Full Scope Simulation of VVER-1000 Blowdown source and Containment Pressurization in a LBLOCA by parallel coupling of TRACE and CONTAIN

Omid Safarzadeh[1], Omid Noori-kalkhoran[2,*], Massimiliano Gei[3]

Lorenzo Morini[2], Rohollah Ahangari[4],

[1]*Faculty of Engineering, Shahed University, Tehran, Iran*

[2]*School of Engineering, Cardiff University, CF24 3AA, Wales, UK*

[3]*Deptartment of Engineering and Architecture, University of Trieste, Trieste, Italy*

[4]*Reactor and Safety Research School, Nuclear Science and Technology Research Institute, Tehran, Iran*

Abstract

Nuclear power plants containment play an important role as last-defined barrier in defense in depth approach against the release of radioactive material to the environment. In this study, a parallel processing couple has been developed to full scope analysis of blowdown source and containment pressurization parameters in a LBLOCA accident. To achieve this goal, primary and secondary loops of a VVER-1000/V446 were first simulated in TRACE V5.0 and steady-state results have been validated against reference data. The second step deals with containment simulation in CONTAIN 2.0 with new modified 30-cells models. A parallel processing interface was developed in MATLAB to couple TRACE and CONTAIN in the break point. Containment average pressure has been fed back to TRACE as forcing function of blowdown source in each time step during pressurization phase (coupling point). Finally, results of blowdown and containment pressurization have been validated against final safety analysis report (FSAR). Results of simulation confirm that the maximum containment pressure can reach 0.36 MPa and 0.395 MPa for this study and FSAR respectively that are lower than the maximum design absolute pressure of 0.46 MPa, so containment maintains its integrity during this accident. Temperature profiles of different control volumes inside containment during accident following the FSAR profiles in terms of shape and value that show the ability of developed parallel coupling to full scope simulation of accidents accurately.

Keywords: LBLOCA, Blowdown source, Containment Pressurization, Parallel coupling, TRACE, CONTAIN

[*] Corresponding Author's email address: NoorikalkhoranO@cardiff.ac.uk

1. Introduction

The reactor containment is intended to keep the public safe from release of radioactive materials during reactor accidents. Maintaining the physical integrity of the containment as a geometrically complex structure with countless joints and penetrations through roots and walls is contributed to the size of the environmental release. The steam explosion in the vessel or reactor cavity by interaction between the corium and the coolant, hydrogen explosion and slow over-pressurization pose a profound challenge of preserving the structural integrity of the containment. These shocks will in turn provide a bigger load that could tear the containment liner and increase the rate of leakage through a penetration.

The temperature and pressure distribution in the containment should be investigated to ensure the operation of the safety systems such as containment spray system and fan coolers to meet the safety requirement and to keep the integrity of containment in a blowdown accident. During an accident in a water reactor, the "blowdown" phase refers to the initial discharge, with a high mass flow rate of high-temperature pressurized coolant from the reactor cooling system into the containment. The intensity of the release is due to the high pressure difference between the cooling system and the containment atmosphere (Noori-Kalkhoran et al., 2016). Two main role-playing factors in such accidents include Blowdown source and as its result, Thermal-Hydraulic distribution inside containment. Both these factors need to be investigated to have a complete perspective of important parameters in containment pressurization accidents. In most of the published studies, only one of these factors has been investigated that can damage the mutual functionality of parameters.

LBLOCA is a type of reactor accident in which coolant is lost from the main legs of primary circuit pipes with a break diameter greater than 0.1 m. This accident is the most dramatic scenario that might happen in that the coolant would be lost massively and rapidly from the reactor vessel and cold-leg pipe due to pressure in the primary circuit in a matter of seconds (Joyce, 2018). In the absence of coolant in the core region and vaporized coolant deposition in the containment, overheating and pressurization would occur for cladding surface and containment building, respectively. Most remedial actions designed to mitigate the effects of an LBLOCA involve rapid-response and sustained supplies of emergency core cooling system and use of spray system to reduce pressure, temperature and radioactive material inside the containment.

TRACE is developed by USNRC to analyze the thermal-hydraulic phenomena in LWRs (NRC, 2007). The field equations implemented in the code are based on the two-fluid model formulation for liquid and vapor phases plus additional equations for boron dissolve and non-condensable transport. The TRACE code has several components for modelling one-dimensional and three-dimensional fluid. This code is specially assessed for the investigation of LBLOCA. Okawa and Furuya, (2019) used TRACE in LBLOCA of LOFT test facility. They were interested in the sensitivity of minimum film boiling temperature for heat transfer model and its influence on the cladding temperature behavior. Radaideh et al. (2019) analyzed a method for reducing fuel temperature during LBLOCA in a BWR. The method was based on a device to limit the reverse flow rate at the inlet of fuel bundles in the core. Chen et al. (2013) investigate the sensitivity of the counter-current flow limitation during LBLOCA in a BWR. TRACE has been used for LBLOCA in AP1000 to validate the Westinghouse results and to investigate the effectiveness of AP1000 design in mitigating LBLOCA accidents (Queral and Jimenez, 2015). The study concluded that TRACE showed lower peak clad temperatures than those provided by Westinghouse. The reason for these differences was due to the conservative assumptions used by Westinghouse. TRACE has been used by (Chen et al., 2013a) to study alternate mitigation strategies for a BWR LOCA with station blackout (loss of onsite and offsite power) similar to the accident that occurred at Fukushima.

Containment pressurization studies can be categorized most in three groups; I) Modelling by developing various thermal-hydraulic (TH) models such as single-cell and multi-cell models in an appropriate programming language (e.g. FORTRAN or MATLAB) II) Simulation by using the available Nuclear codes such as CONTAIN, MELCOR and GOTHIC and III) Application of General CFD fluid codes such as ANSYS-CFX, FLUENT and GASFlOW.

Fernández-cosials et al. (2017) provided the overall peak temperature and pressure of the containment of an AP1000 reactor with a detailed three-dimensional representation of the geometry of the whole building by GOTHIC. Containment pressure distribution has been studied in a VVER-1000 reactor using different methods by (Noori-Kalkhoran et al., 2014b, 2014a; Noori-Kalkhoran et al., 2016). They have applied single- and multi-cell models and CONTAIN 2.0 for simulation of TH parameters inside VVER-1000 containment. Meanwhile, the effects of engineering safety features (ESFs) such as spray system were considered in their evaluation. Kim et al. (2018) have used MELCOR to simulate containment pressurization in a CANDU PHWR containment. They have validated their results against integrated leakage rate test that is a domestic regulatory requirement by nuclear safety and security commission of South Korea.

In addition to nuclear codes, different CFD software such as ANSYS-CFX, FLUENT and GASFLOW have been considered for this simulation. Kaltenbach and Laurien, 2018 developed a model for containment spray and evaluated its effect on temperature and pressure mitigation inside containment by using ANSYS-CFX 16.1(Kaltenbach and Laurien, 2018). They have conducted experimental validation by using THAI multicompartment containment facility. A numerical study has been investigated by (Li et al., 2019) to evaluate the effects of passive containment cooling system on TH behavior of containment. GASFLOW-MPI has been used to achieve this goal and results have been validated against experimental ones.

Among the published articles, few numbers of them have employed coupling codes for reactor coolant system and containment pressurization analysis. Bae et al. have coupled MARS and CONTAIN to simulate TH parameters for the reactor coolant system and containment phenomena respectively(Bae et al., 2021). They have validated their result against integral effect test facility , ATLAS-CUBE. Investigation of TH parameters inside WWER-1000 containment has been conducted by RELAP5/SCDAP and CONTAIN code by Salehi et al. (Salehi and Jahanfarnia, 2020). Although a batch process was applied in this study; firstly break source was simulated using RELAP5/SCDAP and then this break source was fed as CONTAIN input.

In this paper, a comprehensive simulation has been conducted to evaluate the blowdown source coupling with containment pressurization and their effective parameters. TRACE was used to simulate the primary and secondary loops of VVER-1000 reactor and blowdown source. TH parameters and ESFs (such as spray) inside containment were simulated using CONTAIN and employing a new modified 30-cells model. TRACE and CONTAIN were parallelly coupled in break point by developing an interface in MATLAB. This A-to-Z simulation of both blowdown source and containment leads to avoid misleading of mutual factors of blowdown source and containment pressurization process on the results.

2. Simulation

2.1 Simulation of Blowdown source using TRACE

The Bushehr Nuclear Power Plant first unit (BNPP-1) is a pressurized water reactor (PWR) based on the design features of the VVER-1000/446 model, an evolutionary PWR developed by the Russian industry. The BNPP-1 facility could generate 3000 MW thermal power. The fuel assemblies are positioned in a hexagonal core with ability to load 163 fuel assemblies that each of them has consisted of 311 fuel rods and 18 guide/control rods (AEOI, 2007). Figure 1 shows the primary loop of BNPP-1 and some of its parameters.

Characteristic	Value
Reactor nominal thermal power, MW	3000
Coolant flowrate, m^3/h,	84800
Coolant pressure at the core outlet, MPa	15.7
Coolant temperature-reactor outlet,°c	321
Coolant heating in the reactor, °c	30
Number of loops, pcs	4
Number of FA in the core, pcs	163
Fuel height in the core in cold state, m	3.53
Equivalent diameter of the core, m	3.16
Flow area of the core, m^2	4.14

MCP: Main Coolant Pump
SG: Steam Generator
PR: Pressurizer
RPV: Reactor Presuure Vessel

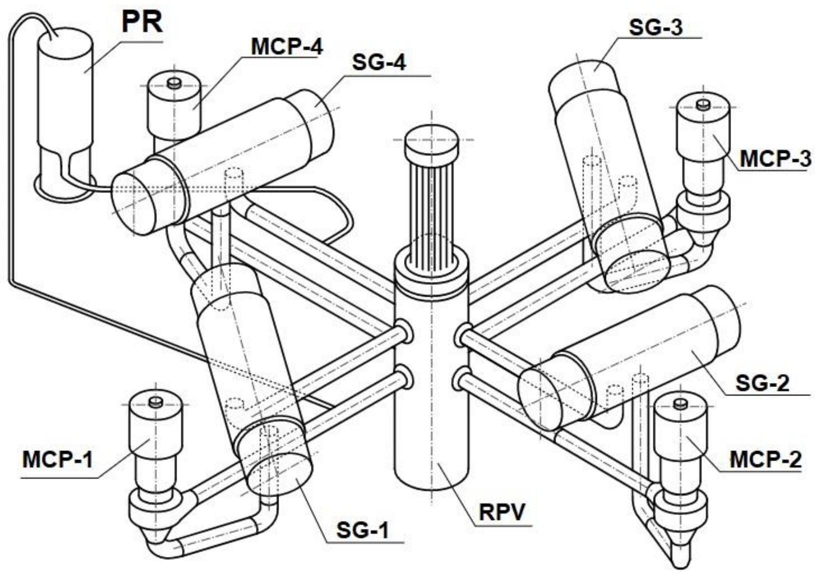

Figure 1. Schematic of BNPP1- primary loop (AEOI, 2007).

The nodalization model of TRACE is performed based on the real geometry and loops configuration of the BNPP-1 as shown in Figure 2. There are four recirculation loops each of which includes a recirculation pump with pipes for pump suction/discharge, a steam generator (SG) and two emergency injection pumps. To show more detail, only 2 of 4 recirculation loops have been shown in Figure 2, the other two loops have the same nodalization symmetrically.

The reactor pressure vessel is modelled by 3D cylindrical VESSEL component. The core located inside the baffle is divided into one radial ring and six azimuthal sections. The active core is also is subdivided into 20 axial meshes. Two meshes are used for the bottom and top reflectors. One radial ring is also used to represent cells in the downcomer region. Another ring is used to model core barrel and baffle regions. The lower plenum and upper plenum are represented by axial meshes, there are 22 axial levels. The outermost radial ring is the downcomer, the inner radial rings stand for the core regions, and the middle radial ring is for barrel and baffle. In each of these cells, flow area, hydraulic diameter, and form loss coefficients are defined in each r, θ, and z-direction allowing for 3D flow path representation. Each hydraulic cell related to the active core region is connected to a heat structure. To represent the average and hot rods heat source, 12 heat structures are used and modeled by HTSTR component. Each heat structure is divided into 20 axial meshes where these meshes are one-to-one mapped to the hydraulic system. Power distribution in the axial direction has a cosine profile. A hot rod peaking factor is 1.65 to one hot fuel assembly.

A pressurizer is also connected to the hot-leg of a recirculation loop. The pressurizer model has a pipe of a pressurizer-type component, eight surge lines connected to the RPV model, power-operated relief valve (PRV) and safety relief valve (SRV). These valves are connected to a break component with a pressure of 0.1 MPa representing the containment. The pressurizer pipe is divided into ten axial cells which are partly filled with steam. The pressure during Each SG model is connected to an azimuthal sector, which are situated above the reactor core.

The recirculation pump is modeled by PUMP component. The pump characteristics such as head, flow rate, velocity and single- and two-phase head and torque are inputted. The SG is modeled by PIPE and HTSTR components.

The SG tube bundle is represented by eight pipes modeled by PIPE component. The HTSTR components are used to thermally connect the primary-coolant that flows inside the tubes with the secondary-coolant that flows outside of these tubes. Each heat structure is radially subdivided into eight cells to properly calculate the heat transfer through conduction between the tube outer and inner surfaces. The secondary side of the SG is modeled by pipes. These pipes are used to show downcomer, hot and cold sides, riser, separator and finally the steam dome. The feedwater systems are modelled using FILL components with nominal flow rates. The main steam line is modeled by a PIPE component. The VALVE components are used for safety relief valve (SRV) and main steam isolation valve (MSIV) with the BREAK component for the outlet boundary. A steam line connected to a break component that represents turbine boundary condition.

ECCS consists of an accumulator, a HPIS and a LPIS. A FILL component is used to set inflow boundary condition of HPCI. The accumulators are connected to the reactor pressure vessel. The accumulator is modeled by PIPE components.

2.2 Simulation of Containment pressurization using CONTAIN

The CONTAIN 2.0 is an integrated analysis tool used for predicting the physical conditions, chemical compositions, and distributions of radiological materials inside a containment building following the release of material from the primary system in a light-water reactor accident (Murata et al., 1997). CONTAIN is categorised as best-estimate codes in the field of containments phenomena simulation.

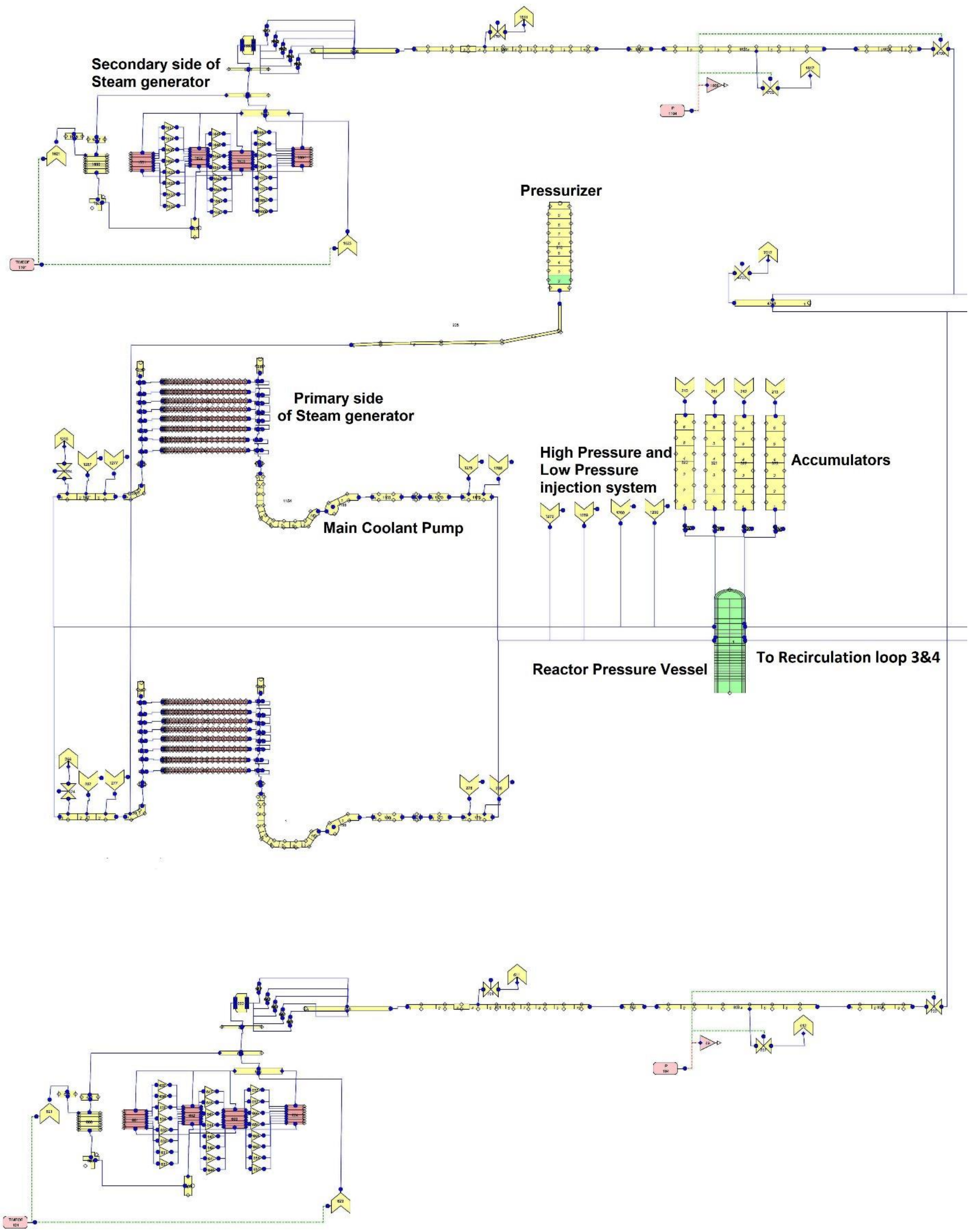

Figure 2. Nodalization of BNPP-1 primary and secondary loops applied in TRACE.

Based on safety requirements, BNPP-1 containment design has been adopted as a double anti-accident containment, an outer cast-in-situ reinforced concrete and an inner steel spherical one. Maximum absolute design pressure at design-basis accident is 0.46 MPa. Absolute pressure for the containment strength test is assumed in compliance with Russian Nuclear Standard No:PNAE G-10-012-89 equal to 0.51 MPa, which admissibility was confirmed by analysis (AEOI, 2007). BNPP-1 containment safety system includes containment spray system, containment vessel isolation system, hydrogen concentration monitoring, hydrogen recombiners, nuclear component cooling system and secured close cooling water system. Depending on the accident intensity and safety procedure, one or some of these safety systems will be employed to tackle the accident progression and its consequences.

A modified 30-cells model (in comparison to previous 23-cells studies; Noori-Kalkhoran et al., 2016, 2014b, 2014a) has been applied in this research to simulate the BNPP-1 containment pressurization as result of LBLOCA. This modified version can cover the spot points of TH parameters and hydrogen distribution (not included in this research) in a more detailed and scientific manner and can help siting of ESFs in more efficient coordinates. A full detailed 3D containment structure has been developed in AutoCAD to extract all the required parameters for different input component of CONTAIN such as control volumes, engineering vents, thermal structures, and spray as ESFs. Figure 3 shows the 3D plan of BNPP-1 containment and its general parameters were listed in Table 1.

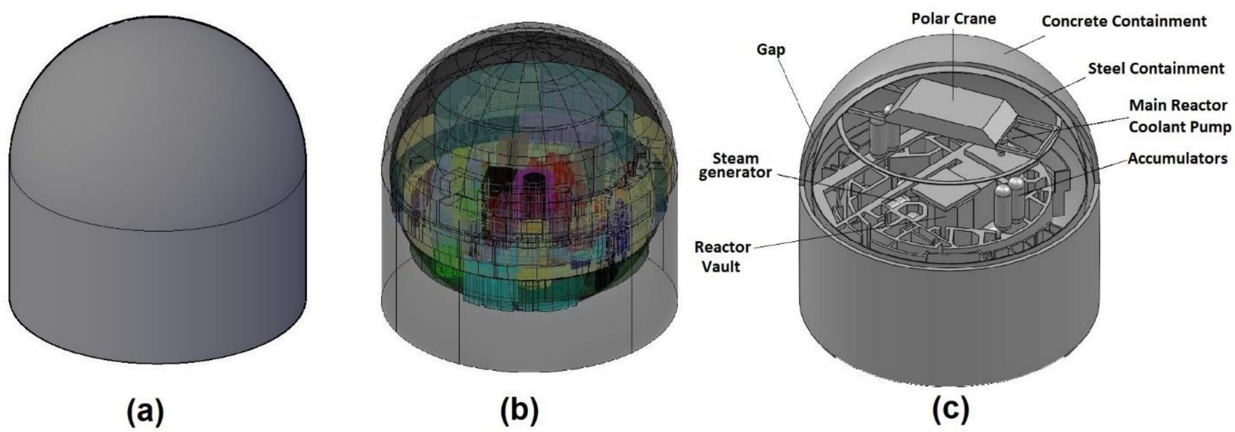

Figure 3. Containment 3D plan, (a) cylindrical concrete containment   (b) layout of concrete and steel containment (c) layout of compartments inside steel containment.

Table 1. Structural and design parameters of BNPP-1 containment and its spray system.

Structural Parameters	**value**
Steel containment inner diameter (mm)	28,000
Steel thickness (mm)	30
Gap thickness(mm)	1650
Concrete thickness(mm)	1750
Containment free volume ($m^{3)}$	71,040
The total area of all the concrete walls (m^2)	18,860
Design Parameters	
Maximum internal pressure at 150 ^{0}C (MPa)	0.46

Maximum pneumatic test pressure at a temperature of up to 60 ^{0}C	0.51
Maximum (averaged over the volume) temperature (^{0}C)	150
Spray System Design Parameters	
Materials sprayed by a nozzles	Boric Acid solution 16
Temperature of sprayed materials, °C	Not more than 90
Design temperature, °C	150
Pressure drop in nozzle, MPa	0.1
Flow rate of sprayed materials, m^3/h	31
Angle of tapered solid cone of spraying, degree	75
Spraying dispersibility, mm	1.2
Conditional flow capacity of supplying pipe branch, mm	50
Conditional flow capacity of outlet pipe branch, mm	30

Control volumes are connected by means of 48 engineering flow paths. In the analysis procedure, all actual civil and process structures of the containment are modelled by 120 thermal structures (walls, roof etc.) that differ from one another in the combination and thickness of the constituent materials. For correctly calculating the heat accumulating properties of the walls and heat transfer processes, the walls are subdivided into several sublayers across their thickness. Figure 4 shows the layout of considered control volumes. It should be noted that control volumes number 5 (Rigs room), 7-10 (Rooms of reactor coolant pumps motors), 14 (Cask pool), 17-21 (Filtration rooms), 26 (Room of High pressure coolers) and 27 (Recovery HX rooms) can't be shown in this Figure as they have been located in the space between the front and rear vertical cross-sections. It was assumed in the calculations that the break is happening in room No. 2 to start the blowdown. Table 2 tabulates compartment volumes and descriptions.

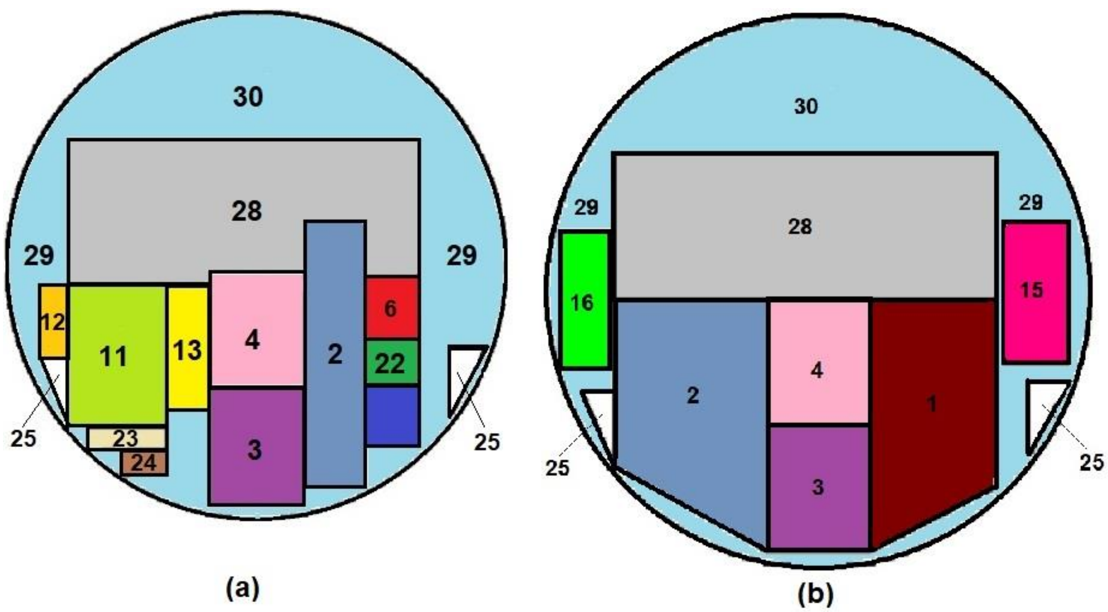

Figure 4. Layout of Control volumes in containment a) Rear vertical cross-section b) Front Vertical cross-section.

Table 2. Compartment volumes and descriptions

No	Composition of the design rooms	Volume (m^3)
1	Sump, Stairs, Rooms of loops 1&2	4966
2	Sump, Stair, Room of loops 3&4	5174
3&4	Reactor Vessel shaft from -2.15 to 21.5 m elevation	457 & 1100
5	Rigs of control protection system, annular corridor from 0 to 180 degree	699
6	Annular corridor from 180 to 360 degree, measurement chamber	786
7-10	Room for Reactor Coolant Pump (RCPs) Motors	270 (each)
11	Fuel Pool	1380
12	Fresh Fuel Storage	677
13	Reactor internals inspection pool	541
14	Cask pool	130
15&16	Ventilation duct and shaft, Room for ventilation systems	761 & 1073
17-21	Room for filters	51 (each)
22	Stair, Pipeline shaft, valve chamber	277
23	I&C room, Spare room, Valve chamber	905
24	Stairs, I&C rooms and sumps of nuclear drain pumps	803
25	Annular pipeline corridors from 0 to 360 degree	784
26	Room for HP coolers	135
27	Room for recovery heat exchanger	35
28	Volume in apparatus hall inside the cylindrical wall	16950
29	Hall volume between the cylindrical wall and containment	6994
30	Hall volume above the cylindrical wall	19340

2.3. Containment Spray system

Containment spray system is designed for operation under emergency conditions arising from leakage of the primary coolant system and leakage of the secondary side inside the containment. Under normal operating conditions the system does not operate and is on standby mode.

During emergency conditions involving a LBLOCA and secondary side ruptures inside the steel containment, the system performs the preset function of pressure and temperature reduction as well as radioactive iodine isotope inside the steel containment. This happens by injection of boric solution into the air space of the steel containment, with concentration being 16 g H_3BO_3 per 1 kg H_2O and iodine-binding reagents. The solution temperature and flow rate are 20-60 °C and 300 t/h per one channel of the system. The pressure setpoint for spray actuation is 1.3 bar or (0.13 MPa).

3. Coupling procedure

Two different approaches are generally utilized to couple two or more nuclear codes; serial integration and couples parallel processing. The serial integration approach includes modifications of the codes, usually by implementing a new module (i.e. subroutines, functions) into the main code. In the parallel processing approach, the coupled nuclear codes are executed separately and exchange the needed data during the calculation (Noori-Kalkhoran et al., 2014). In this study parallel processing couple has been developed to exchange the containment average pressure data between blowdown source (simulated in TRACE) and containment (simulated in CONTAIN).

The BREAK component of TRACE was used as a junction module between TRACE and CONTAIN to exchange the containment average pressure as a forcing function. This component imposes a pressure boundary condition on the primary loop. It can be used anywhere fluid is able to enter or leave the system being simulated, and the pressure distribution as a function of time is known. An interface programmed in MATLAB manages the exchanging the pressure boundary condition in break point and running both codes in appropriate time steps. Figure 5 shows the coupling procedure.

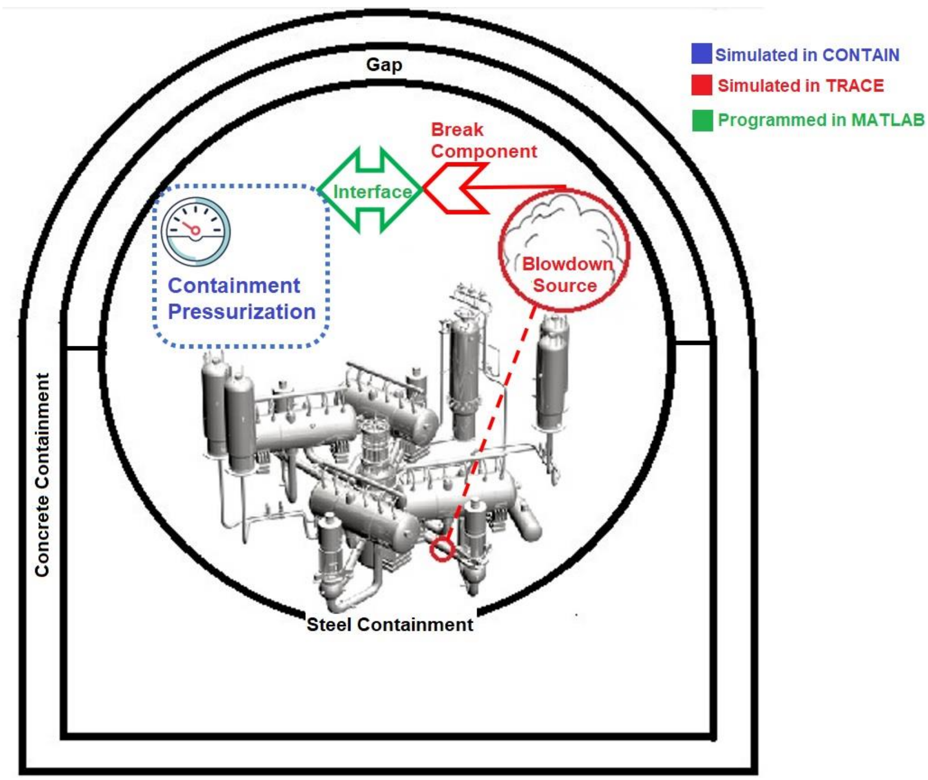

Figure 5. Coupling procedure schematic.

4. Results and discussions

4.1. BNPP-1 steady-state modeling

The model of the facility in the TRACE code was created and then validated using FSAR of the plant. The pressure drop, fluid temperature, and mass flow rate in the normal operation are compared to make sure the geometry and loss coefficients for each component are properly represented. The heat generated in the active part of the core is removed in the SG to achieve the desired steam quality and flow rate. The results can be seen in Table 3. The pressure as well as the temperature is well represented by the model. The comparison demonstrates that the modeling results are within acceptable limits and reliable to be used in transient status and calculation of blowdown source.

Table 3. validation of the main reactor parameters at steady state

Parameter	**FSAR** (AEOI, 2007)	**TRACE**	**Relative Error (%)**
RPV inlet temperature (K)	564.15	561.8	0.4
RPV outlet temperature (K)	594.15	592.6	0.3
Maximum fuel temperature (K)	2156.15	2136.5	0.9
Maximum outer clad surface temperature (K)	625.15	622	0.5
Reactor outlet pressure (MPa)	15.7	15.8	0.6
SG steam pressure (MPa)	6.27	6.25	0.3
Coolant flowrate at core inlet (m^3/hr)	84800	84010	0.9
SG physical water level (m)	2.4	2.4	0.0
PRZ water level (m)	8.17	8.177	0.08
Steam mass flowrate (kg/s)	408.3	408.1	0.05

4.2 Event Sequence

The LOCA is a hypothetical accident that consists of a loss of reactor coolant through a break that occurred in primarily coolant pipe lines including a doubly ended guiltily break on the main coolant pipe. The broken pipe is postulated to occur inside the primary containment (control volume No 2). The pipe break can cause the primary loop to lose pressure and discharge the flow from the main circulation loop to the containment building. This phase of discharge and loss of reactor water level are known as blowdown in which the core heat transfer is degraded. The reactor pressure and water level decrease rapidly during the blowdown phase. This corresponds to the reduction of a large fraction of the coolant inventory in the reactor core, involving subsequent increase in the fuel cladding temperature in short time. The reactor is immediately enforced to decrease the power level by positioning the shutdown rods in the core. The scram signal is generated by reaching primary coolant pressure to 14.7 MPa. The turbine stop valve is being closed as the power production ceased. The control and measurement signals are generated and processed with a delay time. The delay time for scram, turbine valve and shutdown rods movement signals are 0.027 s, 0.6 s, and 1.327 s, respectively. The accumulators start to supply borated water in reactor vessel if the reactor pressure is reached to 5.88 MPa. The cladding temperature is decreased as the cladding surface becomes wetted.

During the depressurization phase, the HPIS and LPIS injecting water also provide more heat removal by diesel generator startup with a delay time of 40 s. The blowdown phase is ended by reaching LPIS flow rate at nominal value. There is a net increase of reactor coolant inventory by ECCS action. This stage of LOCA is known as the refill phase. The final phase of LOCA is known as the reflood phase when the core is being reflooded with water and the mixture level reenters the core region. The cladding temperatures were observed to turn around very shortly after the onset of reflood. The sequence of actuation of systems and devices as well as the actuation set points are given in Table 4.

Table 4. The sequence of events for LBLOCA (AEOI, 2007).

Time, s	Event
0.000	Large break at a cold-leg
0.027	Scram signal generation by reaching reactor coolant pressure to 14.7 MPa
0.036	Startup signal generation for safety systems
0.600	Closing the turbine stop valve
1.327	Movement of shutdown rods
7.0	Opening of pressure safety valves in steam lines
8.8	Injection of borated water by accumulators by reaching reactor coolant pressure to 5.88 MPa
40.0	Start of HPIS and LPIS
55.0	Stop of accumulator injection by reaching the accumulator water level to 1.2 m

4.3 TRACE results

Figure 6(a) shows the rapid decrease in reactor power following break occurrence calculated by TRACE and is compared with the FSAR data (AEOI, 2007) and RELAP5 (Shoushtari, 2010). The core power is obtained by TRACE with use of the point kinetics model, and the power calculated includes decay heat. The reactor power trend is almost the same as those of TRACE, FSAR and REALP5 data. Figure 6(b) compares the pressures in the vessel outlet. As it can be seen, the pressure calculated by TRACE approximately follows the trend of the FSAR data and RELAP5.

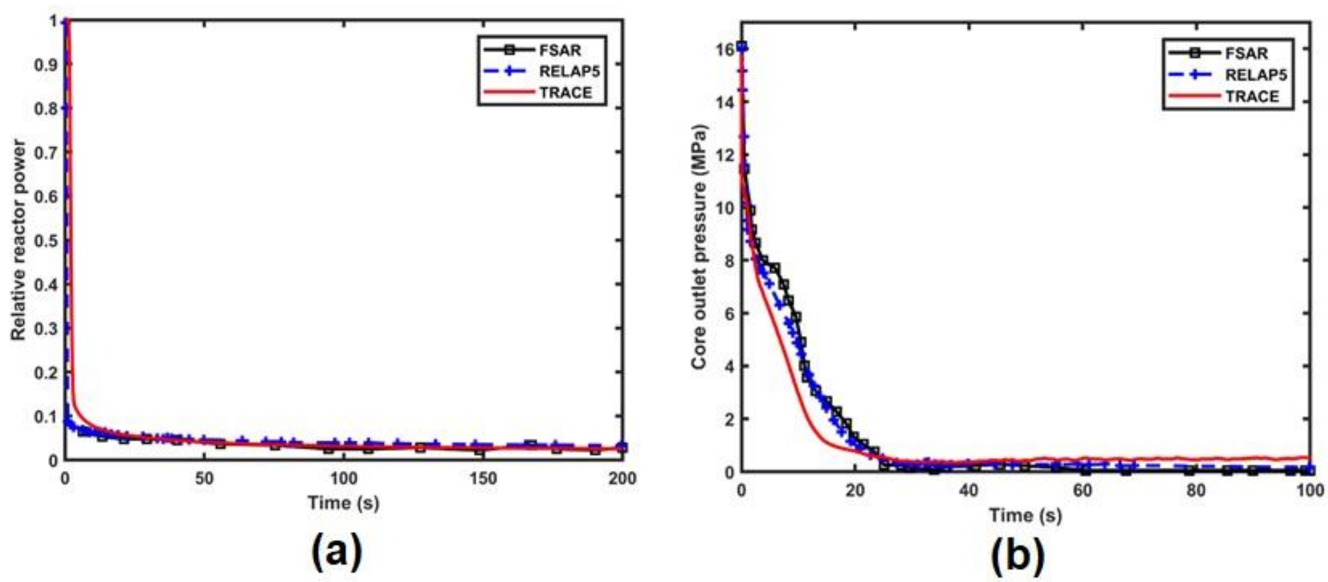

Figure 6. (a) Relative power of reactor, (b) pressure of coolant at core outlet.

Figure 7(a) compares the break mass flow rate with FSAR and RELAP5. This figure reveals that break mass flow rate predicted by RELAP5 is a bit higher than the two others for the period of 20-

40s. The TRACE trend and values agree relatively with the results of the FSAR data. The mass flowrate of ECCS have been compared between TRACE model, FSAR data, and RELAP5 in Figure 7(b).

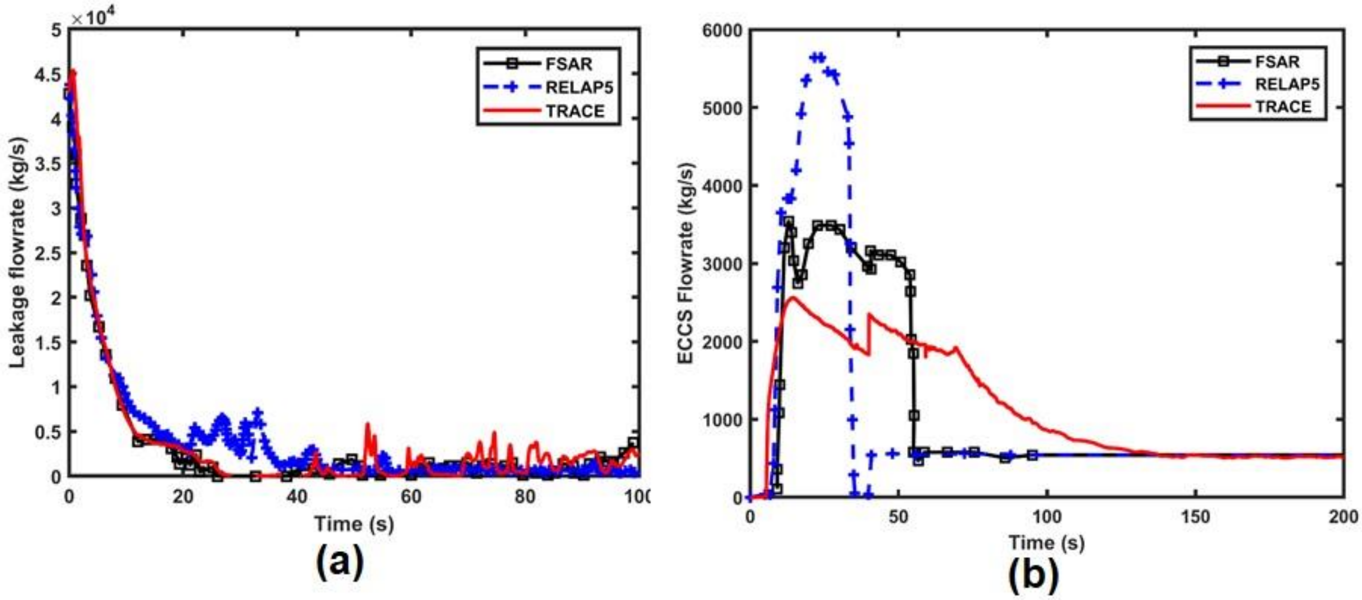

Figure 7. Mass flowrate of: (a) Leakage from break point, (b) ECCS.

Figure 8(a) compares the water levels in pressurizer, demonstrates that the level calculated by TRACE is in agreement with the FSAR data except for the discharge time that occurred 8s onwards. Secondary side pressure of steam generator has been depicted in Figure 8(b). As it can be seen unlike FSAR profile, the pressure is not increased to opening set point of safety valve for both RELAP5 and TRACE. Figure 9(a) presents the results of maximum cladding temperature after the break. The fast-decreasing core flow results in a rapid increase of cladding temperature. The TARCE data follow approximately the same trend with FSAR data at the blowdown and the quenching time-occurred later in reflooding phase, while RELAP5 results for the cladding temperature trend shows a remarkable drop after the peak at the blowdown phase. The quenching time in the reflooding phase has large delay for RELAP5 in comparison to the FSAR and TRACE data. In addition, the REALP5 results do not show any peak of the cladding temperature or quench at the blowdown phase, and the temperature trend remains at a higher level on the whole process until the quench occurs in the reflooding phase. The maximum fuel temperature immediately decreases at the beginning of break because of the fast decreasing in reactor power as shown in Figure 9(b). The fuel temperature will keep a slight increasing before the core reflooding starts. As results of RELAP5 were extracted from (Shoushtari, 2010), the main reasons for discrepancies between RELAP5 and FSAR results can be investigated with step by step comparison of code inputs, nodalizations, applied correlations in codes for heat transfer in fuel and clad, codes structures, etc that are out of the main scope of this article and can be followed by readers.

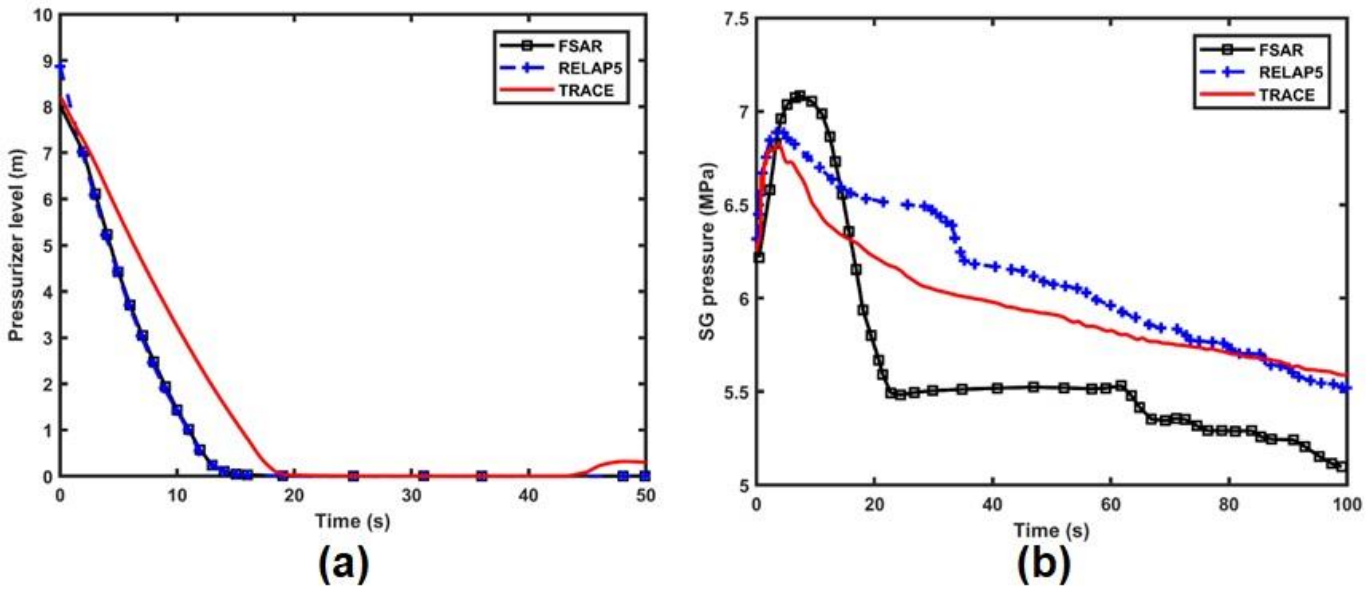

Figure 8. (a) Pressurizer water level, (b) Pressure of secondary side of steam generator

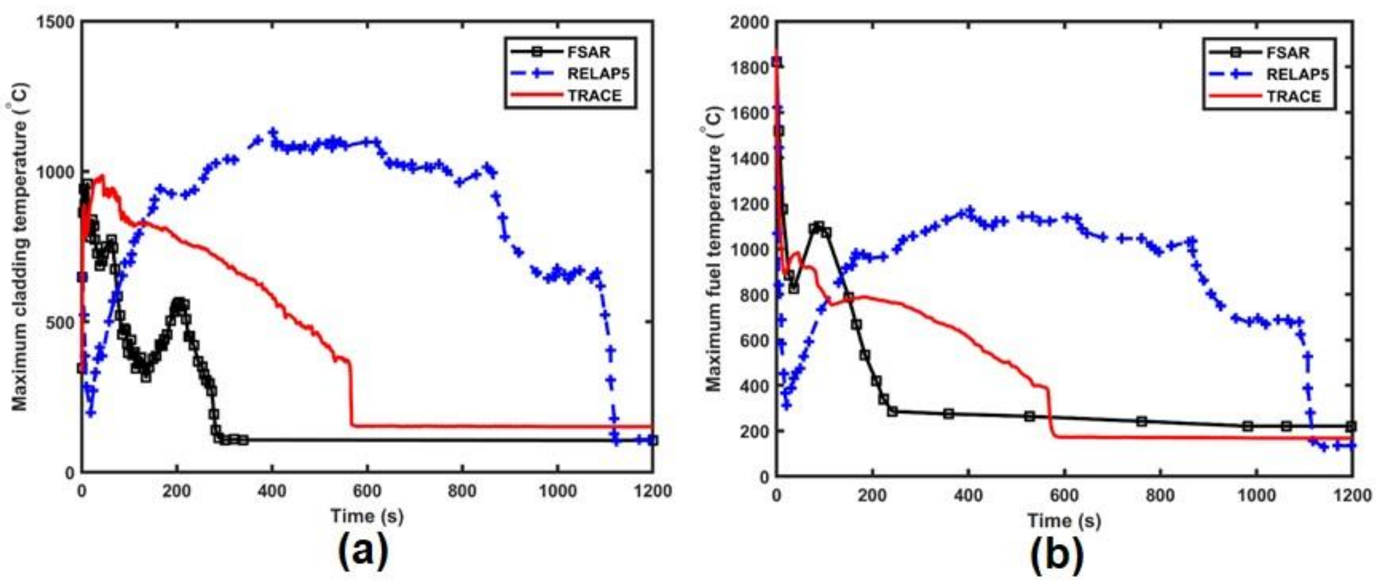

Figure 9. Maximum temperature of: (a) clad, (b) fuel.

4.4 CONTAIN results

The containment system is designed to withstand the effects of a maximum possible safety shutdown earthquake (SSE) including LOCA concurrent with single active failure in a safety system. The containment system is designed to contain the pressure generating as a result of the worst-case LOCA for the containment vessel, a NB 850 mm (break with 850 mm diameter) of main coolant leg split.

In the calculations of postulated accidents, it is conservatively assumed that the containment is almost at atmospheric pressure. The temperatures inside and outside the containment correspond to the maximum design values within the range for normal operation conditions as in restricted access zone (30 °C), in unattended zone (60 °C) and in the annular space (30 °C). The maximum allowable leakage rate through the steel containment equals to 0.25 % of total containment air volume per day at maximum design absolute pressure 0.46 MPa.

Step by step input of blowdown source including break mass and energy of water and steam (resulted by TRACE) is using as CONTAIN input in each time step. The average containment pressure in the same step will be fed back as boundary condition to BREAK component of TRACE (Figure 5). In the initial step, the primary pressure of containment was considered as 0.98 MPa (slightly sub-atmospheric). Figure 10 depicts the mass and energy of blowdown source profiles. As this figure shows, mass and energy profiles of blowdown source resulted from TRACE code greatly following the FSAR profile in both trends and values. It should be noted as these profiles make the main CONTAIN input in each time step, the proximity of FSAR and TRACE results can affect the final TH profiles inside containment compartments (control volumes).

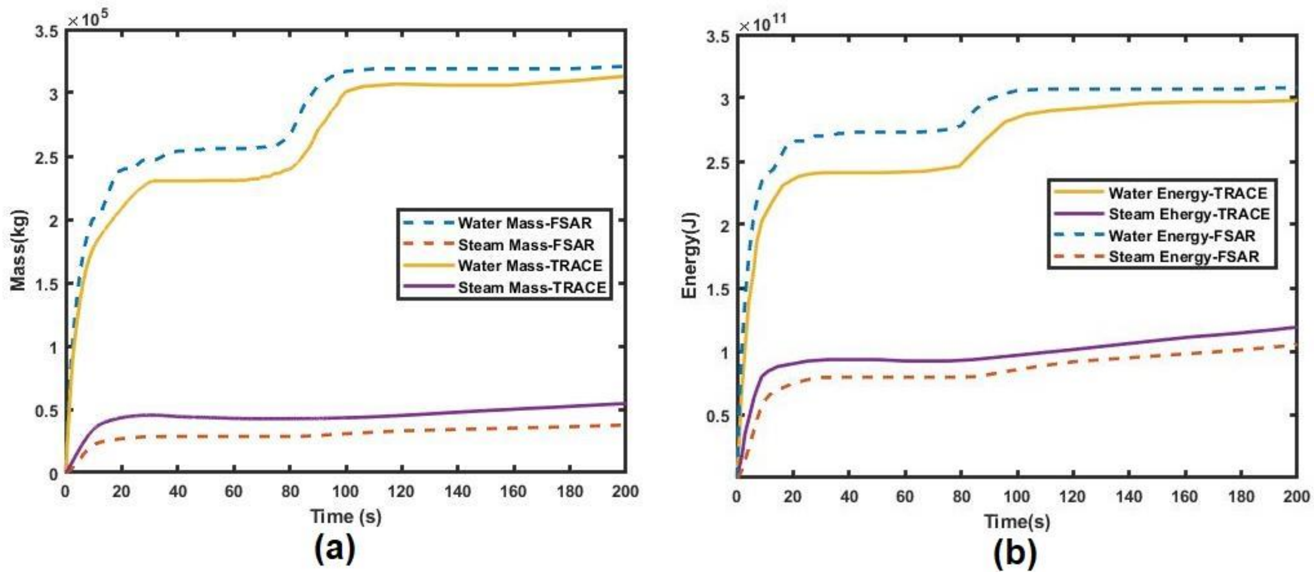

Figure 10. Blowdown source: (a) Mass, (b) Energy.

Pressure is distributing quickly through different containment compartments and flow path (in comparison with heat transfer process). The profiles of pressure inside control volumes during the accident are almost the same in value and trend. As a result, average pressure profile is considered in this accident and has been shown in Figure 11.

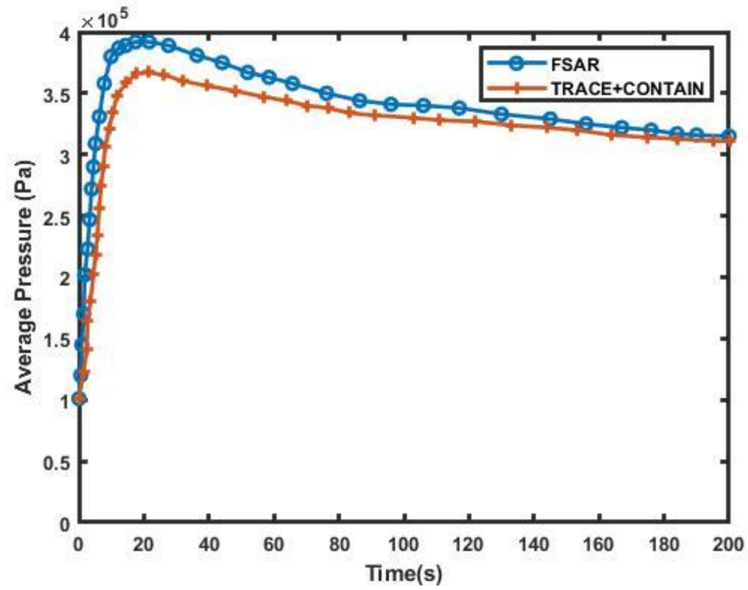

Figure 11. Average pressure profile inside containment as result of blowdown.

As it can be seen in this figure, average pressure profile follows the same trend of steam blowdown profile, although the actuation of spray systems at 0.13 MPa force average pressure to have descending behaviour after its maximum point. Maximum pressure occurs almost at the initial 25-30 seconds of the accident. It is valuable to be noted as containment pressure did not exceed the maximum design pressure of containment in the short time (0-200 second), containment integrity will be maintained during the accident.

Temperature profiles inside control volumes are affected by energy transfer as result of mass transfer between control volumes, heat transfer of heat structures and spray actuation. Although the overall trends (shape) of temperature profiles are the same in different control volumes, the above-mentioned factors can impress the maximum temperature and time functionality of temperature profiles.

Figure 12 shows the as-built 3D geometry of control volume No 7, 12, 25, 28 and 29+30 (29 adjacent with 30) were selected typically among all 30 volumes to validate the temperature profile results (specification of these control volumes can be found in Table 2).

Temperature profile of above control volumes in short time of 0-200 second are presented in Figure 13(a) to (e). As can be seen, all the profiles have almost the same trend. Initially, temperature increases to a maximum amount as result of blowdown source. Initiating spray actuation led to reduction of steam mass and energy (by condensation of steam on spray droplets) and eventually descending the temperature (and pressure) after initial 25-30 seconds.

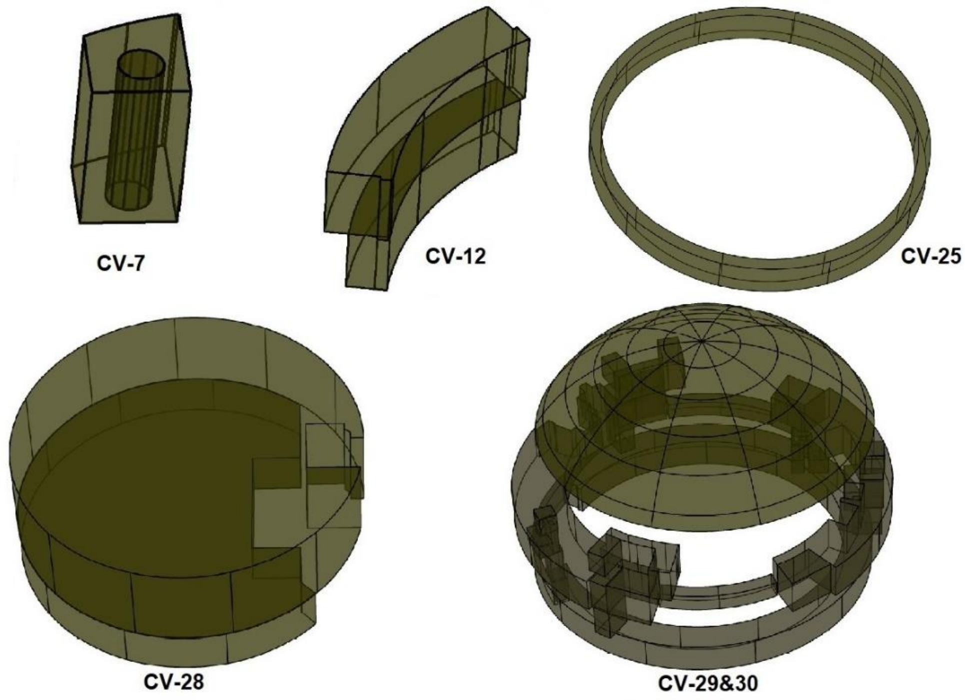

Figure 12. As-built 3D geometry of control volume No 7, 12, 25, 28 and 29&30 (29 adjacent with 30).

Discrepancies between temperature profiles in different control volumes are due to parameters that affect heat transfer phenomena, location of each control volume, its connection respect to the blowdown source (engineering path) and other control volumes. Based on the result of Figures 11 and 13, it can be seen that containment can preserve its integrity during this accident as its TH parameters didn't surpass their maximum design values.

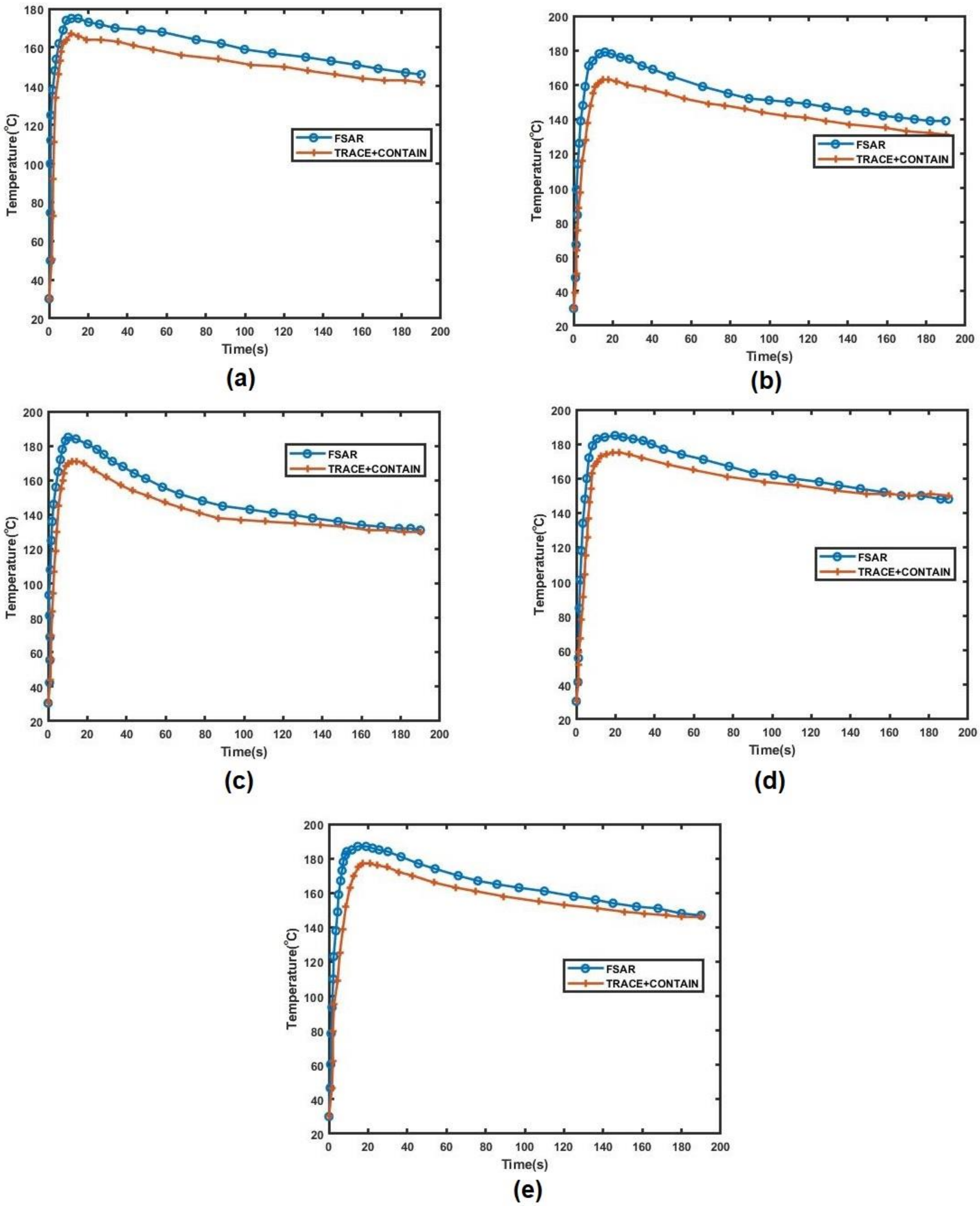

Figure 13. Temperature profiles in control volumes (a) No7, (b) No12, (c) No28, (d) No28 and (e) No29+30.

5. Conclusion

Deterministic safety assessment (DSA) is one of the approaches in safety analysis of nuclear power plants that ensure the beneficiary, stakeholders and public from the safe operation of NPPs. The objective of deterministic safety analysis is to confirm that safety functions can be fulfilled and that the necessary structures, systems and components, in combination with operator actions, are effective in keeping the releases of radioactive material from the plant below acceptable limits (IAEA, 2019). Nuclear simulation codes are one of the DSA main tools used to simulate the different aspects of NPPs in steady-state and transient status. In this article, a comprehensive parallel code coupling was developed to simulate the break source and pressurization of containment in a LBLOCA. As results of this study following remarks can be concluded:

- As it has been implemented in this study, coupling of break source (resulted from TRACE simulation) and containment pressure (resulted from CONTAIN code) -because of their mutual simulation effects- can modify the result of pressurization simulation in comparison to batch simulation (batch simulation is taking break source from reference and simulates pressurization in containment). These modifications in result can be easily investigated by comparison of results of this study by previous studies of authors (Noori-Kalkhoran et al., 2014b, 2014a; Noori-Kalkhoran et al., 2016).
- As it can be seen in figures 6 to 9, the results of this study (using TRACE) for simulation of LBLOCA accident follow the FSAR results that can confirm validation of blowdown source simulation.
- The modified 30-control volumes simulation presents the effect of containment nodalization on the result of pressurization accident. This factor along with the first-mentioned point resulted in more proximity of pressurization profiles and reference ones in contrast with previously published works.
- Figure 11 shows that the maximum pressure inside the containment can hit the 0.36 MPa and 0.395MPa for this study (TRACE coupling with CONTAIN) and FSAR respectively. In both case shapes and values of profiles are in proximity and lower than the maximum design absolute pressure of 0.46 MPa for BNPP. This can confirm the ability of BNPP safety features to cope with the accident and keep the integrity of the containment.
- Pressure and temperature reductions in containment after the maximum points (figures 11 and 13) can confirm the effectiveness of the spray system (as ESFs) on mitigation of parameters during pressurization accident. Undoubtedly without spray actuation both pressure and temperature values could pass their maximum design values and jeopardize the containment integrity.
- Discrepancies of results of this study and RELAP5 with FSAR (FSAR using TETCH-M-97 code for blowdown source simulation and ANGAR code for containment pressurization (AEOI, 2007)) are due to using distinct nuclear simulation codes, different codes' correlations, various heat transfer correlations, different coupling methods, code and input assumptions, combination of errors and different nodalizations that have affected the results.

- As comprehensive coupling needs to simulate a wide variety of components in both loops and containment, it can raise the risk of “combination errors” in the result. Uncertainty analysis and selecting accurate values of stimulation parameters can tackle this issue.
- Comprehensive simulation of containment pressurization accident with considering blowdown source’s simulation creates wide ranges of flexibility in the study of coolant loop parameters effects on containment pressurization. This flexibility can support the designers and analyzer to find the best solution to avoid the accident and mitigating their consequences.

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Figure Captions:

Figure 1. Schematic of BNPP1- primary loop (AEOI, 2007).

Figure 2. Nodalization of BNPP-1 primary and secondary loops applied in TRACE.

Figure 3. Containment 3D plan, (a) cylindrical concrete containment (b) layout of concrete and steel containment (c) layout of compartments inside steel containment.

Figure 4. Layout of Control volumes in containment a) Rear vertical cross section b) Front Vertical cross section.

Figure 5. Coupling procedure schematic.

Figure 6. (a) Relative power of reactor, (b) pressure of coolant at core outlet.

Figure 7. Mass flowrate of: (a) Leakage from break point, (b) ECCS.

Figure 8. (a) Pressurizer water level, (b) Pressure of secondary side of steam generator

Figure 9. Maximum temperature of: (a) clad, (b) fuel.

Figure 10. Blowdown source: (a) Mass, (b) Energy.

Figure 11. Average pressure profile inside containment as result of blowdown.

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