ESCS channel is possible to be used, the latter has a constant hydraulic connection with the secondary loop.

Hereat, a developed circulation is kept in the channel under all operating conditions of the plant with desired use of the heat carried out into ESCS.

4.6. ESCS passive channel in the primary circuit

ESCS passive channel in the primary circuit provides a long-term ($>$ 12h) removal of residual heat under the steam/condensate conditions over the primary circuit at evaporation of cooling water inventory under the containment. Reactor pressure drops together with channel connection due to gas component evacuation from the reactor.

4.7. Device for connection of ESCS passive channel in the primary circuit

Various versions of the device design are possible. This device connects in passive ESCS channel in the primary circuit along steam/condensate path at direct action of mode parameter - reactor pressure growth. Operation setting of the device is adjusted against the value of the reactor operating pressure and the value of connection setting of other channels of emergency heat removal. Putting into operation of the channel restricts the reactor pressure by value $\sim$ 0,45 $P^{vessel}_{ultimate}$ (Fig.5).

4.8. Limiter of primary media outflowing

In the intake pipeline of purification system is provided a passive device reducing water losses from the reactor in the accident with the specified pipeline break outside the boundaries of protective vessel. This device is an additional protection line with respect to failure of double quick-isolating valves of the primary circuit. At direct action of mode parameter - level lowering in the reactor (remaining much above the core) water outflowing from the reactor transfers into steam-water one and then into the steam one. The core is kept under the water level more than 12 hours without reactor makeup.

4.9. Pulse-safety device in the secondary loops (PSD)

PSD is an additional separation line (in respect to heating system, normal shutdown cooling system, ESCS) that realizes principle of heat removal of the reactor protection against pressure excess with implementation of rigid thermal bond between primary and secondary circuits: PSD operation and secondary medium discharge is caused by direct action of mode parameter - emergency growth of the secondary pressure; at unique bond of emergency increase of pressure and temperature in the reactor with secondary pressure and temperature under situations with postulated cessation of heat removal (into heating system, normal shutdown cooling, ESCS) (Fig.5).

4.10. Protective vessel

All-metal protective vessel is a passive device of isolating and protective safety systems. The size of gaps between the protective and the reactor vessels doesn't exceed the value whereat is secured the core under coolant flooded in emergency situations with loss-of-sealing of the primary circuit.

Simultaneously arrangement of equipment and instrumentation for vessel metal control in this gap is being discussed.

Protective vessel is designed for pressure arising in emergency situations with loss-of-sealing of the reactor vessel by maximum possible cross-section.

Possibilities for creation of combined protective vessel with all-metal upper part and concrete metal lower part integrated in design and function with the reactor shaft are also being studied.

5.MAIN RESULTS OF SAFETY ANALYSIS

5.1. At normal operation radiation exposure from AST is significantly ($I0^4$ times) lower than the level of natural background effect (absense of leaks and discharges, tightness of circuits, low neutron fluxes outside the reactor vessel boundaries).

Complex of physical-and-engineering peculiarities and RP inherent properties (discussed above) is aimed at prevention of accidents.

In design accidents safety is secured during continuous time (more than one day) without power consumption and without personnel action with implementation of passive devices and reactor shutdown systems, emergency heat removal, isolation of radioactive products.

In out-of-design accidents with failure of redundant safety systems the core destruction is prevented due to properties of inherent safety of the plant, operation of redundant passive devices with direct action, corrective personnel

actions related to introduction of redundant safety systems (with time reserve for their connection being more than 12 hours).

The statements above said are highlighted by the examples of the most severe out-of-design emergency situations and accidents. Approaches to working out the list of out-of-design accidents, selection of boundaries with respect to the number of failures are being coordinated with IAEA recommendations.

5.2.Emergency situations pertained to interruption of heat removal from the primary circuit.

Protective ability of the plant due to its inherent properties is manifested also in heat removal emergency situations.

Heat storage capacity of the reactor, the secondary and heat circuit serves as the first line in such situations. Redundant three-channel ESCS system in the secondary loops is connected, as it was specified, with high reliability and secures lowering of parameters in the reactor with passive functioning of at least one channel. Time of passive heat removal without personnel interference - more than 24 hours and with regard for make up of ESCS tanks is unlimited in time. Reactor pressure during emergency process does not exceed $I,I\ P_{I\ operating}$.

Residual heat removal system through PSD in the secondary circuit serves as a redundant one. 50% of PSD provide the reactor shutdown cooling, hereat, reactor pressure during the whole emergency process doesn't exceed $1,4\ P_{I\ operating}$. Time of passive heat removal without personnel interference is 12 hours and with regard for make up of the secondary loops is unlimited in time.

Additional redundant ESCS channel in the primary circuit secures its long-term passive cooling (more than 12 hours without making up the cooling water tank, it is unrestricted in time with regard of its makeup).

Hereat, its pressure in the reactor is restricted by value of $0,45\ P_{ultimate}$ where $P_{ultimate}$ - collapsing pressure from evaluations of ultimate strength characteristics of the reactor.

With regard for redundancy and separation of emergency heat removal channels, high reliability of their connection, hydraulic and spatial independence, protection against common-made failures (external effects and personnel errors are included), considerable time reserves (more than one day) to perform corrective actions, emergency situation of complete cessation of residual heat removal from the reactor fails to be technically realized; heat removal emergency situations do not develop within loss-of-coolant accident and haven't radiation consequences.

5.3. Cessation of heat removal into the heating system with postulated EP failure.

This emergency situation is considered to highlight inherent properties of the plant, whereas complex of design decisions implements it into the category of technically unrealized ones.

Emergency situation with cessation of heat removal into heating system at postulated EP nonoperation is pertained to primary circuit heating-up and increase of P_I pressure in spite of ESCS channels connection and PSD operation in secondary loops. Reactor power decrease happens simultaneously due to negative reactivity coefficients, Stabilization of parameters occurs at the reactor pressure level $\sim 4,2$ MPa and power level $\sim 0,1$ N_{nom}.

5.4. Emergency situations with inadvertent reactivity insertion.

5.4.1. In design emergency situations with additional reactivity insertion stipulated by coolant cooling in the core or inadvertent withdrawal of simultaneously moved CPS members, neutron power does not exceed the setting of preventive protection operation.

5.4.2. Out-of-design emergency situations with primary coolant cooling proceed close to the design ones.

Analysis shows that technically unrealized situation with superimposing of simultaneous EP failure do not transfer into accidents with loos-of-coolant from the reactor and radiation consequences. This is attributed to inherent reactor properties (self-control of natural circulation flow rate is proportional to power, sluggishness of cooling processes).

5.4.3. Out-of-design emergency situations of inadvertent withdrawal of CPS members.

Value and rate of positive reactivity insertion is restricted by design decisions both with remote and automatic control (possible quantity of simultaneously moved CPS operating members is resticted).

High reliability of emergency protection is improved by use of passive breakers of power supply of CBS drives that permits to characterize emergency situations with inadvertent withdrawal of CPS operating members without EP operation as technically unrealized.

EP operation excludes transition of similar emergency situation into loss-of-coolant accident and radiation consequences.

Nevertheless, the specified technically unrealized emergency situation is considered in analysis of the plant safety showing the fulfilment of safety provisions on account of inherent properties of the reactor.

5.5. Primary circuit loss-of-sealing accident.

5.5.1. In safety analysis are considered accidents with reactor vessel loss-of-sealing, ruptures of primary pipelines, intercircuit untightness of primary and secondary heat exchangers, etc.

Design accidents of primary circuit loss-of-sealing in connection with availability of protective vessel,isolating valves on primary pipelines in the protective vessel, counterpressure from the secondary side proceed with the core kept above coolant level.

Reactor makeup is not required. Radiation conditions do not exceed limits set for normal operation conditions.

5.5.2. Loss-of-sealing of the reactor vessel in the lower part with untight protective vessel is looked upon as out-of-design accident. Lower part of the reactor and protective vessels has no pipelines with the exception of air removal duct from the protective vessel lower part. Dy5 reducing device and double isolating valves are mounted on the air removal duct. In consideration of the accident, valves are assumed to stay open and heat removal is performed by ESCS system. During continuous time (not less than 12 hours) the core is kept under the coolant level (see Fig.9) without any measures for accident elimination.

Accident elimination measures involve valves closing, reactor make-up and shutdown cooling, opening of blow-off from the reactor.

5.5.3. Out-of-design accident with purification system pipeline rupture (maximum blowdown cross-section) at double isolation valves failure to close. Additionally heat removal is assumed to be performed from the reactor via incomplete number of ESCS channels. This accident is characterized by maximum loss-of-coolant from the reactor. Hereat, time during which corrective actions are to be performed (reactor make-up to keep the core under water flooded) is 24 hours (see Fig.IO).

5.5.4. Accidents with the primary circuit loss-of-sealing with EP failure accompanied are practically excluded. Nevertheless, safety provision on account of inherent plant

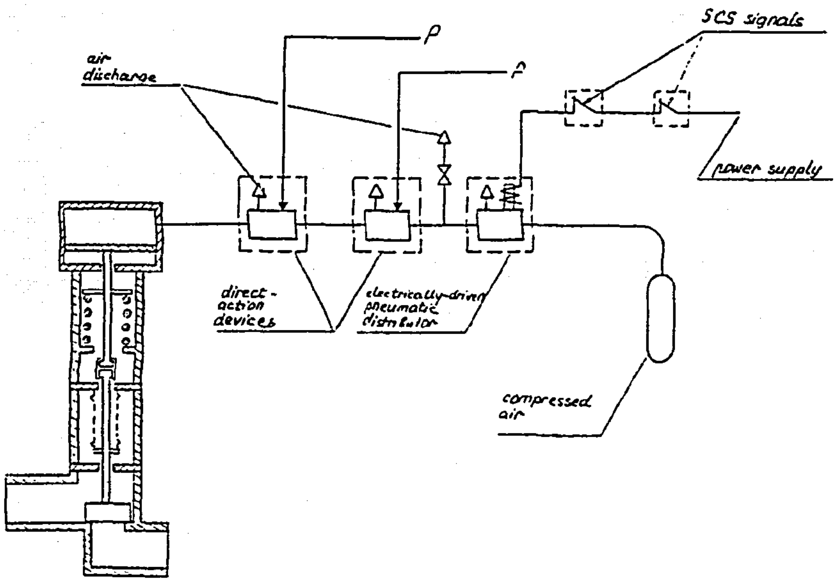


FIG. 9. Schematic diagram of ESCS valve control with electromagnetic drive.

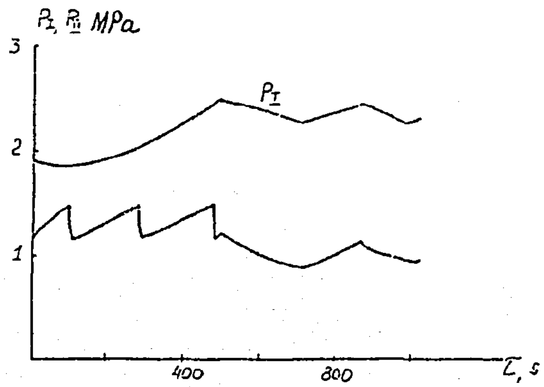


FIG. 10. Heat removal with the help of PSD.

properties in similar technically unrealized accidents has been analyzed. Hereat, it was shown that due to negative steam coefficient of reactivity the core transfers into subcritical state; the core is kept under coolant flood; the reactor stays in safe state during continuous time (not less than 12 hours).

Transition of the reactor into the cold state may be accomplished on account of boric acid solution supply, additional make-up and complete shutdown cooling.

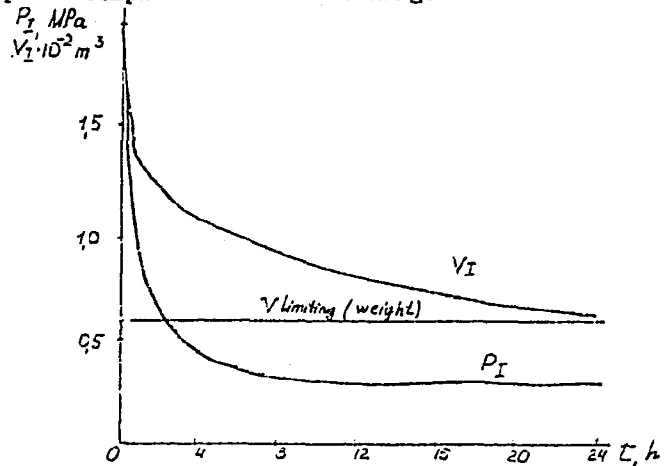


FIG. 11. Loss of sealing of purification system with isolating valves failure.

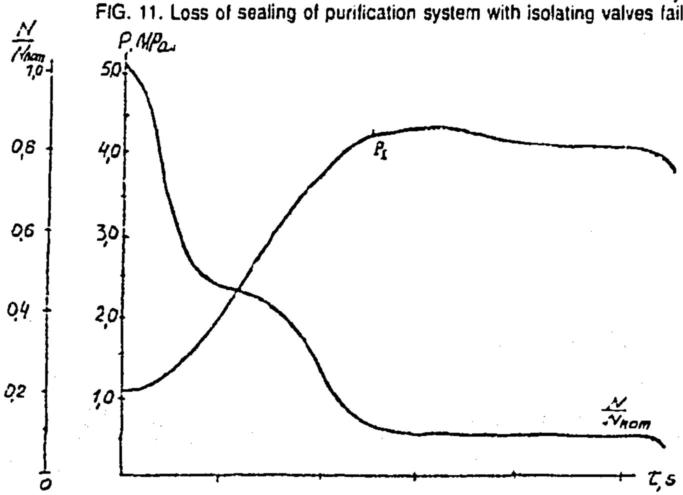


FIG. 12. Cessation of heat removal into the heating system with reactor EP failure (technically unrealized emergency situation).

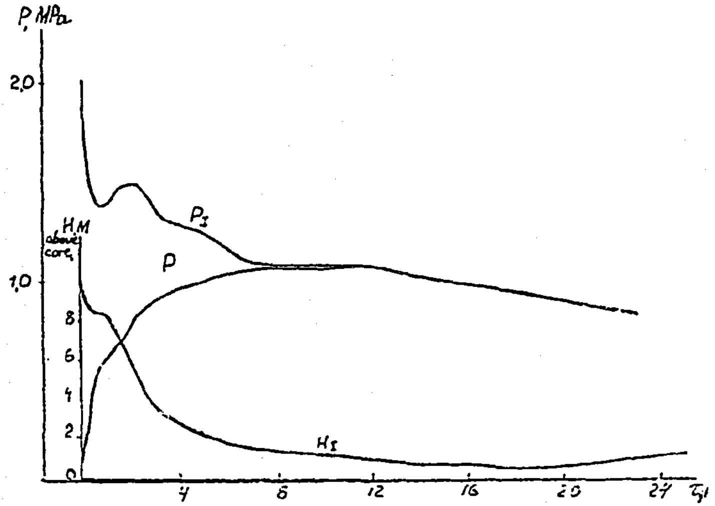


FIG. 13. Loss of sealing of the reactor vessel in the area of the inlet chamber with the valves *open* in the air removal pipeline from low cavity (LC) of mixing chamber (MC).

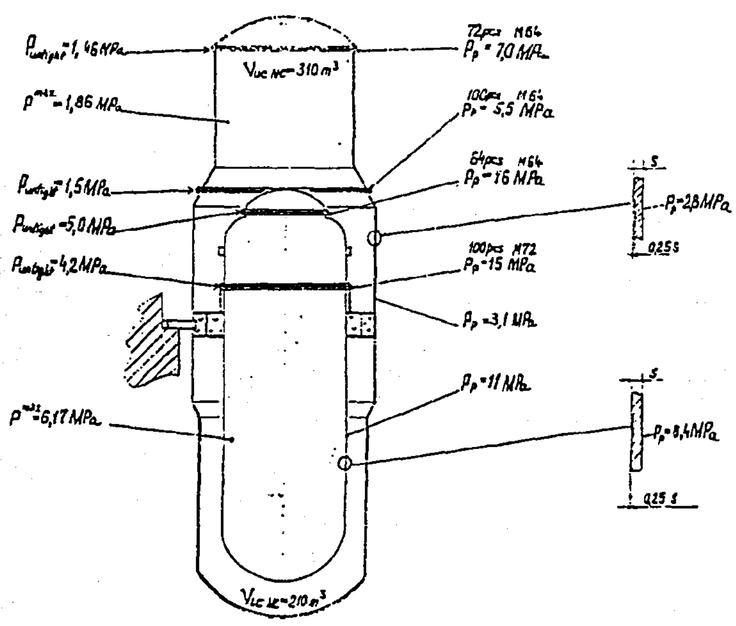


FIG. 14. Limiting characteristics of the MC reactor system.

ULTIMATE STRENGTH CHARACTERISTICS OF THE REACTOR AND PROTECTIVE VESSEL

Hypothetical accidents wherein reactor parameters may go beyond the limits of design values require analyses of ultimate strength characteristics of the reactor and protective vessel. Analysis of stress-strain state of the reactor vessel showed that in the course of operation appearance and growth of fatigue cracks are excluded. Calculations of carrying ability of the reactor vessel with postulated initial flaw of 0,1 S depth gives the following: flaw development to 0,25 S is possible only on the basis of rather great /4.10^6/ number of cycles of "deep" /from 0 to 2,3 MPa/ variation of operating pressure incomparable with number of cycles of any variation of pressure expected during the whole service life of the reactor plant. Further cyclic loading results in realization of criterion "leak before break". Vessel in untight state is capable to withstand a great number of loading cycles /$\sim 10^5$/. Hereat, ultimate size of vessel untightness doesn't exceed Dy 45.

During postulation of unlimited pressure growth it was obtained that auxiliary equipment and primary pipelines are *more* strong than the reactor vessel.

Evaluation of ultimate possibilities of the reactor vessel showed that its "weak" element is a cylindrical shell in the lower part, its minimum thickness equals 50 mm.

Value of destructive pressure determined from the condition of reaching ultimate strength by stresses at its minimum value within the range of variation of vessel material mechanical properties, amounts to $P_p = 11$ MPa.

Consideration of postulated "standard" flaw diminishes destructive pressure to $P_p = 8,4$ MPa.

Destructive pressure for main joint studs at minimum values of mechanical properties amounts to $P_p = 16$ MPa (upper joint) and $P_p = 15$ MPa (lower joint). Tensile stresses in the studs reach the yield point at $P_T = 13\text{-}14$ MPa.

Ultimate pressures in the vessel whereat the joint tightness is preserved are evaluated by the results of hydrotests of the head reactor vessel as well as of bench tests of the upper joint:

$P_H = 4,2$ MPa - for the lower joint

$P_B = 5,0$ MPa - for the upper joint.

Ultimate parameters of the protective vessel are characterized analogously by the following parameters:

$P_{ac} = 3,1 - 3,5$ MPa - for "defect-free" vessel,

$P_{ic} = 2,8$ MPa - for protective vessel with "standard" flaw.

Pressure of its joints opening are evaluated by test results of the head unit:

$P_H^{ac} = 1,5$ MPa - for lower joint

$P_B^{ac} = 1,46$ MPa - for upper joint.

Thus, joints of the reactor and protective vessel play the role of safety valves in consideration of hypothetical accidents.

Steam bleeding through joints of the reactor and protective vessel with flaw rate equal to steam rated capacity of the core on account of residual heat results in pressure limitation in the reactor and protective vessel.

FIG. 14. (cont.)

6. CONCLUSIONS

In AST-500 reactor plant for nuclear district heating plants being in construction and under design are realized design decisions which permit to use widely the passive principles of safety assurance. The reactor plant is characterized by such properties of inherent safety as self-restriction of power (due to negative coefficients of reactivity), increased heat storage capacity, sluggishness of accident conditions, low level of activity in the fuel element, slow development of flaws, etc. Properties of inherent safety in combination with application of passive protection and isolation systems and components permit to realize self-provision of the plant safety during long-term "period of non-interference".

Design materials highlighted that accident with core destruction for the plant of such type is technically unrealized event.

Thus, in AST-500 plant is reached improved safety level that permits to site nuclear district heating plants in the vicinity of town.

SYMBOLS ACCEPTED IN THE TEXT

AST - nuclear district heating plant

CPS - control and protection system

ESCS - emergency shutdown cooling system

NCS - normal cooling system

PV - protective vessel

PSD - pulse-safety device

OM - operating member

EP - emergency protection

EPP - emergency protection panel

LFT - low-frequency transducer

MCR - main control room

RCR - reserve control room

SCS - safety control systems

HTL - heat transfer loop

SP - steam pressurizer

HRS - heat removal system

PP - preventive protection

RC - remote control

AR - automatic regulation

NFME - neutron flux monitoring equipment

LC - low cavity

UC - upper cavity

MC - mixing chamber

ACPR - automatic change over to power reserve