Nuclear Steam-Reheat Options: World Experience

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Abstract

The paper overviews world experience in Nuclear Steam-Reheat (NSR). Calculations of heat transfer to a coolant (light water) at supercritical temperature and subcritical pressure (T = 400 – 625 °C, P = 5.7 – 6.1 MPa) in the Steam-Reheat Channels (SRCh) of the single-reheat cycle layout for the Super Critical Water-cooled Reactor (SCWR) proposed in [1] are presented. The sheath and fuel (UO_2, UC, UC_2) centerline temperatures are calculated at the cosine axial heat-flux profiles.

1. Introduction

Since the 80's, the advancements in metallurgical technology have significantly improved the reliability of SuperCritical (SC) steam turbines. As a result, fossil-fuelled power plants operating at supercritical pressures form the basis of developed countries thermal-power engineering. The gross overall steam-cycle efficiency of SC power plants reached typically 47% – 54%, corresponding to 38% – 43% net plant efficiency (on a Higher-Heating Value (HHV) basis). SC turbines currently used in fossil fueled plants are designed for steam-reheat (SR) cycles. Besides cycle efficiency improvement, SR implementation reduces the steam flow required for a given power output, and furthermore, it reduces the steam moisture content in a Low-Pressure (LP) turbine thus eliminating the need for moisture-removal equipment. The improvements in plant thermal efficiency by NSR introduction were realized at the early stages of Nuclear Power Plants (NPP) development. Experimental reactors with implemented Nuclear Steam-Reheat (NSR) at subcritical pressures were developed in the 60's in Russia, USA and several other countries.

2. NSR in the USA

A very active program for the development and demonstration of NSR was instituted under the direction of the United States Atomic Energy Commission (USAEC). The main challenges considered in reactors with SR-option design were:

a) carry-over fission products or particles from SRChs to a steam turbine in direct-cycle systems;
b) maintenance of the desired power split in the boiling and superheating regions during extended reactor operation;
c) the effective separation of water from the steam prior to its entering the SHChs preventing the corrosion, erosion, and scale build-up on fuel-element surfaces;
d) the performance and corrosion resistance of fuel cladding at high temperatures in a reactor environment;

Reactors, which were constructed under this program included: BOiling Reactor experiment-V (BORAX-V), BOiling NUclear Superheater (BONUS), and Pathfinder. These were designated as integral superheating reactors and identified reactors in which steam is generated and reheated in the same core [2]. Major parameters of these reactors are listed in the Table 1.

Table 1. Major parameters of boiling-water reactor plants with integral reheat design [2].

	BONUS	**BORAX V**	**Pathfinder**
Thermal output, MW_{th}	50	20	200
Gross electric power, MW_{el}	17.5	3.5	66
Net electric power, MW_{el}	16.5	3.5	62.5
Plant steam cycle	Direct	Direct	Direct
Position of reheater region	Peripheral	Central or Peripheral	Central
Nominal operating P, MPa	6.7	4.14	4.14
Saturated steam T, ºC	284	254	254
Reheated steam T, ºC	482	454	441
Thermal power for boiling, MW	37	16.6	157
Thermal power for reheating, MW	13	3.4	43
Gross-cycle efficiency, %	35	–	33
Net-cycle efficiency, %	33	–	31
Max reheat fuel-cladding T, ºC	635	590	677

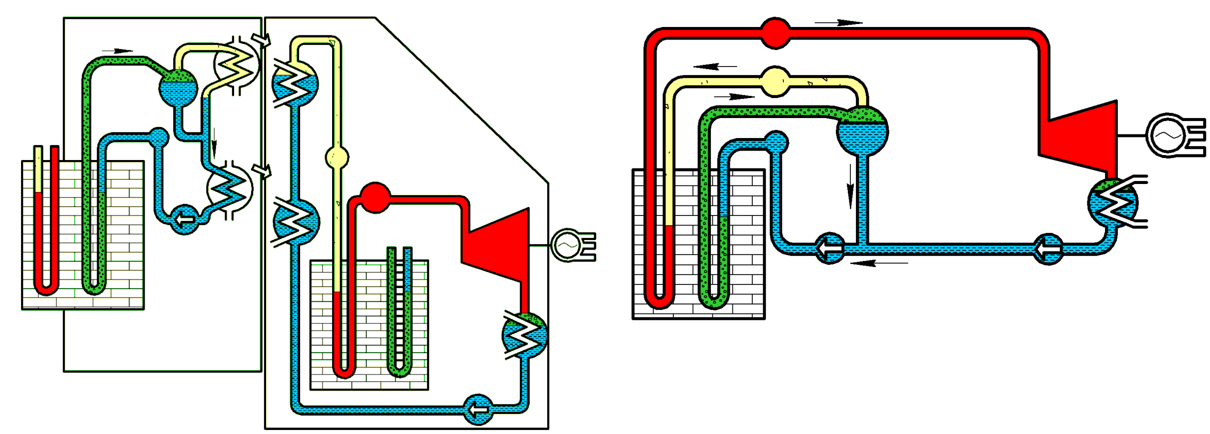
3. NSR in the former Soviet Union

NSR was industrially introduced at the Unit 1 of the Beloyarsk NPP (BNPP) for the first time in the world on April 26, 1964. Unit 2 was started-up on December 29, 1967. Steam parameters at the turbine inlet were the same for both units: P = 8.8 MPa, T = 500 – 510 °C. Main parameters of the units are listed in Table 2.

The second unit differed from the first by a simplified heat-flow diagram and core. Steam generated in Evaporating Channels (EChs) of Unit 2 was separated from water and fed directly into a turbine through (SRChs). Thus, Unit 2 had purely single-circuit heat-flow diagram (see Figures 1 – 2). The BNPP Unit-1 reactor was the uranium-graphite channel type. The reactor used slightly enriched uranium (3.3 – 3.4 %) with a graphite moderator. A number of fuel channels in the first unit was: 730 EChs for preheat and partial evaporation of water of the first circuit and 268 SRChs for steam reheat in the second circuit. A comparison of BNPP Unit 1 characteristics before and after the installation of SRChs is made in Table 3.

Table 2. Main parameters of the Beloyarsk NPP reactors with steam reheat [3].

Parameters	Unit 1	Unit 2
Thermal power, MW_{th}	286	560
Electrical power, MW_{el}	100	200
Uranium load, t	67	50
Specific load, MW_{th}/t	4.3	11.2
Uranium enrichment, %	1.8	3.0
Specific electrical-energy production, MW_{el}·days/t	4000	10000
Total number of operating channels	998	998
Number of SRChs	268	262
Square-lattice pitch, mm	200	200
Core dimensions, m: Diameter	7.2	7.2
Height	6	6

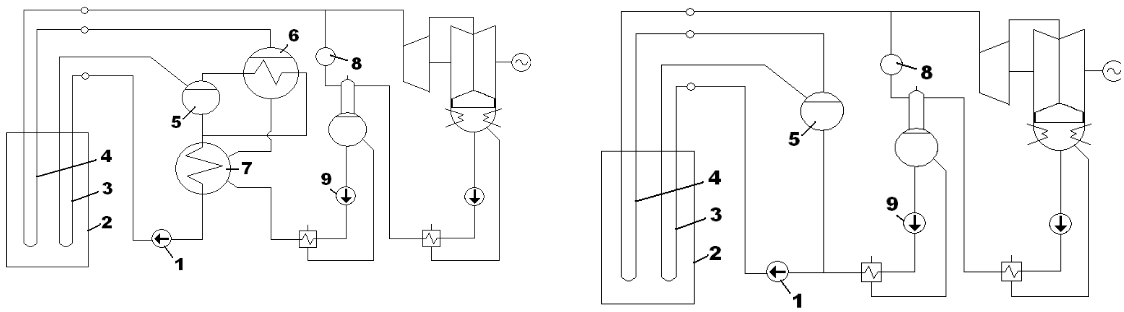


(b)

(a)

Figure 1. Simplified schemes of BNPP Unit 1 (a) and Unit 2 (b) [4]:

Reheated steam
Saturated steam
Water-steam mixture
Water



(a) (b)

Figure 2. Schematic flow diagram of the BNPP Unit 1 (a) and Unit 2 (b) [5]: 1 – circulation pump; 2 – reactor; 3 – EChs; 4 – SRChs; 5 – steam separator; 6 – evaporator; 7 – economizer; 8 – bubbler; and 9 – feed pump.

Table 3. Average parameters of the BNPP Unit 1 before and after installation of Steam-Reheat Channels [6].

Parameters	Before SRChs installation	After SRChs installation
Electrical power, MW_{el}	60 – 70	100 – 105
Inlet-steam *P*, MPa (atm)	5.9 – 6.3 (60 – 64)	7.8 – 8.3 (80 – 85)
Inlet-steam *T*, ºC	395 – 405	490 – 505
Exhaust-steam *P*, kPa (atm)	9 – 11 (0.09 – 0.11)	0.034 – 0.04
Water mass-flow rate, t/h	1400	2300 – 2400
P in separators, MPa (atm)	9.3 – 9.81 (95 – 100)	11.8 – 12.7 (120 – 130)
Gross thermal efficiency, %	29 – 32	35 – 36

4. Steam-reheat option for SCWR [1]

Operation of the BNPP two units showed feasibility of NSR implementation on the industrial scale. It represents valuable experience, because the idea of developing SCWR became attractive again as the ultimate development path for water cooling. It is reasonable to develop a steam cycle for SCW NPP similar to that of fossil-fueled plant, so that conventional SC turbines can be used, because 25 MPa and 600°C are common “steam” parameters in state of the art fossil-fueled power plants. So, the main objective of steam-reheat in nuclear reactors to increase the thermal efficiency of modern NPP from 30 – 35% to about 45 – 48% seems to be attainable.

The results of heat-transfer calculations in SRCh (parameters are listed in Table 4) of the SCW NPP (see Figure 3) are presented in Figures 4−6.

Table 4. Selected parameters of proposed SCW NPP cycle [1].

Parameters	Unit	Description / Value
Power Thermal	MW_{th}	2300
Power Electrical	MW_{el}	1200
Thermal Efficiency	%	52
P_{in} steam (SRChs)	MPa	6.1
P_{out} steam (SRChs)	MPa	5.7
T_{in} coolant (SRChs)	°C	400
T_{out} coolant (SRChs)	°C	625
Power thermal (SRChs)	MW_{th}	430
Number of SRChs	–	80
Total reheated steam flow rate	kg/s	780

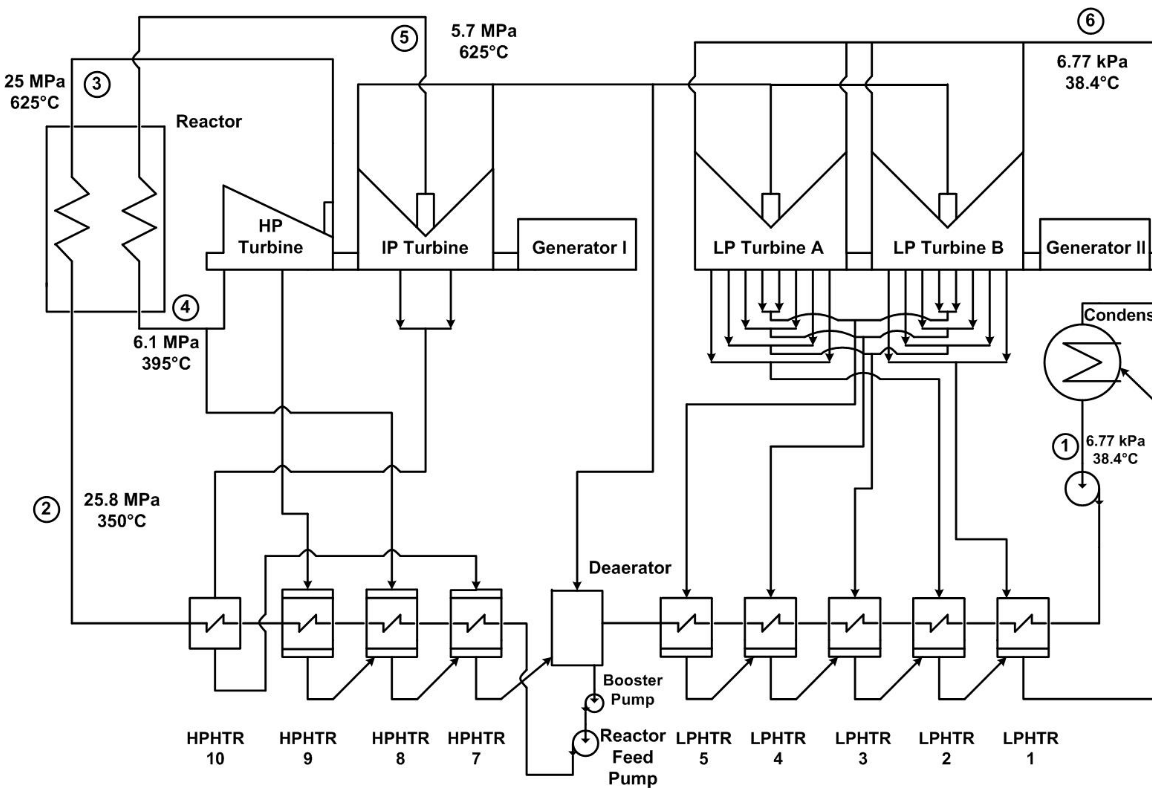


Figure 3. SCW NPP single-reheat cycle [1].

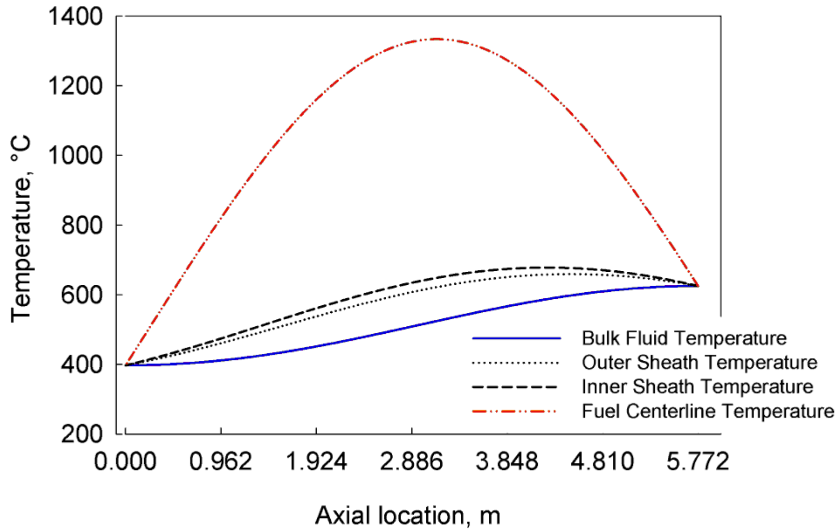


Figure 4. Temperature profiles for UO_2 fuel with cosine axial heat-flux profile.

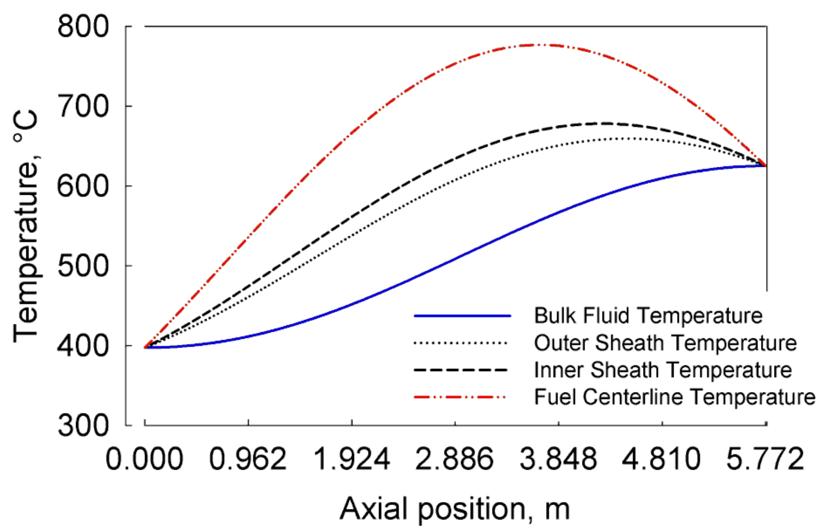


Figure 5. Temperature profile for UC fuel with cosine axial heat-flux profile.

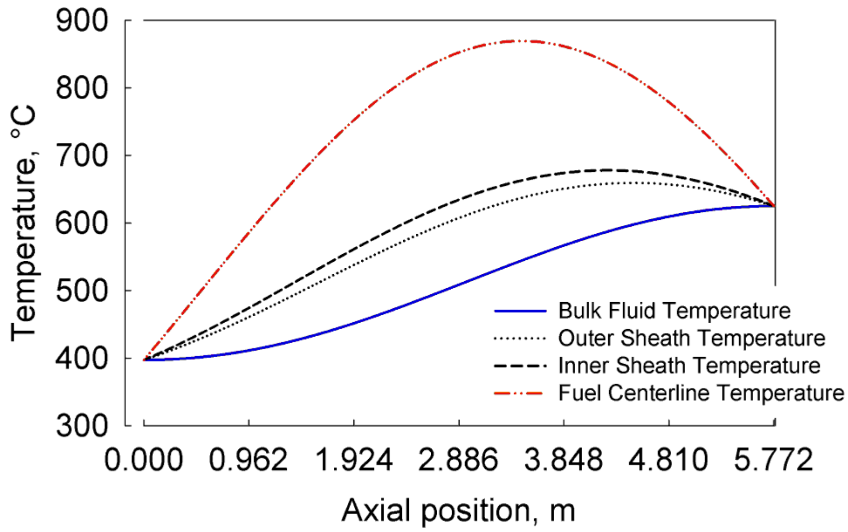


Figure 6. Temperature profile for UC_2 fuel with cosine axial heat-flux profile.

As seen from the figures, in neither cases, the fuel centerline temperature exceeds industry accepted limit of 1850ºC. The highest temperature is achieved in the case of UO_2 fuel (T = 1331ºC), the lowest is in the case of UC fuel (T = 777ºC). Also, maximum outer-sheath temperature doesn't exceed the design temperature limit (T = 678ºC).

7. Conclusion

The first experimental reactors with nuclear steam reheat (NSR) were constructed as early as 1960's. Over two decades of experimental operation of the Beloyarsk NPP showed feasibility of industrial-scale implementation of the nuclear steam reheat. Calculations of heat transfer in steam-reheat channels (SRChs) were performed for a SCW NPP concept proposed in [1]. The results showed that the highest fuel centerline temperature is achieved for UO_2. This temperature, however, is more than 500ºC below the industry accepted limit of 1859ºC.

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