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Expert Panel Report on Proactive Materials Degradation Assessment

Brookhaven National Laboratory

U.S. Nuclear Regulatory Commission
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ABSTRACT

This study is part of the Nuclear Regulatory Commission's (NRC's) Proactive Materials Degradation Assessment (PMDA) program. The main objective was to identify materials and components where future degradation may occur in specific light water reactor (LWR) systems. The approach was to use a structured elicitation drawing on the knowledge of a panel of eight experts and the use of a Phenomena Identification and Ranking Table (PIRT) process. The international panel was given information on the materials, fabrication process, and operational environment for hundreds of different parts of systems in a Westinghouse four-loop design pressurized water reactor and a BWR-5 design boiling water reactor. They considered extensions to other designs as well. The panel developed metrics and used them to evaluate the susceptibility of given parts to different degradation mechanisms as well as the level of understanding for the varying degradation mechanisms for the given part. Inherent in arriving at these judgments of future behavior was an understanding of the prediction methodologies for the various degradation phenomena, calibrated by the component failures that have occurred in the past in the global LWR fleet. Also taken into account were the successes and limitations of mitigation/control approaches that have been used to date. This report includes not only the panel's scoring and their rationale for it, and the conclusions derived from this process, but also considerable documentation of the relevant issues, the latter being found in appendices.

FOREWORD

Today's approach to effective materials degradation management in nuclear power plants involves selecting appropriate materials for the design of components and monitoring these components for potential degradation. The regulatory requirements are identified in the Code of Federal Regulations for both component design and periodic in-service inspections to ensure that design safety margins are maintained throughout component life. Plant technical specifications also include requirements for leakage monitoring and reactor shutdown to provide defense-in-depth to ensure the integrity of the reactor coolant system boundary. Lastly, the NRC issues generic letters, bulletins, and orders to address emergent issues.

Notwithstanding this multifaceted regulatory framework, instances of unexpected materials degradation in nuclear power plants during recent years have led to a heightened interest by the nuclear power industry and the Nuclear Regulatory Commission (NRC) in developing a proactive approach to materials degradation management. To establish a proactive program, information is needed on which components and materials are expected to experience future degradation and by which degradation mechanisms. This report presents the results from a study conducted by the NRC to (1) identify reactor components that could reasonably be expected to experience future degradation, (2) estimate the susceptibility of components to various degradation mechanisms, and (3) assess the degree of knowledge available to develop mitigative strategies.

The research results presented in this report support the regulatory framework by identifying components that are susceptible to degradation. The study used a modified Phenomena Identification and Ranking Table (PIRT) process that embodied, as its central feature, the work of a panel of materials degradation experts from five countries. The study was facilitated by Brookhaven National Laboratory. The panel members' analyses focused on degradation modes associated with the operating environment of specific components. The study identified and evaluated more than 2000 representative components for a typical pressurized water reactor and a similar number for a typical boiling water reactor. The resulting report provides a wealth of information related to passive component materials and their operating environment, operating experience, and susceptibility to potential future degradation. This report also provides state-of-the-art information about materials engineering issues and various degradation mechanisms. As such, the report may be useful to the nuclear industry in implementing proactive materials degradation management (PMDM) programs.

PMDM can be achieved both by implementing actions to mitigate or eliminate materials' susceptibility to degradation, and by implementing effective and timely inspection, monitoring, and repair of susceptible materials. The evaluations of current knowledge and any research needs presented in this report apply only to PMDM via mitigation strategies. The question of whether the available knowledge has been used to develop or implement mitigation strategies was not explicitly addressed by this study. Further, separate NRC programs are addressing inspection effectiveness and the components' relative importance to risk. With this information, the agency will be better able to assess the need for any additional regulatory actions regarding PMDM.

Lastly, although recognized experts conducted this comprehensive study, reactor materials could experience some future degradation that this report does not identify. For example, even though the study considered several cascading degradation scenarios (e.g., boric acid corrosion of manway-retaining bolts caused by flange leakage), the panel concentrated on degradation of components without evaluating the resultant degradation of adjacent components.

Brian W. Sheron, Director
Office of Nuclear Regulatory Research

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EXECUTIVE SUMMARY

Background and Objective

The Nuclear Regulatory Commission (NRC) has undertaken a program to lay the technical foundation for defining proactive actions so that future degradation of materials in light water reactors (LWRs) is limited and, thereby, does not diminish either the integrity of important LWR components or the safety of operating plants. This study is timely since the majority of the U.S. reactor fleet is applying for license renewal, and many plants are also applying for increases in power rating. Both of these changes could increase the likelihood of materials degradation and underline, therefore, the interest in proactive management in the future.

The response of industry and regulators to issues of materials degradation in the past generally has been to develop and approve mitigation actions *after* the degradation has occurred. These mitigation actions have involved increases and improvements in in-service inspection, changes in designs, materials, operating conditions and replacement of degraded components. This reactive approach has maintained the safety of operating reactors, but has proved to be an inefficient and expensive way of managing materials degradation issues for the industry.

The objective of the present program is to identify early the components that are potentially susceptible to future degradation, so that mitigation and/or monitoring and repair actions can be proactively developed, assessed, and implemented before the degradation process could adversely impact structural integrity or safety. Although two processes can be envisioned for the Proactive Materials Degradation Management (PMDM) programs, a) implementation of actions to mitigate or eliminate the susceptibility to materials degradation, and b) implementation of effective inspection, monitoring, and timely repair of degradation, this study concentrated on an identification of components susceptible to future degradation and an assessment of the existing knowledge level for potential development of mitigation actions, i.e. process a) only. This study did not address whether mitigation actions have been developed where adequate knowledge exists, nor the effectiveness of any existing mitigation techniques. The Proactive Materials Degradation Assessment (PMDA) was conducted between August 2004 and August 2005. The first draft of this report was prepared in March 2006.

The impact of the reactive and proactive approaches on management of materials degradation is illustrated via Figure S.1. In the reactive management scenario there is a limited time window following the damage observation during which mitigation actions can be developed before an unacceptable amount of damage is accumulated. This constraint may lead to the deployment of incomplete mitigation strategies. The time constraint is considerably reduced in the proactive management scenario, with the increase in available time for mitigation development being a function of the incubation time before damage starts and the subsequent kinetics of damage accumulation. In order to meet the objective of the proactive management program it is necessary to assess the loci of the various damage vs. time relationships for the multitude of degradation modes, materials, environments, and operating states for the different light water reactor components. This assessment may be based on formulations for the various damage vs. time relationships or, more generally, on the basis of operating and laboratory experience and engineering judgment.

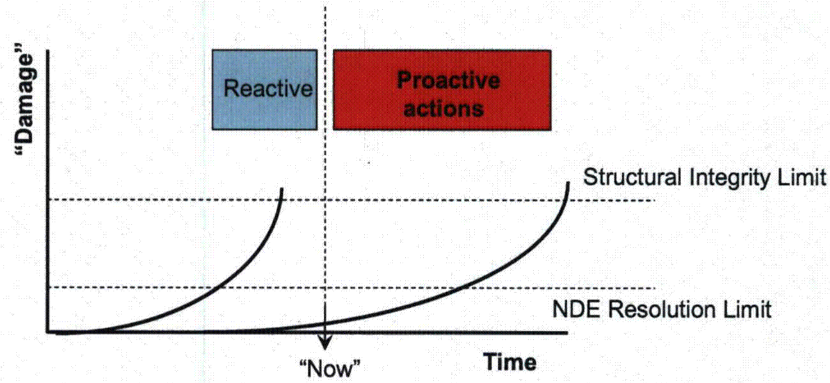

Figure S.1. Schematic diagram illustrating degradation, or damage, development with time, and the differentiation between reactive and proactive actions. Note that the degradation process vs. time is rarely linear, as is often assumed. (NDE=Non Destructive Examination)

Approach and Scope of Report

The overall approach to developing a proactive management capability for materials degradation involves two steps, the first being to identify the components of interest that might undergo future degradation; that is, a Proactive Materials Degradation Assessment (PMDA). The second step is to identify and perform the research projects where needed in order to develop a technical basis for PMDM programs; that is, develop effective mitigation strategies, in-service inspection and monitoring techniques, and repair procedures.

This report covers the PMDA step, which involved a panel of eight materials degradation experts (see Appendix C) from five countries who worked together for a period of about a year, starting in August 2004. The study was facilitated by Brookhaven National Laboratory (BNL) working under contract to the NRC. The panel used a Phenomena Identification and Ranking Table process (PIRT) as described in Section 2.1.

The panel members' analyses focused on degradation modes associated with the operating environment for specific components. A "component" was defined as a continuous section of a system that was of the same material and product form and was subjected to similar stressors (temperature, pressure, irradiation, residual stresses, water chemistry, etc.). The 16 major degradation modes considered are identified in Table S.1. It should be noted that only a few of these degradation modes were specifically addressed in the design codes for the current U.S. LWR fleet.

Table S.1 Degradation Modes considered in Proactive Materials Degradation Analysis

Abbreviation	Degradation Mechanism
BAC	Boric Acid Corrosion
CREEP	Thermal Creep
CREV	Crevice Corrosion (including denting)
DEBOND	De-bonding
EC	Erosion Corrosion Including Steam Cutting and Cavitation
FAC	Flow-accelerated Corrosion
FAT	Fatigue (corrosion/thermal/mechanical)
FR	Reduction of Fracture Resistance
GALV	Galvanic Corrosion
GC	General Corrosion
IC	Irradiation Creep
MIC	Microbially Induced Corrosion
PIT	Pitting Corrosion
SCC	Stress Corrosion Cracking (intergranular, transgranular, irradiation-assisted, strain-induced, hydrogen-embrittlement) and Intergranular Attack
SW	Swelling
WEAR	Fretting/Wear

The scope of the work of the panel encompassed passive components in the primary, secondary and some tertiary systems of Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs), the failure of which could lead to a release of radioactivity, or could affect the functionality of the safety systems. Degradation of such components has occurred in the past and has the potential to affect the economics, safety and public confidence in nuclear power. Examples of significant degradation occurrences that have affected some plants are intergranular stress corrosion cracking of piping, which dominated the loss in capacity factors for BWRs in the period 1982-87 and has since been mitigated (Figure S.2), and flow accelerated corrosion for which analytical tools that help to define in-service inspection programs have been developed (Figure S.3). These examples are not isolated incidents; other degradation modes requiring significant attention have occurred in the past, both in the US and abroad.

This study did not address human-performance aspects (which are often important in determining whether degradation is effectively managed), degradation of fuel cladding, turbines and generators, the consequences of degradation, the failure of active components (such as valves and control rod drives), or issues associated with the mechanics and thermal-hydraulic details of a specific reactor design. Finally, the evaluation of the various inspection techniques for detecting and quantifying the extent of degradation was outside the scope of this particular study. In order to apply results from this study in a regulatory framework, the above areas need to be addressed.

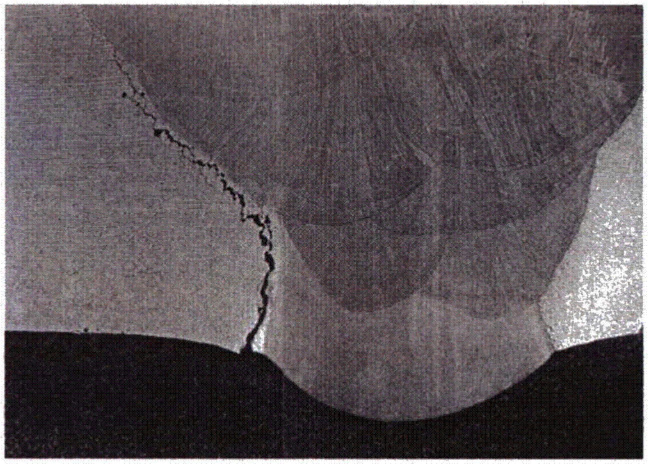

Figure S.2 **Intergranular stress corrosion cracking adjacent to a weld in 28"BWR recirculation pipe as has been observed in many BWR reactor coolant systems.**

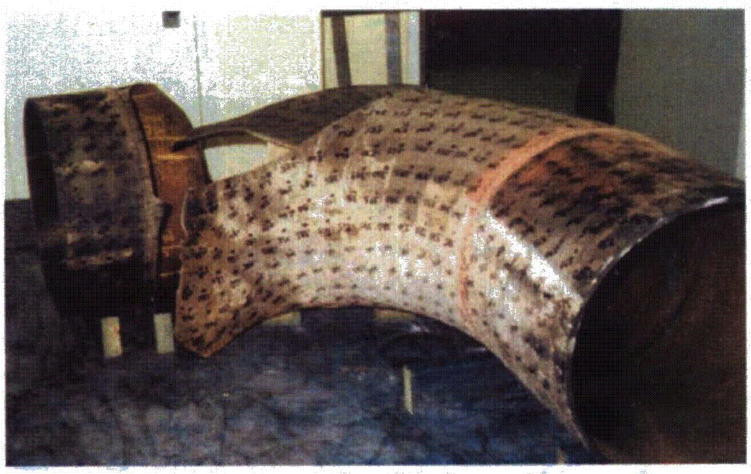

Figure S.3 **Failure by flow accelerated corrosion in the condensate line at Surry-2.**

Deliberations of the Panel

The review of materials degradation in the major systems in PWRs (Reactor Coolant, Emergency Core Cooling, Main Steam, Main and Auxiliary Feedwater, Service Water, etc.) focused on the components in a specific Westinghouse 4-loop design. This review was then extended to cover key differences in materials of construction and plant configurations in other PWR designs including those with once through steam generators (OTSG). An equivalent review of components in a specific General Electric BWR-5 was also performed with, again, an extension to cover other materials/configurations in other BWR designs.

Members of the panel provided their assessment for each combination of component and degradation mode in terms of:

- The degree of **susceptibility** to degradation (or the likelihood of occurrence) under specified system operating conditions. (Note that the use of the word "likelihood" does not imply that the assessments were based on probabilistic arguments)
- Their **confidence** in their assessment of susceptibility
- The extent of **knowledge** of the system interdependencies and, thereby, the predictive capabilities needed to mitigate or "manage" the degradation.

These assessments were quantified using a scoring of 1, 2, or 3, the details of which are discussed in Sections 2.5 and 3.1. For instance, a susceptibility rating of 3 would apply to a demonstrated, compelling evidence for occurrence, or multiple plant observations. Similarly, a knowledge scoring of 3 would be appropriate if there was a sufficiently extensive and consistent data base (preferably backed up by some level of mechanistic understanding) to allow the most important variables and interdependencies to be defined; this level of knowledge would permit a reasonably quantitative prediction of the occurrence of degradation (see Figure S.1) and the potential development of mitigation actions. By contrast, a scoring of 1 for this attribute would indicate that these variables and interdependencies were not quantified to the extent necessary to mitigate the situation. For "susceptibility to degradation", a further score of "0" was used when an expert thought there was no reasonable chance that a specific form of degradation would occur under the specified system operating conditions. (For example, microbiologically-influenced corrosion would be extremely unlikely under the high radiation field and high temperature conditions experienced by near-core reactor vessel internal components.)

To facilitate their judgments the panel members participated in a series of seven week-long meetings with staff from BNL, NRC's Office of Nuclear Regulatory Research and the NRC Technical Training Center. During these meetings the system configuration (i.e., materials of construction, fabrication process, engineering drawings) and operating conditions were defined, and the various degradation modes and operating experience for specific component groups were discussed. These information resources also included documents such as the GALL report, NUREG reports, NRC generic letters and industry guidelines; a full listing of such documents is given in Section 2.2 and Appendices E and F of this report. This information, in addition to the experience of the panel members themselves, led to judgments based on past history, projections into the future and identification of possible modes of degradation which have not yet been observed in service.

Analysis of the Scoring

Scores of susceptibility, confidence, and knowledge were provided by each member of the panel for all relevant (ranging from about three to ten) degradation modes for each of the components for PWRs and BWRs. Further, for each component-degradation mode combination each member provided comments to explain important factors associated with his/her scoring. This scoring (with commentary) was done individually, with no aim at arriving at a consensus opinion (see Appendices D, E.4 and F.4), although major differences in scoring were identified and discussed. Panel members were able to change their scores at any time throughout the study. The confidence and knowledge scores were then aggregated by taking an average of the scores for each component-degradation mode combination. In the case of the aggregation of the "degree of susceptibility" scores, both the statistical average and the statistical mode were considered, and care was taken to ensure that outlier (nonconforming) panel members' susceptibility scores were highlighted. A rule-based decision making process was instituted to ensure that the aggregate susceptibility scores for a given component-degradation mode combination were conservative.

These aggregated results were then used to identify the likelihood for future materials degradation of the various components and to indicate whether the degradation was potentially manageable from a mitigation point of view, or whether additional research would be required to provide this proactive mitigation. These judgments were summarized using a color diagram where the placement of the aggregated values of the degradation susceptibility and knowledge scores gave an indication of the degree of materials degradation mitigation capability for a given combination of component and degradation mode. Moreover, the average confidence score for that particular component/degradation mode combination reflected the uncertainty in arriving at that judgment. A detailed discussion on the development and use of the color diagrams is provided in Section 3.1.

Organization of the Report

The report is structured in the format of Introduction, Methodology, Results of PIRT Evaluation and, finally, Conclusions. However, given the complex decision making process inherent in arriving at these results and conclusions, the report is supplemented by a series of Appendices which describe the reasons behind some results and conclusions. In particular, Appendix A, "Materials Degradation Modes and their Prediction" contains a discussion of some of the corrosion fundamentals behind the degradation modes, for the benefit of readers relatively new to these subjects. Appendix B, "Background Papers," contains a series of reports that describe some of the important issues in more depth so that the reader can appreciate the technical factors behind the panel judgments, and, just as importantly, the associated uncertainties. Table S.2 lists the titles of the background papers included in Appendix B.

Table S.2 List of Background Papers on Specific Degradation Modes

B.1	SCC of Sensitized and Non-Sensitized Austenitic Stainless Steels and Weldments
B.2	IASCC of Stainless Steels and Other Irradiation Phenomena
B.3	SCC and Pitting in Contaminated External Environments
B.4	Thermal Aging and Embrittlement of Cast Stainless Steels
B.5	SCC of Ni Alloy 600 and Alloy 182 and 82 Weld Metals in BWR Water
B.6	SCC of Alloys 600, 690,182, 82,152 and 52 in PWR Primary Water
B.7	Corrosion of Steam Generator Tubes
B.8	Stress Corrosion Cracking of Carbon and Low Alloy Steels
B.9	Environmental Degradation of High Strength Materials
B.10	BWR Water Chemistry Guide; Effects on Materials Degradation and Industry Guidelines
B.11	PWR Primary Water Chemistry Guidelines
B.12	PWR Secondary Water Chemistry Guidelines
B.13	Degradation of Fracture Resistance; Low Temperature Crack Propagation (LTCP) in Nickel- Base Alloys
B.14	Fatigue
B.15	Predicting Failures Which Have Not Yet Been Observed- Microprocess Sequence Approach (MPSA)
B.16	Microbiologically Influenced Corrosion (MIC)
B.17	Flow-Accelerated Corrosion
B.18	Boric Acid Corrosion of Carbon and Low Alloy Steels
B.19	Variability in the Corrosion of Materials in LWR Environments

Analysis of Component-Specific Degradation

The results of the panel's evaluation of component susceptibility to materials degradation and of the knowledge level, are discussed in detail in Section 3.2 and 3.3 for PWR and BWR components, respectively. The Section 4 summary analysis of component degradation was divided between operating modes involving full power operation and non-steady state conditions that are associated with reactor start-up, water chemistry transients and extended plant lay-up. These analyses are summarized for various reactor systems, emphasizing those situations where there is a high degradation susceptibility based on demonstrated, compelling evidence for the occurrence of degradation or multiple plant observations, and those where there is a strong basis for degradation occurrence or known, but limited plant occurrences in addition to a relatively poor understanding of the relevant dependencies affecting degradation. The analyses for the PWR and BWR reactor coolant systems are summarized first since the panel judged the degradation issues in these systems to be more numerous and often more important than those in the other (generally lower temperature) systems.

Generic Materials Degradation Issues

There are several topics that can benefit from further research which cut across the component-specific issues. A good example of such a topic would be a quantitative understanding of the effects of weld metallurgy on the degradation phenomena. This is a wide-ranging issue covering, for instance, the size and distribution of weld defects, compositional and microstructural inhomogeneities, residual stress profiles, and how these alter with welding parameters (weld heat input, number of passes, preheat temperature, etc.). All of these variables can affect the susceptibility to stress corrosion cracking in a given component. These generic issues are discussed in some detail in Section 3.4 in terms of their impact on (a) damage assessment, (b) fracture assessment and, (c) the margin between the extent of damage and the failure criterion. The issues are summarized in Section 4.2.

The Next Step: Proactive Materials Degradation Management

A final generic issue discussed in Section 3.4, "Generic Materials Degradation and Life Management Issues," is the ability to transition from reactive to proactive management of materials degradation at the NRC, the U.S. industry, or their international counterparts. Technically-driven management of materials degradation in LWRs has been part of the professional life of all the panel members involved in this PMDA study–their observations, conclusions and recommendations on this topic are summarized as follows:

- Materials degradation will continue in LWRs, and may increase with license renewal and power uprates. Alternate materials, modified operating conditions, etc. may counteract these factors but, generally, they are not fully qualified and address only a fraction of the degradation modes. The technical reasons for these statements are outlined in detail in this PMDA report. Thus it is concluded that a Proactive Materials Degradation Management (PMDM) phase is needed.[1]

[1] The NRC staff believes that the changed operating conditions due to power uprates are not expected to appreciably reduce the effectiveness of licensees' aging management programs (AMPs) for currently identified mechanisms. For those plants requesting license renewal, the NRC assures that the licensees have developed effective AMPs focused on these mechanisms for the period of extended operation. Nonetheless, proactive materials degradation management may be useful to predict and manage degradation issues not previously encountered and to anticipate degradation of materials of geometries and in locations not previously shown to be susceptible to a particular aging mechanism. This approach could be used to augment the AMPs currently relied on during the practice of inspection, identification, and subsequent management, if warranted.

- Materials degradation typically results from complex phenomena involving metallurgy, electrochemistry, mechanics, radiation damage, physical chemistry, etc., and requires extraordinary experimental sophistication to resolve the interdependencies. This range of expertise rarely exists in any one organization. Thus the development of a PMDM capability would benefit from collaboration between various organizations.

- Prioritization, based on the consequences of degradation, of the proactive tasks on materials degradation management is essential.

- Adequate resources are needed to develop and maintain technical expertise and experimental capability. This applies to all the relevant organizations; reactor designers, regulators, National Laboratories, etc. This seems obvious but in spite of the significant impact of materials degradation over the last 30 years, the level of funding, the available expertise and up-to-date experimental facilities have all decreased. Most key experts are now close to, or are in, retirement. As a consequence, the resources that are available have been concentrated on short term "firefighting" projects and the longer term research and development projects have been delayed or cancelled. This leaves the reactor community vulnerable to a permanent loss of accumulated knowledge and expertise, and with a largely un-mentored workforce with less than five to ten years experience. This is an inadequate basis for addressing the complex questions that need to be answered. *It is imperative that these resource issues be addressed worldwide by government organizations, utilities, vendors and support organizations, and by universities and National Laboratories.*

ACKNOWLEDGEMENTS

This project was completed using an expert panel, and staff at Brookhaven National Laboratory (BNL) and the U.S. Nuclear Regulatory Commission (NRC). The concept for the project was developed at the NRC by Dr. Joseph Muscara who, as the Project Manager, provided guidance and technical direction through the course of the project. He was assisted by Mr. Michael Switzer at the NRC who, among other tasks, served as the scribe at panel meetings. NRC materials engineers, Dr. Todd Mintz, Dr. Amy Hull, and Mr. Hipolito Gonzalez, also worked with Dr. Muscara over the duration of the project. R. Scott Egli from the NRC Technical Training Center (TTC) provided information on pressurized water reactor systems design and operation and other staff from the TTC provided similar information for boiling water reactor systems. At BNL, Dr. Mano Subudhi was the lead technical person and Dr. David Diamond served as project manager, meeting facilitator and report editor. They were assisted at BNL by Dr. William Brown who provided support on database development and reporting, Ray Diaz who helped provide information on systems and components, and Susan Monteleone who was responsible for putting this report into final form. Drs. John Hickling and A. Demma at EPRI supported one of the panel members (R. Jones) and the former provided material for Appendices B.14 and B.18 while the latter provided material for Appendix B.13. Erin Rediger, Assistant to Dr. Roger Staehle, assumed responsibility for obtaining permissions and copyright approvals for the Figures in this report and its appendices that were originally published elsewhere.

We acknowledge with gratitude the help supplied by Guy De Boo and his co-workers at Exelon Corporation in providing detailed drawings and information on reactor systems. Information on stress levels was obtained in part from Engineering Mechanics Corporation of Columbus (EMC^2). Lastly, the NRC and BNL wish to acknowledge the panel members and their organizations who cooperatively shared some or all of their cost and time for participation in this study. The work of the panel members was far and above what was originally expected. Not only did they provide their technical expertise but also, through their hard work and diligence, provided the driving force to assure that the final product would be of use to the NRC and the materials degradation community at large.

We also acknowledge the review and comments on earlier versions of this report by a group of international experts, Drs.: Jacques Daret, personal contribution, France; Hannu Hunninen, Helsinki University of Technology, Finland; Yong-Zhi Wang, CNSC, Canada; Anna Brozova, NRI Rez, Czech Republic; Martin Widera, RWE, Germany; Steve Bruemmer, PNNL, USA; Ron Horn, GE, USA; Tatsuo Funada, JNES, Japan; Pal Efsing, Ringhals AB, Sweden; and Staff of the Naval Reactor Program, USA.

ABBREVIATIONS

AcSCC	Acidic stress corrosion cracking
AFW	Auxiliary feedwater system
AkIGC	Alkaline intergranular cracking
AkSCC	Alkaline stress corrosion cracking
AVT	All volatile treatment
B&W	Babcock & Wilcox
BAC	Boric acid corrosion
BWR	Boiling water reactor
CalSil	Calcium silicate
CASS	Cast austenitic stainless steel
CCW	Component cooling water
CDF	Core damage frequency
CE	Combustion Engineering
CREEP	Thermal creep
CREV	Crevice corrosion
CUF	Cumulative usage factors
CVCS	Chemical and volume control system
DEBOND	De-bonding
EC	Erosion corrosion including steam cutting and cavitation
ECCS	Emergency core cooling system
FAC	Flow-accelerated corrosion
FAT	Fatigue (corrosion/thermal/mechanical)
FIV	Flow-induced vibration
FoM	Figure of merit
FR	Reduction of fracture resistance
FW	Feedwater system
GALV	Galvanic corrosion
GC	General corrosion
GDC	General design criteria
HC	High-cycle (fatigue)
HWC	Hydrogen water chemistry
IASCC	Irradiation-assisted stress corrosion cracking
ICG-EAC	International cooperative group on environmentally assisted corrosion
IC	Irradiation creep
IGSCC	Intergranular stress corrosion cracking
LPSCC	Low potential SCC
LRA	License renewal application

ABBREVIATIONS

LTCP	Low temperature crack propagation
LWR	Light water reactor
MA	Mill annealed
MIC	Microbially induced corrosion
MRP	Materials Reliability Program
MS	Main steam system
NRC	U.S. Nuclear Regulatory Commission
NWC	Normal water chemistry
OTSG	Once through steam generator
P&ID	Piping and instrumentation diagram
PbSCC	Lead-induced stress corrosion cracking
PIRT	Phenomena identification and ranking table
PIT	Pitting corrosion
PMDA	Proactive materials degradation assessment
PMDM	Proactive materials degradation management
PORV	Power operated relief valve
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
R&D	Research and development
RCIC	Reactor core isolation cooling
RCP	Reactor coolant pump
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
RHR	Residual heat removal system
RPV	Reactor pressure vessel
RVI	Reactor vessel internals
RWST	Refueling water storage tank
SAR	Safety analysis report
SCC	Stress corrosion cracking
SG	Steam generator
SGBD	Steam generator blowdown
SI	Safety injection
SR	Stress relieved
SRV	Safety relief valve
SW	Swelling
SWS	Service water system
TGSCC	Transgranular stress corrosion cracking
TT	Thermally treated
WEAR	Fretting/wear

1. INTRODUCTION

1.1 Background

Materials degradation of components in nuclear power reactors has occurred since the inception of the nuclear power industry, and has required substantial investments by the industry in research, mitigation, repair, replacement, and inspection activities; these industry actions have been accompanied by correspondingly strong regulatory involvement and actions. One of the reasons for this situation is that, apart from general corrosion, fatigue and irradiation embrittlement of the pressure vessel, many degradation phenomena were not considered specifically in the design-basis for the current light water reactor (LWR) fleet in the United States. Key among the degradation phenomena not considered were those associated with corrosion events that were localized either because of metallurgical, stress, environmental, or geometrical conditions.

Examples of localized corrosion phenomena that have required industry and regulatory attention include:

- Intergranular stress corrosion cracking of pressurized water reactor (PWR) nickel-base alloy steam generator tubing
- Intergranular stress corrosion cracking of boiling water reactor (BWR) stainless steel piping
- Irradiation-assisted stress corrosion cracking of stainless steel core components in PWRs and BWRs
- Intergranular stress corrosion cracking of nickel-base alloy primary piping in PWRs
- Intergranular stress corrosion cracking of nickel-base alloy vessel penetrations in PWRs
- Flow accelerated corrosion of carbon-steel piping in both PWRs and BWRs
- Boric acid wastage of low-alloy PWR pressure vessel steel

As illustrated in Figure 1.1 [1] for PWR steam generator tube damage and in Table 1.1 [2] for cracking phenomena in BWRs, these materials degradation modes have changed over time, in terms of the transition from one damage mode to the next. In some cases, the evolving degradation was caused by a previous remedy.

These degradation occurrences have been extensively catalogued in conference proceedings [3-21], Institute of Power Operations (INPO) records (EPIX), and U.S. Nuclear Regulatory Commission (NRC) documents such as Licensee Event Reports [22] and the Generic Aging Lessons Learned (GALL) report [23].

The LWR industry has developed mitigation actions and aging management programs to deal with these degradation occurrences. However these activities and additional regulatory actions have been conducted *after* the initial incidents have occurred. This reactive nature of the response has several potential consequences:

- Potential reduction in reliability of important components while the mitigation and appropriate control actions were being developed,
- Increased monetary and time commitments for the NRC and industry due to the unforeseen nature of the incidents that impacted orderly and planned response, and
- Erosion of public confidence in the reliable operation of nuclear power plants.

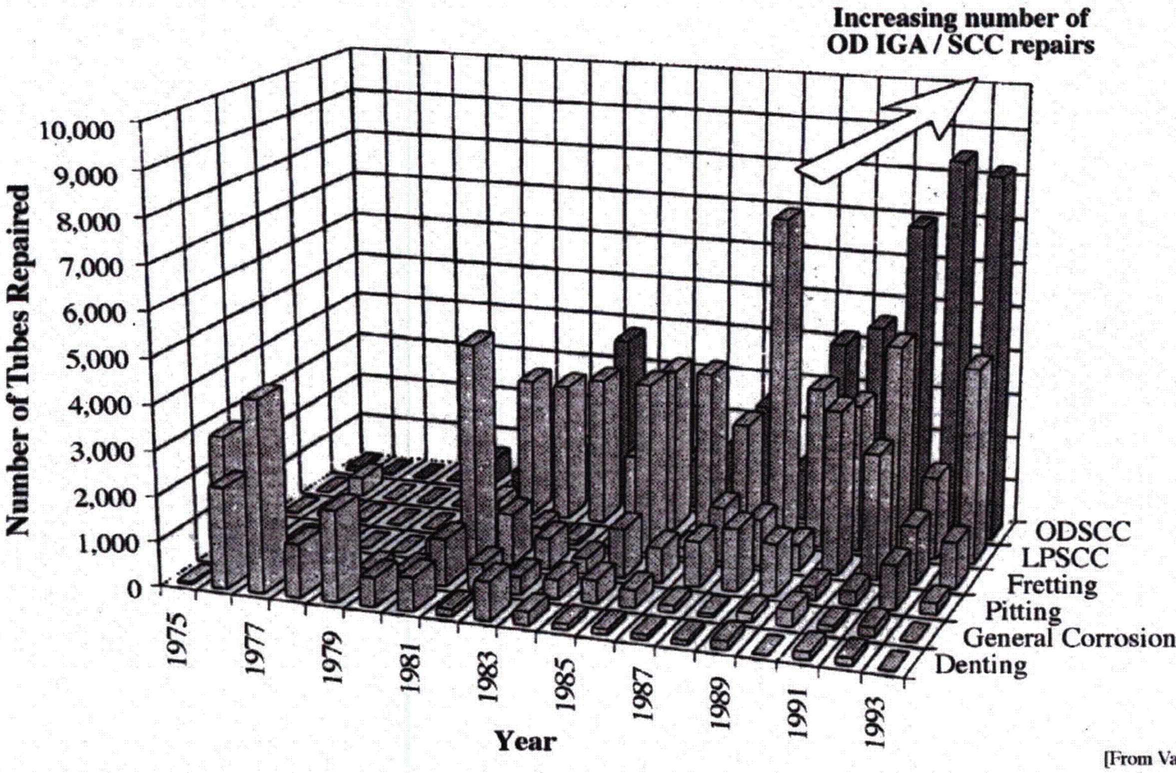

Figure 1.1 **Evolution of damage modes in PWR steam generator tubes [1]** ***(used with permission of EPRI)*** (ODSCC: Outside Diameter Stress Corrosion Cracking, LPSCC: Low Potential SCC, OD IGA: Outside Diameter Intergranular Attack)

Table 1.1 **Evolution of Cracking Incidents in BWR Structures [2]** (SCC: stress corrosion cracking, IGSCC: Intergranular SCC, IASCC: Irradiation-Assisted SCC)

Component and Mode of Failure	Alloy	Time Period
Fuel cladding, irradiation assisted SCC	304	1960s
Furnace sensitized safe ends, IGSCC	304, 182, 600	
Weld sensitized small diameter piping, IGSCC	304	
Weld sensitized large diameter piping, IGSCC	304	
Furnace sensitized weldments & safe ends, IGSCC	182/600	
Low alloy steel nozzles, thermally induced vibration	A508	1980s
Crevice induced cracking	304L/316L	
Jet pump beams, IGSCC	X750	
Cold work induced IGSCC of "resistant" alloys	304L	
Low alloy steel pressure vessel, TGSCC	A533B/A508	
Irradiated core internals, IASCC	304, 316	
IGSCC/IASCC of low carbon and stabilized stainless steel	304L, 316L, 321, 347	2000s

Changes taking place within the LWR fleet may well increase the likelihood of material degradation. For instance, license renewal applications (LRAs) are expected from the majority of U.S. LWR licensees to extend the operating life from the current 40 years to 60 years. This requires regulatory review of the adequacy of current aging management programs at the reactor sites, and of the margins in time-limited aging analyses (TLAA) for pressure vessel embrittlement, and for fatigue that may not have initially accounted fully for the variety of loading, environment and material combinations that are now known to have an effect on these degradation modes.

In many cases, in addition to license renewal, other changes, such as an increase in power output, may be implemented. Such power uprates may be accomplished via core redesign and increased coolant flow and may, therefore, potentially increase the susceptibility to irradiation-assisted stress corrosion cracking (IASCC) of core components, and to flow-accelerated corrosion (FAC) of carbon steel piping and flow-induced vibration (FIV) of other components. Finally, the drive towards longer fuel cycles and decreased outage times may place a constraint on the extent of in-service inspection possible and create a need for improved inspection resolution and probability of detection. If the inspection capabilities and frequencies will be such that degradation cannot be detected in a timely manner by periodic in-service inspection, the use of continuous in-situ monitoring for degradation initiation and growth may provide an alternate method for timely detection of degradation.

As a result, and spurred by the Davis-Besse incident, which involved through-wall corrosion of the pressure vessel low alloy steel, both the NRC and industry have initiated programs that have the overall objective of managing materials degradation *proactively*. This includes early identification of potential degradation occurrence to give sufficient time to develop materials degradation management programs and guidance using mitigative and/or inspection, monitoring and timely repair strategies before costly, loss of integrity or safety incidents occur. The approaches taken by the NRC and industry for proactive materials degradation assessment and management programs are slightly different in terms of timing and focus, but are complementary in terms of achieving the same overall objective. The NRC's Proactive Materials Degradation Assessment (PMDA) was conducted during the period from August 2004 to August 2005. The first full draft of this report was completed by the panel members in March 2006.

The industry approach was launched in May 2003, and is described in the Nuclear Energy Institute's NEI 03-08 "Guideline for the Management of Materials Issues." This document describes how the industry intends to manage proactively the possible degradation of LWR materials and, thereby, provides a management tool that prioritizes the year-by-year resource allocations to those materials degradation issues that are likely to present a business and technical risk to operating LWRs.

The NRC's Proactive Materials Degradation Assessment (PMDA) project is being undertaken to develop information needed to implement action for proactive material degradation management. Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50) sets out the legal requirements with regard to the structural integrity of the reactor pressure boundary materials. For instance in Appendix A, the General Design Criteria (GDC) section of the Code, it is stated (GDC 14) that "the reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture." Moreover, in subsequent sections (GDC 15, 30, 31, 32) there are requirements that the components will have sufficient margin built into the design to preclude such failures under all anticipated operating conditions, even if the components have preexisting flaws. It is further required that all components be capable of being inspected so as to ensure that these structural integrity criteria are met.

The 10 CFR 50 Appendix A provides general design criteria for nuclear power plants; 10 CFR 50.55a requires licensees to adhere to ASME Section XI for more specific In-service Inspection requirements. However, the inspection intervals may not account for many of the particular materials degradation modes, and they may not take into full account the dominant roles that material and environmental combinations may have, as well as the purely mechanical aspects of the degradation. These technology-specific details are usually addressed in NRC Bulletins and Regulatory Guides (and other regulatory instruments) and are generally arrived at in a reactive mode following studies by both the industry and NRC when a particular materials degradation problem emerges.

Presently there is a concerted movement to develop risk-informed, performance-based regulations and associated regulatory guidelines [24-28], which lessen the burden of prescriptive regulations and prioritize actions to those items that present the greatest risk to reactor safety. The risk assessments supporting these programs do not specifically take into account the *details* of time-dependent materials degradation, the evolution of which may vary in a non-monotonic manner dependent on the specific materials and environment conditions. In the current risk-informed ISI programs, these issues are addressed through periodic program updates. However, it may be useful in the longer term to develop a proactive materials degradation management methodology to extend the current probabilistic analyses of materials degradation into a realistic (i.e., does not assume a constant failure rate) time-dependent component in the Probabilistic Risk Assessments (PRA). In addition, maintenance of component integrity, as an important aspect of defense-in-depth, should be seriously considered along with the PRA to limit the potential for failure.

1.2 Objective of Proactive Materials Degradation Assessment (PMDA) Program

The overall objective of the NRC's Proactive Materials Degradation Assessment (PMDA) and subsequent management (PMDM) program is to lay a technical foundation for any actions that may be needed to help ensure that future material degradation does not diminish the integrity of important components or the safety of the operating LWRs. A two-phase approach, which started in the summer of 2004, is being used for this work.

The main objective of the first phase is to identify materials and components where future degradation may occur in specific LWR systems; in some cases the degradation may involve phenomena not yet experienced in the operating fleet, but laboratory data and/or mechanistic understanding indicates that they may be pertinent to future reactor operations.

The objective of the second phase, initiated in summer 2005, is to develop and implement an international cooperative research program for the components and degradation mechanisms of interest to future LWR operation. In Phase 2 of this program, NRC will also consider the knowledge gaps and industry priorities identified in the EPRI BWR and PWR Issue Management Tables and work with industry to establish joint priorities for research supporting proactive materials degradation management. This cooperative program will undertake the proactive R&D needed, starting with the highest priority projects, addressing topics such as:

- Materials and degradation mechanisms,
- Mitigation,
- Repair and replacement, and
- Nondestructive examination and continuous monitoring.

The current study addresses the first phase of the PMDA and PMDM programs. Separate NRC programs are addressing in-service inspection effectiveness and risk importance for components that are found by the current study to be susceptible to future materials degradation.

1.3 The Approach

The approach in the first phase of the PMDA program was to use a structured elicitation process, drawing on the knowledge of a panel of experts to identify various potential degradation modes that could be reasonably expected in the future in selected components of representative PWR and BWR plants. The panel met seven times for week-long meetings to discuss individual analyses and to evaluate the potential degradation modes in specific LWR systems and components. The plant designs considered were a Westinghouse four-loop PWR and a General Electric BWR-5. These design-specific evaluations were then extended to cover the impact of different system conditions (material, environment, stress) associated with other plant designs e.g. Babcock & Wilcox, Combustion Engineering, BWR-6, etc. The analyses were made for specific passive components (identified in Section 2.3) under the full range of normal operating conditions, including start-up, hot-standby, at-power operation, shutdown, etc.

The following conditions and aspects were considered to be outside the scope of the study:

- Material degradation occurring during severe accidents;
- Degradation of fuel and fuel cladding;
- Degradation of balance-of-plant components such as turbines, and the containment structures;
- Specific role of human error that might exacerbate materials degradation; and
- In-service inspection capabilities for different components.

1.4 The Panel

The panel members were drawn from Canada, France, Japan, Sweden and the U.S., and possessed the relevant disciplines (metallurgy, chemistry, engineering, etc), to assess the operating history, practicalities and theory behind materials degradation issues in LWRs. They are listed below along with their affiliation and brief resumes are given in Appendix C. Staff from the NRC Technical Training Center, NRC Office of Nuclear Regulatory Research and Brookhaven National Laboratory supported the panel by providing specific engineering drawings of the various reactor systems, details of the materials of construction and their fabrication conditions, relevant reports, plus extensive plant operational experience.

Peter L. Andresen	General Electric Global Research Center
F. Peter Ford	General Electric Global Research Center (Retired)
Karen Gott	Swedish Nuclear Power Inspectorate (SKI), Sweden
Robin L. Jones	Electric Power Research Institute
Peter M. Scott	AREVA Framatome ANP, France
Tetsuo Shoji	Tohoku University, Japan
Roger W. Staehle	Consultant
Robert L. Tapping	Atomic Energy Canada Limited, Canada

1.5 Organization of Report

Section 2 explains the methodology used by the PIRT panel. It provides an overview of the PIRT process and explains the background information and the degradation mechanisms considered by the panel. Technical challenges faced by the panel members are discussed followed by an explanation of the evaluation process. Lastly, comments are made to explain additional information generated by the panel.

Section 3 provides results of the expert elicitation. These are "reports" from the data base generated by the evaluations carried out by the panel members. This section discusses the way in which these reports can be used to analyze the likelihood of materials degradation and then provides that analysis. It also contains results gleaned from the discussions the panel members had at their meetings. Section 4 contains the conclusions of this study.

Appendix A is a review of materials degradation which provides background information useful for the portion of readers who are not expert in the field. Appendix B is a set of background papers on different degradation mechanisms providing detail and useful references. Appendix C contains short resumes of the panel members. Appendix D. on CD, contains a summary of the evaluations done by the panel members. Lastly, Appendices E and F, also on CD, provide electronic files used/generated by the panel.

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2. METHODOLOGY

2.1 Overview of the PIRT Process Applied to PMDA

The expert elicitation process conducted in this study is based on the Phenomena Identification and Ranking Table (PIRT), which has been used by NRC many times [1]. The PIRT process provides a systematic means of obtaining information from experts and involves generating lists (tables) of phenomena where "phenomena" can also refer to a particular reactor condition, a physical or engineering approximation, a reactor component or parameter, or anything else that might influence some relevant figure-of-merit. The process usually involves ranking of these phenomena using some scoring criteria in order to help determine what is most important. That ranking as well as the information obtained to explain the ranking allows NRC to prioritize research needs for a safety issue or to support some other decision-making process. The PIRT methodology brings into focus the phenomena that dominate an issue, while identifying all plausible effects to demonstrate completeness.

Each PIRT application has been unique in some respect and the current project is again a unique application. The current PIRT can be described in terms of eight steps. The general meaning and application of these steps is explained in detail below:

Step 1: Define the issue that is driving the need, e.g., licensing, operational, or programmatic. The definition may evolve as a hierarchy starting with federal regulations and/or design and safety goals and descending to a consideration of key physical processes. The issue to be addressed in this project is the need to be forward thinking in terms of identifying potential future materials degradation occurrences and keep degradation from adversely impacting the reliability and safety of nuclear power plants. This is explained in detail in Section 1.1

Step 2: Define the specific objectives of the PIRT. The PIRT objectives are usually specified by the sponsoring agency. A clear statement of PIRT objectives is important because it defines the focus, content, and intended applications of the PIRT product. The PIRT objectives should include a description of the final products to be prepared. The objective in the current PIRT is to develop a list of nuclear power plant components and their potential degradation mechanisms for use in various research and regulatory activities. This is explained in more detail in Section 1.2.

Step 3: Define the hardware, equipment and scenario for which the PIRT is to be prepared. Generally, a specific hardware configuration and specific scenario are defined. Experience obtained from previous PIRT efforts indicates that any consideration of multiple hardware configurations or scenarios impedes PIRT development. After the baseline PIRT is completed for the specified hardware and scenario, the applicability of the PIRT to related hardware configurations and scenarios can be assessed. The hardware to be considered in the present PIRT is the components/parts of a Westinghouse four-loop plant and a General Electric BWR-5 design. After this baseline is finished, consideration is given to other designs that might have different degradation assessments. Parts that are identical in design, materials, and environment are lumped together for a particular system. The term scenario for this application is the past and future operating conditions (the environment) for the part. The hardware and scenario are provided as the background information explained in Section 2.2.

Step 4: Compile and review the contents of a database that captures the relevant experimental and analytic knowledge relative to the physical processes and hardware for which the PIRT is being developed. Each panel member reviews and becomes familiar with the information in the database. The background information used by the PMDA panel is the environment (i.e., stressors such as temperature, pressure, irradiation, chemistry, stress, etc.) of each of the parts of the reactor and plant systems considered as well as relevant operational experience. Information on degradation mechanisms is also used. The specific information provided to the panel is explained in Section 2.2.

Step 5: Define the figure-of-merit (FoM). The FoM is the primary evaluation criterion used to judge the relative importance of each phenomenon. Therefore, it must be identified before proceeding with the ranking portion of the PIRT effort. The characteristic of a well-defined FoM is that it is: 1) directly related to the issue(s) being addressed; 2) directly related to the phenomena expected to occur during the scenario; 3) easily comprehended; 4) explicit; and 5) measurable. The FoM in the current PIRT is the degree of susceptibility to given degradation mechanisms. These mechanisms are identified for a given component/part and are the phenomena to be ranked. The objective is to see if any parts are susceptible to these degradation mechanisms. The list of degradation mechanisms used by the panel is discussed in Section 2.3.

Step 6: Develop the importance ranking and rationale for each phenomenon. Importance is ranked relative to the FoM adopted in Step 5. For the current PIRT, in order to understand the "importance" of a degradation mechanism, a scale for susceptibility was developed and the panel members were asked to rank the mechanism with respect to susceptibility and also their confidence in making that ranking call. This is part of the evaluation process that is explained in Section 2.5.

Step 7: Assess the level of knowledge regarding each phenomenon. As with importance ranking, several scales have been used in the past. In the current case in addition to the susceptibility and confidence ranking, the panel members were asked to rank a knowledge level in terms of whether the relevant dependencies have been qualified. This is explained in Section 2.5.

Step 8: Document the PIRT results. The primary objective of this step is to provide sufficient coverage and depth that a knowledgeable reader can understand what was done (process) and the outcomes (results). This report satisfies the documentation requirements. It includes a database with the results of the expert elicitation as well as interpretative material and an explanation of the process used for the elicitation.

As presented above, the PIRT process proceeds from start to end without iteration. In reality, however, the option to revisit any step is available and was often exercised during the PIRT. Steps 4-8 make up the bulk of the effort in this PIRT. Figure 2.1 complements the explanation of these steps showing the series of electronic files used to carry out the PIRT. The background information collected for use by the panel members (Step 4) is first collected (Boxes 1, 2, and 3 on Figure 2.1). It results in an EXCEL workbook with information on parts and a set of drawings both of which are explained in Section 2.2. Because the number of parts is so large, the panel decided to lump parts from a given group into subgroups where they could be described by the same degree of susceptibility to given degradation mechanisms and then evaluated. Box 4 represents the subgrouping and creation of an evaluation workbook, whereas Box 5 takes into account that the evaluation is to be done for a set of applicable degradation mechanisms, explained in more detail in Section 2.3 (Step 5). The evaluation is done to provide more information on the potential degradation mechanisms and hence the panel members each score susceptibility, confidence, and knowledge levels for a particular degradation mechanism for a particular subgroup (Steps 6 and 7). These three metrics are defined in Section 2.5. Boxes 6 and 7 represent the database of evaluations from each of the panel members and the reports, which are different ways of presenting the evaluations. These reports help in the interpretation of the evaluations as is explained in Section 3.

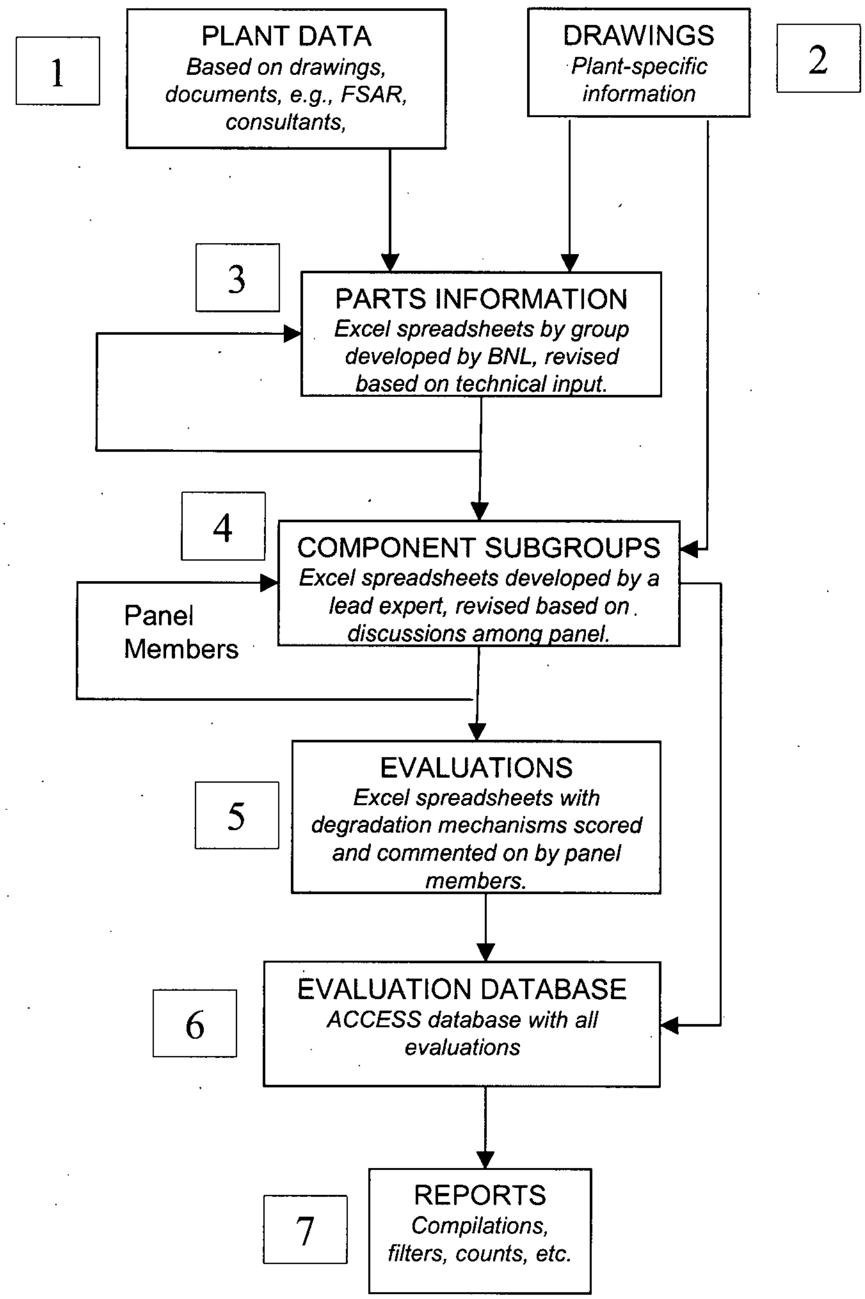

Figure 2.1 Flowchart for files created

2.2 Background Information on Reactor Components

The PWR and BWR reactor and plant systems selected for examination are given in Tables 2.1 and 2.2, respectively. These systems are either safety-related systems or systems where materials degradation can either impact safety or lead to leaks of radioactive water.

Examples of PWR systems that were not included are containment spray, fire protection, emergency diesel generator, main turbine system, and condensate system. Similar examples of BWR systems that were not included are the standby gas treatment system, standby liquid control system, and control room ventilation. Although having some safety significance, failures in these PWR/BWR systems do not directly lead to the leakage of radioactive water and were considered outside the scope of this project. BWR systems that were not included because of their similarity to the corresponding PWR systems are containment penetrations, spent fuel storage, spent fuel pool cooling and cleanup, service water, and component cooling water.

Table 2.1 PWR Systems Considered

General System/Function	System/Component
Reactor Coolant System (RCS)	Reactor Pressure Vessel (RPV)
	Reactor Vessel Internals (RVI)
	Reactor Coolant Pressure Boundary (RCPB)
	Reactor Coolant Pump (RCP)
	Pressurizer, Power Operated Relief Valves (PORVs), Safety Relief Valves (SRVs)
	Steam Generator
Engineered Safety Feature (ESF) Systems	Emergency Core Cooling System (ECCS)—High Head Injection (Charging Mode of Chemical and Volume Control System, CVCS)
	ECCS--Intermediate Head Injection (Accumulators, Safety Injection, SI)
	ECCS—Low Head Injection (Residual Heat Removal System, RHR)
	Refueling Water Storage Tank (RWST)
	Containment Penetrations
Steam and Power Conversion Systems	Main Steam System(MS)
	Feedwater System (FW)
	Auxiliary Feedwater System (AFW)
	Steam Generator Blowdown (SGBD)
Support Systems	Chemical and Volume Control System (CVCS)
	Component Cooling Water (CCW)
	Reactor Coolant Pump Seal Cooling
	Service Water (SW)
	Spent Fuel Pool Cooling and Cleanup
Auxiliary Systems	Spent Fuel Pool and Storage

Table 2.2 BWR Systems Considered

General System/Function	System/Component
Reactor Coolant System (RCS)	Reactor Pressure Vessel (RPV)
	Reactor Vessel Internals (RVI)
	Reactor Recirculation System (RR)
	Recirculation Pump (RP)
Engineered Safety Feature (ESF) Systems	Residual Heat Removal System (RHR)
	Emergency Core Cooling System (ECCS)—Low Pressure Core Spray (LPCS), High Pressure Core Spray (HPCS)
	Condensate Storage Tank (CST)
Steam and Power Conversion Systems	Main Steam System(MS) Automatic Depressurization System (ADS)
	Condensate System
	Feedwater System (FW)
Support Systems	Reactor Water Cleanup (RWC)
Auxiliary Systems	Reactor Core Isolation Cooling (RCIC)
	Control Rod Drive (CRD)

All of the relevant parts or components of the systems in Tables 2.1 and 2.2 were considered. A part/component generally consists of a continuous, uniform section of the same material that experiences the same stressors. Where there are multiple parts in a system of the same material, geometry, and product form such as 20 piping elbows that experience the same stressors (temperature, stresses, radiation, chemical environment, etc.), these parts are considered as a single part. In addition, where there are multiple loops or trains in a system, only one loop or train was considered. Thus, the degradation assessment results from this study apply to a number of locations in a nuclear power plant that is several times the number of components or parts evaluated. In general, piping with diameter of 5 cm (2 in) and larger was included in this study.

The total number of parts for the PWR analysis is 2203 and for the BWR, 1660. In addition to the 1660 parts for the BWR, there were 542 parts analyzed under the PWR study, which are in common with, and the results apply to, the BWR analysis. In order to present information on thousands of parts, it was convenient to divide each system into groups. Each group was defined to make the display of data as comprehensible as possible. Some of the grouping was obvious, e.g., the division of the PWR reactor coolant pressure boundary into groups consisting of the cold leg, hot leg, reactor coolant pump, etc. Some of the grouping was done to take advantage of the fact that drawings were available for a portion of the system. The result is that the systems in Table 2.1 are described in terms of 48 groups of components/parts and those in Table 2.2 in terms of 28 groups. These are listed in Column 3 of Tables 2.3 and 2.4 which provide information on the systems and parts considered.

For each group a spreadsheet was provided with information on all components/parts in that group. Column 4 of Tables 2.3 and 2.4 shows how many parts were considered for each group. For each part, there were 26 types of information provided. Each type is listed in Table 2.5. The most relevant information is the material/fabrication of the part and its environment. The latter includes temperature and pressure of the fluid inside, external environment, and stresses. Not all information was available for all parts, e.g., residual stresses were particularly difficult to find. Sources used to obtain information are listed in Table 2.6.

Tables 2.7a, 2.7b, and 2.7c show a typical spreadsheet; in this case for the power operated relief valves (PORVs). Only nine parts are shown in the tables although 21 parts were considered for that group (see Table 2.3). A workbook containing the 48 spreadsheets for the PWR systems and one with the 28 spreadsheets for the BWR systems are found in Appendices E and F (on DVD), respectively. At the beginning of each workbook is a spreadsheet with generic information relevant to the systems in that workbook.

One important source of information when generating these spreadsheets is drawings, primarily isometric drawings but also piping and instrumentation diagrams [P&IDs]. Some of these drawings were distilled into simpler drawings for use by the panel members in conjunction with the information on parts given on the spreadsheets. An example of the relevant drawings supplied for the parts given in Table 2.7 is found in Figure 2.2. The numbers on these figures correspond to the parts numbers (1-21) for that group (7). The complete sets of 197 drawings for the 48 PWR groups and 179 drawings for the 28 BWR groups are found in Appendices E and F (on DVD), respectively.

The information described above is necessary to understand what degradation mechanisms might be applicable to each part. However, because there are so many parts, the panel decided to lump parts together from a given group where they were considered to be equally susceptible to given degradation mechanisms. This subgrouping led to considering 386 PWR subgroups rather than 2203 parts and 297 BWR subgroups rather than 1660 parts. An example is shown in Section 2.3.

Table 2.3 PWR Groups, Number of Parts and Subgroups per Group

Group	System	Group Name	Number of Parts	# of Subgroups
1	Reactor Coolant System	Cold Leg Piping	28	12
2	Reactor Coolant System	Crossover Leg Piping	18	12
3	Reactor Coolant System	Hot Leg Piping	22	12
4	Reactor Coolant System	Pressurizer	44	15
5	Reactor Coolant System	Pressurizer Spray Piping	36	7
6	Reactor Coolant System	Pressurizer Surge Piping	6	7
7	Reactor Coolant System	Pressurizer Piping to PORVs	21	7
8	Reactor Coolant System	Pressurizer Piping to SRVs	9	6
9	Reactor Coolant System	Reactor Coolant Pump	7	6
10	Reactor Coolant System	Reactor Pressure Vessel	32	12
11	Reactor Coolant System	Steam Generator	71	23
12	Reactor Coolant System	Reactor Vessel Internals	25	12
13	Reactor Coolant System	Stop Valve Loop Bypass Piping	38	7
14	Emergency Core Cooling Systems	RWST Header Piping	56	6
15	Emergency Core Cooling Systems	CVCS Pump Suction Piping	57	6
16	Emergency Core Cooling Systems	SI Pump Suction Piping	48	6
17	Emergency Core Cooling Systems	RHR Pump Suction Piping	77	7
18	Emergency Core Cooling Systems	Accumulator Piping to RCS Cold Leg	41	14
19	Emergency Core Cooling Systems	SI/RHR Piping to RCS Hot Leg	58	12
20	Emergency Core Cooling Systems	RHR Pump Discharge Piping	54	6
21	Emergency Core Cooling Systems	RHR Piping to RCS Cold Leg	30	6
22	Emergency Core Cooling Systems	CVCS Piping to RCS Cold Leg	86	8
23	Emergency Core Cooling Systems	Safety Injection Pump Discharge Piping	53	6
24	Steam & Power Conversion System	Main Steam	89	3
25	Steam & Power Conversion System	Main Feedwater System	56	4
26	Steam & Power Conversion System	Auxiliary Feedwater System	45	2
27	Steam & Power Conversion System	Steam Generator Blowdown Piping	55	3
28	Support System	Service Water Suction Piping from Pond	49	4
29	Support System	Service Water Discharge Piping	70	6
30	Support System	Service Water Piping Inside Containment	56	2
31	Support System	CVCS Pump Piping to Crossover Leg Injection	49	12

Table 2.3 PWR Groups, Number of Parts and Subgroups Per Group (continued)

Group	System	Group Name	Number of Parts	Number Sub-groups
32	Support System	CVCS Normal Letdown Piping	35	6
33	Support System	CVCS Regenerative HX Piping to Letdown HX	56	7
34	Support System	CVCS Letdown HX Piping to VCT	53	8
35	Support System	CVCS Mixed Bed Piping to Filter	48	9
36	Support System	CVCS VCT Piping to Charging Pump Suction	70	7
37	Support System	CVCS Charging Pump Piping to Regenerative HX	53	7
38	Support System	Regenerative HX Piping to Cold Leg	24	6
39	Support System	CVCS Injection Filter Piping to RCP Seals	65	7
40	Support System	CVCS RCP Seal Return Piping Filter	46	6
41	Support System	CCW Piping Surge Tank Piping to CCW HX	72	10
42	Support System	CCW HX Piping to RHR HX	50	6
43	Support System	CCW to Other Loads Outside Containment	56	7
44	Support System	CCW Piping to RCPs Inside Containment	58	7
45	Support System	Spent Fuel Pool Cooling Piping	66	11
46	Support System	Spent Fuel Pool Cleaning Piping	46	7
47	Auxiliary System	Spent Fuel Pool and Fuel Racks	9	7
48	Engineered Safety Features	Containment Penetrations for Process Piping	10	17

Table 2.4 BWR Groups, Number of Parts and Subgroups per Group

Group	System	Group Name	Number of Parts	Number of Sub-groups
1	Reactor Coolant System	Reactor Pressure Vessel Closure Head	24	13
2	Reactor Coolant System	Reactor Pressure Vessel Shell	88	21
3	Reactor Coolant System	Reactor Pressure Vessel Bottom Head	59	15
4	Reactor Coolant System	Core Shroud	47	20
5	Reactor Coolant System	Core Controls	42	17
6	Reactor Coolant System	Jet Pump Assembly	23	14
7	Reactor Coolant System	ECCS Connections	50	11
8	Reactor Coolant System	Steam Separator & Dryer	14	6
9	Reactor Coolant System	Reactor Recirculation System	90	18
10	Emergency Core Cooling System	Low Pressure Core Spray	96	10
11	Emergency Core Cooling System	High Pressure Core Spray - SP Water	83	7
11A	Emergency Core Cooling System	HPCS - CST Water (OTHER PLANT)	--	9
12	Auxiliary System	Reactor Core Isolation Cooling	84	16
13	Engineered Safety Feature System	RHR Suction Line Piping to RHR Pumps	58	9
14	Engineered Safety Feature System	RHR Pump Discharge Piping to RHR HX	78	9
15	Engineered Safety Feature System	RHR Normal Shutdown Cooling Piping	71	5
16	Engineered Safety Feature System	RHR Spray Piping	73	15
17	Steam and Power Conversion System	Main Steam	75	7
18	Engineered Safety Feature System	Cycled Condensate Storage Tank	58	2
19	Steam and Power Conversion System	Feedwater	107	4
20	Auxiliary System	Control Rod Drive	62	12
21	Steam and Power Conversion System	Main Condenser	32	6

Table 2.4 BWR Groups, Number of Parts and Subgroups per Group (continued)

Group	System	Group Name	Number of Parts	Number of Sub-groups
22	Steam and Power Conversion System	Main Condenser Discharge Piping	53	6
23	Steam and Power Conversion System	Condensate Piping to Booster Pump	57	3
24	Steam and Power Conversion System	Condensate Piping to FW Pump	65	6
25	Support System	Reactor Water Cleanup Piping to Pumps	65	9
26	Support System	Reactor Water Cleanup Piping to R/NR HXs	49	15
27	Support System	Reactor Water Cleanup Piping to/from Filters	31	8
28	Support System	Reactor Water Cleanup Piping to Fe-edwater	26	4

Table 2.5 Definition of Information Provided for Each Component/Part

Column	Item	Description
A	System Identification	System description (System Code)
B	Group Identification	Group # —Group description (Group Code)
C	Part Identification	System Code—Group Code
D	Part Number	Sequential unique number within Group
E	Part Description	Description of the part
F	Part Size	Diameter or width
G	Part Thickness	Pipe or component thickness
H	Material A	A-side of a weld or component material specification (form)
I	Material W	Weld material specification (if available)
J	Material B	B-side of a weld material specification (form)
K	Weld Type	Shop or field weld
L	Operating Tempera-ture	Full power temperature
M	Operating Pressure	Full power pressure
N	Operating Flow	Full power flow
O	Design Temperature	Design temperature
P	Design Pressure	Design pressure
Q	Design Flow	Design flow
R	Inside Environment	Flowing liquid, steam or air
S	Outside Environment	Building or surrounding environment
T	Residual Stress	Estimated residual stress due to welding (Sy for thicker pipes and 1.3 Sy for thinner pipes)
U	Normal Stress	Actual or estimated (allowable = 1.5 Sm or 1.2 Sy) normal operating stress
V	Faulted Stress	Actual or estimated (allowable = 3 Sm or 2.4 Sy) faulted condition stress
W	CUF	40-year cumulative usage factor due to plant transients and cyclic loadings
X	Stress Comments	Comments regarding stress values (in columns T, U, V, and W)
Y	Operating Experience	Industry events associated with this part or similar part(s) in other PWR plants
Z	General Comments	Comments on the data included in columns A through Y

Table 2.6 Sources of Information on System Parts

1. "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Volume 2, U.S. Nuclear Regulatory Commission, July 2001.
2. "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," WCAP-14575-A, Westinghouse Electric Company, December 2000.
3. "License Renewal Evaluation: Aging Management Evaluation for Pressurizers," WCAP-14574-A, Westinghouse Electric Company, December 2000.
4. "License Renewal Evaluation: Aging Management Evaluation for Reactor Internals," WCAP-14577, Rev. 1-A, Westinghouse Electric Company, March 2001.
5. "Demonstration of the Management of Aging Effects for the Reactor Vessel," BAW-2251A, B&W Owners Group, August 1999.
6. "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping," BAW-2243A, B&W Owners Group, June 1996.
7. "Demonstration of the Management of Aging Effects for the Pressurizer," BAW-2244A, B&W Owners Group, December 1997.
8. "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," BAW-2248, B&W Owners Group, July 1997.
9. ASME Boiler & Pressure Vessel Code for Nuclear Power Plant Components, Section III, 1996 Version.
10. Plant-Specific License Renewal Applications (LRAs) Submitted to NRC.
11. Plant-Specific Safety Analysis Reports (SARs) on Plant-Specific LRAs Submitted to NRC.
12. Plant-Specific Updated FSAR and Technical Specifications.
13. Plant-Specific Risk-Informed In-service Inspection (RI-ISI) Reports.
14. Plant-Specific Drawings (Includes P&ID, Plan/Elevation, Isometrics), Design Specifications, and System/Component Operating Manuals for Structures, Systems and Components.
15. "Nuclear Power Plant System Sourcebook," Plant-Specific SAIC Report for NRC, SAIC-89/xxxx.
16. Piping and Component Stress Analysis Reports for various Systems, Structures and Components.
17. Pressurizer Stress Reports for Various Pressurizer Models, Westinghouse Electric Corporation, Proprietary Documents.
18. Steam Generator Stress Reports for Various SG Models, Westinghouse Electric Corporation, Proprietary Documents.
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21. V.N. Shah and P.E. MacDonald, "Residual Life Assessment of Major Light Water Reactor Components – Overview," Volume 2, NUREG/CR-4731-VII, Idaho National Engineering Laboratory, November 1989.
22. M Subudhi, J.C. Higgins, and S.M. Coffin, "Review of Industry Responses to NRC Generic Letter 97-06 on Degradation of Steam Generator Internals," NUREG/CR-6754, Brookhaven National Laboratory, December 2001.

Table 2.6 Sources of Information on System Parts (continued)

23.	J.I. Braverman, C.H. Hofmayer, R.J. Morante, et.al, "Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants," NUREG/CR-6679, Brookhaven National Laboratory, August 2000.
24.	G. DeGrassi, "Evaluation of the High density Spent Fuel Rack Structural Analysis for Florida Power and Light Company, St. Lucie Plant – Unit No. 1," BNL Technical Report A-3841-2-2/88, Brookhaven National Laboratory, February 1988.
25.	W. Gunther and K. Sullivan, "Aging Assessment of the Westinghouse PWR Control Rod Drive System," NUREG/CR-5555, Brookhaven National Laboratory, March 1991.
26.	E. Grove and W. Gunther, "Aging Assessment of the Combustion Engineering and Babcock & Wilcox Control Rod Drives," NUREG/CR-5783, Brookhaven National Laboratory, January 1993.
27.	O.K. Chopra and W.J. Shack, "Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels," NUREG/CR-6717, Argonne National Laboratory, May 2001.
28.	O.K. Chopra, "Mechanism and Estimation of Fatigue Crack Initiation in Austenitic Stainless Steels in LWR Environments," NUREG/CR-6787, Argonne National Laboratory, July 2002.
29.	A.G. Ware, D.K. Morton, M.E. Nitzel, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," NUREG/CR-6260, Idaho National Engineering Laboratory, March 1995.
30.	"Pressurized Water Reactor Primary Water Chemistry Guidelines," Volume 1, Revision 5, 1002884, Final Report, Electric Power Research Institute, September 2003 (Revised October 2003).
31.	"Pressurized Water Reactor Primary Water Chemistry Guidelines," Volume 2, Revision 5, 1002884, Final Report, Electric Power Research Institute, September 2003 (Revised October 2003).
32.	"PWR Secondary Water Chemistry Guidelines—Revision 5," TR-102134-R5, Final Report, Electric Power Research Institute, May 2000.
33.	"BWR Secondary Water Chemistry Guidelines – 2000 Revision," TR-103515-R2, Final Report, Electric Power Research Institute, February 2000.
34.	"Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Byron and Braidwood Units 1 and 2 Nuclear Power Plants," WCAP 14559, Rev 1, Westinghouse Electric Company, April 1996.
35.	"NSSS Design Transients," Westinghouse Electric Company Proprietary, April 1974.
36.	"Cold Leg Integrity Evaluation," NUREG/CR-1319, January 1980.
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38.	"SKI-PIPE: SKI's Piping Failure Database Overview of Database Content & Structure," RSA-R-2000-01.5, May 2000.
39.	J. Medoff, "Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components," NUREG-1544, March 1996.
40.	"Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants," NUREG-0531, February 1979.
41.	C. Regan, S. Lee, O.K. Chopra, et al., "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," NUREG-1557, October 1996.

Table 2.7a Section of Spreadsheet for Group 7 (Pressurizer Piping to PORVs in RCS) – Part 1

A	B	C	D	E	F	G
System Identification	**Group Identification**	**Part Identification**	**Part Number**	**Part Description**	**Part Size in cm**	**Part Thickness in cm**
Reactor Coolant System (RCS)	Group 7—Pressurizer to PORVs (PORV)	RCS-PORV-	1	PZR Relief Nozzle Safe End—15 cm (6") Elbow	15 (6")	1.83 (0.719")
Reactor Coolant System (RCS)	Group 7—Pressurizer to PORVs (PORV)	RCS-PORV-	2	15 cm (6") Elbow	15 (6")	1.83 (0.719")
Reactor Coolant System (RCS)	Group 7—Pressurizer to PORVs (PORV)	RCS-PORV-	3	Elbow—Pipe	15 (6")	1.83 (0.719")
Reactor Coolant System (RCS)	Group 7—Pressurizer to PORVs (PORV)	RCS-PORV-	4	15 cm (6") Pipe—5.1 cm (2") Sockolet	5.1 (2")	0.874 (0.344")
Reactor Coolant System (RCS)	Group 7—Pressurizer to PORVs (PORV)	RCS-PORV-	5	5.1 cm (2") Sockolet	5.1 (2")	0.874 (0.344")
Reactor Coolant System (RCS)	Group 7—Pressurizer to PORVs (PORV)	RCS-PORV-	6	15 cm (6") Pipe	15 (6")	1.83 (0.719")
Reactor Coolant System (RCS)	Group 7—Pressurizer to PORVs (PORV)	RCS-PORV-	7	Pipe—Tee	15 (6")	1.83 (0.719")
Reactor Coolant System (RCS)	Group 7—Pressurizer to PORVs (PORV)	RCS-PORV-	8	15x15x7.6 cm (6"x6"x3") Tee		
Reactor Coolant System (RCS)	Group 7—Pressurizer to PORVs (PORV)	RCS-PORV-	9	Tee—15x7.6 cm (6"x3") Reducer	15 (6")	1.83 (0.719")

Table 2.7b Section of Spreadsheet for Group 7 (Pressurizer Piping to PORVs in RCS) – Part 2

H	I	J	K	L	M	N	O	P	Q	R
Material A	Material W	Material B	Weld Type	Operating Temperature °C	Operating Pressure in MPa	Operating Flow	Design Temperature °C	Design Pressure MPa	Design Flow	Inside Environment
SA182 GR.F316L (FORG.)		SA403 GR.WP304 (FITTING)	Field	345	15.5		360	17.1		Reactor Coolant
SA403 GR.WP304 (FITTING)		Not Applicable		345	15.5		360	17.1		Reactor Coolant
SA403 GR.WP304 (FITTING)		SA376 GR.TP304 (SMLS PIPE)	Shop	345	15.5		360	17.1		Reactor Coolant
SA376 GR.TP304 (SMLS PIPE)		SA182 GR.F304 (FORG.)	Field	345	15.5		360	17.1		Reactor Coolant
SA182 GR.F304 (FORG.)		Not Applicable		345	15.5		360	17.1		Reactor Coolant
SA376 GR.TP304 (SMLS PIPE)		Not Applicable		345	15.5		360	17.1		Reactor Coolant
SA376 GR.TP304 (SMLS PIPE)		SA403 GR.WP304 (FITTING)	Shop	345	15.5		360	17.1		Reactor Coolant
SA403 GR.WP304 (FITTING)		Not Applicable		345	15.5		360	17.1		Reactor Coolant
SA403 GR.WP304 (FITTING)		SA403 GR.WP304 (FITTING)	Shop	345	15.5		360	17.1		Reactor Coolant

Table 2.7c Section of Spreadsheet for Group 7 (Pressurizer Piping to PORVs in RCS) – Part 3

S	T	U	V	W	X	Y	Z
Outside Environment	**Residual Stress MPa**	**Normal Stress MPa**	**Faulted Stress MPa**	**CUF**	**Stress Comments**	**Operating Experience**	**General Comments**
Containment Air	223	56.4	193		Pressure+DW+Thermal Data provided by Exelon	LER 255 1993-009: HAZ of the PORV line to pressurizer nozzle safe end weld/ Inconel 600—-PWSCC, IG cracking	
Containment Air		56.3	170		See comment for Part #1		
Containment Air	125	170	170				
Containment Air	125	170	170				
Containment Air		144	144				
Containment Air		170	170				
Containment Air	125	170	170				
Containment Air		53.6	170		See comment for Part #1		
Containment Air	125	170	170				

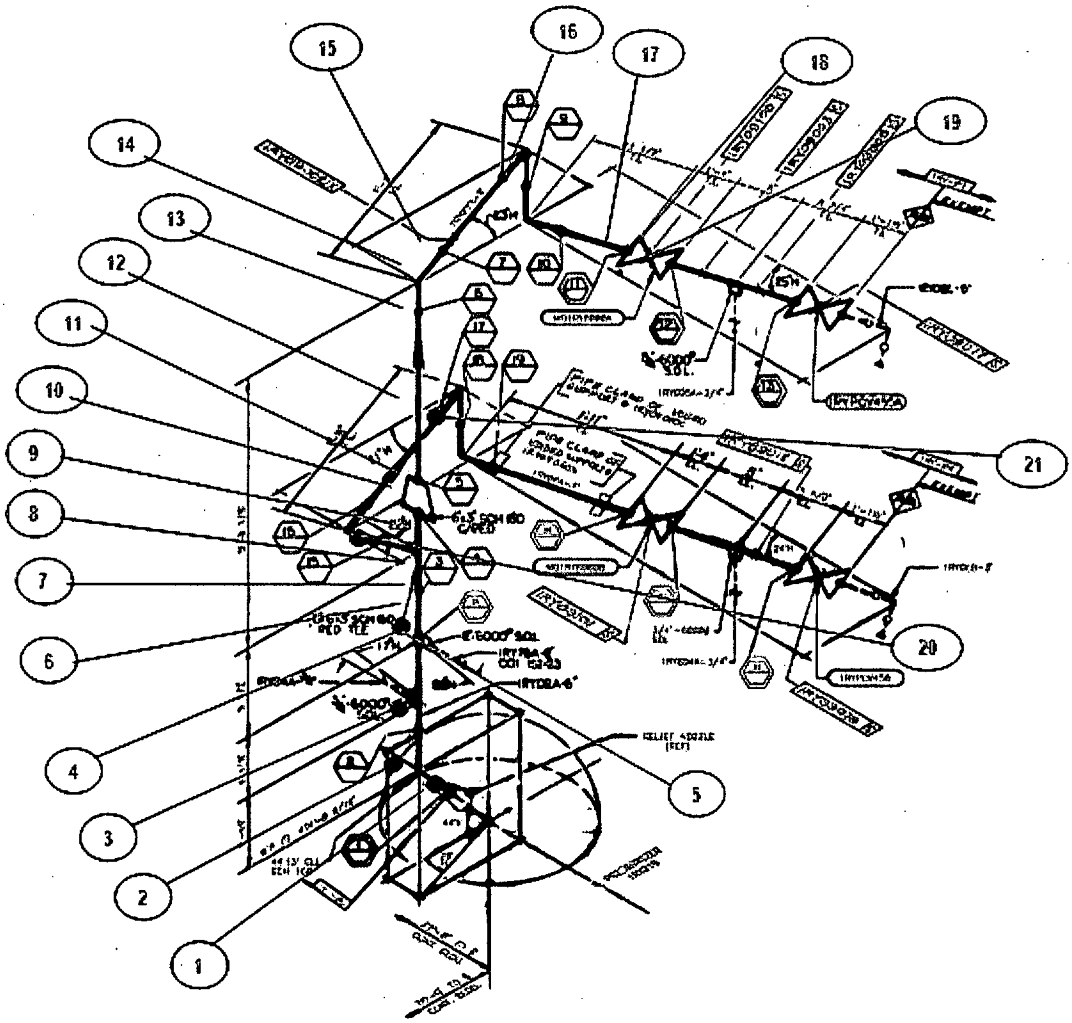


Figure 2.2 Drawing for Group 7 (Pressurizer Piping to PORVs in RCS)

2.3 Degradation Mechanisms

The panel agreed to a set of relevant degradation mechanisms in order to standardize their reporting. The panel members had to decide if these were either applicable, or not, with respect to each of the components/parts lumped into the subgroups. The mechanisms are listed in Table 2.8 along with their abbreviations/acronyms. Each of these mechanisms is discussed in more detail in Appendices A and B.

Table 2.8 Degradation Mechanisms

Abbreviation	Degradation Mechanism
BAC	Boric Acid Corrosion
CREEP	Thermal Creep
CREV	Crevice Corrosion (including denting)
DEBOND	De-bonding
EC	Erosion Corrosion Including Steam Cutting and Cavitation
FAC	Flow-accelerated Corrosion
FAT	Fatigue (corrosion/thermal/mechanical)
FR	Reduction of Fracture Resistance
GALV	Galvanic Corrosion
GC	General Corrosion
IC	Irradiation Creep
MIC	Microbially Induced Corrosion
PIT	Pitting Corrosion
SCC	Stress Corrosion Cracking (intergranular, transgranular, irradiation-assisted, strain-induced, hydrogen-embrittlement) and Intergranular Attack
SW	Swelling
WEAR	Fretting/Wear

The panel assigned degradation mechanisms to each of the subgroups collectively and then left the evaluation of susceptibility, confidence, and knowledge to each individual. An example of the resulting evaluation spreadsheet set up for PWR Group 7 is given in Table 2.9. The complete set of evaluation sheets is found in Appendices E and F (on CD). Column 1 and 2 identify the subgroup according to number and its principal characteristics. The last column shows the actual parts numbers applicable to this subgroup. Note that the total number of parts to be considered was 21 for Group 7 as explained in Section 2.2. Column 3 lists the degradation mechanisms that the panel thought were applicable and should be scored according to susceptibility, confidence and knowledge level. Although there are 16 degradation mechanisms listed in Table 2.8, only a few are relevant to these particular subgroups. The scoring is done in columns 4-6 and Table 2.9 shows a typical evaluation by a panel member. Column 7 provides a rationale for the scoring and Column 8 the evaluator's thoughts on critical factors controlling the occurrence of the degradation mechanism. There were 1222 evaluations (combination of subgroup and degradation mechanisms) for PWRs and 1322 for BWRs to be conducted by each panel member.

Table 2.9 Evaluation Spreadsheet for Group 7 RCS – Pressurizer Piping to PORVs

Identification	Material/Environment combination/Full power temperature/pressure	Degradation mechanisms considered	Susceptibility	Confidence	Knowledge	Rationale for scoring	Critical factors controlling occurrence in plant	Components in this sub-group
		1=low 2=med 3=high						
7.1	All stainless steel components External surfaces when at <150°C containment air Normally dry when at low temp	SCC	1	3	3	Well known phenomenon. Cl from insulation and ocean aerosols, the latter increasing with time	Concern only if wet. Tolerance level for Cl depends on silicate buffer in insulation	All
		PIT	1	3	3	Well known phenomenon. Cl from insulation and aerosols, the latter increasing with time	Concern only if wet. Tolerance level for Cl depends on buffer availability from insulation	All
7.2	Wrought austenitic stainless steel piping Types 304, 316 Stagnant saturated steam/condensate Primary water 345°C, 15.5 MPa	FAT	1	2	1	Doubts on magnitude of environmental effects for fatigue life particularly in view of possible concentration of hydrogen in the steam phase	Fatigue loading of normally stagnant line	6,12
		SCC	0	3	3	No basis either from laboratory data or field experience to expect SCC initiation in wrought SS	Very high level of cold work needed for cracking based on over 30 years experience	
7.3.1	Austenitic components weld HAZs, Type 304 Stagnant saturated steam/condensate 345°C, 15.5 MPa	SCC	2	2	2	Good field experience—possibly some field examples of cracking due to SCC	Not anticipated to be a long term problem due to negligible oxygen	2,5,6,8,10,12,14,18
		FAT	1	2	1	Doubts on magnitude of environmental effects for fatigue life particularly in view of possible concentration of hydrogen in the steam phase	Fatigue loading of normally stagnant line	

Table 2.9 Evaluation Spreadsheet for Group 7 RCS – Pressurizer Piping to PORVs (Cont'd)

Identification	Material/Environment combination/Full power temperature/pressure	Degradation mechanisms considered	Susceptibility	Confidence	Knowledge	Rationale for scoring	Critical factors controlling occurrence in plant	Components in this sub-group
1=low 2=med 3=high								
7.3.2	Austenitic components weld HAZs, Type 316 Stagnant saturated steam/condensate 345°C, 15.5 MPa	SCC	2	2	2	Good field experience—possibly some field examples of cracking due to SCC	Not anticipated to be a long term generic problem due to negligible oxygen	2,5,6,8,10,12,14,18
		FAT	1	2	1	Doubts on magnitude of environmental effects for fatigue life particularly in view of possible concentration of hydrogen in the steam phase	Fatigue loading of normally stagnant line	
7.4	Austenitic to austenitic weld metals, Type 308 Stagnant saturated steam/condensate 345°C, 15.5 MPa	FAT	1	2	2	Doubts on magnitude of environmental effects for fatigue life particularly in view of possible concentration of hydrogen in the steam phase	Fatigue loading of normally stagnant line	1,3,4,7,9,11,13,15,16,17,19,20,21
		FR	1	3	3	Well characterized. Not as susceptible to thermal aging as cast austenitics due to lower ferrite and lower Cr in ferrite phase. Toughness adequate even after aging to lower limit.	Known issue; effective prediction models.	
		SCC	1	3	3	Very good experience; no known cracking due to SCC after more than 30 years operating experience	Not anticipated to be a long term problem due to negligible dissolved oxygen	

Table 2.9 Evaluation Spreadsheet for Group 7 RCS – Pressurizer Piping to PORVs (Cont'd)

Identification	Material/Environment combination/Full power temperature/pressure	Degradation mechanisms considered	Susceptibility	Confidence	Knowledge	Rationale for scoring	Critical factors controlling occurrence in plant	Components in this sub-group
1=low 2=med 3=high								
7.5	Forged austenitic stainless steel nozzles Types 304, 316 Stagnant Saturated steam/condensate 345 °C, 15.5 MPa	FAT	1	2	2	Doubts on magnitude of environmental effects for fatigue life particularly in view of possible concentration of hydrogen in the steam phase	Fatigue loading of normally stagnant line. Surface finish a known influencing factor	2,5,8,10,14,18
		SCC	0	3	3	No basis either from laboratory data or field experience to expect SCC initiation in forged SS. More than 30 years of satisfactory operating experience.	Very high level of cold work	
7.6	Socket welds, Types 304, 316 Stagnant saturated steam/condensate 345°C, 15.5 MPa	FAT	2	3	2	Significant field experience of failures	Depends on design detail and flow induced vibration	4, <2.5 cm instrumentation piping and access plugs
		SCC	1	3	3	No known evidence that field cracking is due to SCC	Not anticipated to be a long term problem due to negligible dissolved oxygen	

2.4 Technical Challenges

In their assessment of materials degradation, the panel members addressed degradation modes such as general corrosion, fatigue and irradiation embrittlement which were part of the original design basis. However the panel also considered numerous degradation modes associated with localized corrosion (e.g., microbiologically-assisted corrosion, crevice corrosion, stress corrosion cracking) that were not originally considered in the initial design-basis evaluations. The technical challenges in addressing these latter degradation modes need to be understood, at least in broad context, before discussing the judgment criteria that were used in the elicitation process. (More detailed discussions of the technical issues are given in Appendices A and B).

A difficulty facing the panel members in assessing the future behavior of a large number of LWR components made from a variety of metallic materials and fabricated by different processes was that the extent of the degradation involves complex interactions between the various metallurgical, environmental and stressing parameters, and this becomes of critical importance when considering localized corrosion. The complexity of these interactions is discussed below, using as an example the initiation and growth of a stress corrosion or corrosion fatigue crack, as illustrated schematically in Figure 2.3.

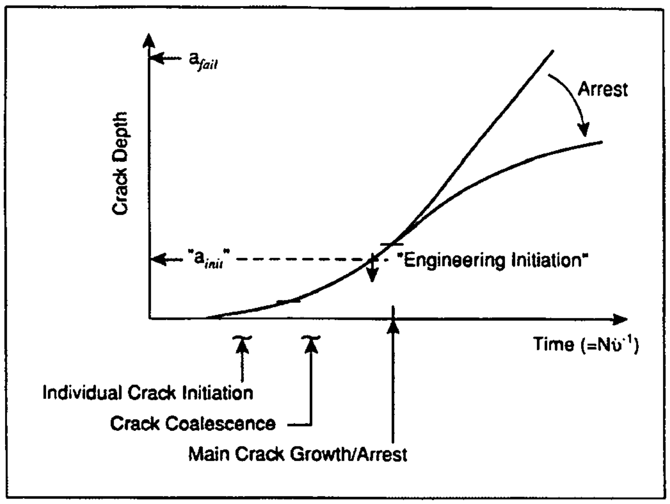


Figure 2.3 Sequence of crack initiation, coalescence and growth during subcritical cracking in aqueous environments [2] (*reprinted with permission from TMS)*

In this case, cracks can initiate on a microscopic level at surface inhomogeneities associated with fabrication or design defects such as scratches, cold worked regions or weld defects, or at corrosion-based artifacts such as pits. In other systems such as more corrosion resistant stainless steels, crack initiation may occur due to intergranular attack caused by the breakdown of passivity over a grain boundary. These corrosion-based defects arise due to time-dependent phenomena, which are precursors to the cracking phenomenon and are largely stochastic in nature. The micron-sized cracks that initiate from these individual surface imperfections may grow or arrest, dependent on the specific material, stress and environment conditions. They may then coalesce, depending on the geometric spacing of the microcracks to form a larger crack. The resultant crack will only be detectable in an engineering structure when its depth is considerably greater, dependent on the specifics of the inspection technique. This fact immediately poses an interpretational challenge to the analyst, since "initiation" is normally defined by the engineer when the crack is detectable (which is typically a depth of 2 mm or greater), whereas the scientist will concentrate on the details of damage accumulation at the micron level. In this current assessment the focus has been on the former interpretation of "initiation."

In some cases, the initiation of microcracks may start very early in life at preexisting surface inhomogeneities such as scratches. In other cases the sequence of events illustrated in Figure 2.3 may be delayed for many years due to the formation of a specific localized chemistry in a crevice, or the development of a "susceptible" material microstructure due to the accumulation of a specific amount of irradiation fluence. The analysis of future degradation behavior is complicated by the difficulty in quantifying these "precursor" events.

Assuming that the local conditions are met for the sequence of events in Figure 2.3 to proceed, it can be argued that the physical process of cracking should exhibit an inherent variance and be appropriately analyzed in a probabilistic manner, since it has been shown [3-5] that the processes that control the early crack initiation process, such as pitting, intergranular attack and crack coalescence, are stochastic phenomena. Thus, several studies [4-14] indicate (Figure 2.4) that (engineering) crack initiation times may be predicted by such a probabilistic approach. Such an approach is pertinent to the prediction of the distribution of damage in a given system, provided the values of the Weibull analysis are known. The point is that, depending on the particular system, the range of crack "initiation" times predictably range over many orders of magnitude, depending on the specific materials, environmental and stress conditions.

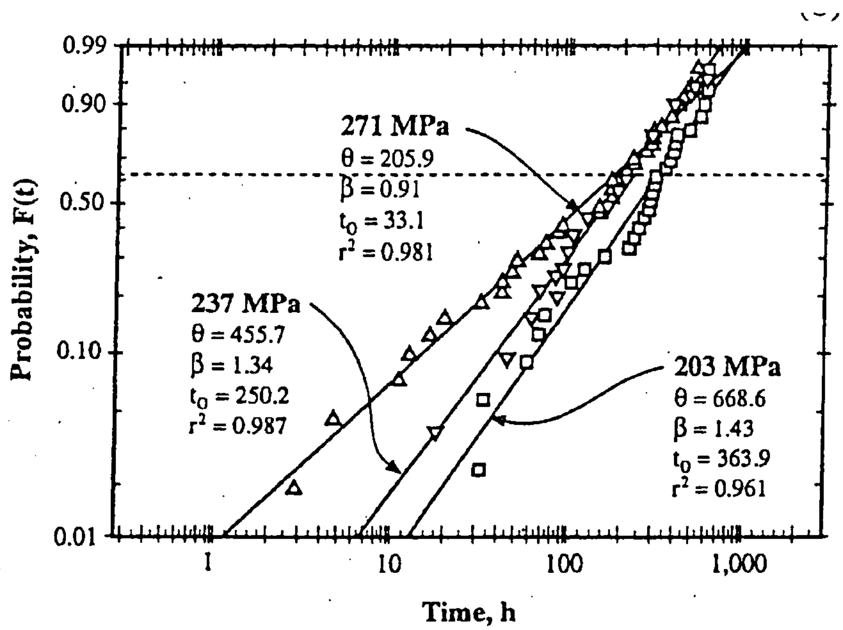


Figure 2.4 **Probability vs. time for initiation of stress corrosion cracks in sensitized type 304 stainless steel in 288°C, 8 ppm oxygenated water; original data from Ref. 9, replotted in Ref. 11.** ***(Courtesy of The Journal of the Society of Materials Science Japan)***

An equally important requirement for a life prediction methodology is the assessment of crack propagation once the cracks are detected and are perhaps 2 mm or greater in depth. The quantitative growth of the crack is a function of the material, stress, and environment conditions over time. The development of such damage algorithms may be based on empirical analyses of a crack propagation rate database, or on knowledge of the mechanism of damage accumulation. There have been considerable advances over the last 20 years in both of these areas for LWR systems [15-22]. For instance, as indicated in Figure 2.5, the cumulative percentage of cracked nickel-base alloys in PWR vessel head penetrations may be assessed via analysis of past failures and a Monte Carlo analysis of the relevant system parameters.

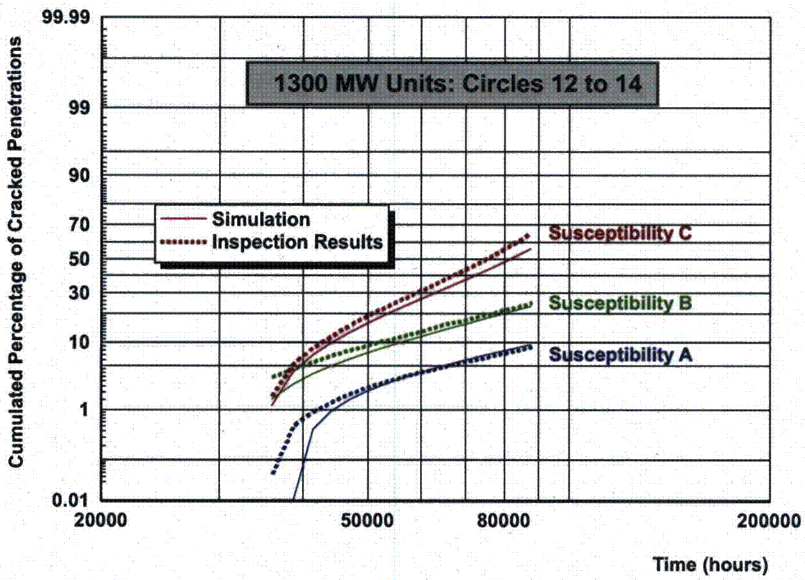


Figure 2.5 **Predicted and observed percentage of the cracked CRDM tubes in French PWR vessel head penetrations as a function of time, material susceptibility and reactor design which determines such parameters as the residual stress, temperature, etc.** The "circles" in the subtitle refer to tubes at a similar angular offset from the centerline of the reactor pressure vessel, and are pertinent to the definition of the residual stress profile [22].

These aspects are discussed in some detail in Appendices A and B, but it is sufficient at this stage to note that the preciseness of the predictions depend on the definition of the governing system parameters. This is illustrated in Figure 2.6 for the growth of stress corrosion cracks in an irradiated type 304 stainless steel BWR core shroud [16]. It is apparent that the extent of observed damage accumulation is reasonably predicted by theoretical trend lines that were developed via an understanding of the mechanism of cracking. It is significant, however, that the two predicted relationships in Figure 2.6 correspond to a relatively minor change in the weld residual stress profile. Such sensitivity emphasizes the need for adequate system definition, and the fact that this is not always possible automatically places a (quantifiable) uncertainty on the crack growth predictions.

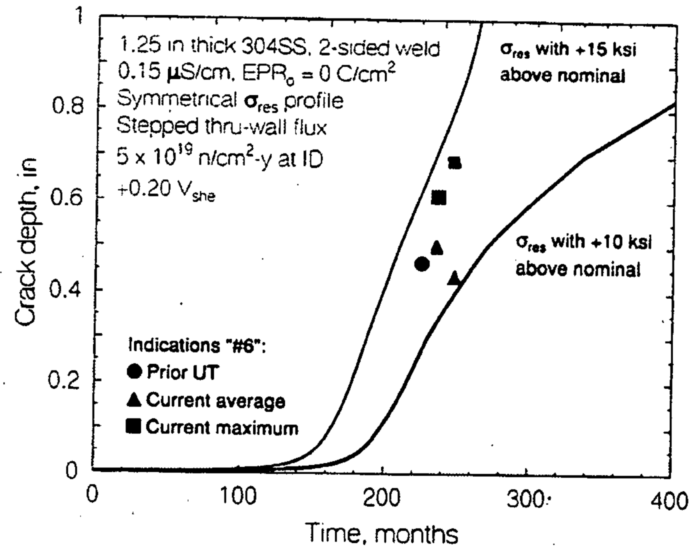


Figure 2.6 **Comparison between the predicted and observed crack depth/operating time relationships for a crack in the HAZ of belt line weld of a core shroud.** The predicted curves correspond to the span of the calculated residual stresses in this component. [16]

These examples outline some of the limitations facing the expert panel in terms of the prediction of stress corrosion cracking, and many of these limitations are mirrored in analyzing degradation by other modes. It is apparent that unless there are well-qualified algorithms (either empirically- or mechanistically-based) between component lifetime and the relevant system parameters, it is unlikely that an assessment of future performance of the component will be completely quantitative.

Fortunately, past experience of materials damage does put these problems into some historical perspective. For instance, the schematic damage vs. time plots in Figure 2.7 indicate three cases that bound the past "reactive" life management style, and set out some of the steps needed for the future "proactive" management style. On the time scale in this figure "now" denotes the boundary between "reactive" and "proactive" life management approaches.

The situation for "Case 1" is epitomized by the intergranular stress corrosion cracking of stainless steel piping observed in BWRs in the 1970s and 1980s. This is now a well-understood problem for which there are well-developed mechanistically-based prediction methodologies (Figure 2.8) for crack growth [15]. Consequently mitigation actions are based on a theoretical understanding of the complex interactions between the material, stress and environment components, and these actions may be implemented with appropriate system control and inspection. With this theoretical base, problems that emerged in the 1980s and 1990s associated with cracking of irradiated stainless steel BWR core components could be more efficiently managed, and mitigation actions, such as noble metal technology, relatively quickly deployed.

This situation is compared with the illustration for "Case 2" in Figure 2.7, which is relevant to the current problem of boric acid corrosion (BAC) of the low alloy pressure vessel steel in PWR vessel head penetrations (i.e., the Davis Besse incident), Figure 2.9. In this case the "precursor" events were the initiation and propagation of a stress corrosion crack in the nickel-base alloys used in the control rod drive mechanism (CRDM) assembly, the introduction of primary water into the annulus between the pressure vessel and the CRDM tube, and the formation of a specific localized chemistry in that crevice. As with all environmentally-assisted damage modes there is an expected distribution in the extent of damage reflecting the stochastic processes occurring early in the damage development, and the uncertainties associated with modeling and system definition.

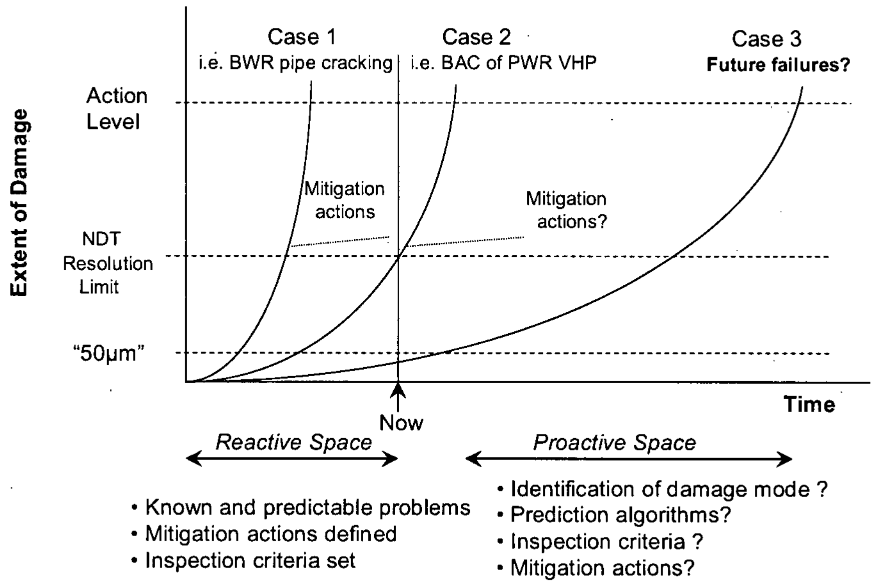


Figure 2.7 Schematic development of damage with time for various "cases" that span reactive and proactive "space"

The practical question here (quite apart from addressing the failure to address the problem in a timely manner) is: "Can we expect other similar incidents, for which the Davis Besse case is part of a wider distribution?" In fact, there have been many similar though less serious incidents including at least one case where almost the entire thickness of a low alloy steel pipe was corroded away to the internal stainless steel cladding due to a boric acid leak from a valve above). Other incidents have also occurred including failures of low alloy steel bolting of flanged joints exposed to high velocity steam jets (steam cutting) from primary side leaks where the degradation was probably mainly caused by erosion with a contribution from BAC where condensation occurred. Unfortunately, we do not know with certitude the specific system conditions that can lead to extensive BAC under boric acid deposits; in many cases relatively little corrosion of carbon or low alloy steel occurs under such deposits. We cannot readily explain, for instance, the fact at Davis Besse (Figure 2.9a) that there was extensive BAC at one penetration (#3), but no BAC damage at an adjacent and nominally identical penetration other than as a possible consequence of different elapsed times from the first occurrence of a leak in the two penetrations. Consequently a defensible proactive mitigation action is difficult to define at this time, and degradation management can be accomplished by reliable inspection and timely repair or replacement.

These two cases represent the ends of a spectrum of situations associated with a reactive approach to life management for components subject to environmental degradation.

"Case 3" in Figure 2.7 illustrates the domain to be considered in a purely proactive assessment. In this domain, damage has not developed "now" to an extent that it is readily detected in plant situations, and the question is: "how long will it be before a particular damage mode becomes detectable?" The answer to this question is challenging, especially for degradation mechanisms not previously encountered in service, or when early indications of damage appears in plants and the rate of future damage must be estimated. The approach in these cases is to draw on an understanding of the mechanisms of degradation, on relevant observations in laboratory experiments, and on experience with degradation that has occurred in other nuclear or non-nuclear systems that bear some comparison to the system under examination.

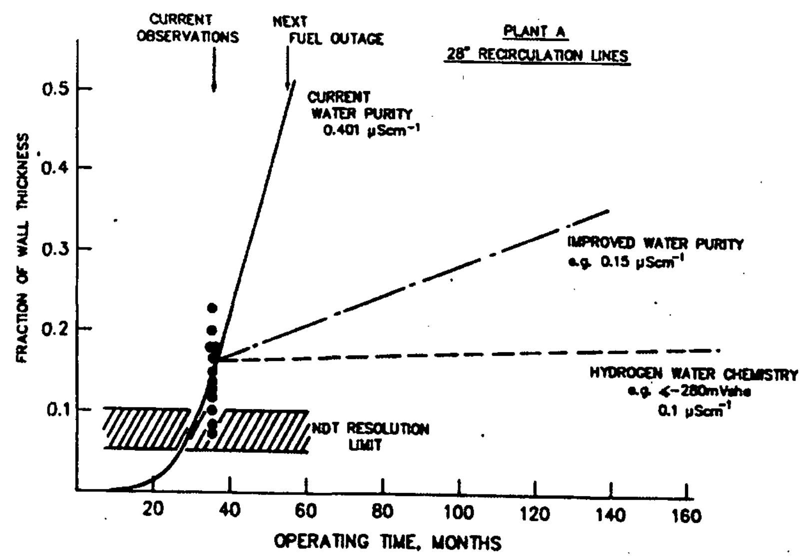
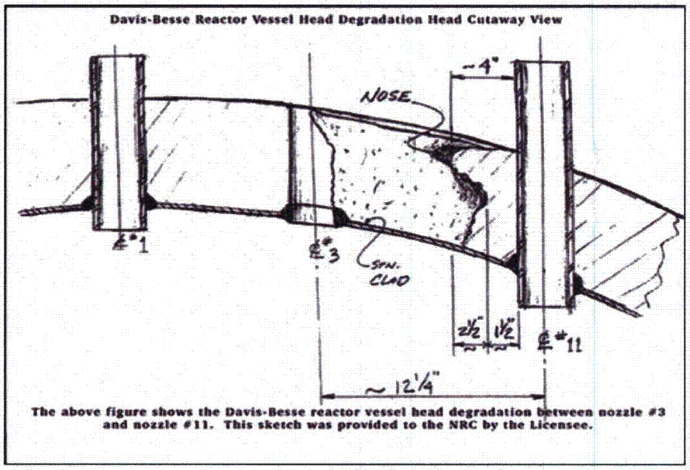
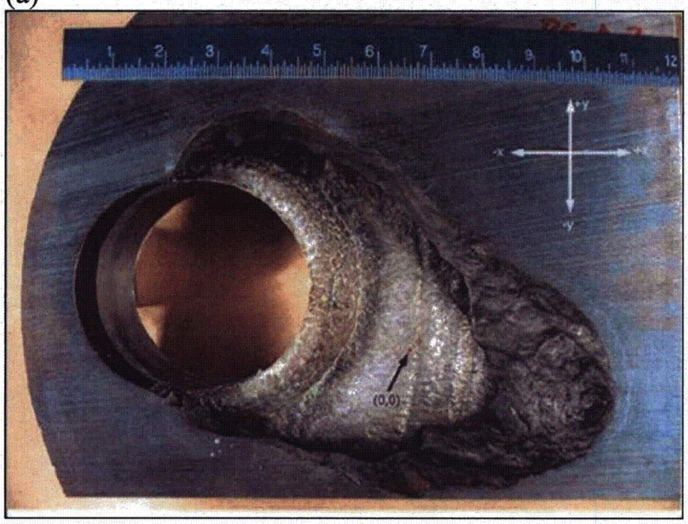


Figure 2.8 **Prediction of extent of observed intergranular stress corrosion cracking in welded Type 304 stainless steel at a reactor operating under specified water chemistry conditions, and the predicted cracking rate associated with different water chemistry regimes. [15]** ***(used by permission of EPRI)***

Potential conclusions from such a proactive assessment might be that future occurrences are to be expected with stress corrosion cracking of nickel-base alloy steam-generator tubes due to the presence of lead and sulfur contamination, or with cracking of PWR pressure vessel head penetration assemblies replaced by alloys 690/52. Thus, it may be desirable to develop a control/mitigation strategy well before generic operational issues arise.



(a)



(b)

Figure 2.9 **(a) Location of boric acid corrosion attack of the low alloy steel pressure vessel at Davis Besse, adjacent to cracked regions in the penetration subassembly that allow access of primary coolant to the low alloy steel. (b) Picture of cleaned top surface of the head, showing the exposed stainless steel cladding which was the sole remaining pressure boundary in this region.**

2.5 Panel Evaluation Process

The expert panel examined the information described in Section 2.2 and the operating history of PWR and BWR system components and, with this input, evaluated the potential for future materials degradation on the basis of a) the components' behavior in the past, and b) the available laboratory data and mechanistic knowledge required to make a judgment of future component behavior.

The experts were asked specifically to give their judgment on the following three attributes for each component/part with a score for each attribute to denote the degree to which that judgment was made. The scores 1, 2, and 3 are sometimes interpreted as low, medium, and high, respectively.

Susceptibility Factor – can significant material degradation develop given plausible conditions?

0 = not considered to be an issue
1 = conceptual basis for concern from data, or potential occurrences under unusual operating conditions, etc.
2 = strong basis for concern or known but limited plant occurrence
3 = demonstrated, compelling evidence for occurrence, or multiple plant observations.

Confidence Level - personal confidence in the judgment of susceptibility

1 = low confidence, little known about phenomenon;
2 = moderate confidence
3 = high confidence, compelling evidence, existing occurrences
Note: "3" is assumed if Susceptibility Factor is "0."

Knowledge Factor – extent to which the relevant dependencies have been quantified

1 = poor understanding, little and/or low-confidence data;
2 = some reasonable basis to know dependencies qualitatively or semi-quantitatively from data or extrapolation in similar "systems";
3 = extensive, consistent data covering all dependencies relevant to the component, perhaps with models; should provide clear insights into mitigation or management of problem.

Inherent in arriving at these judgments of the future behavior of the components is an understanding of the prediction methodologies for the various degradation phenomena, calibrated by the component failures that have occurred in the past in the global LWR fleet. Also taken into account were the successes and limitations of mitigation/control approaches that have been used to date.

The evaluations were done by each panel member following a series of seven week-long meetings attended by the members. During these meetings the various degradation modes and operating experience for specific component groups were discussed. The scoring of each member on the attributes detailed above was done privately, with no aim at arriving at a consensus opinion.

After evaluations by all panel members were completed, the degradation mechanism scoring was discussed so that each panel member could understand the thinking of the other members. No attempt was made to reach a consensus in the scoring, but panel members could change their scores if they wished. Quality assurance was a concern and in addition to having an independent reviewer (BNL) for the evaluations, each of the panel members reviewed their scoring at the end of the process to assure that:

- All listed degradation mechanisms had been scored or purposely left blank
- Comments were consistent with scoring
- There was consistency across groups where the part was similar.

2.6 Other Panel Activities

In addition to the evaluation process described above, it should be noted that the panel members also analyzed materials degradation through their writings and discussions at meetings. Appendix B is a set of background papers on degradation mechanisms written by the panel members. These reports were discussed along with many other aspects of degradation at the panel meetings. The results of this activity are found in Section 3.4, "Generic/Non-component Specific Issues and Associated Research Needs."

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3. DISCUSSION OF PIRT EVALUATION RESULTS

3.1 Introduction

As described in Section 2, the panel members evaluated degradation mechanisms for 386 subgroups of PWR parts and 297 subgroups of BWR parts. In addition to the 297 subgroups evaluated for the BWR, there were 84 subgroups evaluated for PWRs that are in common with, and the results apply to, the BWR evaluation. The scoring and comments were put together in individual tables for each relevant degradation mechanism for each subgroup. Generally there are eight evaluations in each table. For some degradation mechanisms, however, there are fewer as some panel members did not conduct the evaluation, generally because they had insufficient knowledge to perform the evaluation. The resulting 1222 tables for PWRs and 1322 tables for BWRs, compiling the individual evaluations, are found in Appendices E and F, respectively, and were used in drawing conclusions.

Another method used for representing the judgment scores and arriving at general conclusions relating to materials degradation susceptibility and potential management by mitigation is based on aggregating the panel members' scores for susceptibility and knowledge level for each degradation mechanism for each subgroup. This method is illustrated schematically in the colored susceptibility-knowledge diagram of Figure 3.1. In this diagram, three colors are used to depict the degree of susceptibility. The color shading represents the level of knowledge. The susceptibility scale ranges from 0-3, and the knowledge scale from 1-3. The numerical scores 1, 2, and 3 are generally interpreted as low, medium, and high respectively. A score of 0 (zero) for the degree of susceptibility was used when a given panel member judged that there was no reasonable chance that the specific degradation would occur under the given conditions. More precise definitions of the numerical scores for susceptibility, knowledge, and confidence level are given in Section 2.5. Susceptibility scores between 0-1 fall in the green region of the diagram considered to be the low susceptibility region. Susceptibility scores between 1-2 fall in the medium susceptibility yellow region, and scores between 2-3 fall in the high susceptibility red region. The placement of susceptibility scores that fall at the interface between colors is discussed later in this section. Knowledge scores between 1-2 fall in the light shade of the susceptibility color regions (green, yellow, or red) considered to be the low level of knowledge field of the diagram. Knowledge scores between 2-3 fall in the dark shade of the color regions considered to be the high knowledge field of the diagram. Knowledge scores of 2 are assigned to the low knowledge, light shade color field.

Such diagrams are central to the interpretation of the collective judgment of the individual panel members. This interpretation would provide a broad overview of the likelihood of future materials degradation and identify research needs for mitigation actions, thus providing one basis for proactive materials degradation management. For instance, a combination of aggregate scores of "degradation susceptibility" and "knowledge" associated with the light red and light yellow fields in Figure 3.1 would denote the less-than-desirable situation of a component that is likely to undergo degradation, but for which there is insufficient knowledge of the system interdependencies to formulate appropriate mitigation actions. Aggregate scores in the top-right and middle-right (dark-red and dark-yellow) portion of the diagram indicate a high susceptibility to degradation, but there may be sufficient knowledge available to develop mitigation actions. In both cases, implementation of proactive materials degradation management programs could avoid future occurrences in the plants. However, in the first case (light red and light yellow fields), additional proactive actions would be needed to develop the knowledge for managing the potential degradation by mitigation.

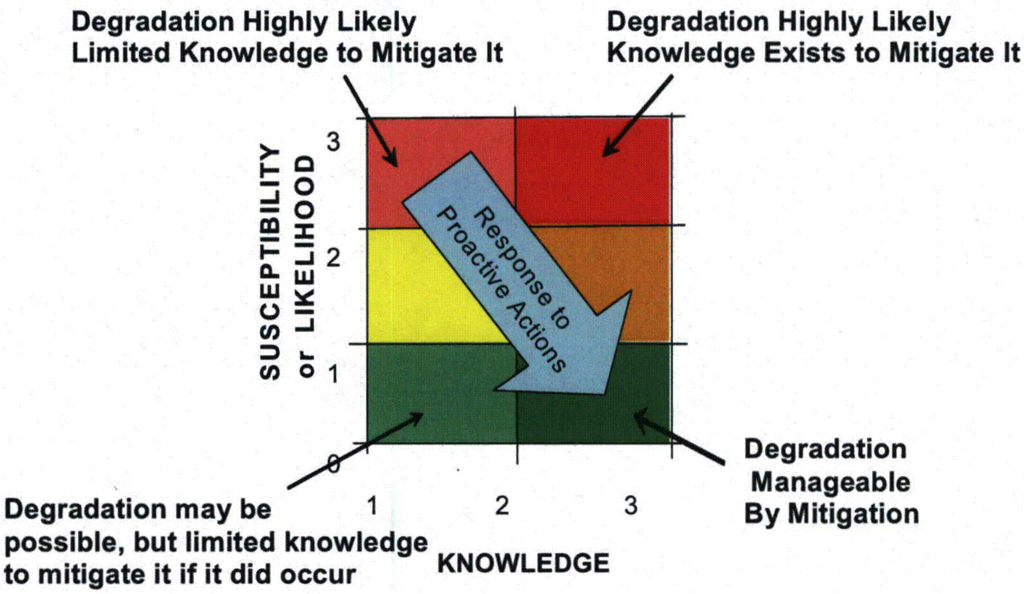


Figure 3.1 Schematic illustrating the combinations of "Damage Susceptibility" and "Knowledge" scores suggesting various life-management responses.

It is recognized that even when adequate knowledge is available for the important parameters and dependencies that affect the degradation mechanism, implementation of degradation management by mitigation may not always be feasible or desirable. For example, it may not be possible nor cost effective to change or control plant operating and materials conditions such as temperature; operating stress; chemical environment; microstructure; residual stress; number, frequency, and level of cyclic loads; etc. In these and other cases, proactive materials degradation management can be achieved through the effective detection, characterization, and monitoring of degradation in susceptible components in conjunction with the timely repair of components before structural integrity or safety is compromised. It is important to note, that this method of proactive materials degradation management will also require the availability of technology and knowledge such as effective in-service inspection and/or continuous crack monitoring techniques in addition to crack initiation and growth rate information for implementing effective timing and frequency of inspections, and timely repairs. It is important to emphasize that the panel members considered the level and kind of knowledge required for PMDM by the mitigation option only. Thus, any discussions, assessments, and evaluations of knowledge and research needs, related to degradation management apply to degradation management by mitigation strategies only. It should also be noted that the panel generally did not evaluate the efficacy and deployment of various mitigating actions. However, BWR components were scored for susceptibility under both "Hydrogen Water Chemistry/NobleChem™" and "Normal Water Chemistry" conditions since all of the U.S. BWRs are currently operating with NobleChem™ or moderate hydrogen water chemistry for most, but not all, of the time.

Combinations of aggregate scores of "degradation susceptibility" and "knowledge" placed in the light-yellow field in Figure 3.1 were of particular interest. These were cases where there was little (or no) evidence of degradation in the plants to date, but there was sufficient evidence from laboratory investigations, for instance, to indicate that degradation in the plant might be expected in the future. Moreover, if such degradation did occur, there is currently insufficient knowledge of the system interdependencies to mitigate it. These situations would fall into the category of issues depicted in Figure 2.7 as potentially benefiting from research studies and proactive actions.

As knowledge of the interdependencies between the degradation susceptibility and the system parameters (temperature, stress, material, etc.) is increased (i.e., moving to the right hand side of the diagram in Figure3.1), the inspection periodicity can be adjusted and mitigation actions can be developed and deployed (as feasible, desirable, or necessary) to decrease the likelihood of degradation.

An example of a susceptibility-knowledge evaluation is given in Figure 3.2 for PWR pressure vessel internals; in this case there are five degradation mechanisms considered for high-strength baffle bolts exposed to irradiation levels of dpa (displacements per atom) values >0.5 (subgroup 12.12). The data points in this diagram represent the average scores of the panel members and place the components in the medium and high susceptibility regions for the various degradation mechanisms indicating that there is a reasonable chance that such degradation modes may occur later in the life of PWRs. Moreover, there is reasonable confidence in these judgments, as indicated by the scoring at the side of the diagram.

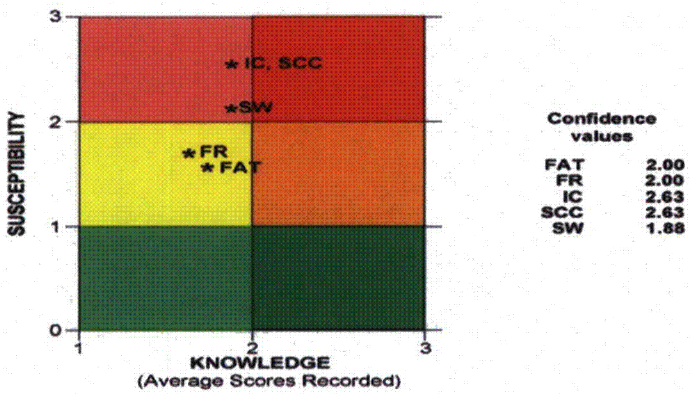


Figure 3.2 **Average scores of the panel members for degradation mechanisms of high-strength baffle bolts in PWR pressure vessel internals at dpa>0.5. The average confidence values for the damage susceptibility are indicated at the right.**

Evaluation of the potential degradation of Alloy 182/82 welds in PWR primary side environments (Subgroup 10.8) is indicated in Figure 3.3. A comparison of Figure 3.2 and Figure 3.3 shows that the situation for these two evaluations is similar, i.e. the components in both cases fall in the medium and high susceptibility regions for the degradation mechanisms evaluated and the level of knowledge tends to be in the low-to-medium level for both cases. It should be noted, however, that review of individual panel member scores in Appendix D (found on attached CD) for these two examples shows that the panel members' "knowledge" ratings in the case of Figure 3.3 extend from high, indicating that there is enough knowledge to develop mitigation actions, to low, suggesting that there is not enough knowledge to accomplish this. This is in contrast to the knowledge ratings for the case of Figure 3.2 where, in general, the knowledge calls from the panel members are more consistent.

Figure 3.4 depicts the panel's average scores for the degradation of the PWR low alloy steel in the pressure vessel shell (subgroup 10.2) due to six different degradation mechanisms. Some of these degradation modes have been the focus of extensive empirical and fundamental studies over the last 35 years and it is not surprising that the "knowledge" scores for fatigue and fracture resistance are high.

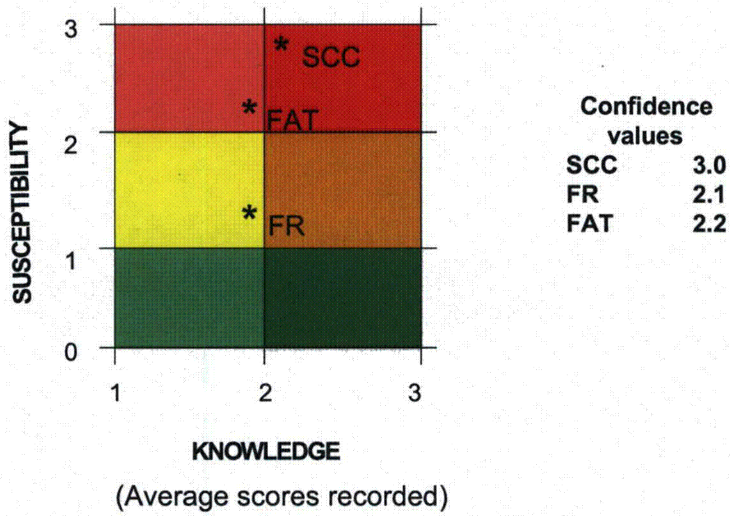


Figure 3.3 Average scores of the panel members for degradation mechanisms of Alloy 182/82 welds in PWR primary side environments. The average confidence values for the damage susceptibility are indicated at the right of the diagram.

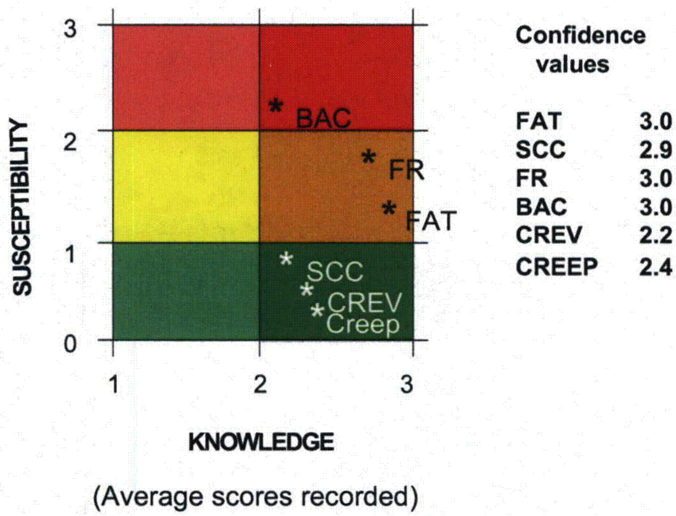


Figure 3.4 Average scores of the panel members for degradation mechanisms for PWR pressure vessel low alloy steel shell.

The individual and aggregated scores for susceptibility, confidence, and knowledge level for each of the (1222+1322) 2544 sets of evaluations are found in Appendix D (on attached CD). A sample from the full list is given in Table 3.1; this compilation is often referred to as a "flag" table. It shows, for several of the subgroups in Group 7 (Reactor Coolant System; Pressurizer Piping to PORVs) each of the degradation mechanisms that were scored, the average scores, the distribution of scores among the panel members, and the statistical mode for the susceptibility factor.

Table 3.1 Sample Summary of Aggregated Evaluations

7 Reactor Coolant System													Pressurizer Piping to PORVs
	Susceptibility					Confidence				Knowledge			
	Average	0	1	2	3	Average	1	2	3	Average	1	2	3
Subgroup 7.1	All stainless steel components External surfaces when at <150°C Containment air Normally dry when at low temp												
PIT	1.00	0	8	0	0	3.00	0	0	8	2.88	0	1	7
SCC	1.00	0	8	0	0	3.00	0	0	8	2.88	0	1	7
Subgroup 7.2	Wrought austenitic stainless steel piping Types 304, 316 Stagnant saturated steam/condens PWR primary water, 653°F, 2250 psia												
FAT	1.13	0	7	1	0	2.38	0	5	3	1.75	3	4	1
SCC	0.80	1	4	0	0	2.60	0	2	3	2.20	0	4	1
Subgroup 7.3.1	Austenitic components weld HAZs Type 304, Stagnant saturated steam/condensate PWR primary water, 653°F, 2250 psia												
FAT	1.38	0	5	3	0	2.00	0	8	0	1.63	3	5	0
SCC	1.50	0	4	4	0	2.50	0	4	4	2.00	2	4	2
Subgroup 7.3.2	Austenitic components weld HAZs Type 316, Stagnant saturated steam/condensate PWR primary water, 653°F, 2250 psia												
FAT	1.38	0	5	3	0	2.00	0	8	0	1.63	3	5	0
SCC	1.38	0	5	3	0	2.38	0	5	3	2.00	2	4	2
Subgroup 7.4	Austenitic to austenitic weld metals Type 308, Stagnant saturated steam/condensate PWR primary water, 653°F, 2250 psia												
FAT	1.25	0	6	2	0	2.00	0	8	0	1.75	2	6	0
FR	1.20	0	4	1	0	2.40	0	3	2	2.00	1	3	1
SCC	1.25	0	3	1	0	2.25	0	3	1	2.00	1	2	1
Subgroup 7.5	Forged austenitic stainless steel nozzles Types 304, 316, Stagnant saturated steam/conder PWR primary water, 653°F, 2250 psia												
FAT	1.38	0	5	3	0	2.00	0	8	0	1.88	1	7	0
SCC	0.83	1	5	0	0	2.50	0	3	3	2.17	0	5	1
Subgroup 7.6	Socket welds Types 304, 308, Stagnant saturated steam/conder PWR primary water, 653°F, 2250 psia												
FAT	2.25	0	0	6	2	2.75	0	2	6	1.75	2	6	0
SCC	1.33	0	4	2	0	2.67	0	2	4	2.17	1	3	2

* Susceptibility at interface between colors with one or more outliers.
X Susceptibility inside color box with one or more outliers.

Figure 3.5 shows an example of a "rainbow" chart from Appendix D. The rainbow charts in Appendix D provide the results for all the subgroups of Components and associated degradation mechanisms evaluated by the panel in terms of the color regions shown in Figure 3.1 and of additional symbols as discussed below. Figure 3.5 shows the assessment for two PWR subsystems, the steam generator and the service water pump discharge piping. The assessment addresses various degradation modes relevant to these two systems (e.g., boric acid corrosion (BAC), stress corrosion cracking (SCC), fatigue (FAT), pitting (PIT), reduction in fracture resistance (FR), etc.). The assessments for these subsystems were chosen for the following illustrative purposes. As expected from the incidences in nuclear plants, "red" colorations are associated with stress corrosion cracking of nickel base Alloy 600 components in the steam generator and with pitting and microbiologically-induced corrosion of steel components in the lower temperature, lower water quality conditions that exist in the service water piping. Equally important are the many system/degradation mode combinations colored "green" or "yellow" for which the expert panel scores indicated a lower potential for future degradation. Only a small fraction of the several thousand PWR and BWR component-degradation mode combinations evaluated by the panel were placed in the "high degradation susceptibility" ("red") categories. It is important to note that, in general, the panel members used a high susceptibility score of three (red) when degradation had been experienced in operating plants and a score of two (yellow) where there was a strong basis for the occurrence of degradation, but limited plant occurrence, thus far. Therefore, in a proactive materials degradation management framework, it is important to consider addressing the components that fall in the red and yellow categories to limit potential future degradation.

Subgroup Description		Degradation Mechanism									
		BAC	CREV	DEBOND	FAC	FAT	FR	MIC	PIT	SCC	WEAR
Steam Generator											
11.1	SS External Surface										
11.2	Shell/Plates, Forgings				*				*	*	
11.3	LAS Nozzles/Welds	*			*				*		
11.4	308/309 SS Chaneel Head Clad			*			*				
11.5	Alloy 600 MA SG Tubes etc.										
11.6	Alloy 600 MA SG Tubes Sec. Side										
11.7	308/309 Dissimilar Welds - Int.										
11.8	Forged 316 SS Nozzles										
11.9	Alloy 600 Divider Plate										
11.10	SA-553 Gr. A Manways										
11.11	Alloy 52/82 Channel Head Clad										
11.12	Alloy 600 TT SG Tubes etc.										
11.13	Alloy 690 TT SG Tubes etc.										
11.14	Alloy 600 TT SG Tubes Sec. Side										
11.15	Alloy 690 TT SG Tubes Sec. Side										
11.16	Alloy 82/182 Dissim. Welds - Int.										
11.17	Type 308/309 Dissim. Welds - Ext.										
11.18	Alloy 82/182 Dissim. Welds - Ext.										
11.19	Alloy 690 Divider Plate										
11.20	CS Drilled Hole TSP		X								
11.21	SS Line Contact/Drilled Hole TSP		*								
11.22	Alloy 600, SA Sensitized SG Tubes										
11.23	Alloy 600, SA Sens. SG Tube Sec.										*
Service Water Pump Discharge Piping											
29.1	CS Comp/Weld/HAZ Ext.										
29.2	CS Comp/Weld/HAZ (Pond)							*	*		
29.3	CCW HX Copper Zinc tubes				X					*	
29.4	CS CCW HX Shell and Tubesheets							*	*		
29.5	CCW HX SS tubes										
29.6	CCW HX Copper Nickel tubes										

NOTES * Susceptibility at color interface with one or more scores higher than interface; [x] Susceptibility inside color box with one or more scores higher than this color box upper interface.

Figure 3.5 Rainbow chart for PWR Steam Generator and Service Water Piping.

As indicated earlier, there was no attempt to reach consensus scores for the components and degradation mechanisms evaluated by the panel. This study placed value on individual members, their scores, opinions, and bases for their calls. When averages are used to place the component and degradation mechanisms in the color regions of the susceptibility-knowledge diagram of Figure 3.1, and subsequently used to compile the flag tables and rainbow charts, the individual member scores and opinions are not evident. Thus, to highlight individual member scores when those scores for the degree of susceptibility were higher than the average or the statistical mode (outliers or non-conforming calls), and when the average score fall on the border between colored regions, the following algorithm was used:

- Determine mode for susceptibility (S) score; for a multimodal case, the higher value of S is chosen for the mode.
- Determine the average for S and knowledge (K) score.
- Take the value of S as the higher of the average and the mode.
- Choose color according to the value of S and K
    - If S = 1 or 2 (i.e., at a color box interface):
        - upper color is chosen if at least one score exists that is higher than the value of S (so called outlier), otherwise lower color is chosen, and
        - An asterisk next to the color indicates the existence of an outlier.
    - If S < 1,or if S < 2 and > 1 (i.e., inside a color box)
        - an "X" next to the color indicates the existence of an outlier (i.e., at least one score exists higher than this color box upper interface)
    - If K = 2, the left column colors (shades) are chosen.

Note that the **bold number** in the flag tables under the susceptibility columns such as in Table 3.1, shows the number of calls at the statistical mode. There are no examples of "outliers" in Table 3.1.

The flag tables and rainbow charts produced in this study using the color scheme of Figure 3.1 and the algorithm discussed above provide a means to gain quickly a broad overview of the consensus that might have been reached if the panel had evaluated collectively, rather than individually, the several thousands of component and degradation mechanism combinations. These tables and charts are used extensively as a basis for the discussion of results in Section 3.2 for PWRs and in Section 3.3 for BWRs where the results are presented in terms of aggregated scores and color regions for different component-degradation combinations. Note that in these discussions, the average score for susceptibility is reported even when the mode was higher but using the mode did not result in placement of the component-degradation mechanism combination in an upper color region as compared to placement by the average score. When reviewing results in Sections 3.2 and 3.3, a number of considerations need to be taken into account. As illustrated in this section under the comparison of Figures 3.2 and 3.3, particularly for the knowledge scores, the color diagrams do not convey the variability and range in panel members scores. In the example discussed for Figure 3.3, the knowledge level scores ranged from low, indicating not enough knowledge is available to develop mitigation actions to high, indicating such knowledge is available. Also with respect to knowledge level, for some cases that fall on either side of the color shade transition from low to high knowledge, there may not be any real difference in the actual knowledge level. For example, in cases where all eight experts assign a level of 2 (or when the average is 2) the average call is placed in the light shade-low knowledge color field; in the cases where seven experts assign a level of 2 and one expert assigns a level of 3, the average call is placed in the dark shade-high knowledge field. These two situations are nearly the same.

Finally, with regard to knowledge level, recall that only a score of 3 indicates that there is enough knowledge available to potentially develop mitigation strategies. A knowledge score of 2 indicates the existence of some knowledge, but not necessarily enough to develop mitigation strategies. Average knowledge scores in the 2 - 2.5 range may have been made up of individual scores containing more 1s and 2s than there were 3s. Thus, although the component and degradation mechanism was placed in the dark shade-high knowledge color field, more panel members' scores may have indicated that not enough knowledge was available to develop effective mitigation strategies. With respect to degradation susceptibility, it is important to note that in general, the panel members assigned a high susceptibility score of 3 (red) only when degradation had been experienced in operating plants, even though a 3 could also have been assigned in the case of a demonstrated, compelling susceptibility problem. A score of 2 (yellow) was assigned when there was a strong basis for the occurrence of degradation, but limited plant occurrence up to now. Thus, the components that fall in the red and yellow color regions (regardless of the knowledge level) should be considered for inclusion in PMDM programs to avoid potential future degradation occurrences and surprises.

For the reasons discussed above, when considering a) components for inclusion in PMDM programs, or b) the current knowledge level and the need for further research to allow development of mitigation strategies, the color charts should be used in combination with a detailed review of individual expert panel member scores, rationale, and comments for the component/degradation mechanisms under consideration. The aggregated panel member scores and the color assignment for the component/degradation mechanism combinations discussed in Section 3.2 and 3.3 imply the following:

- Green - degradation susceptibility and the likelihood of future occurrence is low. In a proactive materials degradation management program, these would be addressed last.
- Yellow - strong basis for occurrence of degradation, but limited plant experience. These should be considered for inclusion in PMDM programs.
- Red - highly susceptible to degradation, has been experienced in operating plants. These should be considered for inclusion in PMDM programs.
- Light shade color fields - Not enough knowledge is available for developing mitigation strategies, and research is needed if such mitigation is desired.
- Dark shade color fields - Enough knowledge <u>may</u> be available for developing mitigation strategies. Individual scores and situations need to be reviewed to determine if additional knowledge is needed to develop mitigation strategies, if desired.

Finally, it should be noted that the panel members did not evaluate the existence, viability, or effectiveness of mitigation or other materials degradation management strategies. Thus, even in cases where the knowledge level is high, it should not be assumed that potential degradation is being addressed, and components that fall in the red and yellow regions should be considered for inclusion in PMDM programs.

3.2 Susceptibility of PWR Components to Materials Degradation

A summary-level discussion is presented below of the results of the PIRT-like assessment of the susceptibility of selected components in a "typical" PWR to the sixteen materials degradation mechanisms defined in Section 2 (Table 2.8). Both external and internal degradation are included for those components for which both are pertinent.

The original aim of the project was to perform the degradation susceptibility assessment component by component, but it very rapidly became evident that this was an unattainable goal. The panel members therefore assessed subgroups of similar components within the groups of parts provided by BNL. It is important to note that the detail of the division into the subgroups differs from group to group. This is partly because of the variable quality of the underlying documentation defining the components and partly because of decisions made by the panel as they grouped together components which they expected would have similar susceptibilities to degradation because of similarities of component type, material composition and microstructure, and service conditions. Within any given subgroup, there was no objective reason for the panel to distinguish between one component and another of the same or similar material.

In all, 386 PWR component subgroups, from the four systems listed below, were evaluated by each member of the panel:

- 138 subgroups were from the Reactor Coolant System (RCS)
- 94 subgroups were from the Engineered Safety Features/Emergency Core Cooling System (ESF/ECCS)
- 12 subgroups were from the Steam and Power Conversion System (SS&PCS)
- 142 subgroups were from the Support and Auxiliary System (S&AS)

The color-coded results of the evaluations for all the subgroups can be found in the so-called "rainbow" and "flag" charts in Appendix D, where they are grouped according to the same four "major systems" as those listed above. In addition, Appendices E and F provide the individual susceptibility and knowledge "calls" by each panelist (together with his/her rationale) for each component subgroup/degradation mechanism combination evaluated. See attached CD for appendices D. E. and F.

As explained in Section 3.1, and illustrated in Figure 3.1, the colors red, yellow and green signify progressively lower panel scores or indications regarding the likelihood of future degradation. This process was biased to give increased weight or highlight minority opinions that a particular degradation mechanism was more likely than the statistical mode or average score would indicate. This was done, even when only one member was of this opinion. The color red is an indication that there is field experience, or demonstrated compelling evidence for the occurrence of the specific degradation mechanism in the specific type of component. The color yellow indicates that there is a good basis for expecting degradation of plant components in the future (based, for example, on laboratory test data, or limited plant observations) whereas the color green indicates that there is a low likelihood of degradation occurrence in the future. The shades of the colors signify the existing level of knowledge based on the average of the panel scores for the degradation mechanism/component combinations. The lighter color indicates a lower knowledge level and a potential need for further research. The darker color indicates a higher knowledge level that potentially can allow, without further research, the development of mitigation strategies.

Overall, it is clearly apparent from the charts in Appendix D that the susceptibility scores indicating the likelihood of future degradation are markedly higher for the RCS component subgroups than for the component subgroups from the other three major systems. For example, 33 RCS subgroups are color-coded red (indicating that there is at least one degradation mechanism with high susceptibility) whereas the total number of red subgroups for the other three major systems is only 15. Similarly, 68 RCS subgroups are color coded light yellow whereas the total number of "light yellows" for the other three major systems is only 49.

It is apparent from the shading in the charts of Appendix D that the panel's knowledge scores are low for significant numbers of susceptible component subgroups:

- 19 of the 48 total red component subgroups are coded light red
- There are 117 light-yellow subgroups and 163 dark-yellow subgroups.

The overall number of component subgroups color-coded green (58) is significantly less than the number color coded dark yellow. The finding of 163 component subgroups color-coded dark-yellow suggests that, at least for some degradation mechanisms and components, the information required for the development of mitigation strategies may be available.

A discussion of the results is presented below for each of the four "major systems." These discussions each contain subsections addressing all the materials degradation mechanisms that apply to each of the component subgroups color coded red and light yellow. For example, when a subgroup falls in the red region because one or more of the relevant degradation mechanisms is in the red region, these degradation mechanisms (red) as well as the other less susceptible degradation mechanisms for the subgroup are discussed. For component subgroups color coded dark yellow (i.e., assessed to have medium susceptibility but high knowledge levels), one or two examples are discussed, but most of the components are only tabulated in order of decreasing susceptibility and increasing knowledge. For the tabulated component groups, and those assessed to have low susceptibility (light or dark green), the reader is referred to the information provided in Appendix D. The discussions concentrate on the particular factors considered relevant for the specific component/degradation mechanism – for a more comprehensive discussion of the degradation mechanisms, readers are referred to Appendices A and B.

3.2.1 Reactor Coolant System

The reactor coolant system (RCS) as defined here includes eight piping systems, the pressurizer, the reactor pressure vessel and its internals, the reactor coolant pump and the steam generator. The RCS piping systems are the cold leg piping, crossover leg piping, hot leg piping, pressurizer spray piping, pressurizer surge piping, pressurizer piping to PORVs, pressurizer piping to SRVs and stop valve loop bypass piping.

The RCS consists of BNL's PWR groups 1 through 13, which the panel reorganized into a total of 138 component subgroups for evaluation purposes. During power operation, most of these component subgroups are exposed to PWR primary water at temperatures in the range 288-327°C (550-620°F) but some of the pressurizer and pressurizer-piping components are exposed to saturated steam/condensate at about 343°C (650°F). In addition, a number of the reactor vessel internal components (and the beltline region of the vessel itself) are exposed to neutron fluxes that can result in moderate or high neutron fluences by end-of-life. Where pertinent, the external environment for the RCS components is containment air, which is expected to contain both moisture and chloride aerosols during outages. External surfaces are generally hot [>121°C (250°F)] and dry during power operation.

The panel's assessment of the 138 total RCS component subgroups resulted in 33 red component subgroups, 68 light-yellow subgroups, 21 dark-yellow subgroups and 16 green subgroups. Stress corrosion cracking and fatigue were the dominant degradation mechanisms but, as discussed below, there were also several other degradation mechanisms identified by the panel.

3.2.1.1 RCS Components with Red Susceptibility

The subgroups in the reactor coolant systems with components falling into the red susceptibility regions are listed in Table 3.2 and illustrated in the modified rainbow chart, Figure 3.6.

Table 3.2 Red Subgroups in the PWR Reactor Coolant System

Component	Subgroups	Degradation Mechanisms Considered
Type 304/316/308 stainless steel socket welds	1.7, 2.7, 3.7, 5.6, 6.6, 7.6	Fatigue Stress corrosion cracking
Type 308/309 Stainless steel dissimilar metal welds	1.9, 2.9, 3.9	Stress corrosion cracking of the external surfaces
Alloy 82/182 dissimilar metal welds	4.6, 10.8, 11.16	Stress corrosion cracking Fatigue Reduction in fracture resistance
Alloy 600 components	4.7, 4.14, 10.9, 11.5, 11.6, 11.9, 11.12, 11.14, 11.22, 11.23	Fatigue Pitting corrosion Stress corrosion cracking Wear
High strength components	9.3, 12.7, 12.12	Fatigue Reduction in fracture resistance Irradiation creep Stress corrosion cracking Swelling
Carbon and low alloy steel components	10.2, 11.20	Boric acid corrosion Thermal creep Crevice corrosion Flow-accelerated corrosion Fatigue Reduction in fracture resistance Stress corrosion cracking
Type 304/316/308 stainless steel components	10.10, 12.4, 12.8, 12.9, 12.10, 12.11	Crevice corrosion Thermal creep Fatigue Reduction in fracture resistance Irradiation creep Stress corrosion cracking Swelling

3.2.1.1.1 Socket Welds

The socket welds which fall into the highest susceptibility category are found in the subgroups 1.7 (cold leg piping), 2.7 (crossover leg piping), 3.7 (hot leg piping), 5.6 (pressurizer spray piping), 6.6 (pressurizer surge piping), 7.6 (pressurizer piping to PORVs), all of which operate at high temperature [269-345°C (517-653°F)] with primary water, or in the case of subgroup 7.6, stagnant saturated steam condensate as the environment. The degradation mechanisms considered are fatigue and stress corrosion cracking. More extensive discussions of these phenomena can be found in Appendices A, B.14 and B.6 respectively.

Socket welds are known to fail in service (but at a low rate). The mode of failure may be a combination of high frequency vibration fatigue, corrosion fatigue and stress corrosion cracking and the panel members assessed socket welds with respect to these degradation mechanisms. The geometry of socket welds makes them prone to low and high cycle fatigue owing to the fact that they are a relatively flexible attachment to a more robust component. The loading will depend on the design details and flow induced vibrations, and there will also be residual welding stresses and possibly other stresses introduced by, for example, excessive grinding during manufacture. The condition of the stainless steel is unknown and may be sufficiently sensitized to be sensitive to stress corrosion cracking in the hydrogenated environment together with the probable presence of cold work and ripple loading.

Subgroup Description		Degradation Mechanism										
		BAC	CREEP	CREV	FAC	FAT	FR	IC	PIT	SCC	SW	WEAR
Type 304/316/308 SS Socket Welds												
1.7	SS 304/308/316 Socket Welds											
2.7	SS 304/308/316 Socket Welds											
3.7	SS 304/308/316 Socket Welds											
5.6	304/308/316 Socket Welds											
6.6	304/308/316 Socket Welds											
7.6	304/308 Socket Welds (Stagnant)											
Type 308/309 SS Dissimilar Metal Welds												
1.9	308/309 Dissimilar Weld - Ext.									*		
2.9	308/309 Dissimilar Weld - Ext.									*		
3.9	308/309 Dissimilar Weld - Ext.									*		
Inconel Alloy 82/182 Dissimilar Metal Welds												
4.6	Alloy 82/182 Dissim. Welds - Int.											
10.8	Alloy 82/182 Dissim. Welds - Int.											
11.16	Alloy 82/182 Dissim. Welds - Int.											
Inconel Alloy 600 Components												
4.7	Forged Alloy 600 Nozzles											
4.14	Alloy 600 (CW) Heater Clad/Welds											
10.9	Forged Alloy 600 Nozzles											
11.5	Alloy 600 MA SG Tubes etc.											
11.6	Alloy 600 MA SG Tubes Sec. Side											
11.9	Alloy 600 Divider Plate											
11.12	Alloy 600 TT SG Tubes etc.											
11.14	Alloy 600 TT SG Tubes Sec. Side											
11.22	Alloy 600, SA Sensitized SG Tubes											
11.23	Alloy 600, SA Sens. SG Tube Sec.											*
High-Strength Components												
9.3	High Strength Parts											
12.7	High Strength Fasteners/Springs							*				
12.12	High Strength Bolts (high fluence)											
Carbon and Low-Alloy Steel Components												
10.2	Shell/Plates, Forgings, Welds											
11.20	CS Drilled Hole TSP			X								
Type 304/316/308 SS Components												
10.10	304/308 CRDM Housing (Stagnant)									*		
12.4	Type 316 CW SS Comp (low fluenc		X							*		
12.8	304 SS Plates/Tubes (high fluence)									*		
12.9	Type 304 SS HAZ (high fluence)									*		
12.10	308 SS Weld Metal (high fluence)									*		
12.11	316 CW SS Comp. (high fluence)											

NOTES: * Susceptibility at color interface with one or more scores higher than this interface; [x] Susceptibility inside color box with one or more scores higher than color box interface.

Figure 3.6 Modified Rainbow Chart Showing Red Subgroups in PWR Reactor Coolant System

The average scores of the panel for susceptibility to fatigue were between 2.25 and 2.38 for fatigue in the socket welds in all the subgroups. The panel average of the knowledge level was 1.75. This knowledge level is insufficient with regard to the details of flow induced vibrations to permit development of adequate mitigation strategies of these components. With regard to fatigue the components were in the light red region.

The susceptibility to stress corrosion cracking was given an average of 1.33 to 1.38. Some members of the panel also indicated the lack of knowledge concerning the possibility of low temperature aging [2] and its effect on stress corrosion cracking susceptibility. The average scores for knowledge were 2.17 to 2.25. With regard to stress corrosion cracking the components were in the dark yellow region.

3.2.1.1.2 Type 308/309 Stainless Steel Dissimilar Metal Welds – External Surface

The stainless steel dissimilar metal welds which fall into the highest susceptibility category are found in the subgroups 1.9 (cold leg piping), 2.9 (crossover leg piping) and 3.9 (hot leg piping). For the external surfaces of these welds, the panel only considered stress corrosion cracking as the degradation mechanism. All the components are insulated and therefore are at or near the operating temperature [293-327°C (559-620°F)].

External stress corrosion cracking is a known issue in the weld metal (308/309 stainless steel). Cracking may even occur before the component is placed into service. These phenomena are further discussed in Appendix B.3.

The panel scored the susceptibility of these components to stress corrosion cracking as statistical mode 2 with 1 higher call which conservatively put the components into the highest (red) susceptibility group. The average panel score of the level of knowledge is 2.14 putting the components in the dark red field.

3.2.1.1.3 Dissimilar Austenitic Welds

The Inconel alloy 82/182 dissimilar metal welds which fall into the highest susceptibility category are found in the subgroups 4.6 (pressurizer), 10.8 (reactor pressure vessel), and 11.16 (steam generator). All of the systems are high temperature [291°C up to 345°C (556°F up to 653°F)] systems exposed to primary side water. It was not always possible to distinguish between stainless steel (SS 308/309) and nickel base alloy (Alloys 182/82) weldments from the list of parts available and, in these cases, the panel described the subgroup with both notations. The scoring was based on the assumption that the welds were Alloy 182/82 since these materials are considered to be much more susceptible to stress corrosion cracking than Type 308/309 stainless steels. The panel assessed stress corrosion, fatigue and reduction in fracture resistance as the potential degradation mechanisms for these components.

The panel described stress corrosion cracking in these components manufactured with alloys 182 and 82 as a generic issue which is expected to occur after exposure to PWR primary water for long periods of time (~ 130,000 effective full-power (EFP) hours). The panel also held the view that there is insufficient understanding of the problem to mitigate the cracking. Some panel members also indicated a need for development of inspection and prediction tools, not least be-

[2] As used in this report, low temperature (thermal) aging refers to exposure of materials for varying times to reactor coolant temperatures or lower. These exposures could lead to microstructural changes that may sensitize materials and render them susceptible to stress corrosion cracking and/or cause embrittlement.

cause of the long and very variable crack initiation times and large dispersion in propagation rates. Sensitization in the dilution zone near the low alloy interface, cracking in the dilution zone with low alloy steels and the possibility of low temperature aging were also listed as contributing factors. More extensive discussions of stress corrosion cracking of dissimilar metal welds can be found in Appendix B.6.

Subgroup 4.6 (pressurizer) is in the light red region and subgroups 10.8 (reactor pressure vessel) and 11.16 (steam generator) are in the dark red region. The panel pointed out that, for dissimilar metal welds manufactured with stainless steel weld metals, the service experience is very much better than for the nickel-base alloys, and that the susceptibility to stress corrosion cracking from the primary water side would correspondingly have been rated much lower.
The average scores of the panel were 2.88 for stress corrosion cracking in the dissimilar metal welds in all the subgroups. The average panel score for the knowledge level was 1.88 to 2.13 putting the components in the light region except for subgroups 10.8 (reactor pressure vessel) and 11.16 (steam generator) which fall just into the dark red region. In general, therefore, there is insufficient knowledge at this time to develop mitigation strategies.

The panel considered that these components are also likely to be affected by fatigue which could be accelerated by the primary water environment. The attachment welds to nozzles at the top of the pressurizer (subgroup 4.6) were felt to be of particular concern. The panel thought that this is only likely to be a problem if the current cumulative usage factor is greater than 0.1 (approximately). However, insufficient information was available to determine if this could be the case. Some panel members were more concerned about the environmental effects on fatigue in the stainless steel welds than in the nickel-base welds. Panel members pointed out that there are limited laboratory data for the stainless steel weld metal, but that it might be possible to use data for the corresponding wrought material.

The susceptibility to fatigue was given average scores of 1.88 to 2.13 putting the component subgroups in the yellow or, for subgroup 10.8 (reactor pressure vessel), in the red region. The panel average knowledge level scores of 2 to 2.25 indicate that there is insufficient knowledge to develop mitigation strategies. Subgroups 4.6 and 11.16 fall into the light and dark yellow fields respectively and 10.8 is in the light-red field.

The panel considered that these components might also be susceptible to a reduction in fracture resistance. This is based on laboratory data which to date are insufficient to exclude the possibility of a type of fracture toughness degradation known as Low Temperature Crack Propagation, see Appendix B.13. The average scores for susceptibility were 1.29 to 1.43, putting all these subgroups into the yellow region. The average scores for the level of knowledge for subgroups 4.6 (pressurizer), 10.8 (reactor pressure vessel) and 11.16 (steam generator) were 1.75 to 1.88. This put these subgroups in the light yellow region indicating that the panel considers that there is insufficient knowledge to develop mitigation strategies for the potential reduction in fracture resistance. Fracture resistance was not considered for stainless steel weld metal.

3.2.1.1.4 Inconel Alloy 600 Components

Alloy 600 components (other than steam generator components) which are in the red region are found in subgroups 4.7 [Alloy 600 forged austenitic nozzles on the pressurizer at 345°C (653°F)], 10.9 (Alloy 600 forged austenitic nozzles on the reactor pressure vessel at 327°C (620°F) maximum) as well as 4.14 (cold-worked Alloy 600 heater cladding and attachment pads in CE plants). The panel evaluated the degradation mechanisms of stress corrosion cracking and fatigue for these subgroups.

The panel noted that Alloy 600, in particular forged or cold-worked material, is vulnerable to stress corrosion cracking in PWR primary water and that this is considered to be a generic problem. The mechanism is not fully understood and the initiation times are long and unpredictable. The average panel scores for the susceptibility of these subgroups were from 2.75 to 3 and the level of knowledge was between 2 and 2.25. This put all the groups in the red region with subgroup 4.7 light red and subgroups 10.9 and 4.14 just into the dark red. The comments of several panel members indicate that this degradation mechanism may require more research together with the development of inspection and prediction tools.

The panel considered that these components were not likely to be subjected to fatigue unless the nominal CUF is already greater than about 0.1. There are some laboratory data available on the environmental effects on fatigue but mostly for wrought materials, and several panel members assume that these will be applicable to the forged components. Surface finish was cited as an influencing factor and some concern was voiced about the stratification of flow that can sometimes affect nozzles. The average panel scores for the susceptibility of these subgroups ranged from 1.13 to 1.75 and the level of knowledge was between 1.88 and 2.5. This put all the groups in the yellow region with subgroups 4.7 and 4.14 in the light yellow and subgroup 10.9 in the dark yellow. The panel scores, therefore, indicate that more research may be required before mitigation actions for corrosion fatigue can be developed for these components.

The subgroups of the steam generator which have been scored where at least one degradation mechanism is in the red region are 11.5 (Alloy 600, MA SG tubes, roll transitions, U-bends, sleeves and plugs), 11.6 (Alloy 600, MA SG tubes secondary side including crevices), 11.9 (Alloy 600, divider plate, primary water), 11.12 (Alloy 600, TT SG tubes, roll transitions, U-bends, sleeves and plugs), 11.14 (Alloy 600, TT SG tubes secondary side including crevices), and subgroups 11.22 (Alloy 600, SA sensitized SG tubes, roll transitions, sleeves and plugs) and 11.23 (Alloy 600, SA sensitized SG tubes secondary side including crevices) for the B&W Once Through Steam Generator (OTSG). The internal environment on the primary side of the steam generator is PWR primary water at temperatures 284-327°C (544-620°F). The environment on the secondary side is PWR secondary side water at temperatures from 284°C (544°F) to potentially almost 327°C (620°F) depending on the concentration and solubility of impurities concentrated by hide-out. See Appendix B.12 for the possible crevice chemistries. The secondary side is exposed to a mixture of water and steam, the amount of steam increasing with increasing elevation in the steam generator. The degradation mechanisms evaluated by the panel for all of these subgroups were stress corrosion cracking and fatigue. In addition, pitting corrosion and wear were evaluated for the secondary side components. The corrosion of steam generator tubes is discussed in detail in Appendix B7.

The panel noted that Alloy 600, in particular highly deformed, forged or cold-worked material, is susceptible to stress corrosion cracking in PWR primary water and that this is considered to be a generic problem as well as for steam generator tubes in both the MA and TT conditions, albeit to different degrees. For the thin-walled steam generator tubes this is a well characterized phenomenon. For subgroups 11.5, 11.9 and 11.12 the average panel score for susceptibility to PWSCC was 2.25 or 2.5 and for the level of knowledge 2.13 to 2.63 putting these components in the dark red field.

Regarding stress corrosion on the secondary side the panel commented that this was a known issue and that both MA and to a much lesser extent TT materials have been reported with degradation. Less is known about the crevice environment on the secondary side, and it was pointed out that the presence of, for example, lead could be an issue. For subgroups 11.6, and 11.14, the average panel scores for susceptibility were 3 and 2.25 and the level of knowledge scores were 2.38 and 2.25 respectively putting these components in the dark red field.

For the B&W OTSGs the comments concerning primary side stress corrosion cracking were similar to those for the other subgroups although the steam generator tubes have been less susceptible to PWSCC due to the pre-service thermal treatment giving rise to some sensitization. For the secondary side stress corrosion cracking the panel pointed out that a considerable amount of cracking has been observed in the free span and superheated regions. For subgroups 11.22, and 11.23 the average panel scores for susceptibility were 2.13 and 2.88 respectively and the level of knowledge scores were 2.75 putting these components in the dark red field.

The panel evaluation of fatigue did not to any great extent distinguish between the primary and secondary sides of the steam generator tubes. Fatigue is not expected to be a widespread problem and initiation will depend on lack of tube support and/or adequate stiffness. It could be accelerated by prior damage due to corrosion mechanisms or tube deformation due to a diode effect at sleeves. One panel member noted that there has been high cycle fatigue of tubes close to the inspection lane in the B&W OTSG. Corrosion fatigue adjacent to the weld of the channel head divider plate, particularly at the triple point between the channel head bowl, divider plate and tube sheet, was listed as a possible degradation mode.

The average panel scores for susceptibility to fatigue for these subgroups ranged from 1.25 to 1.75 putting them all in the yellow region. For the level of knowledge, the panel score was 1.88 for subgroups 11.5 and 11.9 putting them in the light yellow field. The other subgroups were scored from 2.3 to 2.63 putting them in the dark yellow field.

On the secondary side of the steam generator tubes, fretting/wear was evaluated by the panel members whose scores placed this combination in the most susceptible region. The panel noted that wear of steam generator tubes, in particular in the U-bend region, is a known issue and that the problem is design related and depends on anti-vibration bar clearances and flow velocity.

One incidence in the evaluation of fretting/wear in the B&W once-through design was cited as being due to erosion by micron size particles of alumino-silicates in the upper bundle. The panel score for the susceptibility of subgroup 11.23 was statistical mode 2 with one higher call which conservatively put this subgroup in the red category. For the level of knowledge the average panel score was 2.5 therefore this subgroup is in the dark red field.

The panel evaluated pitting as a possible degradation mechanism for the secondary side of the steam generator tubes. Several panel members pointed out that this is no longer an issue, but that ingress of oxygen should be avoided and that pitting might occur in sludge piles under oxidizing conditions during shutdown. The average panel score for susceptibility to pitting was 1 for subgroup 11.6 and 1.13 and 1.25 for subgroups 11.14 and 11.23 putting them in the green and yellow regions. For the level of knowledge the average panel score was 2.75 putting all of the subgroups in the dark green and dark yellow fields.

3.2.1.1.5 High Strength Components

The subgroups included in this category of components are 9.3 (reactor coolant pump internal high strength parts), together with reactor vessel internal subgroups 12.7 (high strength fasteners and springs) and 12.12 (high strength baffle bolts in B&W plants).

Subgroup 9.3 components are made from the high strength materials A-286, 17-4PH, 403 stainless steels and Alloy X750 and 718 nickel base alloys. The type of components is varied and includes the reactor coolant pump shaft. These components are all exposed to PWR primary water at temperatures ~293°C (560°F). The panel considered stress corrosion cracking, fatigue and reduction in fracture resistance to be potential degradation mechanisms.

The panel commented that stress corrosion cracking is a known degradation mechanism for martensitic stainless steels, in particular if they are too hard. The heat treatment and tempering are therefore important for these materials. The average panel score for susceptibility was 2.38 and the level of knowledge was scored as 2.13 placing this subgroup of components in the dark red region.

The panel noted that corrosion fatigue has been cited as the cause of some bolt failures but that this is design specific and initiation can be accelerated if there is any intergranular corrosion. One panel member pointed out that pump shaft failure has been infrequent but that it is probably more likely to be prone to corrosion fatigue and the bolts to stress corrosion cracking. The average panel score for susceptibility to fatigue was 1.88 and the average score for the level of knowledge was 2 putting this subgroup in the light yellow region.

Reduction in fracture resistance is a known problem in particular for 17-4PH depending on the tempering and operating temperature. Low temperature crack propagation has been observed in nickel base alloys such as X750 although it is not certain that the necessary conditions for reduction in fracture resistance (presumably due to hydrogen embrittlement) prevail under normal operating conditions in a PWR. For more discussion of this potential degradation mechanism see Appendix B.13. The average panel score for susceptibility was 2 and the average score for the level of knowledge was 1.63, putting this subgroup in the light-yellow region.

Reactor vessel internals were grouped into several subgroups according to the material condition and the expected end of life fluence level – low ($\leq$ 0.5 dpa) or high ($\geq$ 0.5 dpa). There were no other differences between the environments which were described as PWR primary water at 291-327°C (556-620°F). No dose level was given for subgroup 12.7 which comprises high strength fasteners and springs made from the nickel base Alloys X750 and 718 because the internals parts considered are on the periphery of the nuclear core and are not subject to high neutron doses. Subgroup 12.12 of the reactor internals is listed as a high fluence environment. The degradation mechanisms considered for these subgroups were stress corrosion cracking, swelling, irradiation creep, reduction in fracture resistance and fatigue (not all degradation mechanisms were scored for both the subgroups).

Stress corrosion cracking was considered to be a potential degradation mechanism for both of the subgroups in this category of components. The panel pointed out that there has, in general, been very good field experience of these components, with the exception of Alloy X750 (subgroup 12.7) but that there is insufficient experience and laboratory data to dismiss degradation at high doses since all the contributing factors have not yet been identified. There is also the possibility of synergistic effects between low temperature aging and irradiation which could affect the hardness of the material, which is known to be a contributing factor. For a more comprehensive discussion of stress corrosion cracking in irradiated material, see Appendix B.2. For the high strength nickel base materials in subgroups 12.7 and 12.12, the panel scores for stress corrosion cracking susceptibility were an average of 2 and statistical mode 3 respectively. The corresponding scores for the level of knowledge were average of 2 and 1.88 putting subgroup 12.7 in the light yellow field and subgroup 12.12 in the light red field.

The panel considered swelling as a possible degradation mechanism for subgroup 12.12. As pointed out by the panel, swelling is only considered to be a potential degradation mechanism for components which achieve high fluences (see Appendix B.2 for further discussion of this phenomenon). Several members of the panel also noted that this will only be a localized problem for the components closest to the core. There is insufficient understanding of, for example, the thresholds for the onset of swelling in commercial plants, which is indicated by the score of statistical mode 2 for the confidence by the panel members in their susceptibility call. The average panel score for susceptibility to swelling was 2.14 for subgroup 12.12. The average score for the level of knowledge was 2 putting this subgroup in the light red field.

Irradiation creep is another degradation mechanism that is neutron dose dependent and occurs at high fluences (see Appendix B.2 for further information). The panel considered that it was likely in both subgroups 12.7 and 12.12 but only for highly stressed components. The panel score for the susceptibility of subgroup 12.7 was statistical mode 2 with one higher call and the average score was 2.57 for subgroup 12.12. Subgroup 12.7 was conservatively put in to the higher susceptibility category and is thus colored red. The level of knowledge was scored as 2.18 for subgroup 12.7 putting it in the dark red region and 1.88 for subgroup 12.12 making it light red.

Irradiation also leads to a reduction in fracture resistance of the subgroups subjected to the higher fluences, see Appendix B.2. It was pointed out by the panel that there will be a considerable loss of toughness if there is significant swelling in the components, which is currently difficult to predict. It is also possible that hydrogen can contribute to the reduction in fracture resistance. For the components in subgroups 12.7 and 12.12, the panel also noted that low-temperature crack propagation has been observed in nickel-base alloys such as X750, although it is not certain that the necessary conditions for fracture (exacerbated by hydrogen embrittlement) prevail under normal operating conditions in a PWR. (For more discussion of this potential degradation mechanism, see Appendix B.13.) The average panel scores for susceptibility were 1.5 and 1.71 and for the level of knowledge 1.63 and 1.88 respectively, putting these subgroups in the light yellow region.

3.2.1.1.6 Carbon and Low-Alloy Steel Components

The two subgroups with components made of carbon steel or low-alloy steel are 10.2 (reactor pressure vessel shell plates, forgings, welds, brackets, etc.) and 11.20 (steam generator shell, tube supports and/ or preheater baffles). The degradation mechanisms the panel evaluated were crevice corrosion, reduction in fracture resistance and stress corrosion cracking for both subgroups. Boric acid corrosion, fatigue and thermal creep were also evaluated for the pressure vessel components and flow-accelerated corrosion for the steam generator subgroup. The reactor vessel components are internally clad with stainless steel weld metal protecting them from the primary side water at up to 316°C (600°F). The steam generator components are on the secondary side and experience temperatures of up to 327°C (620°F) during operation.

The panel considered that external boric acid corrosion of low-alloy steel was most likely on the upper or lower heads of the RPV due to leaking nickel-alloy welds. This is a known problem and is discussed further in Appendix B.18. The scores for susceptibility and level of knowledge were both 2.13 putting this in the dark red field.

Flow-accelerated corrosion of the tube supports in the steam generator has been observed in the field. Its occurrence depends on the flow pattern and the particular material composition. Ingress of sea water may aggravate the problem. There are design solutions, predictive models and some amines have also been found to be effective for limiting FAC. The average panel score for susceptibility was 2.25 and for the level of knowledge 2.38 putting subgroup 11.20 in the dark red field.

The reduction in fracture resistance is a known (and extensively studied) issue for the beltline of the reactor pressure vessel and depends on material composition (particularly the copper and phosphorus contents) and neutron dose. The panel noted that the effects of the longer exposure times associated with extended life are not clear and that there may also be an increased risk of stress corrosion cracking because of the hardening effect of radiation (a parallel to cold work) in the event that the internal stainless steel cladding be breached and allow primary water into contact with the RPV. The panel scored the susceptibility as 1.88 and the level of knowledge as 2.88 putting the reduction in fracture resistance of the reactor PV beltline steel in the dark yellow field.

The panel considered that reduction in fracture resistance was not a serious issue for the tube supports in the steam generator. It was pointed out that there may be an effect of the environment but this was thought to be unlikely. The panel scoring for the susceptibility to a reduction in fracture resistance was low, 0.71, and the level of knowledge was high, 2.13.

The panel considered that crevice corrosion could occur in the tube/support plate crevices, in particular if hideout occurs. Crevice corrosion was the cause of denting in the late 1970s, but is not expected to be a problem again since the secondary chemistry remedies adopted were plainly effective in this case. Good chemistry control is important in the control of crevice corrosion and includes the prevention of condenser leaks. The average panel score for susceptibility was 1.5 and for the level of knowledge 2.75 putting crevice corrosion of the tube supports in the dark yellow field.

Stress corrosion cracking in the carbon steel components on the secondary side of the steam generator is highly dependent on the specific water chemistry. It has been observed and was a serious issue in connection with denting, but it is not considered to be a significant future threat. The average panel score for susceptibility was 1.25 and the score for the level of knowledge was 2.25, putting this subgroup in the dark-yellow field.

Crevice corrosion and stress corrosion cracking of the low-alloy steel reactor vessel components were scored in the dark green field (susceptibility 1 and 0.86 and level of knowledge 2.17 and 2.43 respectively). These degradation mechanisms are not considered to be a problem for these components unless the cladding is penetrated.

For the reactor pressure vessel subgroup 10.2, the panel evaluated fatigue and noted that this would only be an issue under very specific loading conditions and that environmental effects would only occur if the cladding were to be penetrated. There is good field experience and the topic has been studied widely. The average panel score for susceptibility was 1.13 and the score for the level of knowledge was 2.88 putting fatigue for this group in the dark yellow field.

Thermal creep was evaluated for subgroup 10.2 and falls into the dark-green field (average susceptibility 0.57 and average knowledge 2.43) indicating that it is not considered a very likely degradation mechanism and that the level of knowledge may be sufficiently high to permit development of mitigation actions. It was evaluated because some highly controversial laboratory test data do not provide completely conclusive assurance that the phenomenon can be disregarded.

3.2.1.1.7 Type 304/316/308 Stainless Steel Components

The subgroups containing stainless steel components are 10.10 (CRDM housing and canopy seals in the reactor pressure vessel) and 12.4, 12.8, 12.9, 12.10, and 12.11 (reactor vessel internals.

In the case of CRDM housing, subgroup 10.10 (reactor pressure vessel), both the base metal and the weld metal were considered together with respect to stress corrosion cracking, fatigue and crevice corrosion. The components are in contact with PWR primary water normally under stagnant conditions in the temperature range of 93-316°C (200-600°F).

The panel considered that the components were potentially susceptible to stress corrosion cracking because of the possibility of non-standard water chemistry including the presence of corrosive species and oxygen trapped after shutdowns in such dead legs although operational practices today have largely eliminated the risk of trapping oxygen. Cases of both transgranular and intergranular stress corrosion cracking have been reported in these components. Other aggravating factors noted by the panel were high residual stresses and cold work. One panel member raised the possibility of low-temperature aging being a factor. The score for the susceptibility of 304/308 CRDM housings to stress corrosion cracking was statistical mode 2 with two higher calls and the level of knowledge had an average of 2.13, putting these components in the dark-red field.

The panel considered that crevice corrosion is possible in principle because of the presence of oxygen and potentially non-standard chemistry conditions post-startup, but that it should not be a major problem except by stimulating pitting if chloride is simultaneously present, which can be a precursor to stress corrosion cracking or corrosion fatigue. The average score for the susceptibility was 1.25 and the average score for the level of knowledge was 2.38, putting the 304/308 CRDM housings in the dark yellow region with respect to crevice corrosion.

The panel considered the possibility of corrosion fatigue in these components because of the non-standard water conditions. For CE plants, corrosion fatigue has been cited as a cause for some failures. One panel member cited a possible contribution from thermal fatigue whilst other panel members found it difficult to identify where the cyclic loading would come from. The average score for the susceptibility was 1.5 and the average score for the level of knowledge was 2.13, putting the 304/308 CRDM housings in the dark yellow field with respect to fatigue.

Stainless steel reactor vessel internals were grouped into several subgroups according to the material condition and the expected end of life fluence level – low ($\leq$ 0.5 dpa) or high ($\geq$ 0.5 dpa). There were no other differences between the environments which were described as PWR primary water at 291-327°C (556-620°F). Only one subgroup fell into the low dpa fluence range: 12.4, Type 316 cold-worked austenitic stainless steel. The other subgroups of the reactor vessel internals which the panel assigned the highest susceptibility to one or more of the degradation mechanisms considered were: 12.8, Type 304 austenitic stainless steel plates or tubes; 12.9, Type 304 austenitic weld HAZs; 12.10, Type 308 austenitic to austenitic weld metal components; and 12.11, Type 316 cold worked austenitic stainless steel components. The degradation mechanisms considered for these subgroups were stress corrosion cracking, swelling, irradiation creep, reduction in fracture resistance, fatigue, and thermal creep (not all degradation mechanisms were scored for all the subgroups).

Stress corrosion cracking was considered to be a potential degradation mechanism for all of the subgroups in this category of components. All of the subgroups were assessed to be in the most susceptible, red, category. The panel pointed out that there has in general been very good field experience for these components but that there is insufficient experience and laboratory data to dismiss susceptibility at high doses since all the contributing factors have not yet been identified. There is also the possibility of synergistic effects between low-temperature aging of CASS materials and irradiation which could affect the toughness of the material, which is known to be a contributing factor. For a more comprehensive discussion of stress corrosion cracking in irradiated material, see Appendix B.2.

The panel's score for susceptibility to stress corrosion cracking was a statistical mode 2 with one or two higher calls or, in the case of subgroup 12.11 (Type 316 cold-worked austenitic stainless steel components) an average of 3. All subgroups except 12.11 were conservatively put in to the higher susceptibility category and are thus colored red. The average scores for the level of knowledge ranged from 2.13 to 2.38 for subgroups 10.10, 12.4 and 12.8, making them dark red, and average scores of 1.75 and 2 for the other three subgroups making them light red.

Fatigue was considered to be a potential degradation mechanism for all the stainless-steel internals subgroups. The panel noted that there has generally been very good field experience with the exception of a few cases of bolts wearing loose. There is also a reasonable amount of laboratory data but the magnitude of the environmental effects for irradiated materials is unclear. In the cold-worked components (subgroup 12.4) the possible relaxation in baffle bolts due to irradiation creep can make them potentially more prone to fatigue when subject to flow-induced vibration.

The average panel score for susceptibility to fatigue was between 1.29 and 1.71. The average panel score for the level of knowledge was 1.5 to 1.75 except for subgroup 12.8 for which it was 2.5. The subgroups 12.4, 12.9, 12.10, and 12.11 are therefore in the light yellow region and subgroup 12.8 is in the dark yellow region. There is however a wide spread in the confidence in the panel scores for the subgroup 12.8. The panel considered swelling as a possible degradation mechanism for subgroups 12.8, 12.9, 12.10 and 12.11. As pointed out by the panel, swelling is only considered to be a potential degradation mechanism for components which achieve high fluences, see Appendix B.2 for further discussion of this phenomenon. Several members of the panel also noted that this will only be a localized problem for the components closest to the core. There is insufficient understanding of, for example, the thresholds for the onset of swelling in commercial plants, which is indicated by the score of 2 for the confidence by all the panel members (except one who scored 1 for one of the groups).

The average panel score for susceptibility to swelling was 2 for subgroups 12.8, 12.9 and 12.10, and 2.14 for subgroup 12.11. The average score for the level of knowledge was also 2 for subgroups 12.8, 12.9 and 12.10, and 1.88 for subgroup 12.11. Subgroups 12.8 (Type 304 austenitic stainless steel plates and tubes), 12.9 (Type 304 austenitic weld HAZs) and 12.10 (Type 308 austenitic to austenitic weld metal components) are therefore in the light yellow region and 12.11 (Type 316 cold worked austenitic stainless steel components) is light red. The panel scores therefore indicate that swelling is a degradation mechanism for which there is currently insufficient knowledge to develop adequate mitigation strategies.

Irradiation creep is another degradation mechanism that is neutron dose dependent; see Appendix B.2 for further information. The panel considered that it was likely in subgroups 12.8 and 12.11 but only significant for highly stressed components such as bolted connections. The panel's average score for the susceptibility of subgroup 12.8 was statistical mode 2; and the score for subgroup 12.11 was statistical mode 3. The average score for the level of knowledge was 2.32 for subgroup 12.8 putting it in the dark yellow region; and 1.8 for subgroup 12.11 making it light red.

Irradiation also leads to a reduction in fracture resistance of the subgroups subjected to the higher fluences, see Appendix B.2. It was pointed out by the panel that there will be a considerable loss of toughness if there is significant swelling (>~6%) in these components. It is also possible that hydrogen can contribute to the reduction in fracture resistance. The average panel score for susceptibility for subgroups 12.8, 12.9, 12.10 and 12.11 was between 1.29 and 1.71, and the average score for the level of knowledge was from 2.13 to 2.43 putting all the subgroups in the dark yellow region.

Long term relaxation (thermal creep) in type 316 cold worked austenitic stainless steels at low fluence, subgroup 12.4, was identified based on laboratory data. Initial stress, temperature and time are listed as contributing factors. The average panel score for susceptibility was 1.25 and for the level of knowledge 2.13 placing this subgroup in the dark yellow field.

3.2.1.2 RCS Components with Light-Yellow Susceptibility

The subgroups in the reactor coolant systems with components falling into the light yellow susceptibility regions are listed in Table 3.3 and illustrated in the modified rainbow chart in Figure 3.7.

Table 3.3 Light Yellow Subgroups in the PWR RCS

Component	Subgroups	Degradation Mechanisms Considered
Type 304/316 Stainless steel weld HAZs	1.3.1, 1.3.2, 2.3.1, 2.3.2, 3.3.1, 3.3.2, 4.10.1, 4.10.2, 5.3.1, 5.3.2, 6.3.1, 6.3.2, 7.3.1, 7.3.2, 8.3.1, 8.3.2, 13.3.1, 13.3.2	Fatigue Stress corrosion cracking
Type 308 Stainless steel welds	1.4, 2.4, 3.4, 4.8, 5.4, 6.4, 7.4, 8.4, 9.2, 12.3, 13.4	Fatigue Reduction in fracture resistance Stress corrosion cracking
Type 308/309 Stainless steel dissimilar metal welds	1.5, 2.5, 3.5, 4.5, 10.4, 11.7	Fatigue Reduction in fracture resistance Stress corrosion cracking
Cast stainless steel components	1.6, 1.10, 2.6, 2.10, 3.6, 3.10, 9.5, 10.7, 12.5, 13.5	Fatigue Reduction in fracture resistance Stress corrosion cracking
Wrought and forged Type 304/316 Stainless steel components	1.8, 2.8, 3.2, 3.8, 4.4, 4.13, 5.5, 6.5, 7.2, 7.5, 8.2, 8.5, 10.5, 11.8, 12.6, 13.6	Creep Fatigue Stress corrosion cracking
Type 308/309 Stainless steel clad components	1.11	Boric acid corrosion Debonding Fatigue Stress corrosion cracking
High strength bolts, studs, etc.	4.9, 9.6, 10.6	Boric acid corrosion Erosion corrosion Fatigue Stress corrosion cracking
Ni-base alloy components	11.11, 11.19	Crevice corrosion Fatigue Reduction in fracture resistance Stress corrosion cracking
Type 308/309 Stainless steel dissimilar metal welds – external surfaces	11.17	Stress corrosion cracking

Subgroup Description		Degradation Mechanism						
		BAC	CREEP	DEBOND	EC	FAT	FR	SCC
Type 304/316 Stainless Steel Weld HAZs								
1.3.1	SS 304 Piping HAZ							
1.3.2	SS 316 Piping HAZ							
2.3.1	SS 304 Piping HAZ							
2.3.2	SS 316 Piping HAZ							
3.3.1	SS 304 Piping HAZ							
3.3.2	SS 316 Piping HAZ							
4.10.1	Type 304 SS HAZ							
4.10.2	Type 316 SS HAZ							
5.3.1	SS 304 Piping HAZ							
5.3.2	SS 316 Piping HAZ							
6.3.1	SS 304 Piping HAZ							
6.3.2	SS 316 Piping HAZ							
7.3.1	SS 304 Piping HAZ (Stagnant)							
7.3.2	SS 316 Piping HAZ (Stagnant)							
8.3.1	SS 304 Piping HAZ (Stagnant)							
8.3.2	SS 316 Piping HAZ (Stagnant)							
13.3.1	SS 304 Piping HAZ							
13.3.2	SS 316 Piping HAZ							
Type 308 Stainless Steel Welds								
1.4	Type 308 SS Weld							
2.4	Type 308 SS Weld							
3.4	Type 308 SS Weld							
4.8	308/316 (CW) Heater Clad/Welds							
5.4	Type 308 SS Weld							
6.4	Type 308 SS Weld							
7.4	Type 308 SS Weld (Stagnant)							
8.4	Type 308 SS Weld (Stagnant)							
9.2	304/308/316 SS Components/Welds							
12.3	Type 308 SS Weld (low fluence)							
13.4	Type 308 SS Weld							
Type 308/309 Stainless Steel Dissimilar Metal Welds								
1.5	308/309 Dissimilar Weld - Int.							
2.5	308/309 Dissimilar Weld - Int.							
3.5	308/309 Dissimilar Weld - Int.							
4.5	308/309 Dissimilar Welds - Int.							
10.4	308/309 Dissimilar Welds- Int.							
11.7	308/309 Dissimilar Welds - Int.							
Cast Stainless Steel Components								
1.6	CASS CF8/CF8M Components							
1.10	CASS CF8/CF8M Piping							
2.6	CASS CF8/CF8M Components							
2.10	CASS CF8/CF8M Piping							
3.6	CASS CF8/CF8M Components							
3.10	CASS CF8/CF8M Piping							
9.5	CASS CF8 Components							
10.7	CASS CF8 Components							
12.5	CASS Components							
13.5	CASS components							
Wrought and Forged Type 304/316 Stainless Steel Components								
1.8	Forged 304/316 SS Nozzles							
2.8	Forged 304/316 SS Nozzles							
3.2	Wrought SS 304/316 Piping							
3.8	Forged 304/316 SS Nozzles							
4.4	Wrought SS 304/316 - Int.							
4.13	Forged 304/316 SS Nozzles							
5.5	Forged 304/316 SS Nozzles							
6.5	Forged 304/316 SS Nozzles							
7.2	SS 304/316 Piping (Stagnant)							
7.5	Forged 304/316 Nozzles (Stagnant)							
8.2	SS 304/316 Piping (Stagnant)							
8.5	Forged 304/316 Nozzles (Stagnant)							
10.5	Forged 304/316 SS Nozzles							X
11.8	Forged 316 SS Nozzles							
12.6	Type 304 SA SS Holdown Spring							*
13.6	Forged 304/316 SS Nozzles							
Type 308/309 Stainless Steel Clad Components								
1.11	SS Clad Ferritic Piping							
High Strength Bolts, Studs Etc.								
4.9	SA-193 Gr B7 Manway Bolts							*
9.6	SA-540 Gr. B24 Bolts							*
10.6	SA-540 Gr B23 Closure Studs							
Inconel Components								
11.11	Alloy 52/82 Channel Head Clad							
11.19	Alloy 690 Divider Plate							
Type 308/309 SS Dissimilar Metal Welds -- Ext. Surfaces								
11.17	Type 308/309 Dissim. Welds - Ext.							

NOTES: * Susceptibility at color interface with one or more scores higher than interface; X Susceptibility inside color box with one or more scores higher than this color box upper interface.

Figure 3.7 Modified Rainbow Chart Showing Light-Yellow Subgroups in the PWR Reactor Coolant System

3.2.1.2.1 Type 304/316 Stainless Steel Weld Heat-Affected Zones

Subgroups containing type 304 and 316 stainless steel piping weld HAZs and which are color-coded light yellow are found in groups 1 (cold leg piping), 2 (crossover leg piping), 3 (hot leg piping), 4 (pressurizer), 5 (pressurizer spray piping), 6 (pressurizer surge piping), 7 (pressurizer piping to PORVs), 8 (pressurizer piping to SRVs) and 13 (stop valve loop bypass piping). The panel decided to subdivide these subgroups to distinguish between these two common austenitic stainless steel materials with respect to their susceptibility to stress corrosion cracking. Subgroups 1.3.1, 2.3.1, 3.3.1, 4.10.1, 5.3.1, 6.3.1, 7.3.1, 8.3.1 and 13.3.1 were Type 304 weld HAZs and subgroups 1.3.2, 2.3.2, 3.3.2, 4.10.2, 5.3.2, 6.3.2, 7.3.2, 8.3.2 and 13.3.2 were Type 316 weld HAZs. The panel did not distinguish between the two materials with regard to fatigue and the other degradation mechanism which was evaluated for these subgroups. Stagnant conditions normally pertain for subgroups 7.3.1, 7.3.2, 8.3.1 and 8.3.2.

With regard to stress corrosion cracking the panel noted that there has been very good field experience for both these materials. The major source for concern is the degree of cold work from for example surface grinding. Drawing an analogy to the BWR experience, some panel members considered that Type 304 might be expected to be more susceptible than Type 316, but that under normal operating conditions it is not expected to be a problem. Ingress of oxygen to regions where it could be trapped could be a concern; one panel member indicated that there is some laboratory evidence of initiation and slow propagation in hydrogenated water even in the non-sensitized, cold worked condition.

The average panel scores for the susceptibility of Type 304 HAZs were from 1.25 to 1.5 and for Type 316 from 1.13 to 1.38. Panel members always had the same score for the level of knowledge in a given group and the average scores were between 2 and 2.25 putting more than half of the subgroups (1.3.1, 1.3.2, 4.10.1, 4.10.2, 5.3.1, 5.3.2, 8.3.1, 8.3.2, 13.3.1 and 13.3.2) in the dark yellow field and the remainder (subgroups 2.3.1, 2.3.2, 3.3.1, 3.3.2, 6.3.1, 6.3.2, 7.3.1 and 7.3.2) in the light yellow field. These scores indicate that there is need for more work before enough knowledge is available to develop potential mitigation strategies for stress corrosion cracking in the HAZs of Type 304 and 316 piping welds.

With regard to fatigue, the panel again noted that there was very good field experience for the components in these subgroups although the adverse effect of corrosion in hydrogenated PWR primary water on fatigue of stainless steel is a known issue not specifically taken into account by present design procedures. The greatest uncertainties by far are in the quantification of cyclic loads and numbers of cyclic transients. Moreover, since fatigue is a cumulative damage process, the absence of significant fatigue failures to date (after ~30 years service) is no guarantee of absence from problems in the future, especially for projected operating license periods up to 60 years. Some panel members pointed to a lack of data concerning HAZs compared with wrought materials. The scoring of the panel members reflected to a small extent the differences in the specific conditions in the different subgroups, for example the known potential for thermal striping in the hot leg piping, albeit of relatively low amplitude, arising from the use of low leakage cores, and dead legs close to the main coolant line which are potentially susceptible to thermal fatigue. The average panel scores for susceptibility to fatigue were between 1.13 and 1.75 and the average scores for the level of knowledge were 1.63 or 1.75 putting all the subgroups in the light-yellow field.

3.2.1.2.2 Type 308 Stainless Steel Welds

The subgroups containing Type 308 stainless steel welds and which are light yellow are 1.4 (cold leg piping austenitic welds), 2.4 (crossover leg piping austenitic welds), 3.4 (hot leg piping austenitic welds), 4.8 (pressurizer, heater cladding/attachment weld), 5.4 (pressurizer spray piping austenitic welds), 6.4 (pressurizer surge piping austenitic welds), 7.4 (pressurizer piping to PORVs austenitic welds), 8.4 (pressurizer piping to SRVs austenitic welds), 9.2 (reactor coolant pump, internals including weldments), 12.3 (reactor vessel internals, austenitic welds) and 13.4 (stop valve loop bypass piping austenitic welds). The degradation mechanisms evaluated were fatigue, reduction in fracture resistance and stress corrosion cracking.

The panel noted that field experience has been good for fatigue issues in these subgroups of components; however there is limited laboratory data for stainless steel weld metal. Corrosion fatigue of stainless steels in deoxygenated water is a known issue but the effects on duplex weld metals are uncertain; there is also a potential effect of cold work from surface grinding on initiation. Some of the parts in subgroups 1.4, 2.4, 3.4, 5.4 and 13.4 are dead legs close to the main coolant line and they may be vulnerable to thermal fatigue. There may also be a contribution to thermal fatigue from thermal striping in the hot leg piping (subgroup 3.4) arising from the use of low leakage cores. Another region in which thermal fatigue may be an issue is the surge line (subgroup 6.4). In the stagnant saturated steam environments of subgroups 7.4 and 8.4 one panel member pointed out that there is less likely to be an environmental contribution to fatigue. Most of the panel members also pointed out that the internal fillet welds of the reactor coolant pump are vulnerable (subgroup 9.2) but that this is design specific. There may also be an issue for the welded reactor vessel internals in CE plants (subgroup 12.3).

The average panel scores for susceptibility to fatigue were between 1.13 and 1.88, and the average scores for level of knowledge were from 1.75 to 2 putting all these subgroups in the light yellow region.

With regard to stress corrosion cracking in Type 308 welds the panel noted that there is very good field experience but that high ferrite containing welds have been observed to crack in the laboratory. Low temperature aging and hardening can lead to stress corrosion cracking under certain conditions but the panel members had divergent opinions as to whether this might be an issue or not at PWR operating temperatures. Overall, the panel members did not expect this to be a problem under normal operating conditions. For subgroup 4.8 (pressurizer, heater cladding/ attachment weld) the panel had a slightly higher score for the susceptibility pointing out that stress corrosion cracking of heavily cold worked stainless steel such as these is a known issue. In addition, the effect of lithium hydroxide concentration in the associated crevices is a concern in connection with boiling at the end of the fuel cycle and during cycle stretch-out.

The average panel scores for susceptibility to stress corrosion cracking were between 1.13 and 1.88, and the average scores for level of knowledge were from 2 to 2.57 putting most of these subgroups in the light yellow region. Subgroups 9.2, 12.3 and 13.4 were in the dark yellow field.

The panel evaluated reduction in fracture resistance for all the subgroups except 4.8. They did not consider that this would be a serious issue but pointed out that there are some laboratory data indicating that there could be a long term issue and an effect on stress corrosion cracking. The average panel scores for susceptibility to reduction in fracture resistance were between 1 and 1.4, and the average scores for level of knowledge were from 1.75 to 2.33 putting most of these subgroups in the light yellow region. Subgroups 1.4 and 9.2 were in the dark yellow field.

3.2.1.2.3 Type 308/309 Stainless Steel Dissimilar Metal Welds

The subgroups containing Type 308/309 stainless steel dissimilar metal welds and which are color-coded light yellow are 1.5 (cold leg piping dissimilar metal welds), 2.5 (crossover leg piping dissimilar metal welds), 3.5 (hot leg piping dissimilar metal welds), 4.5 (pressurizer dissimilar metal welds), 10.4 (reactor pressure vessel dissimilar metal welds) and 11.7 (steam generator dissimilar metal welds, internal surface). The degradation mechanisms evaluated for the internal surfaces of these welds were fatigue, reduction in fracture resistance and stress corrosion cracking.

The experience of stainless steel dissimilar metal welds has been good with regard to fatigue. There could be some environmental and dissimilar expansion coefficient effects which should be considered. The average panel scores for susceptibility to fatigue were between 1.25 and 1.75, and the average scores for level of knowledge were 1.88 putting all these subgroups in the light yellow region.

Some panel members pointed out that there are some laboratory indications that the stainless steel weld metal might be susceptible to a low temperature aging effect leading to increased stress corrosion cracking and reduced toughness. This has been observed mainly in cast austenitic stainless steels to date. The thermal aging is expected to be less than for cast stainless steels because of the lower ferrite content. The average panel scores for susceptibility to reduction in fracture resistance were between 1 and 1.29, and the average scores for level of knowledge were from 1.5 to 2 putting all these subgroups in the light yellow or light green fields.

The field experience shows that there has been no stress corrosion cracking of stainless steel weld metal to date from the primary water side. There is a known issue with possible sensitization in the dilution zone with low alloy steel leading to a potential for intergranular stress corrosion cracking if wetted on the outside. This has been observed in components even before they are put in service. Again, the effect of low temperature aging on stress corrosion cracking was raised as well as the effects of cold work introduced by grinding. The average panel scores for susceptibility to stress corrosion cracking were 1.5, and the average scores for level of knowledge were from 1.5 to 1.75 putting all these subgroups in the light-yellow region.

3.2.1.2.4 Cast Stainless Steel Components

The subgroups containing cast stainless steel piping and components and which are color-coded light yellow are 1.6 and 1.10 (cold leg piping, cast components and piping), 2.6 and 2.10 (crossover leg piping, cast components and piping), 3.6 and 3.10 (hot leg piping, cast components and piping), 9.5 (reactor coolant pump), 10.7 (reactor pressure vessel) 12.5 (reactor vessel internals) and 13.5 (stop valve loop bypass piping). For most of the subgroups the material was designated as CF8 or CF8M. The degradation mechanisms evaluated were fatigue, reduction in fracture resistance and stress corrosion cracking.

The panel pointed out that the cyclic stresses on these cast stainless steel components are limited and that the experience to date has been good. There is a lack of laboratory data to evaluate the effect of the environment on fatigue life. The average panel scores for susceptibility to fatigue were between 1.13 and 1.25, and the average scores for level of knowledge were from 1.75 to 2.5 putting all of these subgroups in the light yellow region except 9.5 which was dark yellow.

Most of the panel considered that thermal aging leading to a reduction in the fracture resistance is well characterized with some predictive models available based on air data. There is, however, a lack of data for the effect of the primary water environment on resistance to stress corrosion cracking of aged material. The ferrite content of the components needs to be high for significant thermal aging to occur, typically more than 20%, and the panel expressed some concern over variations in composition and whether irradiation would have a synergistic effect in the case of the reactor vessel internals (subgroup 12.5). Appendix B.4 addresses these issues in detail.

The average panel scores for susceptibility to reduction in fracture resistance were between 1 and 1.29, and the average scores for level of knowledge were from 2.13 to 2.63 putting all these subgroups in the dark yellow or dark green region.

The panel considered that the stress corrosion of cast stainless steel components is a theoretical concern at present. There is, however, sufficient interest that some testing is being initiated in Europe to assess the long-term susceptibility. The average panel scores for susceptibility to stress corrosion cracking were between 1.13 and 1.38, and the average scores for level of knowledge were from 1.13 to 1.5 putting all these subgroups in the light yellow region.

3.2.1.2.5 Wrought and Forged Type 304/316 Stainless Steel Components

The subgroups containing wrought and forged type 304/316 stainless steel components and which are color-coded light yellow are 1.8 (cold leg piping, forged nozzles), 2.8 (crossover leg piping, forged nozzles), 3.2 (hot leg piping, wrought piping), 3.8 (hot leg piping, forged nozzles), 4.4 (pressurizer wrought components), 4.13 (pressurizer forged nozzles), 5.5 (forged nozzles), 6.5 (forged nozzles, stagnant conditions), 7.2 (wrought piping, stagnant saturated steam), 7.5 (forged nozzles, stagnant conditions), 8.2 (piping, stagnant conditions), 8.5 (forged nozzles, stagnant saturated steam), 10.5 (reactor pressure vessel, forged nozzles), 11.8 (steam generator, forged nozzles), 12.6 (Type 304 hold down spring) and 13.6 (forged nozzles). The degradation mechanisms evaluated for the internal surfaces of these welds were fatigue and stress corrosion cracking. For subgroup 12.6, thermal creep also was evaluated.

The panel noted that there is a large amount of relevant fatigue laboratory data available for these materials but that the magnitude of the environmental effects is not completely clear. There are results which indicate a significant reduction in the fatigue life with low cycle corrosion fatigue. There is an even greater uncertainty in cyclic loads and frequency of load transients. The effect of surface finish was raised by several panel members. The field experience has, however, been good to date although because of the cumulative damage character of fatigue, that is no guarantee of future good behavior. Thermal sleeves in the pressurizer were noted to be potentially vulnerable (subgroup 4.4), as well as the effect of temperature for the systems at the highest temperatures (subgroup 3.2). Components in subgroups 7.2, 7.5, 8.2 and 8.5 were scored lower by several panel members since the lines are stagnant. For the forged components in subgroups 1.8, 2.8, 3.8, 4.13, 5.5, 6.5, 7.5, 8.5, 10.5, 11.8 and 13.6, several members of the panel assumed that the laboratory data for wrought material would be applicable.

The average panel scores for susceptibility to fatigue were between 1.13 and 1.63, and the average scores for level of knowledge were from 1.63 to 2 putting all these subgroups in the light yellow region.

Most of the panel members do not expect there to be any issues with stress corrosion cracking in forged or wrought stainless steel components unless there are components which were heavily cold worked and not solution annealed, for example in subgroup 12.6. For all except subgroups 10.5, 12.6, and 13.6, the average panel scores for susceptibility to stress corrosion cracking were between 0.8 and 1. The average scores for level of knowledge were from 2.25 to 2.75 putting these subgroups in the dark-green region with the exception of subgroups 10.5, 12.6 and 13.6 whose scores placed them in the light yellow, dark yellow, and light green fields respectively.

Thermal creep (relaxation) is a known issue for the hold-down springs in subgroup 12.6 (reactor vessel internals). These components have been replaced in many cases with springs manufactured of martensitic stainless steel. The average panel score for susceptibility to creep in subgroup 12.6 was 2, and the average score for level of knowledge was also 2 putting this subgroup in the light yellow region.

3.2.1.2.6 Type 308/309 Stainless Steel Clad Components

Subgroup 1.11 contains Type 308/309 stainless steel clad ferritic steel piping components found in the cold leg piping systems of B&W and CE plants. The degradation mechanisms evaluated for the internal surfaces of these components were boric acid corrosion, debonding, fatigue and stress corrosion cracking.

Boric acid corrosion is discussed in detail in Appendix B.18. The panel pointed out that this will only be an internal problem if the cladding is breached, and then only at low temperature, or externally if there are leaks in the vicinity of the piping. The average panel score for susceptibility to boric acid corrosion in subgroup 1.11 was 1.25, and the average score for level of knowledge was 1.88 putting this subgroup in the light yellow region.

The panel considered clad debonding to be a theoretical concern which can occur during manufacture, or where the cladding has been damaged or penetrated as a result of mechanical action. The average panel score for susceptibility to debonding in subgroup 1.11 was 0.86, and the average score for level of knowledge was 2.86 putting this subgroup in the dark green region.

The panel noted that there has been very good operating experience regarding fatigue in these components. However it was pointed out that corrosion fatigue of austenitic materials is a known issue. The average panel score for susceptibility to fatigue in subgroup 1.11 was 1, and the average score for level of knowledge was 2.13 putting this subgroup in the dark-green region.

There has also been very good field experience with regard to stress corrosion cracking of these components. There is however the known issue in the dilution zone with the carbon steel. It was not thought that cracks would propagate into the carbon steel. The average panel score for susceptibility to stress corrosion cracking in subgroup 1.11 was 1.13, and the average score for level of knowledge was 2.13 putting this subgroup in the dark-yellow region.

3.2.1.2.7 High-Strength Bolts, Studs, etc.

The subgroups containing high strength bolts, studs, etc. and which are color-coded light yellow are 4.9 (pressurizer, manway bolts made of SA 193 Gr B7), 9.6, (bolts made of SA-540 Gr B24) and 10.6 (reactor pressure vessel, closure studs and nuts made of SA-540 Gr B23). The degradation mechanisms evaluated for these components were boric acid corrosion (except for subgroup 10.6), fatigue and stress corrosion cracking. Erosion corrosion was also evaluated for subgroup 10.6.

Boric acid corrosion was evaluated by the panel for subgroups 4.9 and 9.6. The panel pointed out that boric acid corrosion is always possible if the flange leaks and is ignored. Boric acid corrosion is discussed in detail in Appendix B.18. The average panel scores for susceptibility to boric acid corrosion for subgroups 4.9 and 9.6 were 1.75, and the average scores for level of knowledge were 2.25 putting both these subgroups in the dark-yellow field.

The panel did not consider that there were any loading issues that are likely to result in fatigue of these components. The average panel scores for susceptibility to fatigue for all the subgroups were 0.88, and the average scores for level of knowledge were 2 putting all these subgroups in the light green region.

Overall, the panel members did not consider that stress corrosion cracking is an issue any longer since molybdenum disulfide lubricants were banned. One panel member pointed out that these high strength alloys can be prone to stress corrosion cracking if wetted unless ameliorative actions are taken, and some stress corrosion cracking could then be expected. For subgroups 4.9 and 9.6, the aggregated panel scores for susceptibility to stress corrosion cracking were statistical mode 1 with one higher call, and an average of 0.88 for subgroup 10.6. Subgroups 4.9 and 9.6 were conservatively put in to the higher susceptibility category and are thus colored yellow. The average scores for level of knowledge were from 1.63 to 2.13 putting these subgroups in the light yellow region for subgroups 4.9 and 9.6 and the dark green region for subgroup 10.6.

For subgroup 10.6, the panel evaluated erosion corrosion including steam cutting and cavitation. The panel considered that steam cutting is a known issue and possible if there is a flange leak. The average panel score for susceptibility to erosion corrosion in subgroup 10.6 was 1.75, and the average score for level of knowledge was 2 putting this subgroup in the light-yellow region.

3.2.1.2.8 Wrought and Forged Ni-base Alloy Components

The subgroups containing Inconel components and which are color-coded light yellow are 11.11 and 11.19 (steam generator, Alloy 82, 52 – channel head cladding, and channel head divider plate made of Alloy 690). The degradation mechanism evaluated for these subgroups was stress corrosion cracking. For subgroup 11.11 reductions in fracture resistance and debonding also were evaluated. For subgroup 11.19, fatigue was also evaluated as a potential degradation mechanism.

Stress corrosion cracking was evaluated for both the subgroups 11.11 and 11.19. The panel pointed out that there is a potential generic issue for stress corrosion cracking of Alloy 82. There has been relatively good field experience with only a few incidents in Alloy 82 and none in Alloy 52 or 152. With regard to stress corrosion cracking in Alloy 690, the panel cited good experience to date, and that no problems are expected although the long term experience is limited. The average panel scores for susceptibility to stress corrosion cracking for subgroups 11.11 and 11.19 were 1.5 and 1, and the average scores for level of knowledge were 2.63 and 2.25 respectively putting all these subgroups in the dark yellow or dark green region.

Debonding was evaluated for subgroup 11.11. The panel noted that debonding is for the most part a fabrication issue and thus of theoretical concern for plant aging and which is very unlikely.

The average panel score for susceptibility to debonding in subgroup 11.11 was 1.14, and the average score for the level of knowledge was 2.86 putting this subgroup in the dark yellow region.

Reduction in fracture resistance was evaluated for subgroup 11.11. The panel was divided in its evaluation of the reduction in fracture resistance; several panel members did not consider this to be an issue. Other panel members pointed out that there is laboratory evidence that low temperature aging of weld metal can occur and lead to increased susceptibility to stress corrosion cracking. The panel score for susceptibility to reduction in fracture resistance in subgroup 11.11 was statistical mode 1 with one higher call. Subgroup 11.11 was conservatively put in to the higher susceptibility category and is thus colored yellow. The average score for the level of knowledge was 1.88 putting this subgroup in the light-yellow region.

Fatigue was evaluated for subgroup 11.19. Some panel members considered that there was a possibility of corrosion fatigue, and that the temperature difference across the divider plate might lead to adverse loading conditions. The area adjacent to the weld and the triple point between the channel head bowl, the divider plate and the tube sheet is likely to be the most susceptible region. The average panel score for susceptibility to fatigue in subgroup 11.19 was 1.13, and the average score for level of knowledge was 2 putting this subgroup in the light yellow region.

3.2.1.2.9 Type 308/309 Stainless Steel Dissimilar Metal Welds – External Surfaces

Subgroup 11.17 (steam generator) containing type 308/309 stainless steel dissimilar metal welds is color-coded light yellow when the external surfaces are evaluated with respect to stress corrosion cracking. External stress corrosion cracking is a known issue in the weld metal dilution zone with low alloy steel, leading to sensitization. Cracking may even occur before the component is taken into service. More extensive discussions of these phenomena can be found in Appendix B.3.

The average panel score for susceptibility to stress corrosion cracking was 1.5, and the average score for level of knowledge was 2 putting this subgroup in the light-yellow field.

3.2.1.3 Less Susceptible Subgroups in the RCS

Subgroups in the reactor coolant system which fell into the dark yellow or green regions are shown in the following modified rainbow charts, Figures 3.8 and 3.9 respectively. The subgroups have been sorted in the same manner as for the red and light-yellow component subgroups. For more information on the evaluation and scoring for these subgroups, the reader is referred to Appendices D and E.

Subgroup Description		Degradation Mechanism									
		BAC	CREV	DBND	EC	FAC	FAT	FR	PIT	SCC	WR
Wrought Type 304/316 Stainless Steel Components											
1.2	Wrought SS 304/316 Piping										
2.2	Wrought SS 304/316 Piping										
6.2	Wrought SS 304/316 Piping										
11.21	SS Line Contact/Drilled Hole TSP		*								
13.2	Wrought 304/316 SS Piping									*	
Type 308/309 Stainless Steel Clad Components											
2.11	SS Clad Ferritic Piping										
3.11	SS Clad Ferritic Piping										
4.3	SS 308/309 Cladding										
10.3	Type 308/309 SS Clad/ Welds										
11.4	308/309 SS Chaneel Head Clad			*				*			
Low-Alloy Steel Components											
4.2	Shell/Plates, Forgings, Welds							*			
11.2	Shell/Plates, Forgings					*			*	*	
11.3	LAS Nozzles/Welds	*				*			*		
11.10	SA-553 Gr. A Manways										
Stainless Steel and Inconel Dissimilar Metal Welds -- External Surfaces											
4.11	308/309 Dissimilar Weld - Ext.										
4.12	Alloys 82/182 Dissim. Welds - Ext.									*	
10.11	308/309 Dissimilar Welds - Ext.										
10.12	Alloy 82/182 Dissim. Welds - Ext.										
Inconel Alloy 690 Components											
11.13	Alloy 690 TT SG Tubes etc.										
11.15	Alloy 690 TT SG Tubes Sec. Side										
Type 304 Stainless Steel Weld HAZ											
12.2	Type 304 SS HAZ (low fluence)										

NOTE: * Susceptibility at color interface with one or more scores higher than this interface.

Figure 3.8 Modified Rainbow Chart Showing Dark-Yellow Subgroups in the PWR Reactor Coolant System

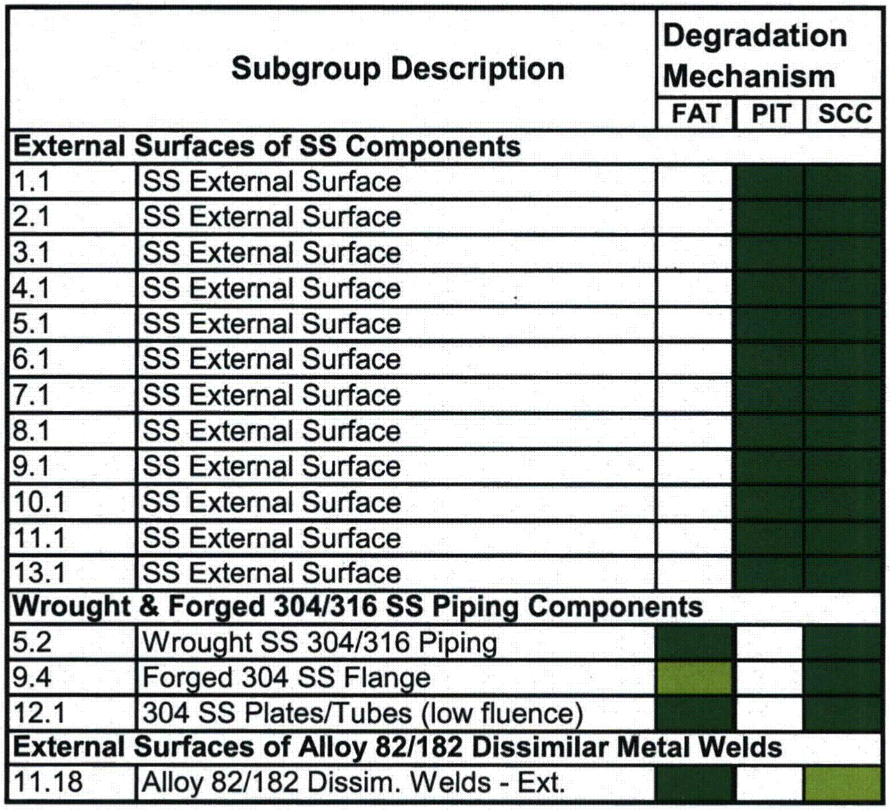


Figure 3.9 Modified Rainbow Chart Showing Green Subgroups in the PWR Reactor Coolant System

3.2.2 PWR Engineered Safety Features/Emergency Core Cooling System

As defined here, the Engineered Safety Features/Emergency Core Cooling System (ESF/ECCS) comprises eleven groups (Groups14 – 23 and Group 48) which include ten piping sub-systems and the containment penetrations for process piping. These groups were organized into 94 component subgroups for assessment purposes.

3.2.2.1 <u>PWR ESF/ECCS Components with Red Susceptibility</u>

The three component subgroups in the emergency core cooling system that fell into the red susceptibility regions are listed in Table 3.4 and illustrated in the modified rainbow chart in Figure 3.10. None of the Engineered Safety Features component subgroups were color-coded red.

Table 3.4 Red Components in the PWR Emergency Core Cooling System (ECCS)

Component	Subgroups	Degradation mechanisms considered
Dissimilar austenitic welds	18.13, 19.10, 22.8	Stress corrosion cracking Fatigue Reduction in fracture resistance

Subgroup Description		Degradation Mechanism		
		FAT	FR	SCC
Alloy 82/182 Dissimilar Metal Welds Exposed to High-T Primary Water				
18.13	308/309, 82/182 Dissim. Weld			
19.10	308/309, Alloy 82/182 Dissim. Weld			
22.8	308/309, 82/182 Dissim. Weld			

Figure 3.10 Modified Rainbow Chart Showing Red Subgroups in the PWR Engineered Safety Features/Emergency Core Cooling System

3.2.2.1.1 Dissimilar Austenitic Welds

The stainless steel dissimilar metal welds which fall into the highest susceptibility category are found the subgroups 18.13 (accumulator piping to RCS cold leg, for CE and B&W plants), 19.10 (SI/RHR piping to RCS hot leg, for CE and B&W plants), and 22.8 (CVCS piping to RCS cold leg, for CE and B&W plants). All of the systems are high-temperature [291-345°C (556-653°F)] systems with primary water environment. It was not always possible to distinguish between stainless steel (Type 308/309) and nickel base alloy (Alloys 182/82) weldments from the list of parts available and the panel then described the subgroup with both notations. The scoring was based on the assumption that the welds were Alloy 182/82 since these materials are considered to be much more susceptible to stress corrosion cracking. The panel assessed stress corrosion cracking, fatigue and reduction in fracture resistance as the potential degradation mechanisms for these components.

The panel described stress corrosion cracking in these components manufactured in the nickel base alloys 182 and 82 as a generic issue which is expected to occur first after exposure to PWR primary water for long periods of time. The panel also held the view that there is an insufficient understanding of the problem to mitigate the cracking. Some panel members also indicated a need for development of inspection and prediction tools, not least because of the long and very variable crack initiation times and wide dispersion in propagation rates. Sensitization, cracking in the dilution zone with low alloy steels, and the possibility of low-temperature aging were also listed as contributing factors. More extensive discussions of stress corrosion cracking of dissimilar metal welds can be found in Appendix B.6.

All of the components were scored in the light-red region. The panel pointed out that, for dissimilar metal welds manufactured with the stainless steel weld metals, the service experience is much better than for the nickel-base alloys, and the susceptibility to stress corrosion cracking would correspondingly have been rated significantly lower.

The average scores of the panel were between 2.5 and 2.88 for susceptibility to stress corrosion cracking in the nickel-alloy dissimilar metal welds in all the subgroups. The average panel scores for the knowledge level were 1.75 to 1.88 putting the components in the light-red region.

The panel considered that these components are also likely to be affected by fatigue, which could be accelerated by the primary water environment. The panel thought that this is only likely to be a problem if the current cumulative usage factor is greater than 0.1 (approximately). However, insufficient information was available to determine if this was the case. Some panel members were more concerned for the environmental effects on fatigue in the stainless steel than in the nickel base welds. Panel members pointed out that there are limited laboratory data for the stainless steel weld metal, but that it might be possible to use data for the corresponding wrought material.

The susceptibility to fatigue was given average scores of 1.75 or 2 putting the component subgroups in the yellow region. The panel scores for the knowledge level were 1.88 or 2 for all the subgroups. Therefore all the subgroups are in the light-yellow field.

The panel considered that these components might also be susceptible to a reduction in fracture resistance. This is based on laboratory data which to date are insufficient to exclude the possibility of a type of fracture toughness degradation known as Low Temperature Crack Propagation; see Appendix B.13. The average scores for susceptibility for all the subgroups were 1.38, putting all these subgroups into the yellow region. The average scores for the level of knowledge were 2.13 or 2.25. These subgroups therefore fall in the dark yellow region, but relatively close to the borderline to the light yellow.

3.2.2.2 PWR ESF/ECCS Components with Light-Yellow Susceptibility

There were no light-yellow component subgroups in the Engineered Safety Features. The 20 component subgroups in the emergency core cooling subsystems falling into the light-yellow region are listed in Table 3.5 and illustrated in the modified rainbow chart, Figure 3.11.

Table 3.5 Light Yellow Components in the PWR Emergency Core Cooling Systems

Component	Subgroups	Degradation Mechanisms Considered
Type 304 Stainless steel socket welds	14.6, 15.6, 16.6, 17.7, 18.7, 19.6, 20.6, 21.6, 22.7	Fatigue Stress corrosion cracking
Cast stainless steel piping components	17.5	Fatigue Reduction in fracture resistance Stress corrosion cracking
Type 308 Stainless steel piping welds and Type 308/309 Dissimilar metal welds	18.5, 18.11, 19.9, 22.5	Fatigue Reduction in fracture resistance Microbiologically induced corrosion Stress corrosion cracking
Type 304/316 Stainless steel piping weld HAZs	18.10.1, 18.10.2, 19.8.1, 19.8.2	Fatigue Stress corrosion cracking
Forged type 304/316 Stainless steel piping components	18.12, 19.11	Fatigue Stress corrosion cracking

	Subgroup Description	Degradation Mechanism			
		FAT	FR	MIC	SCC
Type 304 SS Socket Welds					
14.6	Type 304 Socket Welds				*
15.6	Type 304 Socket Welds				
16,6	304 Socket Welds (Stagnant)				
17.7	304 Socket Welds (Stagnant)				
18.7	Type 304 Socket Welds (Stagnant)				
19.6	Type 304 Socket Welds (Stagnant)				*
20.6	Type 304 Socket Welds				*
21.6	Type 304 Socket Welds				*
22.7	Type 304 Socket Welds				*
Cast SS Piping Components					
17.5	CASS Components (Stagnant)	*	*		*
Type 308 SS Piping Welds & Type 308/309 Dissimilar Metal Welds					
18.5	308/309, 82/182 Dis. Weld - Int.	*			
18.11	Type 308 SS Weld (High T/P)		*		
19.9	Type 308 SS Weld (High T/P)				
22.5	Type 308 SS Weld (High Temp.)				
Type 304/316 SS Piping Weld HAZs					
18.10.1	Type 304 SS HAZ (High T/P)				X
18.10.2	Type 316 SS HAZ (High T/P)				X
19.8.1	Type 304 SS HAZ (High T/P)				X
19.8.2	Type 316 SS HAZ (High T/P)				X
Forged 304/316 SS Piping Components					
18.12	Forged 304/316 Nozzles (Hi T/P)				
19.11	Forged 304/316 Nozzles (High T/P)				
NOTES:	* Susceptibility at color interface with one or more scores higher than interface; [x] Susceptibility inside color box with one or more scores higher than this color box upper interface.				

Figure 3.11 Modified Rainbow Chart Showing Light-Yellow Subgroups in PWR Engineered Safety Features/ Emergency Core Cooling System.

3.2.2.2.1 Type 304 Stainless Steel Socket Welds

The socket welds which fall into the light yellow susceptibility category are found in subgroups 14.6 (RWST header piping), 15.6 (CVCS pump suction piping), 16.6 (SI suction pump piping), 17.7 (RHR pump suction piping), 18.7 (accumulator piping to RCS cold leg), 19.6 (SI/RHR piping to RCS hot leg), 20.6 (RHR pump discharge piping), 21.6 (RHR piping to RCS cold leg) and 22.7 (CVCS piping to RCS cold leg) all of which operate at temperatures 38-177°C (100-350°F), except for 16.6 which operates at ambient temperature, with borated demineralized water as the environment. Subgroups 14.6, 16.6, 17.7, 18.7 and 19.6 normally operate under stagnant conditions. The degradation mechanisms considered are fatigue and stress corrosion cracking. Not all the subgroups were evaluated for stress corrosion cracking. More extensive discussions of these phenomena can be found in Appendix A and Appendices B.14 and B.6 respectively.

Socket welds are known to fail in service (albeit at a low rate) and one panel member indicated that the failure frequency might be lower at these lower temperatures. The geometry of socket welds makes them prone to low and high cycle fatigue owing to the fact that they are a relatively flexible attachment to a more robust component. The loading will depend on the design details and extent of flow-induced vibrations. Some panel members pointed out that the latter were unlikely in stagnant lines, but did not differentiate in their scoring of various subgroups.

The average scores of the panel for susceptibility to fatigue were 2 for the socket welds in all the subgroups. The panel average scores of the knowledge level were also 2, thus with regard to fatigue, the components fall in the light yellow region.

The panel evaluated subgroups 14.6, 19.6, 20.6, 21.6 and 22.7 with respect to stress corrosion cracking. Overall the panel considered that the chemistry conditions were such that stress corrosion cracking would not be expected as long as the chemistry was properly managed. If there is ripple loading some panel members would expect corrosion fatigue to predominate. Since the subgroups include the weld HAZ, according to some panel members, SCC could be experienced and that high stresses and the presence of cold work are aggravating factors despite the absence of oxygen.

For all the subgroups, the scores for susceptibility to stress corrosion cracking were statistical mode 1 with one or two higher calls. These subgroups were conservatively put in the higher susceptibility category and are thus colored yellow. The average scores for the level of knowledge were 2.83 for subgroup 14.6 and 2.75 for the other four subgroups. With regard to stress corrosion cracking, the components were in the dark yellow field.

3.2.2.2.2 Cast Stainless Steel Piping Components

There is only one subgroup with cast stainless steel piping components in the emergency core cooling systems which was classified in the light yellow region: 17.5 (RHR pump suction piping). This subgroup operates at 38-177°C (100-350°F) with an environment of borated demineralized water and is normally stagnant during operation. The degradation mechanisms considered for this subgroup were fatigue, reduction in fracture resistance and stress corrosion cracking.

The panel pointed out that the laboratory data for fatigue in cast stainless steel materials is not as robust as for the wrought materials and that fatigue would only be likely to be a problem if there were flow induced vibrations during operation. The panel's score for susceptibility was statistical mode 1 with one higher call. This subgroup was conservatively put in to the higher susceptibility category and is thus colored yellow. The average score for the level of knowledge was 2.13, putting this component subgroup in the dark-yellow field.

The panel considered that there was little likelihood of thermal aging and the associated reduction in fracture resistance at these low temperatures, although the variations in composition could be a factor, see Appendix B.4. The panel score for susceptibility was statistical mode 1 with one higher call and for the level of knowledge it was 2.13, putting this subgroup in the dark-yellow field.

The panel pointed out that there are sufficient doubts concerning the susceptibility to stress corrosion cracking that a testing program is being considered in Europe on thermally aged CASS. Other panel members noted that there was good field experience and that the temperature was sufficiently low that the potential for thermal aging effect on stress corrosion cracking was inconsequential. The panel score for susceptibility to stress corrosion cracking was statistical mode 1 with one higher call and for the level of knowledge it was 2, conservatively putting this component subgroup in the light-yellow region.

3.2.2.2.3 Type 308 Stainless Steel Piping Welds and Type 308/309 Dissimilar Metal Welds

The stainless steel welds in the emergency core cooling systems which fall into the light-yellow susceptibility field are found in subgroups 18.5 and 18.11 (accumulator piping to RCS cold leg), 19.9 (SI/RHR piping to RCS hot leg) and 22.5 (CVCS piping to RCS cold leg) all of which operate at temperatures 288-316°C (550-600°F) with PWR primary water as the internal environment. Subgroup 18.5 normally operates under stagnant conditions and it was not clear if this was a stainless steel or nickel-base alloy weld. The degradation mechanisms considered for these component subgroups were fatigue, reduction in fracture toughness and stress corrosion cracking and, for subgroup 18.5, also microbiologically induced corrosion.

The panel pointed out that the laboratory data base for fatigue in stainless steel weld metals in PWR primary water is not as extensive as for the wrought materials and that fatigue would only likely be a problem if there were flow-induced vibrations during operation. Dead legs close to the main coolant line in subgroups 18.11 and 19.9 could also be vulnerable to thermal fatigue. The panel scores for susceptibility to fatigue were statistical mode 1 with one higher call for subgroup 18.5, statistical mode 2 for subgroups 18.11 and 19.9, and average of 1.13 for subgroup 22.25. The scores for the level of knowledge were 2, putting all these subgroups in the light-yellow field.

The panel considered that there was little likelihood of thermal aging and the associated reduction in fracture resistance at the low temperatures in subgroup 18.5, although the variations in composition could be a factor; see Appendix B.4. For the other subgroups, which normally operate at higher temperatures, the panel noted that the effect of the environment was not clear but that there is no evidence that the reduction in fracture toughness will be significant under conditions relevant to plant operations. The panel scores for susceptibility to reduction in fracture resistance were average of 1.14 for subgroups 18.5 and 11.9, statistical mode 1 with one higher call for subgroup 18.11, and statistical mode 1 with no higher calls for subgroup 22.5. The level of knowledge scores ranged from 1.75 to 2. All these subgroups, therefore, fall in the light yellow field except for subgroup 22.5 which is in the light green field.

Stress corrosion cracking of the nickel-base alloys which might be included in subgroup 18.5 is a generic problem and is discussed in Appendix B.6. The panel pointed out that the low temperature during operation of this subgroup would probably make it less susceptible. For the other subgroups that were specified as stainless steel Type 308 weld metal, the panel noted that there is very good field experience and that stress corrosion cracking is not anticipated to be a significant problem under normal operating conditions unless low-temperature thermal aging occurs. There was some concern expressed for the dilution zone in carbon or low alloy steels. The average panel scores for susceptibility to stress corrosion cracking ranged from 1.13 to 1.25. The scores for the level of knowledge ranged from 2.13 to 2.25, putting all these subgroups in the dark-yellow field.

The panel included an evaluation of microbiologically induced corrosion for subgroup 18.5 because these lines are normally stagnant and at low temperatures. They considered, however, that the high concentration of boric acid would probably mitigate the issue; the supply of sulfates and nutrients is another important influencing factor. The average panel score for susceptibility to microbiologically induced corrosion for subgroup 18.5 (accumulator piping to RCS cold leg) was 1, and the score for the level of knowledge was 2, putting this subgroup in the light green field.

3.2.2.2.4 Type 304/316 Stainless Steel Piping Weld Heat Affected Zones

The Type 304/316 stainless steel weld HAZs in the emergency core cooling systems which fall into the light-yellow susceptibility category are found in subgroups 18.10.1 and 18.10.2 (accumulator piping to RCS cold leg) and 19.8.1 and 19.8.2 (SI/RHR piping to RCS hot leg) all of which operate at about 316°C (600°F) with PWR primary water as the environment. The degradation mechanisms considered for these subgroups were fatigue and stress corrosion cracking. The panel decided to subdivide these subgroups to distinguish between the two stainless steel materials with respect to their susceptibility to stress corrosion cracking. Subgroups 18.10.1 and 19.8.1 were Type 304 weld HAZs and subgroups 18.10.2 and 19.8.2 were Type 316 weld HAZs. The panel did not distinguish between the two materials with regard to fatigue, the other degradation mechanism evaluated for these subgroups.

The panel pointed out that the laboratory data for fatigue in stainless steel weld HAZs is not as extensive as for the wrought materials and that fatigue would only be likely to be a problem if there were flow induced vibrations during operation. Dead legs close to the main coolant line in these subgroups could also be vulnerable to thermal fatigue. The average panel scores for susceptibility to fatigue were 1.38 and the scores for the level of knowledge were 2, putting all these subgroups in the light-yellow field.

With regard to stress corrosion cracking, the panel noted that there has been very good field experience for both these materials. The major source for concern for some panel members is the degree of cold work from, for example, surface grinding. Drawing an analogy to the BWR experience, these panel members considered that Type 304 might be expected to be more susceptible than Type 316, but that under normal operating conditions it is not expected to a problem. Ingress of oxygen to regions where it could be trapped could be a concern, and one panel member indicated that there is some laboratory evidence of initiation and slow propagation in hydrogenated water even in the non-sensitized but cold worked condition.

The average panel scores for susceptibility to stress corrosion cracking were 1.5 and 1.38 for Type 304 and 316 HAZs respectively. The average scores for the level of knowledge were 2.63 and 2.5 for subgroups 18 and 19 respectively, putting all the subgroups in the dark-yellow region.

3.2.2.2.5 Forged Type 304/316 Stainless Steel Piping Components

The Type 304/316 forged stainless steel piping components in the emergency core cooling systems which fall into the light-yellow susceptibility category are found in subgroups 18.12 (accumulator piping to RCS cold leg) and 19.11 (SI/RHR piping to RCS hot leg) both of which operate at about 316°C (600°F) with PWR primary water as the environment. The degradation mechanisms considered for these subgroups were fatigue and stress corrosion cracking.

The panel pointed out that the laboratory data for fatigue in forged stainless steel are not as extensive as for the wrought materials and that fatigue would only be likely to be a problem if there were flow-induced vibrations during operation. Dead legs close to the main coolant line in these subgroups could also be vulnerable to thermal fatigue. The average panel scores for susceptibility to fatigue were 1.63 and the average scores for the level of knowledge were 1.88, putting these subgroups in the light-yellow field.

	Subgroup Description	Degradation Mechanism					
		CREV	FAT	GC	MIC	PIT	SCC
External Surfaces of SS Components & Dissimilar Metal Welds							
14.1	SS External Surface						
15.1	SS External Surface					X	
16.1	SS External Surface						
17.1	SS External Surface						
18.1	SS External Surface						
19.1	SS External Surface						
21.1	SS External surface						
22.1	SS External surface						
18.8	308/309, 82/182 Dis. Weld - Ext.						
Type 304/316 Stainless Steel Weld HAZs							
14.3	Type 304/316 SS HAZ						
15.3	Type 304/316 SS HAZ						
16.3	Type 304/316 SS HAZ (Stagnant)						
17.3	Type 304/316 SS HAZ (Stagnant)						
18.3	Type 304/316 SS HAZ (Stagnant)						
19.3	Type 304/316 SS HAZ (Stagnant)						
21.3	Type 304/316 SS HAZ						
22.3	Type 304/316 SS HAZ						
23.3	Type 304/316 SS HAZ						
Wrought & Forged 304/316 SS Piping Components							
15.2	Wrought 304/316 SS Piping						*
15.5	Forged 304/316 SS Nozzles						*
17.2	304/316 SS Piping (Stagnant)		*				*
17.6	Forged 304/316 Nozzles (Stagnant)						*
19.2	304/316 SS Piping (Stagnant)		*				
20.5	Forged 304/316 SS Nozzles						
21.2	Wrought 304/316 SS Piping						
21.5	Forged 304/316 SS Nozzles						
22.2	Wrought 304/316 SS Piping						
22.6	Forged 304/316 SS Nozzles						*
23.2	Wrought 304/316 SS Piping						
23.5	Forged 304/316 SS Nozzles						*
Type 308 Stainless Steel Piping Welds							
15.4	Type 308 SS Weld						*
17.4	Type 308 SS Weld (Stagnant)						*
21.4	Type 308 SS Weld						
22.4	Type 308 SS Weld						*
23.4	Type 308 SS Weld						*
Type 304 Stainless Steel Socket Welds							
23.6	Type 304 Socket Welds						*
Components of Containment Penetrations for Process Piping							
48.2	Type 304 Sleeve Dissim. Weld	*					
48.3.1	Penetration Piping - Int.			*		*	
48.5.1	SA106, GR.B Penet. Piping - Int.	*	X	*	*	*	*
48.5.2	SA106, GR.B Penet. Piping - Ext.						
48.7	Fluid Head	*	*	*	*	*	*
48.8	Component HAZ	*	*	*	*	*	*
48.9	Component Welds	*	*	*	*	*	*
48.11	Flanges	*	*	*	*	*	*
48.12	Flange HAZ	*	*	*	*	*	*
48.13	Flange Welds	*	*	*	*	*	*
48.14	CS Bellows		*				*
NOTES	* Susceptibility at color interface with one or more scores higher than interface; X Susceptibility inside color box with one or more scores higher than this color box upper interface.						

Figure 3.12 Modified Rainbow Chart Showing Dark-Yellow Subgroups in PWR Engineered Safety Features/ Emergency Core Cooling System

Subgroup Description		Degradation Mechanism					
		CREV	FAT	GC	MIC	PIT	SCC
External Surfaces of SS Components							
18.9	SS External Surfaces (High T/P)					■	■
19.7	SS External Surfaces (High T/P)					■	■
23.1	SS External surface					■	■
Wrought & Forged 304/316 SS Piping Components							
14.2	Wrought 304/316 SS Piping		■		■		■
14.5	Forged 304/316 SS Nozzles		■		■		■
16.2	304/316 SS Piping (Stagnant)		■		■		■
16.5	Forged 304/316 Nozzles (Stagnant)		■		■		■
18.2	304/316 SS Piping (Stagnant)		■		■		■
18.6	Forged 304/316 Nozzles (Stagnant)		■		■		■
19.5	Forged 304/316 Nozzles (Stagnant)		■				■
Type 308 Stainless Steel Piping Welds							
14.4	Type 308 SS Weld		■		■		■
16.4	Type 308 SS Weld (Stagnant)		■		■		■
18.4	Type 308 SS Weld (Stagnant)		■		■		■
19.4	Type 308 SS Weld (Stagnant)		■				■
Components of Containment Penetrations for Process Piping							
48.1	SS Type 304, LAS, CS Comp.	■	■	■	■	■	■
48.3.2	Penetration Piping - Ext.	■	■	■	■	■	■
48.4.1	Type 304/316L Penet. Piping - Int.	■	■	■	■	■	■
48.4.2	Type 304/316L Penet. Piping - Ext.	■	■	■	■	■	■
48.6	Penet. Piping and Dissim. Welds	■	■	■	■	■	■
48.10	Leak Chase Channel Plug	■	■	■	■	■	■

Figure 3.13 Modified Rainbow Chart Showing Green Subgroups in PWR Engineered Safety Features/ Emergency Core Cooling System

The panel pointed that there has been very good field experience of forged stainless steel components with regard to stress corrosion cracking and that there is no direct laboratory evidence for stress corrosion cracking in these components. Very high levels of cold work and low temperature aging could be aggravating factors, however. The average panel score for subgroups 18.12 and 19.11 were 1.13 and 1.25 for susceptibility to stress corrosion cracking and 2.5 and 2.25 respectively for the level of knowledge. Thus both subgroups fall in the dark-yellow field.

3.2.2.3 ESF/ECCS Less-Susceptible Component Subgroups

Component subgroups in the engineered safety features/emergency core cooling system which fell into the dark yellow or green regions are shown in the modified rainbow charts, Figures 3.12 and 3.13, respectively. The subgroups have been sorted in the same manner as for the red and light-yellow component subgroups. For more information on the evaluation and scoring for these subgroups the reader is referred to Appendices D and E.

3.2.3 Steam and Power Conversion System

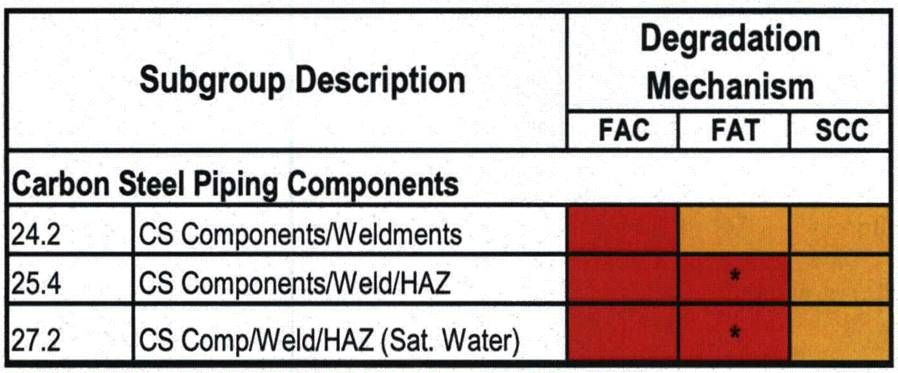
The PWR steam and power conversion system (PWR-S&PCS) as defined here includes groups 24-27, namely the main steam, main feedwater, auxiliary feedwater, and steam generator blowdown subsystems. The PWR-S&PCS was organized into 12 component subgroups for assessment purposes. Three of these subgroups fall into the red susceptibility region and none into the light-yellow region.

3.2.3.1 <u>Steam and Power Conversion System Components with Red Susceptibility</u>

The three steam and power conversion system subgroups with components falling into the red susceptibility regions are listed in Table 3.6 and illustrated in the modified rainbow chart in Figure 3.14.

Table 3.6 Red Components in the PWR Steam and Power Conversion System

Component	Subgroups	Degradation Mechanisms Considered
Carbon steel piping components	24.2, 25.4, 27.2	Flow-accelerated corrosion Fatigue Stress corrosion cracking



<u>NOTES:</u> * Susceptibility at color interface with one or more scores higher than interface.

Figure 3.14 Modified Rainbow Chart Showing Red Subgroups in PWR Steam and Power Conversion System

3.2.3.1.1 Carbon Steel Piping Components

The carbon steel piping components in the steam and power conversion system which fall into the red susceptibility category are found in subgroups 24.2 (main steam), 25.4 (main feedwater) and 27.2 (steam generator blowdown). All the subgroups include welds and HAZs. Subgroup 24.2 operates at 229-277°C (445-530°F) in an environment of saturated steam with < 0.25% moisture. Subgroup 25.4 operates at 232°C (450°F) in an environment of demineralized water at pH 9-10. Subgroup 27.2 operates at 288°C (550°F) in an environment of saturated water from the steam generator. The degradation mechanisms considered for these subgroups were flow-accelerated corrosion, fatigue and stress corrosion cracking.

The panel noted that flow-accelerated corrosion is a well-known phenomenon for carbon-steel components. Some of the important factors noted by the panel are the local moisture content of the steam, high turbulent flow, high blowdown velocities, two-phase environments, temperatures above 93°C (200°F), pH, and the chromium content of carbon steel. Pipe wall thickness measurements can indicate wall thinning and predictive models exist to identify areas at risk. Field experience has been good with a few notable exceptions, those in recent years being attributable to human error. Flow-accelerated corrosion is discussed in more detail in Appendix B.17.

The average panel scores for the three subgroups for susceptibility to flow-accelerated corrosion were between 2.25 and 2.5 and the scores for the level of knowledge ranged from 2.75 to 3. All three subgroups were thus in the dark-red region.

The panel noted that there is generally good field experience to date with regard to fatigue damage but that a few failures have been attributed to fatigue due to excessive vibration. This is most likely to occur in small diameter lines and weldolets. Corrosion fatigue is a known phenomenon in carbon steel piping in the presence of cyclic or ripple loading, but it is often difficult to distinguish it from stress corrosion cracking. High residual stresses, surface finish, corrosive impurities, fabrication defects, and crevices are all factors which contribute to corrosion fatigue.

The average panel score for susceptibility to fatigue was 1.25 for subgroup 24.2 and statistical mode 2 with one higher call for subgroups 25.4 and 27.2. Subgroups 25.4 and 27.2 were conservatively put in the higher susceptibility category and are thus colored red; subgroup 24.2 is in the yellow region. For the level of knowledge, the average panel scores ranged from 2.38 to 2.75, putting all the subgroups in the dark-color field.

The panel did not feel that stress corrosion cracking in these components is likely to be an issue. Carbon steel can sustain stress corrosion cracking at these temperatures but the deoxygenated steam is not likely to support cracking. However for subgroup 24.2, one panel member felt that steam oxidation might be a potential degradation mechanism leading to stress corrosion cracking.

The average panel scores for the subgroups for susceptibility to stress corrosion cracking were between 1.13 and 1.38 and the scores for the level of knowledge were 2.63. All the subgroups were thus in the dark-yellow field.

3.2.3.2 Less-Susceptible Component Subgroups in the Steam and Power Conversion System

There were no steam and power conversion system subgroups in the light-yellow field. Subgroups which fell into the dark yellow or green regions are shown in the following modified rainbow charts, Figures 3.15 and 3.16. The component subgroups have been sorted in the same manner as for the other major systems. More information on the evaluation and scoring for these subgroups can be found in Appendices D and E.

Subgroup Description		Degradation Mechanism							
		BAC	CREV	FAC	FAT	FR	MIC	PIT	SCC
External Surfaces of CS & LAS Components									
24.1	CS/LAS External Surface								
CS & LAS Piping Components									
24.3	LAS Components			X					
26.1	CS Component/Weld/HAZ Ext.								
26.2	CS Component/Weld/HAZ								
27.3	CS Comp/Weld/HAZ (Demin. Water)								*
Alloy 690 Piping Components									
25.3	I-690 Forging/Weld/HAZ					X			*

NOTES: * Susceptibility at color interface with one or more scores higher than interface; [x]Susceptibility inside color box with one or more scores higher than this color box upper interface.

Figure 3.15 Modified Rainbow Chart Showing Dark-Yellow Subgroups in PWR Steam and Power Conversion System

Subgroup Description		Degradation Mechanism				
		FAT	FR	MIC	PIT	SCC
External Surfaces of CS & LAS Components						
25.1	CS External Surface					
27.1	CS Component/Weld/HAZ Ext.					
External Surfaces of Alloy 690 Piping Components						
25.2	I-690 Forging/Weld/ HAZ Ext.		X			

NOTE [x]Susceptibility inside color box with one or more scores higher than this color box upper interface.

Figure 3.16 Modified Rainbow Chart Showing Green Subgroups in PWR Steam and Power Conversion System

3.2.4 Support and Auxiliary System

The support and auxiliary system (SS&AS) as defined here comprises 20 groups, (Groups 28-47), and includes the service water, CVCS, CCW and spent fuel pool subsystems. The panel organized the 20 groups into 142 component subgroups for evaluation purposes.

3.2.4.1 Components in the Support and Auxiliary System with Red Susceptibility

The nine subgroups with components in the Support and Auxiliary System falling into the red susceptibility regions are listed in Table 3.7 and illustrated in the modified rainbow chart in Figure 3.17.

Table 3.7 Red Components in the PWR Support and Auxiliary System

Component	Subgroups	Degradation mechanisms considered
Carbon steel components exposed to untreated service water	28.3, 28.4, 29.2, 29.4, 30.2	Crevice corrosion Erosion corrosion Flow-accelerated corrosion Fatigue General corrosion Microbiologically induced corrosion Pitting corrosion Stress corrosion cracking
Copper zinc heat exchanger tubes	29.3	Flow-accelerated corrosion Microbiologically induced corrosion Pitting corrosion Stress corrosion cracking
Type 304 stainless steel socket welds in CVCS piping	31.11, 32.6, 38.6	Fatigue Stress corrosion cracking

Subgroup Description		Degradation Mechanism							
		CREV	EC	FAC	FAT	GC	MIC	PIT	SCC
CS Components Exposed to Untreated Service Water									
28.3	CS Comp/Weld/HAZ (Pond)						*	*	
28.4	CS Comp/Weld/HAZ (Lake/Sea)								
29.2	CS Comp/Weld/HAZ (Pond)						*	*	
29.4	CS CCW HX Shell and Tubesheets						*	*	
30.2	CS Comp/Weld/HAZ (Pond)						*	*	
Copper Zinc Heat Exchanger Tubes									
29.3	CCW HX Copper Zinc tubes			X					*
Type 304 SS Socket Welds in CVCS Piping									
31.11	304 SS Socket Welds (High T)								*
32.6	Type 304 Socket Welds								*
38.6	Type 304 Socket Welds								*

NOTES: * Susceptibility at color interface with one or more scores higher than interface; x Susceptibility inside color box with one or more scores higher than this color box upper interface.

Figure 3.17 Modified Rainbow Chart Showing Red Subgroups in PWR Support and Auxiliary System

3.2.4.1.1 Carbon Steel Components Exposed to Untreated Service Water

The carbon steel components in the support and auxiliary systems which fall into the red susceptibility category are found in the subgroups 28.3 and 28.4 (service water suction piping from pond, lake, or sea), 29.2 and 29.4 (service water pump discharge piping) and 30.2 (service water piping inside containment). All the subgroups include welds and HAZs except subgroup

29.4, which consists of the heat exchanger shell and tubesheets. These subgroups operate at a maximum of 38°C (100°F) in an environment of untreated service water. Subgroup 28.4 is a lined pipe exposed to salt water and panel members pointed out in their comments that the corrosion degradation mechanisms would only occur if the coating failed. The degradation mechanisms considered for these subgroups were crevice corrosion, erosion corrosion including steam cutting and cavitation, flow-accelerated corrosion, fatigue, galvanic corrosion, microbiologically induced corrosion, pitting corrosion and stress corrosion cracking. Only microbiologically induced corrosion and pitting corrosion were evaluated for all the subgroups.

The panel pointed out that pitting corrosion is to be expected in carbon steel components in this environment. It will occur in stagnant lines and areas in which deposits may accumulate in the presence of oxygen and other impurities such as sulfates and chlorides. The average panel scores for the subgroups for susceptibility to pitting corrosion were 2.43 or statistical mode 2 with one or two higher calls, conservatively putting all these subgroups in the red region. The average scores for the level of knowledge were from 2.71 to 2.88. All the subgroups were thus in the dark-red field.

The panel considered that microbiologically induced corrosion is also to be expected in carbon steel components exposed to raw water and that the same factors as for pitting corrosion are relevant as well as the presence of nutrients. The average panel scores for the subgroups for susceptibility to microbiologically induced corrosion were 2.13 or statistical mode 2 with one or two higher calls, conservatively putting all these subgroups in the red region. The average scores for the level of knowledge ranged from 2.63 to 2.88. All the subgroups were thus in the dark-red field.

The panel considered flow accelerated corrosion as a potential degradation mechanism for subgroups 28.3, 29.2 and 30.2. The panel does not anticipate that FAC will be a serious problem in these systems because of the low temperature and flow rates at which they operate although flow-accelerated corrosion has been observed at lower temperatures than the generally accepted threshold of 93°C (200°F); see Appendix B.17 for more details. The average panel scores for the subgroups for susceptibility to flow-accelerated corrosion were between 1.14 and 1.25 putting all these subgroups in the yellow region. The average scores for the level of knowledge were from 2.4 to 2.75. All the subgroups were thus in the dark-yellow field.

Subgroups 28.4, 29.4 and 30.2 were evaluated with respect to crevice corrosion. For this degradation mechanism, the panel particularly pointed out that the susceptibility of subgroup 28.4 would depend on the nature of the coating. For the other subgroups, the panel considered that crevice corrosion would likely depend on the geometry of the components and would most likely occur where the carbon steel is coupled to other materials such as stainless steel in crevices such as threads, flanges, and the like. The average panel scores for the subgroups for susceptibility to crevice corrosion ranged from 1.75 to 2.29 putting these subgroups in the yellow or red (subgroup 28.4) regions. The average scores for the level of knowledge were 2.71 and 2.75. Subgroups 29.4 and 30.2 were thus in the dark-yellow field and subgroup 28.4 in the dark-red field.

Stress corrosion cracking was also evaluated by the panel for subgroups 28.3, 29.2 and 30.2. The panel considered that stress corrosion cracking could occur in these subgroups despite the low temperature. It will depend on the surface conditions and can also initiate from pits and is most likely in the vicinity of the welds. Some panel members thought that it would be more likely to occur from the outer surfaces in the presence of chlorides from the insulation. The average panel scores for the subgroups for susceptibility to stress corrosion cracking were 1 for subgroups 28.3 and 29.2 and 1.13 for subgroup 30.2 putting these subgroups in the green and yellow regions. The average scores for the level of knowledge were 2.63. Subgroups 28.3 and 29.2 were thus in the dark-green field and subgroup 30.2 was in the dark yellow field.

Fatigue was considered as a potential degradation mechanism for subgroups 29.2 and 30.2. The panel noted that although fatigue is common in carbon steel piping, it would be unlikely in these subgroups unless flow induced vibration occurs. Fatigue can also be aggravated by the presence of high stresses (for example at flanges), corrosive impurities, and oxygen. The average panel scores for the subgroups for susceptibility to fatigue were 1.13, putting these subgroups in the yellow region. The average scores for the level of knowledge were 2.63 putting these subgroups in the dark-yellow field.

General corrosion was only evaluated for subgroup 29.4 (CCW HX shell and tubesheets and fittings). The panel pointed out that unless adequate corrosion protection is in place, wall thinning (both internal and external) can be expected. The average panel score for this subgroup for susceptibility to general corrosion was 1.5 putting it in the yellow region. The average score for the level of knowledge was 2.88, putting it in the dark-yellow field.

Cavitation was only explicitly evaluated for subgroup 30.2. Cavitation is included in the group erosion corrosion, abbreviated as EC. Some panel members considered that cavitation could be a possible degradation mechanism in carbon steel pumps and valves. Also, if particulates are present, then these can contribute to flow accelerated corrosion. The average panel score for this subgroup for susceptibility to cavitation was 0.83 putting it in the green region. The average score for the level of knowledge was 2.5 putting it in the dark-green field.

3.2.4.1.2 Copper Zinc Heat Exchanger Tubes

The copper zinc heat exchanger tubes in the support and auxiliary systems which fall into the red susceptibility category are found in the subgroup 29.3 (service water pump discharge piping). The degradation mechanisms evaluated for this subgroup were flow-accelerated corrosion, microbiologically induced corrosion, pitting corrosion and stress corrosion cracking. The operational temperature is a maximum of 38°C (100°F) and the internal environment is CCW water and the external environment is pond water. One panel member commented that there was insufficient definition of this subgroup to make a quantitative evaluation possible.

The panel considered that the susceptibility to pitting corrosion and stress corrosion cracking would depend on the specific copper alloy used, and that brasses are more susceptible than CuNi alloys. Stagnant flow, the build up of deposits, oxidizing conditions and the presence of chlorides, ammoniacal radicals and other contaminants are all aggravating factors. All of these factors are also aggravating factors for microbiologically induced corrosion although the panel noted that copper alloys are less prone than ferrous materials. The panel commented that copper alloys and brasses are susceptible to flow-accelerated corrosion under conditions of high flow rates [>1.5 m/s (5 fps)] and turbulence.

The average panel score for this subgroup for susceptibility to pitting corrosion was 2.13 putting it in the red region. The average score for the level of knowledge was 2.75 putting it in the dark-red field.

The panel score for this subgroup for susceptibility to stress corrosion cracking was statistical mode 2 with one higher call conservatively putting it in the red region. The average score for the level of knowledge was 2.63 putting it in the dark-red field.

The average panel score for this subgroup for susceptibility to microbiologically induced corrosion was 1.38 putting it in the yellow region. The average score for the level of knowledge was 2.63 putting it in the dark-yellow field.

The average panel score for this subgroup for susceptibility to flow-accelerated corrosion was 1.75 putting it in the yellow region. The average score for the level of knowledge was 2.88 putting it in the dark-yellow field.

3.2.4.1.3 Type 304 Stainless Steel Welds in CVCS Piping

The Type 304 stainless steel welds in the support and auxiliary systems which fall into the red susceptibility category are found in the subgroups 31.11 (CVCS pump piping to crossover leg injection), 32.6 (CVCS normal letdown piping) and 38.6 (CVCS regenerative HX piping to cold leg). All these subgroups are socket welds operating at up to 293°C (560°F) with PWR primary water as the environment. The degradation mechanisms considered by the panel for these subgroups were fatigue and stress corrosion cracking.

The panel pointed out that there has been significant field experience of fatigue failures in socket welds depending on design details and flow induced vibrations. The average panel scores for these subgroups for susceptibility to fatigue were 2.13 putting them in the red region. The average scores for the level of knowledge were 2 putting them in the light-red field.

The panel did not consider that there was evidence to support stress corrosion cracking to be the cause of the field failures. Stress corrosion cracking is not anticipated to be a long term problem with well managed chemistry although it is possible. The panel's score for these subgroups for susceptibility to stress corrosion cracking was statistical mode 1 with one higher call conservatively putting them in the yellow region. The average scores for the level of knowledge were 2.53 or 2.75 putting them in the dark-yellow field.

3.2.4.2 Components in the Support and Auxiliary System with Light-Yellow Susceptibility

The 29 subgroups with components in the support and auxiliary system falling into the light yellow susceptibility regions are listed in Table 3.8 and illustrated in the modified rainbow chart in Figure 3.18.

3.2.4.2.1 Type 308 Stainless Steel Piping Welds

The Type 308 stainless steel piping welds in the support and auxiliary systems which fall into the light yellow susceptibility category are found in the subgroups 31.4 and 31.9 (CVCS pump piping to crossover leg injection), 33.4 (CVCS regenerative HX piping to letdown HX), 34.4 (CVCS letdown HX piping to VCT), 35.4 (CVCS mixed bed piping to filter), 36.4 (CVCS VCT piping to charging pump suction), 37.4 (CCS charging pump piping to regenerative HX), 38.4 (CVCS regenerative HX piping to cold leg) and 39.4 (CVCS injection filter piping to RCP seals). All of these subgroups have PWR primary water as the environment at operating temperatures of 54-143°C (130-290°F) or in the case of subgroups 31.9 and 38.4 the operating temperature is 291°C (557°F). The degradation mechanisms considered by the panel for these subgroups were fatigue, reduction in fracture resistance and stress corrosion cracking.

The panel pointed out that the laboratory data for fatigue in stainless steel weld metal materials is not as extensive as for the wrought materials and that fatigue would only be likely to be a problem if there were flow induced vibrations or thermal cycling during operation. Dead legs close to the main coolant line could be potentially vulnerable to thermal fatigue, in particular. For the subgroups operating at the lower temperatures, one panel member commented that the temperature was too low for there to be a concern about environmental effects on the fatigue of stainless steel.

The average panel scores for susceptibility to fatigue in these subgroups were 1.13 or 1.25 putting them in the yellow region. The average panel scores for the level of knowledge were 1.88 or 2 putting all these subgroups in the light-yellow field.

The panel did not consider that stress corrosion cracking is a serious issue in the subgroups that operate at low temperature. Surface finish and high stress together with impurities and oxygenating conditions, such as the presence of hydrogen peroxide which is introduced as part of the shut down procedure, are aggravating factors. At the higher temperature, the panel pointed out that there is an uncertainty concerning the role of thermal aging on stress corrosion cracking.

The average panel scores for susceptibility to stress corrosion cracking in these subgroups were from 0.5 to 0.75 putting them in the green region, except for the high temperature subgroups, 31.9 and 38.4, where the average score for subgroup 31.9 was 1.13 and the score for subgroup 38.4 was statistical mode 1 with one higher call, putting them in the yellow region. The average panel scores for the level of knowledge were from 2.38 to 2.88 putting all these subgroups in the dark-yellow or dark green fields.

For subgroups 31.9 and 38.4, the panel also considered reduction of fracture resistance as a potential degradation mechanism. Some members of the panel commented that there is some possibility for the thermal aging of this material at the higher operating temperatures for these subgroups. This possibility is, however, very much lower than it would be for cast material or the HAZ-strained material. The average panel scores for susceptibility to reduction of fracture resistance in these subgroups were 1.13 and 1, putting them in the yellow and green regions respectively. The average panel scores for the level of knowledge were 2.17 and 2.5, putting these two subgroups in the dark-yellow and dark-green fields respectively.

Table 3.8 Light Yellow Components in the PWR Support and Auxiliary System

Component	Subgroups	Degradation Mechanisms Considered
Type 308 Stainless steel piping welds	31.4, 31.9, 33.4, 34.4, 35.4, 36.4, 37.4, 38.4, 39.4	Fatigue, Reduction in fracture resistance, Stress corrosion cracking (SCC)
Type 304/316 Stainless steel weld HAZs	31.8, 32.3, 33.3, 34.3, 35.3, 36.3, 37.3, 38.3, 39.3, 40.3	Fatigue, Reduction in fracture resistance, Stress corrosion cracking
Type 304 Stainless steel socket welds	31.6, 33.6, 34.6, 35.6, 37.6, 39.6	Fatigue, Stress corrosion cracking
Forged Type 304/316 Stainless steel piping components	31.10, 32.5, 38.5	Fatigue, Stress corrosion cracking
High strength bolts and studs	31.12	Boric acid corrosion, Fatigue, Reduction in fracture resistance, SCC

Subgroup Description		Degradation Mechanism								
		BAC	CREV	FAC	FAT	FR	GC	MIC	PIT	SCC
External Surfaces of CS & LAS Components and Dissimilar Metal Welds										
28.1	CS Comp/Weld/HAZ External Surface									*
28.2	CS Comp/Weld/HAZ Ext. (Buried)				*					
29.1	CS Comp/Weld/HAZ Ext.									
35.9	Dissimilar Weld Ext.									
Wrought & Forged Type 304/316 SS Piping Components										
31.2	Wrought 304/316 SS Piping									
31.7	304/316 SS Piping (High T)									
32.2	Wrought 304/316 SS Piping									
33.2	Wrought 304/316 SS Piping									
34.2	Wrought 304/316 SS Piping									
35.2	Wrought 304/316 SS Piping									
36.2	Wrought 304/316 SS Piping									
37.2	Wrought 304/316 SS Piping									
38.2	Wrought 304/316 SS Piping									
39.2	Wrought 304/316 SS Piping									
40.2	Wrought 304/316 SS Piping									
31.5	Forged 304/316 SS Components									
33.5	Forged 304/316 SS Components									
34.5	Forged 304/316 SS Components									
35.5	Forged 304/316 Components									
36.5	Forged 304/316 SS Components									
37.5	Forged 304/316 Components									
39.5	Forged 304/316 Components									
40.5	Forged 304/316 Components									
Cast Stainless Steel Piping Components										
35.8	CASS CF8 Components				*					X
36.7	CASS CF8 Components				*					X
Type 308 Stainless Steel Piping Welds										
32.4	Type 308 SS Weld									*
40.4	Type 308 SS Weld					*				
Type 304/316 Stainless Steel Weld HAZs										
31.3	Type 304/316 Piping HAZ									
High Strength Bolts and Studs										
34.7	SA193 Gr B16 or B7 Bolts	*				*				*
34.8	Studs SA453 Gr 660					*				
35.7	SA193 Gr B7 Flange Bolts	*				*				*
37.7	Flange SA193 Gr B7 Bolts	*				*				*
36.6	SA193 Gr B7 Flange Bolts	*				*				*
39.7	Flange SA193 Gr. B16 Bolts	*				*				*
40.6	Flange SA193 Gr. B16 Bolts	*				*				*
CS & LAS Piping System Components										
33.7	CS Letdown HX Shell, Nozzles									
41.2	CS/LAS Elbows									
41.3	CS/LAS Pipe Fittings									
41.4	CS/LAS Valves									
41.5	CS/LAS Piping			*						*
41.6	LAS Flanges			*						
41.7	CS Surge Tank Components									
41.8	SA285 Surge Tank Weld									
41.10	CS CCW HX Nozzles						*	*		
42.2	LAS Elbows		*						*	

NOTES: * Susceptibility at color interface with one or more scores higher than interface; x Susceptibility inside color box with one or more scores higher than this color box upper interface.

Figure 3.18 Modified Rainbow Chart Showing Light-Yellow Subgroups in PWR Support and Auxiliary System

3.2.4.2.2 *Type 304/316 Stainless Steel Weld Heat-Affected Zones*

The Type 304/316 stainless steel weld HAZs in the support and auxiliary systems which fall into the light yellow susceptibility category are found in the subgroups 31.8 (CVCS pump piping to crossover leg injection), 32.3 (CVCS normal letdown piping), 33.3 (CVCS regenerative HX piping to letdown HX), 34.3 (CVCS letdown HX piping to VCT), 35.3 (CVCS mixed bed piping to filter), 36.3 (CVCS VCT piping to charging pump suction), 37.3 (CCS charging pump piping to regenerative HX), 38.3 (CVCS regenerative HX piping to cold leg), 39.3 (CVCS injection filter piping to RCP seals) and 40.3 (CVCS RCP seal return piping to filter). All of these subgroups have PWR primary water as the environment at operating temperatures of 54-143°C (130-290°F) or in the case of subgroups 31.8, 32.3 and 38.3 the operating temperature is 292-293°C (557-560°F). The degradation mechanisms considered by the panel for these subgroups were fatigue and stress corrosion cracking and in some cases, reduction in fracture resistance.

The panel commented that the laboratory data for fatigue in stainless steel weld metal HAZs materials is not as extensive as for the wrought materials but that fatigue would only be likely to be a problem if there were flow induced vibrations or thermal cycling during operation. Dead legs close to the main coolant line could be potentially vulnerable to thermal fatigue. For the subgroups operating at the lower temperatures, one panel member commented that the temperature was too low for there to be a concern about environmental effects on the fatigue of stainless steel. Another panel member commented that the susceptibility to corrosion fatigue may be increased because of thermal aging in the subgroups operating at the higher temperature.

The average panel scores for susceptibility to fatigue in these subgroups were from 1.13 to 1.38, putting them in the yellow region. The average panel scores for the level of knowledge were 1.88 or 2, putting all these subgroups in the light-yellow field.

The panel noted that there has been very good field experience with regard to stress corrosion cracking in Type 304/316 stainless steel weld HAZs in PWRs. There are, however, some laboratory data indicating that cold worked material may be susceptible even under nominal water conditions. At the higher temperature, the effect of possible thermal aging is unresolved. Some panel members pointed out that long initiation times could mean that the problem has not yet been observed. Surface finish, impurities such as chlorides and sulfates, high residual stresses, and oxidants such as hydrogen peroxide which is added during shutdown are identified as aggravating factors. Several panel members considered that the subgroups operating at the lower temperatures are probably not susceptible to stress corrosion cracking although the weld HAZs are high stress regions.

The average panel scores for susceptibility to stress corrosion cracking in these subgroups were from 1.38 to 1.63 putting them in the yellow region. The average panel scores for the level of knowledge were from 2.38 to 3 putting all these subgroups in the dark-yellow field.

Reduction of fracture resistance was evaluated for subgroups 31.8, 32.3, 33.3, 38.3 and 40.3. The panel did not expect this to be an issue for these subgroups but pointed out that data are scarce. Some questions were raised about the in-situ response of HAZ-strained material both at the higher and the lower temperatures. The average panel scores for susceptibility to reduction of fracture resistance were 0.5 or 0.2 for subgroups 33.3 and 40.3 respectively putting them in the green region, and 1.17 for the other three subgroups putting them in the yellow region. The average panel scores for the level of knowledge were from 1.67 to 2 for all the subgroups putting them in the light-yellow or light green fields except for subgroup 33.3 for which the average score was 2.5 putting it in the dark-green field.

3.2.4.2.3 Type 304 Stainless Steel Socket Welds

The Type 304 stainless steel socket welds in the support and auxiliary system which fall into the light yellow susceptibility category are found in the subgroups 31.6 (CVCS pump piping to crossover leg injection), 33.6 (CVCS regenerative HX piping to letdown HX), 34.6 (CVCS letdown HX piping to VCT), 35.6 (CVCS mixed bed piping to filter), 37.6 (CCS charging pump piping to regenerative HX) and 39.6 (CVCS injection filter piping to RCP seals). All of these subgroups have PWR primary water as the environment at operating temperatures of 46-143°C (115-290°F). The degradation mechanisms considered by the panel for these subgroups were fatigue and stress corrosion cracking.

The panel noted that there has been significant field experience of failures in stainless steel socket welds depending on design detail and flow induced vibration. The average panel scores for susceptibility to fatigue were 2 and the average scores for the level of knowledge were also 2 putting all these subgroups in the light-yellow field.

The panel considered that the susceptibility to stress corrosion cracking was low because of the low operating temperatures for these subgroups. The possibility of growth by stress corrosion after fatigue initiates cracking was noted, even though this would be slow. The average panel scores for susceptibility to stress corrosion cracking were from 0.63 to 0.88 and the average scores for the level of knowledge were 2.75 or 2.88 putting all these subgroups in the dark-green field.

3.2.4.2.4 Forged Type 304/316 Stainless Steel Piping Components

The forged Type 304/316 stainless steel piping components in the support and auxiliary system which fall into the light-yellow susceptibility category are found in the subgroups 31.10 (CVCS pump piping to crossover leg injection), 32.5 (CVCS normal letdown piping) and 38.5 (CVCS regenerative HX piping to cold leg). These subgroups operate at temperatures up to 293°C (560°F) and are in contact with PWR primary water. The degradation mechanisms considered by the panel for these subgroups were fatigue and stress corrosion cracking.

The panel expected data for wrought materials to be applicable for fatigue in these forged stainless steel subgroups of components. Although large cyclic loads were not expected by the panel, regions where flow induced vibration or thermal cycling occur could be susceptible to fatigue. The loading contribution to stress corrosion cracking was also noted as was the possible effects of low temperature aging on corrosion fatigue. The average panel scores for susceptibility to fatigue were 1.38 and the average scores for the level of knowledge were 2 putting all these subgroups in the light-yellow field.

The panel commented that there has been good field experience of stress corrosion cracking in forged austenitic stainless steel and that currently there is no laboratory basis to predict that they would be susceptible. The possible effect of low temperature aging on stress corrosion cracking is not clear. If cracks initiate growth is possible. The average panel score susceptibility to stress corrosion cracking for subgroup 31.10 was 0.75 putting it in the green region. For the other two subgroups the panel scores for susceptibility was statistical mode 1 with 1 or 2 higher calls conservatively putting them in the yellow region. The average score for the level knowledge was 2.38 or 2.5 putting the subgroups in the dark-green and dark-yellow fields.

3.2.4.2.5 High Strength Bolts and Studs

The high strength bolts and studs in the support and auxiliary system which fall into the light yellow susceptibility category are found in the subgroup 31.12 (CVCS pump piping to crossover leg injection). The subgroup is normally exposed only to building air at low temperatures. The degradation mechanisms considered by the panel for these subgroups were boric acid corrosion from primary water leaks, fatigue, reduction of fracture resistance and stress corrosion cracking.

The panel considered that boric acid corrosion could be an issue if there were leaking flanges. There is also the possibility of steam cutting due to leaks. However it was pointed out that serious boric acid corrosion will only occur if such leaks are ignored. The average panel score for susceptibility was 1.13 and for the level of knowledge 2.63 putting it in the dark-yellow field.

The panel noted that there are no known fatigue issues but that this could be a potential degradation mode. The average panel score for susceptibility was 1 and for the level of knowledge 2 putting it in the light-green field.

The panel considered that reduction of fracture resistance should not be an issue if the technical specifications for choice of material are met. There could however be a problem if there are primary water leaks, and hydrogen produced by corrosion may also reduce the fracture resistance. The panel score for susceptibility was statistical mode 1 with one higher call and for the level of knowledge 2.5 conservatively putting it in the dark-yellow field.

The panel noted that stress corrosion cracking has occurred in similar components but not since molybdenum disulfide lubricants were banned. This degradation mode was therefore not considered to be a future threat by the majority of the panel. One panel member pointed out that high strength materials are prone to stress corrosion cracking if wetted at low temperatures and that improper heat treatments and high stresses may further exacerbate it. The average panel score for susceptibility was 1.13 and for the level of knowledge 2, putting it in the light-yellow field.

3.2.4.3 Less-Susceptible Subgroups in the Support and Auxiliary System

Subgroups which fell into the dark yellow or green regions are shown in the following modified rainbow charts, Figures 3.19 and 3.20. The subgroups have been sorted in the same manner as for the other major systems. More information on the evaluation and scoring for these subgroups can be found in Appendices D and E.

3.2.5 Stainless Steel External Surfaces

The panel considered that the external stainless steel surfaces of components in contact with containment air or auxiliary building air at ambient temperatures under certain circumstances can be subject to pitting and stress corrosion cracking, the latter being initiated by the former. This is a low temperature phenomenon and requires both wetted surfaces and, in particular, chloride ions which could come from insulation, contamination during maintenance (for example, inadvertent use of chloride-containing tape), or aerosols. Moisture is only likely to be present on the external surfaces during outages or layup since the temperature of these components under operating conditions is normally higher, up to 293°C (560°F) depending on the system. The build-up of aerosols is expected to be most severe for seaside plants, but will increase with time for all plants. The panel noted that if CalSil is removed from plants because of the sump blockage issues, the availability of buffer from the insulation material could be reduced considerably. More extensive discussion of this can be found in Appendix B.3.

Subgroup Description		Degradation Mechanism								
		BAC	CREV	FAC	FAT	FR	GC	MIC	PIT	SCC
External Surfaces of CS & LAS Components and Dissimilar Metal Welds										
28.1	CS Comp/Weld/HAZ External Surface									*
28.2	CS Comp/Weld/HAZ Ext. (Buried)				*					
29.1	CS Comp/Weld/HAZ Ext.									
35.9	Dissimilar Weld Ext.									
Wrought & Forged Type 304/316 SS Piping Components										
31.2	Wrought 304/316 SS Piping									
31.7	304/316 SS Piping (High T)									
32.2	Wrought 304/316 SS Piping									
33.2	Wrought 304/316 SS Piping									
34.2	Wrought 304/316 SS Piping									
35.2	Wrought 304/316 SS Piping									
36.2	Wrought 304/316 SS Piping									
37.2	Wrought 304/316 SS Piping									
38.2	Wrought 304/316 SS Piping									
39.2	Wrought 304/316 SS Piping									
40.2	Wrought 304/316 SS Piping									
31.5	Forged 304/316 SS Components									
33.5	Forged 304/316 SS Components									
34.5	Forged 304/316 SS Components									
35.5	Forged 304/316 Components									
36.5	Forged 304/316 SS Components									
37.5	Forged 304/316 Components									
39.5	Forged 304/316 Components									
40.5	Forged 304/316 Components									
Cast Stainless Steel Piping Components										
35.8	CASS CF8 Components				*					X
36.7	CASS CF8 Components				*					X
Type 308 Stainless Steel Piping Welds										
32.4	Type 308 SS Weld									*
40.4	Type 308 SS Weld					*				
Type 304/316 Stainless Steel Weld HAZs										
31.3	Type 304/316 Piping HAZ									
High Strength Bolts and Studs										
34.7	SA193 Gr B16 or B7 Bolts	*				*				*
34.8	Studs SA453 Gr 660					*				
35.7	SA193 Gr B7 Flange Bolts	*				*				*
37.7	Flange SA193 Gr B7 Bolts	*				*				*
36.6	SA193 Gr B7 Flange Bolts	*				*				*
39.7	Flange SA193 Gr. B16 Bolts	*				*				*
40.6	Flange SA193 Gr. B16 Bolts	*				*				*
CS & LAS Piping System Components										
33.7	CS Letdown HX Shell, Nozzles									
41.2	CS/LAS Elbows									
41.3	CS/LAS Pipe Fittings									
41.4	CS/LAS Valves									
41.5	CS/LAS Piping			*						*
41.6	LAS Flanges			*						
41.7	CS Surge Tank Components									
41.8	SA285 Surge Tank Weld									
41.10	CS CCW HX Nozzles						*	*		
42.2	LAS Elbows		*						*	

NOTES: * Susceptibility at color interface with one or more scores higher than interface; ^{x} Susceptibility inside color box with one or more scores higher than this color box upper interface.

Fig. 3.19 Modified Rainbow Chart Showing Dark-Yellow Subgroups in PWR Support and Auxiliary System

The panel assessed the susceptibility to stress corrosion cracking and pitting for the external surfaces of the RWST header piping, the CVCS pump suction piping, the SI pump suction piping, the RHR pump suction piping, the accumulator piping to the RCS cold leg, the SI/RHR piping to the RCS hot leg, the RHR pump discharge piping, the RHR piping to the RCS cold leg, the CVCS piping to RCS cold leg, and the safety injection pump discharge piping (subgroups 14, 15, 16, 17, 18, 19, 20, 21, 22, and 23 respectively).

Subgroup Description		Degradation Mechanism									
		CREV	FAC	FAT	FR	GALV	GC	MIC	PIT	SCC	WEAR
CS & LAS Piping System Components (cont.)											
42.3	LAS Pipe Fittings	*									
42.4	LAS Valves	*									
42.5	LAS Piping	*								*	
42.6	CS/LAS Lugs and Flanges										
43.2	LAS Elbows	*							*		
43.3	LAS Pipe Fittings	*									
43.4	LAS Valves	*									
43.5	LAS Piping	*								*	
43.6	CS/LAS Flanges and Lugs										
43.7	SA105, SA106 Sockolets	*									
44.2	LAS Elbows	*							*		
44.3	LAS Pipe Fittings	*									
44.4	LAS Valves	*									
44.5	LAS Piping	*								*	
44.6	LAS Flanges and Lugs										
44.7	SA105 Flexible Hose	*					*		*	*	
SS Components in Spent Fuel Pool and Associated Piping											
45.2	SS or Type 304 Ext. (Concrete)							*			
45.3	SS or Type 304 Elbows										
45.4	SS or Type 304 Pipe Fittings										
45.5	SS or Type 304 Valves										
45.6	SS or Type 304 Piping	*									
45.7	SS or 304 Pipe Flanges and Lugs			*							
45.8	SS or Type 304 Weldolet			*							
45.9	SS SFP HX Comp. (tube side)			*							
45.10	SS SFP Pump Components										
45.11	SS SFP HX Comp. (shell side)		*							*	
46.3	SS Type 304/316 Pipe Fittings										
46.4	SS Type 304/316 Valves										
46.6	SS Type 304/316 Flanges										
46.7	SS Type 304/316 Mixed Bed							*			
47.1	SFP Type 304 SS Components						X				*
47.2	SFP Type 304 SS HAZ			*							
47.3	SFPSS Welds			*							
47.6	SFP Floor SS Liner - Int. Surface							*	*		
47.7	SFP Floor SS Liner - Ext. Surface										
Miscellaneous Components											
29.5	CCW HX SS tubes										
29.6	CCW HX Copper Nickel tubes										
41.9	Cast Iron CCW Pump Casing				*						
47.4	Aluminum Boral Panels						*				
47.5	Zr- Alloy Fuel Assembly						*				

NOTES: * Susceptibility at color interface with one or more scores higher than interface; x Susceptibility inside color box with one or more scores higher than this color box upper interface.

Fig. 3.19 (cont'd) Modified Rainbow Chart Showing Dark-Yellow Subgroups in PWR Support and Auxiliary System

The panel assessed the susceptibility to stress corrosion cracking for all these subgroups as low except for subgroup 15 which overall was rated medium because one panel member rated it medium. The panel overall assessment of susceptibility to pitting for all subgroups (except subgroup 23) was medium although all panel members except one had indicated a low susceptibility. The overall panel assessment for pitting susceptibility of subgroup 23 was low. In all cases, both for pitting and stress corrosion cracking, the panel scores indicated the knowledge to be high.

Subgroup Description		Degradation Mechanism					
		CREV	FAT	GC	MIC	PIT	SCC
External Surfaces of CS & LAS Components							
30.1	CS Comp/Weld/HAZ Ext.						
41.1	CS/LAS External Surface						
42.1	CS/LAS External Surface						
43.1	CS/LAS External Surface						
44.1	CS/LAS External Surface						
External Surfaces of SS Components							
31.1	SS External Surface						
32.1	SS External surface						
33.1	SS External surface						
34.1	SS External surface						
35.1	SS External surface						
36.1	SS External surface						
37.1	SS External surface						
38.1	SS External surface						
39.1	SS External surface						
40.1	SS External surface						
45.1	SS or Type 304 External Surface						
46.1	SS or Type 304 Ext. Surface						
304/316 SS Spent Fuel Pool Piping Components							
46.2	SS Type 304/316 Elbows						
46.5	SS Type 304/316 Piping						

Figure 3.20 Modified Rainbow Chart Showing Green Subgroups in PWR Support and Auxiliary Systems.

3.3 Susceptibility of BWR Components to Materials Degradation

A summary-level discussion is presented below of the results of the PIRT-like assessment by the panel of the susceptibility of representative components in a "typical" BWR to the sixteen materials degradation mechanisms defined in Section 2 (Table 2.8). Both external and internal degradation are included for those components for which both are pertinent.

As in the PWR assessment (Section 3.2), the panel assessed the degradation susceptibilities of subgroups of similar components from the groups of parts provided by BNL. It is important to note that the detail of the division into the subgroups differs from group to group. This is partly because of the variable quality of the underlying documentation defining the components and partly because of decisions made by panel members as they grouped together components which they expected would have similar susceptibilities to degradation because of similarities of component type, material composition and microstructure, and service conditions. As before for the PWR assessment, no objective reason was available to distinguish one component from another in the same subgroup of BWR components.

In all, 297 BWR component subgroups containing 1660 parts were defined and evaluated by the panel:

- 135 subgroups were from the Reactor Coolant System (RCS);
- 66 subgroups were from the Engineered Safety Features/Emergency Core Cooling System (ESF/ECCS);
- 32 subgroups were from the Steam and Power Conversion System (SS&PCS);
- 64 subgroups were from the Support and Auxiliary System (S&AS).

The number of component subgroups created and assessed for BWRs was somewhat smaller than for PWRs, in part because five BWR subsystems were excluded from the assessment on the basis that they were so similar to the corresponding PWR subsystems (which the panel already had assessed). Thus a second assessment was unnecessary. The five BWR subsystems excluded were Containment Isolation Penetrations (ESF/ECCS), Spent Fuel Storage (S&AS), Spent Fuel Pool Cooling and Cleanup (S&AS), Service Water (S&AS), and Component Cooling Water (S&AS). These subsystems include an additional 84 subgroups containing 542 parts. A discussion of the PWR assessment results for these subsystems can be found in Section 3.2 in the subsections on the ESF/ECCS and S&AS major systems.

The color-coded results of the panel evaluations for all the BWR subgroups can be found in the "rainbow" and "flag" charts in Appendix D, where they are grouped according to the same four "major systems" as those defined for PWRs. In addition, Appendix E provides the individual susceptibility and knowledge "calls" by each panelist (together with his/her rationale) for each combination of BWR component subgroup/degradation mechanism that was evaluated by the panel.

As explained in Section 3.1, and illustrated in Figure 3.1, the colors red, yellow and green signify panel assessments of progressively lower degradation susceptibility and likelihood of future degradation. The color red is an indication that there is in-plant experience, or a demonstrated compelling problem, of the specific degradation mechanism in the specific type of component. The color yellow indicates that there is a good basis for expecting degradation of plant components in the future (based, for example, on laboratory test data, or limited plant observations) whereas the color green indicates that there is a low likelihood of significant degradation occurrence in the future. The shades of the colors signify the existing level of knowledge based on the average of the panel scores for the degradation mechanism/component combinations. The lighter color indicates a lower knowledge level and a potential need for further research. The darker color indicates a higher knowledge level that potentially could allow, without further research, the development of mitigation strategies.

It is clearly apparent from the charts in Appendix D that the panel's susceptibility scores indicating the likelihood of future degradation of BWR components are markedly higher for the RCS component subgroups than for the component subgroups from the other three major systems. For example, 55 RCS subgroups are color-coded red (indicating that there is at least one degradation mechanism with high susceptibility) whereas the total number of red-coded subgroups for the other three major systems is only 15. Similarly, 7 RCS subgroups are color-coded light yellow whereas the total number of "light yellows" for the other three major systems is only 4.

In contrast to the PWR situation discussed in Section 3.2, it is apparent from the color shading in the charts of Appendix D that the panel's knowledge scores are low for only a small fraction of the highly-susceptible BWR component subgroups:

- Only 1 of the 70 red component subgroups is color-coded light red;

- There are only 11 light-yellow subgroups compared with 199 dark-yellow subgroups.

The total number of BWR component subgroups color-coded green (17) is small compared with the number color-coded dark yellow (199). The finding of 199 dark yellow color-coded subgroups suggests that, for many BWR degradation mechanisms and susceptible components, the information required for the development of mitigation strategies may be available but has not yet necessarily been fully utilized in operating BWRs.

The panel's evaluation for different component and degradation mechanism combinations apply to both conditions of NWC and HWC. However, when the panel considered that there was a benefit or a difference in behavior under HWC (for fatigue and/or stress corrosion cracking), an additional evaluation was conducted for the component and degradation mechanism under HWC conditions. Thus, components exposed to steam did not have separate evaluations for HWC since no benefit was expected in the steam phase. Therefore a blank (absence of color) in the rainbow charts under the HWC column for a given component-degradation mechanism does not reflect the absence of susceptibility. This indicates that the evaluation is the same as for the adjacent column for the degradation mechanism in general (both NWC and HWC).

A discussion of the BWR results is presented below for each of the four "major systems." These discussions each contain subsections addressing all of the combinations of component subgroups and materials degradation mechanisms that were color-coded red or light-yellow. For component subgroups color-coded dark yellow (i.e., assessed to have medium susceptibility but high knowledge levels), one or two examples are discussed, but most of the components are only tabulated in order of decreasing susceptibility and increasing knowledge. For the tabulated component groups, and for those assessed to have low susceptibility (light or dark-green), the reader is referred to the information provided in Appendices D and F. The discussions concentrate on the particular factors considered relevant for the specific component/degradation mechanism -- more comprehensive discussions of each of the degradation mechanisms can be found in Appendix A and Appendices B.1, B.2, B.4, B.5, B.10, B.13, and B.14.

3.3.1 BWR Reactor Coolant System

The BWR reactor coolant system (BWR-RCS) as defined here, includes nine groups (Groups 1-9), namely the reactor vessel top head, shell, and bottom head, the core shroud, the core-control internals, the jet pump assembly, the ECCS connections, the steam separator/dryer, and the reactor recirculation piping. These nine groups were subdivided by the panel into a total of 135 component subgroups for evaluation purposes.

During power operation, the BWR-RCS components are exposed to high-purity reactor water and/or wet steam at temperatures in the range 232-288°C (450-550°F). In some RCS locations, the reactor water can be either mildly oxidizing or mildly reducing, depending on whether or not hydrogen injection is used. Therefore, for fatigue and stress corrosion cracking (which are particularly sensitive to electrochemical potential) separate assessments were made for normal water chemistry (NWC = no hydrogen injection) and hydrogen water chemistry (HWC = hydrogen injection). (For more information about NWC, HWC and the BWR Water Chemistry Guidelines, please see Appendix B.10.) In addition, some of the reactor vessel internal components are exposed to neutron fluxes that will result in moderate fluences (up to ~10 dpa) within the plant design life. The external environment for BWR-RCS components other than internals is containment air, which is expected to contain both moisture and chloride aerosols during plant outages – the external surfaces of RCS components are generally hot [>121°C (250°F)] and dry, in the absence of leaks, during power operation.

The panel's assessment of the 135 BWR-RCS component subgroups resulted in 55 dark red component subgroups, 7 light-yellow subgroups, 68 dark-yellow subgroups and 5 dark green subgroups. No BWR-RCS components were color-coded light red or light green.

3.3.1.1 BWR RCS Components with Red Susceptibility

The subgroups in the BWR reactor coolant system with components falling into the red susceptibility regions are listed in Table 3.9 and illustrated in the modified rainbow chart, Figure 3.21. In both the table and the figure, red-coded component subgroups from throughout the RCS that are similar to each other have been grouped together in six distinct classes so as to minimize repetition in the subgroup-level discussion. It should also be noted that the columns associated with all degradation mechanisms which the panel considered not viable or likely for any of the red color-coded BWR-RCS component subgroups have been removed from Figure 3.21.

It is clearly apparent in Figure 3.21 that, for the BWR-RCS, the degradation mechanism with the highest susceptibility scores from the panel was stress corrosion cracking in (oxidizing) normal water chemistry which was color-coded dark red in 53 of the 55 BWR-RCS dark-red component subgroups. Fatigue in normal water chemistry was the degradation mechanism leading to the other two dark red color codes for BWR-RCS component subgroups. It is noteworthy that stress corrosion cracking and fatigue are the only degradation mechanisms leading to red color codes. It is also noteworthy that none of the substantial number of low-alloy steel component subgroups in the BWR-RCS appears in Figure 3.21--all of these components were color-coded dark-yellow.

In addition to identifying the BWR-RCS subgroup/mechanism combinations color-coded dark red, Figure 3.21 indicates that the panel scored 18 of the "red" RCS component subgroups light yellow with regard to reduction of fracture resistance. This is of potential significance for components that are vulnerable to fatigue or stress corrosion cracking because reduction over time of the fracture resistance of such components would reduce the critical sizes for growing fatigue and SCC cracks, thereby reducing the remaining useful life of the component. Several panel members suggested that additional fracture resistance tests are needed for all of the austenitic stainless steels and nickel-base alloys, particularly tests in environments simulating the full range of service environments expected for RCS components under both NWC and HWC conditions. This is discussed in more detail in Appendices B.4, B.5, and B.13.

3.3.1.1.1 Alloy 82/182 Welds

The Alloy 82/182 welds which fall into the highest susceptibility category are found in the subgroups 1.7 and 1.12 (reactor pressure vessel closure head), 2.10, 2.12, 2.13 and 2.17 (reactor pressure vessel shell), 3.4, 3.5, 3.9, 3.11 and 3.13 (reactor pressure vessel bottom head), 4.9 (core shroud), 5.8 and 5.16 (core controls) and 6.7 and 6.12 (jet pump assembly). All of these subgroups are in contact with reactor water at temperatures from 219-302°C (427-575°F) except for subgroups 1.7 and 1.12, which are in contact with reactor coolant steam at 286°C (547°F). The panel evaluated the degradation mechanisms of crevice corrosion, debonding, fatigue (NWC & HWC), reduction in fracture resistance and stress corrosion cracking (NWC & HWC). Not all of the degradation mechanisms apply to all of the subgroups.

Table 3.9 Red Subgroups in the BWR Reactor Coolant System

Component	Subgroups	Degradation mechanisms considered
Alloy 82/182 dissimilar metal and Inconel welds	1.7, 1.12, 2.10, 2.12, 2.13, 2.17, 3.4, 3.5, 3.9, 3.11, 3.13, 4.9, 5.8, 5.16, 6.7, 6.12	Crevice corrosion Debonding Fatigue (NWC & HWC) Reduction in fracture resistance Stress corrosion cracking (NWC & HWC)
Type 304/316 stainless steel weld HAZs	1.8, 2.19, 3.6, 3.7, 4.2, 4.4, 4.6, 4.8, 4.10, 5.9, 5.17, 6.3, 6.10, 6.11. 6.13, 7.2, 7.5, 7.8, 7.11, 8.3, 9.6, 9.9	Fatigue (NWC & HWC) Reduction in fracture resistance Stress corrosion cracking (NWC & HWC)
Alloy 600 components and weld HAZs	2.8, 2.9, 3.10, 3.12, 5.10, 6.6	Fatigue (NWC & HWC) Reduction in fracture resistance Stress corrosion cracking (NWC & HWC)
Wrought Type 304/316 components	2.11, 4.14, 4.15, 4.17, 5.2, 5.12, 8.1	Fatigue (NWC & HWC) Reduction in fracture resistance Stress corrosion cracking (NWC & HWC) Wear
Alloy X-750 components	4.20, 6.5	Fatigue (NWC & HWC) Reduction in fracture resistance Stress corrosion cracking (NWC & HWC)
Type 308 stainless steel welds and socket welds	8.2, 9.13	Crevice corrosion Fatigue (NWC & HWC) Reduction in fracture resistance Stress corrosion cracking (NWC & HWC)

The panel evaluated stress corrosion cracking under NWC conditions for all of the subgroups. For HWC conditions the subgroups 1.7 and 1.12 were not included since the panel considered that there will be no benefit from HWC in the steam phase. Under NWC steam conditions the panel pointed out that water films will form on the closure head surface and also that there is a significant risk that stress corrosion cracking will occur in particular in Alloy 182 under the oxidizing conditions in the steam space. Moisture content of the steam, stress concentration and the extent of stress relief and sensitization are all factors affecting susceptibility. The average panel scores for the susceptibility to stress corrosion cracking were 2.25 and, for the level of knowledge, the scores were 2.75 and 2.63 for subgroups 1.7 and 1.12 respectively, putting these subgroups in the dark-red field. Although the level of knowledge scores were generally high, one panel member commented that he/she knew of no laboratory test data in the steam environment.

Subgroup Description		Degradation Mechanism							
		CREV	DE-BOND	FAT	FAT-HWC	FR	SCC	SCC-HWC	WEAR
Inconel Alloy 82/182 Dissimilar Metal and Inconel Welds									
1.7	A508 Nozzle to 304 SS Flange 182 Weld								
1.12	Dryer Hold Down Bracket 182 Weld								
2.10	FW Thermal Sleeve/A508 Nozzle 182 Weld	*							
2.12	82/182 Weld Between CS and Alloy 600								
2.13	82/182 Weld Pad Bet. A508 and Alloy 600	*							
2.17	Alloy 182 Attachment Pads								
3.4	Dissimilar Metal Nozzle Weld 182/82								
3.5	Dissimilar Metal J-Weld 182/82								
3.9	82/182 Weld Pad Bet.Thermal Sleeve & Nozzle	*			*				
3.11	Dissimilar 82/182 Welds				*				
3.13	182 Weldments of Inconel to Inconel			X					
4.9	Alloy 182 Shroud Weld Metal								
5.8	Alloy 182 RPV Stub Weld Metal								
5.16	182 Weld Metal In-Vessel Structures								
6.7	Alloy 182 Weld Metal (AHC) (low fluence)								
6.12	Alloy 182 Adapter Weld Metal (low fluence)								
Type 304/316 Stainless Steel Weld HAZs									
1.8	304 SS Flange HAZ						*		
2.19	304 SS HAZ Jet Pump Riser Bracket								
3.6	316NG or 316L SS HAZ						*	*	
3.7	304 SS HAZ Safe End				*				
4.2	Type 304 SS Vertical HAZ				*			X	
4.4	Type 304 SS Circumferential HAZ				*			X	
4.6	Type 304 SS Vertical HAZ (low fluence)				*			X	
4.8	Type 304 SS HAZ (moderate fluence)							*	
4.10	Type 304 SS Shroud HAZ							X	
5.9	Type 304 SS RPV Stub HAZ							X	
5.17	Type 304 SS In-Vessel HAZ								
6.3	304SS Component HAZ (low fluence)				*			X	
6.10	304 SS Riser Brace HAZ (low fluence)				*			X	
6.11	SS Adapter HAZ (low fluence)				*			X	
6.13	SS HAZ on Adapter and Diffuser (low fluence)				*			X	
7.2	Type 304 SS FW HAZ				*				
7.5	Type 304 SS FW Sparger HAZ				*				
7.8	Type 304 SS Core Spray HAZ				*			X	
7.11	Type 304 SS LPACI HAZ				*			X	
8.3	304SS Steam Dryer HAZ			*					
9.6	316 Component HAZ				*				
9.9	Component HAZ 304				*			X	
Inconel Alloy 600 Components and Weld HAZs									
2.8	Alloy 600 Feedwater Safeend								
2.9	Alloy 600 Feedwater Thermal Sleeve								
3.10	Alloy 600 HAZ CRD Stubtube				*				
3.12	Alloy 600 HAZ				*				
5.10	Alloy 600 RPV Stub HAZ								
6.6	Alloy 600 Access Hole Cover (low fluence)						*		
Type 304/316 Stainless Steel Components									
2.11	SS 316 Safeends and Thermal Sleeves								
4.14	Type 304 SS (with 308L welds)							*	
4.15	Type 304 SS Guide Structure (moderate to high							X	
4.17	Type 304 SS Core Plate Structure						*	*	
5.2	304/316 SS Control Rod Blade							X	
5.12	Type 304 SS In-Core Guide (high fluence)								
8.1	304 SS Steam Seaparator and Dryer			*			X		
Inconel Alloy 750 Components									
4.20	X750 Flow Plug Spring							*	
6.5	X750 (mostly HTH) Holddown Beam (low								
Type 308 Stainless Steel Welds and Socket Welds									
8.2	Type 308/L Steam Dryer Weld Metal			*			X		
9.13	Socket Welds SS 308 on 304 and 316	*							

NOTE * Susceptibility at color interface with one or more scores higher than interface; [x] Susceptibility inside color box with one or more scores higher than this color box upper interface.

A blank in HWC column indicates same color (susceptibility and knowledge score) as adjacent column for same mechanism (NWC conditions).

Figure 3.21 Modified Rainbow Chart Showing Red Subgroups in BWR Reactor Coolant System

The other component subgroups in this set are in contact with water and the panel noted that stress corrosion cracking of Alloy 182 is a known issue under oxidizing conditions and that some field failures have been reported. As discussed in more detail in Appendix B.5, the susceptibility of Alloy 182 can be affected by the stress conditions, including the presence of repair welds and thermal stresses, and cold work and weld strains, which can increase the hardness in both the weld metal and the HAZ. The possibility of crevice formation from a fatigue crack was also noted. Another issue is the corrosion cracking in the dilution zone with low alloy steel for some of these subgroups. Both Alloy 82 and 182 are susceptible to stress corrosion cracking, but the panel considers that Alloy 182 is more susceptible. When considering the effect of HWC on stress corrosion cracking susceptibility in the subgroups, the panel commented that they expected them to be less prone to cracking in HWC than in NWC. However, some panel members commented that the long term effects of HWC on stress corrosion cracking are not yet clear and that it is not clear that the low potentials are in fact achievable at all the locations in these subgroups. One panel member commented that hydrogen may even influence the precipitation behavior and oxidation kinetics in Alloy 182. This degradation mechanism is discussed in more detail in Appendix B.5.

The average panel scores for susceptibility to stress corrosion cracking under NWC conditions for this set of component subgroups were between 2.25 and 2.63 and the scores for the level of knowledge ranged from 2.63 to 3. The average panel scores for susceptibility to stress corrosion cracking under HWC conditions for these subgroups were between 1.38 and 1.63 and the scores for the level of knowledge were from 2.5 to 2.88. These subgroups are therefore in the dark-red and dark-yellow fields for NWC and HWC conditions respectively indicating that stress-corrosion cracking could be mitigated, but not necessarily completely by HWC alone.

The panel evaluated fatigue under NWC conditions for all of the component subgroups. For HWC conditions the subgroups 1.7 and 1.12 were not included since the panel considered that there will be no benefit from HWC in the steam phase. The panel did not have access to any fatigue assessments or cumulative usage factors and assumed that the fatigue loading of the pressure vessel top head did not exceed the original design intent. One panel member pointed out that there could be an increased susceptibility under power uprate conditions. The average panel scores for the susceptibility to fatigue were 0.88 and 1.25 for subgroups 1.7 and 1.12 respectively. For the level of knowledge, the average panel scores were 2.88 and 2.63, putting these subgroups in the dark-green and dark-yellow fields respectively.

The panel members' comments for fatigue did not, to any great extent, distinguish between the NWC and HWC environments for these subgroups. Fatigue loading is considered to be minor and thus within the code-allowable limits. Some panel members noted the possibility of thermal loading and that there have been some failures due to thermal fatigue in subgroup 2.10. In other subgroups, the loading will depend upon the precise configuration and residual stresses of the components. For subgroups 3.9 and 3.11 (pressure vessel bottom head), one panel member commented, based on field experience, that thermal stress transients are important. For example, during startup and shutdown, a slow strain rate may increase the fatigue susceptibility. For subgroup 5.16, one panel member commented that there is uncertainty about vibratory loading on the in-vessel control structures. For the HWC conditions, some panel members pointed out that the effect of frequency and hydrogen on the long-term corrosion fatigue of austenitic alloys is not clear. One member commented that if there were an environmental effect on fatigue, the combination of HWC and Noblechem would mitigate it. Additional information on fatigue degradation and on BWR water chemistry can be found in Appendices B.14 and B.10, respectively.

The average panel scores for susceptibility to fatigue under NWC conditions for these subgroups were 1.13 and 1.25 except for subgroup 1.7 which had a score of 0.88. Therefore, all these subgroups fall in the dark yellow field except for 1.7 which is in the dark green field. The level of knowledge scores were from 2.5 to 2.88. The average panel scores for susceptibility to fatigue under HWC conditions for these subgroups had the same range as for NWC conditions with the exception that two subgroups (3.9 and 3.11) had a score of statistical mode 1 with one higher call; subgroups 3.9 and 3.11 were therefore conservatively colored yellow. For the level of knowledge, the scores were slightly lower for HWC than for NWC ranging from 2.38 to 2.75. These subgroups are therefore all in the dark-yellow field for HWC.

Most of the panel members did not consider the reduction of fracture toughness to be a serious concern, but acknowledged there could be some environmental effects. One member of the panel noted his disagreement with that position since there are no data on toughness and fracture resistance of weld metal after long-term exposure to these BWR conditions. Fracture resistance is a larger issue in higher yield strength alloys, and in hardened HAZs, which was pointed out by some panel members. This issue is discussed in more detail in Appendix B.13.

The average panel scores for the susceptibility to reduction of fracture resistance in these subgroups were from 1.25 to 1.5 and for the level of knowledge 2 or 2.13, putting all these subgroups in the light-yellow field except subgroups 1.7, 1.12, 2.10, 2.17, 5.16 and 6.12 which were in the dark-yellow region.

Crevice corrosion was evaluated for subgroups 2.10, 2.13 and 3.9. The panel commented that crevice corrosion can aid stress corrosion initiation, particularly under NWC conditions, but is not a serious concern in itself, given current water chemistry specifications and practice. The panel score for the susceptibility to crevice corrosion for these subgroups was statistical mode 1 with one higher call conservatively coloring them yellow. The average scores for the level of knowledge were 2.75 or 2.88. These groups were in the dark-yellow field.

The panel only evaluated clad debonding for the subgroup 2.17. The panel considered that there could be a theoretical interest for debonding which can occur during manufacture. The average panel score for susceptibility to debonding for this subgroup was 0.88 and the score for the level of knowledge was 2.63, putting it in the dark-green field

3.3.1.1.2 Type 304/316 Stainless Steel Heat-Affected Zones

The Type 304/316 stainless steel heat-affected zones which fall into the highest susceptibility category are found in the subgroups 1.8 (reactor pressure vessel closure head), 2.19 (reactor pressure vessel shell), 3.6 and 3.7 (reactor pressure vessel bottom head), 4.2, 4.4, 4.6, 4.8 and 4.10 (core shroud), 5.9 and 5.17 (core controls), 6.3, 6.10, 6.11 and 6.13 (jet pump assembly), 7.2, 7.5, 7.8 and 7.11 (ECCS connections), 8.3 (steam separator and dryer) and 9.6 and 9.9 (reactor recirculation system). The panel evaluated the degradation mechanisms of fatigue (NWC & HWC), reduction in fracture resistance and stress corrosion cracking (NWC & HWC). All of the subgroups are in contact with reactor water at temperatures 219-288°C (427-550°F) except subgroups 1.8 and 8.3 which are in contact with reactor coolant steam and wet steam respectively at 286-288°C (547-550°F) and were consequently only evaluated under NWC conditions. Subgroups 4.6, 6.3, 6.11 and 6.13 are not expected to achieve more than low fluence by the end of life.

The panel commented, particularly for the subgroups 1.8 and 8.3, which under NWC steam conditions, water films form on the component surfaces. There is a high likelihood that stress corrosion cracking could occur, particularly in the high strength weld HAZs under the oxidizing conditions in the steam space. Stress corrosion cracking has been observed in HAZs in the steam dryers (subgroup 8.3). The panel scores for susceptibility to stress corrosion cracking were statistical mode 2 with one higher call for subgroup 1.8, conservatively putting it into the red region, and an average of 2.28 for subgroup 8.3, also coloring it red. The average panel scores for level of knowledge were 2.88 for both the subgroups putting them in the dark-red field.

For the remaining subgroups in this set the panel commented that stress corrosion cracking is a well-recognized problem in both the plant and laboratory for Type 304/316 stainless steel weld HAZs under the oxidizing conditions pertaining in NWC at reactor temperature, see Appendix B.1 for the typical aggravating factors. Specific comments made by the panel include that thermal stresses are important during start up and shutdown (subgroups 3.6, 7.5, 7.8 and 7.11); that there will be a strong component of cyclic loading for the jet pump riser bracket (subgroup 2.19) and vibration loading for the jet pump adapter (subgroups 6.11 and 6.13); that several of the components in the in-core subgroups are difficult to inspect and in these components irradiation may play a role despite the fact that the fluence is expected to be low. The panel commented that HWC should reduce the susceptibility to stress corrosion cracking and the effects of crevices in these subgroups but that the long term effects are not yet clear. The role of hydrogen in the cracking process is critical but is expected to be less than for the nickel-base alloys.

The average panel scores for susceptibility to stress corrosion cracking under NWC conditions for these subgroups were between 2.25 and 2.5 except for subgroups 1.8 and 3.6 for which the panel score was statistical mode 2 with one higher call conservatively putting them in the red category for susceptibility. For the level of knowledge, the average panel scores were from 2.63 to 3. All these subgroups therefore fall in the dark red field under NWC conditions. The average panel scores for susceptibility to stress corrosion cracking under HWC conditions for these subgroups were between 1.13 and 1.5 except for subgroup 3.6 for which the panel score was statistical mode 1 with two higher calls, and statistical mode 2 with one higher call for subgroup 4.8. The level of knowledge scores ranged from 2.5 to 2.88. These subgroups are therefore all in the dark-yellow field except for subgroup 4.8 which is in the dark-red field under HWC conditions.

The panel evaluated fatigue under NWC conditions for all the subgroups. For HWC conditions, subgroups 1.8 and 8.3 were not assessed because these components are in a steam environment. The panel considered that, for almost all of the subgroups, the fatigue loading was within the code allowable limits. The possibility of environmental effects was noted, and one panel member accentuated this comment pointing out that environmental effects will be significant under low strain rate conditions in particular in subgroups 3.6 and 3.7. Uncertainty in the level of vibration loading was again noted for subgroups 5.17 and 6.3, and thermal fatigue was identified as a potential degradation mechanism for subgroups 7.2, 7.5, 7.8 and 7.11, in particular for the thermal sleeves. For subgroup 8.3, the panel commented that the susceptibility to fatigue loading is very design sensitive. The effect of hydrogen on fatigue and its interaction with ripple loading and dynamic strain aging was noted as a potential concern for HWC conditions.

The average panel scores for susceptibility to fatigue under NWC conditions for these subgroups were between 0.88 and 1.25 except for subgroup 8.3 for which the panel score was statistical mode 2 with two higher calls. These subgroups were therefore in the green (subgroups 1.8, 4.2, 4.4, 4.6, 4.8, 4.10, 6.10, 6.11, 6.13, 7.2, 7.8, 7.11, 9.6, and 9.9), yellow, or conservatively red (subgroup 8.3) regions. For the level of knowledge, the average panel scores were from 2.63 to 3. The average panel scores for susceptibility to fatigue under HWC conditions for subgroups 2.19, 3.6, 4.10, 5.9 and 5.17 were between 0.88 and 1.25. For the other subgroups, the panel score was statistical mode 1 with one higher call putting them conservatively in the yellow susceptibility category. All of the subgroups were in the yellow region except 4.8, which was in the green region. For the level of knowledge, the average panel scores were from 2.38 to 2.75. These subgroups are therefore all in the dark fields of the color regions for both NWC and HWC conditions. More information about fatigue degradation can be found in Appendix B.14.

Most of the panel members did not consider the reduction of fracture toughness to be a very serious concern but agreed that there could be some environmental effects. One member of the panel noted that there are no systematic data on toughness and fracture resistance on weld metal after long term exposure to these conditions. One panel member also expressed a concern about the degree of constraint and lack of data in the environment. This issue is discussed in more detail in Appendix B.13.

The average panel scores for the susceptibility to reduction of fracture resistance in these subgroups were from 1.13 to 1.38 and, for the level of knowledge, were from 2 to 2.17, putting all these subgroups in the dark-yellow field except subgroups 4.2, 4.4, 4.6, 4.8, 5.9, and 6.10 which were in the light-yellow region.

3.3.1.1.3 Alloy 600 Components and Heat-Affected Zones

The Alloy 600 components and heat affected zones which fall into the highest susceptibility category are found in the subgroups 2.8 and 2.9 (reactor pressure vessel shell), 3.10 and 3.12 (reactor pressure vessel bottom head), 5.10 (core controls) and 6.6 (jet pump assembly). All of these subgroups are in contact with reactor water at temperatures between 218-286°C (424-547°F). Subgroup 6.6 is not expected to achieve more than low fluence by the end of life. The panel evaluated the degradation mechanisms fatigue (NWC & HWC), reduction in fracture resistance, and stress corrosion cracking (NWC & HWC).

The panel noted that stress corrosion cracking can occur readily under oxidizing NWC conditions and that the weld HAZs and cold worked and sensitized materials are potentially most susceptible. Typical aggravating factors are discussed in Appendix B.5. The panel commented that they do not expect there to be a large difference in the susceptibility under HWC conditions and that the long term role of hydrogen is not clear. One panel member pointed out that although these subgroups are less susceptible to stress corrosion under HWC conditions, experience from PWRs suggests that it may only be slowed down relative to NWC conditions.

The average panel scores for susceptibility to stress corrosion cracking under NWC conditions for these subgroups were 2.13 or 2.25 except for subgroup 6.6 for which the panel score was statistical mode 2 with one higher call, conservatively coloring it red. For the level of knowledge, the average panel scores were 2.88 or 3. The average panel scores for susceptibility to stress corrosion cracking under HWC conditions for these subgroups were between 1.13 and 1.75 and, for the level of knowledge, were 2.63 or 2.88. These subgroups are therefore in the dark-red and dark-yellow fields for NWC and HWC conditions respectively.

The panel assumed for their judgment of fatigue susceptibility that no significant fatigue loading occurs outside the original design limits. Some events of thermal fatigue have been reported in, for example, the feedwater safe-end sleeve (subgroups 2.8 and 2.9), but this is design specific.

The thermal and mechanical loading is hard to analyze in several of the more complex components, such as the CRDM stub tube (subgroup 3.10), and fatigue can therefore not be dismissed as a potential degradation mechanism. Under HWC conditions one panel member commented that there could be some concern for corrosion fatigue initiation with Noblechem in the higher strength weld HAZs. Again, one panel member noted that the long term role of hydrogen for corrosion fatigue and the possible synergistic effects of dynamic strain aging are not yet clear for Alloy 600.

The average panel scores for susceptibility to fatigue under NWC conditions for these subgroups were 1.13, 1.25, or 1.38 and, for the level of knowledge, were 2.75 or 2.88. The average panel scores for susceptibility to fatigue under HWC conditions for these subgroups was 1.13 or 1.25 except for subgroups 3.10 and 3.12 for which the panel score was statistical mode 1 with one higher call putting these two subgroups conservatively in the yellow region for susceptibility. For the level of knowledge, the average panel scores under HWC were 2.38 or 2.5. These subgroups are therefore all in the dark-yellow field for both NWC and HWC.

Reduction of fracture resistance was not evaluated for subgroup 2.9. In evaluating the other subgroups, most of the panel members commented that there could be a susceptibility for high-strength materials and that there could be an environmental effect. One panel member also expressed a concern about the degree of constraint and lack of data in the environment. This issue is discussed in more detail in Appendix B.13. The average panel scores for the susceptibility to reduction of fracture resistance in these subgroups were from 1.13 to 1.38 and, for the level of knowledge, the scores ranged from 2.13 to 2.25, putting all these subgroups in the dark-yellow field.

3.3.1.1.4 Wrought Type 304/316 Stainless Steel Components

The wrought Type 304/316 stainless steel components which fall into the highest susceptibility category are found in the component subgroups 2.11 (reactor pressure vessel shell), 4.14, 4.15 and 4.17 (core shroud), 5.2 and 5.12 (core controls) and 8.1 (steam separator and dryer). Subgroup 4.14 also included some Type 308 stainless steel welds but the panel's comments regarding that material are not included here. All of the subgroups are in contact with reactor water at temperatures 219-288°C (427-550°F) except subgroup 8.1 which is in contact with wet steam at 288°C (550°F). Subgroup 4.15 is expected to achieve, at most, a moderate fluence by the end of life, subgroup 5.2 at most 4-6 dpa (moderate to high), and 5.12 is a high fluence subgroup. The panel evaluated the degradation mechanisms fatigue (NWC & HWC), reduction in fracture resistance, stress corrosion cracking (NWC & HWC) and, for subgroup 5.12, fretting/wear.

The panel focused on weld HAZs under NWC conditions for stress corrosion cracking of the wrought type 304/316 stainless steels since this is a well-recognized phenomenon and a number of such occurrences have been reported. See Appendix B.1 for typical aggravating factors. For subgroup 5.2, the panel noted that irradiation assisted stress corrosion cracking has been reported. For further discussion of this phenomenon see Appendix B.2. With regard to subgroup 8.1, the panel commented that stress corrosion cracking is possible in wet-steam conditions and will also depend on the level of cold work and the water quality. Subgroup 8.1 was only evaluated for NWC. Panel members commented that, under HWC, the weld HAZs are less susceptible but there was no consensus as to the extent of this improvement and that the long term performance of hydrogen injection on mitigating stress corrosion cracking is not clear.

The average panel scores for susceptibility to stress corrosion cracking under NWC conditions for these subgroups were from 2.13 to 2.5 except for subgroup 4.17 for which the panel score was statistical mode 2 with one higher call conservatively coloring it red, and for subgroup 8.1 for which the average score was 1.63. For the level of knowledge, the average panel scores were from 2.75 to 2.88. The average panel scores for susceptibility to fatigue under HWC conditions for these subgroups were between 1.25 and 1.63 except for subgroups 4.14 and 4.17 for which the panel score was statistical mode 1 with one higher call. For the level of knowledge, the average panel scores were 2.29 to 2.75. These subgroups are therefore in the dark-red and dark-yellow fields for NWC and HWC conditions respectively.

Under NWC conditions, the panel did not expect there to be significant fatigue loading outside the design limits with the exception of thermal fatigue in subgroup 2.11 (nozzle safe-ends and thermal sleeves) for which failures have occurred in plants. The possibility of environmental effects was also noted. For subgroup 8.1, fatigue was identified as a potential degradation mechanism for certain dryer designs and, in particular, for power-uprated plants. Under HWC conditions, panel members commented that there could be some concern for environmental effects at low potentials. Again, one panel member noted that the long term role of hydrogen for corrosion fatigue and the possible synergistic effects of dynamic strain aging are not yet clear for Type 304/316 stainless steel materials.

The average panel scores for susceptibility to fatigue under NWC conditions for these subgroups were 1 or 1.38 except for subgroup 8.1 for which the panel score was statistical mode 2 with one higher call conservatively coloring it red. For the level of knowledge, average panel scores were between 2.25 and 2.88. The average panel scores for susceptibility to fatigue under HWC conditions for these subgroups were 0.86 or 0.88 except for subgroup 2.11 for which it was 1.5. For the level of knowledge, the average panel scores under HWC were from 2.38 to 2.75. These subgroups are therefore all in the dark-green field except for subgroup 2.11 which is in the dark-yellow field for both NWC and HWC. Subgroup 8.1 is in the dark-red field for NWC and was not evaluated for HWC conditions.

Reduction of fracture resistance was not evaluated for subgroup 8.1. In evaluating the other subgroups, most of the panel members commented that there could be a concern regarding an environmental effect on the reduction of fracture resistance and maybe even for low-temperature aging for these stainless steel materials. One panel member also expressed a concern about the degree of constraint and lack of data in the environment and at different fluence levels. This issue is discussed in more detail in Appendices B.2 and B.4. The average panel scores for the susceptibility to reduction of fracture resistance in these subgroups were from 1.25 to 1.5 and the level of knowledge scores were from 2 to 2.14, putting all these subgroups in the dark-yellow field except for subgroup 4.14, which is in the light-yellow field.

Fretting/wear was considered as a potential degradation mechanism for subgroup 5.12 (in-core guide tube assemblies, with no pressure difference across the tube). Panel members commented that this is not a common mode, but that it would be a problem if it occurred. It could be possible if there is contact between the tube and other components or as the result of flow-induced vibration and is, in part, a design issue. The average panel score for susceptibility to wear for this subgroup was 1.29 and, for the level of knowledge the average score was 2.13. This subgroup falls into the dark-yellow field.

3.3.1.1.5 *Alloy X-750 Components*

The Alloy X-750 components which fall into the highest susceptibility category are found in the subgroups 4.20 (core shroud) and 6.5 (jet pump assembly). These subgroups are exposed to reactor water at 274-278°C (525-533°F). Subgroup 6.5 is not expected to achieve more than low fluence by the end of life. The panel evaluated the degradation mechanisms fatigue (NWC & HWC), reduction in fracture resistance and stress corrosion cracking (NWC & HWC).

Most of the panel members did not differentiate in their comments between the susceptibility of Alloy X-750 to stress corrosion cracking in NWC and HWC environments. The susceptibility should be lower under HWC conditions, depending on heat treatment, but the alloy is still expected to be susceptible. Cracking of this material has been reported in operating plants and the susceptibility of this alloy depends upon the final heat treatment as well as on many fabrication and design details. The long term effects of low level irradiation are not known. One panel member commented that HWC/Noblechem will provide effective mitigation. (See Appendix B.10 for more information about HWC and Noblechem.)

Under NWC conditions, the average panel scores for susceptibility to stress corrosion cracking for these subgroups were 2.38 and, for the level of knowledge, the scores were 2.88 and 3. Under HWC conditions, the scores were statistical mode 2 with one higher call or average 2.13 for susceptibility and 2.75 for the level of knowledge. Therefore, these subgroups fall into the dark red field.

For subgroup 4.20 the panel considered that there would be no significant fatigue loading and it therefore would be within the design limits. Under HWC conditions, the long term role of hydrogen in conjunction with dynamic strain aging is not clear. The average panel scores for the susceptibility of subgroup 4.20 to fatigue were 1 and 1.13 and, for the level of knowledge, were 2.88 and 2.5 for NWC and HWC conditions respectively. This component subgroup falls into the dark-green field for NWC and into the dark-yellow field for HWC conditions.

Some vibration and, possibly, ripple loading are expected for subgroup 6.5 which increases the possibility of fatigue. The effect of ripple stresses has not been characterized in conjunction with stress corrosion cracking in Alloy X-750, although there is some evidence for adverse effects from the laboratory and the field. The average panel scores for the susceptibility of subgroup 6.5 to fatigue were 1.63 and 1.25 and, for the level of knowledge, were 2.75 and 2.38 for NWC and HWC conditions respectively. This subgroup falls into the dark-yellow field for both NWC and HWC conditions.

For the reduction of fracture resistance, panel members commented that there could be a long-term reduction in particular if the material is incorrectly heat treated. The panel also pointed out that there is a lack of data concerning the effect of the environment on the reduction of fracture resistance. The average panel scores for the susceptibility of subgroups 4.20 and 6.5 to reduction in fracture resistance were 1.63 and 1.5 and, for the level of knowledge, were 2.25 and 2 respectively. Subgroup 4.20 falls into the dark-yellow field and subgroup 6.5 into the light-yellow field.

3.3.1.1.6 *Type 308 Stainless Steel Welds and Socket Welds*

The Type 308 stainless steel welds and socket welds which fall into the highest susceptibility category are found in the component subgroups 8.2 (steam separator and dryer) and 9.13 (reactor recirculation system). Subgroup 8.2 operates at 288°C (550°F) and is in contact with wet steam. Subgroup 9.13 operates at 288°C (550°F) and the environment is reactor water. The panel evaluated the degradation mechanisms fatigue (NWC) and stress corrosion cracking (NWC) for both subgroups. In addition, reduction of fracture resistance was evaluated for subgroup 8.2 and crevice corrosion, fatigue (HWC) and stress corrosion cracking (HWC) for subgroup 9.13. The subgroups are discussed separately below.

For subgroup 8.2 the panel pointed out that some failures due to fatigue have been seen, in particular in power-up-rated plants. There is significant loading in some areas although it is not well defined. The panel score for susceptibility to fatigue was statistical mode 2 with one higher call and the average score for the level of knowledge was 2.5, conservatively putting this subgroup in the dark-red field.

With regard to stress corrosion cracking of subgroup 8.2, panel members thought that there was not a large likelihood of the Type 308 weld metal being susceptible. One panel member thought that the possibility of low-temperature aging should be considered for this subgroup. The average panel score for susceptibility to stress corrosion cracking was 1.38 and, for the level of knowledge, was 3, putting this subgroup in the dark-yellow field.

For subgroup 8.2, the panel pointed out that the weld metal is not as susceptible to thermal aging as cast austenitic stainless steel and therefore also to reduction of fracture resistance. There could, however, be some adverse effect of the wet steam environment on fracture resistance. There are also some questions about the degree of constraint and the thermal aging kinetics. One panel member pointed out that potential reductions in fracture resistance would be experienced in PWRs before they occurred in BWRs because of the difference in maximum operating temperatures. The average panel score for susceptibility to reduction of fracture resistance was 1.13 and, for the level of knowledge, the average score was 2.13, putting this subgroup in the dark-yellow field.

Panel members noted that fatigue of socket welds (subgroup 9.13) is a design-sensitive issue and that degradation has been observed in the field. This topic was fully discussed by panel members in their assessments of PWR components and summarized in Section 3.2 (Susceptibility of PWR Components). Panel members were therefore relatively sparse with their comments for this BWR subgroup. Panel members pointed out that the environmental effect of HWC will depend on the loading conditions and cyclic frequency. The average panel scores for the susceptibility of subgroup 9.13 to fatigue were 2.5 and 2.25 for NWC and HWC conditions respectively and, for the level of knowledge, the average score was 2.5 for both water chemistries. This subgroup falls into the dark-red field for both NWC and HWC conditions.

Crevice corrosion was considered as a potential degradation mechanism for subgroup 9.13 since there is a possibility that geometrical crevices exist in socket welds. Crevice corrosion could, in this instance, probably be an initiating event for stress corrosion cracking under NWC but probably not under HWC since the concentrating mechanism for acidic anions (essential for both crevice corrosion and cracking) would be absent in the second case. One panel member commented that microstructural changes in Type 308 weld metal after low temperature aging may increase the susceptibility to crevice corrosion. The panel score for susceptibility to crevice corrosion was statistical mode 1 with one higher call and the average score for the level of knowledge was 2.88, putting this subgroup conservatively in the dark-yellow field.

Panel members pointed out that there could be susceptibility to stress corrosion cracking in subgroup 9.13 under NWC conditions if the water purity is not maintained or if the heat affected zones of socket welds were sensitized. Stress corrosion cracking could initiate from pitting and from crevice corrosion. Panel members thought that HWC would decrease the susceptibility to stress corrosion cracking but, for long term performance, hydrogen may not be beneficial. The average panel scores for the susceptibility of subgroup 9.13 to stress corrosion cracking under NWC and HWC conditions were 2.13 and 1.25 respectively and, for the level of knowledge, the scores were 2.88 and 2.75, respectively. This subgroup falls into the dark-red field for NWC and into the dark-yellow field for HWC.

3.3.1.2 <u>BWR RCS Components with Light-Yellow Susceptibility</u>

The seven component subgroups in the BWR reactor coolant system with components falling into the light-yellow susceptibility region are listed in Table 3.10 and illustrated in the modified rainbow chart, Figure 3.22.

Table 3.10 Light-Yellow Subgroups in the BWR RCS

Component	Subgroups	Degradation Mechanisms Considered
Cast stainless steel components	2.18, 5.3	Fatigue (NWC & HWC) Reduction of fracture resistance Stress corrosion cracking (NWC &HWC) Wear
Type 308/308L stainless steel welds	3.8, 4.1, 4.3, 4.5, 4.7	Fatigue (NWC & HWC) Reduction of fracture resistance Stress corrosion cracking (NWC &HWC)

Subgroup Description		Degradation Mechanism					
		FAT	FAT-HWC	FR	SCC	SCC-HWC	WEAR
Cast Stainless Steel Components							
2.18	CF8M Brackets and Guide Rods						
5.3	CF3 A351 CR Guide (low -mod. fluence)		*			*	
Type 308/308L Stainless Steel Welds							
3.8	308 Weldments						
4.1	308/308L Vertical Weld		*			*	
4.3	308/308L Circumferential Weld		*			*	
4.5	Type 308/308L Vertical Weld (low fluence)		*			*	
4.7	Type 308/308L Weld Metal (moderate fluence)		*			*	

NOTE: A blank in the HWC column means the same color (susceptibility and knowledge) score, as the adjacent column for the same mechanism under NWC conditions. * Susceptibility at interface between colors with one or more scores higher than this interface

Figure 3.22 Modified Rainbow Chart Showing Light-Yellow Subgroups in BWR Reactor Coolant System

3.3.1.2.1 Cast Stainless Steel Components

The cast stainless steel components which are colored light-yellow are found in the subgroups 2.18 (reactor pressure vessel shell) and 5.3 (core controls). Both of these subgroups are in contact with reactor water and/or coolant steam at temperatures 274-286°C (525-547°F). Subgroup 5.3 is not expected to achieve more than a low-to-moderate fluence by the end of life. The panel evaluated the degradation mechanisms of fatigue (NWC & HWC), reduction in fracture resistance, stress corrosion cracking (NWC & HWC) and, for subgroup 5.3, wear.

Panel members commented that there is a possibility that stress corrosion cracking can occur in cast stainless steel under NWC conditions if the stresses are sufficiently high. The susceptibility could increase as a result of thermal aging, but there are no data on this. One panel member noted that CF8M appears to be resistant to stress corrosion cracking in oxygenated water, although the reason is not clear. In subgroup 5.3 (control rod guide tube and housing), there may also be a long-term synergistic effect of irradiation. It was noted that the effectiveness of HWC after thermal aging and/or irradiation is not known, but it is expected to be beneficial except for those components in subgroup 2.18 that are exposed to coolant steam.

The average panel scores for the susceptibility of subgroup 2.18 to stress corrosion cracking were 1.5 and 1.25 for NWC and HWC conditions respectively and, for the level of knowledge, were 1.88 for both chemistries. This subgroup falls into the light-yellow field for both NWC and HWC conditions. The panel scores for the susceptibility of subgroup 5.3 to stress corrosion cracking were an average of 1.63 and statistical mode 1 with one higher call conservatively coloring it yellow. The scores for the level of knowledge were 2.13 and 2.38 for NWC and HWC conditions respectively. This subgroup falls into the dark-yellow field for both NWC and HWC conditions.

Panel members expressed some concern about the possible thermal aging of cast stainless steels, which depends on the ferrite content of the component. This phenomenon is discussed in detail in Appendix B.4. Although thermal aging has been studied extensively at higher temperatures, there is little data for fracture resistance in the service environment. The average panel scores for the susceptibility of subgroups 2.18 and 5.3 to reduction in fracture resistance were 1.63 and, for the level of knowledge, were 2.38 and 2 respectively. Subgroup 2.18 falls into the dark-yellow field and subgroup 5.3 into the light-yellow field.

Panel members noted that fatigue failures have been reported in the steam dryers (subgroup 2.18) from some plants in particular after power uprates. For the other components in these subgroups the panel did not consider fatigue loading outside the original design limits. For these subgroups, panel members commented that there could be potential environmental effects under HWC conditions (depending on the frequency of cyclic loading). The average panel scores for susceptibility to fatigue for subgroups 2.18 and 5.3 were 1.38 and 1 and, for the level of knowledge, the scores were 2.75 and 2.88 under NWC conditions. For HWC conditions, the scores were an average of 1.25 and statistical mode 1 with one higher call for susceptibility conservatively coloring it yellow, and 2.5 and 2.63 for the level of knowledge. These subgroups fall into the dark-yellow field except for subgroup 5.3 under NWC conditions which falls in the dark-green field.

The panel considered fretting/wear as a potential degradation mechanism for subgroup 5.3 (control rod guide tube and housing). The panel commented that this is a design issue. If it did occur, it would be a significant problem. The effects of NWC and HWC on tribo-corrosion are not known. The average panel score for susceptibility to wear was 1.14 and, for the level of knowledge, was 2.29, putting this subgroup in the dark-yellow field.

3.3.1.2.2 Type 308/308L Stainless Steel Welds

The Type 308/308L stainless steel welds which are colored light-yellow are found in the subgroups 3.8 (reactor pressure vessel bottom head), and 4.1, 4.3, 4.5, and 4.7 (core shroud). All of these subgroups are in contact with reactor water at temperatures 278-286°C (533-547°F) and subgroups 4.5 and 4.7 are not expected to reach more than a low and a moderate fluence, respectively, by the end of life. The panel evaluated the degradation mechanisms fatigue (NWC & HWC), reduction in fracture resistance, and stress corrosion cracking (NWC & HWC).

Panel members commented that the Type 308/308L weld metal is not as susceptible to thermal aging as cast stainless steel but some panel members indicated that there could be a reduction of fracture resistance in the long term. There is a lack of in-environment test data for this degradation mechanism. The average panel scores for susceptibility to reduction of fracture resistance for these subgroups were 1.13 and 2 for the level of knowledge, putting this subgroup in the light-yellow field.

Panel members noted that there has generally been good experience with respect to stress corrosion cracking of stainless steel weld metals. However, the possible effect of low temperature aging was highlighted by one member as an issue that should not be ignored, nor the synergistic effect with hydrogen under HWC conditions. The average panel scores for the susceptibility of these subgroups to stress corrosion cracking under NWC conditions were 1 and 1.13 and the scores for the level of knowledge were 2.63 and 2.88. For HWC conditions, the average panel scores for susceptibility were 1.13 for subgroup 3.8 and statistical mode 1 with one higher call for the other subgroups. For the level of knowledge under HWC conditions, the average panel scores were 2.5 and 2.75. These subgroups fall into the dark-yellow field for both NWC and HWC conditions except for subgroups 3.8 and 4.1 which fall into the dark-green field.

The panel had no basis available to judge whether fatigue loading would be significant or not in these subgroups and the original design basis was assumed to be correct. However, the panel did judge the expected relative impact of environmental effects of NWC and HWC on fatigue. For the core shroud subgroups, one panel member commented that the role of hydrogen in long-term fatigue is unclear. Under NWC conditions, the average panel scores for these subgroups were 1 for susceptibility to fatigue and 2.88 for the level of knowledge. Under HWC conditions, the panel scores for susceptibility were an average of 0.88 for subgroup 3.8 and statistical mode 1 with one higher call for the other subgroups, putting them conservatively in the yellow region for susceptibility. The average scores for the level of knowledge were 2.63. Therefore, these subgroups fall into the dark-green and dark-yellow fields.

3.3.1.3 Less-Susceptible Subgroups in the RCS

Subgroups in the reactor coolant system which fell into the dark yellow or green regions are shown in the modified rainbow charts, Figures 3.23 and 3.24 respectively. The subgroups have been sorted in the same manner as for the reds and light yellows. For more information on the evaluation and scoring for these subgroups, the reader is referred to Appendices D and E.

3.3.2 BWR Engineered Safety Features/Emergency Core Cooling System

As defined here, the BWR Engineered Safety Features/Emergency Core Cooling System (BWR-ESF/ECCS) consists of 8 groups (Groups 10, 11, 11A, 13-18) namely low-pressure core spray, high-pressure core spray/SP water, high-pressure core spray/CST water, RHR suction line piping to pumps, RHR pump discharge piping to heat exchanger, RHR normal shutdown cooling, RHR cooling water spray piping, and cycled condensate storage tank. The panel broke down these 8 groups into 66 component subgroups for assessment purposes. Nine of these 66 subgroups were scored red in the panel's assessment, 56 were scored dark yellow and 1 was scored green. No BWR-ESF/ECCS component subgroups were scored light yellow.

3.3.2.1 BWR ESF/ECCS Components with Red Susceptibility

The nine subgroups with components in the BWR ESF/ECCS falling into the red susceptibility regions are listed in Table 3.11 and illustrated in the modified rainbow chart, Figure 3.25.

Subgroup Description		Degradation Mechanism									
		CREV	DE-BOND	EC	FAT	FAT-HWC	FR	PIT	SCC	SCC-HWC	WEAR
Carbon and Low-Alloy Steel Components and Welds											
1.1	SA 508 Cl.2 Forging										
1.2	SA-533 Gr.B Rolled Plate					*					
1.5	SA508 Nozzle to SA 533 Plate Welds						*				
1.6	A533B Plate to Plate Welds										
1.9	A508 Cl.1 Nozzle Flanges										
1.11	A533 Gr. B LAS Dryer Hold Down Bracket										
2.1	SA 508 Cl.2 Forging					*					
2.2	SA-533 Gr.B Rolled Plate					*					
2.3	SA508 Nozzle/Flange to SA 533 Plate Welds					*					
2.4	SA508Nozzle to SA 533 Plate Welds					*					
2.5	SA 533 to SA 533 Plate Welds					*					
2.6	SA533 to SA 533 Plate Welds (high fluence)					*					
2.7	SA 533Gr. B Plate					*					
2.14	CS Safe End Extension										
2.16	A 508 MS Nozzle and Safe End			*							
2.20	CRD Return Nozzle Cap Weld				*	*	*				
3.1	SA 508 Cl.2 Forging					*				*	
3.2	SA-533 Gr.B Rolled Plate					*				*	
3.3	SA508 Nozzle to SA 533 Plate Welds					*				*	
9.2	LAS, SA-508 Cl.2 (Forged Ring)	*							X		
9.3	LAS, SA-508 Cl.2 HAZ	*							X		
9.4	Dissimilar metal welds LAS to SS 308										
9.14	Flange SA 182, GR F316 - Carbon Steel								X	*	
Type 308/309 Stainless Steel Welds and Cladding											
1.3	309 SS Cladding										
2.15	309 SS Cladding										
3.15	309 SS Cladding										
5.6	Type 308/L or 309 CRD Weld Metal									*	
5.15	308/L SS In-Vessel Weld Metal									*	
6.2	308/L SS Weld Metal					*				*	
6.9	308L Riser Brace Weld Metal					*				*	
6.14	308/L SS Diffuser Weld Metal					*			X	*	
7.1	Type 308/L SS FW Weld Metal					*				*	
7.4	Type 308/L FW Sparger Weld Metal					*				*	
7.7	Type 308/L SS Core Spray Weld Metal					*				*	
7.10	Type 308/L LPCI Weld Metal					*				*	
9.7	Welds SS 308					*				*	
High Strength Bolts, Studs, Springs, Etc.											
1.4	SA 540 CS Studs, Nuts and Washers										
6.4	XM-19 (Nitronic 50) Bracket Support					*			X		
9.18	Pump Casing Bolting SA 193 GR B7										
Type 304/316 Stainless Steel Components											
1.10	304 SS Dryer Hold Down Bracket										
4.11	Type 304 SS Ring Material							X			
4.12	Type 304 SS Shell Material							X	*		
4.13	Type 304 SS Shell Material (Beltline)							X			
4.16	Type 304 SS Top Guide Wedge							X			
4.18	Type 304 SS Core Plate Bolt					*		X			
4.19	Type 304 SS Flow Plug						*	X			
5.1	Wrought or Cast SS Fuel Support							X			
5.4	Type 304 SS Fuel Bundle Alignment Pin							X	*		
5.5	Type 304 SS CRD Housing Flange							X	*		
5.7	Type 304 SS CRD Housing HAZ								*		
5.11	Type 304 SS Penetrations							X			
5.14	Type 304 SS In-Vessel Structures							X	*		
6.1	304 SS Components							X	*		
6.8	304 SS Slip Fit										
7.3	Type 304 SS FW Sparger							X	*		
7.6	Type 304 SS Core Spray Components							X			
7.9	Type 304 SS LPCI Components										
8.4	ASTM A193, Gr. B8 Steam Dryer							X			
8.5	304 SS Guide Pin							X			
8.6	304 SS RCIC Nozzle							X			
9.5	Safe-end, SS 316NG					*		X	*		
9.8	Straight pipe SS Type 316					*		X	*		
9.10	Straight pipe SS Type 304					*		X	*		
9.11	Elbows SS Type 304					*		X			
9.12	Tee SS Type 304					*		X			
9.16	Reducing Branch SS 304					*		X	*		
9.17	Cap SS 304, 308/L Weld, and HAZ					*					
Cast Stainless Steel Components											
9.15	Cast SS CF-8 and CF-8M										

NOTE * Susceptibility at color interface with one or more scores higher than interface; X Susceptibility inside color box with one or more scores higher than this color box upper interface.

A blank in HWC column indicates same color (susceptibility and knowledge score) as adjacent column for same mechanism (NWC conditions).

Figure 3.23 Modified Rainbow Chart Showing Dark-Yellow Subgroups in BWR Reactor Coolant System

Subgroup Description		Degradation Mechanism			
		FAT	FR	PIT	SCC
Lifting and Stabilizer Lug Welds					
1.13	Lifting Lug Welds				
2.21	Stabilizer Lug Welds				
Carbon and Low-Alloy Steel Components of External Structures					
3.14	SA533B to A508 Forgings Support Skirt		X		
5.13	ASTM A36 & A235 CRD Outside Structure				
All SS Component External Surfaces					
9.1	All SS Components External Surfaces				

NOTE [x] Susceptibility inside color box with one or more scores higher than this color box upper interface.

Figure 3.24 Modified Rainbow Chart Showing Green Subgroups in BWR Reactor Coolant System

Table 3.11 Red Component Subgroups in the BWR ESF/ECCS

Component	Subgroups	Degradation Mechanisms Considered
Carbon steel and low-alloy steel components and welds	10.5, 11A.1, 11A.2, 11A.4, 11A.7, 14.8	Crevice corrosion, Fatigue (NWC) General corrosion, Microbiologically-induced corrosion, Pitting corrosion Stress corrosion cracking (NWC)
Type 304/316 stainless steel weld HAZs	13.3	Fatigue (NWC & HWC) Reduction of fracture resistance Stress corrosion cracking (NWC & HWC)
Carbon steel to brass joint (drywell)	16.10, 16.14	Crevice corrosion, Fatigue (NWC) Galvanic corrosion, General corrosion Microbiologically-induced corrosion Pitting corrosion, Stress corrosion cracking (NWC)

Subgroup Description		Degradation Mechanism									
		CREV	FAT	FAT-HWC	FR	GALV	GC	MIC	PIT	SCC	SCC-HWC
Carbon and Low-Alloy Steel Components and Welds											
10.5	C Steel (higher strength bolts)						X				
11A.1	SA105,106,216,234 - Carbon Steel	*					X	*	*		
11A.2	SA105,106,216,234 - C Steel Weld	*					X	*	*	X	
11A.4	SA105,106,216,234 - Carbon Steel	*									
11A.7	A106,A516 C & LA Steels	*					X	*	*		
14.8	Carbon Steel HX Fittings								*		
Type 304 Stainless Steel Weld HAZs											
13.3	304 SS HAZ										
Carbon Steel - Brass Joint (Drywell)											
16.10	Carbon Steel - Brass joint (Drywell)										
16.14	Carbon Steel - Brass joint (Drywell)										

NOTES: * Susceptibility at color interface with one or more scores higher than interface; ^x Susceptibility inside color box with one or more scores higher than this color box upper interface.
A blank in HWC column indicates same color (susceptibility and knowledge score) as adjacent column for same mechanism (NWC conditions).

Figure 3.25 Modified Rainbow Chart Showing Red Subgroups in BWR Engineered Safety Features/Emergency Core Cooling System

3.3.2.1.1 Carbon and Low-Alloy Steel Components and Welds

The carbon and low-alloy steel components and welds which fall into the highest susceptibility category are found in subgroups 10.5 (low pressure core spray) and 14.8 (RHR pump discharge piping to RHR HX) and, for plants other than the standard plant, in subgroups 11A.1, 11A.2, 11A.4 and 11A.7 (HPCS – CST water). All the subgroups operate at temperatures below 38°C (100°F). For subgroups 10.5 and 14.8, the environment is suppression pool water and for the subgroups 11A.1, 11A.2, 11A.4 and 11A.7 the environment is condensate cooling water. The panel evaluated the degradation mechanisms of crevice corrosion, fatigue, general corrosion, microbiologically-induced corrosion, pitting corrosion, and stress corrosion cracking. HWC is not applicable to these subgroups.

Panel members commented that crevices would be formed under the deposits in these subgroups and that, depending on the water quality, all forms of corrosion could be expected. The average panel scores for the susceptibility of subgroups 10.5 and 14.8 to crevice corrosion were 2.13 and 1.5 respectively, and the average score for the level of knowledge was 2.88 for both subgroups. The panel's scores for the susceptibility of the subgroups 11A.1, 11A.2, 11A.4 and 11A.7 were statistical mode 2 with 1 higher call conservatively coloring them red, and the average score for the level of knowledge was 2.71 for these subgroups. All of the subgroups are in the dark-red field except for subgroup 14.8 that falls in the dark-yellow field.

Panel members commented that pitting corrosion will occur under the deposits in these subgroups if, as assumed in this case, the water quality is poor. One panel member pointed out that there is a correlation between pitting corrosion and microbiologically-induced corrosion and another that the pitting corrosion would be more severe close to the air-saturated condensate storage tank. The panel scores for the susceptibility of subgroups 10.5 and 14.8 to pitting corrosion were respectively an average of 1.88 coloring it yellow, and statistical mode 2 with 1 higher call conservatively coloring it red. For the level of knowledge, the average scores were 2.88 and 2.63 for the two subgroups respectively. The panel scores for the susceptibility of the subgroups 11A.1, 11A.2, 11A.4 and 11A.7 were an average of 1.86 for subgroup 11A.4 putting it in the yellow category, and statistical mode 2 with 1 higher call for the other subgroups putting them conservatively in the red category. For the level of knowledge the average score was 2.71 for all these subgroups. Subgroups 14.8, 11A.1, 11A.2 and 11A.7 are in the dark-red field and subgroups 10.5 and 11A.4 in the dark-yellow field.

Panel members commented that poor water quality in the suppression pool and regular flushing of the systems would make them susceptible to microbiologically-induced corrosion. This degradation mechanism is discussed in detail in Appendix B.16. Subgroups 11A.1, 11A.2, 11A.4 and 11A.7 would not be as susceptible since there are expected to be fewer nutrients in the condensate storage water. The average panel scores for the susceptibility of subgroups 10.5 and 14.8 to microbiologically-induced corrosion were 2 and 1.75 respectively, and the scores for the level of knowledge were 2.88 and 2.75. The panel scores for the susceptibility of the subgroups 11A.1, 11A.2, and 11A.7 were statistical mode 2 with 1 higher call, and an average of 1.71 for subgroup 11A.4; and the score for the level of knowledge was 2.71. Subgroups 11A.1, 11A.2 and 11A.7 are conservatively in the dark-red field and subgroups 14.8, 11A.4, and 10.5 are in the dark-yellow field.

Panel members commented that potentially poor water quality in subgroups 10.5 and 14.8 and the water with dissolved oxygen and carbon dioxide in subgroups 11A.1, 11A.2, 11A.4 and 11A.7 will result in general corrosion of the carbon steel. One panel member pointed out that the corrosion will probably be higher close to the condensate storage tank ends of the lines in subgroups 11A.1, 11A.2, 11A.4 and 11A.7. The panel scores for the susceptibility of subgroups 10.5 and 14.8 to general corrosion were statistical mode 1 with 1 higher call conservatively coloring it yellow, and an average of 1.63 and, for the level of knowledge, the average scores were 2.88 and 2.75 respectively. The panel scores for the susceptibility of the subgroups 11A.1, 11A.2, 11A.4 and 11A.7 were an average of 1.43 to 1.57 and the score for the level of knowledge was 2.71. All the subgroups are therefore, in the dark-yellow field.

Assuming poor water quality in the suppression pool, panel members noted that carbon steel in subgroups 10.5 and 14.8 would not be very susceptible to stress corrosion cracking. It could be possible, but it would be a slow process. Stress corrosion cracking of carbon steel can also occur at these low temperatures in oxygenated water such as the condensate storage water, in particular at welds and in heavily cold worked regions. This degradation mechanism is discussed in more detail in Appendix B.8. The average panel scores for the susceptibility of subgroups 10.5 and 14.8 to stress corrosion cracking were 1.25 and the scores for the level of knowledge were 2.75. The panel scores for the susceptibility of the subgroups 11A.1, 11A.2, 11A.4 and 11A.7 were an average of 1.29 to 1.43 and the scores for the level of knowledge were 2.71. All the subgroups are, therefore, in the dark-yellow field.

Because the ECCS is not operated very often, the panel considered that fatigue loading would be minimal. The average panel scores for the susceptibility of subgroups 10.5 and 14.8 to fatigue were 1.13 and 1 and, for the level of knowledge, were 3 and 2.88 respectively. The panel scores for the susceptibility of the subgroups 11A.1, 11A.2, 11A.4 and 11A.7 were an average of 1 and, for the level of knowledge an average of 3. All the subgroups are in the dark-green field except for subgroup 10.5 which is in the dark–yellow field.

3.3.2.1.2 Type 304/316 Stainless Steel Weld Heat-Affected Zones

Type 304/316 stainless steel weld HAZ which fall into the highest susceptibility category are found in the subgroup 13.3 (RHR suction line piping to RHR pumps). This subgroup operates at 287°C (549°F) and is in contact with reactor water that is normally stagnant. The degradation mechanisms evaluated by the panel were fatigue (NWC & HWC), reduction of fracture resistance and stress corrosion cracking (NWC & HWC).

Under NWC conditions, the panel considered that the stainless steel HAZ are susceptible to stress corrosion cracking depending on the degree of sensitization, cold work and water quality. It was also pointed out that weld strain hardening is important but that dynamic loading conditions are probably not present under stagnant conditions. The panel commented that it is difficult to maintain HWC under stagnant conditions so its effectiveness as a mitigation method could be limited in this subgroup. The average panel score for susceptibility to stress corrosion cracking for this subgroup was 2.13 and was 3 for the level of knowledge under NWC conditions. Under HWC conditions (assuming HWC can be maintained) the panel score for susceptibility was an average of 1.25 and was an average of 2.88 for the level of knowledge. This subgroup falls into the dark-red and dark-yellow fields for NWC and HWC respectively.

Some panel members pointed out that there are insufficient in-environment data for the reduction of fracture resistance and that there could be such an effect. This reduction would be of greater interest for heavily cold worked areas. The average panel score for susceptibility to reduction of fracture resistance was 1.38 and, for the level of knowledge, the average score was 2.5. This subgroup is, therefore, in the dark-yellow field.

The panel assumed that the fatigue loading in this subgroup was probably minimal and well within design limits because of the normally stagnant operating conditions. The panel did not expect that there would be a significant difference under HWC conditions but thought that there might be an environmental effect on fatigue. Under NWC conditions, the average panel score for susceptibility to fatigue was 1 for this subgroup and the score for the level of knowledge was 2.88. Under HWC conditions, the panel score for susceptibility was an average of 1.13 and was an average of 2.63 for the level of knowledge. This subgroup falls into the dark-green and dark-yellow fields for NWC and HWC respectively.

3.3.2.1.3 Carbon Steel to Brass Joint (Drywell)

The carbon steel to brass joints which fall into the highest susceptibility category are found in the subgroups 16.10 and 16.14 (RHR spray piping). When the plant is operating, these are filled with nitrogen but, when the piping system is in operating mode (i.e., spraying), the piping is filled with suppression pool water at 38°C (100°F). Evidently, operation of this system is an extremely rare event that has probably never happened. Nevertheless, the degradation mechanisms evaluated by the panel were crevice corrosion, fatigue, galvanic corrosion, general corrosion, microbiologically-induced corrosion, pitting corrosion and stress corrosion cracking assuming the presence of the suppression pool water. HWC is not relevant in these subgroups.

The panel commented that galvanic corrosion was almost unavoidable with this material combination given poor quality water from the suppression pool. The panel scores for the susceptibility of subgroups 16.10 and 16.14 to galvanic corrosion were an average of 2.5 and, for the level of knowledge were an average of 2.88. Therefore, these subgroups are in the dark-red field.

The panel commented that crevice corrosion was possible in these subgroups if and when the suppression pool water is present. It can occur under deposits or in geometric crevices in conjunction with galvanic attack. The panel scores for the susceptibility of subgroups 16.10 and 16.14 to crevice corrosion were an average of 2.13 and 1.63 respectively and, for the level of knowledge, were an average of 2.88. These subgroups are in the dark-red (16.10) and dark-yellow (16.14) fields.

The joints in these subgroups are sockolets which have often been subject to fatigue damage in other systems. However, the mostly stagnant conditions will reduce their susceptibility. The panel scores for the susceptibility of subgroups 16.10 and 16.14 to fatigue were an average of 1.5 and 1.38 respectively and the scores for the level of knowledge were an average of 2.63. These subgroups are, therefore, in the dark-yellow field.

The panel commented that, since it is not clear that the header is ever completely dry, there will be susceptibility to general corrosion with poor quality water from the suppression pool. The panel scores for the susceptibility of subgroups 16.10 and 16.14 to general corrosion were an average of 1.75 and 1.63 respectively and, for the level of knowledge, an average of 2.75. These subgroups are, therefore, in the dark-yellow field.

Because the header is usually wet, there will be the possibility of microbiologically-induced corrosion with poor quality water from the suppression pool. The panel scores for the susceptibility of subgroups 16.10 and 16.14 to microbiologically-induced corrosion were an average of 1.88 and, for the level of knowledge, were 2.75. These subgroups are, therefore, in the dark-yellow field.

The panel commented that residual poor quality suppression pool water makes pitting corrosion a potential degradation mechanism. The panel scores for the susceptibility of subgroups 16.10 and 16.14 to pitting corrosion were an average of 1.88 and, for the level of knowledge, were 2.75. These subgroups are, therefore, in the dark-yellow field.

The panel commented that the oxidizing conditions and galvanic coupling could lead to stress corrosion cracking which is possible at these low temperatures in poor quality water. The panel scores for the susceptibility of subgroups 16.10 and 16.14 to stress corrosion cracking were an average of 1.5 and 1.25 respectively and, for the level of knowledge, were an average of 2.88. These subgroups are, therefore, in the dark-yellow field.

3.3.2.2 ESF/ECCS Less-Susceptible Component Subgroups

Subgroups in the BWR Engineered Safety Features/Emergency Core Cooling System which fall into the dark yellow or green regions are shown in the modified rainbow charts, Figures 3.26 and 3.27, respectively. The subgroups have been sorted in the same manner as for the red component subgroups. For more information on the evaluation and scoring for these subgroups, the reader is referred to Appendices D and E.

3.3.3 BWR Steam and Power Conversion System

As defined here, the BWR steam and power conversion system (BWR-S&PCS) consists of six groups (Groups 17, 19, 21-24) namely main steam, feedwater, main condenser, main condenser discharge piping, condensate piping to booster pump, and condensate piping to feedwater pump. The panel broke down these 6 groups into 32 component subgroups for assessment purposes. Two of these 32 subgroups were scored red in the panel's assessment, two were scored light yellow and 28 were scored dark yellow. No BWR-S&PCS component subgroups were scored green.

3.3.3.1 Steam and Power Conversion System Components with Red Susceptibility

The two steam and power conversion system subgroups with components falling into the red susceptibility regions are listed in Table 3.12 and illustrated in the modified rainbow chart in Figure 3.28.

3.3.3.1.1 Low-Alloy Steel Bolts for Tee Quencher

The low alloy steel bolts for the tee quencher/sparger that fall into the highest susceptibility category are found in subgroup 17.5 (main steam) which operates at below 32°C (90°F). This subgroup is immersed in the suppression pool water possibly coated with a zinc primer. The degradation mechanisms evaluated by the panel were crevice corrosion, fatigue, pitting corrosion and stress corrosion cracking.

There is some field experience of failure due to stress corrosion cracking (usually regarded as hydrogen embrittlement) but it requires excessive hardness. Stress corrosion cracking of low alloy steel is slow in this temperature range. One panel member pointed out that zinc paint would be an aggravating factor for hydrogen embrittlement. The average panel score for susceptibility to stress corrosion cracking was 2.75 and for the level of knowledge, 2. This subgroup is therefore in the light-red field.

The panel noted that pitting corrosion has been reported, but it is not thought to be a generic issue. Poor water quality will be an aggravating factor and pitting corrosion will increase the susceptibility to stress corrosion cracking. The average panel score for susceptibility to pitting corrosion was 2.13 and for the level of knowledge 2.88. This subgroup is in the dark-red field.

The panel noted that crevices will exist at bolted joints and crevice corrosion can be expected because of poor quality water. Crevice corrosion can also contribute to the stresses due to volume expansion of the corrosion products. The average panel score for susceptibility to crevice corrosion was 1.38 and, for the level of knowledge 2.88. This subgroup is, therefore, in the dark-yellow field.

Subgroup Description		Degradation Mechanism								
		CREV	FAT	FAT-HWC	FR	GC	MIC	PIT	SCC	SCC-HWC
Carbon and Low-Alloy Steel Components and Welds										
10.1	SA105,106,216,234 - Carbon Steel									
10.2	SA105,106,216,234 - Carbon Steel									
10.3	SA105,106,234 - Carbon Steel & Weld									
10.4	SA105,106,216,234 - Carbon Steel									
10.7	SA106,234 - Carbon Steel									
10.8	SA106,234 - Carbon Steel									
10.9	SA105,106,216,234 - Carbon Steel									
10.10	A106,A516 Carbon & Low Alloy Steels									
11.1	SA105,106,216,234 - Carbon Steel									
11.2	SA105,106,216,234 - CS Weld Metal									
11.3	SA105,106,234 - CS Base & Weld									
11.4	SA105,106,216,234 - Carbon Steel									
11.5	SA106,234 - Carbon Steel									
11.6	SA106,234 - Carbon Steel									
11.7	A106,A516 Carbon & Low Alloy Steels									
11A.3	SA105,106,234 - CS Base & Weld									
11A.5	SA106,234 - Carbon Steel									
11A.6	SA106,234 - Carbon Steel									
13.8	Carbon Steel Welds & HAZ									
13.9	SA234 Gr. WPB Components									
14.2	Various Carbon Steel Components									
14.3	Carbon Steel Weld and HAZ									
14.4	SA234 Gr. WPB Weldolet/Sockolet									
14.6	CS w/outside clad of SS									
14.7	Carbon Steel HX Components									
15.2	Various Carbon Steel Components									
15.3	Carbon Steel Welds and HAZ									
15.4	CS - Base Metal, Welds and HAZ									
15.5	CS SA234 Gr. WPB Weldolet/ Sockolet									
16.2	Various Carbon Steels - Base and Weld									
16.3	Carbon Steel Weld HAZ									
16.4	CS SA234 Gr. WPB Weldolet/ Sockolet									
16.5	Various Carbon Steel Components									
16.6	Various Carbon Steel HAZ & Welds									
16.7	Various Carbon Steel Spray Head (Cont.)									
16.8	Carbon Steel SA234 Gr. WPB									
16.9	CS SA234 Gr. WPB Sockolet (Cont.)									
16.12	Various Carbon Steel Spray Head (SP)									
16.13	CS SA234 Gr. WPB Sockolet (SP)									
Type 304/316 Stainless Steel Components, Welds and HAZs										
10.6	Type 304 SS Strainer									
11A.8	Type 304 SS Base Metal									
11A.9	Type 304 SS Weld Metal & HAZ									
13.2	304 SS - Weld and Base Metal									
13.4	304 SS Components									
13.5	Welds SS 308									
13.6	Socket Welds									
14.5	HX Weld in SS, or CS with 1/8" Clad/overlay									
18.2	Stainless Steel Tank (OTHER PLANT)	*								
All Stainless Steel Component External Surfaces										
13.1	All SS Components External Surfaces									
All Carbon and Low-Alloy Steel Component External Surfaces										
13.7	All CS Components External Surfaces									
14.1	All CS Components External Surfaces									
14.9	All CS HX Components External Surfaces									
15.1	All CS Components External Surfaces									
16.1	All CS Components External Surfaces									
Brass Spray Nozzles (Drywell)										
16.11	Brass Spray Nozzle (Drywell)									
16.15	Nozzle Drywell									

NOTES: * Susceptibility at color interface with one or more scores higher than interface; A blank in HWC column indicates same color (susceptibility and knowledge score) as adjacent column for same mechanism (NWC conditions).

Figure 3.26 Modified Rainbow Chart Showing Dark-Yellow Subgroups in BWR Engineered Safety Features/Emergency Core Cooling System

Subgroup Description		Degradation Mechanism			
		CREV	FAT	PIT	SCC
Aluminum Alloys in Cycled Condensate Storage Tank					
18.1	6061-T6 & other Al alloys				

Figure 3.27 Modified Rainbow Chart Showing Green Subgroup in BWR Engineered Safety Features/Emergency Core Cooling System

Table 3.12 Red Components in the BWR Steam and Power Conversion System

Component	Subgroups	Degradation Mechanisms Considered
Low-alloy steel bolts for tee-quencher	17.5	Crevice corrosion Fatigue Pitting Stress corrosion cracking (NWC)
Titanium condenser tubes – outside surface	21.6	Erosion-corrosion

Subgroup Description		Degradation Mechanism				
		CREV	EC	FAT	PIT	SCC
T-Quencher						
17.5	A540 B21 (Hatch 2) T-Quencher	X				
Ti Condenser Tubes -- Outside Surface						
21.6	Titanium tubes, outside of tube		*			

NOTE: * Susceptibility at color interface with one or more scores higher than interface; x Susceptibility inside color box with one or more scores higher than this color box upper interface.

Figure 3.28 Modified Rainbow Chart Showing Red Subgroups in BWR Steam and Power Conversion System

The panel noted that the loading conditions are poorly defined and fatigue could be a credible degradation mode. One panel member commented that corrosion fatigue of high strength bolts should be aggressive. The average panel score for susceptibility to fatigue was 1.13 and for the level of knowledge 2.13. This subgroup is in the dark-yellow field.

3.3.3.1.2 Titanium Condenser Tubes – Outside Surface

The outside surface of titanium condenser tubes are found in the subgroup 21.6 (main condenser) and fall into the highest susceptibility category. This subgroup is exposed to wet steam and condensate. The panel only evaluated erosion corrosion as a degradation mechanism for this subgroup.

The panel noted that droplet erosion is a known issue that is design dependent. If the titanium is exposed directly to incoming steam with droplets, perforation can occur depending on the droplet velocity. This can be managed by using stainless steel tubes in steam inlet areas of the condenser and other design changes to avoid droplet impingement at high velocity. The panel's score for susceptibility to erosion corrosion was statistical mode 2 with one higher call and the average score for the level of knowledge was 2.88. This subgroup is conservatively in the dark-red field.

3.3.3.2 Steam and Power Conversion System Components with Light-Yellow Susceptibility

The two SPCS subgroups with components falling into the light yellow susceptibility region are listed in Table 3.13 and illustrated in the modified rainbow chart, Figure 3.29.

Table 3.13 Light-Yellow Components in the BWR Steam and Power Conversion System

Component	Subgroups	Degradation Mechanisms Considered
Carbon steel socket welds	17.4	Fatigue Stress corrosion cracking (NWC)
Cast/wrought stainless steel venturi	17.7	Erosion-corrosion Fatigue (NWC) Reduction of fracture resistance

Subgroup Description		Degradation Mechanism			
		EC	FAT	FR	SCC
Carbon Steel Socket Welds					
17.4	A234, A106, A105 Weldolet				
Cast/Wrought Stainless Steel Venturi					
17.7	CASS, A351 Type 304, Venturi				

Figure 3.29 Modified Rainbow Chart Showing Light-Yellow Subgroups in BWR Steam and Power Conversion System

3.3.3.2.1 Carbon Steel Socket Welds

The carbon steel socket welds are found in subgroup 17.4 (main steam) and fall into the medium susceptibility category (yellow region) as shown in Figure 3.29 above. The operating conditions for this subgroup are saturated steam at 286°C (547°F). The degradation mechanisms evaluated by the panel were fatigue and stress corrosion cracking.

The panel noted that sockolets often have fatigue issues since they are attached at one end to a robust sturdy member and at the other to a light flexible pipe; however, the fatigue loading conditions are unclear for this subgroup. The panel commented that flow induced vibration is always a potential problem and the effect of ripple stress on stress corrosion cracking is important. Degradation is most likely in small-diameter weldolets and branch lines. The average panel score for susceptibility to fatigue was 2 and the level of knowledge score was 2. This subgroup falls into the light-yellow field.

The panel pointed out that there is a potential for stress corrosion cracking where oxygenated condensate can accumulate and the residual stresses are high. There is also a possibility for corrosion fatigue because of potential cyclic loading if the cyclic frequency is sufficiently low. The average panel score for susceptibility to stress corrosion cracking was 1.88 and the score for the level of knowledge was 2.88. This subgroup falls into the dark-yellow field.

3.3.3.2.2 *Cast/Wrought Stainless Steel Venturi*

The cast/wrought stainless steel venturi which is color-coded light-yellow is in subgroup 17.7 (main steam). The operating conditions are high flow rate saturated steam at 286°C (547°F). The degradation mechanisms evaluated by the panel were erosion corrosion, fatigue and reduction of fracture resistance.

The panel commented that fatigue is an obvious potential degradation mechanism but the fatigue loading was poorly defined. The panel's judgments assumed cyclic loading to be within the original design limits while noting the possibility of flow induced vibration. The average panel score for susceptibility to fatigue was 1.13 and was 2 for the level of knowledge. This subgroup falls into the light-yellow field.

The panel commented that a reduction of fracture resistance due to thermal aging is to be expected at this operating temperature and its magnitude will depend on heat-to-heat variability and operating time. Panel members pointed out that there are insufficient in-environment data for this degradation mode. The average panel score for susceptibility to reduction of fracture resistance was 1.88 and the score for the level of knowledge was 2.75. This subgroup falls into the dark-yellow field.

The panel noted that austenitic alloys are not normally susceptible to erosion corrosion but that this depends on the steam quality and velocity. Droplet abrasion (i.e., erosion) is also possible. The average panel score for susceptibility to erosion corrosion was 1 and the score for the level of knowledge was 2.75. This subgroup falls into the dark-green field.

3.3.3.3 <u>Less-Susceptible Component Subgroups in the BWR SPCS</u>

Subgroups which fell into the dark yellow region are shown in the modified rainbow chart, Figure 3.30. The subgroups have been sorted in the same manner as for the other major systems. More information on the evaluation and scoring for these subgroups can be found in Appendices D and E.

Subgroup Description		Degradation Mechanism							
		CREV	EC	FAC	FAT	GALV	GC	PIT	SCC
Carbon and Low-Alloy Steel Components -- External Surfaces									
17.1	All CS & LAS Components External Surfaces								
Carbon and Low-Alloy Steel Components									
17.2	A106B, A234, A105, A216, A672B70 MS								
17.3	A234, A106, A105 MS Components				*				
19.1	SA105,A106,SA216,A234,A672 Components								
19.2	Carbon Steel - Base, Weld and HAZ								
19.3	Carbon Steel - Weldolet								
21.1	Carbon Steel - Base, Weld and HAZ								
21.2	Carbon Steel - Base, Weld and HAZ								
22.1	SA105,A106,SA106,A234,A672,SA216			*					
22.2	A234 Carbon Steel - Base, Weld and HAZ			*					
22.3	A105,A216 Carbon Steel - Base, Weld and HA			*					
22.4	Pump Carbon Steel - Base, Weld and HAZ								
22.5	Ejector Carbon Steel - Base, Weld and HAZ								
23.1	SA105,A106,SA216,A234,A672 Components								
23.2	SA106,SA216,A234,A672 Sockolet								
23.3	SA105,SA216,A672 Valves								
24.1	SA105,A106,SA216,A234,A672 (low T)								
24.2	SA105,A106,SA216,A234,A672 (high T)								
24.3	SA106,SA216,A234,A672 Sockolet								
24.4	SA105,SA216,A672 Valves								
24.5	Pump Carbon Steel - Base, Weld and HAZ								
Stainless Steel Condenser Tubes									
21.3	Stainless Steel, outside of tube								
21.4	Stainless Steel, inside of tube								
Titanium Condenser Tubes -- Inside Surfaces									
21.5	Titanium tubes, inside of tube								
Miscellaneous Stainless Steel Components									
17.6	Austenitic SS Bimetallic Joint								*
19.4	304 SS Heater Pipes & Flow Elements								
22.6	Stainless Steel Flow Restrictor								
24.6	304 SS Heater Tubes								

NOTE: * Susceptibility at color interface with one or more scores higher than interface.

Figure 3.30 Modified rainbow chart showing the dark yellow subgroups in the BWR Steam and Power Conversion System

3.3.4 BWR Support and Auxiliary System Components

3.3.4.1 <u>BWR Support and Auxiliary System Components with Red Susceptibility</u>

The four BWR support and auxiliary system subgroups with components falling in the red susceptibility regions are listed in Table 3.14 and illustrated in the modified rainbow chart, Figure 3.31.

Table 3.14 Red Components in the BWR Support and Auxiliary System

Component	Subgroups	Degradation Mechanisms Considered
Carbon steel components	12.3, 12.15	Crevice corrosion Fatigue (NWC) General corrosion Microbially-induced corrosion Pitting corrosion Stress corrosion cracking (NWC)
Type 304/316 stainless steel weld heat-affected zones	25.5, 25.7	Fatigue (NWC & HWC) Reduction of fracture resistance Stress corrosion cracking (NWC & HWC)

Subgroup Description		Degradation Mechanism								
		CREV	FAT	FAT-HWC	FR	GC	MIC	PIT	SCC	SCC-HWC
Carbon Steel Components										
12.3	SA105 - Carbon Steel Base & Weld		*							
12.15	SA216 - Carbon Steel	*								
Type 304 Stainless Steel Weld HAZs										
25.5	304 Stainless Steel - HAZ									
25.7	304 Stainless Steel - Base, Weld & HAZ									

NOTE: * Susceptibility at color interface with one or more scores higher than interface; a blank in HWC column indicates same color (susceptibility and knowledge score) as adjacent column for same mechanism (NWC conditions).

Figure 3.31 Modified Rainbow Chart Showing Red Subgroups in BWR Support and Auxiliary System

3.3.4.1.1 Carbon Steel Components

The carbon steel components which fall into the highest susceptibility category are found in the subgroups 12.3 and 12.15 (reactor core isolation cooling). Subgroup 12.3 consists of weldolets and sockolets and has an environment of wet steam at 286°C (547°F) while subgroup 12.15 consists of valves and is in contact with suppression pool water at less than 38°C (100°F) and at atmospheric pressure. The degradation mechanisms evaluated by the panel were fatigue, general corrosion, pitting corrosion and stress corrosion cracking and, for subgroup 12.15, crevice corrosion and microbiologically-induced corrosion also were assessed.

The panel pointed out that there are generic fatigue issues with socket welds such as those in subgroup 12.3 and commented that, depending on the design, these welds could also be subject to thermal fatigue due to eddies as well as flow induced vibrations in the deadlegs. Long pipe runs may also increase the bending stresses and the possibility of cyclic stresses. The panel did not expect there to be any significant fatigue loading in the predominantly stagnant and infrequently used subgroup 12.15. The panel scores for the susceptibilities of subgroups 12.3 and 12.15 to fatigue were statistical mode 2 with one higher call, and an average of 1. For the level of knowledge the panel scores were 2.25 and 3 for subgroups 12.3 and 12.15 respectively. Therefore, subgroup 12.3 is conservatively in the dark-red field and subgroup 12.15 in the dark-green field.

The panel commented that pitting corrosion is a potential degradation mechanism in the oxidizing steam environment in subgroup 12.3, although there are no known service problems. In subgroup 12.15 the panel considered that pitting was very probable due to potentially poor water quality and the possibility of deposits in the pipes. The average panel scores for the susceptibility of subgroups 12.3 and 12.15 to pitting corrosion were 1.25 and 2 respectively and the scores for the level of knowledge were both 2.13. Both subgroups, thus fall in the dark-yellow field.

The panel noted that the oxidizing steam environment will result in general corrosion in subgroup 12.3. The poor water quality in subgroup 12.15 will also result in general corrosion, depending upon the oxygen content, and thus may decrease at locations further from the suppression pool due to a decrease in the oxygen content. The average panel scores for the susceptibility of subgroups 12.3 and 12.15 to general corrosion were 1.13 and 1.88 respectively and, for the level of knowledge the average score was 2.88. Both the subgroups fall in the dark-yellow field.

The panel commented that stress corrosion cracking in subgroup 12.3 was possible in the oxygenated condensate and would depend upon cold work and the sulfur content of the steel, but that it was unlikely to be sustained because of the lack of sufficient stress. For subgroup 12.15, the panel noted that the lower service temperature would reduce the susceptibility. The susceptibility will depend on the source of dynamic loading, such as vibrations, and the quality of the water. The average panel scores for the susceptibility of subgroups 12.3 and 12.15 to stress corrosion cracking were 1.5 and 1.13 and, for the level of knowledge, the average scores were 2.88 and 2.75 respectively. Both the subgroups fall in the dark-yellow field.

The panel noted that the valves in subgroup 12.15 are sufficiently close to the suppression pool for there to be oxygen in water of potentially poor quality. Crevice corrosion will depend upon the presence of geometrical crevices and deposits which form crevices. The panel score for the susceptibility of subgroup 12.15 to crevice corrosion was statistical mode 2 with one higher call conservatively coloring it red, and the score for the level of knowledge was an average of 2.88. This subgroup is thus in the dark-red field.

The panel considered that there was a definite possibility of microbiologically-induced corrosion in subgroup 12.15 due to poor quality water and the regular flushing of the system replenishing nutrients. The panel was not aware of any incidences but noted that they would have increased their susceptibility score if this had been the case. The average panel score for the susceptibility of subgroup 12.15 to microbiologically-induced corrosion was 1.88 and the score for the level of knowledge was 2.88. This subgroup is thus in the dark-yellow field.

3.3.4.1.2 Type 304/316 Stainless Steel Weld Heat-Affected Zones

The Type 304/316 stainless steel weld heat-affected zones which fall into the highest susceptibility category are found in the subgroups 25.5 and 25.7 (reactor water cleanup piping to pumps). Subgroup 25.5 includes a variety of piping component HAZs and subgroup 25.7 consists of weldolets and sockolets. The operating environments for these subgroups are reactor water at 279°C (535°F). The degradation mechanisms evaluated by the panel were fatigue (NWC and HWC), reduction of fracture resistance and stress corrosion cracking (NWC and HWC).

The panel noted that stress corrosion cracking in Type 304 stainless steel weld HAZs is a well-known problem which is discussed in more detail in Appendix B.1. Carbon content, stress, metallurgical condition, water quality and oxidation potential are all contributing factors. Panel members commented that there should be a mitigating effect of HWC in these subgroups although the long-term effects are not yet clear. The average panel scores for susceptibility to stress corrosion cracking under NWC for subgroups 25.5 and 25.7 were 2.25 and 2.38 and the scores for the level of knowledge were 2.75 and 2.88 respectively. Under HWC conditions, the panel scores for susceptibility were an average of 1.25, and were an average of 2.75 for the level of knowledge. These subgroups, therefore, fall into the dark-red (NWC) and dark-yellow (HWC) fields.

For subgroup 25.5, most of the panel members assumed that the fatigue loading was minor. For subgroup 25.7, the panel noted that the fatigue of sockolets and weldolets is a generic problem and the fatigue loading, although uncertain, could be high. The panel did not expect any significant effect of HWC for these subgroups. Again, one panel member pointed out that the long-term effects of hydrogen are unclear. The average panel scores for susceptibility to fatigue under NWC conditions for subgroups 25.5 and 25.7 were 1.13 and 2 and the scores for the level of knowledge were 2.88 and 2.63 respectively. Under HWC conditions, the panel scores for susceptibility were an average of 1.5 and an average of 1.75, and the scores for the level of knowledge were both 2.63. These subgroups, therefore, fall into the dark-yellow field.

The panel pointed out that there is a lack of data for the reduction of fracture resistance in the environment but that this degradation mode could be an issue. The average panel scores for the susceptibility of subgroups 25.5 and 25.7 to reduction of fracture resistance were 1.25 and 1.13 respectively and the scores for the level of knowledge were both 2.25. Both the subgroups, therefore, fall in the dark-yellow field.

3.3.4.2 Support and Auxiliary System Components with Light-Yellow Susceptibility

The two BWR support and auxiliary system subgroups with components falling into the light-yellow susceptibility region are listed in Table 3.15 and illustrated in the modified rainbow chart, Figure 3.32.

Table 3.15 Light-Yellow Components in the BWR Support and Auxiliary System

Component	Subgroups	Degradation Mechanisms Considered
Carbon steel components	12.1, 12.9	Fatigue (NWC) General corrosion Microbially-induced corrosion Pitting Stress corrosion cracking (NWC)

Subgroup Description		Degradation Mechanism				
		FAT	GC	MIC	PIT	SCC
Carbon Steel Components						
12.1	SA105,106,234 - Carbon Steel					
12.9	SA105,106,234 - CS Base & Weld					

Figure 3.32 Modified Rainbow Chart Showing Light-Yellow Subgroups in BWR Support and Auxiliary System

3.3.4.2.1 Carbon Steel Components

The carbon steel components which are colored light-yellow are in subgroups 12.1 and 12.9 (reactor core isolation cooling). The operating conditions for subgroup 12.1 are stagnant wet steam at 286°C (547°F) and for subgroup 12.9, stagnant condensate storage water typically at below 38°C (100°F). The degradation mechanisms evaluated by the panel were fatigue, general corrosion, pitting corrosion and stress corrosion cracking and, for subgroup 12.9, also microbiologically-induced corrosion.

For subgroup 12.1 the panel noted that thermal fatigue was possible due to eddies or flow induced vibration in deadlegs. For subgroup 12.9, which is stagnant most of the time, the panel assumed that there were no significant fatigue loading conditions. The average panel scores for susceptibility to fatigue for subgroups 12.1 and 12.9 were 1.75 and 1.5 respectively and, for the level of knowledge, the average score was 2. These subgroups, therefore, fall into the light-yellow field.

The panel noted that the wet steam condensate is an oxidizing environment and thus significant general corrosion could occur in subgroup 12.1. For subgroup 12.9, the panel commented that the initially high general corrosion rates would decrease with time due to decreasing oxygen content. If the water quality was poor, then general corrosion would be expected to continue. The average panel scores for susceptibility to general corrosion for subgroups 12.1 and 12.9 were 1.13 and 1.75 respectively and, for the level of knowledge, the average scores were both 2.88. These subgroups, therefore, fall into the dark-yellow field.

The panel noted that, given the oxidizing environment, pitting corrosion could occur in subgroup 12.1. For subgroup 12.9, the panel commented that pitting corrosion was possible initially but would be likely to decrease with time due to decreasing oxygen content. If the water quality was also poor, then pitting corrosion would be expected to continue. The average panel scores for susceptibility to pitting corrosion for subgroups 12.1 and 12.9 were 1.25 and 1.13 respectively and, for the level of knowledge, the average scores were 2.13. These subgroups, therefore, fall into the dark-yellow field.

The panel commented that stress corrosion cracking could be possible in these subgroups at welds and other high stress regions if the oxygen content in the condensate was sufficiently high. For subgroup 12.9, the panel commented that the likelihood of stress corrosion cracking will depend on possible sources of dynamic (slow strain rate) loading and possibly also ripple loading due to vibration. The average panel scores for susceptibility to stress corrosion cracking for subgroups 12.1 and 12.9 were 1.13 and, for the level of knowledge, the average scores were 2.88. These subgroups, therefore, fall into the dark-yellow field.

Most of the panel members thought that MIC was unlikely for subgroup 12.9 because of the generally good quality water and the removal of oxygen due to the corrosion of the carbon steel. However, one panel member believed that its occurrence was highly probable in this subgroup. The average panel score for susceptibility to MIC for subgroup 12.9 was 1.13 and, for knowledge, the average score was 2.88. This subgroup thus falls into the dark-yellow field.

3.3.4.3 Less-Susceptible Component Subgroups in the BWR Support and Auxiliary System

Subgroups which fell into the dark yellow and green regions are shown in the modified rainbow charts, Figures 3.33 and 3.34. The subgroups have been sorted in the same manner as for the other major systems. More information on the evaluation and scoring for these subgroups can be found in Appendices D and E.

Subgroup Description		Degradation Mechanism								
		CREV	FAT	FAT-HWC	FR	GC	MIC	PIT	SCC	SCC-HWC
Carbon and Low-Alloy Steel Components and Welds in RCIC System										
12.2	SA105,106,234 - Carbon Steel Weld Metal									
12.4	SA105,106,216,234 - Carbon Steel									
12.5	SA105,106,216,234 - Carbon Steel Weld									
12.6	SA216 - Carbon Steel									
12.7	SA105,106,216,234 - Carbon Steel									
12.8	SA105,106,216,234 - Carbon Steel Weld									
12.10	SA216 - Carbon Steel									
12.11	A516 Gr 70 LAS Pump Casing									
12.12	SA105,106,216,234 - Carbon Steel									
12.13	SA105,106,216,234 - Carbon Steel Weld									
12.14	SA105,106,234 - Carbon Steel Base & Weld									
Carbon and Low-Alloy Steel Components and Welds in Systems other than RCIC										
20.11	Cast CS Nozzles		*							
25.1	SA105,106,216,234 - Components									
25.2	SA105,106,216,234 - CS Weld & HAZ									
25.6	SA105,106,216,234 - Carbon Steel									
25.8	SA105,106,216,234 - Carbon Steel								X	
25.9	A106,A516 Carbon & Low Alloy Steels									
26.1	SA105,106,216,234 - Carbon Steel									
26.2	SA105,106,216,234 - Welds and HAZ									
26.3	SA105,106,216,234 - Sockolet								X	
26.4	SA105,106,216,234 - Valves								X	
26.5	SA105,106,216,234 - HX Nozzle								X	
26.11	SA105,106,216,234 - NR HX Baffles								X	
26.13	SA105,106,216,234 - NR HX Piping									
26.14	SA105,106,216,234 - Carbon Steel (low T)									
27.4	SA216 - Valves									
27.7	SA106 - Carbon Steel Nozzle									
28.1	SA105 - Carbon Steel Nozzle									
28.2	SA106,216,234 - Carbon Steel Piping									
28.3	SA216 - Carbon Steel Valves									
28.4	SA234 - Carbon Steel Weldolet									
Type 304/316 Stainless Steel Components and Welds in RCIC System										
12.16	304 SS RCIC Strainer									
Type 304/316 Stainless Steel Components and Welds in Systems other than RCIC										
20.3	308 SS Welds and HAZ									
20.10	SS Type 304 Tee									
25.3	304 Stainless Steel									
25.4	304 Stainless Steel - Weld Metal									
26.6	304 Stainless Steel - HX Nozzle								X	
26.7	304 stainless steel (assume welds annealed)									
26.8	304 Stainless Steel - Reg. HX Tubesheet									
26.9	304 Stainless Steel - Reg. HX Tubes									
26.10	304 Stainless Steel - Reg. HX Baffles									
26.12	304 Stainless Steel - Reg. HX Piping									
26.15	304 Stainless Steel (low T)									
27.1	304 Stainless Steel and Weld Metal									
27.5	304 SS - Base, Weld and HAZ Sockolet									
27.6	304 SS - Base, Weld and HAZ Nozzle									
27.8	Stainless Steel - Base, Weld and HAZ									

NOTE: * Susceptibility at color interface with one or more scores higher than interface; [x] Susceptibility inside color box with one or more scores higher than this color box upper interface.

A blank in HWC column indicates same color (susceptibility and knowledge score) as adjacent column for same mechanism (NWC conditions).

Figure 3.33 Modified Rainbow Chart Showing Dark-Yellow Subgroups in BWR Support and Auxiliary System

Subgroup Description		Degradation Mechanism			
		FAT	FR	PIT	SCC
Stainless Steel Components -- External surfaces					
20.1	All SS Components External Surfaces				
Wrought and Cast SS Components and Welds					
20.2	SS Type 304 Components				
20.4	Cast SS CF8 Components				
20.5	SS Type 316 Components				
20.8	Cast SS CF8 Valves				
20.9	Welds SS 308				
27.2	304 Stainless Steel HAZ				
Carbon Steel Components and Welds					
20.6	CS A 515 Unclad Pump Outlet				
20.7	Carbon Steel Welds				
20.12	Pump Casing Cast CS				
27.3	SA106,216,234 - Carbon Steel Base, Weld, HAZ				

Figure 3.34 Modified Rainbow Chart Showing Green Subgroups in BWR Support and Auxiliary System

3.4 Generic Materials Degradation and Life Management Issues

The focus of the materials degradation assessment in the previous sections has been on component-specific issues in current LWR designs. That is, material degradation-mode and component combinations likely to be susceptible to future degradation in, for instance, the RCS, ECCS, secondary water and service water systems in PWRs and the corresponding BWR systems. There are, however, generic issues that have a wider significance when managing materials degradation by mitigation. These generic issues and the associated research areas for current LWR designs are discussed in this section in terms of the:
(a) Quantitative prediction of the rate at which damage is accumulated,
(b) Criteria for component "failure,"
(c) Reduction in margin with time between the extent of damage and the "failure" criterion.

A fourth vital component to the materials degradation management scheme is the damage detection capability; this particular aspect was outside the scope of this PMDA. The interactions between these topics are illustrated schematically in Figure 3.35. For instance, damage, shown by the yellow region, increases with time and is shown here in a generally expected form of its rate increasing with time. However, the damage rate, in general, may be linear, parabolic, stepwise, or some other functionality depending on residual stresses, cyclic stresses, and changes in surface environments. Detection is shown in Figure 3.35 as the red horizontal line although, in fact, it may well have a downward slope since these detection capabilities may improve with time. Finally, "failure," denoted by the blue band in Figure 3.35, has several definitions as discussed later; in all cases, however, "failure" may be associated with a specific extent of damage which may well, as illustrated, decrease with operating time due, for instance, to time-dependent reductions in fracture resistance.

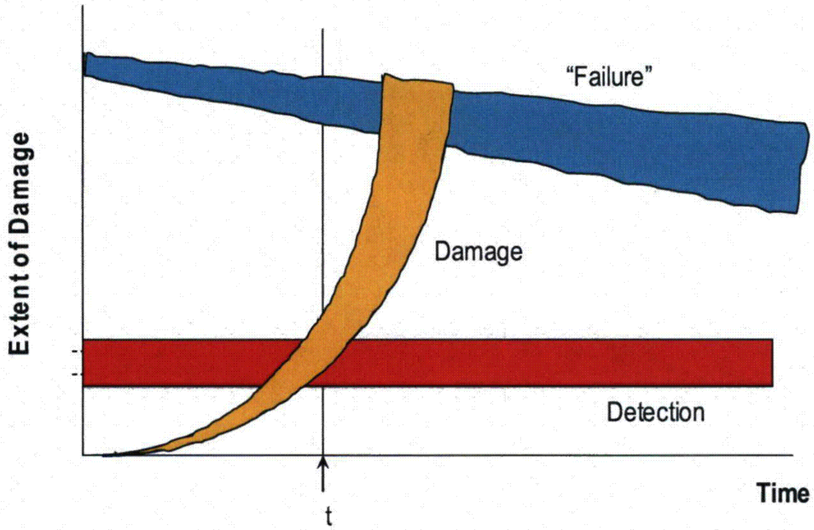


Figure 3.35 Schematic variation of "damage" as a function of time, and its relationship to damage "detection" and to component "failure."

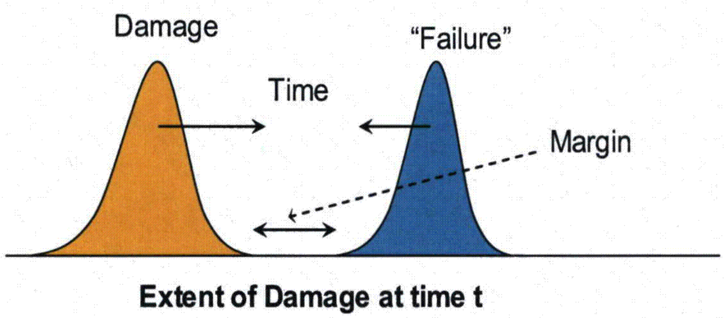


Figure 3.36 Schematic probability density functions for damage and failure as a function of time. Margin is defined as the gap between the greatest value of the damage and the least value of failure.

The relationships in Figure 3.35 are shown as shaded bands, mirroring the dispersion in the various parameters at a given time. As illustrated in Figure 3.36, these dispersions are, at a given time, defined by the probability density function for the amount of damage, with the specific function depending on the degradation mode, the individual plant design, manufacturing history and operating and maintenance conditions. Similarly, there will be a distribution in the extent of damage required to lead to "failure;" an ideal example of this would be the current re-evaluation of the criteria for PWR pressure vessel failure due to pressurized thermal shock, where a significant distribution in damage criteria is associated with the distributions in the material, thermal hydraulic and stress conditions. There will also be a distribution (not shown in Figure 3.36) in the probability of detection, which will vary with defect size, operator experience, etc. Margin is defined as the gap between the greatest value of the detected damage distribution and the least value of failure criterion distribution. The probability of component failure increases as the tails of the two distributions intersect.

An understanding of the changes and distributions of the parameters illustrated in Figures 3.35 and 3.36 is of importance from a degradation management viewpoint, since it provides answers to questions such as: "What is the margin between the current extent of damage and failure;" "How accurately is that margin being monitored;" "How fast is that margin being eroded with time" and, ultimately, "How is the component's functionality affected during normal operation, and what is its capability to prevent design-basis accidents?" The answers to these questions are a measure of the ability to manage proactively the materials degradation issues. This is pertinent even when mitigation actions have been implemented since such actions may fail due to inadequate control of their application.

No distinction is made between generic issues associated with BWRs and PWRs, since it is fairly widely accepted [1, 2] that, although there may be differences in the detailed mechanistic aspects of, for example, degradation in some specific alloy/environment systems (e.g., stress corrosion cracking of nickel-base alloys in PWR primary environment and in BWRs under NobleChem™), in general the various degradation modes in the two reactor designs are governed by a continuum in material (e.g. yield stress, degree of cold work), stress (e.g. residual and applied stress) and environment (e.g. temperature, corrosion potential) conditions. To make a distinction between the two reactor designs in discussing the generic concerns would dilute the importance of addressing these issues.

In this discussion it is assumed that the reactor has been fabricated and is being operated under conditions defined by current regulations, technical specifications and industry guidelines vis-à-vis materials degradation. Some deviations are expected (e.g., water chemistry transients during plant operation) and these are within the discussion scope. However, gross deviations in system conditions due to, for example, human errors (safety culture) are not within the discussion scope. This exclusion is significant, since some of the materials degradation problems that have occurred in the past have been attributed to this root cause. Past examples of materials degradation issues involving procedural or human error include delayed inspection (for instance, flow accelerated corrosion at Mihama-3 Power Station), lack of timely corrective actions (for instance, boric acid corrosion at Davis-Besse Nuclear Power Station), fabrication errors (for instance, abusive grinding and its effect on cracking in BWR core internals and piping systems), and gross, unexpected water chemistry excursions (for instance, resins, thiosulfate, seawater, etc.) leading to localized damage such as stress corrosion cracking.

3.4.1 Damage Assessment

Many of the generic issues discussed are associated with fatigue (including thermal, mechanical and environmental influences) and stress corrosion cracking, since these dominate the judgments of the PMDA panel for both PWR and BWR systems. Items 1-5 below refer to potential research projects for situations where there is strong evidence that degradation might occur, but where there is insufficient knowledge to manage effectively by mitigation the degradation over an extended time period. Item 6 addresses those situations where there is little current evidence of degradation in the plant, but conversely there is no knowledge to assess the likelihood of degradation in the future.

1. ***A quantitative treatment of the sequence of cracking damage accumulation due to localized corrosion (intergranular attack, pitting, etc.), microcrack initiation, crack coalescence, followed by "short" crack propagation.*** This sequence is well recognized

(Figure 3.37), and was discussed in some detail in Section 2.4 in terms of (a) the initiation of microcracks at, for instance, pits, intergranular corrosion sites or regions of surface cold work; (b) microcrack coalescence to form a "single" short crack, followed by its acceleration; and (c) in some cases, deceleration and arrest of the short crack.

During this time period, the defects, although relatively small, are of metallurgically significant dimensions that have a low probability of detection by commercial inspection methods. This sequence has been well quantified for systems peculiar to the gas pipeline industry but, apart from a few isolated instances, such as some preliminary studies of corrosion fatigue crack initiation for carbon steels and stress corrosion cracking of stainless steels in BWR environments, these phenomena have not been quantified for the LWR systems of current interest.

This research can provide valuable input to inspection and repair/replacement decisions.

2. ***A quantitative understanding of the "precursor" conditions required for the onset of cracking after long times.*** In some instances, the sequence of events illustrated in Figure 3.37 start at the time of commissioning of the component. Such instances, which correspond to Case I in Fig. 3.38, include the stress corrosion cracking of sensitized and non-sensitized components in BWR environments; in PWRs, stress corrosion cracking at low potentials (LPSCC) starts at the beginning of life in tubing and piping. A further example could be flow accelerated corrosion of carbon steel components.

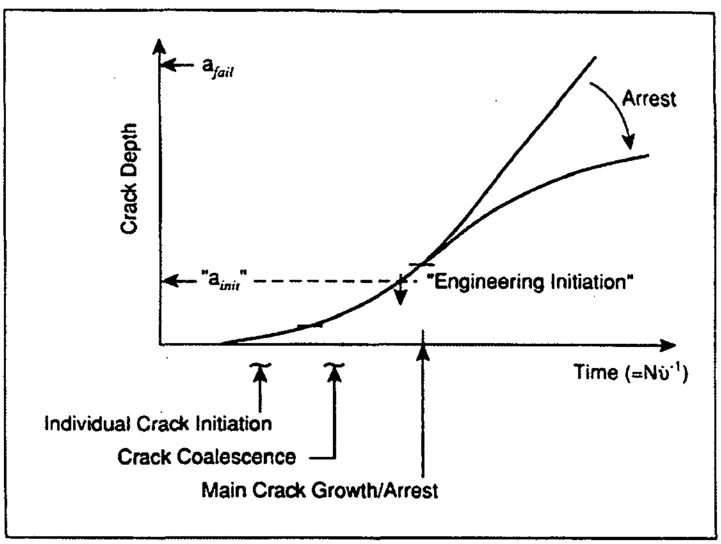


Figure 3.37 Sequence of stochastic events of localized corrosion (e.g., pitting, IGA), crack initiation, coalescence and short crack growth, that are inherent to the definition of "engineering crack initiation" [3].

By contrast, in other systems some specific precursor conditions must develop first before the sequence of crack initiation, coalescence, etc. can start (see Appendix B.15). Examples corresponding to Case II in Figure 3.38 include stress corrosion cracking at the tube support plates in steam generators with drilled hole tube supports. Here, before any cracking can occur, it is necessary that a localized chemistry accumulates which will support the cracking mechanism. A similar example is the creation of corrosion products in the steam generator tubes/carbon steel support plate sufficient to give rise to denting and the creation of stresses that will initiate stress corrosion cracks on the primary side of the tubes. Another example is the creation of specific film compositions that may be required for subsequent crack initiation. This topic received attention decades ago and is currently receiving renewed attention, especially for the initiation of cracks in the nickel-base alloy/PWR primary environment system, as more advanced in-situ surface analysis techniques are developed.

The circumstances for Case III (Fig. 3.38) are of special interest to the occurrence of damage after very long "precursor" times. Here, the precursor events might include the temperature-dependent changes in composition (for example, chromium) at grain boundaries or on the bulk surface thereby producing material chemistries more prone to stress corrosion cracking.

The quantification of such precursor events becomes of importance when attempting to predict future damage in structures in which damage has not yet been noted. A procedure that could be followed in developing the required quantification is suggested in Appendix B.15.

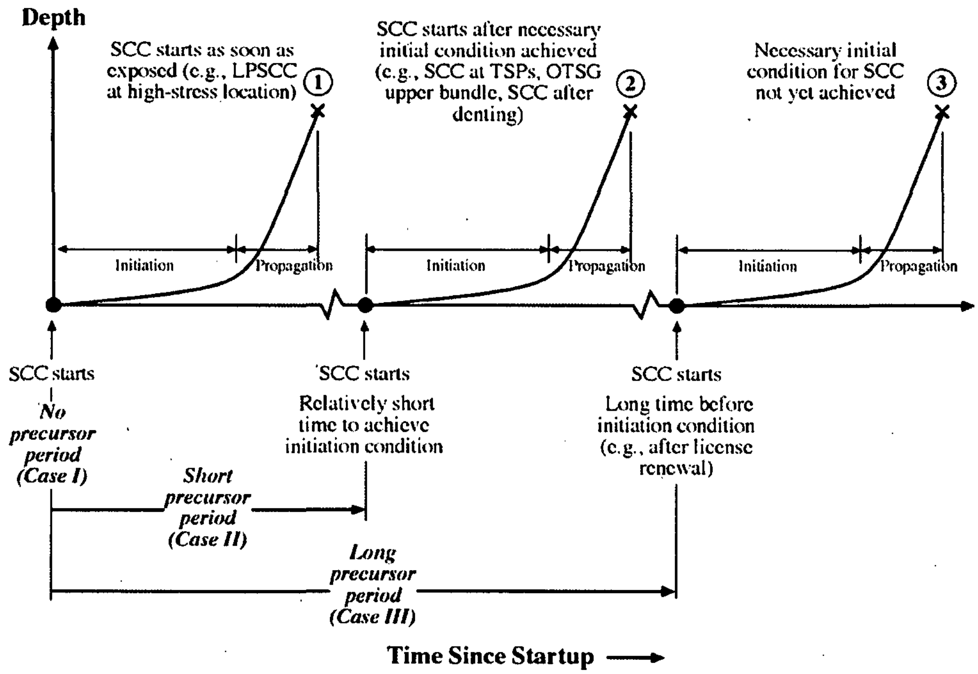


Figure 3.38 Schematic views of three cases for the time-dependence of stress corrosion cracking. (See Appendix B.15)

3. ***Physical interpretation of the statistical parameters associated with the distribution of initiation and propagation times.*** As discussed in Section 2.4 and in Appendix A, modes of corrosion-related degradation (stress corrosion, crevice corrosion, flow-accelerated corro-

sion, etc.) contain stochastic components. This means, as discussed in some depth in Appendix B.19, that even under ideally controlled conditions, the processes which control, for example, stress corrosion cracking are inherently probabilistic owing to the many paths that are available for the initiation and propagation processes. The variability that arises out of the complexity of the sequential paths is *intrinsic* to the degradation mode. A further variability due to *extrinsic* probabilistic factors arises out of uncontrolled variations in metallurgy, chemistry, and structures as well as variations in local environments. Under these particular conditions the degree of variability can be controlled, and this forms part of the discussion of "long crack propagation" and the "adequate definition of the system parameters" discussed in Items 4 and 5 below.

In general, statistical distributions, e.g., Weibull functions that can describe corrosion processes, have three parameters (see Appendix B.19); the space parameter, the shape parameter, and the location parameter. The values of these parameters are usually determined by fitting statistical distributions to data. The resulting distributions are often carried forward to predict the occurrence of failures at some longer time. This approach, as applied to corrosion processes, has been developed by Staehle et al. (See Appendices B.15 and B.19) and is illustrated by the data in Figure 3.39, which have been taken from the indications of cracks of steam generator tubes in an operating PWR during successive outages. Such knowledge is of importance in making management decisions about future repairs, replacement, etc. The analysis may be applied early in the development of damage when only a few incidents have been noted, by applying Bayesian updates as new data is accumulated. This process is aided by knowledge of the relationships between the statistical space, shape and location parameters and the physical system descriptors such as stress, temperature, material, etc.

However, a significant technical challenge in expanding this approach to proactive materials degradation assessment (i.e., predicting degradation *before* it is observed) is the need to develop adequate relationships between the relevant statistical parameters and the *interrelated* physical parameters such as stress, anionic concentration, corrosion potential, etc.

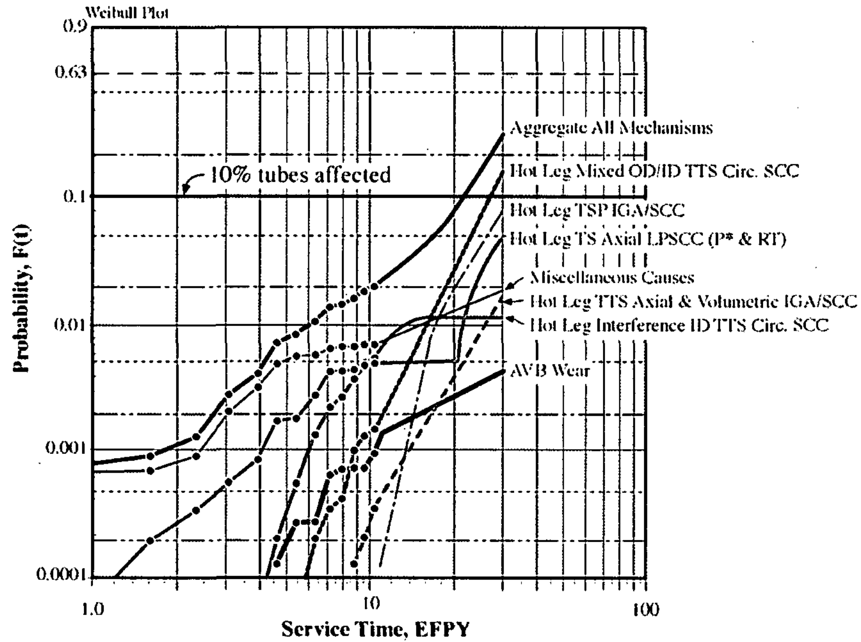


Figure 3.39 Probability vs. equivalent full power years (EFPY) for failures of tubing from a set of SGs in the Ringhals 4 PWR. Designations: TTS = "top of tube sheet." TS = "tubesheet." Circ. SCC = "circumferential SCC." P* = special location where SCC is not serious. RT = "roll transition." AVB = "antivibration bars" (See Appendix B.19 and [4]) ***(© NACE International 1988)***

4. ***A review of the adequacy of quantitative life prediction models for "long (or deep)" crack propagation.*** This need is an adjunct to Item 1 above. Historically there has been a fairly concentrated effort in this area arising out of the need for disposition relationships (i.e., crack propagation rate vs. stress intensity algorithms) that are used in defining inspection periodicities once a crack has been detected. These relationships are usually correlation functions based on an existing crack propagation rate database from plant or laboratory experience. A quantitative understanding of the mechanism of crack propagation will also provide insights into the intrinsic and extrinsic uncertainties in the degradation phenomenon, and help in analyzing the practical impact of the *interactions* of system variables (e.g. stress, temperature) which are not necessarily fully characterized in the experimental programs upon which the data-correlation–based disposition relationships were formulated. This will assist in verifying the adequacy and completeness of both the disposition models and the underlying mechanistic understanding.

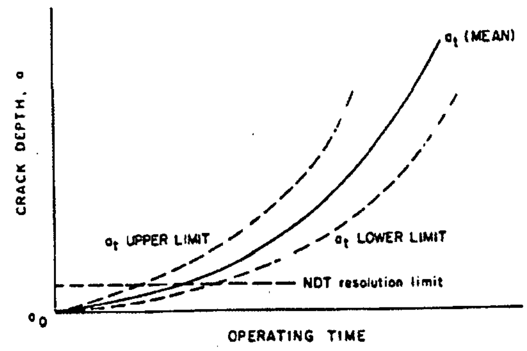
Two examples are given below to illustrate these concerns:

- *Cracking of nickel-base alloys in PWR primary environments.* Continued research can provide an adequate quantitative understanding of all of the relevant variables in the stress corrosion crack propagation rate/stress intensity factor relationships for nickel-base alloys in PWR primary environments, including the interactions between material, stress and environmental parameters. The quantitative definition of the effect of material microstructure (and the associated bulk composition and fabrication history) on the cracking susceptibility is of concern since we cannot predict *a priori* the composition and fabrication conditions that give subsequently "good" and "bad" heats vis-à-vis stress corrosion resistance. A partial reason for this is the fact that the majority of the effort has been expended on developing empirical databases relevant to quantifying and mitigating the problem. An increased amount of effort could result in arriving at a consensus on the cracking mechanism, which is likely to be different from that extensively developed for austenitic alloys in BWRs. This combined correlation-model and mechanistic understanding capability would provide a powerful base for predicting future degradation and formulating mitigation actions in a timely manner.

A further concern in this alloy/environment system is the observation in the laboratory (see Appendix B.13) that enhanced stress corrosion cracking and fracture susceptibility may be noted under very specific alloy composition, environment and stressing conditions that might be relevant to some transient operating conditions. The fact that degradation *increases* at lower temperatures contravenes the generally accepted behavior of temperature-activated processes. The mechanism for this low temperature behavior, and its practical importance to operating PWRs, is still not well understood.

- *The use of "threshold" values of operating parameters below which crack propagation (and initiation) is negligible.* From a degradation management point of view, it will be useful to have threshold values of various system parameters below which degradation is unlikely in a given time period. For instance, threshold values of stress intensity, corrosion potential, neutron fluence, etc. are often quoted for the purpose of setting water chemistry specifications or for evaluating the likelihood of a given degradation mode being possible. In the main, however, these values have been evaluated in "separate effects tests" and are defined when the cracking resistance is "acceptable," without taking into account the fact that the majority (if not all) of these threshold values in fact depend on the values of other parameters. A re-evaluation of this concept of "thresholds" may be needed especially since the definition of "acceptable" cracking resistance may be changing with increased operating time (e.g., license renewal) and changes in reactor operating modes (e.g., water chemistry, power uprates, load following etc.).

5. ***Adequate definition of the "corrosion system" parameters that control the kinetics of environmentally assisted degradation.*** Once a prediction algorithm for the damage process, described in Items 3 and 4 above, has been developed, its practical usefulness will depend strongly on an adequate definition of the material, environment and stress parameters (Figure 3.40). If this is not accomplished then a wide variability in predicted damage is to be expected.



a_t = f [Material, Stress, Environment]

Material	Stress	Environment
Chromium	Primary	Oxidizing Species
Silicon	Static	Corrosion Potential
Sulphur	Dynamic	Anionic Species
Phosphorus	Residual	Type
Nickel	Crack Depth	Activity
Carbon	Neutron Fluence	Flow Rate
%CW		Temperature
Heat Input		Neutron/γ flux
Neutron fluence		Crack Depth

Figure 3.40 Combinations of system parameters that may affect "deep" (i.e., >50-100 µm) crack propagation in austenitic alloys in LWRs. [5]

For instance in the case of stress corrosion cracking of BWR components, the crack propagation rate is a strong function of the corrosion potential, and if corrosion potential is not measured or calculated via knowledge of e.g., oxidant activities, water flow rate, and irradiation flux, then there will be a distribution in the predicted and observed cracking data. This can be assessed via a Monte Carlo analysis of the distribution of the corrosion potential values and how this varies during plant operation. Similar analyses may be conducted for the other relevant system parameters indicated in Figure 3.40. This understanding has been crucial in the development over the last 10 years of data quality control procedures, but it also points out that there are significant gaps in our capability to define adequately the relevant system parameters. Examples of such shortcomings include:

- *Inadequacy of the definition of the tensile stress and its role in controlling crack propagation.* As discussed in Section 2.4, Appendix A and several background papers in Appendix B, the dominant effect of "stress" on cracking of most ductile structural alloys in LWRs is via its effect on the localized plasticity and the effective crack tip strain rate, and how this is defined in the spectrum of values associated with creep (under constant load or displacement loading) and applied strain rates (under, for instance, fatigue loading).

  This realization has led to a considerable amount of research on the changes in cracking susceptibility in various alloy/environment systems due to the effects of absorbed hydrogen (from the corrosion process at the crack tip), cold work, and dislocation morphologies due to various factors (e.g., precipitate free zones at the grain boundary, coherent/incoherent precipitates) on the localized crack tip plasticity.

The development of algorithms describing the relationships between crack tip strain rate and the engineering parameters (such as stress, strain, loading rate, etc.) are under constant development but, from a practical viewpoint, the following concerns probably deserve enhanced attention:

- The effect of high R (ratio of minimum load to maximum load) ripple loading on environmentally assisted cracking, and its codification. As discussed in Appendix B.8 on SCC of carbon and low alloy steels, such ripple loading can significantly increase the crack propagation rates in oxidizing environments via its effect of the crack tip strain rate. Similarly, ripple loading and high load ratio unloading/reloading effects can accelerate growth rates in stainless steels and nickel alloys. Such accelerating factors should be better understood and accounted for if proactive management by mitigation is desired.
- The residual stress adjacent to welds has long been highlighted as a prime mechanical driving force for cracking in welded components. More recent research indicates that it is the residual stress *and strain* profiles that are of importance. This important nuance is not fully embraced in current failure analyses. In the same vein, predictions of future cracking rely extensively on finite element analysis calculations of the residual stress profiles, especially for complex weld geometries involving dissimilar metals. However, there is relatively little validation (against measurements on "mock-ups") of such calculated profiles and their expected distributions as a function of e.g. irradiation assisted relaxation, welding conditions (such as weld heat input, degree of constraint, welding speed, part misalignment), or stress relief heat treatment. As illustrated earlier in Figure 2.6, small changes in the residual stress pattern can have a marked effect on the locus of the crack depth/ time relationship.
- Weld repair is also known to be of significance (witness the implication of the role of weld repairs on the incidence of IGSCC of nickel alloy weldments in PWR primary piping systems). Quantification of these factors will have a marked effect on the accuracy of the prediction of cracking under specific conjoint conditions of material and environment.
- Finally, in this category of residual stress analytical needs, there is the question of predicting the adverse effect of surface cold work in accelerating crack behavior in the region up to 100 μm below the surface; such an effect has been known for decades spanning the effect of surface grinding on the IGSCC of BWR piping in the 1970s to more recent examples in BWR core components. Preliminary analysis indicates that such effects may be predicted quantitatively merely by taking into account the change in the residual stress profile; in a proactive mitigation program, such analyses should be reexamined to account for other known changes, such as microstructural changes (e.g., martensite formation) and increases in yield stress due to bulk cold work, which are known to independently alter the cracking susceptibility.
- The effects of the changes in stress intensity factor with crack length (dK/da), which occurs in essentially all components as cracks develop, and which can lead to dramatic changes in propagation rate not mirrored by consideration of the stress intensity factor alone.

- Characterization of thermal loads and flow-induced vibration, as seen in the significant susceptibility findings of the current study for the assessment of cracking at socket welds in LWRs and of BWR steam dryers.

- *Inadequacy of the definition of material conditions.* Again, it has long been recognized that local material compositional changes due to fabrication and operational conditions can have a marked effect on the susceptibility to environmental degradation. Examples include grain boundary sensitization during stress relief or welding operations, and subsequent low-temperature sensitization or irradiation-induced grain boundary segregation during operation. The understanding of most of these phenomena is at quite a high level and gives qualitative support for the future use of more resistant compositions (e.g., stabilized or L-grade stainless steels) and lays the groundwork for advanced approaches such as "engineered grain boundaries." However, in the PIRT assessments of this study, concerns were expressed regarding the *long-term* degradation resistance which requires a more detailed quantitative understanding of many of these materials condition issues. Such issues include:
  - Details of weld metallurgy (matrix, grain boundary and interdendritic microstructure and microchemistry) especially for some of the replacement nickel base alloys that are prone to hot shortness during welding.
  - The kinetics of grain boundary composition changes of stainless steels under irradiation conditions, with special attention to silicon and perhaps molybdenum. In this regard, attention should be paid to the effect of dose rate in addition to cumulative dose, since embrittlement may occur at lower doses when it is accumulated at a low dose rate.
  - The specification of alloying elements not heretofore recognized as pertinent to degradation modes. An example in this category is aluminum and nitrogen in low alloy steels which gives rise to dynamic strain aging under specific strain rate and temperature conditions and has a deleterious effect on stress corrosion and corrosion fatigue in oxidizing environments.

- *Inadequacy of the definition of environmental conditions.* It is apparent from the discussions in the background papers of Appendix B and in Appendix A that a precise definition of the environmental conditions is required for any defensible prediction and management of the various degradation modes. Such a definition depends on (a) knowledge of the global or system environment via monitoring, (b) qualified algorithms to translate that definition for monitored localities to unmonitored localities, and (c) prediction algorithms to define the creation of localized environments (which may be affected by system phase changes, changes in oxidizing conditions, heat transfer, etc.). All of these aspects have been researched in some depth over the last 25-30 years, and have led to a justifiable definition of guidelines for water chemistry control in both PWRs and BWRs. During this PIRT assessment, several issues were identified which indicated that further research may be needed in the areas of:
  - The definition of crevice chemistries in PWR steam generators, with special focus on the presence of lead and low-valence sulfur combinations that may markedly increase the stress corrosion cracking susceptibility of nickel base alloys, including thermally treated alloy 690.
  - The chemical environment in occluded regions such as the annulus between the nickel-base alloy CRDM penetration tube and the low-alloy steel PWR pressure

vessel head. Extensive boric acid corrosion occurred in this region at the Davis Besse plant at rates that were not expected in this particular assembly geometry. Thus, it is important to determine the conditions under which such high rates of corrosion can occur, especially with regard to the physical chemistry of concentrated boric acid and the possible contributing effects of flow impingement. Boric acid corrosion models for geometries specific to PWR pressure vessel and pressurizer penetrations would be useful.

- The impact of bulk chemistry transients on changes in crack tip chemistry and the resultant effect on the duration of increased cracking susceptibility. This is of particular importance for chloride transients on the stress corrosion crack propagation rate in low-alloy steels in oxidizing environments, where, as discussed in Appendix B.8, laboratory data show that the increase in susceptibility and the duration of the effect is markedly greater than for sulfate transients.

It is noted that environmental transients are frequently associated with maintenance, operating, or mechanical transients, and it is often difficult to separate these transient contributions, especially the links between stress and environmental transients, including shutdowns and layups.

6. ***Completeness of identification of degradation modes.*** Numerous hypotheses have been proposed over the last 30 years to identify, understand and predict environmental degradation (and particularly environmentally assisted cracking) in LWR systems. However, there are isolated incidences, either in the laboratory or in the plant, which challenge the belief that all degradation mechanisms and modes are adequately identified and understood. Examples of such degradation issues, which are significant in assessing the future degradation behavior of reactor components, include:

    - The enhanced cracking susceptibility of some nickel-base alloys in hydrogenated water at temperatures below approximately 150°C (302°F) (see Appendix B.13). This is of concern since some of these alloys are replacement alloys for 600/182/82 in PWR primary circuits. Nominally, these effects are generally attributed to localized hydrogen embrittlement at a pre-existing crack tip. However, the specific system conditions leading to this behavior, their relevance to LWR operations and their possible applicability to wrought stainless steels or cast stainless steels have not been examined. The impact of these observations to the fracture resistance of these materials is discussed later.

    - Review of the adequacy of current models of flow accelerated corrosion, especially when combined with possible galvanic effects that might be associated with weldments. Enhanced FAC at welds has been observed in piping and equipment (e.g. feedwater heaters) at over 30 plants globally covering BWR, PWR and VVER designs. The precise reason for this enhancement is currently unknown.

    - Mechanisms and quantitative prediction capability for biological fouling and MIC in service water systems, which has, in some circumstances, led to lack of functionality of vital service water systems.

    - The possibility that the presence of one degradation mode may promote susceptibility to a further mode. An example of this is the hypothesis that absorption of hydrogen during flow accelerated corrosion of a carbon steel component may increase the susceptibility of that component to cracking, especially if that component is cold worked.

- Accumulation of apparently minor chemical impurities or microstructural changes which may change the details of the accepted damage mechanism and, thereby, the prediction of the rate of future damage of the targeted (or other) components. Examples of such concerns include:
  - The reasons for the observation (in isolated plant and laboratory experience) of an intergranular cracking morphology for low alloy steel in high temperature water, where both stress corrosion and fatigue cracks are usually transgranular. There is interest in acquiring such an understanding because intergranular cracking usually represents an increase in cracking susceptibility. Changes in cracking morphology are usually associated with specific changes in *localized* grain boundary composition under given environmental/stressing conditions. Unless the reasons for the morphology change to the intergranular mode in the low alloy steel/high temperature water system are known, then the disposition algorithms for transgranular cracking cannot be considered conservative.
  - Near-surface composition changes which may act as a precursor to subsequent cracking. An example would be depletion of chromium on the surface of Alloy 690 from exposure to high temperature water, which may promote crack initiation in this replacement alloy.
  - The kinetics of the reduction of sulfate anions by hydrazine to sulfide anions, which may promote cracking of steam generator alloys and other components (e.g., turbine materials).

3.4.2 Fracture Assessment

"Failure" may be defined by a range of metrics spanning catastrophic rupture of a pipe due to a double-ended guillotine break initiated by propagating cracks to, at the other end of the spectrum, a relatively minor leak that, nonetheless, exceeds the plants technical specifications or regulatory criteria. "Failure" may also be attributed to the removal from service of a component for safety reasons, even though that component may not be markedly degraded; an example of this would be the plugging of steam generator tubes adjacent to a cracked tube because they may be equally susceptible to failure and could fail unexpectedly at any time. Alternatively, failure may be associated with adverse economic impacts due to inspection requirements or the extensive plugging of steam generator tubes. Given this wide range in definitions of "failure," spanning structural integrity to economics criteria, the expert panel confined its scope to those issues that could potentially affect the integrity of the materials used in the plant, in accordance with the guidance that "the reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture."

Within that scope, potential areas for future research considered during the PMDA panel discussions related to failure assessment included:

- ***Validity of fracture resistance values measured in air compared to the values obtained in the operating aqueous environment.*** The decrease in fracture resistance with time for low alloy pressure vessel steels and stainless steels exposed to fast neutron irradiation, or duplex cast stainless steel due to thermal decomposition of the delta ferrite phase are well recognized. Of some concern is not only the scatter in these data, but also the fact that the fracture resistance associated with surface breaking flaws can be lowered when measured in water, especially when tested at reasonably slow strain rates. The presumption here is that subcritical crack propagation is occurring during the fracture mechanics test. Thus there is an interest in clearly understanding the physical meaning of J_i (J value at SCC initiation) and dJ/da (slope of J/R curve) measured in an aqueous environment.

- ***Effect of dissolved H on fracture resistance of Ni-base alloys.*** As discussed in Appendix B.13 on low temperature crack propagation and mentioned earlier in this section, higher chromium content nickel-base alloys may exhibit markedly lower fracture resistance values after they have been exposed to hydrogenated high temperature water, and are tested at temperatures below 150°C (302°F) at specific applied strain rates. The mechanism for this embrittlement is still being evaluated, as is the relevance of these conjoint conditions to operating conditions in the PWR primary system. Potentially these findings are of importance since many of the replacement alloys may be affected by this phenomenon.

- ***Extension of hydrogen embrittlement concerns to duplex stainless steels and thermally aged higher ferrite content cast austenitic stainless steel (CASS).*** Hydrogen embrittlement of ferritic steels at low operating temperatures is a well-recognized and researched phenomenon. By analogy with the preceding item on nickel alloys, it can be hypothesized that there is a synergy between hydrogen embrittlement and the considerable embrittlement associated with thermal decomposition of delta ferrite in cast stainless steels containing more than 20% ferrite and, to a lesser extent, with the lower ferrite content duplex stainless steels.

3.4.3 Margin Assessment

In order to effectively manage materials degradation, it is necessary to have an adequate margin between the distributions of the damage assessment and the failure criteria indicated schematically in Figure 3.36. A more defensible definition of that margin could be developed that improves on the classical "engineering judgment" where there is sometimes a lack of rigor with regard to input assumptions including the synergistic effects between the material, environment and stress conditions. Two generic developments to be considered are therefore: (1) a definition of an acceptable frequency-consequence combination associated with loss of functionality due to degradation of the structural material, and (2) an adequate inspection/monitoring system that defines the current extent of damage.

- ***Incorporation of material degradation (aging effects) into probabilistic risk assessments (PRAs).*** There has been development and use of risk-informed guidance for on-line maintenance, changing technical specifications, and in-service inspection. This places reliance on probabilistic risk assessments to assure that any changes in plant operation do not significantly increase the core damage frequency. Such assessments do not currently account for the fact that the material in safety-related and safety-significant components may be undergoing degradation at a rate that changes with operational time. Incorporation of aging effects into PRAs has been discussed for almost a decade, and preliminary applications have been made [6]. Such developments should be expanded beyond a simple Bayesian methodology developed for the NRC (e.g., DORIAN), that takes into account the increasing availability of quantitative time dependent materials degradation algorithms.

A related issue is how to manage materials degradation where no defects have yet been detected, but may remain in service for 60 years or more. The classic example is that of Alloy 800 SG tubing, with only a few cracks after 30 years of service. The problem is, therefore, how to provide proactive management options other than to maintain good inspection practices, when there is no reactor database, and only highly accelerated or aggressive laboratory test data available.

- ***Inspection capabilities.*** Evaluation of inspection capabilities was outside the scope of this PMDA panel study. However it is vital that there be adequate volumetric inspection capabilities, especially for the complex weld geometries associated with reactor pressure vessel and pressurizer penetrations. Development of on-line crack detection and monitoring capabilities should also be encouraged since degradation may develop at changing rates depending on the specific plant operating conditions, even for the same material under similar operating conditions.

3.4.4 The Next Step: Proactive Materials Degradation Management

A final generic issue is the ability to transition from reactive to proactive management of materials degradation at the NRC, the U.S. industry, or their international counterparts. Technically-driven management of materials degradation in LWRs has been part of the professional life of all the panel members involved in this PMDA study–their observations, conclusions and recommendations on this issue are as follows:

- Materials degradation will continue in LWRs, and may increase with license renewal and power uprates, coupled with an emphasis on the short term economics of operation. The former factors increase the operating time and/or the severity of the environment, e.g., from increased irradiation fluence and flux, thermal aging and increased coolant flow rates. Alternate materials, modified operating conditions, etc. may counteract these factors but, generally, they are not fully proven and address only a fraction of the degradation modes. The technical reasons for these statements are outlined in detail in this PMDA report. Thus it is concluded that a Proactive Materials Degradation Management (PMDM) phase is needed.[3]

- Materials degradation typically results from complex phenomena involving metallurgy, electrochemistry, mechanics, radiation damage, physical chemistry, etc. The complexity is exacerbated by the interdependencies among these phenomena, which lead in turn to the need for extraordinary experimental sophistication. The plethora of materials degradation modes, disciplines involved, and experimental complexity underscores the need for proactivity, including development of quantitative interdependencies, fundamental knowledge, and experimental techniques needed for high quality, reproducible data.

[3] The NRC staff believes that the changed operating conditions due to power uprates are not expected to appreciably reduce the effectiveness of licensees' aging management programs (AMPs) for currently identified mechanisms. For those plants requesting license renewal, the NRC assures that the licensees have developed effective AMPs focused on these mechanisms for the period of extended operation. Nonetheless, proactive materials degradation management may be useful to predict and manage degradation issues not previously encountered and to anticipate degradation of materials of geometries and in locations not previously shown to be susceptible to a particular aging mechanism. This approach could be used to augment the AMPs currently relied on during the practice of inspection, identification, and subsequent management, if warranted.

This range of capabilities rarely exists in any one organization. Thus the development of a PMDM capability would benefit from collaboration between various organizations

- Prioritization of the proactive tasks on materials degradation management is essential. This must be based on the consequences of a failure as well as the likelihood that definitive insights can be obtained, the need to perform some studies over extended periods of time, the need to evaluate emerging concerns, etc. This is not to promote poorly directed scientific studies, but well-defined, in-depth, quantitative evaluation of important degradation phenomena and their consequences.

- Adequate resources are needed to develop and maintain technical expertise and laboratory capability. This seems obvious but in spite of the significant impact of materials degradation over the last 30 years, the level of funding resources and of the associated expertise and experimental facilities has decreased. Most key experts are now close to, or are in, retirement. This leaves the reactor community vulnerable to a permanent loss of accumulated knowledge and expertise, and with a largely un-mentored workforce with less than five to ten years experience. This is an inadequate basis for addressing the complex questions that need to be answered. Research and development (R&D) funding has decreased, and a higher fraction allocated to short term, "firefighting" projects, which undermines both the status quo and the foundations for proactive materials degradation management. *It is imperative that these resource issues be addressed worldwide by governmental organizations, utilities, vendors and support organizations, and universities and National Laboratories.*

3.5 References for Section 3

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[2] P.L. Andresen "Conceptual Similarities and Common Predictive Approaches for SCC in High Temperature Water Systems," Paper 258 *Corrosion 96*

[3] F.P. Ford, "Prediction of Corrosion Fatigue Initiation in Low Alloy Steel and Carbon Steel/Water Systems at 288°C," Proceedings of Fourth International Symposium on Environmental Degradation in Nuclear Power Systems – Water Reactors, Jekyll Island, Eds. D. Cubicciotti, E. Simonen, National Association of Corrosion Engineers, August 6-10, 1989.

[4] Unpublished data provided by L. Bjornkvist of Vattenfall and J. Gorman of Dominion Engineering.

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[6] C.L. Smith et al., "Incorporating Aging Effects into Probabilistic Risk Assessment- A Feasibility Study Utilizing Reliability Physics Models," NUREG/CR-5632, U.S. Nuclear Regulatory Commission, August 2001.

4. CONCLUSIONS

Members of the expert panel who conducted the Proactive Materials Degradation Assessment undertook the following actions:

- Detailed reviews of the designs of the primary, secondary and some tertiary systems in a representative Westinghouse 4-loop PWR and the corresponding systems in a representative General Electric BWR-5. Differences in materials of construction and component configuration between these specific plants and other PWR and BWR designs were also reviewed.

- An evaluation of the materials degradation modes in the system components based on laboratory and plant operating experience worldwide.

- An assessment of *future* materials degradation based on judgments of the current predictive capabilities (both theoretical and experiential).

The panel members integrated their individual judgments to numerical indices of the degree of susceptibility, the confidence in this call, and the existing level of knowledge that could lead to development of mitigation actions. The methodology for this process is given in Section 2 of this report. This numerical output was supported by written comments to justify the individual scores, and background papers were included to describe the "state of the art" associated with various degradation modes of prime interest. Attention was drawn to individual panel member's scores for degradation susceptibility that reflected a higher susceptibility than the panel's statistical mode or average would indicate.

This process lasted approximately one year starting in August 2004 and involved seven week-long panel discussions at the NRC, with individual assessments being made by the expert panel members in the intervening time.

Based on these actions the expert panel came to the following conclusions regarding both component-specific and generic issues associated with materials degradation in PWRs and BWRs.[4]

4.1 Analysis of Component-Specific Degradation

With regard to the component-specific degradation analyses, the conclusions below concentrate on two scenarios:

- Degradation mode-component combinations where there was a high susceptibility to degradation, based in large part on multiple observations in operating plants, and thus it would be prudent to include these components in proactive materials degradation management (PMDM) programs regardless of the knowledge level.
- Degradation mode-component combinations where there was little (or no) evidence to date of degradation in the plants, but where there was sufficient evidence from laboratory investigations, for instance, to indicate that degradation in the plants might be expected in the future. Therefore, it would also be prudent to consider these components

[4] The Foreword to this report addresses the current regulatory framework for dealing with materials degradation. It also discusses ongoing NRC research programs to help evaluate the safety significance of materials degradation which will be used in evaluating the need for any additional regulatory guidance.

for inclusion in PMDM programs. In addition, cases are identified where the knowledge level of the system interdependencies is low and additional proactive actions (research) may be warranted if PMDM by mitigation is desired.

The analysis of the degradation of components considered operating modes involving full power operation and non-steady state conditions such as reactor start-up, water chemistry transients and extended plant lay-up. The main conclusions of these analyses are summarized below for the various reactor systems, emphasizing those situations where there is high degradation susceptibility based mainly on past plant experience, and those where there is a strong basis for degradation occurrence but no significant plant experience and where there is a low level of knowledge. The analyses for the PWR and BWR Reactor Coolant Systems are addressed first since the panel judged the degradation issues in these systems to be often more numerous and more important than those in the other (generally lower temperature) systems.

4.1.1 PWR Reactor Coolant System (RCS) under Full Power Operating Conditions

The PWR reactor coolant system (RCS) includes the pressurizer, the reactor pressure vessel and its internals, the reactor coolant pump, the primary and secondary sides of steam generator and eight piping subsystems. The RCS piping subsystems are the cold leg piping, crossover leg piping, hot leg piping, pressurizer spray piping, pressurizer surge piping, pressurizer piping to PORVs, pressurizer piping to SRVs and stop valve loop bypass piping.

During full power operation most of the components in the primary side of the RCS are exposed to PWR primary water at temperatures in the range 288°C to 327°C (550-620°F), but some of the pressurizer and pressurizer-piping components are exposed to saturated steam/condensate up to about 343°C (650°F). A number of the reactor vessel internal components (and the belt-line region of the vessel itself) are also exposed to neutron fluxes that can result in moderate or high neutron fluences as high as 80 dpa at 40 years reactor life.

Components on the secondary side of the steam generator may be exposed to temperatures in the range 293°C to 315°C (560-600°F), steam and a variety of aqueous environments depending on the particular water chemistry regime that may have been used at various times in the operating life; in all cases however, aggressive environments can form in some locations such as crevices, heat transfer surfaces, etc.

On the basis of the aggregated scores, it is concluded that there is high degradation susceptibility in some components in the PWR RCS under full power operating conditions. These situations are discussed in detail in Section 3.2.1.1. The dominant degradation modes were stress corrosion cracking and fatigue. High susceptibility to other degradation modes in specific components such as wear of steam generator tubing, irradiation-induced creep of high strength fasteners and flow-accelerated corrosion of carbon steel components were also identified. More general areas of high susceptibility to degradation are listed below. It should be noted that the listing given below is not in any order of priority.

- Stress corrosion cracking of Alloy 82/182 weldments throughout the primary system and especially in the highest temperature components such as the pressurizer.
- Stress corrosion cracking of Alloy 600MA and 600TT steam generator tubes and Alloy 600 forged components and, especially, cold worked or mechanically-strained material such as steam generator tube expansion transitions, small radius U-bends, and the high-temperature components of the pressurizer.

- Corrosion of low-alloy steel components in boric acid concentrate originating from primary water leaks, specifically in the annuli between the Alloy 600 penetrations and the low-alloy steel reactor vessel or pressurizer.
- Irradiation-induced creep and stress corrosion cracking of austenitic stainless steels at more than 0.5 dpa, (e.g., baffle bolts, and other high strength fasteners) and swelling in internals components that reach higher temperatures and much higher doses.
- Fatigue due to unanticipated vibration and thermal fluctuations, e.g., in socket welds (which occur throughout the RCS), primary circuit deadlegs, and baffle bolts after irradiation-induced relaxation of pre-stress.

In addition to the above degradation modes, all of which have been observed in operating plants, other potential degradation of components was identified on the basis of laboratory studies or other industrial experience but for which there is incomplete knowledge of the relevant parameters and dependencies that affect the degradation phenomena. These particular findings are fully discussed in Section 3.2.1.2. Stress corrosion cracking and fatigue issues again dominate. Some more specific areas of potential degradation were also identified such as creep of stainless steel hold-down springs and for erosion-corrosion of high strength ferritic closure bolts. More general areas of potential degradation included the following:

- Although Alloy 690TT has exhibited good resistance to stress corrosion cracking compared with Alloy 600 MA during more than16 years of use in steam generator tubing, some panel members questioned its long term resistance in other PWR primary water applications (e.g., pressure vessel penetrations). The potential for this degradation increases when considering effects of cold work and the use of the alternate welding alloys (Alloys 52 and 152) which present welding challenges associated with, for example, lack of fusion and ductility dip cracking.
- Stress corrosion cracking of severely mechanically-strained stainless steels in PWR primary water, e.g., weld shrinkage strains in heat affected zones, pressurizer heater cladding, and cold-bent piping elbows.
- Cyclic thermal loading in the primary circuit hot leg caused by the difference in temperature between primary coolant streams exiting the center and periphery of low-leakage cores which, despite turbulence, can persist up to the steam generator channel head. While studies have shown that this cyclic loading does not produce significant thermal fatigue issues, the possible effect of ripple loading on other degradation mechanisms (such as stress corrosion cracking) could be a concern.
- Secondary side degradation (e.g., stress corrosion cracking) of steam generator tubes by potentially corrosive species (e.g., lead and low-valency sulfur ions) not previously appreciated to be widespread in the secondary system and which may also affect "more resistant" tubing materials such as Alloy 690.
- Potential degradation related to end of fuel cycle primary water chemistry, especially during cycle stretch-out, when the concentration of boron may be reduced below the minimum recommended value, thereby leading to possible localized corrosion due to excess LiOH, in, e.g., the boiling crevices in the pressurizer.
- Accelerated environmental effects on fatigue of austenitic stainless steels and nickel alloys at low electrochemical corrosion potentials, as demonstrated in the laboratory and therefore anticipated to apply in the field.
- Less-than-anticipated fatigue resistance of some components due to the current uncertainty in the characterization of the cyclic loading conditions, particularly thermal loading, and the material response under those specific conditions.

- Reduced fracture resistance and increased susceptibility to stress corrosion cracking of cast stainless steels (and perhaps ferrite-containing stainless steel cladding and welds) as a result of thermal aging.
- Abusive grinding or machining during weld preparation or surface finishing and their adverse effects on stress corrosion and fatigue crack initiation.

4.1.2 BWR Reactor Coolant System (RCS) under Full Power Operating Conditions

The BWR reactor coolant system (RCS) components include the reactor pressure vessel, the core internals (core-control, shroud, jet pump), ECCS connections, steam dryer/separator, and the recirculation piping. During full power operation these components are exposed to high-purity reactor water and/or wet steam at temperatures in the range 232°C to 288°C (450-550°F). The electrochemical potential of these components depends on, among other things, whether hydrogen injection is used. Therefore, corrosion fatigue and stress corrosion cracking, which are particularly sensitive to electrochemical potential, were assessed separately for normal water chemistry (NWC) and hydrogen water chemistry (HWC). In addition, some of the internal components in reactor vessels are exposed to neutron fluxes that produce moderate fluences (up to ~10 dpa) at the end of the current 40-year plant design life.

On the basis of the aggregated scores, there is high degradation susceptibility under full power operating conditions in the BWR RCS components listed below. These results are fully discussed in Section 3.3.1.1. As in the PWR case discussed earlier, the listing given below is not in any order of priority.

- Stress corrosion cracking of Alloy 82/182 weldments used in joints between ferritic and austenitic components (e.g., thermal sleeves, attachment pads) and particularly Alloy 182 under NWC conditions.
- Stress corrosion cracking of Type 304/316 stainless steel heat affected zones (even under HWC conditions) at moderate fluences.
- Fatigue of Type 304/316 stainless steel in steam dryers under NWC and HWC conditions. This is probably design-specific, but as of now, is not quantified across the whole fleet.
- Stress corrosion cracking of Alloy 600 components and heat affected zones throughout the reactor coolant system, particularly under NWC conditions.
- Stress corrosion cracking of Alloy X750 components throughout the reactor coolant system under both NWC and HWC conditions.
- Fatigue due to unanticipated vibration and thermal fluctuations in socket welds throughout the reactor coolant system.

As concluded above for the PWR RCS, other potential degradation of components was identified for the BWR RCS on the basis of laboratory studies or other industrial experience and for which there is incomplete knowledge of the relevant parameters and dependencies that affect the degradation phenomena. These findings are fully discussed in Section 3.3.1.2.

- Cast stainless steel components (e.g., CF8M brackets and guide rods and CF3 control rod guides), may experience a reduction in fracture resistance as a result of thermal aging. For the most part, however, the BWR RCS components operate at lower temperatures than PWR RCS components and, therefore, possible problems would be expected to become evident in PWRs earlier than in BWRs. Also, the possibility of stress corrosion cracking occurring in cast stainless steel components should not be dismissed without further investigation of the possible synergistic effect of thermal aging on this degradation mode.

- The long term effects of HWC on stress corrosion cracking are not yet clear and it is also not clear that the low potentials needed for mitigation are in fact achieved at all the high-susceptibility locations. The role of hydrogen in the stress corrosion cracking process of austenitic stainless steels is critical but is expected to be less than in high strength nickel-base alloys. The effect of hydrogen on fatigue and its interaction with ripple loading and dynamic strain aging was noted as a potential contributor to degradation under HWC conditions.
- Several panel members suggested that additional fracture resistance tests are needed for all of the austenitic stainless steels and nickel-base alloys, particularly tests in environments simulating the full range of high temperature service environments under both NWC and HWC conditions. Type 308 stainless steel weld metal is not as susceptible to thermal aging as cast austenitic stainless steel and therefore is also less susceptible to reduction of fracture resistance. There could, however, be some adverse effect of wet steam environments on fracture resistance.
- Accelerated environmental effects on fatigue of austenitic stainless steels and nickel alloys at low electrochemical corrosion potentials under HWC conditions. These effects have been demonstrated in the laboratory and therefore are anticipated to apply in the field.
- Stress corrosion cracking of severely cold worked or mechanically-strained stainless steels in BWR water, e.g., weld shrinkage strains in heat affected zones, cold bent elbows.
- Abusive grinding or machining during weld preparation or surface finishing and their adverse effects on stress corrosion and fatigue crack initiation.

4.1.3 LWR Systems other than the RCS under Full Power Operating Conditions

4.1.3.1 PWR Systems

The analyses of PWR systems other than the RCS indicated in general a lower likelihood of degradation primarily because these systems mostly operate at a lower temperature than the RCS. However, as summarized below, the lower operating temperatures plus the existence of different degrees of water chemistry control, led to the identification of degradation modes that are favored under these different operating conditions.

It was concluded that in the ***PWR Emergency Core Cooling System***, the components with a high degree of susceptibility to degradation were far fewer than in the RCS, primarily because these systems operate (apart from subsystems connected directly to the RCS cold and hot legs) at significantly lower temperatures than the RCS. The following component-degradation mode combinations, however, have a high degree of susceptibility (as discussed in detail in Section 3.2.2):

- Stress corrosion cracking of dissimilar metal Alloy 82/182 weldments in piping connected to the RCS cold leg of CE and B&W plants. Such components operate at high temperature (291°C up to 345°C (556-653°F)) in primary water and are judged to have a high susceptibility to cracking. Some of the dissimilar metal weldments in this system are of Type 308/309 stainless steel and were judged to be of much lower susceptibility.

In the lower temperature portions of this system, the following component-degradation mode combinations (as discussed in section 3.2.2.2) were identified on the basis of laboratory studies or other experience and for which there is incomplete knowledge of the relevant parameters and dependencies that affect the degradation phenomena:

- Thermal fatigue in non-isolable deadlegs attached to the primary coolant circuit.
- Fatigue of socket welds throughout the ECCS due to unanalyzed thermal stresses and flow-induced vibrations.
- Stress corrosion cracking of cast stainless steel components in, for example, the RHR pump suction piping.

The ***PWR Steam and Power Conversion Systems*** include the main steam, main feedwater, auxiliary feedwater and steam generator blowdown subsystems. These subsystems are fabricated primarily with carbon and low alloy steel components that are exposed to steam, condensate or demineralized water. The degradation modes of high susceptibility are flow-accelerated corrosion and fatigue of low carbon steel components, welds and heat-affected zones, discussed in section 3.2.3.

The ***PWR Support and Auxiliary Systems*** include the Service Water, Chemical and Volume Control (CVCS), Component Cooling Water (CCW) and Spent Fuel Pool subsystems. These subsystems are fabricated primarily with carbon steels, copper base alloys and stainless steels that are exposed to untreated and treated water or (for the CVCS lines) primary water, all at relatively low temperatures. The degradation modes, such as general corrosion, pitting, stress corrosion cracking, crevice corrosion and microbiologically-influenced fouling and corrosion, are recognized on the basis of extensive experience in the nuclear and other industries. However, predicting their occurrence in advance is still uncertain. Failures in these systems can have a significant impact on the functionality of RCS components. (For instance, chloride contamination of the steam generator system can occur as a result of turbine condenser leakage, etc.) The following high susceptibility component-degradation mode combinations (which are discussed in Section 3.2.4) have been observed in these PWR systems:

- Microbiologically influenced corrosion of carbon steels in service and raw water and on external surfaces of buried piping, which may lead to localized corrosion and penetration of safety-significant components due to pitting, fouling and stress corrosion cracking.
- Pitting and crevice corrosion of buried carbon steel piping and penetrations through concrete, and of Cu-Zn brass (as opposed to Cu-Ni alloy) heat exchanger tubing.
- Fatigue of socket welds throughout the support systems, but especially in the CVCS system due to unanticipated flow assisted vibration.
- Stress corrosion cracking of Cu-Zn brass heat exchanger tubing used in the service pump water discharge piping.

The following component-degradation mode combinations (as discussed in section 3.2.4.2) were identified on the basis of laboratory studies or other experience and for which there is incomplete knowledge of the relevant parameters and dependencies that affect the degradation phenomena:

- Fatigue of austenitic stainless steels.

- Possible reduction in fracture resistance of the constrained region adjacent to welds in stainless steels.
- Stress corrosion of high-strength studs and bolts in the CVCS system.

4.1.3.2 BWR Systems

The analyses of BWR systems other than the RCS were simplified by excluding the following five systems since they were considered to be no different from the corresponding PWR systems discussed above: containment isolation penetrations, spent fuel storage, spent fuel pool cooling and cleanup, service water, and component cooling water.

The ***BWR Engineered Safety Features/Emergency Core Cooling System*** includes the low- and high-pressure core spray lines, the condensate storage tank, and the RHR piping and subsystems. The components with a high degree of susceptibility to degradation were far fewer than in the RCS, primarily because these predominantly carbon and low alloy steel systems operate in condensate cooling water or suppression pool water at significantly lower temperatures (below 38°C (100°F)). However, the generally poor quality water, especially in the suppression pool increases the susceptibility to degradation modes such as microbiologically influenced corrosion and pitting. The assessments of the degree of susceptibility to degradation are covered in detail in Section 3.3.2.1. The following component-degradation mode combinations were found to have a high or medium degree of susceptibility:

- Carbon and low-alloy steel components and welds exposed to suppression pool water are susceptible to crevice corrosion, general corrosion (medium), pitting corrosion, stress corrosion cracking (medium) and microbiologically influenced corrosion (medium).
- Depending on the degree of sensitization, Type 304/316 stainless steel weld heat-affected zones are susceptible to stress corrosion cracking in these systems where it is difficult to maintain HWC as a mitigation measure.
- If they are wetted, carbon steel to brass joints in the drywell are susceptible to galvanic corrosion and crevice corrosion.

The ***BWR Steam and Power Conversion Systems*** include the main steam and feedwater lines, and the main condenser and its associated piping. The components in these subsystems components are fabricated primarily from carbon and low alloy steel, and are exposed to steam, condensate or demineralized water. The following component-degradation mode combinations were found to have high susceptibility:

- Low-alloy steel bolts for the T-quencher/sparger above the suppression pool are susceptible to pitting corrosion and stress corrosion cracking or hydrogen embrittlement. Such degradation has been observed and is exacerbated by the fact that the high strength bolts are exposed to poor quality suppression pool water in conjunction with the use of a zinc primer coating.
- The outside surfaces of titanium condenser tubes are susceptible to droplet erosion if directly exposed to incoming wet steam. This is a design-specific problem that has been mitigated by fabricating the outrow of tubing (i.e., the row that takes the impact) from stainless steel.

Two component-degradation mode combinations were identified on the basis of laboratory studies or other experience and for which there is incomplete knowledge of the relevant parameters and dependencies that affect the degradation phenomena:

- Fatigue of carbon steel weldolets and the possible effect of "ripple loading" on stress corrosion cracking in saturated steam at 286°C (547°F) in the main steam line.
- Fatigue of cast stainless steel venturis in the main steam line due to unanalyzed cyclic loading.

The ***BWR Support and Auxiliary Systems*** are fabricated primarily of carbon steels and stainless steels and are exposed to a variety of environments ranging from wet steam at 286°C (547°F) to suppression pool water at less than 38°C (100°F). The degradation modes, such as general corrosion, pitting, stress corrosion cracking, crevice corrosion and microbiologically-influenced fouling and corrosion, are mostly understood on the basis of extensive experience in the nuclear and other industries. As with the corresponding PWR system, however, failures in these systems can impair the functionality of other important systems or components. The following component-degradation mode combinations (as discussed in Section 3.3.4.1) were found to have high susceptibility:

- Fatigue of carbon steel socket welds in 286°C (547°F) steam.
- Crevice corrosion of ferritic valve components exposed to oxygenated suppression pool water with poor chemistry control.
- Stress corrosion cracking adjacent to Type 304/316 stainless steel weldments in the reactor water clean up piping to pumps with water at 279°C (535°F) under NWC conditions.

The knowledge base for mitigating these known issues is available. However, as discussed in Section 3.3.4.2.1, for components with medium susceptibility to degradation, there was one issue for which it was considered that there was insufficient knowledge: Fatigue of carbon steel components in stagnant wet steam at 286°C (547°F) and in stagnant condensate storage water at less than 38°C (100°F). In both cases cyclic loading would not be expected to be a major problem. However, there was uncertainty concerning the extent of thermal fatigue due to eddies or flow induced vibration in dead legs.

4.1.4 Non-Steady State Operating Conditions in PWRs and BWRs

Non-steady state, or transient, operating conditions have long been associated with increased susceptibility to degradation of various types. The underlying reasons include changing chemical and stressing conditions during reactor start-up, oxygen inleakage, water chemistry transients due to resin bed leakages (especially in the BWR RCS which relies crucially on the maintenance of stringent water chemistry specifications), and relaxed water chemistry control during extended lay-up. During the analyses of the LWR components summarized above, it was assumed that the current operating specifications were being followed and that the potential degradation during these "non-steady state" operating conditions was being adequately controlled and managed. However, it was concluded that there were three situations which merited further attention. One was applicable to both BWRs and PWRs, while the other two were specific to the reactor type:

- An issue common to both BWRs and PWRs was the deposition of chloride contamination on external surfaces originating from aerosols (especially at marine sites), PVC tapes used in NDE, glues etc., which can lead to transgranular stress corrosion initiating on the outside surface of stainless steel components during shutdown conditions. Such phenomena are well known, and guidance in Nuclear Regulatory Commission Regulatory Guide 1.36 specifically addresses the effect of different chloride/silicate combinations in the insulation on cracking susceptibility. The reason why this is highlighted is that there is the possibility that, in order to mitigate the problem of PWR sump screen blockage, some utilities might opt for removing fibrous calcium silicate (CalSil) insulation. Although such an action may well lessen the sump screen blockage problem, it may, without sufficient analysis, reintroduce the older problem of pitting and transgranular stress corrosion cracking of stainless steel piping.

- A potential issue that, at present, is specific to nickel-base components in the PWR primary circuit is the susceptibility of some of the nickel-base alloys to hydrogen embrittlement at temperatures below 200°C (392°F). This leads to reduced fracture resistance and accelerated subcritical crack growth under temperature and strain rate conditions that may exist during transient operating conditions (specifically when shutting down the plant). Elements of these issues, which are discussed in Appendix B.13, have been demonstrated in the laboratory, but no occurrences have yet been observed in operating plants. There is some concern that the same vulnerability may exist for some nickel-base alloy components in the BWR RCS but, as yet, no laboratory tests have been conducted under BWR conditions. It has also been hypothesized that a similar susceptibility could exist for thermally aged cast austenitic stainless steel (CASS) and/or sensitized stainless steels under either (or both) PWR and BWR conditions.

- As pointed out above, the integrity of BWR RCS components is dependent on the control of impurity contents in the coolant and their effect on stress corrosion cracking, corrosion fatigue and irradiation-assisted stress corrosion cracking. It is for this reason that there are prescriptive water chemistry purity control guidelines, described in Appendix B.10, with action responses should these guidelines be exceeded. The judgments made in the present study have been made on the assumption that the chemistry guidelines are followed for both "Normal Water Chemistry" and "Hydrogen Water Chemistry/NobleChem™" operating modes. If these guidelines are exceeded during non-steady state conditions, then many of the judgments of "low cracking susceptibility" made in this study would automatically change to "high cracking susceptibility." Fortunately the knowledge basis for managing most of these situations is high enough that "off chemistry" time limitations are defined in the water chemistry guidelines. The one situation which is currently in question, however, and which is discussed in Appendix B.8, is the increase in stress corrosion cracking susceptibility in ferritic piping and pressure vessel steels associated with relatively minor chloride transients and dynamic loading.

4.2 Generic Materials Degradation Issues

It was concluded that there are several topics that can benefit from further research that cut across the component-specific issues summarized above. A good example of such a topic would be the need for a quantitative understanding of the effects of weld metallurgy on the degradation phenomena. This is a wide-ranging issue covering, for instance, the size and distribution of weld defects, compositional and microstructural inhomogeneities, residual stress profiles, and how these alter with welding parameters (weld heat input, number of passes, preheat temperature, etc.). All of these variables can affect the susceptibility to stress corrosion cracking in a given component.

The listing given below is undoubtedly incomplete but covers topics that were most often discussed at the panel meetings, and which are important to the objective of achieving a proactive materials degradation management capability. The technical details behind these generic issues were fully discussed in Section 3.4, and were categorized in terms of research in (a) damage assessment, (b) fracture assessment and, (c) the margin between the extent of damage and the failure criterion.

4.2.1 Damage Assessment

- A treatment of the sequence of cracking damage accumulation due to localized corrosion (intergranular attack, pitting, etc.), microcrack initiation, and crack coalescence, followed by crack propagation.

- An understanding of the "precursor" conditions required for the onset of modes of aggressive corrosion such as stress corrosion cracking.

- An understanding of the influence of uncertainty in the magnitude of physical parameters associated with the distribution of failure times (e.g., as predicted by the Weibull distribution).

- A review of the adequacy of quantitative life prediction models for "long" (or "deep") crack propagation.

- An adequate definition of the "corrosion system" parameters that control the kinetics of environmentally assisted degradation.

- A clarification of the completeness of identification of degradation modes and synergies associated with combinations or sequences of corrosion and mechanical stresses. This interest reflects on the fact that changes in the system conditions may alter the details of the degradation mechanism upon which the life prediction methodology is based and, thereby, impacts on the ability to identify currently unforeseen component degradation. For instance the kinetics of cracking in cold worked components may be increased due to absorbed hydrogen introduced by flow-accelerated corrosion on an adjacent region.

4.2.2 Fracture Assessment

- An evaluation of the validity and/or the limitations on the use of fracture resistance values measured in air compared to those measured in the operating aqueous environment.

- The applicability of the "hydrogen embrittlement" phenomenon discussed in the component specific section above for PWR primary system nickel-alloy components to other reactor systems (such as BWRs on hydrogen water chemistry) and other materials. Concerns are primarily for the low temperature fracture resistance of high strength Ni-base alloys, cold worked materials, thermally aged duplex cast and sensitized stainless steels.

4.2.3 Margin Assessment

- The incorporation of material degradation into probabilistic risk assessments (PRAs).
- Inspection reliability. Assessment of this topic was outside the scope of the panel's task. However, during the panel discussions, it was reiterated that there must be adequate inspection capabilities, especially for the complex geometries associated with reactor pressure vessel and pressurizer penetrations. Use of continuous in-situ monitoring capabilities should also be encouraged given the fact that the development of degradation may occur at changing rates depending on the specific plant operating conditions, even for the same material under similar operating conditions.

4.3 The Next Step: Proactive Materials Degradation Management

A further generic issue discussed in Section 3.4 *"Generic Materials Degradation and Life Management Issues"* is the ability to transition from reactive to proactive management of materials degradation at the NRC, the U.S. industry, or their international counterparts. Technically-driven management of materials degradation in LWRs has been part of the professional life of all the panel members involved in this PMDA–their observations, conclusions and recommendations on this topic are summarized as follows:

- Materials degradation will continue in LWRs, and may increase with license renewal and power uprates. Alternate materials, modified operating conditions, etc. may counteract these factors but, generally, they are not fully qualified and address only a fraction of the degradation modes. The technical reasons for these statements are outlined in detail in this PMDA report. Thus it is concluded that a Proactive Materials Degradation Management (PMDM) phase is needed.[5]
- Materials degradation typically results from complex phenomena involving metallurgy, electrochemistry, mechanics, radiation damage, physical chemistry, etc. and requires extraordinary experimental sophistication to resolve the interdependencies. This range of expertise rarely exists in any one organization. Thus the development of a PMDM capability would benefit from collaboration between various organizations.
- Prioritization, based on the consequences of degradation, of the proactive tasks on materials degradation management is essential.
- Adequate resources are needed to develop and maintain technical expertise and experimental capability. This applies to all the relevant organizations: reactor designers, regulators, National Laboratories, etc. This seems obvious but in spite of the significant impact of materials degradation over the last 30 years, the level of funding, the available expertise and up-to-date experimental facilities have all decreased. Most key experts

[5] The NRC staff believes that the changed operating conditions due to power uprates are not expected to appreciably reduce the effectiveness of licensees' aging management programs (AMPs) for currently identified mechanisms. For those plants requesting license renewal, the NRC assures that the licensees have developed effective AMPs focused on these mechanisms for the period of extended operation. Nonetheless, proactive materials degradation management may be useful to predict and manage degradation issues not previously encountered and to anticipate degradation of materials of geometries and in locations not previously shown to be susceptible to a particular aging mechanism. This approach could be used to augment the AMPs currently relied on during the practice of inspection, identification, and subsequent management, if warranted.

are now close to, or are in, retirement. As a consequence, the resources that are available have been concentrated on short term "firefighting" projects and the longer term research and development projects have been delayed or cancelled. This leaves the reactor community vulnerable to a permanent loss of accumulated knowledge and expertise, and with a largely un-mentored workforce with less than five to ten years experience. This is an inadequate basis for addressing the complex questions that need to be answered. *It is imperative that these resource issues be addressed worldwide by governmental organizations, utilities, vendors and support organizations, and universities and National Laboratories.*