Chapter 8

Operation and Performance Analysis of Steam Generators in Nuclear Power Plants

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Additional information is available at the end of the chapter

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Abstract

Steam generators are components in which heat produced in the reactor core is transferred to the secondary side, the steam supply system, of the nuclear power plant (NPP). Steam generators (SGs) have to fulfil special nuclear regulatory requirements regarding their size, selection of materials, pressure loads, impact on the NPP safety, etc. The primary-side fluid is liquid water at the high pressure, and the fluid on the secondary side is saturated water-steam mixture at the pressure twice as low. A special attention is given to preserving the boundary between the contaminated water in the primary reactor coolant system and the water-steam mixture in the secondary system. A brief overview of the SG design, its operation and the mathematical correlations used to quantify heat transfer is given in the chapter. Results of the SG transient behaviour obtained by the simulation with the best-estimate computer code RELAP5, developed for safety analyses of NPPs, are also presented. Two types of steam generators are analyzed: the inverted U-tube SG, which is commonly found in the present-day pressurized water reactors and the helical-coil SG that is part of the new-generation reactor designs.

Keywords: steam generator, nuclear power plant, reactor safety, numerical analysis, SG operation, inverted U-tube SG, helical-coil SG

1. Introduction

Steam generators (SGs) are nuclear power plant components (NPPs) in which the steam, driving the turbine, is produced. They are heat exchangers where the heat produced in the reactor core is transferred to the secondary side, the steam system, of the nuclear power plant. Their other important function is to provide the barrier between the reactor coolant, which may be contaminated with radioactive fission products, and the environment. The thin tubes with a large surface area act both as heat transfer elements and fission product barriers. The priority of a nuclear safety philosophy is to maintain a sufficient water level in the SGs to avoid the damage of the tubes and release of radioactive fluid from the NPP. Thus, there are two functional requirements placed on the steam generators. The first is to act as a heat sink for the reactor core to prevent any core damage. The second functional requirement is to generate the flow rate of steam from the feedwater supply at the temperature, pressure and enthalpy conditions necessary to efficiently drive the steam turbine/electric generating system.

There are few steam generator designs depending on the type of the nuclear power plant [1]. Here, it is necessary to mention that the discussion inside the chapter is focused solely on heat exchangers in the so-called pressurized water reactor (PWR) NPPs where both the primary and secondary fluids are water. There are some other NPP types with gases, salts, or liquid metals used as reactor coolants, but they are not going to be presented. There are also light water reactor power plants where water boils directly inside the reactor core, boiling water reactors (BWRs), which do not have the steam generators at all, because the steam is produced in the reactor itself. Nonetheless, PWRs are the most common type of NPPs making up to 63% of the total number worldwide, while BWRs make up 18% of the total.

The most common design is the SG with the inverted U-tube heat exchanger bundle, where steam separation equipment is located inside the top shell of the SG [2]. The primary water flows upwards through the tubes first, making a bend, and then flows downwards back into the reactor coolant system (RCS) piping. On the secondary side, roughly a quarter of water, injected through the feedwater system, evaporates, and the remaining water is recirculated back into the boiling region; therefore, that type of SG is called recirculating steam generator (RSG). The American company Babcock and Wilcox (B&G) developed the once-through steam generator (OTSG), a vertical shell counterflow straight-tube design which directly generates superheated steam as the feedwater flows through the SG in a single pass. The primary-side water enters at the steam generator at the top, flows through the generator in unbent tubes and exits at the bottom. There is also a horizontal type of the SG which has the horizontal cylindrical housing and horizontal coils. The steam is dried at the top of the housing by gravitational separation. That type of SG is used mostly in Russian types of PWR reactors called vodo vodijanoj energetičeskij reactor (VVER). The last important design, to be later analyzed in the chapter, is the vertical SG with helical tubes, which has the intention to be used in the future advanced modular and high-temperature reactor systems [3]. The helical-coil design offers compactness and increased heat transfer. At the SG outlet, the steam is superheated, which results in higher thermal efficiency.

2. Description and operation of recirculating steam generators

2.1. Design configuration and flow

A recirculating steam generator consists of a bottom head of a hemispherical shape, the central cylindrical part where the heat exchange occurs and the upper part, the steam dome. The lower head is divided into inlet and outlet plenums by a vertically oriented plate. The primary coolant, after leaving the reactor, enters the lower plenum and then flows upwards through the tube bundle area, the riser, in the cylindrical part of the SG. Water evaporation takes place on the secondary side of the riser. The tube bundle is the interface between the primary and secondary circuits. It is composed of U-shaped tubes with approximate height of 11 m. Water in the reactor coolant system thus flows first upwards and then downwards through the tube sheet, transferring the heat to the secondary fluid. The cooled down water is then poured into the other half of the header tank before flowing back to the reactor. Primary coolant enters the steam generator at 588–603 K on the hot-leg side and leaves at about 560 K on the cold-leg side. The steam generator is shown in **Figure 1** and the schematic drawing of the flow paths inside the RSG in **Figure 2**. The main RSG characteristics and parameters are shown in **Table 1**.

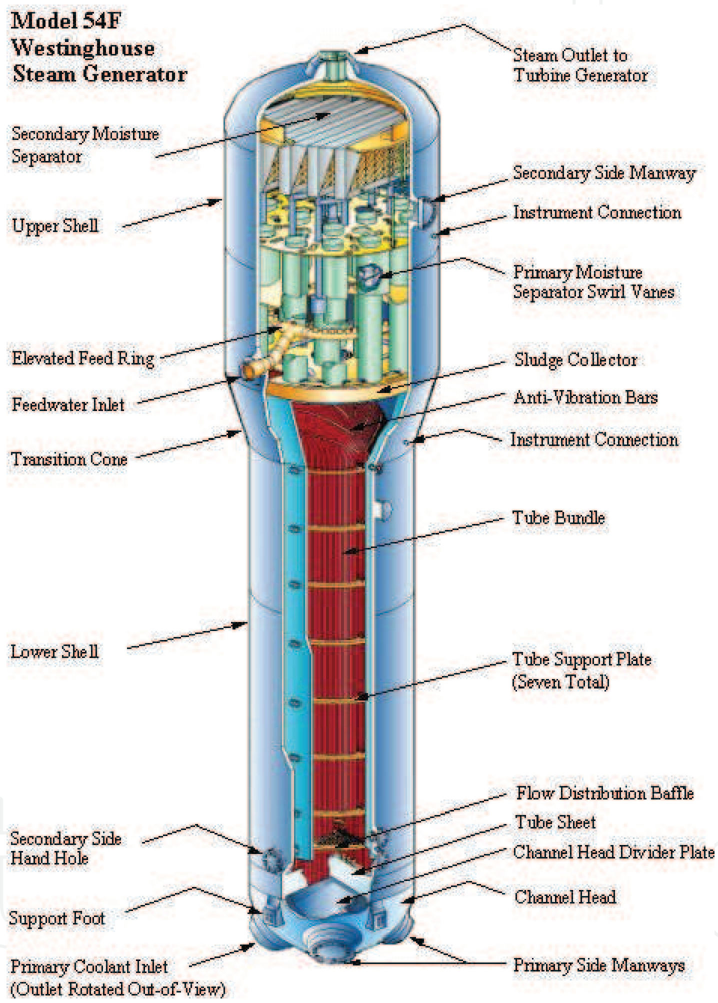

Figure 1. Recirculating steam generator (taken from Ref. [2]).

On the secondary side, feedwater enters the steam generator in the upper shell through a nozzle via a feed ring and is mixed with water draining from the moisture-separating equipment. Then, the water flows downwards through the downcomer to the tube sheet, a vertical plate separating the lower header and riser sections. The downcomer is situated between the tube bundle shroud and the SG outer shell. After reaching the tube sheet, water goes up, flowing below the shroud wall, in the central riser section where it is heated by the primary water flowing inside the U-tubes. Since the fluid in the secondary side, outside the tubes, is saturated, secondary-side water evaporates and a two-phase flow is established near the top of the tube bundle. The buoyancy force caused by a difference in densities of water inside the downcomer and the two-phase mixture in the tube bundle section ensures fluid circulation in the SG. As the secondary fluid flows upwards the riser, the steam quality increases up to 30%. This is not sufficient for a safe turbine operation as the droplets of water in the steam could damage the turbine blades. Thus, the steam that exits the tube bundle goes first into steam-moisture separators, composed of swirl vanes, and steam dryers, in the form of chevron separators, before it goes out of the steam generator through a nozzle at the top of the SG dome. The steam lines direct the steam flow into the turbine. The water collected by the separation devices falls down to the riser or is, for the most part, directed to the downcomer. After the drying process, the steam is saturated with a residual humidity, the moisture carryover, of less than 0.0025. Dry and saturated steam leaves the steam generator vessel and enters the steam lines.

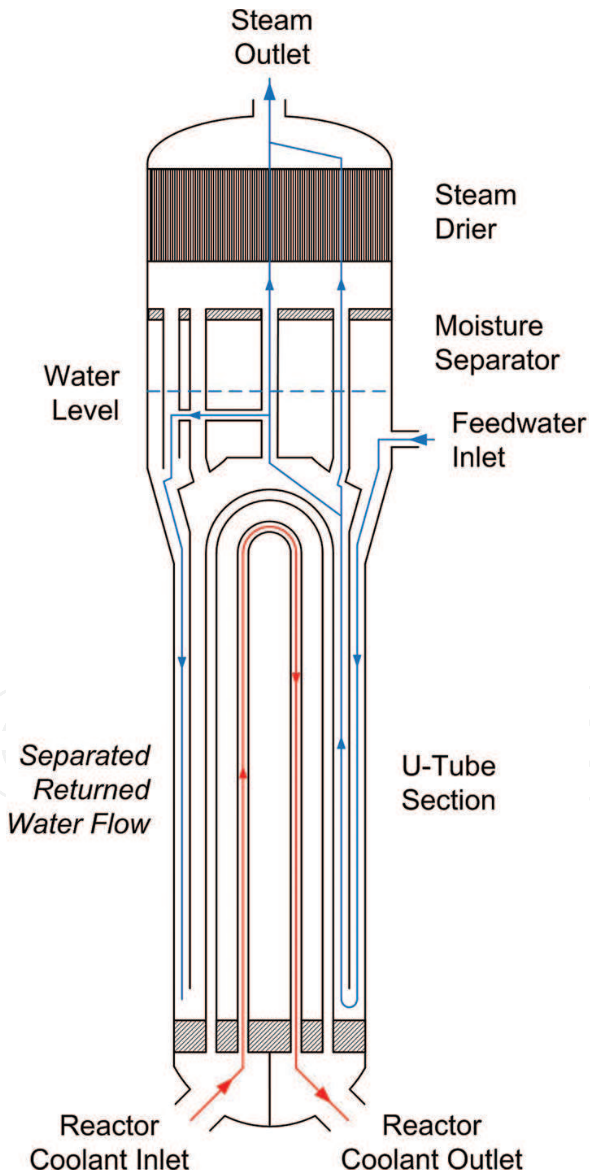

Figure 2. Simplified RSG flow paths.

Parameter	Value
Weight [tons]	330
Height [m]	20.7
Lower/upper diameter [m]	3.5/4.5
Heat transfer area [m^2]	7177
Number of U-tubes	5428
Tube inner/outer diameter [cm]	1.69/1.9
Power [MWt]	1000
Primary operating pressure [MPa]	15.5
Secondary operating pressure [MPa]	6.3
Primary inlet temperature [K]	598
Primary outlet temperature [K]	559
Secondary feedwater temperature [K]	492
Secondary steam temperature [K]	552
Primary coolant flow [kg/s]	4700
Secondary steam flow [kg/s]	545
Circulation ratio	3.7
Tubing material	Inconel 600

Table 1. Typical RSG dimensions and operating parameters.

About 25% of the secondary coolant is converted to steam on each pass through the generator, and the remainder is recirculated. That amount of recirculated coolant is described by the important design parameter called circulation ratio which is defined as the ratio between the total flow rate through the tube bundle and the flow rate of steam exiting the steam generator.

2.2. Regulation and control

In order to achieve the normal steam generator operation, a variety of parameters are subjected to regulation. Steam and feedwater flows need to be balanced; otherwise, the SG would overfill (if the feedwater flow is larger than the steam flow) or dry out (in the case the steam flow is larger than the feedwater flow). The amount of steam leaving the steam generator depends on the electrical load demand by the consumers. The turbine power is a linear function of the reactor coolant average temperature. The reactor temperature regulation system maintains the temperature by adjusting the position of the reactor control rods taking into account the signal of the turbine power. For the fast load changes, the excess steam is directed directly into the turbine condenser, thus bypassing the turbine. In that way, the steam generator pressure is being kept below the safety limits. Additionally, for the SG pressure control, power-operated relief and safety valves, mounted on the steam lines, are used. The relief valves are motor-operated valves, while safety valves are passive components.

The most important SG parameter subjected to regulation is the SG level. If the level is too low, the insufficient heat removal by the secondary side may cause evaporation of the reactor coolant, thus overheating of the reactor core. On the other hand, if the level is too high, the steam exiting the steam generator would carry water droplets (the void fraction would be higher than zero) which can be damaging to the turbine. The SG level is maintained by the feedwater flow by means of controller which continuously compares measured feedwater flow with steam flow and a compensated steam generator downcomer water level signal with a water level set point. A functional diagram of the steam generator level control system is shown in **Figure 3**.

The measured steam generator level is compensated by a lag controller ($1/(1+\tau_1 s)$) and subtracted from a desired reference SG level. That signal is then corrected by a proportional-integral (PI) controller ($K_1(1+1/\tau_2 s)$) and added to a difference between steam flow and feedwater flow signals. The resulting signal goes through a final PI correction ($K_2(1+1/\tau_3 s)$) before being used for feedwater flow control. Parameters K_1 and K_2 are scaling factors and τ_1, τ_2 and τ_3 time constants depending on the design of nuclear power plant control system.

According to **Figure 3**, in the case the reference level signal is larger than the measured level or the steam flow is larger than the feedwater flow, the feedwater flow will be increased by increasing the feedwater control valve area. In the opposite case, the control valve flow area will be decreased.

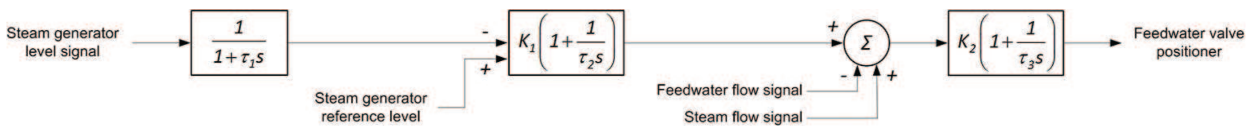

Figure 3. Functional diagram of the SG level control system.

Two types of water levels are measured inside the steam generator: the narrow range (NR) level and the wide range (WR) level. The term “water level” should not be taken literally since no free water surface in the SG secondary side can be established. The fluid is in a state of boiling, and in the area above the tube bundle, steam quality steadily rises from the top of the U-tubes to the inlet in the steam separators. Thus, the level is deduced from the pressure difference, pressures being measured at two different heights. The level is affected by variations in the fluid density as well as residual pressure drops.

In general, the SG level is a measure of a pressure difference inside the steam generator compared to a pressure difference between the liquid and gas phases. It is calculated by the expression:

$$SGLVL[\%] = 100 \cdot \frac{\Delta p - \Delta p_{0\%}}{\Delta p_{100\%} - \Delta p_{0\%}} \tag{1}$$

where $\Delta p_{0\%} = \rho_g gh$ and $\Delta p_{100\%} = \rho_l gh$.

The height in the expressions for the pressure difference is the distance between the measurement taps. For both the narrow and wide range measurements, the upper tap is in the

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separator area. The lower tap for the NR level measurement is just below the U-tube bend area, near the top of the downcomer. For the WR level measurement, the lower tap is at the bottom of the downcomer. The total height for the NR level measurement is about 5 m and for the wide range 15 m.

The narrow range level is used to control the feedwater flow rate. Except in the case of extreme events, such as very fast transients caused by accidental depressurization, the narrow range level is maintained constant by the control system. The feedwater flow rate and temperature, and thermal hydraulic SG conditions, have much larger influence on the wide range level. Therefore, the WR level is only used as a level indicator for the NPP operators during slow running transients or during the plant shutdown and start-up operation modes after an outage. Dependence of the WR level on the dynamic SG pressure prevents it to be used for the control of the NPP performance [4].

3. Numerical analysis and performance results of the RSG

The mathematical simulation of the SG normal and transient operation and behaviour was performed with system computer code RELAP5 [5] on the example of a two-loop PWR power plant Krško [6].

Correlations for the heat transfer calculation in steam generators are given first. Then, SG modelling requirements and mathematical models of the plant are presented, followed by analytical results of SG performance in accident conditions during the loss of the NPP heat sink.

3.1. Physical models of thermal hydraulic and heat transfer phenomena

3.1.1. Thermal hydraulic conditions

The RELAP5 code is a six-equation one-dimensional, nonhomogeneous, nonequilibrium transient system code. It solves mass, momentum and energy conservation differential equations for the two phases, gas and liquid, hence, the six conservation equations. The equations will not be presented here because they are standard fluid conservation equations, although they include many fine transport mechanisms in order to realistically simulate thermal hydraulic system behaviour [5]. For example, momentum equations take into account wall friction, momentum transfer due to interface mass transfer, interface frictional drag and force due to virtual mass. The interface mass and heat transfer terms are also treated by the mass and energy conservation equations.

Efficient and reliable SG operation requires efficient steam separation. Separators must be capable of achieving very low moisture carryover. High carryover will result in turbine efficiency losses as well as the potential for turbine blade erosion. Efficient steam separator design also requires that the primary separation stage has the low pressure drop and low steam carryunder in the downcomer flow in order to support efficient recirculation through the tube bundle. Furthermore, to allow flexibility in water level operation, the separators must be able to operate over a wide range of water levels.

The computational separator model consists of a special hydraulic volume component with junction flows (**Figure 4**). A steam-water inflowing mixture is separated by defining the quality of the outflow streams using empirical functions. The void fraction of the vapour at the separator outlet junction J_1 depends on the void fraction in the separator thermal hydraulic volume according to the left curve in **Figure 5**. If the vapour void fraction in the separator volume is larger than the input parameter, labelled as VOVER, the outlet void fraction will be 1.0, and pure gas vapour will be released out of the separator. If the separator vapour void fraction is less than the value of VOVER, then the outflow is going to be a two-phase mixture of gas and liquid. Thus, changing the VOVER parameter, the code user can control the state of fluid leaving the separator. In the case of VOVER parameter being small, the separator is going to act as an ideal separating device, discharging pure vapour, and in the case of VOVER having a high value, close to 1.0, the separator component will perform as a standard junction releasing fluid in the same state as is entering the separator.

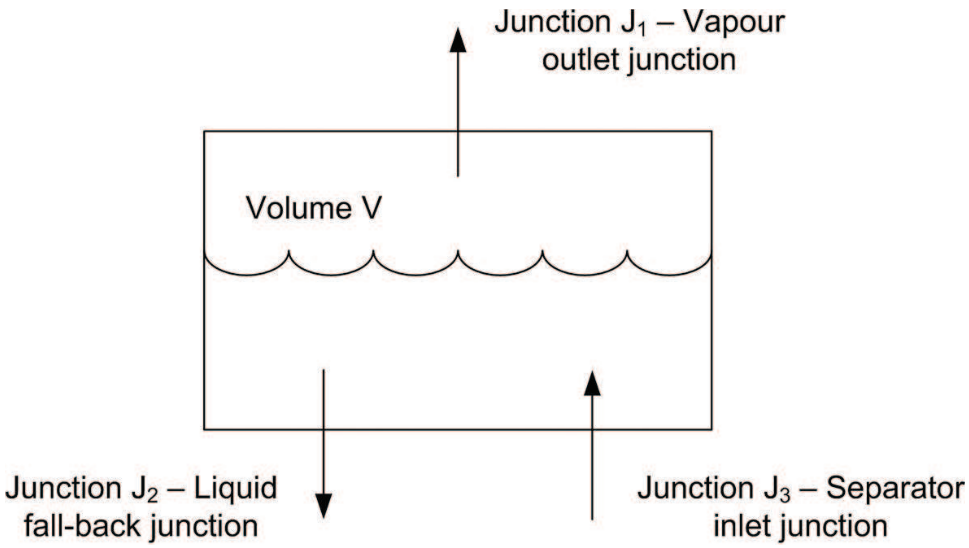

Figure 4. Typical separator volume and junctions.

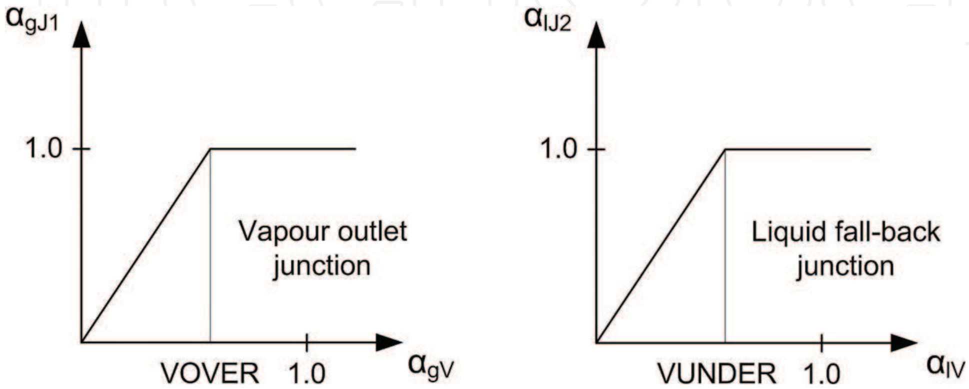

Figure 5. Outlet junctions void fractions.

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The flow of the separator liquid drain junction is modelled in a manner similar to the steam outlet except that pure liquid outflow is assumed when the volume liquid void fraction is greater than the value of VUNDER (**Figure 5**). Typical values of VOVER and VUNDER parameters are 0.001 and 0.1, respectively.

3.1.2. Correlations for the heat transfer calculation

Steam generator operation depends on the wall-to-fluid heat transfer on the secondary side. During the steady-state operation, the steam generator water level is constant, but during the transient, it can vary in the large range; thus, both convective and boiling heat transfers occur across the U-tubes. The total wall heat flux is composed of convective heat transfer to gas and liquid phases, boiling and condensation heat fluxes. During boiling, the saturation temperature based on the total pressure is the reference temperature, and during condensation, the saturation temperature based on the partial pressure is the reference value. The expression for a heat flux is given as

$$q_{\text{wall\_total}} = h_{g,g}\left(T_w - T_g\right) + h_{g,spt}\left(T_w - T_{spt}\right) + h_{g,spp}\left(T_w - T_{spp}\right) + h_{l,l}\left(T_w - T\right)_l + h_{l,spt}\left(T_w - T_{spt}\right), \tag{2}$$

where

- $h_{g,g}$ is the heat transfer coefficient (HTC) from the wall to the gas phase, and the reference temperature is the gas temperature.
- $h_{g,spt}$ is the HTC from the wall to the gas phase, and the reference temperature is the saturation temperature at the total fluid pressure.
- $h_{g,spp}$ is the HTC from the wall to the gas phase, and the reference temperature is the saturation temperature at the partial pressure of steam.
- $h_{l,l}$ is the HTC from the wall to the liquid phase, and the reference temperature is the liquid temperature.
- $h_{l,spt}$ is the HTC from the wall to the liquid phase, and the reference temperature is the saturation temperature at the total fluid pressure.
- T_w, T_g and T_l are the wall surface, gas and liquid temperatures, respectively.
- T_{spt} and T_{spp} are the saturation temperatures at the total fluid pressure and at the partial pressure of steam, respectively.

Correlations used to calculate heat transfer in the steam generators, depending on the heat transfer regimes, are given in **Table 2**.

For the convective heat transfer, the correlation is given by Churchill and Chu [7]:

$$Nu = \left(0.825 + \frac{0.387 Ra^{1/6}}{\left(1 + \left(\frac{0.492}{\Pr}\right)^{9/16}\right)^{8/27}}\right)^2, \tag{3}$$

where Nu, Ra and Pr are Nusselt, Rayleigh and Prandtl numbers, respectively.

Heat transfer phenomena	Correlation
Convection to noncondensable steam-water mixture, supercritical, single-phase liquid or gas flows	Churchill and Chu [7]
Subcooled or saturated nucleate boiling	Forster and Zuber [8]
Subcooled or saturated transition boiling	Chen et al. [9]
Subcooled or saturated film boiling	Bromley [10]
Condensation heat flow	Nusselt [11]

Table 2. Wall heat transfer correlations.

The Forster-Zuber correlation [8] for the nucleate boiling heat transfer coefficient is given as

$$h = 0.00122 \frac{k_l^{0.79} c_{pl}^{0.45} \rho_l^{0.49} g^{0.25}}{\sigma^{0.5} \mu_l^{0.29} h_{\mathrm{lg}}^{0.24} \rho_g^{0.24}} \Delta T_w^{0.24} \Delta p^{0.75}, \tag{4}$$

where k is the thermal conductivity, c_p the specific heat, ρ the density, g the gravity acceleration, σ the surface tension, μ the viscosity, h_{lg} the enthalpy of boiling, ΔT_w the difference between the wall and fluid temperatures and Δp the difference between the saturation and total pressures.

The Chen transition boiling model [9] considers the total transition boiling heat transfer to be the sum of individual components, one describing wall heat transfer to the liquid and a second describing the wall heat transfer to the vapour. The model is expressed as

$$q = q_l A_l + 0.0185\, \mathrm{Re}^{0.83}\, \mathrm{Pr}^{1/3} (T_w - T_g)(1 - A_l), \tag{5}$$

where q is the transition boiling heat flux and A_l is the fractional wall-wetted area.

The Bromley [10] correlation for the heat transfer coefficient during film boiling is given as

$$h = 0.62\alpha \left(\frac{g \rho_g k_g^2 (\rho_l - \rho_g) h_{\mathrm{lg}} c_{pg}}{L(T_w - T_{spt})\, \mathrm{Pr}_g} \right)^{0.25}, \tag{6}$$

where α is the void fraction and L the characteristic length.

Finally, the Nusselt [11] correlation for the condensation heat transfer coefficient is given as

$$h = 1.1006\, k_l \left(\frac{g \rho_l \Delta\rho}{\mu_l^2\, \mathrm{Re}_l} \right)^{1/3}. \tag{7}$$

3.2. Numerical simulation: the NPP model and results

3.2.1. The RELAP5 computational model of the power plant

The nodalization scheme (mathematical model) of the two-loop PWR nuclear power plant, analyzed herein, is shown in **Figure 6**. The boxes represent hydraulic control volumes (CVs) connected by junctions. The reactor is in the middle of the figure, connected by pipes with two steam generators (SG1 and SG2). The scheme also includes other important NPP components and systems: the pressurizer; the safety injection system; the main feedwater; and auxiliary feedwater systems, steam lines, etc.

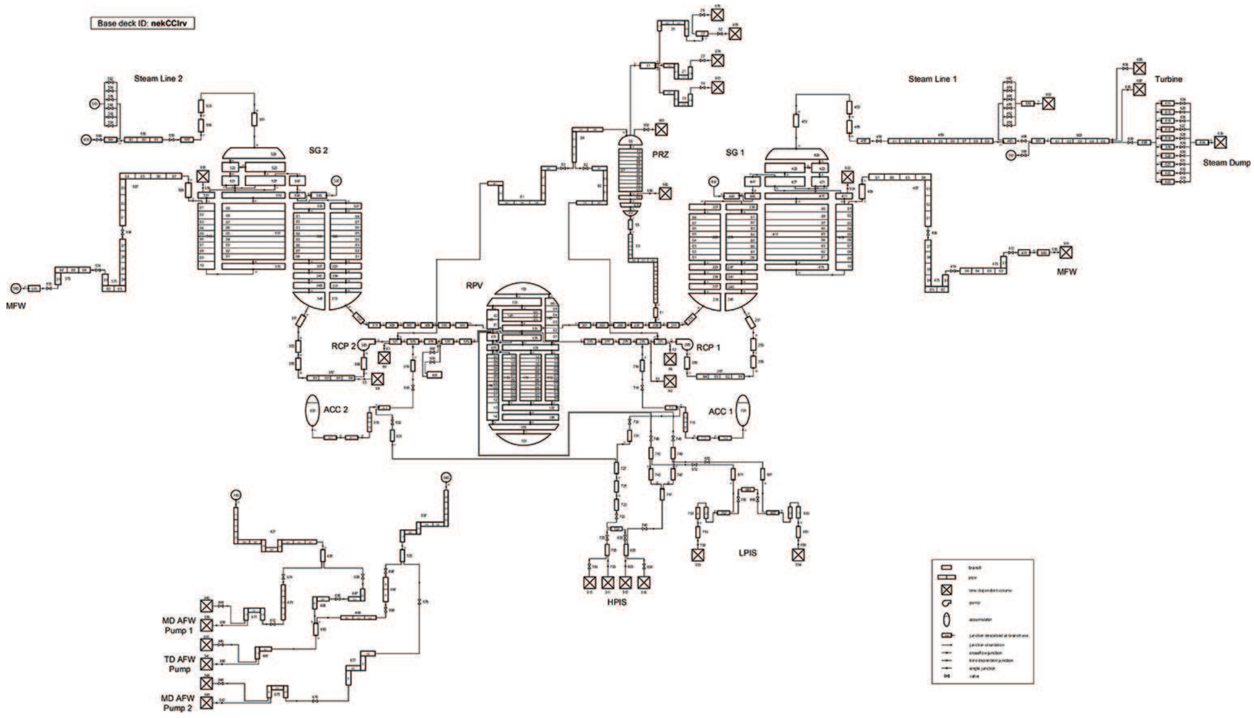

Figure 6. RELAP5 nodalization of the NPP Krško.

Regarding the steam generator model, on the primary side, the inlet plenum is represented with two control volumes (215 and 217), the tube sheet inlet with one CV (219), the U-tube section with 20 control volumes (upward part of U-tubes—CVs 223, 22501–22508, 227, and downward part of U-tubes—CVs 233, 23501–23508, 237), the tube sheet outlet with one CV (241) and the outlet plenum with two CVs (243 and 245).

On the secondary side, the downcomer is represented with 11 control volumes (411, 41301–41310), the riser section also with 11 CVs (415, 41701–41709+-, 419), the separator with one CV (421), the steam plenum with dryer with one CV (423), the separator bypass region with two CVs (425, 427) and, finally, the steam generator dome with one control volume – 429. This is the model of the SG1. The SG2 has the same model with the different numbering.

3.2.2. Steady-state calculation

The first step in the NPP and the SG model development is the qualification of the code nodalization. This is done by comparing plant's main operating parameters with computer steady-state simulation at full power. Parameters of interest are primary and secondary system pressures, reactor coolant, feedwater and steam mass flow rates, reactor coolant temperatures, pressurizer and steam generator liquid levels, primary and secondary system fluid masses, heat transfer inside SGs, circulation ratio, etc. If the calculated values differ less than approximately 1% than the real plant data, we can say that nodalization is qualified for the plant safety analyses.

The steam generator qualification process of the RELAP5 model includes calculation of pressures, temperatures, fluid flows and liquid levels inside the SG. Additionally, geometrical representation of the computational model and calculation of SG conditions at partial loads need also to correspond to manufacturer data, as shown in **Figures 7** and **8**, respectively. The SG pressure at a lower load is higher than the pressure at a higher load. As the pressure increases, water evaporates at a slower rate, and a total SG secondary fluid mass is 50% higher at a 10% load than at a 100% load. The steam flow at a full load is 540 kg/s, while at a 10% load, the flow is only 40 kg/s, achieved by increase of pressure and decrease in feedwater temperature. The pressure difference of 1 MPa, as observed in **Figure 8**, results in steam temperature change of only 10 K. The highest impact of load change is on the circulation ratio which decreases from the value of 41 at a 10% load to a value of 3.7 at a full load.

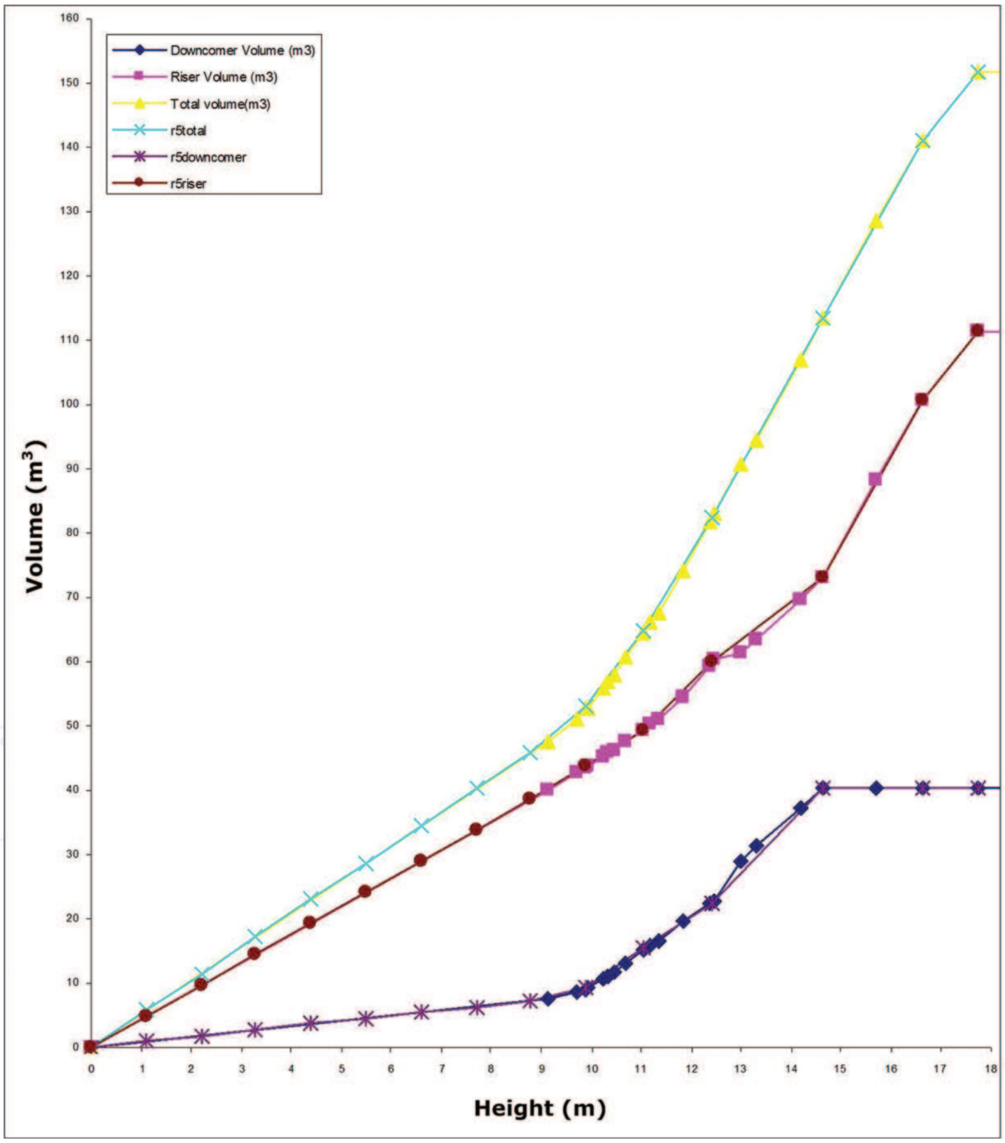

Figure 7. SG secondary side volume versus height.

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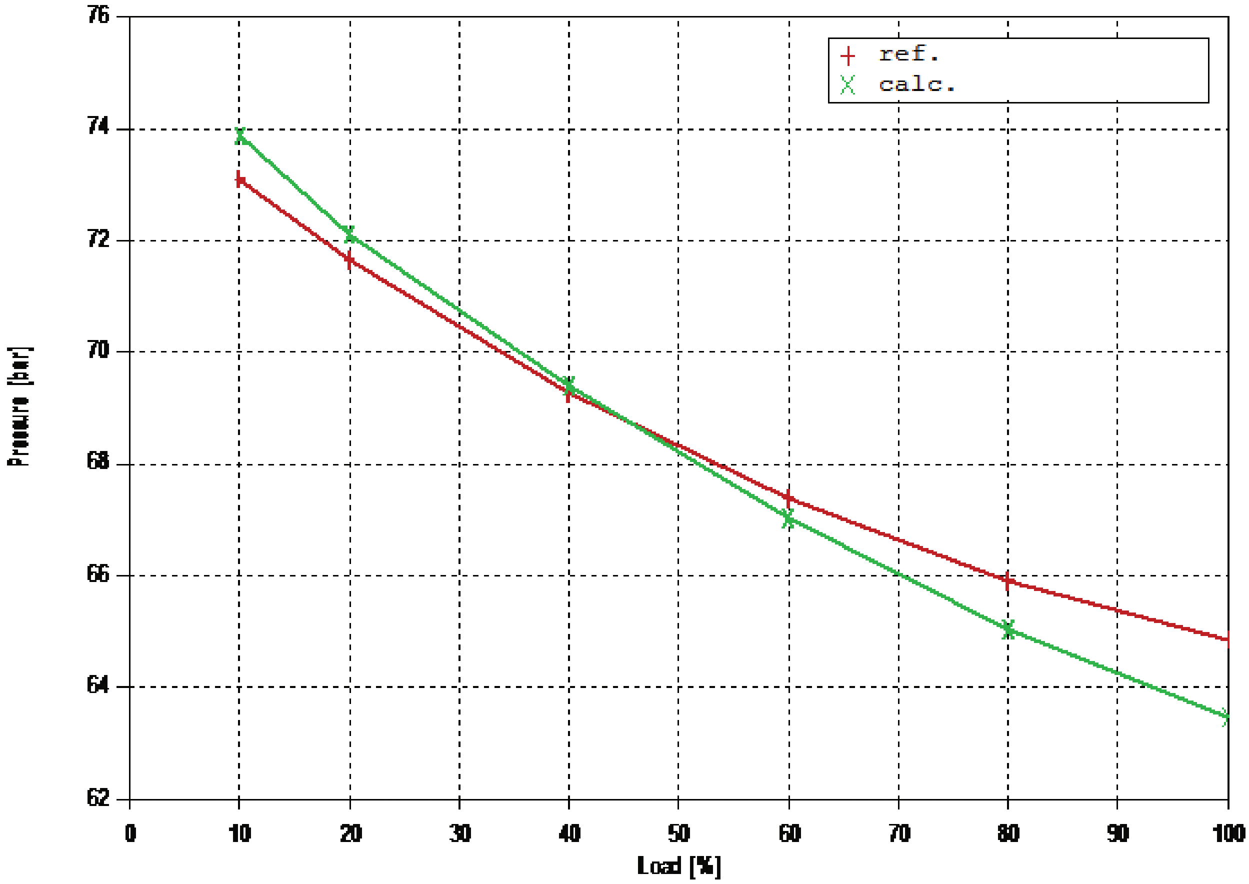

Figure 8. SG steam pressure versus load.

3.2.3. Transient calculation

In order to illustrate SG dynamic behaviour, a representative accident of the loss of all electrical power was selected. The unavailability of AC power supply to NPP systems means that important operational and safety components, such as big pumps which provide driving force to primary and secondary system coolant flows, will not work. There will not be feedwater flow into the SG, and the water level will decrease. On the primary side, the reactor core will heat up and subsequently, if in the meantime, no power restoration occurs, melt. Plant conditions will be further complicated by the loss of reactor coolant through damaged reactor coolant pump seals. The seals are normally cooled by the high-pressure water flow provided by the charging pump which, without electrical motor drive, does not operate.

There are two ways of mitigating accident consequences. First, there is a passive steam-driven auxiliary feedwater (AFW) pump that can provide water for the cooling of steam generators. The steam is taken directly from the SG outlet. In addition, the plant operator can reduce pressure by controlling SG relief valves and prolong the time to the core damage because of increased cooling of the reactor coolant system (RCS). Power supply needed for those actions is provided by the accumulator batteries installed at the plant. The three scenarios ((1) no AFW flow, no SG depressurization; (2) AFW flow, no SG depressurization; (3) AFW flow, SG depressurization) were analyzed according to the aforementioned mitigating options.

The AFW system water injection provided secondary-side heat sink. The SG pressure was maintained at 8 MPa by the operation of SG safety valves (**Figure 9**). Natural circulation was established in the primary system after the stoppage of coolant pumps, heat source being the core decay heat and heat sink of the steam generators. Primary system water evaporated in the core, and steam condensed in the SG U-tubes. The heat (**Figure 10**) was transferred to the secondary-side boiling water which level (**Figure 11**) was maintained by injection from the AFW system. Oscillatory behaviour was due to operation of safety valves, and continuous short-term steam releases to keep pressure at 8 MPa. Large condensate storage tank (CST) water inventory (860 m^3) provided AFW flow for almost 70 h. Depletion of CST inventory led to dry out of the SGs. The CST tank could be filled up at any time during the accident, but in this conservative analysis, no provision of maintaining the CST water inventory was taken into account. Soon after the CST depletion, the RCS was heated up, water boiling in the core accelerated and the core started to uncover. If there is no immediate action to inject water in the core, the core will melt.

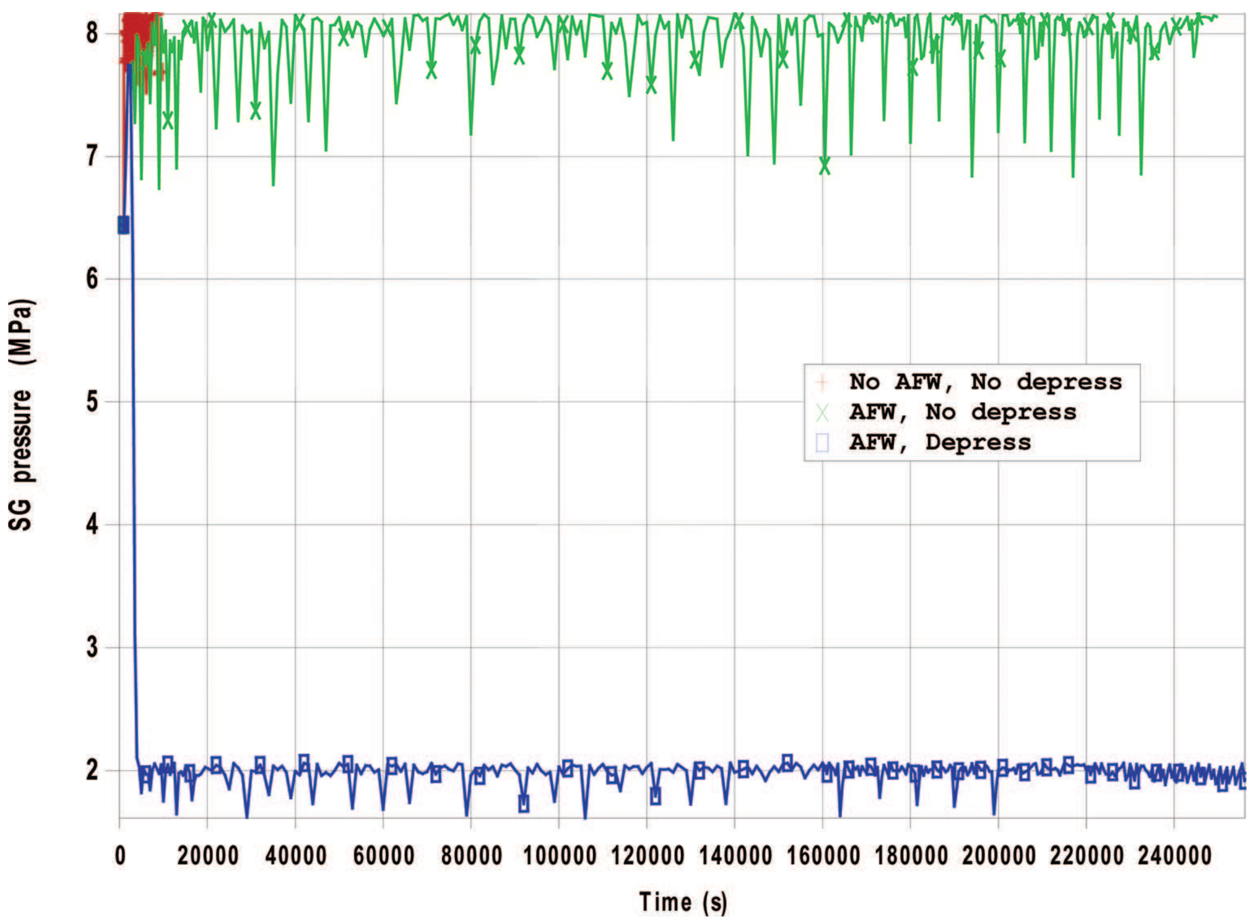

Figure 9. SG pressure.

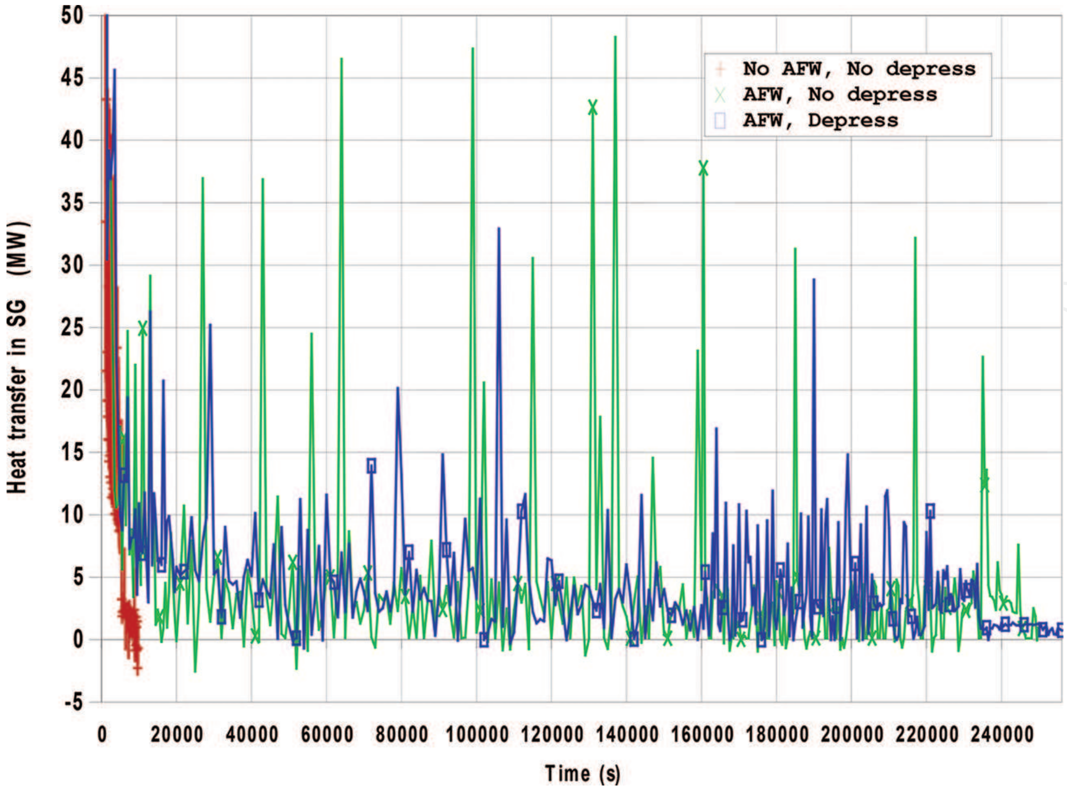

Figure 10. Heat transfer in steam generators.

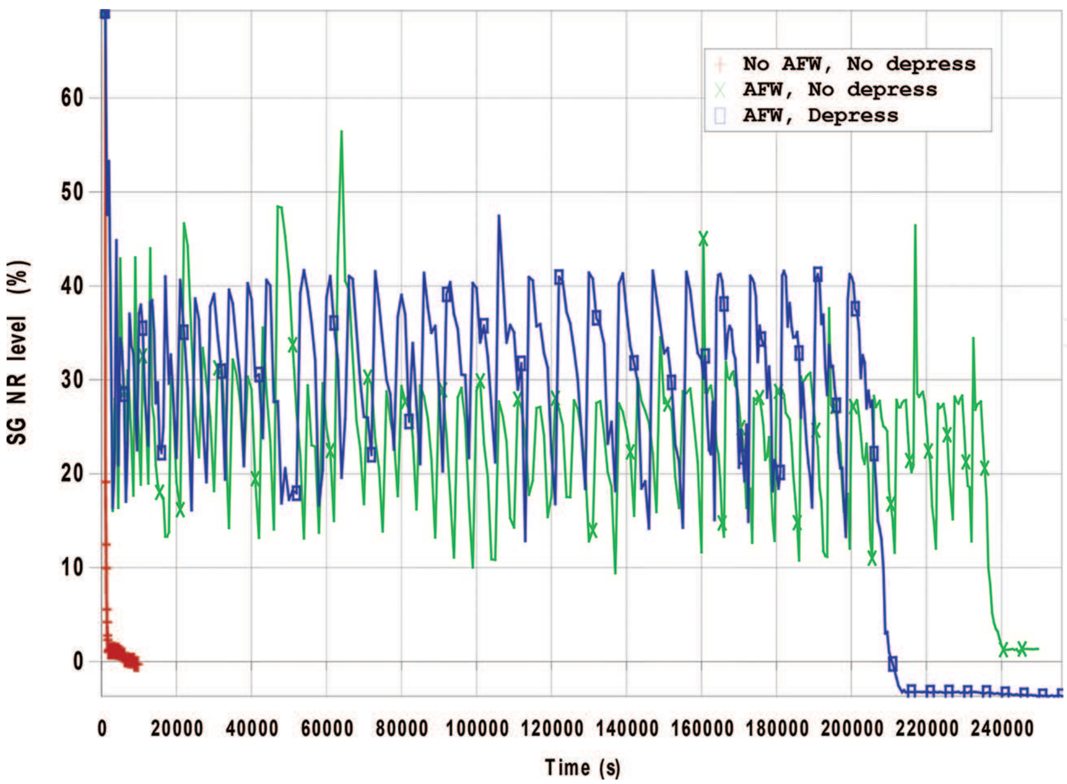

Figure 11. SG narrow range water level.

Operator action to rapidly depressurize secondary side to 2 MPa using SG relief valves led to fast primary-side cooldown and depressurization. The primary and secondary systems were closely coupled with little difference in pressures and temperatures. Both systems were in saturated conditions, temperature depending on the saturation pressure. Since the core decay heat was only a couple of percents of the total power and the heat transfer area in the SG was very large, the temperature difference was only few kelvins, and, thus, pressures were almost the same. With the operator action, the CST was emptied at ~202,500 s. That was 30,000 s earlier than in the case without any operator actions due to higher AFW flow required during SG steam extraction to satisfy prescribed RCS cooldown rate. Reducing the primary pressure to 2 MPa enabled the actuation of accumulators' water injection which quenched the core. Therefore, although secondary-side heat sink was lost earlier, the larger RCS water inventory increased the time to core damage. In the case with no AFW flow, the core heat-up started after 2.2 h. In the case with the AFW availability, the core temperatures started to increase after 65 h for the case without SG depressurization, and after 70 h when the operator controlled the SG pressure.

4. Helical-coil steam generators

4.1. Description and main characteristics

Many future nuclear reactor projects, especially innovative small- and medium-sized reactor systems [12], are expected to use helically coiled pipes for the steam generators. Their favourable characteristics justify the helical tube SG development in the nuclear field. In particular, helical tubes provide enhanced heat and mass transfer rates, a higher critical heat flux during boiling and evaporation and a better capability to accommodate the thermal expansion, in addition to allow a more compact design of the SG. Helical coils are particularly attractive for small and medium modular reactors (SMRs) since many of them adopt an integral layout. Compactness and efficiency improvement become particularly important for this type of reactors, as all the primary system components are located inside the reactor vessel.

The helical coil SG design and operation will be explained on the example of the IRIS reactor [13]. The international reactor innovative and secure (IRIS), an integral, modular, medium-sized (335 MWe) PWR, has been under development since the turn of the century by an international consortium led by Westinghouse and including over 20 organizations from nine countries. IRIS features an integral vessel that contains all the major reactor coolant system components, including the reactor core, pumps, the steam generators and the pressurizer. The unique IRIS safety-by-design approach provides a very powerful first level of defence in depth approach by eliminating accidents, at the design stage, or decreasing their consequences and probabilities when outright elimination is not possible. There are no primary system pipings, and a large-break loss-of-coolant accident (LOCA), related to a double-ended break of a primary leg pipe, is avoided. The passive safety systems increase plant safety by providing core decay heat removal during accident conditions even where there is no electrical power supply.

Steam generators are located in the space between the core barrel, precisely the shroud enclosing the riser section, and the reactor vessel wall. There are eight SG units in total, designed as once-through heat exchanging units (**Figure 12**). They are made of helical tubes with

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secondary fluid flowing inside the tubes. The feedwater header is located at the bottom of the SG module, while the steam header is positioned at the top of the steam generator. The tubes are wrapped around the inner supporting column. The primary cooling water flows outside of the tubes, through the tube bundle. The primary reactor coolant pumps are installed above the steam generators and drive coolant from the top to the bottom of the SG. Thus, a counter-current flow regime is developed inside the SG, the primary coolant flows from the top to the bottom of the SG and the secondary fluid flows in the opposite direction.

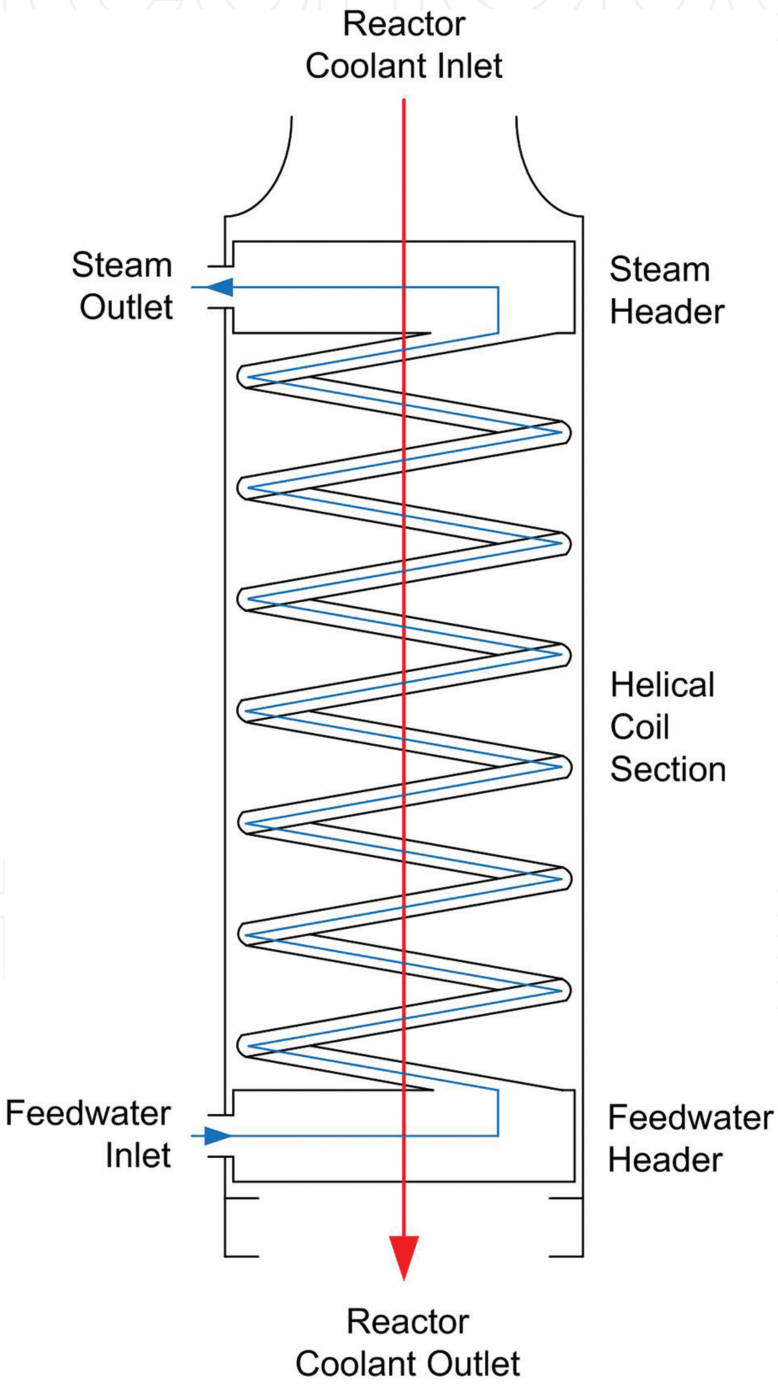

Figure 12. Simplified IRIS helical-coil SG flow paths.

The tubes are set up in annular rows and connected to steam and feedwater headers (**Figure 13**), which, on the other hand, are connected to feedwater and steam lines piping by nozzles mounted on the external surface of the reactor vessel wall. At the tube inlet, orifices which reduce fluid flow are installed in order to maintain uniform flow distribution across the tubes and to prevent parallel channel flow instabilities. The pressure drops at these orifices are of the same order of magnitude as the pressure drops in the tubes. The main IRIS reactor helical-coil SG characteristics and parameters are shown in **Table 3**.

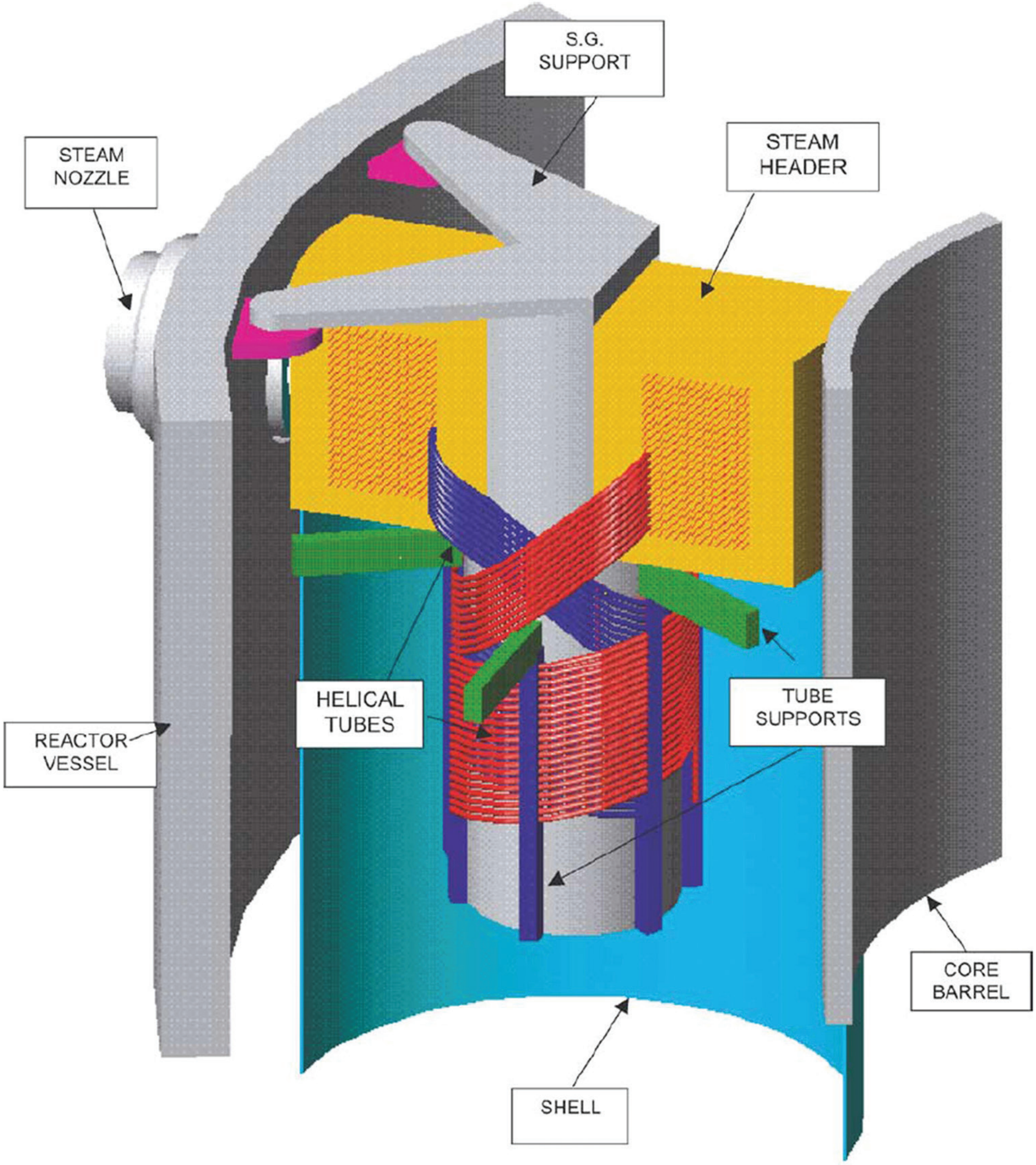

Figure 13. IRIS SG steam header (taken from Ref. [13]).

Parameter	Value
Height [m]	8.5
External shell inside diameter [m]	1.62
Internal shell outside diameter [m]	0.61
Number of helical rows	21
Number of tubes	655
Tube bundle average length [m]	32
Tube inner/outer diameter [cm]	1.32/1.75
Power [MWt]	125
Primary operating pressure [MPa]	15.5
Steam outlet pressure [MPa]	5.8
Primary inlet temperature [K]	602
Primary outlet temperature [K]	565
Secondary feedwater temperature [K]	497
Secondary steam temperature [K]	590
Primary coolant flow [kg/s]	589
Secondary steam flow [kg/s]	62.5
Tubing material	Ni-Cr-Fe alloy

Table 3. IRIS helical-coil SG dimensions and operating parameters.

Primary and secondary fluid temperature profiles are shown in **Figure 14**. The primary fluid is water, and its temperature continuously increases along the height of the SG (decreases in the flow direction). On the secondary side, subcooled water enters the tubes, heats up, boils and produces superheated steam in the upper part of the tubes. During boiling, steam temperature slightly decreases due to a small pressure drop across the tubes. The total pressure drop on the secondary side, from the feedwater inlet to the steam outlet, is 323 kPa and in the boiling part of the tubes 140 kPa. The void fraction at point when superheating starts is 0.98.

4.2. Computational model and numerical simulation of the IRIS reactor

The RELAP5 code was used to model the facility and simulate accident conditions. Although the code lacks appropriate correlations for the heat transfer coefficient in helical pipes, comparison with detailed calculation [14] shows a good agreement between the results. The worldwide experience and quality models inside the RELAP5 were the reasons for selecting it to perform IRIS preliminary safety assessment studies [15].

The nodalization of the IRIS nuclear power plant is shown in **Figure 15**. On the left side of the figure, the reactor core is in the lower part (CV 110), the riser in the middle part (CVs

120–123) and pressurizer in the upper part (CV 130). On the right side, the pump (CV 191) is below the pressurizer and connected to the steam generator (CV 211). The SG annular dead space surrounding the SG shell and the inner inactive region are modelled as CV 240 and CV 241, respectively. The lower downcomer region (CV 101) is below the steam generator. The secondary side of the steam generator is modelled with CV 271. All eight steam generators are modelled, but only one is presented in the figure. The primary side of the SG was represented with 25 control volumes and the secondary side with 50 control volumes. That fine nodalization was obtained during the optimization process that resulted with stable and correct steady-state performance. Parallel channel flow instabilities were eliminated by modifying pressure loss coefficients. The heat transfer coefficient was also adjusted using multiplication factor to accommodate the inability of the RELAP5 to model curved helical-coil geometry. In that geometry eddy flow currents are created inside the coils, which promote mixing of the fluid and, thus, increase heat transfer capability at the expense of higher pressure drops [16].

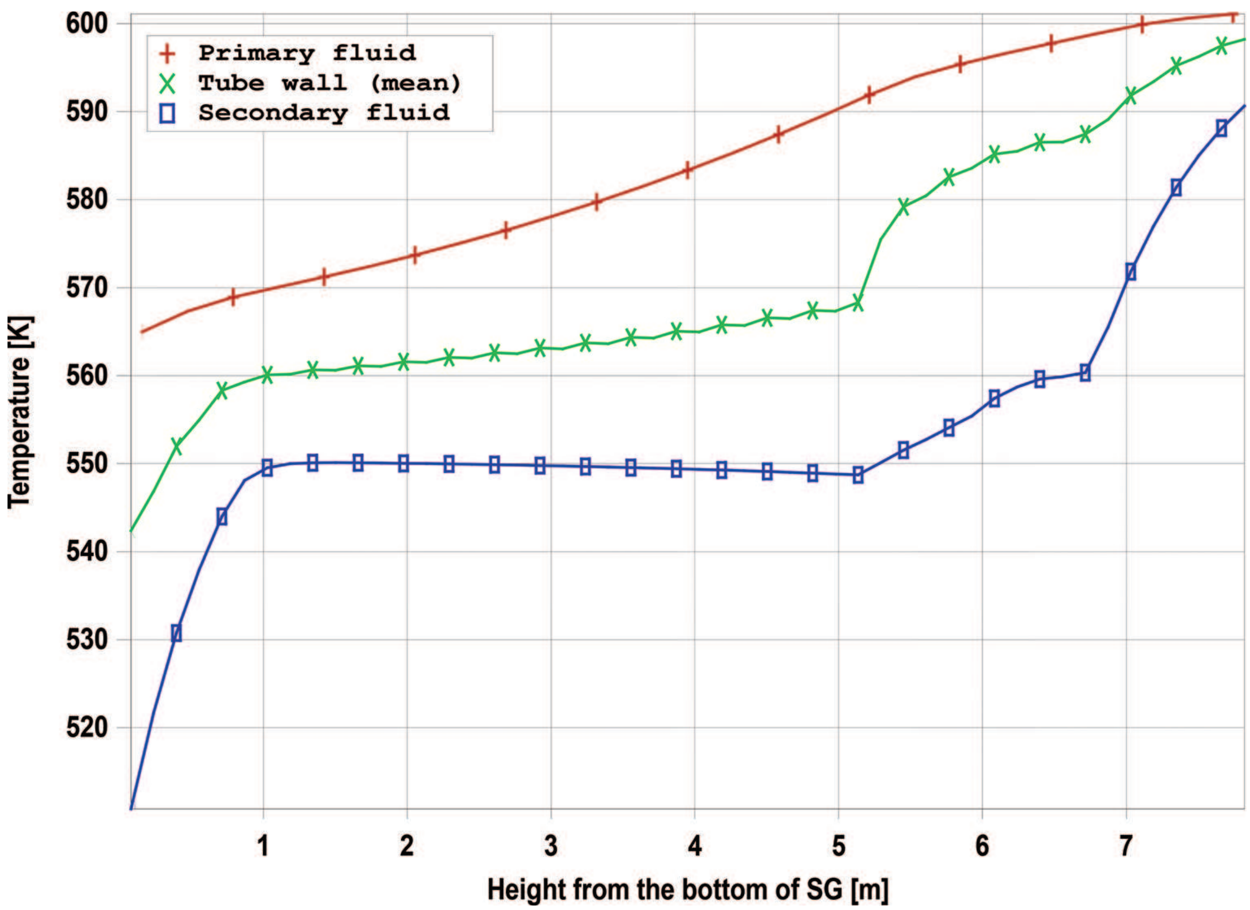

Figure 14. Primary and secondary fluid temperature profiles in the IRIS NPP.

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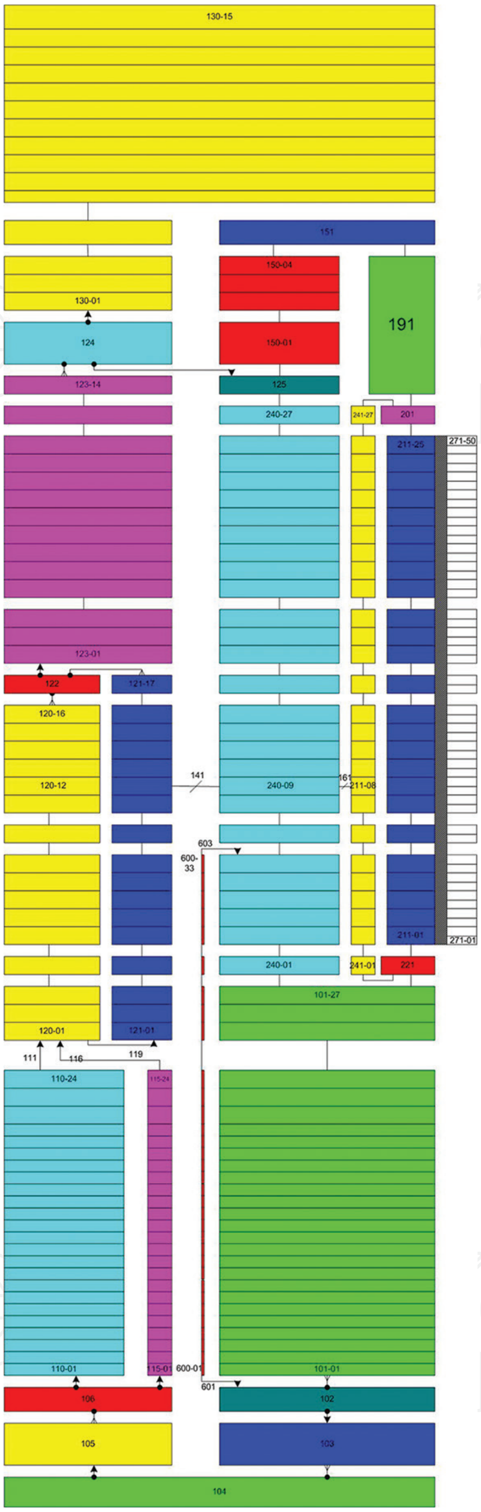

Figure 15. RELAP5 nodalization of the IRIS reactor vessel.

A small-break LOCA was analyzed to demonstrate steam generator behaviour in accident conditions. The initiating event was the rupture of a chemical and volume control system pipe connected to the upper annular pump suction plenum of the reactor vessel. IRIS is designed to limit the loss of coolant from the vessel rather than relying on systems to inject water into the reactor vessel. It has a compact, small diameter, high-design pressure containment that assists in limiting the blowdown from the reactor vessel by providing a higher back pressure in the initial stage of the accident. Furthermore, four trains of passive emergency heat removal systems (EHRS) help to depressurize the primary system by condensing steam, coming out of the reactor core, on the steam generators tubes (depressurization without the loss of mass), and to remove the decay heat. Finally, it features automatic depressurization system to condense steam, released from the top of the pressurizer, in the pressure suppression pool located inside the containment.

Following the initiating event, the LOCA mitigation signal is rapidly actuated, the reactor and reactor coolant pumps are tripped and the four EHRS subsystems are actuated by closing the main feed and steam isolation valves and by opening the fail-open valves in the EHRS return lines connected to the SG feed lines. The EHRS is composed of pipes, valves and heat exchangers submerged in the water tank outside the containment. After the initial decrease of the coolant inventory in steam generators caused by the isolation of the feedwater flow, the EHRS operation restores the SG secondary fluid mass (**Figure 16**) and enables heat removal out of the steam generators (**Figure 17**). It does not take long for the situation to stabilize to ensure safe reactor conditions. Equalization of reactor vessel and containment pressures marks the end of the blowdown phase and start of a long-term cooling phase by the continued operation of the EHRS, with the pressure being slowly reduced as the core decay heat decreases.

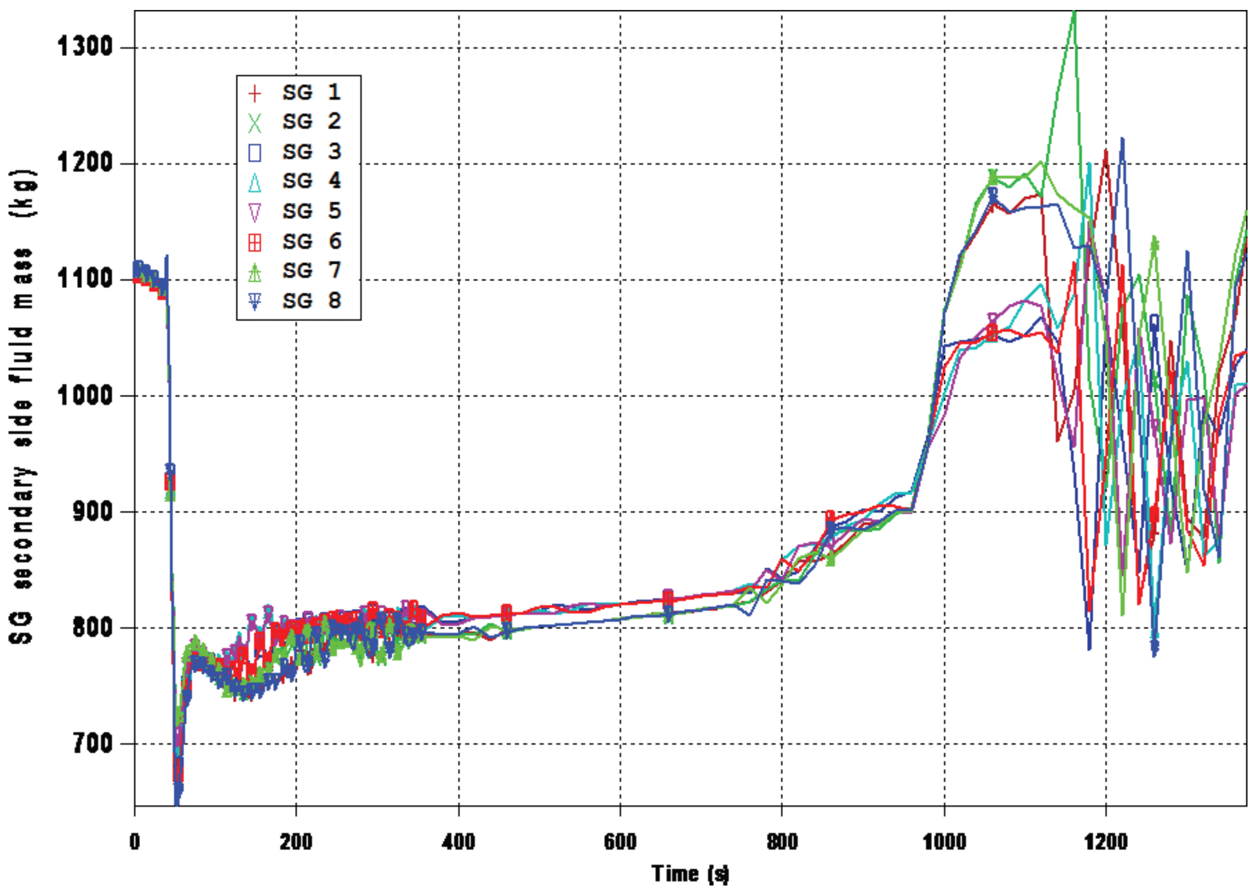

Figure 16. SG secondary-side fluid mass in IRIS NPP.

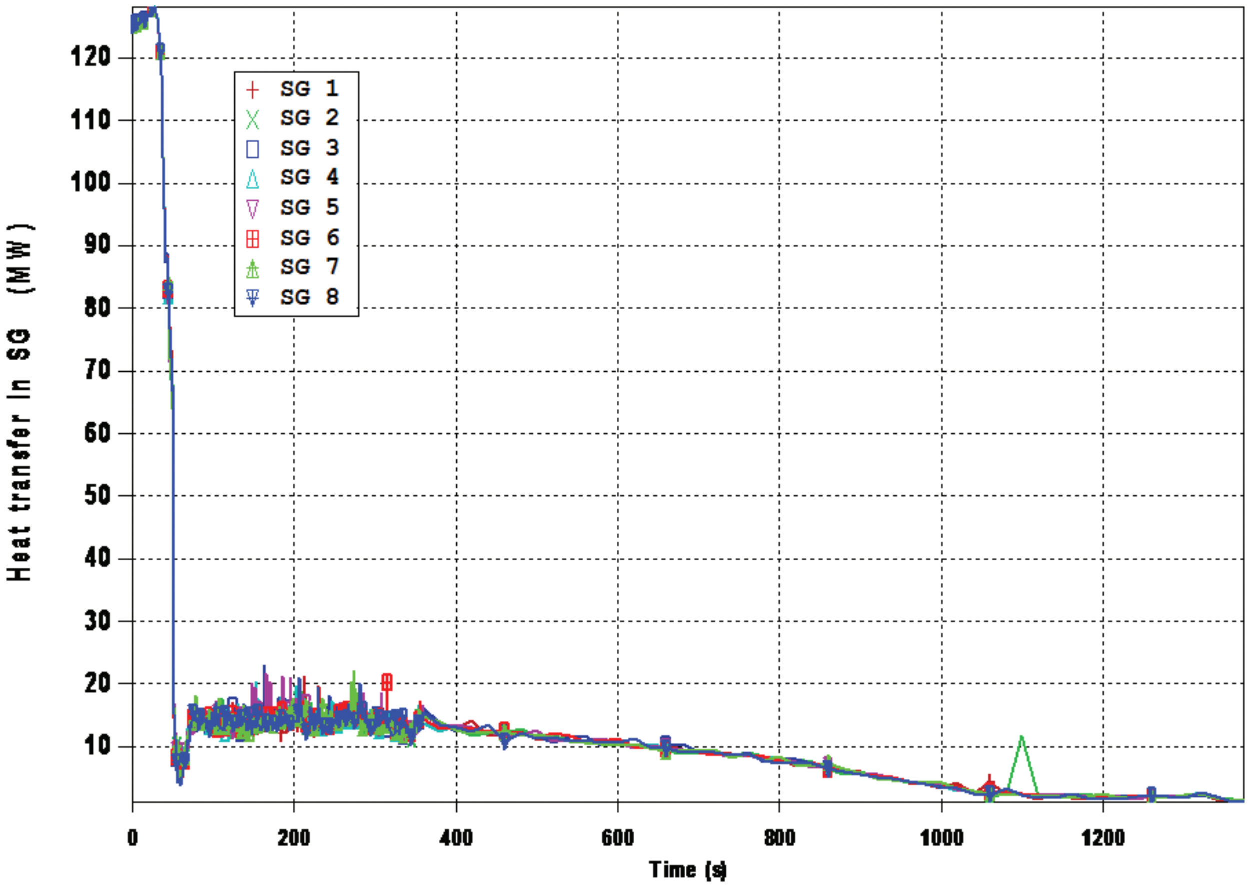

Figure 17. Heat transfer in IRIS NPP steam generators.

5. Conclusions

Steam generators (SGs) are nuclear power plant components where the steam, which drives the turbine, is produced. They also represent barrier that prevents radioactive fission products to escape outside the containment building.

In order to ensure safe operation of a nuclear power plant, SG parameters, the steam flow, steam pressure and temperature, feedwater temperature, circulation ratio and total inventory mass, have to be maintained within prescribed values. These values depend on the operating window and the thermal load. The pressure is controlled by the relief and safety valves and the inventory by the feedwater flow.

The inverted U-tube steam generators have quite large water mass in the secondary side which is important during accidents of loss of the secondary heat sink. In that design, primary system water which is at a higher pressure than the secondary fluid flows inside the tubes, while the secondary fluid is on the outer side. In the helical-coil SGs, the situation is opposite: the primary water is on the tubes' outside surface keeping the tubes in the state of compression. The loss of the secondary flow is here more critical due to limited amount of water, but this is compensated with passive auxiliary safety systems.

In the event of losing the primary coolant pumps, the natural circulation between the reactor core and the SGs can ensure safe cooling of the reactor core by establishing the two-phase flow inside the tubes, with only minimal operator actions required. Thus, the design of the nuclear steam generators and the auxiliary systems enables safe NPP operation, not only during the normal plant operation but also during the accident conditions.

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