CADERNOS UniFOA
Edição 53 | Dezembro 2023

e-ISSN: 1982-1816

Data de submissão: 10/08/2022 Data de Aprovação: 31/08/2022

Assessment of structural reliability in steam generator tubes in nuclear power plants: a study with sanicro 69 alloy

Avaliação da confiabilidade estrutural em tubos de gerador de vapor em usinas nucleares: um estudo com a liga sanicro 69

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Abstract

Steam generators are integral components in nuclear power plants, particularly in those utilizing pressurized water reactors. They are instrumental in the conversion of thermal energy into electricity. These generators must maintain a strict separation between the primary and secondary coolant systems to ensure the release of pure, non-radioactive steam. The structural integrity of Sanicro 69 alloy tubes that transport primary coolant is paramount. To assess the structural reliability, condition monitoring and operational assessment structural limits are employed, focusing on each individual crack's failure probability. In this study, a structural reliability assessment is conducted considering uncertainties in working pressure, crack and tube geometries, and mechanical properties. This study employs a Monte Carlo simulation incorporating goodness-of-fit to enhance the distribution model's accuracy. The assessment results indicate low failure probabilities at room temperature. However, the failure probabilities are increased at elevated temperature, underscoring the importance of proactive maintenance and inspection strategies. This study's robust methodology ensures the validity of these assessments, emphasizing the significance of structural integrity in steam generators in nuclear power plants, particularly at high operating temperatures.

Keywords:

Steam generators, structural reliability, Sanicro 69 alloy, Monte Carlo simulation, failure probability.

Resumo

Os geradores de vapor são componentes integrais em usinas nucleares, particularmente naquelas que utilizam reatores de água pressurizada. Eles são instrumentais na conversão de energia térmica em eletricidade. Esses geradores devem manter uma separação estrita entre os sistemas de refrigeração primário e secundário para garantir a liberação de vapor puro e não radioativo. A integridade estrutural dos tubos de liga Sanicro 69 que transportam o refrigerante primário é primordial. Para avaliar a confiabilidade estrutural, são empregados métodos de condição como encontrado e avaliação operacional, focando na probabilidade de falha de cada fissura individual. Neste estudo, é realizada uma avaliação de confiabilidade estrutural considerando incertezas na pressão de trabalho, geometrias de trinca e tubo e propriedades mecânicas. O estudo emprega uma abordagem da simulação de Monte Carlo, incorporando testes de aderência para aumentar a precisão do modelo. Os resultados indicam baixas probabilidades de falha à temperatura ambiente. Entretanto, as probabilidades de falha aumentam à temperatura elvada, destacando a importância de estratégias proativas de manutenção e inspeção. A robusta metodologia do estudo garante a validade dessas avaliações, enfatizando a importância da integridade estrutural em geradores de vapor em usinas nucleares, particularmente em altas temperaturas de operação

Palavras-chave:

Geradores de vapor, confiabilidade estrutural, liga Sanicro 69, simulação de Monte Carlo, probabilidade de falha.

1 INTRODUCTION

Steam generators play a pivotal role in nuclear power plants, especially those utilizing pressurized water reactors (PWRs). Their significance lies in the heat transfer from the primary coolant system (pressurized water) to the secondary coolant system (NRC, 2018). The primary coolant system is heated in the reactor core and subsequently circulates through the steam generator, where it transfers heat to the water of the secondary coolant system, generating steam. This steam is then used to drive a turbine that powers an electric generator, converting thermal energy into electricity. The steam is subsequently condensed and returns to the steam generator as feedwater (GREEN; HETSRONI, 1995).

It is crucial that the steam generated is pure and free from any radioactive materials, as it is released from the containment structure of the plant. However, the reactor's primary fluid contains radioactive materials, making it imperative to maintain a complete separation between these two fluids. The structural integrity of the tubes that transport the primary fluid through the steam generator is essential to ensure this separation. These tubes are typically made from a type-690 alloy, a high-performance alloy known for its excellent corrosion resistance and thermal stability (SMITH; KLEIN, 1985). Therefore, the proper maintenance of these tubes is one of the main concerns in the design, construction, operation, and maintenance of steam generators in PWR nuclear power plants (GREEN; HETSRONI, 1995).

The maintenance of the steam generator tubes must demonstrate compliance with the guidelines of the Electric Power Research Institute (EPRI), which requires two structural limits of the structural integrity of the steam generator tubes. The first, known as condition monitoring (CM), aims to determine the fitness for operation at the end of the recently completed inspection interval, by evaluating compliance with structural performance and leak rate criteria using inspection results as input. The second one involves conducting an operational assessment (OA) to demonstrate that applicable structural performance and leak rate criteria will be met during the next inspection interval of the steam generators. The operational assessment is simply a tube integrity analysis performed on a population of degraded tubes projected to exist at the end of the next inspection interval. It is important to emphasize that these structural limits are fundamental to ensuring the safety and efficiency of the nuclear power plant (EPRI, 2015).

In *MIRANDA* et al. (2006), the structural integrity of the steam generator tubes of Inconel 600 was carried out according to the CM and OA structural limit curves, at just one temperature. The structural integrity assessment using structural limit curves presents some restrictions. First, it does not provide clearly the failure probability value for each detected crack. And then, it does not consider global failure probability of a set of detected cracks which can increase or decrease the steam generator failure risk as a whole. *MIRANDA* et al. (2009) performed the structural integrity of the tubes of Inconel 600 according to the OA structural limit curve, using the multi-cycle approach to calculate the global failure probability for the steam generator. CIZELJ and ROUSSEL (2011) used another structural limit model to establish the structural integrity of the steam generator tubes of Inconel 600, and predicted the maximum number of defective tubes to be left in the steam generator. Recently, FRANCISCO et al. (2022) assessed the structural reliability of the steam generator tubes to compare the global failure probability of tubes of Inconel 600, 690 and 800. In the present study, the global failure probability was calculated according to the structural reliability method, taking into account the variability in working pressure, crack and tube geometries and mechanical properties of the steam generator tubes of Sanicro 69 alloy, at room and elevated temperatures.

2 METHODOLOGY

The primary objective of this study is to calculate a global failure probability of a set of detected cracks in the steam generator tubes between the P14a and P15a shutdowns of the Angra 1 nuclear power plant.

It is important to note that the structural integrity analysis of the steam generator tubes using the CM and OA curves is performed as specified by EPRI (Electric Power Research Institute). This analysis is crucial to understand the current condition of the tubes and identify possible areas of concern (EPRI, 2015).

In this study, the CM and OA curves were constructed using Monte Carlo simulation. Uncertainties were incorporated considering probability distributions for each random variable (LEWIS, 1996; D'AGOSTINO, 1986). To further refine the distribution model, goodness-of-fit tests were used to determine the probability distribution for each random variable. These tests were the first step in this study, allowing to accurately represent the behavior of each random variable. By determining the best-fitting probability distribution for each variable, the uncertainties inherent in the problem can be more accurately modeled (MONTGOMERY, 2002). The next step was to calculate the failure probability for each detected crack. This calculation, once validated, allow to use the procedure to determine the global failure probability of a set of detected cracks of the steam generator.

The failure probability of the tube is dictated by the variability in the working pressure, the crack and tube geometries and the mechanical properties. The failure probability is calculated from data characterized by independent statistical distributions. The equation that needs to be solved is the Equation 1 (THOFT-CHRISTENSEN E BAKER, 1982)

$$P_f = P[g(X_1, X_2, \ldots, X_n) \leq 0] = \int\int\int_{g(X_1,X_2,\ldots,X_n)\leq 0} f_{X_1,X_2,\ldots,X_n}(x_1, x_2, \ldots, x_n)\, dx_1 dx_2 \ldots dx_n \quad (1)$$

where is the limit state function, which depends on the random variables denoting the working pressure, the crack and tube geometries and the material properties, is the joint probability density function of the random variables (THOFT-CHRISTENSEN and BAKER, 1982).

Once the calculation of the failure probability for each detected crack has been performed, it was possible to calculate the global failure probability of the steam generator, taking into account a set of detected cracks (THOFT-CHRISTENSEN E BAKER, 1982). This global failure probability, which refers to the probability of failure of the equipment as a whole, was determined based on the CM and OA curves, as established by EPRI (2015). For the calculations, an inspection interval of 1 year was considered. The failure probabilities were calculated both for room temperature (24 °C) and for elevated temperature (350 °C), given that the steam generator operates at a temperature of 330 °C. These assessment results are crucial to ensure the structural integrity of the steam generator and to properly plan maintenance activities.

A Python program was developed to perform the calculations. The program receives data from an external file, which can be easily replaced with data from other shutdowns. The inspection interval can also be easily changed to generate results according to the needs of the plant. This flexibility allows for dynamic adjustments and ensures that the analysis remains relevant under various operational scenarios.

3 RESULTS AND DISCUSSION

This section presents the results and discussion of the study began by solving the Equation 1 using the Monte Carlo simulation. This simulation is a statistical technique that allows for the modeling of uncertain phenomena. It operates by generating a large number of realizations for the involved random variables (Table 1), and then evaluates the limit state function for each realization (ROBERT; CASELLA, 2004). This section explores these results in depth and offers a thorough analysis of their significance.

The results of the goodness-of-fit tests, presented in Table 1, allowed for the determination of the best fitting distributions for each random variable involved in assessing the structural reliability of steam generator tubes.

Table 1. Random variables and its distributions at room and elevated temperatures.

Parameters	Distributions	Mean values (24 °C / 350 °C)		Standard deviation (24 °C / 350 °C)	
L (length)	Lognormal	10.19 mm		0.289 mm	
(yield strength)	Normal	312.58 MPa	236.56 MPa	5.52 MPa	9.05 MPa
(tensile strength)	Normal	695.66 MPa	576.73 MPa	6.63 MPa	10.06 MPa
h (relative depth)	Normal	44.42%		9.16%	
c1	Normal	1.104		0.0705	
(working pressure)	Normal	28.95 MPa		1.50 MPa	
r (radius)	Normal	8.43 mm		0.017 mm	
t (thickness)	Normal	1.0923 mm		0.034 mm	

Obs1: c is the mean value from burst test results on pulled tubes.

Source: prepared by the authors (2023)

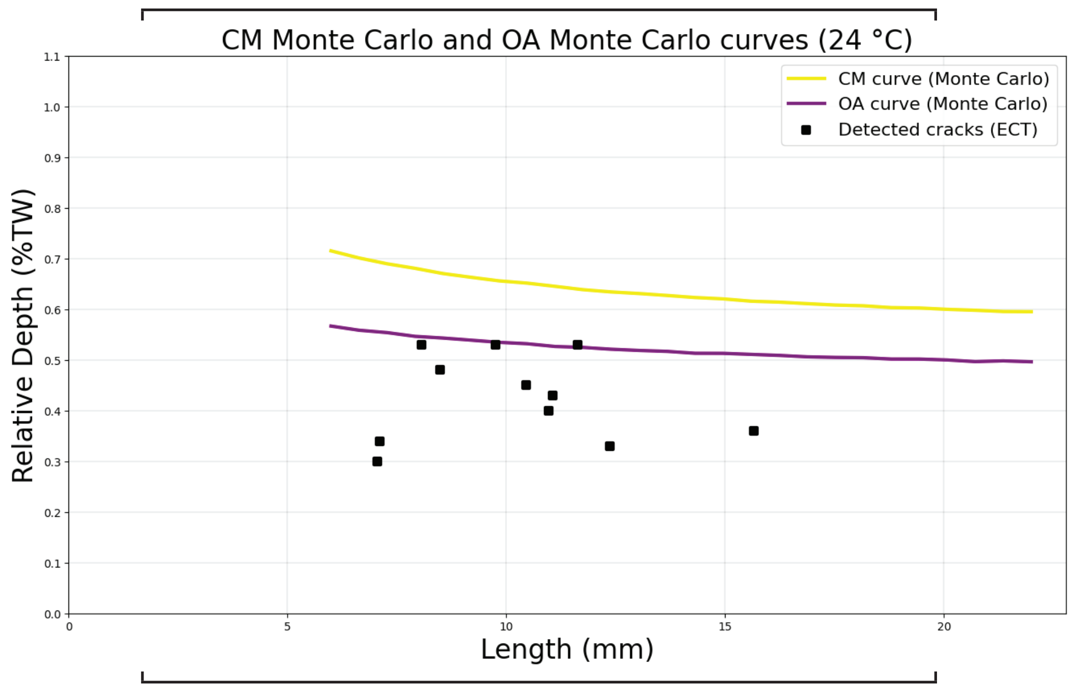
These probability distributions enabled the incorporation of each variable's uncertainties into the failure probability calculations of the steam generator tubes, leading to more realistic and less conservative outcomes.

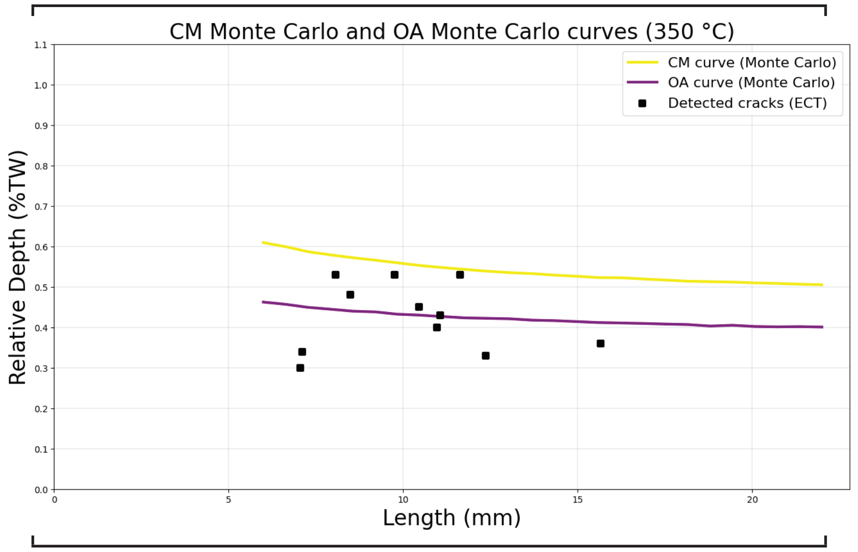
The assessment of the structural integrity of the steam generator tubes can be presented graphically by the CM and OA curves. In the graphics, these curves represent the structural limits considering a failure probability of 5%. In Figure 1, the detected cracks by the Eddy Current Testing (ECT) are located according to their failure probabilities in relation to the curves, for the room and elevated temperatures.

In the structural reliability assessment, 10,000 simulations were conducted to obtain an accurate estimate of the failure probability. Each simulation represents a possible realization for the system, and the proportion of simulations that result in failure provides an estimate of the failure probability (ROBERT; CASELLA, 2004).

The failure probabilities were calculated for axial cracks that were detected by the ECT in some row (Row) and column (Col) at location 01H from the P15a shutdown of the Angra 1 nuclear power plant, at room temperature and elevated temperature. The results of the 10,000 simulations are summarized in Table 2. It can be observed that the failure probabilities calculated in the structural reliability assessment are in good agreement with the location of the respective cracks presented in Figure 1.

Figure 1. The CM and OA structural limit curves.





Source: prepared by the authors (2023)

Table 2. Failure probability for each crack at room and elevated temperatures.

Room temperature				
Row	Col	Relative Depth (%)	Failure probability - CM (%)	Failure probability - OA (%)
10	9	0.48	0.00	0.00
21	32	0.43	0.00	0.00
10	45	0.36	0.00	0.00
46	49	0.53	0.00	1.42
40	52	0.33	0.00	0.00
8	60	0.53	0.00	2.55
47	60	0.53	0.00	4.35
40	63	0.30	0.00	0.00
46	63	0.40	0.00	0.00
8	76	0.45	0.00	0.00
10	80	0.34	0.00	0.00
Elevated temperature				
Row	Col	Relative Depth (%)	Failure probability - CM (%)	Failure probability - OA (%)
10	9	0.48	0.00	23.18
21	32	0.43	0.00	4.27
10	45	0.36	0.00	0.00
46	49	0.53	0.01	62.68
40	52	0.33	0.00	0.00
8	60	0.53	0.73	70.49
47	60	0.53	3.67	77.33
40	63	0.30	0.00	0.00
46	63	0.40	0.00	0.11
8	76	0.45	0.00	10.60
10	80	0.34	0.00	0.00

Source: prepared by the authors (2023)

Finally, the global failure probability of the steam generator tubes was calculated. This evaluation incorporated a collective set of all externally axial cracks that were detected by the Eddy Current Testing (ECT) between the P14a and P15a shutdowns. This was an important step for understanding the total structural integrity of the steam generator and for planning any necessary intervention or maintenance. The outcome of this evaluation can be found in Table 3.

Table 3. Global failure probability at room and elevated temperatures.

Temperature (°C)	Global failure probability - CM (%)	Global failure probability - OA (%)
24	0.00	0.03
350	0.03	8.16

Source: prepared by the authors (2023)

These results indicated that the global failure probability of the steam generator, considering all externally axial cracks detected between P14a and P15a shutdowns, was significantly influenced by the operating temperature.

At room temperature, the global failure probability was negligible for the CM and OA structural limits. However, at elevated temperature of 350 °C, which was above the operational temperature of 330 °C, the global failure probability increased to 0.03% for CM and 8.16% for OA.

These findings underscored the necessity of proactive maintenance strategies and rigorous inspection methodologies to ensure the structural integrity of steam generator tubes, particularly under high-temperature operating conditions. The maximum failure probability of 5% established by EPRI further reinforces the reliability of these assessments.

To understand the reason for the increase in failure probability with temperature, a statistical characterization of sample data from tensile tests performed on Sanicro 69 alloy was carried out (SEKULSKI, 2019). SANDVIK provided the results of 102 tensile tests at both room and elevated temperatures. The average yield strength values were 313 MPa at 24 °C and 237 MPa at 350 °C, and the average tensile strength values were 696 MPa at 24 °C and 578 MPa at 350 °C. These values aligned with the minimum mechanical property requirements established by the ASME SB-163 code case N-20-3 (ASME, 2022).

The statistical characterization, presented in Table 4, shows that even though Sanicro 69 alloy has good mechanical properties, it experiences a substantial reduction in its mechanical properties with increasing temperature (SMC, 2004). This reduction in mechanical properties with temperature increase is the reason why the failure probability also increases with temperature. As the material's strength decreases, its ability to withstand operational stresses also reduces, leading to a higher likelihood of failure (LI et al., 2022).

Table 4. Statistical characterization of Sanicro 69 alloy mechanical properties.

Statistical parameters	Yield Strength		Tensile Strength	
	24 °C	350 °C	24 °C	350 °C
Sample size:	102	102	102	102
Sample mean:	312.58 MPa	236.56 MPa	695.66 MPa	576.73 MPa
Median:	313.00 MPa	236.00 MPa	695.50 MPa	577.00 MPa
Mode:	310.00 MPa	236.00 MPa	698.00 MPa	583.00 MPa
Standard deviation:	5.52 MPa	6.63 MPa	9.05 MPa	10.06 MPa
Coefficient of variation:	1.77 %	2.8 %	1.3 %	1.74 %
Variance:	30.44 MPa²	43.99 MPa²	81.87 MPa²	101.11 MPa²

Source: prepared by the authors (2023)

CONCLUSION

The structural integrity of steam generator tubes is paramount to the safe and efficient operation of nuclear power plants, especially in pressurized water reactors (PWRs). These heat exchangers play a pivotal role in converting the thermal energy generated in the primary coolant system into electricity. Maintaining a rigorous separation between the primary and secondary cooling systems is crucial to ensure the purity of the generated steam and prevent the release of radioactive contaminants.

Proper maintenance of the tubing, typically made from a high-performance Sanicro 69 alloy, is one of the main concerns in the design, construction, operation, and maintenance of the steam generators in PWR nuclear power plants. Structural integrity assessment such as condition monitoring and operational assessment are crucial for ensuring the safety and efficiency of nuclear power plants.

The comprehensive approach of this study, which included the evaluation of the probability of failure for each crack individually and structural reliability analysis, underscores the importance of proactive maintenance and rigorous inspection strategies. The results reveal that the probability of failure is low at room temperature but is significantly increased at elevated temperatures above operational temperature. This highlights the need for maintenance strategies that consider high-temperature conditions and underscores the importance of periodic inspection to ensure tubing integrity.

The robust approach of failure probability calculations demonstrates the accuracy of these assessments and underlines the critical importance of the structural integrity of steam generator tubes in nuclear power plants. To ensure safe and efficient operation of these facilities, it is essential to adopt proactive maintenance and inspection strategies, particularly under high-temperature conditions. The confidence reinforced by the maximum failure probability of 5% established by EPRI further strengthens the validity of these assessments, which have the potential to contribute significantly to nuclear power generation safety.

ACKNOWLEDGEMENTS

This study was financed in part by Eletrobrás Termonuclear S/A through the scholarship for Matheus Felipe Silva Oliveira.

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