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D3.3 – Definition report of SB LOCA + SG tubing break benchmark

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Summary

This Deliverable 3.3 on “Definition report of SB LOCA + SG tubing break benchmark” is a part of CAMIVVER, WP3, Task 3.3: “SB LOCA specification + SG line break specification”. The report describes plant specific scenario for the VVER-1000 in accordance with the CAMIVVER Grant agreement, NUMBER 945081 [1].

This report is serving as a definition report of a transient which will be analysed in the WP7.3. The main initiating event is selected to be a Small Break Loss of Coolant Accident (SB LOCA) located in the cold leg between a main coolant pump (MCP) and the reactor vessel inlet. Additionaly, a steam generator tubing break in one of the SGs which do not belong to the damaged loop is induced in the scenario from the beginning of the transient.Simultaneously with both initiating events is assumed station blackout (SBO) which will simplify the transient and also will lead to more severe conditions.

Consecutively in the report are presented plant specific initial and boundary conditions, as well as the expected processes, phenomena and acident progressions for the considered initiating events if they ocure independently.

The reference nuclear power plant (NPP) is Kozloduy NPP Unit #6 equiped with a VVER-1000 / V320 reactor type. All geometric data and plant specific equipment characteristics are presented in the Deliverable D3.2 [2]. In the plant transient modelling parameters for the fuel cycle #8 will be used.

The selected combination of events will allow to investigate important phenomena and processes during the accident progression. The calculated important plant parameters by different codes will be used in organizing a benchmarck and will help in developing and updating models for RELAP5, TRACE and CATHARE3 computer codes. It is noticeable that among these codes, the RELAP5 relies on several validation works concerning NPP tests and transients analyses, for instance a PSB integral facility simulating VVER-1000 behaviour [4]. In this way the objective of Task 7.3 is to check TRACE and CATHARE3 availability to produce consistent results compared to RELAP5. Task 7.3 will also allow evaluating improvements related CATHARE3 and TRACE advanced models or related to 3D vessel modelling.

The loss of primary circuit coolant will challenge core cooling especialy when this event is attended by station blackout. This challenge will have a higher magnitude at the end of fuel cycle. From the other side loss

of primary coolant will cause primary pressure reducing below the secondary side pressure, which will cause reversing of flow from the secondary side to primary circuit. This will lead to boron dilution in primary coolant if the event occurs in the beginning of fuel campaign. Opposing these two processes requires aditional analysis for selecting the plant state in developing the scenario. All this is discussed in this document and as a result, the end of the fuel cycle was chosen due to the greater risk to the nuclear power plant unit.

The main objectives of this Deliverable is to prepare a plant specific scenario for the VVER-1000 and to provide information of all equipment and systems expected to be initiated during the progression of accident including operator actions if they are assumed based on existing emergency operating procedures.

Approval

Version	First Author	WP leader	Project Coordinator
1	P. Groudev (INRNE) 30/08/2021	P. Groudev (INRNE) 30/08/2021	D. Verrier (Framatome) 30/08/2021

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Abbreviations

AC/DC	Alternating current / Direct current
ASSL	Automatic Step by Step Load
AFW	Auxiliary feed water
BDBA	Beyond Design Basis Accident
BOC	Beginning of cycle
BRU-A	Steam dump to atmosphere
BRU-K	Steam dump to condenser
BZOK	Fast Acting Isolating Valve (FAIV)
CSF	Critical safety functions
DBA	Design basis accident
DG	Diesel Generator
ECCS	Emergency core cooling system
EOC	End of fuel cycle
EOPs	Emergency operating procedures
IE	Initiating event
FW	Feed water
HAs	Hydro accumulators
HRS	Heat removal system
HPP	High pressure pump
INRNE	Institute for Nuclear Research and Nuclear Energy
LOCA -	Loss of coolant accident
MCP	Main Coolant Pump
MSH	Main Steam Header
MSIV	Main Steam Intercept Valve
NPP	Nuclear power plant
PSB	Large-scale integral test facility simulating behaviour of NPP with VVER-1000 reactor
PRZ	Pressurizer
PWR	Pressurizer water reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RL	Auxiliary feed water
RPV	Reactor pressure vessel
RV	Relief valve

RPLC	Reactor Power Limitation Controller
RPC	Reactor Power Controller
SAMG	Severe Accident Management Guidelines
SAR	Safety analyses report
SB LOCA	Small Break Loss of Coolant Accident
SBO	Station blackout
SCRAM	Emergency shutdown of the reactor (Safety control rod assembly moving)
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SVs	Safety Valves
TB10	System for boron injection
TK pumps	Make-up System
TSV	Turbine Stop Valve
TQ12, 22, 32	Low pressure injection pumps
TQx2	Low pressure system injection
TQx3	High pressure system injection
TQx4	High-High pressure system injection (piston type)
TQ40S	Valve for connection of HRS to primary side
TQ40S08, 09	HRS safety valves
VVER	Water-Water Cooled Reactor
WP-1	Warning Protection #1

1. Introduction

This Deliverable 3.3 on “Definition report of SB LOCA + SG tubing break benchmark” is a part of CAMIVVER, WP3, Task 3.3: “SB LOCA specification + SG line break specification”. The report describes plant specific scenario for the VVER-1000 in accordance with the CAMIVVER Grant agreement, NUMBER 945081 [1].

The activities in Task 3.3 are mainly oriented to provide data for WP7. In WP7 is considered to be performed Small Break Loss of Coolant Accident (SB LOCA) inducing steam line pipe break in one of SGs which do not belong to the damaged loop. Simultaneously with initiating events are assumed station blackout which will allow the simplification of transient and leading to more severe conditions. The selected combination of events will allow to explore phenomena which will cause positive reactivity injection in the reactor core due to leakages of clean water from secondary side of damaged steam generator (SG) to the borated coolant in primary circuit. To have a high magnitude of such phenomenon is needed the accident to happen in the beginning of fuel cycle, when the boron concentration is higher. From the other side in the beginning of cycle the residual power is lower, while the maximum residual power is observed in the end of cycle. If the SB LOCA happens at the end of fuel cycle simultaneously with a station blackout, the main concern will be removing a residual heat and “core cooling” Critical Safety Function (CSF). Based on this, the most challenging conditions are at the end of cycle.

The loss of primary circuit coolant will cause primary pressure reducing below the secondary side pressure, which will cause a reverse flow from secondary to primary circuit. This will lead to boron dilution in primary coolant. The objective of this task is to prepare plant specific scenario for the VVER-1000 and to provide information of all equipment and systems expected to be initiated during the progression of accident including operator actions if they are assumed based on existing emergency operating procedures. The selected accident is important due to the expected phenomena as a possibility of not symmetric reactivity insertion in reactor core. This will allow to investigate capabilities to simulate of 3D kinetics and will be checked by comparing code to code. The scenario is important also for analysis with 1D system codes, due to the importance of assessing on overall jump of reactivity and reactor power in such situation. The suggested scenarios will allow to investigate stratification flow in coolant loops, loops seal cleaning, reactor core cool down at natural circulation, loss of natural circulation as well as integral effects of primary and secondary circuits. The selected accident is interesting, as the initiating events are considered separately as postulated events in the safety analyses report (SAR). The combination could involve practically challenging of all critical safety functions (CSFs). It is important that primary to secondary leakage is the accident that could lead directly to bypassing the containment and radioactive release. The benchmark will allow to investigate modelling of natural circulation, reactor core uncovery, heat-up of reactor core internals, SG pipe line breaks as well as integral effects important to the safety assessment, by comparing code to code.

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2. Description of the initiating event

The initiation of transient is occurred with simultaneous combination of two specific leakages. The first one is a loss of coolant from primary circuit due to a small break at cold leg, while the second one is from primary to secondary side which appears due to a single tubing break. Both are defined as postulated events. In addition to both breaks, a total loss of power (station blackout (SBO)) is considered. It will simplify the transient and will create more severe conditions, which will lead to reactor core heat up.

For a better understanding of the plant responses to the combination of the selected initiating events, below are described separately the main phenomena and processes provoked independently, if each one initiating event occurs. The combination and interference of these phenomena and processes of the initiating events will result in the preparation of the expected scenario.

The primary leakage provokes a loss of primary inventory. The primary pressure decreases and reaches a value to which the reactor SCRAM is actuated. As a result, the turbine stop valves are closed and steam generator feed-water system is stopped. The secondary pressure begins to rise and reaches the set point of BRUK opening. They remain open and regulate the secondary pressure close to the pressure before TSV closure. The SG water level decrease leads to the switch-on of Auxiliary feed-water system, which restore and maintain SG levels about the nominal. RCP trip and Primary make-up/blow-down is isolated, when the sub-cooling margin is violated. The primary coolant leakage and its non-compensation leads to core uncovering, fuel heat up and violation of the fuel integrity. This section describes the main characteristics and assumptions in preparation of scenario for VVER-1000, Unit 6 KNPP. The selection of reference NPP and corresponding unit is determined based on the information prepared in Deliverable 3.2 [2].

2.1. Defining of main phenomena and processes in case of SB LOCA at cold leg

2.1.1. Break size determination

The selection of a break size and its location is based on simulation of the specific expected phenomena and processes explained in the chapter above. The break size is determined based on the experience in performing of SB LOCA transients and existing literatures. In most of the analyses the place of the break is located at cold leg between the RPV and MCP due to maximum loss of coolant and avoiding the core after the SG. The selection of size is consistent with avoiding effective work of HAs. Based on that, the size is selected below the range of 40-60 mm, where the effectiveness of HAs is reduced sufficiently in injection of borated water, what we try to avoid. The HAs are designed to work effectively above this margin. The reason for this is to avoid borating of primary circuit in case of reversing flow from SG secondary side to primary circuit. This could have effect mainly in case of transient in the beginning of fuel cycle for challenging subcriticality. In case of calculation at the end of fuel cycle there will be no insertion of positive reactivity due to the missing of boron in primary circuit, but will challenge significantly core cooling in case of failure of HPP (TQx3) or station blackout. As the primary objectives of this task is creating transient with maximum challenge to the reactor coolant system, it should be selected EOC.

In case of significant integral flow rate from SG to primary circuit before injection of HAs, the selected transient will be interesting for further assessment at BOC.

The cold leg break is selected with an equivalent diameter of 30 mm with a downward orientation and with a location between the TQx3 pump injection point and reactor vessel (RCP's discharge side). The break discharge coefficients (DC) are accepted at default i.e. all DC=1.0. This is important in comparison of integral break flow rates calculated with different computer codes.

2.1.2. Main phenomena and processes during a SB LOCA

The cold leg break accident is characterized with a loss of primary inventory and depressurization of primary circuit. This usually leads to decrease of the reactor and pressurizer water levels and respectively to the heat up of the reactor core. One of the first effects of the primary depressurization reflects on reducing of saturation coolant temperature and losing of subcooling of 10 °C. Additionally, reducing of coolant flow through the reactor core due to the leakage of primary coolant before entrance in the reactor vessel, leads to increasing of core coolant temperature. The loss of subcooling dT= (Ts-Tf) < 10°C activates the safety systems by activation of program “Automatic Step by Step Load”. All safety injection system will start to work on recirculation mode and will inject borated coolant in the primary circuit after reaching their corresponding points. This set point will lead to switching off all MCPs and transition to natural circulation.

The reactor SCRAM will be activated due to reducing the primary pressure to the certain point or increasing of core exit temperature to the certain point or after losing of subcooling and switching off 3 out of 4 MCPs when reactor power is above 75%. As a result, the turbine stop valves (TSV) are closed and steam generator feed-water system is stopped, too. The isolation of turbogenerator will happen 15 seconds after the reactor SCRAM. The secondary pressure will begin to rise and will reach the set point of BRU-K opening. They will remain open and will regulate the secondary pressure close to the pressure before TSV closure. The SG water level decrease leads to the switch-on of Auxiliary feed-water system, which restore and maintain SG levels about the nominal. The logic of BRU-K operation is presented and explained in Deliverable 3.2 [2].

After reaching 5.88 MPa, injection of borated water from hydro accumulators (HAs) will be activated. All four HAs will start to inject borated water in upper and down comer volumes. The SB LOCA is characterized with establishing of a primary side pressure plateau after reaching a secondary pressure and after some time will continue further reduction. The time for supporting such plateau will depend mainly from a break size and blow down rate.

The make-up and let down system will try to support primary pressure and Pressurizer water level by increasing TK pumps flow rate from 25 m^3/h to the 80 m^3/hr. The maximal flow rate is 120 m^3/h, but in the analyses, is assumed maximal flow rate of 80 m^3/h. Primary make-up/let-down system is isolated, when the sub-cooling margin is violated.

If the water level in pressurizer drops below 4.0 m all PRZ heaters will switched off.

After establishing of a natural circulation, the residual heat will be removed mainly by steam generators work at natural circulation and partially by the leakage of the coolant. Loss of coolant will lead to loss of natural circulation, dry out of hot legs, with subsequently core uncovery and dry out of cold legs, until complete dry out of the reactor core. The heat-up of reactor structures will be observed with a losing of heat remove capability. After reaching 600.0 °C reactor core exit temperature, it will be expected beginning of hydrogen generation which will increase rapidly after reaching of 1200.0 °C.

During the transient progression it will be observed stratification of coolant in the cold leg and if a break is located in the middle of the cold leg elevation, the break flow rate will start to decrease after reducing the level below the level of break due to voiding of the break. In this case the break is located on the bottom of cold leg (down oriented) and this effect will not be observed.

The collecting of coolant at the lower part of cold leg called “seal”, the cooldown of core will be reduced until evaporation of this water or “clean up” of cold legs.

The calculation usually is terminated at around 1200.0 °C if there is no any injection from active safety systems. At that moment the reactor core will be fully uncovered and plant state will be in the area of SAMG application, which will start after 650.0 °C.

2.1.3. Assumptions

The 30 mm cold leg break located between RPV and MCP is assumed as accident initiator.

- Reactor power, pressure and temperature are assumed equal to nominal values.
- The reactor protection system operation is assumed.
- Plant state correspond to EOC.
- Steam bypass (BRU-K) operation by Program 1 is allowed. BRU-K will open and maintain the steam header pressure equal to the pressure 1 sec before turbine stop valves closing.
- In this scenario, BRU-SN is not taken into account, because BRU-K valves keep secondary pressure close to this of regulation by BRU-SN.
- Auxiliary Feed Water System (RL pumps) is available for SG feeding.
- After appearance of "YZ" signal for unit protection, sealing water delivery to the RCP's is stopped. As a bounding assumption no operator actions for closing the sealing letdown isolation valves are undertaken and from RCP's shaft seals a leakage will begin with a 2.5 m^3/h.
- Primary Make-up and blow down system are isolated when sub-cooling margin is violated.
- One HPP (TQx3) is available with injection not in the damaged loop.

2.1.4. Expected scenario for SB LOCA

1. Cold leg break occur close to the RPV with an equivalent diameter of 30 mm (ID 30).
2. Pressurizer heaters switch on automatically to support primary pressure.
3. Flow rate of Make - up pumps (TK) increasing automatically. The flow rate is set to reach a maximum of 80 m^3/h (intentionally it is set less than the designed flow of 120 m^3/h). With an assumed density of 997 kg/m^3, the make - up system (i.e. pump TK) delivers 22.1 kg/s. Injected water from the makeup system is heated to a fluid density of 797 kg/m^3 by the makeup drain line heaters. Primary make-up/blow-down is isolated, when the sub-cooling margin of 10 °C is violated.
4. Reactor SCRAM initiated by low primary pressure or high core exit temperature or by loss of 3 from 4 MCPs due to loss of subcooling.
5. MCPs are switched off due to loss of subcooling.
6. Activation of ASSL program after loss of subcooling of 10 °C – all safety systems are prepared to work.
7. Turbine Stop Valve (TSV or MSIV - Main Steam Intercept Valve RA 11 -14 SO1) tripped and isolated Turbine.

8. The secondary pressure begins to rise and reaches the set point of BRU-K opening. They remain open and regulate the secondary pressure close to the pressure before TSV closure.

9. The SG water level decrease leads to the switch-on of Auxiliary feed-water system, which restore and maintain SG levels about the nominal.

10. Starts to cooldown reactor coolant system by injection of borated water with boron concentration of 16g/kg

by one HPP (TQ13) pump after reaching its set points of 10.79 MPa (110 kgf/cm^2).

11. If the pressure drops to 5.88 MPa (60 kgf/cm^2) it will activate all HAs.

12. Calculation termination after establishing stable decreasing of core exit temperature.

In case of safety injection systems failure, the primary coolant leakage and its non-compensation will lead to core uncovering, fuel heat up and violation of the fuel integrity.

2.2. Defining of main phenomena and processes in case of SG tube line break.

The investigated accident is considered as a postulated. Such accidents could provide a direct release path for contaminated primary coolant to the environment via the secondary side. Accumulation of water in the secondary side can also lead to an overfill condition which can severely aggravate the radiological consequences and increase the likelihood of complicating failures. Timely operator intervention is necessary to limit the radiological releases and prevent steam generator overfill.

2.2.1. Selecting of break size and location.

The break size depends on the number of failed tubing in a SG. It is possible to have primary to secondary side leakage in case of SG collector head lifting. Such initiating event is considered with low level frequency compared to tubing break. The initiating event of the analyses is selected to be a double-ended one-pipe rupture with equivalent diameter 13 mm (ID 13) in a Steam Generator #1 close to the cold collector.

2.2.2. Modelling Assumptions.

The broken tube is located in the middle row of the tube bundle. The base model should be modified according to this calculation work package to represent the SGTR. It will be appropriate only the broken tube to be directly affected by the break. Fluid velocity in the modeled broken tube should be independent from the fluid velocity in the intact tubes. Likewise, heat transfer across the walls of the broken tubes is independent from that of the intact tubes. In a real steam generator, it will be expected the fluid velocity in the broken tube to be greater than in the intact tubes.

2.2.3. Nuclear power plant expected behaviour during SG tube rupture.

The response of the VVER-1000/V320 nuclear steam supply system to Steam Generator Tube Rupture (SGTR) initiating event is characterized with a very specific feature. Since the primary system pressure is initially much greater than the steam generator pressure, reactor coolant flows from the primary into the secondary side of the affected steam generator. In response of this loss of reactor coolant, the pressurizer level and RCS pressure decrease. Normally, charging flow automatically will increase and pressurizer heaters will energize in an effort to stabilize pressure and level. However, if leakage and coolant cooling shrinkage exceed the capacity of the charging system and the pressurizer heaters, reactor coolant pressure will continue to decrease and eventually lead to an automatic reactor trip signal. For the expected case however, break flow rate will not lead to the rapidly decreasing of pressure so that reactor trip on low primary pressure will not occur.

On the secondary side, leakage of contaminated primary coolant will increase the activity of the secondary coolant resulting in high radiation indications from the air ejector radiation monitor, blow down line radiation monitors and main steam line radiation monitors. As primary coolant accumulates in the affected steam generator, normal feed water flow is reduced automatically to compensate high steam generator level. Consequently, a mismatch between steam flow from and feed-water flow to the affected steam generator may be observed. This potentially provides early confirmation of a tube failure event and also it identifies the affected steam generators. The water flow level in the affected steam generator may not be significantly greater than the intact steam generators prior to reactor trip, as the normal feed-water control system automatically compensates the changes in steam flow rate and steam generator level due to primary to secondary leakage. The steam generator level may drop following the reactor trip.

The automatic systems alone will not terminate the primary to secondary leakage. Once a tube failure has been identified, recovery actions begin by isolating feed-water flow to the affected steam generator, switching off the MCP of this SG and isolating steam flow after depressurization to 70 kg/cm^2.

The main differences between Westinghouse strategies for SGTR vs. VVER SGTR strategy is in the earlier isolating of steam flow from the affected SG in Westinghouse Reactors. In VVER approach, the affected SG is isolated from MSH after depressurization of primary side to 70 kg/cm^2. In this way the operators prevent the possibility for opening of damaged SG SVs (or its BRU-A) and prevent radiological release in the environment. Keeping the primary pressure higher than the pressure in the damaged SG, the operators support a small leakage from primary to secondary side to prevent filling primary side with not borated water. After reaching the safety concentration of the boric acid in the primary circuit, there is no problem to terminate leakage from primary to secondary side.

2.2.4. Expected scenario in case of SGTR

1. Steam Generator Tube Break with ID 13 in the middle level of tubing package at cold leg side.

2. Pressurizer heaters switch on automatically for supporting PRZ water level and primary pressure. (PRZ heaters will continue to work to the end of calculation).

3. Flow rate of Make - up pumps (TK) increasing automatically. The flow rate is set to reach a maximum of 80 m^3/h (intentionally it is set less than the designed flow of 120 m^3/h as never is observed flow rate above 100 m^3/h). With an assumed density of 997 kg/m^3, the make - up system (i.e., pump TK) delivers 22.1 kg/s. Injected water from the makeup system is heated to a fluid density of 797 kg/m^3 by the makeup drain line heaters.

3. Reactor SCRAM initiated by the operator (The action is taken because of radiation detected on the secondary side of the Steam Generator). Estimated time is 10 minutes after the beginning of the transient.

4. Turbine Stop Valve (TSV or MSIV - Main Steam Intercept Valve RA11 -14S01) tripped and isolated Turbine.

5. The operator starts to borate primary circuit 3 min after SCRAM with TK pumps.

6. The operator checks subcooling (if there are the conditions for Actuation of Automatic Step by Step Load (ASSL) according to the saturated temperature ΔT_s<10 ^oC.)

7. The operator checks primary side pressure: if P_I< P_{sat}+15[kgf/cm^2], then the operator stops MCPs.

8. The operator checks SG levels (undamaged SGs). The Pressure in secondary side is supported by BRU-K.

9. The operator stops MCP at the primary loop #1 with damaged SG.

10. The operator isolates the ruptured SG #1 from the feed-water - 300 s. after SCRAM.

11. The operator fills in the undamaged SGs (SGs #2, #3 and #4) up to level 3.5 m.

12. The operator reduces pressure in secondary side up to 52 kgf/cm^2 with 60^{o}C/h. by BRU-Ks.

13. The operator reduces primary side pressure up to 70 kgf/cm^2 by spray into the Pressurizer from cold leg (using Δp of MCPs). Operator must maintain subcooled safety margin above 10 ^{o}C between the core exit temperature and core exit saturated temperature t_s

14. The operator closes Fast Acting Isolating Valve (i.e. BZOK) of the failed SG at 70 kgf/cm^2 primary circuit pressure.

15. The operator checks boron concentration (Reaching the safety concentration of the boric acid in the primary circuit).

16. The operator continues the cool down of the primary system by steam dump to turbine condenser (BRU-K) with speed of 60^{o}C/hour and maintaining the safety margin to the boiling temperature of the hot loop more than 15^{o}C. Primary system is depressurized by injection of cold leg fluid into the Pressurizer Sprays. Pressurizer is filled up to 11.0 m.

2.3. Discussion of main phenomena and processes in case of total loss of all AC and DC power (Station Blackout - SBO).

Station Blackout can take place as a result of losses of external sources of electric power and switching-off of the NPP generator. The reason of given failure can be the infringements of stability of electric power network, short circuits in external electric network or short circuits in network of station. The total disappearance of electric power on NPP station causes switching-off of the own need main consumers.

Station Blackout is classified as a BDA accident and in the past when was designed VVER 1000/V320 was not considered in the reactor design. After involving a Symptom Based Emergency Operating Procedures (SB EOP) instead of the Event Oriented Procedures (EOP), the SBO is considered in the Safety Analyses Report (SAR).

The immediate consequences of the total loss of AC and DC power (SBO), if not accompanied by some other complicated event, such as a loss of reactor coolant, loss of secondary coolant or steam generator tube rupture, are not severe. However, should AC and DC power either from the grid or the emergency diesels not be restored quickly, the consequences to plant and public safety can potentially be extreme.

2.3.1. Event description

Should a SBO occur there are a couple of events that will follow this event: switching off all Main Coolant Pumps (MCPs); actuation of the Reactor Protection System (RPS) after 0.4+1.2s due to “Three of Four MCPs switched off” and after this signal all control rods drop in 2-4 s to the bottom of the core; the Pressurizer Heaters switch off; the Main Isolating Valve (MIV) closes in 1s due to electrical protection actuation (condenser vacuum loss); the Make-up system stops 2 s after the blackout, the draining line is closed; the Feed Water Pumps switch off after 5 s due to condenser vacuum. The Automatic Loading Sequence System (ALSS) starts 2 s after the blackout and 15 s after the blackout starts the available DG. The sequence of the ALSS events is described in the scenarios.

The secondary side pressure will increase and reach a set point of opening the BRU-As. The numbers of secondary side Relieve and Safety valves are 12 (and four turbine bypass valves – BRU-K). BRU-Ks are not available to work due to loss of condenser vacuum. In the past the BRU-A on SG #3 was not available too, because it was not connected to accumulator battery. The other 3 BRU-As were available during the first 30 minutes, after that they will failed in closed position, except BRU-A #4. When there was a loss of all AC power, BRU-A #4 will fail in closed position after 30 min., too, or it will continue to work when DG is available. Now, all BRU-As are equipped with powerful accumulators and can continue to work many hours. In case of BRU-As failure, all SGs are equipped with a two SVs (in total 8 SVs).

After dryout all SGs, the pressure in primary circuit will increase and reach a set point of opening the first Pressurizer Safety valve. The Safety valve will try to support the pressure in primary circuit opening at 185 kg/cm^2 and closing at 176 kg/cm^2. Cycles repeatedly, even under single phase liquid conditions will increase probability of failure of this valve. The likely timing of this event could be estimated in an engineering judgment. The most challenging condition could be the Pressurizer Safety valve to fail open when the Pressurizer is filled up. The Pressurizer Safety valve failure is considered as the induced failure.

The other important problem during the SBO is that the loss of high-pressure seal injection flow from the motor driven pumps will result in out leakage from the reactor coolant system along the RCP shafts. Without power this leakage cannot be replaced and a continuous loss of reactor coolant occurs in time. Loss of RCP seal cooling potentially can also cause degradation of the sealing capability of the RCP seals as a result of overheating, but conducted experiments at KNPP demonstrated that the seals will survive at least 24 hours. In this way the leakage out of the RCS could be consider approximately 2.5 m^3 /h per RCP. The problem with RCP seal leakage will be discussed additionally in item of model assumptions and appropriate equivalent diameter will be provided.

2.3.2. Assumptions

Steam Generator Water Level:

In nominal operation the Steam Generator water level is assumed to be 2.4 m. In Blackout analyses is accepted to have the minimum water level of 1.71 m in all SGs, as this level still allows to support a 100% of the reactor power.

This assumption will lead to earlier core damages and it will cause challenges to the heat sink and core cooling.

Reactor Power:

Reactor power in the initial state is assumed to be nominal with account for the measurement and control accuracy. Decay heat – ANS-79-1 +6%. Burnup status – corresponding to the end of life.

Feed Water Pumps:

The turbine driven feed water pumps are assumed to stop in 5 s.

Auxiliary Feed Water Pumps:

Before Blackout event the auxiliary feed water pumps will not be considered to work and their flow rates will not be taken into account.

Coolant temperature, primary side pressure, primary coolant flow, steam pressure, feed-water temperature, pressurizer level are assumed to be nominal.

2.3.3. Expected scenario in case of SBO

The initial and boundary conditions should be considered in correspondence with presented information above.

1. Switching off the all four MCPs due to SBO.
2. Actuation of the Reactor SCRAM after 0.4+1.2 s due to “Three of Four MCPs switched off” and after this signal all control rods drop in 2-4 s to the bottom of the core.
3. The Main Isolating Valve (MIV) closes in 1 s due to electrical protection actuation (condenser vacuum loss)
4. The BRU-Ks are not available due to loss of condenser vacuum
5. The Feed Water Pumps switch off after 5 s due to condenser vacuum.
6. The Make-up system stops 2 s after the blackout and the draining line (Let down system) is closed.
7. The Pressurizer Heaters switch off.
8. The Automatic Loading Sequence System starts 2 s after the blackout. (This should lead to: 15 s after the blackout starts the available DG (if this is conserved in the scenario); 20 s after the blackout start TQ12, 13, 14D01 pumps (low, high, high-high pressure pumps).45 s after the blackout starts TQ11D01 pump (sprinkler pump); 55s after the blackout starts TX10D01 pump (EFWP); TQ14D01 starts to inject after 3 min (the operator closes its recirculation valves); TQ13D01 head pressure – 110 kg/cm^2. TQ12D01 head pressure – 26 kg/cm^2).
9. The secondary pressure increases. BRU-A for SG #3 is not available. The three BRU-As for the SG #1, #2 and #4 open when P2=74 kg/cm^2, trying to support P2=68 kg/cm^2 and they close when P2=64 kg/cm^2. All three BRU-As are available up to 1800 sec. until AC power is available After this time only the BRU-#4 and SVs of the SG are available. More conservatively could be accepted failure of all BRU-A and activation of SVs.
10. After the blackout, a leakage from the RCP seals 2.5 m^3/h for each seal is initiated.
11. Due to the BRU-A work (or work of the SVs of the SG) the SG levels decrease. When the water levels of the SG #1; 2; 3; 4 reach L<130 cm starts an injection of EFW with supporting water level L=2.3 m. The limit of the maximum EFW flow, is 150 m^3/h. If the water level is L<110cm then the injection stops.

In case of later availability of DG (for example after core exit temperature is higher than 350.0 °C (plant state condition is in the “orange path” concerning the Critical Safety Function (CSF) “core cooling”, is considered procedure “feed and blead”. This could be done by depressurization of primary circuit and injection of borated water by HPP or LPP. In case of core exit temperature above 650.0 °C, the operators will leave the EOPs and entrance in the SAMGs.

3. Developing of a scenario with a combining the initiating events

3.1. Discussion of expected phenomena and processes.

The selected initiating event is a SB LOCA with ID 30 along with a total station blackout. Simultaneously is initiated a SG tubing break. The analyses of a total blackout became a very important assessments after the Fukushima accident. Combining SBO with two small break leakages: one from primary and other from primary to secondary side make these analyses quite complicated from one side and very important and interesting from the other side due to challenging all critical safety functions (CSF). The scenario has to combine two design basic accidents, such as small break loss of coolant accident and primary to secondary leakage with a beyond design basis accident (BDBA) such as total station blackout, where all the onsite and offsite alternating current (AC) electric power failed.

Should a SBO happened, all active safety systems will not be available. Removing of a residual heat from the reactor core will be organized by natural circulation due to switching off all MCPs. The inertia in the flywheel will allow MCPs to stop after approximately 240s. The homological curve is available in [2], but it could also be used and rundown characteristic of MCPs [2] to simulate transition to natural circulation.

The SBO will isolate Make-up/let down system, feed water system and will make not available PRZ heaters, as well as BRU-Ks.

The secondary side pressure will begin to rise and reaches the set point of BRU-As opening. They remain open and regulate the secondary pressure close to 68 kgf/cm^2. If the BRU-As will be assumed to fail due to they are electrically driven, then the SG SVs will start to support secondary pressure by cycling between their set points of opening and closing.

The ASSL (Automatic Loading Sequence System) will be activated in 2 s after the blackout. (This should lead to: 15s after the blackout starts the available DG (if this is conserved in the scenario); 20s after the blackout start TQ12, 13, 14D01 pumps (low, high, high-high pressure pumps).45s after the blackout starts TQ11D01 pump (sprinkler pump); 55 s after the blackout starts TX10D01 pump (EFWP); TQ14D01 starts to inject after 3 min (the operator closes its recirculation valves); TQ13D01 head pressure – 110 kg/cm^2. TQ12D01 head pressure – 26 kgf/cm^2).

The loss of coolant from primary side will reduce additionally capability for removing of residual heat from the reactor core. Loss of coolant will cause also reducing of PRZ water level, as well as reducing of primary pressure. Reducing of primary pressure as well as increasing of core exit temperature will generate signals duplicating the first one (switching off 3 out of 4 MCPs and power more than 75%) for reactor SCRAM.

Reducing the primary pressure will cause activation of HAs. When the primary circuit pressure drops to 5.88 MPa (60 kgf/cm^2) HAs will start injection of borated water with 16 g/kg in the primary circuit.

The leakage from primary to secondary side will not lead directly to activation of safety systems, except activation of signal for N16 in the secondary side. The operator will recognize leakage from primary to secondary side by reasons explained in the Chapter 2.2. Because of that after decreasing of primary side pressure below the pressure in the secondary side of faulted SG, the operator will isolate faulted SG by FW and steam. The operator will close BZOK at faulted SG. Some of the requested operator actions already are happened. The FW is already closed due to SBO.

Residual heat will be removed from the reactor core and primary side by natural circulation until SGs are effective and there is enough coolant in primary circuit.

The other interesting phenomena is cold loop clearance. After dryout of hot loop and after that cold loop between MCPs and RPV, the water in lower part of cold leg will seal flow path of the steam and will reduce the existing small cooling of the reactor core. This will accelerate heat-up of the reactor core which will cause cleanup of cold loop seal.

Selecting a break in cold leg with down orientation will allow blowout from the break until complete voiding of the break. After voiding the break, the flow rate from the break will reduce significantly. This will lead to reducing residual heat removing and possible primary pressure increasing or temporary stabilizing at existing level of primary side pressure, which will restrict HAs work.

After activation of HAs the primary circuit will start not only to cooldown the reactor core, but also to borate it. In this way the influence of reverse flow from SG to primary circuit on creating of positive reactivity for BOC will be reduced significantly. For investigation of necessity in using BOC characteristics in reactor core could be investigated initially by 1D assessment. In case of creating risk with significant insertion of positive reactivity and reaching “o” reactivity for reactor core will be run and 3D assessment for BOC.

After loss of natural circulation, it will start core uncovery, heat up, hydrogen generation. The heat up of reactor internals will delay primary pressure decreasing and it will restrict effective injection of borated coolant in the reactor vessel. This will intensify furthermore reactor core heat up.

3.2. Initial and boundary conditions.

The combination of selected initiating events should be investigated using nominal parameters for a coolant temperature, primary and secondary side pressure, primary coolant flow, steam pressure, feed-water temperature and pressurizer level. As the main initiating event is a loss of coolant accident and SG tubing break and SBO are involved events, the SG water level will be assumed to be nominal.

In a LOCA accident the main concern is a residual heat removing. The most challenging condition in this situation will be the burnup status at the end of fuel campaign (EOC). The EOC is also the most challenging for the SBO, while for the SG tube rupture, the begging of campaign is the most challenging condition if the accident progression will cause significant revers of flow of clean SG water to the borated primary side water and will lead to boron dilution in the primary circuit. The boron dilution will be bigger if the primary pressure will be longer time below the secondary side pressure. The injection of borated water from the HAs, which will be activated after reaching 5.88 MPa, will minimize the effect of the revers flow from secondary to primary leakage.

After discussion with experts, it could be recommended investigation of SG tube rupture at the end of fuel cycle due to the same concern, as for the other two initiating events – cooldown of the reactor coolant system and faulted SG. It is possible to have significant differences in the temperature regimes in intact and faulted SG, which should be avoided.

Based on all above, the end of fuel cycle has been selected. The integral flow rate from the secondary side of SG to primary side will show the magnitude of possible challenges for boron dilution. If the risk is significant, it will be recommended recalculation for the begging of cycle.

Reactor Power:

Reactor power in the initial state is assumed to be nominal with account for the measurement and control accuracy. Decay heat – ANS-79-1 +6%. Burnup status – corresponding to the end of cycle.

Steam Generator Water Level: The Steam Generator water level is assumed to be 2.4 m due to combining of more than 2 initiating events (IE).

Feed Water and Auxiliary Feed Water Pumps: The feed water pumps are assumed to stop in 5 sec.

Coolant temperature, primary side pressure, primary coolant flow, steam pressure, feed-water temperature, pressurizer level are assumed to be nominal.

Steam dump to atmosphere: all 4 BRU-As

Passive safety system: all 4 HAs and all 8 SG SVs

Active safety systems: One HPP (TQ13) is available, but will activate after DG is available. Water temperature in tanks 40 ^{0}C

3.3. Modelling assumption

MCP seals will not be modelled to simplify, as this is important mostly for SBO scenario without loss of coolant.

Modelling of separately heat structure of faulted tube line for more correct assessment blow down through the tubing break.

3.4. Developing of the scenario.

1. SB LOCA (ID 30) along with a total station blackout occur at 0.0s. Simultaneously is initiated a SG tubing break.

1. Switching off all four MCPs due to SBO.

2. Actuation of the Reactor SCRAM after 0,4+1,2 s due to “Three of Four MCPs switched off” and after this signal all control rods drop in 2-4 s to the bottom of the core.

3. The Main Isolating Valve (MIV) closes in 1 s due to electrical protection actuation (condenser vacuum loss)

4. The BRU-Ks are not available due to loss of condenser vacuum.

5. The Make-up system stops 2 s after the blackout and the draining line (Let down system) is closed.

6. The Feed Water Pumps switch off after 5 s due to condenser vacuum.

7. The Pressurizer Heaters switch off.

8. The Automatic Loading Sequence System starts 2 s after the blackout

9. Establishing of natural circulation and cooldown of reactor coolant system at natural circulation.

10. Opening of BRU-As at 74 kgf/cm^2 and beginning to support secondary pressure at 68 kgf/cm^2. The SG SVs will open in case of BRU-A failure.

11. Isolating damaged SG by closing BZOK after primary pressure is below the pressure in the faulted SG. FW is isolated already due to SBO.

12. Total dryout of PRZ.

13. Activation of HAs at 5.88 MPa and injection of borated water into RPV.

14. Loss of natural circulation.

15. Termination calculation after reaching of core exit temperature above 1200.0 °C.

Possible operator action: activation of DG and one HPP (TQx3) after 350.0 °C for injection of borated water. The characteristics of HPP and tanks are presented in [2].

4. List of important parameters for comparison code by code

Here is presented the list with important parameters for comparison code to code

1. Primary pressure (at core exit and in top of Pressurizer), [MPa]
2. Core exit coolant temperature, [K].
3. Reactor core water level, [m].
4. Pressurizer water level, [m].
5. SG water level of faulted SG, [m]
6. SG water levels of intact SGs, [m]
7. Integral break flow rate from primary circuit, [kg]
8. Integral break flow rate from primary to secondary side, [kg]
9. Reactor power, [MW]
10. Heat transfer of faulted SG, [MW]
11. Core exit cladding temperature, [K]
12. Core exit fuel temperature, [K]
13. Maximum core fuel temperature, [K]
14. Coolant flow rates in hot and cold legs, [kg/s]
15. Hydrogen generation, [kg]
16. Reactivity (in case of transient in the beginning of life), [$]

Additionally, will be compared important points during accident progression as:

1. Time for total dryout of Pressurizer, [s].
2. Time for loss of natural circulation, [s]
3. Time for initiating HAs injection, [s]
4. Time for hot leg dryout, [s].
5. Time for beginning of core uncovery, [s].
6. Beginning of hydrogen generation, [s]
7. Total dryout of reactor core, [s]
8. Total dryout of reactor vessel, [s]

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