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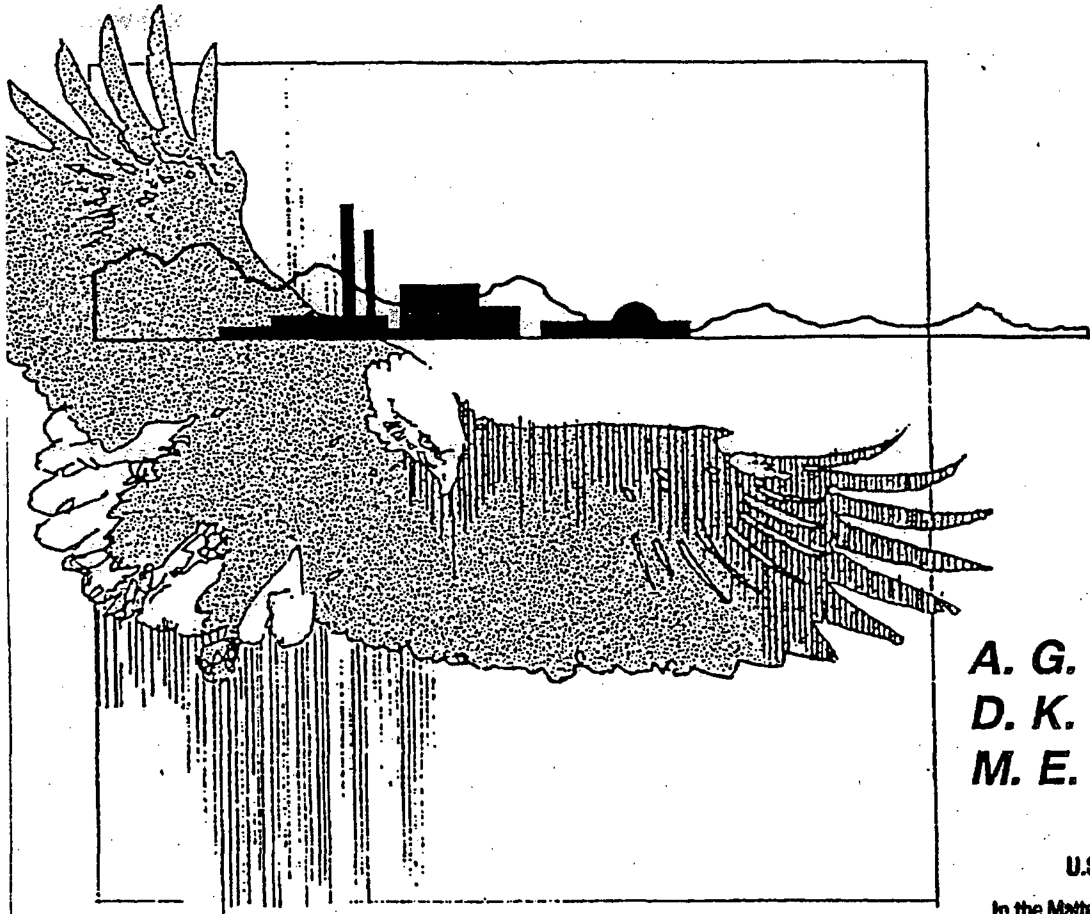

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Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components

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ABSTRACT

Recent test data indicate that the effects of the light water reactor (LWR) environment could significantly reduce the fatigue resistance of materials used in the reactor coolant pressure boundary components of operating nuclear power plants. Argonne National Laboratory has developed interim fatigue curves based on test data simulating LWR conditions, and published them in NUREG/CR-5999. In order to assess the significance of these interim fatigue curves, fatigue evaluations of a sample of the components in the reactor coolant pressure boundary of LWRs were performed. The sample consists of components from facilities designed by each of the four U.S. nuclear steam supply system vendors. For each facility, six locations were studied, including two locations on the reactor pressure vessel. In addition, there are older vintage plants where components of the reactor coolant pressure boundary were designed to codes that did not require an explicit fatigue analysis of the components. In order to assess the fatigue resistance of the older vintage plants, an evaluation was also conducted on selected components of three of these plants. This report discusses the insights gained from the application of the interim fatigue curves to components of seven operating nuclear power plants.

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EXECUTIVE SUMMARY

ES-1. Introduction

Recent test data indicate that the effects of the light water reactor (LWR) environment could significantly reduce the fatigue resistance of materials used in the reactor coolant pressure boundary components of operating nuclear power plants. The American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code fatigue curves used for the design of these components were based primarily on strain-controlled fatigue tests of small, polished specimens at room temperature in air. Although adjustment factors were applied to the best-fit curves to account for effects such as size, surface finish, environment, and data scatter, some of the recent test data indicate that these factors may not have been sufficiently conservative to account for environmental effects.

In a separate project funded by the United States Nuclear Regulatory Commission (USNRC), the Argonne National Laboratory (ANL) has developed interim fatigue curves based on test data of small, polished specimens cycled to failure in water simulating LWR conditions, and published them in NUREG/CR-5999. In order to assess the significance of the interim fatigue curves in NUREG/CR-5999, fatigue evaluations of a sample of the components in the reactor coolant pressure boundary were performed. The sample consists of components from facilities designed by each of the four U.S. nuclear steam supply system (NSSS) vendors. For each facility, six locations were studied, including two locations on the reactor pressure vessel.

In addition, there are older vintage plants where components of the reactor coolant pressure boundary were designed to codes, such as United States of America Standard (USAS) B31.1, that did not require an explicit fatigue analysis of the components. Since the Code of Federal Regulations currently references the ASME Boiler and Pressure Vessel Code which includes a fatigue evaluation of the components of the reactor coolant pressure boundary (unless certain exemption requirements are met), this has led to a concern regarding the adequacy of the fatigue resistance of these older vintage plants. In order to assess the fatigue resistance of the older vintage plants, an evaluation was also conducted on selected components of these plants. The components selected were the same as in the newer vintage plants. A comparison of the magnitudes of the cumulative usage factors (CUFs) between older and newer vintage plants, and the results of the application of the NUREG/CR-5999 interim fatigue curves to six components in each of the three older vintage plants are presented in this report.

ES-2. ASME Code Section III Fatigue Methodology

In the 1960s Codes and Standards specific to nuclear power plants were developed. Section III, *Nuclear Vessels,* was first issued in 1963 as a separate code. All of the vessel analyses reviewed in this NUREG/CR were performed using the 1965 or later editions of Section III. Prior to 1969, nuclear piping was designed using United States of America Standard (USAS) B31.1; from 1969 to 1971, plants were designed with USAS B31.7-1969 as the standard; and the ASME Code has been used thereafter. The rules of B31.7 were incorporated in NB-3600 of the 1971 edition of Section III.

The ASME Code, Section III, NB-3200 elastic fatigue analysis is applicable to any component, but is generally used exclusively for vessels, fairly frequently for nozzles, but rarely for piping. If neither the elastic or simplified elastic-plastic methods can demonstrate that the ASME Code limits are satisfied, NB-3200 allows a fully plastic analysis. (However, the time and expense needed to perform such an analysis makes this option a last resort.) For Class 1 piping, the ASME Code (Article NB-3600 of Section III) provides for protection against fatigue failures caused by elastic and plastic cycling similar to NB-3200; however, more detailed equations are given leading to a simpler, but generally more conservative, analysis approach.

ES-3. NUREG/CR-5999 Interim Fatigue Curves

The NUREG/CR-5999 figures are very small, use a log-log scale, and contain no background grid. This makes the values very difficult to read from the graphs. Dr. W. J. Shack of ANL supplied us with a spreadsheet with the data points used to construct the interim fatigue curves for use in this project. The spreadsheet values were used to perform the CUF calculations in Section 5 of this report.

In order to assess the increase in the CUF using the interim fatigue curves, values for the numbers of cycles on the ASME Code fatigue curve were divided by the numbers of cycles at corresponding stresses on the interim fatigue curves (using the ANL spreadsheet values). The ASME Code method of interpolating between values was used. The factor of increase depends on the alternating stress intensity. The factor of increase for stainless steel is as high as a factor of 17. For carbon and low-alloy steels in low-oxygen environments, the maximum factor of increase is only about 2.75. For carbon and low-alloy steels in high-oxygen environments at saturated (0.001%/s) strain rates, the maximum factors of increase are about 13, 30, and 55 at temperatures of 200, 250, and 288°C, respectively. The lowest maximum increase of about 3.5 occurs at high strain rates (0.1%/s) at 200°C.

In order to be able to accurately interpolate between the temperature and strain rate values on the interim fatigue curves, studies were carried out to determine appropriate interpolation formulas. The ratios of the numbers of cycles for the three strain rates at the three temperatures on the high-oxygen curves were plotted. In addition, the ratio of the values for the three temperatures at the three strain rates were plotted. From these curves we deduced that interpolation relations can be determined irrespective of alternating stress intensity.

Since the ratios were not dependent on the alternating stress intensity, a value of 55 ksi was chosen to determine the relations between strain rate, temperature, and number of cycles. The logarithms of strain rate and numbers of cycles have a linear relationship, and the temperature and the logarithm of the numbers of cycles are linearly related.

Subsequent to the issue of NUREG/CR-5999, ANL transmitted revised best-estimate fatigue curves for stainless steel (in equation form) to the NRC. The revised curves are strain rate-, temperature-, and material-dependent and differ for Type 316NG and other types of stainless steel. However, none of the stainless steel components investigated as part of this project are Type 316NG stainless steel.

The ANL best-estimate curves were converted to design curves comparable to ASME Code fatigue design curves by reductions of a factor of 1.5 on stress or 20 on cycles, whichever is less. The revised curves increase the CUF by a factor of about 5 (1%/s strain rate) to 11 (0.001 %/s strain rate) over CUFs computed using the ASME Code fatigue design curves. The NUREG/CR-5999 interim fatigue curves increase the CUF by as much as a factor of 17 over the ASME Code design fatigue curves. However, for low strain rates, the revised curves would result in a higher CUF for alternating stress intensities above 90 ksi. The 1%/s strain rate was achieved during tests in which the specimens were loaded by mechanical cycling. It is highly unlikely that such a high strain rate could be achieved during thermal cycling. No strain rates approaching 1%/s were calculated in this study. A 1%/s strain rate corresponds to an equivalent elastic stress rate of 283,000 psi/s.

ES-4. Approach

The components chosen for the evaluation of the five PWR plants [B&W, Combustion Engineering (one older and one newer vintage), and Westinghouse (one older and one newer vintage)] are as follows:

1. Reactor vessel shell and lower head

2. Reactor vessel inlet and outlet nozzles

3. Pressurizer surge line (including hot leg and pressurizer nozzles)

4. Reactor coolant piping charging system nozzle

5. Reactor coolant piping safety injection nozzle

6. Residual heat removal (RHR) system Class 1 piping.

The components chosen for the evaluation of the two BWR plants [General Electric (one older and one newer vintage)] are as follows:

1. Reactor vessel shell and lower head

2. Reactor vessel feedwater nozzle

3. Reactor recirculation piping (including inlet and outlet nozzles)

4. Core spray line reactor vessel nozzle and associated Class 1 piping

5. RHR Class 1 piping

6. Feedwater line Class 1 piping.

For both PWR and BWR plants, these components are not necessarily the locations with the highest design CUFs in the plant, but were chosen to give a representative overview of components that had higher CUFs and/or were important from a risk perspective. For example, the reactor vessel shell (and lower head) was chosen for its risk importance.

NUREG/CR-5999 includes one fatigue curve for stainless steel, but several curves for carbon/low-alloy steels which are based on the sulfur content of the steel and the oxygen level in the coolant. For the five PWR plants, the curves for high-sulfur steel and a low-oxygen environment (typical for PWRs) were used. For the two BWR plants, the curves for high-sulfur steel and a high-oxygen environment were used. The high-oxygen (greater than 100 ppm) environment considered in the selected curves is consistent with the water chemistry in BWRs without hydrogen water chemistry. Neither of the two BWR plants evaluated have used hydrogen water chemistry.

If the CUF for a component exceeded 1.0 using the NUREG/CR-5999 interim fatigue curves, potential changes that could be used to reduce the CUF were sought. In reviewing the licensees' calculations, we found 17 potential changes that could be used to reduce the CUF. Several changes were found from review of the licensees' calculations that might increase the CUF. These mainly consisted of changes to the ASME Code since the edition of record for the plants' licensing bases, and the anticipated numbers of cycles for some transients exceeding the number of design basis cycles.

ES-5. Component Evaluations

The stress results from existing analyses were used to determine revised CUFs based on the NUREG/CR-5999 curves. Since the licensees' design basis analyses were based on the ASME Code of record, it was uneconomical for the licensee to attempt to reduce the CUF to lower and lower values by removing conservative assumptions once the Code requirements were met. Given more funding and time, further calculations could have been performed to reduce the existing stress values by using more realistic loadings or more detailed analysis models. These reduced stresses would result in lower CUFs. Therefore, high CUF values obtained using the NUREG/CR-5999 interim fatigue curves do not reflect the lowest CUF, since in every case where the CUF was greater than 1.0, we have listed one, and in most cases several, steps that could be taken to reduce the CUF by additional analyses and monitoring.

The details of the evaluations for six components for each of the seven plants surveyed are described in Sections 5.1 through 5.7 in the body of the report. It appears that the two most difficult areas to reduce the CUF to lower values are PWR surge lines, which are subject to thermal stratification, and BWR tees joining RHR, recirculation, RCIC, RWCU, feedwater, etc. lines where hot and cold coolant mixing occurs. The results and conclusions of these evaluations are summarized in ES-6. below.

ES-6. Conclusions

The conclusions from applying the NUREG/CR-5999 interim fatigue curves to the fatigue analyses of seven LWRs (five PWRs and two BWRs) are divided into three parts. Conclusions relating to PWR and BWR plants, and conclusions from comparing plants designed to B31.1 versus plants designed to the ASME Code.

ES-6.1 Applications to PWR plants

1. The anticipated number of cycles are less than the design basis number of cycles for all key transients, notably heatup and cooldown transients and power changes. (For example, the design analyses accounted for load following whereas the plants are being operated as base-loaded.)

2. After removing conservative assumptions and using anticipated numbers of cycles, the CUFs for all the reactor vessel components (shell and lower head, inlet and outlet nozzles) were less than 1.0 for a 40-year life. In two cases, an Alloy 600 instrumentation nozzle and a lower head core support block, the CUFs (1.113 and 1.337, respectively) were slightly above 1.0 for 60 years.

3. The CUFs for the stainless steel surge lines of all five plants exceeded 1.0 for 40 years. The most significant transient for surge lines is thermal stratification which was not accounted for in the original design basis. The surge lines were reanalyzed for fatigue in response to NRC Bulletin 88-11. Fatigue monitoring was used to determine temperature differences and numbers of cycles during times of thermal stratification. More refined analyses to later (circa 1986) editions of the ASME Code, including removal of conservative assumptions, were used by the licensees to reduce the CUF below 1.0 using ASME Code fatigue curves. However, there remain conservative assumptions that could be used to further reduce the CUF. Four of the five analyses used NB-3600 piping methods. A detailed finite analysis of the regions with high CUFs, and, if needed, plastic analyses, could be used to reduce the CUF. The B&W plant's analysis already has incorporated an NB-3200 plastic analysis. Probably the best way to reduce the CUF is more precise monitoring of the individual surge lines. The stratification transients used in the analyses are mainly based on owners group submittals that conservatively define a set of enveloping stratification transients that will apply to several plants.

4. After removing conservative assumptions and using anticipated numbers of cycles, the 40-year CUFs for the stainless steel charging and safety injection nozzles were below 1.0 for 7 of the 10 cases. The other three (two charging and one safety injection nozzle) had CUFs ranging from 1.3 to 4.9 for a 40 year life. The numbers of key transients for these two components (for example, loss of letdown and loss of charging) are not counted on a regular basis as are transient cycles important to overall plant operation (for example, heatups and reactor trips); consequently, it was difficult to estimate anticipated numbers of cycles. It appears that the number and severity of these key cycles are conservative and further studies based on plant operation could be used to reduce the CUF. Based on our results of the CUFs for charging and safety injection nozzles of an older vintage plant using the 1992 ASME Code edition NB-3600 and NB-3200 methods, it appears that by using NB-3200 methods contained in the 1992 ASME Code, the CUFs for all nozzles could be reduced than 1.0.

5. The 40-year CUFs for RHR lines were less than 1.0 for four of the five plants. The fifth plant included cycles for thermal stratification in the RHR line, which were not considered for the other four plants. Excluding thermal stratification, the CUF for the fifth plant would have been compa-

rable to the other four plants. The analysis of the fifth plant used NB-3600 piping methods. A detailed finite analysis using NB-3200 methods, and, if needed, a plastic analysis, could be used to reduce the CUF. Probably the best way to reduce the CUF is fatigue monitoring of the RHR line. The stratification transients may conservatively define a set of enveloping stratification transients or valve leakage.

6. For carbon and low-alloy steel components, the NUREG/CR-5999 interim fatigue curves increased the CUF by an average factor of 2.2 times the design basis CUF. This was before any adjustments based on conservative assumptions removal and anticipated cycles were made. For stainless steel and Alloy 600, the average multiplication factor is 9.2.

ES-6.2 Applications to BWR plants

1. The anticipated number of cycles exceed the design basis numbers of cycles for some transients, notably startup and shutdowns. However, the anticipated number of cycles is less than the design basis number of cycles for other transients such as power changes (the design analyses accounted for load following whereas the plants are being operated as base-loaded.)

2. After removing conservative assumptions and using anticipated numbers of cycles, the CUFs for the reactor vessel shell and lower head were less than 1.0 for 40- and 60-year lives. The core spray nozzle CUF was less than 1.0 for the 40- and 60-year lives of the newer vintage BWR plant, but was greater than 1.0 (2.305) for the older vintage BWR plant for 40 years. Although CUFs for the recirculation nozzles were not calculated using NUREG/CR-5999, the design basis CUFs were 0.002 for the newer vintage plant and 0.300 for the older vintage plant (using very conservative lumped transients). No problem would be expected in reducing the CUFs below 1.0.

3. The 40-year CUF for the feedwater nozzle exceeded 1.0 for both plants. The CUF range was from about 1.9 to 3.2. (The CUF for the thermal sleeve on the BWR/6 plant was about 5). Although we incorporated transient definitions, anticipated cycles, strain rates, and temperatures according to the information available, there remains a great deal of uncertainty concerning these values. There also remain conservative assumptions that could be used to reduce the CUFs. Two studies based on fatigue monitoring of BWR feedwater nozzles in other plants showed that the monitored CUF was a factor of 30 to 50 less than the design basis CUF.

4. The 40-year CUF for the recirculation system is less than 1.0 for the newer vintage BWR, and slightly exceeds 1.0 for 60 years (1.245). The CUF for the older vintage BWR is 3.898. Both CUFs were calculated using NB-3600 methods, and were for tees. Based on our experience with comparing NB-3200 and NB-3600 methods for nozzles, we believe that an NB-3200 analysis and fatigue monitoring would reduce the CUF below 1.0.

5. The CUF for the feedwater lines are 3.688 and 6.980 (at tee locations). The CUF for the tee was calculated using NB-3600 methods. Based on our experience with comparing NB-3200 and NB-3600 methods for nozzles, we believe that an NB-3200 analysis and fatigue monitoring would reduce the CUFs below 1.0.

6. The CUF for the BWR/6 RHR line is 11.26 in a straight run of piping. All transients that contributed to the CUF involved thermal stratification. The analysis used NB-3600 piping methods. A detailed finite analysis using NB-3200 methods, and, if needed, a plastic analysis, could be used to reduce the CUF. Probably the best way to reduce the CUF is more precise monitoring of the RHR line. The stratification transients may con-

servatively define a set of enveloping stratification transients.

ES-6.3 CUF Evaluations for Piping Components Designed to the B31.1 Piping Code

1. The design of PWR components and the transients to which they are subjected to are similar for older and newer vintage plants. An exception is the Westinghouse 3- and 4-loop plants that we studied, which had different safety injection piping configurations. Consequently, we reviewed transients from both the newer vintage Westinghouse and the Combustion Engineering plants to ensure that the transients we used were representative for the older vintage Westinghouse plant.

   The design of some of the BWR systems were not similar for the older vintage (BWR/4) and newer vintage (BWR/6) plants that we reviewed. Several key locations of hot and cold coolant mixing, which on the BWR/4 plant are on piping that would be considered Class 1 today, are included in the Class 2 portions of the BWR/6 piping. We reviewed transients from both a BWR/6 and another BWR/4 plant to ensure that the transients we used were representative for the older vintage BWR plant.

2. While we did not perform additional fatigue evaluations of PWR surge lines because the licensees had already analyzed these lines for fatigue in response to NRC Bulletin 88-11, the results of the fatigue evaluations and CUFs for older and newer vintage plants appear comparable.

3. The charging and safety injection nozzles for one older vintage PWR were analyzed using detailed finite element models (both contained thermal sleeves). The CUF using both the ASME Code and NUREG/CR-5999 curves were less than 1.0.

4. The design basis CUFs for two older vintage PWR RHR lines that we analyzed, including representative transients from other PWRs, were low and comparable to the other PWRs (not including thermal stratification effects).

5. The design basis CUFs for the older vintage BWR plant recirculation, RHR, and feedwater lines that we analyzed, including representative transients from other BWRs, were less than 1.0. The 40-year CUFs using the NUREG/CR-5999 curves were above 1.0 for the recirculation and feedwater lines. The comparable CUFs were above 1.0 for the newer vintage BWR, also, but only about half those computed for the older vintage BWR.

6. The older vintage plants piping typically have thicker walls and larger diameters than do newer vintage plans. This causes higher thermal stresses in the older vintage plants' piping. Thermal stresses were found to be the major type of stress contributor to the CUF. Some stress indices are a function of the pipe diameter and thickness, but this is expected to have only a minor effect on the CUF.

ES-6.4 Overall Conclusion

We were able to show that by removing conservative assumptions and using anticipated numbers of cycles, the CUF could be reduced to below 1.0 for most components, both for older and newer vintage plants. For components which we were not able to reduce the CUF below 1.0, several additional steps that could be taken to further reduce the CUF were listed. The two major remaining steps mentioned were (1) more detailed finite element analyses or (2) fatigue monitoring of the transients. Whereas using ASME Code NB-3200 versus NB-3600 analysis methods will assist with regions of axial thermal gradients, we did not find that the CUF could be reduced when the majority of the stress was caused by radial thermal gradients. A major problem with NB-3200 analyses is that minimal guidance is provided by the ASME Code regarding fatigue strength reduction factors for welds. Analysts typically do not apply fatigue strength reduction factors for welds on nozzles made in the shop. For field welds, the NB-3600 stress indices can be used, but they may be too conservative. A plastic analysis in which the strains are computed, rather than using the K_e factor to adjust the elastic stresses, will lower the CUF.

The best method to lower the CUF for the few worst locations appears to be fatigue monitoring. For most of the cases where the CUF exceeded 1.0, neither actual numbers of cycles that the plant is experiencing nor the magnitude of temperature differences or thermal shocks were known. Therefore, worst-case design assumptions were used. By using realistic numbers of cycles and severity of transients, we believe that the CUF could be reduced sufficiently without resorting to more detailed analysis methods. However, in some cases, for example where thermal stratification exists, a combination of fatigue monitoring and more refined analyses may be needed.