Enclosure 3

Staff Responses to Public Comments on Draft Regulatory Guide DG-1144 (proposed new Regulatory Guide 1.207), "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," and Draft NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (Draft Report for Comment)"

Sources for Comments

I: Comments from Ronnie L. Gardner, AREVA NP, Inc. (ML062920056)
II: Comments from Takao NAKAMURA, The Kansai Electric Power Co., Inc. (ML062790143)
III: Comments from James H. Riley, Nuclear Energy Institute (ML062790136)
IV: Comments from C.L. Funderburk, Dominion Resources Services, Inc. (ML062790144)
V: Comments from Makoto HIGUCHI, Ishikawajima-Harima Heavy Industries Co., Ltd. (ML062790138)
VI: Comments from Robert E. Brown, GE Energy Nuclear (ML062790141)
VII: Comments from Gerry C. Slagis, G.C. Slagis Associates, Consulting Engineering (ML062620349)
VIII: Comments from Kenneth R. Balkey, Nuclear Codes and Standards, American Society of Mechanical Engineers (ML062790139)

#	Source	Specific Comment*	NRC Comment Resolution
1	I - 1a	AREVA agrees with laboratories fatigue tests results concerning demonstration of the role of pressurized water reactor (PWR) environment on the low cycle fatigue (LCF) behavior of reactor materials. However, AREVA is not aware of any operating experience that supports the need for these conservative design rules. The NRC should cite specific examples where operating events associated with a significant environmental effect have been at the root cause of fatigue failure. The NRC should also cite where in the fatigue analyses supporting the original design, it was necessary to account for environmental effect to demonstrate the need for this regulatory guidance.	Numerous examples of fatigue cracking of nuclear power plant components have been reported. Electrical Power Research Institute (EPRI) Report TR-106696 (Reference 44 of NUREG/CR-6909) provides some examples. The exact role of the environment on all of the reported fatigue cracks is difficult to assess because of the lack of detailed information regarding the stresses in these components. However, the EPRI report attributes environmentally assisted fatigue as the cause of PWR feedwater nozzle cracking. Therefore, the staff disagrees with the comment to revise the final NUREG/CR-6909 to cite more examples of operating events associated with a significant environmental effect as the root cause of fatigue failure.

#	Source	Specific Comment*	NRC Comment Resolution
			The implication of the AREVA comment is that laboratory testing is not representative of actual components in service. Although full-scale testing of operating plant components would be desirable, laboratory testing of small-scale test specimens is the only practical method to establish design fatigue curves. In fact, it is the basis for the current fatigue curves in the American society of Mechanical Engineers (ASME Code). Paragraph NB-3121 of the ASME Code states that the tests on which the ASME fatigue curves are based did not include tests in the presence of corrosive environments that might accelerate fatigue failure. Therefore, the ASME Code recognizes that additional criteria may be needed to account for fatigue life corrosive environments. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
2	I - 1b	The Regulatory Analysis states that the "costs associated with implementing this guidance are expected to be minimal." AREVA believes that an increase in the Cumulative Fatigue Usage Factors (as suggested in DG-1144) will lead to more analyzed piping break locations, to more installed pipe whip restraints, and to designs that will be more detrimental for normal (thermal expansion) operating conditions. In addition, there will be more restrictions on the Design Transients (in the Functional Specifications) and the analyses will have to be performed with added accuracy, such as performing elasto-plastic finite element analyses, to be able to reduce the conservatism inherent to the current design and analysis methods. However, it is not usual to perform elasto-plastic finite element analyses at a design stage and this added complexity to new plant designs is unwarranted. Analysis costs will increase significantly owing to the involved nature of the F(en) calculation, particularly related to the determination of strain rate. This method will also require more detailed analyses of piping and components due to the severe nature of the F(en) penalty. For example, it can be anticipated that more locations in stainless steel piping will have to be evaluated using finite element approaches (NB-3200) instead of the traditional simplified rules in NB-3600.	AREVA states that implementation of the guidance will lead to more analyzed pipe break locations and more installed pipe whip locations. The staff agrees that the guidance could lead to more postulated pipe break locations based on the fatigue usage factor. It is not the intent of the staff to increase the number of pipe whip restraints added to new plant designs. Instead, the staff expects applicants for new reactor designs to minimize the necessity for pipe whip restraints by ensuring adequate separation during the initial plant design. However, the staff will consider a justified modification with the appropriate technical basis of the fatigue criteria for the postulation of pipe breaks if implementation of the current criteria results in a significant increase in the number of required pipe whip restraints. AREVA also states that the guidance will place more restrictions on the Design Transient Functional Specifications. The staff agrees that changes in the Design Transient Function Specifications could be used to lower the fatigue usage of a component. However, AREVA did not provide any specific examples in which the Design Transient Functional Specifications would be impacted by the guidance. The staff is not aware of changes made to operating plant functional specifications as a result of implementing environmental fatigue guidance during license renewal reviews.

#	Source	Specific Comment*	NRC Comment Resolution
			AREVA states that the guidance will also require more detailed analyses of the piping components using finite element approaches. The staff agrees that a greater use of finite element analyses may be required to demonstrate acceptable fatigue usage of components. However, based on its review of environmental fatigue evaluations performed for plant license renewal applications, the staff also believes that only a limited number of locations will require the more detailed finite element analysis procedures. Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
3	I - 1c	The practice reported in NUREG/CR-6260 applied to several plants and identified locations of interest for consideration of environmental effects using the fatigue design curves that incorporated environmental effects. Section 5.4 of NUREG/CR-6260 identified the following component locations to be most sensitive to environmental effects for PWRs. 1. Reactor vessel shell and lower head 2. Reactor vessel inlet and outlet nozzles 3. Surge line 4. Charging nozzle 5. Safety injection nozzle 6. Residual Heat Removal system Class 1 piping It is not understandable why the guidance for new plants, in spite of better materials, more modern nondestructive testing technologies, and improved manufacturing process, is not restricted to a limited number of locations. In lieu of evaluating the entire Class 1 systems for the environmental effects on fatigue, AREVA believes an approach that parallels the license renewal approach would provide more reasonable assurance that the environmental effects are bounded sufficiently.	AREVA contends that the guidance for the new plants should parallel the license renewal approach, which involved evaluating a sample of components reported in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,” issued March 1995. The selection of the sample components reported in NUREG/CR-6260 considered high fatigue usage and plant risk for the existing operating nuclear power plants, as documented in SECY-95-245, “Completion of the fatigue Action Plan,” dated September 25, 1995. As stated in the resolution of Generic Safety Issue 190 (GSI-190): The results of the probabilistic analyses, along with the sensitivity studies performed, the iterations with industry (NEI and EPRI), and the different approaches available to the licensees to manage the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is closed. This conclusion is based primarily on the negligible calculated increases in core damage frequency in going from 40 to 60 year lives. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

#	Source	Specific Comment*	NRC Comment Resolution
			These considerations do not apply to new reactor licensing. The proposed guidance applies to new reactor designs that NUREG/CR-6260 did not evaluate. Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
4	I - 1d	AREVA does not believe the NRC should establish very conservative design rules without peer consensus. The fact that consensus has not been reached in the industry highlights both that the research is not sufficiently finalized to be conclusive and that the correct method of treatment of environmental effects is not clearly established. For example, there is not enough evidence to support the combination of all detrimental effects. It is not appropriate to treat simultaneously all the detrimental effects of size, surface finish, loading history, data scatter, material variability, dissolved oxygen in the water, strain rate, and temperature to calculate the environmental fatigue penalty. AREVA believes that there are cases where, when one effect is taken at its worst (at saturation), the other effects do not further negatively affect the fatigue resistance of the component. Therefore, AREVA believes that for fatigue the "Cumulative Penalties" methodology is overly conservative.	AREVA states that the NRC should not establish very conservative design rules without peer consensus. AREVA states that it is not appropriate to treat simultaneously all detrimental effects in developing the fatigue curves. However, AREVA does not provide any data to support its argument. The staff based its method on a careful statistical evaluation of a significant quantity of specimen test data. A further statistical evaluation of the parameters used to adjust laboratory data to account for actual components supplemented this evaluation. Data indicate that, at a 95% confidence level, there is less than a 5% probability of fatigue crack initiation. The staff does not consider this criterion to be overly conservative. In fact, application of the method results in carbon steel air fatigue curves that are less conservative than the existing ASME Code fatigue curves. Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
5	I - 1e	The current ASME Code fatigue methodology is overly conservative. Examples of the conservatism that are inherent to methodology include: • use of conservative values for fatigue strength reduction factors, • the piping stress indices, • the piping stress methodology, • use of Tresca criterion for the calculation of the stress intensity, • use, in the design methodology, of minimum specified mechanical properties in place of representative materials properties, • the fatigue plasticity penalty factor (Ke), • design transients are more severe than the actual transients, • grouping various transients into analysis sets in which each set is bounded by the most severe transient in the set, and • there are fewer transients during the plant lifetime than specified in the Functional Specs.	AREVA states that the current ASME Code methodology is overly conservative and cites several specific provisions of the ASME Code methodology to support its argument. AREVA recommends that the staff consider the entire analysis method rather than just the material aspects. The staff did consider the other aspects of the ASME Code fatigue methodology. The ASME Code allows for design-by-analysis or design-by-rule. The design-by-rule procedures contain simplified formulas that, as a consequence of simplification, are conservative for most applications. However, the designer also can use the design-by-analysis provisions, which can eliminate much of the conservatism inherent in the application of the simplified formulas.

#	Source	Specific Comment*	NRC Comment Resolution
		It would be preferable to review the whole methodology rather than limiting efforts to the materials aspects.	The staff does not agree with all of the conservatism associated with the ASME Code fatigue methodology cited by AREVA. For example, the number of design transients included in the functional specifications has not always been conservative. NUREG/CR-6260 discusses several cases in which the number of actual transient cycles exceeded the number of design transient cycles. The staff also does not agree with AREVA about the use of the Tresca condition for calculating stress intensity. The Tresca criterion is based on the maximum shear stress theory, and it is considered the appropriate criterion for evaluating the fatigue life of pressure vessels (J. F. Harvey, "Theory and Design of Pressure Vessels," Second Edition, page 284). Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
6	I - 1f	There is no guidance in DG-1144 or CR-6909 regarding how to treat carbon steel and low alloy steel, which are "protected" from the primary coolant environment by stainless steel (or Alloy 690) cladding. AREVA believes it is reasonable to assume that there will not be any environmental effects on clad carbon steel and low alloy steel. For completeness, the guidance should address this subject.	The ASME Code allows the designer to neglect the presence of the cladding if its thickness is less than 10% of the total thickness of the component, as stated in paragraph NB-3122 of the ASME Code. The designer should assume that the environmental effects apply to the underlying carbon steel material for those cases in which the cladding is neglected. However, if acceptable fatigue usage of the cladding can be demonstrated, the designer may assume that the underlying carbon steel is protected from the primary coolant environmental effects. Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
7	I - 2a	The majority of the LCF tests were performed at high temperature on polished specimens in the NUREG/CR-6909. About ninety percent of the tests were done at high temperature (between 260 °C and 325 °C) in isothermal conditions with triangular strain signals leading to constant strain rates. These test conditions are not representative of realistic thermo-mechanical loadings applied on components during operation. Indeed, the triangular form of cycles with two slopes and a constant temperature chosen for the laboratory fatigue tests is very different from the actual cycles applied during operating transients, which contain successions of high strain rates and low strain rates with a variable temperature. Because the tests performed in the laboratory	Initiation of a fatigue crack is a local property that depends on local values of strain range and, in light-water reactor environments, local values of strain rates, temperatures, material properties, and water chemistry. Thus, initiation in a specimen test is directly relevant to initiation in a component. Evaluating the effect of these parameters on crack initiation in specimens is relevant to its effect on initiation in a component. It is true that many of the cases of thermomechanical loading of interest do occur under nonisothermal conditions. Fatigue test data obtained in Japan have demonstrated that the damage accumulation in isothermal tests is representative of that which occurs under nonisothermal histories.

#	Source	Specific Comment*	NRC Comment Resolution
		specimens are not representative of in-service reactor components, it is not clear that the F(en) factors derived from those tests apply to the components and operating conditions in a nuclear plant.	<u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
8	I - 2b	After a micro-structural crack has formed, the crack depth is approximately 0.3 mm and a surface finish effect is no longer required, since the fatigue process occurs at the crack tip. Surface finish effect were only established in air. It is supposed to affect the fatigue life by a factor of three. NUREG/CR-6909 recommends treating the environmental effect on a rough surface by multiplying F(en) factor by approximately 3 but this accumulation is not proven by sufficient data obtained on representative surface at various strain amplitudes in PWR environment.	Limited tests on austenitic stainless steels (SSs) indicate that, in both air and PWR environments, the fatigue life of rough specimens is a factor of 3.0 lower than that of smooth polished specimens (Figure 51 of NUREG/CR-6909). Unless additional tests indicate otherwise, the staff considers the effect of surface roughness on the fatigue life of these steels to be the same in both air and PWR environments. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
9	I - 2c	Loading sequence effects should not be considered as an additional penalty for the factor of 12.0, as suggested in NUREG/CR-6909. During normal operation of the nuclear power plant, the cycles are reasonably well distributed for the entire life of the plant. Therefore, the Loading Sequence effect is not required. Furthermore, such a loading sequence is not supported by reviewed and accepted experimental results.	The staff disagrees with this comment. Several studies in the literature indicate loading sequence effects on fatigue life; NUREG/CR-6909 lists a few examples. In a variable loading sequence, the presence of a few cycles at high strain amplitude causes the fatigue life at smaller strain amplitude to be significantly lower than that at constant-amplitude loading (i.e., the fatigue limit of the material is lower under variable loading histories). NUREG/CR-6909 considers a factor of 1.2 – 2.0 on life to account for such effects. A recent study of Type 316NG and Ti-stabilized Type 316 SS indicates that these margins (a factor of 1.2 – 2.0 on life) may not be conservative (PVP2006-ICPVT11-93833). Strain-controlled tests in air and PWR environments with constant or variable strain amplitude indicate a decrease by a factor of 3.0 or more in fatigue life under variable amplitude compared to that under constant amplitude. Although the strain spectrum used in the study was not intended to be representative of real transients, it does represent a generic case and demonstrates the effects of loading sequence on fatigue life. If it could be demonstrated that variation in strain amplitude is indeed unlikely for an application, an argument to eliminate the factor in that particular case could be made and would be considered in a case by case basis. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
10	I - 2d	There should be a real threshold for both temperature and strain rate. In other words, below a certain temperature (150˚C or 180˚C), or above a certain strain rate (0.4 percent or 1 percent per second) penalty F(en) value should be 1.0. That has been shown clearly in the Figure 12 of the 2005 PVP Paper No. 71409 and in the Figure 10 of the 2005 PVP Paper No. 71410. These two technical papers are from William J. O'Donnell, William John O'Donnell, and Thomas P. O'Donnell.	The existing fatigue S-N data indicate that Fen does not equal unity (1.0) at high strain rates or very low temperatures. For example, see Figures 12 and 16 of NUREG/CR-6909. Figure 10 of 2005 PVP Paper No. 71410 and Figure 12 of 2005 PVP Paper No. 71409 are proposed water environment fatigue design curves for carbon and low-alloy steels, and austenitic SSs, respectively. O'Donnell et al. developed these design curves , which do not necessarily represent the available fatigue data in light-water reactor coolant environments. Test data show a moderate environmental effect consistent with the computed Fen correction factors obtained from the equations (see the NRC response to comment 18). Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
11	I - 2e	The proposition of a new fatigue curve in air is based on insufficiently supported test results and some of which were obtained on unrepresentative materials. For instance, a paper cited in the NUREG/CR-6909 [reference 105] is used as data for this fatigue curve to analyze mean stress effect. Nevertheless, the material used in this reference has an inordinate high reduction of fatigue strength due to mean stress. In section 5.1.1 of NUREG/CR-6909, for example, it can be possible to obtain three different best-fit mean S - N curves for austenitic stainless steels types 304, 316 or 316 NG. Other authors like Jaske and O'Donnell in 1977 or Tsutsumi in 2000 (see PVP 2000 - Vol. 410-2) have also proposed best fit mean S - N curve expressions for similar austenitic stainless steels (304, 316, 310, and 347), which are different from those proposed in NUREG/CR-6909. Significant differences of about +-20 percent are noticed on the fatigue life according to the best-fit S - N curve selected which shows that the S - N curve determination is a function of the chosen materials and associated fatigue test database. NUREG/CR-6909 does not sufficiently demonstrate that the tested materials and fatigue test data used for the definition of a reference best-fit mean S - N curve are representative of modern materials.	The database used to develop the new air curve is much larger and more developed for a greater number of representative materials than the database upon which the existing ASME Code fatigue design curves are based. It is an updated version of the Pressure Vessel Research Council (PVRC) database; Table 1 of this report lists the sources. The data were obtained on smooth specimens tested under strain control with fully reversed loading (i.e., R = -1) in compliance with consensus standard approaches for the development of such data. The database for austenitic SSs consists of some 520 tests on Types 304, 316, 304L, 316L, and 316NG SS; about 220 tests for Type 304 SS; 150 tests for Type 316 SS; and 150 tests for Types 316NG, 304L, and 316L SS. All austenitic SSs used in these studies comply with the compositional and strength requirements of the ASME Code specifications. The analysis did not use data from the study by Wire et al. (Ref. 105 of NUREG/CR-6909); thus, Table 1 does not include these data. The results of Wire et al. were obtained from either load-controlled tests, or the strain cycling was not reversed fully.

#	Source	Specific Comment*	NRC Comment Resolution
		The fatigue E-N data are typically expressed by using one equation to cover the two domains (ie., LCF and high cycle fatigue (HCF)). The proposed modification of the reference mean S - N curve comes from the consideration of recent fatigue test results corresponding to the HCF domain, whereas, for reactor components, design studies are mainly concerned with the LCF domain.	The literature has proposed several different best-fit mean S-N curves for austenitic SSs, including (1) Jaske and O'Donnell, Trans. ASME Pressure Vessel Technol. 99, 1977 (Ref. 70 of NUREG/CR-6909), (2) Diercks, Trans. ASME Pressure Vessel Technology 101, 1979, (3) Chopra, NUREG/CR-5704, 1998 (Ref. 37 of NUREG/CR-6909), Tsutsumi et al., PVP Vol. 410-2, 2000 (Ref. 28 of NUREG/CR-6909), and (4) Solomon and Amzallag, PVP 2005-71063, 2005. These curves differ by up to 50%, particularly in the 104- to 107-cycle regime. The analyses by Jaske and O'Donnell and by Diercks are based on the Jaske and O'Donnell database. The details regarding the database used by Tsutsumi et al. are not available. The database used in NUREG/CR-5704 included the Jaske and O'Donnell data, the data obtained in Japan (including the JNUFAD database), and some additional data obtained in the United States. The database used in NUREG/CR-6909 to develop the new air mean curve for austenitic SSs is an updated version of the database used in NUREG/CR-5704. In NUREG/CR-5704, two separate best-fit curves were developed, one for Types 304 and 316 SS and another for Type 316NG SS. However, to be consistent with the approach taken by the current ASME Code, NUREG/CR-6909 presented a single best-fit mean curve. The values of the constants in the modified Langer equation differ from those included in the models proposed by Jaske and O'Donnell or Tsutsumi et al. because of the different database used for the analysis. The current data set includes many more heats, including several heats of Types 304L, 316L, and 316NG SS. The proposed curve yields an R2 value of 0.851 when compared with the available data; the R2 values for the Tsutsumi et al., Jaske and O'Donnell, and ASME Code mean curves are 0.839, 0.826, and 0.568, respectively. Figure 30 of Section 5.1.1 of the report clearly demonstrates that the current ASME Code curve is nonconservative with respect to the PVRC database. For example, all of the data for Type 316 SS and nearly 80% of the data for Types 304 and 316N SS are below the ASME Code mean curve. The staff revised the final NUREG/CR-6909 to include these additional details. <u>Disposition:</u> The final NUREG/CR-6909 has been revised to reflect the additional details discussed in this comment/resolution.

#	Source	Specific Comment*	NRC Comment Resolution
12	I - 2f	The conclusions in NUREG/CR-6909 regarding evaluations of the mean stress effect seem to be solely based on the paper published by Bettis Bechtel Inc. (see PVP 1999 - Vol. 386). This paper suggests - for an austenitic stainless steel type 304 - that the mean stress effect can reach 26 percent of the strain amplitude in the LCF domain and in the intermediate domain of fatigue life (N < 10^6cycles). This evaluation of the mean stress effect seems too conservative and is probably mainly due to the selection of the tested materials by the Bettis Bechtel Inc. laboratory, which are not representative of modern materials. In fact, this result is essentially based on a fatigue test program performed on two stainless steel type 304 materials with very different tensile and fatigue properties. The new reference design fatigue curve in air is established in section 5.1.1 of NUREG/CR-6909 by using insufficiently supported data, since portions of the data were obtained on unrepresentative material. The hot yield strength of the tested materials can for example vary as much as 100 percent (152 to 338 MPa at 288 °C). This strong scatter of mechanical properties is attributed to variations in cold working from the surface to the center of the forgings supplied for the study. In these conditions, depending on the cold working level, it is well known that the material can present significant variations of its fatigue life in the LCF domain and in the intermediate domain. Fatigue strength results were obtained by AREVA for N = 10^7 cycles on standard polished specimens in air at room temperature on a 304L austenitic stainless steel. These results (JIP 2006—Paris, May 30–31, June 1, 2006) have shown that, in the case where progressive deformation and cold work associated to loading conditions are very limited, the maximum reduction of endurance limit is of about 10 percent, compared to 26 percent found in reference [105] cited in NUREG/CR-6909.	As discussed in the NRC's response to comment 11, the current ASME Code mean curve for austenitic SSs is not consistent with the existing fatigue S-N data. At strain amplitudes < 0.3% (stress amplitudes < 585 MPa), the ASME Code mean curve predicts significantly longer fatigue lives than those observed experimentally for several heats of austenitic SSs with composition and tensile strength within the ASME specifications. In addition, the 106-cycle fatigue limit (i.e., the stress amplitude at a fatigue life of 106 cycles) for the current ASME Code mean curve is 389 MPa, which is greater than the monotonic yield strength of austenitic SSs (about 303 MPa). Consequently, the current ASME Code design curve for austenitic SSs does not include a mean stress correction for fatigue lives below 106 cycles. However, the study by Wire et al. at the Bettis Bechtel laboratory (Ref. 105 of NUREG/CR-6909) and a more recent study sponsored by EDF, France, at GE (Solomon et al., PVP2005-71064; 3rd International Conference on Fatigue of Reactor Components, Seville, Spain, 2005) on the effect of residual stress on fatigue life clearly demonstrate that mean stress can decrease the 106-cycle fatigue limit of the material; the extent of the effect depends on the cyclic hardening behavior of the material and the resultant decrease in strain amplitude developed during load-controlled cycling. Strain hardening is more pronounced at high temperatures (e.g., 288–320 °C) or at high mean stress (e.g., > 70 MPa). Therefore, as observed by Wire et al. and Solomon et al., fatigue life for load-controlled tests with mean stress is actually increases at high temperatures or large values of mean stress. In both studies, under load control, mean stress effects were observed at low temperatures (150 °C) or at relatively low mean stress (< 70 MPa).

#	Source	Specific Comment*	NRC Comment Resolution
			NUREG/CR-6909 quotes the study by Wire et al. to demonstrate that a mean stress correction is needed for austenitic SSs in the low- to intermediate-cycle regime (i.e., 104–106 cycles). Researchers performed fatigue tests on two heats of Type 304 SS to establish the effect of mean stress under both strain control and load control. The strain-controlled tests indicated “an apparent reduction of up to 26% in strain amplitude in the low- and intermediate-cycle regime (< 106 cycle) for a mean stress of 138 MPa.” However, as the authors point out, mean stress and cold work both affected the results. Although the composition and vendor-supplied tensile strength for the two heats of Types 304 SS were within the ASME specifications, the measured mechanical properties showed much larger variations than those indicated by the vendor properties. Wire et al. state, “at 288 °C, yield strength varied from 152–338 MPa. These wide variations are attributed to variations in (cold) working from the surface to the center of the thick cylindrical forgings.” After separating the individual effects of mean stress and cold work, the results of Wire et al. indicate a 12% decrease in strain amplitude for a mean stress of 138 MPa. These results are consistent with the predictions based on conventional mean stress models such as the Goodman correlation. The staff revised the final version of NUREG/CR-6909 to include these additional details. In its response to comment 11, the staff addressed the comment that “the new reference design fatigue curve in air is established in section 5.1.1 of NUREG/CR-6909 by using insufficiently supported data, since portions of the data were obtained on unrepresentative material.” <u>Disposition:</u> The final NUREG/CR-6909 has been revised to reflect the additional details discussed in this comment/resolution.

#	Source	Specific Comment*	NRC Comment Resolution
13	I - 2g	In NUREG/CR-6909 section 5.1.5, the surface finish conditions reproduced on LCF test specimens by using a 50-grit sandpaper to obtain circumferential striations - with an average surface roughness of 1.2 µm is not sufficiently representative of those obtained on reactor components. In fact, the roughness parameter alone is not sufficient to ensure that surface finish is representative of those obtained during manufacturing of components. In addition, only two tests that were performed on rough specimens were reported in NUREG/CR-6909. This is not sufficient to determine a roughness surface effect. Fatigue tests performed on turned and ground specimens by AREVA (S. Petitjean—Fatigue 2002) have shown that the radius at the bottom of machining striations is a second critical parameter to characterize the surface roughness, in addition to average value of roughness amplitude. In conclusion, the reduction factor attributed to surface finish that comes from only one surface condition and a limited number of tests cannot be used for real components.	The NRC considered the average surface roughness of 1.2 µm for the test specimens used in the Argonne National Laboratory (ANL) study to determine the effects of surface finish on fatigue life in light-water reactor environments to be reasonably representative of those obtained on reactor components. The roughness of machined surfaces or natural finishes can range from approximately 0.2 to 4.0 µm. For example, for machining processes, typical surface finish is in the range of 0.2–3.0 µm for cylindrical and surface grinding, 0.8–3.0 µm for finish turning and drilling, and 1.6–4.0 µm for milling. For fabrication processes, surface finish values for extrusion and cold rolling can range from 0.8 to 4.0 µm. Fatigue crack initiation is very sensitive to surface finish. Although the height, spacing, shape, and distribution of surface irregularities are all important, the average surface roughness is the most important observed parameter for fatigue crack initiation (Maiya and Busch, 1975; Maiya, 1975). In an air environment, the fatigue life of rough specimens, depending on the average surface roughness, can be up to a factor of 3.0 lower than that of smooth polished specimens. Limited tests on austenitic SSs indicate that, in PWR water, the fatigue life of rough specimens is also a factor of approximately 3.0 lower than that of smooth polished specimens (see Figure 51 of NUREG/CR-6909). Unless additional tests indicate otherwise, the staff considers the effect of surface roughness on the fatigue life of these steels to be the same in air and PWR environments. It should be noted that the factor of 12.0 does consider the uncertainties associated with material variability, surface finish, and the effect of size. The factor was developed by considering distributions of material fatigue strength, surface finish effects, and size effects. Monte Carlo sampling was performed from these distributions to develop a distribution representing the cumulative distribution of these effects on fatigue life. The value of 12 corresponds to the 95th percentile of this distribution. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
14	I - 2h	The majority of the LCF tests on polished specimens in NUREG/CR-6909 were performed at high temperature. Ninety percent of the tests were performed at high temperature (between 260 °C and 325 °C) and in isothermal conditions with triangular strain variations leading to constant strain rates. These test conditions are not fully representative of realistic thermo-mechanical loadings applied on components during operation. Indeed, the triangular form of cycles with two slopes and a constant temperature chosen for the laboratory fatigue tests is very different from the actual cycles applied during operating transients, which contain successions of high strain rates and low strain rates with a variable temperature. Because the tests performed in the laboratory specimens are not representative of inservice reactor components, it is not clear that the Fen factors derived from those tests apply to the components and operating conditions in a nuclear plant.	See the NRC response to comment 7.
15	I - 3	AREVA recognizes the environmental effects demonstrated by laboratory fatigue tests on reactor materials. Nevertheless, AREVA believes that alternative methods for fatigue analysis provided in NUREG/CR-6909 and DG 1144 are too conservative and should not be used for the design of new reactors. The four main reasons for this recommendation are: a. NUREG/CR-6909 only deals with materials aspects of environmental fatigue, and addresses it with a very conservative approach, while the whole methodology of fatigue is already treated at design stage with a conservatism that cannot be removed. b. The concept of cumulative penalties, which leads to multiply by the environmental factor F(en), the reduction factor of 12, which already integrates surface finish, size effect, material variability, and loading sequence effect is too severe. In addition, AREVA believes that combining some of these effects is not justified.	See the NRC response to comments 4 and 5.

#	Source	Specific Comment*	NRC Comment Resolution
		c. There are too many uncertainties in the transposition of the specimen fatigue test results obtained in a PWR environment to component fatigue. For example, the results gathered in NUREG/CR-6909 are linked to laboratory tests for which the loading conditions are simple but not representative of the field operating conditions, where the loading parameters history (e.g., temperature gradient, pressure, strain rate, and dissolved oxygen) is much complex. d. Past fatigue failures observed in nuclear power plants were due to failure of the designer/analyst to consider the actual loading conditions, such as thermal stratification, turbulent penetration, and thermal mixing. These past fatigue failures were not attributed to the fact that the designer/analyst used either a non-conservative methodology or non-conservative Design Fatigue Curves. In other words, there is no field experience on steel components, either in-air or in LWR environment, that points to the necessity to modify the current Design Fatigue Curves. AREVA agrees that if, in the future, it becomes apparent that the environmental effects have an impact on component fatigue for the current fleet of nuclear power plants or for the new nuclear power plants, additional methods may need to be applied to the fatigue analyses.	
16	I - 4	In summary, AREVA NP is not aware of any operating experience that supports the need for these conservative design rules. Nor does AREVA believe that the NRC should establish very conservative design rules without industry peer consensus. The guidance for new plants should be restricted to a limited number of locations consistent with the approach taken for license renewal reviews. It would be preferable to review the whole methodology, including a new methodology for selecting the list of design transients relevant for environmental analysis, rather than limiting efforts to the materials aspects. Finally, if the NRC continues with the guidance in DG-1144 and NUREG/CR-6909 as written, considerable flexibility should be provided for the use of alternative methods to those provided.	See the NRC response to comment 1.

#	Source	Specific Comment*	NRC Comment Resolution
17	II - C1	Introduction of environmental effects can improve the accuracy of fatigue evaluation. Given the introduction of environmental effects, safety margins of fatigue design curve should be reviewed in a more reasonable way. The proposed safety margins of 2 on strain amplitude and 12 on cycle seem to be too high. I consider that lower safety margins can be allowed through the evaluation of the scatter in the data obtained from the tests both in the air and under the water. For example, I propose that safety margins of 1.5 on strain amplitude alone should be adopted when we introduce environmental effect. The technical basis of this safety margin is presented in PVP-2003-1775. This paper addresses the scatter in the test data only. The difference between test conditions and actual conditions needs to be also considered. However the current fatigue curve is based on the initiating point of a 3mm crack in smooth specimen. We have determined that such difference can be offset by the conservativeness of these assumptions applied in the codes.	See the NRC response to comment 4.
18	II - C2 (1)	Environmental fatigue correction factors of carbon steels, low-alloy steels and austenitic stainless steels have the following values in the case of no environmental effect. Fen,nom = exp(0.632) = 1.88 (for carbon steels) Fen,nom = exp(0.702) = 2.02 (for low-alloy steels) Fen,nom = exp(0.734) = 2.08 (for austenitic stainless steels) Fen should be 1.0 in the case of no environmental effect.	The staff does not agree with the assertion that there is no environmental effect under these conditions. The test data show a moderate environmental effect consistent with the computed Fen correction factors obtained from the equations. For additional information, see the NRC response to comment 10. Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
19	II - C2 (2)	The Japanese study (EFT Program) indicates that fatigue lives of austenitic stainless steels shows clear difference in PWR and BWR environment. The NUREG report developed the environmental fatigue correction factors by using conservatively low-DO environmental data which cause lower fatigue life. However the PWR and BWR environmental fatigue correction factors of austenitic stainless steels should be separated based on the test data. Otherwise the equation for austenitic stainless should be expressed by DO such as carbon steels and low-alloy steels.	The data from the EFT program have not been published. In open literature, most of the data in high-Dissolved Oxygen (DO) water have been obtained on Type 316NG SS. Limited data on sensitized Type 304 SS indicate that the environmental effect on fatigue life is the same in high-DO and low-DO water (see Figure 49 in NUREG/CR-6909). Unless additional data show otherwise, the staff considers the environmental effects on the fatigue life of austenitic SSs to be the same in both high-DO and low-DO environments.

#	Source	Specific Comment*	NRC Comment Resolution
20	II - C2 (3)	The Japanese study (EFT Program) also indicates that Fen of nickel-chromium-iron alloy (Inconel) is smaller than Fen of austenitic stainless steel (SS). DG-1144 doesn't mention Fen of Inconel and I assume that you use that of SS. It is too conservative to apply Fen of SS to the evaluation of Inconel. You should add Fen of Inconel in Reg. Guide. (Ref: PVP2006-93194)	The staff agrees with this comment. Disposition: The final versions of the regulatory guide and NUREG/CR-6909 include the Fen method for nickel-alloy materials (e.g., Alloy 600, Alloy 690).
21	II - D1	I agree that the regulatory guide will apply only to new construction plants. However, the applicability of this regulatory guide to actual plants needs to be investigated carefully since it entails drastic review of the current fatigue evaluation. In particular, the design transient conditions should be entirely revised applying this RG. Therefore, it is necessary to assure a sufficient period of leading time for investigation before applying the regulatory guide to an actual construction plant.	The staff does not agree that the design transient conditions will need to be revised in order to apply this regulatory guide. The staff experience with license renewal reviews indicates that the guidance can be applied using the current plant design transient conditions. Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
22	III - 1	Since DG-1144 utilizes a similar Fen methodology to that evaluated in MRP-47, Rev.1, the issues identified in MRP-47, Rev. 1 are considered to be equally applicable to the DG-1144 methodology. Some, but not all, of the issues raised in MRP-47, Rev. 1 have been specifically addressed in DG-1144. Based on this, the MRP would like to see clarification on the remaining issues included in DG-1144 or the supporting document (DRAFT REPORT FOR COMMENT NUREG/CR-6909 (ANL 06/08), "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," (July 2006.). Please clarify the following specific issues: a. "Linking" of transients pairs is not straight-forward and can lead to significant differences in results (refer to Figure 1). The MRP thinks that the recommendations made in Section 4.2.2 of MRP-47, Rev. 1 are an acceptable means of addressing linking of transients with respect to: • Situations where the starting and ending stress points between two linked transients are not equal. • Establishing the rate of change for the discontinuity between linked transients. • Computing the strain rate for linked transients.	The staff disagrees with the comment to revise the final version regulatory guide to endorse the recommendations from the mentioned sections of MRP-47, Rev. 1. DG-1144 provides guidelines for incorporating the impact of the environment on the fatigue life of components in light-water reactors. The level of analytical detail that is discussed in items a through g is beyond the scope of this regulatory guide. Regarding item h, the evaluation should use the metal temperature. The metal temperature should be available to the designer so that the designer can compute through-wall temperature stresses used in the ASME Code evaluations. Disposition: No changes were made to the final version of the regulatory guide.

#	Source	Specific Comment*	NRC Comment Resolution
		Please revise the text of DG-1144 to state that the recommendations made in Section 4.2.2 of MRP-47, Rev. 1 are an acceptable means of addressing linking of transients, or provide alternate recommendations. b. Please revise the text of DG-1144 to state that cycle counting methods other than those typically employed in ASME Code Section III calculations, such as Rainflow Cycle Counting, are acceptable for use in fatigue analyses associated with DG-1144. c. The MRP thinks that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 are an acceptable means for addressing the effect on strain rate from the elastic-plastic correction factor (Ke). Please revise the text of DG-1144 to state that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 are an acceptable means of addressing the effect on strain rate from Ke, or provide alternate recommendations. d. The MRP thinks that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 are an acceptable means for addressing stratification loads. Please revise the text of DG-1144 to state that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 are an acceptable means of addressing stratification loads, or provide alternate recommendations. e. The MRP thinks that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 are an acceptable means for addressing seismic loads. Please revise the text of DG-1144 to state that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 are an acceptable means of addressing seismic loads, or provide alternate recommendations. f. The MRP thinks that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 are an acceptable means for addressing pressure and moment loads. Please revise the text of DG-1144 to state that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 are an acceptable means of addressing pressure and moment loads, or provide alternate recommendations.	

#	Source	Specific Comment*	NRC Comment Resolution
		g. Environmental fatigue is typically linked to dissolved oxygen. As noted in MRP-47, Rev. 1, this involves inappropriate over-simplification and ignores the key role of other water chemistry parameters such as conductivity (or more correctly, level of dissolved anionic impurities) and pH. NUREG/CR-6909 notes (for example, in Section 5.2.6) that water chemistry effects have been appropriately incorporated into the model except for off-normal water chemistry conditions. Please define off-normal water chemistry conditions and provide specific guidance on what should be done to evaluate such conditions. h. NUREG/CR-6909 includes definitions for temperature for use with the Fen expressions. Please revise the text of DG-1144 to state which temperature (metal or fluid) is to be used in environmental fatigue evaluations. If it is the metal temperature, please provide guidance in DG-1144 on alternatives for cases when metal temperature is not available.	
23	III - 2	In the Introduction of DG-1144, the NRC states: "This draft regulatory guide provides guidance for use in determining the acceptable fatigue life of ASME pressure boundary components, with consideration of the light-water reactor (LWR) environment. In so doing, this guide describes a methodology that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable to support reviews of applications that the agency expects to receive for new nuclear reactor construction permits or operating licenses under 10 CFR Part 50, design certifications under 10 CFR Part 52, and combined licenses under 10 CFR Part 52 that do not reference a standard design. Because of significant conservatism in quantifying other plant-related variables (such as cyclic behavior, including stress and loading rates) involved in cumulative fatigue life calculations, the design of the current fleet of reactors is satisfactory, and the plants are safe to operate."	Section D, "Implementation," of DG-1144 clearly states that the regulatory guide only applies to new plants and that no backfitting is intended. <u>Disposition:</u> No changes were made to the final version of the regulatory guide.

#	Source	Specific Comment*	NRC Comment Resolution
		The above text is not clear on what constitutes "new nuclear reactor construction." During the August 2006 meetings of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code in Henderson, NV, the NRC clarified that DG-1144 requirements will only apply to new plant construction, and that the requirements did not apply to repair or replacement component design for operating reactors. Please revise the text of DG-1144 to state that environmental fatigue rules do not apply to repair or replacement component design for operating reactors.	
24	III - 3	In Appendix A of NUREG/CR-6909, reference is made to two papers that may be used for guidance: (1) Mehta, H. S., "An Update on the Consideration of Reactor Water Effects in Code Fatigue Initiation Evaluations for Pressure Vessels and Piping," Assessment Methodologies for Preventing Failure: Service Experience and Environmental Considerations, PVP Vol. 410-2, R. Mohan, ed., American Society of Mechanical Engineers, New York, pp. 45–51, 2000. (2) Nakamura, T., M. Higuchi, T. Kusunoki, and Y. Sugie, "JSME Codes on Environmental Fatigue Evaluation," Proc. of the 2006 ASME Pressure Vessels and Piping Conf., July 23–27, 2006, Vancouver, BC, Canada, paper # PVP2006–ICPVT11–93305. While both of these papers describe Fen methodologies and their application to fatigue analyses, the Fen formulas contained in these papers differ from those specified in DG-1144 and supporting document NUREG/CR-6909. Please revise the text of DG-1144 to state that the Fen methods and formulas specified in either of the above two documents are acceptable alternatives to the methodology specified in DG-1144.	The staff disagrees with this comment. The papers listed in NUREG/CR-6909 are for reference only. Section C, "Regulatory Position," of the regulatory guide contains the method endorsed by the staff. Disposition: No changes were made to the final version of the regulatory guide.

#	Source	Specific Comment*	NRC Comment Resolution
25	III - 4	DG-1144 does not provide any specific methods for evaluating Ni-Cr-Fe material. Alloy 600 and Alloy 690 materials, for example, have regularly been used in operating nuclear plants. It is assumed that this practice will continue for new reactors. Ni-Cr-Fe rules that have previously been applied by some license renewal applicants are specified in the following documents: • O. Chopra, "Status of Fatigue Issues at Argonne National Laboratory," Presented at EPRI Conference on Operating Nuclear Power Plant Fatigue Issues & Resolutions, Snowbird, UT, August 22–23, 1996. • EPRI TR-105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations," December 1995. Please revise the text of DG-1144 to state that the rules defined in the above two documents are acceptable for use in evaluating Ni-Cr-Fe materials (including Alloy 690).	See the NRC response to comment 20.
26	III - 5	DG-1144 specifies rules for fatigue analysis for new reactor design. It is assumed that new reactors will need to be certified in accordance with ASME Code, Section III, in order to receive an N-stamp or similar certification prior to entry into service. The fatigue rules specified in DG-1144 currently differ from the fatigue rules specified in ASME Code, Section III. At this point in time, there is no reason to believe that the ASME Code will adopt methodology into Section III that is consistent with the methodology specified in DG-1144. Please revise the text of DG- 1144 to state how these differences are to be reconciled to allow proper certification of nuclear components for new reactors.	The person making the comment appears to be concerned that implementation of the regulatory guide criteria would impact the design certification under Section III of the ASME Code. The staff does not consider this to be a problem. As indicated in the NRC response to comment 1, paragraph NB-3121 of the ASME Code states that the ASME Code design fatigue curves did not consider corrosive environments. The regulatory guide provides guidance to address the environmental conditions for the purpose of nuclear power plant licensing. The designer can easily compute the fatigue usage using the existing ASME Code fatigue curves if necessary for ASME Code design certification. Disposition: No changes were made to the final version of the regulatory guide.

#	Source	Specific Comment*	NRC Comment Resolution
27	III - 6	Page A.3 of NUREG/CR-6909 states the following: "When the results of detailed transient analyses are available an average temperature (i.e., average of the maximum and minimum temperatures for the transients) may be used to calculate Fen. The maximum temperature can be used to perform the most conservative evaluation." We are not clear on the definition of "average" temperature and how it would be used in each of the recommended methods of evaluation. As an example, consider a fluid temperature transient that step changes from 550 °F to 100 °F and pairs with a Zeroload (zero stress at 70 °F) transient. Based on the guidance in Appendix A of NUREG/CR-6909, we understand the following with respect to the use of an average temperature: • The Fen would be computed based on the following for an "average strain rate" approach: • An average strain rate may be determined using the difference between the peak stress for the cooldown transient and zero, and the time from the beginning of the transient until the peak stress occurs. • An average transient temperature of 310 °F (i.e., average of 70 °F and 550 °F) for this postulated transient pairing may be used. • Alternatively, the Fen would be computed based on the following for a "modified rate approach" (as described in Section 4.2.14 of NUREG/CR-6909): • An integrated strain rate may be determined using Equation (28) of NUREG/CR-6909 for the tensile portion of the cooldown transient. • For each integration step, the average temperature during the integration step is used in Equation (28). Alternatively, as a simplification, the average transient temperature of 310 °F (i.e., average of 70 °F and 550 °F for the two transients being evaluated) may be used for all integration steps.	The person making the comment requested guidance for defining the "average" temperature to be used for the Fen calculation. This regulatory guide does not intend to define the exact details of the analytical method. However, the average temperature used in the calculations should produce results that are consistent with those that would be obtained using the modified rate approach described in Section 4.2.14 of NUREG/CR-6909. In the case of a constant strain rate and a linear temperature response, the average temperature would be the arithmetic average. However, when the temperature response is not linear, an arithmetic average would not be appropriate. Disposition: No changes were made to the final version NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
		Alternatively, for either of the above examples, the maximum temperature of 550 °F could be used to provide a conservative assessment. Please expand the text Appendix A of NUREG/CR-6909 to state that the above examples are an acceptable means of addressing average temperature, or provide alternate recommendations.	
28	III - 7	For cumulative usage factor (CUF) due to rapid thermal cycling, such as the cycling typically evaluated for boiling water reactor (BWR) feedwater nozzles, the MRP thinks that Fen = 1.0 is appropriate. Similar to dynamic loading practices, this approach is based on the premise that the cycling due to rapid thermal cycling occurs too quickly for environmental effects to be significant. Please revise the text of DG-1144 to state that the application of Fen = 1.0 is an appropriate treatment of rapid thermal cycling fatigue effects in environmental fatigue analyses, or provide alternate recommendations.	The staff disagrees with the recommendations provided. The calculation of Fen should use the appropriate formula provided in the regulatory guide. The staff notes that stresses associated with rapid thermal cycling will probably fall below the strain threshold (Fen = 1.0) provided in the regulatory guide. <u>Disposition:</u> No changes were made to the final version of the regulatory guide.
29	III - 8	There is no guidance in DG-1144 regarding how to treat carbon steel or low alloy steel that is protected from the primary coolant environment by stainless steel (or Alloy 690) cladding. The MRP thinks it is reasonable to neglect the effects of the cladding and perform environmental fatigue assessment of the underlying base material, consistent with ASME Code, Section III methodology where the structural effects of cladding are neglected when the cladding is less than 10% of the component wall thickness. Please revise the text of DG-1144 to state that the cladding may be neglected in environmental fatigue analyses, or provide alternate recommendations.	See the NRC response to comment 6.
30	IV - 1	The terms “life” and “number of cycles” have been used interchangeably throughout the text.	The staff finds both terms appropriate within the context of the regulatory guide. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
31	IV - 2	In the Introduction of DG-1144, the NRC states: "This draft regulatory guide provides guidance for use in determining the acceptable fatigue life of ASME pressure boundary components, with consideration of the lightwater reactor (LWR) environment. In so doing, this guide describes a methodology that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable to support reviews of applications that the agency expects to receive for new nuclear reactor construction permits or operating licenses under 10 CFR Part 50, design certifications under 10 CFR Part 52, and combined licenses under 10 CFR Part 52 that do not reference a standard design. Because of significant conservatism in quantifying other plant -related variables (such as cyclic behavior, including stress and loading rates) involved in cumulative fatigue life calculations, the design of the current fleet of reactors is satisfactory, and the plants are safe to operate." During the August 2006 meetings of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code in Henderson, NV, the NRC clarified that DG-1144 requirements only applied to new plant construction, and that the requirements did not apply to replacement component design for operating reactors. Recommended change—Explicitly clarify that DG-1144 does not apply to repaired or replaced components for currently operating plants.	See the NRC response to comment 23.
32	IV - 3	DG-1144 does not provide any specific methods for evaluating Ni-Cr-Fe material. Alloy 600 and Alloy 690 materials, to name a few, have regularly been used in operating nuclear plants. It is assumed that this practice will continue for new reactors. Ni-Cr-Fe rules that have previously been applied by some license renewal applicants are specified in O. Chopra, "Status of Fatigue Issues at Argonne National Laboratory," Presented at EPRI Conference on Operating Nuclear Power Plant Fatigue Issues & Resolutions, Snowbird, UT, August 22–23, 1996. These rules are similar to those found in EPRI TR-105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations," December 1995. Recommended change—Clarify that these rules are acceptable for use to evaluate Ni-Cr-Fe materials (including Alloy 690).	See the NRC response to comment 20.

#	Source	Specific Comment*	NRC Comment Resolution
33	IV - 4	DG-1144 specifies rules for fatigue analysis for new reactor design. It is assumed that new reactors will need to be certified in accordance with ASME Code, Section III, in order to receive an N-stamp or similar certification prior to entry into service. The fatigue rules specified in DG- 1144 currently differ from the fatigue rules specified in ASME Code, Section III. At this point in time, there is no reason to believe that the ASME Code will adopt methodology into Section III that is consistent with the methodology specified in DG-1144. Recommended change—Clarify how these differences will be reconciled to allow proper certification of nuclear components for new reactors.	See the NRC response to comment 26.
34	IV - 5	DG-1144 does not provide guidance for how analyses of rapid thermal cycling fatigue, such as those typically evaluated for boiling water reactor (BWR) feedwater nozzles, are to be evaluated for environmental effects? Recommended change—Clarify that environmental effects do not apply to fatigue from sufficiently rapid cycles such as the thermal cycling fatigue noted.	See the NRC response to comment 28.
35	IV - 6	Page 4, Paragraph 2, Lines 5–8: It is stated that Figures 9, 10 and 37 in NUREG/CR-6909 are prepared using margins of 12 for life and 2 for stress for carbon steel, low alloy steel, and austenitic stainless steel respectively. It appears from page 47 of Draft NUREG/GR-6909 a margin of 20 on cycles was used in Figure 37. It is possible that Figure 37 of NUREG/CR-6909 was constructed with a margin of 12 on cycles and the text in page 47 is in error. Please clarify. Recommended change—The DG-1144 statement will be correct if the draft NUREG/CR- 6909, page 47, Paragraph 5.1.8, Line 10, “20 on cycles” is changed to “12 on cycles.”	The staff agrees with this comment. <u>Disposition:</u> The correction was made to the final version of NUREG/CR-6909.
36	IV - 7	Page 1, Paragraph 2, The last sentence should read, “The cumulative usage factor (CUF), calculated on the basis of Miner's rule, is the... ...exceed 1.” Recommended change—Insert, “, calculated on the basis of Miner's rule,”.	The staff agrees with this comment. <u>Disposition:</u> The recommended change was made to the final version of NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
37	IV - 8	In Appendix A of NUREG/CR-6909, reference is made to two papers that may be used for guidance: (1) Mehta, H. S., "An Update on the Consideration of Reactor Water Effects in Code Fatigue Initiation Evaluations for Pressure Vessels and Piping," Assessment Methodologies for Preventing Failure: Service Experience and Environmental Considerations, PVP Vol. 410-2, R. Mohan, ed., American Society of Mechanical Engineers, New York, pp. 45–51, 2000. (2) Nakamura, T., M., Higuchi, T. Kusunoki, and Y. Sugie, "JSME Codes on Environmental Fatigue Evaluation," Proc. of the 2006 ASME Pressure Vessels and Piping Conf., July 23–27, 2006, Vancouver, BC, Canada, paper # PVP2006-ICPVT11-93305. While both of these papers describe Fen methodologies and their application to fatigue analyses, the methodologies differ from those specified in DG-1144 and supporting document NUREG/CR-6909. Recommended change—Please clarify that the Fen methods specified in either of the above two documents are an acceptable alternative to the methodology specified in DG-1144 providing that the Fen formulas provided in DG-1144 are used in lieu of those in the referenced documents.	See the NRC response to comment 24.
38	IV - 9	The methodology used by DG-1144 is an environmental fatigue multiplier (Fen) approach, very similar to the approach being used by license renewal applicants, as documented in NUREG/CR-5704 (ANL-98/31), "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999, and NUREG/CR-6583(ANL-97/18), "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," March 1998.	See the NRC response to comment 22.

#	Source	Specific Comment*	NRC Comment Resolution
		The MRP provided guidance for performing plant specific environmental fatigue evaluations for plants pursuing license renewal in EPRI TR-1012017, “Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47 Revision 1),” April 2005, which Dominion understands is being provided to the NRC by the MRP as part of its comments to the draft Regulatory Guide and NUREG. The intent of MRP-47, Rev. 1 was to unify the process used by applicants to address environmental effects in the License Renewal Application, and provide specific guidance on the use of currently accepted environmental fatigue evaluation methodologies. As a result of industry application of the Fen relationships, MRP-47, Rev. 1 identified several practical issues associated with the application of the Fen methodology to typical industry fatigue evaluation problems. These issues have led to application of a variety of different solutions applied by analysts depending upon the analyst or the level of detail available in the existing fatigue evaluations. This varied approach has led to non-consistent application of the Fen approach between plants, and some amount of confusion amongst the industry. Since DG-1144 utilizes a similar Fen methodology to that evaluated in MRP-47, Rev. 1, the issues identified in MRP-47, Rev. 1 are considered to be equally applicable to the DG-1144 methodology. Some, but not all, of the issues raised in MRP-47, Rev. 1 have been specifically addressed in DG-1144. Based on this, clarification on the remaining issues included in DG- 1144 or the supporting document (DRAFT REPORT FOR COMMENT NUREG/CR-6909 (ANL 06/08), “Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials,” July 2006.) are appropriate. Please clarify the following specific issues: a. “Linking” of transients pairs is not straight-forward and can lead to significant differences in results (refer to Figure 1) in part due to the following items: . Treatment of cases where the starting and ending stress points are not equal needs to be clarified. . The rate of change assumed for the discontinuity between transient's needs to be addressed? . What is strain rate?	

#	Source	Specific Comment*	NRC Comment Resolution
		Recommended change—Clarify that the recommendations made in Section 4.2.2 of MRP-47, Rev. 1 for addressing linking of transients are an acceptable approach. b. Recommended change—Clarify that cycle counting methods other than those typically employed in ASME Code Section III calculations, such as Rainflow Cycle Counting, are acceptable for use in fatigue analyses associated with DG-1144. c. There is an effect on strain rate from the elastic-plastic correction factor (Ke). How should this be evaluated? Recommended change—Clarify that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 for addressing Ke effects are acceptable. d. How are stratification loads to be addressed using the Fen methodology? Recommended change—Clarify that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 for addressing stratification loads are acceptable. e. How are seismic loads to be addressed using the Fen methodology? Recommended change—Clarify that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 for addressing stratification loads are acceptable. 1. How are pressure and moment loads to be addressed using the Fen-methodology? Recommended change—Clarify that the recommendations made in Section 4.2.6 of MRP-47, Rev. 1 for addressing stratification loads are acceptable.	

#	Source	Specific Comment[*]	NRC Comment Resolution
		g. Recommended change—Environmental fatigue is typically linked to dissolved oxygen. As noted in MRP-47, Rev. 1, this involves inappropriate over-simplification and ignores the key role of other water chemistry parameters such as conductivity (or more correctly, level of dissolved anionic impurities) and pH. NUREG/CR-6909 notes (for example, in Section 5.2.6) that water chemistry effects have been appropriately incorporated into the model except for off-normal water chemistry conditions. Please define off-normal water chemistry conditions and provide specific guidance on what should be done to evaluate such conditions. h. Recommended change—NUREG/CR-6909 includes definitions for temperature for use with the Fen expressions. Please clarify whether the temperature to be used should be the metal temperature or the fluid temperature. If it is the metal temperature, please provide guidance on alternatives for cases when metal temperature is not available.	
39	V - 1	4.2.7 Sulfur Content in Steel Equation (20) gives S*=0.015 at DO > 1.0 ppm regardless of sulfur content. This value seems too much severe for lower sulfur steels to compare Japanese model (Figure 2 in PVP2006 93194) but the environmental condition of DO > 1.0 ppm is unusual in plant operation and thus this influence on the evaluation of environmental fatigue of actual plants may be very small. Equation (20) also gives S* = 0 at S = 0 and DO (Ü1.0 ppm, this is not supported by experimental data. Japanese model (PVP2006 93194) indicates Fen = 12.3 at the conditions of S = 0%, 289 C, DO = 0.7 ppm and 0.001%/s. The NRC model is too much unconservative for low sulfur steels.	The staff agrees that the environmental condition of DO > 1.0 ppm is unusual in plant operation; the influence of S* = 0.015 at DO > 1.0 ppm is primarily for evaluating laboratory data and not actual plant operation. Regarding S* = 0 at S = 0, in practice, S content is never zero; there is always a trace amount of S (e.g., 0.001 wt.%) in the steel. For S = 0.001 wt.%, DO = 0.7 ppm, temperature = 289 °C, and strain rate = 0.001%/s, Equation (20) yields a Fen value of 2.4 and 2.6 for carbon steel and low-alloy steel, respectively. Therefore, the staff finds the Fen method acceptable for low-sulfur steels. Disposition: No changes were made to the final version of NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
40	V - 2	4.2.12 Statistical Model & 4.2.13 Environmental Fatigue Correction Factor Equation (18) does not equal to Equation (15) without environmental effects. It seems not reasonable. Why the constant A of these equations are not same? The same situation can be seen in the Equations (19) and (16). Caused by this reason, Fen does not equal to 1.0 for Equations (25) and (26) when the environmental effect is zero. In this NUREG, the difference of Fen value between carbon and low alloy steels becomes very little, there is no necessity to apply different equations for these steels. It seems enough to use the same equation (averaged of equations (18) and (19)). Japanese model gives only one equation for these steels.	See the NRC response to comment 18. The staff agrees that the difference in the Fen values between carbon steels and low-alloy steels is small; the Fen correlations in Section 4.2.13 of NUREG/CR-6909 are based on the analysis of the available data. Disposition: No changes were made to the final version of NUREG/CR-6909.
41	V - 3	5.2.5 Dissolved Oxygen Japanese fatigue data indicate that fatigue life of stainless steels in water was not influenced by the dissolved oxygen concentration itself but influenced strongly by the water chemistry of PWR and BWR. Additionally, several Japanese data indicate that any fatigue life reduction of fully sensitized type 304 SS cannot be observed at the conditions of 289 C BWR water, DO=0.2 and 0.01 ppm and 0.001%/s strain rate. The mechanism of fracture is considered to be fully fatigue because the fracture surface of sensitized 304 SS seems perfectly transgranular and SCC does not influence this failure. The effects of sensitization on fatigue of stainless steel need not be considered for fatigue evaluation in BWR water. Current NRC model gives too severe Fen for stainless steel in BWR water. Based on these results, the different equations should be applied for PWR and BWR, respectively, as similar as Japanese model.	See the NRC response to comment 19.
42	V - 4	5.2.12 Statistical Model & 5.2.13 Environmental Fatigue Correction Factor Equation (32) does not equal to Equation (30) without environmental effects. It seems not reasonable. Why the constant A of these equations are not same? Caused by this reason, Fen does not equal to 1.0 for Equations (32) when the environmental effect is zero. This is the same comment as 4.2.12. Current NRC model gives too severe Fen for stainless steel in BWR water. The different equations should be applied for PWR and BWR, respectively, as similar as Japanese model.	See the NRC responses to comments 18 and 19.

#	Source	Specific Comment*	NRC Comment Resolution
43	VI - 1	The use of the DG1144 guidance will result in an increase in the calculated cumulative fatigue usage in the ASME Code stress analyses of the NSSS piping. As a result. more locations are likely to exceed a fatigue usage of 0.1, the threshold specified for pipe break postulation in EMEB3-11. This in turn will result in the postulation of significantly more break locations and thus lead to the design and installation of more pipe whip restraints. Pipe whip restraints can hinder access for inservice-inspections, thus an increase in the number of pipe whip restraints may have an adverse effect on plant safety. Therefore, the staff is requested to consider an increase in the fatigue usage threshold. GE notes that the specified fatigue usage threshold for pipe break postulation in ANSI 58.2 was 0.4. GE suggests the staff consider an increase in the fatigue usage threshold for pipe break postulation, for example to -0.8, under the proposed guidance in DG1144.	See the NRC response to comment 2.
44	VI - 2	Neither the DG1144 nor NUREG/CR-6909 currently addresses Ni-Cr-Fe materials (e.g., Alloy 690 & Alloy 600), GE suggests the staff consider adding clarification to the DG1144 or NUREG/CR-6909 to address materials such as these, or communicate plans for future revisions that would include guidance for these materials.	See the NRC response to comment 20.
45	VII - 1	Draft Regulatory Guide DG-1144 should be withdrawn. The DRG provides “guidelines” for performing the ASME Section III Class I code compliance analyses for suitability of pressure-retaining components for cyclic conditions (fatigue analysis). Two specific revisions to the Section III requirements are being made by NRC. NRC is revising the Section III design fatigue curve for stainless steel. And NRC is specifying environmental correction factors to be used with the code fatigue curves to account for the effects of LWR coolant environments. The DRG is based on evaluation of research test data given in NUREG/CR-6909. The research test data used as a basis for the DRG is incompatible with the Section III design fatigue curves. The wrong failure criterion was used, the test specimens were much smaller than those used to establish the Section III design fatigue curves, and effects of elevated temperature and variable strain rates are included in the data.	The staff disagrees with this comment. The person making the comment states that the regulatory guide should be withdrawn, in part, because the research data upon which it is based are incompatible with the Section III design curves. As discussed in the NRC response to comment 1,paragraph NB-3121 of the ASME Code states that the tests on which the ASME fatigue curves are based did not include tests in the presence of corrosive environments that might accelerate fatigue failure. The staff considers the regulatory guidance for environmental fatigue as supplementing the existing ASME Code fatigue requirement in an area not addressed by the Code. The person making the comment also expressed a concern with the staff’s modification of the ASME Code SS air curve. The staff discussed its technical concerns regarding the lack of conservatism in the existing SS air curve in NUREG/CR-6909. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
		In the late 70s, there was a proposal to revise the Section III design fatigue curve for stainless steel based on more recent test data. This proposal was rejected by the Section m committee. There were sound, documented, technical arguments for rejecting the proposal from experts on fatigue design. Those experts on fatigue design are no longer with us. However, the same arguments apply to rejection of the DRG revision of the stainless steel design fatigue curve. As a regulatory agency, I can understand NRC taking exception to Section III requirements if there is a verified safety concern. But, I don't see a verified safety concern. I suspect that there is no legal basis for NRC to unilaterally revise Section m, which is how I interpret the guidelines in the DRG. In addition, based on my review of the DRG and NUREG/CR-6909, I have to conclude that the NRC staff does not understand the code methodology for fatigue design. As stated in the first paragraph of this letter, I request that the DRG be withdrawn. If there is a technical concern with the Section III requirements, I recommend that NRC officially document that concern with supporting technical data for consideration by the Section III committee. Please don't provide research data that is incompatible with the design methodology.	
46	VII - 2	Background on Section III fatigue design: The Section III fatigue design methodology was developed in the late 50s early 60s. There are three main parts—elastic analysis methods to predict stress, specific procedures for cyclic evaluation and cumulative damage, and a design fatigue curve. All three parts are dependent on each other. The Section III design fatigue curves (to 106 cycles) were developed in the early 60s. Langer was the principal contributor and the curves are based on the low-cycle strain fatigue work of Coffin and applicable fatigue data from the late 50s. The Section III methodology is an extension of well-established machine design methods for fatigue. The main improvement was the use of strain-controlled cyclic data to establish the material S-N curve. The other improvement was the "cyclic operation" procedures. For design, the sequence of events in a nuclear plant is unknown, and therefore, the cyclic history is an unknown. A conservative procedure is defined [NB-3222.4] to ensure that the worse possible cyclic stresses are evaluated.	See the NRC response to comment 45.

#	Source	Specific Comment*	NRC Comment Resolution
		The Section III material S-N curve used for fatigue design is based on the tests performed in the late 50s. These were uniaxial, fully-reversed, strain-controlled cyclic tests on "small polished specimens" [from the ASME Criteria document] to separation (specimen fracture). The stainless steel data included 146 experimental values [Langer, 1962]. A best-fit curve was determined based on the Coffin relationship. The design fatigue curve is "...based on the best-fit curve and with a safety factor of either 2 or 20 on cycles, whichever is more conservative at each point. It is believed that these safety factors are sufficient to cover the effects of size, environment, surface finish, and scatter of data." [Langer, 1962]. In 1977, a technical paper on fatigue by Jaske & O'Donnell was published [77-PVP-121]. This paper included "new technology and data." It also included load-controlled axial and bending-fatigue data. A revised design curve was proposed for stainless steel. This proposal was reviewed by the Section III code committee. The code committee correspondence shows that the experts on fatigue design concluded that there was no need to revise the Section III stainless steel low-cycle design fatigue curve.	
47	VII - 3-1	The NRC environmental correction factors are not appropriate for use with the Section III design fatigue curves. The NUREG/CR-6909 data is from cyclic tests on a much smaller specimen size, of a different configuration (tubular), with failure defined as 25% load drop. The Section III design fatigue curves are based on cyclic tests of much larger, solid, specimens with failure defined as separation. The NRC use of 25% load drop data, which is essentially crack initiation data, is such a fundamental error as to be inexcusable.	It is widely recognized that the difference in fatigue life for specimens tested to 25% load drop and those tested to failure is relatively small. There is excellent agreement between the mean curves in air for carbon and low-alloy steels generated using the current specimen design and test techniques and the original mean air curves used to develop the ASME Code Section III of the ASME Code fatigue design curves for carbon and low-alloy steels. Thus, in air, the differences in test technique and specimen design have no effect, and the procedures used in NUREG/CR-6909 give results that are fully consistent with the current ASME Code. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
48	VII - 3-2	The NRC environmental correction factors include variable strain rate effects. For use with the Section III design fatigue curves, material testing should be performed at the same strain rate as the original tests. Variable strain rate effects should not be included in design since the cyclic strain history is not known. The Section III cyclic operation analysis procedures are sufficiently conservative to account for strain rate effects.	The fatigue design curves are applied to cyclic loads that result in a wide range of strain rates. If the fatigue life does not depend on strain rate, as is the case in air, the results for a single strain rate may be used. However, in water environments, fatigue life does depend on strain rate, and the designer must use fatigue lives corresponding to the actual strain rates of interest. Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
49	VII - 3-3	The NRC environmental correction factors include temperature effects. For use with the Section III design fatigue curves, material testing should be performed at room temperature. If temperature effects are a technical concern, NRC should provide technical data to ASME for consideration.	In an air environment, the effect of temperature on fatigue life over the range of temperatures of interest is small and need not be considered explicitly. In a water environment, the effect is large, and, thus, temperature is considered explicitly in assessing the environmental effect. Disposition: No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
50	VII - 3-4	The NRC revision of the Section III stainless steel air curve is not valid. It is inappropriate to collect cyclic data that is not consistent with the original test methods, and plot that data to construct a design fatigue curve. According to NUREG/CR-6909, the stainless steel air data include the Jaske and O'Donnell data, the JNUFAD database from Japan, studies at EDF in France, and the results of Conway et al. and Keller. The Jaske and O'Donnell data has already been considered by ASME (as discussed in the background section above) and the decision was made not to revise the Section III design fatigue curve for stainless steel. I have not reviewed the other data cited, but I expect that the test methods are inconsistent with the test methods used to construct the Section III design fatigue curves.	The staff disagrees with this comment. The fatigue life must be obtained under conditions appropriate to its use. A fatigue life obtained in air can be used under conditions for which the fatigue life is similar to the life in air. Disposition: No changes were made to the final version of NUREG/CR-6909.
51	VII - 3-5	The test data to determine environmental effects is not being properly evaluated. Each unique set of test parameters should be individually evaluated. The test parameters are specimen material, specimen size, specimen configuration, strain rate, and temperature (I assume that all tests are uniaxial, fully reversed, and strain-controlled). Each unique test parameters should be tested in air and in water. The reduction in cycles (the environmental factor) should be calculated for each unique set of test parameters.	This is a question of experimental design. Test data are available over a wide range of conditions that encompass all those relevant to light-water reactor environments. Disposition: No changes were made to the final version of NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
52	VII - 3-6	There is a simple test to properly determine LWR coolant environmental effects on the Section III design fatigue curve. Use polished bar specimens of the same size as used in the original tests. Do fully reversed, strain-controlled tests to separation at the same strain rate used in the original tests. Perform testing at the same strain amplitude in air and in LWR water. Compare the cycles to failure to determine environmental effects. I expect that these tests will demonstrate that the environmental effects are well within the 2/20 factors.	This would be an appropriate approach if the strain rates in all components under all cyclic loadings were the same in the original tests or if the fatigue life did not depend on strain rate. Because this is not the case, the proposed approach is inappropriate. <u>Disposition:</u> No changes were made to the final version of NUREG/CR-6909.
53	VII - 4-1	The DRG endorses a new stainless steel air design curve for use with the Section III Class 1 fatigue analysis rules. The stated reasons are "nonconservatism of the current ASME stainless steel air design curve" and "More recent evaluations of stainless steel test data indicate that the ASME curve is inconsistent with the appropriate test materials and conduct of the fatigue test." Both of these stated reasons are incorrect. Rather than the ASME curve being inconsistent, the more recent testing is inconsistent with the test methods used to construct the design fatigue curve. Because more recent testing uses much smaller specimens does not mean that the Section III design fatigue curve is incorrect. It means that the more recent testing is not directly applicable to construction of a Section III design fatigue curve. If the more recent testing includes load-controlled data, that data is not directly applicable. If the more recent testing uses 25% load drop as a failure criterion, that data is NOT applicable for construction of a Section III design fatigue curve. DG-1144, Page 3—discussion of margins of 2 on strain and 20 on cyclic life—"(including temperature differences between specimen test conditions and reactor operating experience)"—I believe the temperature comment is not correct. I have not seen the temperature effects discussed in this manner in the historical literature. I think there was a different "consideration" for temperature effects. Page 3—"More recent fatigue test data from the United States, Japan, and elsewhere show that the LWR environment can have a significant impact on the fatigue life..."—The data show a significant impact on crack initiation of a very small test specimen. To equate significant impact on crack initiation of a very small test specimen with significant impact on low-cycle fatigue life of a nuclear component is wrong and misleading in the extreme.	The staff believes that the ASME Code Section III design seeks to avoid initiation of fatigue cracks. Initiation of a fatigue crack is a local property that depends on local values of strain range, and, in light-water reactor environments, on local values of strain rates, temperatures, and material and water chemistry. Thus, initiation in a specimen test is directly relevant to initiation in a component, and evaluating the effect of subfactors such as surface roughness on specimens is relevant to its effect on initiation in a component. It is implausible to assume that the original designers intended the factors of 2 and 20 to encompass the differences in the fatigue life to failure, with failure defined as total structural failure, among the test specimens used to develop the curve and every component, ranging from small pipes to large pressure vessels, to which it might be applied. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
		Page 3—"...the researchers analyzed existing data to predict fatigue lives as a function..."—The data does NOT predict fatigue lives of nuclear components. The data predicts the occurrence of crack initiation in a very small material test specimen. Page 4—There is a discussion of "an evaluation of the ASME design curve margins."—"...the researchers reviewed data available in the literature to assess the subfactors..."—It is meaningless to evaluate specimen data to evaluate the subfactors. To evaluate the subfactors requires evaluation of actual nuclear component fatigue failure data. Page 4—"This methodology involves a strain-based integral for evaluating conditions for which temperature and strain rate change, resulting in variation of Fen over time." From a design standpoint, the strain-based integral approach has no technical basis and is nonsense. The specimen tests are done at controlled strain rates. To extrapolate from those controlled strain rates to a cyclic evaluation of a nuclear component with unknown cyclic history is unreasonable and meaningless.	
54	VII - 5-1	"The evaluations used in resolving GSI-166 and GSI-190 relied on conservatism in the existing component fatigue analyses." I have been involved in Class I piping fatigue analyses on many different nuclear plants. My experience leads me to believe that, unless knowledgeable engineers with expertise in code fatigue evaluations performed the analyses, the existing component fatigue analyses are probably unconservative. And therefore, I question the resolution of GSI-166 and GSI-190. Definition of the design thermal transients is the only conservatism that I am aware of.	The person making the comment questions the technical adequacy of the fatigue evaluations used for the resolution of GSI-166 and GSI-190. NUREG/CR-6909 documents the fatigue evaluations that form the basis for the resolution of GSI-166 and GSI-190. The person making the comment has not provided specific information that challenges the conservatism of the NUREG/CR-6260 fatigue evaluations. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.

#	Source	Specific Comment*	NRC Comment Resolution
55	VII - 5-2	"The staff based this conclusion primarily on the negligible calculated increase in core damage frequency in extending a plant's operating life from 40 to 60 years." and "Argonne National Laboratory (ANL) developed statistical correlations that can be used to evaluate the fatigue life of ASME Code components in LWR environments." These two statements are alarming to me. It appears that NRC is taking the statistical evaluation of small material specimen data to predict the failure probability of actual nuclear plant components. If this is the case, I question the technical capability of the NRC staff and the technical adequacy of the core damage frequency calculations.	The person making the comment questions the technical evaluation associated with the resolution of GSI-190 because it relied on the ANL statistical evaluation of small material specimens to predict the failure probability of actual nuclear power plant components. The ANL statistical evaluation of the specimen test data was used to estimate the probability of fatigue crack initiation of the components. The GSI-190 evaluation used the ANL statistical evaluation of the specimen test data because sufficient test data are not available for full-size components. Specimen data were used to construct the ASME Code design fatigue curves that are currently used to evaluate full-size components. The specimen test data used to construct the ASME Code design fatigue curves were adjusted to account for the differences between small laboratory specimens and actual full-size components. The GSI-190 study made similar adjustments to the ANL specimen data. The person making the comment did not suggest a viable alternative to this procedure. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.
56	VIII - 1	ASME will consider adopting the proposed regulatory guide approach in the format of a Code Case. This action will enable a thorough review by ASME constituents. If consensus approval is obtained, an agreement between the NRC and ASME about one acceptable method of addressing the impact of environmental fatigue would be achieved. ASME will continue to develop other Code Cases covering alternative ways of addressing this impact. The voting process on this Code Case is underway and it is anticipated that any comments and/or objections will be resolved in a timely manner and the Code Case will be issued early in 2007. Once these Code Cases are issued, ASME requests the NRC to endorse these Code Cases in a revision of the Regulatory Guide 1.84. In this manner, ASME plans to foster continued cooperative development of acceptable ways to deal with the impact of environmental fatigue in a timely manner. The goal will be to develop a method that can be implemented in the component design in a straightforward manner, without undue conservatism, that will provide assurance of adequate fatigue life when environmental factors are present.	The NRC staff will consider endorsing available ASME Code cases through its normal process for revising Regulatory Guide 1.84. <u>Disposition:</u> No changes were made to the final version of the regulatory guide and NUREG/CR-6909.

*. Comments are quoted directly from the letter submitted by the source.