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Selected aspects of the choice of live steam pressure in PWR nuclear power plant

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Abstract In commercially available generation III and III+ PWR (pressurized water reactor) reactors, pressure of steam produced in steam generators varies in a relatively wide range from 5.7 to 7.8 MPa. Therefore, it is important to ask which value of steam pressure should be used for a specific unit, taking into account different location conditions, the size of the power system and conditions of operation with other sources of electricity generation.

The paper analyzes the effect of steam pressure at the outlet of a steam generator on the performance of a PWR nuclear power plant by presenting changes in gross and net power and efficiency of the unit for steam pressures in the range of 6.8 to 7.8 MPa. In order to determine losses in the thermal system of the PWR power plant, in particular those caused by flow resistance and live steam throttling between the steam generator and the turbine inlet, results concerning entropy generation in the thermal system of the power plant have been presented.

A model of a nuclear power plant was developed using the Ebsilon software and validated based on data concerning the Olkiluoto Unit 3 EPR (evolutionary power reactor) power plant. The calculations in the model were done for

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design conditions and for a constant thermal power of the steam generator. Under nominal conditions of the Olkiluoto Unit 3 EPR power unit, steam pressure is about 7.8 MPa and the steam dryness fraction is 0.997. The analysis indicates that in the assumed range of live steam pressure the gross power output and efficiency increase by 32 MW and 0.735 percentage point, respectively, and the net power output and efficiency increase by 27.8 MW and 0.638 percentage point, respectively.

In the case of all types of commercially available PWR reactors, water pressure in the primary circuit is in the range of 15.5−16.0 MPa. For such pressure, reducing the live steam pressure leads to a reduction in the efficiency of the unit. Although a higher steam pressure increases the efficiency of the system, it is necessary to take into account the limitations resulting from technical and economic criteria as well as operating conditions of the primary circuit, including the necessary DNBR (departure from nucleate boiling ratio) margin. For the above reasons, increasing the live steam pressure above 7.8 MPa (the value used in EPR units that have already been completed) is unjustified, as it is associated with higher costs of the steam generator and the high-pressure part of the turbine.

Keywords: Live steam pressure; Efficiency and power output of a nuclear power plant; Nuclear power plant performance; Entropy balance

Nomenclature

h_i	–	steam enthalpy at the inlet of the i-th group of stages, kJ/kg
h_{Ro}	–	water enthalpy at the reactor outlet, kJ/kg
h_{Ri}	–	water enthalpy at the reactor inlet, kJ/kg
h_{ti}	–	steam enthalpy at the outlet of the i-th group of stages for isentropic expansion, kJ/kg
$\dot{m}$	–	mass flow rate, kg/s
$\dot{m}_i$	–	mass flow rate at the inlet, kg/s
$\dot{m}_p$	–	mass flow rate through the pump, kg/s
$\dot{m}_o$	–	mass flow rate at the outlet, kg/s
$\dot{m}_{\mathrm{R}}$	–	mass flow rate of water fed to the reactor, kg/s
$\dot{m}_{\mathrm{ti}}$	–	mass flow rate of steam at the i-th group of stages, kg/s
$\dot{m}_{\mathrm{doc}}$	–	mass flow rate from the technical documentation, kg/s
$\dot{m}_{\mathrm{ebs}}$	–	mass flow rate from calculations in the Ebsilon program, kg/s
n	–	number of pumps in the system, item
P_{b}	–	gross electric power output, MW
P_n	–	net electric power output, MW
P_{pi}	–	electric power needed to run the i-th pump in the system, MW
p_1	–	current steam pressure at the steam generator outlet, MPa
p_{1r}	–	reference (nominal) steam pressure at the steam generator outlet, MPa
p_{3r}	–	reference (nominal) steam pressure at the inlet of the HP part of the turbine, MPa
p_{R}	–	pressure of water (coolant) in the primary circuit of the reactor, MPa
$\dot{Q}_{\mathrm{b}}$	–	heat flux through the boundary of the control volume, kW
$\dot{Q}_{\mathrm{bR}}$	–	heat flux to the environment in the reactor, kW

$\dot{Q}_{\mathrm{R}}$	–	thermal power of the reactor, MW
$\dfrac{dS_{CV}}{dt}$	–	change of entropy rate in the control volume, kW/K
$\dot{S}_{\mathrm{gen}}$	–	entropy generation rate, kW/K
s_i	–	specific entropy at the inlet, specific steam entropy at the inlet of the i-th group of stages, kJ/kg/K
s_o	–	specific entropy at the outlet, kJ/kg/K
s_{Ro}	–	specific entropy at the reactor outlet, kJ/kg/K
s_{Ri}	–	specific entropy at the reactor inlet, kJ/kg/K
T_{b}	–	boundary temperature, K
T_{bR}	–	ambient temperature, K
$\mathrm{TTD}_{\mathrm{SG}}$	–	terminal temperature difference at the steam generator, °C
t_1	–	temperature of steam produced in steam generators, °C
t_i	–	value of the temperature of the i-th mass flow rate, °C
$\Delta\dot{m}$	–	difference of mass flow rate, $= \dot{m}_{\mathrm{doc}} - \dot{m}_{\mathrm{ebs}}$, kg/s
x	–	dryness fraction
ΔT_{b}	–	temperature margin to the boiling (saturation) point in the primary circuit, °C
ΔT_{SG}	–	drop in water temperature in the steam generator on the primary side, °C
ΔP_{ON}	–	auxiliary power of the power plant, MW
Δp_p	–	pressure rise across the pump, Pa
$\delta\dot{m}$	–	relative difference of mass flow rate, $= \Delta\dot{m}/\dot{m}_{\mathrm{doc}}$, %

Greek symbols

η_{b}	–	gross efficiency of the power unit
η_G	–	efficiency of the generator
η_i	–	isentropic efficiency of the i-th group of stages
η_m	–	mechanical efficiency of the turbine
η_n	–	net efficiency of the power unit
η_p	–	efficiency of the pump
η_s	–	efficiency of the electric motor
ρ	–	density of the medium flowing through the pump, kg/m^3

Abbreviations

ACR1000	–	advanced Candu reactor
AP1000	–	advanced passive pressurized water reactor
APR1400	–	advanced power reactor
CANDU	–	Canadian pressurized heavy-water reactor
DNBR	–	departure from nucleate boiling ratio
EDF	–	Électricité de France
EPR	–	evolutionary pressurized reactor
HP	–	high pressure part of steam turbine
LP	–	low pressure part of steam turbine
PWR	–	pressurized water reactor
VVER	–	vodo-vodyanoi energetichesky reaktor

1 Introduction

A nuclear power plant is planned to be built in Poland in the coming years. Nuclear power plants with pressurized water reactors (PWR) have the largest share in global electric power generation (73%) [1], which is why a PWR power plant will be most likely constructed in Poland. A number of PWR technologies are currently commercially available [2, 3], such as the Électricité de France (EDF) evolutionary pressurized reactor (EPR) [4], the American advanced passive reactor (AP1000) [5], the Russian vodo-vodyanoi energetichesky reaktor (VVER1200) [6], the Korean advanced power reactor (APR1400) [7], and the Canadian pressurized heavy-water reactor (Canada Deuterium Uranium – CANDU) [8]. A characteristic parameter of a nuclear power plant that determines its efficiency and power output is steam pressure at the steam generator outlet (at the turbine inlet). Values of efficiency and power output will be among the key parameters considered when selecting the nuclear power plant technology to be used in Poland. If we look at developments in PWR technologies over the years, an increase in steam pressure can be observed. For PWRs, steam pressure at the steam generator outlet was about 5.8 MPa for the first types of steam generators, 6.35 MPa for Konvoi [9], 6.75 MPa for P4 68/19, and 7.23 MPa for N4 73/19 steam generators [9]. For the EPR (Olkiluoto Unit 3), steam pressure at the steam generator outlet is about 7.8 MPa [10,11]. For the AP 600 and AP 1000 reactors, steam pressure at the steam generator outlet is 5.74 MPa [11,12] and 5.76 MPa [13], respectively. As far as VVER reactors are concerned, steam pressure at the steam generator outlet is 4.61 MPa for VVER 440 [14], 6.3 MPa for VVER 1000 [14, 15], and 7.0 MPa for VVER 1200 [15–17]. In the case of the Korean APR1400 reactor, steam pressure at the steam generator outlet is 7.3 MPa [7]. For the CANDU family of reactors, steam pressure at the steam generator outlet is 4.7 MPa for CANDU 6 [18], 5.0 MPa for CANDU 9 [19], and 6.0 MPa for the Advanced Candu Reactor ACR 1000 [20].

The pressure rise downstream of steam generators is due to thermodynamics, since for higher steam pressures the whole thermal system achieves greater efficiency and power output [21]. Despite the steam pressure rise, the net efficiency of PWR plants is in the range of 31% to 36% [22, 23] and is lower than in the case of conventional steam power plants [24–26]. Lower efficiencies with nuclear power plants result mainly from low parameters of steam fed to the turbine and the fact that the steam is saturated. In conventional steam power plants, steam is superheated and in new power units, it is most often at supercritical levels (25.0 to 30.0 MPa, 580 to 650°C). In nuclear power units, steam is at a much lower pressure (6.0 to 8.0 MPa) and nearly saturated or, more precisely, at a very low level of humidity. Since the steam parameters in nuclear power plants are lower, a smaller enthalpy drop in the turbine is available. Various adjustments of nuclear power plants have been investigated in order to increase the efficiency of the whole thermal system of nuclear power units. Modifications mainly concern combined systems, such as a nuclear power plant operating with a steam turbine [22,27–29], with a conventional steam power plant [30,31], in partial cogeneration mode to produce district heat [23,32–34], with water desalination plants [35], or with cogeneration of hydrogen [36,37]. Modifications in the structure of a nuclear power plant include using an additional steam superheater [38,39] and determining media parameters in order to maximize the power output and efficiency [40].

For nuclear power plants alone, calculations mainly involve the design point [22,41–44]. Energy and exergy analyses at the nominal load of a power unit can be found in the literature [45–51]. It is more difficult, however, to find papers concerning off-design conditions of nuclear power plants, with an exception of papers discussing the effect of cooling water temperature on the performance of a nuclear power plant [52–56]. There are also very few publications providing power output and efficiency characteristics of nuclear power plants as functions of pressure of live steam fed from steam generators, which is why the effect of steam pressure on the performance of the EPR (Olkiluoto Unit 3) nuclear power plant is analyzed in this paper. The paper also analyzes whether excess losses occur in machines and equipment comprising the thermal system of the EPR nuclear power plant based on entropy generation.

2 A mathematical model of a nuclear power plant

The impact of steam pressure on the performance of a nuclear power plant was examined for a nuclear power plant with an EPR reactor and a thermal system based on the power unit constructed in the Olkiluoto Unit 3. The thermal diagram of the nuclear power plant according to the Rankine cycle can be seen in Fig. 1. There are two main cooling circuits in a PWR/EPR nuclear power plant: the primary and the secondary one. The primary circuit comprises a nuclear reactor, a steam generator, a feed-water pump,

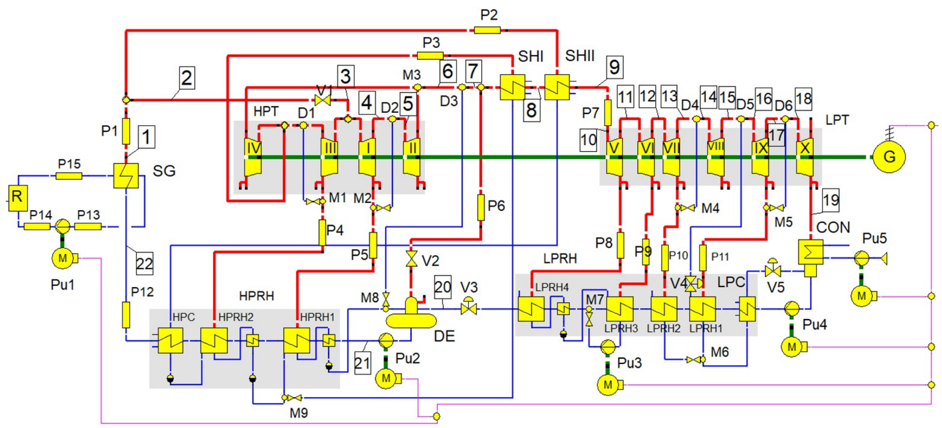

Figure 1: A diagram of the thermal system of an EPR (Olkiluoto Unit 3) nuclear power plant: R – nuclear reactor, SG – steam generator, HPT – the high-pressure part of the steam turbine, LPT – the low-pressure part of the steam turbine, SHI, SHII – superheater, first and second stages, CON – steam condenser, DE – deaerator, LPRH – low-pressure regenerative heat exchangers, HPRH – high-pressure regenerative heat exchangers, P – pipes, V – valves, Pu – pumps, M – mixers, HPC – high pressure cooler, LPC – low pressure cooler, D – drains, G – power generator.

and pipelines between the components. The primary circuit is where water at about 15.5–16.0 MPa circulates to remove heat generated in the reactor during the fission reaction of nuclear fuel. The secondary circuit includes the steam generator which connects the primary and secondary circuits by transferring heat from the primary to the secondary one. The secondary circuit comprises a steam turbine with its high-pressure (HP) and low-pressure (LP) parts. An integrated two-stage steam separator-superheater is placed between the HP and LP parts. From the steam generator outlet within the secondary circuit, nearly saturated steam (steam dryness fraction $x = 0.997$) is fed to the HP part of the turbine. Following expansion in the HP part, steam contains moisture that must be removed so that water droplets cause no damage of blades in the LP part of the turbine due to erosion. Water droplets are removed at the separator between the HP and the LP part of the turbine, and then steam is superheated in the two-stage steam superheater. The superheated steam is fed to the LP part of the turbine where it expands and where work is performed. In the LP part of the turbine, an internal moisture separation occurs to prevent steam at the outlet from exceeding the acceptable level of humidity. Following the expansion in the LP part of the turbine, steam flows to the condenser where it condenses. The condensate is pumped by a condensate pump through four LP regenerative heat exchangers to a deaerator. From the deaerator, water is pumped by a feed-water pump and flows through three HP regenerative heat exchangers up to the steam generator. In the steam generator, the heat transferred from the primary circuit is used to convert water into nearly saturated steam, and the cycle starts again.

A model of the nuclear power plant was developed using the Ebsilon software [57]. The model consists of complete components including groups of stages of the turbine, regenerative heat exchangers, a steam condenser, pipelines, pumps, and valves. The complete components employ mass and energy balances, relations for pressure drops, heat transfer equations according to Péclet's law, and (for groups of stages) the Fugel-Stodola steam flow capacity equation. The Ebsilon software offers no special components dedicated to a nuclear reactor, which is why the nuclear reactor was modelled as a source of heat fed to the primary circuit. With respect to operating conditions of the primary and secondary circuits, the nuclear power plant model was validated on the basis of data concerning the Olkiluoto Unit 3. This data set was presented at a lecture delivered by Framatome at the Institute of Heat Engineering [58]. Under nominal conditions of the Olkiluoto Unit 3 EPR power unit, steam pressure at the steam generator outlet is about 7.8 MPa, the gross power output of the system is 1714 MW, and the gross efficiency is 39.31%. Table 1 lists parameters under nominal conditions: steam pressure is about $p_1 = 7.8$ MPa and the steam dryness fraction is $x = 0.997$ at selected points of the unit's thermal system shown as a thermal diagram in Fig. 1. Values of pressure, efficiency of groups of stages of the turbine, and the terminal temperature difference of heat exchangers were input to the model developed in the Ebsilon software. Using the data, the software calculates values of mass flow rates, efficiency, and power output of the system. Therefore, the model was validated by comparing the power output and efficiency of the system and mass flow rates for selected points in the thermal diagram (Table 1).

The model provided the same efficiency and gross power output of the whole system under nominal conditions; the relative error for the mass flow rates was lower than 0.33%, which proves that the model matches well the data obtained from the technical documentation.

Steam pressure at the steam generator outlet (p_1) is determined from:

$$p_1 = p_s\,(t_1)\,, \tag{1}$$

$$t_1 = t_s\,(p_{\mathrm{R}}) - \Delta T_{\mathrm{b}} - \Delta T_{\mathrm{SG}} - \mathrm{TTD}_{\mathrm{SG}}\,, \tag{2}$$

where p_s and t_s refer to saturation conditions. Steam pressure at the steam generator outlet is affected by the pressure in the primary circuit, the margin to the boiling (saturation) point in the primary circuit, water temper-

Table 1: Pressure, temperature, specific enthalpy, the mass flow rate, dryness fraction, and specific entropy at selected points of the thermal system of the EPR (Olkiluoto Unit 3) power plant accompanied by validation of the model based on the mass flow rates.

Flow number	Pressure	Temperature	Specific enthalpy	Mass flow rate ($\dot{m}_{\mathrm{ebs}}$)	Dryness fraction	Specific entropy	Mass flow rate ($\dot{m}_{\mathrm{doc}}$)	Difference of mass flow rate $\Delta\dot{m} = \dot{m}_{\mathrm{doc}} - \dot{m}_{\mathrm{ebs}}$	Relative difference of mass flow rate
	MPa	°C	kJ/kg	kg/s	–	kJ/kg/K	kg/s	kg/s	%
1	7.8000	293.2	2757.9	2471.3	0.997	5.752	2467.8	–3.5	–0.14
2	7.5600	291.1	2757.9	2312.8	0.995	5.762	2309.7	–3.2	–0.14
3	6.9000	284.9	2757.9	2312.8	0.989	5.793	2309.7	–3.2	–0.14
4	1.9960	212.3	2557.7	1009.2	0.873	5.844	1008.7	–0.5	–0.05
5	1.9960	212.3	2561.2	1007.1	0.874	5.851	1006.3	–0.7	–0.07
6	0.9990	179.8	2458.4	1930.8	0.842	5.881	1927.9	–3.0	–0.15
7	0.9990	179.8	2761.1	1638.3	1.000	6.550	1632.9	–5.4	–0.33
8	0.9790	220.0	2876.6	1541.6	1.000	6.805	1541.2	–0.4	–0.03
9	0.9640	277.8	3005.0	1541.6	1.000	7.058	1541.2	–0.4	–0.03
10	0.9510	277.6	3005.0	1541.6	1.000	7.064	1541.2	–0.4	–0.03
11	0.4430	196.8	2852.0	1470.6	1.000	7.107	1471.5	0.9	0.06
12	0.2130	128.4	2722.6	1377.5	1.000	7.139	1379.7	2.2	0.16
13	0.0615	86.6	2537.2	1283.4	0.949	7.198	1287.1	3.6	0.28
14	0.0615	86.6	2543.7	1279.6	0.952	7.216	1282.8	3.2	0.25
15	0.0303	69.4	2454.6	1279.6	0.927	7.265	1282.8	3.2	0.25
16	0.0303	69.4	2483.8	1262.5	0.939	7.351	1263.8	1.3	0.10
17	0.0131	51.2	2384.4	1211.5	0.912	7.410	1214.6	3.1	0.25
18	0.0131	51.2	2418.9	1192.5	0.927	7.516	1193.4	0.8	0.07
19	0.0029	23.5	2271.7	1195.5	0.890	7.672	1193.9	–1.6	–0.14
20	0.9495	143.5	604.6	1563.3	0.000	1.775	1562.0	–1.3	–0.08
21	9.6110	179.4	764.9	2471.3	0.000	2.122	2467.8	–3.5	–0.14
22	9.0040	230.0	993.0	2471.3	0.000	2.600	2467.8	–3.5	–0.14

ature drop in the steam generator on the primary side, and the terminal temperature difference at the steam generator. The margin to the boiling point in the primary circuit depends on the coolant temperature at the reactor outlet and on the departure from nucleate boiling ratio (DNBR), which essentially influences the operational safety of the reactor. Water temperature drop in the steam generator on the primary side is related to its geometry, in particular to the heat transfer surface area (7960 m^2 for the steam generator operating with the EPR reactor), and heat transfer effectiveness. These two properties determine the overall heat transfer coefficient and the logarithmic mean temperature difference of the steam generator. For a given power output of the reactor (about 4300 MW), these properties determine the mass flow rate of water in the primary circuit (about 21.266 kg/s). When the temperature drop (ΔT_{SG}) is smaller, the cooling water mass flow rate is higher. The terminal temperature difference ($\mathrm{TTD}_{\mathrm{SG}}$) at the steam generator depends on heat transfer and optimization of the steam generator's design. If the terminal temperature difference is smaller, the design of the steam generator is better and the heat transfer is more effective, which makes it possible to achieve a higher pressure of live steam.

A diagram of temperature *vs.* the heat rate at the steam generator in the EPR plant under pressure in the primary circuit (p_{R}) equal to 15.5 MPa can be found in Fig. 2. On the primary side, water flows inside the tubes of the steam generator and its heat is transferred to the secondary fluid. On the secondary side, there is water at the steam generator inlet that

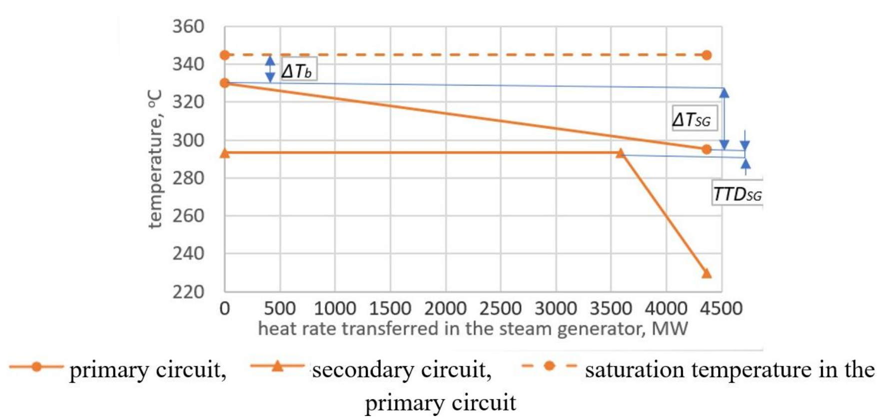

Figure 2: Temperatures, including the saturation temperature under pressure in the primary circuit, *vs.* the heat rate in the EPR.

takes heat from the primary circuit; the properties of the fluid at the steam generator outlet are close to those of saturated steam. The rise in water temperature in the reactor (the temperature drop on the primary side of the steam generator, ΔT_{SG}) is 34.8°C. The saturation temperature under pressure present in the primary circuit (15.5 MPa) is marked as a line in Fig. 2. Knowing the saturation temperature in the primary circuit, one can calculate the margin between water temperature and its boiling point (ΔT_{b}) which amounts to about 15°C. The terminal temperature difference ($\mathrm{TTD_{SG}}$) is about 2°C. When these optimized values of temperatures and mass flow rates in the primary circuit and the steam generator were used, temperature t_1 of about 293°C and pressure p_1 of about 7.8 MPa could be achieved in the EPR reactor.

The effect of steam pressure at the steam generator outlet on the performance of the PWR/EPR plant was determined for a constant flow rate of heat transferred in the steam generator. Calculations were performed for the design operating parameters of the power unit, which means that a minor modification of the system would be required in the area of the steam generator and the HP part of the turbine in order to change pressure p_1. Steam pressure at the steam generator outlet was adjusted between 6.8 and 7.8 MPa in steps of 0.1 MPa. A change in steam pressure at the steam generator led to a change in pressure at the turbine inlet. The total pressure drop between the steam generator outlet and the inlet of the HP part of the turbine was calculated from the following formula:

$$\Delta p_{13} = \frac{\left(p_{1r} - p_{3r}\right) p_1}{p_{1r}} \,. \tag{3}$$

Steam produced in the steam generator expands in the HP and LP parts of the steam turbine, generating mechanical power at the turbine shaft. The mechanical power is then converted into electric power in the generator. For the nuclear power plant considered, the gross electric power is a sum of power outputs generated at each group of stages of the steam turbine and can be written in the form

$$P_{\mathrm{b}} = \left\{ \sum_{i=1}^{i=n} \dot{m}_{\mathrm{ti}} \left[h_i\left(p_i, t_i\right) - h_{\mathrm{ti}}\left(s_i, p_{i+1}\right) \right] \eta_i \right\} \eta_m \eta_G \,. \tag{4}$$

For the calculated gross electric power output (4), the gross efficiency of the power unit was linked with the heat generated in the nuclear reactor

(the heat transferred to coolant)

$$\eta_{\mathrm{b}} = \frac{P_{\mathrm{b}}}{\dot{m}_{\mathrm{R}}(h_{\mathrm{Ro}} - h_{\mathrm{Ri}})} \,. \tag{5}$$

The net electric power is equal to the gross electric power less the auxiliary power used to operate the power plant

$$P_n = P_{\mathrm{b}} - \Delta P_{\mathrm{ON}} \,. \tag{6}$$

The auxiliary power of the power plant is defined as the total electric power that has to be supplied to all pumps in the system (condensate pumps, feed-water pumps in the secondary circuit, water pumps in the primary circuit, and cooling-water pumps)

$$\Delta P_{\mathrm{ON}} = \sum_{i=1}^{i=n} P_{pi} \,. \tag{7}$$

Note that the auxiliary power determined from Eq. (7) covers only the loads within the thermal system of the power unit without any auxiliary systems of the power plant. This is also the case with the net power output and efficiency expressed by Eq. (9). Other auxiliary loads are insignificant for the topic of this paper.

The electric power required to run the pump is calculated from the formula

$$P_p = \frac{\dot{m}_p \Delta p_p}{\rho \eta_s \eta_p} \,. \tag{8}$$

For the calculated net electric power output (6), the net efficiency of the power unit was defined as

$$\eta_n = \frac{P_n}{\dot{m}_{\mathrm{R}}(h_{\mathrm{Ro}} - h_{\mathrm{Ri}})} \,. \tag{9}$$

Irreversible processes take place in machines, devices and power systems, for example as a result of heat flow, flow resistance, and mixing of fluids. Due to the irreversible processes, less energy can be converted into work (power). The measure of the process irreversibility is entropy generation [21, 59, 60]. By calculating the entropy generation for machines and equipment that form part of a power system, it is possible to determine in which components of the system the entropy generation is the highest or is higher than expected. After identifying the machinery where entropy generation is the largest or larger than expected, solutions can be proposed to reduce this quantity. Entropy generation is calculated both for complete thermal systems of nuclear power plants [47, 61–64] and for selected components such as a nuclear reactor [65, 66] or a steam generator in order to choose geometrical parameters [67].

The paper analyzes the rise of entropy generation (losses) at the EPR power unit to assess the extent to which live steam pressure, particularly the pressure loss between the steam generator and the inlet at the turbine blades, affects the losses compared to losses at other components of the thermal system of the unit. The reason to carry out this analysis is a relatively low pressure of steam from steam generators compared to conventional power units.

The general entropy balance relations can be expressed for control volumes as [21]

$$\sum \frac{\dot{Q}_{\mathrm{b}}}{T_{\mathrm{b}}} + \sum \dot{m}_i s_i - \sum \dot{m}_o s_o + \dot{S}_{\mathrm{gen}} = \frac{dS_{CV}}{dt}\,. \tag{10}$$

The entropy balance relation for a general steady-flow process can be obtained by setting $ds_{CV}/dt = 0$ and rearranging to give [21]

$$\dot{S}_{\mathrm{gen}} = \sum \dot{m}_o s_o - \sum \dot{m}_i s_i - \sum \frac{\dot{Q}_{\mathrm{b}}}{T_{\mathrm{b}}}\,. \tag{11}$$

In a steady state, assuming that the mass flow rate at the inlet is equal to the one at the outlet and that heat losses to the environment are negligibly small, the entropy generation rate for a group of turbine stages, the pump and the valve is calculated as the product of the rate of mass flow through machinery and the difference between the specific entropy at the outlet and at the inlet of the machinery according to the formula

$$\dot{S}_{\mathrm{gen}} = \dot{m}\,(s_o - s_i)\,. \tag{12}$$

The entropy generation rate for a shell-and-tube heat exchanger in which heat is transferred between two fluids, assuming that heat losses to the environment are negligibly small, is equal to

$$\dot{S}_{\mathrm{gen}} = \dot{m}_c\,(s_{co} - s_{ci}) - \dot{m}_h\,(s_{hi} - s_{ho})\,, \tag{13}$$

where index c refers to the colder fluid and index h to the hotter fluid.

The entropy generation rate in the reactor caused by heat flow in the reactor and heat losses to the environment can be expressed as

$$\dot{S}_{\mathrm{gen}} = \dot{m}_{\mathrm{R}} \left(s_{\mathrm{Ro}} - s_{\mathrm{Ri}} \right) - \frac{\dot{Q}_{\mathrm{bR}}}{T_{\mathrm{bR}}} . \tag{14}$$

The following values were taken: ambient temperature of 20°C which is equal to the annual average ambient temperature in the nuclear reactor building, and the flow rate of the reactor heat lost to the environment equal to 0.5% of the flow rate of heat transferred to water in the reactor.

3 Results

The calculation results are true for pressure changes under nominal conditions, i.e. for the design operating parameters of the power unit. To achieve them, one would have to make a certain adjustment in the area of the steam generator and the HP part of the turbine. Calculations were done for steam pressures between 6.8 and 7.8 MPa. The dryness fraction of steam from the steam generator and steam pressure at the outlet of the HP part of the turbine are also constant. The flow rate of heat transferred from the reactor in the steam generator was constant and equal to about $\dot{Q}_{\mathrm{R}} = 4300$ MW.

With the increase in the steam pressure, the heat of evaporation is decreasing (Table 2). The mass flow rate of water at the steam generator inlet slightly rises so that the constant value of the flow rate of heat transferred in the steam generator is maintained. As the mass flow rate of water at the steam generator inlet increases, so does the power supplied to the feed-water pump according to Eq. (8) (Fig. 3).

For a constant flow rate of heat in the steam generator, a rise in live steam pressure leads to a slight drop in the heat of evaporation; this is followed by an increase in water mass flow rate at the steam generator inlet so that the whole flow rate of heat transferred in the steam generator can be removed. Another consequence of a higher steam pressure is an increase in water pressure downstream of the feed-water pump. The increase in both water pressure downstream of the feed-water pump and the mass flow rate of water flowing through the pump results in a higher entropy generation rate at the feed-water pump (Table 2). The increase in steam pressure at the steam generator causes the steam temperature to rise, thus elevating the mean heat removal temperature on the secondary side; this results in a lower logarithmic mean temperature difference and thus in a lower entropy generation rate at the steam generator [21, 59].

Table 2: The effect of steam pressure on the parameters in the area of the steam generator, and entropy generation rates at the steam generator and the feed-water pump.

Pressure of live steam	Temperature of steam at the steam generator outlet	Mass flow rate of water at the steam generator inlet	Steam enthalpy	Heat of evaporation	Temperature of water at the steam generator inlet	$\dot{S}_{\text{gen}}$, the feed-water pump	$\dot{S}_{\text{gen}}$, the steam generator
MPa	°C	kg/s	kJ/kg	kJ/kg	°C	kW/K	kW/K
7.8	293.2	2471.3	2757.9	1450.5	230.0	13.42	363.39
7.7	292.4	2469.0	2759.3	1456.8	230.0	13.26	373.76
7.6	291.4	2466.6	2760.7	1463.1	229.9	13.09	384.05
7.5	290.5	2464.3	2762.1	1469.4	229.9	12.93	394.56
7.4	289.6	2462.0	2763.5	1475.8	229.8	12.76	405.30
7.3	288.7	2459.6	2764.9	1482.2	229.7	12.60	415.82
7.2	287.7	2457.4	2766.2	1488.5	229.7	12.44	427.03
7.1	286.8	2455.1	2767.5	1494.9	229.6	12.27	438.03
7.0	285.8	2452.9	2768.8	1501.4	229.5	12.11	449.58
6.9	284.9	2450.7	2770.1	1507.8	229.5	11.95	461.08
6.8	283.9	2448.5	2771.3	1514.3	229.4	11.79	472.82

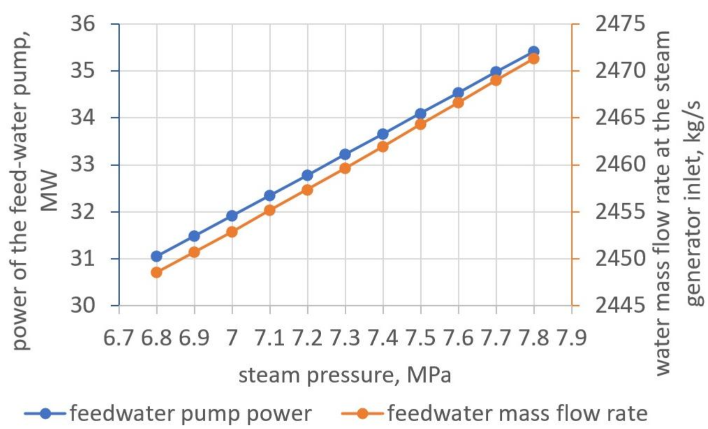

Figure 3: The effect of steam pressure on the mass flow rate of water at the steam generator inlet and on the power of the feed-water pump.

The curve of expansion in the turbine for a steam pressure of 7.8 MPa at the steam generator outlet is displayed in Fig. 4. The expansion curve shows the steam dryness fraction downstream of the HP part of the turbine (point 6), steam conditions downstream of the separator-superheater (point 9), and at the outlet of the LP part (point 19). Between the outlet of the steam generator (SG) and the inlet of the HP part of the turbine (HPT), pressure is falling across the pipeline connecting the steam generator and the turbine and across the valve (points 1 to 3).

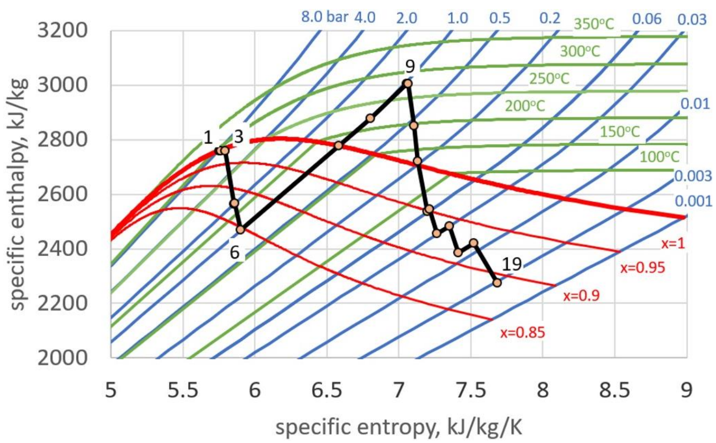

Figure 4: The curve of expansion in the turbine for a steam pressure of 7.8 MPa at the steam generator outlet. Points 1 to 6 – steam expansion in the HP part; points 6 to 9 – moisture separation and the interstage reheating; points 9 to 19 – steam expansion in the LP part with internal moisture separation.

As steam pressure increases, the specific enthalpy of saturated steam (at the turbine inlet) initially rises up to the maximum at a pressure of about 3.1 MPa and then it falls (Figs. 4 and 5). The value of specific enthalpy at the turbine inlet is important as this is the starting point of steam expansion in the turbine. The higher the specific enthalpy, the more enthalpy drop at the turbine is available and hence more power can be generated. Steam generators in nuclear power plants are currently operated at pressures greater than 4.0 MPa. When pressure exceeds 3.1 MPa, despite a slight enthalpy drop at the turbine inlet, the isentropic enthalpy drop at the HP part of the turbine increases, assuming that pressure at the outlet of the HP part of the turbine is constant. As steam pressure rises, the isentropic enthalpy drop at the HP part increases at a progressively smaller rate (Fig. 5). As can be seen in Fig. 5, adjusting the steam pressure over 7.8 MPa seems unjustified since there is a reduction in specific steam enthalpy at the turbine inlet and additionally the increase in the isentropic enthalpy drop in the HP part of the turbine is relatively small.

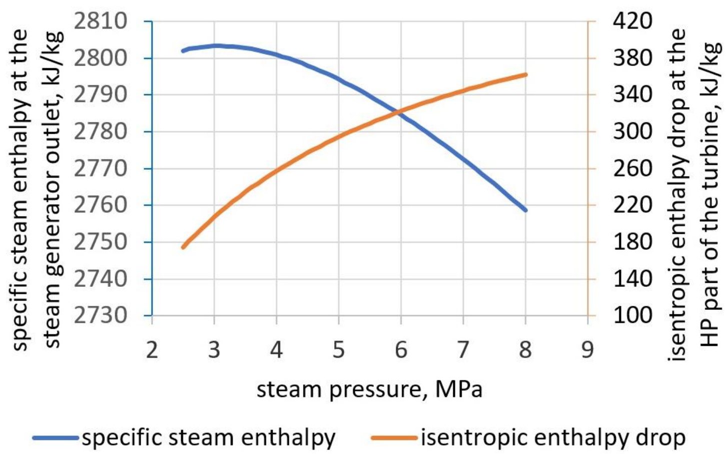

Figure 5: Specific enthalpy of saturated steam and the isentropic enthalpy drop at the HP part of the turbine as functions of steam pressure.

The gross and net power output and efficiency as functions of steam pressure (p_1) are shown in Figs. 6 and 7. At higher steam pressures the gross power output and efficiency tend to go up. At steam pressures varying between 6.8 MPa and 7.8 MPa, the gross power output increases by 32 MW and efficiency rises by 0.735 percentage point. The net power output and efficiency increase by 27.8 MW and 0.638 percentage point, respectively.

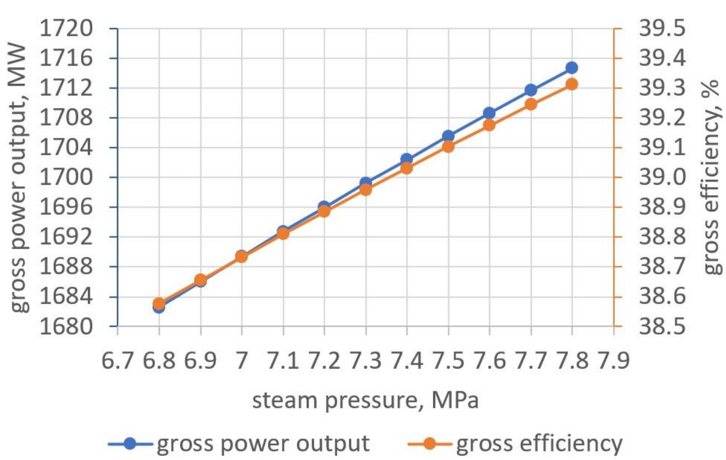

Figure 6: The effect of steam pressure on the gross power output and efficiency of the system.

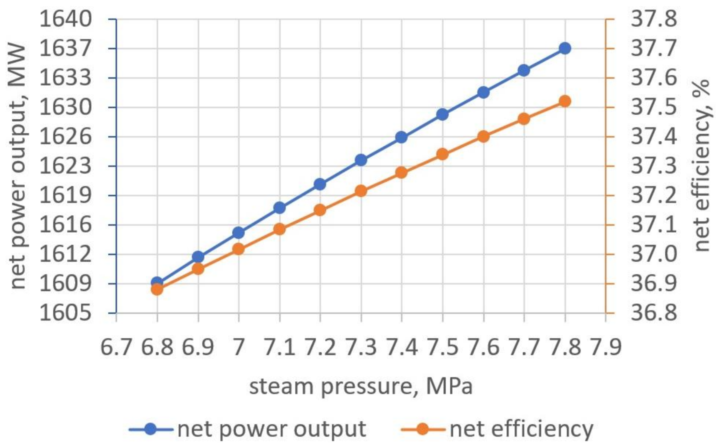

Figure 7: The effect of steam pressure on the net power output and efficiency of the system.

The highest entropy generation rate, equal to 7335.74 kW/K, was found at the nuclear reactor, while at other components of the thermal system it amounts to 1808.34 kW/K. The entropy generation rates and their percentages at each component of the thermal system are listed in Table 3. Since the entropy generation rate at the reactor is significantly higher than at the other components of the system, the percentages refer to the total entropy generation of the system excluding the entropy generation at the reactor to make it easier to interpret the entropy generation in the secondary system.

Table 3: Entropy generation rates at each component of the thermal system.

Component	$\dot{S}_{\text{gen}}$	Share
	kW/K	%
Steam Turbine	728.78	40.30
Heat exchangers including steam condenser	538.48	29.78
Steam generator	363.40	20.10
Valves	73.62	4.07
Pipes	70.03	3.87
Pumps	29.75	1.65
Mixers	4.28	0.24
Drains	0.00	0.00
Sum	1808.34	100.00

The second highest entropy generation rate of 728.78 kW/K occurs at the steam turbine. The total entropy generation rate of 538.48 kW/K was found at heat exchangers which comprise the steam separator-superheater (73.037 kW/K – SHI – 34.72 kW/K, SHII – 38.317 kW/K), the steam condenser (306.652 kW/K), LP and HP regenerative heat exchangers (97.469 kW/K and 40.841 kW/K, respectively), and the deaerator (20.481 kW/K). The next component of the system in terms of entropy generation is the steam generator where the entropy generation rate is 363.395 kW/K.

The percentage of the entropy generation rate at each component of the system excluding the entropy generation rate at the nuclear reactor can be found in Fig. 8.

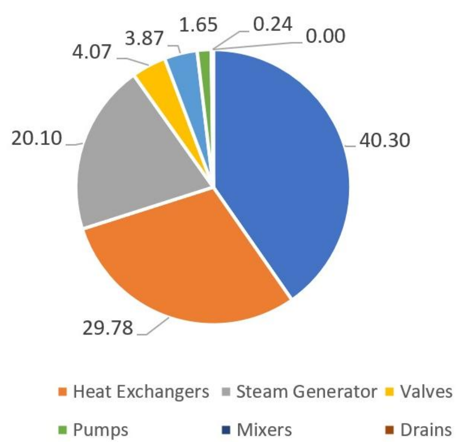

Figure 8: Percentage of the entropy generation rate at each component of the system excluding the entropy generation rate at the nuclear reactor.

The entropy generation analysis shows that the highest entropy generation rate occurs at the reactor which is followed by the steam turbine and the heat exchangers. These results match those available in the literature [47, 68]. Calculations indicate that the entropy generation rates at the valves and all pipelines in the system (Table 3, Fig. 8) are close to each other, which is confirmed by data found in [47]. The analysis shows that the entropy generation rate at the valves is higher than that at all the pipelines. Higher entropy generation will occur on the valves when throttling is present. In EPR, such throttling occurs on the main control valve (V1, Fig. 1) even at the full load of the power plant for operational reasons. One of the reasons is the need to ensure lower flow resistance on the valve when steam generator pressure decrease, which occurs during many years of operation. To provide a detailed analysis of the entropy generation rate at the valves, specific entropies at the inlet and downstream of the valves, mass flow rates across the valves, the entropy generation, and the entropy generation rates are listed in Table 4.

Table 4: Specific entropies upstream and downstream of the valves, mass flow rates across the valves, the entropy generation, and the entropy generation rates.

Valves	Specific entropy at the inlet, s_i	Specific entropy at the outlet, s_o	Mass flow rates across the valve, $\dot{m}$	Entropy generation, Δs	Entropy generation rate, $\dot{S}_{\text{gen}}$
	kJ/kg/K	kJ/kg/K	kg/s	kJ/kg/K	kW/K
V1	5.762	5.793	2312.82	0.03085	71.3521
V2	6.596	6.607	102.54	0.01118	1.1464
V3	1.774	1.775	1563.26	0.00056	0.8716
V4	0.947	0.954	17.48	0.00716	0.1251
V5	0.419	0.419	186.18	0.00067	0.1253
				Sum	73.6204

The entropy generation rate at the main control valve V1 indicates a relatively high level of steam throttling at this valve. The regulatory need to throttle the steam on the main control valve, increasing the flow resistance of fresh steam, causes a relative pressure drop from the steam generator to the turbine inlet, which increases as the pressure at the outlet from the steam generator is lowered. At a steam pressure of a steam generator of 7.8 MPa as in an EPR power plant, the relative loss is much lower than, for example, at a pressure of 5.76 MPa as in an AP1000 power plant.

4 Conclusions

The paper deals with the effect of steam pressure at the steam generator outlet (the turbine inlet) on the performance of the thermal system of an EPR nuclear power plant, i.e. on its power output, efficiency, and entropy generation. The choice of steam pressure has an impact on the efficiency of the unit, and the analysis of entropy generation allows us to locate losses in individual elements of the thermal system. The model of the nuclear power plant was developed using the Ebsilon software and validated based on data concerning the Olkiluoto Unit 3 EPR.

An increase in live steam pressure from 6.8 to 7.8 MPa leads to an increase in the gross power output from 1682.6 MW to 1714.6 MW, i.e. by 32 MW, and in the gross efficiency of 0.735 percentage point. In terms of the net power output and efficiency, the increment is 27.8 MW and 0.638 percentage point, respectively. The increase in efficiency is therefore moderate, which is due to factors such as the shape of the saturation curve (Fig. 5) for the analyzed range of pressures. The curve has its maximum enthalpy at a pressure of about 3.1 MPa. For higher pressures, the available enthalpy drop is increasing, but its increments are becoming smaller (Fig. 5).

If live steam pressures are higher, the shell, the tube assembly of the steam generator, and pipelines between the steam generator and the turbine need to have thicker walls, which translates into higher costs of construction. For the applied maximum coolant pressures in the primary circuit of 15.5–16.0 MPa, the live steam pressure of 7.8–8.0 MPa seems to be the limit pressure and is constrained by technical, economic and operating conditions in the primary circuit (a sufficient DNBR margin).

In practice, the efficiency of a power unit depends on steam pressure at the inlet to the turbine which is lower than the pressure generated in steam generators due to hydraulic resistance in pipelines and throttling in valves. On the basis of the entropy analysis performed, a relatively high generation of entropy was demonstrated at the control valve upstream of the steam turbine. Nevertheless, the high steam throttling in this valve is necessary due to ageing of the steam generator (declining steam pressure in the steam generator over the entire period of operation of the unit) and varying operational needs. It should be noted, however, that for a high live steam pressure the corresponding throttling losses across pipelines and valves between the steam generator and the turbine are relatively low (much lower than e.g. for reactors with a pressure of 6.0 MPa). The range of throttling in the main control valve in the EPR plant is adjusted to a change (drop) in pressure provided by the steam generator over the lifetime of the unit.

For the above reasons, the relatively high steam pressure of 7.8 MPa assumed for the EPR power plant ensures both high efficiency and flexible operation of the unit throughout the entire lifetime of the power plant.

The experience collected over many years of operation of power plants with PWR reactors, safety analyses, heat transfer studies, and optimization of the design of steam generators in EPR nuclear power plants made it possible to optimize the margin to the boiling (saturation) point in the primary circuit (ΔT_b), the drop in water temperature in the steam generator on the primary side (ΔT_{SG}), and the terminal temperature difference in the steam generator (TTD_{SG}) so that the current live steam pressure of about 7.8 MPa was achieved. This is the highest value of pressures used in new PWR reactors and it ensures that a high energy efficiency of a power unit is delivered and safety requirements are met. By taking advantage of the margin of the control range of steam pressure at the turbine inlet, which is controlled by the control valve (V1), one can maintain a high efficiency over long-term operation of an EPR power plant.

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