Enhancing Structural Safety of Reactor Pressure Vessels through Probabilistic Fracture Mechanics Modeling

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Abstract: This study presents the development and application of a Probabilistic Fracture Mechanics (PFM) code to assess the failure probability of nuclear reactor pressure vessels (RPVs) with pre-existing cracks. RPVs, made of low alloy steel and subjected to various aging mechanisms such as fatigue and stress corrosion cracking, are especially vulnerable in the beltline region where nuclear fission occurs. A Python-based PFM code was developed to evaluate failure risk considering a single initial crack, whose size follows a log-normal distribution. The final (critical) crack size is calculated using applied design stresses and plane strain fracture toughness (KIC), which is modeled based on Nil Ductility Temperature (NDT) and neutron fluence, using IAEA and ASME recommendations. Stress input- membrane, bending, thermal, and seismic- are determined analytically using ANSYS simulation tools and Bangladesh National Building Code (BNBC) seismic guidelines. Failure probabilities are evaluated for both vertical and horizontal crack orientations, and results are benchmarked against PRAISE-JNES code outputs. The study finds that vertical cracks pose a higher failure risk, and that increased temperature and pressure significantly raise failure probability. The proposed PFM framework offers a reliable and compliant tool for probabilistic safety assessment of RPVs.

Key-Words: - PFM- code, Fracture Toughness, Nil ductile Temperature, BNBC code.

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1 Introduction

The Bangladeshi government intends to construct a nuclear power station near Ishwardi Pabana, where two Russian VVER 1200 generation III+ reactors are being build [1]. The reactor pressure vessels for VVER 1200 reactor are constructed using of low ferritic alloy steels [2] [3]. Austenitic steel is used for RPV cladding because of its unique qualities, which include ductility, corrosion resistance, cryogenic toughness, strength, and hardness. The inner surface is made of an anti-corrosive substance, such as stainless steel [4] ,to protect it from the corrosive environment. Aside from its' distinctive characteristics, austenitic steel impacts the behavior of RPV crack initiation [5].With the construction of Bangladesh's first nuclear powerplant underway, probabilistic risk assessment of the RPV is a rising sector in power plant Engineering.

The beltline area of a reactor pressure vessel (RPV) is exposed to neutron irradiation which results in material embrittlement and a decrease in fracture toughness. This increases the possibility of brittle fracture which is refers to a rapid and sudden failure of a material with minimal plastic deformation, typically occurring under stress, low temperatures, or in the presence of material flaws. under mechanical and thermal stresses by elevating the ductile-to-brittle transition temperature (DBTT) [6]. In order to assure structural integrity and extended reactor operation, safety precautions and surveillance systems monitor embrittlement [7]. Pressurized thermal shock (PTS) and loss-of-coolant accidents (LOCA) are examples of failure events that can be prevented with proper aging control. [8]For nuclear reactors to remain safe in the long run the RPV must withstand high pressure and temperature. [4] [9] Assessing stresses in the beltline region is critical, as it typically experiences the greatest stress concentration, potentially leading to material failure, fatigue, or deterioration. A thorough evaluation ensures the structural stability, safety, and longevity of the system, enabling optimal design and dependable performance under operational conditions [10] [11].

Pressurized thermal shock (PTS) transients pose a threat to the structural integrity of reactor pressure vessels (RPVs) and must be properly controlled.

Nuclear power plant aging and failure risks can possibly be statistically evaluated using Probabilistic Fracture Mechanics (PFM). [12] By taking into account variables like irradiation embrittlement and transient thermal stresses, PFM programs like FAVOR (ORNL), PRAISE (LLNL), and PINTIN-CAM help in estimating of RPV failure probability. To enhance integrity management, these algorithms make utilization of crack development models and Monte Carlo simulations. Their efficacy has been demonstrated by studies that appear in publications such as ASME Pressure Vessel Technology and Nuclear Engineering and Design. [13] Standards for RPV evaluation have been established by regulatory agencies including the NRC, IAEA, and ASTM. PFM-driven analysis promotes risk-informed decision-making, enhances safety, and extends plant life. The primary goal of this research is to estimate a reactor pressure vessel's (RPV) lifespan by applying fracture mechanics concepts and algorithms to analyze the behavior of a crack in the beltline area. This was accomplished by creating a 3D model of the RPV in SOLIDWORKS using appropriate design data. In order to analyze the stress distribution under different transient situations, a crack was assumed into the beltline region and boundary conditions were set up in ANSYS [14]. For further examination, the gathered stress data was assembled. employing a Probabilistic-Fracture Mechanics (PFM) technique [15], which made it possible to estimate how long the RPV will endure. The ultimate goal of this research is to assure the pressure vessel's structural integrity and long-term safety, minimizing catastrophic failures and improving the reactor reliability. In this study, we started by organizing the data and setting up the design parameters and boundary conditions. We then ran a stress analysis on the reactor using ANSYS. After that, we used the PFM code to calculate the probability of failure, which helped us estimate how long the reactor might last before failure, giving us a better understanding of its overall reliability.

2 Design and Modelling

2.1 Material Properties

Austenitic steel is extensively utilized to reinforce various components of pressure vessels (RPVs) in VVER reactors, with cladding typically composed of 18 Cr/Ni steel welded to the RPV to enhance its durability. Stress fractures may occur at the junction between the base steel and cladding, particularly during pressure surges, which can compromise the reactor's structural integrity [16] The carbon content in stainless steel is a key factor in enhancing its strength and hardness, with higher carbon levels contributing to greater strength. Vanadium carbides further improve the material's strength, resistance to thermal aging, and contribute to a refined grain structure [17]. Additionally, the material's strength, resistance to thermal aging, and very tiny grain size are provided by vanadium carbides. Welding nickel-alloyed steels presents significant challenges, primarily due to the risk of hot cracking, necessitating high preheating. Additionally, combining carbon and chromium can improve corrosion resistance. [18]

Table 1. Chemical composition of RPV Base Material [wt%]

Material ID	**C**	**Si**	**Mn**	**Ni**	**S**	**P**	**Cr**	**N**	**Nb**
1st Layer	-	0.5	1-2	12-14	-	-	23.26	-	-
Sv07Kh25N13	0.09	1	-	-	0.018	0.025	-	0.05	-
2nd Layer	0.05	0.2	1.8	9.5	-	-	18.5	-	0.9
Sv08Kh19N10G2B	0.1	0.45	2.2	10.5	0.02	0.03	20.5	0.05	1.3

Table 2. Chemical composition of RPV cladding Material [wt%]

Material ID	**C**	**Mn**	**Si**	**P**	**Cr**	**Ni**	**Mo**
15Kh2NMFA	0.13-18	0.3 - 0.6	0.17 - 0.37	≤ .002	1.8 - 2.3	1.0 - 1.5	0.5 - 0.7

2.2 Design of RPV

The Reactor Pressure Vessel (RPV) profile provides important details on its size and composition. The vessel is massive, as is common for conventional nuclear reactors, measuring 11,185 mm in height and 4,250 mm in radius. Its 197.5 mm wall thickness guarantees that it can tolerate the extreme heat and pressures within the reactor [19]. The RPV is a massive structure that weighs 330 tons and was designed to remain intact under harsh operating conditions. With a height of 3,550 mm, the beltline region is especially crucial [20] ,since it is the area most susceptible to material deterioration and fracture due to the highest radiation levels and thermal loads from the reactor core [21] [22]

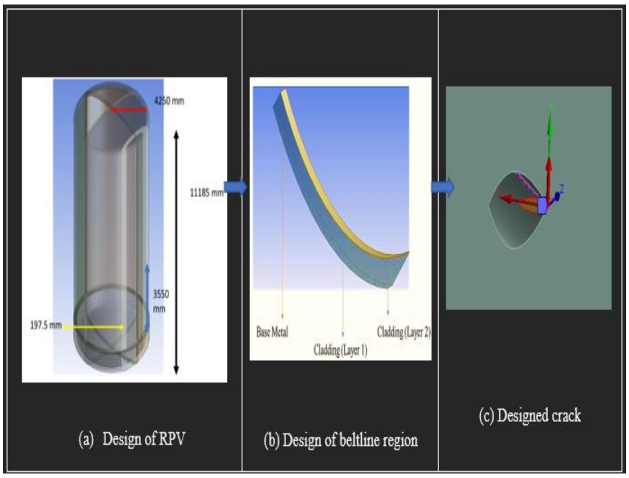
2.3 Beltline region Parameters

The beltline specimen profile has a height of 110 mm and a radius of curvature of 4250 mm, which represents the curved geometry of the reactor pressure vessel wall. The vessel's wall thickness is 197.5 mm, which provides the structural integrity required to withstand high pressure and temperature conditions [23]. Furthermore, the vessel's inner surface is protected by two layers of cladding, each 4 mm thick, which are normally engineered to resist corrosion and radiation damage. This combination of wall thickness and coating enhances the reactor vessel's long-term sustainability and safety. [24]

After assuming an arbitrary crack, a 3D RPV was modeled with the assumed crack profile and all the boundary conditions were set for normal reactor operational conditions in ANSYS. Approximate pressure of 16.2MPa and temperature of 350℃ was considered for the estimation of stresses in ANSYS software along with other boundary conditions which are mentioned in Table 3. [25]

Table 3 crack profile

Serial	Parameters	Value
1.	Type	Arbitrary
2.	Length	6 mm
3.	Depth	30 mm
4.	Division	6
5.	Crack plane	z-x plane
6.	Crack depth	X-axis



(a) Design of RPV (b) Design of beltline region (c) Designed crack

Figure 1. Flow of the design.

Figure 1 (a) depicts the design of the Reactor pressure vessel, figure 1(b) depicts the belt line region, figure 1(c) depicts the created arbitrary crack

3 Methodology

3.1Probabilistic fracture mechanics analysis

Reactor pressure vessel (RPV) cracks can develop and widen over time as a result of a number of variables, including the material composition, the construction of the pipes, the environment, and even the welding technique. Since the 1980s, rules have employed probabilistic failure analysis (PFM) to forecast this harm and guarantee safety. This approach takes into account the possibility of fracture development and propagation to enhance nuclear power plant safety [26]. The stress intensity factor, K is

$$\boldsymbol{k = f\sigma\sqrt{\pi a_c}} \qquad \textbf{(1)}$$

The fracture toughness for a material at a specific thickness can be approximated as:

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

Where f is a geometry factor for the specimen and flaw, σ is applied stress and a is the flaw size. If the specimen is assumed to have an "infinite" width, then f≅ 1.0. For, a small single edge notch f=1.12 When the flaw size a is so sized that the crack to propagate and cause failure then the stress intensity factor is then called critical stress intensity factor K_c. [27]

The sample's thickness affects fracture toughness; as thickness rises, fracture toughness K_c falls to a constant value. The plane strain fracture toughness, or K_{ic}, is the name given to this constant number.
The crack growth rate at higher stress intensities

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

Therefore,

$$N = \frac{2[(a_c)^{(2-n)/2} - (a_i)^{(2-n)/2}]}{(2-n)Cf^n\Delta\sigma^n\pi^{n/2}} \qquad (4)$$

Where, N is the number of cycles, C and n is the empirical constants depends upon the material property, ac is the critical crack size, ai is the initial crack size and f is the geometry factor. [27]

A fracture caused by a fatigue process is presumed to have formed in the early phases of plant operation. In pipes made of carbon and stainless steel, the first fracture size is distributed logarithmically. The log-normal distribution provided for the fracture depth a may be stated as follows:

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

$\mu_a = 0.294mm$ $\sigma_a = 1.61$ where, μ_a is the mean value, σ_a is the standard deviation of crack and a is the crack depth. [27]

3.2 Stresses analysis

Fracture toughness is notably affected by stress. In this study, various types of stresses were examined, including membrane stresses such as hoop stress, axial stress, radial stress, and dead weight stress, in addition to thermal and seismic stresses. According to Bangladesh's construction code, the country is divided into two seismic zones, with Zone 2, which includes Dhaka, experiencing moderate seismic activity. Seismic stress on the reactor pressure vessel is considered by evaluating forces resulting from both vertical and horizontal earthquake movements. The seismic load is calculated by incorporating the zone coefficient, soil properties, and the expected ground acceleration. An additional equation is used to determine the seismic shear force, which factors in the pressure vessel's weight, height, and response. There is increasing concern regarding the effects of even minor earthquakes on the pressure vessel, as such events occur with greater frequency. The analysis considers distances from the epicenter of frequent earthquakes, with magnitudes of 6, ranging from 10 km to 200 km. The unique seismic hazards in Bangladesh are taken into account in this study. [28] [29]

Figure 2. Seismic map of Bangladesh.

The procedure for analyzing a Reactor Pressure Vessel (RPV) using finite element analysis (FEA) in ANSYS, with an emphasis on fracture modeling and static structural analysis, is outlined in the image. Initially, the engineering material is selected, and the RPV design model is set up in the static structural section, where the material properties are defined. During the setup phase, the mesh size and quantity are determined, boundary conditions are applied, and the fracture module is incorporated to facilitate fracture analysis. The model is then processed in ANSYS, where all configurations are implemented. Ultimately, the output results, including hoop and axial stresses, are solved to evaluate the RPV's structural integrity and identify potential fracture zones. This procedure ensures that the RPV is capable of withstanding operational stresses and conditions in a safe and reliable manner. Figure 3 shows us the overall stress analysis procedure

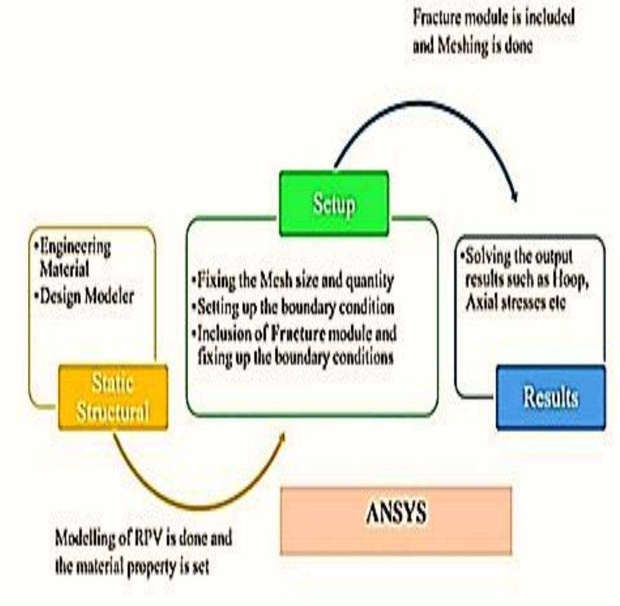


Figure 3. Developed stress determination procedure.

3.3 Failure Evaluation

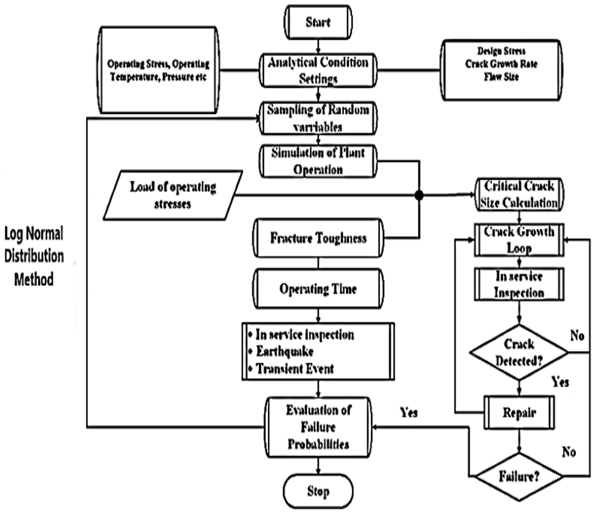


Figure 4. Flow chart of the PFM code.

The flowchart shown in Figure 4 outlines a structured methodology for evaluating the failure probabilities of a system, considering factors such as operating stresses, temperature, pressure, and other relevant conditions. The procedure begins with the establishment of analytical conditions, which encompass operating stresses, temperature, pressure, and additional parameters. This is followed by the sampling of random variables to simulate the plant's operation. Subsequently, fracture toughness and operating time are assessed, and the critical crack size is determined through the consideration of design stress, crack growth rate, and flaw size. A crack growth analysis loop is then initiated, incorporating in-service inspections, along with assessments of potential events such as earthquakes and transient disturbances. Following this, failure probabilities are evaluated, and in the event that a crack is detected, the system undergoes repair. If no crack is identified, normal operations are resumed. When a failure be detected, the process is concluded. This comprehensive approach ensures the detection and remediation of potential cracks, thereby safeguarding the structural integrity and operational safety of the system [30].

4 Results and Analysis

4.1 Determination of the initial cracks

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 failure probability for a given fracture size. Figure 5 illustrates that when the initial defect size grows from 0.5 mm to 4 mm, the failure probability rises, but beyond 4 mm, it is seen that the probability of the failure increases up to 60%. Therefore, we had to consider an initial crack size beyond 4mm

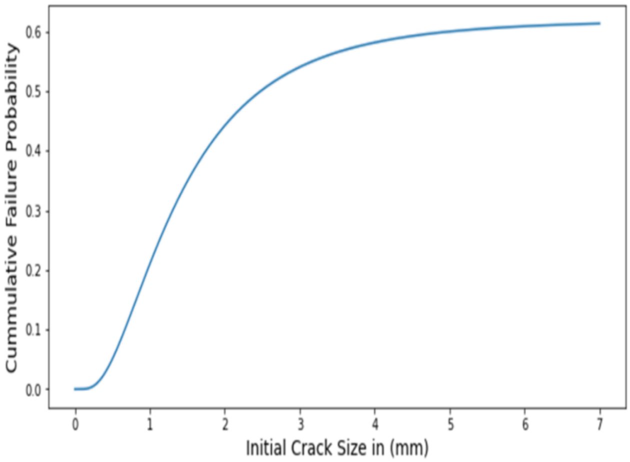


Figure 5. Failure probablity vs Initial crack.

4.2 Overall Stress Assessment using ANSYS

Using the proper boundary conditions and suitable operating parameters we calculated overall stresses at the crack tip using finite element method for both circumferential and the axial crack

Table 3. Stresses during transient situation (increase) at circumferential crack.

Ser	Stress Due to Fatigue Load	Value (MPa & °C) Normal Operation	Value (MPa & °C) 25% more than Normal Operation	Value (MPa & °C 50% more than Normal Operation	Value (MPa & °C) 100% more than Normal Operation
1.	Dead Weight Stress, σ_w	1.12	1.12	1.12	1.12
2.	Hoop Stress, σ_h	43.815	54.093	67.62	86.55
3.	Axial Stress, σ_a	52.084	84.029	105.04	134.45
4.	Radial Stress, σ_r	-	-	-	-
5.	Thermal Stress, σ_θ	61.042	77.326	93.61	126.18
6.	Seismic Stress, σ_s	24.67	24.67	24.67	24.67
Total, σ_t		183.271	241.778	292.6	373.51

The table illustrates how different stress components rise with increasing operational loads under various loading scenarios. While hoop, axial, and thermal stresses increase dramatically, dead weight stress stays constant at 1.12 MPa. The total stress increases from 183.27 MPa (normal) to 373.51 MPa (100% increased load). Radial stress is not taken into account, and seismic stress remains constant at 24.67 MPa. Particularly in high-stress settings like reactor pressure vessels, this information is essential for evaluating structural integrity and failure concerns.

Table 4. Stresses during transient situation (increase) at axial crack.

Serial	**Stress Due to Fatigue Load**	**Value (MPa & °C) Normal Operation**	**Value (MPa & °C) 25% more than Normal Operation**	**Value (MPa & °C 50% more than Normal Operation**	**Value (MPa & °C) 100% more than Normal Operation**
1.	Dead Weight Stress, σ_w	1.12	1.12	1.12	1.12
2.	Hoop Stress, σ_h	173.15	216.43	259.72	346.29
3.	Axial Stress, σ_a	51	63.68	76.41	101.88
4.	Radial Stress, σ_r	-	-	-	-
5.	Thermal Stress, σ_θ	61.83	78.32	94.82	127.81
6.	Seismic Stress, σ_s	1.01	1.01	1.01	1.01
Total, σ_t		288.65	361.1	433.62	578.65

Table 4 illustrates, the impact of increasing temperature and pressure on structural integrity by showing the rise in different stress components under various operating circumstances. At 100% increased load, axial stress (σ_a) and hoop stress (σ_h) rise sharply to 346.29 MPa and 101.88 MPa, respectively, while dead weight stress (σ_w) stays constant at 1.12 MPa. While seismic stress (σ_s) remains constant at 1.01 MPa, thermal stress (σ_θ) exhibits a similar trend, increasing from 61.83 MPa to 127.81 MPa. This assessment does not take radial stress (σ_r) into account. At the maximum load, the total stress (σ_t) rises significantly from 288.65 MPa under normal circumstances to 578.65 MPa. This highlights how important stress management is in high-pressure settings, such as reactor pressure vessels, to avoid material failure.

4.3 Life time assessment using PFM code for transient situation

Now putting all the values of stresses according to the flowchart mentioned in figure 4 we can assess the probability of failure of RPV in different transient condition.

As operational temperatures and pressures are elevated, the likelihood of reactor failure increases significantly. Under normal conditions, the probability of failure gradually rises over time. However, with a 100% increase in the design temperature and pressure, the failure probability swiftly approaches the threshold within only a few years of operation.

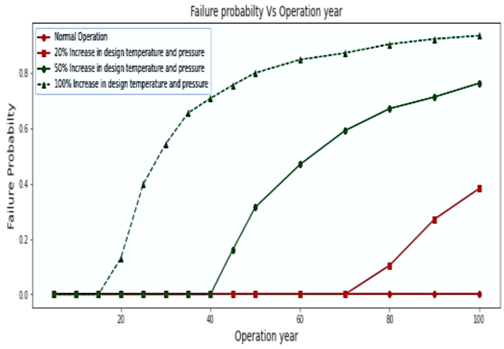


Figure 6. Failure probability vs Operation year (Increase) for circumferential crack.

A 20% increase in these parameters would lead the reactor to approach failure after 70 years of operation, with the failure process accelerating if the parameters are raised by 50%. Figure 6 clearly illustrates the strong correlation between increased operational conditions and the accelerated probability of failure. Conversely, when the temperature and pressure are reduced, failure does not occur, resulting in a zero probability of failure in such conditions.

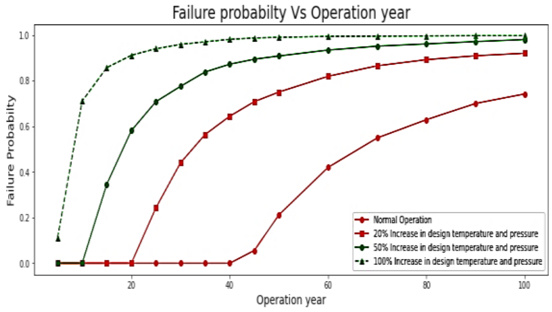


Figure 7. Failure probability vs Operation year (Increase) for axial crack.

Reactor failure is far more likely as operating temperatures and pressures increase. The failure probability grows steadily over time under normal circumstances, but after a few years of operation, it quickly approaches the threshold when the design temperature and pressure are increased by 100%. The reactor will be on the verge of failure after 40 years of operation if the parameters are increased by 20%. This process will proceed more quickly if the parameters are increased by 50% While lowering temperature and pressure will not result in failure, figure 7's graph indicates a strong correlation between more conditions and a higher probability of failure. In this case, the probability of failure will be zero in decreasing circumstances. If we compare both cracks, we can see that axial cracks are more vulnerable than circumferential crack.

5 Conclusion

The development of this Probabilistic Fracture Mechanics (PFM) code marks a significant milestone as the first of its kind in our country, enabling advanced failure probability analysis of reactor pressure vessels (RPVs) with pre-existing cracks. Simulation results reveal that stress, and hence failure risk, increases sharply with elevated temperature and pressure emphasizing the need to operate within 25% of design limits for enhanced safety. The study confirms that higher plane strain fracture toughness (K_{IC}) improves ductility, reduces Nil Ductility Temperature (NDT), and lowers the chance of initial crack formation under neutron exposure. Seismic loading was found to critically impact circumferential cracks more than vertical ones, underscoring the necessity of targeted inspections and timely repairs. The approach offers a robust tool for both operators and regulators to maintain reactor integrity within safety margins and regulatory compliance. Future improvements should incorporate realistic reactor geometry, residual and liquefaction stresses, and more advanced crack modeling. Additionally, expanding the material property database will make the tool more adaptable for real-world nuclear applications.

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