Comparative Analysis of Material Design and Performance Steam Generators for Nuclear Power Plants: Risk Assessment and Optimization Strategies

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ABSTRACT

Steam generators play a crucial role in nuclear power plants by converting thermal energy into steam to drive turbines. This study examines the material design and performance of steam generator configurations, through Failure Mode and Effect Analysis (FMEA) and Fishbone Diagram methodologies. The objective is to identify potential material failures, assess operational risks, and propose optimal strategies for the development of steam generators. The findings reveal that each model exhibits unique performance characteristics and failure risks. The helix model demonstrates superior heat transfer efficiency but is susceptible to structural degradation due to tensile stress introduced during fabrication. Conversely, the shell-and-tube model offers higher resilience against extreme pressure conditions, but it faces a greater risk of corrosion in the tube materials. These insights provide recommendations to enhance design robustness and optimize material selection, ensuring improved operational efficiency and safety. This research makes a significant contribution to the advancement of nuclear technology in Indonesia, particularly in optimizing steam generator configurations for improved reliability. The results serve as a valuable reference for strengthening nuclear power plant resilience and promoting the development of sustainable, environmentally friendly energy solutions.

Keywords: Steam Generator, Failure Mode and Effect Analysis (FMEA), Fishbone Diagram, Material, Performance.

I. INTRODUCTION

The operation of nuclear power plants relies heavily on steam generators, which are essential components of the heat transfer process that convert thermal energy into steam to power turbines. The dependability and efficiency of steam generators directly impact the longevity, safety, and general performance of a plant. Steam generator design and material selection optimization have gained a lot of attention as nuclear power remains a crucial source of low-carbon energy. Engineering and materials science developments present fresh chances to improve their effectiveness, robustness, and ability to withstand operating stresses. However, harsh operating circumstances, such as high temperatures, fluctuating pressures, and harsh chemical environments, can cause material deterioration and system failure in steam generators. Thus, a thorough comprehension.

Nuclear power facilities typically use one of two steam generator configurations: the shell-and-tube model or the helix model. Every design has intrinsic benefits and drawbacks regarding mechanical integrity, heat transfer efficiency, and susceptibility to failure mechanisms like fatigue, corrosion, and stress-induced cracking. Since the helix model has a larger surface area and better flow dynamics, it is well known for having higher heat transmission capabilities. On the other hand, tensile stress imposed during manufacture and operational cycling makes it more susceptible to structural failures. On the other hand, many nuclear reactor designs choose the shell-and-tube form because of its well-known strong structural stability and resistance to high-pressure conditions.

Using techniques like Failure Mode and Effect Analysis (FMEA) and Fishbone Diagram analysis, failure hazards in steam generators have been thoroughly investigated. These methods enable the systematic identification of potential failure modes, evaluation of their causes and effects, and development of mitigation plans. Specifically, FMEA offers a systematic framework for assessing risk priority numbers (RPN), which aid in ranking failure mechanisms according to their seriousness, frequency, and detectability. Analysis of fishbone diagrams also helps to identify contributory elements, including material selection, manufacturing flaws, operational stressors, and environmental impacts, that may result in failures. Applying these approaches together provides a comprehensive strategy for enhancing the safety and dependability of steam generators.

Numerous investigations have examined the performance of materials in nuclear steam generators, emphasizing important failure processes such as thermal fatigue, intergranular attack, pitting corrosion, and stress corrosion cracking (SCC). In particular, corrosion is still a major problem since it can weaken the integrity of tubes, resulting in leaks, decreased heat exchange efficiency, and even safety risks. Although the creation of sophisticated coatings and corrosion-resistant alloys has demonstrated promise in addressing these problems, further experimental research and field data analysis are required to confirm their long-term efficacy.

This study aims to conduct a comparative analysis of the material design and performance of steam generators, employing FMEA and Fishbone Diagram methodologies to identify critical failure risks and propose optimization strategies. By systematically evaluating failure mechanisms, risk factors, and material properties, this research seeks to provide valuable insights into the selection and enhancement of steam generator designs for improved efficiency and reliability in nuclear power plants.

This research systematically investigates the operational vulnerabilities and performance dynamics of helical versus shell-and-tube steam generator (SG) configurations in nuclear power systems. By employing failure mode and effects analysis (FMEA) and fishbone diagram techniques, the study delineates critical failure pathways and proposes mitigation frameworks to enhance design robustness. A multidimensional assessment of degradation mechanisms—spanning material fatigue, thermal cycling, and mechanical stress interactions—serves to advance predictive models for SG reliability. The findings aim to establish actionable guidelines for optimizing safety margins and operational longevity in next-generation nuclear reactors.

II. METHODOLOGY

The objectives of this work are structured around three primary goals, the first is to perform a comparative analysis of design-specific failure probabilities, emphasizing geometric and material-driven risk factors. The Second is to evaluate how advanced alloy compositions and fabrication techniques impact long-term durability under extreme operational conditions; and the third is to propose targeted design modifications and maintenance protocols that mitigate identified risks while optimizing thermal and operational efficiency. By bridging empirical data with risk modeling, this research seeks to provide actionable insights for advancing SG technologies, supporting the development of resilient nuclear energy infrastructures aligned with global safety and sustainability imperatives. This research stands out due to its integrated comparative approach, which combines failure risk assessment, material performance evaluation, and design optimization. Additionally, the use of risk assessment techniques such as FMEA strengthens the systematic evaluation of failure probabilities and their impact on nuclear plant operations.

This study employs a qualitative research methodology, utilizing a literature review as the primary data collection method. Data is derived from diverse academic and technical sources, including books, scholarly journals, IAEA (International Atomic Energy Agency) documents, and publications from national seminars. The research aims to analyze specific conditions of the steam generator unit under investigation, particularly focusing on the mechanisms of damage formation in steam generator shell & tube structures and helical steam generator tubes. A comparative analysis is conducted to examine how variations in tube geometry contribute to distinct failure modes, which compromise component integrity and overall system performance. The descriptive approach is applied to systematically identify, categorize, and explain the nature of structural changes, degradation patterns, and operational impacts caused by these damages. By synthesizing theoretical insights and empirical findings from existing literature, the study seeks to provide a comprehensive understanding of the relationship between tube design variations and failure mechanisms in steam generator systems.

This research is particularly relevant in the context of Indonesia's emerging nuclear energy initiatives, where advancements in steam generator technology can significantly contribute to national energy security and environmental sustainability. The findings from this study can serve as a knowledge base for future innovations in nuclear reactor engineering and maintenance strategies. The following sections of this paper will elaborate on the research methodology for failure risk assessment, provide a comparative analysis of material performance and failure characteristics, and explore optimization strategies aimed at enhancing steam generator dependability. The conclusions derived from this study will offer actionable recommendations for industry professionals and policymakers, fostering ongoing improvements in nuclear power plant infrastructure and operations.

III. RESULTS AND DISCUSSION

In this discussion, the steam generator data comes from various sources, while the reactor types focusing on include CANDU, VVER (Russia), Siemens (Germany), Westinghouse (USA), and HTGR (China). We collected this data and analyzed it using two methods: Failure Mode and Effect Analysis (FMEA) and Fishbone Diagrams. Now, FMEA is a systematic approach used to spot potential failures in a process, product, or system before they happen. In this study, we applied FMEA specifically to examine steam generator failures that could impact the SG's materials or overall structural integrity. Meanwhile, the Fishbone Diagram is a visual tool that helps us dig into the root causes of a specific problem or effect. organizing and identifying all the different factors that might be contributing to an issue. For our steam generators, we used this method to group damage causes into categories like chemistry, operational issues, human factors, material problems, design flaws, and maintenance practices and ultimately, the findings from both of these methods, analyze them together, and discuss them in more depth.

III. A. Analysis of Steam Generator Failure in CANDU Reactors

TABLE 1, Analysis Failure in CANDU Reactors

Failure Mode	Brief Description	Primary Controls	S	O	D	RPN
PH/SCC	Intergranular cracking due to residual stress, sensitized Alloy 600MA, and high temperatures.[1]	Alloy 800NG/690TT, routine eddy current inspections.	9	4	3	108
Fretting/Wear	Mechanical wear from fluid-induced vibrations or AVB contact.	Reinforced AVB design, debris removal, and inspections.	7	4	3	84
Pitting	Localized corrosion (chloride/sulfate) in the secondary system.[2]	Secondary water chemistry control (high pH, hydrazine).	6	3	2	36
High Cycle Fatigue	Cyclic cracking in U-bend regions due to flow-induced vibrations.	Vibration analysis, AVB reinforcement, and ultrasonic tests.	9	4	3	108
IGA/OD/SCC	Secondary-side cracking from impurities (Cl^-/sulfate) under deposits.[3]	Chemical cleaning, deposit control, Alloy 800NG/690TT.	9	4	3	108
Wastage	Generalized thinning due to chemical corrosion (phosphate/magnetite). [4]	Secondary chemistry conversion (AVT).	6	1	2	12

Based on the table above, the detailed explanation includes Severity (S) rated from 1 to 10 (with 1 indicating minimal impact and 10 indicating catastrophic failure, such as PWSCC with S=9 due to the high risk of primary-to-secondary leakage), Occurrence (O) rated from 1 to 10 (with 1 being very rare and 10 being almost certain, such as Fretting/Wear in older designs with O=4 due to frequent incidents before AVB modifications), Detection (D) rated from 1 to 10 (with 1 being easily detectable and 10 being hard to detect, such as Pitting with D=3 as it is easily identified via eddy current testing), and the Risk Priority Number (RPN) calculated as S × O × D, where high RPN values exceeding 100 indicate critical risks that necessitate priority action.[5]

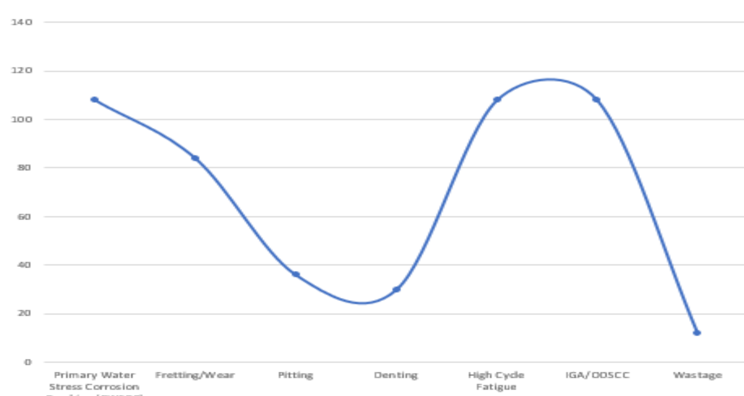

Figure 1. Damage/Failure Value Graph

Based on the graph, the highest damage values are found in High Cycle Fatigue and IGA/SCC. Therefore, this CANDU steam generator has several risks that need to be considered. The highest risks are found in Fretting/Wear and IGA/ODSCC, with an RPN value of 140 due to suboptimal AVB design and deposit accumulation. There is also a risk of High Cycle Fatigue with an RPN value of 108 in the U-bend area. To mitigate this, materials such as Alloy 690TT/800NG should be used to reduce PWSCC and ODSCC. The AVB design must be optimized to reduce fretting. Chemical control should maintain a high pH (>9.5) and reduce oxygen using hydrazine. Routine cleaning, including chemical cleaning, should be performed to remove deposits and impurities.[6]

Low-risk damages that should still be considered in the CANDU steam generator include wastage, with an RPN of 12, as it is already controlled through AVT, and denting, with an RPN of 30, due to the use of stainless steel supports. Furthermore, the damage factors in the CANDU steam generator are illustrated in the fishbone diagram as shown below.

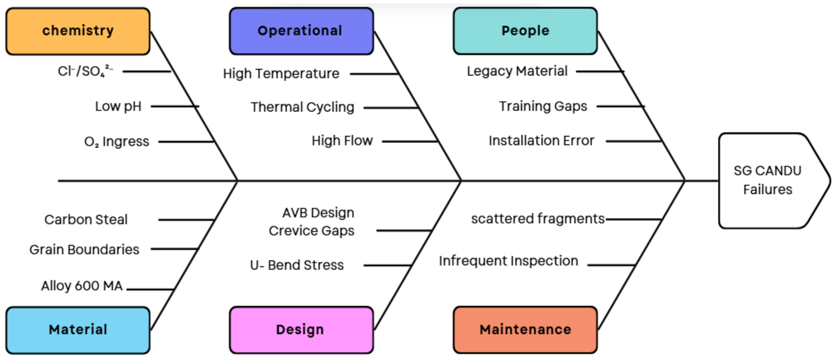

Figure 2. Fishbone CANDU Failure

The system issues are primarily due to the inherent weaknesses of the old material, Alloy 600MA, and carbon steel components, which lead to denting; design flaws, including suboptimal AVB and gaps between components, resulting in fretting and fatigue; chemical factors such as contamination and low pH, causing localized corrosion, including pitting and cracking such as ODSCC; inadequate maintenance practices, characterized by a lack of cleaning and inspection, allowing deposit buildup and undetected failures; operational conditions, involving temperature, flow, and thermal cycles, that exacerbate material degradation; and historical human factors, encompassing errors in material selection and installation, which contribute to ongoing risks.

III. B. Damage Analysis of Russian VVER Steam Generator

Table 2, Analysis Failure in Russian VVER

Failure Mode	Brief Description	Primary Controls	S	O	D	RPN
Tube Degradation	Stress corrosion cracking (SCC) or wear can cause leaks and contamination.[7]	Eddy current inspections, water chemistry control	8	5	3	120
Primary Secondary Leakage	Micro-cracks leading to secondary circuit contamination.[8]	Leak detection systems, isolation of damaged tubes	9	4	3	108
Thermal Fatigue	Cyclic cracking from operational temperature transients.[9], [10]	Operational design optimization, thermographic inspections	7	5	4	140
Flow-Accelerated Corrosion	Wall thinning due to high-velocity abrasive flow.[11]	Tube thickness monitoring, flow optimization	8	4	3	96
Uneven Feedwater Distribution	Hot spots and corrosion from damaged distributors.[12]	Feedwater distributor calibration and inspections	9	4	3	108

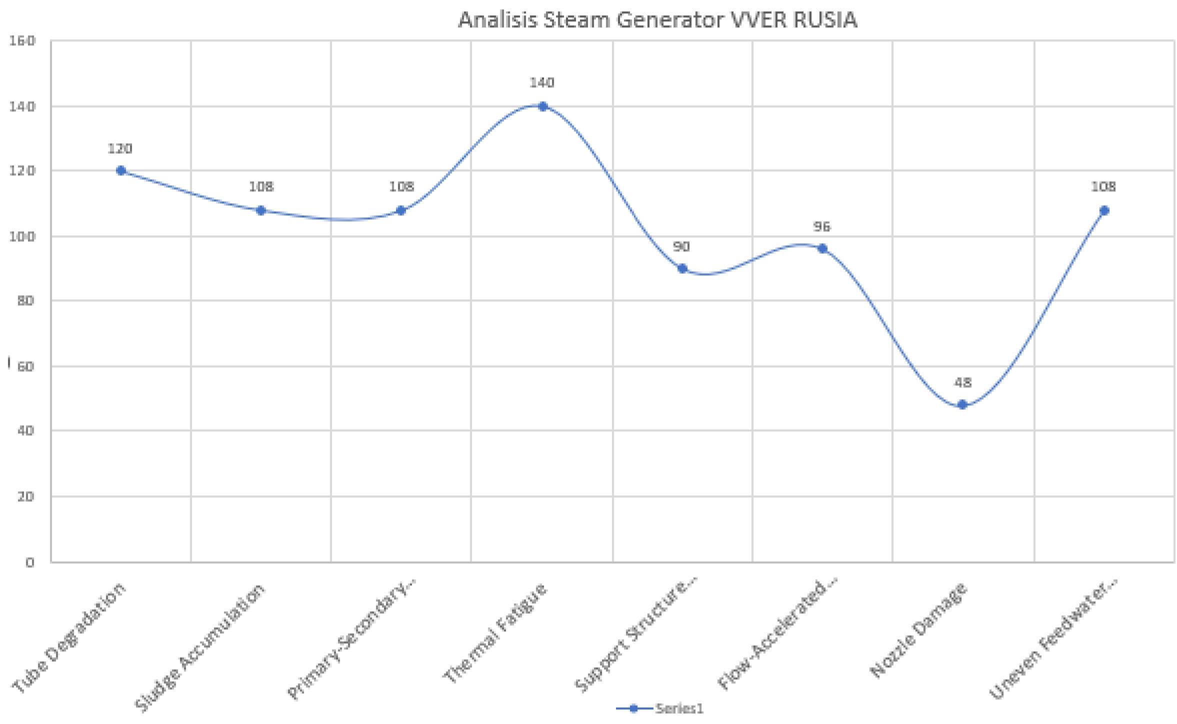

Figure 3. Damage/Failure Value Graph

Based on the graph, the damage case occurs due to thermal fatigue, which is caused by cyclic stress on tubes or components that can lead to structural failure though such failures are generally detected before reaching a critical leak. The issue originates from operational conditions, specifically sudden temperature changes due to operational transients. Additionally, design limitations, such as a lack of flexibility in the system to accommodate thermal expansion, contribute to the problem. The material used (Inconel 600/800) is also susceptible to cyclic cracking in high-temperature environments.

The lowest damage risk is associated with nozzle damage, with an RPN value of 48. This is because VVER nozzles are generally resistant to fatigue; however, cracking can occur due to fabrication defects, potentially leading to leaks.

If the causes of damage in the Russian VVER Steam Generator are mapped using a fishbone diagram, they are illustrated in the following image.[13]

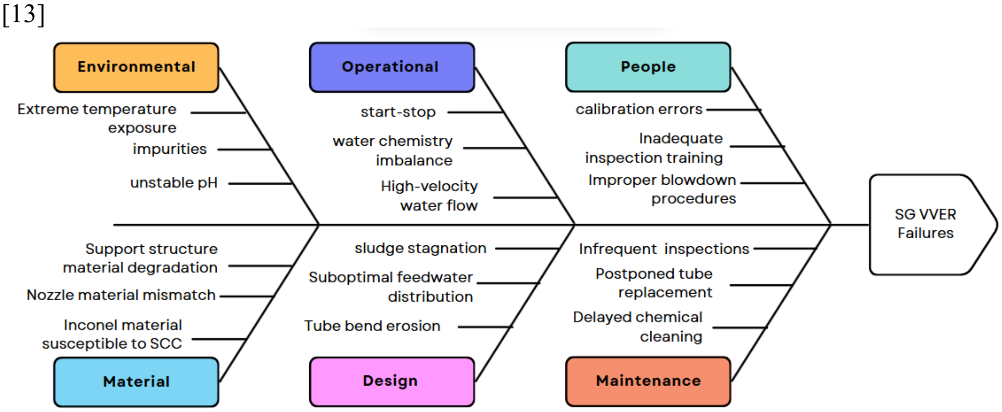

Figure 4. Fishbone Russian VVER Failure

Material failures, such as Stress Corrosion Cracking (SCC) in Inconel and wear of tube supports, are primary causes of tube degradation and structural deformation; the horizontal design of the VVER promotes sludge stagnation on the tube sheet, and uneven feedwater distribution leads to hot spots; operational transients, including sudden temperature changes, and chemical imbalance in water accelerate thermal fatigue and corrosion; contaminated feedwater, such as with chloride ions, and unstable pH trigger corrosion and deposit buildup; human factors, like calibration errors or improper blowdown procedures, increase the risk of sludge accumulation and uneven water distribution; and irregular inspections and delayed cleaning worsen tube degradation and deposit accumulation.

III. C. **Damage Analysis of Siemens Steam Generator**

Table 3. Analysis Failure in Siemens

Failure Mode	Brief Description	Primary Controls	S	O	D	RPN
Tube Corrosion	Primary-secondary leakage and contamination due to water chemical imbalance or thermal/oxidation corrosion.[14]	Annual UT inspections, water chemistry monitoring	8	5	5	200
Tube Wear (Fretting)	Micro-cracks and leakage risk from fluid-induced vibrations or friction with supports.[15], [16]	Visual and eddy current testing (ECT)	6	5	5	180
Tube Denting	Flow restriction and crack risk from mechanical pressure or under-deposit corrosion.[17]	Eddy current testing (ECT) and profilometry	8	4	4	128
Primary Water Stress Corrosion Cracking	Leakage risk due to residual stress and corrosive environments.[18]	PWSCC-resistant materials (Alloy 690), ECT inspections	8	4	4	128
Flow-Induced Vibration	Accelerated wear from two-phase flow instability or suboptimal anti-vibration design.[19], [20], [21]	In-situ vibration analysis, support modifications	7	4	3	84
Tube Rupture	Catastrophic failure leading to LOCA and radioactive release.[22]	Automatic isolation systems, intensive ECT	9	4	7	252

Based on the table above, the assessment according to the highest SOD values reveals that Severity (S) is rated from 1, indicating minimal effect, to 10, indicating catastrophic failure such as a Loss of Coolant Accident (LOCA) and radioactive release; Occurrence (O) is rated from 1, indicating very rare, to 10, indicating almost certain, with corrosion due to secondary water chemical imbalance as a common example; and Detection (D) is rated from 1, indicating easily detected, to 10, indicating difficult to detect.

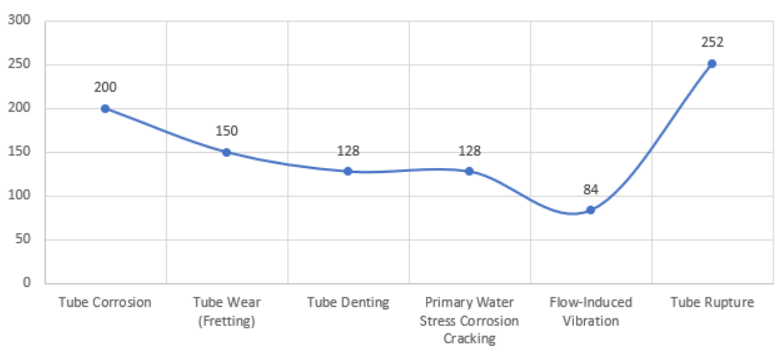

Figure 5. Damage/Failure Value Graph

Based on the graph, the highest risk condition is tube rupture, which leads to catastrophic failure (tube rupture) and can cause a Loss of Coolant Accident (LOCA), a massive loss of reactor coolant. This has the potential to trigger radioactive release into the environment and a severe nuclear accident. Tube rupture requires immediate action due to its extreme severity, presenting a direct threat to public safety, facility damage, and severe reputational impact. Detection delays in the case of cracks or corrosion leading to rupture often go undetected until the final stages due to limitations in inspection technology (e.g., ECT cannot detect microcracks in hidden areas).

The lowest risk is tube plugging, which results in tube blockage that reduces thermal efficiency but does not pose an immediate safety threat. It can lead to decreased system performance and increased operational costs. However, tube plugging can be detected with strict plugging criteria and ECT inspection, which is very effective in identifying blocked tubes.

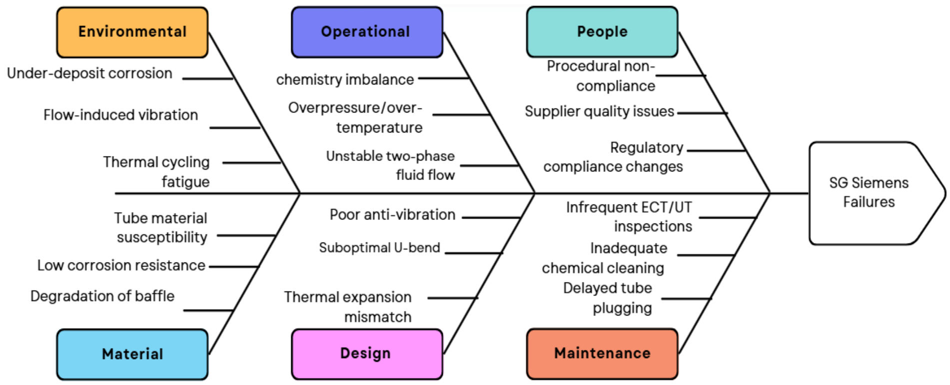

Figure 6. Fishbone Siemens Failure

The system issues are primarily due to the suboptimal design of the anti-vibration bar, leading to excessive vibrations; the use of Alloy 600, which is susceptible to Primary Water Stress Corrosion Cracking (PWSCC); operational factors such as water chemistry parameters, pressure/temperature, and fluid flow stability; failures in maintenance activities, including inspections, cleaning, and plugging criteria; environmental factors like corrosion, vibrations, and thermal fatigue; and a combination of human and external factors, such as training, procedures, supplier quality, and regulations.[13]

III. D. Damage Analysis of Westinghouse Steam Generator

Table 4. Analysis of Westinghouse Steam Generator

Failure Mode	Brief Description	Primary Controls	S	O	D	RPN
Stress Corrosion Cracking (SCC)	Primary-to-secondary leakage due to a corrosive environment, mechanical stress, or susceptible materials.[23], [24]	Eddy current inspections, corrosion-resistant alloys	9	4	3	108
Fouling (Deposits)	Reduced heat transfer and overheating from secondary water impurities (Fe, Cu, CaCO33).[25], [26]	Water chemistry control, periodic flushing	6	6	5	180
Crevice Corrosion	Structural damage from stagnant water and chloride/sulfate concentration in crevices.[27], [28]	Minimal crevice design, corrosion-resistant materials	7	5	4	140
Phosphate Wastage	Tube wall thinning due to erosion and high phosphate concentration.[29]	Chemistry monitoring, inspections	8	4	4	128
Primary Water Stress Corrosion Cracking (PWSCC)	Axial cracks in U-bends from low-pH water and residual stress.[30]	PWSCC-resistant Alloy 690, strict inspections	9	4	4	90

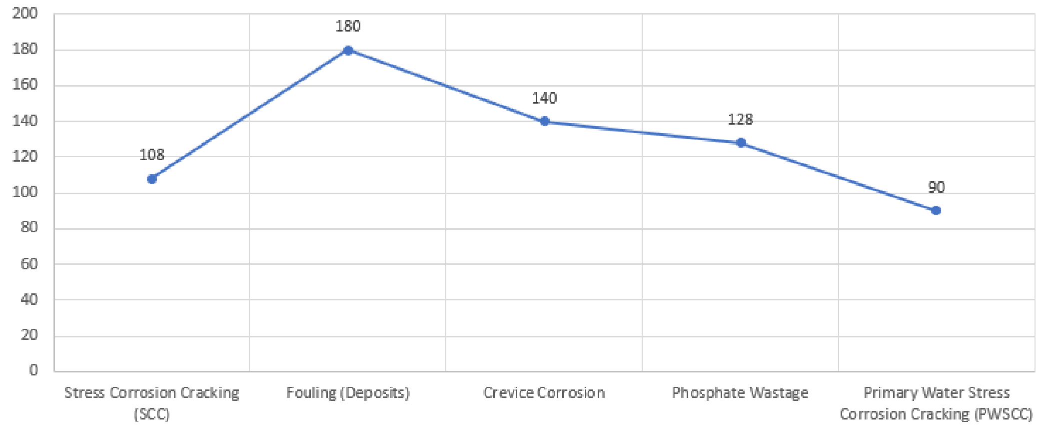

Figure 7. Damage/Failure Value Graph

The highest risk occurs in fouling, with an RPN value of 180. This is caused by Fe/Cu ion contamination in the secondary water and suboptimal filter design. The impact includes the accumulation of deposits, which then obstruct water flow, leading to secondary overheating. This event can result in a reduction in nuclear plant efficiency. If left unaddressed, extreme fouling could trigger an emergency shutdown for cleaning. The high RPN value is due to the difficulty in detecting fouling or the delays in detection.

The lowest risk is associated with PWSCC (Primary Water Stress Corrosion Cracking), with issues arising from Alloy 600's vulnerability in low pH environments and thermal stresses. As a result, microcracking can occur, leading to primary leakage, causing contamination of the secondary circuit. If this happens, a forced shutdown would be required. The RPN value for PWSCC is relatively low (90) because its detection is easier to carry out.

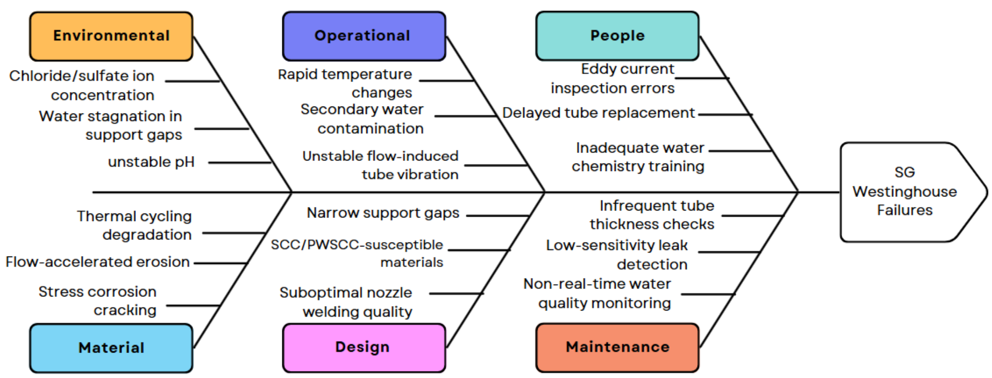

Figure 8. fishbone Westinghouse Failure

The system issues are due to several factors: design and manufacturing flaws, such as the use of Alloy 600, which is susceptible to SCC/PWSCC, and suboptimal nozzle welding quality; material degradation, including stress corrosion cracking in high pH environments and degradation caused by thermal cycling; operational factors, such as sudden temperature changes during startup and shutdown and secondary water contamination with elements like Fe, Cu, and $CaCO_3$, which exacerbate degradation; environmental conditions, including chloride and sulfate ion concentrations in water and uncontrolled pH levels in both primary and secondary circuits, that contribute to corrosion; human factors, such as inspection errors or procedural mistakes in performing eddy current inspections, which can lead to undetected issues; and inadequate inspection and monitoring, with a leak detection system that is not sensitive enough and water chemistry monitoring that is not real-time, delaying the detection of potential problems.

III. E. Damage Analysis of HTGR Steam Generator

Table 5. Analysis of HTGR Steam Generator

Failure Mode	Brief Description	Primary Controls	S	O	D	RPN
Creep Deformation	Pipe wall thinning and leakage risk due to sustained thermal/mechanical loads and inadequate creep-resistant materials.[31], [32], [33]	High-temperature alloys (Inconel 617), ultrasonic inspections	9	3	4	108
Thermal Fatigue	Cracking and fluid leakage from thermal cycling (startup/shutdown) and extreme gradients.[34]	Thermal expansion-compatible design, CFD analysis	8	4	5	160
Oxidation Corrosion	Material thinning and structural weakness caused by helium impurities (H22O, CO) at >800°C.[35], [36]	High-purity helium, anti-oxidation coatings (Al22O33)	8	4	4	128
Stress Corrosion Cracking	Secondary leakage due to residual stress and corrosive water (O22/Cl⁻).[37], [38]	pH control (9–10), SCC-resistant alloys (Alloy 800H)	9	4	3	108

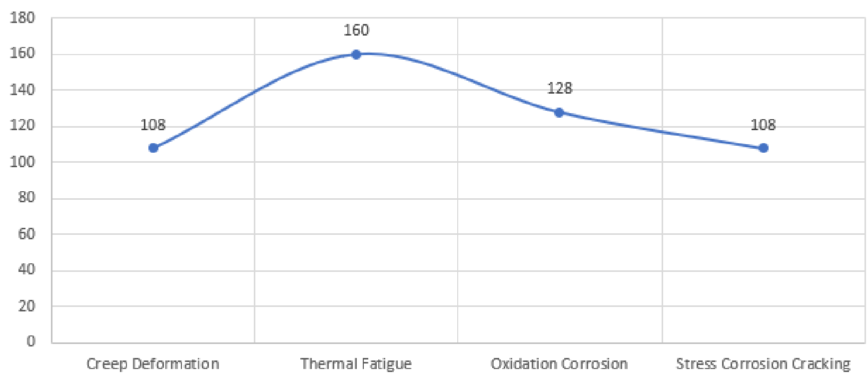

Figure 9. Damage/Failure Value Graph

The highest risk is Thermal Fatigue, which occurs due to repeated thermal cycles, such as those during startup/shutdown, causing material expansion/contraction. Extreme temperature gradients in the helical pipe area can lead to intergranular/circular cracks in the pipe, secondary or primary fluid leakage, and a reduction in heat transfer efficiency by up to 30%. The high RPN value is caused by high-pressure helium/water leakage, which poses a risk of reactor cooling loss. Thermal cycles are an inevitable part of reactor operation. Microcracks are difficult to detect without regular non-destructive testing (NDT). The lowest risk in this HTGR Steam Generator is Creep Deformation caused by long-term thermal/mechanical load at temperatures above 800°C. This is due to the material, Inconel 617/Alloy 800H, designed to withstand creep up to 950°C. Deformation can be detected via ultrasonic wall thickness measurement. The next lowest risk is Stress Corrosion Cracking (SCC) with an RPN value of 108, caused by a combination of residual stress and a corrosive environment (e.g., chloride-containing water). However, chemical control can be strictly maintained, ensuring the secondary water's pH stays between 9–10 and oxygen levels remain below 5 ppb. The material used, Alloy 800H, has high SCC resistance. When broken down in a fishbone diagram, HTGR damage can occur due to the following factors.

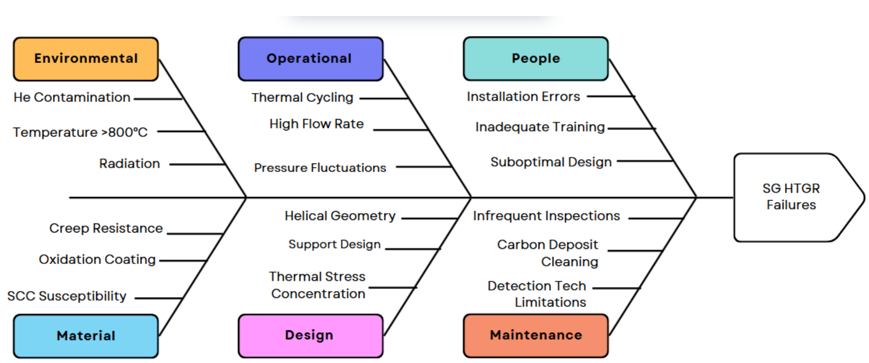

Figure 10. fishbone HTGR Failure

The system issues are primarily due to material problems, including plastic deformation caused by high temperatures, surface oxidation of the pipe from exposure to helium impurities, and cracking resulting from a combination of stress and a corrosive environment; design flaws, such as stress concentration at bend areas and uncontrolled helium flow vibration; operational challenges, like thermal cracking due to temperature gradients and turbulence and vibration caused by high flow velocity; environmental factors, including helium contamination from water or CO that triggers corrosion and material degradation due to neutron exposure; maintenance shortcomings, specifically the failure to detect microcracks early; and human factors, notably installation errors in the alignment of helical pipe sections.

III. F. Damage analysis of steam generator

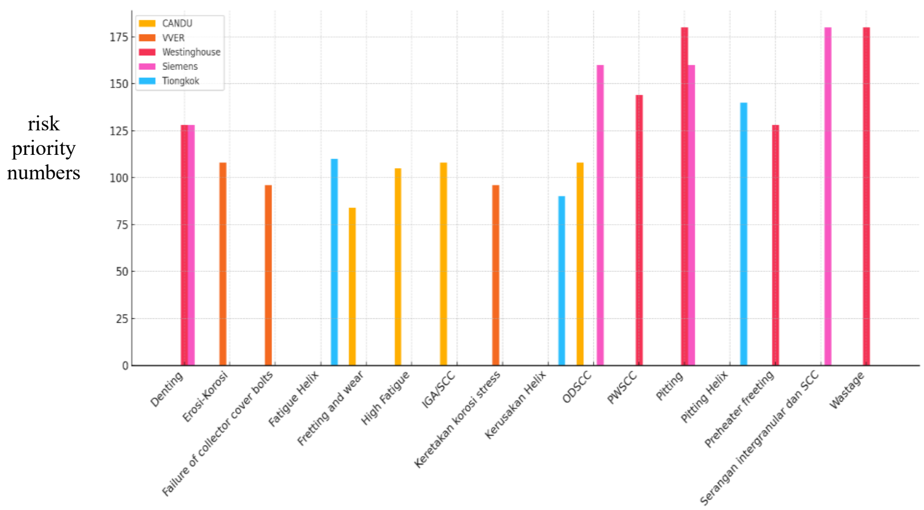

Figure 11. Diagram Of Damage Across Various Designs

The study diagram explains the distribution of damage across various designs in five plants (CANDU, VVER, Westinghouse, Siemens, and China). Based on the available data, it can be concluded that for the CANDU design, the most frequent damages are IGA/SCC and ODSC, with equal percentages, indicating that corrosion is the main challenge in this design. In the VVER design, the most common damage is erosion-corrosion, suggesting that the material and water handling must be given more attention. For the Westinghouse design, phosphate wastage and pitting are the dominant types of damage, signaling issues with water chemistry control and contamination. In the Siemens design, intergranular attack, SCC, and pitting are the most frequent types of damage, indicating challenges in managing corrosion stress and chemical environment control. In the Chinese design, the main damage is helix pitting, indicating that the helix tube design requires special attention to material fatigue and corrosion. Overall, pitting and corrosion occur frequently in various steam generator designs, requiring more attention in material development and water control methods.[13]

The chart serves as a visual tool for risk assessment, enabling stakeholders to identify and prioritize failure modes with the highest RPN values. The prominence of Siemens reactors with RPN values of 150 for multiple failure modes underscores the need for targeted interventions, such as enhanced maintenance, improved material selection, or design modifications, to mitigate these risks. Westinghouse and China, with RPN values of 125 for specific modes, also require attention, particularly for Dewetting, ODSCC, and PWSCC, which are critical in nuclear safety contexts.

The moderate RPN values for CANDU and VVER (around 100) suggest lower immediate risks, but these still warrant monitoring to prevent escalation. The absence of data for certain failure modes across some reactor types may indicate either low relevance (e.g., Helical Pretensioner not applicable to CANDU) or a lack of data, which could be an area for further research. This comparison facilitates informed decision-making in risk management, ensuring resources are allocated efficiently to address the most critical risks first, thereby enhancing the safety and reliability of nuclear reactor operations.

In the CANDU Steam Generator, High Cycle Fatigue and Intergranular Attack/Outside Diameter Stress Corrosion Cracking (IGA/ODSCC) exhibit the highest Risk Priority Number (RPN) of 108, indicating a significant failure risk due to cyclic stress and under-deposit corrosion. Fretting/Wear (RPN 84) arises from fluid-induced vibration and suboptimal design of the Anti-Vibration Bar (AVB). This is further exacerbated by the use of older materials (Alloy 600MA), which are highly susceptible to Stress Corrosion Cracking (SCC). Inadequate AVB design and the accumulation of chemical deposits (chlorides/sulfates) aggravate corrosion processes. Therefore, it is recommended to replace the material with corrosion-resistant alloys such as Alloy 690TT or 800NG, optimize AVB design, and enhance routine chemical cleaning to reduce deposit buildup.

In the Russian VVER Steam Generator, Thermal Fatigue (RPN 140) is predominantly caused by extreme operational temperature transients, while Tube Degradation (RPN 120) is attributed to SCC and abrasive flow. The horizontal design leads to sludge stagnation and uneven feedwater distribution, and the use of Inconel 600/800 makes the tubes vulnerable to cyclic cracking. Thus, improvements are necessary in feedwater distribution, thermal flexibility of the design, the use of high-temperature-resistant materials, and the implementation of routine thermographic inspections.

In the Siemens Steam Generator, Tube Rupture (RPN 252) poses a high risk of Loss of Coolant Accident (LOCA), and Tube Corrosion (RPN 200) results from chemical imbalance in the water. These issues stem from ineffective AVB design, the vulnerability of Alloy 600 to Primary Water Stress Corrosion Cracking (PWSCC), and operational factors such as fluctuating water chemistry and flow-induced vibrations. Therefore, it is recommended to replace materials with Alloy 690, increase the frequency of Eddy Current Testing (ECT), optimize the AVB design, and implement real-time water chemistry control.

In the Westinghouse Steam Generator, Fouling (RPN 180) due to Fe/Cu ion contamination reduces thermal efficiency, while Crevice Corrosion (RPN 140) is triggered by water stagnation and chloride concentration. These issues are caused by suboptimal secondary filter design, the susceptibility of Alloy 600, and poor inspection procedures with non-real-time chemical monitoring. Therefore, there is a need to improve water chemistry control, implement regular deposit cleaning, use corrosion-resistant materials such as Alloy 800H, and optimize joint design.

Compared to these, in the HTGR Steam Generator, Thermal Fatigue (RPN 160) is caused by startup/shutdown cycles and extreme temperature gradients, while Oxidation Corrosion (RPN 128) results from helium impurities at temperatures above 800°C. This is likely due to the plastic deformation of Inconel 617/800H at elevated temperatures and the stress concentration in the helical tube design. To address these issues, it is recommended to apply anti-oxidation coatings (e.g., Al_2O_3), maintain helium purity, and conduct Computational Fluid Dynamics (CFD) analysis to optimize flow distribution and reduce vibration.

IV. CONCLUSIONS

In conclusion, the comparative analysis highlights the diverse performance and risk profiles of steam generator designs across nuclear reactor types. Helix models offer superior heat transfer but are prone to structural degradation, as seen in high RPN for helix-related failures in Westinghouse and Siemens, while shell-and-tube models, prevalent in these reactors, are pressure-resilient but face higher corrosion risks, evidenced by elevated RPN for SCC and wastage. Optimization strategies should focus on material selection, design enhancements, and tailored maintenance to mitigate these risks, ensuring improved safety and efficiency. This research underscores the importance of customized risk management for each reactor type, contributing to the advancement of nuclear technology in Indonesia and globally, promoting sustainable energy solutions.

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REFERENCES

[1] P. Liu *et al.*, “Quantitatively evaluating the contribution of intergranular carbides, Cr-depleted zone, and grain boundary to intergranular stress corrosion cracking of Alloy 600 in a simulated boiling water reactors environment with high oxygen concentrations,” *Acta Mater*, vol. 269, 2024, doi: 10.1016/j.actamat.2024.119794.

[2] D. Lucan, “Behaviour of the steam generator tubing in water with different pH values,” *Nuclear Engineering and Design*, vol. 241, no. 4, pp. 1172–1176, Apr. 2011, doi: 10.1016/j.nucengdes.2010.03.029.

[3] D. Lucan, “CANDU steam generator corrosion due to the impurities of concentration and deposition,” in *Steam Generators: Design, Types and Applications*, 2017, pp. 197–225.

[4] C. W. Turner, M. Huang, and A. McKay, “Hideout, hideout return and crevice chemistry in nuclear steam generators,” in *Steam Generators for Nuclear Power Plants*, Elsevier Inc., 2017, pp. 273–321. doi: 10.1016/B978-0-08-100894-2.00012-1.

[5] N. Fasya Fashihah Khalid, Z. Mohd Salleh, I. Amin Ishak, R. Hanizam Madon, I. Taib, and M. Fairuz Marian, “Advances in Fluid, Heat and Materials Engineering Analysis of the Performance of a Pressure Vessel Structure: A Numerical Method Approach,” vol. 2, pp. 23–35, 2024, doi: 10.37934/afhme.2.1.2335.

[6] R. L. Tapping, J. Nickerson, P. Spekkens, and C. Maruska, “CANDU steam generator life management,” 2000. [Online]. Available: www.elsevier.com/locate/nucengdes

[7] C. L. Verona and R. L. Higginson, “Stress corrosion cracking of CrNi martensitic stainless steels for steam turbine blades,” *Materials at High Temperatures*, vol. 26, no. 1, pp. 79–83, 2009, doi: 10.3184/096034009X438202.

[8] O. Horner, E.-M. Pavageau, F. Vaillant, and O. De Bouvier, “Laboratory results of stress corrosion cracking of steam generator tubes in a ‘Complex’ environment - an update,” in *EUROCORR 2004 - European Corrosion Conference: Long Term Prediction and Modelling of Corrosion*, 2004. [Online]. Available: https://www.scopus.com/inward/record.uri?eid=2-s2.0-84867263338&partnerID=40&md5=fbd37e50d3e7ebfbe10f008da44e8df3

[9] R. Adibi-Asl and R. Seshadri, “Thermal hot spot damage assessment in pressure vessels,” in *American Society of Mechanical Engineers, Pressure Vessels and Piping Division (Publication) PVP*, 2016. doi: 10.1115/PVP2016-63903.

[10] R. Adibi-Asl, M. Rana, R. Seshadri, and C. Joshi, “Simplified procedure to assess hot spots in pressure vessels,” in *American Society of Mechanical Engineers, Pressure Vessels and Piping Division (Publication) PVP*, 2020. doi: 10.1115/PVP2020-21825.

[11] IAEA, “Assessment and management of ageing of major nuclear power plant components important to safety: Steam generators,” 1997.

[12] D. Féron and J.-M. Olive, *Corrosion issues in light water reactors: Stress corrosion cracking*. 2007. doi: 10.1533/9781845693466.

[13] Andini Efriyanto Putri, “ANALISIS PERBANDINGAN DESAIN, MATERIAL DAN KINERJA STEAM GENERATOR MODEL HELIX DAN MODEL SHELL&TUBE REAKTOR NUKLIR ,” *https://repository.pnj.ac.id* , 2024.

[14] W. Tina, E. Donaldson, and T. E. Dickinson, “FAILURE ANALYSIS OF OTSG TUBE IN A COGENERATION POWER PLANT,” in *American Society of Mechanical Engineers, Power Division (Publication) POWER*, 2023. doi: 10.1115/POWER2023-110584.

[15] S.-H. Jeong, B.-J. Lee, and Y.-Z. Lee, “Wear transitions of tube-support materials for a nuclear steam generator through sliding wear test and fretting wear test,” in *Key Engineering Materials*, 2006, pp. 430–433. doi: 10.4028/0-87849-412-x.430.

[16] C. Y. Park and J. K. Lee, “Wear scar progression of impact-fretting at elevated temperature for steam generator tubes in nuclear power plants,” in *Key Engineering Materials*, 2006, pp. 1251–1254. doi: 10.4028/0-87849-415-4.1251.

[17] S. S. Hwang, H. Pyo Kim, and J. Soo Kim, “Evaluation of the burst characteristics for axial notches on SG tubings,” *Nuclear Engineering and Design*, vol. 232, no. 2, pp. 139–143, 2004, doi: 10.1016/j.nucengdes.2004.06.003.

[18] H. P. Kim, S. S. Hwang, D. J. Kim, J. S. Kim, Y. S. Lim, and M. K. Joung, “Stress corrosion cracking of a kori 1 retired steam generator tube,” in *EUROCORR 2004 - European Corrosion Conference: Long Term Prediction and Modelling of Corrosion*, 2004. [Online]. Available: https://www.scopus.com/inward/record.uri?eid=2-s2.0-84867251478&partnerID=40&md5=236c0c98f1562647c939d005d45850fe

[19] E. Merzari *et al.*, “High-fidelity simulation of flow-induced vibrations in helical steam generators for small modular reactors,” in *17th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, NURETH 2017*, 2017. [Online]. Available: https://www.scopus.com/inward/record.uri?eid=2-s2.0-85052517479&partnerID=40&md5=78e05efd82aabb56498acf686842d478

[20] S. Khushnood, Z. M. Khan, M. A. Malik, Z. Koreshi, and M. A. Khan, “Modeling and simulation of cross-flow induced vibration in a multi-span tube bundle,” in *Proceedings of the International Conference on Nuclear Engineering (ICONE12)*, 2004, pp. 253–264. doi: 10.1115/icone12-49161.

[21] L. Yang, X. Li, Y. Shi, and H. Chang, “Cfd simulated calculation of the test section of flow induced vibration for U-tubes in the two-phase flow,” in *17th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, NURETH 2017*, 2017. [Online]. Available: https://www.scopus.com/inward/record.uri?eid=2-s2.0-85052469457&partnerID=40&md5=0a2134dc39956a85f56de3e8cda0e45e

[22] C. W. Turner, M. Huang, and A. McKay, “Hideout, hideout return and crevice chemistry in nuclear steam generators,” in *Steam Generators for Nuclear Power Plants*, 2017, pp. 273–321. doi: 10.1016/B978-0-08-100894-2.00012-1.

[23] C. L. Verona and R. L. Higginson, “Stress corrosion cracking of CrNi martensitic stainless steels for steam turbine blades,” *Materials at High Temperatures*, vol. 26, no. 1, pp. 79–83, 2009, doi: 10.3184/096034009X438202.

[24] J. Wilson, "Controlling alloy 600 degradation," *Nuclear Plant Journal*, vol. 27, no. 2, pp. 36–37, 2009, [Online]. Available: https://www.scopus.com/inward/record.uri?eid=2-s2.0-67651096011&partnerID=40&md5=7ee084e80a493012b95f39765958fd35

[25] C. W. Turner, "Implications of steam generator fouling on the degradation of material and thermal performance," in *15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors 2011*, 2011, pp. 2162–2174. [Online]. Available: https://www.scopus.com/inward/record.uri?eid=2-s2.0-84867609048&partnerID=40&md5=96d24544168c9e4401ec680dceb6d175

[26] J. Mei, Y. Han, P. Chu, F. Xue, and Q. Peng, "Survey on fouling of steam generators in nuclear power plants," *Corrosion Science and Protection Technology*, vol. 30, no. 4, pp. 435–440, 2018, doi: 10.11903/1002.6495.2017.201.

[27] C. B. Bahn *et al.*, "Impurity concentration behaviors in a boiling tubesheet crevice: Part I. Open crevice," *Nuclear Engineering and Design*, vol. 225, no. 2–3, pp. 129–144, 2003, doi: 10.1016/S0029-5493(03)00179-1.

[28] C. B. Bahn, K. E. Kasza, J. Park, and W. J. Shack, "Experimental simulation of Na and Cl hideout in steam generator crevices," *Nuclear Engineering and Design*, vol. 250, pp. 156–166, 2012, doi: 10.1016/j.nucengdes.2012.05.019.

[29] L. Qiu, G. R. Burton, S. Rousseau, and J. Qian, "Kinetics and Thermodynamics of Sulfate Adsorption on Magnetite at Elevated Temperatures," *J Solution Chem*, vol. 48, no. 11–12, pp. 1488–1502, 2019, doi: 10.1007/s10953-019-00933-8.

[30] Y. Lu, "Effect of hazardous impurities on steam generator tube degradation," in *International Conference on Nuclear Engineering, Proceedings, ICONE*, 2010, pp. 283–291. doi: 10.1115/ICONE18-30120.

[31] R. I. Jetter, T.-L. Sham, and R. W. Swindeman, "The impact of material and design criteria on the assessment of negligible creep," in *American Society of Mechanical Engineers, Pressure Vessels and Piping Division (Publication) PVP*, 2010, pp. 717–726. doi: 10.1115/PVP2009-77953.

[32] R. I. Jetter, T.-L. Sham, and R. W. Swindeman, "Application of negligible creep criteria to candidate materials for HTGR pressure vessels," *Journal of Pressure Vessel Technology, Transactions of the ASME*, vol. 133, no. 2, 2011, doi: 10.1115/1.4001919.

[33] S. Wu, Y. Chen, Y. Yang, Y. Zhu, Y. Liu, and Y. Fan, "Validation analysis of high pressure turbine creep deformation for Ultra-supercritical steam turbine," in *Proceedings of the ASME Turbo Expo*, 2017. doi: 10.1115/GT2017-63059.

[34] T. Kokufuda and N. Kasahara, "Study on simplified prediction method of thermal ratchet deformation based on parallel bar model," in *American Society of Mechanical Engineers, Pressure Vessels and Piping Division (Publication) PVP*, 2014. doi: 10.1115/PVP2014-28631.

[35] W. Zheng *et al.*, "Oxidation behaviors of the high temperature alloys in the impure helium and argon," in *International Conference on Nuclear Engineering, Proceedings, ICONE*, 2021. doi: 10.1115/ICONE28-63659.

[36] H. Li *et al.*, "Study on Decarburization and Oxidation Corrosion Behavior of T-22 Alloy in Impure Helium of High-temperature Gas-cooled Reactor," *Hedongli Gongcheng/Nuclear Power Engineering*, vol. 43, no. 4, pp. 38–45, 2022, doi: 10.13832/j.jnpe.2022.04.0038.

[37] F. N. Smith, "DAMAGE CAUSED BY STRESS CORROSION CRACKING Stress corrosion cracking is an insidious form of damage that can occur when a susceptible metal is subjected to a tensile stress in a specific environment.," *Advanced Materials and Processes*, vol. 182, no. 7, pp. 27–29, 2024, doi: 10.31399/asm.amp.2024-07.p027.

[38] Q. Wang, W. Zheng, H. Yin, S. Li, X. He, and T. Ma, "Corrosion of high temperature alloys in the primary coolant of HTGR under very-high temperature operation," in *International Conference on Nuclear Engineering, Proceedings, ICONE*, 2020. doi: 10.1115/ICONE2020-16074.