Advanced WWER-1000 reactor plant for nuclear power plant

Yu.G.Dragunov, S.B.Ryzhov, A.M.Rogov
FSUE OKB “GIDROPRESS”, Podolsk, Russia

Abstract. A concept and main trends in the development of WWER-1000 reactor plant are addressed. Measures and design solutions aimed at safety enhancement and improvement of technical-and-economic figures of reactor plant are specified. Evolutionary changes are highlighted for specific design units of the equipment, ways for the units at issue to be perfected are indicated. Configuration of advanced WWER-1000 RP design which meets up-to-date Russian and international requirements is covered.

1. Introduction

Federal state unitary enterprise - experimental and design organization “GIDROPRESS”, being General Designer of the advanced WWER-1000 RP, for more than half a century has been developing the designs of the equipment and systems intended for nuclear power engineering and industry. FSUE OKB “GIDROPRESS” is the author of a large number of the designs of various installations and equipment. Our designs have been implemented at the nuclear power plants not only in Russia but abroad as well: in Finland, Germany, Bulgaria, Hungary, Slovakia, Czech Republic, Ukraine, Armenia, Kazakhstan. Under the designs developed at the FSUE OKB “Gidropress” 85 reactor plants of various types have been constructed including 40 abroad. FSUE OKB “Gidropress” holds the TÜV NORD certificate; the latter certifies that quality assurance system accepted at the enterprise corresponds to the international standard ISO 9001-2000.

2. Development of reactor plant design

In developing advanced WWER-1000 RP, design reactor plant for AES-91 was accepted as a basis. Development of the concept started in 1977 in co-operation between the specialists of Saint-Petersburg Institute “Atomenergoproject” and Finnish company Imatran Voima International Ltd.(presently named as Fortum Engineering Ltd). The prototype of the design is that of the commercial reactor plant with WWER-1000 (V - 320). Design is based on evolutionary path of safety enhancement and at the same time it differs considerably from the prototype of standard NPP with RP V-320. In particular, there are some differences in lay-out and configuration of the equipment, in approaches to ensuring nuclear and radiation safety, main equipment has been modified.

Since 1991 the work on improving RP AES-91 concept has made an impressive headway. A series of meaningful design improvements has been introduced into the designs of equipment and systems (reactor, steam generator, RCP set, reactor scram system, etc.) which enabled to achieve the safety level meeting the up-to-date requirements.

The design is based on safety criteria covered in normative-technical documentation valid in Russia as well as in IAEA recommendations. Principle of defence-in-depth is the basis of safety assurance accepted in the design. It means application of the system of barriers on the way of propagation of ionizing radiation and radioactive substances into the environment and the system of technical and organizational measures for protecting the barriers and maintaining their effectiveness, and directly for protecting population.

Measures for protecting the barriers involve:
— use of up-to-date norms and standards;
— consideration of operational feedback of standard reactor plant;
— quality assurance at all the stages of RP life cycle;
— monitoring for the state of barriers during operation by up-to-date means and methods of inspection;
— maximum use of properties of reactor inherent safety peculiar to reactors of WWER type;
— control of technological process in such a way so as to provide non-exceeding of design safety limits under all design conditions including accident conditions.

The design uses deterministic approach assuming that safety is ensured in normal operation and at any initiating event considered in the design with regard for single failure principle. Besides, the design provides measures on management of beyond design basis accidents and mitigation of their consequences. The designing strategy is directed to prevent origination and, if it is not managed, to provide possibility of management of severe accidents.

The design characteristics of the plant together with management measures of severe accidents provide the safety indices required by regulatory documentation. In particular:
— frequency of severe core damage less than $1 \times 10^{-5}$ per reactor/year;
— frequency of ultimate radioactive release less than $1 \times 10^{-7}$ per reactor/year.

Safety enhancement and improvement of technical-and-economic figures in the advanced RP design is ensured with the factors as follows:
— improvement of nuclearphysics properties of the core and structure of the critical reactor units;
— possibility of transition to uranium-gadolinium fuel;
— provision of coolant temperature negative coefficients of reactivity;
— introduction of new systems of monitoring and diagnostics of the equipment;
— improvement of RP special systems (CPS, NFME, ICIS);
— realization of the “leak before break” concept for the RP pipelines;
— use of removable thermal insulation of modular type;
— provision of maximum average burnup of FA of 55 MWday/kgU;
— decrease in duration of repair downtime and increase in load factor and so on.

3. Basic characteristics of the reactor plant

Reactor coolant circuit involves reactor, four circulation loops each of them consists of steam generator, reactor coolant pump set as well as main coolant pipeline connecting loop equipment with the reactor. Reactor plant schematic lay-out is shown in Fig. 1.
Basic characteristics of the advanced reactor plant as compared with those of standard RP V - 320 are addressed in Table 1.

Table 1. Basic characteristics and parameters of reactor plants

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Advanced RP</th>
<th>V-320</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power nominal (MW)</td>
<td>3000</td>
<td>3000</td>
</tr>
<tr>
<td>Service life of RP basic equipment (years)</td>
<td>60</td>
<td>30</td>
</tr>
<tr>
<td>Steam capacity under nominal conditions (T/h)</td>
<td>1470×4</td>
<td>1470×4</td>
</tr>
<tr>
<td>Pressure of steam generated under nominal load at SG steam header outlet (MPa)</td>
<td>6.27</td>
<td>6.27</td>
</tr>
<tr>
<td>Temperature of steam generated under nominal load (°C)</td>
<td>278.5</td>
<td>278.5</td>
</tr>
<tr>
<td>Time of fuel residence (fuel cycle) in reactor core (year)</td>
<td>3 - 4*</td>
<td>3</td>
</tr>
<tr>
<td>Average fuel burn-up fraction (under equilibrium fuel cycle) (MW day/kg uranium)</td>
<td>50 – 55*</td>
<td>40.2</td>
</tr>
<tr>
<td>Time of operation at nominal power during year (effective), h</td>
<td>8400*</td>
<td>7000</td>
</tr>
<tr>
<td>Consideration of LBB concept (leak before break)</td>
<td>Considered</td>
<td>Not considered</td>
</tr>
<tr>
<td>Consideration of ATWS</td>
<td>Considered</td>
<td>Not considered</td>
</tr>
<tr>
<td>Coolant temperature at the core outlet, °C</td>
<td>321*</td>
<td>320</td>
</tr>
<tr>
<td>Coolant temperature at the core inlet, °C</td>
<td>291*</td>
<td>289.7</td>
</tr>
<tr>
<td>Coolant flow rate through reactor, m³/h</td>
<td>86000</td>
<td>84800</td>
</tr>
<tr>
<td>Inner diameter of reactor vessel (cylindrical part), mm</td>
<td>4195</td>
<td>4150</td>
</tr>
<tr>
<td>Wall thickness in the area of the core, mm</td>
<td>195</td>
<td>192.5</td>
</tr>
<tr>
<td>Length, mm</td>
<td>11185</td>
<td>10885</td>
</tr>
<tr>
<td>Number of CPS control rods, pcs.</td>
<td>85-121*</td>
<td>61</td>
</tr>
<tr>
<td>Steam generator</td>
<td>PGV-1000MK</td>
<td>PGV-1000M</td>
</tr>
<tr>
<td>Inside diameter of steam generator vessel, m</td>
<td>4.2</td>
<td>4.0</td>
</tr>
<tr>
<td>Reactor coolant pump set</td>
<td>GTCNA-1391</td>
<td>GCN-195M</td>
</tr>
</tbody>
</table>

* Parameter values are specified in the course of detailed project report development

4. Basic solutions on the reactor plant equipment

4.1. Reactor

Reactor is a vertical pressurized vessel (the vessel with a top head) having internals, core, control members and ICIS sensors inside. Reactor is intended for conversion of energy of nuclear fuel fission into thermal and its transfer to the primary coolant of two-circuit reactor plant of the power unit of nuclear power plant. Reactor type – pressurized water-cooled water moderated, of vessel type, with thermal neutrons. The structure of the nuclear reactor is shown in Fig. 2.

Reactor design of the advanced reactor plant as compared with standard RP V-320 takes into account improvements, applied for new projects of WWER-1000 reactors as well as it applies new solutions aimed at extension of design service life of the reactor vessel to 60 years. The following is referred to the first improvements:

— completely new programme of surveillance-specimens (arrangement of irradiated SS directly on RV wall);
— limitation of nickel content in welds;
— limitation of harmful impurities in base metal and welds;
— decrease in $T_{\text{no}}$ of nozzle zone shells down to minus 35°C.

Modernization of the reactor vessel in respect to new designs of WWER-1000 RP is performed retaining design solutions of the upper part of the vessel i.e. of the flange and nozzle zone as well as keeping geometry of the bottom with the aim of using developed manufacturing process (i.e. reference character is practically maintained in respect to WWER-1000 operating units). At the same time,
internal diameter of vessel cylindrical part is lengthened to 4195 mm that makes possible to reduce neutron flux to the reactor vessel to the extent for the total fluence value not to be exceeded in terms of provision of brittle failure resistance during operation period of 60 years.

4.2. Core

Core with uranium-gadolinium fuel based on the fuel assembly of the second generation FA-2 which is characterized by enhanced stability to shape variation under high burnups is assumed to be used in the project of advanced reactor plant. This FA (fig.3) has a structure with rigid welded frame wherein spacing grids are welded to zirconium guiding channels.

The basic conceptual solutions on FA structure of the second generation - FA - 2:
— spacing grids rigid to bending with the lengthened height to 30 mm provide geometrical stability of the frame and FA – 2 itself;
— special design of spacing grid cells provides decrease in slipping forces of fuel rods and reduces mechanical loads to the frame;
— strengthened upper and lower end pieces (head and tail-piece) improve stability of FA - 2.
FA - 2 is considered as a structure which according to its technical characteristics provides possibility for:

- achievement of high burn-ups - 55 MW days / kg of uranium and more;
- operation in 4 - 5 year fuel cycles;
- operation during extended fuel cycles.

It provides a certain “flexibility” for NPP in choosing fuel cycles.

Types of fuel cycles being of interest to Customers nowadays are given in Table 2.
Table 2. Perspective fuel cycles

<table>
<thead>
<tr>
<th>Fuel cycle</th>
<th>4×1 year</th>
<th>3×350 days</th>
<th>2×1,5 years</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of FA to be loaded (pcs.)</td>
<td>42</td>
<td>54</td>
<td>82</td>
</tr>
<tr>
<td>Average enrichment in of FA to be loaded in $^{235}U$ (% weight)</td>
<td>4,31</td>
<td>4,2</td>
<td>4,56</td>
</tr>
<tr>
<td>Duration of operation of equilibrium fuel cycle (eff. day)</td>
<td>296,6</td>
<td>330</td>
<td>526</td>
</tr>
<tr>
<td>Average burn-up of FA to be unloaded (MW day/kg of uranium)</td>
<td>47,2 (52,1)*</td>
<td>46,3 (49,7)*</td>
<td>44,6 (47,9)*</td>
</tr>
<tr>
<td>Specific consumption of natural uranium, (g/MW day)</td>
<td>198</td>
<td>-</td>
<td>224</td>
</tr>
</tbody>
</table>

* Maximum values of fuel burn-up are given in brackets

### 4.3 CPS drive ShEM-3

Mounting of 121 CPS drives ShEM-3 is provided for on the reactor top head. Pitch electromagnetic drive with position indicator is used for control rods motion. Control rod assemblies contain 18 absorbing rods. Absorbing rods are made of stainless steel tubes filled with absorbing material. Boron carbide is used for the upper part and dysprosium titanate – for the lower part. Control rod drives are placed on the reactor top head. Drawing of drive ShEM-3 is shown in Fig. 4.
The following has been done as compared with the drive of standard power unit:

— power electromagnetic system is optimized and dynamic drive characteristic is improved under motion conditions;
— drive service life is extended to 30 years whereas the prototype mechanical part has a service life of 20 years, service life of electromagnets – 10 years and that of position indicator – 5 years;
— pulling force is increased (– twice as much);
— PPI is used with indication of pitch position every 20 mm of extension shaft motion (LPI in ShEM drive provides indication of position of extension shaft with 350 mm pitch).

4.4 Steam generator

Steam generator (Fig. 5) represents a heat exchanging device of horizontal type with submerged heat exchanging surface. The problem of ensuring a reliable operation of PGV-1000 tubing is pertained, primarily, to the state of water chemistry at power units with WWER –1000. Under violation of water chemistry as well as under undue chemical cleaning there are cases with damages of heat exchanging tubes related to their increased fouling with operational deposits and clogging of separate zones of intertube space with sludge.

Experience has shown that in steam generators of WWER-440 type with corridor arrangement of tube bundles, less number of heat exchanging tubes have been plugged so far than at steam generators of WWER-1000 type having a staggered arrangement of tubes in the bundle, with the operating life of steam generators of WWER-440 type being twice as much. It is also worth mentioning that at the initial period steam generators PGV-440 operated with feed - and blowdown water having quality far worse than that at the PGV-1000. That is why OKB “Gidropress” has developed steam generator PGV-1000M with increased diameter of the vessel and sparse corridor arrangement of tubes in heat exchanging bundle to enhance the reliability of operation of heat exchanging tubes of PGV-1000 and bring water chemistry of SG and the secondary circuit, on the whole, to conformity with the requirements imposed to them by leading nuclear countries.

Application of sparse corridor arrangement of tubes in heat exchanging bundle makes possible to:

— increase a circulation rate in the tube bundle that will reduce the probability of damage of heat exchanging tubes due to decrease in the rate of deposits increase on heat exchanging tubes and concentration of corrosion-active impurities under them;
— decrease the possibility of clogging the shell-side with sludge;
— make easy access into the shell-side for inspection of heat exchanging tubes and their cleanup, if necessary;
— increase the water inventory in steam generator;
— increase the space under the tube bundle to facilitate sludge removal.
4.5 Reactor coolant pump set

The RCP set represents a vertical centrifugal single-stage pump set GCNA-1391 (fig. 7), consisting of a hydraulic casing, internals, electric motor, upper and lower spacers, supports and auxiliary systems. The reactor coolant pump set is intended to establish the coolant circulation in the primary circuit of the reactor plant.

RCP set has:
— main thrust bearing with water cooling and lubricant;
— double-speed electric motor, that reduces loads to transformer during startup (this provides possibility of stepwise startup);
— seal, which is capable to provide non-exceeding of nominal leakage (50 L/h) during RCP set trip without 24 hour-cooling at nominal parameters of the primary circuit.

The RCP set electric motor bearings are greased with non-flammable lubricant.

In designing GCNA-1391 operational experience of GCN-195M was widely used. The latter operates presently at all NPP power units with WWER-1000 in operation. GCNA-1391 as compared with the prototype has the following advantages:
— use of water as lubricant and cooling of the main thrust bearing of the pump and incombustible lubricating liquid in electric motor as well as plate coupling not requiring lubricant that makes possible not to use oil system and completely exclude fire hazard:
— use of independent cooling circuit of the lower bearing operating during RCP set outage by natural circulation principle that makes possible not to use auxiliary pump.

Design changes performed in GCNA-1391 as compared with GCN-195M are directed to enhancement of reliability and independent operation in respect to station systems.
5. Measures on decrease in the cost, load factor increase, reduction in the scope of in-service maintenance and improvement of its conditions

The project of advanced WWER-1000 reactor plant is developed with regard for the latest requirements and achievements in nuclear power engineering. The project has a significant reference character of the equipment and systems applied in respect to operating NPP with WWER-1000 as well as to NPP under construction in China that makes possible to reduce RP cost due to standardizations and unifications of the equipment, technological components and materials being in use, and also due to experience in designing, manufacturing and operation.

Engineering solutions accepted in the design enables to provide the required level of RP reliability and safety due to the balanced number of active and passive safety systems as well as measures aimed at preventing and mitigating the consequences of accidents including the severe ones that is confirmed by IAEA peer review of RP design for NPP in China.

A number of new engineering solutions were suggested in the design which results in improvement of the RP economic figures. First and foremost, it is extension in service life of the main equipment, decrease in probability of reactor plant component failures, reduction in repair and maintenance, increase in FA burnup, optimization of fuel cycle which leads to decrease in operational expenses and to increase in load factor.

With this, a decrease in the duration of RP scheduled maintenance outage for refuelling, inspection, maintenance and repair takes place as a result of optimization of repair cycle structure, including increase in intervals between repairs; improvement of management in preparation for maintenance and repair and their execution, improvement of the quality of personnel training, optimization of work scope, optimization of conducting operating conditions under RP shutdown; application of high-effective means of process equipment for maintenance and repair (automated monitoring systems, multi-position power-nut drivers, repair equipment etc.), increase in the rate of performance of fuel handling procedures (increase in velocity of fuel handling machine and polar crane in the reactor hall); improvement of maintenance and repair technologies, improvement of the corresponding regulatory documents.

Upon the whole, measures implemented in the project of the reactor plant together with common-station measures make possible to decrease the cost price of the supplied electric power in comparison with the operating units with WWER-1000.

6. Conclusion

The project of advanced RP has been analyzed with the purpose for compliance with the basic requirements of the European Utilities Requirements (EUR), and this analysis has shown that it, basically, meets the above-stated requirements. Compliance of the project with up-to-date international requirements with regard for extended service life of the main equipment shows its fairly high potential to be realized during construction of NPP both in Russia and abroad.