Aging Management of Environmental Fatigue for Carbon/Low-Alloy Steels

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Enclosure 3

Objectives

- Present industry basis for resolving environmental fatigue issue for carbon and low-alloy steel components during license renewal
- Reach agreement with NRC management on a process for NRC review of industry basis
- Discuss aging management of fatigue for carbon and low-alloy steel during license renewal
- Establish post-meeting actions

Discussion Outline

- **Background**
- **Aging management for fatigue of carbon and low-alloy steel components during license renewal**
- **Supporting presentations**
  - Re-evaluation of NUREG/CR-6674
  - Review of laboratory and component/structural fatigue data
- **Conclusions**
- **Actions**

Background

- **Significant industry activities over the past decade to investigate the effects of reactor water environment on fatigue life of metal components.**
- **Since mid-2000 the effort has been coordinated under the EPRI MRP Fatigue Issue Task Group (ITG):**
  - Focal point for industry technical efforts
  - Technically represents NEI License Renewal Working Group
  - Provides the technical interface with NRC staff
  - Provides guidance to license renewal applicants
  - Provides results to ASME Code for consideration
- **MRP environmental fatigue program structured to address near-term industry needs and resolve long-term technical issues**

Issue Definition

- NRC Staff and industry have yet to reach consensus on the need to explicitly account for the effects of reactor water environment as part of an overall fatigue management program during the license renewal period

Background

- **Environmental fatigue found by NRC to be risk insignificant and no generic regulatory action required for:**
  - Current 40-year plant operating life (SECY-95-245)
    - Most high fatigue locations could be excluded with more detailed analyses
    - Fatigue failure of piping is an insignificant contributor to core melt frequency

Background

- **Environmental fatigue found by NRC to be risk insignificant and no generic regulatory action required for:**
  - 60-year operating life (Thadani GSI-190 closeout memorandum to Travers, December 26, 1999)
    - ALWR
      - ✓ Sufficient conservatism in the fatigue analyses performed for the generic 60-year ALWR life to account for environmental effects.
    - License renewal period of existing plants (40-60 years)
      - ✓ NUREG/CR-6674 evaluated effect of environmental fatigue on overall risk through 60 years of operation
      - ✓ Concluded that environmental fatigue not a risk-significant issue for 20 years of additional operation
      - ✓ Predicted increase in frequency of pipe leakage in 40-60 year period

Current Status

- Because of increased leakage potential, the closeout of GSI-190 requires license renewal applicants to address environmental fatigue in aging management programs.
- Significant resources expended by industry and NRC to address this issue
- NRC Staff and industry have yet to reach consensus on the need to explicitly account for the effects of reactor water environment as part of an overall fatigue management program during the license renewal period

MRP Program Activities

- **Near-term guidance to address environmental fatigue in a license renewal application (MRP-47 published October 2001)**
  - Previous applicants utilized different approaches
  - Consistent application of previous approaches desired to simplify process
  - RAIs received June 26, 2002; responses under development
  - MRP-47 assumes consideration of environmental fatigue is necessary (until results of long-term efforts are known)
- **Long-term activities to address identified technical issues**
  - Assess relevance of previous work on environmental fatigue to current plant operating conditions
  - Determine scope of aging management program necessary to address environmental fatigue in license renewal.
  - Results of long-term activities may necessitate change in near-term guidance document (MRP-47).

Ongoing Long-Term MRP Activities

- **Re-evaluate NUREG/CR-6674 risk study and conclusions**
  - Review assumptions and inputs to the original risk study for conservatism and applicability
  - Review basis for determining significance of predicted component leakage
  - Re-evaluate components for changes in leakage probabilities using revised assumptions and other inputs
- **Review and evaluate available environmental fatigue data**
  - Reconcile available laboratory data with structural/component test results and plant operating experience (MRP-49 published in December 2001)

Current Aging Management Approach

- **Plants rely on existing programs to effectively manage fatigue**
  - Compliance with current fatigue licensing basis through:
    - Cycle counting and comparison to design limits
    - Fatigue monitoring to count and categorize thermal cycles
    - Re-analysis, if necessary, to account for actual cycles and transient severity
  - Structural integrity of fatigue-sensitive locations
    - Formal inservice examination requirements provided in each plant's inservice inspection programs
    - Augmented inspection/evaluation, if necessary, based on plant operating experience and/or regulatory enforcement actions
    - Inspection scope and frequency expanded if flaws detected
    - Risk-informed considerations now developed
- **The industry believes this approach is adequate for license renewal period**

Basis for Continuing Current Aging Management Approach During License Renewal

- **Results from MRP long-term activities indicate no need for any formal consideration of environmental fatigue for carbon and low-alloy steel components**
  - Results of NUREG/CR-6674 risk study re-analysis (MRP-74)
    - Fatigue failure is an even less significant contributor to increases in core damage frequencies ($<10^{-10}$) and is well below the NRC threshold (10^{-6}) for being risk significant
    - Several orders of magnitude reduction in crack initiation and leakage probabilities
    - Predicted 60-year leakage probabilities are not significant and are below previously NRC accepted leakage probabilities at 40 years in NUREG/CR-6674
      - ✓ Maximum 40-year leakage probability from NUREG/CR-6674
        - ✓ 0.41
      - ✓ Maximum 60-year leakage probability from MRP re-evaluation
        - ✓ 0.0014

Basis for Continuing Current Aging Management Approach During License Renewal

- **Results from MRP long-term activities indicate no need for any formal consideration of environmental fatigue for carbon and low-alloy steel components**
  - MRP evaluation of available laboratory, component, and structural data generated under simulated reactor water environmental conditions showed behavior consistent with margins in ASME Code fatigue design curve
    - Flow rate identified as critical variable, but not simulated in typical laboratory environmental fatigue tests
    - Component/structural tests are more representative of actual plant operating conditions (oxygen content, flow rate, size effects, surface finish)
  - Existing fatigue design process is sufficiently conservative to account for a potential environmental effect
    - ASME Code methods
    - Design transient definition, number of cycles and severity

Conclusions

- MRP evaluation concludes consideration of environmental fatigue effects for carbon and low-alloy steel components, as stipulated in GSI-190 closeout memorandum, is not warranted
- All carbon/low-alloy steel 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
- MRP evaluation of austenitic stainless steel locations is continuing

RE-EVALUATION OF RESULTS FROM NUREG/CR-6674 "FATIGUE ANALYSIS OF COMPONENTS FOR 60-YEAR PLANT LIFE" FOR FERRITIC STEEL COMPONENTS

Presentation to NRC

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September 18, 2002

Enclosure 4

OUTLINE

- **Review NUREG/CR-6260 fatigue analysis**
  - Application of NUREG/CR-5999 "Interim Fatigue Curves to Selected Nuclear Power Plant Components" - published March 1995
- **Review NUREG/CR-6674 probabilistic evaluation**
  - Fatigue Analysis of Components for 60-Year Plant Life - published June 2000
- **Present results of re-evaluation for carbon and low-alloy components**
- **Provide conclusions that core damage frequencies and leakage probabilities much less than previously documented**

BACKGROUND - NUREG/CR-6260

- **NUREG/CR-5999 (ANL 1993) proposed modified fatigue curves to consider environmental effects for carbon, low alloy and austenitic stainless steels**
- **Idaho National Engineering Laboratories (INEL) contracted by NRC**
  - Obtained stress analyses for six representative components for seven older/newer plant types
  - Performed analysis for some piping where design was per ANSI B31.1
  - Assessed effect of modified curves for 40 years
  - Where CUF high removed conservatism/considered actual cycles
  - Projected results to 60 years (multiplier of 1.5 x 40-year CUF)
- **Could not explicitly show that for all components CUF < 1.0**
  - Stated that detailed transient monitoring or refined analysis could be used to demonstrate CUF < 1.0

BACKGROUND - NUREG/CR-6260

Results for ferritic components

VENDOR/VINTAGE	LOCATION	40 YR DESIGN CUF	NUREG/CR-5999 CUF		
			DESIGN	CONSERVATISMS REMOVED AND/OR EXPECTED CYCLES	60 YR EXTRAPOLATION
CE NEW	RPV LOWER HEAD/SHELL	0.007	0.014	-	0.021
CE NEW	INLET NOZZLE	0.182	0.475	-	0.712
CE NEW	OUTLET NOZZLE	0.377	0.835	0.472	0.708
CE NEW	CHARGING NOZZLE	0.05	0.104	-	0.156
CE NEW	SAFETY INJECTION	0.898	2.101	0.457	0.686
CE OLD	RPV LOWER HEAD/SHELL	0.008	0.013	-	0.02
CE OLD	INLET NOZZLE	0.073	0.172	-	0.258
CE OLD	OUTLET NOZZLE	0.284	0.554	-	0.831
B&W	RPV LOWER HEAD/SKIRT	0.12	0.223	-	0.335
B&W	OUTLET NOZZLE	0.9	2.148	0.469	0.704
B&W	HOT LEG SURGE NOZZLE	0.592	1.092	0.47	0.705
B&W	CORE FLOOD NOZZLE	0.345	0.632	-	0.948
W NEW	RPV LOWERHEAD	0.012	0.018	-	0.027
W NEW	INLET NOZZLE	0.11	0.29	-	0.435

BACKGROUND- NUREG/CR-6260

Results for ferritic components (continued)

VENDOR/VINTAGE	LOCATION	40 YR DESIGN CUF	NUREG/CR-5999 CUF		
			DESIGN	CONSERVATISMS REMOVED AND/OR EXPECTED CYCLES	60 YR EXTRAPOLATION
W NEW	OUTLET NOZZLE	0.398	0.658	-	0.987
W OLD	RPV AT SUPPORT	0.29	0.891	-	1.33
W OLD	INLET NOZZLE	0.28	0.496	-	0.744
W OLD	OUTLET NOZZLE	0.431	1.161	0.347	0.52
GE NEW	RPV LOWER HEAD	0.2	11.702	0.628	0.942
GE NEW	FW NOZZLE SAFE-END	0.301	1.73	1.881	2.822
GE NEW	CORE SPRAY SAFE-END	0.05	0.675	0.436	0.654
GE NEW	RHR LINE PIPE	0.407	11.26	-	16.89
GE NEW	FEEDWATER ELBOW	0.435	3.746	3.688	5.532
GE OLD	RPV LOWER HEAD	0.032	2.063	0.079	0.119
GE OLD	FW NOZZZLE BORE	0.7	9.859	3.168	4.752
GE OLD	CORE SPRAY NOZZLE	0.023	0.441	0.52	0.78
GE OLD	FW LINE RCIC TEE	0.427	5.016	6.98	10.47

BACKGROUND - NUREG/CR-6674

- **Pacific Northwest National Laboratories (PNNL) assessed the effects of water environment for 60 year life: leakage + core damage frequency**
  - INEL results for stresses/environmental conditions
  - Fatigue curves for carbon steel/LAS taken from NUREG/CR-6335 (1995)
  - Used an enhanced version of the pc-PRAISE code to evaluate probabilities of CUF exceeding 1.0 and through-wall cracking
  - Through-wall stresses and component geometry (diameter/thickness) based on reasonable assumptions; NUREG/CR-6260 had only surface stress amplitudes
- **Key conclusions**
  - Some components had through-wall crack probabilities ≈ 0.05/year
  - Probabilities of throughwall cracking in water environment approached 1.0 for some components for both 40 and 60 years
  - Core damage frequencies < 10^{-6} per year; much less in most cases

BACKGROUND - NUREG/CR-6674

Results for ferritic components

VENDOR/VINTAGE	LOCATION	40 YR DESIGN CUF	NUREG/CR-5999 CUF	NUREG/CR-6674 PROBABILITIES					
				40-YEAR AIR		40-YEAR WATER		60-YEAR WATER	
			60 YR EXTRAPOLATION	PTW CRACK	CDF	PTW CRACK	CDF	PTW CRACK	CDF
CE NEW	RPV LOWER HEAD/SHELL	0 007	0.021	8.40E-23	1.65E-24	6.71E-15	1.13E-14	1.44E-12	1.91E-14
CE NEW	INLET NOZZLE	0 182	0.712	1.75E-07	6 94E-14	5 90E-05	2 03E-11	9.01E-04	2 05E-10
CE NEW	OUTLET NOZZLE	0 377	0.708	1.00E-07	6 75E-14	1.74E-03	9 65E-10	2.90E-02	6.93E-09
CE NEW	CHARGING NOZZLE	0 05	0.156	6 48E-11	8 32E-17	2.61E-06	2.77E-12	5.50E-05	4.05E-11
CE NEW	SAFETY INJECTION	0 898	0 686	1.22E-06	8.30E-13	1.00E-06	1.88E-12	1.90E-05	7.50E-12
CE OLD	RPV LOWER HEAD/SHELL	0.008	0.02	4.85E-24	9.56E-26	6.36E-16	1.08E-17	1.85E-13	1.86E-14
CE OLD	INLET NOZZLE	0 073	0 258	7.99E-11	3 40E-17	4.11E-07	1.58E-13	1.33E-05	3.59E-12
CE OLD	OUTLET NOZZLE	0 284	0 831	6 72E-04	1.91E-10	7.05E-02	2 42E-08	3.53E-01	6.13E-08
B&W	RPV LOWER HEAD/SKIRT	0 12	0.335	4.07E-09	6 24E-11	7.85E-06	1.04E-07	7.52E-04	1.36E-06
B&W	OUTLET NOZZLE	0.9	0.704	2.92E-03	6.72E-10	1.83E-01	5 25E-08	4.55E-01	9.03E-08
B&W	HOT LEG SURGE NOZZLE	0 592	0.705	ESTIMATED CUF - STRESSES NOT AVAILABLE					
B&W	CORE FLOOD NOZZLE	0 345	0 948	NOT ANALYZED SINCE STAINLESS STEEL SAFE-END LOCATION MORE CONTROLLING					
W NEW	RPV LOWERHEAD	0.012	0 027	1.04E-19	1.80E-21	7.52E-13	1.24E-14	9 64E-11	1.50E-12
W NEW	INLET NOZZLE	0 11	0 435	7.03E-10	3.00E-16	9.17E-07	3.51E-13	2 84E-05	7.64E-12

BACKGROUND - NUREG/CR-6674

Results for ferritic components (continued)

VENDOR/VINTAGE	LOCATION	40 YR DESIGN CUF	NUREG/CR-5999 CUF	NUREG/CR-6674 PROBABILITIES					
				40-YEAR AIR		40-YEAR WATER		60-YEAR WATER	
			60 YR EXTRAPOLATION	PTW CRACK	CDF	PTW CRACK	CDF	PTW CRACK	CDF
W NEW	OUTLET NOZZLE	0.398	0 987	3 00E-04	6.76E-11	3 65E-01	8.57E-08	7.42E-01	1.22E-07
W OLD	RPV AT SUPPORT	0 29	1.33	1.06E-09	1.38E-11	7.20E-07	8.44E-09	1.11E-05	9 15E-08
W OLD	INLET NOZZLE	0 28	0.744	4.48E-04	1.28E-10	4 38E-03	2.03E-09	5 04E-02	1.07E-08
W OLD	OUTLET NOZZLE	0.431	0.52	7.77E-03	1.89E-09	9 33E-03	4.21E-09	9 60E-02	2.04E-08
GE NEW	RPV LOWER HEAD	0 2	0 942	3 59E-18	6.03E-20	7.88E-12	1.25E-13	6 82E-10	8 26E-12
GE NEW	FW NOZZLE SAFE-END	0.301	2.822	2.00E-06	1.88E-14	1.31E-03	3.57E-11	1.47E-02	1.84E-10
GE NEW	CORE SPRAY SAFE-END	0 05	0 654	6 54E-12	3.67E-19	1.45E-07	7.09E-15	3 25E-06	1.09E-13
GE NEW	RHR LINE PIPE	0.407	16 89	2.08E-01	1.35E-11	4.10E-01	2.54E-11	6.21E-01	2.03E-10
GE NEW	FEEDWATER ELBOW	0.435	5 532	1.00E-05	2.25E-11	1.01E-03	3.04E-09	1.46E-02	5 06E-09
GE OLD	RPV LOWER HEAD	0.032	0.119	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GE OLD	FW NOZZZLE BORE	0 7	4.752	4.53E-05	8.69E-14	1.00E-05	3.75E-14	8 80E-04	1.46E-12
GE OLD	CORE SPRAY NOZZLE	0.023	0 78	4.44E-14	1.72E-22	1.91E-08	6.41E-17	8 84E-07	2.14E-15
GE OLD	FW LINE RCIC TEE	0.427	10.47	4.30E-06	2 00E-13	2.99E-03	1.04E+10	5 92E-02	8 30E-10

BACKGROUND - GSI 190 CLOSURE

- **Generic Safety Issue 190 was closed following December 26, 1999 memo from Thadani to Travers**
  - Low core damage frequencies were predicted even with most recent fatigue test data
  - However, since studies indicated an increase in leakage frequency, recommendation was that licensees address environmental effects in support of license renewal
- **Actions by NRC were reasonable based on conservative results presented in NUREG/CR-6674**
  - High probability of leakage from some components

RE-EVALUATION OF NUREG/CR-6674

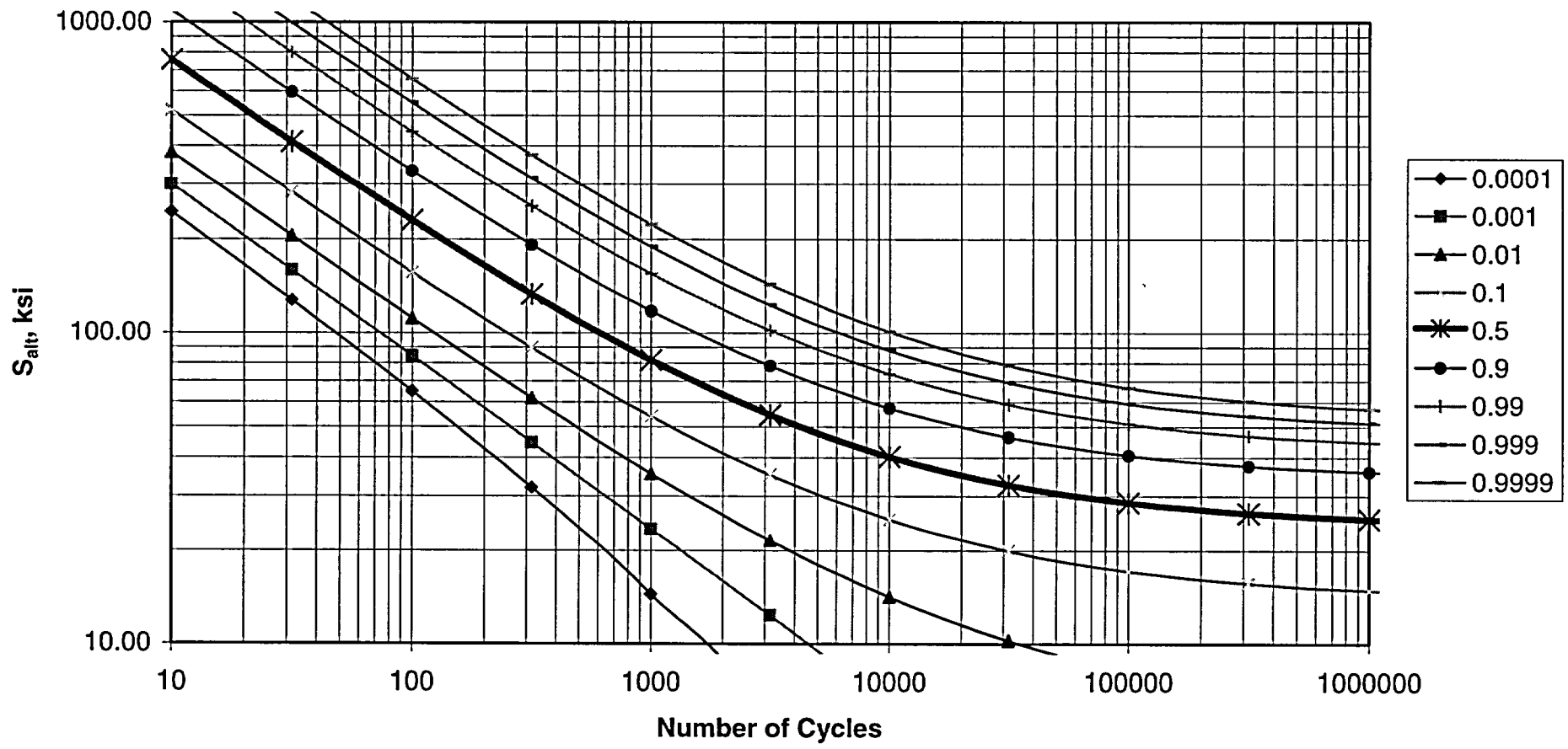
- **Project initiated to re-evaluate fatigue initiation and leakage probabilities, and core damage frequencies presented in NUREG/CR-6674**
- **Re-evaluation included**
  - Critical review of input to NUREG/CR-6674
  - NRC version of pc-PRAISE further modified by Engineering Mechanics Technology
    - Benchmarked against NUREG/CR-6674 results
  - Revised initiation/leakage probability predictions
- **Initial effort concentrated on carbon and low-alloy steel components**

DIFFERENCES FROM NUREG/CR-6674

- **Standard deviation at high-cycle end of fatigue curve was reduced**
  - Available data suggest standard deviation ≈ 0.1 S_{alt} at 10^6 cycles
  - NUREG/CR-6674 assumed standard deviation of 0.325 S_{alt} for LAS and 0.277 S_{alt} for CS (from NUREG/CR-6335)
    - Represents physically impossible material behavior
- **Significantly affects components with large number of low-stress cycles**
  - e.g., RPV nozzles designed for daily load following

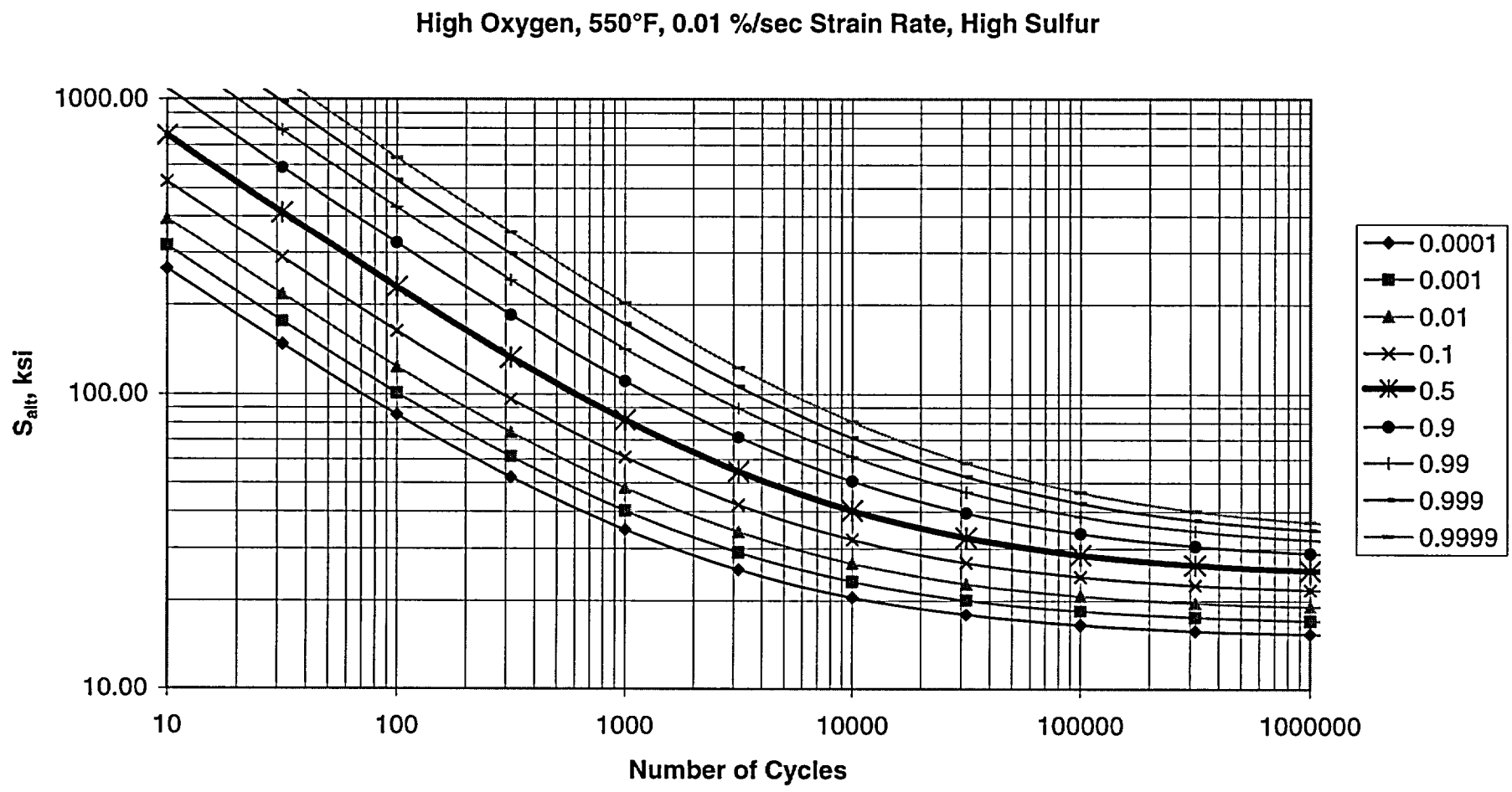
RE-EVALUATION OF NUREG/CR-6674

- **Low-alloy steel fatigue curve from NUREG/CR-6674**



RE-EVALUATION OF NUREG/CR-6674

- **Low-alloy steel fatigue curve from NUREG/CR-6674 with modified endurance limit variance**

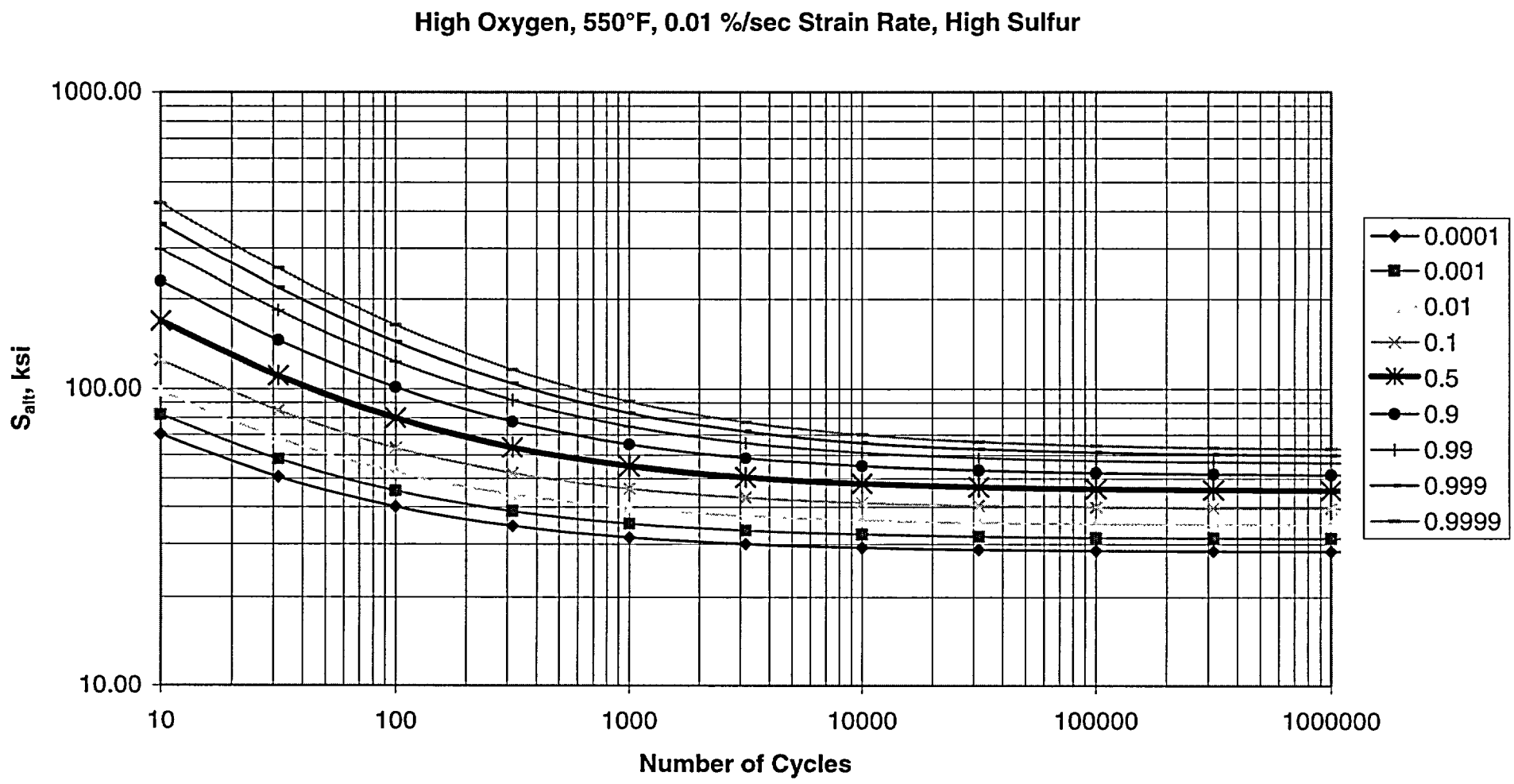


DIFFERENCES FROM NUREG/CR-6674

- **Latest environmental test data were considered - Original work based on NUREG/CR-5999 (1993) and NUREG/CR-6335 (1995)**
  - NUREG/CR-6583 published March 1998
  - NUREG/CR-6717 published May 2001
- **New curves are more penalizing at low-cycle end and slightly less penalizing at high-cycle end**
- **All carbon and low-alloy steel components were re-evaluated using modified endurance limit and updated fatigue curves**

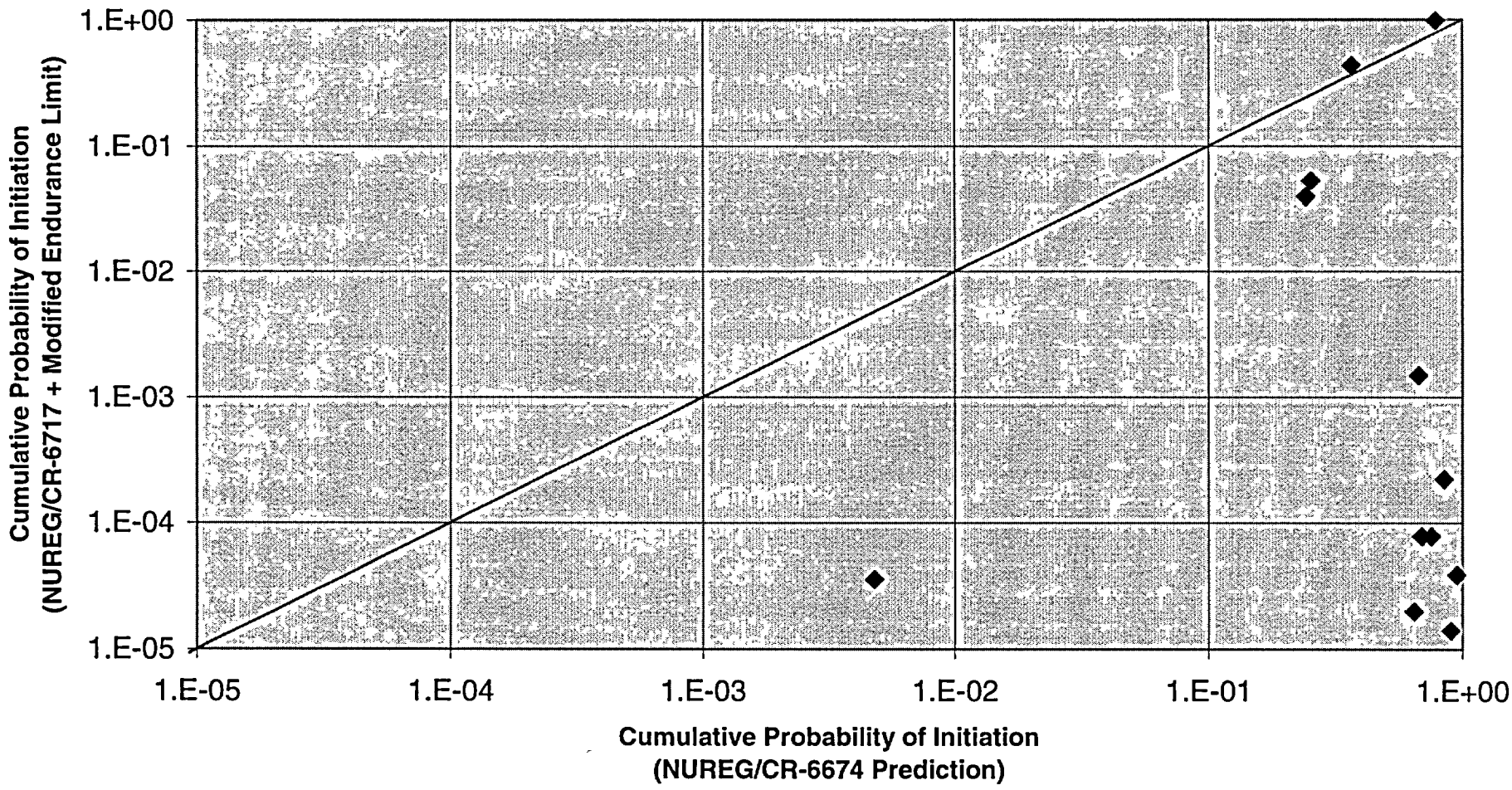
RE-EVALUATION OF NUREG/CR-6674

- **Low-alloy steel fatigue curve with modified variance/latest data fit**



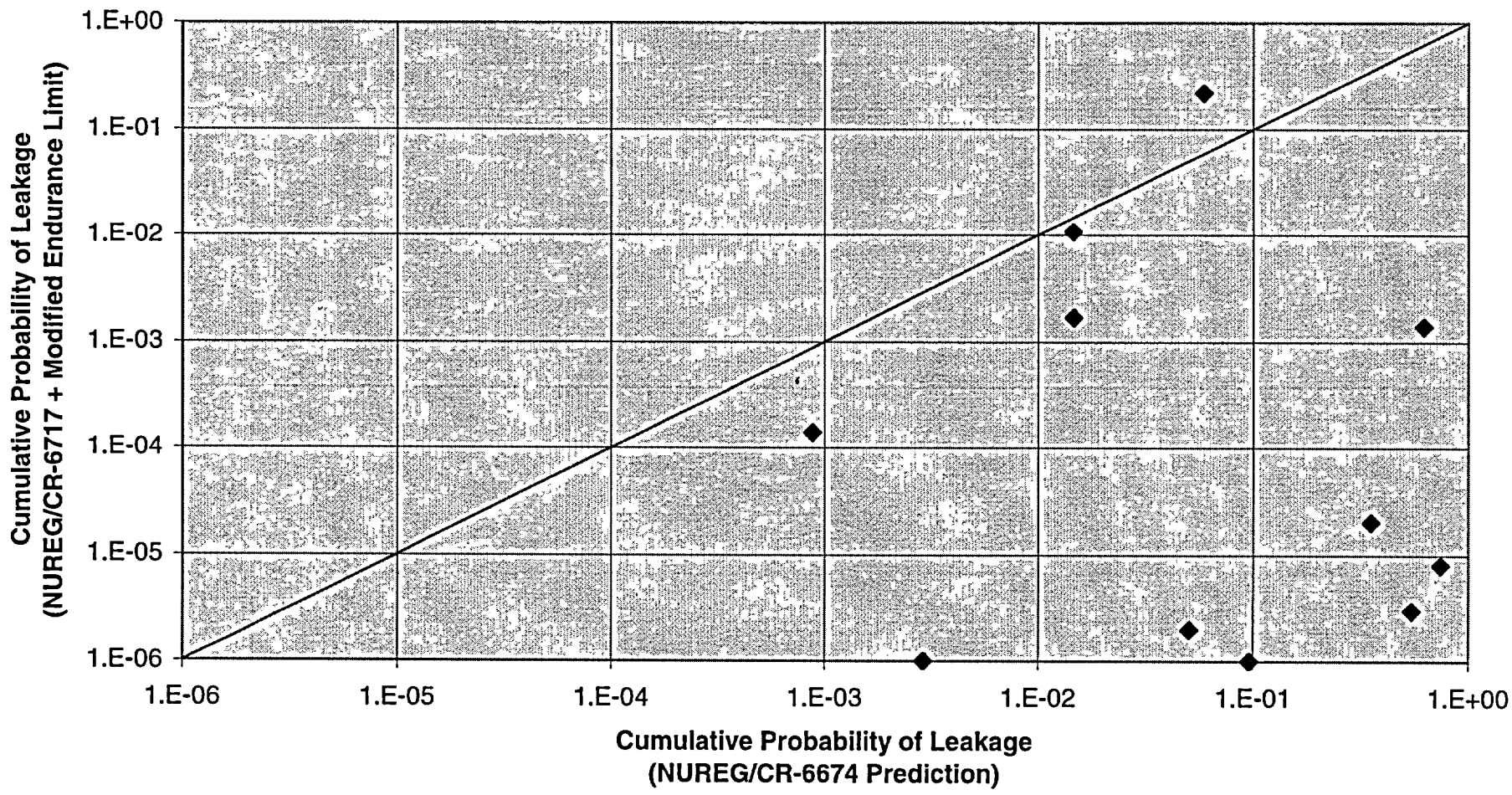
RE-EVALUATION OF NUREG/CR-6774

- **Effect of updated fatigue data and endurance limit modification on probability of initiation at 60 years**



RE-EVALUATION OF NUREG/CR-6674

- **Effect of updated fatigue data and endurance limit modification on probability of leakage at 60 years**



Further Refinements From NUREG/CR-6674

- **Locations with predicted leakage probabilities > 10^{-3} were further evaluated**
  - Re-evaluation based on NUREG/CR-6260 temperatures instead of 590ºF defaults considered in NUREG/CR-6674 for some BWR components
    - Actual location operating temperatures do not approach 590°F for these components
- **For locations where detailed stress reports were available, actual geometry, stresses, strain rates and reduced cycles were used**

RE-EVALUATION OF NUREG/CR-6674

- **Detailed analysis of RPV outlet nozzles conducted to incorporate available stress report and cycle information**

Older CE Plant

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	All Modifications
Initiation			
40 Years	0.591	1.1×10^{-4}	$<10^{-6}$
60 Years	0.846	2.2×10^{-4}	$<10^{-6}$
Leakage			
40 Years	0.071	$<10^{-6}$	$<10^{-6}$
60 Years	0.353	2.0×10^{-5}	$<10^{-6}$

B&W Plant

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	All Modifications
Initiation			
40 Years	0.774	6×10^{-6}	$<10^{-6}$
60 Years	0.899	1.4×10^{-5}	$<10^{-6}$
Leakage			
40 Years	0.183	2×10^{-6}	$<10^{-6}$
60 Years	0.544	3×10^{-6}	$<10^{-6}$

RE-EVALUATION OF NUREG/CR-6674

- **Evaluation of two new GE components with actual temperatures from NUREG/CR-6260**

FW Nozzle Safe-End

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Consideration of Actual Temperatures
Initiation			
40 Years	0.104	0.0139	0.0004
60 Years	0.253	0.0533	0.0024
Leakage			
40 Years	0.0013	0.0001	1×10^{-6}
60 Years	0.0147	0.0017	3.5×10^{-5}

FW Line Elbow

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Consideration of Actual Temperatures
Initiation			
40 Years	0.159	0.140	0.0032
60 Years	0.365	0.434	0.0192
Leakage			
40 Years	0.0010	0.0003	2×10^{-6}
60 Years	0.0146	0.0107	1.8×10^{-4}

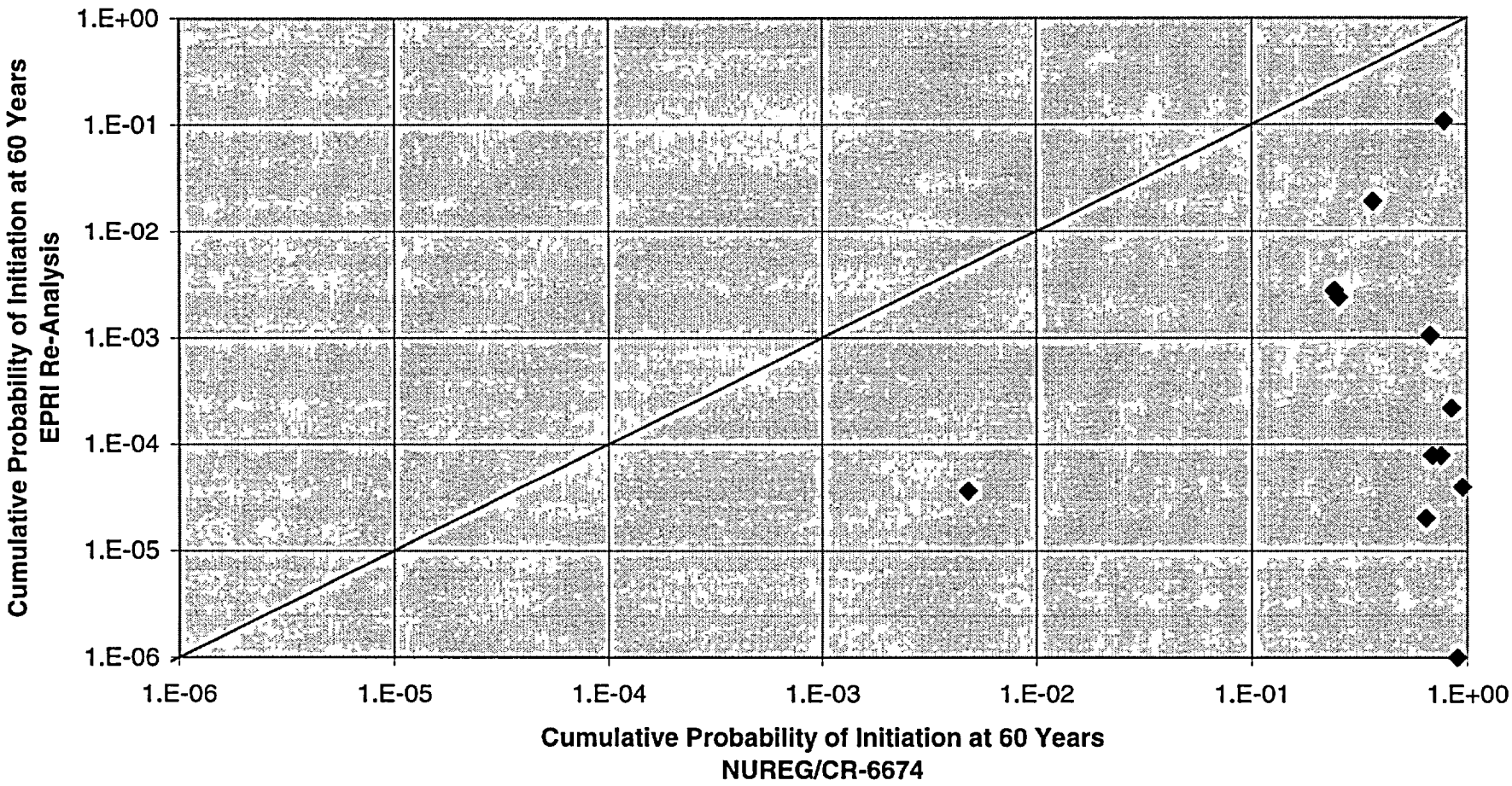
RE-EVALUATION OF NUREG/CR-6674

- **Evaluation of GE feedwater line RCIC tee with actual temperatures, cycles and strain rates**

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Consideration of Actual Temperatures, OBE Cycles and Strain Rates
Initiation			
40 Years	0.376	0.791	0.025
60 Years	0.782	0.981	0.108
Leakage			
40 Years	0.0030	0.0169	6×10^{-5}
60 Years	0.0592	0.2190	0.00139

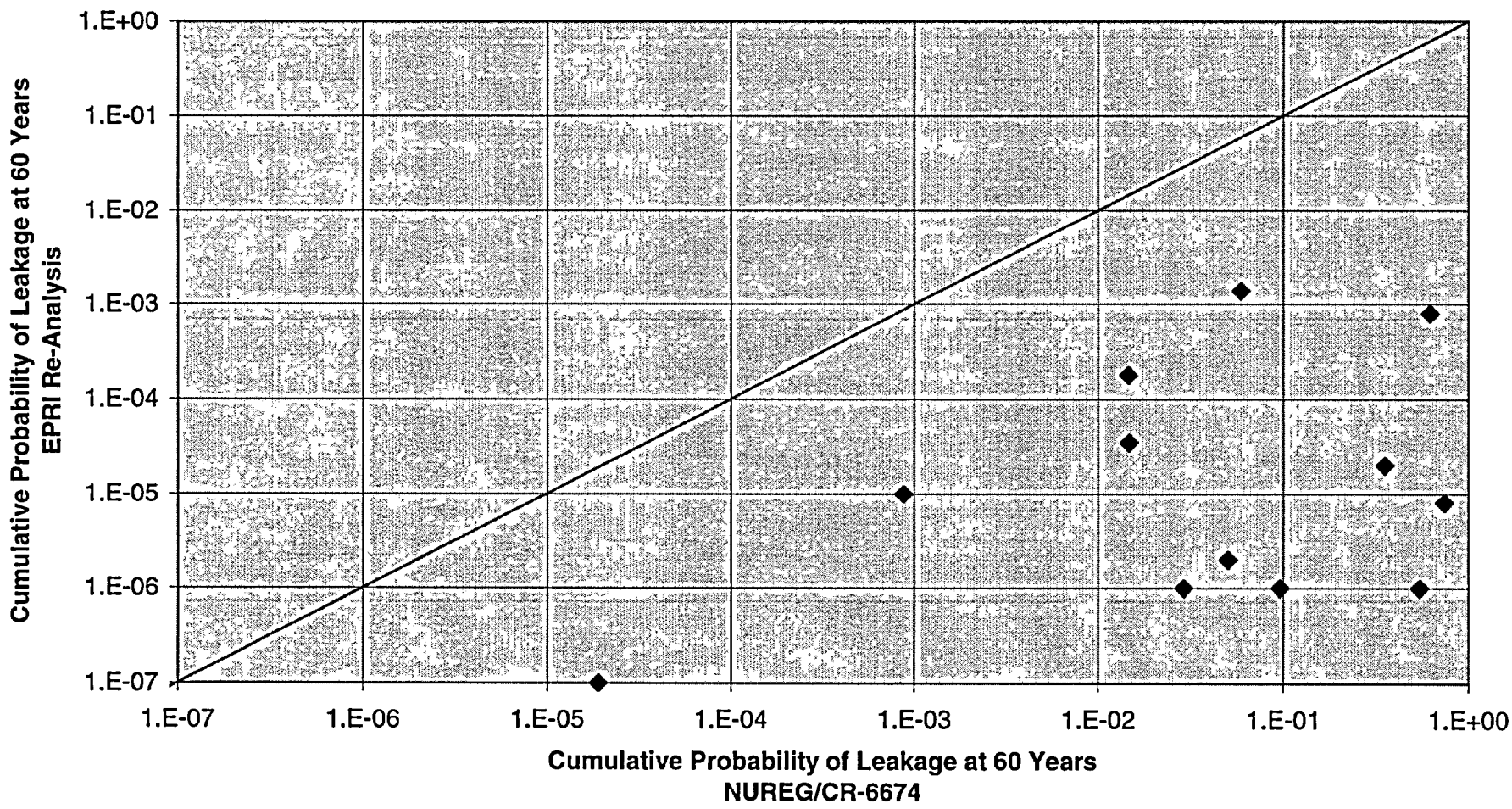
RE-EVALUATION OF NUREG/CR-6674

- **Final 60-year initiation probabilities**



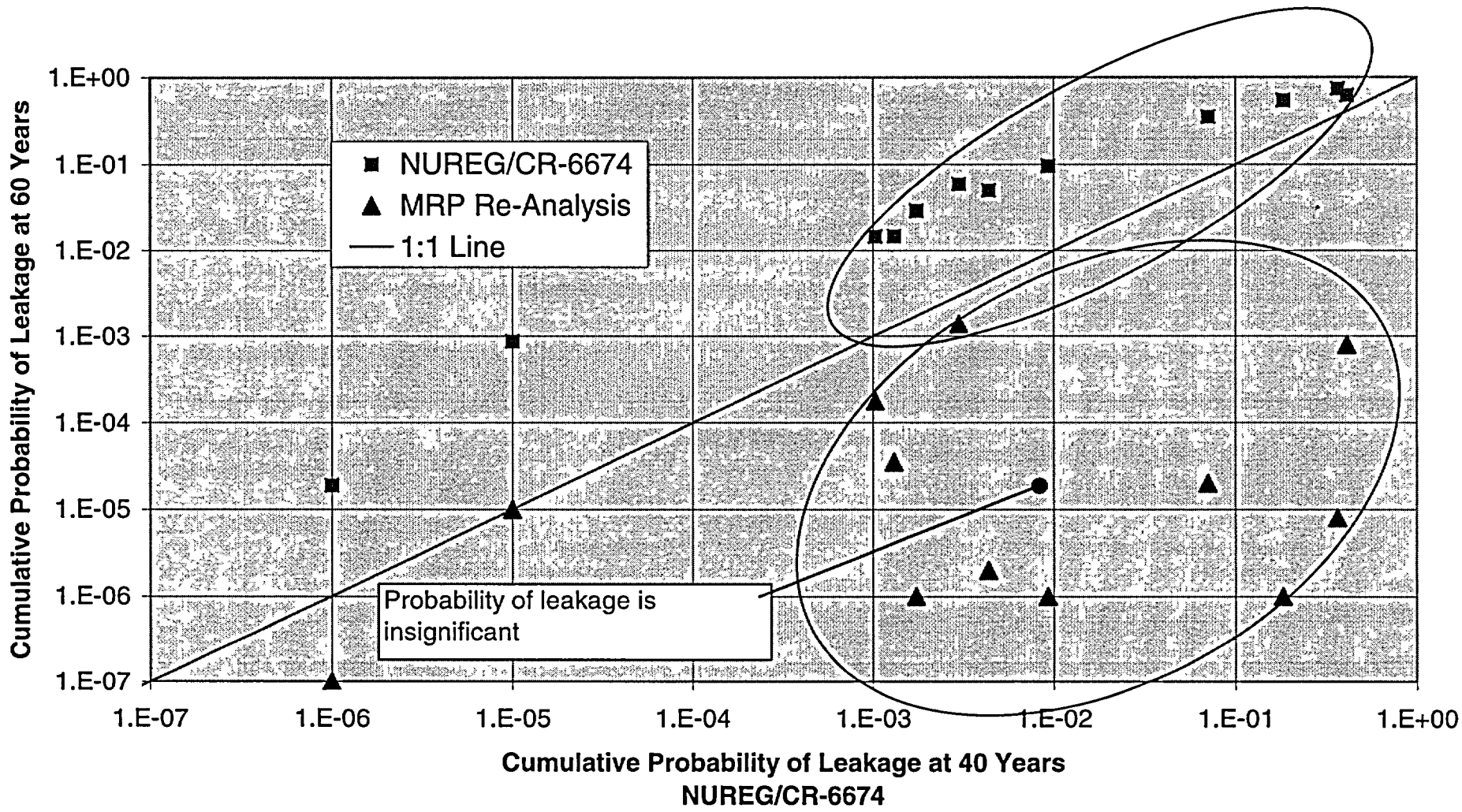
RE-EVALUATION OF NUREG/CR-6674

- **Final 60-year leakage probabilities**



RE-EVALUATION OF NUREG/CR-6674

- **Comparison of revised analysis to NUREG/CR-6674 results**



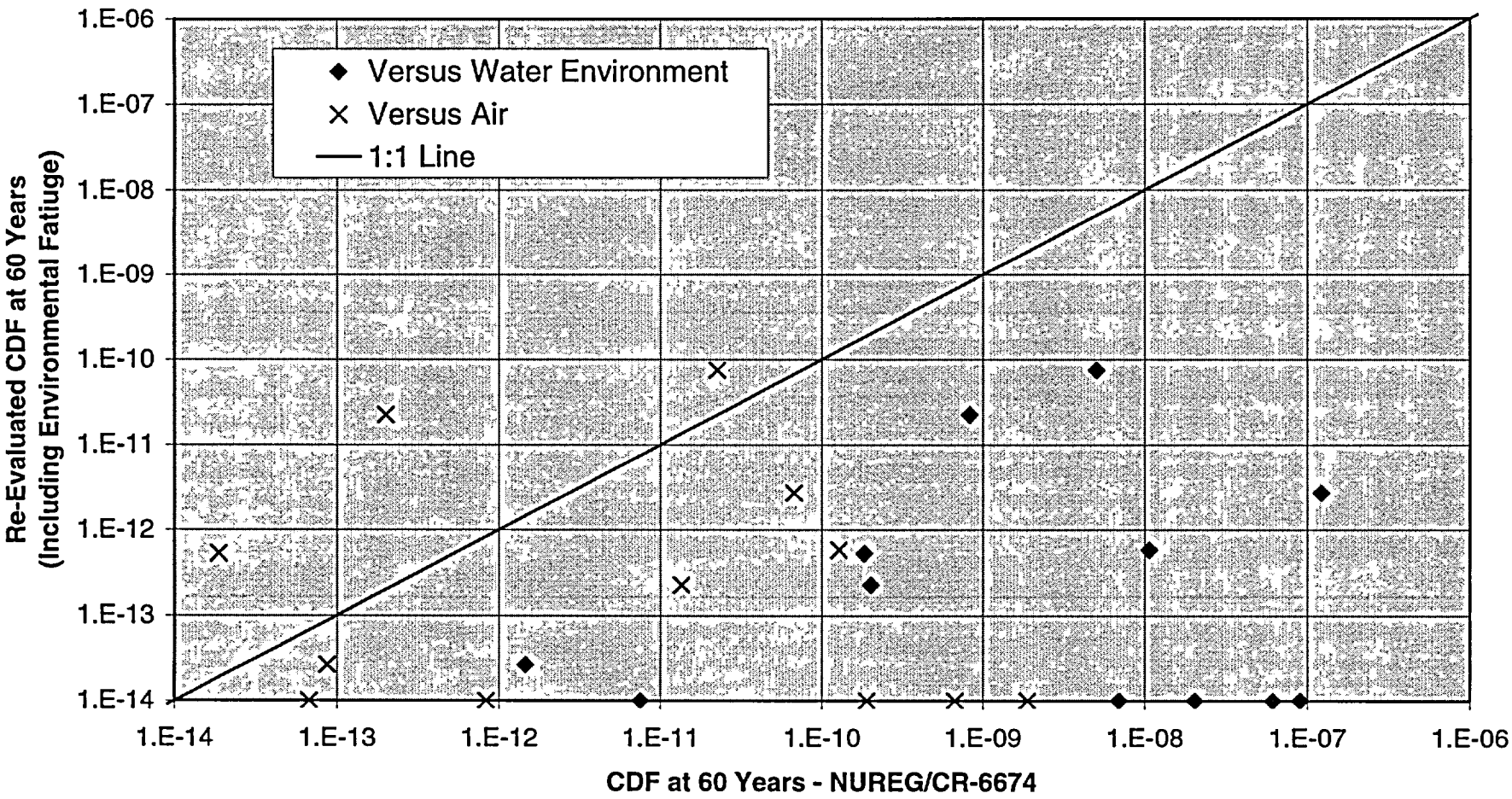
RE-EVALUATION OF NUREG/CR-6674

- **Core damage frequency re-evaluation**

Plant/Location	PNNL Results			Re-Evaluation	
	Air	Water		Water	
	CDF(40)	CDF(40)	CDF(60)	CDF(40)	CDF (60)
B&W RPV OUTLET NOZZLE	6.72E-10	5.25E-08	9.03E-08	0	0
CE-NEW RPV OUTLET NOZZLE	6.75E-14	9.65E-10	6.93E-09	0	0
CE-NEW SAFETY INJECTION NOZZLE	8.30E-13	1.88E-12	7.50E-12	0	0
CE-OLD RPV OUTLET NOZZLE	1.91E-10	2.42E-08	6.13E-08	0	0
GE-NEW FEEDWATER NOZZLE SAFE END	1.88E-14	3.37E-11	1.84E-10	2.12E-14	5.23E-13
GE-NEW RHR LINE STRAIGHT PIPE	1.35E-11	2.54E-11	2.03E-10	4.13E-14	2.26E-13
GE-NEW FEEDWATER LINE ELBOW	2.25E-11	3.04E-09	5.06E-09	2.70E-11	7.50E-11
GE-OLD RPV FEEDWATER NOZZLE BORE	8.69E-14	3.75E-14	1.46E-12	3.08E-16	2.65E-14
GE-OLD FEEDWATER LINE - RCIC TEE	2.00E-13	1.04E-10	8.30E-10	8.77E-12	2.25E-11
W-NEW RPV OUTLET NOZZLE	6.76E-11	8.57E-08	1.22E-07	4.06E-13	2.71E-12
W-OLD RPV INLET NOZZLE	1.28E-10	2.03E-09	1.07E-08	8.28E-14	5.78E-13
W-OLD RPV OUTLET NOZZLE	1.89E-09	4.21E-09	2.04E-08	2.70E-13	0

RE-EVALUATION OF NUREG/CR-6674

- **Revised core damage frequency**



CONCLUSIONS

- **Probabilities of initiation and leakage significantly less than reported in NUREG/CR-6674**
  - More realistic fatigue curve endurance limit variance
  - Use of latest environmental fatigue test data fits
  - Use of actual temperatures
  - Examination of detailed stress analysis showed that probabilities could be further reduced
- **60-year leakage probability considering environmental effects for ferritic components not significant**
  - Less than 40-year values from NUREG/CR-6674 accepted by NRC
  - Maximum 60-year leakage probability of 0.0014
- **Maximum core damage frequency < 1 x 10^{-10}/year**

Review of Laboratory, Component and Structural Environmental Fatigue Data

R. E. Nickell
Applied Science & Technology, Poway, CA

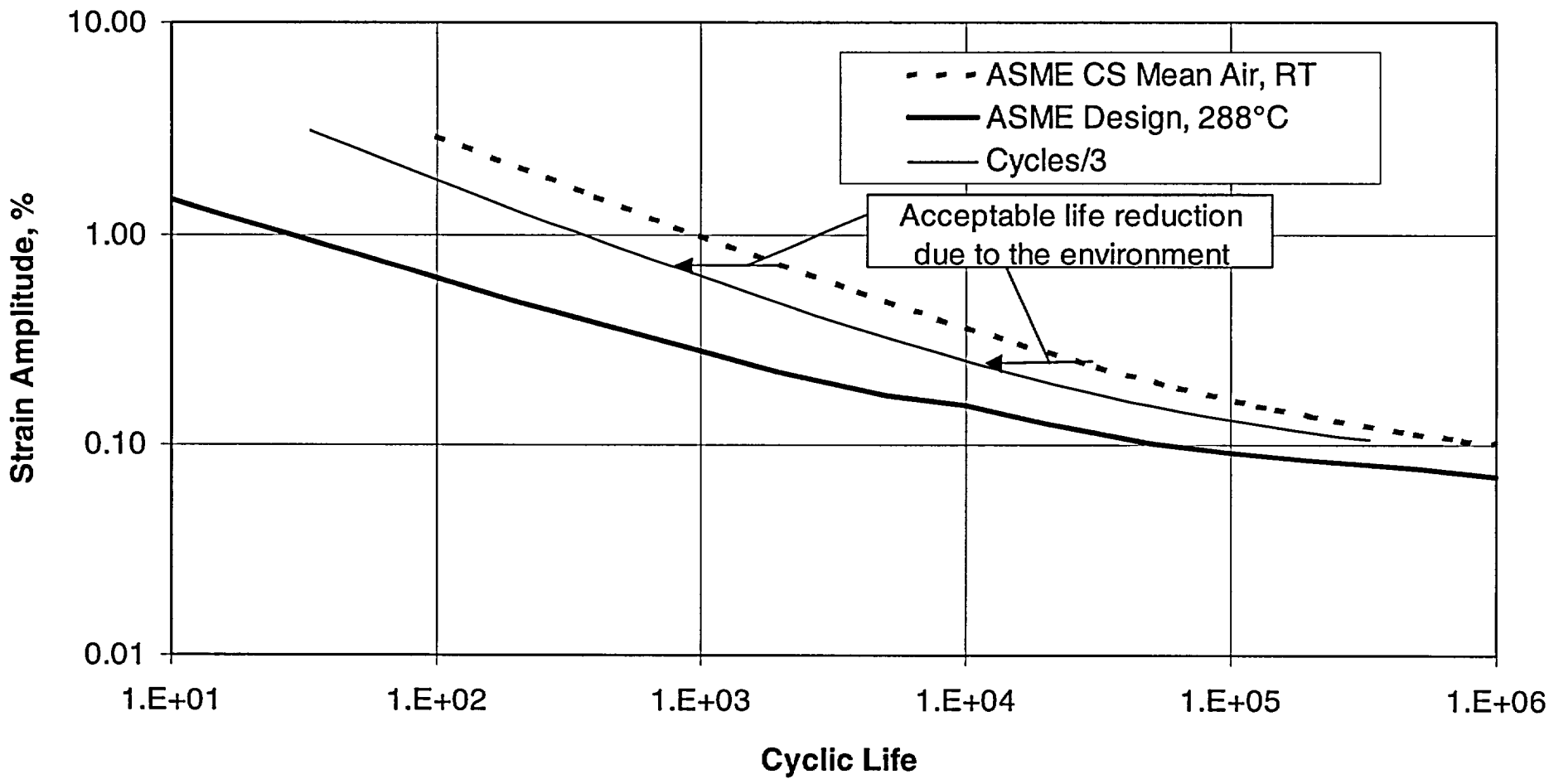
September 18, 2002
NRC Headquarters

Enclosure 5

Objectives

- Demonstrate that data are consistent with the revised probabilities of through-wall cracking (and leakage) for component locations evaluated in NUREG/CR-6674
- Show that laboratory test data obtained under simulated reactor water environmental conditions for carbon and low-alloy steels are within margins in ASME Code fatigue design curve
- Show that structural/component fatigue test data with one surface in contact with water environment exhibit behavior consistent with margins in ASME Code fatigue design curve
- Show that operating experience has not revealed significant increases in fatigue failures ascribed to reactor water environmental effects as a function of increasing length of service of nuclear power plant components

Carbon/Low-Alloy Steel Laboratory Data
(Environmental Shift, No Size/Roughness Effect)



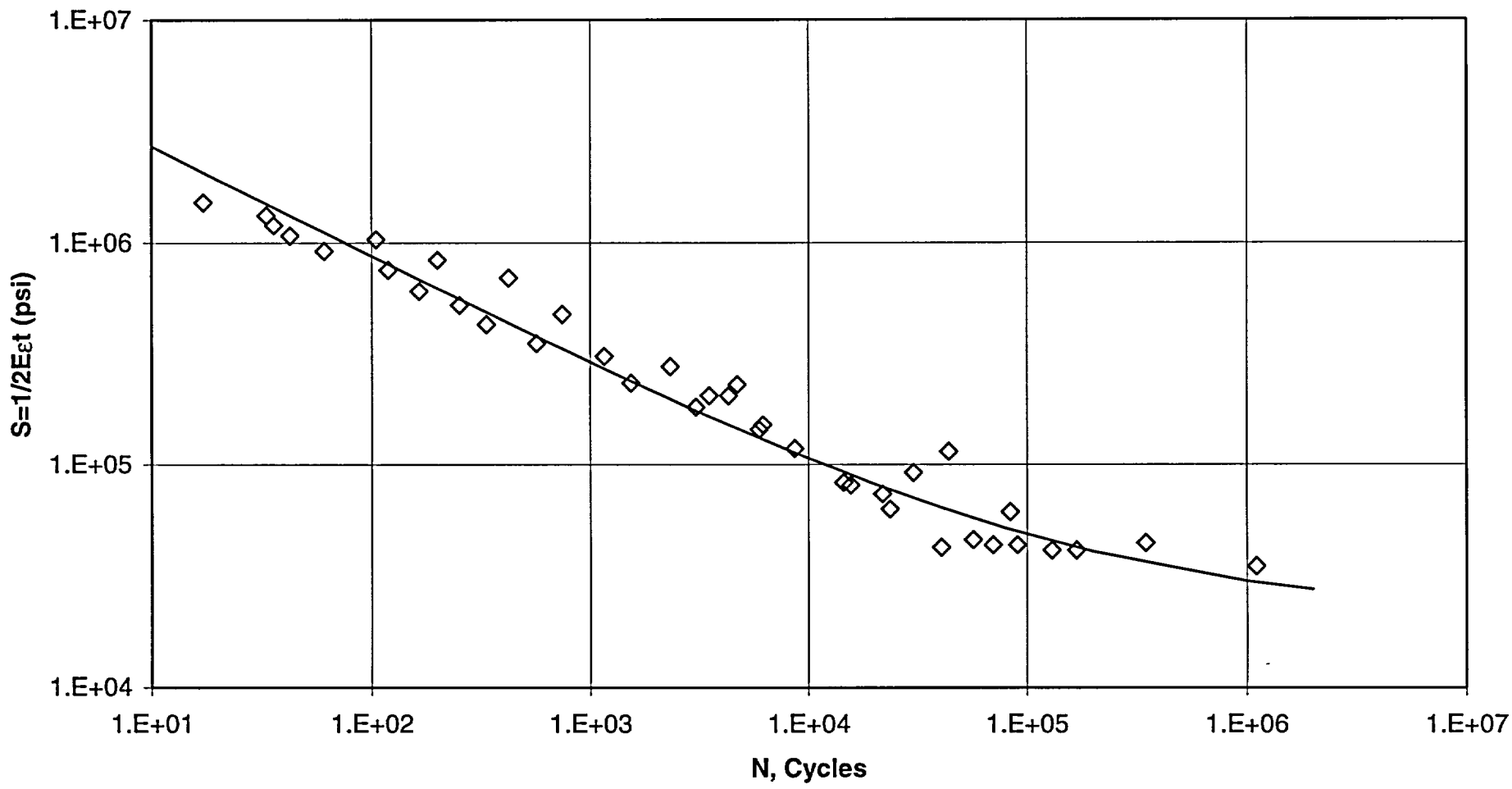
Evaluation of Laboratory Data

- **EPRI published a comprehensive review of laboratory data and the relationship to component/structural fatigue tests in December 2001 (MRP-49)**
- **Performed critical review of laboratory testing/environments and reconciled them with operating experience**
  - Available structural/component test data are consistent with the majority of the laboratory test data and are within margins in ASME Code fatigue design curve; size effects and surface finish effects reduce the fatigue life only slightly
  - Flow rate (rather than the trickle flow for most laboratory testing) has pronounced beneficial effect in operating environments

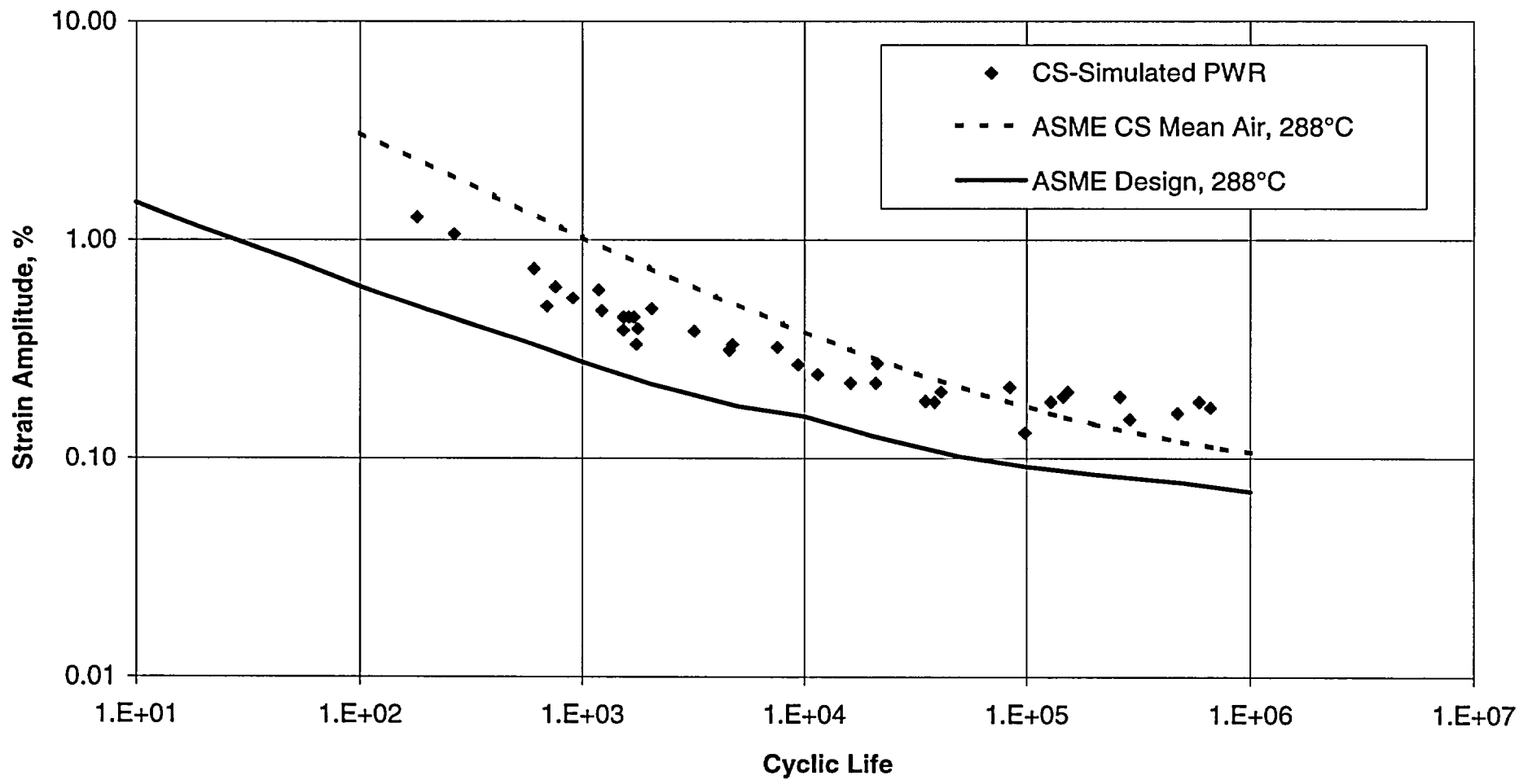
Evaluation of Laboratory Data

- **Findings for Carbon Steel Laboratory Data and Component-Scale Tests**
  - Laboratory simulated reactor water data are within the margin in ASME Code fatigue design curve for environmental effects
  - PVRC component-scale carbon steel tests demonstrate additional shift for size effects and surface finish effects
  - Component-scale flow rate tests with trickle flow compromise full ASME Code margin of 20 at low-cycle end of design curve
  - Moderate flow rate carbon steel component-scale tests shift fatigue life to appropriate position to the right of the ASME Code design curve

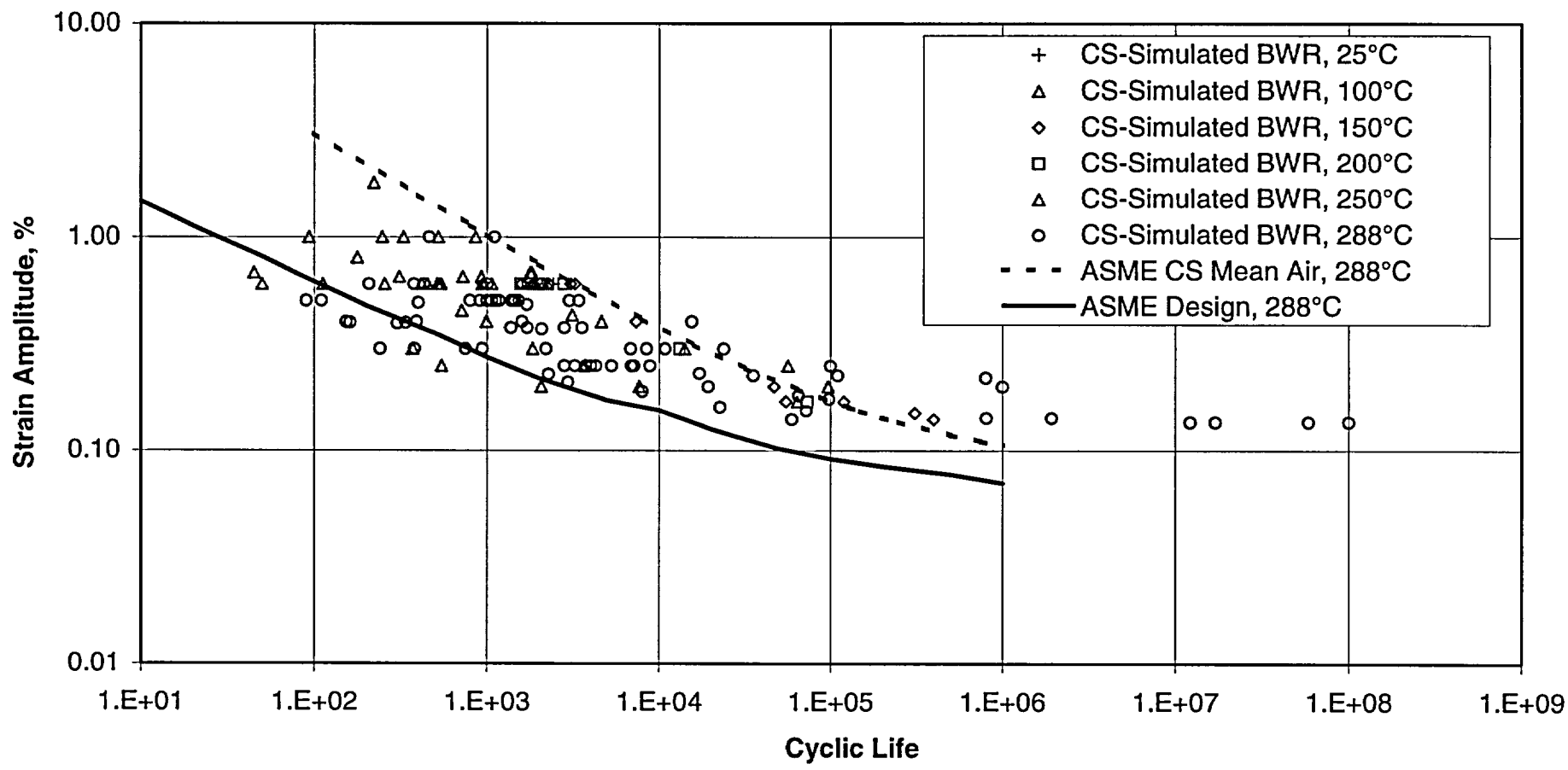
Carbon Steel Laboratory Air Data
(ASME Code Background Document)



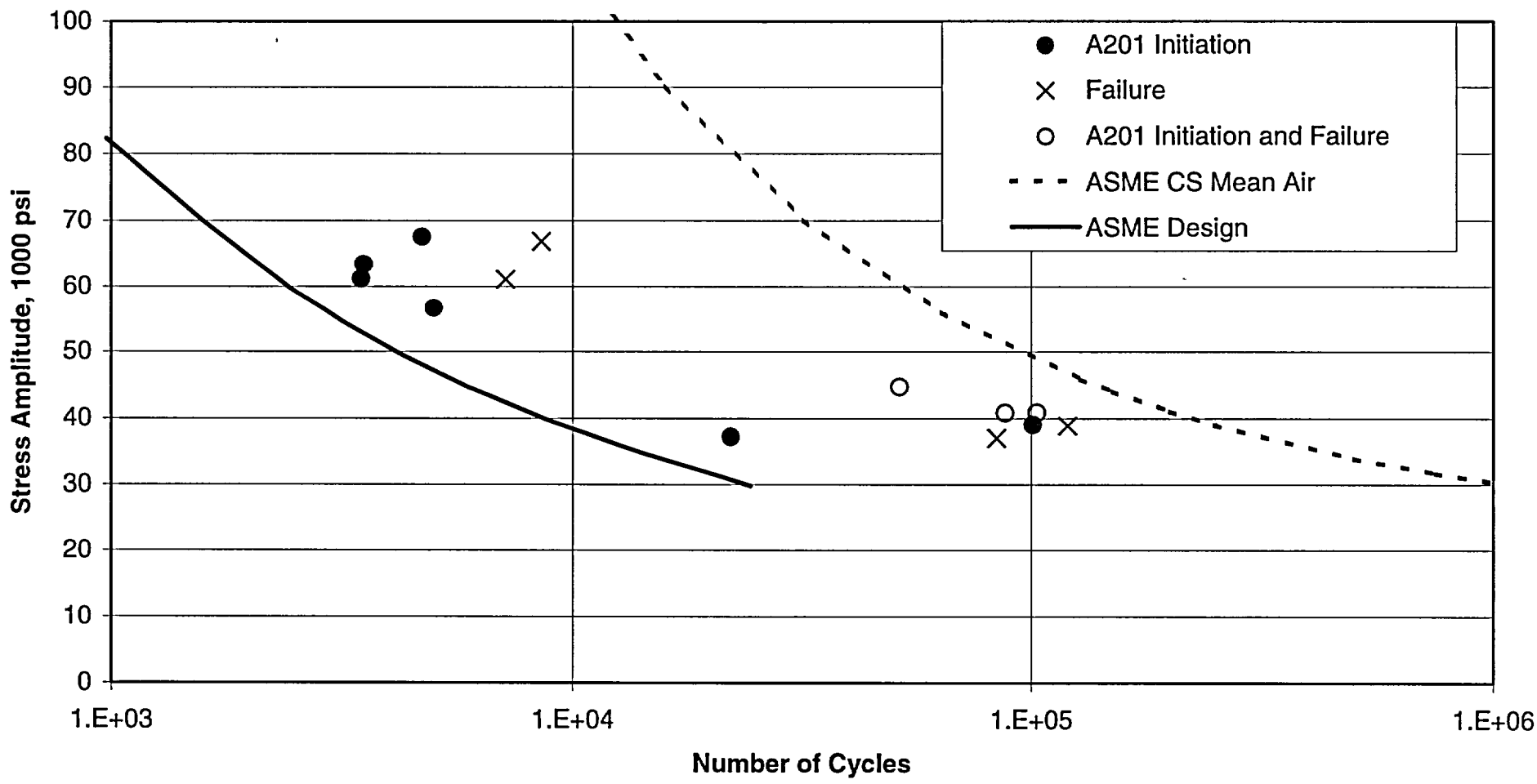
Carbon Steel Simulated PWR Data
(PVRC Data at 288°C)



Carbon Steel Simulated BWR Data
(PVRC Data, All Conditions)



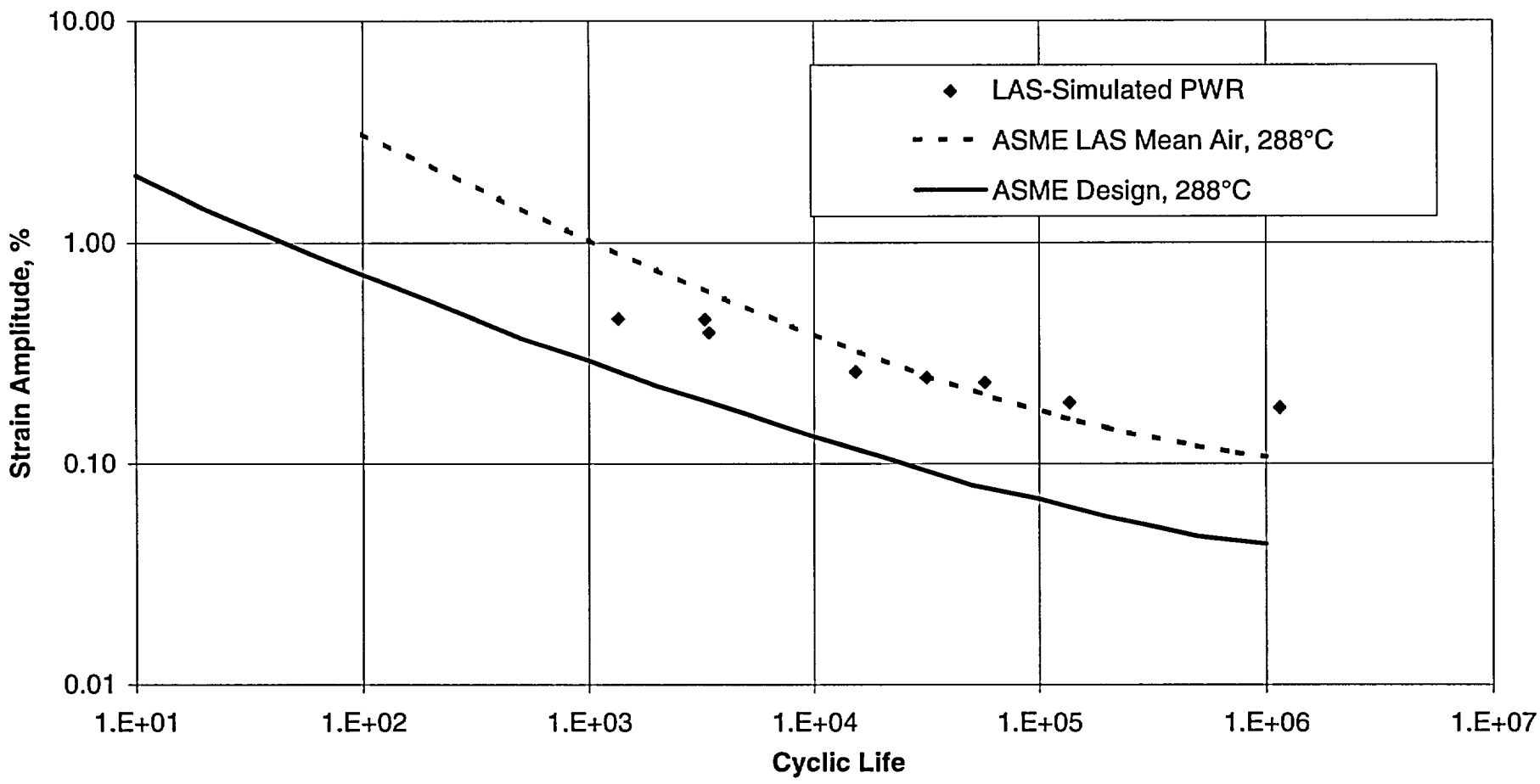
PVRC Carbon Steel Component Tests
(Environmental Shift + Size/Roughness Effect)



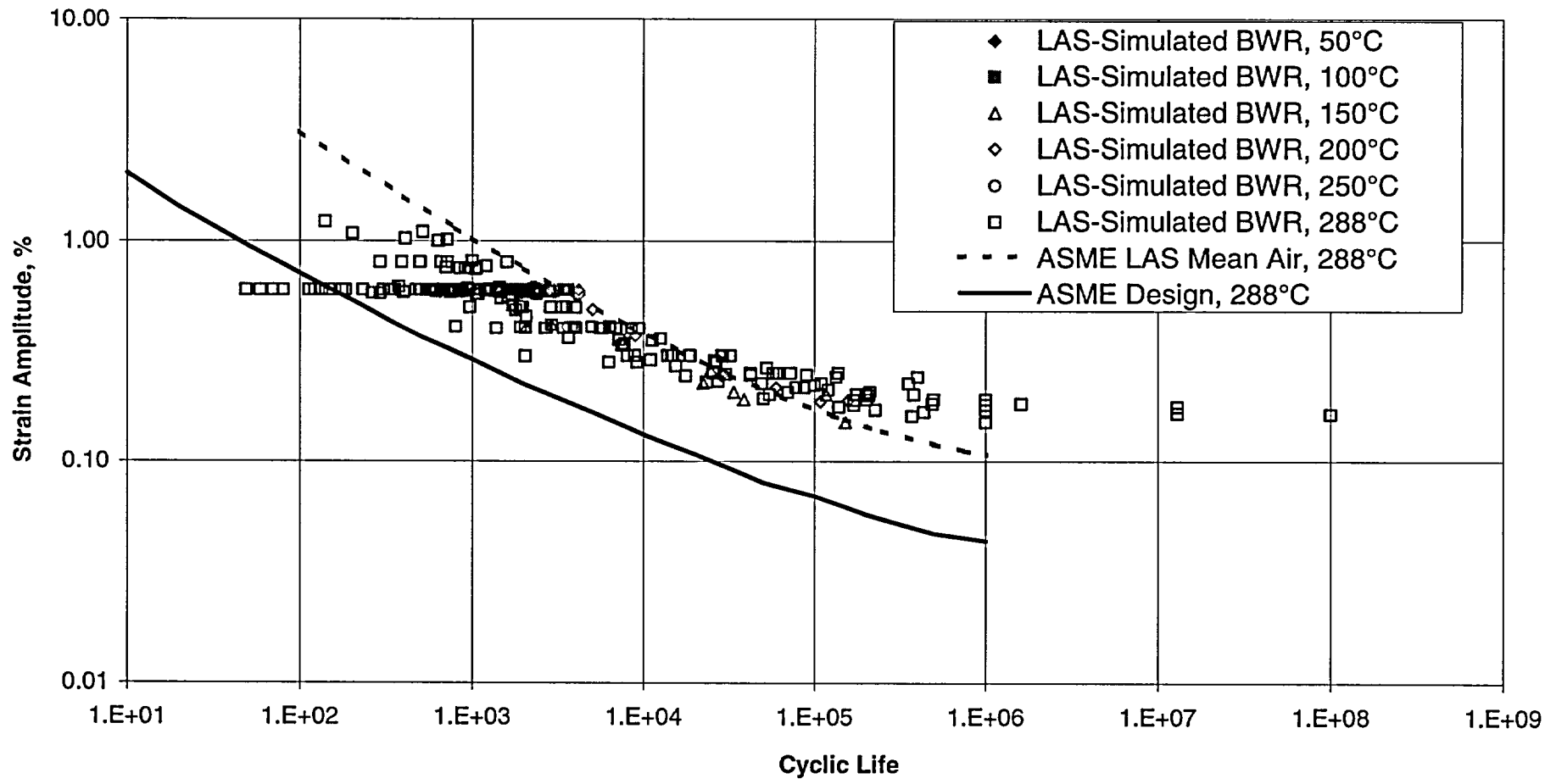
Evaluation of Laboratory Data

- **Findings for Low-Alloy Steel Laboratory Data and Component-Scale Tests**
  - Laboratory simulated reactor water data are within the margin in ASME Code fatigue design curve for environmental effects
  - PVRC component-scale low-alloy steel tests demonstrate additional shift for size effects and surface finish effects

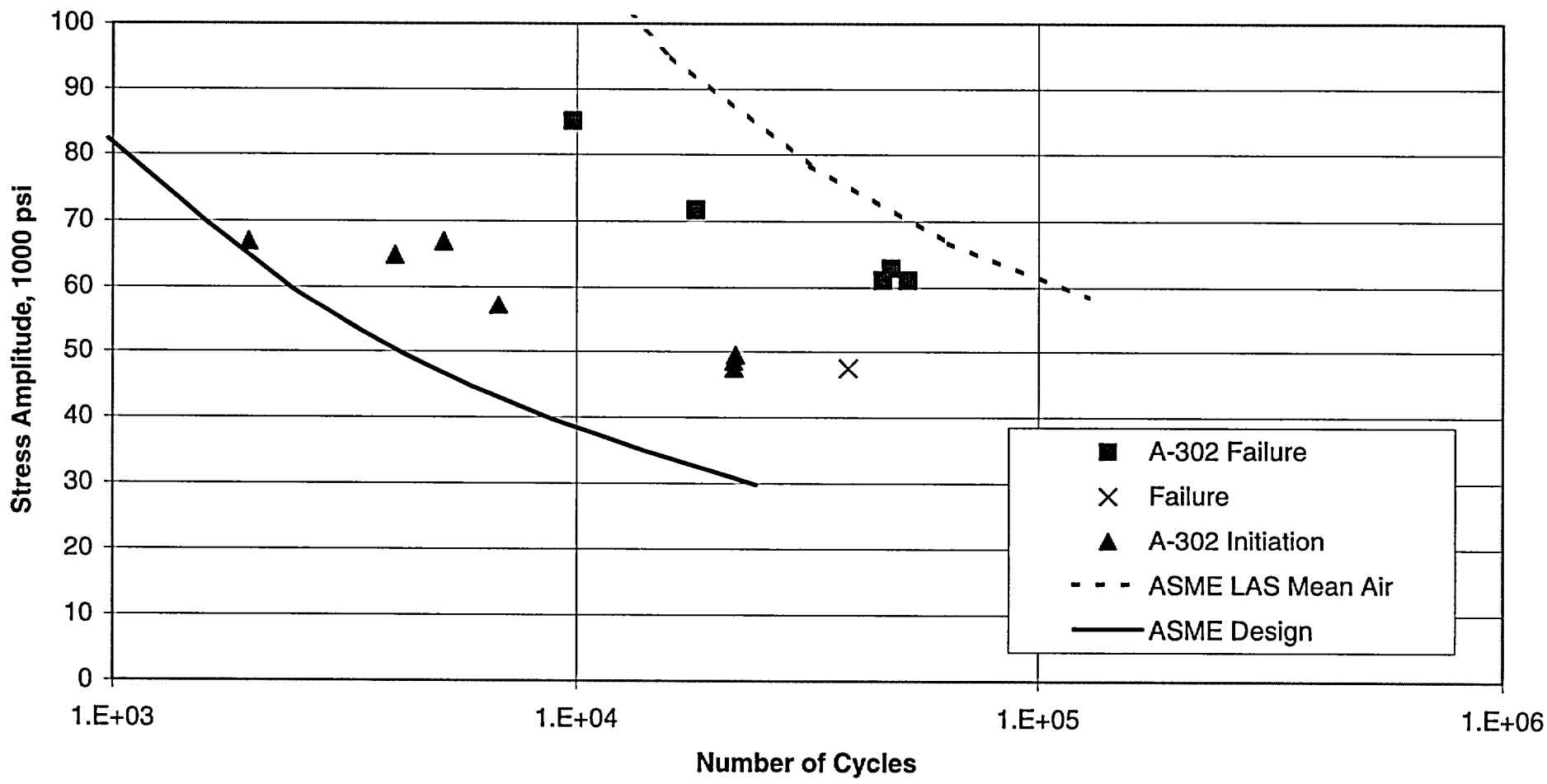
Low-Alloy Steel Laboratory PWR Data



Low-Alloy Steel Laboratory BWR Data



PVRC Low-Alloy Steel Component Tests
(Environmental Shift + Size/Roughness Effect)



Flow Rate Effects

Laboratory Data Versus Operating Experience

- Absence of flow and uniformity of strain at low strain rate appears to be the major difference
- Laboratory data are obtained at high strain amplitudes and very low strain rates on cylindrically-shaped (uniform surface strain) test specimens, under simulated reactor water flow chemistries and very low flow velocities.
- Fatigue crack initiation mechanism appears to be rupture of the protective oxidation/passivation layer at high strain, with reoxidation/repassivation prevented by sustained straining at very low strain rates and low oxidizing potential.

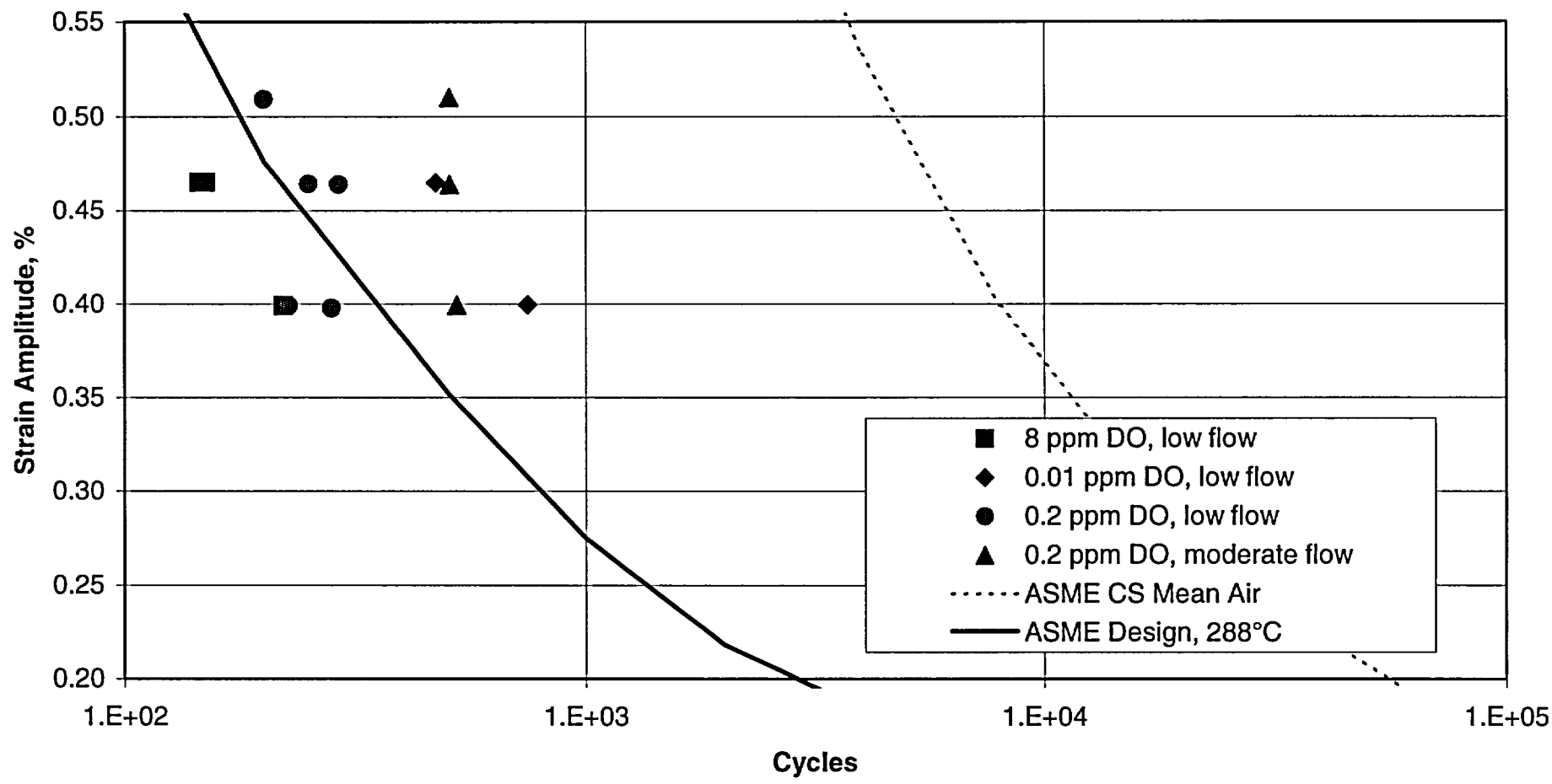
Flow Rate Effects

Laboratory Data Versus Operating Experience

Full-Scale Component Tests on Carbon Steel with Flow at Temperature

- **KWU Tube Tests (Now Framatome ANP GmbH), Erlangen, Germany**
- **Strain-Induced Stress Corrosion Cracking of Non-Post-Weld Heat Treated Cold-Formed Bends and Welded Joints in High-Temperature Water, E. Lenz and A. Liebert, BMFT 11 B 504/2, May 1986.**
  - 1200 Liters/hr (20 Liters/min) recirculating flow rate at 240°C with controlled dissolved oxygen concentration.
  - 34-mm nominal diameter tubes with 3.6-mm wall thickness bent 180°.
  - Cross-sectional area = 1.407 in^2; maximum flow rate = 14.45 in/sec (0.4 m/sec).

KWU Carbon Steel Component Flow Tests
(Variable Flow, Variable DO)



Operating Plant Failure Data

- EPRI TR-110102, Nuclear Reactor Piping Failures at U. S. Commercial LWRs: 1961 – 1997, Draft Report, July 1998
- 4064 failure event records; about 2200 leaking events and about 1800 non-leaking events
- Mostly IGSCC (1227 failures), erosion and flow-assisted corrosion (1003 failures), and vibratory fatigue (475 failures) events
- The 636 fatigue failures are apportioned into 475 vibratory fatigue failures, 120 thermal fatigue failures, and 13 corrosion fatigue failures; 28 failures had no second-level failure description

Operating Plant Failure Data

- All of the corrosion fatigue events were detected within the first 13 years of plant operation (4 in BWRs and 9 in PWRs)
- Similarly, the majority of the thermal fatigue events were detected within the first ten or eleven years of operation, caused by unanticipated transients not included in the design basis
- Similarly, most vibratory fatigue events were detected within the first fifteen years of plant operation, with low event frequency since 1985
- Systems affected by corrosion fatigue were component cooling water (2), feedwater (2), RCS (2), core spray (1), safety injection (1), service water (1), small instrument line (2), steam line (1), and containment cooling (1)

Operating Plant Failure Data Result

- **Available U. S. failure data for nuclear power plant piping components does not support the observations that would be reached from the conservative (trickle flow) laboratory data – there is no trend of accelerating corrosion fatigue failures in U. S. operating nuclear power plants**
  - In general, the laboratory fatigue data under simulated reactor water environmental conditions fall within the margins already accounted for by a portion of the ASME Code factor of 20
  - The effect of reactor water flow rate further mitigates against potential environmental fatigue failures
  - The ASME Code fatigue design procedures and industry practice with respect to design-basis transient definitions are sufficiently conservative to more than compensate for reactor water environmental effects

Data Evaluation Conclusion

- **Data are consistent with and support the revised probabilities of through-wall cracking (and leakage) for component locations evaluated in NUREG/CR-6674**
  - Laboratory test data obtained under simulated reactor water environmental conditions show that almost all data for carbon and low-alloy steels are within margins in ASME Code fatigue design curve
  - Structural/component fatigue test data with one surface in contact with water environment showed behavior accounted for by margins in ASME Code fatigue design curve
  - Operating experience has not shown significant increases in fatigue failures ascribed to reactor water environmental effects as a function of increasing length of service of nuclear power plant components

Aging Management of Environmental Fatigue for Carbon/Low-Alloy Steels

Wrap-up

Michael R. Robinson
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September 18, 2002
NRC Headquarters

Enclosure 6

Summary

- NUREG/CR-6674 probabilistic calculations showed that, even with essentially bounding assumptions, the number of carbon/low-alloy steel component locations at which fatigue cracks would initiate and grow to a significant size are very few
- Industry recalculations of these probabilities, based on more realistic assumptions, show two to five orders of magnitude reduction of crack initiation and through-wall cracking probabilities
- U. S. nuclear power plant piping failure data do not exhibit a clear trend supporting environmental fatigue concerns
- Review of laboratory and component/structural test data show that no explicit treatment of reactor water environmental effects is needed for carbon/low-alloy steel components

Conclusions

- MRP evaluation concludes consideration of environmental fatigue effects for carbon and low-alloy steel components, as stipulated in GSI-190 closeout memorandum, is not warranted
- All carbon/low-alloy steel 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
- MRP evaluation of austenitic stainless steel locations is continuing

Actions

- **MRP will provide draft Interim Staff Guidance**
  - ISG to include technical basis
    - Re-analysis of NUREG/CR-6674 (MRP-74)
    - Environmental data review (MRP-49)
  - ISG to include changes to GALL and SRP
  - ISG to be provided by December 31
- **Coordinate revision to near-term guidance provided in MRP-47**
  - Deferral of RAI process pending ISG and ongoing austenitic stainless steel activities