NUCLEAR STEAM-REHEAT OPTIONS: RUSSIAN EXPERIENCE

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Abstract

Concepts of nuclear reactors cooled with water at supercritical pressures were studied as early as the 1950s and 1960s in the USA and Russia. After a 30-year break, the idea of developing nuclear reactors cooled with SuperCritical Water (SCW) became attractive again as the ultimate development path for water cooling. The main objectives of using SCW in nuclear reactors are: 1) to increase the thermal efficiency of modern Nuclear Power Plants (NPPs) from 30 – 35% to about 45 – 50%, and 2) to decrease capital and operational costs and hence decrease electrical-energy costs.

SCW NPPs will have much higher operating parameters compared to modern NPPs (pressure about 25 MPa and outlet temperature up to 625ºC), and a simplified flow circuit, in which steam generators, steam dryers, steam separators, etc., can be eliminated. Also, higher SCW temperatures allow direct thermo-chemical production of hydrogen at low cost due to increased reaction rates.

To achieve higher thermal efficiency a nuclear steam reheat has to be introduced inside a reactor. Currently, all supercritical turbines at thermal power plants have a steam-reheat option. In the 60's and 70's, Russia and some other countries have developed and implemented the nuclear steam reheat at subcritical-pressure in experimental reactors. There are some papers, mainly published in the open Russian literature, devoted to this important experience. Analysis of the Russian literature on nuclear steam-reheat option is presented in the current paper.

Keywords: Supercritical water-cooled nuclear reactor, Steam reheat.

1. Introduction

An idea of obtaining superheated steam directly in a nuclear reactor attracted attention in the very first stages of power-reactor development. Already in 1950s, during a discussion of possible alternatives to the reactor of the first NPP in the world (Obninsk, Russia), a reactor with nuclear steam reheat was considered, but it was postponed as technically insufficiently prepared option. The successful start-up in 1954 and operational tests of the Obninsk NPP reactor served as a basis for realization of the idea of nuclear steam reheat. A pressure-channel-type reactor was chosen to be developed as a more suitable design in the constructional sense compared to a pressure-vessel reactor. In the pressure-channel reactor it is possible to implement two types of fuel channels: one for evaporating water and another for steam reheat (Dollezhal' et al. 1958). The choice of a water-graphite channel reactor provided the following advantages:

- freedom of installation in the reactor fuel channels of various purposes and differentiated action on the physical and heat-engineering characteristics of the core;
- on-line refueling for more effective use of the fuel in the case of a sufficiently good equalization of power distribution throughout the core;
- the use of various designs of the fuel channels (removable and non-removable), sleeve and fuel-rod elements;
- the use of a progressive single-circuit engineering layout with the input of steam from the reactor to a turbine;
- increase in the reactor power on the basis of standard elements without fundamental restrictions from technical and safety reasons.

The most important technological challenge in developing a reactor with nuclear steam reheat was to design fuel elements, which would permit steam production at a temperature of 500 − 540°C and pressure of 8.8 − 12.7 MPa, and heat fluxes up to 1.2 MW/m^2 with acceptable neutron-physics characteristics and an economically practical depletion of the uranium. Another challenge was to maintain equalized power distribution and the power ratio for generation and reheat of steam, based on necessary thermal balance.

Problems noted for nuclear steam reheat have essentially been successfully solved in the design and upon the construction of the first Beloyarsk NPP (BNPP) reactors. Experimental testing of the most important elements of the reactor, its physical characteristics, thermalhydraulic processes, and transitional engineering conditions were conducted on special testing rigs and in experimental loops of the Obninsk NPP. A start-up of the first reactor with the nuclear steam reheat (100 MW$_{el}$) occurred in 1964, followed up with a start-up of the second reactor (200 MW$_{el}$) in 1967. The gross thermal efficiency of both units was about 37 − 38%. The reactors were identical in the structural sense, but differed only in the capacity and external engineering layout (Baturov et al. 1978).

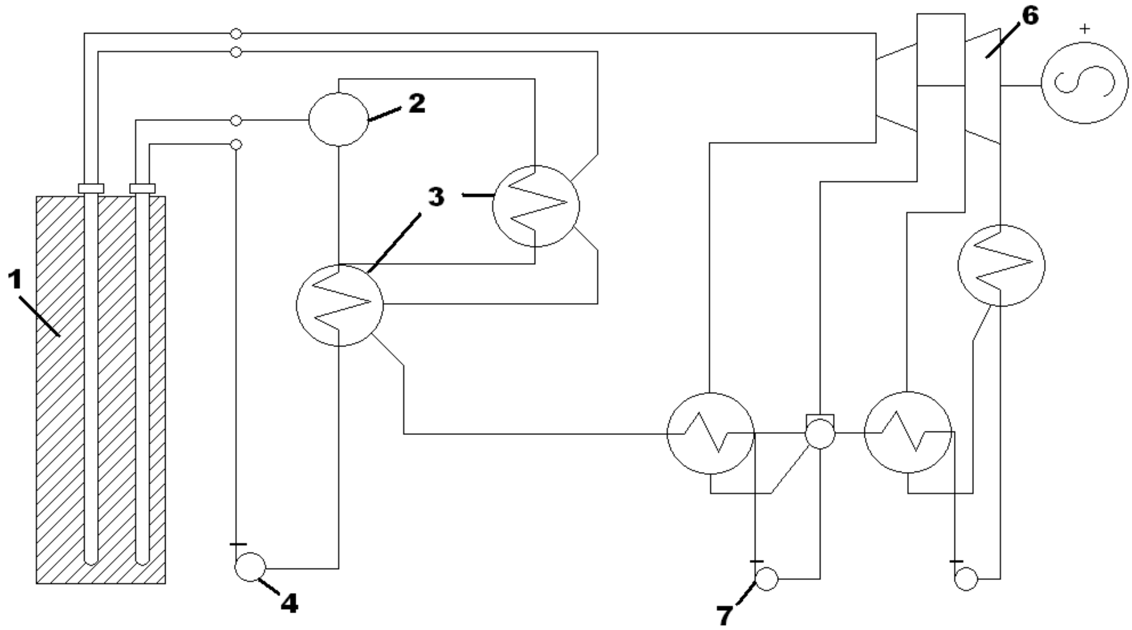

Figure 1. BNPP Unit 1 layout (Dollezhal' et al. 1958): 1 – reactor; 2 – steam separator; 3 – steam generator; 4 – main circulation pump; 6 – turbo-generator; 7 – feed pump; 8 – intermediate steam reheater.

2. Overview of Performance and Principal Characteristics of the BNPP

2.1 General Characteristics

The BNPP reactor was a uranium-graphite channel-type reactor with high-pressure steam reheat. The reactor used slightly enriched uranium; the moderator was graphite; and the number of fuel channels was 998: 730 evaporating channels provided preheating and partial evaporation of water of the first circuit and 268 steam reheat channels used for steam reheat in the second circuit. The superheated steam at a subcritical pressure passed directly, i.e., without any intermediate heat exchanger, from the reheat channels into the turbine connected through a shaft with an electrical generator. The core of the reactor (7.2 m in diameter and 6-m high) was surrounded by a graphite reflector (0.8-m thick) (Samoilov et al. 1976). Basic thermal-power characteristics attained with the first and second power units are listed in Table 1.

Table 1. Main thermal power parameters of the BNPP Unit 1 and Unit 2 (Dollezhal' et al. 1974).

Parameters	First unit	Second unit
Electrical power, MW_{el}	100 – 105	180 – 190
Thermal power, MW_{th}	285 – 290	490 – 515
Outlet steam T, °C	505 – 510	515 – 518
Outlet steam P, MPa	8.6 – 8.8	6.9 – 7.4
P in steam separators, MPa	11.8 – 12.3	11.3 – 11.6
Uranium Enrichment, %	3.3	3.4

2.2 Fuel Elements

Fuel elements of the Steam-Reheat Channels (SRChs) were expected to endure severe working conditions, as concluded from specifications and parameters of the reactor. In designing the fuel elements, certain mutually exclusive requirements were encountered, which were to meet with a compromise solution.

Development of the steam-reheat fuel element proceeded in several directions. A tubular fuel element with a stainless steel sheath and a uranium dioxide-based composite fuel were chosen for more extensive development after preliminary technological tests on production and experimental runs. The initial tubular design was later replaced with a U-shaped design of the SRCh. Efforts to improve the physical and thermal characteristics of the reactor led to the further modernization of the channel and modification of the fuel elements. One of the fuel bundles in the channel was removed and steam was reheated in five fuel bundles: first successively passing down along three bundles and then passing up along two other bundles. The inner tube of the fuel element was increased to a size of ø16×0.7 mm, the outer to ø23×0.3 mm. Thermal and technical characteristics improved sharply after this reconstruction (see Table 2) as the result of reduced matrix material volume in the fuel element and increased cross-section opening of the channel (Samoilov et al. 1976).

Table 2. Average parameters of the BNPP Unit 1 before and after SRChs installation (Dollezhal' et al. 1969).

Parameters	Before SRChs installation	After SRChs installation
Electrical power, MW_{el}	60 – 70	100 – 105
Gross thermal efficiency, %	29 – 32	35 – 36
Electrical power for internal needs, %	10 – 12	7 – 9
Turbine inlet steam P, MPa	5.9 – 6.3	7.8 – 8.3
Turbine inlet steam T, ºC	395 – 405	490 – 505
Exhausted steam P, kPa	9 – 11	3 – 4
Mass-flow rate of water in 1^{st} circuit, t/h	1400	2300 – 2400
P in separators, MPa	9.3 – 9.81	11.8 – 12.7

Since 1967, 300 SRChs have been installed in the BNPP Unit 1 and about 30 have been taken out for various reasons (failure due to errors of the service personnel, damage in the coolant tract and manufacturing faults, as well as for test inspections and monitoring). The remaining SRChs continued satisfactory operation, none of them having failed as a result of radiation damage to the fuel bundles or the incompatibility of their materials; premature extractions of the channels have been reduced to individual cases in each year. Over 450 SRChs have been installed in the reactor of the BNPP Unit 2 since 1967. Over the period 1967 − 1976 only eight SRChs were prematurely extracted, because of the breakdown of service conditions or for test inspections and monitoring.

The probability of the fault-free operation of the SRChs at designed power generation level (720 MW·days per channel) was over 0.96, accounting for channels failures due to constructional / technological reasons (experimental channels and channels failed because of the breakdowns in service conditions were not taken into consideration). The high reliability of the bundles in SRChs ensured stable operation of the reactor in the nuclear steam-reheat mode, and constituted one of the main factors in ensuring a high efficiency and relatively low net cost in electrical power production (Baturov et al. 1978).

2.3 Fuel Channels

Fuel channels of several types (steam-reheat and Evaporating Channels (EChs)), were used in the reactors of the BNPP Units 1 and 2. These channels ensured the operational reliability and high fuel burn-up needed in full-scale nuclear-power facilities. Damage suffered earlier by EChs was due to flaws in the welded joint, leaks in the expansion loops, poor sealing of the fuel bundles at the ends, chlorine corrosion under stress, and disturbances in operating conditions. After those shortcomings were eliminated, the service life of the fuel channels became 5 to 7 years at burn-up levels higher than ratings (Dollezhal' et al. 1974).

Improved operational reliability of the EChs was aided by the permissible output level criteria worked out and successfully applied, and by the attainment of free of burn-out cooling conditions. Operating experience showed that EChs malfunctions attributable to inadequate safety margin under burn-out were entirely avoided. Long-term performance of the SRChs failed to disclose any flaws or deficiencies in their design. Isolated instances of SRChs malfunctioning were mainly due to disturbances in operating conditions.

Table 3. Design parameters and operating conditions of SRChs (Dollezhal' et al. 1964).

Parameters	BNPP Unit 1	U-shaped channels with 6 fuel elements, BNPP Unit 2	
		Downward fuel bundles	Upward fuel bundles
Max channel power, kW	368	767	
Min channel power, kW	202	548	
Steam flow at maximum power per channel, t/h	1.9	3.6	
Steam flow at minimum power per channel, t/h	1.04	2.57	
Channel inlet *P*, MPa	10.8	12.9	12.2
Channel outlet *P*, MPa	9.81	12.3	10.8
Channel inlet steam *T*, °C	316	328	397
Channel outlet steam *T*, °C	510	399	508
Max heat flux, MW/m^2	0.56	0.95	0.79
Max steam velocity, m/s	57	76	112
Max *T*, °C:			
inner fuel-element sheath	530	426	531
fuel composite	550	482	565
graphite	725	735	

The excellent reliability of the SRChs made it possible to raise the steam temperature on a portion of these channels from 520 to 560°C at the exit. The actual burn-up levels in the EChs and in the SRChs were far higher than the ratings. For example, the average burn-up of uranium unloaded in 1973 was 13.7 MW·days/kg in the EChs and 23 MW·days/kg in the SRChs.

Deposits on surfaces of the fuel-element sheath consisting primarily of corrosion products (as much as 70% iron) occurred during operation of the BNPP Unit 2 and were attributed to the presence of carbon steels in the condensate and feedwater lines. The deposits were removed quite easily by washing the loops down with special solutions.

Monitoring fuel-channel output and reloading showed, that the impressive cost savings at the BNPP were attained by maintaining the maximum available power-output level combined with high heat-transfer reliability on part of the fuel channels. The BNPP Unit 1 went into the partial refueling mode in 1971. The reason was that the SRChs with steam parameters rated up to 545°C were installed in the reactor in 1967. Since 1971, there have been three to four partial refuelings (averaging 155 full days), each involving 32 fuel channels. There were an average of 35 fuel channels refueled every 80 to 85 full days in the BNPP Unit 2. Refueling of 30 to 35 fuel channels took an average of 48 to 72 hours, and had no effect on the duration of the planned overhaul and maintenance, which took 5 to 7 full days. The fuel channels were reloaded in such a way that axial and azimuthal symmetry of the reactor loading pattern remained unaltered; a certain relationship between the output of the SRChs and EChs zones, the optimum steam-reheat temperature, and the optimum thermal efficiency, and the minimum possible variation factor in the power distribution coefficient were maintained.

Fulfillment of these requirements allowed attaining a more uniform distribution of heat loads throughout the reactor. Furthermore, mutual repositioning of fuel channels during the reloading, and alterations in the relationships between the reloaded EChs and SRChs (depending on the heat loads on the fuel channels, the hydraulic characteristics of the fuel channels, and on the enrichment and burn-up percentage of the uranium) permitted an increase in the heat-transfer reliability of the fuel channel performance, and higher thermal efficiency. A variation factor of the power distribution was 1.20 to 1.25 for the EChs and 1.12 to 1.15 for the SRChs over the first years of operation of the BNPP Unit 2, and those values were brought up respectively to 1.28 − 1.35 and 1.20 − 1.30, with the increased difference in the outputs of fresh and burned-up fuel channels. The shaping of smoothed-out power-distribution fields over the core radius was achieved by the use of control rods in the first few months after the refueling. Within the last 2 to 3 weeks preceding a refueling, when the control rods were completely withdrawn from the core, the power distribution field was shaped in accordance with the pre-specified distribution of breeding properties throughout the core, which was taken into account and corrected in fuel channel refueling. The neutron fields in the BNPP Unit 1 and Unit 2 were significantly smoothed out over the core height, because of the non-uniform uranium burn-up. In the case of most fresh or spent-fuel channels, the variation factor of the neutron field with respect to height was 1.18 to 1.23. In the case of fuel channels located near the partially withdrawn control rods, this coefficient was slightly higher, but still in the range of 1.24 to 1.26 for both Unit 1 and Unit 2. In the beginning of the operating period, the variation factor of the neutron field with respect to height was 1.41 to 1.44 for both Unit 1 and Unit 2 (Dollezhal' et al. 1974).

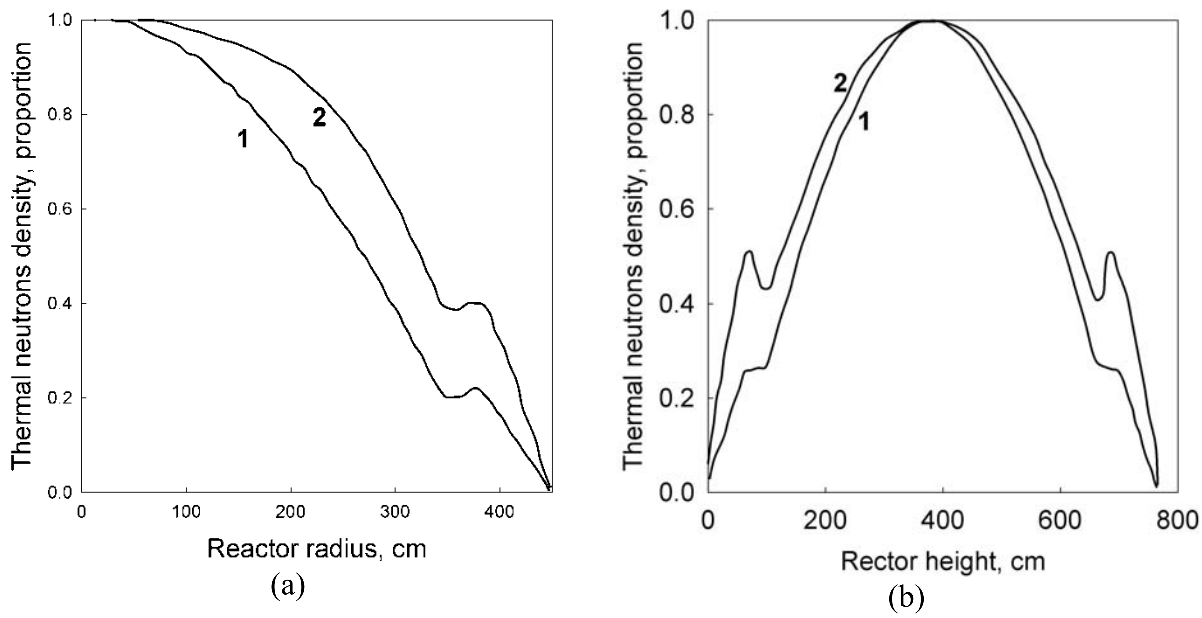

Figure 2. Thermal-neutron-density distribution along radius (a) and height (b) of the BNPP Unit 2 (Dollezhal' et al. 1964):
1 – beginning of the operating period; 2 – end of the operating period;

2.4 Control and Safety Systems

Control and safety systems performed satisfactorily on the whole. Among the disadvantages to consider were the low positioning of the power drives (beneath the reactor), which resulted in radioactive contaminants getting into the space accommodating the power drives, and building up there. Failures and malfunctions of the control and safety systems were due to jamming of control rods. Steel channels of the control and safety systems were replaced with zirconium channels at BNPP Unit 1, and the number of manual control devices was reduced to 52, thereby cutting down non-productive neutron losses. In the first five years of operation, malfunctions of electromagnets holding shutdown (scram) rods in place were observed, and those malfunctions resulted in a loss of power output, and sometimes in emergency reactor shutdowns. Special design electromagnets then installed featured sufficient reliability and permitted trouble-free operation of the reactors. The least reliable of the components of the control and safety systems were suspensions of ionization chambers and hydraulic shutdown system (Dollezhal' et al. 1974).

2.5 Radiation Conditions

Radiation conditions prevailing directly on the premises of the NPP itself and in its immediate environment were satisfactory. Radioactive discharges vented to the atmosphere remained below levels setup in public-health regulations. The turbo units of the BNPP were not provided with radiation shielding, since the maximum intensity of radiation was 1 to 10 μr/s in the high-pressure cylinder and 0.2 to 8 μr/s in the low-pressure cylinder. Radiation-dose rates were 0.05 to 0.1 μr/s in rooms, where personnel were constantly at work, and 0.3 to 12 μr/s in rooms occupied by personnel only part time. Dose rates in rooms not used by personnel, and near evaporating-loop equipment of the BNPP Unit 1, as measured while the reactor was shut down, amounted to 25 to 200 μr/s and to 15 to 20 μr/s near the equipment of the reheat-loop. Dose rates measured near equipment of the evaporating-loop of the BNPP Unit 2 were 15 to 200 μr/s, compared to 5 to 50 μr/s near equipment of the condensate-feedwater line. Radiation levels from these components were lowered by washing out the loops and by deactivation of individual pieces of equipment and deactivation of the loops completely. Operating experience showed that exposure of personnel of the NPP took place during repair and overhaul work, and was principally due to the radiation emitted by radioactive corrosion products deposited on the surfaces of the piping and process equipment (Dollezhal' et al. 1974).

3. Conclusions

The operating experience of the BNPP Units 1 and 2 showed a possibility of reliable and safe industrial application of nuclear steam reheat up to outlet temperatures of 510 − 540°C. The introduction of nuclear steam reheat was economically justified in cases where steam reheat was up to 500°C and higher with the use of stainless-steel-sheath fuel elements. The overall summary of the BNPP operation provides highly valuable information about problems associated with full-scale industrial implementation of nuclear steam reheat, and also provides important experience for current development of SCWR concepts with the nuclear steam-reheat option to achieve higher thermal efficiencies.

Acknowledgements

Financial supports from the NSERC/NRCan/AECL Generation IV Energy Technologies Program and NSERC Discovery Grant are gratefully acknowledged

Nomenclature

P	pressure, Pa
T	temperature, ºC

Subscripts

el	electrical
th	thermal

Abbreviations and Acronyms

BNPP	Beloyarsk Nuclear Power Plant
ECh	Evaporating Channel
NPP	Nuclear Power Plant
SCW	Supercritical water
SRCh	Steam Reheat Channel

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