STRUCTURAL INTEGRITY ANALYSIS OF VVER-1200 REACTOR VESSEL WALL DUE TO PRESSURIZED THERMAL SHOCK

MOHAMMED SARIM SALMAN KARIM

M.Sc. ENGINEERING THESIS

DEPARTMENT OF NUCLEAR SCIENCE & ENGINEERING
MILITARY INSTITUTE OF SCIENCE AND TECHNOLOGY
DHAKA, BANGLADESH

MARCH 2024

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A Thesis Submitted in Partial Fulfillment of the Requirements for the Degree of Bachelor of Science in Nuclear Science & Engineering

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DECLARATION

I hereby declare that the study reported in this thesis entitled as above is my own original work and has not been submitted before anywhere for any degree or other purposes. Further, I certify that the intellectual content of this thesis is the product of my own work and that all the assistance received in preparing this thesis and sources have been acknowledged and/or cited in the reference Section.

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ABSTRACT

STRUCTURAL INTEGRITY ANALYSIS OF VVER-1200 REACTOR VESSEL WALL DUE TO PRESSURIZED THERMAL SHOCK

Reactor Pressure Vessel (RPV) is the most crucial and irreplaceable part of any Nuclear Power Plant (NPP). So, maintaining the integrity and safety of RPV during any kind of transient or accidental conditions when it can be detrimental e.g., Pressurized Thermal Shock (PTS), need to address as highest priority. Analyzing the PTS event is of utmost importance to ensure the safety and integrity of the VVER-1200, a Gen-III+ reactor model of the Russian Federation, during hypothetical transient scenarios. To analyze the PTS event, Beaver Valley-105 case which is Main Steam Line Break (MSLB) type Loss of Coolant Accident (LOCA), is taken into consideration for this study. A computational fluid dynamics (CFD) analyses was done by ANSYS Fluent considering the injection of cold-water from emergency core cooling system (ECCS) to the hot primary coolant system through cold leg. The beltline region is considered for the modeling to analyze the temperature evolution in high pressure at both coolant and inner wall surface during the transient event and found 167◦C temperature coolant at 200 sec simulations after ECCS water mixing. Sudden increase of pressure in primary system which can lead the scenario to PTS at low temperature is analyzed by ANSYS Mechanical to evaluate the stress profile. The stress analysis for the transient event, which caused PTS due to an instantaneous elevation to a primary system pressure of 16.2 MPa, has been done. It is observed that maximum stress, e.g. Equivalent (Von mises) stress was found 663 MPa and was generated at the edge of nozzle area and the stress values exceed the ultimate yield strength at 16.2 MPa pressure. Furthermore, a comprehensive postulated crack modelling has been done considering the temperature and pressure of the selected transient case. Three different semi-elliptic crack model are analyzed at selected transient event and for elevated primary pressure of 18, 20 and 22 MPa respectively. Finally, the results of stress intensity factor (SIF) at 100 ◦C obtained for these three postulated crack cases have found, were analyzed with respect to fracture toughness value, K_{IC} of initial state and end of service life state to assess the service life. Here, fracture toughness, K_{IC} is calculated as per ASME and IAEA TECDOC considering the neutron fluence and Nil Ductility temperature values. SIF results (87-118 Mpa.m^1/2) were found below the limiting fracture toughness values, K_{IC} at case II for every pressure conditions. For case I & III, SIF values exceed the K_{IC} values limit at 22 MPa and 20 & 22 MPa respectively. A further study was done to determine the maximum allowable critical temperature for crack for these three cases.

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STRUCTURAL INTEGRITY ANALYSIS OF VVER-1200 REACTOR VESSEL WALL DUE TO PRESSURIZED THERMAL SHOCK

নিউক্লিয়ার পাওয়ার প্ল্যান্টের (NPP) সবচেয়ে গুরুত্বপূর্ণ এবং অপরিবর্তনীয় অংশ হল রিঅ্যাক্টর প্রেসার ভেসেল (RPV) । সুতরাং, যেকোনো ধরনের ক্ষতিকারক ক্ষণস্থায়ী বা দুর্ঘটনাজনিত পরিস্থিতিতে যেমন, প্রেসারাইজড থার্মাল শক (PTS) এবং অন্যান্য, আরপিভির (RPV) এর সামগ্রিকতা এবং নিরাপত্তা বজায় রাখা সর্বোচ্চ অগ্রাধিকারে স্থান পায়।, রাশিয়ান ফেডারেশনের নকশাকৃত জেনারেশন-৩+ (Gen-III+) চুল্লি ভিভিইআর-১২০০ (VVER-1200) মডেল এর নিরাপত্তা এবং সামগ্রিকতা নিশ্চিত করার জন্য কাল্পনিক ক্ষণস্থায়ী পরিস্থিতিতে PTS ইভেন্ট বিশ্লেষণ করা অত্যন্ত গুরুত্বপূর্ণ। PTS ইভেন্ট বিশ্লেষণ করতে, এই গবেষণার জন্য Beaver Valley-105 কেস বিবেচনা করা হয়েছে যা হচ্ছে মূল বাষ্পীয় পাইপের ভাঙ্গনের (Main Steam Line Break) (MSLB) ফলে শীতলীকরণ তরলের হ্রাস পাওয়া জাতীয় দুর্ঘটনা (Loss of Coolant Accident) (LOCA)। এনসিস ফ্লুয়েন্ট দ্বারা একটি গণনীয় প্রবাহী গতিবিদ্যা-এর (CFD) বিশ্লেষণ করা হয়েছে যেখানে জরুরি কোর শীতলীকরণ ব্যবস্থা (ECCS) থেকে শীতল বাহুর মাধ্যমে উষ্ণ প্রাথমিক শীতলীকরণ ব্যবস্থায় শীতল তরল প্রবেশ করে। ক্ষণস্থায়ী ঘটনার সময় শীতল তরল এবং ভিতরের প্রাচীরে উচ্চ চাপে তাপমাত্রার বিবর্তন বিশ্লেষণ করার জন্য বেল্টলাইন অঞ্চলটিকে মডেলিংয়ের জন্য বিবেচনা করা হয় এবং ECCS এর শীতল তরল মিশ্রনের ফলে ২০০ সেকেন্ড সিমুলেশন পর শীতলীকরন তরল এর তাপমাত্রা ১৬৭ ◦ C পাওয়া যায়। প্রাথমিক সিস্টেমে নিম্ন তাপমাত্রায় হঠাৎ চাপ বৃদ্ধি যা পরিস্থিতিকে পিটিএস (PTS)-এ নিয়ে যেতে পারে এবং এই ঘটনার ফলে উদ্ভুত স্ট্রেস (stress) প্রোফাইল মূল্যায়ন করতে এনসিস যান্ত্রিক (ANSYS Mechanical) দ্বারা বিশ্লেষণ কর্ম করা হয়। প্রাথমিক সিস্টেমে ১৬.২ মেগা প্যাসকেল তাত্ক্ষণিক চাপ সৃষ্টির কারণে PTS সংঘটিত হওয়ার ক্ষণস্থায়ী পরিস্থিতির জন্য স্ট্রেস (stress) বিশ্লেষণ করা হয়েছে। এটা দেখা যায় যে শীতল বাহুর অগ্রভাগের প্রান্তে সর্বাধিক চাপ তৈরি হয়েছিল এবং স্ট্রেসের মান ১৬.২ মেগা প্যাসকেল চাপে চূড়ান্ত ফলন শক্তিকে অতিক্রম করে উদাহরণস্বরূপ সমতুল্য স্ট্রেসের মান ৬৬৩ মেগা প্যাসকেল তৈরি হয়েছিল। এছাড়া, নির্বাচিত ক্ষণস্থায়ী পরিস্থিতিতে তাপমাত্রা এবং চাপ বিবেচনা করে একটি বিস্তৃত প্রাক বিদ্যমান চিড় (crack) মডেলিং করা হয়েছে। নির্বাচিত ক্ষণস্থায়ী পরিস্থিতিতে যথাক্রমে ১৮, ২০ এবং ২২ মেগা প্যাসকেল -এর উচ্চতর প্রাথমিক চাপের জন্য

তিনটি ভিন্ন আধা-উপবৃত্তীয় চিড় (crack) মডেল বিশ্লেষণ করা হয়। পরিশেষে, রিঅ্যাক্টর প্রেসার ভেসেল-এর জীবনকাল মূল্যায়ন করার জন্য প্রাথমিক অবস্থার এবং পরিষেবা জীবনের শেষ অবস্থার ক্ষেত্রে ১০০ ◦ C তে এই তিনটি প্রাক বিদ্যমান চির কেসের জন্য বিশ্লেষণ করে চাপ প্রবলতা গুণনীয়ক (stress intensity factor) (SIF) এর ফলাফলগুলি ফ্র্যাকচার টাফনেস ভ্যালু, KIC এর সাথে তুলনা করা হয়েছে। এখানে, নিউট্রন ফ্লুয়েন্স এবং শূন্য নমনীয়তা তাপমাত্রার মান বিবেচনা করে আমেরিকান সোসাইটি অফ মেকানিক্যাল ইঞ্জিনিয়ারস' (ASME) এবং আন্তর্জাতিক পরমাণু শক্তি এজেন্সির কারিগরি ডকুমেন্ট (IAEA TECDOC) অনুযায়ী ফাটল-এর শক্ততা (K_{IC}) গণনা করা হয়েছে। দ্বিতীয় চিড় মডেলে প্রতিটি চাপের অবস্থার জন্য চাপ প্রবলতা গুণনীয়ক (SIF) এর ফলাফলগুলি (৮৭-১১৮ মেগা প্যাসকেল √ মিটার) ফাটল-এর শক্ততা (K_{IC}) মানের নীচে পাওয়া গেছে। প্রথম অ্যান্ড তৃতীয় চিড় মডেলে SIF মানগুলি যথাক্রমে ২২ মেগা প্যাসকেল এবং ২০ এবং ২২ মেগা প্যাসকেল -এ ফাটল-এর শক্ততা (K_{IC}) মান সীমা অতিক্রম করে। এই তিনটি ক্ষেত্রে চিড় মডেলে ভঙ্গুরতার সর্বোচ্চ গ্রহণযোগ্য সংকটপূর্ণ তাপমাত্রা নির্ধারণের জন্য আরও একটি কাজ করা হয়েছে।

ACKNOWLEDGEMENTS

All praise and gratitude belong to almighty Allah for His infinite mercy, which have enabled me to successfully complete my journey and come to an end. I am profoundly grateful to His unwavering guidance and blessings.

I would like to express my utmost gratitude to Dr. Debashis Datta, my supervisor, and Lt Col Md Altab Hossain, PhD, my co-supervisor, for their invaluable support, guidance, and the opportunity they provided me throughout my every step in analyzing the structural integrity of VVER-1200 reactor pressure vessel wall due to Pressurized Thermal Shock (PTS). Without their continuous support and assistance throughout the whole time, it would not have been possible to complete the journey. Working alongside them has been an absolute honor and privilege that I will forever cherish.

I would like to extend my deepest appreciation and gratitude the Head of the department of Nuclear Science and Engineering, Col Ashfaquer Rahat Siddique, BGBMS, for his wisdom, inspiration and continuous support that I have received throughout the entirety of this endeavor. I would like to thank NSE department for their invaluable cooperation and guidance throughout this remarkable journey.

Lastly, I would like to thank my parents and beloved wife for their unwavering support, encouragement, and constant presence in my life has been nothing short of remarkable. Their boundless understanding and endless patience have not only shaped my journey but have also left an indelible mark on my heart.

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k_{eff}	Effective multiplication factor
E	Young's modulus
α	Coefficient of linear thermal expansion
υ	Poisson's ratio
λ	Thermal Conductivity
C_p	Heat Capacity
ρ	Density
$\bar{u}_i$	Mean velocity
u_i'	Fluctuating velocity
φ	Scalar concentration
k	Turbulence kinetic energy
ε	Dissipation rate
G_k	Generation of turbulence kinetic energy resulting from mean velocity gradients
G_b	Generation of turbulence kinetic energy caused by buoyancy
Y_M	Contribution of fluctuating dilatation in compressible turbulence to the overall dissipation rate
σ_{eq}	Equivalent (Von Mises) Stress
σ_1	Maximum principal stress
σ_2	Middle principal stress
σ_3	Minimum principal stress
K_I	Stress intensity factors

a/c	Aspect ratio
a	Minor radius/ crack depth
c	Major radius
b_0, b_1, b_2, b_3, b_4	Coefficients for the polynomial approximation of the stresses
i_0, i_1, i_2, i_3, i_4	Influence coefficients
Q	Crack shape correction factor
t	Wall thickness
J	J- integrral
RT_{NDT}	Reference nil ductility temperature
T	Temperature
Bi	Crack front length
ΔRT_{NDT}	Nil ductility temperature shift
f	Neutron fluence
CF	Chemistry factor

CHAPTER 1
INTRODUCTION

1.1 Background of The Study

Structural integrity maintaining of reactor pressure vessel (RPV) is an obligatory part for ensuring the safety of nuclear reactor operation. Several factors such as material properties, operating pressure and temperature, corrosion and material degradation, weld integrity, fatigue, design etc. are responsible to achieve the integrity of nuclear reactor vessel. These factors also vary from different reactor design and model. During the assessment of the integrity of RPV, pressurized thermal shock (PTS) that can occur during emergency water cooling requires careful consideration. One of the most critical challenges that could potentially impact the RPV's structural integrity is referred as PTS (Ruan and Morishita, 2021). Over time and with recent advancements, the VVER-1200 reactor in the AES-2006 nuclear power plant design has evolved into an evolutionary Russian design of Generation III+ reactors. According to the brochure of VVER-1200 of ROSATOM, Service life of VVER-1200 is 60 years (ROSATOM, n.d). And structural integrity must have to maintain throughout the service life to ensure the safety of the plant, operating personnel and public; prevention of accidents; long term reliability and financial safety.

Active and passive design safety features of VVER-1200 including two-channel structure of active safety systems with redundant emergency pumps in each channel, a double containment, a core melt localization device, a passive heat removal system, etc. provides more reliability of the nuclear reactor to withstand any abnormal condition (Asmolov et al., 2017). However, the analysis of pressurized thermal shock (PTS) is a crucial matter that needs to be addressed for any reactor to ensure structural integrity.

When a system with a higher temperature suddenly undergoes rapid cooling under high pressure, followed by re-pressurization, this event can potentially result in a thermal shock due to a temperature gradient. Furthermore, the increase in stress caused by the re-pressurization event can be referred to as pressurized thermal shock (PTS) (Uitslag-Doolaard et al., 2020). For a VVER-1200 nuclear reactor, the operational temperature of the primary coolant is 298°C

(IAEA, Status report 108, 2011). Under transient or accidental conditions, relatively cold emergency core cooling system (ECCS) water may be injected into the reactor coolant system (RCS), which can increase the possibility of rapid cooling at a higher system pressure. If re-pressurization occurs during the event with a lower RCS temperature, it could worsen the situation and potentially lead to brittle fracture of the reactor pressure vessel (RPV) material. Due to the wide range of aspects that must be considered, a comprehensive analysis of PTS requires a multidisciplinary approach. This approach involves various disciplines such as probabilistic risk analysis (PRA), materials research, thermal-hydraulics, structural mechanics, fracture mechanics, and materials testing and inspection (Boyd, 2008).

According to the studies of US NRC, the conclusions are made that primary side breaks have a greater impact on through-wall crack frequency (TWCF) than secondary side breaks. Another complicated incident that creates challenges is when a primary side valve opens up and then closes again. The overall significance of these events is determined by combining the likelihood (occurrence frequency) and severity of this transient event. The findings are briefly summarized in Fig. 1.1 (EricksonKirk et al., 2005).

Transient Class		Transient Severity			Transient Likelihood	TWCF Contribution
		Cooling Rate	Minimum Temperature	Pressure		
Primary Side Pipe Breaks	Large-Diameter	Fast	Low	Low	Low	Large
	Medium-Diameter	Moderate	Low	Low	Moderate	Large
	Small-Diameter	Slow	High	Moderate	High	~0
Stuck-Open Valves, Primary Side	Valve Recloses	Slow	Moderate	High	High	Large
	Valve Remains Open	Slow	Moderate	Low	High	~0
Main Steam Line Break		Fast	Moderate	High	High	Small
Stuck-Open Valve(s), Secondary Side		Moderate	High	High	High	~0
Feed-and- Bleed		Slow	Low	Low	Low	~0
Steam Generator Tube Rupture		Slow	High	Moderate	Low	~0
Mixed Primary & Secondary Initiators		Slow	Mixed		Very Low	~0
Color Key	Enhances TWCF Contribution		Intermediate		Diminishes TWCF Contribution	

Fig. 1.1: Summary of Factors Leading to TWCF (EricksonKirk et al., 2005)

Integrity analyses for PTS is a cross-disciplinary activity such as thermal-hydraulic (TH) analyses, computational fluid dynamics (CFD) analyses, structural analyses, fracture mechanics analyses, risk assessment that calls for a wide range of abilities. The evaluation of PTS can be greatly impacted by a number of plant-specific parameters, including the distinctive properties of vessel walls and embrittlement, operator procedures, control system set points, safety injection flow rates, injection geometry, and other system specifics. TH system codes e.g., TRACE, RELAP5, ATHLET, or CATHARE are used to predict the overall system behavior during the postulated events. For multi-dimensional effects analyzing in the downcomer region, CFD analyses is done by different simulation software e.g., OpenFOAM, STAR-CCM+, ANSYS etc. for more accuracy to predict the temperature and heat transfer in the RPV wall during thermal and mechanical loading. Lastly, computation of thermal stresses and performing probabilistic fracture mechanics analyses. Since the 1980s, computer codes have been developed to facilitate the probabilistic analysis of Reactor Pressure Vessels (RPVs). The most frequently used of these codes are PASCAL, created by Japan, and FAVOR, created by Oak Ridge National Laboratory (Thamaraiselvi and Vishnuvardhan, 2020).

To analyze the PTS event specific to the VVER-1200 reactor, a transient event based on a Loss-of-Coolant Accident (LOCA) is considered, simulated through Computational Fluid Dynamics (CFD) and thermo-mechanical analysis. Main Steam Line Break (MSLB) LOCA can have less potential for TWCF but transient likelihood of this event and transient severity factors which may enhance TWCF contributions are relatively higher. Hence, this MSLB LOCA event in PTS event need to analyze for both design-based scenario and hypothetical scenario.

During the transient, ECCS injection is highly 3D complex mixing phenomena which requires 3D analyses. Thus, the conventional one-dimensional system thermal-hydraulic codes, which have been used for safety analyses for the past three decades, are insufficient to accurately capture the complex three-dimensional flow characteristics, particularly the anisotropic flow patterns within the downcomer, and the subsequent thermal shock within the Reactor Pressure Vessel (RPV). PTS has the potential to initiate crack propagation within a reactor vessel, particularly in scenarios where the material's fracture toughness has been diminished due to prolonged exposure to neutron irradiation during operation. So, evaluating temperature and stress profile of vessel inner part during the transient event is very important. Moreover, during the end of life of the reactor, the fractur toughness can be reduced for the embrittlement and aging effect. So, such crack propagation needs to assess by determining the SIF for postulated cracks for the selected transients and furthermore hypothetical over-pressurized conditions.

Maximum allowable critical temperature of brittleness margin for such hypothetical scenarios need to address to avoid any unwanted situation.

1.2 Problem Statement

The occurrence of a PTS event during a LOCA can be highly detrimental to the integrity of the RPV wall, especially if pre-existing cracks exist on its surface. In this study, we are considering a Main Steam Line Break (MSLB) type of LOCA due to its rapid cooling rate, high pressure, and high likelihood of transient conditions, making it a critical PTS scenario. In this hypothetical scenario, the primary system pressure rises beyond allowable limits, both during normal operation and the LOCA transient, leading to the initiation of a PTS scenario at a low temperature of the primary coolant. This situation creates a high-stress environment within the inner RPV wall. The potential for the propagation of surface cracks on the wall during this hypothetical scenario poses a significant threat to the integrity and service life of the irreplaceable RPV, even up to its design life. Neutron fluence and the nil ductility temperature have been taken into account for VVER-1200 reactor in the analysis.

1.3 Significance of the Study

The structural integrity analysis of the VVER-1200 RPV in response to a PTS event is of highest importance for assessing the RPV's service life. This analysis not only elucidates the mixing of Emergency Core Cooling System water and primary hot coolant at the beltline region of the RPV but also examines the temperature evolution during a LOCA event. In a hypothetical scenario involving high primary pressure followed by a LOCA transient at low temperatures, conditions that are not permissible under normal or transient operation, this analysis provides insights into the stress distribution within the inner RPV wall. Additionally, it conducts a service life assessment, taking into account a postulated crack profile on the RPV wall's surface, which allows for predictions regarding integrity throughout the design life of the vessel. Furthermore, this study resolves the uncertainty surrounding the maximum allowable critical temperature for brittleness observed during the hypothetical high-pressure PTS event. The outcomes of this research hold significant concern for both reactor operators and regulatory bodies in the context of PTS event analysis. From a regulatory safety perspective concerning RPV integrity, this analysis represents a major step in the long-term operating experience assessment of the VVER-1200 reactor.

1.4 Scope of the Study

This study enables us to obtain results and assessments regarding the structural integrity of the VVER-1200 reactor's RPV wall during PTS events. PTS events encompassing various initiating events and plant-specific conditions, providing diverse insights into reactor behavior. In particular, a hypothetical scenario is analyzed involving high primary pressure during transient events, such as LOCA, to enhance the integrity and robustness of the RPV during accidental conditions. Additionally, within this scenario, the analysis of pre-existing cracks on the RPV wall's surface assists in assessing the vessel's service life, extending beyond the design life if necessary. Moreover, this study allows us to determine the maximum allowable critical temperature for brittleness, which has practical applications in the design and improvement of the RPV in accidental conditions.

1.5 Objectives

The objectives of the research work are stated below:

i. To analyze the 3D temperature distribution of downcomer coolant and Reactor Pressure Vessel (RPV) wall for a loss of coolant accident (LOCA) induced single-phase pressurized thermal shock (PTS) scenario for a generic VVER 1200 reactor by ANSYS FLUENT.
ii. To assess the stress profile of the RPV wall for the PTS scenario by ANSYS Mechanical.
iii. To investigate the crack propagation on the RPV wall to evaluate the service life during single-phase PTS scenario.

1.6 Organization of the Thesis

Chapter 1:

This chapter provides essential information on the VVER 1200 reactor, concentrating on its significant components including the reactor pressure vessel (RPV), and demonstrates the importance of ensuring structural integrity to make sure the reactor's safe operation for an extended period of time.

Chapter 2:

This chapter provides a comprehensive review of the existing literature on thermo-mechanical studies and the determination of the ductile-brittle transition temperature (DBTT) in accidental conditions using various codes and methodologies used in various countries. The objectives of this work are outlined as well, depending on this literature review.

Chapter 3:

This chapter presents an outline of the research work conducted in this study. The theory of governing equations utilized in ANSYS FLUENT for computational fluid dynamics (CFD) calculations, as well as the theory of stress analyses employed in ANSYS Mechanical, are discussed in this section. Besides, the theory of stress intensity factor (SIF) used for evaluating for postulated crack profile and DBTT evaluation are described to analyze fracture behavior and assessing structural integrity under different operation conditions.

Chapter 4:

Finally, the 3D fluid mixing temperature and adjacent RPV wall temperature during the LOCA transient is evaluated by ANSYS FLUENT. And, the stress profiles are assessed by ANSYS Mechanical for different pressurized conditions. Thereafter, postulated cracks in the crucial zones are analyzed and SIF determined. In addition, maximum allowable critical temperature of brittleness for different crack profiles are evaluated.

Chapter 5:

The research's outcomes are compiled in this chapter, and recommendations are made for future improvements.

CHAPTER 2
LITERATURE REVIEW

2.1 Synopsis

To obtain a better observation on PTS event occur at RPV, different field of analyses e.g., Thermal hydraulics, CFD, stress analyses, fracture analyses are coupled with necessary accuracy. Structural integrity of RPV is subject to ageing effects, embrittlement and extreme environment. For safe operation and achieve integrity and higher safety during accident condition, safety must be ensured for brittle failure in appropriate manner. Assessment methodologies which are being used for current operating reactors around the world are found ASME Code section XI, Appendix A (ASME, 2004), RSE-M Code (ASSOCIATION FRANÇAISE POUR, 1997), KTA Code (KERNTECHNISCHER AUSSCHUSS, 1996) for flaw analyses and Russian utility procedure MRKR-SKhR-2004 (MRKR-SKhR, 2004), VERLIFE Unified Procedure (VERLIFE, 2008) or international guidelines (IAEA, 2005), US NRC PTS screening criteria (Faidy, 2007) for PTS rules. The required analyses that may take into account during operation lifetime of reactor and periodically updates can be found in IAEA-TECDOC-1627 (2010). These PTS analyses are very crucial for regulatory purpose, licensing activities, in-service maintenance program.

2.2 Factors Affecting PTS Event

The complex and non-uniform mixing phenomena of PTS analysis, along with the various scenarios studied, prevent a unanimous conclusion regarding the most suitable CFD codes or turbulence models. However, certain studies may suggest using an anisotropic turbulence model. It is significant to note that these studies investigate on resolving flow distribution and coolant temperature (Uitslag et al., 2020). Furthermore, the RPV wall undergoes continuous neutron exposure, which induces embrittlement of the vessel due to irradiation effects (Chexal et al., 1982). This exacerbates the crack growth during PTS occurrences. Therefore, conducting a structural integrity assessment of the RPV for PTS event becomes crucial to ensure its stability against thermal shock throughout its operational lifespan.

Different transients are responsible for occurring PTS event in an RPV. LOCA in primary or secondary circuit is one of the significant transients which can lead to PTS phenomena in RPV. Incorporated safety systems, both active and passive systems are activated during any LOCA according to the safety design. Though the main goal of these emergency cooling systems e.g., active and passive ECCS, hydro-accumulators etc. are to cool down the reactor core within the temperature and pressure design limit. Rapid cooling from ECCS injection gives rise to thermal shock in the downcomer section. Loss of coolant accident (LOCA) event occurs in the primary circuit of reactor. It can be differentiated in three types such as large break LOCA (LBLOCA), medium break LOCA (MBLOCA) and small break LOCA (SBLOCA). LBLOCA is considered as the design-based accident (DBA) and LBLOCA is a realistic accident which has proved in Three Mile Island accident (Liu et al., 2010). SBLOCA is emphasized and considered a critical event for PTS because depressurization occurs very slowly to initiate the HPIS and probability of occurrence is high.

2.3 Experimental Studies

Various experimental reference data concerning thermal and momentum mixing behavior in PTS-related scenarios have been generated in several facilities, such as ROCOM facility (Kliem et al., 2008), the TOPFLOW facility (Prasser et al., 2006), Cylindrical Core Test Facility (CCTF), the Upper Plenum Test Facility (UPTF), the and the Slab Core Test Facility (SCTF) (Damerell and Simons, 1992). These experiments encompass a diverse range of flow regimes and Froude numbers, incorporating single-phase and two-phase scenarios, thereby generating reference databases for various specific phenomena. Nonetheless, none of these experiments address the heat transfer into the structures. The experimental TOPFLOW-PTS project is constructed to investigate the thermal hydraulic phenomena for an emergency core cooling scenario during a loss-of-coolant-accident (LOCA) in a nuclear reactor circuit (Peturaud et al., 2011).

2.4 CFD analysis for PTS events

To analyze the non-uniform cooling effects, different CFD codes can be used for complex flow distribution during transient event. This CFD can also consider the multidimensional features which can be used for PTS analyses. ECCS injection into the primary coolant system is modeled by different CFD simulation (ANSYS FLUENT, ANSYS CFX, NEPTUNE CFD) and compared to validate the experimental setup of TOPFLOW-PTS (Apanasevich et al., 2014).

Willemsen and Komen (2005) as well as Willemsen and Lycklama (2006) directed their attention towards studying the stratification in the cold leg using CFX, specifically in a single-phase PTS scenario. Their research highlighted the significant role of turbulence production or destruction caused by buoyancy in such scenarios. Boros and Aszodi (2008) examined the thermal stratification within the primary circuit of a VVER-440 reactor, utilizing the CFD code CFX. Their findings emphasized the crucial significance of having an ample amount of operational data for conducting such analyses effectively. Depending on a number of variables, such as the size of the leak, its precise position, and the operational parameters in place, the state of the cooling fluid inside a Nuclear Power Plant (NPP) might change from single-phase to two-phase circumstances. (Lucas et al., 2009). CFD results for a hypothetical medium break loss of coolant accident (MBLOCA) happening in the hot leg were given by Sharabi et al. in 2016. They pointed out how exceptionally time-consuming CFD models are. High-resolution CFD approaches like LES or DNS offer improved accuracy in resolving flow patterns and heat transfer, with examples of LES simulations in Bieder and Giovanna Rodio (2019) and Loginov et al. (2011), and a recent DNS example in Lai et al. (2020).

2.5 Stress Evaluation for PTS events

Different international projects and experts has been involved in study of RPV under PTS to assess the integrity and used various codes and methods. Thermal profile and stress profile which are affecting RPV are the main outcome of the thermal hydraulic analyses in these studies. One dimensional different code like RELAP (RELAP, 1999), TRACE (TRACE, 2007) is widely employed for TH analyses. The integrity assessment of an RPV under an overcooling scenario necessitates a thorough examination of a wide range of influencing factors to ensure a comprehensive safety analysis. Pugh and Bass (2000) outlined numerous influencing factors that can affect the integrity of an RPV during a PTS-induced event. The focus of the current study revolves around the fracture assessment of the RPV.

Different analytical analyses simply depend on the fractures and structural geometries but didn’t count the actual situation experienced by RPV during PTS in LOCA event. (Fillery and Hu, 2012; Ma et al., 1994; Oliveira and Wu, 1987). Kral and Vyskocil (2018) used FLUENT with URANS modeling to simulate a PTS scenario in the VVER-440 and VVER-1000 reactors. They subsequently compared these findings with the outcomes of their RELAP5 system TH calculations. Except for circumstances like a main steam line break, where the difference in the cold plume zone reached tens of degrees, the observed variances were generally quite small. The cooling flow is non-homogeneous in nature and develops form which is known as the "cooling plume" (Qian et al., 2016).

In transient simulations, the temporal scale is a key factor in understanding dynamic processes. According to Wang et al. (2017), the transient behavior can last for more than ten thousand seconds after a Small Break Loss of Coolant Accident (SBLOCA), whereas the coolant mixing phenomenon only lasts for about ten seconds. Previous studies have been done for MBLOCA and LBLOCA under PTS event considering single phase liquid water (Gonzalez et al., 2015; Sharabi et al., 2016). Again, structural integrity of RPV due to PTS is evaluated by 3D CFD approach associated with the probabilistic fracture mechanics (PFM) analysis (Pin-Chiun et al., 2019)

In an NPP, RPV is the only irreplaceable part which is the heat generating component. The RPV is exposed to extreme radiation environment, high temperature, high pressure during its operational time. During its operation, it undergoes different embrittlement process, in which, neutron embrittlement is one the most significant which deteriorate its strength. With the time being, the RPV materials fracture toughness decreases due to the neutron embrittlement (Tipping, 2010). Significant event like PTS can exaggerate the existing crack propagation in RPV wall, especially in the beltline region adjacent to the nuclear fuel and reactor core. In extreme condition, the RPV may face partial or complete wall fracture.

A detailed thermal hydraulic analysis has been done by USNRC for its Pressurized Thermal Shock Rebaseline Program considering the associated transient sequences of important risk for PTS events for the Oconee-1, Beaver Valley-1, and Palisades Nuclear Power Plants by the RELAP5/MOD3.2.2 gamma computer program (U.S. NRC, 2004). These TH analyses are taken into account for the initial and boundary conditions of CFD analyses in different studies.

2.6 Fracture Assessment due to PTS

For structural integrity assessment, fracture analysis is the last stage where the conclusion is made where the integrity will sustain or not. There are a number of models accessible that consider the chemical composition and neutron fluence into account in order to evaluate the fracture toughness of materials at different temperatures. These models encompass the Master Curve (ASTM-E1921-02, 1997), the FAVOR model (Williams et al., 2004), and the ASME model (ASME, 1995). These frameworks, which have been widely used in the field, provide useful tools for assessing fracture toughness under various situations. Fracture analysis has two parts, they are deterministic and probabilistic analyses. Deterministic analyses require crack size, load, and strength for the calculations as deterministic parameters and assessment is done with the help of ASME code or other establish codes. Stress profile of the inner wall and SIF along different postulated crack due to specific event can be obtained and evaluated with the fracture toughness (Sun, X., Chai, G. and Bao, Y., 2018). Probabilistic analyses have been used to determine the rupture risk associating different parameters like nil ductility temperature, In Service Inspection (ISI) for PTS case in PWR. Different regulatory guidelines has been developed by IAEA and regulatory bodies on PTS and fracture assessment by using PFM. Some codes of PFM have been developed such as FAVOR (Fracture Analysis of Vessels – Oak Ridge), PASCAL (PFM Analysis of Structural Components in Aging LWR), R-PIE (Reactor - Probabilistic Integrity Evaluation). Thermomechanical stress distribution on RPV wall fracture analyses of postulated crack in the inner wall need the accurate assessment to identify the mechanical problems. Different commercial finite element computer codes are used to analyze these parts. ABAQUS (ABAQUS, 2012) code usually use in two approaches, i.e., conventional FEM and extended FEM (XFEM) to calculate the SIF. On the other side, conventional approach required the finite element mesh around the crack executed in the geometry. RPV modeling for PTS event has been done by Mora et al. (2019) using 3D-XFEM. A multi-step simulation approach is taken by CFD-Fluent for thermal–hydraulic simulation, ABAQUS-FEM for thermo-mechanical and fracture mechanics analysis with ABAQUS-XFEM used for a LBLOCA accident.

Gonzalez et al. (2016) carried out an uncoupled analysis that was divided into two parts. In order to estimate the temperature distribution, they first addressed the pure thermal issue during the LOCA. They then figured out the mechanical issue to evaluate the stress field brought on by both thermal loading and pressure. Finally, as required by the existence of stress concentrations, a fracture mechanics analysis was carried out in certain regions of interest. Since the strength of material of RPV decrease with respect of operational time, so it is required to analyze the effect of PTS at the end of the design life. However, Annor-Nyarko, Xia, and Ayodeji (2022) has done a thorough thermomechanical analyses of a RPV under PTS due to frequent anticipated transient which is inadvertent actuation of safety injection valve. Under PTS event, the crack initiation and propagation process are studied by XFEM. A re-pressurization on RPV make the critical challenge to the strength of the structure (Sun, Chai, and Bao, 2017). Qian and Niffenegger (2013) conducted a study that involved the assessment and comparison of the effectiveness of the FAVOR and PASCAL codes. Additionally, they provided a case study employing the FAVOR code to illustrate their findings. Qian and Niffenegger (2014) utilized the FAVOR code to assess the failure probability of an RPV under PTS conditions during LOCA. Their study concluded that taking into account the observed crack distribution could result in a higher failure probability compared to assuming cracks of specific sizes.

2.7 Research Gap

Previous studies have examined postulated transient scenarios (PTS) for various reactor types, including PWR, VVER-440, and VVER-1000. These investigations have encompassed all stages of PTS, considering thermal-hydraulic behavior, computational fluid dynamics (CFD) phenomena, and stress evolution. Most of these studies have primarily concentrated on the analysis of design basis accidents and related calculations. However, there has been limited analysis of the VVER-1200 reactor, taking into account its unique design, material properties, and neutron fluence estimations. Several experimental studies have been conducted in the past to analyze aspects such as heat transfer, temperature distribution, crack propagation, and PTS phenomena over different time scales. Additionally, various software and code-based analyses have been carried out to investigate thermal hydraulics during PTS transients, along with associated CFD and thermo-mechanical analyses, in order to understand reactor behavior during these events. Different codes and tools have been employed in different fields of study. Hypothetically, in the event of an accident, the possibility of pressure exceeding the allowable limit for the VVER-1200 reactor presents a research gap that needs addressing. Contributing to this context, potential design improvements could include establishing a maximum allowable temperature limit for preventing reactor pressure vessel (RPV) for crack propagation, defining a standard crack size for consideration, and setting regulatory limits and long-term operation management to ensure the reactor's integrity in extreme accidental conditions.

2.8 Summary

The existing literature on postulated transient scenarios (PTS) for various reactor types has focused mostly on design basis accidents, with limited analysis of the VVER-1200 reactor. Experimental studies have observed heat transfer, temperature distribution, and crack propagation, while software-based analyses have investigated thermal hydraulics and thermo-mechanical behavior. However, there is a research gap regarding the possibility of higher pressure in the VVER-1200 reactor during accidents above permissible limits. In order to address this, possible design improvements might include defining standard crack sizes for consideration, establishing regulatory limits and long-term operational oversight for improved reactor integrity during extreme conditions, and setting maximum temperature boundaries to avoid reactor pressure vessel (RPV) crack propagation.

CHAPTER 3
METHODOLOGY

3.1 Pressurized Thermal Shock (PTS) Modelling

Pressurized Thermal Shock (PTS) modeling is a complex analytical and computational process to assess the structural integrity and safety of reactor components, particularly reactor pressure vessels (RPVs). PTS events occur when a sudden and significant temperature change, combined with high pressure, creates stress in the RPV that could potentially lead to the initiation and growth of cracks. These events are a critical safety concern in the operation of nuclear power plants. The comprehensive analysis of PTS is a multidisciplinary work that demands diverse skills and expertise. Figure 3.1 presents a flowchart illustrating a method for conducting PTS analysis, outlining the various steps involved in the process.

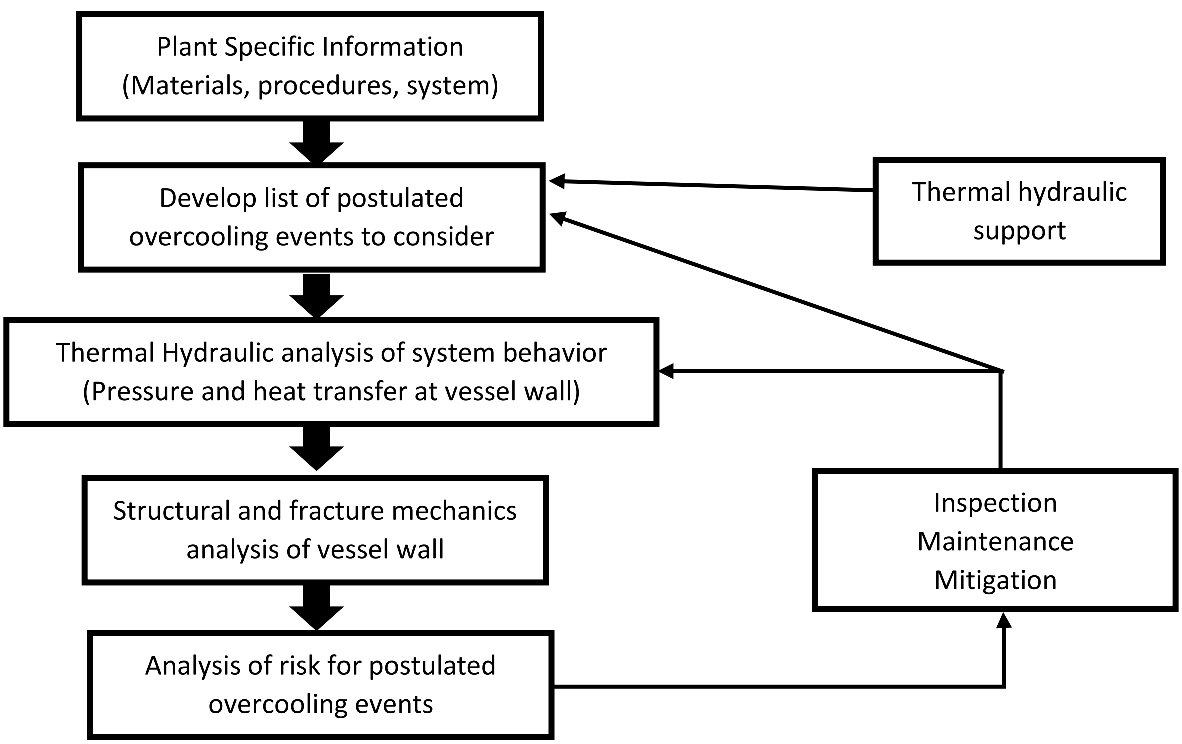


Fig. 3.1: An Analysis Approach to PTS (Christopher Boyd, THICKET, 2008)

3.2 Methodology for assessing integrity due to PTS event

To conduct this research, the analysis is done by four steps to assess the PTS event during LOCA scenario. The steps are descried shortly below:

1. 3D Model Construction: Construction of a 3D Reactor Pressure Vessel (RPV) model is carried out using ANSYS Design Modular, followed by analysis using a symmetric construction model.

2. CFD and Fluid-Structure Interaction (FSI) Analysis: 3D study of fluid flow and temperature of the RPV CFD model is done by ANSYS FLUENT. And for FSI modeling, ANSYS Mechanical is coupled with ANSYS FLEUNT.

3. Stress Analysis: Stress analysis of the computational model is done by ANSYS Mechanical to study mechanical and thermal stress distribution.

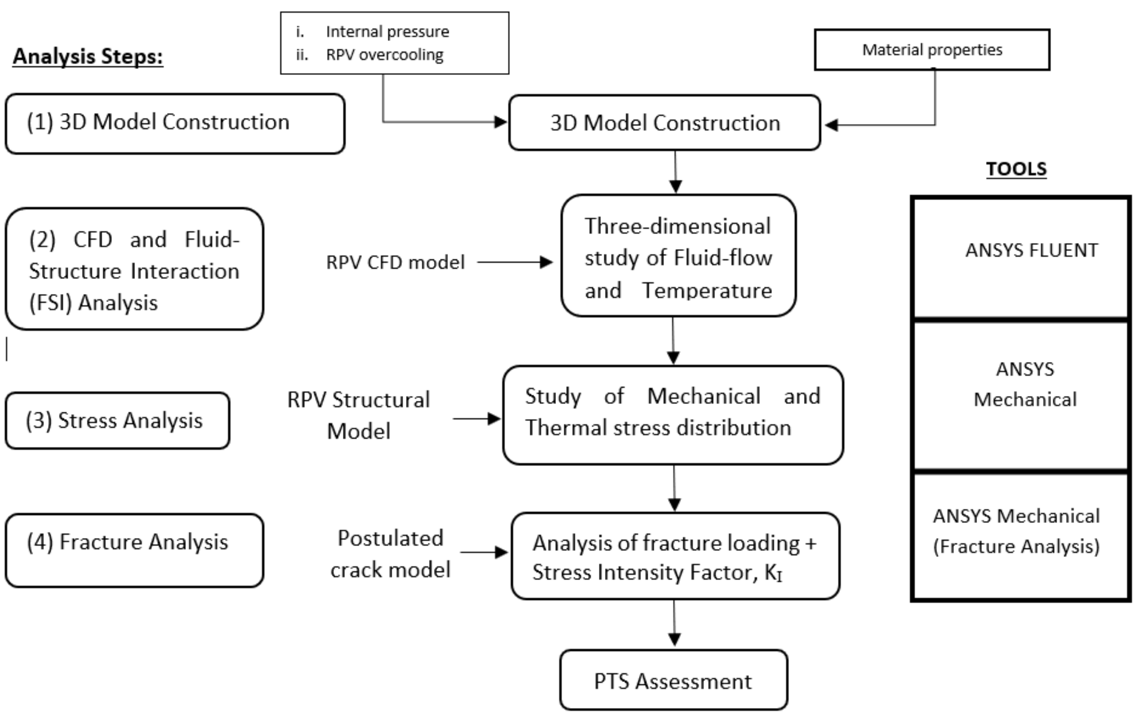


Fig. 3.2: PTS assessment methodology

4. Fracture Analysis: The computational model is subjected to fracture analysis by incorporating a postulated crack profile. This allows for the examination of fracture loading and the calculation of stress intensity factors.

After all these steps, final assessment is done to evaluate the PTS scenario for MSLB LOCA event. The flowchart illustrating the research methodology for this study is presented in Fig. 3.2.

3.3 Transient Scenario Selection for PTS Analysis

To analyze a potential PTS scenario created by LOCA event, a case has been selected from the study done by U.S. Nuclear Regulatory Commission (USNRC, NUREG- 1806, Vol. 1, 2007). In this study, LOCA due to Main Steam- Line Breaks (MSLB) were identified as one of the most significant cases for arising the risk to propagate through wall crack. In this research work, Beaver Valley Main Steam Line Break from Hot Zero Power (Beaver Valley Case 105) is taken as the transient scenario from the analysis of US Nuclear Regulatory Authority considering its temperature and pressure variation and possible PTS event to analyze for VVER-1200 (USNRC, NUREG/CR-6858, 2004).

3.3.1 Main Steam Line Break (MSLB)

One of the main steam lines, which transport steam from the reactor coolant system to the turbines, rupturing is known as a Main Steam Line Break (MSLB), a catastrophic accident scenario in VVER-1200. The event may cause the reactor coolant system to rapidly depressurize, endangering reactor safety and possibly resulting in core damage. Emergency core cooling systems are among the safety systems that can be activated by a quick loss of pressure during an MSLB in order to lessen the severity of the accident. However, core cooling is potentially affected if these systems don't work properly or if the accident exceeds their capacity, which could cause the nuclear fuel to overheat and melt. The MSLB is among the design-basis accidents taken into account in VVER safety analyses.

In this transient scenario (BV Case-105), due to the Main Steam Line Break from Hot Zero Power, the Reactor control pumps (RCP) are tripped.

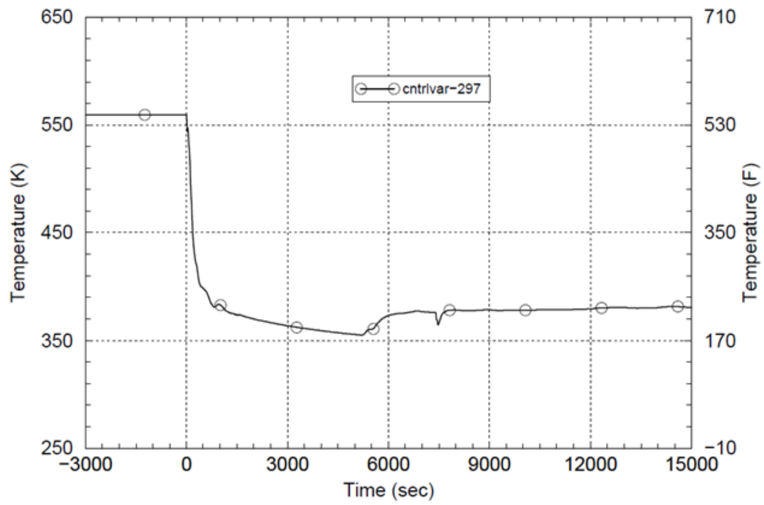


Fig. 3.3: Average Downcomer Fluid Temperature during Transient Scenario: LOCA- Main Steam Line Break from Hot Zero Power (BV Case 105) (USNRC, NUREG/CR-6858, 2004).

The temperature and pressure fluctuation which are most important parameters, are presented in Figure 3.3 and 3.4 respectively.

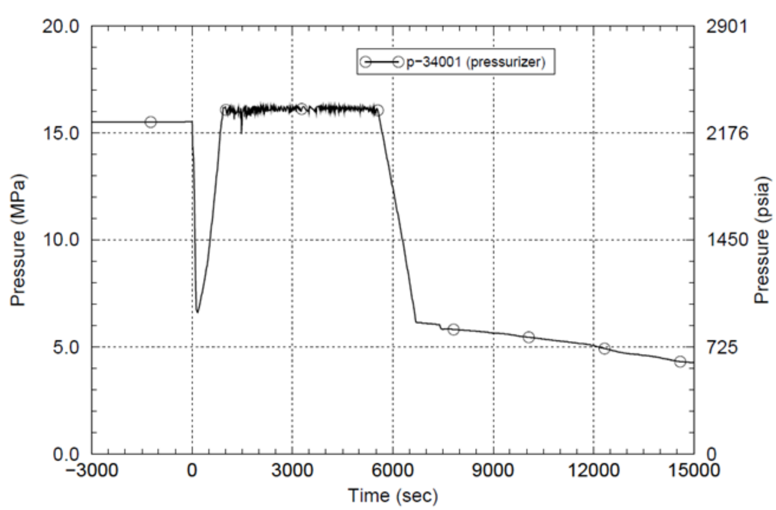


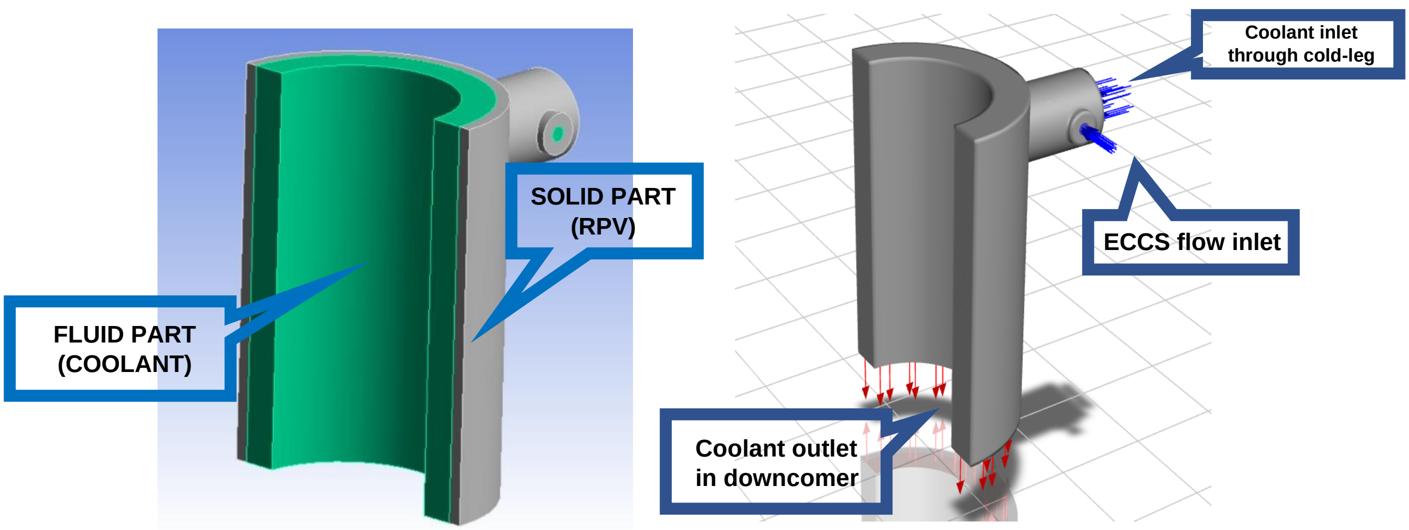
Fig. 3.4: Primary system pressure during Transient Scenario: LOCA- Main Steam Line Break from Hot Zero Power (BV Case 105) (USNRC, NUREG/CR-6858, 2004).

At the beginning of the transient, temperature and pressure decreases rapidly, but after a few times, the system pressure increases to around 16 MPa. The minimum downcomer fluid temperature reached to 355 K at 16.2 MPa primary pressure.

3.4 Computational Model

A simplified 3D model of VVER-1200 reactor pressure vessel is designed in ANSYS design modular. The model comprises the beltline region of vessel, cold leg, ECCS injection region connected to cold leg and associated primary and ECCS coolant. The model is designed according to the design of V-491 model of VVER-1200 (IAEA Status Report 108, 2011).

Inlets of the system are taken as the inlet of cold leg coolant and ECCS injection and outlet of the domain is taken as the end of downcomer fluid. In this 3D model, hot leg is not considered.



(a) (b)

Fig. 3.5: Simplified symmetric 3D model of VVER-1200 RPV showing (a) solid & fluid regions and (b) fluid inlet & outlet path

In the simulation, it is considered that the entire flow ultimately directed toward the downcomer. Figure 3.5 shows the simplified symmetric 3D model of VVER-1200 which is considered for simulation with detailed description showing different parts of the model. The entire domain of the RPV is divided into two domains, which are depicted in Figure 3.5, namely, the fluid and the solid regions. Different parameters of RPV computational model that has been used in this study are shown in the following Table 3.1.

Table 3.1: Different parameters of VVER-1200 used for computational model (IAEA Status Report 108, 2011)

Parameter	VVER-1200, unit	Computational Model, Value, unit
Vessel diameter	4.232 m	4.2 m
Vessel height	11735 mm	6000 mm (Belt line region)
Thickness	197.5 mm	200 mm
Cold leg diameter	850 mm	850 mm
ECCS inlet diameter	225 mm	225 mm

3.5 Material Properties

During PTS analyses of VVER-1200, the base material is taken Steel 15H2NMFA which is being used in the computational model for this analysis (Anosov et al., 2018). The thermo-mechanical properties used for the RPV base material in both CFD and thermos-mechanical analyses for PTS event in this research are presented in Table 3.2.

Table 3.2: Material properties of VVER-1200 used in computational model (IAEA-EBP-WWER-08 (Rev. 1), 2006)

Temperature, ◦C	Young's modulus (10^3 MPa), E	Coefficient of linear thermal expansion (10^{-6} K^{-1}), α	Poisson's ratio, υ	Thermal Conductivity (W/K m), λ	Heat Capacity (J/kg K), c_p	Density (kg/m^3), ρ
20	208		0.3	35.0	446.9	7830
50			0.3	35.5	458.9	7822
100	201	11.6	0.3	36.1	478.8	7809
150			0.3	36.6	499.7	7795
200	193	12.0	0.3	36.8	520.4	7780
250			0.3	36.6	541.2	7765
300	183	12.6	0.3	36.2	562.0	7750
350	177.5		0.3	35.6	584.6	7733

3.6 Computational Fluid Dynamics Analysis of PTS

During any transient event such as MSLB, the flow of hot primary coolant and cold ECCS water in the downcomer region as well as the inlet region is anisotropic. This flow requires turbulence modelling which is done in this present study. The temperature profile of fluid in downcomer region is analysis by computational fluid dynamics modeling. The primary fluid is heated by the nuclear reactor from the core and heat transfer is done by convection to reactor pressure vessel wall. ANSYS FLUENT (version 2022 R1) solver is used to determine the temperature profile of fluid and vessel inner wall (ANSYS, 2009).

3.6.1 Turbulence model

In order to solve the turbulent flow considered in this research analysis, standard k-ε model is used in single phase fluid flow. Also, enhanced near wall treatment is employed in the turbulence model with considering thermal effects and pressure gradient effects for the present study. It is assumed that at a considerable distance from the walls, turbulence tends to exhibit near-isotropic behavior, making the k-ε model a suitable choice, comparable to other models in its effectiveness. The simulation is performed for 200 sec using Unsteady Reynolds Averaged Navier-Stokes (URANS) equations for time dependent solution by ANSYS FLUENT CFD simulation.

3.6.2 Theory and Mathematics for CFD Analysis

To solve the CFD analysis by ANSYS FLUENT, it is necessary to carry out Navier-Stokes equations and other relevant equations by fluent solver. The applied energy, continuity and momentum equation are showing in Equation (3.1) to (3.3):

Continuity Equation:

$$\rho\left(\frac{\partial u}{\partial x}+\frac{\partial v}{\partial y}+\frac{\partial w}{\partial z}\right)=0 \tag{3.1}$$

Momentum Equation:

$$\rho\left(\frac{\partial u}{\partial x}+\frac{\partial u}{\partial y}+\frac{\partial u}{\partial z}\right)=\frac{\partial}{\partial x}\left(\mu\frac{\partial u}{\partial x}\right)+\frac{\partial}{\partial y}\left(\mu\frac{\partial u}{\partial y}\right)+\frac{\partial}{\partial z}\left(\mu\frac{\partial u}{\partial z}\right)-\frac{\partial p}{\partial x} \tag{3.2}$$

$$\rho\ \left(\frac{\partial v}{\partial x}+\frac{\partial v}{\partial y}+\frac{\partial v}{\partial z}\right) = \frac{\partial}{\partial x}\left(\mu\frac{\partial v}{\partial x}\right)+\frac{\partial}{\partial y}\left(\mu\frac{\partial v}{\partial y}\right)+\frac{\partial}{\partial z}\left(\mu\frac{\partial v}{\partial z}\right)-\frac{\partial p}{\partial y}$$

$$\rho\ \left(\frac{\partial w}{\partial x}+\frac{\partial w}{\partial y}+\frac{\partial w}{\partial z}\right) = \frac{\partial}{\partial x}\left(\mu\frac{\partial w}{\partial x}\right)+\frac{\partial}{\partial y}\left(\mu\frac{\partial w}{\partial y}\right)+\frac{\partial}{\partial z}\left(\mu\frac{\partial w}{\partial z}\right)-\frac{\partial p}{\partial z}$$

Energy equation:

$$\rho\ \left(u\frac{\partial T}{\partial x}+v\frac{\partial T}{\partial y}+w\frac{\partial T}{\partial z}\right) = \frac{\partial}{\partial x}\left(\frac{k_f}{c_p}\frac{\partial T}{\partial x}\right)+\frac{\partial}{\partial y}\left(\frac{k_f}{c_p}\frac{\partial T}{\partial y}\right)+\frac{\partial}{\partial z}\left(\frac{k_f}{c_p}\frac{\partial T}{\partial z}\right) \tag{3.3}$$

Here, c_p= Specific heat of fluid [J/kg K], $\rho =$ Density [kg/m^3], $T =$ Temperature, $\mu =$ Dynamic viscosity [(N·s)/m^2] and u, v, and w in x, y and z directions velocity components

During turbulence modeling of Standard $k-\varepsilon$ model, for Reynolds averaging, the mean and fluctuation components of the solution variables in the Navier-Stokes equations are separated (ANSYS, 2013). For the velocity components:

$$u_i = \bar{u}_i + u_i'$$

Where $\bar{u}_i$and u_i' are the mean and fluctuating velocity components (i=1,2,3)

Similar results hold for pressure and other scalar quantities:

$$\varphi_i = \bar{\varphi}_i + \varphi_i'$$

In this case, φ stands for a scalar, such as pressure, energy, or species concentration.

The fluent solver used continuity and momentum Equation (3.4) and (3.5) for the discretization showed as follows:

$$\frac{\partial \rho}{\partial t} + \frac{\partial}{\partial x_i}(\rho u_i) = 0 \tag{3.4}$$

$$\frac{\partial}{\partial t}(\rho u_i) + \frac{\partial}{\partial x_j}(\rho u_i u_j) = -\frac{\partial p}{\partial x_i} + \frac{\partial}{\partial x_j}\left[\mu\left(\frac{\partial u_i}{\partial x_j} + \frac{\partial u_j}{\partial x_i} - \frac{2}{3}\delta_{ij}\frac{\partial u_l}{\partial x_l}\right)\right] + \frac{\partial}{\partial x_j}\left(-\rho\overline{u_i u_j}\right) \tag{3.5}$$

Equation (3.4) and Equation (3.5) from ANSYS Fluent Theory Guide (2013), which are commonly referred to as the Reynolds-averaged Navier-Stokes (RANS) equations. These equations possess a similar structure to the instantaneous Navier-Stokes equations, but with the velocities and other solution variables now representing ensemble-averaged or time-averaged values. The inclusion of additional terms accounts for the influence of turbulence. However, to complete Equation (3.5), the term $-\rho\overline{u_i u_j}$ is necessary to model the Reynolds stresses, which play a vital role in achieving closure for the Equation (3.5) equations. The transport equations used to obtain the turbulence kinetic energy (k) and its dissipation rate (ε) for Standard k-ε model are as follows as Equation (3.6) and Equation (3.7):

$$\frac{\partial}{\partial t}(\rho k) + \frac{\partial}{\partial x_i}(\rho k u_i) = \frac{\partial}{\partial x_j}\left(\left(\mu + \frac{\mu_t}{\sigma_k}\right)\frac{\partial k}{\partial x_j}\right) + G_k + G_b - \rho\varepsilon - Y_M + S_k \tag{3.6}$$

$$\frac{\partial}{\partial t}(\rho\varepsilon) + \frac{\partial}{\partial x_i}(\rho\varepsilon u_i) = \frac{\partial}{\partial x_j}\left(\left(\mu + \frac{\mu_t}{\sigma_\varepsilon}\right)\frac{\partial \varepsilon}{\partial x_j}\right) + C_{1\varepsilon}\frac{\varepsilon}{k}(G_k + C_{3\varepsilon}G_b) - C_{2\varepsilon}\rho\frac{\varepsilon^2}{k} + S_\varepsilon \tag{3.7}$$

In these equations, G_k signifies the generation of turbulence kinetic energy resulting from mean velocity gradients. G_b represents the generation of turbulence kinetic energy caused by buoyancy. Y_M denotes the contribution of fluctuating dilatation in compressible turbulence to the overall dissipation rate. $C_{1\varepsilon}, C_{2\varepsilon}$ and $C_{3\varepsilon}$ are constants. σ_k and σ_ε are the turbulent Prandtl numbers for k and ε, respectively. S_k and S_ε are user-defined source terms. The model constants have following default values for simulation:

$$C_{1\varepsilon} = 1.44, C_{2\varepsilon} = 1.92, \sigma_k = 1.0 \text{ and } \sigma_\varepsilon = 1.3$$

3.6.3 Mesh Configuration

For mesh setup, hex dominant method is taken for the CFD model. In order to achieve the better accuracy, edge sizing is done at the concerned nozzle region. The mesh configuration used for the computational model of CFD analyses is given below in Table 3.3.

Table 3.3: Main parameters of mesh configuration for CDF model

Parameters	Value (Unit)
Element order	Quadratic
Element size	70 mm
Node No	584627
Element no	164874

Orthogonal quality	0.78
Skewness factors	0.32

Mesh configuration for symmetric CFD model of RPV I divided into two parts. One is for fluid domain and one is for solid domain. Figure 3.6 shows the hex dominant mesh configuration of solid domain, fluid domain and around the edge of nozzle.

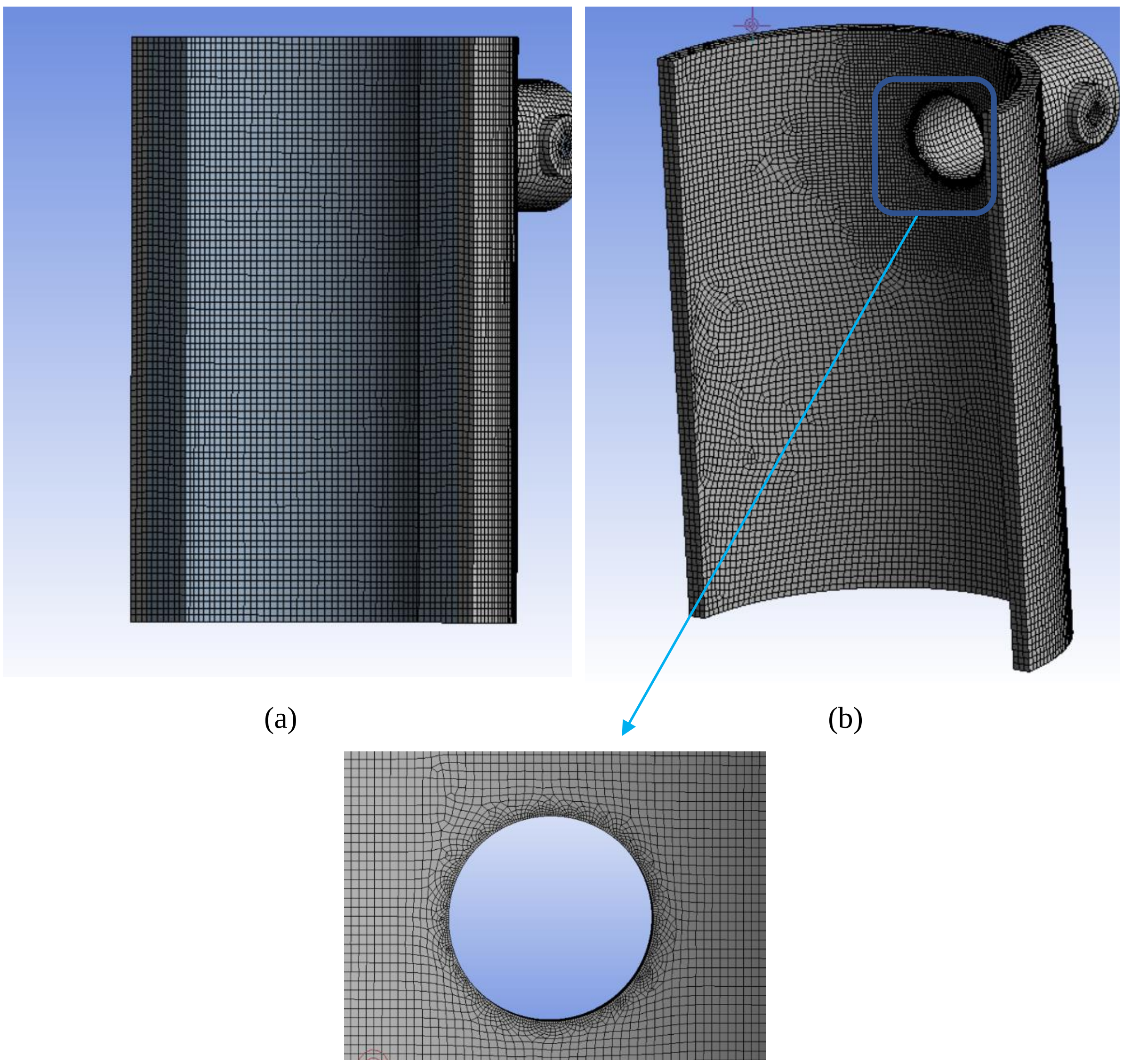

(a) (b) (c)

Fig. 3.6: Mesh configuration of (a) fluid domain (b) solid domain & (c) around edge of nozzle

3.6.4 Initial and Boundary Conditions

The CFD simulation is executed by ANSYS FLUENT considering the following computational setup given in Table 3.4.

Table 3.4: Main Parameters of initial and boundary conditions for CFD simulation

Parameters (Unit)	Value
Transient Pressure (MPa)	7
Inlet velocity of ECCS (m/s)	0.4
Inlet temperature of ECCS (◦C)	10
Inlet temperature (◦C)	298
Flow type of primary coolant	Stagnant
Void fraction	0
Inlet mass flow rate of ECCS (kg/s)	80
Transient primary coolant temperature (◦C)	298
Viscosity (Pa-s)	0.2
Coolant Density (kg/m^3)	1000

Pressure based solver is used for this transient CFD analyses with a time step size of 0.25 sec. The simulation is run for 200 sec. The low average CFL values for the simulation is kept below the 0.5 for the base mesh. In FLUENT setup, the velocity inlet- pressure outlet combination is used for boundary condition. The solver implemented a second-order upwind discretization method to solve the governing PDEs. The pressure-velocity coupling was evaluated using the SIMPLEC scheme, and a convergence criterion lower than 10^6 was employed.

3.7 Thermo-Mechanical Analysis of PTS

To analyze the stress profile generated in the inner wall of RPV during PTS event, ANSYS Mechanical is used (ANSYS, 2013). RPV base material that have been used in thermo-mechanical analysis is same as table 3.2. Thermo-mechanical analysis is only done for solid part which is RPV. The temperature evolution with respect to time and other conditions from FLUENT are imported to Static Mechanical Analysis system of ANSYS Mechanical and imposed as the boundary conditions. The finite element method (FEM) is carried out with the discretized equations at each node for the elements generated after meshing.

3.7.1 Theory and Mathematics for Stress Analysis

Equivalent (Von Mises) Stress: The Equivalent (Von Mises) Stress is a measure of the equivalent stress that combines the effects of normal and shear stresses. It is often used to assess the overall stress level in the material and evaluate its potential for yielding or plastic deformation (ANSYS, 2013). Mathematical expression for calculating equivalent stress is given in Equation (3.8).

$$\sigma_{eq} = \frac{1}{\sqrt{2}}\sqrt{(\sigma_x - \sigma_y)^2 + (\sigma_y - \sigma_z)^2 + (\sigma_z - \sigma_x)^2 + 6(\sigma_{xy}^2 + \sigma_{yz}^2 + \sigma_{zx}^2)} \tag{3.8}$$

If the Von Mises stress of the vessel exceeds its yield strength, it indicates that the material has reached or exceeded its elastic limit and may undergo plastic deformation. In this situation, the vessel may experience permanent deformations or even failure, depending on the severity of the stress levels and the material's ductility. When the yield strength is exceeded, the material's ability to return to its original shape diminishes, and it may exhibit plastic deformation. If the stress continues to increase beyond the yield point, the material may experience localized necking or further deformations until it eventually fractures.

Therefore, if the Von Mises stress exceeds the yield strength of the vessel, it indicates that the structural integrity of the vessel is compromised, and there is a risk of plastic deformation or failure. The Von Mises stress is primarily used for ductile materials. If the Von Mises stress of a vessel exceeds the ultimate yield strength of the material, it indicates that the stress levels have reached or exceeded the maximum capacity of the material to resist deformation before failure. In this scenario, the material may experience significant plastic deformation, which can lead to structural failure or rupture.

Principal Stress: The principal stresses represent the maximum and minimum stress values acting on a specific plane or point within the material to determine the failure potential of the material and identify areas where the stress levels approach or exceed the material's fracture toughness (ANSYS, 2013).

The three normal stresses that remain are called the principal stresses:

σ_1 - Maximum

σ_2 - Middle

σ_3 - Minimum

Equivalent stress is related to the principal stresses by the Equation (3.9):

$$\sigma_{eq} = \left[\frac{(\sigma_1 - \sigma_2)^2 + (\sigma_2 - \sigma_3)^2 + (\sigma_3 - \sigma_1)^2}{2}\right]^{1/2} \tag{3.9}$$

Principal stress is applicable to both ductile and brittle materials. For ductile materials, the principal stresses are relevant for analyzing yielding, plastic deformation, and failure mechanisms. The stress states at critical points can be evaluated using the principal stresses to assess the material's behavior under different loading conditions.

For ductile materials, exceeding the yield strength implies that the material will undergo plastic deformation. It may exhibit permanent deformations, such as elongation, bending, or bulging. The vessel may experience localized yielding or necking, indicating that the material has reached its elastic limit and will not fully recover its original shape upon unloading. For ductile materials, exceeding the ultimate yield strength may lead to extensive plastic deformation and structural failure. The material may experience localized necking, significant deformations, and even complete rupture, compromising the integrity and functionality of the vessel.

For brittle materials, which have limited ductility, exceeding the yield strength can result in more abrupt consequences. Brittle materials tend to fail suddenly and without significant plastic deformation. For brittle materials, surpassing the ultimate yield strength can result in catastrophic failure without significant plastic deformation. The high stress concentration may lead to crack initiation and propagation, ultimately causing the vessel to fracture suddenly.

Hoop Stress: A form of mechanical stress known as "hoop stress" (σh) is applied to rotationally symmetric objects, such as pipes and tubing. It is the result of circumferentially acting forces. The stresses caused by internal pressure (P) in a cylindrical cylinder are shown in Figure 3.7. Figure 3.7 shows a section of the pipe with the hoop stress indicated on the right side.

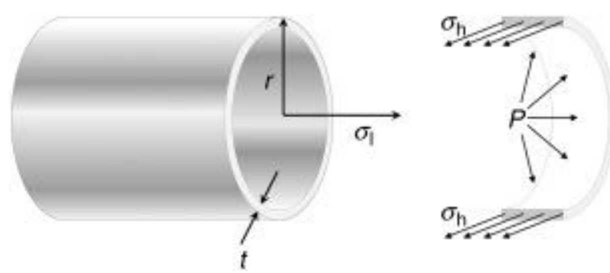


Fig. 3.7: Illustration of hoop stress & longitudinal Stress (McKeen, L.W., 2016.)

The following is the classic Equation (3.10) for hoop stress caused by internal pressure in a thin-walled cylindrical pressure vessel.

$$\sigma_h = \frac{Pr}{t} \quad (3.10)$$

Here, σ_h denotes the hoop stress, P represents internal pressure, r is radius of the cylinder and t is the wall thickness of cylinder.

Longitudinal Stress: When equal and opposite forces are applied to two cross-sectional areas of the cylinder, the stress endured by the cylinder is referred to as longitudinal stress. The longitudinal stress under the same conditions of Figure 3.7 is given by Equation (3.7).

$$\sigma_l = \frac{Pr}{2t} \quad (3.11)$$

3.7.2 Mesh Configuration

The mesh configuration used for the computational model of ANSYS Mechanical analyses is given below in Table 3.5.

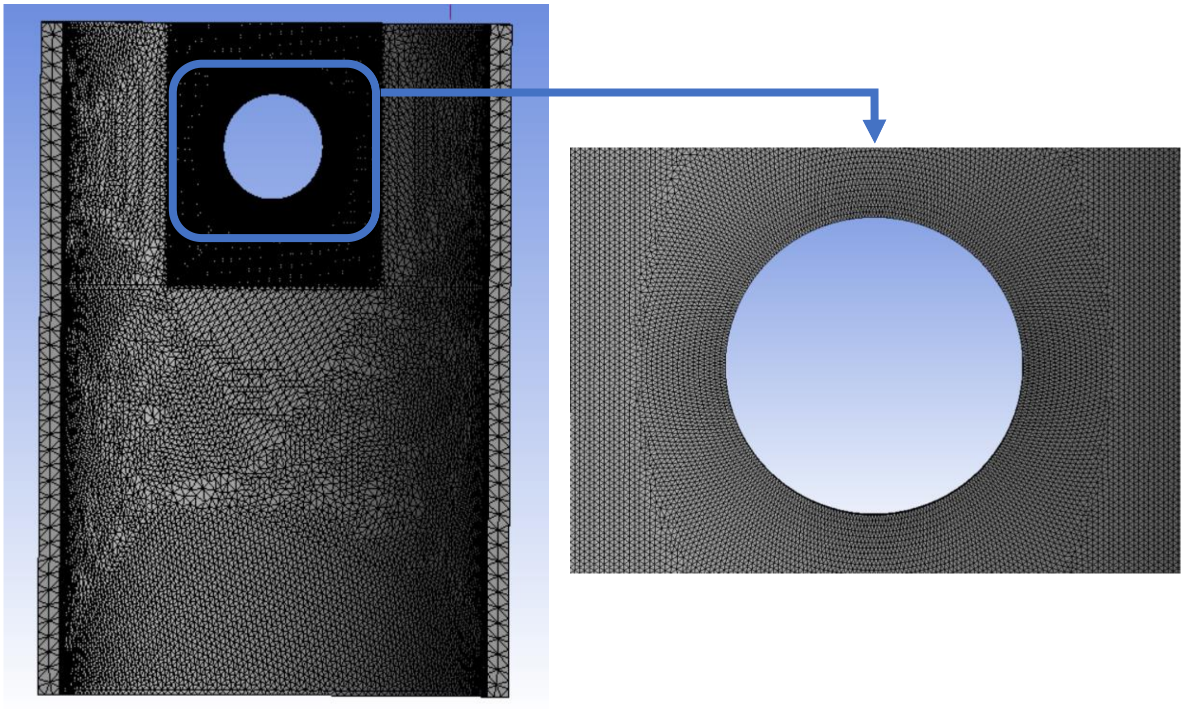
Table 3.5: Main parameters of mesh configuration for ANSYS Mechanical analyses

Parameters	Value (Unit)
Element order	Quadratic
Element size	70 mm
Node No	373146
Element no	216029
Orthogonal quality	0.78
Skewness factors	0.32

For mesh setup, tetrahedron element shape is taken for the model. In order to achieve the better accuracy, edge sizing and mesh refinement is done at the concerned nozzle region. To enhance the solution accuracy, regular and structured mesh as possible is employed. This approach aims to ensure a uniform distribution of grid points throughout the domain, minimizing irregularities and optimizing the precision of the solution. Figure 3.8 shows the tetrahedron mesh configuration of RPV for Thermo-mechanical analysis at full solid domain & near inlet nozzle edge with fine refinement of mesh.

3.7.3 Initial and boundary conditions

thermo-mechanical analyses, inner pressure of the system is preliminary taken as 7 MPa at the end of the simulation decreasing from initial pressure 16.2 MPa. Stress profile has been evaluated for 7 MPa without any postulated crack. it is conservatively assumed to consider a constant pressure of 16.2, 18, 20 & 22 MPa in the fracture analyses. The constraints applied on the RPV model during stress analyses is depicted in Figure 3.9.



(a) (b)

Fig. 3.8: Mesh configuration of RPV for Thermo-mechanical analysis (a) full solid domain & (b) near inlet nozzle edge fine refinement of mesh

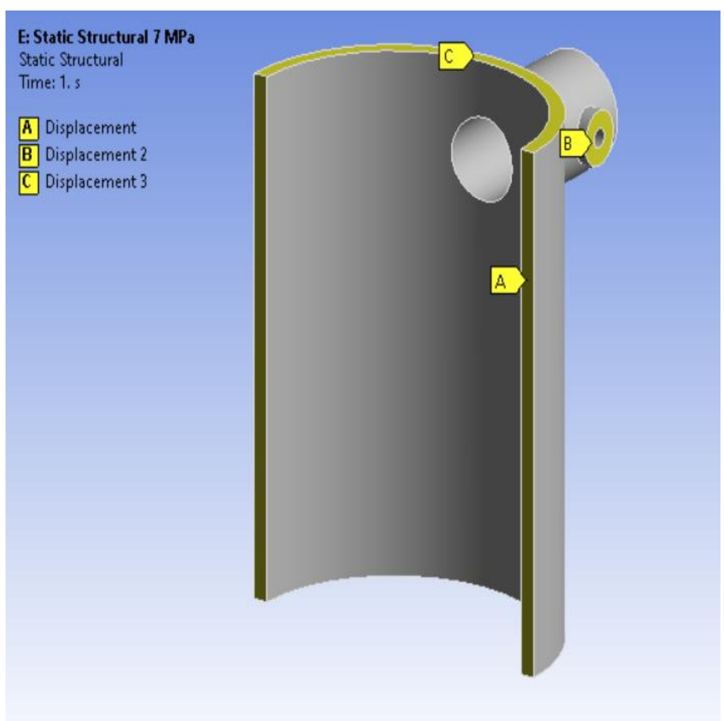


Fig. 3.9: Constraints applied on RPV model

The RPV wall temperature of computational model is imported from the result of ANSYS FLUENT. Here, the wall temperature of RPV is taken into account during the stress analysis for transient case. Figure 3.10 shows RPV wall temperature imported from ANSYS FLUENT result which is coupled with ANSYS Mechanical.

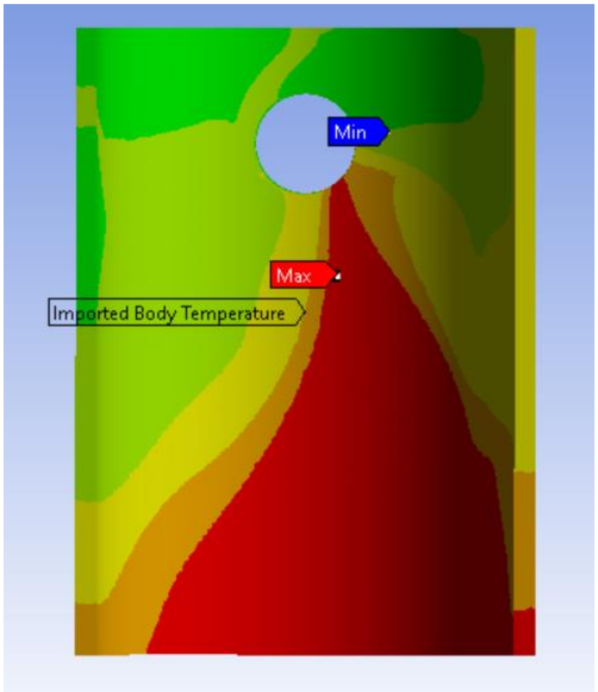


Fig. 3.10: RPV wall temperature imported from ANSYS FLUENT result

3.8 Postulated Crack Analyses

3.8.1 Semi-elliptic crack model

The semi-elliptical crack shape is modelled with a constant ratio between the arc crack length and the crack depth. The term "semi-elliptic" refers to the fact that the crack shape resembles half of an ellipse. Semi-elliptic cracks can lead to significant stress concentrations at their tips.

A semi-elliptic crack model is a structural analysis and engineering concept used to assess the integrity and safety of materials and components, particularly in the context of fracture mechanics. This model is commonly employed to predict the behavior and failure of structures containing semi-elliptical cracks, which are discontinuities or flaws in the material characterized by a curved, half-elliptical shape. The stress concentration factor is a critical parameter that engineers and analysts consider when assessing the structural integrity of a component with such a crack. The analysis of semi-elliptic cracks often involves the principles of fracture mechanics.

This includes calculating stress intensity factors (K_I) and critical crack lengths to predict the crack's behavior and whether it might lead to structural failure. Fig. 3.11 depicts a curvilinear coordinate system for surface semi-elliptic crack.

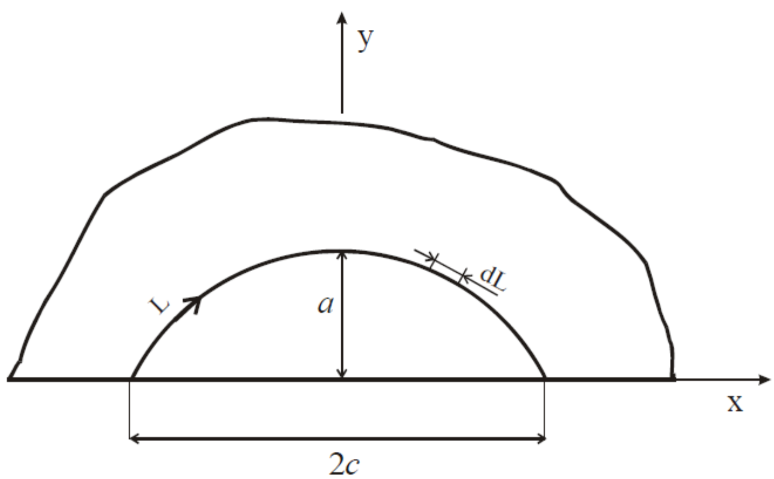


Fig. 3.11: Curvilinear coordinate system for surface semi-elliptical crack (IAEA-EBP-WWER-08 (Rev. 1), 2006)

3.8.2 Crack dimensions for analysis according to RUSSIAN, US standard and generic model

Three type of crack profile is inserted at the RPV inner wall surface and analyzed them individuality. Semi-elliptic crack is taken for the analyses for all the three cases. Crack case I & III are taken from the Russian and ASME standards. And a generic crack model is taken for case II. Details of the crack models are presented in the following Table 3.6.

Table 3.6: Three different pre-existing crack profile

No.	Standard	Crack depth	Crack length	Major & minor radius in computational model	Aspect ratio, a/c
Case I	Russian	¼ of RPV wall thickness	¾ of RPV wall thickness	Major Radius, c = 75 mm Minor radius, a = 50 mm	0.7
Case II	Generic	-	-	Major Radius, c = 45 mm Minor radius, a = 15 mm	0.3
Case III	ASME code	¼ of RPV wall thickness	1.5 of RPV wall thickness	Major Radius, c = 150 mm Minor radius, a = 50 mm	0.7

Postulated crack position is situated at under the nozzle area in axial direction for all the three cases. The cracks are examined under the PTS scenario and stress intensity factors (SIF) at crack points are determined from the analyses.

3.8.3 Stress Intensity factor (SIF)

Stress Intensity factor (SIF) is a parameter which is used to characterize the loading condition at the postulated flaw. It is denoted by K_C. SIF values are essential for assessing the structural integrity and failure behavior of materials containing cracks or flaws. They provide crucial information about the stress concentration and potential for crack propagation in a given structure or component.

The calculation of the SIF is typically based following Equation (3.12) to (3.14):

(i) Engineering Method: In this method, the stress intensity values for specific defect and vessel shapes are calculated analytically or tabulated.

The calculation of the SIF is typically based on influence functions:

$$K_I = \sqrt{\frac{\pi a}{Q}}.\ \left[b_0.\,i_0 + b_1.\,i_1.\frac{a}{t} + b_2.\,i_2.\ \left(\frac{a}{t}\right)^2 + \ b_3.\,i_3.\ \left(\frac{a}{t}\right)^3 + \ b_4.\,i_4.\ \left(\frac{a}{t}\right)^4\right] \tag{3.12}$$

(ii) Finite Element Analyses: In this method, the crack is directly modelled in the mesh and subjected to the appropriate loadings. The SIF is calculated from the J-integral value according to one of the following formulae:

$$K_I = \sqrt{J.\ E}\ \text{; for plane stress (only for surface point)} \tag{3.13}$$

$$K_I = \sqrt{\frac{J.E}{1-v^2}}\ \text{; for plane strain assumption (other points)} \tag{3.14}$$

For this present study, latter one is taken subjected to the appropriate loadings for crack analyses.

3.8.4 Mesh refinement configuration around the crack

A minor adjustment was made to the mesh in the vicinity of the crack to address the oscillations observed in the Stress Intensity Factors (SIF). A minor refinement was made to the mesh in the vicinity of the crack to minimize numerical errors in SIF. As a result, node number and mesh element no increased to around 216213- 193587 (node) and 132742- 127490 (element Number) respectively. Figure 3.12 shows semi-elliptical crack model & Mesh refinement around crack front which is modelled at the wall surface of RPV. The evaluated SIF values of crack points at 100 C for all of the three cases are compared with the fracture toughness values whether it exceed the limit or not. Specifically, K_{1C} values are taken into consideration for compare, as it provides the maximum SIF during the analyses. All of the three crack cases are analyzed for the PTS event in maximum estimated pressure during MSLB and elevated pressurized situation up to 22 MPa.

3.9 Fracture Toughness evaluation

3.9.1 Fracture toughness

Fracture toughness is a material property that represents the resistance of a material to crack propagation and failure under applied stress. It typically quantifies the amount of stress required to propagate a pre-existing crack. The SIF values are compared to the fracture toughness to assess the criticality of cracks and predict their behavior under different loading conditions. In this study, fracture toughness is calculated according to the analytical approximation relationship shown in Equation (3.15) from ASME XI, appendix G:

$$K_{1C} = 36.5 + 22.783 \exp[\, 0.036\, (T - RT_{NDT})] \qquad (3.15)$$

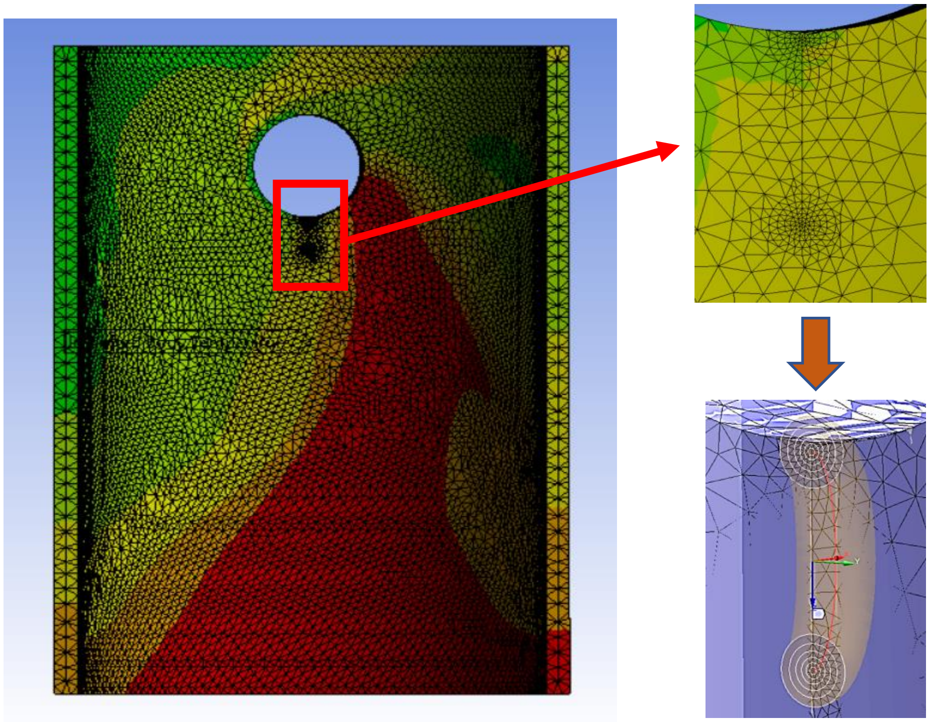


Fig. 3.12: semi-elliptical crack model & Mesh refinement around crack front

3.9.2 Fracture Toughness Re-evaluation

When crack front length, Bi is > 25 mm; re-evaluation of the K_{1c} value is recommended. According to IAEA-EBP-WWER-08 (Rev. 1), Equation (3.16) shows the re-evaluation of the K_{1c} value calculation expression:

$$[K_{1C}]_{Bi} = (25/Bi)^{1/4}.([K_{1C}]_i - K_{min}) + K_{min} \quad (3.16)$$

For semielliptical surface crack, crack front length, Bi of postulated defects can be calculated using the following equation (3.17):

$$Bi = 2c.\left[1 + 4.6\left(\frac{a}{2c}\right)^{1.65}\right]^{0.5} \quad (3.17)$$

This correction for a postulated crack front length is performed only for relations in the range of 25 mm ≤ Bi ≤ 150 mm. For values Bi > 150 mm, the value of Bi = 150 mm is taken,

3.9.3 Reference Nil-Ductility Temperature

The Reference Nil-Ductility Temperature (RT_{NDT}) is a parameter which is used to assess the brittleness of materials, particularly metals, under specific testing conditions. It is a reference temperature point at which a standard Charpy impact test specimen, usually made of a specific material, typically steel, fractures in a completely brittle manner, exhibiting no ductile (tough) behavior. At temperatures below the RT_{NDT}, the material is considered to behave in a brittle manner. This means that when subjected to sudden impact or loading, it will fracture without significant deformation or plasticity. The RT_{NDT} is an important parameter especially for applications in which the material will be exposed to low temperatures. It helps engineers and designers assess the suitability of a material for use in environments where brittle fracture could lead to catastrophic failures, such as in structural components of bridges, pipelines, or pressure vessels. It is crucial to choose materials with RT_{NDT} values that are below the minimum expected service temperature to ensure the material retains adequate toughness and ductility.

The above relationship between fracture toughness and temperature conservatively correlates the critical SIF values to a temperature parameter associated with the reference nil-ductility temperature (RT_{NDT}). Here Equation (3.18) shows the mathematical equation to calculate RT_{NDT}.

$$RT_{NDT} = RT_{NDT}o + \Delta RT_{NDT} + Margin \tag{3.18}$$

$RT_{NDT}o$ is the initial nil-ductility reference temperature. And the safety margin is expressed by equation (3.19).

$$Margin = 2\sqrt{\sigma_1^2 + \sigma_\Delta^2} \tag{3.19}$$

Here, σ_1 is the standard deviation obtained from the set of data used to establish the mean. σ_Δ is 28 °F for welds and 17 °F for base metal. Considering, initial RT_{NDT0} is $-35 \circ C$ and initial fluence, f = $1 \times 10^{19}\ n/cm^2$ and after 60 years (design life), $\Delta RT_{NDT} = 160 \circ C$ at fluence, f = $4.22 \times 10^{19}\ n/cm^2$ (ROSATOM, n.d).

3.9.4 Ductile-to-brittle transition temperature (DBTT) shift or ΔRT_{NDT}

This transition temperature, RT_{NDT} can be influenced by various factors, and one of them is neutron embrittlement. Neutron embrittlement is a phenomenon that occurs in RPV steels when materials are exposed to high levels of neutron irradiation over an extended period. Neutrons can interact with the atomic structure of the material, causing changes that may lead to increased brittleness, reduced fracture toughness, and a shift in the RT_{NDT}. To calculate RT_{NDT}, the values of ΔRT_{NDT} at initial and following service years are essential to identify. Here, ΔRT_{NDT} is calculated using predictive formula is used, shown in Equation (3.20), which is defined in US Nuclear Regulatory Guide 1.99 Revision 2.

$$\Delta RT_{NDT} = CF.\ f\,(0.28 - 0.10 \log f) \tag{3.20}$$

Here, CF = Chemical Factor ($\circ\ F$), a function of copper and nickel content which is given in US Nuclear Regulatory Guide 1.99 Revision 2. And f = neutron fluence

3.9.5 Integrity Assessment condition

Integrity assessment depends on obtained SIF values for different transient scenarios or postulated accidents. If SIF values stay below the limiting fracture toughness values, then it is concluded that integrity of the structure is conserved. But, if SIF values exceed the limiting fracture toughness values, it indicates a critical condition that can lead to brittle fracture and catastrophic failure of the material or component.

The RPV integrity is assured if the following Equation (3.21) according to the IAEA-EBP-WWER-08 (Rev. 1) is fulfilled:

$$K_I < [K_{IC}]_{Bi} \tag{3.21}$$

Brittle fracture is characterized by the rapid propagation of a crack without significant plastic deformation or warning, often resulting in sudden and catastrophic failure. When the SIF exceeds the fracture toughness, it means that the applied stress or loading is sufficient to drive crack propagation. The crack will extend through the material under the influence of the applied load. Unlike ductile materials, which undergo plastic deformation and absorb energy before fracture, this phenomenon can result in a sudden release of stored energy and can lead to explosive or violent failures of vessel, particularly in such high-stress applications. Exceeding fracture toughness values is a significant safety concern which can result in unexpected and catastrophic failures and lead to potential harm to people and property.

3.10 Summary

This chapter provides a comprehensive overview of the methodology employed for the analysis of Pressurized Thermal Shock (PTS) behavior and its associated analyses. A thorough PTS analysis necessitates the consideration of fluid mixing behavior, various types of stress generation due to thermal shock, and fracture analysis using a standard crack model. In the context of analyzing the VVER-1200 model, this chapter encompasses the description of its material properties, general design data, and all other relevant necessary data and records. Furthermore, this chapter elucidates the underlying theory and mathematics used as the foundation for analysis, employing tools such as ANSYS FLUENT and ANSYS Mechanical. It also delves into the analytical calculation of the nil ductility temperature, assisted by nil ductility temperature shift calculations as per ASME and USNRC regulatory guidelines. These calculations are instrumental in determining the limit for evaluating fracture toughness in the VVER-1200 Reactor Pressure Vessel (RPV) wall.

CHAPTER 4
RESULTS AND DISCUSSION

4.1 3D Computational Fluid Dynamics Result & Analysis

During PTS event, the non-uniform mixing phenomena of cold ECCS water and hot primary coolant is analyzed by 3D CFD simulation modelling with the help of ANSYS FLUENT. The simulation is done for 200 sec to get the temperature distribution of inner vessel due to mixing of coolants. The study investigated the 3D temperature profile evolution along the downcomer and inner wall surface of the RPV during the coolant mixing event due to transient event. The initial and boundary conditions are described in details in Methodology part. The ECCS water is injected into the primary circuit within the 225 mm diameter nozzle. From Figure 4.1, it shows that the initial center line temperature of inlet nozzle of primary circuit which is around 298 ◦C or 571 K decreases with time. And after 200 sec simulations, it reaches to 167◦C or 440 K for cold (10 ◦C) ECCS water mixing.

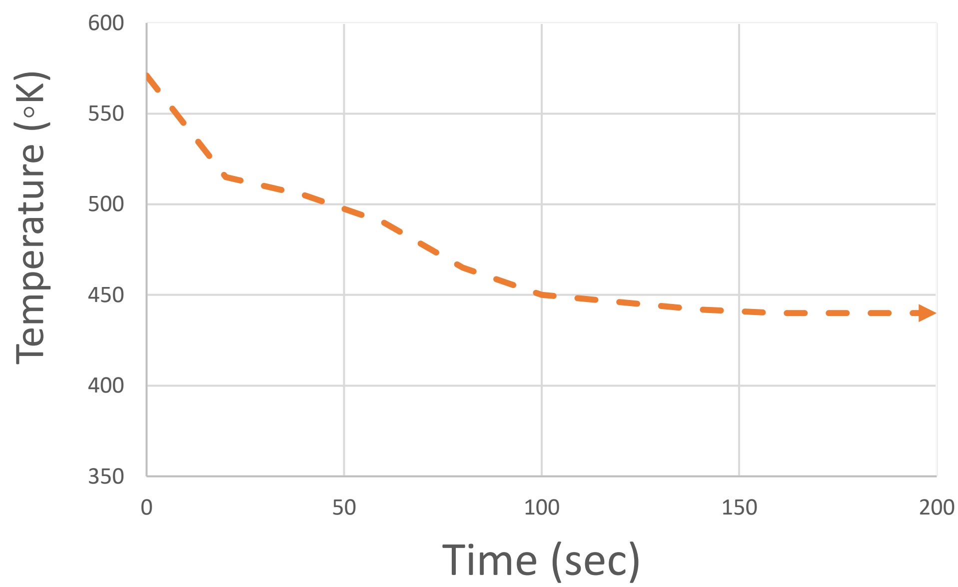


Fig. 4.1: Center line temperature of inlet nozzle region with respect to time (sec)

3D temperature evolution of fluid at different time spans up to 200 sec is analyzed with the constant pressure of 7 MPa. No pressure drop & rise event is considered in this simulation approach. Cross sectional plane view at different time step of temperature contour is showed in Figure 4.2. The cold ECCS water entering into the primary loop with the velocity of 0.4 m/s and how it cools down the stagnant hot primary coolant in downcomer region can be demonstrated by Figure 4.2. The minimum temperature can be found at the inlet of ECCS.

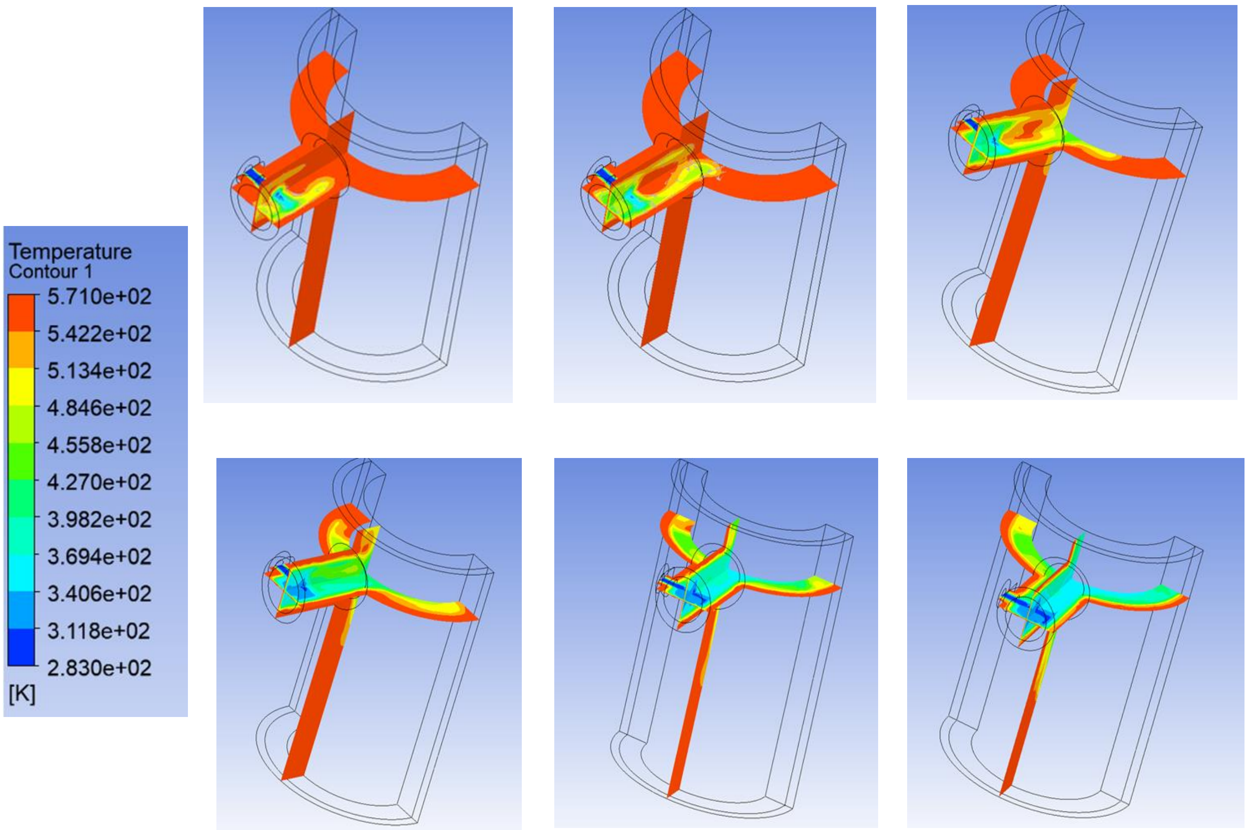


Fig. 4.2: Cross sectional plane contour at different time step

From Figure 4.3, temperature development of the fluid region in downcomer and cold leg part due to ECCS injection up to 200 sec can be obtained from simulation results. The temperature distribution in fluid region is non-homogeneous. The lowest temperature obtained in this analysis is found to be 361.4 K or 88.4 ◦C after 200 sec. And the maximum temperature difference reaches at 210 ◦C at the fluid region. Temperature in the whole computational domain varies from 10 °C up to 298°C. Surface temperature of inner RPV wall is obtained in ANSYS FLUENT simulation. Temperature development in fluid part due to ECCS injection and mixing, the inner wall also undergoes a temperature change and experience thermal gradients. Figure 4.4 shows the temperature evolution contour of RPV wall surface at different timesteps up to 200 sec. It is observed that up to 50 sec, the cooling effect is only shown at the cold leg part and nozzle area.

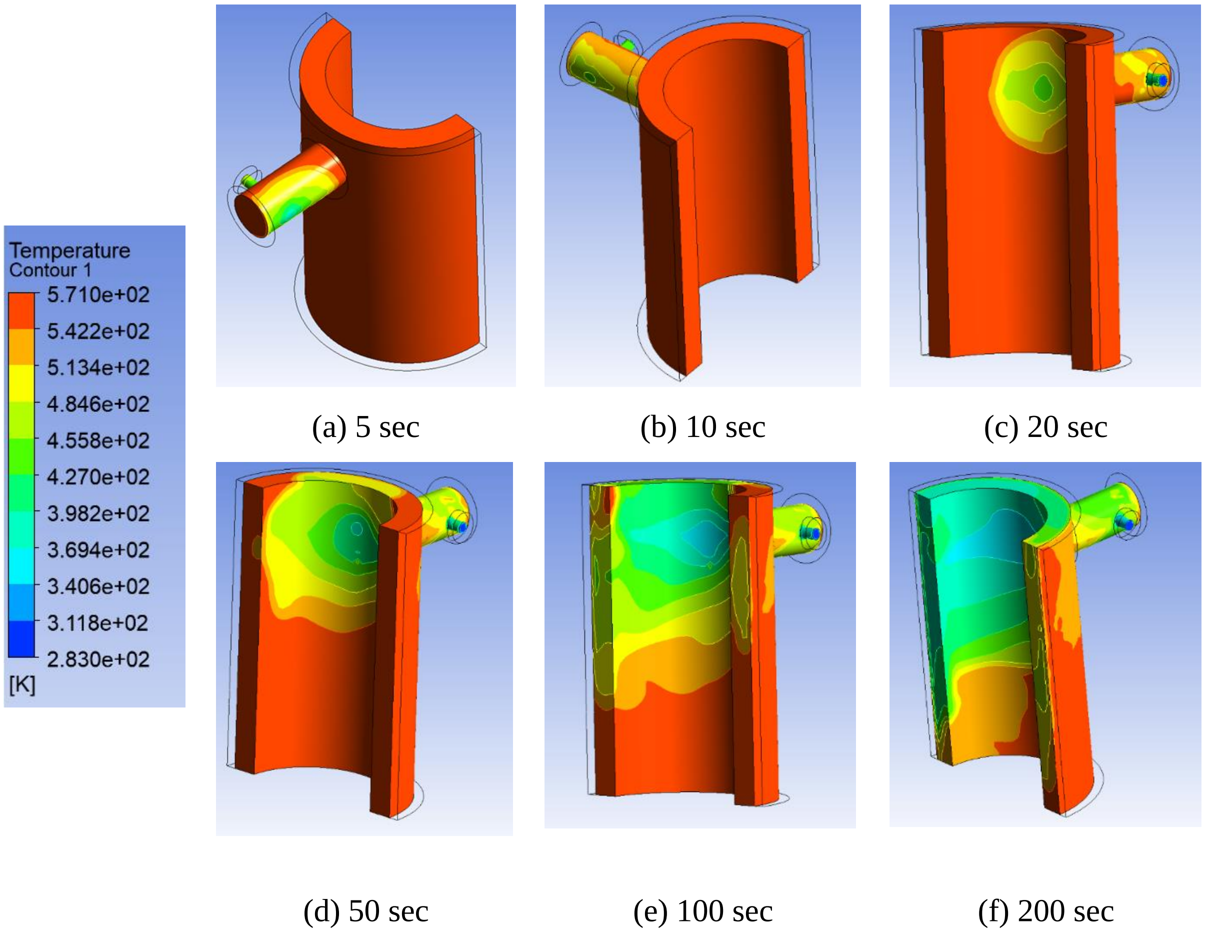


Fig. 4.3: Temperature evolution contour of fluid profile at different time step

Additionally, no cooling plume is experienced on the inner surface of the RPV. Two cooling plumes are observed on the inner surface of the RPV wall at 100 seconds and afterwards. Thermal gradients at the inner wall increase with an increase in ECCS injection mixing time and total mass injection. The temperature variation at inner wall surface also varies within 10 °C to 298°C.

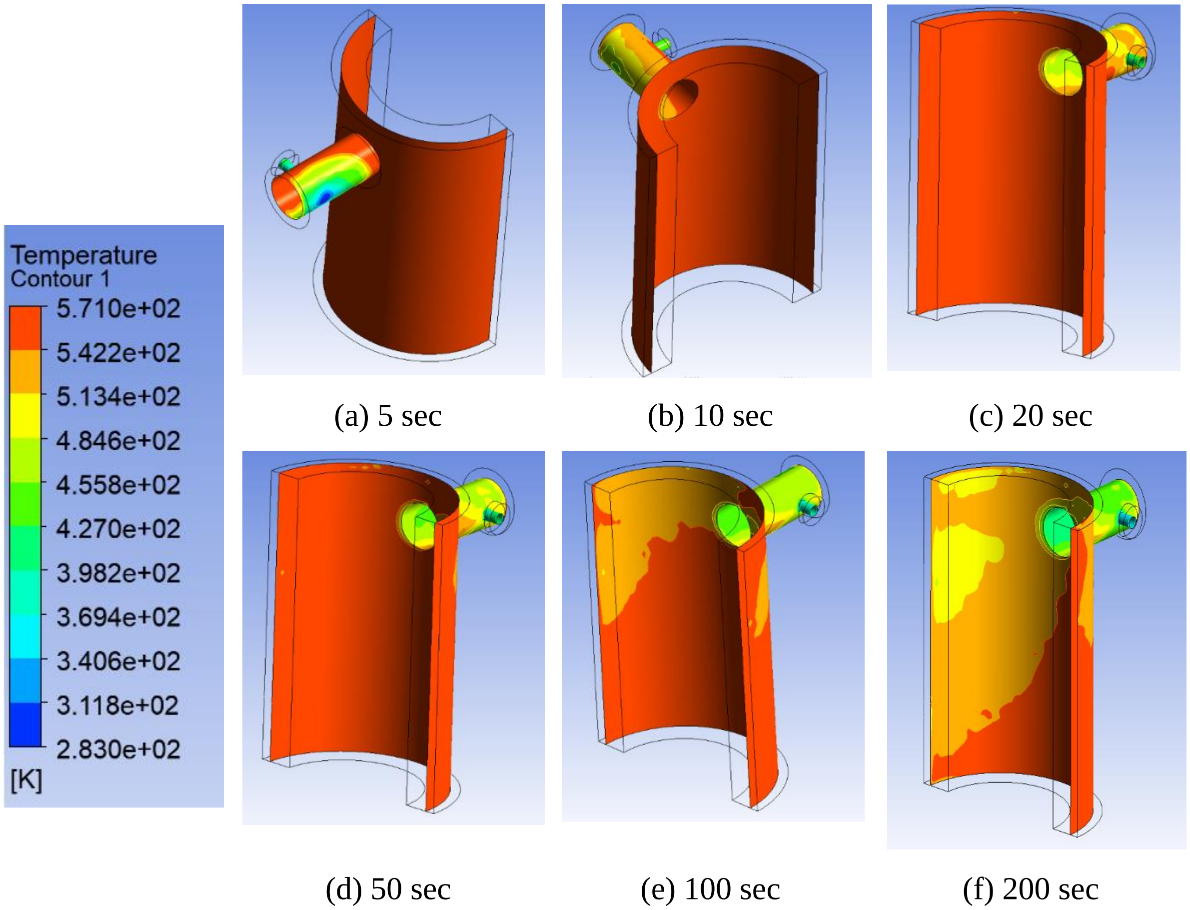


Fig. 4.4: Temperature evolution contour of RPV wall surface at different time step

4.2 3D Stress Profile Analysis of RPV Wall

The 3D CFD results obtained from the ANSYS FLUENT is coupled with ANSYS Mechanical. It is done to evaluate the stress profile at RPV inner wall during the ECCS injection at cold leg for a MSLB LOCA transient case. The thermal shock results in stress generation at the internal surface of the RPV wall. Stress analysis by ANSYS Mechanical is done by Finite Element Method (FEM) where the 3D model is discretized with tetrahedron elements and solve numerically to obtain the results. For stress analysis, time-dependent temperature is imported from CFD results.

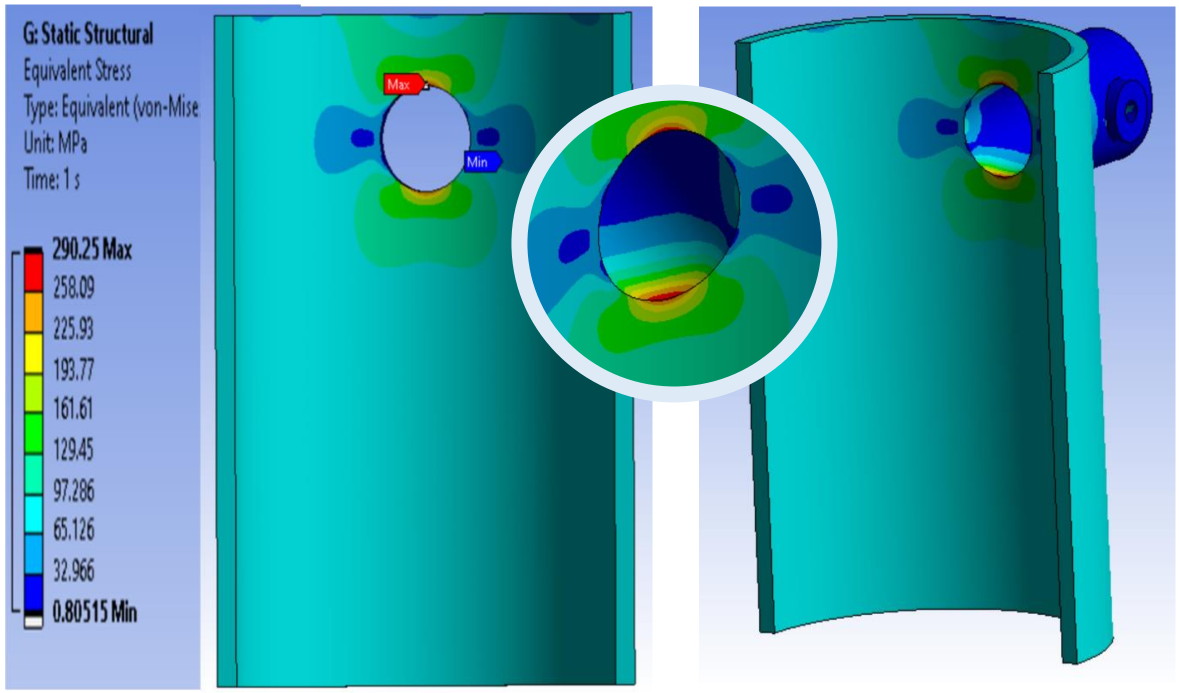


Fig. 4.5: Equivalent (Von mises) stress contour of symmetric RPV model at 7 MPa pressure

And pressure rise and drop event and different constraints are applied as input parameters to solve the model for stress calculations. Stress analysis for this PTS event is done for VVER-1200 considering material properties in ANSYS Mechanical. Meshed structure of CFD and Mechanical analysis are not same Discretized elements and nodes in Mechanical analysis automatically adapts the temperature results from CFD results by using python scripts. The pressure applied at the RPV is distributed uniformly while the temperature profile is result for the heat transfer problem. The evolution of equivalent (Von mises) stress is depicted in Figure 4.5 and Figure 4.6 at 7 and 16.2 MPa respectively. Equivalent (von mises) stress analysis is important to understand plastic deformation behavior under various loading conditions of ductile material to evaluate the integrity. The stress distribution in RPV for 7 MPa due to the mixing of ECCS injection with primary coolant has range from 0.80515 MPa to 290 MPa. And, the stress distribution at RPV increases with increased pressure at 16.2 MPa for the same fluid mixing phenomena and has a range of stress between 1.77 MPa to 66 MPa. For both of the case maximum stress generate at the corner edge of the nozzle region.

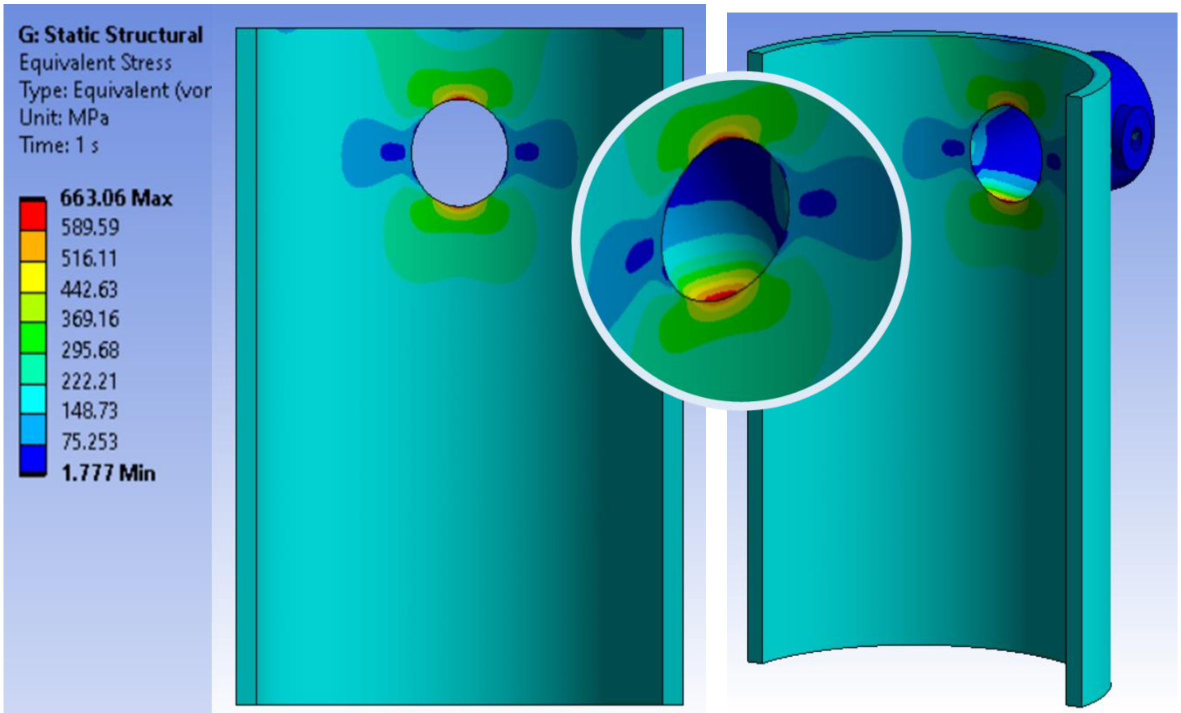


Fig. 4.6: Equivalent (Von mises) stress contour of symmetric RPV model at 16.2 MPa pressure

In this result, the pressure drop during the PTS event is not taken into account. The analysis is conducted at 7 MPa at the end of 200 seconds and is also repeated for 16.2 MPa. This indicates a rapid pressure change from 7 MPa to 16.2 MPa instantly and associate results are shown here. From the Figure 4.5 and Figure 4.6, it is obtained that, the maximum generated Von-mises stress didn't exceed the tensile yield strength and ultimate yield strength limit at 7 MPa but at 16.2 MPa, generated stresses exceed the limit for both parameters.

As a result, it can be concluded here that due to exceeding the ultimate yield strength at 16.2 MPa in a sudden pressure spike, several potential situations may arise. RPV material can exhibit significant plastic deformation and necking might occur after reaching the ultimate yield strength limit. As the RPV materials are more ductile in properties, the chance of sudden catastrophic failure is less than the brittle materials. Exceeding the limit of ultimate yield strength by von mises stress, postulated crack may propagate under favorable conditions. The materials around the crack tips may tend to deform plastically which blunt the crack and slow its propagation. But during the event of DBTT shift and experience of coolant temperature less than NDT, can worsen the situation showing brittle material properties. Thus, sudden rise of pressure during ECCS injection for a transient event can compromise the structural integrity of the vessel.

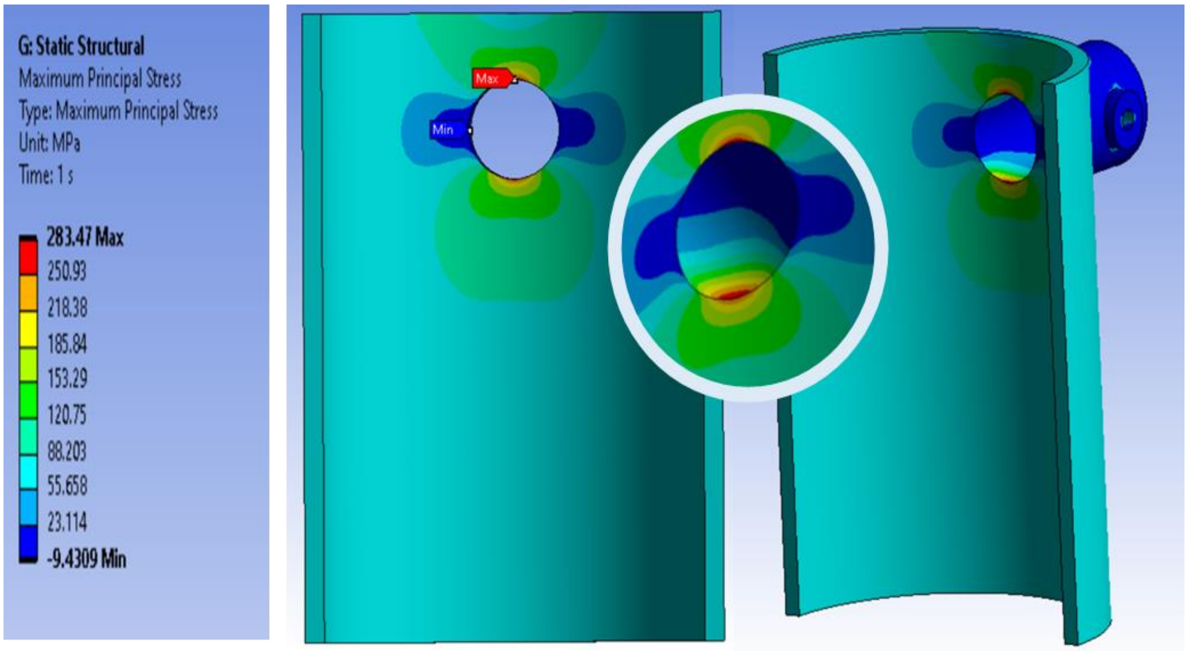


Fig. 4.7: Maximum Principal stress contour of symmetric RPV model at 7 MPa pressure

Maximum principal stress can evaluate specially the integrity of brittle materials. If the material of RPV obtain the NDT during any transient event, ductile to brittle shift can be observed and material may show brittle behavior. In this analysis, the maximum generated principal stress at 7 MPa didn't exceed the tensile yield strength limit depicted in Figure 4.7. And at 16.2 MPa, the limit of ultimate yield strength is exceeded shown in Figure 4.8. If the RPV material do not reach the DBTT at this time, ductile properties of material will be evident. From the result at 16.2 MPa, RPV material may likely undergo plastic deformation. It can also carry some more load after exceeding the yield strength due to its ability to undergo plastic deformation. Localized necking might occur in the high stress regions. The failure probability of the RPV will increase in plastic deformation until a critical point is reached. If the RPV material show the brittle behavior, material can be failed abruptly. Crack can initiate and propagate rapidly in areas where high stress concentrates. There might be no clear warning before failure, and the failure would be sudden and catastrophic.

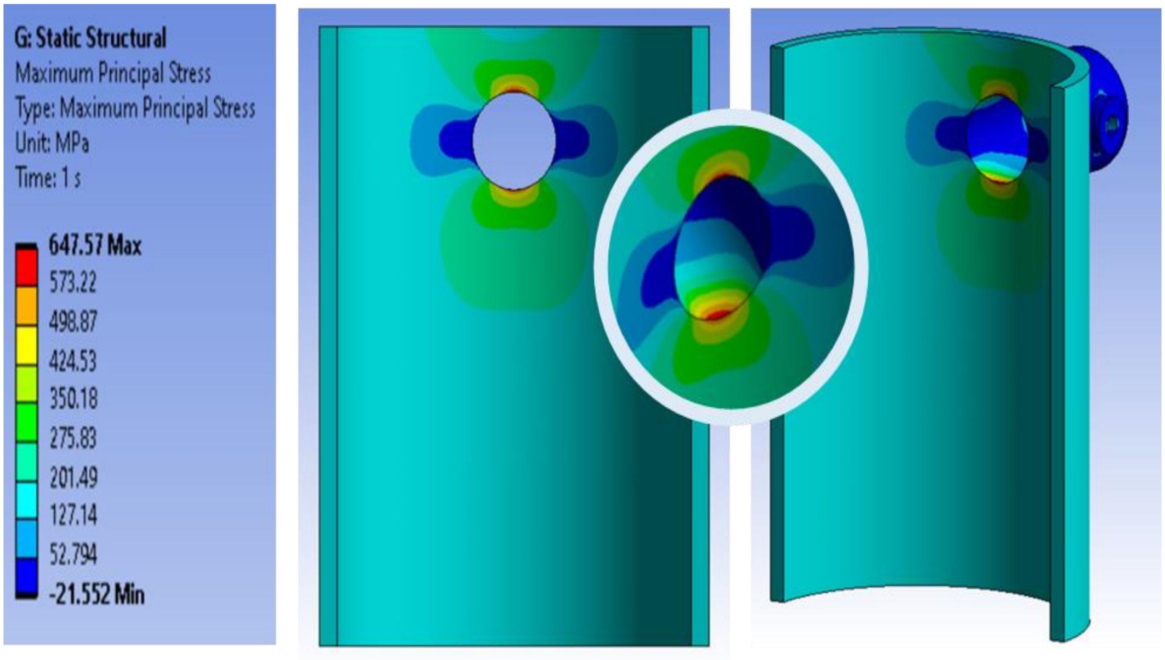


Fig. 4.8: Maximum Principal stress contour of symmetric RPV model at 16.2 MPa pressure

4.3 Validation of Stress Results

Results obtained from the ANSYS Mechanical on different stresses are validated with the analytical results. Hoop stress and longitudinal stresses during 7 MPa and 16.2 MPa pressure are obtained from ANSYS Mechanical and compared with the analytical results using the classical equation for hoop and longitudinal stress discussed in Chapter 03.

Table 4.1: Validation of Hoop stress simulation result with analytical result

HOOP STRESS			
Pressure	Analytical result	FEA result	Error
7 MPa	70 MPa	73.21 MPa	4.38%
16.2 MPa	162 MPa	164.38 MPa	1.44%

Table 4.2: Validation of longitudinal stress simulation result with analytical result

LONGITUDINAL STRESS			
Pressure	Analytical result	FEA result	Error
7 MPa	35 MPa	36.704 MPa	4.64%
16.2 MPa	81 MPa	84.57 MPa	4.22%

Resultant simulation values are very close to the analytical results and the errors are below 5%. Detailed results are described in Table 4.1 and 4.2 shows the good agreement of the correct calculation of simulation model.

4.4. Mesh Dependency Test

Mesh dependency test is done for ANSYS Mechanical part where generated stress in the inner part of RPV vessel is observed. For this research work, mesh dependency test is done for equivalent (Von-mises) stress and Mesh node no. during internal pressure of 7 MPa ECCS mixing phenomenon at PTS event. This analysis begins with coarse mesh and it is refined to a finer mesh for each run until results do not change significantly and depend on the mesh size anymore. From Figure 4.9, it can be observed that from the third point of mesh node no, the resultant equivalent (Von-mises) stress does not change significantly with changing mesh up to order of 5. This mesh dependency test provides critical criterion for ensuring the accuracy and reliability of simulation results.

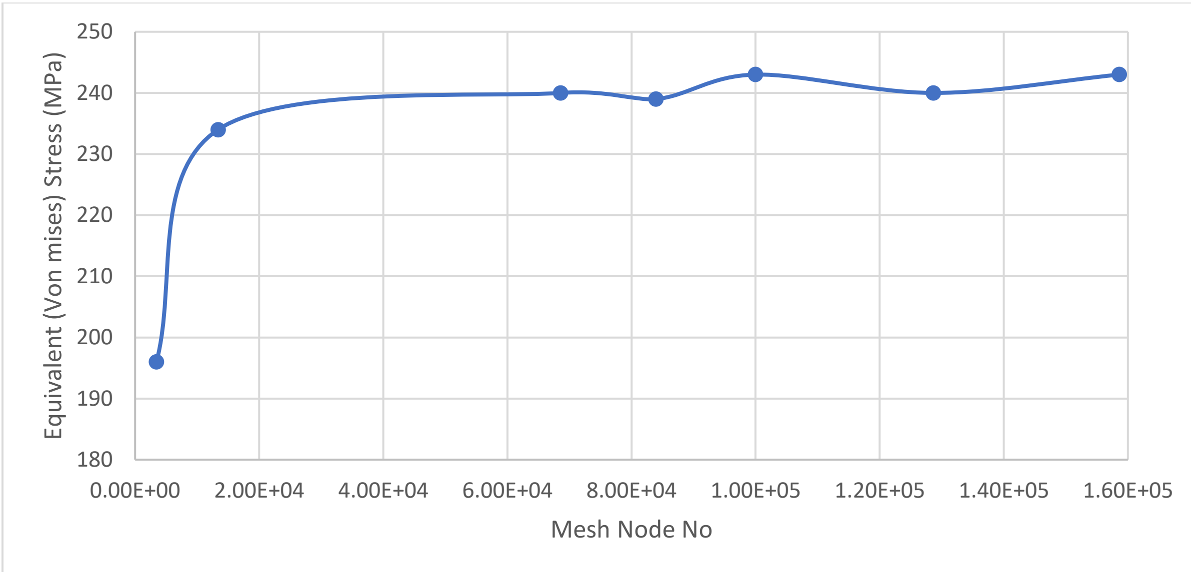


Fig. 4.9: Mesh Dependency Test

4.5 Stress analysis at different path of RPV

During the simulated transient event of MSLB LOCA, the stress variation along the RPV wall thickness due to different pressurized situation has been evaluated.

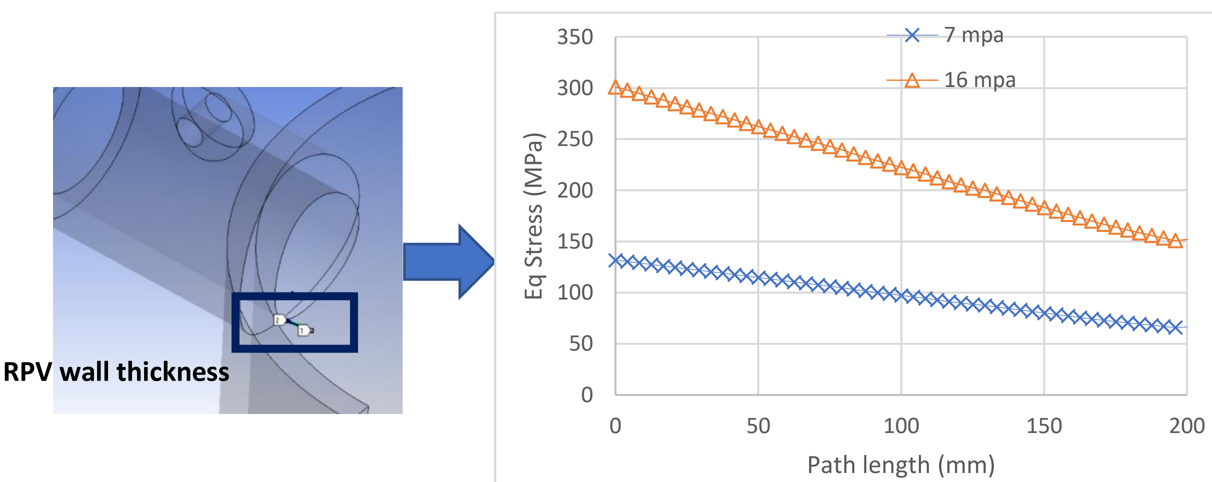


Fig. 4.10: Equivalent (Von mises) stress graph along the RPV wall below the inlet nozzle at 7 MPa & 16.2 MPa

As the study found that around the corner nozzle area the generated stress is significantly high, so the stress variation has been evaluated just below the corner of inlet nozzle of RPV inner wall. In this line of interest, at lower pressure, maximum equivalent (Von mises) stress is found 141 MPa and at higher pressure, stress is found around 310 MPa (Figure 4.10). Maximum stress equivalent (Von mises) stress is found at the inner surface wall for both pressurized situations. And equivalent stress curve almost linearly decreased along the vessel thickness of 196 mm. Figure 4.11 depicted the maximum principal stress generated at the RPV wall along the vessel thickness in the same region as Figure 4.10 and found similar trend of graph. Maximum principal stress is obtained 144 MPa and 314 MPa at 7 MPa pressure at 16.2 MPa pressure event respectively. Results of stresses are well below to the limit of tensile yield strength of vessel material, 490 MPa.

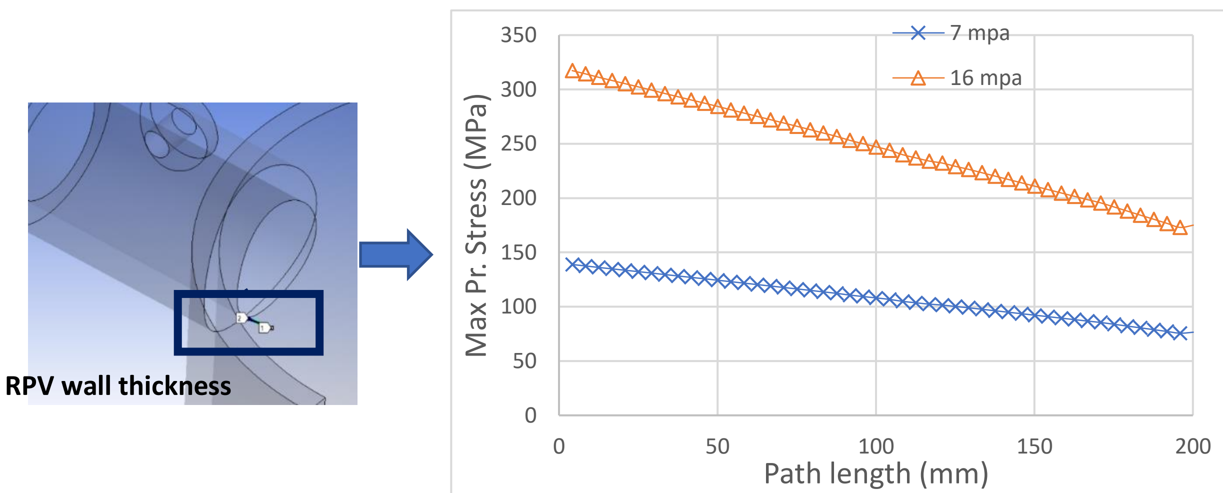


Fig. 4.11: Maximum Principal stress graph along the RPV wall below the inlet nozzle at 7 MPa & 16.2 MPa

The belt region of RPV inner part which experience the most extreme environment throughout the operating period, is also evaluated for equivalent stress along the vertical length. Equivalent stress gradually decreases along the length of vessel at belt line region. Peak stress is found 637 MPa at the nozzle corner of cold leg and gradually decreases. At around 800 mm below from the initial point, equivalent stress come to a constant value. The transient simulation is done for 200 sec, so fully mixing of coolants is not available. Figure 4.13 and 4.14 depicts the

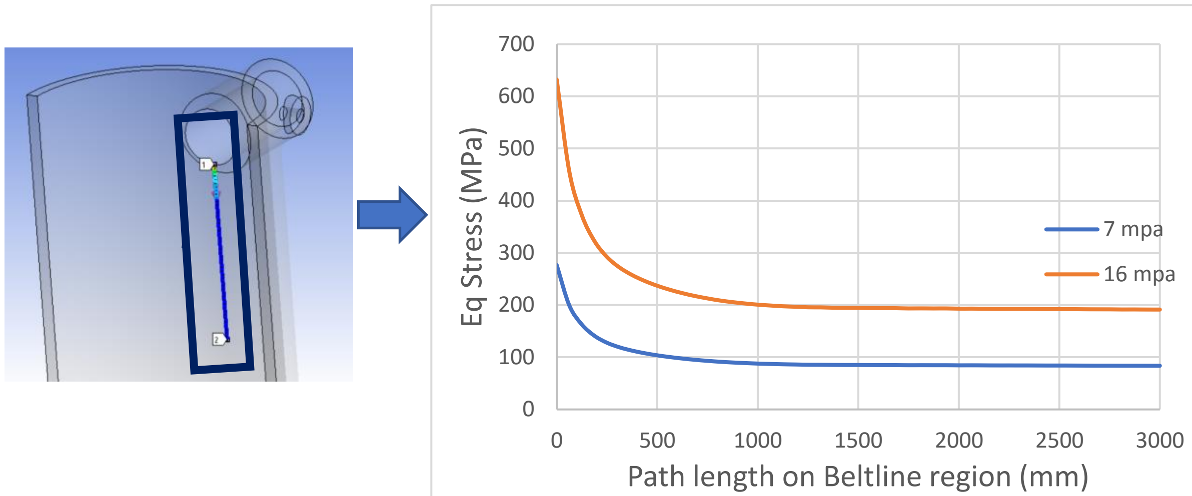


Fig. 4.12: Equivalent (Von-Mises) stress graph along the upper part of inner vessel part through the inlet nozzle section at 7 MPa & 16.2 MPa

equivalent stress results evaluated for the nozzle upper and lower region where maximum stress generated. These graphs gradually decrease with the path length of cold leg.

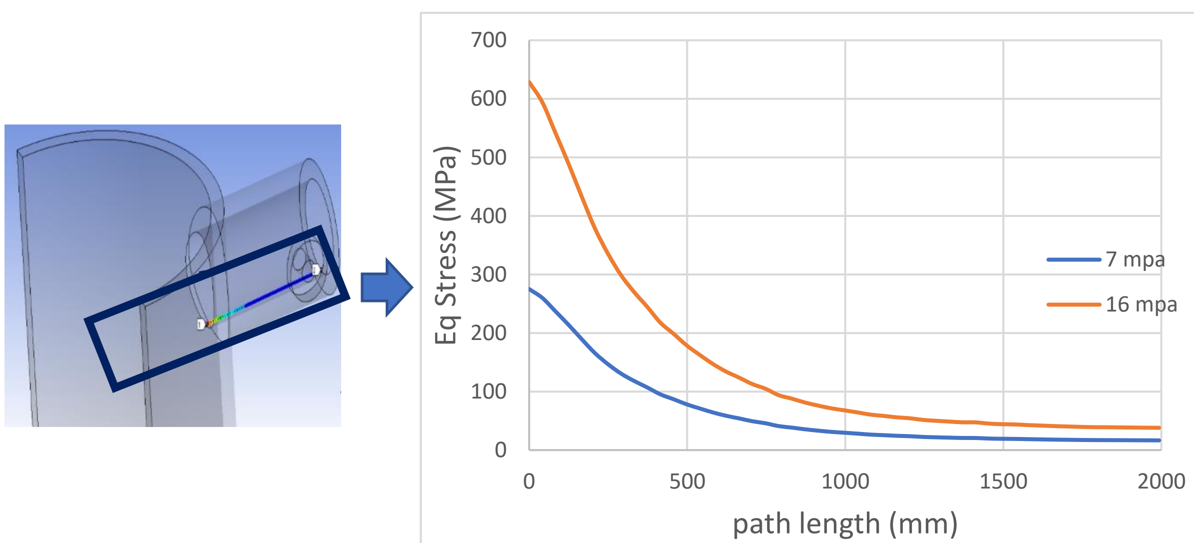


Fig. 4.13: Equivalent (Von-Mises) stress graph along the lower part of inner vessel part through the inlet nozzle section at (a) 7 MPa & (b) 16.2 MPa

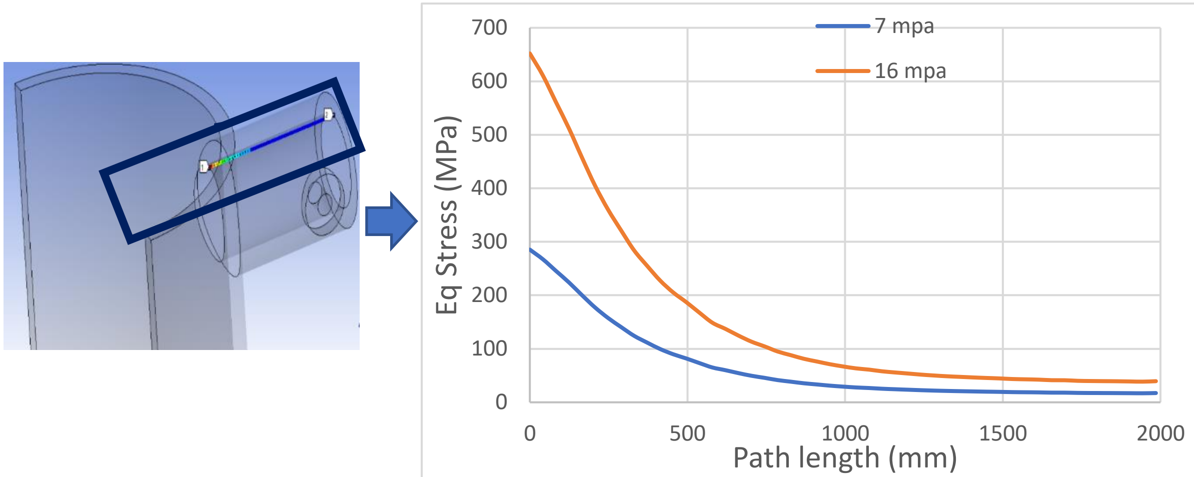


Fig. 4.14: Equivalent (Von-Mises) stress graph along the upper part of inner vessel part through the inlet nozzle section at (a) 7 MPa & (b) 16.2 MPa

4.6 Fracture assessment of postulated crack

Fracture assessment due to the PTS event has been evaluated for three crack cases that has been described in Chapter -03: Methodology part. The postulated semi-elliptical crack is inserted at most stress generated region of RPV which is the corner of nozzle part. Stress intensity factor (SIF) has been evaluated along the crack length for 7 MPa and 16.2 MPa during the end of ECCS injection for transient event. Figure 4.15, 4.16 and 4.17 depicts the SIF results of K_1 values for crack case I - III for above mentioned situation. For all the three cases, it is found that, maximum SIF has been generated for higher pressure that is 16.2 MPa and has been varied with the different crack profile.

4.6.1 Stress Intensity Factor (SIF) results for Crack: Case-I

For case- I, where, where aspect ratio, a/c=2/3, maximum SIF is obtained 194 MPa. m^1/2 at 16.2 MPa pressure. Maximum SIF locates at the nearest corner point of the nozzle where maximum stress has been generated.

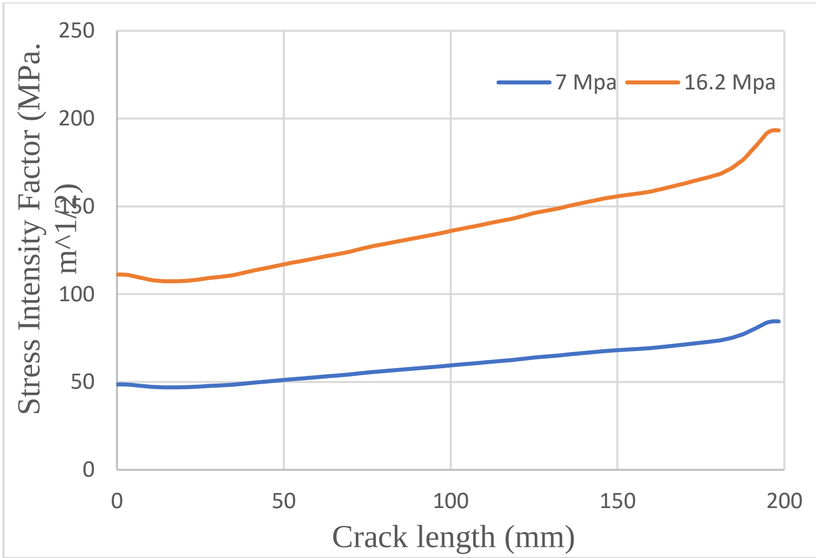


Fig. 4.15: SIF results at different pressure along crack length for Crack: case I

4.6.2 Stress Intensity Factor (SIF) results for Crack: Case-II

For case II, where, where aspect ratio, a/c=1/3, SIF increases with crack length and found 100 MPa. m^1/2 maximum at the middle of the crack position during 16.2 MPa and again decreases to 72 MPa. m^1/2 at the crack tip on the surface of the inner part. For 7 MPa pressure, SIF varies from 26 to 44 MPa. m^1/2 along the crack length.

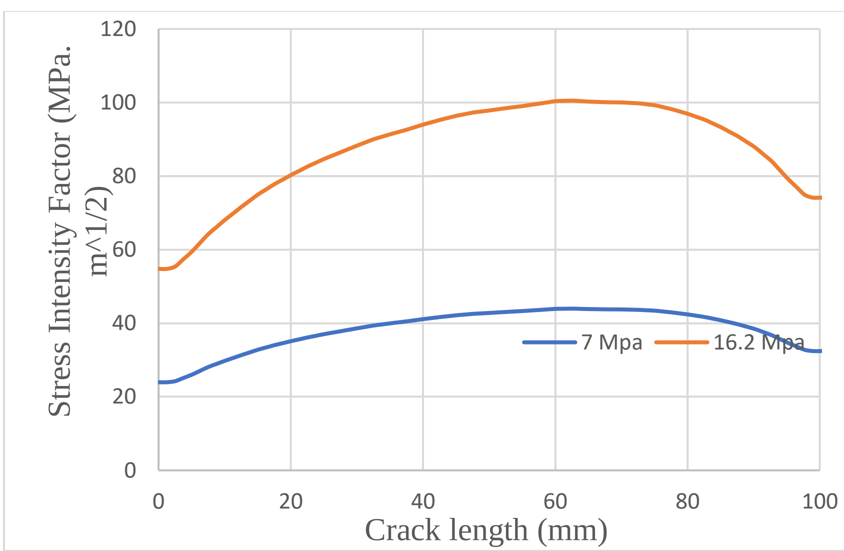


Fig. 4.16: SIF results at different pressure along crack length for Crack: case II

4.6.3 Stress Intensity Factor (SIF) results for Crack: Case-III

For case III, where aspect ratio, a/c=1/3, maximum SIF is found few millimeters before the crack tip of inner surface at the nozzle end. Maximum SIF is found 157 MPa. m^1/2 at 16.2 MPa. All these SIF values need to assess with the maximum allowable material toughness values to determine whether the crack will propagate or not. Material toughness value depends on several factors including ageing effects. Along with reactor operation, neutron embrittlement, rate of loading, temperature, high pressure, material composition, fatigue and other factors determine the material toughness. Mostly, neutron embrittlement and high temperature has large impact on changing the material toughness.

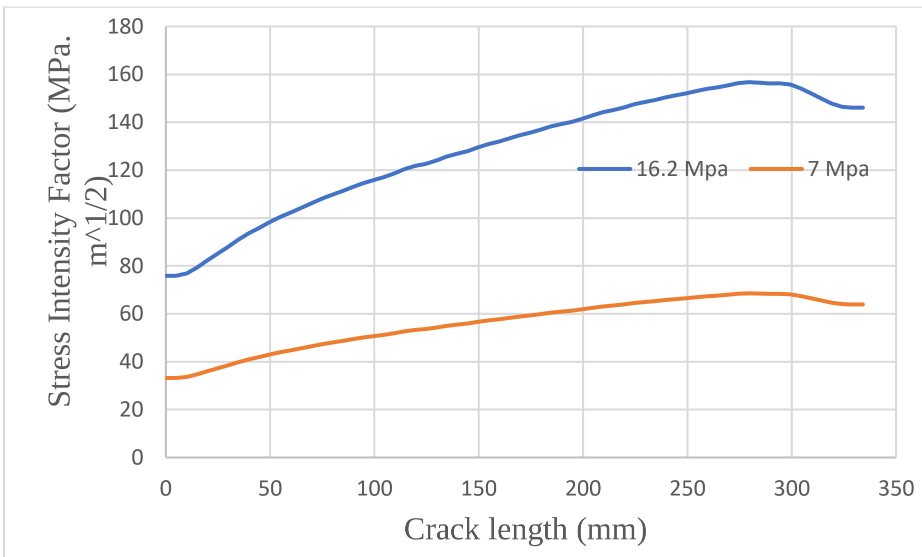


Fig. 4.17: SIF results at different pressure along crack length for Crack: case III

Again, if a postulated crack exists on the reactor pressure vessel (RPV) wall, it will lead to a significant concentration of stress at the crack point. The stress generated at the crack point increases as the pressure within the RPV rises. Figure 4.18 depicts the higher peak generated at crack point with respect to increased pressure.

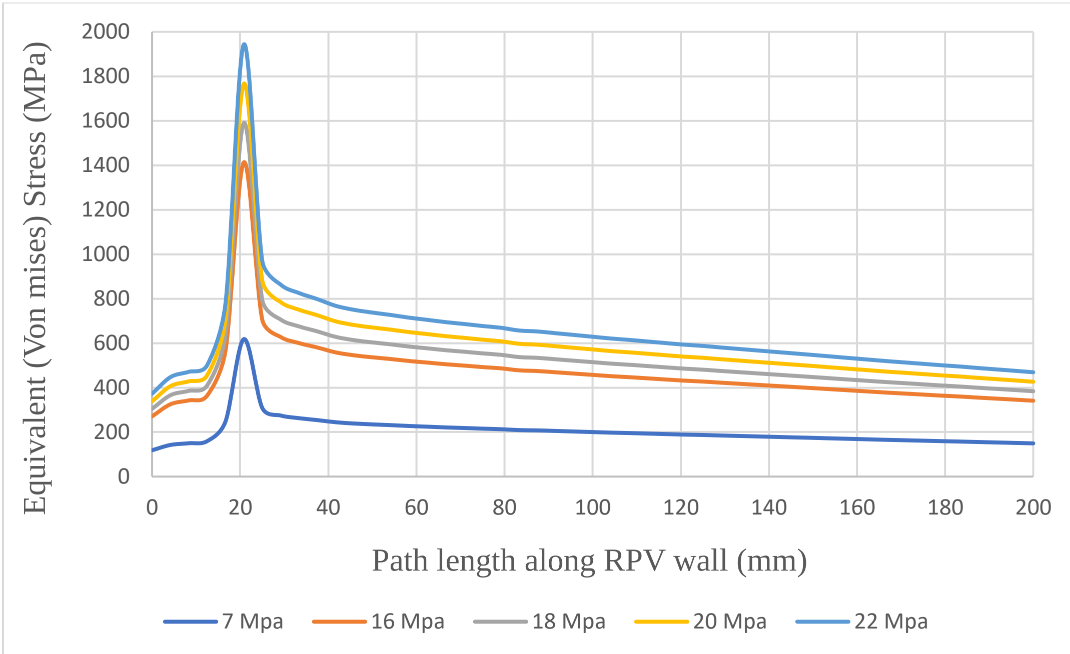


Fig. 4.18: Stress variation along the path length of RPV wall thickness for case-III postulated crack at different pressure level

4.7 Service life assessment of RPV during PTS

For service life assessment due to PTS event, it is conservatively considered as the fast neutron fluence subjected to the inner part of the reactor vessel. Fast neutron fluence also varies according to reactor design and development of reactor design with time. For VVER 1200, maximum fast neutron fluence at the end of the design life (60 years) is considered $4.22 \times 10^{19}\ n/cm^2$ where the nil-ductility reference temperature, $\Delta RT_{NDT}\ was\ 160\ \circ\ C$. Fracture toughness value, K_{IC} is calculated with the help of ASME code.

4.7.1 Crack : Case- I

For crack case- I, the analyses is done to assess stress intensity factor (SIF) for different pressure environment at different lifetime e.g., initial stage, after 60 years. From Figure 4.19, transient stress intensity factor, K_I obtained for different pressure environment compared with the fracture toughness limiting values during PTS event. SIF values increased with respect to increasing pressure. And the SIF values are obtained at the temperature of 100 ◦C where

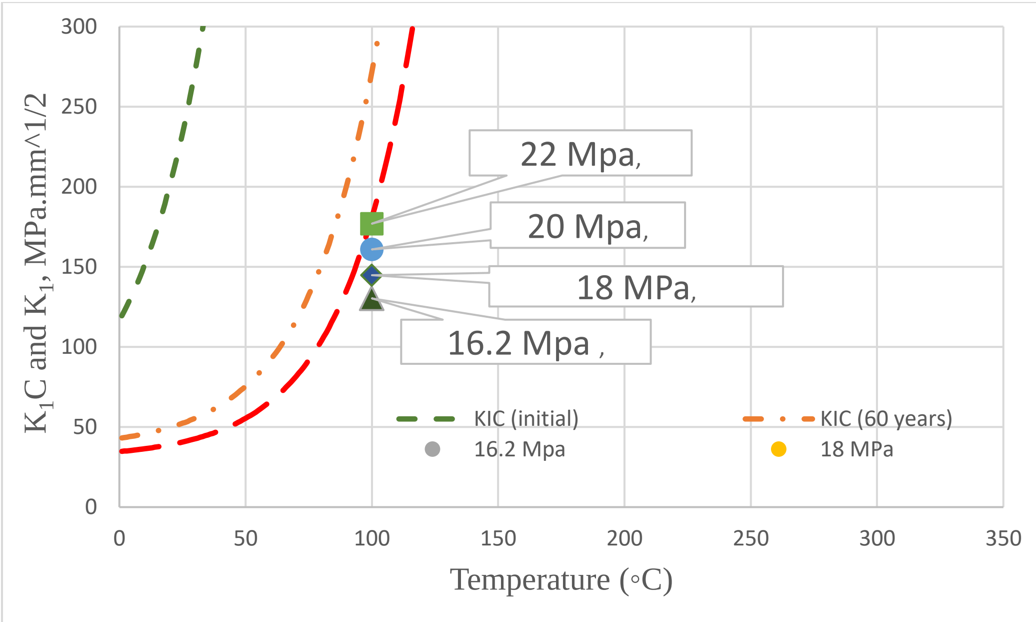


Fig. 4.19: K_1 results for crack: case - I in different pressure environment at different lifetime K_{IC} values

resulting value indicates the maximum allowable critical temperature of brittleness. Before sudden rise of the inner pressure, the lowest temperature of the system was taken 100◦C. And, after the initiation of sudden pressure and reach to 16.2 MPa, the SIF value found to be 130.33 Mpa.m^1/2. When the system pressure increases to 18, 20 and 22 MPa respectively, the SIF values obtained from the simulation are 144.78, 160.88 and 176.98 Mpa.m^1/2 respectively. From Table 4.3, the determined SIF values for crack case- I, compared with the fracture toughness values, K_{IC}, it is determined whether K_I values exceed K_{IC}. From Figure 4.19, it can be observed that only K_I value of 22 MPa pressure exceed the K_{IC} and propagate the crack for this crack case.

Table 4.3: Comparison of maximum allowable critical temperature of brittleness to assess service life in transient event for crack: case I

Pressure, MPa	Stress Intensity Factor (SIF), Mpa.m^1/2	Status at Initial stage DBTT= -35 ◦C	Status at DBTT after 60 years operation	maximum allowable critical temperature of brittleness, ◦C
16.2	130.33	Not Exceed	**Not Exceed**	60 ◦C
18	144.78	Not Exceed	**Not Exceed**	56 ◦C
20	160.88	Not Exceed	**Not Exceed**	53 ◦C
22	176.98	Not Exceed	**Exceed**	50 ◦C

So, during PTS scenario with increasing pressure, up to 20 MPa the crack profile of crack case-I will not propagate for not exceeding the fracture toughness values. For this considered design of VVER 1200 and its material properties, according to the crack case-I, it can withstand up to 20 MPa pressure during PTS scenario. If the design improvement requires to withstand 22 MPa pressure in a PTS event, the maximum allowable critical temperature of brittleness need to limit within 50◦C at the end of design life for reactor for case-I.

4.7.2 Crack : Case- II

For case-II crack profile, the assessment is done to evaluate the design life of RPV inner wall at the end of design life (60 years) during PTS event. From Figure 4.20, it can be observed that sudden pressure rises during ECCS mixing phenomenon due to LOCA event, maximum SIF values are below the fracture toughness values and not liable for crack propagation.

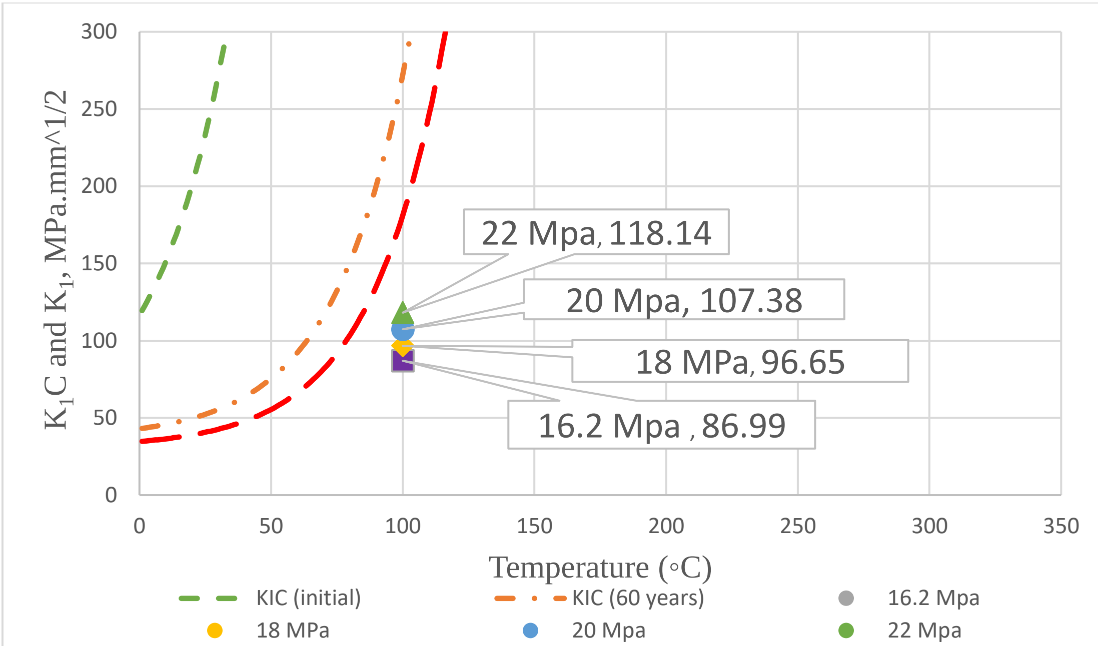


Fig. 4.20: Fracture toughness results for crack: case II in different pressure environment

Sudden pressure rise is considered from 16.2 MPa to 22 MPa. For this crack profile, lowest maximum allowable critical temperature of brittleness value can be obtained 64 ◦C according to the simulation result.

Table 4.4: Comparison of maximum allowable critical temperature of brittleness to assess service life in transient event for crack: case II

Pressure, MPa	Stress Intensity Factor (SIF), Mpa.m^1/2	Status at Initial stage DBTT= -35 ◦C	Status at DBTT after 60 years operation	maximum allowable critical temperature of brittleness, ◦C
16.2	86.99	Not Exceed	Not Exceed	78 ◦C
18	96.65	Not Exceed	Not Exceed	73 ◦C
20	107.38	Not Exceed	Not Exceed	69 ◦C
22	118.14	Not Exceed	Not Exceed	64 ◦C

Table 4.4 is showing the generated maximum SIF values and maximum allowable critical temperature of brittleness whereas they either exceed the fracture toughness values for 16.2, 18, 20 and 22 MPa respectively.

4.7.3 Crack : Case- III

For case-III crack profile, the assessment is done in the same procedure to evaluate the design life of RPV inner wall at the end of design life (60 years) during PTS event. From Figure 4.21, it can be observed that sudden pressure rise at 20 & 22 MPa in primary loop generates SIF values which exceed the fracture toughness values. It can cause crack propagation for this crack profile. For 16.2 & 18 MPa pressure rise event, the SIF values do not exceed the fracture toughness values. For this crack case, maximum allowable critical temperature of brittleness is found 46◦C at 22 MPa pressure environment during PTS event. For design improvement of RPV to withstand 22 MPa pressure for this crack case, DBTT value need to be within 46◦C temperature. From Table 4.5, maximum SIF values and maximum allowable critical temperature of brittleness whereas they either exceed the fracture toughness values for 16.2, 18, 20 and 22 MPa respectively can be found.

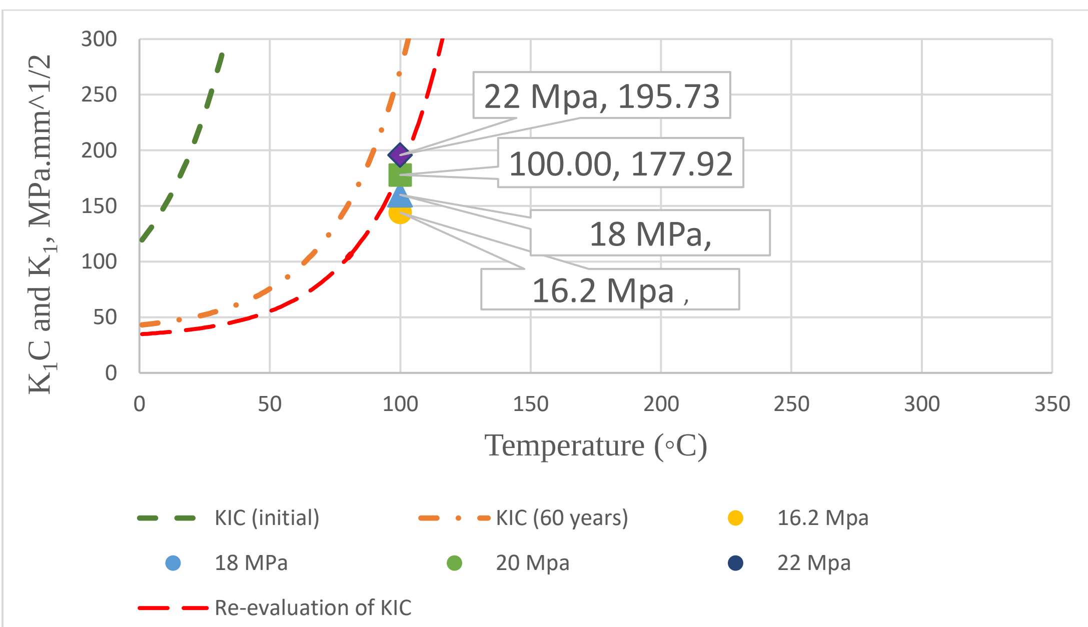


Fig. 4.21: Fracture toughness results for crack: case III in different pressure environment

Table 4.5: Comparison of maximum allowable critical temperature of brittleness to assess service life in transient event for crack: case III

Pressure, MPa	Stress Intensity Factor (SIF), Mpa.m^1/2	Status at Initial stage DBTT= -35 ◦C	Status at DBTT after 60 years operation	maximum allowable critical temperature of brittleness, ◦C
16.2	144.12	Not Exceed	Not Exceed	56 ◦C
18	160.03	Not Exceed	Not Exceed	53 ◦C
20	177.92	Not Exceed	Exceed	49 ◦C
22	195.73	Not Exceed	Exceed	46 ◦C

4.8 Summary

This chapter presents the results of the analysis and assessments conducted to evaluate the structural integrity of the VVER-1200 Reactor Pressure Vessel (RPV) throughout this study. The analysis involved several key steps:

I. Utilizing ANSYS FLUENT, Computational Fluid Dynamics (CFD) simulations were performed to determine the mixing temperature at 200 seconds into the Loss of Coolant Accident (LOCA) transient, specifically in the belt line region where the Emergency Core Cooling System (ECCS) water interacts with the hot primary coolant.

II. Subsequently, equivalent stress and maximum principal stresses were evaluated to assess the impact of the thermal shock associated with the previous CFD work.

III. Finally, fracture assessments were conducted for three different crack profiles according to various standards. This assessment considered hypothetical high-pressure transient LOCA scenarios followed by PTS events.

Upon completion of these assessments, the structural integrity of the RPV was evaluated for its entire design life of 60 years. Additionally, the maximum allowable temperature to prevent brittleness in the RPV was determined. This chapter provides a comprehensive overview of the methodology, analysis, and calculations performed to assess the VVER-1200 RPV's integrity under these extreme conditions.

CHAPTER 5
CONCLUSIONS AND FUTURE RECOMMENDATIONS

5.1 Conclusions

Pressurized Thermal Shock (PTS) analysis for a Main Steam Line Break (MSLB) LOCA in a VVER-1200 reactor has been conducted for the first time in our country. This analysis, as indicated in the case study, aids in assessing and evaluating the integrity of the Reactor Pressure Vessel (RPV) during a PTS event. It takes into account material properties, Ductile-to-Brittle Transition Temperature (DBTT) values, and neutron fluence results at various stages of the reactor's service life. During a PTS event, the evaluation of pre-existing vertical surface cracks to monitor their propagation and the determination of the maximum allowable critical temperature for brittleness play vital roles in maintaining the safety and integrity of the reactor.

Based on this research work, the following conclusions can be derived:

I. For a generic VVER-1200 reactor model, a LOCA-induced Pressurized Thermal Shock (PTS) scenario is analyzed, focusing on the temperature distribution within the Reactor Pressure Vessel (RPV) and the cold leg region. The computational 3D model is assumed to be symmetric and analyzed accordingly for this study. The mixing of hot stagnant coolant in the RPV at 298°C with cold water from the Emergency Core Cooling System (ECCS) at 10°C is simulated using ANSYS FLUENT. After a transient analysis lasting 200 seconds, the temperature at the center point of the cold leg stabilizes at 167 °C (440 K). The analysis also reveals the presence of two cooling plumes around the nozzle area of the RPV wall, indicating the cooling pathway for the vessel wall.

II. The stress profile of the RPV wall is obtained during the PTS scenario induced by a LOCA event. Stress profiles are obtained for pressures ranging from the initial low pressure of 7 MPa to the higher pressure of 16.2 MPa, which is the design pressure for the VVER-1200. The observed results indicate that the maximum equivalent (von Mises) stress and principal stress are generated at the vertical edge of the nozzle region. Specifically, it is observed that the maximum equivalent stress and principal stress reach their peaks at 16.2 MPa, coinciding with a sudden rise in system pressure. At this point, the generated stresses measure 663 MPa and 647 MPa, respectively. Additionally, it's worth noting that the stress values decrease in all directions, including through the vessel thickness, from the maximum stress zone at the nozzle.

III. An investigation has been conducted to assess the service life of the Reactor Pressure Vessel (RPV), taking into account the potential presence of cracks on the inner wall surface during a Pressurized Thermal Shock (PTS) event. Three different crack profiles have been considered and subsequently simulated to obtain the Stress Intensity Factor (SIF) values. These SIF values have then been rigorously evaluated against the limiting fracture toughness values, accounting for neutron fluence and results from the Ductile-Brittle Transition Temperature (DBTT) assessments, spanning from the initial years of service to the intended design service life of 60 years.

IV. The investigation encompassed a range of pressures, extending from the design pressure to higher pressures up to 22 MPa. The results of this analysis indicate a noteworthy trend: SIF values increase in tandem with higher pressure conditions within the system during PTS events. Additionally, the investigation has allowed for the determination of the maximum allowable critical temperature for brittleness, serving as a crucial benchmark.

V. It is evident from this investigation that in higher primary pressure systems within the RPV during PTS events, there is a heightened risk of the propagation of postulated vertical cracks. In conclusion, it is necessary to modify and enhance the RPV material by decreasing the critical temperature for brittleness or reducing neutron fluence. This will enable it to withstand the elevated primary high-pressure conditions during PTS events.

VI. This study will assist operators and regulators in identifying and evaluating the comprehensive behavior of system parameters during a PTS event within the service life and potentially even beyond the service life.

5.2 Future Scopes

In future, this work can be enrich with the help of using more realistic code development and more dedicated software regarding PTS event. Experimental results of such events and two phase flow can be analyzed through this methodology can also be done in future. Moreover, the following improvements can be approached.

More practical assumptions

During the simulation, the computational model of the Reactor Pressure Vessel (RPV) was simplified to a hollow, thin cylinder with a cold leg, whereas a real-world reactor possesses a more intricate geometry, with various types of elements attached to it. In the stress calculations, the influence of this complex geometry and the presence of these elements were not taken into account. Furthermore, the modeling did not incorporate considerations for bending and the weldment's corners. In future, these factors and considerations can be incorporated for a more comprehensive and accurate representation.

Consideration of System Environment Change with Time

In this simulation approach, the system environment, such as pressure, is treated as a constant during the CFD analysis. However, during PTS analysis, the primary system pressure undergoes rapid changes over time, along with other system environment parameters. In the future, more advanced codes or simulation software may be employed to account for variations in pressure, temperature, and other relevant system environment factors, thereby yielding more realistic simulation results.

Inclusion of Neutronics analysis and coupling with the analysis

Neutronics analysis of the VVER-1200 is of utmost importance for comprehending and assessing the integrity of the RPV across its various service life stages. In this study, neutron fluence calculations have been conducted, taking into account different reference and analytical calculations. The inclusion of neutronics analysis in this work would contribute to a better understanding and yield more accurate results.

Simulation of other relevant transient scenarios and develop a dataset

Additional relevant transient scenarios of PTS can be simulated, leading to the development of a comprehensive dataset for PTS evaluation across different service life stages and conditions. Such an effort would prove invaluable in assisting both operators and regulators in effectively addressing PTS transient scenarios.

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