AGING AND REACTOR WATER EFFECTS ON FATIGUE LIFE

By

W. J. O'Donnell

J. S. Porowski

E. J. Hampton

M. L. Badlani

G. H. Weidenhamer

D. P. Jones

J. S. Abel

Brian Tomkins

ABSTRACT

Methods of including aging effects and reactor water enhanced crack propagation rates in Codified S-N fatigue life assessment curves are presented and illustrated. Such methods are essential because it is not feasible to produce experimentally based S-N life evaluation curves for all of the reactor materials of interest under all of the relevant cyclic rate and environmental conditions of interest within available finite research funding.

Reactor water environmental effects are known to accelerate fatigue crack growth rates in reactor pressure vessel and piping materials. Recently developed advanced elastic-plastic fracture mechanics technology [Ref. 1] is used herein as a means of correcting S-N fatigue life evaluation curves for measured environmental crack growth rate effects. As an important illustration, ASME Code Section XI reactor water crack growth rate curves are used to generate revised new Section III and VIII fatigue design curves for A106 reactor piping. Reactor water effects on the fatigue life are found to be quite significant, and their inclusion in the S-N curves greatly improves the technical basis for assessing the residual component life which meets ASME Code safety margins for cumulative fatigue.

INTRODUCTION

Recognizing the increasing importance of nuclear plant license renewal, NRC published a Program Plan for Nuclear Plant Aging Research in July of 1985 [Ref. 2]. The research described in this Plan is intended to resolve issues related to the aging and service wear of equipment and systems at commercial reactor facilities and their possible impact on plant safety. Considerable emphasis is on the mechanisms of material and component degradation during service. One of the major program goals is to identify and characterize aging and service wear effects which, if unchecked, could cause degradation of structures, components, and systems and thereby impair plant safety.

Nuclear utilities need a sound technical basis for making plant, system and component repair and replacement decisions [Ref. 3]. Since a new plant requires many years to plan and construct, decisions concerning the expected operating life of existing plants must be made long before the end of the 40 year licensed life of the plant. The reliability and dependability of operating plant components are very important to utilities operating nuclear power plants. Forced outages are disruptive and costly.

In spite of recent advances, inspection technology is not yet totally reliable, and there are areas not accessible for inspection. S-N technology provides a reliable means of evaluating the safe operating fatigue life without depending on inspection results. It also covers vibratory and thermal high cycle loading conditions which could result in the rapid propagation of cracks too small to be detected by in-service inspection.

This report focuses on the development of evaluation methods which can be used to accurately predict the reliability and structural integrity of components, systems and piping. These criteria are intended to assure that regulatory and Code[1] safety margins used to design and license the plant are maintained during the extended period of operation, and that the reliability and dependability of the hardware is not seriously compromised by corrosion-assisted fatigue.

The fatigue design evaluation criteria for nuclear components in Section III of the ASME Code, Ref. [99], are based on 30-year-old technology and do not account for important environmental and aging factors which must be considered in evaluating the service life of components and piping. On-going systematic long-term research by NRC and others is based on fracture mechanics technology for the initiation, propagation and instability of cracks. The analytical methods developed in Ref. [1] combine the existing Codified S-N technology with elastic-plastic crack propagation technology to include reactor water environmental effects on crack growth rates in improved S-N criteria for nuclear components. These methods can also be extended to account for imperfections and residual stresses in weldments.

Most LWR plants have not experienced all the hypothetical transients and earthquakes included in the original fatigue design analyses. Thus, these plants will not have experienced as much fatigue usage or damage as anticipated in the original design stage, and there should be margin for life extension beyond the original 40-year life. However, the results of environmental degradation testing during the past fifteen years have shown

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[1] "Code" in this report refers to the ASME Boiler and Pressure Vessel Code.

that such effects are more deleterious than anticipated when the ASME Code adopted the current S-N fatigue design curves. Therefore, environmental and aging effects must be taken into account when reevaluating safe operating life. Improved fatigue life evaluation curves for A106 piping steel which include reactor water crack growth rates from Section XI of the ASME Code are obtained herein using the methods of Ref. [1].

Extensive data analyses are needed to evaluate the effect of reactor water environments on the crack initiation phase of fatigue failure and to evaluate the effects of weldment imperfections and residual stresses. The resulting new S-N curves will provide a means for utilities to evaluate current fatigue damage in their plants, taking into account the cumulative damage from operating transients and cycles which the plant has experienced. The safe residual life can then be evaluated using the same curves in order to assure compliance with Code safety margins. This plant life extension approach is applicable even where in-service inspections are not feasible. Of course, it does not account for fabrication or stress corrosion-induced cracking, which can be found by in-service inspection and evaluated using Section XI criteria.

Considerable fatigue data on BWR environmental cracking effects was generated in an EPRI-sponsored program [Refs. 4 and 5]. Much of this data was obtained on notched specimens and cannot be generalized to other geometries. The elastic-plastic crack analysis approach herein provides a method of using that data to evaluate environmental effects on crack initiation and propagation. Such results can then be used to generate S-N fatigue curves.

Research carried out under NRC sponsorship [Refs. 6 - 12] provides crack growth data directly usable in the general method of Ref. [1] developed herein for including such results in S-N curves. The effects of various material, environmental and operating parameters on crack initiation and propagation are being evaluated in on-going research worldwide (see for example Refs. [5 - 45]). Thus, these results can now be used to improve the S-N fatigue life evaluation curves.

COMPONENT FATIGUE TEST EXPERIENCE

Numerous fatigue tests have been run on pressure vessels, piping and components following the famous early work on piping components by Markl [Refs. 46 thru 49] beginning in the 1940's. The Pressure Vessel Research Committee and Atomic Energy Commission sponsored early low cycle fatigue tests on full size pressure vessels incorporating a variety of nozzle configurations of interest to the reactor designer and pressure vessel industry at large [Refs. 50 thru 53].

The local stress conditions were carefully measured using brittle coatings and nearly 900 strain gages. Peak stresses occurred at the inside corners of the nozzles and fatigue cracks developed at these locations. Crack initiation and crack growth were observed. A comparison of the crack initiation and failure points with the ASME Code design curves provides an indication of the safety margins provided by the current design criteria. No crack progressed thru-the-wall in less than three times the allowable number of design cycles.

While this comparison indicates that the current ASME Code design curves do provide the intended safety margins (see also Ref. [53]), the tests did not include the effects of reactor water corrosion-assisted fatigue crack growth anticipated during the operation of Light Water Reactors. Laboratory test results indicate that crack growth rates of LWR ferritic steels in reactor water at 550°F may be an order of magnitude higher than in air for certain material and water chemistry conditions with ΔK in the intermediate range. Moreover, the tests did not include the spectrum of thermal and vibration cycles experienced under actual reactor operating conditions.

Since these tests were run, numerous such tests have been run on various components worldwide. In the U. K., where elastic shakedown is used as a basic design criteria, extensive tests have been run on full-size components.

Cracks often initiate early at acceptable imperfections in weldments. While some of these cracks stop propagating as they progress beyond the stress raising influence of the imperfection, others may continue to propagate, limiting the fatigue life. In the high-cycle regime, cracks initiate late in the fatigue life, and propagate thru-wall in a short time.

The NRC sponsored a Battelle study [Ref. 54] and evaluation of available fatigue test data on piping products which was intended to determine the margin of safety of the ASME Code design curves. These tests of piping products form important bases and justification for the fatigue evaluation procedures used for nuclear power plants. However, as pointed out in the report, the tests were run over a relatively short period of time (days or weeks) as compared to the 40-year design life of a nuclear plant. Accordingly, the tests do not encompass environmental effects. Corrosion-assisted fatigue crack growth is chemistry, temperature and cyclic rate dependent. NUREG/CR-0325 [Ref. 54] concludes: "The most important discrepancy exists in assessing the effect of the environment over the 40 years of PWR plant life. If the environmental attack is significant, the cold leg behavior may not be conservative with respect to the fatigue test results. To accurately evaluate the margin of safety inherent in the cold leg piping, an analysis technique that accounts for the environmental factors must be employed."

As part of the Nuclear Plant Aging Research Program, the NRC sponsored an ORNL study [Refs. 55 - 58] of available sources of LWR operating experience contained in Licensee Event Reports (LER's) to identify and evaluate age-related events and trends which could result in compromising a safety function. Fatigue, corrosion, vibration and cracks were found to be very significant factors. Data collected [Ref. 55] on domestic commercial nuclear power plants covering 1969 to 1983 yielded 324 failures specifically reported as fatigue failures. 75 percent of these failures were judged to be degraded failures which placed plant operation outside the Technical Specifications, and 25 percent were judged as catastrophic failures. An additional 259 failures were keyworded "crack", 165 were reported as "vibration" failures, 110 as "stress-corrosion" failures and 414 as "corrosion" failures. These results indicate that corrosion effects contributed to a large percentage of the failures, and that corrosion-assisted fatigue should be included in the S-N design curves.

Typically, fabrication-related problems tend to show up early in the operating life of a plant. This is followed by a long period of relatively trouble-free operation until fatigue and other aging phenomena begin to produce failures. Such failures then tend to occur with increasing frequency as the systems, components and piping approach the end of their useful lives.

AGING EFFECTS ON FATIGUE LIFE

The effects of aging on fatigue life have yet to be quantified and methods of doing so are described herein. Concerns include thermal aging and the associated loss of toughness, reduced ductility, via increased notch sensitivity.

Thermal embrittlement, well known in cast stainless steels, is expected to be far less significant in wrought materials. The associated loss in toughness and ductility would not be expected to have a significant effect on fatigue life, except in cast stainless steels and possibly in weldments.

Fatigue data correlates very consistently with fracture ductility, following the Langer-Coffin relation from a tensile test to the high cycle regime:

$$S_a = \frac{E}{4\sqrt{N}} \ln\left[\frac{100}{100 - R.A.}\right] + S_e$$

where

E = elastic modulus (psi)

N = number of cycles-to-failure

S_e = endurance limit (psi)

R.A. = Reduction in Area in tensile test

This equation was originally derived by using the true strain at fracture in a tensile test as a point on the S_a vs. N fatigue failure curve, assuming the existence of a well-defined endurance limit. It provides a good fit of strain-controlled fatigue data for most materials out to 10^6 cycles. Of course, more complex equations with additional parameters can be used to achieve better statistical fits to the S_a vs. N data. Beyond 10^6 cycles, where fatigue is controlled by crack initiation, the effects of loading sequence, mean stress effects, environmental and cyclic rate effects introduce major complexities, which are important for thermal mixing and vibratory loads.

However, the above equation is the basis of the existing ASME Code fatigue design curves and can be used to include any reduction in ductility due to aging in the S-N fatigue life evaluation curves. This is illustrated in Fig. (1), which shows how the

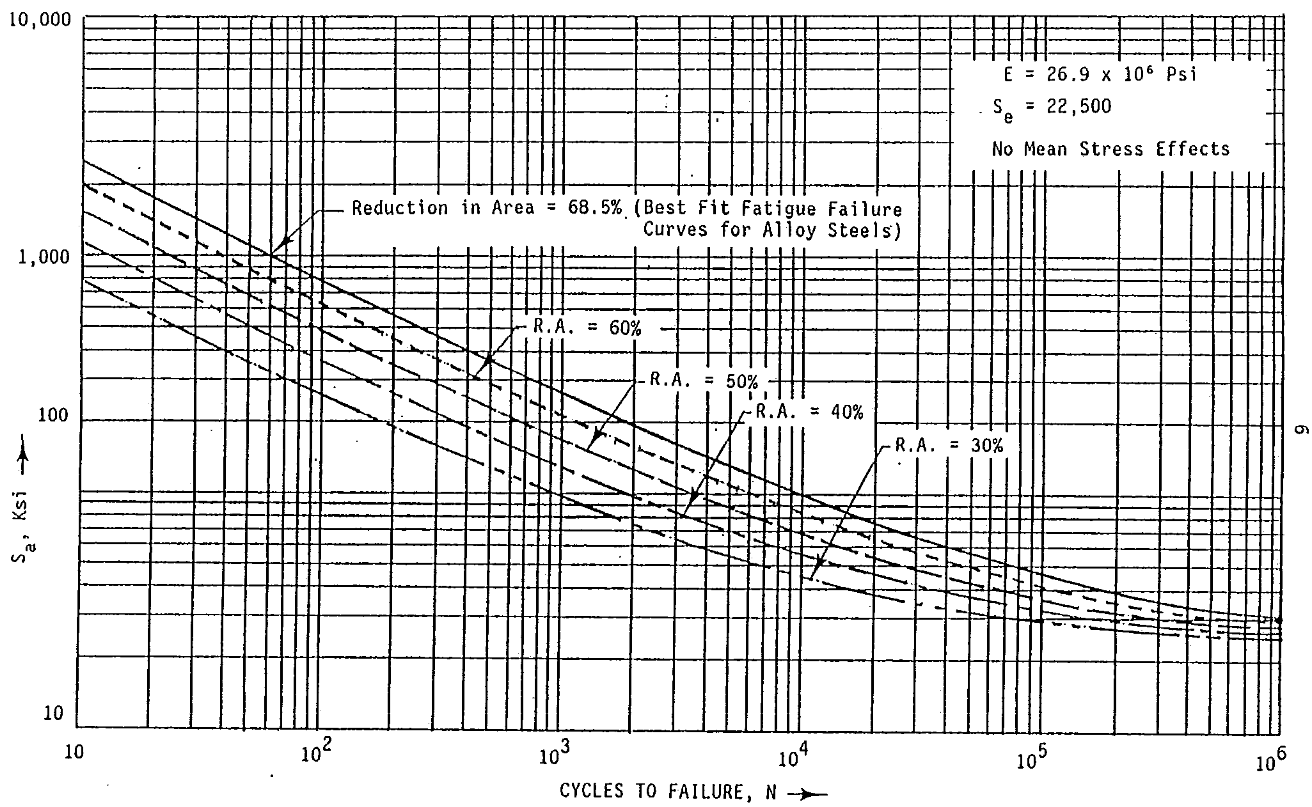

Fig. 1 Illustrates Effects of Hypothetical Reductions in Ductility Due to Aging

failure curves for low-alloy steels would be affected by hypothetical reductions in ductility from Reduction in Area = 68.5% to 60%, 50%, 40%, and 30%, respectively. Losses in ductility of this magnitude due to aging are not anticipated at temperatures below the creep regime.

Concerns have been raised that the crack initiation resistance of weldments could conceivably be reduced by aging. Such effects would reduce the fatigue evaluation curves in the high cycle regime, but would not affect the low cycle regime where crack propagation constitutes more than 90 percent of the failure life. Of course, the methods developed herein to correct fatigue life evaluation curves by considering failure as the sum of the initiation and propagation phases can be used directly to include any change in initiation resistance in the total fatigue life curves.

Increased notch sensitivity has also been cited as a potential aging effect. Fatigue notch sensitivity has been found to be a monotonically increasing function of ultimate strength. This may be largely due to the correlation between strength and ductility, wherein higher strength generally correlates with lower ductility. In fatigue design evaluation practice, fatigue strength reduction factors, K_f, are usually taken to be equal to the local elastic stress concentration factor, K_t. The latter is actually an upper bound for the fatigue strength reduction factor, applicable for very notch sensitive materials, and/or strain raisers having large enough dimensions to grow a crack which is of sufficient size to keep growing beyond the "shadow" of the notch. Most steels used in reactor pressure boundary components are not very notch sensitive. The common practice of using conservatively large fatigue strength reduction factors in fatigue design life evaluations is usually sufficient to cover increases in notch sensitivity with aging.

CYCLIC PLASTICITY IN NUCLEAR COMPONENTS

The process of material fatigue and fracture is one of damage accumulation which can be both local and global. The complexity of the geometric and loading conditions of importance in nuclear components and piping requires that fracture processes be treated within a general and consistent approach. This approach must include consideration of the crack initiation phase as well as both short and long crack propagation. The effects of general plasticity must be included because nominal secondary stresses are permitted to exceed yield and typically do exceed yield in nuclear components and piping.

Linearized secondary stresses in Class (1) nuclear components are limited to $3S_m$ by Section III of the ASME Boiler and Pressure Vessel Code. S_m is the lesser of 2/3 of yield or 1/3 of the ultimate, so that $3S_m$ is about twice yield[2]. Nominal (linearized) thru-the-wall thermal gradient stresses and local stresses which include stress concentration factors are limited only by fatigue, and can be several times yield. Therefore it is essential that any fatigue or fracture analysis methods used to evaluate the safe useful life of nuclear components and piping must be capable of including the effects of plasticity beyond that which can be assessed according to linear elastic fracture mechanics (LEFM) with minor corrections for local plastic zones. The S-N fatigue life evaluation approach with the simplified elastic-plastic strain concentration factors, K_e, in Section III of the ASME Code, includes the effects of such plastic cycling.

[2] In austenitic stainless steels, S_m can be as high as 90 percent of yield where distortions are not critical, so that $3S_m$ can be 2.7 times yield.

LOW-CYCLE FATIGUE CRACK GROWTH

Existing ASME Code Section XI (Ref. [98]) flaw evaluation methods are limited to LEFM ΔK concepts, with local plastic zone corrections which are quite small. Elastic-plastic ΔJ crack propagation technology has made great strides in recent years in both its theoretical and experimental facets.

In order to include both elastic and plastic conditions, Rice [Refs. 59 - 61], Hutchinson [Ref. 62] and others analyzed the energy available to drive the crack per unit crack extension. For linear elastic behavior, J is equal to the energy release rate per unit area of crack extension, G. For nonlinear elastic conditions, J is the potential energy difference between two identically loaded bodies possessing slightly different crack lengths. Thus, for either linear or nonlinear elastic behavior, J is the energy at the crack tip per unit area of crack extension, or the crack driving force.

With irreversible plastic straining, J is no longer equal to the energy available for crack extension. However, by defining J in the same manner for nonlinear elastic and elastic-plastic conditions, J remains a measure of the intensity of the entire elastic-plastic stress-strain field surrounding the crack tip. In the context of the deformation theory of plasticity, the J-integral denotes the energy released by a unit increase in crack area.

Begley and Landes [Refs. 63 and 64] did considerable early work developing the J-integral as an analytical tool for elastic-plastic cracks using its compliance characteristic. The J-integral has also proven to be of value as a means of evaluating the stress intensity from finite element analyses, using the σ and ε values at locations other than the crack tip. Rice [Refs. 59, 60, 65 and 66] solved two-dimensional crack problems with plastic deformations using a J-integral integration around the tip of the crack. For the power law stress-plastic strain relation, Hutchinson [Ref. 62], and Rice and Resengren [Ref. 67] showed that crack-tip stress and strain singularities are functions of J (see Ref. [68]).

Crack Tip Opening Displacement (CTOD), C*, concepts have also been applied and can be shown to be equivalent to J [Ref. 69] for certain cases. Sehitoglu and Morrow [Ref. 70] have used the cyclic range of the CTOD to characterize thermal fatigue crack growth for AISI 1070 carbon steel.

ΔJ may be meaningfully defined [Refs. 71 through 74] for cyclic loading, where all loads, stresses, displacements and strains in the line integral definition of J are replaced by the