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Review of Degradation Mechanisms in EPRI Risk - Informed Inservice Evaluation Procedure

Prepared for:

Electric Power Research Institute

Prepared by:

Structural Integrity Associates, Inc.

Reviewers:	N. G. Cofie	Date: 11/26/96
	A. F. Deardorff	Date: 11/26/96
	A. J. Giannuzzi	Date: 11/26/96
	for G. J. Licina	Date: 11/26/96

ATTACHMENT 5

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1.0 INTRODUCTION

This report documents the findings of an independent third party review performed by Structural Integrity Associates (SI) on the degradation mechanisms considered in the EPRI risk-informed in service inspection procedure for application to nuclear power plant piping components. The degradation mechanisms and other aspects of the EPRI risk-informed process are discussed in EPRI-TR-106706 [1]. Recognizing the wide range and different forms of degradation mechanisms associated with nuclear plant piping, a team of experts with various backgrounds in the nuclear industry were assembled to review the various attributes of the degradation mechanisms considered in Reference 1 for accuracy, completeness and applicability. The resumes of the reviewers are presented in Appendix A of this report and their expertise is briefly summarized below.

N. G. Cofie, Ph.D. - Over 15 years of experience in the nuclear industry. Expert in fatigue, stress and fracture mechanics analysis. Active participant on ASME Section XI on issues relating to pipe and vessel flaw evaluation.

A. F. Deardorff, M. S., P. E. - Over 20 years experience in the nuclear industry. Expert in thermal hydraulics, fatigue, stress and fracture mechanics. He is well known in ASME Section XI for its contributions on fatigue, pipe wall thinning and vessel integrity issues. He was an active participant in the preparation of the EPRI fatigue management handbook and the EPRI TASCS program.

A. J. Giannuzzi, Ph.D. P.E. - Over 20 years experience in the nuclear industry. Expert in corrosion, metallurgy, material selection, repair and replacement. Authority on intergranular stress corrosion cracking (IGSCC) and various mitigating practices to prevent IGSCC. Active participant in ASME Section XI on repair of degraded piping components.

G. J. Licina, BS, - Over 20 years experience in the nuclear industry. Expert in corrosion, metallurgy and material selection. Authority on microbiologically influenced corrosion (MIC) and other forms of localized corrosion such as pitting.

Appendix B contains a copy of Table 4-1 from TR-106706, listing all of the degradation mechanisms considered in the document, the criteria for assessing whether the mechanism is potentially active for the piping system being evaluated, and the materials, product forms, or specific locations where the mechanism is likely to be operative. The list was constructed from the listing of pipe failures in Commercial U.S. Nuclear Power Plants compiled by Jamali [2, 3]. The list is presented in a slightly different form in Appendix C.

2.0 TECHNICAL APPROACH

The independent review was based upon the degradation mechanisms that are known to affect piping systems in nuclear plants using a similar "coarse screen" as was constructed for EPRI report NP-5461 [4]. The review thus provided a look at the degradation mechanisms "from the ground up" and incorporated expert experience in dealing with the gamut of degradations that occur in nuclear plants. The review also evaluated the listing in EPRI TR-106706 based upon its implementation to selected systems at ANO, the PWR pilot plant in the Risk Informed Inspection study. The utility's attempts to apply the list and criteria to their plant systems indicated that some categories were too broad, some criteria were unclear, and that the only mechanism deemed capable of producing a large leak was Flow Accelerated Corrosion (FAC). This review also compared the EPRI TR-106706 results to those from the so-called "B-J Code Case", which has been approved by ASME Code, Section XI and has been subjected to a very extensive review by the industry at large. That Code Case had similar objectives to the work described in both NP-5461 and TR-106706. Finally, the reviewers interacted on several occasions with EPRI, Sartrex (one of the preparers of EPRI TR-106706) and personnel from one of the on-going pilot plant studies (ANO) soliciting their input so as to produce a list of degradation mechanisms that had some level of agreement from all of the involved parties.

The method provided in TR-106706 for evaluating the applicability of the listed mechanisms are essentially binary in nature. That is, a degradation mechanism is considered potentially operative or not; no "shades of gray" are permitted. Because of the binary nature of the degradation mechanism assessment, the criteria provided should be, and in most cases, are conservative to assure that a potentially operative mechanism will not be overlooked during the initial screening. We find this approach to be reasonable and meaningful for an initial screening of the degradation mechanisms, where the primary objective is completeness.

Criteria and methodology that consider all components in a system and all of the potential operating conditions, both normal and off-normal, would render that first cut or coarse screening too complicated. As such, our evaluation of the criteria took the binary nature of the procedure at this first level into consideration.

The EPRI process, however, goes beyond this binary approach in the element selection process in that a more "continuous" type process is used to select the most susceptible locations for inspection after the initial degradation mechanism and consequence evaluation. This approach was found to be very reasonable and practical because it limits the degree of damage mechanism assessment on the "front end" of the evaluation for systems or subsystems where those mechanisms will eventually be found to be inconsequential to failure after the initial evaluation. However, the approach allows the inspection to be focussed at areas of a system that are considered the most susceptible to any identified damage mechanism. The review also took into consideration this element selection aspect of the EPRI risk-informed ISI process.

3.0 RECOMMENDATIONS

On the whole, the review by the various participants concluded that the degradation mechanisms outlined in EPRI TR-106706 capture most of the active mechanisms that potentially affect nuclear power plant piping. A few additional observations are discussed below with respect to thermal fatigue, vibrational fatigue and corrosion-related mechanisms. In addition, a reorganization of some of the mechanisms is proposed.

3.1 Thermal Fatigue

The thermal fatigue section of the degradation mechanisms in EPRI-TR-106706 was appropriately subdivided into thermal stratification, cycling and striping (TASCS), and thermal transients. We believe that this subdivision is appropriate to distinguish the relatively "low cycle" thermal transient events that have typically been designed for in the piping Stress Reports and the TASCS events that are associated with high cycles and generally not accounted for in the original piping design. The basis for the proposed TASCS criteria is that from the EPRI fatigue management handbook [5] and the EPRI TASCS report [6]. These criteria are judged to be sound. Based on input received from the ongoing pilot programs, the criteria in this section were clarified, taking guidance from the fatigue management handbook.

We find the section on thermal transients to be reasonably conservative for this binary approach. During the element selection process, however, the user can use the severity of the thermal transients and the frequency of occurrence to determine the most susceptible locations to inspect and this could, in fact, provide the basis for eliminating some thermal transients as potential sources of degradation.

We find the criteria based solely on the temperature differential for the thermal transients acceptable. It is believed, however, that there may be a few cases where there is potential for hot water injection or reflood into a cold component which will initially result in compression stress on the inside of the pipe. Fatigue usage is insensitive to the sign of the stress. Moveover, if the temperature is large enough, this will cause compressive yielding which will subsequently result in tensile residual stresses on the inside of the pipe once the temperature differential is removed. This, of course, will involve a much higher ΔT than that currently specified in EPRI TR-106706. For this binary approach, we recommend that the ΔT specified for the thermal transient be converted to absolute numbers to account for the possibility of inside surface tensile stresses developing from a hot fluid on a cold pipe. This approach is conservative for the initial screening. However, the user may choose to use the actual temperature differential to provide realistic assessment of the level of compressive stresses compared to the yield strength of the component.

3.2 Vibrational Fatigue

Though a very common failure mechanism in nuclear power plant piping, vibrational fatigue was not specifically made part of the evaluation process in the EPRI risk-informed procedure. Most documented vibrational fatigue failures in power plant piping, however, indicate that they are restricted to socket welds in small bore piping (less than 2 inch nominal pipe size) which does not fall under current ASME Section XI volumetric inspection programs. Vibrational fatigue failures have not been observed in large bore piping welds. It is also well documented that most of the damage in vibrational fatigue failures occurs in the initiation phase and once a crack forms, the propagation is so fast that failure of the component can occur very rapidly. As such, vibrational fatigue failures cannot be avoided by the risk-informed ISI process being considered by EPRI or for that matter by any ISI program. We, therefore, agree with the observation in the EPRI TR-106706 that vibrational fatigue should not be included in the risk-informed program but that it should be treated as an entirely separate program taking guidance from the work documented in the EPRI fatigue management handbook.

3.3 Corrosion-Related Mechanisms

The majority of mechanisms listed in EPRI TR-106706 are related to corrosion. We believe that most of the corrosion-related mechanisms that are associated with nuclear plant piping components have been included in the report. However, for completeness, three additional degradation mechanisms were considered for possible inclusion into the list of mechanism. These mechanisms are pitting, general corrosion and galvanic corrosion. After careful consideration of these mechanisms, only pitting was included in the potential list of mechanisms as discussed below.

3.3.1 Pitting

Even though this mechanism had presumably been considered under TR-106706 as part of microbiological influenced corrosion (MIC), pitting can occur without the presence of living organisms or organic material. This mechanism should be included in the list of potential degradation mechanisms and can be combined with MIC and crevice corrosion under one general title called "Localized Corrosion".

3.3.2 General Corrosion

General corrosion occurs in ferritic piping and results in an essentially uniform wall loss around the circumference of the pipe. Though this is an active mechanism for ferritic piping, we consider it to be too broad to be included in the EPRI risk-informed process. That is, general corrosion will be operative for **all** ferritic piping, thus, including it as a degradation mechanism would not really accomplish anything. In general, Class 1 piping is designed with an allowance for general corrosion. The major concern for a risk-informed ISI program would be to demonstrate that that corrosion allowance is adequate. Examinations of piping for pitting, crevice corrosion and flow assisted corrosion (FAC) would uncover general corrosion if it occurs. As such, we believe that no special treatment of this mechanism is further required in the EPRI risk-informed process.

3.3.3 Galvanic Corrosion

Galvanic corrosion occurs as a result of the potential difference developed in a conductive solution if two dissimilar materials are in contact either physically or through an external electrical circuit. The potential difference produces electron flow between the two materials and the corrosion rate for the less corrosion resistant material will be increased. Galvanic corrosion occurs in the vicinity of connection of carbon steel with stainless steel or other more noble metals in conducting solutions and as such, it is a potentially achieve mechanism for piping systems with dissimilar metal joints. Though an active mechanism, we believe that galvanic corrosion will be adequately addressed under other forms of localized corrosion since the susceptible regions for localized corrosion due to galvanic effects addresses fittings, welds, heat affected zone (HAZ), base metal and dissimilar metal joints. Hence no further specific evaluation is required for this mechanism.

3.3.4 Rearrangement of Corrosion-Related Mechanisms

In order to streamline all the corrosion-related mechanisms, a new organization of these mechanisms is proposed. This new organization is somewhat consistent with that used in ASME Code Case N-560 which was recently been approved by the ASME Boiler and Pressure Vessel Code Committee for Category B-J welds in class 1 piping. This Code Case has similar objectives to the EPRI risk-informed ISI process and has gone through extensive review by the industry.

Corrosion cracking, primary water stress corrosion cracking and intergranular stress corrosion cracking (from Table 4-1 of EPRI TR-106706) are proposed to be combined under one general topic called "Stress Corrosion Cracking" with the following subsections:

- Intergranular Stress Corrosion Cracking (IGSCC) -BWR
- IGSCC - PWR
- Transgranular Stress Corrosion Cracking (TGSCC)
- External Chloride Stress Cracking Corrosion (ECSCC)
- Primary Water Stress Corrosion Cracking (PWSCC)

A gene . category called "Localized Corrosion" is proposed with the following subsets:

- MIC
- Pitting
- Crevice Corrosion

4.0 RESOLUTION OF COMMENTS AND CONCLUSION

Based on the observations made in Section 3.0 of this report, several discussions were held among EPRI, SARTREX and some of the participants in the on-going pilot plant studies. Based on these discussions, and valuable input from these organizations, a final list of degradation mechanisms to be considered in the EPRI risk-informed process was established. This list shown in Appendix D of this report includes the resolution of all the issues discussed in Section 3 of this report.

This table, though different in arrangement, is not very different in content than the original list of mechanisms in Table 4-1 of TR-106706. The only new mechanism that appears in this new table is pitting which for most practical purposes was covered by MIC. All other mechanisms listed in this new table were also covered in Table 4-1 of TR-106706, albeit under a different general heading.

The attributes for the mechanisms remain similar to those identified in EPRI TR-106706. Additional clarification is provided to differentiate between oxidizing conditions and initiating contaminants which tend to exacerbate the corrosion phenomenon and potentially produce corrosive effects outside the range of parameters generally accepted for oxidizing conditions alone.

The Criteria and Susceptible Regions sections of the revised table are very similar to those in EPRI TR-106706 and are intended to be sufficiently general that all potentially active degradation mechanisms are considered in the binary, coarse screening but not so broad that degradation will be identified where it is unlikely to occur. The specific operating conditions for the system provide especially important information for comparison to the criteria defined for corrosion mechanisms.

We conclude from this review that the mechanisms contained in EPRI TR-106706 are adequate for the purpose of the risk-informed ISI process. The new recommended table agreed upon with all participants of this project provides only an enhanced rearrangement of these mechanisms and does not materially change any of the attributes and criteria. Hence, even though some of the on-going pilot studies have been performed under the TR-106706 table, the use of the new recommended list of mechanisms will not invalidate any of the completed work performed under the original TR-106706 table.

5.0 REFERENCES

1. Gosselin, S. R., et al., "Risk-Informed Inservice Inspection Evaluation Procedure", EPRI-TR-106706, June 1996.

2. Jamali, K., "Pipe Failures in U.S. Commercial Nuclear Power Plants", EPRI TR-100380, Electric Power Research Institute, Palo Alto, CA, July 1992.

3. Jamali, K., "Pipe Failures Update Study", EPRI TR-100380, Electric Power Research Institute, Palo Alto, CA, April 1993.

4. Copeland, J. F., et al., "Component Life Estimation: LWR Degradation Mechanisms", EPRI NP-5461, September 1987.

5. Riccardella, P. C., et al., "EPRI Fatigue Management Handbook", EPRI TR-104534, Electric Power Research Institute, Palo Alto, CA, December 1994.

6. Roarty, D.H, et.al, "Thermal Stratification, Cycling, and Striping (TASCS)", EPRI TR-103581, Electric Power Research Institute, Palo Alto, CA, March 1994.

APPENDIX A

Resume's of Reviewers

N. G. Cofie
A. F. Deardorff
A. J. Giannuzzi
G. J. Licina

Dr. Nathaniel G. Cofie

Associate

Education

BS, Civil Engineering, University of Science & Technology (1975)
MS, Civil Engineering, Stanford University (1977)
Degree of Engineer, Stanford University (1979)
PhD, Civil Engineering, Stanford University (1983)

Professional Associations

Member - American Society of Mechanical Engineers (ASME)
Member - ASME Section XI Working Group on Pipe Flaw Evaluation

Professional Experience

1990 to present	Structural Integrity Associates, San Jose, CA Associate
1981 to 1990	NUTECH, San Jose, CA Staff Consultant
1979 to 1981	Stanford University, Palo Alto, CA Research Assistant
1977 to 1979	URS/John A. Blume & Associates, San Francisco, CA Engineer
1975 to 1976	University of Science & Technology, Ghana Research Engineer

Summary

Dr. Cofie has been involved in engineering for nuclear power plant components and conventional structures since 1975. He is an expert on the inelastic modeling of materials for structural applications. He has considerable experience in the application of finite element analysis, fracture mechanics, leak-before-break analysis and fatigue analysis. He is well versed in the requirements of the ASME, AISC, ACI and UBC codes.

While at NUTECH, Dr. Cofie was the technical leader of the fracture mechanics group. He was involved with several aspects of stress corrosion problems in the BWR industry, including development and implementation of induction heating stress improvement (IHSI) and weld overlay repairs design and implementation.

He worked on leak-before-break analyses, feedwater nozzle cracking evaluations, and fatigue analyses of several nuclear power plant components. Before joining the fracture mechanics group, Dr. Cofie worked in the structural engineering group as a project engineer and individual contributor on several projects. Examples are BWR Mark I hydrodynamic and seismic loads evaluation of the vent system and suppression chamber, finite element analyses of reactor vessel and high pressure injection nozzles, structural analyses of flued head penetrations, and structural evaluation of spent fuel casks and canisters. He also served as project engineer in charge of several test programs, including a program to evaluate materials to be used in the design of a tension leg platform for off-shore structures.

As a research assistant at Stanford University, Dr. Cofie was involved with several projects in dealing with full-scale and component testing of structures under severe inelastic cyclic loading. He was also involved with various material testing programs.

As an engineer with URS/John A. Blume & Associates, Dr. Cofie was involved with development of finite element models for static and dynamic analyses of piping systems for nuclear power and chemical plants. He also gained experience in the structural analysis and design of pipe supports.

As a research engineer at the University of Science & Technology, Dr. Cofie participated in the design of various reinforced concrete, steel and timber structures. He performed geotechnical analysis and design of building foundations and retaining walls. He also served as a teaching instructor in structural analysis and design.

Arthur F. Deardorff, P. E.

Associate

Education

BS, Mechanical Engineering, Oregon State University (1964)
MS, Mechanical Engineering, University of Arizona (1966)

Professional Associations

Registered Mechanical Engineer, State of California
American Nuclear Society
American Society of Mechanical Engineers
ASME Section XI Subcommittee
- Member - Working Group on Erosion-Corrosion
- Member - Task Group on Fatigue in Operating Plants
- Member - Task Group on Implementation of Risk-Based Inspection
- Member - Sub-Group Water Cooled Systems

Professional Experience

1987 to present	Structural Integrity Associates, San Jose, CA Associate
1976 to 1987	NUTECH, San Jose, CA Supervising Engineer
1970 to 1976	General Atomic Company, San Diego, CA Senior Engineer
1966 to 1970	The Boeing Company, Seattle, WA Engineer

Summary

Mr. Deardorff has been involved in specification, design, analysis and testing of nuclear power plant systems and structures since 1970. At Structural Integrity Associates, he has been actively involved in projects related to fatigue monitoring, fatigue and fracture mechanics, erosion-corrosion, thermal-hydraulics, expert systems, ASME Section III design analysis, and other related topics. He has made major contributions in developing ASME Section XI methods and criteria for evaluating thinned piping and for assessing fatigue in operating nuclear plants. He is actively involved in other Section XI committee activities relating to inspecting and evaluating nuclear plant power plant components and systems. He has directed several fatigue monitoring projects and had developed many of the enhancements to the **FatiguePro** fatigue monitoring system. He has consulted to the Electric Power Research Institute in several major fatigue-related projects.

Prior to joining SI, he was involved in design, analysis and testing of nuclear plant vessels, piping and containment systems. He performed evaluations of piping systems for effects of intergranular stress corrosion cracking and performed testing and analysis to develop new Induction Heating Stress Improvement (IHSI) techniques. He developed methodology for predicting leakage through pipe cracks, and has performed leak-before-break evaluations. He has also performed loading and structural evaluations of containment structures for hydrodynamic effects. In the late 70's, he was deeply involved in the containment structure analysis and testing and in formulation of the industry's approach to resolution of the Boiling Water Reactor Mark I Containment issue.

Mr. Deardorff's areas of expertise lie in the areas of fracture mechanics, fluid mechanics and heat transfer, stress analysis, dynamics, ASME Boiler and Pressure Vessel Code applications, and reactor systems evaluation, with a strong academic background in thermal-hydrauiics and fluid system. He has a good understanding of both Pressurized and Boiling Water Reactor systems and structures and has been involved in several projects related to fossil-fired power plants. Over the years, he has developed the reputation of being able to provide practical engineering solutions to complicated problems involving mechanical/structural integrity issues.

Dr. Anthony J. Giannuzzi, P. E.
Associate

Education

BS, Physics, LeMoyne College (1964)
MS, Solid State Science and Technology, Syracuse University (1967)
PhD, Solid State Science and Technology, Syracuse University (1969)

Professional Associations

Professional Corrosion Engineer, State of California
American Society of Mechanical Engineers
ASME Section XI Subcommittee
    Member - Working Group on Welding and Other Special Processes

Professional Experience

1983 to present	Structural Integrity Associates, San Jose, CA Vice President
1979 to 1983	Electric Power Research Institute, Palo Alto, CA Project Manager
1978 to 1979	NUTECH, San Jose, CA Project Manager
1972 to 1978	General Electric Company, San Jose, CA Principal Engineer
1969 to 1972	Aerojet Nuclear Systems Company, Sacramento, CA

Summary

Dr. Giannuzzi has been involved in solving materials and corrosion problems for the nuclear industry since 1969. One of the world's leading authorities on intergranular stress corrosion cracking of stainless steel in aqueous systems, Dr. Giannuzzi was employed by the Electric Power Research Institute in the Nuclear Systems and Materials Department for three-and- one-half years prior to joining Structural Integrity Associates in 1983. At EPRI, Dr. Giannuzzi was task leader and principal investigator involved in development and qualification of all the Boiling Water Reactor IGSCC piping remedies. This activity included primary responsibility for qualifying and producing material specification for the alternative materials (Types 316NG and 304NG stainless steels), qualifying the induction heating stress improvement (IHSI) remedy, qualifying heat sink welding, last pass heat sink welding and the weld overlay, and performing the investigations to determine the causes of and remedies to IGSCC in Type 304 stainless steel piping.

In addition to his BWR IGSCC responsibility at EPRI, Dr. Giannuzzi has had the lead responsibility for investigating the causes of low pressure large steam turbine stress corrosion cracking in nuclear and fossil steam turbines and has been involved in projects associated with bolt and fastener reliability, steam and water piping erosion-corrosion and has been active in projects related to primary-side and secondary-side corrosion of steam generators. Dr. Giannuzzi has also been the lead project manager responsible for all materials-related failure analysis activities in the Nuclear Systems and Materials Department and was a member of the EPRI Three Mile Island Unit 2 task force.

Prior to his employment at EPRI, Dr. Giannuzzi was employed as a senior consultant at NUTECH. While at NUTECH, he formed the stress corrosion cracking group and developed the methodology used to estimate likely locations of IGSCC in stainless steel piping systems. He also was involved in the earliest investigations involving PWR boric acid corrosion and assisted in the final formulation of the NRC I-E Bulletin 79-02 which established criteria for inspection of the boric acid system piping.

From 1972 to 1978, Dr. Giannuzzi worked as a principal development engineer at the General Electric Company Nuclear Energy Division. His responsibilities while at GE involved investigation of alternative materials and processes to alleviate the IGSCC problem in stainless steel piping. He managed the initial weld residual stress measurement and analyses activities which lead to the development of the residual stress remedies to IGSCC.

From 1969 to 1972, Dr. Giannuzzi worked for the Aerojet Nuclear Systems Company developing materials for use in the nuclear rocket engine (NERVA).

In 1983, Dr. Giannuzzi founded Structural Integrity Associates with Dr. P. C. Riccardella and Dr. T. L. Gerber. His activities at Structural Integrity have included nuclear plant life extension studies, temper bead welding development on low alloy steels, and selecting of remedies to IGSCC in BWRs. Dr. Giannuzzi is a member of the ASME Section XI Working Group on Welding and Other Special Processes and has chaired a Task Group on Alternative Repair Methods for Erosion-Corrosion Damage in Carbon Steel Piping. He is currently chairman of a Task Group on "Laser Welding of Steam Generator Tubes" for structural repairs of SG tubes.

George Licina
Associate

Education

BS, Metallurgical Engineering, University of Illinois (With High Honors)
Graduate Work, Materials Science, San Jose State University

Professional Associations & Awards

Alpha Sigma Mu - Metallurgy Honorary Society
Tau Beta Pi - Engineering Honorary Society
Patent No. 4166019 - Electrochemical Oxygen Meter
Patent No. 4139421 - Method of Determining Oxygen Content
Patent No. 5246560 - Apparatus for Monitoring Biofilm Activity
General Manager's Award - General Electric Company, Advanced Nuclear Systems Technology Operation, 1984

Professional Experience

1986 - present	Structural Integrity Associates, San Jose, CA Associate
1972 - 1986	General Electric Company, San Jose, CA Senior Engineer

Summary

Mr. Licina's experience at Structural Integrity Associates has dealt primarily with the degradation and environmental compatibility of power plant materials under a variety of operating conditions. These degradation mechanisms include corrosion and environmentally assisted cracking in BWR, PWR, and various raw water environments and embrittlement of pressure vessel steels and high performance alloys. Mr. Licina is a recognized authority on microbiologically influenced corrosion and has authored reference documents on that topic for the Electric Power Research Institute and numerous utilities. Plant-specific activities include metallurgical and fracture mechanics evaluations of nuclear steam generators, heat exchangers, valve stem cracking, BWR pipe replacement, and irradiation embrittlement of reactor pressure vessels, and the use of electrochemical methods for predicting and monitoring corrosion in power plant environments.

Mr. Licina has also integrated technical and regulatory requirements into guidelines for field certification of materials in nuclear plants and developed a methodology and approach for nuclear life extension issues.

He has authored more than thirty publications in these technology areas and is the author of two patents involving the determination of oxygen levels in liquid sodium systems and a third for an on-line method for monitoring biofilm activity in cooling water environments.

Mr. Licina served as lead engineer and program manager on a number of important development programs at the General Electric Company including:

- Control Rod Blade Surveillance and Lifetime Evaluation
- Stress Corrosion Cracking of Cr-Mo Steels
- Carbon Transport Effects on Steels in Liquid Sodium Systems

APPENDIX B

Degradation Mechanisms Table Presented in EPRI TR-106706

Table 4.1
Degradation Mechanisms

Degradation Mechanism	Criteria	Susceptible Regions
Thermal fatigue		
(a) Thermal stratification, cycling, stripping (TASCS)	(a) Areas where hot and cold fluid can mix where: Operating temp > 220°F(CS) or 270°F(SS), NPS >1 inch Vertical rise <45°F, and ΔT> 50°F or Richardson number > 4.0	(a) Nozzles, branch pipe connections, safe ends, welds, heat-affected zones (HAZ), base metal, regions of stress concentration
(b) Thermal transient	(b) Operating temp > 220°F(CS) or 200°F(SS), and ΔT> 150°F(CS) or 200°F(SS), and ΔT > T allowable	(b) Nozzles, branch pipe connections, safe ends, welds, HAZ, base metal, regions of stress concentration
Corrosion cracking		
(a) Chloride cracking	(a) Areas exposed to chloride contamination where temperatures >150°F and tensile stresses	(a) Base metal, welds and HAZ
(b) Crevice corrosion cracking	(b) Areas that contain crevices that can result in oxygen depletion and concentration of impurities	(b) Base metal, welds and HAZ
Primary water stress corrosion cracking	· Mill annealed Alloy 600 · Cold worked or cold worked and welded · Exposed to primary water temperature greater than 620°F	Nozzles, welds, HAZ without stress relief, thermowells

Table 4.1 (cont.)
Degradation Mechanisms

Degradation Mechanism	Criteria	Susceptible Regions
Intergranular stress corrosion cracking (IGSCC)		
(a) IGSCC - BWRs	(a) Generic Letter 88-01	(a) Austenitic steel welds and HAZ
(b) IGSCC - PWRs	(b) High oxygen, stagnant flow	(b) Austenitic steel welds and HAZ
Microbiologically influenced corrosion (MIC)	· Presence or intrusion of organic material · Untreated water · Low flow · Operating temperatures of 20 to 120°F · pH <10	Fittings, welds, HAZ, and base metal, especially regions containing crevices
Erosion-cavitation	· $(p_d - p_v)/\Delta p < 5$, and V > 30ft/sec. and fluid temperature < 250°F · EPRI TR-10318, T2, provides additional guidance	Fittings, welds, HAZ, and base metal
Flow-accelerated corrosion (FAC)	Evaluated in accordance with plant FAC program	Evaluated in accordance with plant FAC program

APPENDIX C

Supplement to Degradation Mechanism Table Presented in EPRI TR-106706

DEGRADATION MECHANISM WORKSHEET
April 3, 1996

1. __THERMAL FATIGUE__ (TASCS & Transients):

Base metal and weld regions ARE NOT SUSCEPTIBLE to degradation from thermal fatigue if any of the criteria in both A and B are true:

A. Thermal Stratification, Cycling and Striping (TASCS)

There is no potential for low flow, and no pipe segments containing hot fluid connected to segments containing cold fluid, and no pipe segments connected to components containing steam,

or

The pipe segment has a slope > 45° from horizontal (or elbow into a VERTICAL pipe),

or

NPS < 1 inch,

or

The calculated or measured $\Delta T < 50$ F,

or

$R_i < 4$,

AND

B. Transients

The design or operating temperature $T < 270$ F (S.S.) or $T < 220$ F (C.S.),

or

There is no potential for cold water injection onto a hot component,

or

$\Delta T < 200$ F (S.S.),

or

$\Delta T < 150$ F (C.S.),

or

$\Delta T < \Delta T$ allowable (per Fatigue Handbook)

2. *STRESS CORROSION CRACKING (SCC):*

A. General SCC (Internal):

Welds and weld heat affected zones at the inner surface of austenitic stainless steel pipe ARE SUSCEPTIBLE to degradation from SCC if all of the following are true:

Stagnant, intermittent, or low flow,

and

Operating temperatures T > 120 F,

and

Water chemistry IS NOT monitored.

and

There is the potential for in leakage from connected systems containing brackish or untreated water, or there is a history of or potential for contamination by chlorides, fluorides, sulfides, etc.

B. Chloride SCC (External):

The outer surface of austenitic stainless steel pipe within 5D of probable leak paths (e.g. valves with stems) IS SUSCEPTIBLE to degradation from chloride SCC if the pipe in that region is covered with non-metallic insulation that IS NOT in compliance with Reg. Guide 1.36.

The outer surface of austenitic stainless steel pipe IS SUSCEPTIBLE to degradation from corrosion cracking if it is exposed to wetting from chloride bearing environments such as sea or brackish water.

C. Crevice SCC:

Regions that ARE SUSCEPTIBLE to crevice SCC include thermal sleeves as shown in Figures 5.2.2.2 & 5.2.2.3 of EPRI Report TR-106218.

3. *PRIMARY WATER STRESS CORROSION CRACKING (PWSCC)*

Inconel (alloy 600) IS SUSCEPTIBLE to degradation from PWSCC if both of the following are true:

The material is mill-annealed and either cold worked or cold and welded without stress relief,

and

Is exposed to primary water at T > 620 F.

4. *INTERGRANULAR STRESS CORROSION CRACKING (IGSCC)*

A. BWRs

Welds and weld heat affected zones in BWR piping ARE SUSCEPTIBLE to degradation from IGSCC if they are examined as part of the existing plant IGSCC program, which was defined in accordance with NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping'.

B. PWRs

The criteria in IE Bulletin 79-17, Rev. 1 are used to determine if welds and weld heat affected zones in austenitic stainless steel pipe in PWRs are susceptible to degradation from IGSCC. Accordingly, welds and weld heat affected zones in austenitic stainless steel pipe in PWRs ARE SUSCEPTIBLE to degradation from IGSCC if the following are true.

> The pipe segments are in stagnant oxygenated borated water systems [1].

and

> The carbon content of the austenitic steel pipe material is equal to or greater than 0.05 wt %, as determined form the material certification reports.

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1. The term "stagnant, oxygenated borated water system" refers to those systems serving as engineered safeguards having no normal operating functions and containing essentially air saturated borated water where dynamic flow conditions do not exist on a continuous basis. However, these systems must be maintained ready for actuation during normal power operation. Systems or portions of systems that are flushed at least once every three months need not be classified as stagnant for purpose of this evaluation.

5. *MICROBIOLOGICALLY INFLUENCED CORROSION (MIC)*

Carbon steel welds, weld heat affected zones and base metal, and austenitic steel welds and heat affected zones ARE SUSCEPTIBLE to degradation from MIC if the following are true:

The materials are in raw water systems, transport systems, or storage tanks, or other systems containing untreated water where

pH < 10,

and

There is low or intermittent flow, especially in regions of geometric discontinuities,

and

The operating temperature is between 20 F and 120 F

6. *EROSION-CAVITATION*

Regions within 5D downstream of throttling or pressure reducing valves or orifices, ARE SUSCEPTIBLE to degradation from erosion-cavitation if all the following are true:

Operating temperature < 250 F,

and

Flow > 100 hrs/yr (approximately 2% of plant operating time),

and

V > 30 ft/s,

and

$(P_d - P_v) / \Delta P < 5$, where

V = flow mean velocity at the inlet of the unit,
P_d = static pressure downstream of the unit,
P_v = vapor pressure, and
ΔP = pressure differential across the unit.

7. *FLOW ACCELERATED CORROSION (FAC)*

Regions in piping segments ARE SUSCEPTIBLE to degradation from FAC if they are examined as part of the existing plant FAC program, which was defined in accordance with NRC generic letter 89-08.

APPENDIX D

SI Recommended Table of Degradation Mechanisms

Revised Table 4.1 for EPRI TR-106706

Degradation Mechanism Criteria and Susceptible Regions

Degradation Mechanism		Criteria	Susceptible Regions
TF	TASCS	– nps > 1 inch, and – pipe segment has a slope < 45° from horizontal (includes elbow or tee into a vertical pipe), and – potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids, or potential exists for leakage flow past a valve (i.e., in-leakage, out-leakage, cross-leakage) allowing mixing of hot and cold fluids, or potential exists for convection heating in dead-ended pipe sections connected to a source of hot fluid, or potential exists for two phase (steam / water) flow, or potential exists for turbulent penetration in branch pipe connected to header piping containing hot fluid with high turbulent flow, and – calculated or measured $\Delta T > 50°F$, and – Richardson number > 4.0	nozzles, branch pipe connections, safe ends, welds, heat affected zones (HAZ), base metal, and regions of stress concentration
	TT	– operating temperature > 270°F for stainless steel, or operating temperature > 220°F for carbon steel, and – potential for relatively rapid temperature changes including cold fluid injection into hot pipe segment, or hot fluid injection into cold pipe segment, and – $\lvert\Delta T\rvert > 200°F$ for stainless steel, or $\lvert\Delta T\rvert > 150°F$ for carbon steel, or $\lvert\Delta T\rvert > \Delta T$ allowable (applicable to both stainless and carbon)	
SCC	IGSCC (BWR)	– evaluated in accordance with existing plant IGSCC program per NRC Generic Letter 88-01	austenitic stainless steel welds and HAZ
	IGSCC (PWR)	– operating temperature > 200°F, and – susceptible material (carbon content ≥ 0.035%), and – tensile stress (including residual stress) is present, and – oxygen or oxidizing species are present OR – operating temperature < 200°F, the attributes above apply, and – initiating contaminants (e.g., thiosulfate, fluoride, chloride) are also required to be present	
	TGSCC	– operating temperature > 150°F, and – tensile stress (including residual stress) is present, and – halides (e.g., fluoride, chloride) are present, or caustic (NaOH) is present, and – oxygen or oxidizing species are present (only required to be present in conjunction w/halides, not required w/caustic)	austenitic stainless steel base metal, welds, and HAZ

Table Legend

Thermal Fatigue (TF)
- Thermal Stratification, Cycling, and Striping (TASCS)
- Thermal Transients (TT)

Stress Corrosion Cracking (SCC)
- Intergranular Stress Corrosion Cracking (IGSCC)
- Transgranular Stress Corrosion Cracking (TGSCC)
- External Chloride Stress Corrosion Cracking (ECSCC)
- Primary Water Stress Corrosion Cracking (PWSCC)

Localized Corrosion (LC)
- Microbiologically Influenced Corrosion (MIC)
- Pitting (PIT)
- Crevice Corrosion (CC)

Flow Sensitive (FS)
- Erosion-Cavitation (E-C)
- Flow Accelerated Corrosion (FAC)

Revised Table 4.1 for EPRI TR-106706

Degradation Mechanism Criteria and Susceptible Regions

Degradation Mechanism		*Criteria*	*Susceptible Regions*
SCC	*ECSCC*	– *operating temperature > 150°F, and* – *tensile stress is present, and* – *an outside piping surface is within five diameters of a probable leak path (e.g., valve stems) and is covered with non-metallic insulation that is not in compliance with Reg. Guide 1.36, or an outside piping surface is exposed to wetting from chloride bearing environments (e.g., seawater, brackish water, brine)*	*austenitic stainless steel base metal, welds, and HAZ*
	PWSCC	– *piping material is Inconel (Alloy 600), and* – *exposed to primary water at T > 620°F, and* – *the material is mill-annealed and cold worked, or cold worked and welded without stress relief*	*nozzles, welds, and HAZ without stress relief*
LC	*MIC*	– *operating temperature < 150°F, and* – *low or intermittent flow, and* – *pH < 10, and* – *presence/intrusion of organic material (e.g., raw water system), or water source is not treated w/biocides (e.g., refueling water tank)*	*fittings, welds, HAZ, base metal, dissimilar metal joints (e.g., welds, flanges), and regions containing crevices*
	PIT	– *potential exists for low flow, and* – *oxygen or oxidizing species are present, and* – *initiating contaminants (e.g., fluoride, chloride) are present*	
	CC	– *crevice condition exists (e.g., thermal sleeves), and* – *operating temperature > 150°F, and* – *oxygen or oxidizing species are present*	
FS	*E-C*	– *operating temperature < 250°F, and* – *flow present > 100 hrs/yr, and* – *velocity > 30 ft/s, and* – $(P_d - P_v) / \Delta P < 5$	*fittings, welds, HAZ, and base metal*
	FAC	– *evaluated in accordance with existing plant FAC program*	*per plant FAC program*

Table Legend

Thermal Fatigue (TF) – *Thermal Stratification, Cycling, and Striping (TASCS)* – *Thermal Transients (TT)* *Stress Corrosion Cracking (SCC)* – *Intergranular Stress Corrosion Cracking (IGSCC)* – *Transgranular Stress Corrosion Cracking (TGSCC)* – *External Chloride Stress Corrosion Cracking (ECSCC)* – *Primary Water Stress Corrosion Cracking (PWSCC)*	*Localized Corrosion (LC)* – *Microbiologically Influenced Corrosion (MIC)* – *Pitting (PIT)* – *Crevice Corrosion (CC)* *Flow Sensitive (FS)* – *Erosion-Cavitation (E-C)* – *Flow Accelerated Corrosion (FAC)*

RAI Open Issues from Meeting on September 9-10, 1998

RAI No.	Issues Identified / Additional Information or Clarification Requested
1.0	More clearly define alternative requested to existing Code requirements (i.e., substitution of Code Case N-578 using EPRI TR-106706) and specify timeframe (i.e., remainder of 2nd Interval and 3rd Interval) for which its applicable; Clarify content of Table 1-1 including crediting of existing plant FAC program for MFW and MSS
2.0	More clearly define contents of Table 2-1 and provide basis for scope of systems considered
3.0	No changes identified
4.0	Provide a more explicit description of the "enhancements" utilized in the ANO-2 application versus EPRI TR-106706 and Code Case N-578
5.0	Editorial comment – change "risk analysis" to "RI-ISI" in last sentence
6.0	Clarify meaning of "initial screening" versus "full analysis"; justify exclusion from 'full analysis" based upon inclusion in existing FAC program and ensure consideration of other damage mechanisms
7.0	No changes identified
8.0	Review critical flooding water level in penetration room area; review pressure at which outside door latch is assumed to fail; review timing of valve motion; review whether door failures are credited for flood prevention elsewhere in the analysis
9.0	No changes identified
10.0	Editorial comment - remove everything after first sentence
11.0	Determine whether the ventilation dampers are credited as a flood prevention measure (propagation from General Access area to ECCS pump rooms) in any of the consequence evaluations
12.0	Provide Sketch of penetration room area and address impacts of pipe breaks in this area; verify 5 minute operator response time for breaks in ECCS pump rooms
13.0	Check recoveries for loss of SW system (T7)
14.0	No changes identified
15.0	No changes identified
16.0	Provide limited validation results
17.0	Editorial comment – in the 2nd paragraph replace "ANO-2" evaluation with "SW" evaluation
18.0	Provide limited validation results
19.0	Provide limited validation results
20.0	No changes identified
21.0	Document assumptions for support dependencies
22.0	~~No changes identified~~ revise SG isolation wording WDR 9/16
23.0	No changes identified
24.0	Describe controls for backup trains during SDC operation for midloop and other configurations; confirm SOPP meets RI-ISI assumptions
25.0	Evaluate industry water hammer events for applicability to ANO-2 using EPRI/NRC guidelines
26.0	No changes identified
27.0	Need input from staff

ATTACHMENT 6

RAI Open Issues from Meeting on September 9-10, 1998

RAI No.	Issues Identified / Additional Information or Clarification Requested
28.0	Editorial comment – in the "defense in depth" discussion clarify that "high" refers to the consequence category and not the risk rank; clarify that "poor performing equipment" refers to item with a high failure rate; clarify that high consequence or high failure potential always result in a risk significant segment
29.0	No changes identified
30.0	Better define review process utilized including; 1) function performed by plant project team in the performance and review of system calculations, and 2) independent, integrated plant review performed on completed system calculations; expand on meaning of "all aspects"; remove implication that EPRI guidelines are not sufficient to ensure identification of risk significant segments; reference internal plant validation/verification process
31.0	No changes identified
32.0	Identify any existing relief requests for dissimilar metal welds
33.0	No changes identified
34.0	No changes identified
35.0	No changes identified
36.0	Expand upon reliance on SIR-96-097 for damage mechanism assessment
37.0	No changes identified

FAX

Date 9/14/98

Number of pages including cover sheet 3

TO: BILL RECKLEY

Phone

Fax Phone 301-415-3061

FROM: MIKE PATERAK

Arkansas Nuclear One

Entergy Operations

1448 S.R. 333

Russellville, AR 72801

Phone (501) 858-4922

Fax Phone (501) 858-4685

CC:

REMARKS: ☐ Urgent ☐ For your review ☐ Reply ASAP ☐ Please Comment

BILL,

SORRY FOR THE DELAY.

MIKE

ANO-2 RI-ISI Questions
Clarifications provided by the NRR staff to the licensee by facsimile on 09/17/98

Below are the talking points regarding the two RAIs whose resolution was not clarified Thursday the 10'th with ANO-2.

Information for RAI 27 - Delta Risk Methodology

1) What process did ANO-2 itself rely upon to review the method? Did you bring to bear any of your own internal expertise to approve the methodology?

2) Request the executive summaries of the EdF benchmark and UM review.

3) Request that the licensee state that they have found the work to be of sufficient quality and accuracy to support the conclusions drawn in the submittal.

Information for RAI 12 - Human interactions

1) Please confirm that for all isolations credited that 1) there are Control Room alarms to which the operators respond by investigations or actions which would identify or confirm the leak, 2) the response and/or isolation is directed by procedure, and 3) the isolation manipulations can be taken from the control room.

2) Please confirm that the interaction between responding to an unrelated initiating event and an on-demand pipe rupture is considered during the evaluation to determine the appropriateness of crediting isolation as one full train.

ATTACHMENT 7