FAILURE PROBABILITY ANALYSIS OF A PWR PRESSURE VESSEL NOZZLE BY USING PROBABILISTIC FRACTURE MECHANICS CODE

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Abstract

This research focuses on failure probability of nuclear grade steel considering an arbitrary crack, which was employed in the PWR pressure vessel and nozzle region construction. Nuclear grade steel materials, "SA-533 Gr.A Cl.1" which is often used in PWR pressure vessel constructions, was taken into consideration for this study. Because they can absorb high temperatures (about > 330°C), enormous pressures (about > 16.2 MPa), and significant levels of neutron flux throughout the operating period, the inner wall of pressure vessels and nozzle structures is vulnerable to failure. Our main objective is to ascertain the probability of failure of the arbitrary crack significant criteria, such as high temperature, pressure in operational and transient situation. We calculated total stresses (Longitudinal, Thermal, Radial, etc.) on the crack point and later we assessed the failure probability of the PWR pressure vessel in both operational and transient condition using PFM code.

Keywords: PFM (probabilistic fracture mechanics), Nozzle, PWR (Pressurized Water Reactor), Operational condition, Transient condition

1. Introduction

The Russian VVER 1200 generation III+ reactor will be used in the nuclear power plant that the Bangladeshi government plans to construct close to Ishwardi Pabna. The ferritic low alloy steel reactor pressure containers for the VVER 1200 reactor were made in Russia. Because of its special properties—ductility, corrosion resistance, cryogenic toughness, strength, and hardness austenitic steel is used for RPV cladding. As a compromise for the first generation of VVER reactors, the inner surface is constructed of an anti-corrosive material, such as stainless steel, to shield it from the corrosion environment [1]. Austenitic steel affects the behavior of RPV fault initiation in addition to its unique properties. With the construction of Bangladesh's Rooppur Nuclear Power Plant, these codes of the probabilistic risk assessment of the RPV are a growing field in power plant engineering [2] [3].

A reactor pressure vessel is a vertical pressure vessel (one with a top head) that contains the core, control rods, and in-core instrumentation sensors. Reactor pressure vessels (RPVs) are one of the most crucial components of pressurized water reactors (PWRs), and they get embrittled after extended periods of operation under hostile conditions [4].

The nozzle area is important in a variety of applications because it influences flow rate, velocity, and pressure. In fluid mechanics, it is used to regulate the discharge of a fluid, which affects parameters such as coolant flow, pressure, neutron flux, fuel element cooling, and reactor control [5] [6].

Initiating a fracture in a nuclear RPV is inevitable and can lead to catastrophic situations. This study examines the onset and propagation of fractures in RPV cladding material under various design stress and fracture toughness conditions using a probabilistic approach [7]. Following then, the possibility of cumulative crack failure was seen. This study aims to analyze the crack propagation rate and failure probability of the RPV material [6].

2. Design and Modeling

According to proper design parameter we designed a model of reactor using the SolidWorks software after that we open that designed model in the ANSYS software and inserted the material properties and chemical composition there. Then we cutdown the area of our interest from the designed reactor.

2.1 RPV And Nozzle Material Characteristics

The nozzle shell region of a pressurized water reactor (PWR) is normally built of high-strength, low-alloy steel. This steel is typically an alloy, such as ASTM A533 or A302, that contains components such as nickel, chromium, and molybdenum. These materials are chosen for their superior mechanical qualities, such as high tensile strength and resistance to radiation damage, which are critical for dealing with the reactor vessel's harsh circumstances, such as high pressure and temperature. In our study we used ASTM A533 (SA533 Grade A) as the nozzle material [8]. Its mechanical and chemical properties are given in Table 1 and Table 2 respectively.

Table 1. Mechanical properties of Nozzle material

Material name	Density	Tensile strength	Yield strength	Modulus of elasticity	Poisson ratio	Bulk modulus	Specific heat capacity	Thermal conductivity
SA533 Grade A	7.80 g/cc	550-690 MPa	345 MPa	200 GPa	0.29	160 GPa	0.470 J/g-C	52 W/m-K

Table 2. chemical properties of the Nozzle Material in wt%

Material name	carbon	Iron	Nickle	Manganese	Molybdenum	Phosphorus	Silicon	Sulphur
SA533 Grade A	0.25	97.55	0.18	1.15 – 1.5	0.40 -0.60	0.0345	0.040	0.040

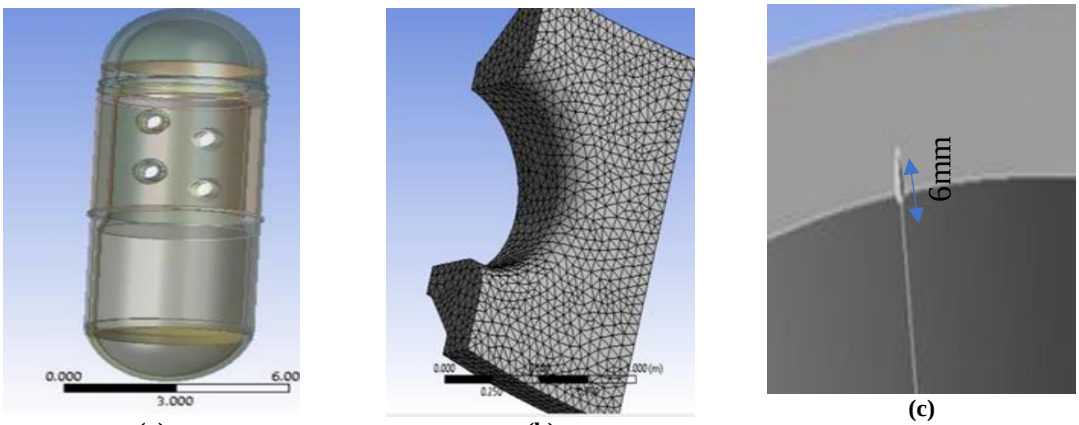


Fig. 1. (a) Reactor vessel, (b) Cross section of Nozzle, (c) Crack parameters

Here Fig. 1. (a) depicts the reactor vessel from which we cut down the section of the nozzle area depicted in Fig. 1. (b), where we used tetrahedral mesh for further analysis at the surface of the nozzle, we considered a 6mm crack size depicted at Fig. 1. (c).

3. Methodology

We can frame the total work into two following sections. At first, we analyzed total 4 kinds of stresses on the crack tip and using these stresses we analyzed the probabilistic fracture mechanics analysis.

3.1 Stress analysis

Using the Ansys software with proper boundary conditions we assessed different types of stresses developed on the nozzle shell. Using proper boundary conditions (Temperature 330°C, Pressure 16.2 MPa) We determined Thermal, Axial, Longitudinal and Dead weight stress formed at the tip of the crack. Stress determining process is depicted in Fig. 2.

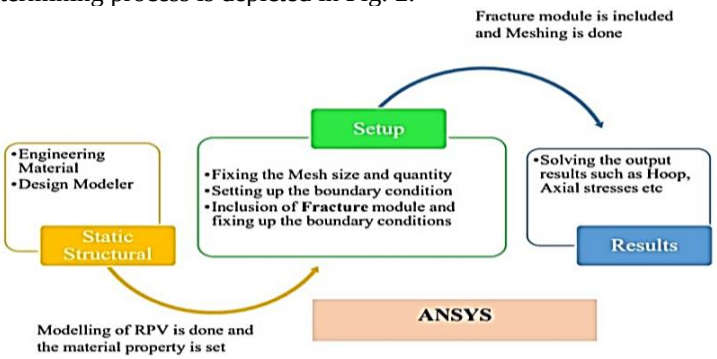


Fig. 2. Determining different stresses using ANSYS software

This figure gives us an overall observation of how the static structural module and associated setup led us to determine our expected results.

3.2 Probabilistic Fracture Mechanics Analysis

The behavior of cracks in a Reactor Pressure Vessel (RPV) varies with the service circumstances. Factors influencing this difference include the chemical makeup of the material, pipe design, climatic circumstances, and welding procedures [3]. PFM has received a lot of interest in the nuclear sector for its capacity. The factor of stress intensity is denoted as K

$$K = f\sigma\sqrt{\pi a} \tag{1}$$

The fracture toughness of a material at a particular thickness may be estimated as:

$$K_c = K_{IC}\left(1 + B_k e^{-\left(\frac{A_t t}{t_0}\right)^2}\right) \tag{2}$$

In equations number (1) and (2), f represents the specimen's geometrical factor, σ stands for the employed stress, and a represents the size of defects. Material Constants are A_t, B_k. It is believed to have an infinite breadth of the specimens, $f \cong 1.0$. For a modest single-edge notch, $f = 1.12$. As fault size causes crack propagation and failure, the stress intensity factor is referred to as the critical stress intensity factor (K_c) [3]. The fracture toughness of a sample is proportional to its thickness, with K_c decreasing to a fixed value. The fixed value is known as the plane strain fracture toughness (K_{IC}) [9].

The crack propagation rate at greater stress intensities

$$\frac{da}{dN} = C(\Delta K)^n \tag{3}$$

$$Therefore, N = \frac{2\left[(a_c)^{(2-n)/2} - (a_i)^{(2-n)/2}\right]}{(2-n)Cf^n\Delta\sigma^n\pi^{n/2}} \tag{4}$$

Here in equations (3) and (4), N stands for number of cyclic load C and n are both empirical constant, a_c denotes critical crack size, a_i denotes initial crack size and f denotes geometrical factor. It is a presumption that a crack can be formed due to fatigue [3]. Stainless and carbon steel pipelines exploit a log-normal distribution for initial fracture size [9]. The log-normal distribution for fracture depth a may be represented as equation 5.

$$f(a) = \frac{1}{\sqrt{2a\sigma\pi}} e^{-\frac{1}{2}\frac{\left(ln\frac{a}{\mu_a}\right)}{\sigma_a}} \tag{5}$$

Here the mean value μ_a= 0.294, Standard deviation σ_a = 1.61, a is the initial crack depth

4. Evaluation Of Failure

The size, shape, and operational circumstances of a PWR reactor all have an impact on failure due to the fracture. Initial crack size is estimated using the log-normal distribution based on material parameters and reactor conditions, with fracture toughness defined by the stress intensity factor (K_{IC}) for thin-walled vessels. Fatigue can cause cracks to expand to critical sizes, potentially leading to failure [3] [10].

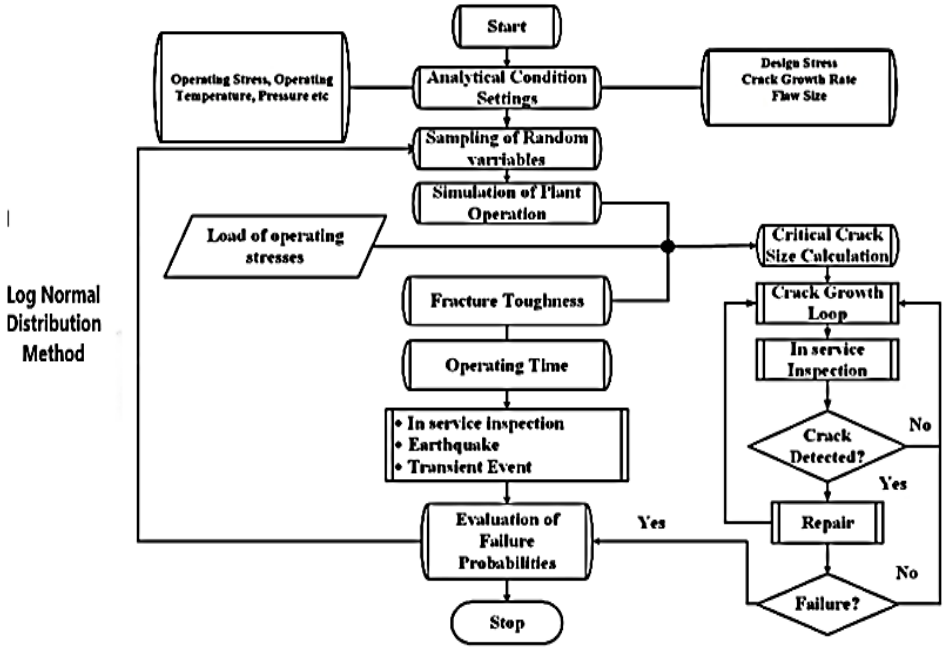


Fig. 3. Crack Evaluation Flowchart

The algorithm depicted in Fig. 3., calculates the probability of failure for a pressure vessel by simulating vulnerable initial fracture sizes with the log-normal distribution approach. It takes operational stress and operating circumstances as inputs and utilizes the computed crack sizes to predict failure over time. The code is written in Python and is based on changes to the equations mentioned in equation (1) and (5) [11].

5. Result and Analysis

As mentioned in the previous section, the log-normal distribution technique was used to forecast the initial fracture size. The cumulative distribution function of the log-normal function was used to calculate the cumulative failure probability for a given fracture size [3].
Figure 4 illustrates that the failure risk increases dramatically from 0.5 mm to 4 mm, reaching up to 60% at 4 mm. This suggests that the best option for first assessments of the defect size is a 4 mm fracture size. We examined four different kinds of stresses near the fracture tip under various transitory circumstances using ANSYS, taking into account both increase and decrease of pressure and temperature.

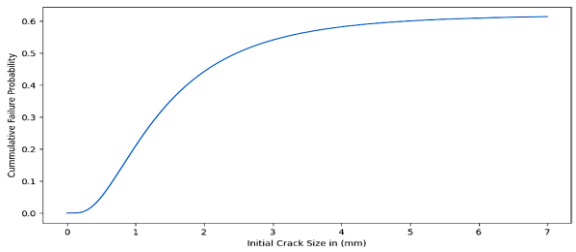


Fig. 4. Failure probability vs Initial Crack

5.1 Evaluation of changes of stresses in various transient conditions

Using boundary conditions (Temperature 330℃, Pressure 16.2 MPa) Thermal, Axial, Longitudinal and Dead weight stress formed at the tip of the crack were analyzed.

Table 3. Stresses during transient situation (increase)

Serial no	Name of stresses	Value at normal operation MPa	Value at 20% more (temperature and pressure) MPa	Value at 50% more (temperature and pressure) MPa	Value at 100% more (temperature and pressure) MPa
1	Dead Weight Stress, σ_w	1.12	1.12	1.12	1.12
2	Hoop Stress, σ_h	22.3	26.76	33.45	44.6
3	Axial Stress, σ_a	8.6	10	12.9	17
4	Radial stress	0	0	0	0
5	Thermal stress	139.25	169.2	215	290
6	Total stress (Excluded axial stress)	162.67	193.3	249.52	329.51

From Table 3 it can be estimated total stresses acting on the crack tip at different normal and abnormal situations will be 162.67MPa, 196.3MPa (20% increase), 249.52Mpa (50% increase), 329.51Mpa (100% increase) correspondingly. As the formed axial stress is acting along the crack tip it will not perform a significant role in the propagation of the crack. The other stresses that act perpendicular along the crack tip will cause the propagation of the crack.

Table 4. Stress during transient situation (decrease)

Serial no	Name of stresses	Value at normal operation MPa	Value at 20% more (temperature and pressure) MPa	Value at 50% more (temperature and pressure) MPa
1	Dead Weight Stress, σ_w	1.12	1.12	1.12
2	Hoop Stress, σ_h	22.3	17.84	11.2
3	Axial Stress, σ_a	8.6	6.88	4.30
4	Radial stress	0	0	0
5	Thermal stress	139.25	110	69.87
	Total	162.67	128.96	82.11

Here table 4 shows, how particular stresses go down while the reactor is being operated at low pressure and temperatures than the normal condition. Now putting all the values of stresses according to the flowchart mentioned in Fig. 3. the probability of failure of RPV in different transient conditions was calculated. As we aren't getting any value that can hamper the rigidity of the pressure vessel, we didn't go through the detail evaluation of it.

5.2 Analysis of failure Probability

As we considered operational temperatures and pressures rise at same rate, the probability of reactor failure grows dramatically. Under normal conditions, failure probability gradually increases over time.

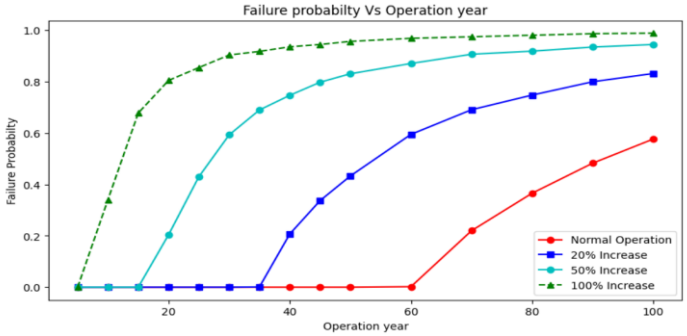


Fig. 5. Failure probability vs Operation year (Increase)

However, with a 100% increase in design temperature and pressure, it swiftly approaches threshold after few years of operation. It can be seen that, if the parameters are increased 20% the reactor will be leaning over failure nearby 40 years of operation, this process will be faster according to graph nearby 20 years of normal operation if the parameters are increased 50% the graph shows a significant relationship between increased circumstances and faster failure probabilities in Fig. 5. while the decreasing temperature and pressure will not cause failure for the same model because the vessel is made to be withstand in normal condition which is more higher comparing to the decreased value.

6. Conclusion

In this study, the probabilistic mechanics approach was applied to estimate the failure probability of the PWR nozzle area. The specimen model of a PWR nozzle area with an arbitrary crack was developed from where the various stresses were calculated which were later given as input to the PFM code made using Python language to find the failure probability for that crack over the years. Operating conditions were further enhanced towards transients to find out the failure probabilities for different unwanted situations. Hence, it can be said that this work can significantly help to build an overall failure prediction tool in the future with real-time data.

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