RPE Radiation Physics & Engineering
An International Peer-Reviewed Journal
Published by K.N. Toosi University of Technology
http://rpe.kntu.ac.ir
Research Article

Radiation Physics and Engineering 2020; 1(4):17–27
https://doi.org/10.22034/rpe.2020.104840

Investigation of molten material retention during the large and small breaks LOCA and station blackout accident

Mohsen Salehi[a,*], Mohsen Shayesteh[b]

[a] *Young Researchers and Elites club, Science and Research Branch, Islamic Azad University, Tehran, Iran*
[b] *Physics Department, Imam Hossein University, Tehran, Iran*

HIGHLIGHTS

- The RELAP5/SCDAP is modeled for three-loop PWR NPP plant.
- SBLOCA, LBLOCA, and SBO are analyzed as three significant severe accidents.
- The main parameters such as creep rupture of the surge line of PRZ and lower plenum failure time are reported.
- The IVR strategy is investigated for SBO, and small and large LOCA.

ABSTRACT

The analysis deals with the assessment of best estimate code RELAP5/SCDAP mod3.4 in the simulation of double-ended loss coolant accident as a LBOCA, 4 in break as a SBLOCA an SBO accident with considering except accumulator water where no core cooling water systems are available. The reference plant is SURRY nuclear power plant as a Westinghouse three-loop nuclear power plant. In order to mitigation accident, the in-vessel retention strategy was investigated for the prevention of lower plenum failure. It has been concluded that during the SBLOCA, LBLOCA conditions bottom of active fuel is uncovered at 6340 s and 2160 s, respectively. It occurred for two times at 11650 s and 15608 s in SBO. At 6792 s and 57002 s in the LBLOCA and SBO due to reaching melting point and in the SBLOCA at 15215 s due to lower plenum creep rupture, failure of the reactor pressure vessel occurred. The results show that hydrogen production in the SBO is more than the other two cases. For the prevention of the lower plenum failure, the in-vessel molten material retention strategy is investigated as a passive system. The results show that lower plenum heat flux can be kept below the critical heat flux and its integrity is preserved in two cases of this analysis.

KEYWORDS
SBLOCA
LBLOCA
SBO
Creep rupture
In-vessel cooling

HISTORY
Received: 7 September 2019
Revised: 6 December 2019
Accepted: 19 December 2019
Published: October 2020

1 Introduction

Prior to the 1970s, safety studies and analysis of nuclear power plants were essentially limited to the DBA (design basis accident) discussions. The BDBA (beyond design basis accident) and severe accidents were considered incredible and were thought to be of such low probability. So that it was unnecessary to show because these consequences did not harm the public. Following the 1979 accident at Three Mile Island Nuclear Plant in the USA, this viewpoint was altered. Concern over such events was increased by the 1986 accident at Chernobyl in Ukraine. Severe accidents are now considered in the design of all nuclear plants. Finally following the Fukushima Disaster on March 11, 2011, the Passive safety system was developed for mitigation accidents (Gu, 2018). A severe accident in the nuclear reactor is an event that causes multiple failures. These failures consist of the fuel rods damage inside of the reactor core, core degradation, the molten core relocation into the reactor pressure vessel (RPV) lower head, lower head failure, and molten core material slumping into the reactor vessel cavity. In this event, soluble and non-condensable fission products released into the RPV, containment, or the environment.

In the nuclear power plant, the ECCS (emergency core cooling system) supplies the emergency reactor core water in the event. This system is composed of an active and passive system. HPIS (high-pressure injection system), LPIS (low-pressure injection system), and containment spray system are known as the active system that needs to power for actuation. On the contrary, of this system, the accumulator (ACC) and PAR (passive autocatalytic

*Corresponding author: Mo.Salehi@iran.ir

recombiner) as a passive system are operated without the need to the source of power.

Following the loss of coolant accident (LOCA), the pressure and water level of RPV decrease and the containment pressure increases. The absence of core cooling water supply by ECCS leads to core heatup and melting. If the active part of ECCS is not available, the ACCs coolant injection to the system due to pressure setpoint are able to delay the in-vessel core melt progression and lower head of RPV failure (Salehi and Shayesteh, 2017). On the other hand, in the station blackout (SBO) accident, the pressure and water level increase. Following the creep rupture occurrence in the surge line of pressurizer (PRZ), the pressure and water level decrease rapidly. The pressure reaches ACC setpoint and then injection of boric water occurs into the reactor core. It should be noted that the high pressure and temperature in the reactor cooling system generate stresses. These stresses are sufficiently great to cause a creep rupture in the hot leg, steam generator (SG) tubes, PRZ surge line, etc. (Jian and Xuewu, 2007).

Many deterministic analyses are investigated to severe accident process using nuclear computer code such as MELCOR, CONTAIN and RELAP5/SCDAP. This article is focused on RELAP5/SCDAP as the best estimate code. Park et al. (Park et al., 2013) analyzed LOCA without active part of the safety system for the APR1400. In this paper, the effect of the safety injection tank on in-vessel core degradation is evaluated. In another study, Salehi et al. (Salehi and Jahanfarnia, 2016) have studied SBLOCA without safety injection system in VVER-1000/V444 and the final safety analysis report are developed. In some of the studies, the RELAP5/SCDAP results have been compared with other codes such as ASTEC (Kalchev et al., 2005).

Following TMI accident where although 62 tons of the molten core material slumping into the lower plenum, the IVR (in-vessel retention) strategy began. During this accident, the core melts quenched with water inside the vessel and it terminated without lower plenum failure (Chan, 2006). It should be noted that in-vessel melt coolability and retention include three general concepts. These concepts are quenching of the core in situ, coolability of in-vessel particulate beds and coolability of the in-vessel melt pool (Ma et al., 2016). The only operating reactor in the world that has the in-vessel retention of corium as an approved severe accident management measure is the VVER-440 Finnish Loviisa plant. The approach selected takes advantage of the unique features of the plant, such as low power density, a reactor vessel without penetrations and ice-condenser, which ensures a flooded cavity (Seghal, 2012). There are a large number of studies about this strategy and several types of the reactor such as CANDU (Mladin et al., 2010), small Integrated IPWR (Jiang et al., 2019), PWR (Salehi and Shayesteh, 2017) and VVER 1000 (Jahanfarnia and Salehi, 2016).

This analysis deals with simulation of large break loss of coolant accident (LBLOCA), small break loss of coolant accident (SBLOCA), and SBO for a three-loop Westinghouse nuclear power plant (NPP) using RELAP5/SCDAP mod 3.4. The objective of this analysis is to study the behavior of nuclear power plant during the long-term after LOCA and SBO without any source of power for safety injection system (SIS) actuation that may cause overheating of the core and imply a risk of core damage. In the next step of this analysis, the IVR strategy is investigated as a method for mitigation of accidents.

2 Materials and Methods

2.1 General plant description

Surry NPP Unit 1 is the reference plant. It is one of the plant analysis in the severe accident risks (Commission et al., 1990) safety analysis study. According to this document, SBO and LOCA formed more than half of the total mean core damage frequency. It should be noted that according to the United States nuclear regulatory commission website (NRC, 2018), the operation license expires in 2032. The reactor is a typical Westinghouse three-loop pressurized water reactor (PWR). Thermal power is 2441 MW_{th} and the core consists of 157 fuel assemblies. The furl rod array in assembly is 15×15. The active part of fuel height is 3.66 m. There are a U-tube SG, a reactor coolant pump (RCP), and associated piping for each coolant loops. The surge line of PRZ, cold and hot leg pipes are constructed of stainless steel. The tubes of SG are made of Inconel. The power operated relief valves (PORV) and safety relief valves (SRVs) are located at the top of PRZ and can be used to relieve excess pressure in reactor coolant system (RCS). There are an ACC for each loop and containing 29100 kg of borated water at 48.85°C. It should be noted that it is attached to each cold leg. The accumulators, which are initially pressurized to 4.24 MPa with a nitrogen gas cover, are the only operational part of ECCS during RPV Pressure drop. A large, dry, subatmospheric pressure containment building surrounds the reactor system. The simplified schematic of Surry containment and loops are shown in Figs. 1 (Pilch et al., 1995) and 2 (USNRC Technical Training Center).

2.2 SCDAP/RELAP5 code description and input modeling

The RELAP5/SCDAP computer code is designed to calculate the severe accident situations of the overall RCS thermal-hydraulic response, the core damage progression, the reactor vessel heat-up and damage, and the fission product release and transport. The code is the result of merging the RELAP5 MOD3 (RELAP5, 1995), SCDAP as a severe core damage analysis package models (Allison and Beers, 1984), and the COUPLE model (Lemmon, 1980) that calculates the heat up of the debris and surrounding structures in the reactor lower head vessel after a core slumping to the lower head occurs.

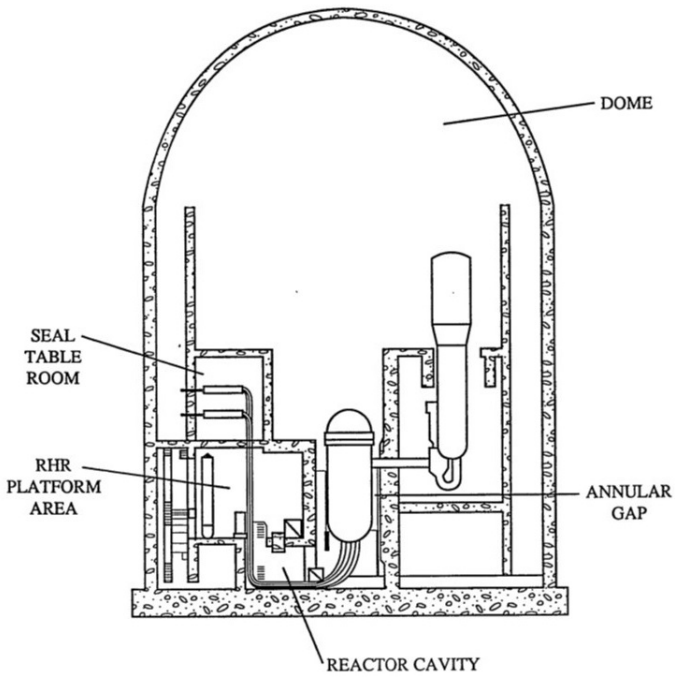


Figure 1: Simplified schematic of Surry containment.

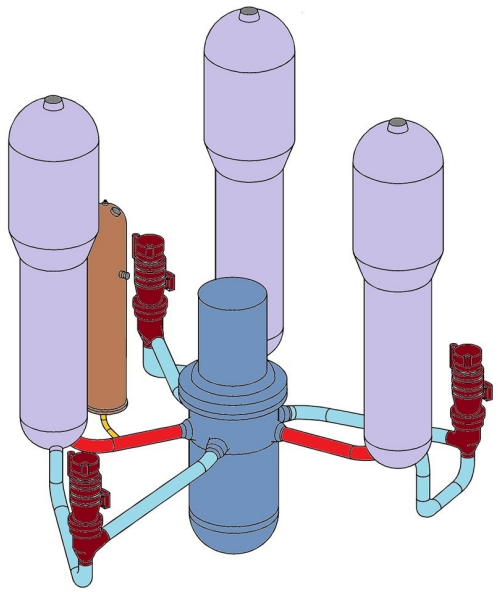


Figure 2: Simplified schematic of Surry loops.

The overall RCS thermal-hydraulics, the control system interactions, the reactor kinetics, and the transport of noncondensable gases are calculated by RELAP5 models. The heat-up and the damage progression in the core structures and the lower head of the reactor vessel are calculated by the SCDAP model. Treatment of the core in this model includes fuel rod heat-up, ballooning and rupture, fission product release, rapid oxidation, Zr melting, UO_2 dissolution, ZrO_2 breach, flowing and freezing of the molten fuel and cladding, and debris formation and behavior. It should be noted that RELAP5 calculates the flow losses in porous debris. The COUPLE model takes into account the decay heat and internal energy of the newly fallen or formed debris and then calculates the transport by conduction of this heat in the radial and axial directions to the wall structures and the water surrounding the debris. The most important use of this model is to calculate the heatup of the vessel wall and consequently, the time at which the vessel may rupture can be determined.

The input model for reactor plant simulation is a combination of RELAP5, SCDAP, and COUPLE input models. Heat structures for the fuel rods and lower part of the reactor vessel in the RELAP5 input model were replaced by SCDAP and COUPLE input models, respectively. The RELAP5 nodalization of the reactor NPP and lower plenum (LP) COUPLE model are shown in Fig. 3. As shown, flow channels of the reactor core are represented using 5 radial segments and 10 axial nodes. Each flow channel is connected to its neighboring flow channels by crossflow junctions, thus flow in the lateral direction through the core is modeled. The core bypass is represented by a RELAP5 control volume (pipe) containing five sub-volumes and the downcomer is represented by a RELAP5 control volume (pipe) with seven sub-volumes. One RELAP5 control volumes are used to represent the lower plenum and seventeen RELAP5 control volumes to represent the upper plenum region of the reactor vessel. The sub-atmospheric containment design pressure and failure setpoints are 0.412 MPa and 0.756 MPa, respectively (Hessheimer and Dameron, 2006). The containment is modeled by a single volume and a simple heat structure is assumed for heat transfer to the environment.

In the SCDAP input model, the reactor core was divided into five regions according to five flow channels in RELAP5 and for this analysis by grouping similarly powered fuel assemblies together. In this model, there are 10 components for the fuel and the control rods modeling. The numbers of axial nodes for the fuel and control rods are l0 in each component to accurately simulate ballooning and relocation after rupture of the fuel cladding, and the number of radial nodes for the fuel and control rods is 6 and 5, respectively. Figure 4 is a cross-section of the core illustrating each region and its average radial power peaking factor. It should be noted that baffle or shroud is modeled.

A two-dimensional finite element mesh was used to represent the reactor vessel lower head in the COUPLE model. The lower part of the reactor vessel is divided into 221 nodes and 192 elements. The lower head wall is divided into five heat conduction elements. The remaining elements are initially filled with primary coolant, which can either boil away or be displaced by core debris. Convection and radiation heat transfer are modeled at all interfaces between the coolant and core debris. The outside surface of the lower head is assumed to be adiabatic in without the IVR state. In contrast to this state, the outside of LP wall is connected to an insulator. The heatup of the debris that slumps into the LP and the heatup of the vessel wall in the axial and radial directions are calculated.

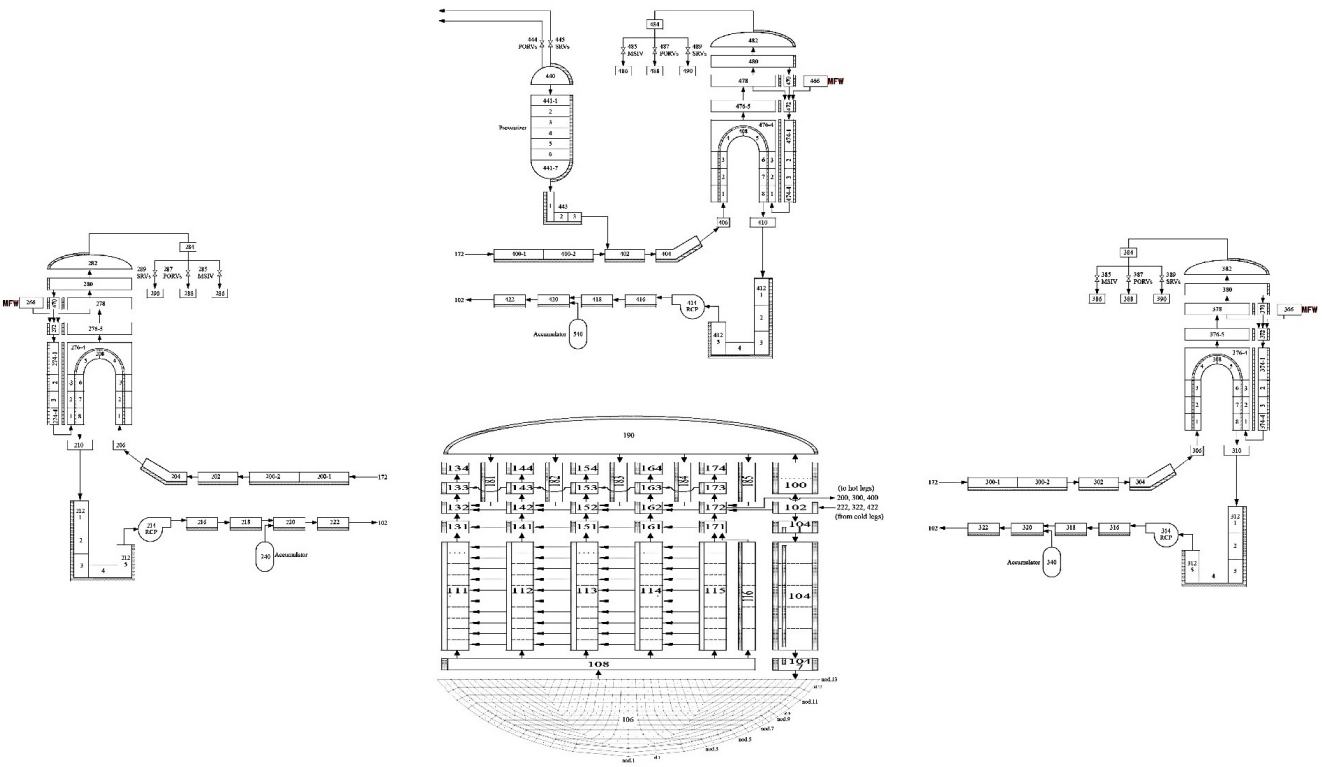
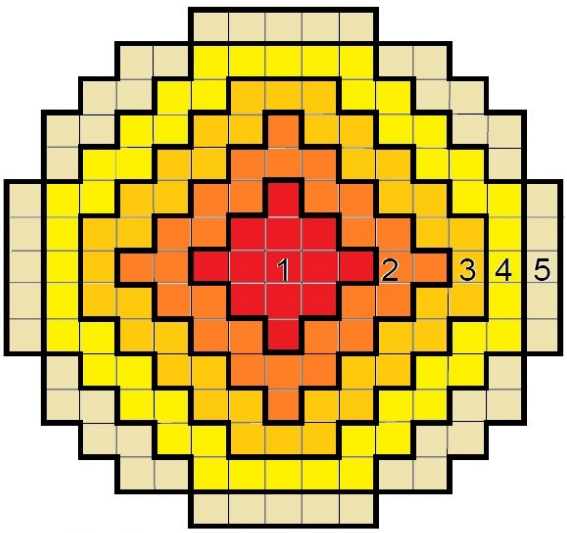


Figure 3: Nodalization of the Surry NPP and lower head coupling model.



Region	Radial Peaking Factor
1	1.190
2	1.164
3	1.118
4	1.1005
5	0.762

Figure 4: Cross-sections of core showing fuel assembly grouping and radial peaking factors

The most important output of this model is the calculation of the vessel wall temperature from which the time for creep rupture failure, or melting of the lower head, can be estimated. The model accounts for the time-dependent debris bed and considers spatially varying initial internal energy, decay heat porosity, particle size, and effective (thermal and radiative) conductivity of the porous material. A value of 10000 W/m.k is currently used, which is designed to simulate essentially perfect contact that would be expected if liquefied corium filled all voids associated with the surface roughness of the lower head. The reactor lower head will fail due to melting (Nunez-Carrera et al., 2012) or creep rupture. Two different theories are applied for creep rupture: Larson-Miller (Larson, 1952) and Manson-Haferd (Manson and Haferd, 1953).

It should be noted that in addition to the reactor lower vessel creep rupture, a model based on creep rupture theory is used to calculate the damage and nearness to rupture of structural components selected by the code user, such as a steam generator U tube, hot leg, PRZ surge line. The particular theory to be applied is dependent on the material composition and stress. For 316 stainless steel and Inconel 600 materials, the Larson-Miller theory is used. For A-508 Class 2 carbon steel, in the lower range of stress the Manson-Haferd theory and the Larson-Miller theory for the higher range of stress are applied. The creep damage is evaluated by Eq. 1:

$$D_c(t + \Delta t) = D_c(t) + \frac{\Delta t}{t_r(t)} \tag{1}$$

where $D_c(t)$ is the creep damage at time t, Δt is the time step at the current problem time (s), $t_r(t)$ is the time required for the structure to fail by creep rupture at the current state of temperature and stress (s), and t is the problem time (s). If the value of D_c is zero, the structure has not experienced any creep damage. If the value

is one, the structure has failed due to creep damage. The equation for calculating tr is depended on the material composition and stress. The equations are shown in Table 1. The lower head, stress term in the equations is calculated by Eq. 2:

$$\sigma = \frac{(P_i r_i^2 - P_0 r_0^2)}{(r_0 + r_i)(r_0 - r_i)} + 0.5(P_i - P_0) \quad (2)$$

where σ is the stress in creep rupture equations (converted from Pa to ksi), P_i indicates the internal pressure on structure (Pa), P_0 is the external pressure on structure (Pa), r_i is the inner radius of structure (m), and r_0 is the outer radius of structure (m).

Table 1: Equations for calculating the time to creep rupture.

Material	Range of stress (σ)	Equation for creep rupture time *
Carbon Steel A-508	$0.0 < \sigma < 14.0$	$t_r = 10^{[\frac{T-1503.69}{P}+3.499]}$ P=158.233 $\log(\sigma) - 255.346$
	$\sigma \geq 14.0$	$t_r = 10^{[\frac{P}{T}+20]}$ P=9603.0 $\log(\sigma) - 46454.0$
316 stainless steel	$0.0 < \sigma < 52.0$	$t_r = 10^{[\frac{P}{T}+20]}$ P=-13320.0 $\log(\sigma) + 54870.0$
	$\sigma \geq 52.0$	$t_r = 10^{[\frac{P}{T}+20]}$ P=-64000.0 $\log(\sigma) + 142000.0$
Inconel 600	$0.0 < \sigma$	$t_r = 10^{[\frac{P}{T}+20]}$ P=-11333.0 $\log(\sigma) + 43333.0$

* t_r: time to rupture (h), T: average temperature of structure (Raumure), σ: stress in structure (ksi)

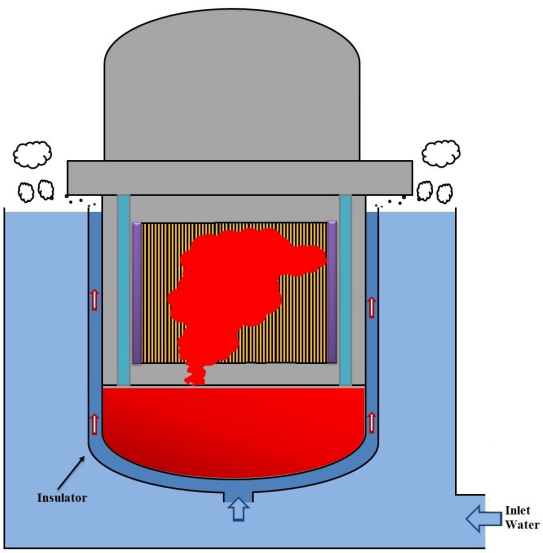


Figure 5: Schematic view of IVR strategy.

The molten core may relocate to the RPV lower head in a severe accident nuclear power plant. The molten core material could cause a failure of the lower head of the RPV, if sufficient internal or external cooling of the vessel could not be provided. In IVR strategy, there is an annulus between the vessel wall and cavity wall that it is called insulator. It serves as the riser for the two-phase flow around the vessel. A set of nucleate boiling heat transfer correlations were implemented to calculate the transfer of heat from a flooded lower vessel head for a hemispherical surface. The set of subcooled nucleate boiling correlations were developed for calculation of heat flux from a hemispherical surface in Eq. 3:

$$q = a\Delta T + b\Delta T^2 + c\Delta T^2 \quad (\frac{\mathrm{MW}}{\mathrm{m}^2}) \quad (3)$$

where a, b, and c are position-dependent coefficients from Table 2 for the orientation shown in Fig. 5 and ΔT is the difference between the wall surface temperature and the pool saturation temperature (°K). Nucleate boiling curves derived from Eq. 3 are valid from a ΔT of ~ 4°K to the ΔT associated with the critical heat flux (CHF) for subcooled boiling given by Eq. 4

$$q_{chf} = 0.4(1 + 0.036\Delta T_{sub})(1 + 0.021\theta - (0.007\theta)^2) \quad (4)$$

where q_{chf} is the critical heat flux in $\frac{\mathrm{MW}}{\mathrm{m}^2}$, ΔT_{sub} is the degree of subcooling and θ is the surface contact angle in degrees for the orientation. Schematic view of IVR strategy is shown in Fig. 5.

3 Results and discussion

The steady-state conditions obtained from the simulation were used as initial conditions for the transient calculation therefore it was performed to verify the input nodalization. Table 3 shows a comparison of the Surry unit.1 operating design parameter (Manson and Haferd, 1953) with the simulation results. In addition, the RELAP5/SCDAP results are compared with acceptable error (Petruzzi and D'Auria, 2008). The simulation values except the hot leg temperatures are approximately in good agreement with the design values. The discrepancies are limited to acceptable margins. Due to the unavoidable approximations in the constitutive equation, the limited capabilities of numerical solution methods, error in setting up the nodalization, uncertainties in the knowledge of boundary and initial conditions, and thermo-hydraulic system code calculation is affected by unavoidable error arising.

The accident started at 100 s. The LBLOCA and SBLOCA initiated with double-ended rupture and 4-inch break in the cold leg of the loop C (with PRZ), between the RCP and RPV. Also due to loss of power, the SBO initiated without break in the coolant system but in this accident the creep rupture model for PRZ surge line, SG U-tube and hot leg is available. The time sequence of events during three accidents considering the failure of HPIS and LPIS is presented in Table 4. The simulation results for these events are shown in Figs. 5 to 12.

It is assumed, immediately following the initiation event the reactor trip signal is initiated and after 2s control rod falls into the core, and the power decreases rapidly to decay heat power (approximately 7% of its nominal value) as shown in Fig. 6. If core cooling is inadequate during the accident, the decay power let to the core heat up and degradation.

Table 2: Subcooled nucleate boiling correlation coefficients as a function of position on the exterior surface of the reactor vessel lower hemispherical head.

Subcooled Boiling Correlation	Data Reported at	Correlation Applied at	Correlation Coefficients		
			a	b	c
1	$\frac{L}{D} = 0$	$0 < \frac{L}{D} \leq 0.1$	0	319	-2.83
2	$\frac{L}{D} = 0$	$0.1 < \frac{L}{D} \leq 0.275$	4016	430	-4.13
3	$\frac{L}{D} = 0$	$0.275 < \frac{L}{D} \leq 0.425$	0	337	2.61
4	$\frac{L}{D} = 0$	$0.425 < \frac{L}{D} \leq 0.625$	0	891	-9.04
5	$\frac{L}{D} = 0$	$\frac{L}{D} > 0.625$	0	529	-0.08

Table 3: Comparison of the Surry operation conditions with RELAP5/SCDAP results.

No.	Parameters	Surry Design Parameters	RELAP5/SCDAP	| Error |* (%)	Acceptable Error (%)
1	Power, MW_{th}	2441.0	2441.0	0.0	2.0
2	Primary system pressure, MPa	15.51	15.51	0.0	0.1
3	Core inlet coolant temperature, °C	282.44	281.18	0.45	0.5
4	Core outlet coolant temperature, °C	318.67	316.22	0.77	0.5
5	Tube side design flow rate, $kg.s^{-1}$	12738.39	12731.22	0.06	2.0
6	Pressure of steam generator secondary side, MPa	5.4	5.4	0.0	0.1
7	Temperature of steam generator secondary side, °C	268.94	269.44	0.19	0.5
8	Initial mass of UO_2, kg	79650.8	80964.7	1.65	-
9	Initial mass of Zr, kg	16465.4	15968.9	3.02	-

*[(Design Parameter-Calculated Value)/ (Design Parameter)] %

Table 4: Time sequence of event.

No.	Parameters	Time (s)			Interlocking, Setpoint for Actuation and other Causes
		SBO	SBLOCA	LBLOCA	
1	LOCA Occurs in Loop C Cold Leg, Scram Signal Generation Closer of Turbine Stop Valve	100	100	100	Initiating Event, Turbine Trip
2	Beginning of Control Rod Motion	102	102	102	Reactor Scram
3	Trip of RCP and MFW	103	103	103	RCP and MFW Pump Trip (3.0 s After Turbine Trip)
4	Top of Active Fuel Uncovered	9400 & 13184	600 & 3640	100	RPV Water Level < 6.7 m
5	ACC Injected	12784	840	101	RPV Pressure < 4.2 MPa
6	ACC Empty	12834	4740	140	-
7	Reaching of Fuel Rod Cladding Temperature value of 1200 °C	11800	5840	350	Core Uncovery and Oxidation Processes
8	First FP Releases	12050	5680	320	Cladding Rupture
9	Initial Relocation to LP	15358	9100	2440	Melt through Structure
10	Bottom of Active Fuel Uncovered	11650 & 15608	6340	2160	RPV Water Level < 3.06 m
11	LP Dryout	36937	14560	4920	Vaporization of all Water in LP
12	LP Failure (End of Calculation)	57002	15215	6792	Reaching to Melting Point/Creep Rupture

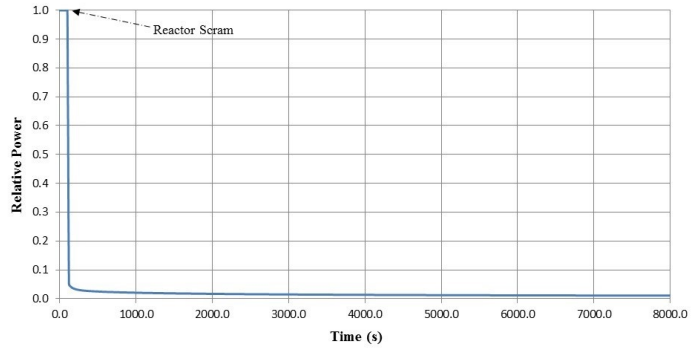


Figure 6: Relative core power.

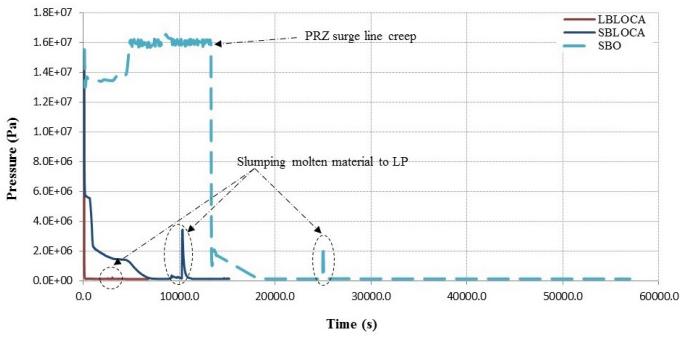


Figure 7: Pressurizer pressure.

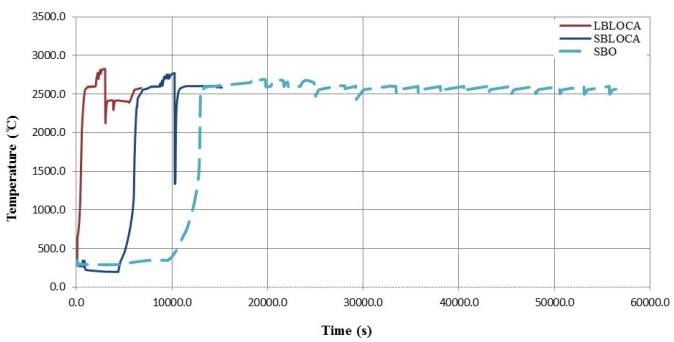


Figure 8: Core maximum surface temperature.

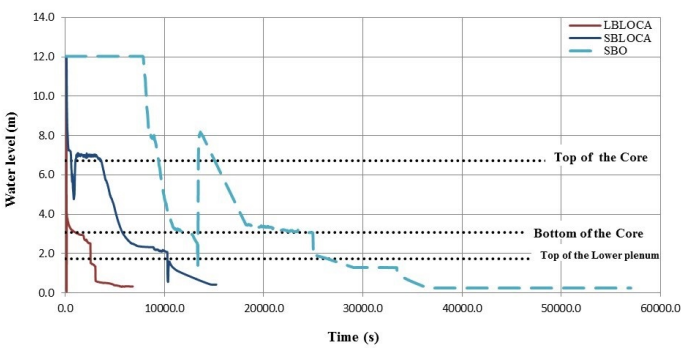


Figure 9: RPV collapsed liquid water level.

The pressure of PRZ is shown in Fig. 7. Following the accident the pressure decreases in the LBLOCA and SBLOCA. In contrast to LOCAs the pressure in the first step decreases due to reactor scram and then because of saving decay heat energy in the coolant the pressure increases in the SBO. Due to pressure and temperature increasing the PRZ surge line ruptured and pressure decrease rapidly about 12784 s there are some pressure picks that are due to molten material slumped to LP. The increasing scale is related to temperature and mass of debris. It should be noted that ACC injection was caused decreasing rate of pressure drop in SBLOCA and SBO.

The Core maximum surface temperature of each modeling components is depicted in Fig. 8. The timing of the progression of the core damage is indicated by the surface temperature of core component. The core damage can occur when at any location the temperature of the clad exceeds about 726.85°C. In the SBLOCA, LBLOCA and SBO this condition occurred at 5050 s, 110 s and 11458 s respectively. According to PRZ pressure (Fig. 7) after the creep rupture of surge line, the component temperatures increased in the core. At the temperature, about 2596.7°C, melting of the core component began, which was calculated to occur for SBLOCA at 6330 s, LBLOCA at 3150 s and SBO at 13443 s. It should be noted that when molten core material is slumping to the LP the temperature rapidly decreases. As shown, in SBLOCA more than two other cases molten material slumped into the LP.

Figure 9 shows the RPV collapsed liquid level. Except SBO, there is a sharp drop in the water level at the beginning of the event for two other cases. The water level was increased by the ACCs coolant injection and the decreased again when these tanks empty at 101 s in LBLOCA condition, at 840 s in SBLOCA condition and in SBO at 12784 s. In LBLOCA condition the core is uncovered after the accident and ACCs actuation cannot refold the total height of it. If reactor core uncovered, heatup and molten core material occurred, and then molten material relocated into the LP. Heat transfer from this hot debris boils away the remaining water inventory as shown this figure. In the SBLOCA compared to the other two accidents, the performance of ACCs is better in providing core coolant inventory. The void fraction of LP is illustrated in Fig. 10. As to be seen when the molten material slumps into the LP the entire inventory of water boils.

Since the main mechanism of RPV hydrogen production is the reaction between vapors and fuel cladding, hydrogen will not be produced as long as the cladding that is not exposed to the vapor as shown in Fig. 11. So the hydrogen production begins after or during the core uncover process. As shown in this figure approximately 178.6 kg, 206.3 kg, 607.8 kg of hydrogen was produced during the SBLOCA, LBLOCA and SBO respectively.

When the cladding temperature reaches about 1072.0°C at 5680 s in SBLOCA, 1139.9°C at 320 s in LBLOCA and 1208.2°C at 12050 s in SBO first fission products released from the fuel rod gap to RCS. The total of Xe, Kr, He, H2 as non-condensable fission products and CsI, CsOH as soluble fission products, fission products releases to coolant in three cases are shown in Fig. 12.

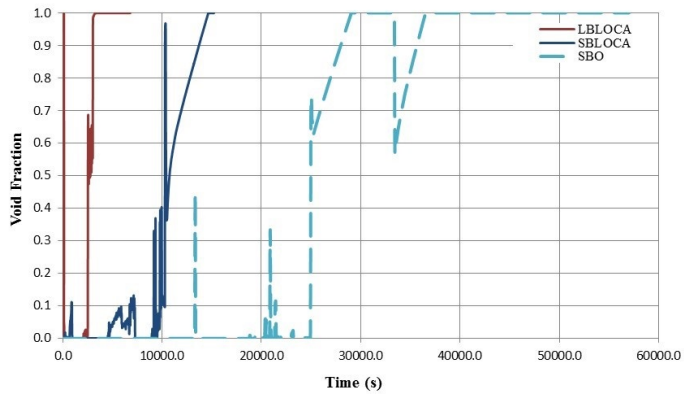


Figure 10: Void fraction of LP.

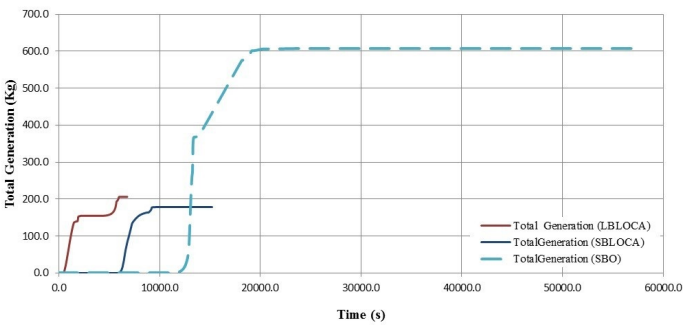


Figure 11: Total generation of H_2.

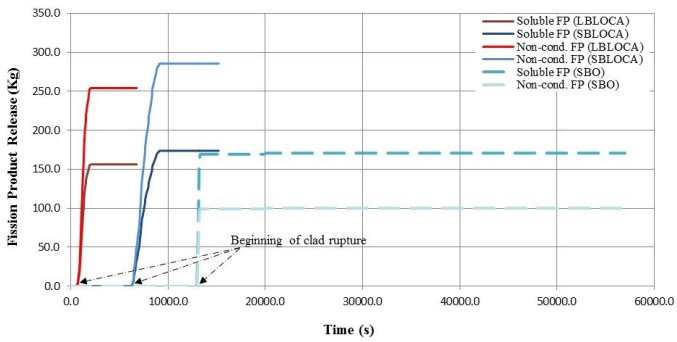


Figure 12: Mass of soluble and non-condensable fission products release.

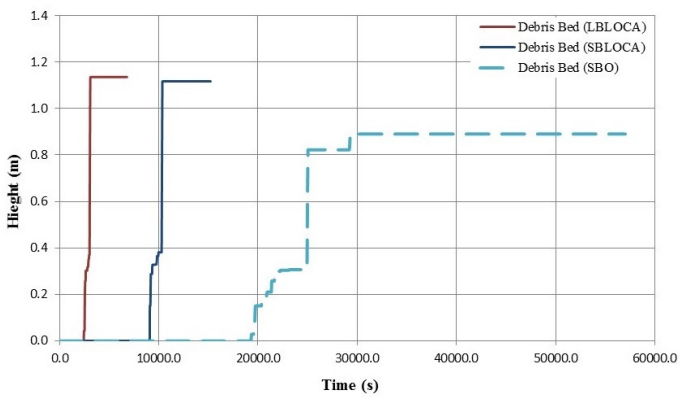


Figure 13: Debris height.

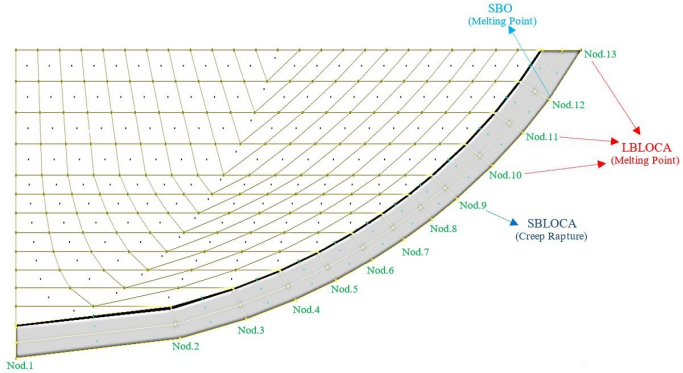


Figure 14: Failure locations in COUPLE mesh.

Figure 13 shows debris height. The occurrence of the molten pool marks a significant increase in the reactor core damage. The height of the debris in LP, increases from zero to 1.14 m and 1.11 m during the 3100 s and 10360 s, in LBLOCA and SBLOCA respectively. In the SBO accident, the debris height reaches to 0.89 m at 29337 s. The first increase is related to Ag alloy, of control rods and other increasing related to Zr, ZrO_2, and UO_2.

Following falling debris, the LP can be failed due to wall melting point or creep rupture. Figure 14 shows the location of failure in the COUPLE model mesh points. The LP failed due to reaching the melting point in LBLOCA (nodes 10, 11, and 13) and SBO (node 12) at 6792 s and 57002 s respectively. The creep rupture failure occurred in SBLOCA (node 9) at 15215 s.

The molten material can be relocated into the cavity when the failure of LP occurred. The IVR strategy is used for preventing slumping molten material into the cavity. As shown in Fig. 15 the heat flux in all angles is kept below the CHF by using the IVR strategy in the wet cavity condition. Thus, the IVR strategy can be useful for maintaining RPV integrity.

The mass flow rate of the insulator as a part of the IVR strategy is shown in Fig. 16. This strategy in this analysis is assumed as a passive system. The heat flow of molten material transferred to the insulator channel when it slumped to the LP. Due to heat transfer, different temperatures occurred in this channel and natural flow formed.

The maximum debris temperature in LP is illustrated in Fig. 17. It can be seen the debris temperature in SBO is very high compared to other accidents. The average heat transfer coefficient and the average temperature in the outer surface of LP are shown in Figs. Fig. 18 and 19, respectively.

In the three accidents, material and temperature distribution of LP during the IVR strategy are shown in Figs. 20, 21 and 22. According to simulation 45.59, 47.51 and 39.87 ton UO_2, 2.34, 2.63, and 4.0 ton ZrO_2, 2.52, 4.74 and 2.14 ton Zr are relocated into the lower plenum in LBLOCA, SBLOCA, and SBO respectively. The mass of slumping Ag-In-Cd is 2.09 ton for each large and small breaks. This mass is 2.06 ton in SBO. As shown in these figures the outer wall of LP temperature remained in near the surrounding coolant temperature.

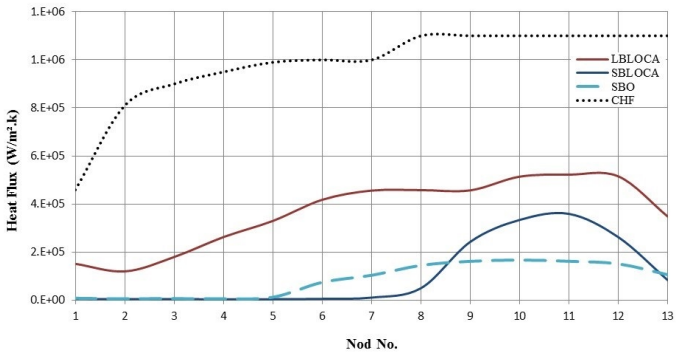


Figure 15: LP heat flux distribution.

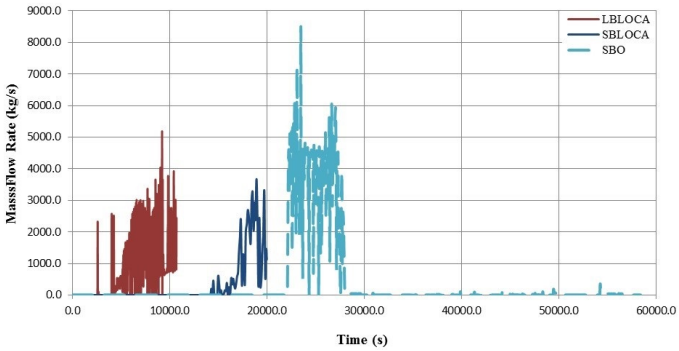


Figure 16: Insulator mass flow rate.

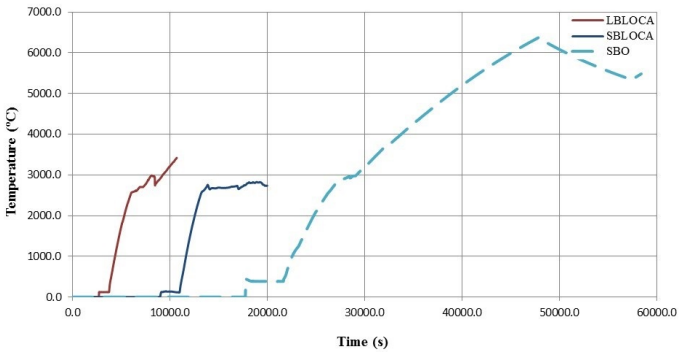


Figure 17: Maximum debris temperature in LP.

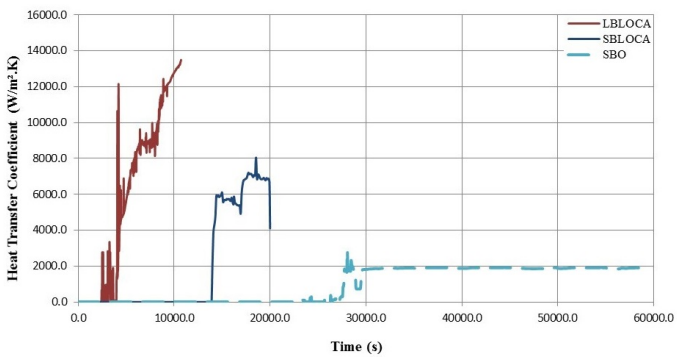


Figure 18: Average heat transfer coefficient at outer surface of LP.

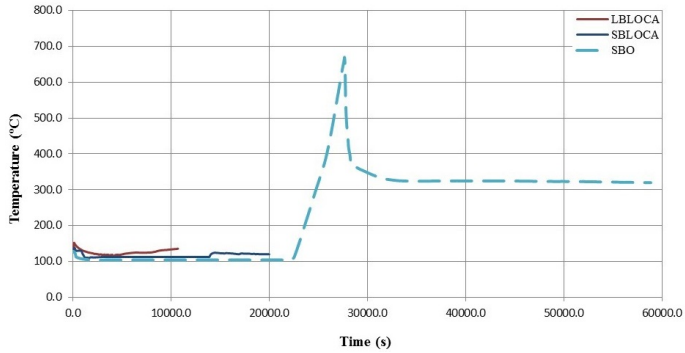


Figure 19: Average temperature of outer surface of LP.

4 Conclusion

The best estimate simulation from the initiating events of 4 in. and double-ended LOCA and SBO in loss of power condition have been carried out with RELAP5/SCDAP code. According to this condition, no source of power is available for SIS, ECCS, and containment spray system pump actuation. Only the ACCs are available as a passive emergency core cooling system. The results have shown that in the LBLCOCA condition phenomena occur earlier than the SBLOCA and SBO. In the small and large breaks accident the pressure and water level in the core decrease but in the station blackout accident the pressure increases. The pressure and water level decreased due to creep rupture failure in pressurizer surge line. Main concern in the SBO accident is amount of hydrogen production. The results show in the SBO condition, the hydrogen was produced about three times more than LOCA conditions. The location of failure in lower plenum was in the upper node of the COUPLE model in all of accidents. In LBLOCA comparison with the two other cases, LP was failed.

In the present article, the IVR strategy has been evaluated for the retention of the relocated core materials within the vessel. Three failure are reported for LP in this paper. The LBLOCA and SBO failures are related to reaching melting point temperature and SBLOCA is due to the creep rupture failure. The IVR strategy has a significant effect on LP integrity if the cavity is full of the coolant before the beginning of the slumping to the lower plenum.

The present results can provide guidelines in establishing detailed severe accident management procedures to mitigate severe core damage accidents in SBLOCA, LBLOCA and SBO without active part of ECCS of the Surry NPP or other nuclear power plants.

Acknowledgment

Dad and Mom, thank you for your encouragement, support, and love.

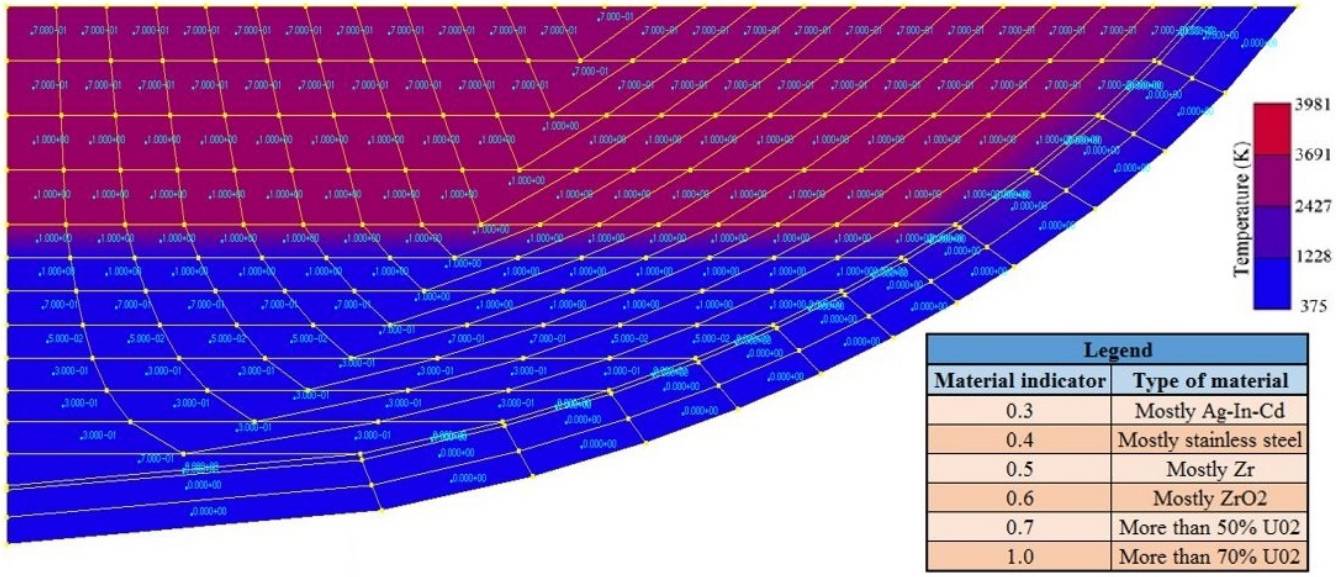


Figure 20: Material and temperature (K) distribution of lower head during IVR strategy in SBLOCA.

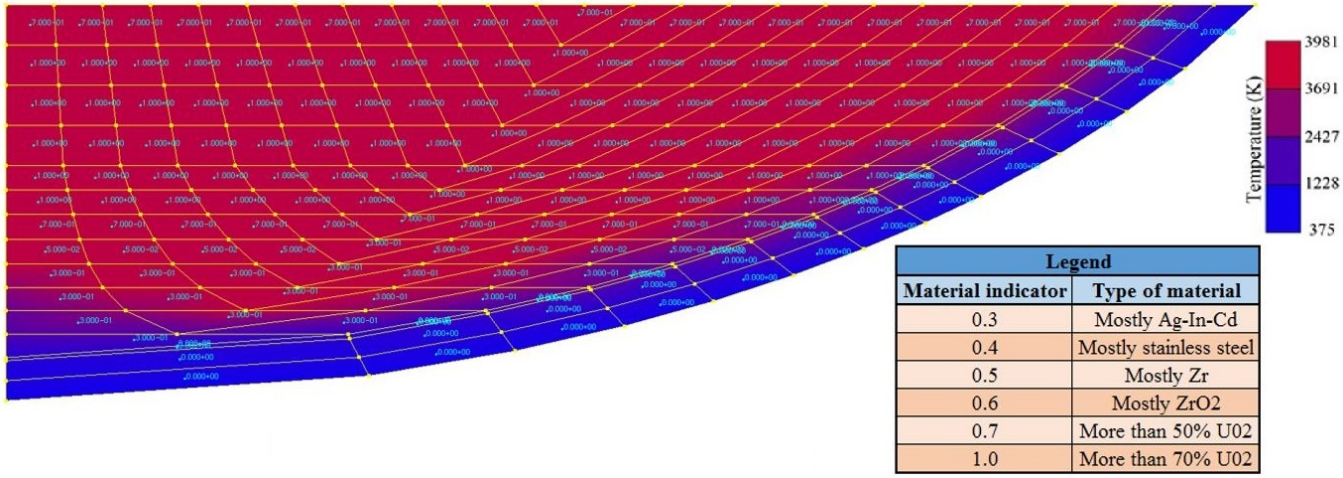


Figure 21: Material and temperature (K) distribution of lower head during IVR strategy in LBLOCA.

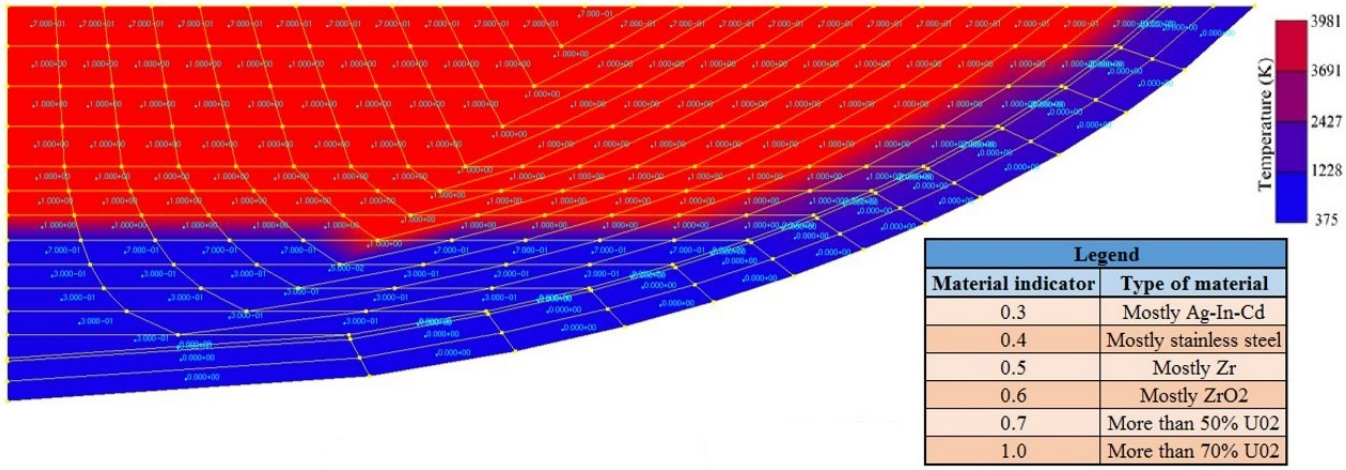


Figure 22: Material and temperature (K) distribution of lower head during IVR strategy in SBO.

References

Allison, C. and Beers, G. (1984). Scdap/mod1/v0: A computer code for the analysis of LWR vessel behavior during severe accident transients. Technical report, IS-SAAM-83-002.

Chan, S. L. (2006). Assessment of S/R5/MOD3. 3 models of stratified molten pool and debris-to-vessel contact resistance by TMI-2 lower-head creep rupture analyses. *Nuclear technology*, 156(2):191–212.

Commission, N. R. et al. (1990). Severe accident risks: An assessment for five us nuclear power plants: Appendices A, B, and C. Technical report, Nuclear Regulatory Commission.

Gu, Z. (2018). History review of nuclear reactor safety. *Annals of Nuclear Energy*, 120:682–690.

Hessheimer, M. F. and Dameron, R. (2006). *Containment Integrity Research at Sandia National Laboratories*. Division of Fuel, Engineering & Radiological Research, Office of Nuclear .

Jahanfarnia, G. and Salehi, M. (2016). The in-vessel melt retention strategy using the RELAP5/SCDAP code for VVER-1000 reactor. *The 8th international conference on sustainable development through nuclear research and education*.

Jian, D. and Xuewu, C. (2007). Analysis of hot leg natural circulation under station blackout severe accident. *Nuclear Science and Techniques*, 18(2):123–128.

Jiang, N., Cong, T., and Peng, M. (2019). Margin evaluation of in-vessel melt retention for small IPWR. *Progress in Nuclear Energy*, 110:224–235.

Kalchev, B., Dimov, D., Tusheva, P., et al. (2005). Comparative severe accident analysis of WWER 1000/B 320 LOCA DN100 computed by computer codes ASTEC V1. 1 and SCDAP/RELAP5.

Larson, F. R. (1952). A time-temperature relationship for rupture and creep stresses. *Trans. ASME*, 74:765–775.

Lemmon, E. (1980). Couple/fluid: A two-dimensional finite element thermal conduction and advection code. Technical report, EGG-ISD-SCD-80-1.

Ma, W., Yuan, Y., and Sehgal, B. R. (2016). In-vessel melt retention of pressurized water reactors: historical review and future research needs. *Engineering*, 2(1):103–111.

Manson, S. and Haferd, A. M. (1953). A linear time-temperature relation for extrapolation of creep and stress-rupture data.

Mladin, M., Dupleac, D., Prisecaru, I., and Mladin, D. (2010). Adapting and applying SCDAP/RELAP5 to CANDU in-vessel retention studies. *Annals of Nuclear Energy*, 37(6):845–852.

NRC (2018). Surry power station, Unit 1. Technical report, https://www.nrc.gov/info-finder/reactors/sur1.html.

Nunez-Carrera, A., Camargo-Camargo, R., Espinosa-Paredes, G., et al. (2012). Simulation of the lower head boiling water reactor vessel in a severe accident. *Science and Technology of Nuclear Installations*, 2012.

Park, R.-J., Ha, K.-S., Kim, H.-Y., et al. (2013). Detailed evaluation of safety injection tank effects on in-vessel severe accident progression in a small break LOCA without safety injection. *Annals of Nuclear Energy*, 58:54–59.

Petruzzi, A. and D'Auria, F. (2008). Thermal-hydraulic system codes in nulcear reactor safety and qualification procedures. *Science and Technology of Nuclear Installations*, 2008.

Pilch, M., Allen, M., Bergeron, K., et al. (1995). The probability of containment failure by direct containment heating in surry. Technical report, Nuclear Regulatory Commission, Washington, DC (United States). Div. of .

RELAP5 (1995). Relap5/m0d3 code manual, the RELAP5 development team. Technical report, RELAP5/M0D3 Code Manual. NUREG/CR- 5535, INEL-95j0174.

Salehi, M. and Jahanfarnia, G. (2016). Small break LOCA analysis without emergency core cooling systems using the RELAP5/SCDAP code in VVER-1000 reactor. *Annals of Nuclear Energy*, 87:299–307.

Salehi, M. and Shayesteh, M. (2017). The accumulator effects on in-vessel severe accident progression of a three loop pwr nuclear power plant in a SBLOCA without safety injection system. *Annals of Nuclear Energy*, 107:1–16.

Seghal, B. R. (2012). Nuclear safety in light water reactors: Severe accident phenomenology, ed.