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Conservative evaluation of radionuclides release for VVER-440 and VVER-1000 type reactors

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A R T I C L E I N F O

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A B S T R A C T

In this work, we present the preliminary results of radionuclide release assessment outside the containment under LOCA and PRISE accidents of design basis and design extension conditions categories for pressurized water VVER-440&1000 type reactors with a conservative approach to specifying the source term. ATHLET and COCOSYS calculation codes are used for analysis. The discussion concerns the main results of calculations depending on the type of reactor unit. In conclusions, the possible approaches are outlined to reduce conservatism and improve the methodology in certain areas for further studies.

1. Introduction

As follows from numerous studies, the loss of primary coolant accidents (LOCA) including Primary to Secondary leaks (PRISE) are the initiating events (IE) potentially leading to the largest release of the fission products (FP) to the environment. Although such accidents with significant releases of the FPs to the environment occur quite rarely, appropriate simulation tools should be used for analyses to consider the complexity of the physical processes of the FP propagation in the coolant of the first and second circuits, as well as in the surrounding atmosphere. Appropriate simulation tools should be used to analyze such accidents to consider the complexity of transient progression including possibility of core degradation, fission product release to reactor coolant, transport within the reactor cooling circuit, transfer to the containment and potentially to the environment.

The reviews of safety analysis approaches after the Fukushima Daiichi Nuclear Power plant (FDNP) situations is motivated by the importance of strengthening the global assessment of the safety level of Nuclear Power Plants (NPP) by considering specific situations more serious than those envisaged by the design of the plants. In particular, additional events or combinations of events have to be taken into account. This has been stated by the IAEA (2016) through the definition of the Design Extension Conditions (DEC). The project “Reduction of Radiological Consequences of design basis and design extension Accidents” (R2CA) aims at assessments of radiological consequences (RC) of design basis accidents (DBA) and design extension conditions (DEC-A) reactor accidental situations, focusing on two main categories of accident: the Loss of Coolant Accidents (LOCA) and the Steam Generator Tube Rupture (hereinafter referred to as PRISE) accidents.

In this paper, a short description of the ATHLET and COCOSYS models developed in the frame of the R2CA (Reduction of Radiological Consequences of Design Basis and Design Extension Accidents) project together with preliminary results of simulation of the LOCAs under DBA and DEC-A are presented. Accident simulations were performed for VVER-1000 and VVER-440 reactor types. Their main purpose was to estimate in a conservative manner the characteristics of the FP release into the environment under DBA versus DEC-A situations.

The main rated operational characteristics of VVER-1000 and VVER-440 reactor units are given in Table 1.

2. Tools and methodologies

The reactor plant models were developed to simulate the primary and secondary circuits parameters behavior during transients, as well as to determine the parameters of the coolant release (mass and energy) into the containment with using of ATHLET and COCOSYS models developed by ARB. A brief description of the models is presented below.

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2.1. ATHLET model

The thermal–hydraulic computer code ATHLET 3.2 was developed by GRS (Gesellschaft für Anlagen und Reaktorsicherheit gGmbH, Germany) for the analysis of operational conditions, abnormal transients and all kinds of leaks and breaks in nuclear power plants. The code covers the whole spectrum of design basis and beyond design basis accidents (without core degradation) (Lerchl, et al., 2019).

2.1.1. ATHLET model for VVER-1000/V-320

Nodalization scheme of the VVER-1000/V-320 reactor unit primary side model is shown in Fig. 1. Each of the four cold legs (P1-CL, P2-CL, P3-CL, P4-CL) is connected to the corresponding sector of downcomer (PV-DC-11, PV-DC-21, PV-DC-31, PV-DC-41 and then to PV-DC-12, PV-DC-22, PV-DC-32, PV-DC-42). The reactor lower plenum is represented by volumes PV-LP-1, PV-LP-2, PV-LP-3.

The reactor core is simulated by three parallel channels. Thermo Fluid Objects (TFO) PV-COR-FA simulates "hot" fuel assembly in the central part of the reactor core. TFOs PV-COR-IN and PV-COR-OUT simulate 108 and 54 fuel assemblies located in its internal and external part, respectively.

TFOs PV-BP-xx simulate core bypasses through the spacer ring, through baffle trains, through a gap between the baffle and the core barrel, through the guide trains and central tubes, PTU tubes.

The reactor upper plenum is simulated by TFOs PV-UP-xx with splitting of upper space to take into account actual design from the top of the reactor core to the PTU upper plate. The volume located under the reactor cover is simulated by TFO PV-UH.

The Pressurizer (PRZ) system (Fig. 2) is simulated with PRZ vessel (P0-PRESS), surge line (P0-SURGE), spray lines (P0-SPRAY, P0-SPRAY1, P0-SP-CON, P0-SP-THIN), steam discharge line (P0-SV1, P0-SV2, P0-SV3) into babbler tank (BUBBL-PO, BUBBL-OP, BUBBL-DN, BUBBL-UP) with cooling system (BUBBL-PROM), PRZ electric heaters.

The four loops of rector unit are simulated with four single loops. Nodalization of hot and cold legs of all loops is taken to be the same (Fig. 1, only loop #1 is shown). The hot legs are simulated with TFOs P1-HL, P2-HL, P3-HL, P4-HL. There is a connection with PRZ surge line on the hot leg #4. The cold legs are simulated with volumes P1-CL, P2-CL, P3-CL, P4-CL. Reactor coolant pump (RCP) is installed on each cold leg. There is a connection with PRZ spray lines on the cold leg #1.

The SG nodalization (Fig. 3) includes a part of primary side (internal volume of SG headers and tubing, Fig. 1) and a part of secondary with simulation of heat transfer through the tubing heat structure. SG Hot Headers are simulated with TFOs P(1–4)-SG-IN(1–9) and Cold Headers with P(1–4)-SG-OUT(1–9). The secondary side of each SGs is simulated with six TFOs: the lower part with first bundle of tubing (S(1–4)-DN), volumes within the boundaries of tube bundles S(1–4)-PO, volumes between banks of tube bundles S(1–4)-OP (downcomer), volumes above tube bundles S(1–4)-TOP, upper steam volumes S(1–4)-UP.

The nodalization of main steam line system (Fig. 4) is represented by: four steam lines S(1–4)-MSL from SGs to Turbine stop/control valves; Main Steam Header (MSH) S0-GPK(1,2); connecting pipelines from corresponding steam lines to MSH, BRU-K, BRU-A and SG Safety Valves (SV).

Nodalization of the main and auxiliary feed water systems is represented by: two main feed water pumps, two auxiliary feed water pumps, main feed water collector, feed water lines, main and auxiliary control valves and check valves.

Nodalization of the emergency feed water system is represented by: three emergency feed water pumps (EFWP), EFW collectors, EFW supply pipelines, control valves and check valves.

ATHLET model also includes Emergency Core Cooling System (ECCS) that consists of the active and passive components. The active parts are three systems with three independent trains each: Low pressure injection system (LPIS) TQ12(22,32); High pressure injection systems (HPIS) TQ13(23,33) and TQ14(24,34). One LPIS train (TQ12) is connected to the cold and hot legs of loop #1 and the other two are connected to the HA lines. HPIS trains are connected in pairs to cold legs: TQ13,14 to cold leg #1; TQ23,24 to cold leg #4; TQ33,34 to cold leg #3. The passive parts are four Hydroaccumulators (HA). Two of them are connected to the upper plenum of the reactor, and the other two are connected to the reactor downcomer.

2.1.2. ATHLET model for VVER-440/V-213

Nodalization scheme of the VVER-440/V-213 reactor unit primary side is shown in Fig. 5. Each of the six cold legs P(1–6)-CL is connected to the corresponding sector of downcomer V-DCL-P(1–6). The reactor lower plenum is represented by volumes V-LP1, V-LP2, V-LP-3.The 276 working fuel assemblies of the reactor core are represented by two thermo fluid objects V-FIX-HOT and V FIX AVER. TFO V-FIX-HOT simulates a "hot" assembly with one "hot" fuel pin. TFO V-FIX-AVER simulates the rest of the core working "average energy" assemblies. The 37 "ARCs" control assemblies are simulated by TFO V-FOLLOW. Inter assemblies gaps are simulated by V-INTERGAP volume. TFO V-UP5 simulates the peripheral part of the upper plenum with hot legs attached. TFOs V-UP(2–4) represent the rest of the upper plenum volume. The volume located under the reactor cover is simulated by TFO V–UHEAD.

The Pressurizer system (Fig. 6) is simulated with PRZ vessel TFO PRESSUR, PRZ nozzle TFO SURGE1, surge line TFO SURGE, spray line TFO SPRAY, and associated thermal structures, including those for electric heaters. The injection into PRZ is controlled by a single valve V-SPRAY. PRZ discharge system is simulated by valve PRES-SV.

The six loops of rector unit are simulated with six single loops. Nodalization of hot and cold legs of all loops is taken to be the same (Fig. 5, only loop #1 is shown). The hot legs are simulated with TFOs P (1–6)-HL. There is a connection with PRZ surge line on the hot leg #6. The cold legs are simulated with volumes P(1–6)-CL. Reactor coolant pumps (RCP) are installed on each cold leg. There is a connection with PRZ spray lines on the cold leg #6.

The SG nodalization includes a part of primary side (internal volume of SG headers and tubing, Fig. 5) and a part of secondary (Fig. 7) with simulation of heat transfer through the tubing heat structure. SG Hot Headers are simulated with TFOs P(1–4)-SG-IN(B,M)(1,2) and Cold Headers with P(1–4)-SG-EX(B,M)(1,2). The tubing is divided in height into 4 bundles. The secondary side of each SGs is simulated with five TFOs: S(1–4)-SG-BOT, S(1–4)-SG-TOP, S(1–4)-SG-HOT, S(1–4)-SG-COLD and S(1–4)-SG-DFL.

The nodalization of main steam line system (Fig. 8) is represented by: six steam lines S(1–4)-MSL from SGs steam collectors S1(1–4)-MSL to Turbine stop/control valves S(1,2)-MTV; pipelines of Main Steam Header S0-COLL; connecting pipelines from MSH to BRU-K and BRU-A.

Table 1
Main operational characteristics of VVER-1000 and VVER-440 Reactor Units.

Primary Circuit Parameters	Units	VVER-1000	VVER-440	Secondary Circuit Parameters	Units	VVER-1000	VVER-440
Core power	MW	3000	1375	SG thermal power	MW	750	230
Primary pressure	MPa	15.7	12.4	SG pressure	MPa	6.2	4.7
CL/HL coolant temperature	°C	289/320	267/297.9	Feed water temperature	°C	220	220
Coolant flow rate	m^3/h	84800^{+4000}_{-4800}	42700 ± 400	SG steam capacity	t/h	1470	450
Pressurizer level	m	8.77	5.96	SG level	m	2.25	2.1

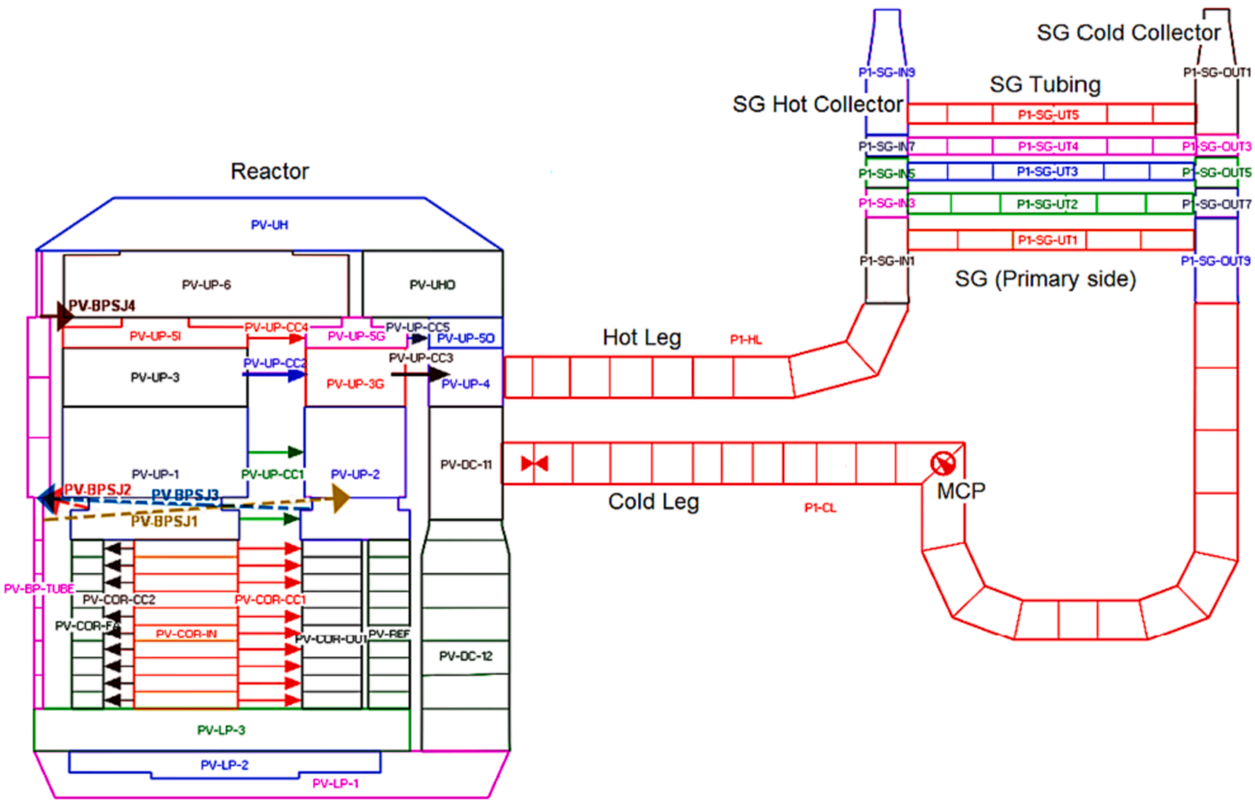

Fig. 1. Primary circuit.

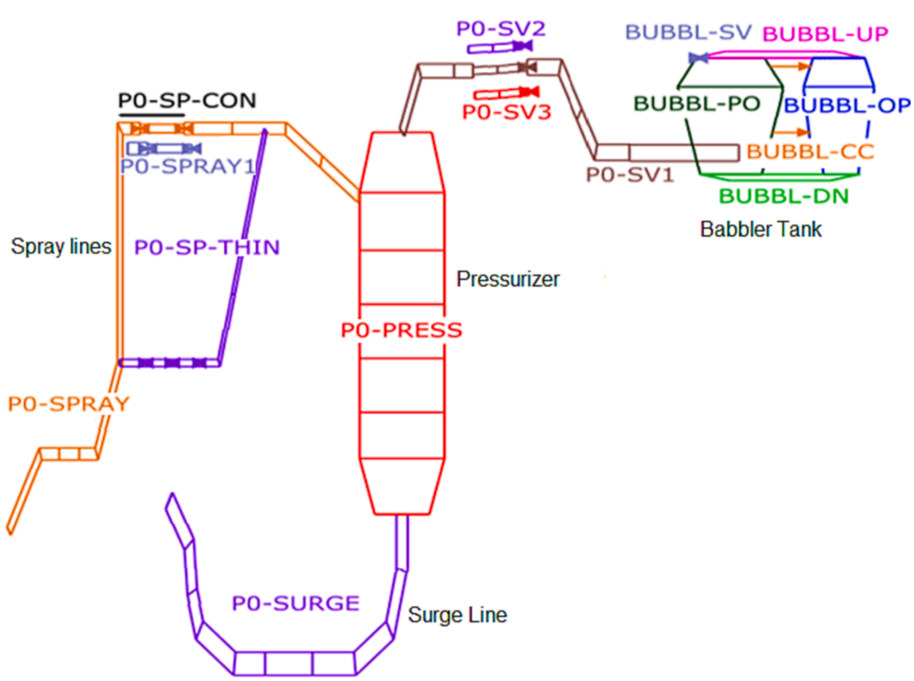

Fig. 2. Pressurizer.

Nodalization of the main and auxiliary feed water systems is represented by: four main feed water pumps, two auxiliary feed water pumps, feed water semicollectors, feed water lines, main and auxiliary control and check valves.

Nodalization of the emergency feed water system is represented by: three emergency feed water pumps, EFW semicollectors, EFW supply pipelines, control and check valves.

The ATHLET VVER-440 model also includes Emergency Core Cooling System that consists of the active and passive components. The active parts are two systems with three independent trains each: High Pressure Injection Pumps NAP1(2,3) and Low Pressure Injection Pumps NOR1(2,3). One LPIS train is connected to the cold and hot legs of loop #4 and the other two are connected to the HA pipelines. HPIS trains are connected to cold legs: NAP1 to cold leg #2; NAP2 to cold leg #3; NAP4

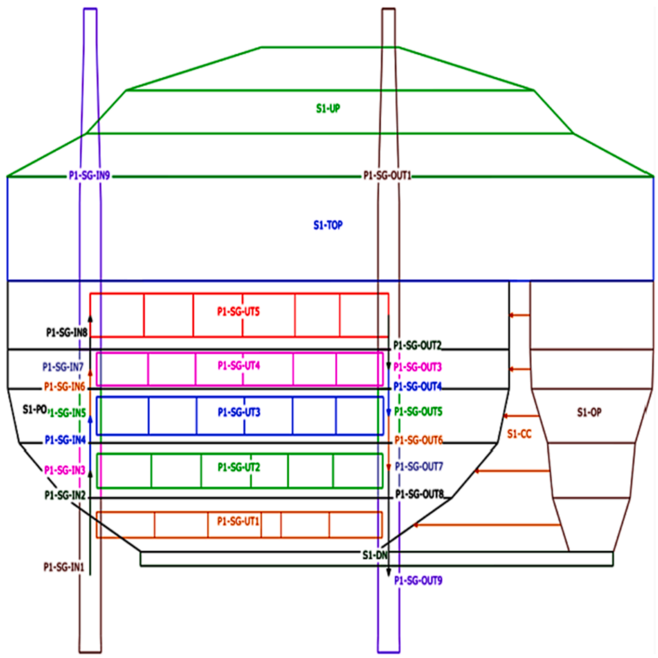

Fig. 3. Steam generator.

to cold leg #6. The passive parts are four Hydroaccumulators. Two of them are connected to the upper plenum of the reactor, and the other two are connected to the reactor downcomer.

2.2. COCOSYS model

The COCOSYS is a lumped parameter computer code developed and maintained at GRS for best-estimate analysis of light-water reactor containments during severe accidents. A feature of the COCOSYS code is the extensive consideration of interactions between the various developing phenomena, such as the thermal hydraulics processes, hydrogen combustion, and aerosols and nuclides behavior (Arndt et al., 2020). The containment model of the COCOSYS 3.0 code was used for calculating the containment parameters and the activity distribution in the containment and release to the environment.

2.2.1. COCOSYS model for VVER-1000/V-320

The base case COCOSYS model consists of the 95 nodes, 281 junctions and 324 heat structures. The nodalization scheme of the COCOSYS model is shown on Fig 9. The model contains a spray system and a ventilation system. There is also the possibility of taking water from the sump with HPIS and LPIS pumps.

2.2.2. COCOSYS model for VVER-440/V-213

The VVER-440/V-213 COCOSYS model consists of a hermetic compartments system, a pressure suppression system (bubble condenser (BC) tower), and six nodes simulating air traps. The nodalization scheme is shown on Fig. 10. The BC tower is modelled with twenty nodes: eight volumes (4 axial layers) represent the space in front and at the sides of the water trays, six nodes model twelve water trays, and six nodes were introduced to separately represent the volumes below the trays. The volumes below the trays were needed to properly simulate the water removal from the trays and the passive spray reservoir. The air locks are represented with six different nodes, which are separated from adjacent water trays by the check valves.

3. Test scenarios

3.1. DBA scenarios

3.1.1. DBA LOCA scenarios for VVER-1000 and VVER-440

The primary leaks from cold leg to the containment (LOCA) with 2 × 850, 350, 100, 50 mm diameters are simulated for the VVER-1000.

LOCA leaks with 2 × 500, 250, 100, 50 mm diameters are simulated for the VVER-440.

For DBA LOCA analyses conservative assumptions (IAEA-EBP-WWER-01, Guidelines for Accident Analysis of WWER Nuclear Power Plants, 1995; Analysis, 2002) are applied for both thermohydraulics and radiological consequences simulations.

Assumptions regarding the selection of initial and boundary conditions when creating scenarios for VVER-1000 and VVER-440 are conceptually the same (individual differences are noted below where necessary).

For thermohydraulics 104% for VVER-1000 (and 103.5% for VVER-440) of nominal reactor power, maximum peaking factors for “hot” fuel assembly (FA) and “hot” fuel rod power distribution (with symmetric axial profile) are used. The core decay heat release is increased by 10% of the standard ANSI/ANS-5.1*-1979 (American National Standard for Decay Heat Power in Light Water Reactors, 1979). The reactivity coefficients are taken to ensure the maximum integrated power in transients. Minimum reactor flow rate, PRZ and SG levels; maximum primary and secondary pressure, feed water temperature, containment pressure and temperature are applied.

No credit for personnel actions; loss of the Unit power supply; a single failure to start one (of three) diesel generator with a dependent failure of one (of three) train of each safety system of LPIS, HPIS, EFW and containment spray (all with minimum flowrate characteristics); no primary make-up; not taken into account of AFW operation; non-working containment recirculation ventilation systems are postulated as conservative boundary conditions. Additionally, for leak 2 × 850 mm (VVER-1000) and leak 2 × 500 mm (VVER-440) the failures of second trains of the HPIS and LPIS and one HA supplying to the reactor downcomer are taken, for containment spray one of the two remaining trains is inoperative (in repairing).

For radiological consequences, the FP instant release from the fuel to the primary circuit of the entire inventory of the gas gap at the beginning of the transient plus total primary circuit coolant activity (including spike) are specifying the source term. The entire inventory of the gas gap as well as total activity of radionuclides in primary coolant (including spike) are obtained from the industry guideline of the Ukrainian operator (Standard procedure, xxxx) (for VVER-1000) or relevant safety analysis reports (for VVER-440). The total primary circuit coolant activity corresponds to the steady state. Spike effect implies a thirtyfold increase in the concentration of iodine isotopes compared to steady state. The entire inventory of the gas gap corresponds to the time immediately after the reactor shutdown at the end of the fuel cycle and the assumption that the core contains the same amount of fuel assemblies of one, two, three and four years of operation. Such approach for specifying the source term meets the requirements of the nuclear regulation of Ukraine for the report on the safety analysis (SAR) of NPP unit.

FP release from primary to the containment through the leak from cold leg is calculated in proportion to the coolant mass release (so after the total mass of the primary coolant to the containment becomes equal to the initial primary inventory, the output of activity from the primary circuit stops). FP dilution in the primary coolant with ECCS boric water is not taken into account. Design leakages from containment to the environment 0.3% per day of total gas mass at a design pressure 0.49 MPa (5 kgf/cm^2) are applied for VVER-1000, and 20.0% per day at a design pressure 0.13 MPa (1.3 kgf/cm^2) are applied for VVER-440.

3.1.2. DBA PRISE scenarios for VVER-1000 and VVER-440

PRISE leaks as result of SG collector cover lift-up (100, 60, 40, 20

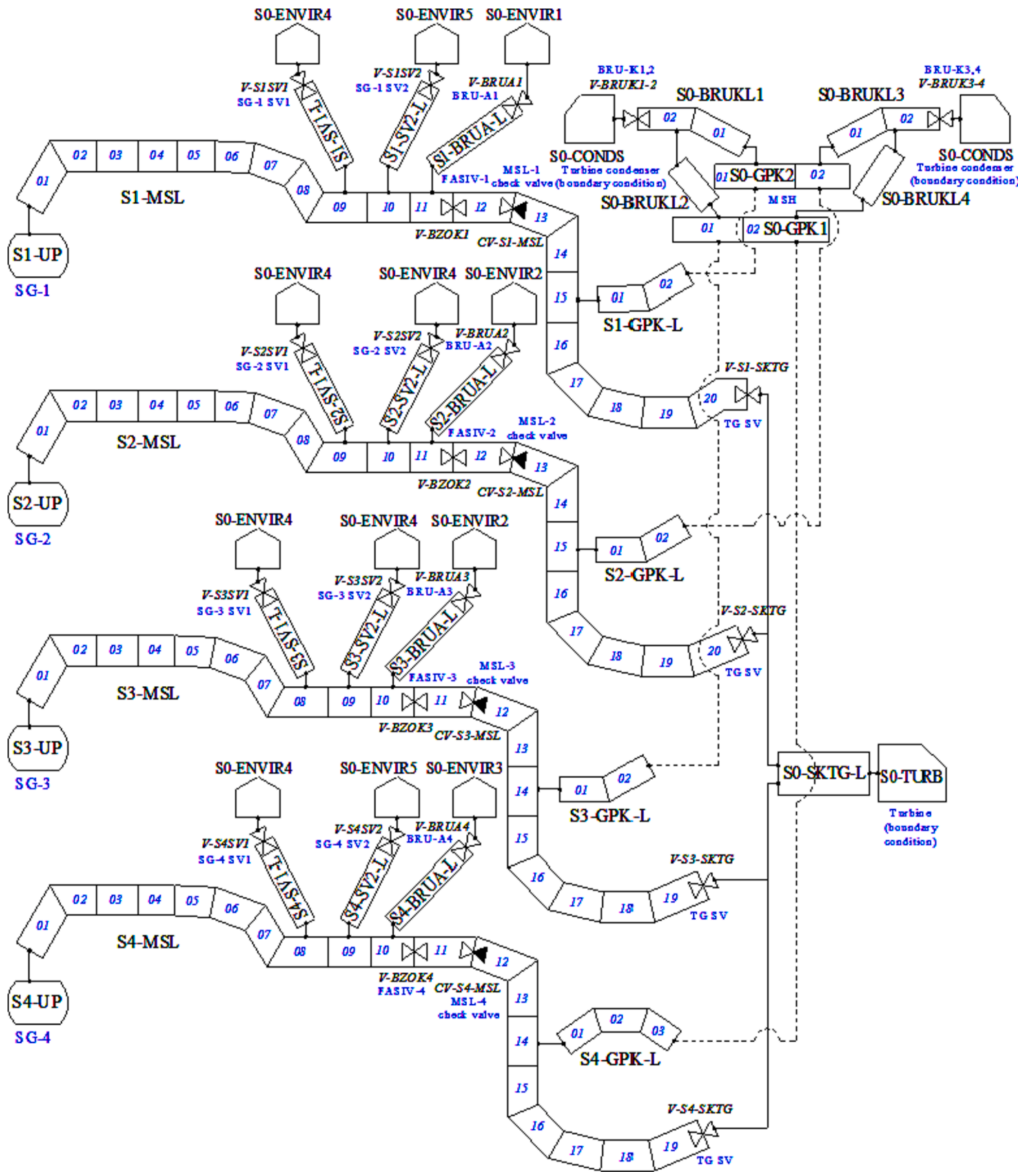

Fig. 4. Main steam lines.

mm) and guillotine break of one, two or three tubes (2 × 13, 2 × 2 × 13, 3 × 2 × 13 mm) are simulated for the VVER-1000.

PRISE leaks as result of SG collector cover lift-up (107, 60, 40, 20 mm) and guillotine break of one, two or three tubes (2 × 13, 2 × 2 × 13, 3 × 2 × 13 mm) are simulated for the VVER-440.

For DBA PRISE leaks analyses of VVER-1000 and VVER-440 conservative assumptions are applied for both thermohydraulics and radiological consequences simulations.

Assumptions regarding the selection of initial and boundary conditions when creating scenarios for VVER-1000 and VVER-440 are conceptually the same (individual differences are noted below where necessary).

For thermohydraulics 104% for VVER-1000 (103.5% for VVER-440) of nominal reactor power, maximum peaking factors for "hot" fuel assembly (FA) and "hot" fuel rod are used. The core decay heat release is increased by 10% of the standard ANSI/ANS-5.1-1979 (American National Standard for Decay Heat Power in Light Water Reactors, 1979). The reactivity coefficients are taken to ensure the maximum integrated power during transients. Minimum reactor flow rate; maximum primary and secondary pressure, PRZ and SG levels, feed water temperature are applied.

Loss of the Unit power supply; a single failure of emergency SG BRU-A stuck open for VVER-1000 (or SV for VVER-440) with non-localized leak from the primary circuit through the SG and steam pipelines to the environment; maximum flowrate characteristics of LPI and HPI pumps; no primary make-up; not taken into account of AFW operation are postulated as conservative boundary conditions.

There are personnel actions after 1800s: all FASIV closure, all HPI pumps stop, 2 (of 3) LPI pumps stop, EFW supply of intact SGs and cooldown with maximum speed through their BRU-As for VVER-1000 (or through SVs for VVER-440), open all PRZ SVs. Additionally for VVER-1000, (after primary pressure <18 kgf/cm^2 and hot legs temperature < 150 °C) personnel begin of maintaining the PRZ level of 5–6 m by the last LPI train and activate scheduled cooldown system. Additionally for VVER-440, (after primary pressure < 6 kgf/cm^2 and hot legs temperature <150 °C) personnel stop the last of LPI pump and activate scheduled cooldown system.

For radiological consequences, total primary circuit activity (including spike) are specifying the source term (see also section 2.1.1). FP release from primary to emergency SG and further through steam

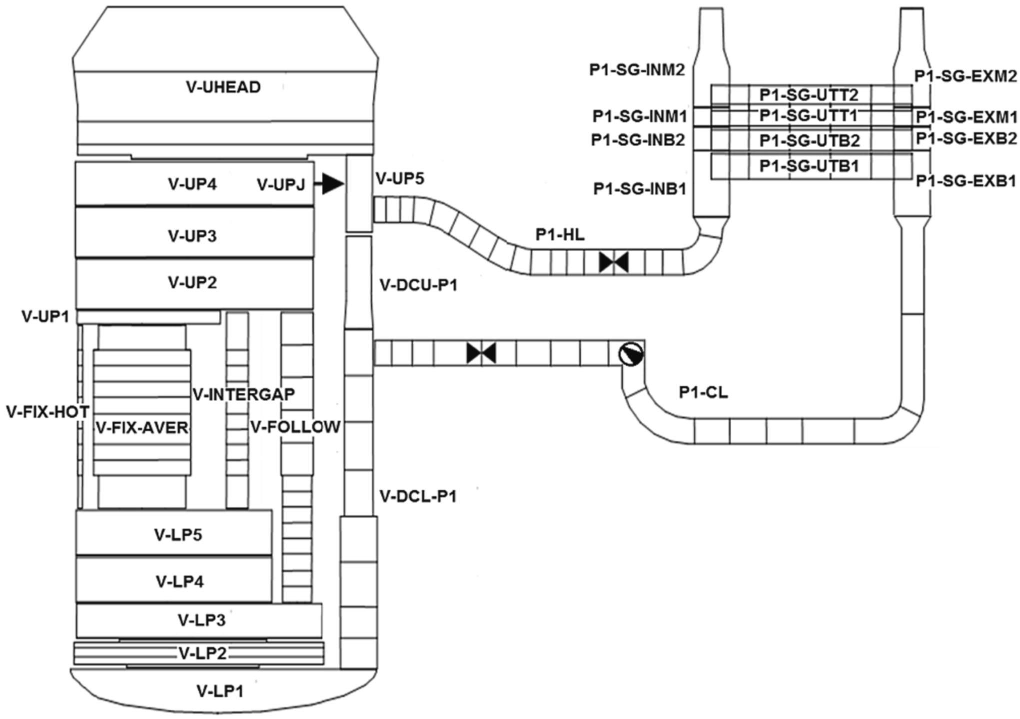

Fig. 5. Primary circuit.

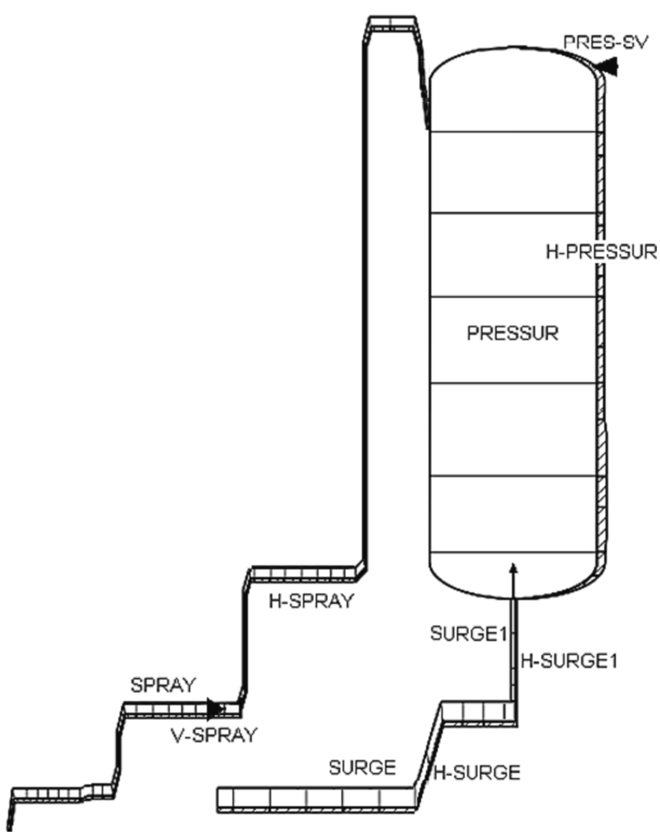

Fig. 6. Pressurizer.

pipelines and stuck open BRU-A into the environment is calculated in proportion to the coolant mass release (so after the total mass of the primary coolant to the environment becomes equal to the initial primary inventory (about 250 t), the input of activity to the environment stops).

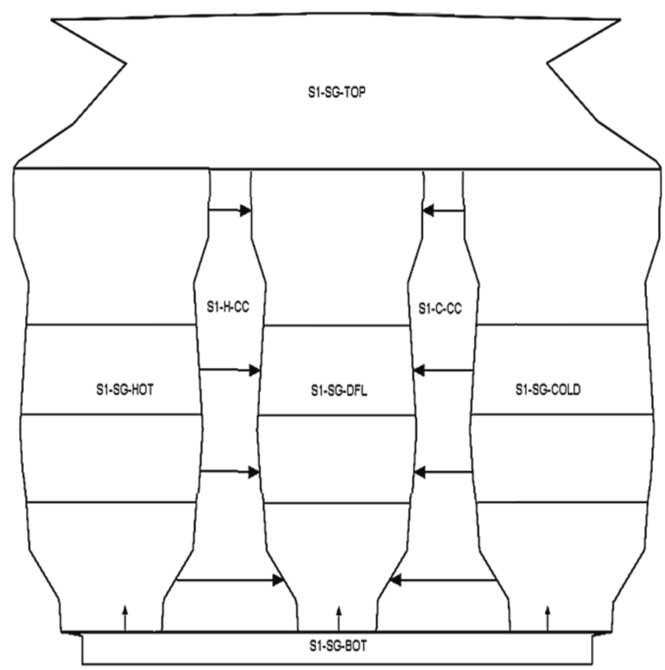

Fig. 7. Steam generator (secondary side).

FP dilution in the primary coolant by ECCS boric water and by FW of secondary is not taken into account.

3.2. DEC-A scenarios

3.2.1. DEC-A LOCA scenarios for VVER-1000 and VVER-440

The primary leaks from cold leg to the containment (LOCA) with 2 × 850, 350, 100, 50 mm diameters are simulated for the VVER-1000.

LOCA leaks with 2 × 500, 250, 100, 50 mm diameters are simulated

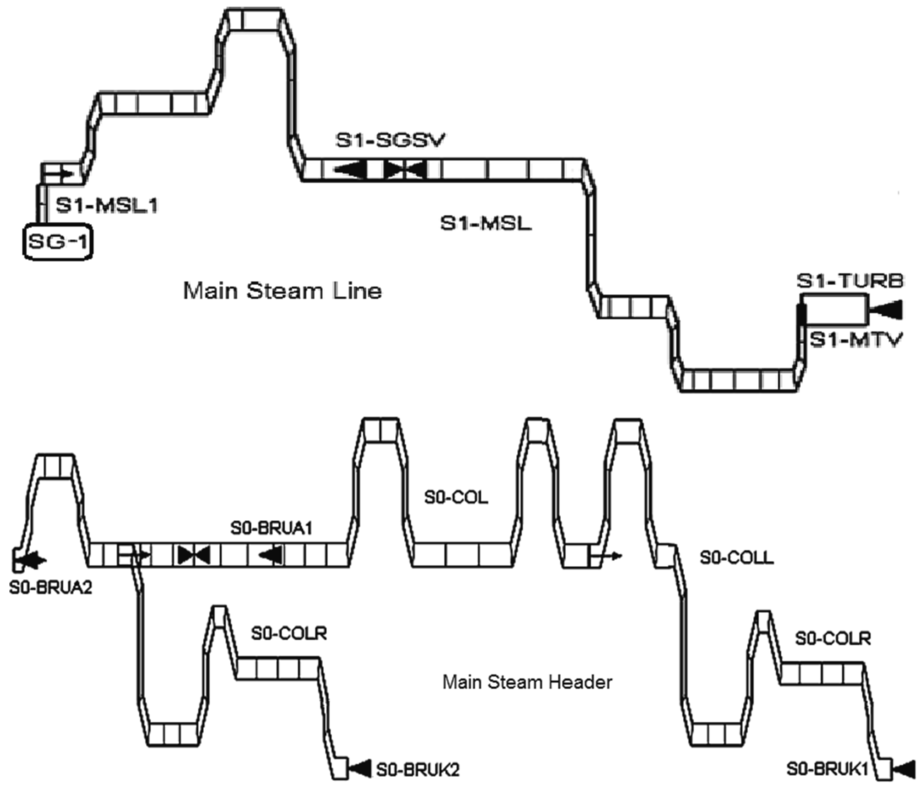

Fig. 8. Main steam lines system.

for the VVER-440.

For DEC-A LOCA thermohydraulics simulations a realistic approach for initial conditions is adopted with nominal primary and secondary initial parameters (Analysis, 2002), except for the application of maximum peaking factors for "hot" fuel assembly and "hot" fuel rod power distribution.

The DEC-A LOCA boundary conditions for thermohydraulics and radiological consequences are the same as those of the DBA, except the containment spray system is completely inoperative.

3.2.2. DEC-A PRISE scenarios for VVER-1000 and VVER-440

PRISE leaks as result of SG collector cover lift-up (100, 60, 40, 20 mm) and guillotine break of one, two or three tubes (2 × 13, 2 × 2 × 13, 3 × 2 × 13 mm) are simulated for the VVER-1000.

PRISE leaks as result of SG collector cover lift-up (107, 60, 40, 20 mm) and guillotine break of one, two or three tubes (2 × 13, 2 × 2 × 13, 3 × 2 × 13 mm) are simulated for the VVER-440.

For DEC-A PRISE leaks thermohydraulics simulations a realistic approach for initial conditions is adopted with nominal primary and secondary initial parameters (Analysis, 2002), except for the application of maximum peaking factors for "hot" fuel assembly and "hot" fuel rod.

The DEC-A PRISE boundary conditions for thermohydraulics and radiological consequences are the same as those of the DBA, except the nominal flowrate characteristics of HPI and LPI pumps are used, and for VVER-1000 additional failure to close an emergency SG SV after its opening is assumed, and for VVER-440 additional failures to close of emergency SG FASIV and to close of one BRU-A after its opening are applied.

4. Main outcomes of Athlet/Cocosys simulations

This section presents the main results of simulation emergency scenarios in par. 2 using the developed models described in par. 1.1 and par. 1.2.

4.1. DBA analyses

4.1.1. LOCA DBA calculations

Simulations with ATHLET/COCOSYS for the DBA LOCA scenarios include leaks 2 × 850, 350, 100, 50 mm for VVER-1000 and leaks 2 × 500, 250, 100, 50 mm for VVER-440.

The main attention is paid to leaks the double-ended guillotine breaks of 2 × 850 and 2 × 500 mm, during which the most unfavourable conditions are observed in relation to the integrity of the reactor core, the growth of parameters in the containment, as well as the FP release into the environment. The most conservative scenario for maximum peak cladding temperature was determined using a sensitivity analysis by varying the leak discharge coefficients. There are two specific peaks of cladding temperature for both VVER-1000 and VVER-440 (Fig. 11). The cladding temperature reaches the first peak of 1098 °C for VVER-1000 and 700 °C for VVER-440 in the first seconds of the transient at the outflow stage as a result of a heat transfer crisis in the core after the coolant circulation through the reactor has stopped and heat removal from the fuel rod has deteriorated. The second peaks of 1019 °C (at 100 s) for VVER–1000 and 1045 °C (at 970 s) for VVER-440 were reached during the re-flooding stage at low levels in the reactor (Fig. 12).

The main LOCA DBA calculation results with ATHLET/COCOSYS that may challenge the FP release are presented in Table 4. The parameters of liquid and water discharge into the containment, ECCS injection into the primary are given over a period of 1500 s. The aactivity releases to the environment are shown in Table 4 at 1500 s of transient for VVER-1000 and at 1500 s, 30000 s for VVER-440. Progression of FP release to the environment for VVER-1000 and VVER-440 are shown in Fig. 13.

At the beginning of the transient, the main part of the released activity from primary into the containment is concentrated in the form of aerosols. The activity of the aerosols within the containment as well as their release to the environment decreases as FP passes from aerosols to water due to the spray system operation.

The FP mass and activity in the containment atmosphere practically

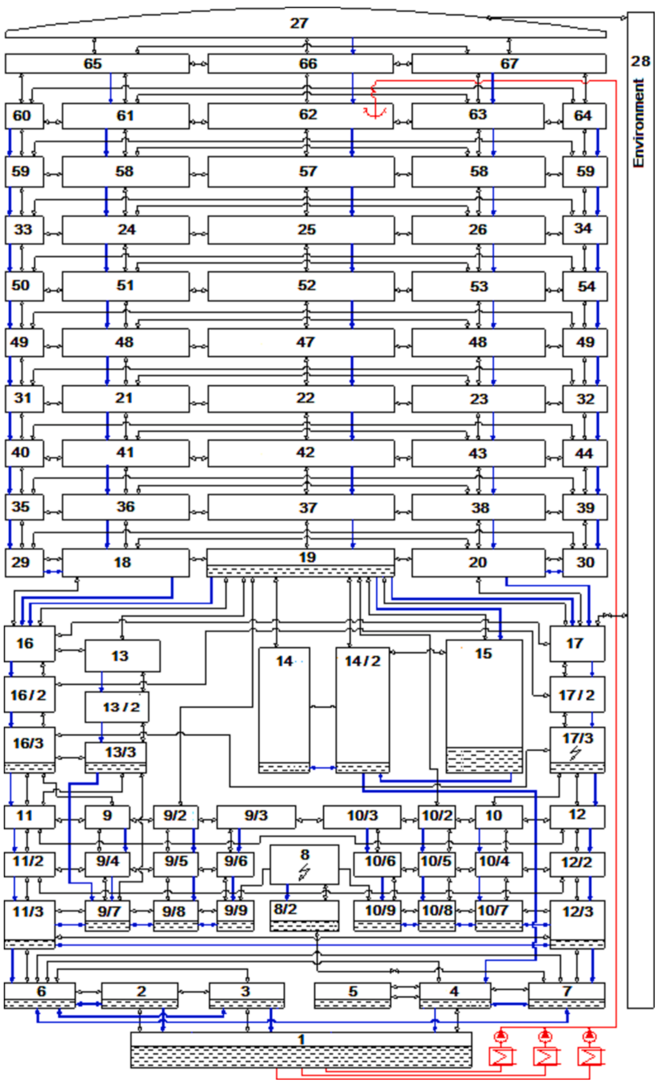

Fig. 9. VVER-1000/V-320 COCOSYS nodalization.

do not decrease during the transient. Thus, the release of the FP air fraction into the environment will continue until the pressure in the containment falls below the ambient pressure.

4.1.2. PRISE DBA calculations

Simulations with ATHLET for the DBA PRISE scenarios include leaks as result of SG collector cover lift-up, as well as due to guillotine break of SG tubes (see par. 2.1).

As a result of the PRISE leak primary coolant with FP passes to emergency SG and then releases to the environment through stack open relief valve (postulated failure). Three ECCS trains operation contribute to a high injection flow rate resulting in rapid emptying of the ECCS tanks with permanent loss of cooling water through the SG break (Fig. 14). Appropriate measures must be taken by personnel to terminate irretrievable losses of the coolant and to reduce the radiation release as well. Some of such actions are simulated after 1800 s (see par. 2.1). It allows maintaining the possibility of long-term heat removal from the reactor core and minimizing the release to the environment.

The main PRISE DBA calculation results with ATHLET that may challenge the FP release are presented in Table 5. The parameters are given over a period of 3600 s. Progression of FP release to the environment for VVER-1000 and VVER-440 are shown in Fig. 15. Results demonstrate that the release into the environment is from 16 to 100% of the primary coolant activity for VVER-1000 and from 44 to 100% for VVER-440.

In the course of the analyzed transient processes, no conditions are reached for damage to the fuel cladding. Therefore, an approach that takes into account only the activity of the primary circuit in the FP release is justified. The calculation of the duration of FP release to SG does not take into account the dilution of the primary coolant with ECCS water from HPI, LPI and HA, which could significantly reduce the radioactive release and extend it over time. Also it should be noted that with a decrease in the leak diameter, the water fraction of the release decreases, which can also affect the qualitative composition of radionuclides and the overall activity of the release to the environment.

4.2. DEC-A analyses

4.2.1. LOCA DEC-A calculations

Simulations with ATHLET/COCOSYS for the DEC-A LOCA scenarios include leaks 2 × 850, 350, 100, 50 mm for VVER-1000 and leaks 2 × 500, 250, 100, 50 mm for VVER-440. Unlike DBA, for DEC-A LOCA simulations nominal primary and secondary initial parameters are adopted. Boundary conditions are the same as those of the DBA, except the containment spray system is completely inoperative.

To define scenario with maximum peak cladding temperature a sensitivity analysis was fulfilled by varying the leak discharge coefficients. Cladding temperature curves of double ended guillotine breaks with two specific peaks for both VVER-1000 and VVER-440 are shown in Fig 16).

The main LOCA DEC-A calculation results with ATHLET/COCOSYS that may challenge the FP release are presented in Table 6. The parameters of liquid and water discharge into the containment, ECCS injection into the primary are given over a period of 1500 s. The activity releases to the environment are shown in Table 6 at 1500 s of transient for VVER-1000 and at 1500 s, 30000 s for VVER-440. Progression of FP release to the environment for VVER-100 and VVER-440 are shown in Fig. 17.

At the beginning of the transient, the main part of the released activity from primary into the containment is concentrated in the form of aerosols. Since the work of all spray pumps is not considered, the removal rate of radionuclides from aerosols into water is significantly reduced in comparison with DBA case.

The FP mass and activity in the containment atmosphere practically do not decrease during the transient. Thus, the release of the FP air fraction into the environment will continue until the pressure in the containment falls below the ambient pressure.

4.2.2. PRISE DEC-A calculations

Simulations with ATHLET for the DEC-A PRISE scenarios include leaks as result of SG collector cover lift-up, as well as due to guillotine break of SG tubes (see par. 2.1).

As a result of the PRISE leak primary coolant with FP passes to emergency SG and then releases to the environment through two stack open relief valves (postulated failures). Three ECCS trains operation contribute to a high injection flow rate resulting in rapid emptying of the ECCS tanks with permanent loss of cooling water through the SG break (Fig. 18). Appropriate measures must be taken by personnel to terminate irretrievable losses of the coolant and reduce the radiation release. Some of such actions are simulated after 1800 s (see par. 2.1). It allows maintaining the possibility of long-term heat removal from the reactor core and minimizing the release to the environment.

The main PRISE DEC-A calculation results with ATHLET that may challenge the FP release are presented in Table 7. The parameters are given over a period of 3600 s. Progression of FP release to the environment for VVER-1000 and VVER–440 are shown in Fig. 19. Results demonstrate that the release into the environment is from 16 to 100% of the primary coolant activity for VVER-1000 and from 43 to 100% for VVER-440.

In the course of the analyzed transient processes, the clad

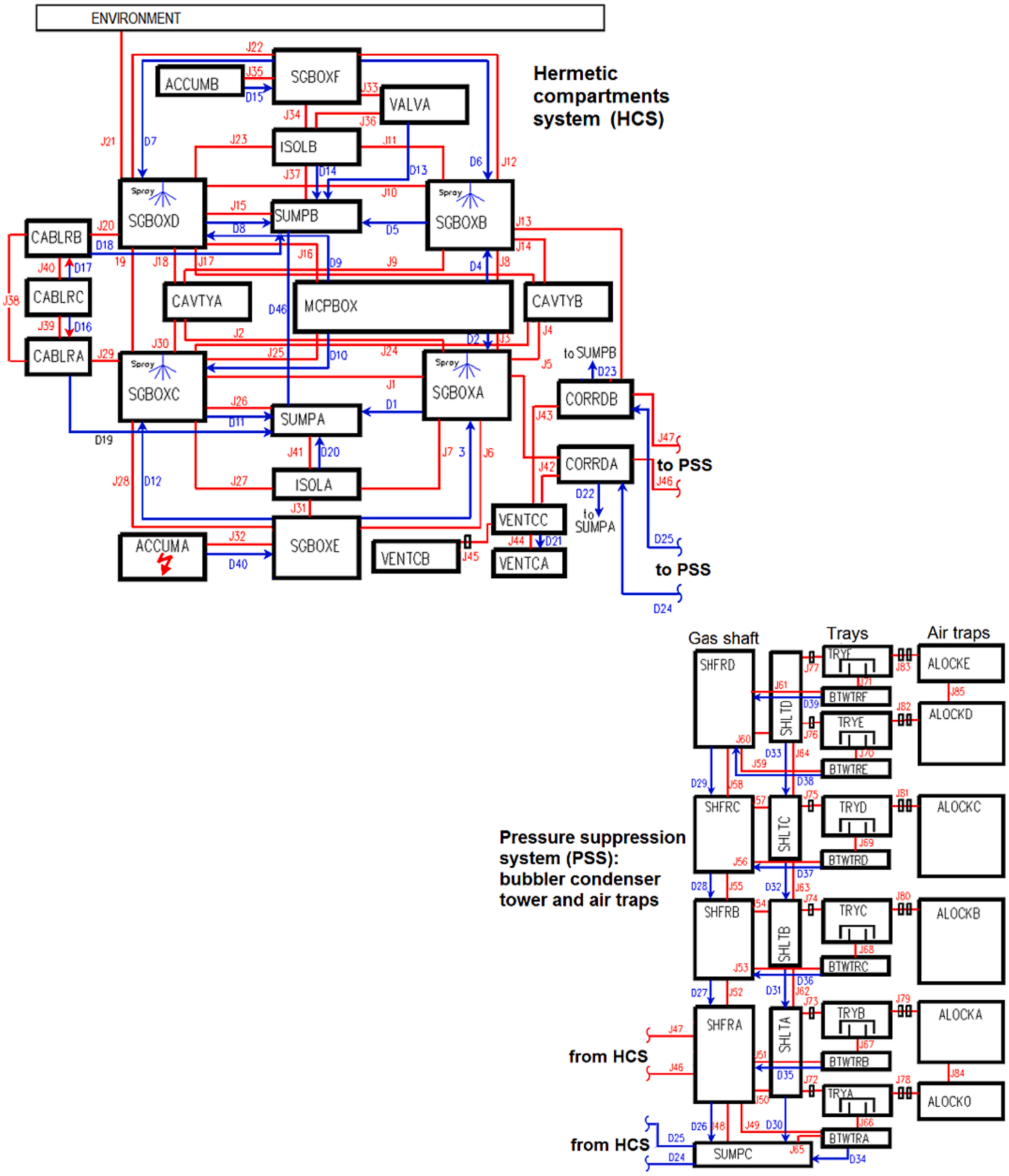

Fig. 10. VVER-440/V-213 COCOSYS nodalization.

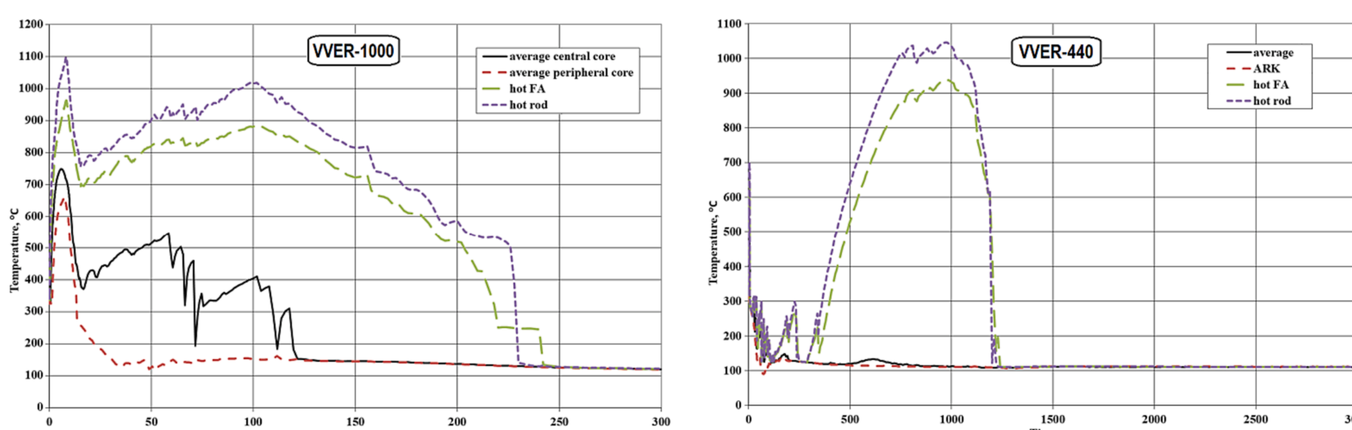

Fig. 11. Maximum outside surface temperature of fuel rod claddings.

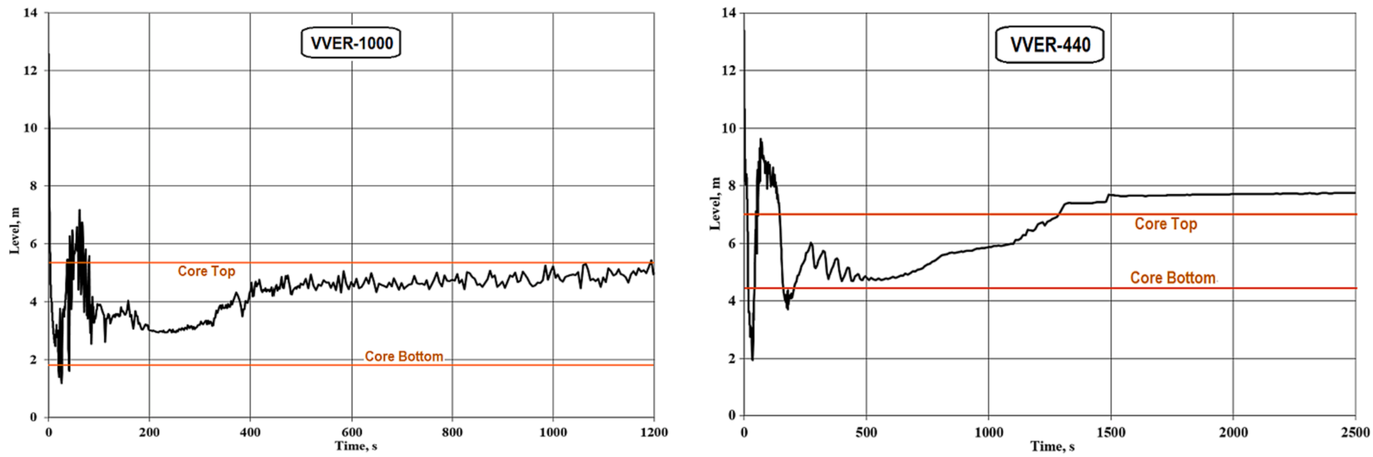

Fig. 12. Level of coolant in reactor.

Table 4
Main results of LOCA DBA calculations with ATHLET/COCOSYS.

Parameters	VVER-1000				VVER-440			
	2 × 850	350	100	50	2 × 500	250	100	50
Max peak cladding temperature, °C	1098	813	354	354	1045	389	338	338
Percentage of clad oxidation depth (hot rod/ hot FA), %	1.2/ 0.47	0.15/ 0.14	0.14	0.14	2.3/0.2	0.153	0.153	0.153
Mass of hydrogen generated, g	7.5	0.01	0	0	3.8	0	0	0
Liquid release to containment, t	590	987	284	274	454	561	36	201
Vapour release to containment, t	111	31	39	0.02	31	13.4	2.5	0
ECCS injection into primary, t	478	696	150	130	373	476	239	160
Duration of FP release from primary to containment, sec	24	100	1170	1383	30	72	650	1190
Max containment pressure, kgf/cm^2	3.7	3.1	2.2	1.8	1.8	1.8	1.54	1.8
Max containment temperature, °C	138	141	110	91	119	118.5	110.5	98
Maximum release flow rate to environment, g/ s	1.2	1.1	0.7	0.55	24.5	24.5	22	20
Activity release to environment (at 1500 s/ 30,000 s), Bq	5.7e + 11	5.4e + 11	7.3e + 8	3.6e + 8	7.5e + 12/9.5e + 13	9.7e + 12/1.2e + 14	2.5e + 13/1.2e + 14	2.0e + 13/1.3e + 14

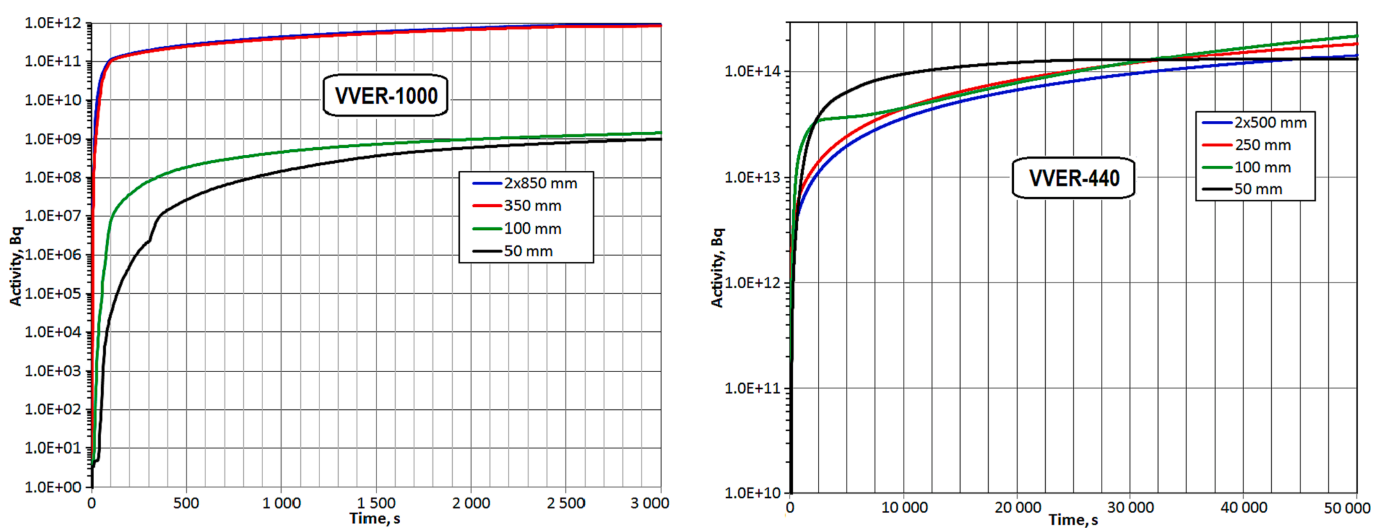

Fig. 13. FP release to the environment.

temperature did not increase significantly, and therefore no conditions are reached for damage to the fuel cladding. Therefore, only the activity of the primary circuit is considered in the FP release to the environment. The duration of FP release to SG does not take into account the dilution of the primary coolant with ECCS water, which could significantly reduce the radioactive release and extend it over time. Also it should be noted that with a decrease in the leak diameter, the water fraction of the release decreases, which can also affect the qualitative composition of radionuclides and the overall activity of the release to the environment.

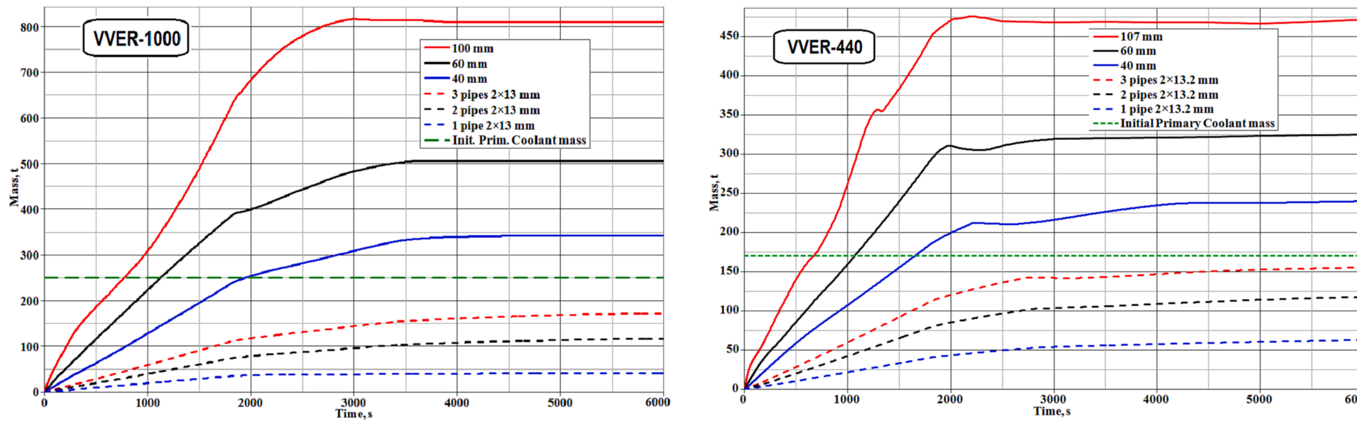

Fig. 14. Mass flow of coolant through SG break.

Table 5
Main results of PRISE DBA calculations with ATHLET.

Parameters	VVER-1000						VVER-440					
	100	60	40	3 × 13	2 × 13	1 × 13	107	60	40	3 × 13	2 × 13	1 × 13
Coolant mass to SG/ environment, t	814/ 755	506/ 443	335/ 273	156/125	105/122	40/91	468/ 449	320/ 308	228/ 191	144/107	106/68	56/51
ECCS injection into primary, t	843	641	617	451	422	339	451	386	358	275	269	213
Duration of FP release to SG, s	778	1134	1971	>9000	>9000	>9000	664	1064	1655	>9000	>9000	>9000
Water mass to environment, t	670	369	184	0	0	0	442	296	161	67	21	0
Steam mass to environment, t	56	74	89	125	122	91	7	16	30	40	47	51
Activity release to environment, Bq	2.4e + 14	2.4e + 14	2.4e + 14	1.5e + 14	1.0e + 14	3.9e + 13	4.6e + 13	4.6e + 13	4.6e + 13	3.9e + 13	2.9e + 13	1.5e + 13

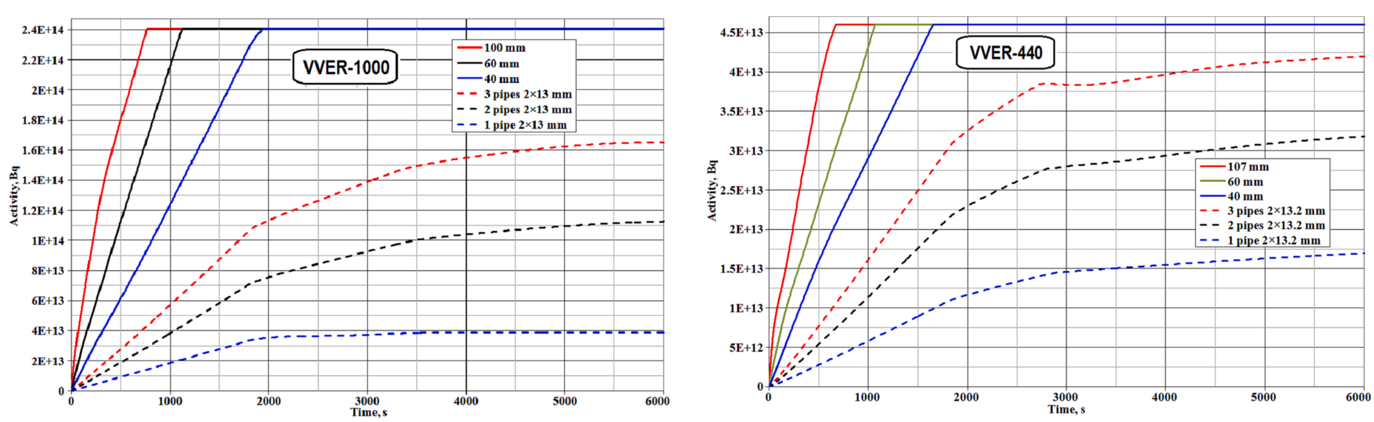

Fig. 15. FP release to the environment.

5. Conclusions

The aim of this paper was to present preliminary results of evaluating the FP release into the environment under DBA and DEC-A conditions for LOCA and PRISE events with a conservative approach to specifying the main contributors to the source term for pressurized water VVER–440 and VVER–1000 reactor types.

For both DBA and DEC-A the entire contents of the fuel rods gas gap plus maximum permitted primary circuit coolant activity at the beginning of the transient are assumed to be released from the primary under the LOCA events. Design primary circuit coolant activity (including spike) defines the source term for the PRISE events. FP deposition and dilution in the primary coolant with ECCS boric water is not taken into account. The rate of the FP release from the primary circuit is calculated in proportion to the mass release of the coolant.

For both DBA and DEC-A LOCA the operation of the containment recirculation ventilation systems is not taken into account. For PRISE events, selected personnel actions were simulated to prevent the complete loss of the ECCS water needed for core cooling and to reduce the release of the FP to the environment.

As a general rule, safety analyzes focus only on leaks with the maximum rupture diameters, as the worst cases with the most adverse consequences. Consideration of the ranges of possible leak diameters made it possible to assess the change in quantitative indicators, as well as to see the general trends in the change in radiological consequences depending on the size of the leak. For accidents with a leak from the

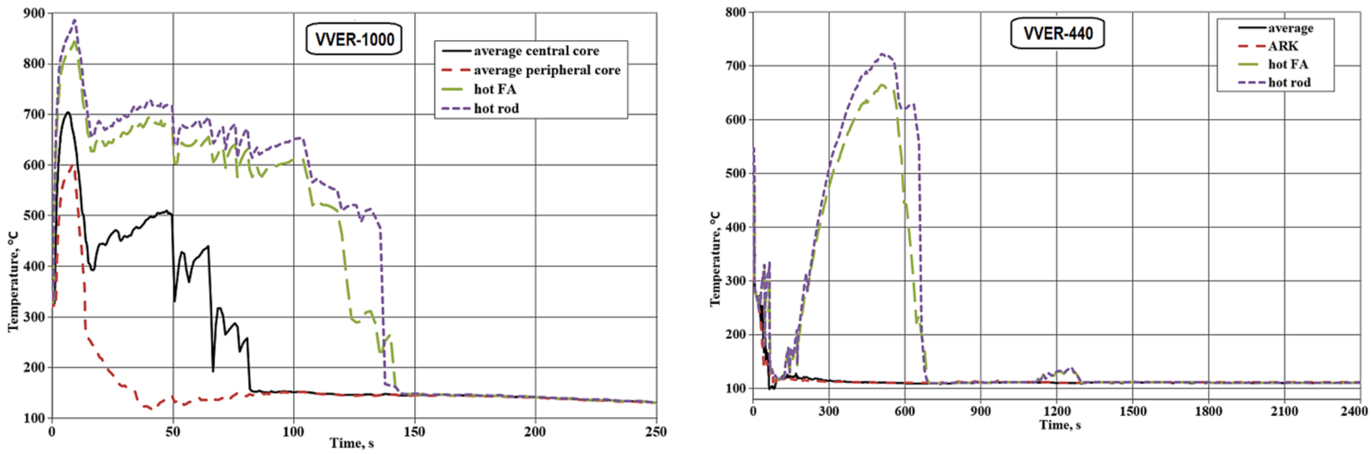

Fig. 16. Maximum outside surface temperature of fuel rod claddings.

Table 6
Main results of LOCA DEC-A calculations with ATHLET/COCOSYS.

Parameters	VVER-1000				VVER-440			
	2 × 850	350	100	50	2 × 500	250	100	50
Max peak cladding temperature, °C	886	343	346	345	722	379	335	335
Percentage of clad oxidation depth (hot rod/ hot FA), %	0.21/ 0.18	0.14	0.14	0.14	0.2/0.15	0.15	0.15	0.15
Mass of hydrogen generated, g	1.83	0	0	0	0.1	0	0	0
Liquid release to containment, t	510	1088	303	274	490	677	368	200
Vapor release to containment, t	97	29	33	0	23	9	2	0
ECCS injection into primary, t	540	940	166	142	401	581	272	169
Duration of FP release from primary to containment, s	6	97	844	1391	33	74	330	1250
Max containment pressure, kgf/cm^2	3.7	3.1	2.5	2.3	1.74	1.84	1.62	1.3
Max containment temperature, °C	137	139	108	102	116	121	116	98
Maximum release flow rate to environment, g/ s	1.2	1.0	0.8	0.7	20.0	20.0	20.0	20.0
Activity release to environment (at 1500 s/ 30,000 s), Bq	1.1e + 12	1.2e + 12	1.2e + 12	5.8e + 11	7.2e + 12/1.3e + 14	7.5e + 12/1.5e + 14	8.9e + 12/1.7e + 14	1.9e + 13/1.3e + 14

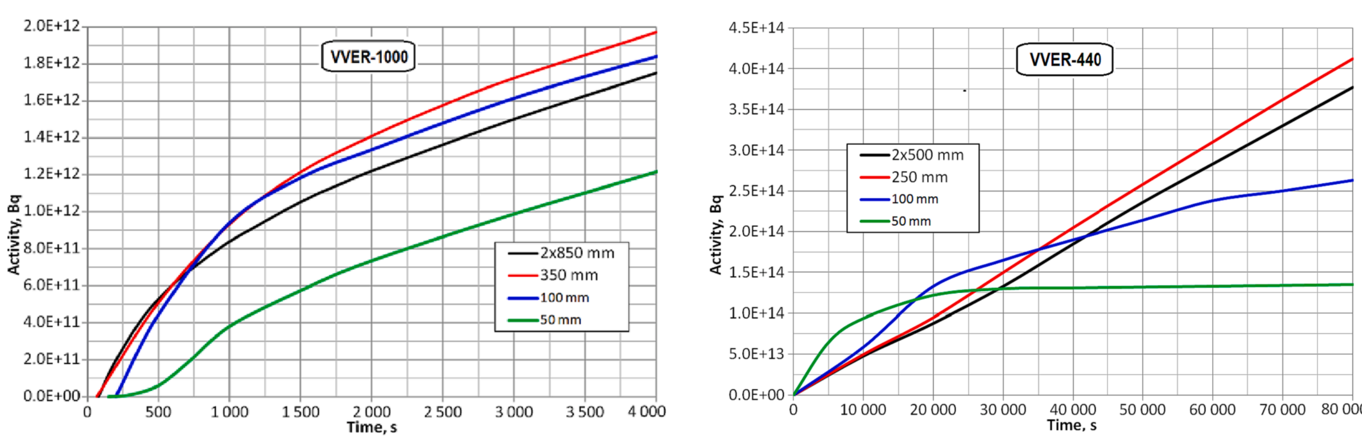

Fig. 17. FP release to the environment.

primary to the secondary, the results obtained confirmed that leaks with a maximum diameter cause the worst consequences in terms of FP release to the environment (Fig. 19). In the case of LOCA scenarios, the situation is quite different. Within the selected conservative approach and assumptions, the above pattern is valid only for LOCA DBA VVER-1000 (Fig. 17). The results obtained showed that for LOCA DBA VVER-440, as well as for LOCA DEC-A VVER-1000 and VVER-440, leaks with a maximum diameter are not the “worst”, i.e. do not lead to the worst results (Figs. 15, 19). For LOCA DEC-A VVER-1000, the failure of the containment spray system leads to the worst consequences for a leak with a diameter of 350 mm. For LOCA DBA and DEC-A VVER-440, the features of a pressure suppression system (bubble condenser tower) have a significant impact on the worst radiological consequences in terms of the dominance of smaller leak diameters at various stages of the accident.

Significant increase in the cladding outer surface temperature occurred only at large leaks 2 × 850 mm (1098 °C), 350 mm (813 °C), 2 × 500 mm (1045 °C, VVER-440) and only for the hot channel. Such

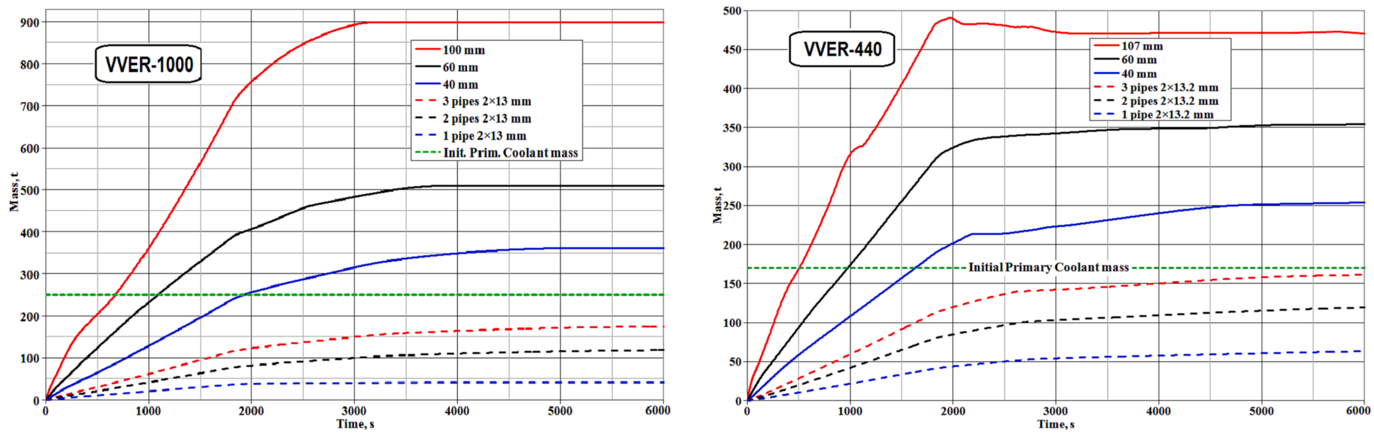

Fig. 18. Mass flow of coolant through SG break.

Table 7
Main results of PRISE DEC-A calculations with ATHLET.

Parameters	VVER-1000						VVER-440					
	100	60	40	3 × 13	2 × 13	1 × 13	107	60	40	3 × 13	2 × 13	1 × 13
Coolant mass to SG/ environment, t	899/ 843	506/ 444	339/ 278	160/138	107/132	40/91	470/ 452	347/ 326	233/ 185	147/94	107/83	56/64
ECCS injection into primary, t	921	663	565	437	369	319	477	371	374	275	255	223
Duration of FP release to SG, s	675	1091	1930	>9000	>9000	>9000	463	915	1544	5840	>16000	>16000
Water mass to environment, t	774	350	169	0	0	0	437	299	144	43	29	0
Steam mass to environment, t	69	94	109	138	132	91	15	27	41	51	54	64
Activity release to environment, Bq	2.4e + 14	2.4e + 14	2.4e + 14	1.5e + 14	1.0e + 14	3.9e + 13	4.6e + 13	4.6e + 13	4.6e + 13	4.0e + 13	2.9e + 13	1.5e + 13

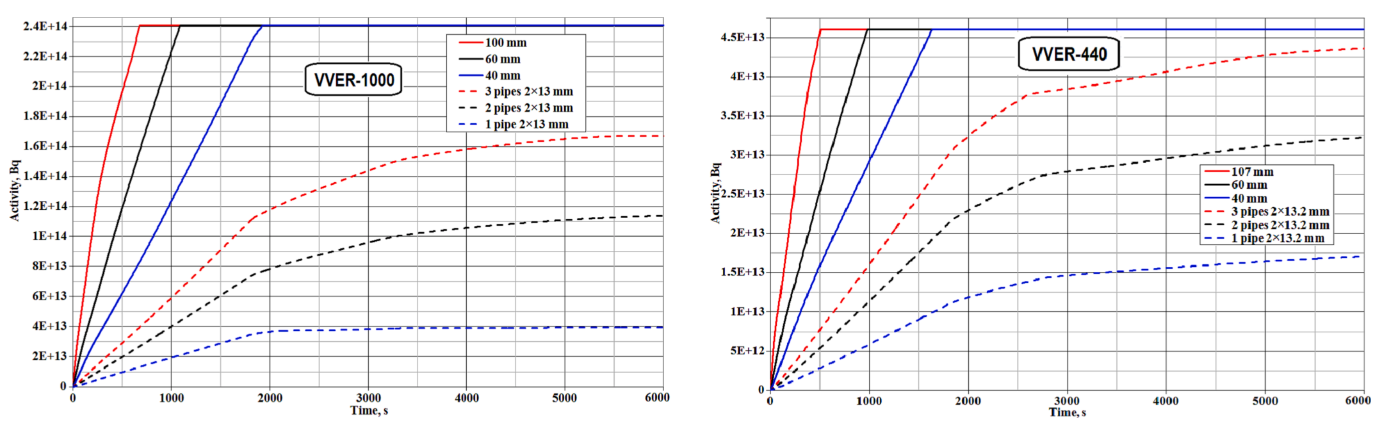

Fig. 19. FP release to the environment.

temperatures can only lead to shape deformation (only clad ballooning) of the fuel cladding. Because the conditions for fuel cladding damage are not reached during the analyzed LOCA transients, this can be used as a technical basis for revising the FP source term of the LOCA events.

Since the methodologies used to analyze LOCA and PRISE events are very conservative, it would be advisable to take into account the following aspects in future studies to reduce this conservatism:

- accounting for the dilution of the primary coolant with ECCS boric water;
- accounting for the containment recirculation ventilation system operation;
- for LOCA, considering use for the FP release estimate of the activity of damaged fuel rods only instead of the activity of the entire core FP inventory in the fuel gas gap;
- for PRISE, considering optimized emergency actions of the plant personnel.

The mentioned assumptions will provide more realistic estimates of the FP release into the environment by reducing excessive conservatism in the applied approaches for analysis of LOCA and PRISE events in DBA and DEC-A.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

Data will be made available on request.

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