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PROBABILISTIC ASSESSMENT OF CRACK FAILURE OF REACTOR PRESSURE VESSEL (RPV) CLADDING MATERIAL

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ABSTRACT

To design a Reactor Pressure Vessel (RPV), material property like crack must be considered as it is an unavoidable property of materials. Presence of crack in materials must be kept within limit to prevent material's failure. So, crack propagation must be analyzed and observed. In this paper, crack propagation due to stress and materials fracture toughness of reactor pressure vessel cladding has been observed to estimate cumulative probability of crack failure using Probabilistic Fracture Mechanics (PFM). Average crack size is guessed as 3 mm and geometry factor is considered as 1.12 to analyze edge crack. Final crack analysis range has been found to be 1.8 mm with crack propagation rate of ± 30% of its average size. Variation of critical crack size and crack initiation point for several design stresses and fracture toughness has been investigated with probabilistic fracture mechanics technique. The observed crack propagation by calculating final crack size and the cumulative crack failure probability of the reactor pressure vessel materials are presented in this work.

Key words: Reactor pressure vessel, Cladding, Crack, Stress, Fracture toughness

INTRODUCTION

Bangladesh is going to install two generation- III$^+$ VVER -1200 model reactor in Rooppur Nuclear Power Plant (RNPP) project. Russian designed VVER-1200 reactor pressure vessels are made of ferritic low–alloy steels. The inner surface of these RPV is coated with anti-corrosive material to protect it from corrosion environment that was not considered in case of first generation VVER reactors (Timofeev and Karzov, 2006). Austenitic steel is used for RPV cladding material because of its special characteristics like ductility, corrosion resistance, cryogenic toughness, strength and hardness (Timofeev and Ulin, 1992). Besides the special characteristics, austenitic steel has influence on the behavior of RPV defects initiation. Crack initiation and propagation in RPV is influenced by the mechanical strength experienced by RPV cladding (Blauel *et al.* 1997). Ensuring structural integrity of RPV is essential to ensure the safe operation of the reactor

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throughout its lifetime (Kim *et al.* 2016). Structural integrity and safety of RPV may be evaluated by either deterministic fracture mechanics (DFM) or probabilistic fracture mechanics (PFM) approach (Huang et.al, 2016). Due to higher conservatism in DFM, PFM has been incorporated into the regulations to rationally evaluate plant life (Yagawa *et al.* 1997). The regulations and standards have been taken into account the PFM approach to evaluate structural integrity of RPV and RPV cladding including several cases (Vesely *et al.*1978, U.S. Nuclear, 1987). Depending on the PFM approach several number of computer programs has been developed worldwide to assess the structural integrity of RPV cladding and piping (ASME, 1991, Muhammed, 2003, Williams *et al.* 2004, Saarenheimo and Simola 2004, Onizawa *et al.* 2004). Safety aspects of crack initiation in nuclear RPV is inconceivable and emergency conditions may arise due to its consequence. In this paper, crack initiation and crack propagation in RPV cladding material has been investigated with the help of NUREG manual and for several design stress and fracture toughness conditions, cumulative crack failure probability has been observed. This investigation is conducted to estimate crack propagation rate and crack failure probability of the VVER-1200 RPV cladding material so that it can be used as data base of the forthcoming nuclear power project of the country.

VVER RPV AND CLADDING MATERIAL CHARACTERISTICS

The VVER reactors are Russian designed reactors that incorporate some salient features over previously designed light water reactors (LWRs) (Dwiddar *et al.* 2014). It consists of vessel itself, vessel head, support ring, thrust ring, closure flange, sealing joint and surveillance specimens (Series, I.N.E., 2009). The RPV cladding is austenitic steel in general that is manufactured from forged rings and cladding is deposited by welding process. Typically, 18 Cr/Ni type austenitic stainless steel is applied as cladding material. The potential crack propagation initiates from the cladding and the RPV base steel when it experiences highest peak of stress during several transient conditions. The higher toughness of cladding material can provide appreciable strength to the RPV (Sauter, 1983). The RPV cladding is made with two layers of different chemical compositions by automatic strip welding. The chemical compositions according to layers are shown in Tables 1 and 2 (Revka *et al.* 2012).

Table 1. Chemical composition in wt % for RPV cladding materials.

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

Table 2. Chemical composition for VVER-1000 base materials.

Material ID	C	Mn	Si	P	Cr	Ni	Mo	V
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	-

From Table 2, it is seen that VVER-1000 RPV uses type 15Kh2NMFA steel that contains almost no vanadium and 1-1.5 mass percentage of nickel.

Though vanadium carbides make the material relatively resistant to thermal ageing, fine grained and strong, it is more difficult to weld than nickel alloyed steels and requires very high preheating to avoid hot cracking (Abbasi and Shokuhfar, 2007). The key component of any stainless steel is the amount of carbon present in it. To obtain high strength and hardness, carbon content is deliberately increased. Corrosion resistance can also be improved by combining carbon with chromium (Kermani, 2001). From the work done by Gillemot *et al.* 2007, a very clear idea can be obtained about RPV cladding material characteristics.

PROBABILISTIC CRACK FAILURE ANALYSIS AND DISCUSSION

It is known that crack behavior is dynamic depending upon various situation that a RPV encounters throughout its service life. It depends on various conditions like materials, piping geometry, environmental conditions, welding mechanism and so on. Considering all of these situations PFM has been incorporated into the regulations early 1980s (Code of federal, 1985). Probabilistic failure analysis has drawn attention all over the world because of its structural formation. The flow chart that is used to analyze crack initiation and to determine the cumulative crack failure probability for specific conditions are shown in **Fig. 1**.

In this analysis, average crack size has been considered as 3 mm. From NUREG manual (Harris et.al, 1992), critical crack ≤ 3mm and crack will propagate with ± 30% of its average size. The geometry factor for the crack analysis is 1.12 for the present study since edge location is considered for the present analysis. Fracture toughness value, critical crack size and cumulative probability has been observed based on equation (1) (Donald *et al.* 2004).

$$N = \frac{2 \times \left[(a_f)^{\frac{(2-n)}{2}} - (a_i)^{\frac{2-n}{2}} \right]}{(2-n) \times C \times f^n \times (\Delta\pm)^n \times r^{\frac{n}{2}}} \qquad (1)$$

where, N is the number of cycle per year, a_i is the initial crack size, a_f is the final crack

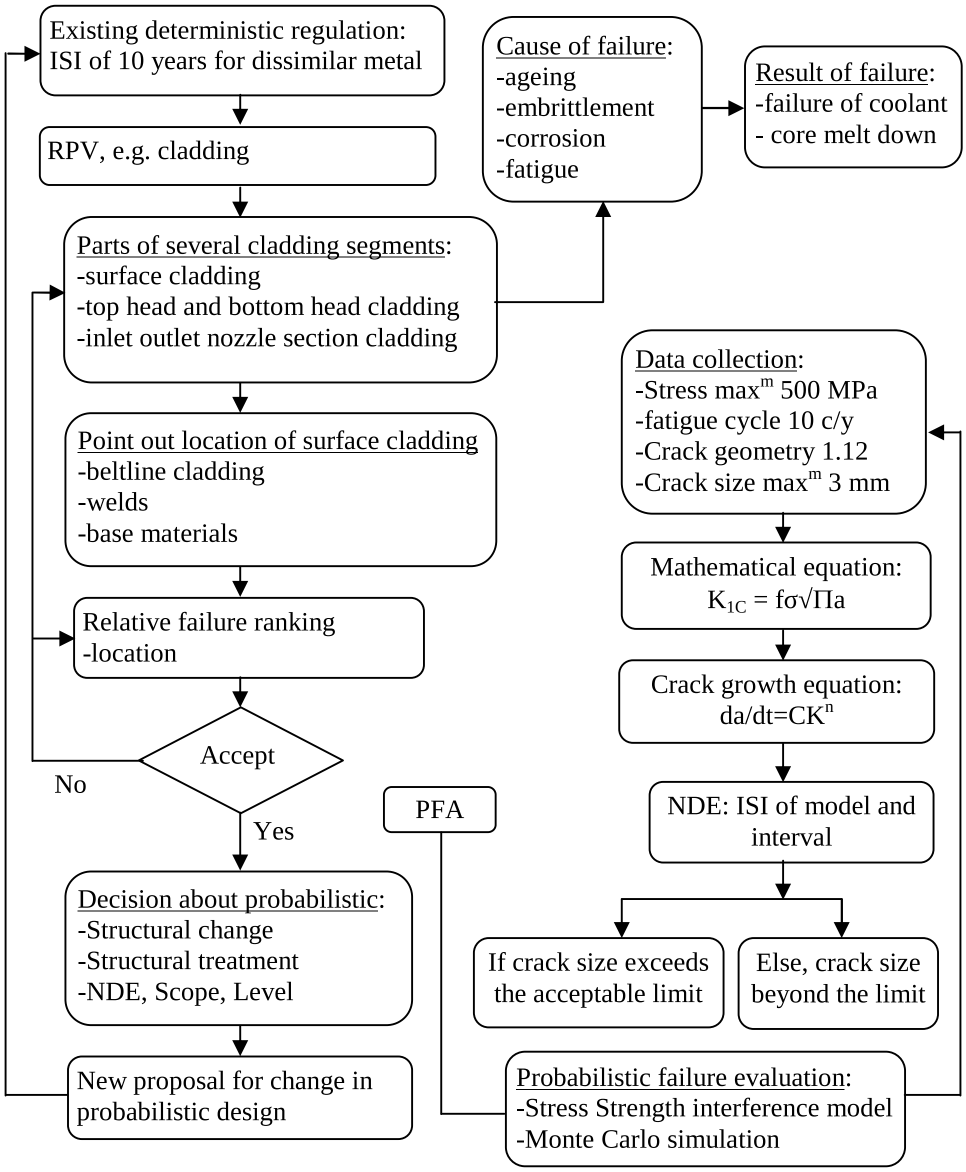

Fig. 1. Flow chart of probabilistic failure assessment of RPV cladding (Zubchenko, 2003).

size, *f* is the geometry factor and ± is the stress. According to the propagation rate with its average size, final analysis range is 1.8 mm with a considerable deviation of 0.0002 mm and a single analysis gives 9002 number of final crack size. Considering design stress 1000 MPa fixed, critical flaw size has been found to be 3.65592 mm and the final crack size is supposed to be 3.01582 mm. Again considering design stress at 1000 MPa, critical crack size has been analyzed varying fracture toughness value [Fig. 2].

Critical crack size is directly related to the fracture toughness value as it is seen in Fig. 2. The size of the critical crack increases as the fracture toughness value increases. Cumulative probability of crack failure is also analyzed considering fracture toughness at a fixed value while design stress is continuously changed [Fig. 3]. In this case, fracture toughness is considered fixed at 100 MPa$\sqrt{m}$. It is also observed that, cumulative probability of crack failure is directly proportional to the applied stress. It is also observed that, if design stress is used as applied load during operation then the crack will propagate at maximum rate and material fails after 20 years only for 100 MPa$\sqrt{m}$ fracture toughness. If plain strain fracture toughness (K_{1C}) is used, then material fails just after the application of this load.

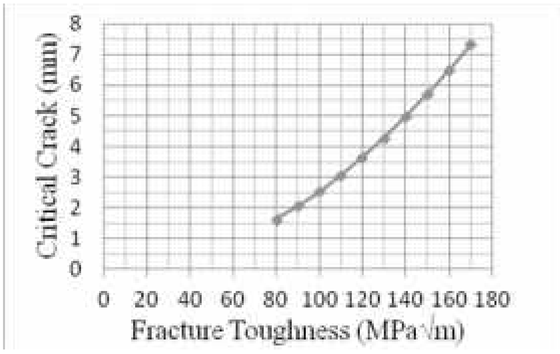

Fig. 2 Relation between fracture toughness and critical crack.

Fig. 3 demonstrates that, cumulative probability for 274 MPa stress is higher than 190 MPa stress and it rises sharply with time. Stress calculation for longitudinal and circumferential direction of RPV was found to be 190 MPa and 274 MPa respectively which depicts that, crack failure probability is higher at circumferential direction than longitudinal direction. Relation between crack failure starting point and the operation year of the reactor is shown in Fig. 4. It is observed that, crack failure starts at high initial value for low stresses and this starting point decreases as stress increases. Another observation is that, the larger the stress, the sharper the crack failure starting point drop to downward with time.

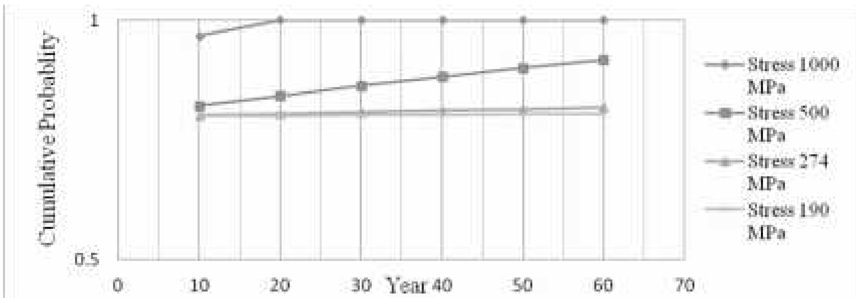

Fig. 3. Cumulative probability with varying stresses.

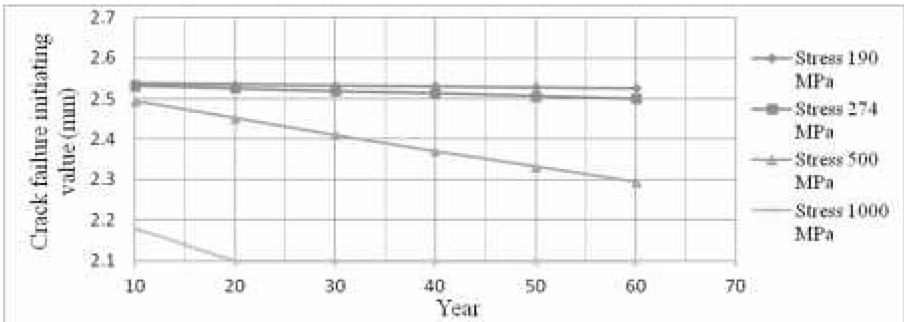

Fig. 4. Crack failure initiating value for different stresses.

As it is done by Degueldre (2017), cumulative crack failure probability with time is observed for specific fixed fracture toughness values following passive technique [Fig. 5]. It shows that crack failure probability is inversely proportional to the fracture toughness. If considered fracture toughness increases, then crack failure probability shows decreasing tendency. Cumulative crack failure probability becomes low for higher fracture toughness but for lower fracture toughness, it rises slowly compared to the higher fracture toughness.

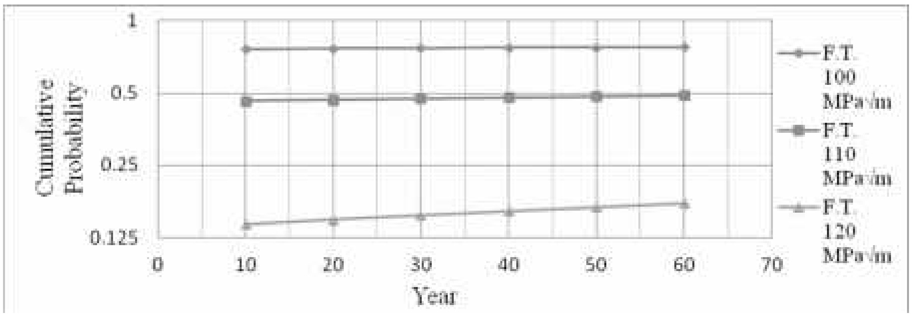

Fig. 5. Cumulative probability at 274 MPa stress with several fracture toughness.

Fig. 6 shows the relation between crack failure starting point and the reactor operation year for several fracture toughness conditions. It is observed that, at higher fracture toughness condition, crack failure starts from a high crack initiating point but it sharply goes downward with time compared to the lower fracture toughness condition.

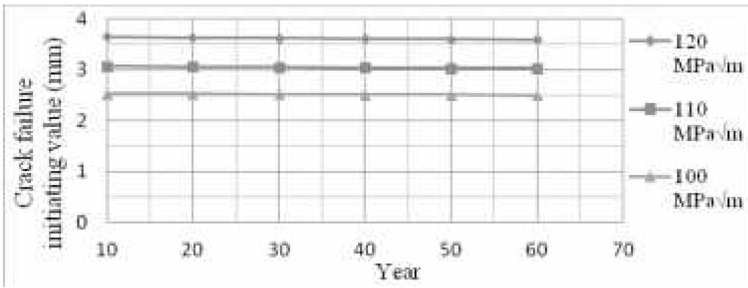

Fig. 6. Crack initiating value at 274 MPa stress with several fracture toughness.

274 MPa is considered as stress, 100 MPa$\sqrt{m}$ as fracture toughness and design stress as a varying quantity to estimate crack failure probability (Fig. 7). In this case, cumulative crack failure probability is observed with reactor operation year for several design stress conditions. Crack failure probability for a constant fracture toughness condition shows increasing tendency as design stress increases. Unless increasing materials fracture toughness, it is not a good practice to increase design stress at a higher value because material suffers from aging problem and degraded at an earlier life (Horsten *et al.* 2001). Fig. 8 shows that higher design stress allows the material to reach its critical crack size earlier and crack initiating point is more steady than the lower design stress condition.

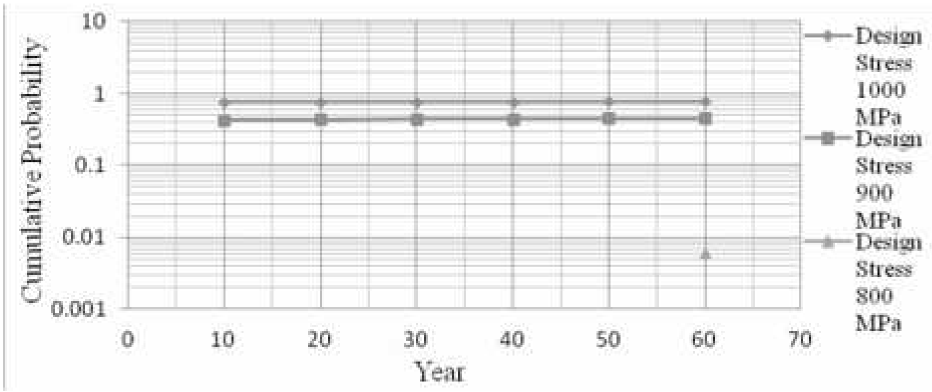

Fig. 7. Cumulative crack failure probability for several design stresses.

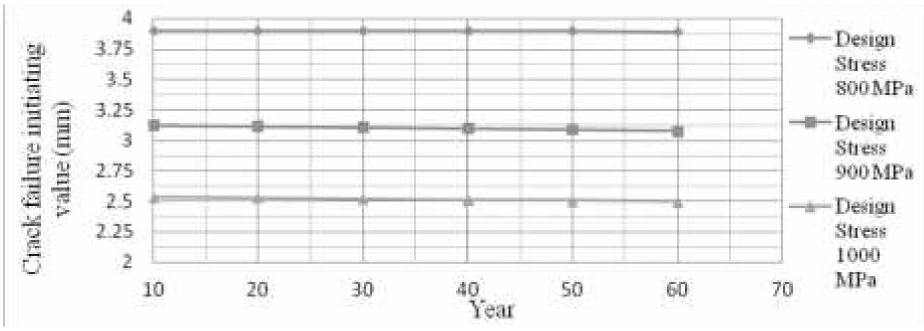

Fig. 8. Crack failure initiating value for several design stresses.

Fracture toughness changes linearly with design stress for a fixed size critical crack. Hence higher fracture toughness is allowable for using higher stresses that eliminates materials boundary limitation of using higher design stress value (Lu and Zheng, 2017).

CONCLUSION

To estimate practical safety margins of RPV cladding, PFM plays a vital role (Spencer *et al.* 2016). It is because efficient utilization and extension of existent power plant becoming more important day by day. From the present analysis, it is found that critical crack size increases with fracture toughness. Crack initiating point has been found larger for higher fracture toughness values though it decreases sharply with time than lower fracture toughness. On the other hand, cumulative crack failure probability is lower for higher fracture toughness but increases sharply with time. At constant critical crack size (3 mm), fracture toughness can be increased as per demand with the increase of design stress value.

REFERENCES

Abbasi, S.M. and A. Shokuhfar. 2007. Improvement of mechanical properties of Cr-Ni-Mo-Cu-Ti stainless steel with addition of vanadium. *Journal of iron and steel research, International*, **14(6)**: pp.74-78.

ASME, Boiler and pressure vessel code, Section XI, A5300, 1991.

Blauel, J.G., L. Hodulak, G. Nagel, W. Schmitt and D. Siegele. 1997. Effect of cladding on the initiation behaviour of finite length cracks in an RPV under thermal shock. Nuclear engineering and design, 171(1-3), pp.179-188.

Code of Federal Regulation, title 10, part 50, section 50.61 and appendix G; 1985.

Degueldre, Claude André. 2017. "Characterization Using Passive or Interactive Techniques."The Analysis of Nuclear Materials and Their Environments. Springer, Cham, 35-256.

Donald, R. A., P. P. Pradeep and D. K. Bhattacharya. 2004. *Essentials of materials science and engineering*. Thomson Publication.

Dwiddar, M. S., A. A. Badawi, H. H. Abou-Gabal and I. A.El-Osery. 2014. From VVER-1000 to VVER-1200: investigation of the effect of the changes in core. In the third international conference on physics and technology of reactors and applications, Tetuan, Morocco.

Gillemot, F., M. Horváth, G. Úri, T. Fekete, E. Houndeffo, B. Acosta, L. Debarberis and H.W., Viehrig. 2007. Radiation stability of WWER RPV cladding materials. *International journal of pressure vessels and piping*, **84(8)**: 469-474.

Harris, D.O., D.D. Dedhia and S.C. Lu. 1992. Theoretical and user's manual for pc-PRAISE: A probabilistic fracture mechanics computer code for piping reliability analysis (No. NUREG/CR-5864; UCRL-ID--109798).

Horsten, M.G. and W.P. Belcher. 2001. January. Fracture toughness and tensile properties of irradiated reactor pressure vessel cladding material. In Effects of Radiation on Materials: 20th International Symposium. ASTM International.

Huang, Kuan-Rong, Chin-Cheng Huang and Hsiung-Wei Chou. 2016. Deterministic and Probabilistic Fracture Mechanics Analysis for Structural Integrity Assessment of Pressurized Water Reactor Pressure Vessel. *Journal of Pressure Vessel Technology* **138(3)**: 031202.

Kermani, B., M. Dougan, J.C. Gonzalez, C. Linne and R. Cochrane. 2001. January. Development of low carbon Cr-Mo steels with exceptional corrosion resistance for oilfield applications. In CORROSION 2001. NACE International.

Kim, J., B. Lee, T. Kim and Y. Chang. 2016. Development of Probabilistic Fracture Mechanics Analysis Codes for Reactor Pressure Vessels Considering Recent Embrittlement Model and Calculation Method of SIF: Progress of the Work. In ASME 2016 Pressure Vessels and Piping Conference (pp. V06AT06A042-V06AT06A042). *American Society of Mechanical Engineers.*

Lu, Wei, and Zheng. 2017. He. "RPV Structure Integrity Assessment With Surface Cracks Under PTS." 2017 25th International Conference on Nuclear Engineering. American Society of Mechanical Engineers.

Muhammed, A. 2003. STRUREL results for SCC. NURBIM Report D4/Appendix A2. UK:TWI.

Onizawa, K., K. Shibata, M. Suzuki, D. Kato and Y. Li. 2004. Embedded crack treatments and fracture toughness evaluation methods in probabilistic fracture mechanics analysis code for the PTS analysis of RPV. In: Proceedings of ASME pressure vessels and piping division conference, ASME-PVP 481, p. 11-7.

Revka, V., L. Chyrko, Y. Chaikovskiy and O. Trygubenko. 2012. Different approaches to estimation of RPV material embrittlement. Current Problems in Nuclear Physics and Atomic Energy, 226.

Saarenheimo, A. and K. Simola. 2004. Independent review of NURBIT. NURBIM Report D4/Appendix D2. Finland: VTT Industrial Systems.

Sauter A. Influence of cladding on linear elastic RPV-analysis during loss of coolant accident. Nucl Eng Des 1983, **76(3)**: 347.

Series, I.N.E. 2009. Integrity of reactor pressure vessels in nuclear power plants: assessment of irradiation embrittlement effects in reactor pressure vessel steels. no. NP.

Spencer, Benjamin, *et al.* 2016. Probabilistic Fracture Mechanics of Reactor Pressure Vessels with Populations of Flaws. No. INL/EXT--16-40050. Idaho National Laboratory (INL), Idaho Falls, ID (United States).

Timofeev, B.T and V.P. Ulin. 1992. Mechanical properties and fracture resistance of anticorrosive cladding on pressure vessel. In: Proceedings of the 7th international conference on pressure vessel technology, vol. 1. Dusseldorf, p. 1014.

Timofeev, B.T. and G.P. Karzov. 2006. Assessment of the WWER-440/V-213 reactor condition. Int J. Press Vessels Piping, **83**: 216.

U.S. Nuclear Regulatory Commission. Regulatory guide 1.154: format and content of plant specific pressurized thermal shock safety analysis reports for pressurized water reactor; 1987.

Vesely, W.E., E.K. Lynn and F.F. Goldberg. 1978. The OCTAVIA computer code: PWR reactor pressure vessel failure probabilities due to operationally caused pressure transients. NUREG: CR-0258.

Williams, PT, T.L. Dickson and S. Yin. 2004. Fracture analysis of vessels – Oak Ridge, FAVORv04.1, computer code: theory and implementation of algorithms, methods and correlations. USNRC Report NUREG/CR-6854. TN. USA: Nuclear Regulatory Commission.

Yagawa, G., S. Yoshimura, N. Soneda and M. Hirano. 1997. Probabilistic fracture mechanics analyses of nuclear pressure vessels under PTS events. *Nuclear Engineering and design*, **174(1)**: 91-100.

Zubchenko, A. S. (2003). i dr. Marochnik staley i splavov [Database of steel and alloy]. M. Mashinostroenie.

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