Pressurized Thermal Shock Analysis for OPR1000 Pressure Vessel

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1. Introduction

This study aims to analyze the effect of pressurized thermal shock (PTS) on the reactor pressure vessel (RPV) during an emergency core coolant injection in a loss-of-coolant accident (LOCA) to check on the RPV integrity for a pressurized water reactor (PWR). Thus the study provides a brief understanding of the analysis procedure and techniques using ANSYS, such as the acceptance criteria, selection and categorization of events, thermal analysis, structural analysis including fracture mechanics assessment, crack propagation and evaluation of material properties. PTS may result from instrumentation and control malfunction, inadvertent steam dump, and postulated accidents such as small-break (SB) LOCA, large-break (LB) LOCA, main steam line break (MSLB), feedwater line breaks and steam generator overfill. In this study our main focus is to consider only the LB LOCA due to a cold leg break of the Optimized Power Reactor 1000 MWe (OPR1000). Consideration is given as well to the emergency core cooling system (ECCS) specific sequence with the operating parameters like pressure, temperature and time sequences. The static structural and thermal analysis to investigate the effects of PTS on RPV is the main motivation of this study. Specific surface crack effects and its propagation is also considered to measure the integrity of the RPV.

A PTS event scenario due to the LB LOCA by a cold leg breaking of the PWR type OPR1000 is shown in Fig. 1.

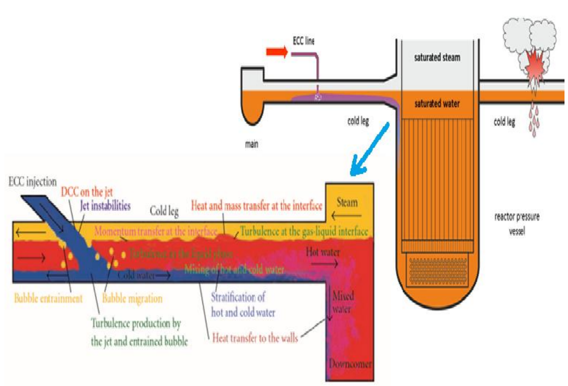


Fig. 1. Simplified PTS event scenario due to LBLOCA in a cold lag of RPV.

2. Background, Motivation and Focus

RPV is considered as the brain of a NPP (Nuclear Power Plant). As like brain, RPV cannot be replaced during the lifetime of the NPP. The integrity of the RPV has to be maintained throughout the plant life since there are no feasible provisions which would mitigate a catastrophic vessel failure. Adequate approach to the RPV integrity assessment provides a basis for safe operation and for timely implementation of preventive and corrective measures if necessary [1].

Before the late 1970s it was assumed that the most severe thermal shock in a PWR vessel would be required to withstand would occur during a LB LOCA. In this type of overcooling transient, room-temperature emergency core coolant would flood the reactor vessel within a few minutes and rapidly cool the vessel wall. The resulting temperature difference across the vessel wall would cause thermal stresses. However, the addition of pressure stresses to the thermal stresses was not considered, since it was expected that during a LB LOCA the system would remain at low pressure [2].

In 1978, the occurrence of a non-LOCA-type event at the Rancho Seco NPP in California revealed that during some types of overcooling transients the rapid cool down could become pained by depressurization of the primary system, which would composite the effects of the thermal stresses [2]. As long as the fracture resistance of the reactor vessel remains relatively high, such transients are not expected to cause the reactor vessel to fail. However, after the fracture toughness of the vessel is gradually reduced by neutron irradiation, severe PTS might cause a small flaw already existing near the inner surface of the wall to propagate through the wall. Depending on the progression of the accident, such a through-the-wall crack (TWC) could lead to core melting. Following the Rancho Seco incident, the US Nuclear Regulatory Commission (USNRC) designated PTS as an unresolved safety issue (A-49) [3].

In this study due to the limitation of high speed CPU, main focus is given on the static analysis of PTS though the transient analysis of PTS is more significant to get better understanding. A simple RPV geometry with uniform thickness is used but in reality there are many internal components exists inside the vessel and the effects of those components during PTS is not insignificant. The gradient of temperature flux is not maintained properly as the fluid flow path inside the RPV is not considered exactly.

3. RPV Integrity Study

PTS analysis, which is a part of RPV structural integrity assessment, is associated with large thermal down shocks of the RPV after a certain time. Here, the material, the design rules, the transient loads are similar (not identical) to PWR. In Fig. 2 RPV of a typical PWR is represented with and without internal components.

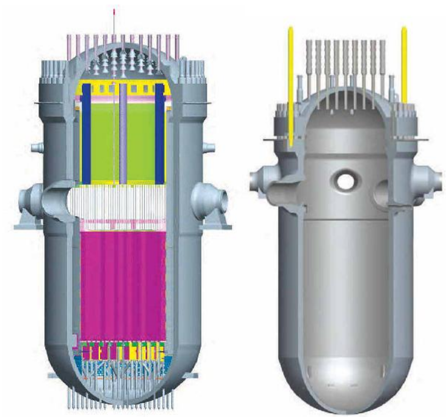


Fig. 2: Reactor Pressure Vessel (RPV) of PWR

The assessment methodologies have been developed around different codes [4]:

a) Flaw evaluation procedures: ASME Code section XI, RSE-M Code, KTA Code;
b) Specific PTS rules: Russian utility procedure MRKR-SKhR-2004, VERLIFE Unified Procedure or international guidelines (IAEA guidelines for WWER PTS analysis, US NRC PTS screening criteria).

Detailed analysis needs are connected to the fact that for some PTS events, the final temperature can be lower than the irradiated materials' ductile to brittle transition temperature. During the life of a RPV, the following analyses are made and periodically updated [5]:

a) Flaw evaluation procedures: ASME Code section XI, Appendix A , RSE-M, KTA Code;
b) Design analysis with a codified evaluation of a postulated hypothetical deep crack, for all type of design transients;
c) Pressure-temperature (P-T) curve evaluation to define the maximum allowable pressure for different rates of temperature variation with respect to the current coolant temperature;
d) Flaw evaluation for any indications discovered during in-service inspection;
e) PTS screening evaluation or generic detailed analysis;

This study focus is on the core shell. Though in original RPV, which are all manufactured using circumferential welding of forged rings and rolled plate with longitudinal welds, in this study no welding joint are considered.

4. PTS Events and Analysis

After LOCA in a nuclear reactor, emergency cooling water is injected into the main pipes of the primary circuit that is under pressure by ECCS. Pipes connected to the RPV with the steam generator are called hot legs, where the heated water flows away from the reactor; and pipes in between main coolant pump and RPV are called cold legs, where the colder water flows towards the reactor.

A safety issue is thermo-mechanical stresses introduced to the RPV wall by sudden contact with the cold liquid represented in Fig. 1. After reactor shut-down, the RPV and the connecting pipes with saturated steam and water inside still have temperatures well above 295°C while the emergency core cooling (ECC) water may be below 50°C. If this ECC water was flowing badly mixed into the RPV down comer, the vessel wall would be exposed to severe thermo-mechanical loads. The sudden cooling of the wall under pressure (also referred to as PTS) could potentially lead to cracking. The risk of an RPV failure both depends on the actual structural mechanical properties of the wall material as well as on the thermal hydraulic phenomena governing the fluid mixing in the main coolant pipe. The main initiating PTS events identified in the literature are Small Break(SB) LOCA, Large Break(LB) LOCA, Main Steam Line Break (MSLB), Loss of Main Feed Water (LOFW), Steam Generator Tube Rupture (SGTR), and Loss Of Heat Sink (LOHS). In this study only LB LOCA for PTS event is considered.

3.1 LB LOCA Sequences for PTS

The LB LOCA and ECCS system operation consist of five main phases: Blow-down, Bypass, Refill, Re-flood and Long-term-cooling that are represented in Fig. 3 and Table 1. The ECCS component specification of OPR1000 is also presented in Table 2.

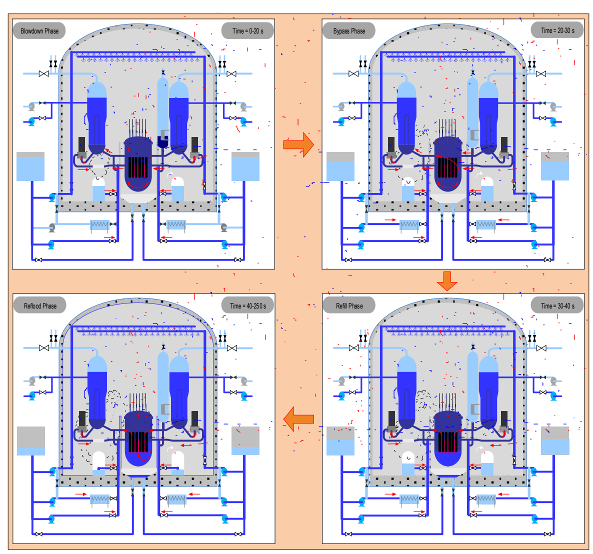


Fig. 3. LBLOCA event scenario in PWR.

Table 1. : LBLOCA event sequences and parameters

Steps/Sequences	Time (Sec)	Operating Pressure & Temp.
Initial Condition	0	175 Bar, 327 °C
Accident (Before ECCS)	0 ~ 240s	20 Bar, 171 °C
Blow Down Phase	240 ~ 260s	41.37 Bar, 148.8 °C
By pass phase	260 ~ 270s	27.58 Bar, 121 °C
Refill Phase	270 ~ 280s	24.132 Bar, 93 °C
Reflood phase	280 ~ 490s	20.684 Bar, 65 °C

Table 2. : ECCS system specification for LBLOCA event in OPR1000

ECCS System Specification			
Steps/Sequences	SIT	LIPSI	HIPSI
Quantity	4	2	2
Design Pr(psig)	700	750	2050
Design Temp	200(°F)	400(°F)	350(°F)
Thermal shock withstand	--	40 ~ 300°F within 10s	40 ~ 300 °F within 10s
Flow Rate(gpm)	13898	4200	815

3.2 PTS Event Sequences for ANSYS Analysis

The PTS event sequences for large break loss-of-coolant accident (LBLOCA) due to one cold lag line break on RPV having five main phases: Blow-down, Bypass, Refill, Re-flood and Long-term-cooling used in this study are represented in Fig. 4 and Table 3.

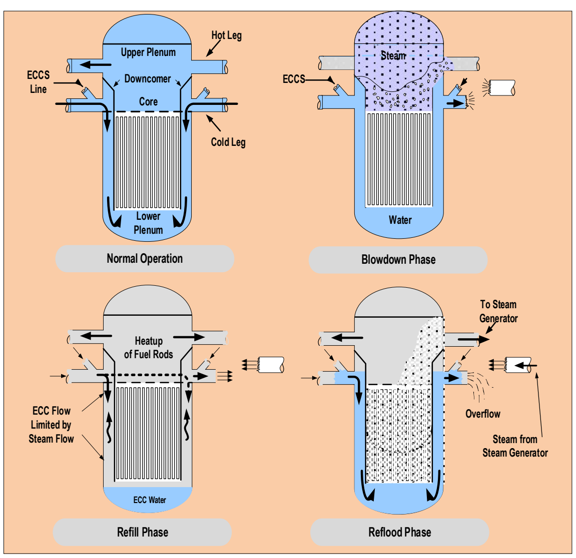


Fig. 4: LBLOCA event scenario in RPV.

The time duration of each stage is reduced (presented in Table 3) from the original time duration showed in Table 1 for reducing the computation time in ANSYS.

Table 3. : Simplified LBLOCA event sequences and operating parameters for ANSYS analysis

Steps/Sequences	Time (Sec)	Operating Pr. & Temp.
Initial Condition	0	175 Bar, 327 °C
Accident (Before ECCS)	0 ~ 20s	20 Bar, 171 °C
Blow Down Phase	20 ~ 40s	41.37Bar, 148.8 °C
By pass phase	40 ~ 50s	27.58Bar, 121 °C
Refill Phase	50 ~ 60s	24.132Bar, 93 °C
Re-flood phase	60 ~ 70s	20.684Bar, 65 °C

3.3 RPV and Internal Crack Geometry

The original and simplified PWR reactor pressure vessel geometry are shown in Fig. 5 and Fig. 6 respectively. To make the RPV geometry simple we consider the vessel wall thickness is uniform 25cm that is represented.

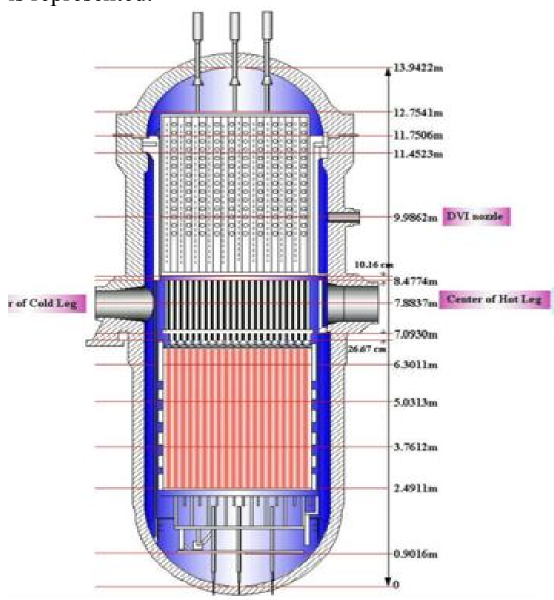


Fig. 5: RPV Original Geometry.

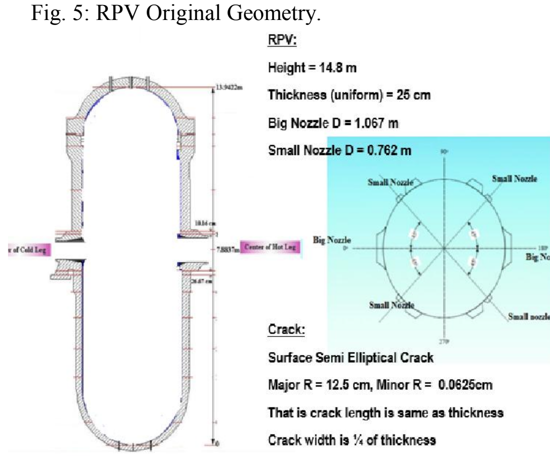


Fig. 6: RPV Simplified Geometry for ANSYS Analysis.

3.4 ANSYS Project Development and Configuration

Stress analysis of RPV due to PTS can be performed by means of a FE structural mechanics code (such as ANSYS) applied to a complete model ANSYS grid, where both pressure and thermal loads are considered. For this purpose it is needed to transfer main CFD results (in particular: temperature profile) to the ANSYS model, in order to evaluate the thermal stresses inside the RPV. Once the stress profile in the RPV wall is known, FM analysis is then performed, for the calculation of the stress intensity factor at crack tip (KI) to be compared with the material fracture toughness curve that is, obtained from specific tests, to establish whether crack propagation is stable or unstable.

Here RPV PTS analysis using ANSYS followed by mainly two steps: Steady-state Thermal analysis and Static Structural analysis. In the thermal analysis the temperature and pressure in each state is needed to configure properly. Using the thermal analysis the static structural analysis is accomplished with detail solution configuration.

3.5 RPV & Internal Crack Geometry Development

ANSYS geometry is developed by using the simplified RPV geometry values (Fig. 6). Here RPV of OPR1000 is considered as the references. This RPV has two big size outlet nozzle and four small size inlet nozzles. Here, break is considered at a cold lag pipe (small size) nozzle of the RPV. The developed geometry is shown in Fig. 7. Here, height is 14.8m, thickness (uniform) is 25cm, big nozzle dia. is 1.067 m and small nozzle dia. is 0.762m.

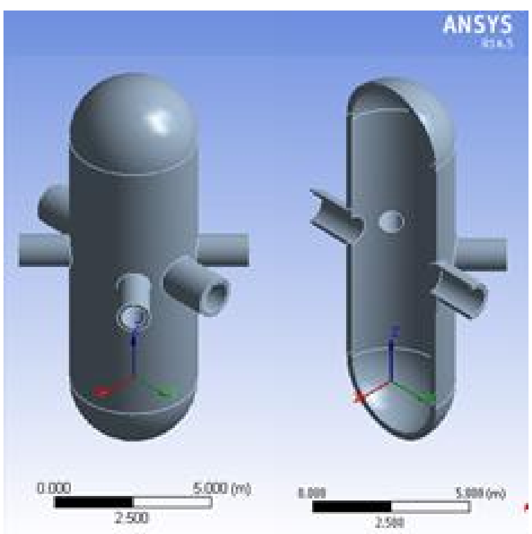


Fig. 7: RPV geometry developed in ANSYS

A surface semi elliptical surface crack is considered inside the RPV in the ANSYS geometry which location and shape is represented in Fig. 8. The creak length is almost equivalent to the thickness of RPV and width is one-forth (¼) of the RPV thickness.

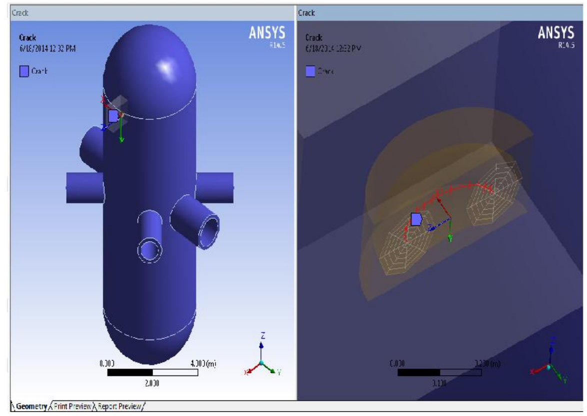


Fig. 8: RPV geometry with crack developed in ANSYS

3.6 RPV Material Specification

Though the original RPV have internal stainless steel (SS) cladding, in this study the cladding is not considered to make the geometry simple. Material chosen for RPV is general structural steel (low grade steel). Material total volume is 43.907 m^3, Mass is 344670kg.

3.7 RPV Geometry Meshing

The RPV geometry mesh is shown in Fig. 9. As the geometry is very large size, the minimum mesh size (4cm) that support by the available CPU is considered for this study. The surface crack mesh size is also set to the minimum value with considering CPU limitation.

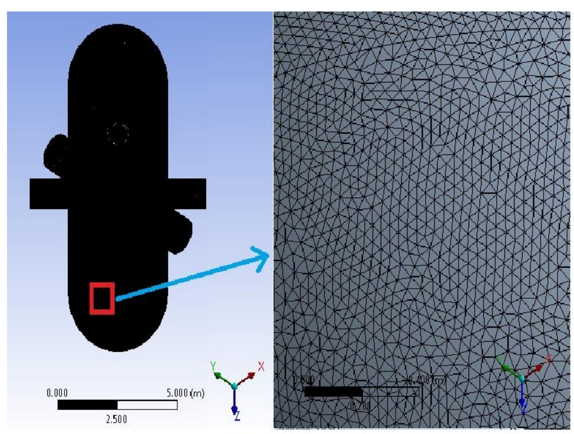


Fig. 9: RPV geometry mesh in ANSYS

3.8 RPV Geometry Structural Support

The RPV geometry structural support is shown in Fig. 10 and earth gravity is considered through the vertical axis. The structural support is considered at bottom head cover uniformly for simplicity of analysis.

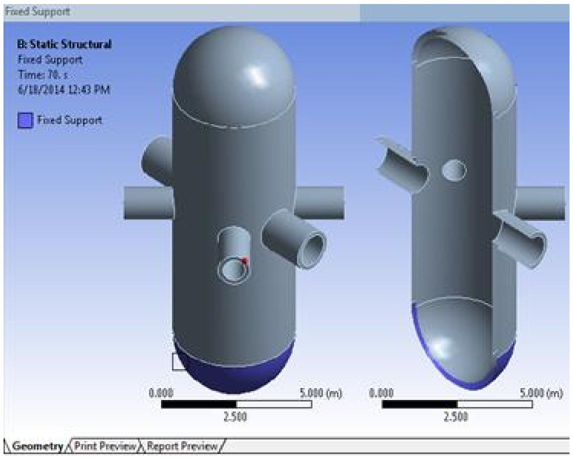
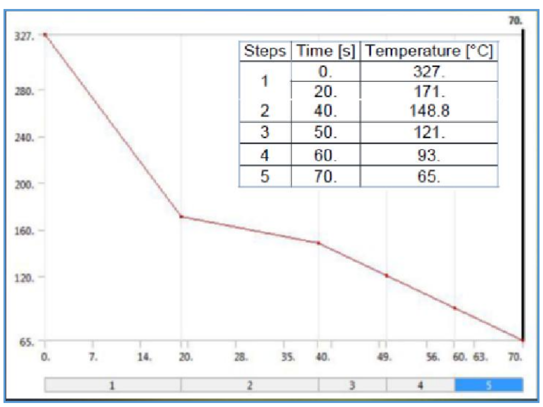


Fig. 10: RPV geometry static structural support in ANSYS

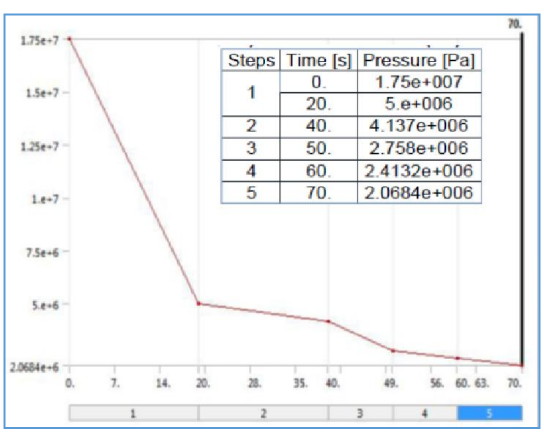
3.9 RPV PTS ANSYS: Temperature & Pressure Profile

The temperature and pressure profile for RPV PTS ANSYS analysis is shown in Fig. 11 and 12. Here five steps with linear slope are considered. Before the PTS event the temperature and pressure are considered as same during the operating condition of the nuclear reactor. OPR1000 operating pressure in RPV and temperature at hot leg coolant is about 175 Bar, 327 ^{o}C.



Steps	Time [s]	Temperature [°C]
1	0.	327.
	20.	171.
2	40.	148.8
3	50.	121.
4	60.	93.
5	70.	65.

Fig. 11: Temperature profile for RPV PTS analysis



Steps	Time [s]	Pressure [Pa]
1	0.	1.75e+007
	20.	5.e+006
2	40.	4.137e+006
3	50.	2.758e+006
4	60.	2.4132e+006
5	70.	2.0684e+006

Fig. 12: Pressure profile RPV PTS analysis

5. ANSYS Solution and Results

ANSYS solution and results for PTS of RPV is represented in two group Steady-state Thermal analysis Static Structural analysis in Fig. 13~15 and Fig. 16~25 respectively. The grid sensitivity result shows that percentile error of this ANSYS CFX result is good from moderate grid points. Considering the calculated result and running time, minimum mesh face size is reasonable choice.

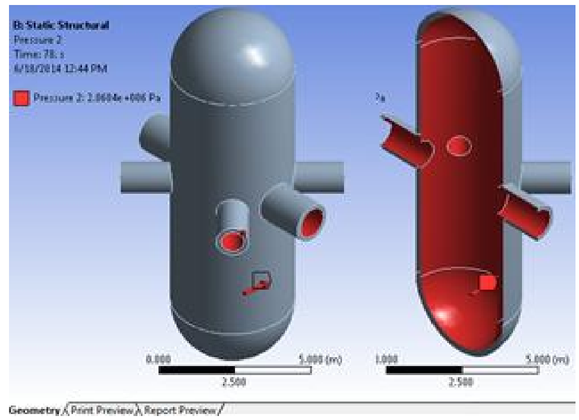


Fig. 13: Steady-state Thermal analysis: Final Pressure

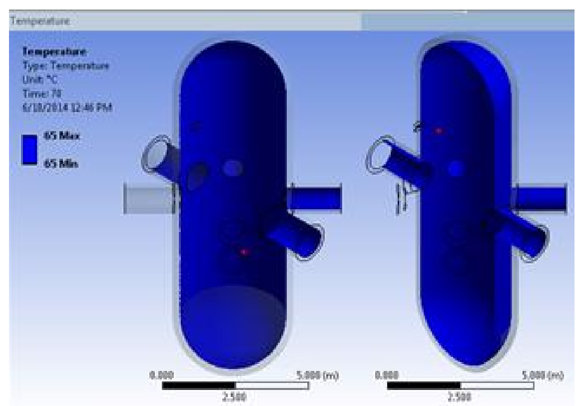


Fig. 14: Steady-state Thermal analysis: Final Temperature

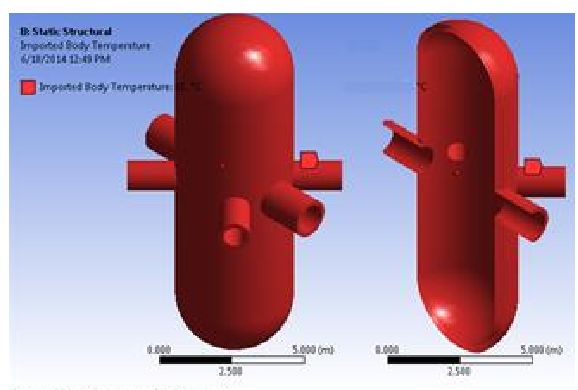


Fig. 15: Steady-state Thermal analysis Body for Static Structural analysis

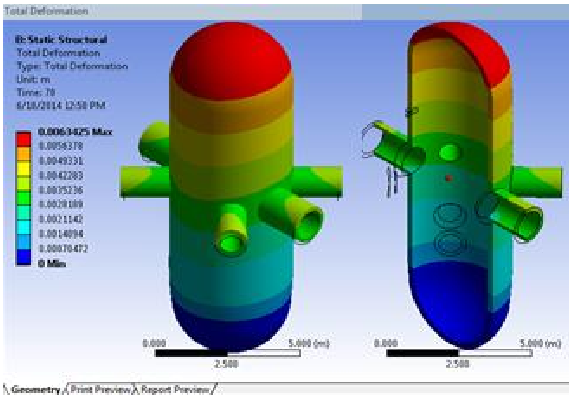


Fig. 16: Total deformation

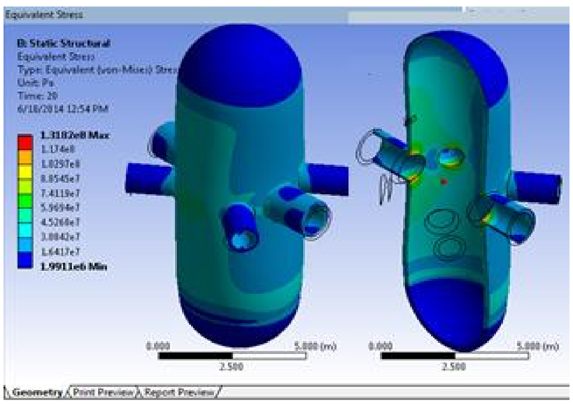


Fig. 17: Von-Mises Stress

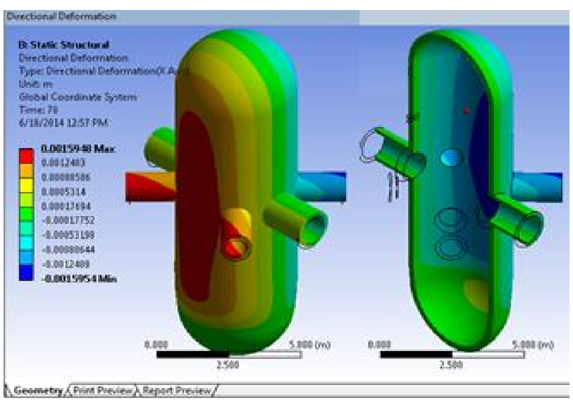


Fig. 18: Directional deformation(x axis)

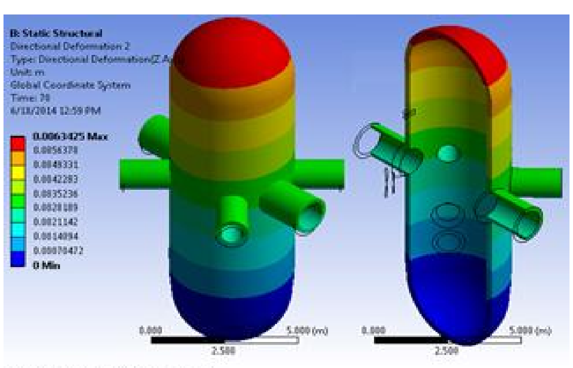


Fig. 19: Directional deformation (z axis)

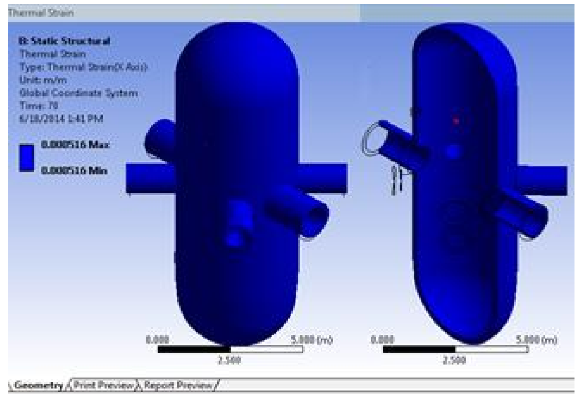


Fig. 20: Thermal strain

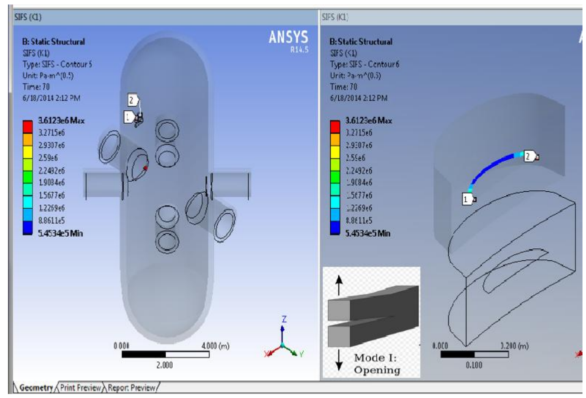


Fig. 21: SIF(Stress Intensity Factor), K_1

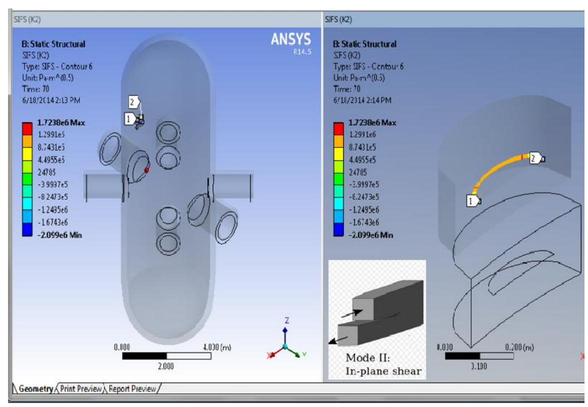


Fig. 22: SIF(Stress Intensity Factor), K_2

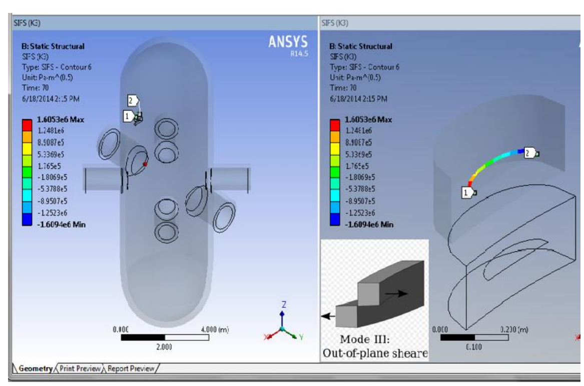


Fig. 23: SIF(Stress Intensity Factor), K_3

6. Conclusions and Recommendations

This study describes the procedure for pressurized thermal shock analysis due to a loss of coolant accidental condition and emergency core cooling system operation for reactor pressure vessel. Different accidental events that cause pressurized thermal shock to nuclear RPV that can also be analyzed in the same way. Considering the limitations of low speed computer only the static analysis is conducted. The modified LBLOCA phases and simplified geometry can is utilized to analyze the effect of PTS on RPV for general understanding not for specific specialized purpose. However, by integrating the disciplines of thermal & structural analysis, and fracture mechanics analysis a clearer understanding of the total aspect of the PTS problem has resulted. By adopting the CFD, thermal hydraulics, uncertainties and risk analysis for different type of accidental conditions, events and sequences with proper mathematical models and boundary conditions, the PTS analysis can be improved further.

ACKNOWLEDGEMENT

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