UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 21, 2003

ORGANIZATION: Nuclear Energy Institute

SUBJECT: SUMMARY OF MEETING WITH THE NUCLEAR ENERGY INSTITUTE (NEI) TO DISCUSS STAFF'S REQUEST FOR ADDITIONAL INFORMATION (RAI) ON ENVIRONMENTAL ASSISTED FATIGUE

On July 24, 2003, the U.S. Nuclear Regulatory Commission (NRC) staff met with the Nuclear Energy Institute (NEI), the Electric Power Research Institute (EPRI), and other industry representatives to discuss the staff's Request for Additional Information (RAI) on EPRI technical report that addresses environmental assisted fatigue for carbon and low alloy steels. This RAI is in response to a meeting held on September 18, 2002, that discussed aging management of environmental fatigue for carbon and low alloy steels. At the meeting, the industry requested the staff to review the above environmental fatigue issues under interim staff guidance (ISG). By letter dated January 17, 2003, the NEI submitted the industry recommendation of fatigue environmental effects, as ISG-11, for staff review (See ADAMS Accession No. ML030300144). The staff has reviewed the industry recommendation on the ISG-11, and subsequently issued the aforementioned RAI on June 30, 2003 (See ADAMS Accession No. ML031810630).

Since the industry recommendation is based on an EPRI technical report, "Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74)," EPRI was preparing the staff's RAI response. EPRI provided a presentation on the staff's RAI response during the meeting and their interpretation of what was needed to respond to the RAI. The staff discussed and provided the following comments:

- In its RAI, the staff questioned whether the standard deviation used by EPRI for the fatigue endurance limit was consistent with the data presented in Figure 14 of Attachment 1 to the ISG submittal. The industry representatives presented a reassessment of the data presented in Figure 14, in order to support the assumption of the standard deviation used in the EPRI study. This reassessment involved adjusting some of the data using a modified Goodman approach to account for mean stress effects. This modification reduced the apparent data scatter. However, the staff review of the adjustment (i.e., after meeting) found that the adjustment was not performed properly for some data points. The test data was adjusted using the maximum values assumed in constructing the ASME fatigue design curves instead of the actual values of mean stress reported for the test data. Had the mean stress adjustment been performed properly, the resulting data scatter would have been greater.

- In its RAI, the staff noted that the assumption used for the EPRI standard deviation for the high-cycle end of the fatigue curves came from a general handbook recommendation as opposed to the standard deviation derived by Argonne National Laboratory (ANL) from its assessment of nuclear power plant carbon steel materials. In response to a staff question, the industry representatives indicated that a regression analysis of the existing carbon steel test data had not been performed to justify the

standard deviation assumption used in the EPRI study. The staff pointed out that regression analyses performed by EPRI in fatigue studies of socket welded components did not support the standard deviation assumption used in the EPRI study. The staff further indicated that additional fatigue data of actual components, such as butt welded joints, may also exist.

- The industry representatives agreed that an adjustment should be made to account for difference between smooth specimen test data and actual components, and that there should be a mean stress adjustment. The mean stress adjustment should use the same values of material yield and ultimate stress that were used in the ANL studies. The industry representatives indicated that the threshold assumptions did not impact the results, and that the revised study would be performed without any thresholds. The industry representatives also agreed to provide the revised fatigue usage factors for any components where the cycles or stresses were modified from those used in the original PNNL (Pacific Northwest National Laboratory) study.

- In its RAI, the staff noted that PNNL had modified the ANL fatigue crack initiation correlation to account for potential multiple initiation sites. PNNL had calibrated this modification using test data from 9-inch diameter vessel tests. The staff asked how the adjustment was applied in the EPRI study. The industry representatives indicated that there was no change to the adjustment that was used in the PNNL study. The staff asked whether the same adjustment was valid given the ANL correlation had been modified in the EPRI study.

Enclosed please find the meeting agenda (Enclosure 1), the list of meeting attendees (Enclosure 2), and the presentation made by industry representatives that was discussed during the meeting (Enclosure 3). The industry representative stated that they plan to provide their response within the next 30 days. A draft of this meeting summary was provided to the NEI to give them an opportunity to comment prior to being issued. If you have any questions concerning this proposed ISG, please contact Peter J. Kang, at (301) 415-2779.

Peter J. Kang

Peter J. Kang, Reactor Systems Engineer
License Renewal Section
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No.: 690

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/RA/
Peter J. Kang, Reactor Systems Engineer
License Renewal Section
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

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OFFICE	PM:RLEP:DRIP	LA:RLEP:DRIP	EMEB:DE	SC:RLEP:DRIP
NAME	PKang	M Jenkins	JFair	SWest
DATE	8/21/03	8/21/03	8/21/03	08/21/03

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NUCLEAR ENERGY INSTITUTE

Project No. 690

cc:

Mr. Joe Bartell
U.S. Department of Energy
NE-42
Washington, DC 20585

Ms. Christine S. Salembier
Commissioner
State Liaison Officer
Department of Public Service
112 State St., Drawer 20
Montipelier, VT 05620-2601

Mr. Fred Emerson
Nuclear Energy Institute
1776 I St., N.W., Suite 400
Washington, DC 20006-3708

Mr. Stephen T. Hale
Florida Power & Light Company
9760 S.W. 344 St.
Florida City, FL 33035

Mr. William Corbin
Virginia Electric & Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060

Mr. Frederick W. Polaski
Manager License Renewal
Exelon Corporation
200 Exelon Way
Kennett Square, PA 19348

George Wrobel
Manager, License Renewal
R.E. Ginna Nuclear Power Plant
1503 Lake Rd.
Ontario, NY 14519

Mr. David Lochbaum
Union of Concerned Scientists
1707 H St., NW, Suite 600
Washington, DC 20006-3919

Ronald B. Clary
Manager, Plant Life Extension
V.C. Summer Nuclear Station
Bradham Blvd.
P.O. Box 88
Jenkinsville, SC 29065

Mr. John B. Herman
Manager - Nuclear Licensing
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

Mr. Paul Gunter
Director of the Reactor Watchdog Project
Nuclear Information & Resource Service
1424 16th St., NW, Suite 404
Washington, DC 20036

Mr. Hugh Jackson
Public Citizen's Critical Mass Energy &
Environment Program
215 Pennsylvania Ave., SE
Washington, DC 20003

Mary Olson
Nuclear Information & Resource Service
Southeast Office
P.O. Box 7586
Asheville, NC 28802

Talmage B. Clements
Manager - License Renewal
Nuclear Engineering Services
CP&L
410 South Wilmington St.
Raleigh, NC 27602

Mr. Charles R. Pierce
Manager - License Renewal
Southern Nuclear Operating Company
P. O. Box 1295
Birmingham, AL 35201

Mr. Garry G. Young
Manager, License Renewal Services
1448 SR 333, N-GSB-45
Russellville, AR 72802

Meeting Agenda
License Renewal Meeting to
Discuss Staff's Request for Additional Information (RAI) on
EPRI Technical Report, "Material Reliability Program (MRP-74)" for
Fatigue Environmental Effects

Room O-14B6
July 24, 2003
(1:30 PM-3:30 PM)

1. Welcome/Introductions	10 minutes
2. Discussion of Enclosed RAI	90 minutes
3. Public comments	10 minutes
4. Summary	10 minutes

Enclosure 1

REQUEST FOR ADDITIONAL INFORMATION
FOR PROPOSED INTERIM STAFF GUIDANCE (ISG)
FOR FATIGUE ENVIRONMENTAL EFFECTS

1. The proposed ISG is based on re-evaluation of the carbon and low alloy steel components originally evaluated by Pacific Northwest National Laboratory (PNNL) and presented in NUREG/CR-6674, "Fatigue Analysis of Components for 60-Year Plant Life." This re-evaluation is presented in EPRI Report, "Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74)." EPRI claims that more realistic assumptions were used in the re-evaluation of these components and the use of these more realistic assumptions results in probabilities of crack initiation and leakage that are significantly less than indicated in NUREG/CR-6674. The most significant change made to the original study was in the standard deviation assumed for the endurance limit strain in the PNNL study. EPRI proposed to replace the standard deviation used in the PNNL study with a much smaller standard deviation. EPRI cites a typical value of fatigue data scatter proposed by Wirsching (Probabilistic Structural Mechanics Handbook, edited by C. Sundararajan, Chapman & Hall, New York, NY 1995, Chapter 7) as the basis for the change. This reference is general in nature and not directly applicable to carbon and low alloy steels used in nuclear power plants. The standard deviation for the endurance limit strain used in the PNNL study is based on a statistical evaluation of test data relevant to carbon and low alloy steels described in NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments and NUREG/CR-6717," "Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels." Provide the following additional information regarding the EPRI endurance limit strain and its standard deviation:

   a. The revised probabilistic fatigue curves do not appear consistent with the data for carbon and low alloy steels. For example, compare probabilistic curves developed using the EPRI assumption for the standard deviation of the endurance limit with the data presented in Figure 14 of Attachment 1 of the submittal.

   b. The study does not appear to adjust the endurance limit strain to account for the differences between smooth specimen data and actual components. The ANL correlation used by PNNL was developed to account for this difference. Provide the basis for not adjusting the endurance limit to account for the difference between the specimen data and actual components.

   c. The EPRI report indicates that a strain threshold was used in the evaluation but does show how the threshold was applied. The EPRI Report, page 3-11, references NUREG/CR-6717 for the strain threshold values used for the evaluation. As discussed in NUREG/CR-6717, the thresholds are strain levels at which environmental effects are considered moderate. These thresholds were proposed for use in the development of fatigue design curves. NUREG/CR-6717 also indicates that the threshold strain is approximately 20 percent higher than the fatigue limit (endurance limit) of the steel. Therefore, the threshold strain should be related to the endurance limit. Additionally, the proposed 0.07 percent threshold strain for the carbon and low alloy steel design curves has not been universally accepted at this time. For example, some fatigue researchers have

proposed using the endurance limit strain of 0.042 percent as the threshold value. As a consequence, the use of a fixed threshold strain in the probabilistic study is questionable. Explain how the strain threshold values were used in the evaluations presented in Chapter 4 of the EPRI report. Provide the results of the EPRI evaluations without using strain threshold values.

d. The strain thresholds are discussed on page 26 of NUREG/CR-6717. NUREG/CR-6717 indicates that after mean stress effects are taken into account, a threshold strain amplitude of 0.07 percent provides a 90 percent confidence level for both carbon and low alloy steels. As discussed previously, the threshold strain is approximately 20 percent higher than the endurance limit of the steel. Consequently, the 10 percent probabilistic fatigue curve should approach a strain amplitude of approximately 0.06 percent at 10E6 cycles. The 10 percent probability curve shown in Figure 3-11 of the EPRI report is not consistent with a strain of 0.06 percent. Discuss this discrepancy between Figure 3-11 of the EPRI report and the data assessment contained in NUREG/CR-6717.

2. The EPRI report, page 3-3, indicates that the ANL adjustment of ln(4), used to account for the differences between laboratory specimens and actual components, was included in the study in accordance with the discussion in the PNNL study. Section 4.7 of the PNNL study indicates that the ln(4) value was adjusted to account for the potential for multiple crack initiation sites. The PNNL study further indicates that the adjustment was calibrated against the data from the 9 inch diameter vessel tests described in the ANL report. Describe how this adjustment was applied in the EPRI study.

3. The EPRI report, page 3-11, provides a procedure to account for mean stress effects. Show how this procedure was implemented in the evaluations presented in Chapter 4 of the report. Discuss the consistency of the mean stress adjustment used in the Chapter 4 evaluations with the mean stress adjustment discussed in NUREG/CR-6717.

4. Several of the component evaluations presented in Chapter 4 of the EPRI report use stresses and cycle counts that are different than those used in the PNNL study. The changes affect the calculated environmental fatigue usage factors for these components. Provide the environmental fatigue usage factors based on the revised component stress and cycle assumptions. Discuss the actions that would be required by a license renewal applicant to address components with these usage factors.

5. The submittal references the evaluation of the component fatigue tests contained in EPRI Report MRP-49. The evaluation of the component fatigue test data is similar to the evaluation contained in EPRI Technical Report, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47)," Draft Revision G dated June 5, 2001. This report was submitted to the NRC by NEI letter dated July 31, 2001. The staff transmitted a request for additional information regarding the evaluation of the component fatigue tests by letter dated June 26, 2002. The staff has not received a response to its request for additional information. Indicate how the relevant June 26, 2002, staff comments have been addressed in the test data evaluation contained in EPRI Report MRP-49.

NRC Meeting Attendance List

MEETING WITH NUCLEAR ENERGY INSTITUTE (NEI)
TO DISCUSS STAFF'S REQUEST
FOR ADDITIONAL INFORMATION (RAI)
ON ENVIRONMENTAL ASSISTED FATIGUE

July 24, 2003

Name	Organization
PT. Kuo	NRC
K.S. West	NRC
J.R. Fair	NRC
O.K. Chopra	ANL
K.C. Chang	NRC
Y.Y. Liu	ANL
Paul Donavin	American Electric Power
Michael Guthrie	Progress Energy LR
Douglas Kalinousky	NRC
Makutesnarol Srinivasan	NRC
Gary Hammer	NRC
Jit Vora	NRC
Carol Moyer	NRC
Ram Subbaratnam	NRC
Peter J. Kang	NRC
Leslie Spain	Dominion Generation
Stan Rosinski	EPRI
Arthur Deardorff	Structural Integrity Assoc.(W/EPRI)
Fred Emerson	NEI
Mike Robinson	Duke Power
Bob Nickell	Consultant EPRI
J. Michael Davis	Duke Power
Fred Polaski	Exelon
Jim Riley	NEI
Gary Adkins	TVA
Eric Blocher	Parsons Power
Mark Ackerman	FENOC
William Stuard	Constellation Nuclear Services
T.J. Kim	NRC

Enclosure 2

Discussion of Response to RAI on ISG-11

EPRI Materials Reliability Program
Fatigue Issue Task Group

July 24, 2003
Washington, DC

EPRI

Question 1

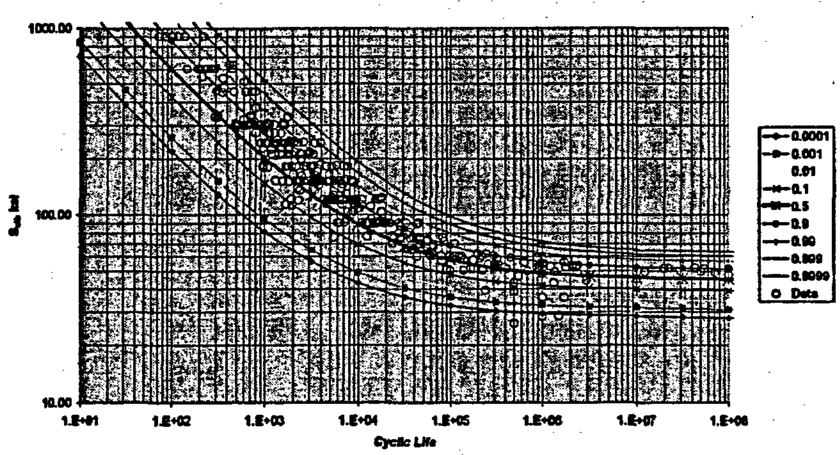
1. *The proposed ISG is based on re-evaluation of the carbon and low alloy steel components originally evaluated by Pacific Northwest National Laboratory (PNNL) and presented in NUREG/CR-6674, "Fatigue Analysis of Components for 60-Year Plant Life." This re-evaluation is presented in EPRI Report, "Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74)." EPRI claims that more realistic assumptions were used in the re-evaluation of these components and the use of these more realistic assumptions results in probabilities of crack initiation and leakage that are significantly less than indicated in NUREG/CR-6674. The most significant change made to the original study was in the standard deviation assumed for the endurance limit strain in the PNNL study. EPRI proposed to replace the standard deviation used in the PNNL study with a much smaller standard deviation. EPRI cites a typical value of fatigue data scatter proposed by Wirsching (Probabilistic Structural Mechanics Handbook, edited by C. Sundararajan, Chapman & Hall, New York, NY 1995, Chapter 7) as the basis for the change. This reference is general in nature and not directly applicable to carbon and low alloy steels used in nuclear power plants. The standard deviation for the endurance limit strain used in the PNNL study is based on a statistical evaluation of test data relevant to carbon and low alloy steels described in NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments and NUREG/CR-6717, "Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels." Provide the following additional information regarding the EPRI endurance limit strain and its standard deviation:*

EPRI

Question 1A

- Question
  - *The revised probabilistic fatigue curves do not appear consistent with the data for carbon and low alloy steels. For example, compare probabilistic curves developed using the EPRI assumption for the standard deviation of the endurance limit with the data presented in Figure 14 of Attachment 1 of the submittal.*
- Response
  - Most of data are above the mean curve
  - Only a few points below the mean curve
  - Detailed review of data shows low points resulted from high R-ratio testing
  - Correction for mean stress effects shows that observed low points are within scatter band for remainder of data
  - Therefore, the standard deviation chosen to represent the data is conservative

Original LAS Data (MRP-49) Plotted vs. MRP-74 Modified Equations



Evaluation of Low-Strain Amplitude LAS Air Data

- Effective S_{alt} corrected using modified Goodman approach

$$S_{alt,eff} = S_{alt}\frac{S_{uts}}{S_{uts} - S_{ys} + S_{alt}} \quad (S_{alt} < S_{ys})$$

- Used S_{uts} = *100 ksi* and S_{ys} = *70 ksi* (NUREG/CR-6335)

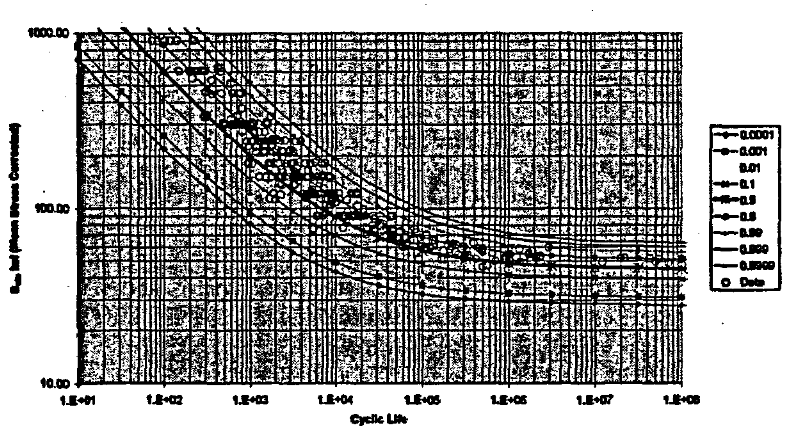
Data Source (MPR-49)	S_{alt}, ksi	Cyclic Life	R-ratio	Effective S_{alt}, ksi
Endou	26.1	500,000	0.14	46.5
Kou	28.5	1,000,000	0.19	48.7
Kou	28.8	1,500,000	0.05	48.0
Kou	36.1	1,010,000	0.19	54.6
Kou	36.1	1,700,000	0.05	54.6
Endou	38.8	500,000	0.14	56.4
Kou	39.4	242,000	0.19	56.8
Endou	41.5	496,900	0.14	58.0

Note: Cyclic Life stated to be 25% Area Reduction; all material A508 Class 3-4

EPRI

Corrected LAS S_{alt} Plotted vs. MRP-74 Modified Equations

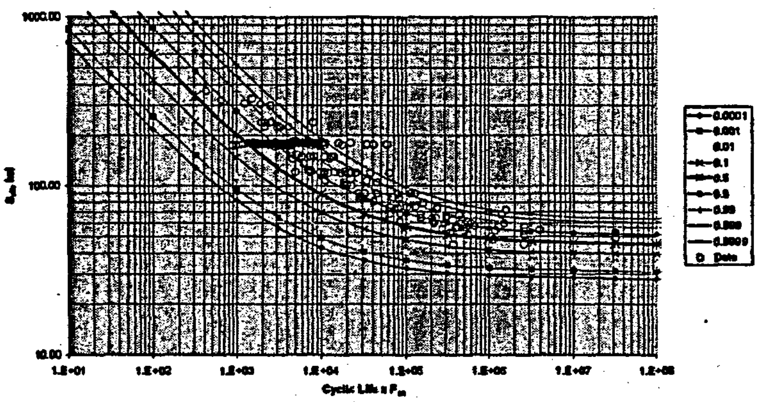
MRP-49 Air Data (corrected)



EPRI

Evaluation of Other Data

MRP-49 Simulated BWR Data Show Similar Trend



- **Review of additional plots (NUREG/CR-6335, -6583, -6717) show no instances of data where endurance limit data fall significantly below mean curves for carbon or low-alloy steel data**

Question 1B – Adjustment for Difference Between Specimens and Actual Components

- **Question**
  - ***The study does not appear to adjust the endurance limit strain to account for the differences between smooth specimen data and actual components. The ANL correlation used by PNNL was developed to account for this difference. Provide the basis for not adjusting the endurance limit to account for the difference between the specimen data and actual components.***
- **Response**
  - **Explanation for modified fatigue curves from NUREG/CR-6335**
    - **Size/geometry effect............... 1.4 on cycles / 1.25 on strain**
    - **Surface finish......................... 3.0 on cycles / 1.3 on strain**
    - **Factors on strain not judged to be cumulative**
    - **Effects of surface finish/size/geometry would be bounded by lower 5% quantile curve in probabilistic model**
  - **The factor on strain was not originally included in MRP-74; the factor of 4 on cycles was included; no effect on low-cycle end of fatigue curve**
  - **The approach for strain correction in NUREG/CR-6335 is overly conservative at low quantiles and is non-conservative relative to data scatter in the positive direction**
  - **Recent evaluation of the factor on strain demonstrates that overall conclusions do not change**

Question 1C - Thresholds

- Question
  - *The EPRI report indicates that a strain threshold was used in the evaluation but does show how the threshold was applied. The EPRI Report, page 3-11, references NUREG/CR-6717 for the strain threshold values used for the evaluation. As discussed in NUREG/CR-6717, the thresholds are strain levels at which environmental effects are considered moderate. These thresholds were proposed for use in the development of fatigue design curves. NUREG/CR-6717 also indicates that the threshold strain is approximately 20% higher than the fatigue limit (endurance limit) of the steel. Therefore, the threshold strain should be related to the endurance limit. Additionally, the proposed 0.07% threshold strain for the carbon and low alloy steel design curves has not been universally accepted at this time. For example, some fatigue researchers have proposed using the endurance limit strain of 0.042% as the threshold value. As a consequence, the use of a fixed threshold strain in the probabilistic study is questionable. Explain how the strain threshold values were used in the evaluations presented in Chapter 4 of the EPRI report. Provide the results of the EPRI evaluations without using strain threshold values.*
- Response
  - Approach is described in Section 3.2 of MRP-74 (per NUREG/CR-6717)
    - Used ramp between $\varepsilon = 0.07\%$ and $\varepsilon = 0.08\%$ to gradually apply environmental effects
  - Sensitivity studies showed that effect of applying thresholds is not significant
    - See Figures 4-5 and 4-6 of MRP-74
  - Major contributors to fatigue/crack growth are due to load-set pairs where thresholds do not apply (S_{alt} > threshold)

Questions 1D and 3 – Mean Stress Effect

- Question 1D
  - *The strain thresholds are discussed on page 26 of NUREG/CR-6717. NUREG/CR-6717 indicates that, after mean stress effects are taken into account, a threshold strain amplitude of 0.07% provides a 90% confidence level for both carbon and low alloy steels. As discussed previously, the threshold strain is approximately 20% higher than the endurance limit of the steel. Consequently, the 10 percent probabilistic fatigue curve should approach a strain amplitude of approximately 0.06% at 10E6 cycles. The 10 percent probability curve shown in Figure 3-11 of the EPRI report is not consistent with a strain of 0.06%. Discuss this discrepancy between Figure 3-11 of the EPRI report and the data assessment contained in NUREG/CR-6717.*
- Question 3
  - *The EPRI report, page 3-11, provides a procedure to account for mean stress effects. Show how this procedure was implemented in the evaluations presented in Chapter 4 of the report. Discuss the consistency of the mean stress adjustment used in the Chapter 4 evaluations with the mean stress adjustment discussed in NUREG/CR-6717.*

Mean Stress Adjustment

- Response
  - Approach used in MRP-74 provides a more rigorous approach for mean stress adjustment
- In NUREG/CR-6335 and –6717, a modified Goodman approach was used to adjust fatigue curves down; implemented in modified curve fit equations

$$S'_a = S_a\left(\frac{S_{uts} - S_{ys}}{S_{uts} - S_a}\right) \quad for\ S_a < S_{ys}$$

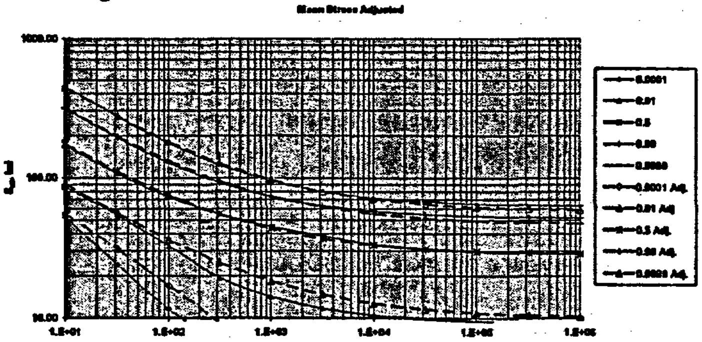
- MRP-74 stresses adjusted upward to enter basic data curve fit equation

$$S'_{alt} = S_{alt}\left(\frac{S_{uts}}{S_{uts} - S_{ys} + S_{alt}}\right) \quad for\ S_{alt} < S_{ys}$$

- Both arrive at essentially the same results as shown on the following pages for S_{uts} = 100 ksi and S_{ys} = 70 ksi

Mean Stress Effects

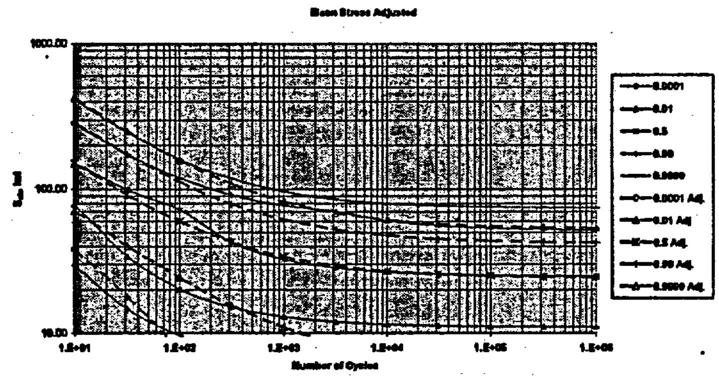
- Comparison of effective fatigue curves using NUREG/CR-6635 data fits and modified fatigue curves for carbon steel



  - Environmental effects at 550°F. Solid lines are mean-stress corrected data curves and dotted lines are NUREG/CR-6335 Equation 18. Both use NUREG/CR-6335 endurance limit data scatter
- Approach used in MRP-74 is more conservative at low quantiles – correction made to modified fatigue curve not to mean

Mean Stress Effects

- Comparison of effective fatigue curves using NUREG/CR-6635 data fits and modified fatigue curves for low alloy steel



  - Environmental effects at 550°F. Solid lines are mean-stress corrected data curves and dotted lines are NUREG/CR-6335 Equation 18. Both use NUREG/CR-6335 endurance limit data scatter.

- Curves are comparable

EPRI

Mean Stress Effects

- NUREG/CR-6335 mean stress effects are based on bounding S_{ys} and S_{uts} values consistent with ASME Code

Material	S_{ys}	S_{uts}
CS	40 ksi	80 ksi
LAS	70 ksi	100 ksi

- MRP-74 mean stress effects are based on lower values

Material	S_{ys}	S_{uts}
CS	28.3 ksi	60 ksi
LAS	30.8 ksi	70 ksi

- Recent evaluation of modified mean stress effects (same values as NUREG/CR-6335) demonstrates that overall conclusions do not change

EPRI

Mean Stress Effects

- **Minimum Code yield and ultimate tensile strengths for materials evaluated in NUREG/CR-6260**

Material	Type	UTS, ksi	YS, ksi			
			70°F	400°F	500°F	600°F
A-508 Cl 2	LAS	80	50	44.5	43.2	42.0
A-182 F1	LAS	70	40	33.7	32.5	31.4
A-336 LAS	LAS	80	50	44.5	43.2	42.0
A-508 Cl1	CS	70	36	30.8	29.1	26.6
A-336 Cl1	CS	70	40	33.7	32.5	31.4
A-333 Gr 6	CS	60	35	30.0	28.3	25.9
A-106 Gr B	CS	60	33	30.0	28.3	25.9

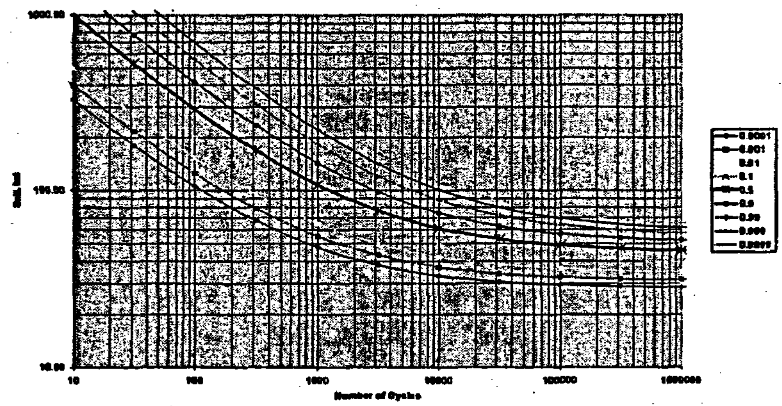
- **Evaluation shows that reduction of S_{ys} on the order of 15% is justified at 550°F for A-508 Cl2 rather than using bounding Code values**
  - **similar for other materials**
  - **could also be a random value**
- **This approach, if applied, would remove additional conservatism**

Further Consideration on PFM Fatigue Curves

- Fatigue curves should be based on fatigue data variability
  - No corrections for mean stress or surface finish/size/etc.
  - Include a coefficient of variation representative of data scatter observed in testing
    - Cycle variation controls for mid-range stress/strain region
    - Stress variation controls for endurance limit region
- Make corrections to input stresses and data fatigue curves to account for other factors
  - Mean stress effects
  - Surface/size/surface finish effects

Example – Typical fatigue curve

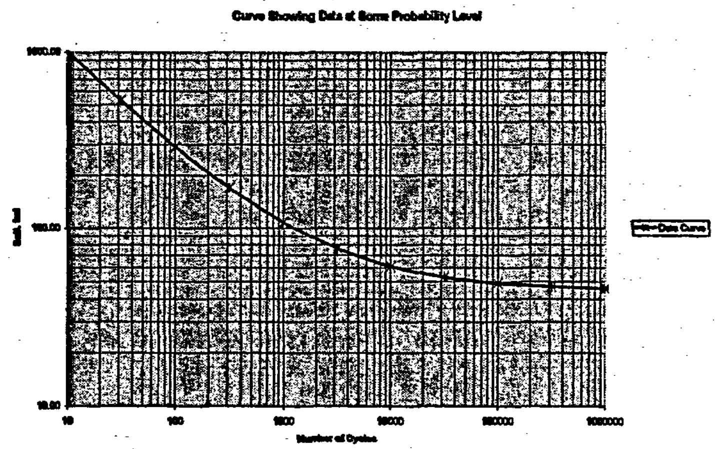
- From fatigue data curve equation, use Monte Carlo sample to choose a specific quantile for each sample analysis – typical quantiles shown below



EPRI

Example – Typical fatigue curve

- Following represents fatigue data curve at a PFM quantile equal to 0.5 (as example)



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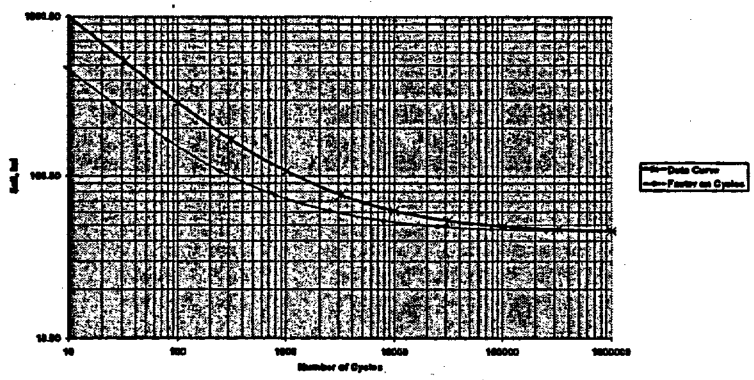
Example – Typical fatigue curve

- Determine shift of curve (in cycle direction) due to surface finish, size, roughness etc. – could be a PFM variable
  - (like factor of ln(4) in NUREG/CR-6335)



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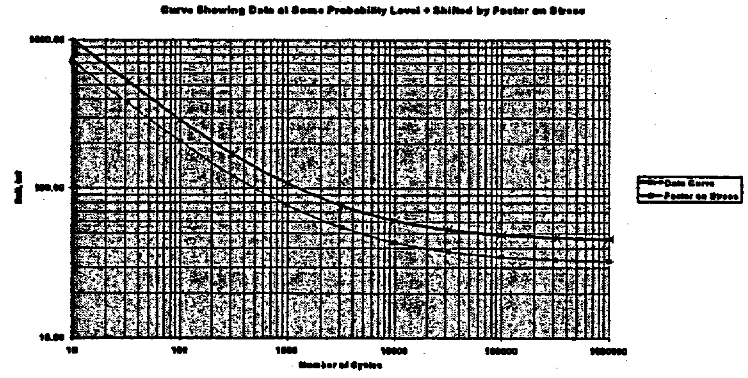
Example – Typical fatigue curve

- Determine shift of curve (in stress direction) due to surface finish, size, roughness, etc. – could be a PFM variable
  - Allowable number of cycles determined by multiplying stress amplitude by shift factor (e.g., 1.3)



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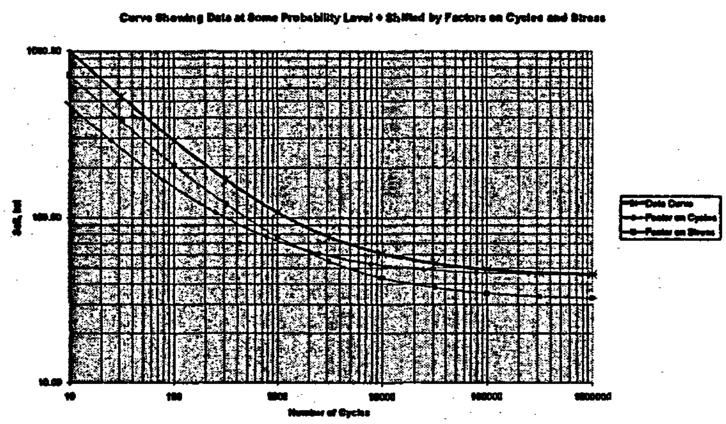
Example – Typical fatigue curve

- Typically, factor on cycles controlling at low cycles and factor on stress controlling on high cycles



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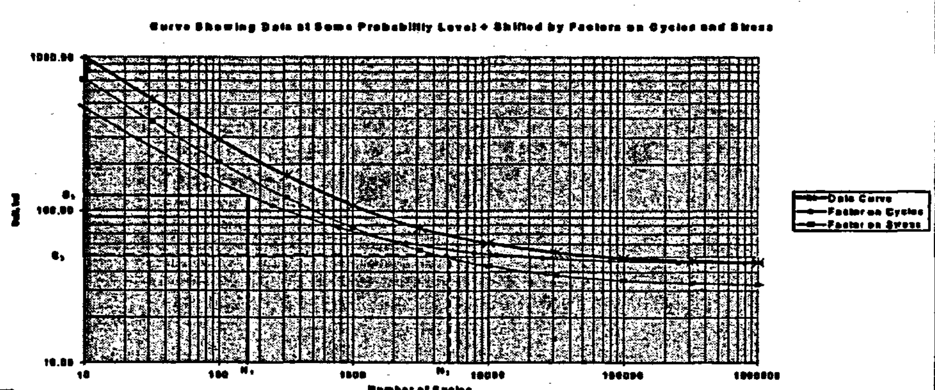
Example – Determine Number of Allowable Cycles

- Adjust stress for mean stress effects (as previously addressed) and determine number of allowable cycles
  - Two examples shown below



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Question 2 – Adjustment of Cycles by Factor of 4

- **Question**
  - *The EPRI report, page 3-3, indicates that the ANL adjustment of ln(4), used to account for the differences between laboratory specimens and actual components, was included in the study in accordance with the discussion in the PNNL study. Section 4.7 of the PNNL study indicates that the ln(4) value was adjusted to account for the potential for multiple crack initiation sites. The PNNL study further indicates that the adjustment was calibrated against the data from the 9-inch diameter vessel tests described in the ANL report. Describe how this adjustment was applied in the EPRI study.*
- **Response**
  - Adjustment was performed using exactly the same approach as for the PNNL study

EPRI

Question 4 – Cycle Counts

- **Question**
  - *Several of the component evaluations presented in Chapter 4 of the EPRI report use stresses and cycle counts that are different than those used in the PNNL study. The changes affect the calculated environmental fatigue usage factors for these components. Provide the environmental fatigue usage factors based on the revised component stress and cycle assumptions. Discuss the actions that would be required by a license renewal applicant to address components with these usage factors.*
- **Response**
  - Reductions in cycles are consistent with typical plants evaluated in NUREG/CR-6260 reflecting the way plants really operate
  - In MRP-74, for older vintage CE plant RPV outlet nozzles, case with expected cycles was presented only to show extremely low leakage probability with typical cycles
  - For B&W plant outlet nozzle, loading/unloading reduced from 48,000 cycles (3 times per day) to 2,080 (once per week to conservatively bound plant operation)
  - Inclusion of daily loading/unloading was conservatively included in original plant analyses for many components – not representative of how plants really operate

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Cycle Counts

- Following tables show NUREG/CR-6260 reactor vessel outlet nozzle usage calculations
  - Design basis
  - Plant loading/unloading conservatively reduced to once per week
  - Anticipated plant cycles
- Should be no requirement by license renewal applicant to address components with reduced cycles considered
  - Cycle reduction in NUREG/CR-6260 typical of plant operation
  - US nuclear plants are base-loaded and do not cycle daily or even weekly

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Cycle Counts

CUF results for newer-vintage CE plant reactor vessel outlet nozzle using NUREG/CR-5999 interim fatigue curve (from Table 5-4 of NUREG/CR-6260)

Load Pair	S_{alt} (adjusted)	N	n	u	n[1]	u[1]	n[2]	u[2]
Cooldown/plant load	61.51	1458	500	0.343	500	0.343	90	0.062
Leak test/plant unload	60.85	1534	200	0.130	200	0.130	200	0.130
Heatup/plant load	39.24	4263	500	0.117	500	0.117	90	0.021
Plant load/unload	20.84	58406	13800	0.236	1880	0.032	1880	0.032
Plant unload/upset	19.82	73899	480	0.007	480*	0.007	480*	0.006
Plant unload/OBE	14.32	265677	200	0.001	200*	0.001	200*	0.001
Plant unload/step load	12.71	729360	520	0.001	520*	0.001	520*	0.001
			CUF	0.835	CUF[1]	0.631	CUF[2]	0.253

Notes:
1. Plant loading/unloading reduced to 2080 (once per week for 40 years) – asterisked cycles conservatively retained
2. Anticipated cycles from Table 5-1of NUREG/CR-6260, plus Note 1 above

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Cycle Counts

CUF results for older-vintage CE plant reactor vessel outlet nozzle using NUREG/CR-5999 Interim fatigue curve with effect of modified cycles (from Table 5-30 of NUREG/CR-6260)

Load Pair	S_{alt} (adjusted)	N	n	u	n[1]	u[1]	n[2]	u[2]
Loss of secondary pressure/hydrotest	74.46	508	5	0.010	5	0.010	0	0
Hydrotest A/hydrotest B	38.46	4610	5	0.001	5	0.001	2	0.0004
Heatup/loss of load	32.41	8518	40	0.005	40	0.005	40	0.005
Heatup/loss of flow	31.73	9207	40	0.004	40	0.004	40	0.004
Heatup/cooldown	31.53	9423	420	0.045	420	0.045	21	0.0023
Cooldown/plant loading	29.70	11737	80	0.007	80	0.007	80	0.007
Reactor trip/plant loading	25.83	21348	400	0.019	400	0.019	92	0.0044
Plant loading/plant unloading* *erroneously listed as reactor trip in NUREG/CR-6260	23.79	31396	14520	0.462	1600	0.051	30	0.0095
			CUF	0.554	CUF[1]	0.143	CUF[2]	0.033

Notes:
1. Plant loading/unloading reduced to 2080 (once per week for 40 years)
2. Anticipated cycles from Table 5-27 of NUREG/CR-6260

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Cycle Counts

CUF results for B&W reactor vessel outlet nozzle using NUREG/CR-5999 Interim fatigue curve (from Table 5-51 of NUREG/CR-6260)

Load Pair	S_{alt} (adjusted)	N	n	u	n[1]	u[1]	n[2]	u[2]
Heatup/cooldown	37.96	4853	240	0.049	240	0.049	155	0.032
Step load/reactor trip	22.15	43885	480	0.011	4890	0.011	214	0.005
Plant loading/unloading	17.24	138590	48000	0.346	2080	0.015	2080	0.015
All other	16.69	155130	9850	0.063	9850	0.063	9850	0.063
			CUF	0.459	CUF[1]	0.138	CUF[2]	0.115

Notes:
1. Plant loading/unloading reduced to 2080 (once per week for 40 years)
2. Anticipated cycles from Table 5-83 of NUREG/CR-6260, plus Note 1 above

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Cycle Counts

CUF results for older-vintage Westinghouse reactor vessel outlet nozzle using NUREG/CR-5999 interim fatigue curve (from Table 5-87 of NUREG/CR-6260) – reductions for reactor vessel inlet nozzle similar

Load Pair	S_{alt} (adjusted)	N	N	U	n^1	u^1	n^2	U^2
Heatup/cooldown	17.22	139150	350	0.003	350	0.003	172	0.0012
Plant loading/unloading	18.89	92576	14100	0.152	2080	0.023	734	0.0074
OBE A/OBE B	20.94	57109	400	0.007	400	0.007	0	0
Combination	32.78	8179	2760	0.337	2760	0.337	2760	0.3374
			CUF	0.499	CUF[1]	0.370	CUF[2]	0.357

Notes:
1. Plant loading/unloading reduced to 2080 (once per week for 40 years)
2. Use of actual plant cycles from Table 5-83 of NUREG/CR-6260 (except combination)

Cycle Counts

CUF results for newer-vintage Westinghouse reactor vessel outlet nozzle using NUREG/CR-5999 interim fatigue curve (from Table 5-67 of NUREG/CR-6260)

Load Pair[1]	S_{alt} (adjusted)	N	n	u	n^2	u^2
	48.85	1825	60	0.044	60	0.044
	45.40	2407	10	0.004	10	0.004
	44.34	2642	20	0.008	20	0.008
	39.94	3977	20	0.005	20	0.005
	34.39	7012	70	0.010	70	0.010
	29.31	12320	130	0.011	130	0.011
	28.30	14013	160	0.011	150	0.011
	27.09	17075	60	0.003	50	0.003
	26.99	17374	30	0.002	30	0.002
	21.37	51918	40	0.001	40	0.001
	20.20	67603	1930	0.029	1930	0.029
	20.20	67603	2000	0.030	2000	0.030
	20.13	68712	9270	0.135	2080	0.030
	18.85	93499	60	0.001	60	0.001
	18.44	103654	230	0.002	230	0.002
	18.35	106059	10	0.000	10	0.000
	18.05	114581	80	0.001	80	0.001
	17.84	127614	160	0.001	160	0.001
	17.84	127614	26400	0.207	2080	0.016
	17.05	144035	2000	0.014	2000	0.014
	16.39	165234	400	0.002	400	0.002
	15.89	180237	13200	0.073	2080	0.012
	15.37	20-7136	13200	0.064	2080	0.010
	14.90	231047	80	0.000	80	0.000
	14.84	234349	80	0.000	80	0.000
	14.70	242295	70	0.000	70	0.000
			CUF	0.858	CUF[1]	0.247

Notes:
1. Licensee's load pairs were smeared and very difficult to read, so not provided in NUREG/CR-6260
2. High cycle transients conservatively assumed to be reduced to once per week (2080 cycles)

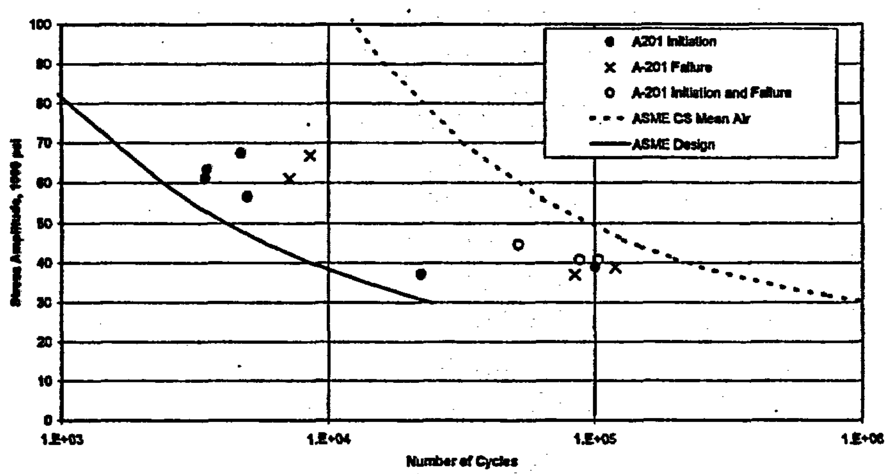
Question 5 – Evaluation of Component Fatigue Tests

- Question
  - *The submittal references the evaluation of the component fatigue tests contained in EPRI Report MRP-49. The evaluation of the component fatigue test data is similar to the evaluation contained in EPRI Technical Report, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47)," Draft Revision G dated June 5, 2001. This report was submitted to the NRC by NEI letter dated July 31, 2001. The staff transmitted a request for additional information regarding the evaluation of the component fatigue tests by letter dated June 26, 2002. The staff has not received a response to its request for additional information. Indicate how the relevant June 26, 2002, staff comments have been addressed in the test data evaluation contained in EPRI Report MRP-49.*
- MRP-47 RAI on Evaluation of Component Fatigue Tests
  - *The evaluation of component tests did not include the data from the Southwest Research Institute tests on vessels that is shown on page 4 of NUREG/CR-6583. This test data does not appear to support conclusion regarding the maximum effect of size and surface finish on fatigue life. Explain why this data was omitted from the evaluation. Provide an assessment of the Southwest Research Institute using the same method used to assess the KWU tube tests and the General Electric pipe tests.*

Response to Question 5

- Appendix C of MRP-47 evaluated preliminary component-scale fatigue data with at least one surface in contact with oxygenated water at temperature
- MRP-49 included component-scale fatigue data with at least one surface in contact with oxygenated water at room temperature – the same PVRC/SWRI carbon and low-alloy steel pressure vessel fatigue data shown on page 4 of NUREG/CR-6583
- The evaluation process for these pressure vessel fatigue test data was identical to the process used to evaluate the KWU 180-degree bend fatigue tests and the General Electric pipe tests. The results for all three sets of component fatigue tests are completely consistent

Results From the PVRC Fatigue Testing of Full Size Carbon Steel Pressure Vessels



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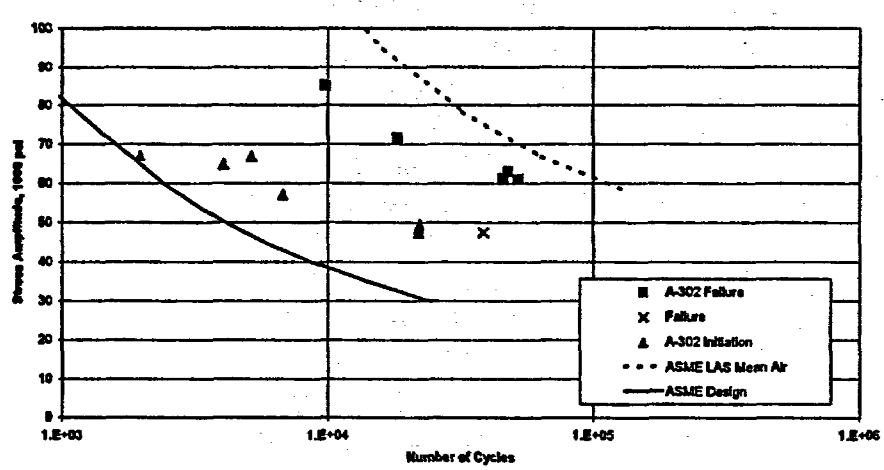
Results From the PVRC Fatigue Testing of Full Size Low-Alloy Steel Pressure Vessels



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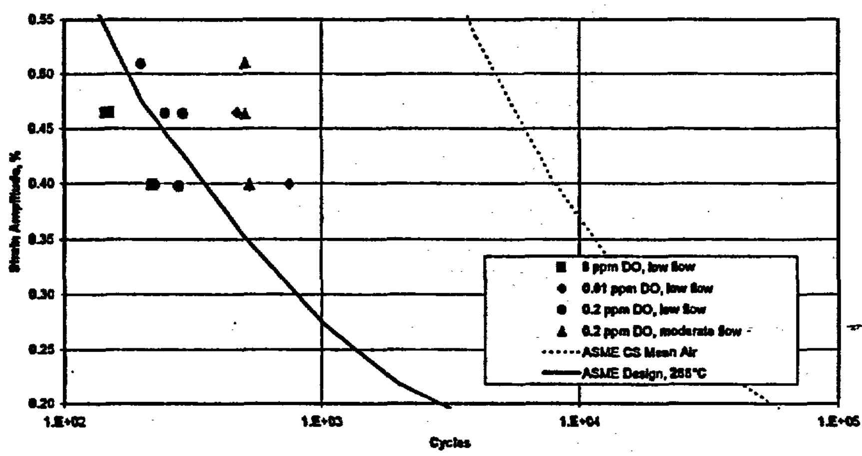
KWU Component Scale Test Results for Carbon Steel



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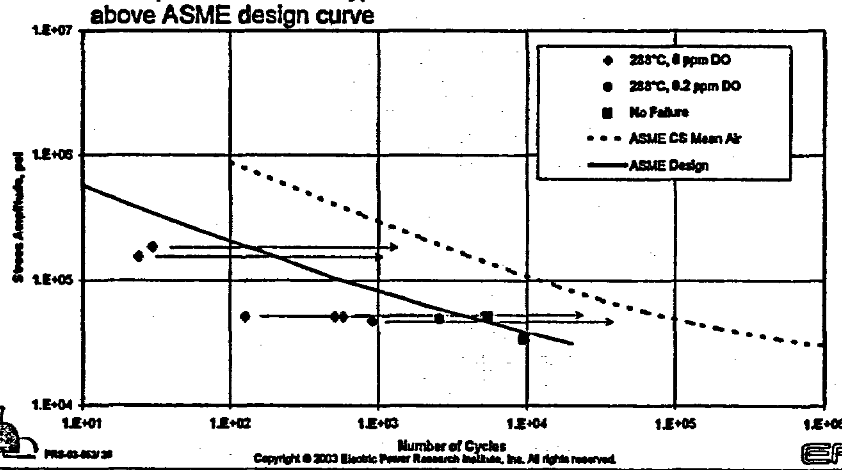
Results From GE Fatigue Testing of Butt-Welded Pipe Under Simulated BWR Conditions

- Tests performed on sample with 11 welds *in series*
  - Only first failure reported; remaining welds exhibited higher fatigue life
  - Tests performed under typical BWR environmental conditions failed at or above ASME design curve



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Response to Question 5 (continued)

- Results from the PVRC tests were reported in MRP-49 (Figures 2-35 and 2-36). While some of the crack initiation results approach the ASME Code fatigue design curve, none of the cycles to crack initiation were less than the design curve. Since the intent of the ASME Code fatigue design curves is to predict the mean line for crack initiation in actual vessels, these tests demonstrate that the margins used by the ASME in developing the design curves are conservative, even with exposure of vessel inner surfaces to oxygenated water.
- Figure 2 from NUREG/CR-6583 shows essentially the same data, but does not show the comparison between the ASME design curve and air curve, so that the reader may have difficulty determining whether the data points are located appropriately or not.

Response to Question 5 (continued)

- NUREG/CR-6583 states on page 4 that "These results demonstrate that the current Code design curves do not necessarily guarantee any margin of safety." However, the intent of the ASME Code fatigue design curve is to predict the mean line for crack initiation in actual vessels, and these data support that premise.
- The data points are consistent, based on a combination of surface roughness, component size effect, and environmental effects. None of the data compromise the ASME Code design curve in spite of the effects of surface roughness, component size effect, and water environment effects, provided that flow rate and very high levels of dissolved oxygen are taken into consideration.
- Even considering data not representative of normal plant operation, MRP-74 demonstrates that the risk of component leakage is acceptably low.

Conclusions

- Extensive review of available laboratory and component/structural data generated under simulated reactor water environment conditions suggests behavior consistent with margins in ASME Code design curve for C/LAS
- Results from MRP-74 demonstrate that explicit consideration of EF in fatigue aging management programs is not necessary during license renewal period
  - Analysis included potential effects of environmental fatigue
  - Insignificant contributor to core damage frequencies
  - Insignificant increase in predicted leakage probabilities
- All C/LAS fatigue locations can continue to rely on existing plant programs to track component fatigue usage through the license renewal period and remain in compliance with all NRC regulatory requirements

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Schedule

- To be discussed

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