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REVIEW OF NHR ACTIVITIES IN THE RUSSIAN FEDERATION

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Abstract

NHR development activities in the ex-USSR were initiated in the 1970s mainly due to a growing deficiency of organic fuels needed for heating large cities in the European part of the country. Construction of two pilot nuclear district heating plants with AST-500 NHRs was started in the early 1980s, and by 1989 the first unit in Gorky NDHP was nearly 90% completed. Current activity in this field is concentrated on upgrading the AST-500 design and on the development on this basis of a whole series of heating-only and co-generation reactor plants of unit power ranging from 30 to 600 MW. A brief description of the AST-500 reference NHR design features is given, as well as of the R&D activities that have been carried out for the design decisions and safety validation.

1. INTRODUCTION

The fraction of fossil fuels being consumed in the Russian Federation for the generation of low grade heat for industrial and district heating purposes amounts to 30-40% of the total consumption, where the most qualified fuels such as natural gas and oil comprise the main part. Therefore, the use of nuclear fuel for heating and hot water supply could become a significant contributor to the improvement of the fuel consumption balance of the country. It would thus relax the problems related to the remoteness of the most energy-consuming regions from the main fuel production areas, which are mainly located in the East of Russia.

Besides, it was realized that the substitution of fossil fuel by nuclear one would give an important social-economic effect through attenuation of the adverse influence of the traditional power technology on the environment, particularly in large cities.

By the early 80s favourable conditions existed in the ex-USSR for the introduction of nuclear reactors into the district heating grids. Such conditions were promoted by the following factors:

- positive experience that had been gained by the national power industry in centralized district heating using large energy sources and heat-distributing grids;

- a stable tendency to increase the heat load concentration and unit power of district heating sources;

- deficiency in fossil fuels and large expenditures for its long-distance transportation;

- mastering of large-scale manufacture of equipment for NPPs by the national industry;

- experience gained in construction and operation of powerful nuclear power plants with steam extraction for heating of nearby satellite towns.

All these factors stimulated the start of the first NHR of AST-500 type development and pilot heating plants construction [1].

The analysis performed to assess both the current and prospective levels of municipal heating loads and their concentration showed that in the European regions of Russia there are several tens of large heat-consuming centers with heating loads of more than 1000 GCal/hr. For the majority of the regions, the anticipated increment of heating load for the nearest future allowed to use twin-unit district heating stations of a total thermal power around 860 GCal/hr. A feasibility study showed the competitiveness of such a plant with fossil-fuelled heating stations.

Pilot twin-unit nuclear district heating stations with 500 MW(t) AST-500 reactors were intended to provide heat for the highly-populated districts of the large industrial cities of Gorky, Voronezh and Arkhangelsk. All these cities suffered from an acute shortage of fossil fuel which decelerated the municipal activity in the field of domestic building.

A minor construction activity is now under way for the Voronezh NDHP only, while the Gorky NDHP has been cancelled in 1991 due to strong protests of concerned local population. Recently, the designs for the advanced commercial NHRs AST-500M and AST-600 were developed, and a feasibility study for the construction of such reactors is being carried out for the Khabarovsk NDHP (Far East of Russia).

According to the recently issued Concept of Nuclear Power Development in the Russian Federation [10], the pilot twin AST-500 NDHP should be realized until 2000 (construction work completion and power start-up). In the same period of time, pilot small power electricity and heat co-generation nuclear plants are planned to be constructed to satisfy the energy needs of remote regions in the Far North and the Far East of the country.

2. AST-500 REFERENCE DESIGN CONCEPT AND PRINCIPAL DECISIONS

The AST-500 reactor plant principal flow diagram, the main engineering decisions and the main parameters are given in Table 1 [2,3].

The AST-500 NHR conception was preceded by the development of NHR-specific requirements for the safety of such nuclear plants. They were adopted as a supplement to the basic National Code for NPP safety. These requirements envisage the necessity to exclude fuel melt at loss of integrity of any reactor-related pressurised system and to account for external impacts like air crash and shock wave. Besides, stringent requirements to radiological safety were introduced, including a collective dose limit specification [4, 5].

A PWR-type nuclear reactor with positive intrinsic properties, particularly as it concerns nuclear safety, such as neutron power self-control and self-limitation, was adopted for the AST-500 design.

The chain reaction self-control capability in virtue of the negative temperature, power and void reactivity feedbacks was enhanced additionally in the AST-500 thanks to a rejection of soluble boron reactivity control and to the use of burnable poison in the core.

The feature of the NHR as a source of low-grade energy gave the opportunity to adopt much lower parameters for the primary system (T=200°C, P=2.0 MPa) compared to the traditional light water reactors. Conforming to the specific heating purposes, the required unit power of NHRs can be considerably lower than that generally accepted in contemporary nuclear power (up to 500 MW), depending on the specific heat load characteristics of the heating grid connected to a given NHR. Along with the reduction of coolant parameters, the

TABLE 1: AST-TYPE NHP DEVELOPMENT HISTORY IN THE EX-USSR

First phase (1977-1986)

* AST-500 Basic Design development, R&D activity
* Pilot AST-500 units construction in Gorky and Voronezh

Second phase (1986-1989)

* AST-500 design updating
* Peak of construction activity on both Sites
* Gorky NHP independent safety review (IAEA Pre-OSART mission)

Third phase (after 1989)

* Voronezh NHP design updating
* AST-500 advanced design development
* Series of AST-type NHP designs development
* Public acceptance and financing problems growing

NHP CURRENT ACTIVITY

* Feasibility study for Khabarovsk NHP (2 x AST-500M)
* Voronezh-NHP licensing (environmental impact review)
* No construction activity
* Minor design and R&D activity on advanced heating reactors
* AST-based co-generation (electricity and heat) NPPs development
* Activity to influencing public acceptance

NHRs PREMISSES IN THE RF

* DH - major contributor to fossil fuels consumption
* Considerable remoteness of fuel production fields
* Availability of powerful district heating grids
* Growth of heating loads and their concentration
* Positive experience in DH from operational NPPs

NHRs Motivation in the RF

* Large - scale substitution of fossil fuels
* Environmental benefits
* Scale down of fuel transport flows
* Competitive generating cost
* Improvements in working conditions and standard of life

NHRs IN DISTRICT HEATING GRIDS

Heat power appr. 50% of max. heating load

* 4500-5500 eff. hr operation per year
* At least two NHRs on a site
* Load - follow mode of operation
* Direct water rated temp. - 150°C

AST-500 BASIC DESIGN OBJECTIVES

* Simplicity and Reliability
* Invulnerability to Multiple Failures
* Immunity to Personnel Errors
* Large Grace Period
* Corium Confinement in RPV
* Exclusion of Population Evacuation

NHP Specific Safety Requirements

* Fuel melt exclusion
* External impacts consideration
* Reduced design limits for fuel failure
* Minimized radiological impacts

AST-TYPE NHR BASIC DESIGN FEATURES

* Integral PWR in guard vessel
* Natural convection of primary coolant
* 3-circuit heat transport scheme
* Low coolant parameters
* Low core power density

HEAT REMOVAL PRINCIPLE FOR PRIMARY CIRCUIT OVERPRESSURE PROTECTION

* Passive self-actuated safety systems
* Proven technology (PWR, operating or tested prototypes)

AST-TYPE NHPs BASIC DESIGN DATA				
	Parameter	Design Concept		
		AST-500M	AST-200	AST-30B
1.	Heat output, MW	500(600)	200	30
2.	Primary coolant			
	- pressure, MPa	2.0	2.0	0.54
	- inlet/outlet temperature, °C	131/208	147/208	70/144
3.	Secondary coolant			
	- pressure, MPa	1.2	1.2	1.08
	- IHX inlet/outlet temperature, °C	88/160	113/155	58/109
4.	Grid circuit			
	- pressure, MPa	2.0	2.0	1.57
	- GHX inlet/outlet temperature, °C	70/150	70/150	50/95

AST-500 Design Validation

* NHR - related experimental base
* NHR thermal - hydraulics
* natural circulation
* reactor hydrodynamics
* boiling crisis
* two-phase flows
* coolant outflow at LOCAs
* NHR safety at accidents
* RPV/GV structural mechanics and seismic stability
* RCS chemistry
* NHR neutron physics
* Functional tests of equipment and components

Design Objectives for Advanced AST Development

* Gaining better economics
* Simplification and standardization of design decisions
* Improving reactor equipment reliability
* Attaining longer plant lifetime
* Improving stability to personnel errors, internal and external impacts

core power density was reduced by 3-4 times down to 27 kW/l and the fuel specific heat rating down to 10 kW/kg. Such combination of parameters assures that the energy accumulated in the core is low which, in turn, assures that operating transients are slow. Besides, it also reduces considerably in-core accumulated radioactivity and delays the release of radioactive products from the relatively "cold" fuel ($T < 500°C$).

Natural convection of primary coolant in all operational modes is the essential feature of the AST-500 reactor. It ensures the independence of both the primary system due to complete elimination of active means for forced circulation of coolant and power consumers. It enables to exclude the complicate transient modes which are characteristic for reactors having forced coolant circulation and which are caused by start-ups and stops of reactor coolant pumps, which results in unfavourable thermal impacts upon the reactor structures.

To solve the complex safety problems, an integral design was adopted for the NHR, with arrangement of the primary heat exchangers and the pressurizer immediately in the reactor pressure vessel [6-9]. This design eliminates large-diameter primary pipelines. Besides, all auxiliary reactor-related pipes are arranged in the upper part of the reactor vessel. The height of water above the core up to the outlet nozzles is approximately 8 m, the water volume for evaporation is approximately 130 m^3.

The enlarged water gap between the core and the reactor pressure vessel decreases significantly the fast neutron fluence on the vessel down to approximately $10^{16} n/cm^2$, thus eliminating the problem of radiation embrittlement of the vessel's steel.

The large specific water inventory per unit power in an integral reactor provides for accumulation of large amounts of heat, and determines a considerable inertia in case of accidental events associated with loss of heat removal from the reactor. Due to the reactor heat accumulation capability, the pressure limit would be achieved only after 2 hours into an accident. The availability of such a time margin allows to eliminate any automatic actions for the actuation of the AST-500 heat removal systems.

The essentially new engineering decision for the AST-500 was to use a guard vessel that houses the entire reactor unit. Its main function is to keep the core covered with water and thus to exclude fuel element melting in reactor vessel loss-of-integrity accidents. The guard vessel (GV) eliminates the need for an emergency make-up system. At the same time the GV serves as a radioactivity confinement system which, in contrast to conventional containments, localizes radioactive products in a small volume in the immediate vicinity of the reactor.

The passive heat removal principle was used and validated in-depth for reactor overpressure protection. The capability to reduce the coolant temperature by the heat removal means available, and in a such way to decrease effectively the reactor pressure due to the close thermal coupling between the primary and secondary circuits through the built-in heat exchangers, provides the reactor with a reliable overpressure protection without the use of safety valves. So, the necessity of blowing off primary coolant in transients was eliminated.

The emergency residual heat removal system operates under natural convection of water in all circuits up to the ultimate heat sink, with no need of power for operation over several days. One of the ERHR channels is capable to provide a pressure reduction in reactor in accidents. As reliability is concerned, this system is not inferior to the reactor emergency protection system.

Pilot-operated relief valves installed on the secondary system pressurizers are actuated both to a primary pressure signal and directly by the action of the secondary medium pressure. The number of valves and the flow rate characteristics were chosen to ensure a primary system overpressure protection by removal of heat through the PORVs. So, a back-up heat removal channel is provided.

A twin isolation valve system is provided to limit discharges of primary coolant in case of any pipeline rupture, or loss of integrity of primary system components located beyond the GV boundary. The valve type was chosen to provide their closure without external energy supply. They close due to the force of a precompressed spring following a compressed air blow-off.

A remotely-operated boron injection system is provided for bringing the reactor to a subcritical state under cold, clean conditions, should a large number of control rods fail to insert into the core (back-up safety system).

For assured protection of the heat consumers, a three-circuit flow scheme is used for heat transport from the reactor. A pressure barrier is provided between the secondary and the grid circuits, thereby preventing radioactive products to ingress into the heating grid at primary/secondary HXs loss-of-integrity accidents.

3. NHR-RELATED EXPERIMENTAL BASE

All components and systems of the AST-500 reactor plant were tested comprehensively using appropriate rigs and test facilities created by the reactor plant designer (OKBM) and his sub-contractors. Their characteristics and operating processes were studied in depth on various models, analogues and prototypes [11, 12]. Many institutions were involved in these activities.

To check on a system level, and to validate the thermal-hydraulic characteristics of the NHR coolant natural convection circuit, as well as to investigate the plant emergency modes, special test facilities were created, including multi-channel thermophysical rigs, at OKBM, large-scale circulation loops at CKTII and at the Kurchatov Institute of Atomic Energy (IAE). The tests were performed in the entire range of the plant parameters' variation. The mechanisms of reactor coolant natural convection were studied depending on the power of the core simulators, pressure level, etc.

The reactor hydrodynamics were investigated at the Central Air-Hydrodynamic Institute, using a 1:4 mock-up of the AST-500 and air for the simulation of coolant flow.

Experimental studies of boiling crisis have been carried out independently on various thermal-physical rigs at OKBM, IAE and CKTII. The tests have been performed with six experimental assemblies representing the real fuel rods with bundles of electrically heated simulators.

The analysis of the thermal-technical reliability of the AST-500 reactor core which was carried out according to a special technique proved the availability of large design margins for fuel assembly heat power.

Investigations of two-phase flow and pressurizer operation, as well as of coolant outflow after primary circuit loss-of-integrity accidents, were carried out by the Power Research Institute.

The basic experimental work for the NHR safety validation concerning primary circuit loss-of-integrity accidents was performed at CKTII using a reactor model allowing to study thermal and hydraulic processes in the primary circuit at accidents. These experiments gave data on the variation of all essential thermal-hydraulic parameters characterizing the emergency process development during AST-500 primary circuit LOCAs. Comparison of the obtained experimental results with analytical data has shown that the mathematical models and the calculation codes used ensure a sufficient accuracy for design purposes.

A study of the reactor pressure vessel rupture mechanism has been performed by the Machinery Research Institute, CNIIMASH, and by OKBM.

The development and optimization of the water chemistry technology for the AST-500 primary circuit also required a large number of experiments. So, radiation-chemical processes have been investigated at the in-pile experimental loop of the MR test reactor at the Kurchatov Institute. The tests were carried out under both non-boiling and boiling conditions. They allowed to validate the need of primary circuit make-up with hydrogen to suppress coolant radiolysis and oxygen concentration build up. Physical simulation and investigation of gas-exchange processes in the primary circuit with a built-in steam-gas pressurizer have been carried out on special test rigs at OKBM. Based on the experiments, the technique and calculation codes were developed for the analysis of the gas distribution in a pressurizer and of the steady-state gas concentration in primary coolant.

Experimental studies of the neutron-physics characteristics of the AST-500 core were performed initially on small critical facilities using physical models which were composed of full-scale fuel assemblies, and then with a full-scale core model at the core manufacturer's site. On the models "cold" and "hot" critical experiments were performed for verification of the calculation codes which allowed to carry out the necessary check calculations. The analysis has shown that the complex of neutron-physics computer codes allows to define with high engineering accuracy the basic physical characteristics of the AST-500 core models.

According to the program of the AST-500 core acceptance tests, experiments were carried out for the determination of the basic neutron-physics characteristics of the initial core at the full-scale critical facility under conditions practically completely corresponding to the real ones. The experimental data analysis has shown that the basic neutron-physics characteristics of the AST-500 core determining the nuclear safety correspond to the design specifications and satisfy the regulatory requirements for nuclear safety.

In the phase of reactor plant basic design development, investigations of the reactor unit seismic stability have been performed using a 1:4 model. The tests confirmed the reactor seismic stability up to a magnitude 8 earthquake (to MSK-64 scale).

All reactor-related equipment items have been tested comprehensively during their acceptance tests before their delivery to the construction site.

Resulting from those extensive R&D activities the safety limits and margins have been comprehensively investigated, particularly for: critical heat flux, multichannel hydraulic instability, single channel flow stability, xenon instability etc. The design margins were proven adequate. A considerable empirical data base has been gained for AST-500 design validation and optimization.

4. CONCLUSION

Resulting from the NHR activities in the RF, the basic design of the AST-500 has been developed, comprehensively tested and mastered in manufacturing. Appropriate construction and installation technologies were proven on an industrial scale with the two pilot NDHPs in Gorky and Voronezh.

The IAEA Pre-OSART mission to the Gorky NDHP has confirmed the sound basis of the plant design and its enhanced safety. That conclusion covered the AST-500 design basis, specific design solutions, their experimental validation, as well as the reactor-related equipment manufacture, erection and construction quality.

Significant upgrading potential of the reference design approaches was revealed in the course of subsequent optimization and R&D activities, which allowed to recently develop the advanced NHR AST-500M with better economics and safety. On this basis a whole series of safe and reliable nuclear reactors of a wide unit power range can be developed for application in various district heating and co-generation systems.

It is noteworthy that the NHR development has played a significant role in forming an advanced safety concept, and design approaches have been implemented for the development of a new generation of enhanced safety nuclear power reactors.

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