Nuclear district heating plants AST-500. Present status and prospects for future in Russia

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Abstract

Development activity in the field of nuclear reactors for district heating purposes was initiated in the ex-USSR in the middle 1970s. It was stimulated by a growing deficiency of fossil fuels needed to provide for the heating of large cities in the European regions of the country. Construction of two pilot twin-reactor nuclear district stations was started in the early 1980s in the cities of Gorky and Voronezh. Nuclear heating reactor AST-500, developed by OKB Mechanical Engineering, has a number of unique engineering features, e.g. integral lay-out of primary system's components, natural coolant circulation, steam-gas self-pressurizing system, additional safety vessel around the reactor unit, etc. All these specific features, which were adopted in order to assure enhanced safety of the reactor plant, required fulfilment of an extensive R&D program. Description of the reference AST-500 design features is given in this paper, as well as respective R&D activities carried out for the design solutions and safety validation. Currently, activity in the field is directed to AST-500 design upgrading and development and also to a whole series of heating-only and co-generation reactor plants ranging from 30 to 600 MW(th) for the new generation nuclear stations. © 1997 Elsevier Science S.A.

1. Introduction

It is now 20 years since practical activity was initiated in the ex-USSR aimed at creating nuclear plants of the AST-500 type intended to provide a low-potential heat supply for large dwelling complexes (Skvortsov and Sidorenko, 1980). The need for such an energy source was validated by the following reasons and preconditions.

The fraction of fossil fuel being spent in the Russian Federation for generation of low-potential heat for both industrial and district heating purposes amounts to 30–40% of total consumption, where the most qualified kinds of fuel such as natural gas and oil comprise the major fraction. Therefore, the use of nuclear fuel for heating and hot water supply might become a significant contributor to improvement of the national fuel consumption balance, which was characterised by considerable remoteness of the most-energy-consuming regions from the main fuel-production ones, situated mainly in the East of Russia (Concept of nuclear power development in the Russian Federation, 1992).

Besides, it was realised that substitution of fossil fuel for nuclear would give an important social-economic effect through attenuation of the adverse impact of the traditional power industry on the environment, particularly of large cities.

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By the early 1980s favourable conditions were established in the ex-USSR for introduction of nuclear reactors into the district heating grids, promoted by the following factors:

1. Positive experience had been gained by the national power industry in centralised district heating using large energy sources and heat-distributing grids;
2. There was a stable tendency to the increase in heating loads concentration and unit power of centralised district heating sources;
3. Deficiency of fossil fuel and large expenditures for its long-distance transportation;
4. Experience had been gained in construction and operation of powerful nuclear power plants with steam extraction for heating of nearby satellite towns.

According to the decision of the USSR Government in 1977 a number of leading enterprises and institutions were involved in development of pilot nuclear heating stations: OKB Mechanical Engineering was the chief designer of the reactor plant, the Kurchatov Institute was the scientific supervisor, the VNIPET and NIAEP Institutes were the general designers of pilot stations (architect-engineers).

The respective heat-only reactor plant of 500 MW(th) unit power was named AST-500. Design of the AST-500 reactor plant was developed with both its location in the close vicinity of large cities and the necessity of assuring a qualitatively higher level of safety compared with existing NPPs taken into account. Therefore, during the design development the highest priority was given to technical solutions which would provide net safety of the reactor plant by virtue of appropriate design solutions, application of passive safety systems and maximum reliance on natural processes for safety functions fulfilment. Such an approach presented the possibility of creating the reactor plant with unique self-protection properties.

2. AST-500 basic physical-engineering features

The AST-500 reactor plant overall view, principal flow diagram, engineering solutions and main parameters are given in Figs. 1–3 and Table 1 (Mitenkov et al., 1985; Egorov et al., 1980).

The AST-500 NHR design development was preceded by adoption of the NHR-specific requirements for safety, that was adopted as supplementary to the basic national Code for NPP safety. These requirements envisage the necessity of excluding the fuel meltdown at loss of integrity of any reactor-related pressure vessel, taking into account external impacts like aircrash and shock wave, as well as more stringent requirements for radiological safety including collective dose limit specification (Nikiporetz et al., 1983; Mitenkov et al., 1988).

The PWR-type nuclear reactor with its intrinsic positive properties, particularly as concerns its nuclear safety, such as neutron power self-control and self-limitation, was accepted for the AST-500 design development. The chain reaction self-control capability in virtue of negative temperature, power and void reactivity feedbacks was enhanced additionally in AST-500 by refusal from soluble boron control and the use of burnable poison in the core.

The peculiarity of NHR as a source of low-grade energy gave the opportunity to accept much lower parameters for the primary system \((T = 200^\circ C, P = 2.0 \text{ MPa})\) compared with traditional light water reactors. To conform to specific heating purposes the required unit power of NHR might be considerably lower than that generally accepted for contemporary nuclear power, i.e. 500–600 MW, depending on specific heat load characteristics of a heating grid connected to the given NHR. Along with the coolant parameter reduction, the core power density was reduced by 3-4 times down to 27 kW l\(^{-1}\) and the fuel specific heat rating - down to 10 kW kg\(^{-1}\). Such a combination of parameters ensures low energy accumulation in the core that predetermines quiet development of operating transients, and also considerably reduces in-core accumulated radioactivity. It also delays radioactive products release from relatively ‘cold’ fuel \((T < 500^\circ C)\) into a fuel-cladding gap.

Natural convection of primary coolant in all operating modes is one of the key features of the AST-500 reactor. It ensures the independence of the primary system due to complete elimination of the active means for forced circulation of coolant.
and related power consumers, it enables the exclusion of the complicated transient modes, which are characteristic of reactors with forced coolant circulation, where reactor coolant pump start-ups and stops result in unfavourable thermal impacts upon the reactor structures.

To solve the reactor plant safety problems on a system level, an integral design was adopted for
the NHR with arrangement of the primary/secondary heat exchangers and pressurizer immediately in the reactor pressure vessel (Gureeva et al., 1989; Mitenkov and Samoilov, 1992). This structure design eliminates large-diameter primary piping, in addition all auxiliary reactor-related pipes are arranged in the upper part of the reactor vessel. The height of water above the core to outlet nozzles is ca. 8 m, so water volume for evaporation is ca. 130 m$^3$.

The enlarged water gap between the core and the reactor pressure vessel decreases its exposure significantly by fast neutrons down to ca. $10^{16}$ n cm$^{-2}$, and removes the problem of radiation embrittlement of the vessel's steel.

A large specific (per one MW) water inventory in the integral reactor provides for accumulation of large amounts of heat and determined considerable inertia of accident processes associated with loss of heat removal from the reactor. Due to the reactor plant heat accumulation capability only the pressure limit would be achieved in 2 h from the accident initiation. The availability of such a time margin gives the possibility of excluding any automatic actions for the AST-500 heat removal systems actuation.

Essentially new engineering features of AST-500 is application of a guard vessel houses the entire reactor unit. Its main function is to keep the core covered with water and exclude fuel meltdown during the reactor vessel loss-of-integrity accidents. The guard vessel (GV) eliminates the need of an emergency make-up system. At the same time, GV serves as a radioactivity confinement system, which is contrast to a conventional containment localises radioactive products in relatively small volume in the immediate vicinity of the reactor.

The emergency residual heat removal system operates at natural convection of water in all circuits up to an ultimate heat sink, with no need for external power for operating during several days. One of the ERHR channels is capable of providing pressure reduction in the reactor in accidents. As concerns the reliability, this system is not inferior to the reactor emergency protection system.

The capability of reducing the coolant temperature by heat removal means available and in a such way to decrease effectively the reactor pressure due to close thermal coupling between the primary and secondary circuits through the built-
in heat exchanger provides the reactor overpressure protection by means of the heat-removal principle. So, the need for installation of safety valves in the primary system and primary coolant blow off in transients was excluded.

The dual isolation valve system is provided to limit discharges of radioactive primary coolant in a case of any pipeline rupture or loss on integrity of the primary system’s equipment located beyond the GV boundary. The valves type was selected to provide their closure without external energy supply. They are closed under the precompressed spring action following a compressed air blow-off.

To protect heat consumers, a three-circuit flow scheme is used for heat transport from the reactor with a pressure barrier between circuits that prevent radioactive products ingress into the heating grid in the primary/secondary HXs loss-of-integrity accidents.

3. AST-500 design solutions validation

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 leading institutions, their characteristics and operating processes were in-depth studied on various models, analogues and prototypes (Falikov et al., 1994). In order to study on a system level and validate thermal-hydraulics of the NHR-specific coolant natural convection circuit, as well as to investigate various accident transients, special test facilities were created including a multi-channel thermophysical rig in OKBM, large-scale circulation loops in CKTI and the Kurchatov Institute of Atomic Energy (IAE). The investigations were performed in the entire range of the plant operational parameters variation. The reactor coolant natural convection mechanisms were studied de-
depending on the power of the core simulators, pressure level, etc.

The reactor hydrodynamics was investigated in Central Air-Hydrodynamic Institute using 1:4 mock-up of AST-500 and air for simulation of coolant flow.

Experimental study of boiling crisis has been carried out independently at various thermal-physical rigs in OKMB, IAE and CKTI. The tests have been performed using six experimental assemblies which represented bundles of electricity heated full-size simulators of fuel rods.

Analysis of thermal-technical reliability of the AST-500 reactor core carried out by a special technique has shown the availability of large design margins for fuel assemblies thermal power.

Investigation into two-phase flows and the pressurizer operation, as well as coolant outflowing in the primary circuit loss-of-integrity accidents were carried out by the Power Research Institute of the Russian Academy of science.

Basic experimental work for the NHR safety validation in the primary circuit loss-of-integrity accidents was performed in CKTI using the reactor model that allowed the study of thermal and hydraulic processes in the primary circuit under conditions of their mutual influence. These experiments gave data on variation of all essential thermal-hydraulic parameters, characterising progression of the primary circuit loss-of-integrity accidents.

Based on the produced experimental data the computer codes for calculation analysis of that kind accidents in the AST-500 were validated.

Comparison of the experimental data with calculated ones has shown that the mathematical models and the calculation codes used provide the accuracy sufficient for the design purposes.

Study of the reactor pressure vessel fracture mechanics has been performed by the Machinery Research Institute, CNIIMASH and OKBM.

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

Experimental studies of the neutron-physical characteristics of the AST-500 core were performed on critical facilities using physical models which were composed of full-scale fuel assemblies and on the full-scale core model at the core manufacturer. ‘Cold’ and ‘hot’ critical experiments were performed on the models for verification of the calculation codes that allowed the necessary check calculations to be carried out. The analysis has shown that the complex of neutron-physical computer codes allows the definition with a high engineering accuracy of the basic neutronic characteristics of the AST-500 core models.

<table>
<thead>
<tr>
<th>Reactor type</th>
<th>Integral PWR with natural coolant circulation</th>
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<tbody>
<tr>
<td>Reactor thermal power (MW)</td>
<td>500</td>
</tr>
<tr>
<td>Installed heat output capacity (Gcal h⁻¹)</td>
<td>430</td>
</tr>
<tr>
<td>Primary coolant temperature, °C</td>
<td>134/208</td>
</tr>
<tr>
<td>Pressure in steam-gas pressurizer (MPa)</td>
<td>1.96</td>
</tr>
<tr>
<td>Secondary circuit temperature °C</td>
<td>88/158</td>
</tr>
<tr>
<td>Pressure in secondary circuit pressurizer (MPa)</td>
<td>1.2</td>
</tr>
<tr>
<td>Grid circuit water temperature °C</td>
<td>64/144</td>
</tr>
<tr>
<td>Refuelling interval (years)</td>
<td>2</td>
</tr>
<tr>
<td>Main components life time (years)</td>
<td>Up to 60</td>
</tr>
<tr>
<td>Equipment seismic stability (points to MSK-64 scale)</td>
<td>8</td>
</tr>
</tbody>
</table>
According to the program of the AST-500 core acceptance test the experiments were carried out for determination of the basic neutron-physical characteristics of the initial core on the full-scale critical facility. The experimental data analysis has shown the basic neutron-physical characteristics of the AST-500 core that determine nuclear safety and correspond to the design ones and satisfy the regulatory requirements for nuclear safety.

In the phase of the reactor plant basic design development the investigation into the reactor unit seismic stability was performed using the 1:4 model. The tests confirmed the reactor seismic stability up to magnitude 8 earthquake intensity (to the MSK 64 scale).

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

From 1987 to 1994 OKBM proceeded with activity aimed at perfection of the AST-500 design for commercial (serial) units of nuclear district heating stations and improvement of the reactor plant economics. In that period basic design of advanced reactor plants AST-500M and AST-600 have been performed in which the following innovative features were implemented:

1. CRDM capable of operating in a gas environment;
2. back-up liquid absorbent injection system with passive principle of operation;
3. a set of measures for enhancement of reliability of emergency heat removal from the core, e.g.: strengthening of secondary circuit's equipment with respective increase in a safety valve actuation setpoint; introduction of a passive ERHR channel on the reactor which is capable of performing additionally such safety functions as the reactor de-pressurisation and self-shutdown;
4. Self-actuated device for actuation of safety systems, which enhance control systems resistance against common-cause failures (e.g. fires, etc.) and personnel errors.

Other systems and equipment have been improved as well, i.e. the reactor pressure vessel (a large diameter lower joint eliminated), primary-to-secondary HX (heat exchange surface increased that allowed the increase of the reactor power up to 600 MW), primary coolant purification system (simplified), the reactor core (optimised fuel cycle).

In connection with public requirements, the ex-USSR government addressed itself in 1988 to the IAEA request to perform an independent international safety review for the lead nuclear district heating station in Gorky (GNDHS). The review was carried out according to the program of OSART missions with consideration of both the station design and construction activity organisation, quality of equipment, preparation of personnel and documentation for operation. The main findings of the review mission were positive.

Recommendations were given by an IAEA review team aimed at further enhancement of the station safety and the quality of remaining construction work, and on preparation for operation. Many of the recommendations supported and supplemented the work and investigations that were started before by the station designers.

4. Status, tasks and prospects for nuclear heating

Analysis performed to assess both the current and prospective levels of municipal heating loads and their concentrations showed that in the European regions of Russia there are several tens of large heat-consuming centres with heating loads more than 1000 Gcal h\(^{-1}\). For majority of the regions anticipated increment of heating load for the nearest future allows application of twin-unit district heating stations with total heat output of 860 Gcal h\(^{-1}\). Feasibility studies showed the competitiveness of such plants with conventional fossil-fuelled heating plants.

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

Following this feasibility studies were performed for development of NDHSs in Arkhangelsk, Brjansk and Khabarovsk.
Realisation of the state program for NDHSs deployment in the ex-USSR was disturbed at the end of the 1980s by the social-political system reconstruction in the country and subsequent economic difficulties. In 1990 decisions were adopted by the regional administrations on the termination of NDHSs construction in Gorky and Voronezh. Until that time two sets of AST-500 reactor plant components were delivered to the Gorky site and one set to the Voronezh one. In that time building-mounting work on a first unit of Gorky NDHS was 83% completed (in cost terms) and 31% on the Voronezh site.

However, already in 1994 under the pressure of acute problems in a district heating of Voronezh caused by the NDHS non-starting the regional administration initiated the station environment impact review by a public commission composed of qualified local scientists and experts. After consideration of the design materials, including a safety validation report a conclusion was made by the commission about the plant safety for both local population and the environment. Recommendations were also formulated for the administration to take a decision on the station construction proceeding and start-up.

In 1995 the RF Ministry for Nature performed a state environmental review for the NDHS. The State Review Committee confirmed the previous findings about the possibility of continuing station construction. Particularly, the committee's conclusion says: '... the design materials submitted justify the environmental safety of the station are sufficiently convincing as a whole, especially as concerns construction-technological features of the new reactor plant AST-500 possessing enhanced safety. Under the normal operational conditions the station will be much less hazardous for the environment and human health than the operations of existing coal, oil or gas-fired heating stations in Voronezh. Therefore, the station construction is completely substantiated.'

On the basis of the positive conclusion of the indicated review missions the regional and city administrations adopted the decision to proceed with station construction activity, with a view to putting it into operation.

So, work on the station site was resumed in 1996. The programs have been developed on the station design updating (conforming with requirements of norms and rules being valid in Russian nuclear power) and reactor plant equipment inspections. Inspection of the equipment available at the site has just been started aiming at its preparation for erection. Simultaneously, the design documentation and pertinent licensing materials are being prepared for submission to the State regulatory body in order to receive the permission for the station construction proceeding.

The next phase in the development of nuclear heating technology is associated with Siberia. In connection with forthcoming decommissioning plutonium-production reactors on a site of the Siberian Chemical Combine in Seversk (Tomsk region) according to the USA-RF strategic weapons stocks reduction agreement, the Ministry for Atomic Energy of the RF envisages deployment on the site of a twin-unit (AST-500) NDHS capable of substituting the indicated reactors being operated as sources of heat for the existing district heating grid of Tomsk.

Lately OKBM, NIAEP and VNIPLET (both are architect-engineering institutions) performed a feasibility study to validate the station construction project and prepared necessary materials for receipt of the safety authority permission for the station siting and building. In order to accelerate the station construction, it is proposed to utilise the components available at the Gorky NDHS. For this aim the work on validation of the first reactor unit dismantling feasibility was carried out, necessary structures and technological tools were designed, and labour consumption for the equipment dismantling, inspection and de-bugging activities was evaluated. Storage conditions for the reactor plant components (delivered 1984–1989) allow its satisfactory state to be expected.

Presently, main task for designers is a receipt of authorised licences from GAN for proceeding construction activity at the Voronezh NDHS and start of construction work for the NDHS at Seversk. It is planned to begin the work on the both sites in 1997-1998.
5. Conclusion

A nuclear district heating plant is the most preferable as a source of heat for district heating purposes for both environmental and social points of view. A great potential of knowledge and technological experience of Russian scientists and engineers has been accumulated in the AST-500, particularly that gained in the field of marine nuclear reactors where OKBM possesses the unique well-proven technology.

A methodology of system level investigations into safety adopted presently in the common practice conformably to design analysis of different complex industrial systems (e.g. power engineering, chemistry etc.) was applied to the AST-500 in a full extent. A deterministic approach (i.e. 'all is possible') to postulated accidents analysis was used, as well as probabilistic assessments of safety with analysis of hundreds accident sequences.

All these facts give grounds to affirm that the AST-500 has a guaranteed safety. All feasible measures have been implemented to minimise accident frequency. More important that radiological consequences of any accident including a most severe one is limited to the background level of radiation. The consequences and respective risks are much less than that associated with other kinds of industrial activity (chemistry, transport, power). So, in the AST-500 those tasks have been solved that were raised before the world nuclear power by Chernobyl accident.

References