VVER-440 reactor thermal power increase. Up-to-date approaches to substantiation of the core heat-engineering reliability

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Presently, increasing the power of the Units is of an urgent problem for nuclear power plants with WWER-440 reactors. Improving the fuel assembly designs and calculated codes creates all the prerequisites to fulfil this purpose.

The decrease in the core power peaking is reached by using the profiled fuel assemblies, burnable absorber integrated into the fuel, the FA with the modernized interface attachment, modern calculated codes that allows to reduce conservatism of the RP safety substantiation.

A wide spectrum of experimental study of behaviour of the fuel having reached burn-up (50-

60) MW days / kg U under the transients and accident conditions was carried out, the post-irradiated examination of the fuel assemblies, fuel rods and fuel pellets with four and five annual operating fuel cycle were performed as well and confirmed the high reliability of the fuel, the presence of large margins of the fuel stack state that contributes to reactor operation at the increased power.

The results of the work carried out on implementing the five and six annual fuel cycles show that the limiting fuel state as to its serviceability in the WWER-440 reactors is far from being reached. The neutronics and thermal-hydraulic characteristics of the cores of the V-213 RP power units in operation are such that the actual (calculated and measured) power peaking factors for the fuel assemblies and fuel rods, as a rule, are less than the maximum - design and limiting values given in specifications for delivery of fuel assemblies. This factor is a real reserve for increasing the power.

Presently there is an experience of the increased power operation of Kola NPP, Units 1, 2, 4 and Rovno NPP, Unit 2.

The “Loviisa” NPP Units are operated at 109 % power.

By the present moment the Russian experts had gained an experience in substantiating the core operation at 108 % power for “Paks” NPP, Unit 4.

From the viewpoint of power producers an increase in power of the Units results in the essential economic benefit by producing the additional electric power at the low expenses for additional modernization.

When changing over to the increased power operation it is reasonable for the second generation fuel assemblies with the increased fuel stack height to be used that allows to decrease the average linear heat rate. The average linear heat rate for the various power levels, types of the fuel assemblies and cores is given in Table 1. It is seen from the Table that, for example, for the "large" core containing 349 fuel assemblies the average linear power even for the power unit of 114 % corresponds to the average linear power of the “small” core operating at the design power of 100%.

Thus, the fuel ensure the prerequisites to increase the power of the Units.

Table 1

Core	Quantity of fuel assembly		q_{lcp} (W/cm)			
			N=100%	N=107%	N=109%	N=114%
Large	349	Second generation fuel assembly	125	134	136	143
		Standard fuel assembly	128	137	140	
Small	313		143	153	156	

An experience of substantiations to be performed and operation of the Units at the increased power

In 1977 two Kola NPP Units of the first phase (V-230 reactors that were put into operation in 1973 and 1975, respectively) were changed over into the increased power level of 107% Nном.

These Units were in operation at power of 107% Nном till 1984. Then because of placing the shield assemblies into the core to decrease the neutron fluence on the reactor vessel, the Units of the first phase were in 100 % power operation.

Till 1981 for all the modified Units of V - 230 type the fuel assemblies were delivered according to the original documentation being specific for each modification of the Unit or NPP. After 1981 the fuel assemblies unified as to their design and documentation started to be delivered for all the WWER-440 Units. The design of the fuel assemblies delivered for the V-213 Units was assumed as the basis. In the working fuel assemblies the sheathes with the unloading openings in the upper and lower parts began to be used. The unified fuel assemblies started to be delivered at the Kola NPP of the first phase in the period when they have been already in operation at power up to 107%Nном.

By the results of the increased power operation of the Units no abnormal phenomena have been revealed. The coolant activity was within the ranges not exceeding the sum of iodines from 1.10^{-5} up to 6.10^{-4} Ci/l.

At the Kola NPP, Unit 2 there was an activity increase not related to the power increase as the subsequent events showed. In the course of the further studies the large vibrating loads acting on the fuel assemblies were revealed and a special complex of the work was carried out to eliminate them.

The increased power operation of the Kola NPP of the first phase was terminated by the results of the reactor vessel metal examination and followed by the subsequent placing of the shield assemblies around the core periphery.

Both the operating experience of the Kola NPP, Units 1 and 2 and also an additional calculated substantiation of safe operation within the scope of the requirements of Regulations of Gosatomnadzor were taken into account during operation of the Kola NPP, Unit 4 from 1987 to 1988 and Rovno NPP, Unit 2 from 1985 to 1987 at power of 107%Nном (these reactors are referred to V-213 type).

In 2002 the substantiation of operation of the Unit 4 core at 108% power was performed.

When changing over the above Units of V-213 type into the increased power operation the fuel assemblies started to be loaded into the core with the fuel rods having the increased helium pressure up to (0,5-0,7) MPa under their claddings.

Decommissioning the Units being in the increased power operation was not due to any disturbances in safety or deviations from the requirements of the regulatory documents, but after the events at the Chernobyl NPP, Unit 4.

Since 1977 the “Loviisa” NPP WWER-440 reactors have been in operation at power of 109 % (1500 MW (th)) with the fuel assemblies (working ones and CPS FAs) enriched in 3,6 % U-235 when observing the design thermal and physical restrictions except for the coolant heating-up (the procedure of the hot point monitoring is used).

Additional conditions for increasing the power of the Kola NPP, Units 1 and 2

The additional conditions for increasing the power of the Kola NPP, Units 1 and 2 are as follows:

- coolant heating should not exceed the design one under the rated power;
- under the transients the automatics, protection and interlockings keep the Unit parameters within the permissible design limits;
- carrying out the reference balance tests over the secondary circuit;
- determining the measuring errors of the mass measuring thermocouples and hourly output of this information with indicating the hottest fuel assembly and setpoint value;
- carrying out the neutronic calculations for 107% power;
- determining the flowrates through the fuel assemblies;
- elaborating the « Tables of the operating conditions of the Units »;
- change in protections and interlockings.

These requirements and the measures arising from the former were dictated by the fact that the first phase of the Kola NPP had differences from the “General provisions for ensuring safety of nuclear power plants during designing, construction and operation”, accepted after its commissioning.

The V-213 Units were in the increased power operation according to the valid instructions and supplements on the increased power operation.

The supplements to the operating manuals included the additional kinds of monitoring and an increased frequency of taking the standard monitoring parameters, and also an indication of the actions related to the possible increase in the coolant activity. In this case the requirement was also imposed that in case of development of the maximum permitted power level the coolant flowrate at SVO-1 shall be within the ranges (20-40) t/hours. The KAPRI system for the on-line monitoring of power fields in the core was introduced. During the power growth after the load shedding the on-line monitoring of both the temperature values at the fuel assembly outlet using ICIS "Gindukush" and the linear heat rate values by the KAPRI formats was carried out. Upon reaching the zero boric acid concentration in the primary coolant at the end of life with changing over to regulating by the control group, the power peaking factors were analyzed each 10 cm of the group rise in order not to allow for the power of the fuel assemblies and linear heat rate to be exceeded.

The main results of substantiation of increase in power of the “Paks” NPP, Unit 4 up to 1485 MW

In 1998 substantiation of operation of the generalized unit in the four annual fuel cycle with the use of the WAs average enriched in 3,82 % and the FAs average enriched in 3,6 % was made for the “Paks” NPP. In 2002 substantiation of the Unit 4 core operation at 108 % Nnom was made as well.

To substantiate the “Paks NPP” Unit 4 core operation at 108 % the document “Report on the design requirements” has been elaborated; it contains the design requirements the fulfillment of which is necessary to ensure the RP normal operation, the reactor core functioning under both the anticipated operational occurrences and design basis accidents under the increased power conditions of the “Paks” NPP, Unit 4.

The following documents were the normative base:

- U.S. Nuclear Regulatory Commission, “ Standard Review Plan “, Section 4.2 (Fuel System Design), NUREG - 0800, Rev.2, July 1981;
- “General provisions for ensuring safety of nuclear power plants (OPB - 88/97) PNAE G - 01-011-97, Moscow, 1997;
- “Nuclear safety regulations for reactor units of nuclear power plants (PBYa RP NPP - 89) PNAE G - 1-024-90, Moscow, 1990;
- Report on design requirements.

Within the framework of substantiation of possibility of the power increase at the “Paks” NPP, Unit 4 the summary report has been made on fulfilling the design requirements. The report deals with the results of substantiation of the strength, physical, thermal-hydraulic aspects and the safety analysis aspects during the accidents.

- Fuel assembly strength aspects

The calculations of the base design were used due to the identity of the basic operational parameters affecting the strength.

- Physical design

The physical design deals with the chosen “Paks” NPP loadings under the increased power conditions of Unit No.4. In the first transitive 18-th loading the thermal power was

1430 MW, beginning from the 19-th loading the power was 1485 MW.

The detailed neutronics on the chosen loadings are given here as well. During the stationary loadings in the two-year irradiated fuel assembly being near the core centre, for the fifth day there occurs the power release increase up to Kqmax=1,37.

On the basis of the above-stated neutronic calculations the fuel assembly nominal power of 6,3 MW and fuel rod of 54,5 kW are set.

For thermal-hydraulic calculations of stationary normal operating condition at the power of 1485 MW the conservative distributions of fuel assembly relative powers are given, where the maximum value of relative power in the fuel assembly, fuel rod, pellet can be realized: K_q^{max}=1,37; K_r^{max}=1,465; K_0^{max}=2,05. During these relative powers the fuel assembly, fuel rod, pellet have the maximum power. The calculations showed that the powers given correspond to the operational limits under the conditions of the power raised up to 1485 MW.

On the basis of calculation the initial data are obtained for analysis of the accidents (the maximum and minimum values of the various neutronics at power of 1485 MW, conservative axial distribution of the worth of emergency protection, working group) are given. The prescribed fuel assembly maximum power of 6,58 MW and that of fuel rod of 57 kW corresponding to the safe operation limits are confirmed by the calculations.

Besides, the calculation results confirm the fulfillment of the requirements imposed on the physical part of the design concerning the reactivity control with both the regulating groups and boron solution.

- Thermal-hydraulic design

To confirm that the design requirements for the DNBR are met, the thermohydraulic calculation of the steady-state operating conditions of “Paks” NPP, Unit 4 was carried out at the raised power up to 1485 MW considering the fuel assembly cooling conditions and restrictions on the maximum coolant temperature in a jet at the hottest fuel assembly outlet during the normal operation.

Under the considered steady-state normal operating condition with the “Paks” NPP RP nominal parameters for the transient and equilibrium fuel loadings the DNBR is not less than 2,56.

Coolant flowrate through the fuel rod bundle of the working assembly and ERC is within the design range of 100-130 m^3/h and amounts to 109 m^3/h and 115,8 m^3/h, respectively.

The maximum coolant temperature in a jet at the hottest working assembly outlet is equal to 323°C and does not exceed the saturation temperature of 326,3°C under the reactor core outlet pressure of 12,26 MPa.

The analysis of the cell-by-cell calculation results showed that an increase in the control rod pitch from 12,2 mm to 12,3 mm results in improving the heat-exchange conditions in the heat-powered regular (inter-fuel rod) cells of the working assembly due to increasing the coolant mass velocities and decreasing the hydraulic maldistribution as a whole along the fuel rod bundle cross-section.

The coolant temperature in a jet at the hottest working assembly outlet with the limiting neutronics at the end of life of the fuel loading decreases from 325,1°C to 323,0°C, and DNBR increases from 2,47 to 2,56.

The results of calculation of thermal and hydraulic characteristics of the core and reactor under the considered conditions for the worst combination of the parameter deviations: i.e. reactor thermal power is 1515 MW (102% of 1485 MW), coolant pressure at the core outlet is 12,06 MPa (by 0,2 MPa below the nominal value of 12,26 MPa), coolant flowrate minimum value through the reactor of 39300 m^3/h) and the limiting neutronics, confirm the core reliable cooling under the considered steady-state normal operating condition (DNBR is not less than 2,36).

The coolant flowrate through the fuel rod bundle of the fuel assemblies exceeds the minimum allowable value of 100 m^3/h and amounts to 105,9 m^3/h.

The maximum coolant temperature in a jet at the hottest working assembly is 325,0°C and does not exceed the saturation temperature of 325,1 °C under the reactor core outlet pressure of 12,06 MPa.

- Safety analyses during the postulated accidents

Safe state of the fuel rods during the design basis accidents is regulated by the criteria specified in the report on the design requirements with consideration of the requirements of normative documents and given in Table 2.

Table 2 – Safety criteria

Criterion	The purpose of introducing the criterion
Maximum cladding temperature is not more than 1200 ^{o}C Maximum local depth of cladding oxidation is not more than 18 % of its initial thickness	The absence of occurrence of self-sustaining steam-zirconium reaction is necessary to ensure the core cooling ability. Restriction of cladding embrittlement is necessary to avoid the fuel rod fragmentation in case of flooding, to ensure the possible core unloading.
The zirconium fraction having reacted with the steam in the core is not more than 1 % of its mass in the fuel rod claddings	Limitation of the hydrogen quantity having formed during the steam-zirconium reaction is necessary for the explosive mixture formation not to be allowed
Fuel temperature is less than the melting temperature: $T_{ПЛ}$ = (2840 - 0,56∗B) ^{o}C, where B-burn-out, MW·day/kg U	No interaction of the melt fuel and cladding is required to keep the cooled core geometry and ensure its possible unloading
Fuel rod section-averaged enthalpy not more than 230cal/g (963kJ/kg)	No fragmentation of fuel rods under the conditions of quick energy release during the accident followed by reactivity rise is necessary to keep the cooled core geometry and ensure its possible unloading

From the viewpoint of fulfilling the above-stated criteria in safety analysis of the "Paks" NPP, Unit 4 core fuel rod behaviour during operation at the increased power of 108 % in the course of the design basis accidents the following determining conditions are considered:

- accident with a break of the main coolant pipeline of Dnom 500 equivalent diameter;
- accident with ERC assembly ejection.

Besides, the condition is considered with both the control rod drop by gravity and imposing the failure to scram; this condition by the Russian classification can be attributed to a class of the beyond design-basis accidents, as it is caused by two independent initiating events.

The analyses of these conditions were carried out on the basis of the results of the neutronic calculations and thermal-hydraulic calculations performed by the Hungarian experts.

The calculated analysis results showed that the above-stated safety criteria regulating the fuel rod state during the accident with the primary circuit leak of diameter Dnom 500 are met , namely:

- maximum fuel temperature in the fuel rod - 1456,1°C; it is less than the melting temperature;
- maximum fuel rod cladding temperature - 872,5 °C; it is less than1200 °C;
- maximum local oxidation depth of the fuel rod cladding - 0,438 %; it is less than18 %;
- evaluation of relative amount of the oxidized zirconium in the core does not exceed 0,163 % of its mass in the fuel rod claddings - less than 1 %;

The performed analyses of the accidents with both ejection of the ERC fuel assembly and the control rod drop by gravity with imposing the failure to scram confirm the meeting of the safety criteria given in Table 2. The results of the given analyses are given in Table 3

Table 3

Results of the accident analysis with the ERC assembly ejection	Results of the accident analysis with control rod drop by gravity and imposing the failure to scram
Maximum fuel temperature in the fuel rod of 1517°C	Maximum fuel temperature in the fuel rod of 2484°C
Maximum average-radial fuel enthalpy of 76,8 cal/g (321,8 kJ/kg)	Maximum average-radial fuel enthalpy of 143,4 cal/g (600,4 kJ/kg)
Maximum fuel rod cladding temperature of 698,5°C	Maximum fuel rod cladding temperature of 1184,5°C
No essential cladding oxidation occurs	Maximum local oxidation depth of fuel rod cladding of 6,81 %
	Evaluation of relative amount of the oxidized zirconium of the hottest fuel rod cladding of 1,12%.

Besides to check how the safety criteria are fulfilled the additional calculated analyses of the determining condition of the group with an increase in the heat removal from the secondary circuit side, that one of the group with a decrease in the heat removal from the secondary circuit side and that one of the group with a decrease in the primary coolant flowrate are carried out.

As the determining condition for the group with an increase in the heat removal from the secondary circuit side, that one is chosen with a steamline rupture on the section from the SG upstream of the MSIV and resulting in the maximum possible increase in the heat removal by the secondary circuit out of the given group of the conditions. The heat removal increase leads to decrease in the coolant temperature at the core inlet, that due to the action of coolant temperature feedback of reactivity results in a quick reactor power increase and DNBR decrease. Thus, the given condition was analyzed from the viewpoint of checking the fulfillment of the acceptance criterion by the no DNB in the core.

The acceptance criteria and the RP parameter extreme values reached during the considered transient are given in Table 4.

Table 4

Acceptance criterion	Result of analysis
Departure from nucleate boiling ratio (DNBR) is not less than 1,0 with confidence probability of 95%.	Minimum DNBR is 1,63
Maximum fuel temperature does not exceed 2570 °C.	Maximum fuel temperature is 1777°C
Pressure in the primary circuit systems does not exceed 15,1 MPa.	Maximum pressure is 13,32 MPa
Pressure in the secondary circuit systems does not exceed 6,15 MPa.	Pressure during the calculated transient does not exceed the initial one

As the determining condition for the group with a decrease in the heat removal from the secondary circuit side that one is chosen with an inadvertent MSIV closing on the steam line of one SG and that shall confirm the possibility of accumulation of large heat quantity in the primary circuit and its transfer to the secondary circuit when there is the closed MSIV on one steam line from the viewpoint of the acceptance criterion on the non-exceeding of the permissible secondary pressure.

Two alternative calculations of the condition were performed.

The calculation results show that in all the alternative calculations the changes in the reactor plant parameters during operation at power of 1485 MW, except for the secondary pressure, do not fall beyond the ranges of the acceptance criteria to be assigned for the given initiating event.

For version 1 of the calculation wherein the RCP trip is not taken into account by the fact of generating the signal from the tips on the MSIV signalling on the seating of the MSIV, the secondary pressure exceeds the prescribed acceptance criterion on the secondary pressure. The second version of the calculation deals with the generation of the signal to trip the RCP by the fact of MSIV closing. Such a scenario is given for version 2 of the calculation. For this scenario the acceptance criterion on the secondary pressure is met.

Due to the revealed possible situation it is concluded that the change-over of the “Paks” NPP, Unit 4 into the increased power level from the viewpoint of the condition under consideration does not result in any contradictions with the acceptance criteria if the condition of RCP trip in case of the MSIV closing is assumed.

The acceptance criteria and the extreme values of the RP parameters reached under the considered transient are given in Table 5.

Table 5

Acceptance criterion	Result of the analysis	
	Version 1	Version 2
DNBR is not less than 1,0 with confidence probability of 95%	Minimum DNBR of 1,93	Minimum DNBR of 1,62
Maximum fuel temperature does not exceed 2570 °C	Maximum fuel temperature of 1677 °C	Maximum fuel temperature of 1680 °C
Pressure in the primary circuit systems does not exceed 15,1 MPa	Maximum pressure of 13,24 MPa	Maximum pressure of 13,87 MPa
Pressure in the secondary circuit systems does not exceed 6,15 MPa	Maximum pressure in the SG is 6,16 MPa	Maximum pressure in the SG 5,85 is MPa

The determining condition from the viewpoint of no-DNB reactor core cooling for the group with a decrease in the coolant flowrate in the primary circuit is that one followed by the trip of three RCPs as under this condition there occurs the maximum decrease in the coolant flowrate through the reactor core without a direct RCP trip signal in response to the reactor scram.

As the determining acceptance criterion we have assumed the criterion on non-reaching the DNB in the reactor core.

The calculation results show that the changes in the reactor plant parameters during operation at power of 1485 MW do not fall beyond the ranges of the acceptance criteria to be prescribed for the given initiating event. The acceptance criteria of acceptance and the RP parameter extreme values reached under the considered transient are given in Table 6.

Table 6

Acceptance criterion	Result of the analysis
DNBR is not less than 1,0 with confidence probability of 95%	Minimum DNBR is reached at 31,2 s and amounts to 1,08.
Maximum fuel temperature does not exceed 2570 °C (for the burnt-out fuel)	Maximum fuel temperature is reached at 30,8 s and amounts to 1772 °C.
Pressure in the primary circuit systems does not exceed 110% of the design one and amounts to 15,1 MPa	Maximum pressure in the primary circuit is reached at 31,7 s and amounts to 13,84 MPa.
Pressure in the secondary circuit systems does not exceed 110% of the design one and amounts to 6,15 MPa	Maximum pressure in the secondary circuit is reached at 70,0 s and amounts to 5,84 MPa.

- Design of the fuel rod

During the fuel system analysis under the normal operating conditions the following factors are considered:

- limiting stresses, deformations and loads of the fuel rod structural components;
- oxidation and hydrogenation of fuel rod claddings;
- dimensional changes in the fuel rods;
- internal pressure of gases in the fuel rod;
- hydraulic loads;
- fretting-wear;
- fuel temperature.

The acceptance criteria applied should ensure serviceability of both the fuel rods proper and their reliability as the fuel assembly structural component.

The fulfillment of the criteria according to the requirements of regulatory documents proves to be true by a calculated way.

The substantiation made showed the sufficient thermophysical and strength reliability of the fuel rods under the steady-state operating conditions while increasing the power of power unit up to 1485 MW.

Table 8 presents the intentions and stages of implementation of the WWWER-440 Unit power increase by the Operating Organizations.

Table 8

NPP Units	Power level	Stage of implementation
Units 1-4 «Dukovany» NPP	105%	Supplement to the Contract
Units 3 and 4 «Bohunice» NPP	107%	Supplement to the Contract is under discussion
Units 1 and 2 «Mochovce»	105-107%	Intention
«Paks» NPP Units	108%	Substantiation has been performed
Units 3 and 4 «Mochovce» NPP	114%	RP design
Units 3 and 4 Kola NPP	110%	Intention
Units 1 and 2 «Loviisa» NPP	109%	Operation

Conclusion

The achieved level of the WWER-440 fuel improvement, the available experience of substantiation and increased power operation of the Units allow for the thermal power up to 1500 MW (considering the equipment features of the different Units) to be ensured at the V-213 RP Units.