

# High-Temperature Water Corrosion Fatigue Mechanism of Nuclear-Grade Low-Alloy Steel and Environmental Fatigue Design Model: Post-print

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## Abstract

Through simulation of high-temperature, high-pressure circulating water corrosion fatigue experiments in nuclear power environments, this study investigated the environmental fatigue damage patterns and control mechanisms of nuclear-grade low-alloy steel, constructed a fatigue design model incorporating environmental effects, presented environmental fatigue design curves suitable for engineering applications, established an environmental fatigue safety assessment procedure for actual nuclear power plant components, and provided an implementation example.

## Full Text

### Corrosion Fatigue Mechanism of Nuclear-Grade Low Alloy Steel in High-Temperature Pressurized Water and Its Environmental Fatigue Design Model

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## Abstract

The service degradation and life assessment of key components in light water reactor nuclear power plants (NPPs) mainly depend on the accumulation of service property data of component materials, understanding of environmental degradation mechanism, and construction of evaluation models or methods. The current ASME design fatigue code does not take full account of the interactions of environmental, loading and material' s factors. In the present work, based on the corrosion fatigue tests in simulated NPPs' high temperature pressurized water, the environmental fatigue behavior and dominant mechanism of nuclear-grade low alloy steel have been investigated. A design fatigue model was constructed by taking environmentally assisted fatigue effects into account and the corresponding design curves were given for the convenience of engineering applications. The process for environmental fatigue safety assessment of NPPs' components was proposed, based on which some tentative assessment cases have been given.

**Keywords:** nuclear-grade low alloy steel, high temperature pressurized water, corrosion fatigue, design model, environmental fatigue safety assessment

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## 1. ASME Fatigue Design Curves and EAF Effects

The service degradation and life assessment of key equipment in light water reactors (LWRs) nuclear power plants throughout their entire lifecycle, particularly for the reactor coolant system (RCS) pressure boundary, represents a critical issue affecting both operational safety and economic viability. Most commercial LWRs worldwide have been in service for 20-40 years and now face aging and life extension challenges. However, current understanding of the degradation mechanisms, life design, and prediction models for nuclear equipment materials remains limited, necessitating urgent research efforts. From the perspectives of nuclear safety and engineering design, ensuring structural integrity of the RCS throughout its service life requires reasonable design curves. Nuclear power countries have established their own design standards, such as ASME BPV in the United States, RCC in France, RSK in Germany, and MITI 501 in Japan [1], for the design, operation, and life assessment of nuclear power plant pressure boundaries. China' s commercial nuclear power plants have relatively short service histories, and due to the lack of effective simulation experimental techniques and damage evaluation methods, the design, manufacturing, operation, supervision, damage assessment, and life extension of key equipment must currently

rely on foreign standards.

Recent experimental studies [2-7] have demonstrated that even the ASME fatigue design curves, widely adopted in pressure boundary equipment engineering design, may have insufficient safety margins under certain combined loading and environmental conditions. Consequently, in 2007, the U.S. Nuclear Regulatory Commission (NRC) issued Regulatory Guide 1.207 [8], requiring that fatigue assessments for new LWR structural materials must consider environmentally assisted fatigue (EAF) effects, presenting new challenges for designers, owners, regulators, operators, and researchers. Therefore, investigating the service degradation mechanisms, life design, and prediction models for domestic nuclear materials is crucial for enhancing China's nuclear safety design and regulatory capabilities while promoting key technology independence.

The pressure boundary components in LWRs nuclear power plants operate in high-temperature pressurized water environments. Due to thermal stratification, start-up/shutdown cycles, and flow-induced vibration, these components may experience environmental fatigue damage [9-11]. As a potential failure mode for nuclear equipment, corrosion fatigue is critical for safety design, manufacturing, operation, inspection, regulation, safety assessment, and life management. The currently applied ASME fatigue design curves were developed by fitting low-cycle fatigue data obtained in room-temperature air, then dividing strain by 2 or life by 20, taking the minimum value [12]. While the factor of 20 includes data scatter (2), size effect (2.5), surface finish and environment (4), it does not adequately account for LWR service environment effects. Since Japanese researchers first discovered in the 1970s that high-temperature pressurized water environments affect material fatigue performance, decades of experimental research have confirmed that under specific combinations of environment, loading, and material conditions, high-temperature pressurized water significantly reduces the fatigue life of nuclear structural materials (carbon steel, low alloy steel, stainless steel, and nickel-based alloys), demonstrating pronounced EAF effects [2-7,10,11]. For example, low alloy steel with sulfur content above 0.15 wt% exhibits fatigue life reductions of over 100 times in water at 300°C with dissolved oxygen greater than  $1 \times 10^{-6}$  and strain rates below  $10^{-3} \% \cdot s^{-1}$  compared to air [6,10].

In response, Argonne National Laboratory in collaboration with the Japan Nuclear Energy Safety Organization and other institutions attempted to establish the ANL and JSME models considering environmental factors. The ASME Committee also began addressing EAF effects, issuing Code Cases N-761 and N-792 in 2010, recommending consideration of EAF in design curves and discussing revisions to the original ASME fatigue design curves. Based on the ANL fatigue model, the NRC issued Regulatory Guide 1.207 in 2007 [8], requiring new nuclear power plants to fully consider LWR environment effects on structural material fatigue performance and recommending use of the ANL model's environmental correction factor  $F_{en} = N_{air} / N_{env}$ , (where  $F_{en}$  is the environmental factor,  $N_{air}$  is fatigue life in room-temperature air, and  $N_{env}$  is fatigue

life in high-temperature pressurized water) to calculate cumulative fatigue damage factors  $U = \Sigma(F_{en}, U)$  (where  $F_{en}$ , is the environmental factor for damage process  $i$ ,  $U$  is the cumulative fatigue damage factor at room temperature for process  $i$ , and  $i = 1, 2, \dots, n$ ).

As China actively develops nuclear power and strives for technological independence in design, manufacturing, operation, and assessment, it faces EAF challenges. Neither ASME' s N-761 and N-792 nor the American ANL and Japanese JSME models include environmental fatigue strength data for domestic nuclear structural materials. Therefore, accumulating environmental fatigue strength data for domestic materials, developing independent fatigue life models and design curves, and evaluating environmental fatigue safety margins for domestic LWR components is of great significance.

## 2. Simulated Nuclear High-Temperature Pressurized Water Corrosion Fatigue Experiments

The service environment of LWR pressure boundaries primarily involves high-temperature pressurized circulating water with specific water chemistry conditions, making simulation experimental apparatus the prerequisite for research. Key challenges include: (1) high-temperature pressurized water environment simulation technology, such as environments up to 360°C and 20 MPa; and (2) control of dissolved oxygen, dissolved hydrogen, pH, conductivity, and ion concentrations.

Experimental results show that fatigue crack initiation resistance in high-temperature pressurized water depends on strain rate: lower strain rates result in lower crack initiation resistance. At high strain rates ( $0.1 \% \cdot s^{-1}$ ), fatigue cracks grow in a zigzag pattern with macroscopic branching, and failure typically occurs through continuous propagation of a single crack [Figure 2: see original paper]. As strain rate decreases to  $0.01 \% \cdot s^{-1}$ , cracks become straighter, and at even lower rates ( $0.001 \% \cdot s^{-1}$ ), cracks propagate completely straight with numerous straight secondary cracks. Statistical analysis reveals that the number of cracks per unit surface area in the gauge section increases as strain rate decreases.

[Figure 3: see original paper] shows typical fatigue fracture morphologies [3]. At high strain rates, fracture surfaces are extremely rough with step-like features, showing fan-shaped quasi-cleavage patterns related to MnS inclusions (FIG. 3d), tear ridges, and terrace-like cracking features (FIG. 3b), consistent with hydrogen-induced cracking mechanisms observed in hydrogen-charged materials or stress corrosion experiments [18,20,24]. At low strain rates, fracture surfaces become relatively flat (FIG. 3e), with cracks perpendicular to the loading axis and slight crack arrest markings along the propagation direction (FIG. 3f), consistent with film rupture/slip dissolution mechanisms [21,25]. These results indicate that the controlling mechanism for corrosion fatigue cracking in nuclear-grade low alloy steel transitions from hydrogen-induced cracking to

film rupture/slip dissolution as strain rate decreases. Dissolution of MnS inclusions or increased dissolved oxygen content accelerates electrochemical corrosion processes within the fatigue crack system, thereby promoting corrosion fatigue cracking in high-temperature pressurized water.

#### 4. Environmental Fatigue Design Model and Design Curves for Nuclear-Grade Low Alloy Steel

Although extensive experimental results confirm that LWR high-temperature pressurized water environments affect fatigue damage mechanisms and life of structural materials, incorporating EAF effects rationally into design curves remains a critical challenge. For nuclear-grade low alloy steel, constructing an environmental fatigue design model requires quantifying all factors affecting fatigue life in high-temperature pressurized water based on substantial experimental data.

Using corrosion fatigue strength data for domestic nuclear-grade low alloy steel SA508-III and analyzing similar foreign materials (ASTM A508 Cl. 3, A533B Cl. 1), this work developed an environmental fatigue design model. Following the authors' previous methodology for nuclear-grade stainless steel [26] and ASME standards, the fatigue design model is based on the simplified Langer equation [27]. Assuming environmental degradation multiplies the air fatigue life by influence factors  $X$ , the Langer equation with environmental effects becomes:

$$\ln(\frac{N_{25}}{N_{air}})$$

where  $N_{25}$  is the fatigue life corresponding to peak stress dropping to 75% of maximum peak stress,  $\epsilon$  is strain amplitude,  $C$  is a material fatigue strength constant,  $P$  is a material- and temperature-related constant,  $B$  is a material constant, and  $Q$  is an environment-related constant. Determining coefficients  $P$ ,  $B$ ,  $Q$  and boundary conditions for influence factors  $X_1, X_2, \dots, X_n$  yields the environmental fatigue life design equation.

From air fatigue data for nuclear-grade low alloy steel (FIG. 4a), the best-fit equation is:

$$\ln(\frac{N_{25}}{N_{air}}) = 36.13 + 0.132 \ln(\epsilon) - 1.786 \ln(\frac{N_{25}}{N_{air}})$$

Fitting error analysis (FIG. 4b) shows good correlation. High-temperature pressurized water corrosion fatigue experiments indicate that primary factors affecting environmental fatigue life include strain rate factor  $X_{\dot{\epsilon}}$  (FIG. 5a), dissolved oxygen factor  $X_{DO}$  (FIG. 5b), steel sulfur content factor  $X_S$  (FIG. 5c), temperature factor  $X_T$  (FIG. 5d), and thermal aging factor  $X_t$  (FIG. 5e).

Combining air and environmental fatigue life models with boundary values of these factors yields the environmental fatigue life model:

$$\ln(\frac{N_{25}}{N_{air}}) = 4.154 - 1.786 \ln(\frac{N_{25}}{N_{air}}) + X_T X_t$$

$$X = 0 \quad (\dot{\epsilon} > 1 \% \cdot s^{-1})$$

$$X = \ln(\dot{\epsilon}) \quad (-6.908 \leq \dot{\epsilon} \leq -1 \% \cdot s^{-1})$$

$$X = \ln(0.001) \quad (\dot{\epsilon} < 0.001 \% \cdot s^{-1})$$

$$XDO = 0 \quad (wDO < 0.05 \times 10^{-6}) \quad XDO = \ln(25wDO) \quad (0.05 \times 10^{-6} \leq wDO \leq 0.5 \times 10^{-6}) \quad XDO = 2.526 \quad (wDO > 0.5 \times 10^{-6})$$

$$XS = \ln(1000 wS) \quad (wS \leq 0.015\%)$$

$$XS = 2.708 \quad (wS > 0.015\%)$$

$$XT = 0 \quad (T < 150^\circ C)$$

$$XT = (T - 150)/100 \quad (150 \leq T \leq 325^\circ C)$$

$$XT = 1.607 \times 10^{-6} \exp(0.01T) \quad (T > 325^\circ C)$$

$$Xt = 0 \quad (ta = 0)$$

$$Xt = \exp(-6.32 \times 10^{-5} ta) \quad (ta > 0)$$

where wDO is dissolved oxygen content (mass fraction), wS is steel sulfur content,  $\dot{\epsilon}$  is strain rate, T is environmental temperature, and ta is thermal aging time.

For engineering application, strain amplitude in the environmental fatigue design model was converted to stress by multiplying by the elastic modulus and corrected for mean stress using the Goodman relationship. Considering experimental data scatter, specimen size, and surface roughness effects, the curves were conservatively adjusted by dividing fatigue life by 20 and stress by 2, taking the minimum values to obtain environmental fatigue design curves (FIG. 6). The resulting environmental fatigue design curves are more conservative than ASME standard curves. Comparison with experimental data shows that some high-temperature pressurized water corrosion fatigue data points fall below the ASME standard design curve, indicating insufficient safety margins under certain conditions, while the environmental fatigue design curve covers all experimental data points with adequate safety margins.

## 5. Environmental Fatigue Safety Assessment Process for Actual Components and Trial Implementation

Engineering fatigue assessment primarily calculates the cumulative fatigue damage factor U based on Miner's linear fatigue cumulative damage theory [28]:  $U = \sum(n_i/N_i)$  ( $i = 1, 2, 3, \dots, n$ ), where  $N_i$  is the number of cycles to failure under stress  $S_i$ , and  $n_i$  is the actual number of cycles at stress  $S_i$ . According to ASME codes, components are considered safe if  $U < 1$  and unacceptable if  $U > 1$ .

FIG. 7 illustrates the environmental fatigue life assessment process for specific NPP components. The procedure involves: (1) selecting the component for assessment; (2) identifying typical operational transients (shutdown, startup, heat-up, cool-down, pressure testing, inspection, reactor trip, refueling, etc.); (3) obtaining design cycles ( $n_1, \dots, n_n$ ) from design documents; (4) acquiring stress, temperature, and pressure versus time curves for each transient using

design data and field monitoring; (5) performing finite element analysis to obtain stress distributions and determine maximum/minimum stress values ( $P_1, P_2, \dots, P$ ) at critical locations; (6) calculating stress amplitude  $S = (S_1, S_2, \dots, S)$  per ASME procedures; (7) determining fatigue life  $N = (N_1, N_2, \dots, N)$  from ASME design curves; (8) obtaining environmental fatigue life  $N = (N_1, N_2, \dots, N)$  from the proposed environmental fatigue design curve; (9) calculating cumulative damage factors  $U = \Sigma(n/N)$  and  $U = \Sigma(n/N)$ . If both  $U$  and  $U$  are less than 1, the component is safe. If both exceed 1, the component is unacceptable. If  $U < 1$  but  $U > 1$ , EAF risk exists and requires further detailed analysis.

TABLE 1 presents a trial assessment for a boiling water reactor carbon steel feedwater pipe safe end [29–31]. Parameters related to operational transients ( $S, T, \dots$ ) and design cycles were obtained from literature [29–31]. Following the assessment process in FIG. 7, the calculated cumulative damage factors are  $U = 0.317$  and  $U = 2.81$ . While  $U < 1$  based on ASME design curves,  $U > 1$  using the environmental fatigue design curve, indicating potentially insufficient safety margins and potential failure risk requiring attention.

Using typical design basis cycles and anticipated cycles for 40 and 60 years (TABLE 2) [31], the environmental fatigue cumulative damage factors for a pressurized water reactor inlet nozzle are  $U_{40} = 0.472$  and  $U_{60} = 0.708$  (TABLE 3), both less than 1. This indicates adequate safety margins for service conditions and potential for life extension from 40 to 60 years.

## Conclusions

This work quantified key factors affecting the high-temperature pressurized water corrosion fatigue life of nuclear-grade low alloy steel, including strain rate, dissolved oxygen content, steel sulfur content, temperature, and thermal aging time. An environmental fatigue design model incorporating EAF effects was constructed, and engineering-applicable environmental fatigue design curves were developed. The established environmental fatigue design curves for nuclear-grade low alloy steel are more conservative than ASME standard curves with adequate safety margins. An environmental fatigue safety assessment process for actual NPP components was established with trial implementation cases provided.

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