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Expanded Materials Degradation Assessment (EMDA)

Volume 1: Executive Summary of EMDA Process and Results

Office of Nuclear Regulatory Research

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NUREG/CR-7153, Volume 1 – Expanded Materials Degradation Assessment (EMDA): Executive Summary of EMDA Process and Results

NUREG/CR-7153, Volume 4 – Expanded Materials Degradation Assessment (EMDA): Aging of Concrete and Civil Structures

Page 26 in Volume 1 and Page 86 in Volume 4 incorrectly state that there is operational experience indicating the occurrence of alkali-silica reactions in the containment at the Davis Besse Nuclear Power Station. The correct plant should be Seabrook Nuclear Power Plant.

The corrected sentences should read as follows:

Though this degradation is well documented by the operating experience (for bridges and dams in particular) and scientific literature, its high ranking in the EMDA analysis describes the need to assess its potential consequences on the structural integrity of the containment, considering the recent operating experience at Seabrook and other plants.

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ABSTRACT

In NUREG/CR-6923, “Expert Panel Report on Proactive Materials Degradation Assessment,” referred to as the PMDA report, NRC conducted a comprehensive evaluation of potential aging-related degradation modes for core internal components, as well as primary, secondary, and some tertiary piping systems, considering operation up to 40 years. This document has been a very valuable resource, supporting NRC staff evaluations of licensees’ aging management programs and allowing for prioritization of research needs.

This report describes an expanded materials degradation assessment (EMDA), which significantly broadens the scope of the PMDA report. The analytical timeframe is expanded to 80 years to encompass a potential second 20-year license-renewal operating-period, beyond the initial 40-year licensing term and a first 20-year license renewal. Further, a broader range of structures, systems, and components (SSCs) was evaluated, including core internals, piping systems, the reactor pressure vessel (RPV), electrical cables, and concrete and civil structures. The EMDA uses the approach of the phenomena identification and ranking table (PIRT), wherein an expert panel is convened to rank potential degradation scenarios according to their judgment of susceptibility and current state of knowledge. The PIRT approach used in the PMDA and EMDA has provided the following benefits:

- Captured the status of current knowledge base and updated PMDA information,
- Identified gaps in knowledge for a SSC or material that need future research,
- Identified potential new forms of degradation, and
- Identified and prioritized research needs.

As part of the EMDA activity, four separate expert panels were assembled to assess four main component groups, each of which is the subject of a volume of this report.

- Core internals and piping systems (i.e., materials examined in the PMDA report) – Volume 2
- Reactor pressure vessel steels (RPV) – Volume 3
- Concrete civil structures – Volume 4
- Electrical power and instrumentation and control (I&C) cabling and insulation – Volume 5

Factors considered in the assessment included the reactor environment, existing operational experience and laboratory data and models of materials behavior. Each separate assessment provided an analysis of key degradation modes for current and expected future service, key degradation modes expected for extended service, and suggested research needs to provide technical information for operation up to 80 years. The scope, background, and analysis for each of these four areas are described in detail in other companion volumes. This volume provides a summary of the findings in each panel.

FOREWORD

According to the provisions of Title 10 of the *Code of Federal Regulations* (CFR), Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," licensees may apply for twenty-year renewals of their operating license following the initial forty-year operating period. The majority of plants in the United States have received the first license renewal to operate from forty to sixty years and a number of plants have already entered the period of extended operation. Therefore, licensees are now assessing the economic and technical viability of a second license renewal to operate safely from sixty to eighty years. The requirements of 10 CFR, Part 54 include the identification of passive, long-lived structures, systems, and components which may be subject to aging-related degradation, and the development of aging management programs (AMPs) to ensure that their safety function is maintained consistent with the licensing basis during the extended operating period. NRC guidance on the scope of AMPs is found in NUREG-1800 "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR) and NUREG-1801, "Generic Aging Lessons Learned (GALL) Report."

In anticipation of reviewing applications for reactor operation from sixty to eighty years, the Office of Nuclear Reactor Regulation (NRR) requested the Office of Nuclear Regulatory Research (RES) to conduct research and identify aging-related degradation scenarios that could be important in this timeframe, and to identify issues for which enhanced aging management guidance may be warranted while allowing for prioritization of research needs. As part of this effort, RES agreed to a Memorandum of Understanding with the U.S. Department of Energy (DOE) to jointly develop an Expanded Materials Degradation Assessment (EMDA) at Oak Ridge National Laboratory (ORNL). The EMDA builds upon work previously done by RES in NUREG/CR-6923, "Expert Panel Report on Proactive Materials Degradation Assessment." Potential degradation scenarios for operation up to forty years were identified using an expert panel to develop a phenomena identification and ranking table (PIRT). NUREG/CR-6923 mainly addressed primary system and some secondary system components. The EMDA covers a broader range of components, including piping systems and core internals, reactor pressure vessel, electrical cables, and concrete structures. To conduct the PIRT and to prepare the EMDA report, an expert panel for each of the four component groups was assembled. The panels included from 6 to 10 members including representatives from NRC, DOE national laboratories, industry, independent consultants, and international organizations. Each panel was responsible for preparing a technical background volume and a PIRT scoring assessment. The technical background chapters in each volume summarizes the current state of knowledge concerning degradation of the component group and highlights technical issues deemed to be the most important for subsequent license renewal.

The subject of the present volume is an executive summary of the specific PIRT process, results, and any knowledge gaps identified in the respective PIRT assessments. These detailed backgrounds discussions, PIRT findings, assessments and comprehensive analysis for each of these component groups are presented in the companion volumes.

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ABBREVIATED TERMS

%, percent

°C, degrees Celsius

°F, degrees Fahrenheit

γ, gamma

γ', gamma prime

Δ, delta; denotes change

$\Delta\sigma_y$, change in yield strength

σ, sigma; denotes variability

τ, UMD recovery time

ϕ, flux

ϕ**t**, fluence

$<T_{dam}>$, total average damage energy per atom

0.5T, ½T compact tension specimen

1TC(T), 1T compact tension specimen

3/4-t, three-quarters of the way through the vessel

3DAP, three-dimensional atom probe

41J, 41 joules (absorbed energy level in which Charpy v-notch specimen reaches the ductile-to-brittle transition temperature)

AAR, alkali-aggregate reaction

ADP, annealing demonstration project

AERE, Atomic Energy Research Establishment (UK)

AFCEN, French Society for Design and Construction and In-Service Inspection Rules for Nuclear Islands

AMP, aging management program

AMR, aging management review

ANO-1, Arkansas Nuclear One Unit 1

APT, atom probe tomography

ASME, American Society of Mechanical Engineers

ASTM, American Society for Testing and Materials

at %, atomic percent

ATI, ATI Consulting

ATR, Advanced Test Reactor

B&W, Babcox and Wilcox

BAC, boric acid corrosion

BR3, Belgian reactor 3

BWR, boiling water reactor

C, carbon

C&LAS, carbon and low alloy steels

CASS, cast austenitic stainless steel

CFR, *Code of Federal Regulations*

Cl$^-$, chloride ion

cm, centimeter

Cr, chromium

CR, cold rolled

CRD, control rod drive

CRDM, control rod drive mechanism

CREEP, thermal creep

CREV, crevice corrosion

CRIEPI, Central Research Institute of Electric Power Industry (Japan)

CRP, Cu-rich precipitates

Cu, copper

CUF, cumulative fatigue usage factor

CVCS, chemical and volume control system

CVN, Charpy V-notch

CW, cold-worked

DBTT, ductile-to-brittle transition temperature

DEBOND, debonding

DH, dissolved hydrogen

DOE, U.S. Department of Energy

dpa, displacements per atom

E, neutron spectrum flux

EBSD, electron backscatter diffraction

EC, erosion–corrosion

ECCS, emergency core cooling system

ECP, electric chemical potential

E_d, displacement threshold energy

EDF, Electricite de France

EDS, energy-dispersive X-ray spectroscopy

EK, Erickson Kirk

Emb., Embrittlement

EMDA, Extended Materials Degradation Assessment

Env., environmental

EONY, Eason, Odette, Nanstad, and Yamamoto

EPMDA, Extended Proactive Materials Degradation Assessment

EPR, electrochemical potentiokinetic reactivation

EPRI, Electric Power Research Institute

eV, electron volt

FAC, flow-accelerated corrosion

FAT, corrosion fatigue

Fe, iron

f_p, volume fraction

FR, fracture resistance

GALL, generic aging lessons learned

GALV, galvanic corrosion

GC, general corrosion

h, hour

HAZ, heat-affected zone

HC, high cycle

HSSI, Heavy-Section Steel Irradiation

HSST, Heavy Section Steel Technology

HWC, hydrogen water chemistry

HWR, heavy water reactor

I&C, instrumentation and controls

IA, irradiation assisted

IAEA, International Atomic Energy Agency

IASCC, irradiation-assisted stress corrosion cracking

IC, irradiation creep

IG, intergranular

IGC, intergranular corrosion

IGF, intergranular fracture

IGSCC, intergranular stress corrosion cracking

IMP, Implementation

IMT, Issue Management Table

in., inch

INL, Idaho National Laboratory

IPA, integrated plant assessment

IVAR, irradiation variables

JAEA, Japan Atomic Energy Agency

JAERI, Japan Atomic Energy Research Institute

JMTR, Japan Materials Testing Reactor

JNES, Japan Nuclear Safety Organization

JPDR, Japan Power Demonstration Reactor

K, stress intensity

keV, thousand electron volt

K_{Ia}, crack-arrest toughness

K_{Ic}, fracture toughness

K_{Jc}, elastic-plastic fracture toughness at onset of cleavage fracture

LAS, low alloy steel

LBP, late-blooming phase

LC, low cycle

LMC, lattice Monte Carlo

LRO, long-range ordering

LTCP, low-temperature crack propagation

LTO, long-term operation

LWR, light water reactor

LWRS, Light-Water Reactor Sustainability

LWRSP, Light Water Reactor Sustainability Program

MA, mill-anneal

MDM, materials degradation matrix

MeV, million electron volts

MIC, microbially induced corrosion

MF, matrix feature

MIG, metal inert gas (welding)

Mn, manganese

MO, Mader and Odette

Mo, molybdenum

MOU, memorandum of understanding

MOY, Mader, Odette, and Yamamoto

MPa√m, stress intensity factor; fracture toughness in units of megapascal square root meter

MPC, Materials Properties Council

n/cm², fluence

n/cm²·s, flux

NE, DOE Office of Nuclear Energy

NEI, Nuclear Energy Institute

Ni, nickel

NMCA, noble metal chemical addition

NOSY, Nanstad, Odette, Stoller, and Yamamoto

NPP, nuclear power plant

NRC, U.S. Nuclear Regulatory Commission

NWC, normal water chemistry

ORNL, Oak Ridge National Laboratory

P, phosphorous

PA, proton annihilation

PIA, postirradiation annealing

PIRT, phenomenon identification and ranking technique

PIT, pitting

PLIM, Nuclear Power Plant Integrity Management

PMDA, Proactive Materials Degradation Assessment

PMMD, proactive management of materials degradation

PNNL, Pacific Northwest National Laboratory

PRA, primary recoil atom

PRE, Prediction of Radiation Embrittlement

PREDB, Power Reactor Engineering Database

PSF, Poolside Facility

PT, penetration test

PTS, pressurized thermal shock

PWHT, post-weld heat treatment

PWR, pressurized water reactor

PWROG, Pressurized Water Reactor Owners Group

PWSCC, primary water stress corrosion cracking

R&D, research and development

RADAMO, SCK-CEN TR model and corresponding TR database

RCS, reactor coolant system

RES, NRC Office of Nuclear Research

RHRS, residual heat removal system

RIS, radiation-induced segregation

RPV, reactor pressure vessel

RSE-M, Rules for In-Service Inspection of Nuclear Power Plant Components (France)

RT, reference temperature

SA, solution anneal

SANS, small-angle neutron scattering

SCC, stress corrosion cracking

SCK-CEN, Studiecentrum voor Kernenergie—Centre d'Etude de l'Énergie Nucléaire (Belgian Nuclear Research Centre)

SE(B), single-edge, notched bend

SEM, scanning electron microscopy

SG, steam generator

SIA, self-interstitial atom

SIS, safety injection system

SM, Stationary Medium Power

SMF, stable matrix feature

SR, stress relaxation

SS, stainless steel

SSC, system, structure, and component

SSRT, slow strain rate test

SW, swelling

T_0, fracture toughness reference temperature

T_{41J}, ductile-to-brittle transition temperature measured at 41 joules of Charpy impact energy

TEM, transmission electron microscopy

TG, transgranular

Th, thermal

T_i, irradiation temperature

TIG, tungsten inert gas (welding)

TiN, titanium nitride

TLAA, time-limited aging analysis

TMS, The Minerals, Metals and Materials Society

TR, test reactor

TT, reference transition temperature; thermal treatment

TTS, transition temperature shift

UCSB, University of California, Santa Barbara

UK, United Kingdom

UMD, unstable matrix defect

UNS, Unified Numbering System

U.S., United States

USE, upper-shelf energy

UT, ultrasonic test

VS, void swelling

VVER, Voda-Vodyanoi Energetichesky Reaktor (Water-Water Energetic Reactor)

WEAR, fretting/wear

Wstg., wastage

wt %, weight percent

Zn, zinc

1. INTRODUCTION

Nuclear reactors present a very harsh environment for components service. Components within a reactor core must tolerate high temperature water, stress, vibration, and an intense neutron field. Degradation of materials in this environment can lead to challenges in required performance, and in some cases, sudden failure. Materials degradation phenomena within a nuclear power plant are very complex. There are many different types of materials that make up different components: over 25 different metal alloys can be found within the primary and secondary systems, not to mention the concrete containment vessel, instrumentation and control, and other support facilities. When this diverse set of materials is placed in the complex and harsh environment coupled with varying types of loading, degradation over an extended life is indeed quite complicated. Clearly, materials degradation could potentially impact the safe operation of a reactor. Routine surveillance and component replacement can mitigate these factors, although failures can still occur. While all components can, in theory be replaced, it may not be practical or economically favorable. Therefore, understanding, controlling, and mitigating materials degradation processes are key priorities for extending the reactor operating life.

According to the provisions in Title 10 of the Code of Federal Regulations (10 CFR), Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” plant-operating licenses may be renewed for periods of 20 years after the initial 40-year licensing term. The licensee must provide reasonable assurance that the plant-licensing basis will continue into the extended operating period. An integral part of ensuring that the plant can continue to operate safely is demonstrating that the effects of aging-related degradation on structures, systems, and components (SSCs) are well understood and can be adequately managed. As necessary, this includes the implementation of aging management programs (AMPs), which may involve inspection and mitigation approaches for affected SSCs, among other strategies. Likewise, time-limited aging analyses (TLAAs), which involve assumptions about the plant operating life, such as fatigue calculations, are reassessed for the extended operating period.

The majority of U.S. plants have received a first license renewal to operate for up to 60 years and some have entered the extended operating period. At present, industry is considering the feasibility of pursuing subsequent license renewal to operate from 60 to 80 years. While applications for subsequent license renewal may not be prepared for several years, both NRC and DOE have an interest in acting proactively to identify issues that may affect the ability of plants to operate for up to 80 years. Nuclear Regulatory Commission (NRC) must provide guidance to applicants on the expected contents of a subsequent license renewal application and develop the technical bases for making safety determinations in the license review. Through the Light Water Reactor Sustainability (LWRS) Program, the Department of Energy (DOE) undertakes research to understand the fundamentals of component aging, thereby supporting industry in sustaining the domestic fleet as an economic and strategic resource. Given the common interests, NRC and DOE have put in place a Memorandum of Understanding (MOU) to cooperate on research activities related to long-term operations. One activity initiated under the MOU is the Expanded Proactive Materials Degradation Analysis (EMDA), which is the subject of the present report.

1.1 BACKGROUND

To address aging-related degradation, NRC has taken the approach of attempting to proactively identify scenarios that could affect reactor components so that appropriate mitigation actions could be taken before plant safety was compromised. An important part of this initiative was undertaken in the timeframe of 2004 to 2005 with NUREG/CR-6923, “Expert Panel Report on Proactive Materials Degradation Assessment,” referred to as the PMDA report [1]. The PMDA report was conceptualized as an expert elicitation process to identify degradation processes that could affect mainly primary, secondary, and some tertiary plant systems for operation up to 40 years. The PMDA report followed the phenomena identification and ranking technique (PIRT) process, wherein degradation scenarios (i.e., system, material, operating environment, degradation mechanism) were ranked according to the probability of occurrence, level of knowledge concerning that process, and confidence in scores. Over 3,000 scenarios were scored. It was intended that the outcomes of the PMDA report could be used to prioritize research and identify potential gaps for which enhanced regulatory guidance may be needed.

The degradation scenarios most prominently highlighted in the PMDA report were those classified as high probability of occurrence but low knowledge. For pressurized water reactors (PWRs), these included fatigue for socket welds, stress corrosion cracking (SCC) for dissimilar metal welds, and radiation effects for stainless steels. The only boiling water reactor (BWR) scenario in this category was SCC of a low-alloy steel bolt in the main steam system. Given lesser prominence were scenarios classified as high susceptibility but high knowledge. For PWRs, these included fatigue and flow-accelerated corrosion (FAC) of carbon steel components in secondary systems, SCC of Alloy 600 steam generator tubes, and microbially induced corrosion (MIC) or pitting of carbon steel service water piping. For BWRs, these included crevice corrosion, MIC, and pitting carbon steel in the condensate storage tank and service water piping, SCC of Alloy 600/82/182 thermal sleeves, nozzles, safe ends, and attachment pads, and SCC of austenitic stainless steel welds for core internal components. Finally, intermediate susceptibility and low knowledge degradation scenarios for PWRs included fatigue or SCC of austenitic stainless steel, including for internal components and piping systems. For BWRs, most of these related to loss of fracture resistance in austenitic stainless steel or Alloy 82/182 welds. Other degradation scenarios were ranked as lower susceptibility and/or higher knowledge.

The findings from the PMDA report were used as inputs to develop the most recent revision of NUREG-1801, “Generic Aging Lessons Learned (GALL) Report, published in 2010 [2]. Aging Management Programs (AMPs) found in the GALL Report were identified to address the significant degradation scenarios identified in the PMDA report. Moreover, it was recognized that most of the components susceptible to degradation are inspected as part of the in-service inspection programs found in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, associated code cases, or industry guidance. The scenarios in the PMDA also align well with NRC research priorities, as there are established research programs evaluating such issues as irradiation effects on stainless steels, primary water stress corrosion cracking (PWSCC) of Ni alloy welds, and environmentally assisted fatigue

Given the utility of the PMDA report, DOE and NRC staff recognized that a similar approach could be used to support the development of technical bases for subsequent license renewal, therefore motiving the idea of the EMDA. EMDA represents a significant broadening of scope relative to PMDA. First, the analytical timeframe is extended from 40 years to 80 years, encompassing the subsequent license renewal-operating period. Second, the materials and systems addressed in EMDA are generally extended to all of those which fall within the scope of aging management review for license renewal. Thus, in addition to piping and core internals, EMDA also includes the reactor pressure vessel (RPV), electrical cables, and concrete structures. Given the time and financial resources that were anticipated for such an undertaking, as well as the mutual interest in the findings, NRC and DOE agreed that this would be a suitable activity to pursue under the research MOU.

A diverse expert panel was assembled for each of the four assessments. Each panel was composed of at least one member representing the regulator, industry [e. g., the Electric Power Research Institute (EPRI), vendors], the U.S. national laboratories, academia, and an international aging degradation expert.

1.2 SUMMARY OF POTENTIAL MATERIALS DEGRADATION ISSUES FOR LONG TERM OPERATION

Components serving in a nuclear reactor power plant must withstand a very harsh environment including extended time at temperature, neutron irradiation, stress, and/or corrosive media. The many modes of degradation are complex and vary depending on location and material. However, understanding and managing materials degradation is a key for the continued safe and reliable operation of nuclear power plants. Extending reactor operation to beyond 60 years will increase the demands on materials and components. While operation beyond 60 will add additional time and neutron fluence, the primary impact will be increased susceptibility to known degradation modes, although new mechanisms are possible.

For the reactor core and primary systems, several key issues have been identified. Thermo-mechanical considerations such as aging and fatigue must be examined. Irradiation-induced processes must also be considered for higher fluences, particularly the influence of radiation induced segregation (RIS), swelling, and/or precipitation on embrittlement. Corrosion takes many forms within the reactor core and piping systems, although irradiation assisted stress corrosion cracking (IASCC) and PWSCC are of high interest in extended life scenarios. Research in these areas can build upon other ongoing programs in the light water reactor (LWR) industry as well as other reactor materials programs (such as fusion and fast reactors) to help resolve these issues for extended LWR life. In the secondary systems, corrosion is extremely complex. Understanding the various modes of corrosion and identifying mitigation strategies is an important step for long-term service.

For reactor pressure vessels, a number of significant issues have been identified for future research. Relatively sparse or nonexistent data at high fluences, for long radiation exposure (duration), and resulting high embrittlement create large uncertainties for embrittlement predictions. The use of test reactors at high fluxes to obtain high fluence data is not the most direct representation of the low flux conditions in RPVs. Late-blooming phases (LBPs), especially for high nickel welds, have been observed and additional experimental data are needed in the high fluence regime where they are expected. Other discussed issues include specific needs regarding application of the fracture toughness master curve, data on long term thermal aging, attenuation of embrittlement through the RPV wall, and the development of an embrittlement trend curve based on fracture toughness measurements.

Concrete structures can also suffer undesirable changes in properties with time, including adverse performance of its cement paste matrix or aggregate constituents under environmental influences (e.g., physical or chemical attack). Changes to embedded steel reinforcement as well as its interaction with concrete can also be detrimental to concrete’s service life. Aging effects can be exacerbated if improper concrete specifications were used at the time of construction. A number of areas of research would help assess the long-term integrity of the reactor concrete structures.

Cable and cable insulation systems play an important role in the safety and operation of a nuclear power plant. Degradation of polymer insulation due to the combined effects of mechanical stress, elevated temperature, irradiation and high humidity environments (or complete submergence) has been observed, although there may be knowledge gaps for reactor long term operation.

The companion volumes to this executive summary provide much greater detail for each of these major material systems.

1.3 DESCRIPTION OF THE PIRT PROCESS

The expert elicitation process for the EMDA is based on the same PIRT process that was employed for PMDA. This process has been used in many industries for ranking and prioritizing any number of issues. The PIRT process provides a systematic means of obtaining information from experts and involves generating lists (tables) of phenomena where "phenomena" can 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, which is related to reactor safety. The process usually involves ranking of these phenomena using a series of scoring criteria. The results of the scoring can be assembled to lead to a quantitative ranking of issues or needs.

Each PIRT application has been unique in some respect and the current project is unique in its application. The current PIRT can be described in terms of several key steps. These are described for the generic process below, although each panel made minor adjustments, based on the needs of that material system and the operational environment and expected interactions. Such adjustments are described below.

As part of this activity, four expert panels were assembled to evaluate each of the four main component groups: core internals and piping systems, the RPV, electrical cables, and concrete and civil structures, respectively. Each panel included 8–10 leading experts with a variety of perspectives (including regulatory, academia, industry, and international experience). To ensure a diverse set of background and expertise, each panel was assembled to ideally include

- At least one member from regulatory bodies, including the U.S. NRC
- At least two members representing industry (Electric Power Research Institute [EPRI], vendors, etc.)
- At least one member from the U.S. DOE national laboratories
- At least one member from academia
- At least two members from outside the United States

Members from non-nuclear fields were also selected for the concrete and civil structure panel. The NRC and DOE cooperatively selected and assembled the various panels.

The EMDA report volume for each component group consists of a technical background assessment to summarize the current state of knowledge concerning the relevant degradation scenarios, and then the PIRT scoring and analysis. Ideally, the technical background assessments provide the context and rational for which scenarios were scored and how they were ranked. For the core internals and piping volume, the existing PMDA report was used as a starting point for identifying important degradation scenarios and additional discussion focused on the potential changes that might be experienced during subsequent operating periods. For the other volumes, given that there was no preexisting PIRT, the latest technical literature was reviewed, and experts used their judgment to identify the important degradation scenarios. Generally, one panel member was assigned to write each chapter of the technical background assessment, which may focus on a particular material or degradation mode, after which the chapters were peer reviewed by the entire panel. Subsequent discussion amongst the entire panel was also used to identify key themes and revisions to the technical background assessments were made accordingly. These assessments are listed as the opening chapters of each volume in the EMDA. It is important to note that these background assessments are not intended to be all-encompassing primers on particular degradation modes or material systems. Detailed information and background assessments exist in other publications and it is beyond the scope of this project to reproduce them here. Rather, the discussions presented are intended to introduce the subject and context for the evaluation of key modes of degradation for subsequent operating periods. The reader is referred to the publications listed in the background Chapters of each Volume for more in-depth technical information.

Based on the input from the technical background volume, the panels then developed a PIRT matrix with a list of degradation scenarios to score. A degradation scenario generally encompasses a particular material, system, component, or subcomponent (depending on the categorization scheme devised by the panel), the environmental condition to which that material is exposed, and the degradation mode, which that material may experience, based on laboratory and operational data. It was recognized that such data do not exist for reactor operational periods beyond 40 or so years, thus posing a considerable challenge for the expert panels to extrapolate reactor operation for greater than 60 years. Some materials are used in different components and experience different environments or may experience multiple degradation modes in a single location. Each material, environment, degradation mode is scored as a distinct scenario. The number of degradation scenarios varies widely by component group, from less than 50 for the cables to over a 1,000 for the piping and core internals.

After the scoring matrix was developed, panelists independently scored the degradation scenarios in three categories that were originally used in the PMDA report: Susceptibility, Confidence, and Knowledge. The Susceptibility score rates the likelihood that degradation will occur, on a scale from 0 (not considered to be an issue) to 3 (demonstrated, compelling evidence for occurrence, or multiple plant observations). The Knowledge score rates the expert's current belief of how adequately the relevant dependencies have been quantified through laboratory studies and/or operating experience, on a scale from 1 (poor understanding, little and/or low-confidence data) to 3 (extensive, consistent data covering all dependencies relevant to the component). Finally, the Confidence score measures the expert's *personal* confidence in his or her judgment of Susceptibility, on a scale from 1 (low) to 3 (high).

After completion of scoring and identification of "outliers," the panels were reassembled for discussion of the scoring. In most panels, this was done in a face-to-face meeting, but this was not required in all cases. During this discussion, each degradation mode and related scoring was discussed with the "outliers" being of highest priority. In these discussions, the scoring panelist presented rationale for any scores that differed from the average. The objective was not to develop a consensus score or force conformity among the panelists. The primary goal of this discussion was to foster debate and exchange differing points of view. This debate and discussion among panelists was an important part of the process to ensure all points of view were considered, including consideration of any new information on the subject area which was not previously considered, and accounted for in the final scoring. After compiling any changes in scoring following this debate, the PIRT scoring was tabulated to determine relative needs and priorities.

After compiling any changes in scoring following this debate, the PIRT scoring was tabulated to determine relative needs and priorities. In this process, the average Susceptibility and average Knowledge scores were plotted versus each other on a simple plot. An example plot of Knowledge versus Susceptibility is shown in Figure 1.1. The left side of the plot with the lighter shading is indicative of low Knowledge, while the darker shading on the right side of the plot is indicative of high Knowledge. The labeled areas in the corners of the plot indicate the high Knowledge, low Susceptibility; high Knowledge, high Susceptibility; and low Knowledge, high Susceptibility areas discussed above. Moving from upper right to lower left can be accomplished via additional research and development to understand and predict key forms of degradation.

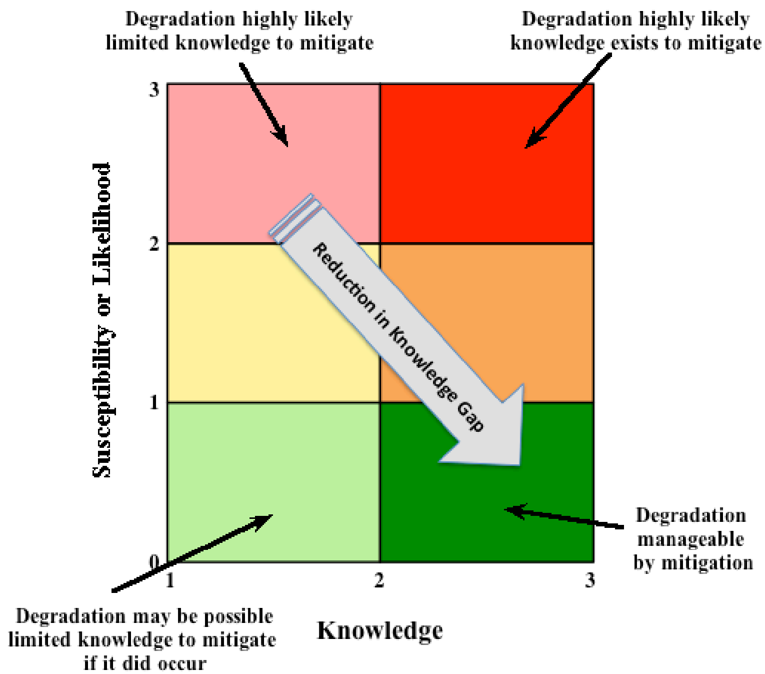

Figure 1.1. Schematic illustrating the combinations of Susceptibility and Knowledge scores suggesting various life management responses.

The different domains of these plots highlight key areas of concern, including:

- Low Knowledge, high Susceptibility degradation modes are indicated by the pink shading in Figure 1.1 and represent modes of degradation that could be detrimental to service with high Susceptibility scores (>2) and low Knowledge scores (<2). These scores indicate gaps in understanding for degradation modes that have been demonstrated in service. Low Knowledge and moderate Susceptibility also indicate gaps in knowledge, although with lower consequences. These scoring regions are useful in identifying potential knowledge gaps and areas requiring further research into mechanisms and underlying causes to predict occurrence.

- High Knowledge, high Susceptibility degradation modes are shown in red in Figure 1.1 and represent areas that could be detrimental to service with high Susceptibility scores (>2) and high Knowledge scores (>2). These modes of degradation are well understood and have likely been observed in service. While there may be some mechanistic understanding of the underlying causes, re-confirmation for extended service and research into mitigation or detection technologies may be warranted.

- High Knowledge, low Susceptibility degradation modes (dark green in Figure 1.1) are those that are relatively well understood and of low consequence to service with low Susceptibility scores (<1) and high Knowledge scores (>2). These modes of degradation are adequately understood and may been observed in service. Mitigation and maintenance can currently manage this form of degradation. Research on these modes of degradation is a lower priority.

Other combinations of Knowledge and Susceptibility are of course possible and fit between the cases listed above in terms of priority.

Finally, the results of the PIRT scoring were compared to the background technical chapters to ensure all of the important modes of degradation and points were captured. Revisions were then made to the supporting chapters and analysis to ensure adequate discussion of key topics, outcomes, and underlying causes. Thus, the technical basis information for conducting PIRT and the results of the PIRT were re-iterated to ensure that coverage and consistency is maintained in the various PIRT subject areas.

Given the diversity of materials and systems considered by each panel, some minor variations to the process described above were implemented by each panel. These changes and their motivation are listed specifically in subsequent sections of this volume and in the appropriate material system volume.

1.4 ORGANIZATION OF THIS EMDA VOLUME

This volume of the EMDA provides an overall summary and key findings from each the four expert panel assessments covering core internals and primary piping, the reactor pressure vessel, concrete and civil structures, and cable and cable insulation systems. In the sections to follow, the scope of each panel, specifics of the PIRT process utilized, and detailed findings from that panel are presented.

2. CORE INTERNALS AND PIPING SYSTEMS

This technical area is broad and is an extension of the PMDA report, which covered the same material systems. These material systems include low-alloy steels, wrought stainless steels, Alloy 600 and its weldments, Alloy 690 and its weldments, cast austenitic stainless steels, and liner materials. Components using these materials must serve in a variety of environments spanning a broad range of water chemistry temperature, and stress conditions. For many components, irradiation may also exist. The expert panel considered and scored over 1,000 different material/environment/degradation combinations (451 for PWR and 599 for BWR). Volume 2 of this report provides detailed background assessments and PIRT scoring details for all of these materials and degradation modes.

2.1 SUMMARY AND BACKGROUND OF KEY DEGRADATION ISSUES

The reactor core is a very hostile environment, combining the effects of stress, high temperature water environments, and irradiation. Components in this environment are also often the most critical for safe and reliable operation, as the failure of a core internal component may have severe consequences. In general, service beyond 60 years will increase time of exposure in a range of temperature and neutron fluence, leading to potentially increased susceptibility and severity for known degradation mechanisms (although the emergence of new mechanisms are also possible). Therefore, understanding the materials performance and degradation mechanisms is a key to ensure adequate component performance. The issues described below represent those that may warrant additional attention in reactor operation beyond 60 years and are grouped into three key areas: corrosion, thermal aging and fatigue, and irradiation-induced effects. While the susceptibility for each of these key concerns is highly dependent upon specific material and environment, these have been observed in service for many key materials, such as those used for pressure boundary components. These materials, mechanisms, and components were described in considerably more detail in Volume 2: Piping and Core Internals, organized by key classes of materials.

2.1.1 Corrosion and Stress Corrosion Cracking Issues

In addition to elevated temperatures, intense neutron fields, and stress, components must also be able to withstand a corrosive environment. Temperatures typically range from 288 ºC (550 ºF) in a BWR up to 360 ºC (680 ºF) in a PWR, although other water chemistry variables differ more significantly between the BWR and PWR’s.

Corrosion is a complex form of degradation that depends on temperature, material condition, material composition, water purity, water pH, chemical species present, and gas concentrations. The operating corrosion mechanism will vary from location to location within the reactor core and a number of different mechanisms may be operating at the same time. These may include general corrosion mechanisms such as uniform corrosion, boric acid corrosion (BAC), flow accelerated corrosion (FAC), and/or erosion corrosion that will occur over a reasonably large area of material in a fairly homogenous manner. Localized corrosion modes occur over much smaller areas, but at much higher rates than general corrosion and include crevice corrosion, pitting, galvanic corrosion, and microbially induced corrosion (MIC). Finally, environmentally assisted cracking (EAC) includes a combination of other forms of degradation, which are closely related to localized or general corrosion with the added contribution of stress, temperature and/or irradiation. In a LWR, a number of different environmentally assisted cracking mechanisms are observed: intergranular stress-corrosion cracking (IGSCC), transgranular stress corrosion cracking (TGSCC), PWSCC, IASCC and low-temperature crack propagation (LTCP).

While all forms of corrosion are important in managing the safe operation of a nuclear reactor, IASCC has received considerable attention over the last four decades due both to its severity and unpredictability. Despite over thirty years of international study, there does not exist a consensus on the underlying mechanism of IASCC, although more recent work in the open literature has identified several possible causes. These forms of degradation are discussed in considerably more detail in Volume 2 of the EMDA.

Components in the secondary (steam generator) side of a PWR are also subject to degradation. While the secondary side of the reactor does not have the added complications of an intense neutron irradiation field, the combined action of corrosion and stress can create many different forms of failure. The majority of steam generator systems in U.S. power plants today originally used Alloy 600 (a Ni-Cr-Fe alloy) for tubes and some other components, although service experience showed many failures in tubes through the 1970s. In the last 20 years, most steam generators have been replaced with units that have Alloy 690 tubes, which shows more resistance SCC. In addition to the base material, there are weldments, joints, and varying water chemistry conditions leading to a very complex component. Indeed, the array of modes of degradation varies with location. In a single steam generator examined by Staehle and Gorman [3], twenty-five different modes of corrosion degradation were identified. Stress-corrosion cracking is found in several different forms, and may be the limiting factor for extended service. The integrity of these components is critical for reliable power generation in extended operation, and as a result, understanding and mitigating these forms of degradation is important. Adding additional service period to these components will allow more time for corrosion to occur. The various forms of corrosion must be evaluated as in the PMDA report, with a special attention to those that may be life limiting in extended service.

2.1.2 Thermal Aging and Fatigue

The effects of elevated temperature service in metal alloys have been examined for many years. Possible effects include phase transformations that can adversely affect mechanical properties. Extended time at elevated temperature may permit even very slow phase transformations to occur. This is of particular concern for cast stainless steel components where the formation of a brittle alpha-phase can result in a loss of fracture toughness and lead to brittle failure. The effects of aging on other components are also of concern and should be examined. The effort required for identifying possible problems can be reduced, though, by using modern materials science modeling techniques and experience from other industries.

Fatigue refers to an aging degradation mechanism where components undergo cyclic stress. Typically, these are either low-load, high frequency stresses or high-load, low frequency stresses generated by thermal cycling, vibration, seismic events, or loading transients. Environmental factors may accelerate fatigue and eventually may result in a component failure. In a light water reactor, components such as the pressure vessel, pressurizer, steam generator shells, steam separators, pumps, and piping are among the components that may be affected. The PMDA report identified fatigue as an issue for a number of different components and subsystems for both PWR and BWR's. This area of degradation was also identified by the panelists of this effort and is discussed in considerable detail in Volume 2.

Due to the potential for thermal aging and fatigue damage during extended lifetimes, the assumptions and limits considered at the design phase for core internal structures should also be examined. During the initial plant design, each component was designed with a load to expected and specific lifetimes and operating conditions using established guidelines (typically those in Section III of the ASME Boiler and Pressure Vessel Code). An 80-year reactor lifetime corresponds to over 600,000 hours of service (at a 90% service factor) while most creep data used in design comes from tests operating much less than 100,000 hours. The extension of lifetimes beyond these initial design considerations should be carefully examined.

2.1.3 Irradiation-Induced Effects

Over the forty-year lifetime of a light water reactor, internal structural components may experience neutron flux to ~10^{22} n/cm^2/s in a BWR and ~10^{23} n/cm^2/s in a PWR (E > 1 MeV), corresponding to accumulated neutron dose of ~7 displacements per atom (dpa) and 70 dpa, respectively. Extending the operating period of a reactor will increase the total neutron fluence to each component. Fortunately, radiation effects in stainless steels (the most common core constituent) are also the most examined as these materials are also of interest in fast-spectrum fission and fusion reactors where higher fluences are encountered.

The neutron irradiation field can produce large property and dimensional changes in materials. This occurs primarily via one of five radiation damage processes: Radiation-induced hardening and embrittlement, phase instabilities from radiation-induced or -enhanced segregation and precipitation, irradiation creep due to unbalanced absorption of interstitials vs. vacancies at dislocations, volumetric swelling from cavity formation, and high temperature helium embrittlement due to formation of helium-filled cavities on grain boundaries. For light water reactor systems, high temperature embrittlement and creep are not common problems due to the relatively (for creep) lower reactor operating temperature. However, radiation embrittlement, phase transformation, segregation, and swelling have all been observed in reactor components.

Radiation-induced segregation and phase transformations: Under irradiation, the large concentrations of radiation-induced defects will diffuse to defect sinks such as grain boundaries and free surfaces. These concentrations are far in excess of thermal-equilibrium values and can lead to coupled-diffusion with particular atoms. In engineering metals such as stainless steel, this results in radiation-induced segregation of elements within the steel. For example, in Type 316 stainless steel (SS), chromium (important for corrosion resistance) can be depleted at areas while elements like nickel and silicon are enriched to levels well above the starting, homogenous composition. While radiation-induced segregation does not directly cause component failure, it can influence corrosion behavior in a water environment. Further, this form of degradation can accelerate the thermally driven phase transformations mentioned above and also result in phase transformations that are not favorable under thermal aging (such as gamma or gamma-prime phases observed in stainless steels). Additional fluence may exacerbate radiation-induced phase transformations and should be considered. The wealth of data generated for fast-breeder reactor studies and more recently in LWR-related analysis will be beneficial in this effort.

Radiation-induced swelling and creep: The diffusion of radiation-induced defects can also result in the clustering of vacancies, creating voids. If gas atoms such as He enter the void, it becomes a bubble. While swelling is typically a greater concern for fast reactor applications where it can be life-limiting, voids have recently been observed in LWR components such as baffle bolts. The motion of vacancies can also greatly accelerate creep rates, resulting in stress relaxation and deformation. Irradiation-induced swelling and creep effects can be synergistic and their combined influence must be considered. Longer reactor component lifetimes may increase the need for a more thorough evaluation of swelling as a limiting factor in LWR operation. As above, data, theory, and simulations generated for fast reactor and fusion applications can be used to help identify potentially problematic components.

Radiation-induced embrittlement: Radiation embrittlement results in an increase in the yield and ultimate tensile strength of the material. This increase in strength comes with a corresponding decrease in ductility. This hardening can be caused by the changes in the alloy's microstructure including radiation-induced segregation, phase transformations, and swelling. Ultimately, hardening and loss of ductility will result in reduced fracture toughness and resistance to crack growth. Extended reactor lifetimes may lead to increased embrittlement issues.

2.2 SPECIFICS OF PIRT PROCESS FOR CORE INTERNALS AND PIPING SYSTEMS PANEL

The expert elicitation process conducted for each panel is based on the PIRT process. As noted above, the inspiration and methodology for this specific panel is most directly based on that found in the PMDA report.

For the PMDA report, eight experts were utilized for conducting PIRT. For the current activity, 8-10 experts were selected for each of the key panels, and in the case of this volume, 9 experts participated in this exercise. To ensure a diverse set of background and expertise, each panel was assembled to include the institutional affiliations noted above. The panelists selected for this core internals and piping panel had an average of over 40 years' experience in the field and several participated in developing the PMDA report. Selection and assembly of panel experts was performed with NRC and DOE input and approval.

The panelists followed the process covered in section 1.6 above, based on the previous PMDA effort. However, there are also key differences in the PIRT assessment in this work versus the previous PMDA activity that should be noted.

Of particular importance is the PIRT scoring. In the PMDA report, scoring was done on an individual component basis, or groups of components with similar characteristics. For a reference reactor design, a detailed component list was created for both a BWR and PWR plant. The environment was assessed for each component and then relevant degradation modes were considered. For the PMDA report, over 3000 material/environment/degradation modes were considered and scored. However, upon analysis as part of EMDA, it was noted that the same material/environment/degradation mode groupings were scored repeatedly and identically for multiple components or systems, introducing considerable redundancy into the scoring matrix.

For EMDA, considerable effort was made to reduce this scoring redundancy. The original scoring sheets from the PMDA report were obtained and sorted by material and environment. Common components/environments were then condensed into a common entry. For example, in the PMDA report, Type 316 SS heat-affected zones (HAZ) in primary PWR with no irradiation appeared in 17 different entries, although the panelist scores were identical. In this activity, 316 SS HAZ in primary water were scored only a single time. This effort reduced the total number of scoring categories from greater than 3000 to 1020 scoring categories, giving the panel more time to focus on substantive technical concerns.

As a result, this distillation of scoring categories provided a much more efficient process and reduced redundancy. However, it also precludes direct comparison of unique scores for an individual component between the two activities. To retain this capability, the part and component description from *NUREG/CR-6923* were retained as a reference and these cross-references can be found as part of the scoring summary for each category provided in Volume 2, Appendices A through K.

2.3 KEY FINDINGS FOR CORE INTERNALS AND PIPING SYSTEMS

The expert panel deliberated and identified key forms of degradation and potential concerns for extended service operations. Volume 2 of the EMDA report provided expert background assessments of corrosion, stress corrosion cracking, thermal effects, and irradiation for key material systems in core internal and piping systems. Based on the technical background assessments, the panel then developed a PIRT matrix with a list of degradation scenarios to score. Panelists independently scored each of 1020 distinct degradation scenarios in three categories: susceptibility, confidence, and knowledge. Subsequent debate and discussion among panelists was an important part of the process to ensure all points of view were considered. Finally, the results of the PIRT scoring were compiled and used to identify potential knowledge gaps for extended service conditions.

As part of the PIRT analysis, 451 categories were scored for PWR degradation and 569 categories were scored for BWR degradation. Only a small fraction of scores fall into the low Knowledge regime for both PWR and BWR cases. Indeed, only 57 out of 1020 categories were scored in the low Knowledge categories. The vast majority of scores (>75% for both PWR and BWR) fall into the high Knowledge, moderate Susceptibility category. This indicates that the panelists felt the majority of degradation modes considered are well known and manageable to some extent.

Low Knowledge, high Susceptibility degradation modes are those that could be detrimental to service with high Susceptibility (>2) scores and low Knowledge scores (<2). These scores indicate gaps in understanding and can be considered to be identified research to inform degradation mechanisms and underlying causes to predict occurrence during long-term operation. A total of 27 categories were scored in this grouping as part of the PIRT analysis (less than 3%). All of these categories were related to high fluence irradiation effects on core internals. It is important to note that this PIRT process makes no judgment or evaluation on the number of components or significance to structural integrity or safety for a given component, material, or degradation mode. This caveat should be considered when making research priorities and other high Knowledge categories should also be evaluated in that process.

The expert assessment of the background information for irradiation effects identified several modes of degradation that could be key during subsequent operating period. These included the influence of more direct irradiation effects such as hardening, potential phase transformations, swelling and irradiation creep, which may play more significant roles at high fluences. These changes may also have a significant effect on irradiation-induced embrittlement and stress corrosion cracking although the understanding of the interdependencies and synergies at high fluences are yet to be fully developed. These assessments were confirmed following analysis of the PIRT scoring. All 27 low Knowledge, high Susceptibility categories (summarized in Tables 9.4 and 9.5 of Volume 2) are related to fracture resistance, swelling, and SCC effects at high fluence for stainless steels and high strength bolting in core internal applications. The panelists also identified other experience (e.g. swelling experience with 316 SS in fast reactors) that supports the possibility that these forms of degradation will occur in subsequent operating periods.

For stainless steel components, the assessment of background information identified a number of possible knowledge gaps including SCC effects in low-potential environments, effect of stagnant and off-normal water chemistries, crack growth in weld metals, and crack initiation effects under different loading conditions over long-life times. These were similar gaps as identified by the PIRT scoring process for PWR and BWR environments, which included:

- Effect of irradiation on fracture toughness, irradiation creep, swelling, and SCC for Type 304, 316, 347, and 308/309 SS weldments

- SCC susceptibility at very long lifetimes for 304, 316, and 308/309 weldments, particularly in BWR normal water chemistry (NWC) environments

- Potential impact of poor water chemistry control in service water on crevice corrosion, pitting, and MIC for 304, 317, and 308/309 SS weldments

- Cumulative impact of corrosion and fatigue on component integrity for 304 and 316

Alloy 600 has been used for LWR components and piping applications due to low corrosion rate, general resistance to SCC, and thermal expansion coefficient that is similar to that of low-alloy RPV steel. Over the last two decades, there have been numerous incidents of stress corrosion cracking and that is expected to continue with extended service. No low-Knowledge areas were identified in either PWR or BWR environments. However, several outstanding issues were raised by the expert panel for additional consideration, including:

- SCC was identified as a high Knowledge, high Susceptibility mode of degradation in all primary and secondary PWR environments and in BWR NWC and hydrogen water chemistry (HWC) environments for Alloy 600 and Alloy182/82 weldments. This is a known issue for these alloys.

- Wear was identified as a high Knowledge, high Susceptibility mode of degradation in secondary coolant environments for Alloy 600. This is a known form of degradation.

- A reduction in fracture resistance in 182/82 welds at lower temperatures has been noted in laboratory testing although the mechanism is not completely understood.

Today, wrought Alloy 690 and its associated weld metals (Alloy 152, 52, 52M, and other variants) have become the common replacement and repair materials for Alloy 600 and Alloy 182/82 weld metals, primarily due to their superior resistance to primary side SCC. No knowledge gaps were identified for Alloy 690 or 152/52 weldments under subsequent operating periods in PWR environments via the PIRT process. The panelists did note that SCC, fatigue cracking, and pitting should be minimal for Alloy 690, although good water chemistry must be maintained.

Carbon and low alloy steels are widely used, important materials and were the focus of considerable discussion in the expert assessment and PIRT scoring activities. Three specific areas of concern were noted. These include potential lack of understanding in key driving factors and predictive tools for fatigue crack initiation, flow-accelerated corrosion, and stress corrosion cracking. Synergistic effects must also be considered when evaluating long service life integrity. No significant knowledge gaps were identified for carbon and low-alloy steels in PWR or BWR environments following analysis of the PIRT scoring. However, several trends and common themes were identified. These are consistent with the background assessment and included:

- Carbon and low-alloy steels are highly susceptible to BAC of carbon steel but only in the event of a leak of primary coolant. This is a well-known form of degradation.

- Crevice corrosion, pitting, microbial-induced corrosion, and general corrosion of carbon steel and low-alloy steel was identified as a high Knowledge mode of degradation, but only in the event of loss of water-chemistry control or failure of protective features such as liners or cathodic protection. These are well-known forms of degradations.

- Flow-accelerated corrosion is a well-known form of degradation for low-alloy and carbon steels, but can be exacerbated in elbows and changing water chemistry and flow conditions and longer service life and exposure to FAC conditions may increase susceptibility.

- Stress-corrosion cracking and fatigue are possible for these alloys, the Susceptibility was scored low (near 1) for most environments considered. Changes in loading or increases in chemical conditions (such as chloride content) may drive increased susceptibility over a long operating period.

Today, cast austenitic stainless steels (CASS) are used in a variety of applications in both BWRs and PWRs including for reactor coolant, auxiliary system piping, reactor coolant pump casings, reactor coolant valve bodies and fittings. The expert background assessment identified the effects of long-term thermal aging and subsequent degradation on mechanical properties, fracture resistance, and/or corrosion properties as a research need. The PIRT process also identified the effects of long-term thermal aging for extended operating as a knowledge gap.

Other materials beyond these major classes are also in use in a variety of environments and were evaluated as part of the PIRT process. While these materials comprise the majority of LWR components, other materials are also very important. Several knowledge gaps were identified for high strength bolting in both BWR and PWR systems. Specifically, the impact of irradiation on fracture toughness, irradiation creep, swelling, and SCC of high strength bolting materials used in core internal applications and SCC susceptibility over very long lifetimes were noted. No significant gaps in knowledge for extended service were identified for closure studs in PWRs or BWRs. Further, no significant gaps were identified for CuZn tubes, CuNi tubes, BORAL® panels, Zr-fuel assemblies, or 405/409 SS in PWRs nor any gaps identified for brass tubes, Ti-tubing, or Al 6061-T6 components in BWRs. The importance of maintaining good water chemistry control was noted for each material system.

In addition to the specific technical issues for specific material degradation modes and material systems, the expert panel also felt strongly about several other key potential considerations. The items below are not specific material degradation issues and represent the personal opinion of the majority (if not entirety) of the expert panel. While not technical, the expert panel felt strongly that these topics might also ultimately be gaps for extended operation and should be mentioned here. Knowledge retention and transfer is a key factor in capturing knowledge from past generations to future operators, regulators, and researchers who will support extended service operations. Knowledge" represents the subset of information which is known with some certainty, and “expertise” involves greater subtlety associated with the much larger myriad of information that is a combination of complex, not well distilled, ambiguous, and even conflicting. Sustaining expertise is a much more challenging process than transferring knowledge. Similarly, a loss of laboratory capacity could be limiting when trying to close knowledge gaps.

3. REACTOR PRESSURE VESSEL STEELS

NRC regulations require that RPV steels maintain conservative margin for fracture toughness so that flaws do not threaten the integrity of a RPV during either normal operation and maintenance cycles or under accident transients such as pressurized thermal shock (PTS). Neutron irradiation degrades fracture toughness, in some cases severely. Thermal aging, although not generally considered a significant issue for 40 or 60 years of operation, must be an additional consideration for extended operating life to 80 or more years. Regulations in 10 CFR, Part 50 [4], as well as discussion or recommendations in Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code [5], and Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" [6] recognize that embrittlement has a potential for reducing toughness below acceptable levels.

The last few decades have seen remarkable progress in developing a better understanding of irradiation embrittlement mechanisms, including the development of physically based and statistically calibrated models of Charpy V-notch (CVN)-indexed transition-temperature shifts. Those semi-empirical models account for key embrittlement variables and their interactions, including the effects of copper (Cu), nickel (Ni), phosphorous (P), fluence (ϕt), flux (ϕ), and irradiation temperature (T_i). Models of the evolution of nanoscale precipitates, rich in Cu, manganese (Mn), and Ni, are quantitatively consistent with experimental observations of the complex interplay between those elements and other embrittlement variables. The models have provided early warnings of potential technical challenges, such as the contribution of Mn and Ni in high-Ni steels to embrittlement by so-called "late blooming" phases, and have enabled the assessment of outliers in the Transition Temperature Shift Database as well as other contradictory observations. However, these models and the present understanding of radiation damage are not fully quantitative and do not take into consideration the potential contribution of all potentially significant variables and aging technical issues.

Over the past three decades, advances in fracture mechanics have led to a number of consensus standards and codes for determining the fracture-toughness parameters needed for development of databases that are useful for statistical analysis and establishment of uncertainties. The CVN toughness, however, is a qualitative measure that must be correlated with the fracture toughness (K_{Ic}) and crack-arrest toughness properties (K_{Ia}) necessary for structural integrity evaluations. Where practical, direct measurements of fracture-toughness properties are desirable to reduce the uncertainties associated with correlations. Moreover, sufficient fracture-toughness data have been obtained to permit probabilistic determinations. However, specimen-size-effect issues must be resolved to enable the use of typical surveillance specimens for reliable determinations of fracture toughness, applicable at the component level.

Such progress notwithstanding, significant technical issues still need to be addressed to reduce the uncertainties in the data and understanding of the changes in RPV material properties following neutron irradiation. The issues regarding irradiation effects are the most significant issues for RPVs. Of the many significant issues discussed, the following are those deemed to have the most impact on the current RPV material behavior evaluation process:

- High fluence, prolonged irradiation duration, and flux effects
- Material variability
- Alloys with high-Ni content

- The fracture toughness master curve
- The bias in reference toughness derived from precracked Charpy specimens
- Neutron attenuation or through-thickness irradiation effect
- Modeling and microstructural analysis
- Thermal annealing and re-irradiation
- Thermal aging

As part of this expert panel evaluation, all of these issues were discussed in detailed background assessment chapters. Volume 3 of this report provides considerably more depth and detail these key degradation factors and materials of interest.

3.1 SPECIFICS OF PIRT PROCESS FOR REACTOR PRESSURE VESSEL STEEL PANEL

The expert elicitation process conducted for each panel is based on the PIRT process. For the RPV evaluation, the expert panel utilized a recent industry-led activity as a starting point. Specifically, the industry degradation management matrices and issue management tables for RPV steels were recently updated to cover 80 years of operating life. These provided a good starting point to organize possible degradation mechanisms and develop the form of the PIRT tables to be used in this EMDA for boiling water reactor (BWR) and pressurized water reactor (PWR) vessels. From this starting point, the panelists independently determined whether degradation mechanisms for consideration should be added, removed, or modified. For example, Tables 1.1 and 1.2 of Volume 3 were derived from the EPRI Materials Degradation Matrix (MDM) [7] and used to identify the overall array of degradation mechanisms for the entire pressure boundary, including the RPV, pressurizer, steam generator channel head, tubesheet surfaces exposed to primary water, divider plate, and primary piping system. This array was then reduced to isolate only elements of related to the RPV. Those details have been extracted from the EPRI Issue Management Tables (IMTs) [8, 9], rearranged, and summarized in Appendices A (BWR) and B (PWR) of Volume 3. The specifics of the subcomponents provide a detailed resource for the reader to determine a specific location where a mechanism may be important, but generally will not be covered in the individual discussions of the degradation mechanisms.

Following this development of a scoring matrix, the nine members of the expert panel provided scoring for Knowledge, Confidence and Susceptibility for each category. Overall, 54 categories were scored in the RPV panel. All other aspects of the PIRT process described above in section 1.3 were then applied to the RPV materials and relevant modes of degradation.

3.2 KEY FINDINGS FOR REACTOR PRESSURE VESSEL STEEL PANEL

Scoring was completed and compiled for 54 distinct categories of material/degradation issues related to the RPV. There were several notable trends observed for PWRs and BWRs. The highest susceptibilities at extended lifetimes for PWRs and BWRs are embrittlement of carbon and low-alloy steel base metal and welds; however, the knowledge of the phenomena and mechanism as well as the confidence in that assessment were ranked from high to very high, indicating that significant progress has been made in understanding embrittlement. However, as addressed in Chapter 6 of Volume 3, which addresses neutron embrittlement, significant issues and uncertainties remain. Moreover, while there may be mechanistic understanding of the underlying causes, confirmation for extended service and research into mitigation or detection technologies may also be warranted. These knowledge gaps and areas of uncertainty are listed below.

3.2.1 Environmental Effects on Fracture Resistance

Although degradation of RPV materials due to environmental effects is considered unlikely, hydrogen embrittlement could lead to a reduction of fracture resistance of RPV materials. Based on very limited data, this mechanism should not present a concern for LWRs under normal operating conditions. However, if future relevant test data and extended operating experience indicate that 60 year operation of RPVs could cause hydrogen buildup, then an assessment of hydrogen buildup and the development of subsequent mitigation procedures for 80 year operation may be needed. Based on the current data available, a hydrogen level of 4 ppm and higher in the RPV material could become a contributor to the overall degradation in fracture resistance of the RPV.

3.2.2 Thermal Embrittlement of RPV Steels

It has been observed that the HAZs of higher-temperature low-alloy steel (LAS) components are prone to thermal aging, with the pressurizer experiencing the highest temperature. It typically operates at 343 °C (650 °F) and could undergo a significant shift in HAZ ductile-brittle transition temperature (DBTT) (rivaling the RPV irradiation embrittlement shift). For that reason, the pressurizer, if fabricated from LAS, and portions of the RPV, that operate at high temperature, could be prone to thermal aging and have significant stresses. The RPV components that reach higher temperatures [315 °C (~600 °F)] consist of the RPV flange, the nozzle shell ring, and the outlet nozzles of all plants as well as the vessel heads of some reactors which have head temperatures near the hot leg temperature of about 315 °C (~600 °F). However, many of the RPV heads of the U.S. plants, including the heads in all hot head plants, have been replaced thereby resetting the aging. The nozzle shell ring and outlet nozzles receive a low neutron dose rate exposure, which could synergistically combine with thermal aging, potentially creating greater-than-expected embrittlement. That region, known as the extended beltline, is undergoing pressure-temperature curve evaluation by the Pressurized Water Reactor Owners Group (PWROG); however, thermal aging shift in DBTT is not currently considered in the PWROG evaluation.

Several opportunities to better understand the effects of thermal aging exist. Combustion Engineering pressurizers, fabricated with materials similar to RPV materials and operated at about 343 °C (650 °F), have been retired at Saint Lucie 1 (Fall 2005), Millstone 2 (Fall 2006), and Fort Calhoun (Fall 2006) and may be available for examination. Because the pressurizer is the reactor coolant system (RCS) pressure boundary component that reaches the highest temperature, any thermal-aging embrittlement seen would provide a leading indicator for the rest of the RCS. Moreover, even without baseline properties, relatively high DBTT, evidence of grain boundary P segregation, and intergranular fracture indicating thermal aging could be determined using the retired pressurizer material. Examination of LAS pressurizer HAZs would provide information on the extent of the long-term embrittlement of a component that has experienced reactor operation. This information could be used to determine if there is a need to address thermal aging embrittlement for operation up to 80 years.

Over the last 20 years, a number of steam generators have been replaced. The bottom heads of Westinghouse designed steam generators were fabricated from SA-508 forgings, the same type as the RPV. Moreover, the same bottom head bowl forging has a cold leg and a hot leg nozzle welded to it. For that reason, retired steam generator bottom bowl nozzle HAZs could be examined with the properties and microstructure of the cold leg side (where no thermal ageing is expected) compared to the hot leg side (where thermal aging is possible). The HAZ of the same material could also be evaluated for evidence of long-term thermal aging.

Within the PWROG research program, a 300,000 h thermal aging exposure of the Arkansas Nuclear One-Unit 1 RPV head is projected to be reached in 2017. Although the aging temperature is relatively low, the material has an exceptionally long aging time and the mechanical properties and microstructure have been well documented, making it a unique candidate for evaluation. The panel recommended that some of these materials be tested to assess any changes in the transition temperature, HAZ microhardness, and microstructure.

3.2.3 Long-Term Integrity of Dissimilar Metal Welds

Unless data from the initial 40-year operating period and the first license renewal indicate that the SCC factors are insignificant, the following issues need to be considered as part of the evaluation of whether the operating time can be extended to 80 years:

- Effect of long-term thermal aging on the susceptibility of Alloy 82 weld metals to SCC
- Effect of long-term operation on the susceptibility of Alloy 152 and 52 weld metals to SCC
- Effect of alloying elements and the compounds formed during heat treatment on the susceptibility of LASs to SCC under BWR conditions
- Validity of the crack growth data for LAS and in the SCC disposition curves
- Crack behavior at the fusion weld line between Ni alloy weld metal and LAS
- Effect of neutron irradiation on the susceptibility of LAS to SCC

3.2.4 Environmental Assisted Fatigue

Fatigue issues for the RPV generally are insignificant and seldom as important as the associated piping connected to the RPV. Fatigue in water environments at regions where cumulative usage factor (CUF) values are projected to be significant at 80 years of operation may require monitoring or assessment. The panel felt that the development of the relationship between laboratory test data under conditions simulating reactor operating stresses and loading sequences is needed to improve confidence that environmental fatigue does not becomes a significant factor for long term operation.

3.2.5 Neutron Embrittlement

A summary of the extensive recommendations provided by the expert panel is listed below by topic.

- *Flux effects at high neutron fluence recommendations:* Although high-fluence surveillance data may eventually provide a sufficient basis for timely informed decisions regarding extended operation for PWRs expected to reach very high fluences, developing a significant database on high-fluence effects from test reactor (TR) data provides a useful complement to the surveillance transition temperature shift (TTS) data. Currently, the Eason, Odette, Nanstad, and Yamamoto (EONY) model has been incorporated into 10 CFR, Part 50.61a for limited applications. The use of accelerated, higher-flux TR data ultimately requires improved understanding and modeling of flux effects because high-flux irradiation may result in artifacts in TTS that would not be encountered in low-flux (e.g., power reactor) irradiations. Therefore, systematic and objective research on flux effects on TTS at intermediate and high fluence is recommended to resolve uncertainties. Simultaneously, efforts to obtain surveillance specimens from very high fluence irradiations (e.g., the Palisades vessel at a relatively high lead factor, and high-Ni weld specimens from Swedish power reactors) should continue.

- *High-nickel effects and other potential high-fluence embrittlement mechanisms recommendations:* Because LBPs may result in significantly increased embrittlement not predicted by current embrittlement models, additional research is recommended to determine: (1) the conditions leading to the formation of LBP; and, (2) the severity of the corresponding embrittlement. Other contributors to hardening, especially self-interstitial atom cluster dislocation loops, may also be important at high fluence.

- *Thermal annealing and reirradiation recommendations:* To better understand the effects of annealing, material characterization, and modeling, data is needed and includes high dose rate experiments, post-annealing reirradiation, microstructural characterization of reirradiation effects, temper embrittlement of HAZ, and characterization from reirradiated surveillance programs.

- *Attenuation of embrittlement recommendations*: (1) If generic approaches to attenuation are to be used, it is recommended they be improved relative to the approach used in Revision 2 of Regulatory Guide 1.99. Further, assessment of the uncertainties in predicted TTS associated with generic methods is recommended. (2) The issues raised regarding the use of a generic attenuation procedure argue for use of plant-specific approaches to attenuation. Vessel-specific approaches based on computed dpa and dpa rates converted to effective fluence and flux for use in TTS models would not be difficult to implement because the required neutronics calculations are generally available. (3) Assembling a catalogue of experimental studies pertinent to the issue of attenuation, and compiling a corresponding database that can be systematically analyzed using the outlined procedures are recommended as a very high priority.

- *Master Curve fracture toughness recommendations:* The most significant issue impeding more comprehensive use of the Master Curve in RPV embrittlement monitoring and structural integrity analysis is the effect of specimen size on T_0. In particular, it is important to establish how T_0 values measured using Charpy-sized specimens can be used to reliably predict the transition behavior of much larger structures.

- *Embrittlement beyond the beltline recommendations:* Assuming that the current definition of beltline based on a fluence limit continues to be accepted, the physical extent of the "beltline" implied by this fluence limit will be expanded during operation for up to 80 years to include regions of the RPV where there are nozzle penetrations and shell thickness transitions. This will require: (1) an evaluation of the extended beltline materials that exceed the 1×10^{17} n/cm^2 fluence and the development of a better understanding of the properties of these relatively under-characterized materials; (2) the inclusion of representative material of the extended beltline in surveillance programs; (3) an assessment of thermal embrittlement in the hot leg nozzle HAZs since there may be a synergistic effect with low flux irradiation; and (4) an assessment of the albedo effect, in which neutrons that pass through the RPV wall reflect off the concrete and stream up through the cavity to the nozzle area. Since the fluence on the outside of the nozzles and RPV can be as high as or higher than the fluence on the inside of the RPV in that region, accurately modeling the fluence in the various areas around the nozzles/extended beltline becomes more important.

4. CONCRETE AND CIVIL STRUCTURES

As concrete ages, changes in its properties will occur as a result of continuing microstructural changes (e.g., slow hydration, crystallization of amorphous constituents, and reactions between cement paste and aggregates), as well as environmental influences. These changes do not have to be detrimental to the point that the concrete will not be able to meet its functional and performance requirements. Concrete structures can also suffer undesirable changes in properties with time, including adverse performance of its cement paste matrix or aggregate constituents under environmental influences (e.g., physical or chemical attack). Changes to embedded steel reinforcement as well as its interaction with concrete can also be detrimental to concrete's service life. Aging effects can be exacerbated if improper concrete specifications were used at the time of construction. A number of areas of research would help assess the long-term integrity of the reactor concrete structures.

In general, the performance of reinforced concrete structures in nuclear power plants (NPPs) has been good. Incidents of degradation initially reported generally occurred early in the life of the structures and primarily have been attributed to construction/design deficiencies or improper material selection. Although the vast majority of these structures will continue to meet their functional or performance requirements during the current and any future licensing periods, it is reasonable to expect that there may be isolated examples where, as a result primarily of environmental effects, the structures may not exhibit the desired durability, (e.g., water-intake structures and freezing/thawing damage of containments), without some form of intervention. The details of the assessment of concrete and civil structures are found in Volume 4 of this report.

4.1 SPECIFICS OF PIRT PROCESS FOR CONCRETE AND CIVIL STRUCTURES PANEL

As noted above, each PIRT application has been unique in some respect and the current project is, again, a unique application. The approach followed by the civil structures and concrete panel used the methodology described above in Section 1.3 and consisted of the following steps.

1. First a list of relevant structures and components was prepared, and a hierarchical identification of the various degradation modes was developed and logged in for each. Four classes of structures and components were identified, together with related degradation modes and mechanisms. The four component classes were containment concrete, containment steel structures, spent fuel pool (SFP) and transfer canal, and the cooling tower. Descriptions of relevant structures, materials of construction, and durability mechanisms and processes are given in Chapters 2, 3, and 4, respectively, of Volume 4. Safety-related structures of primary importance and their related degradation modes were identified. Crosscutting issues associated with NPP containments were also identified.

2. Next a spreadsheet reflecting these degradation modes and mechanisms was developed. For each of the identified for classes of structures and components (described below), each panel then provided an assessment of the level of knowledge, susceptibility, confidence, and structural significance for each degradation mode and mechanism. This assessment is detailed in the spreadsheet included in Appendix A of Volume 4.

3. From the spreadsheet, the mean, median, and standard deviation were determined for each potential degradation mode/mechanism.

To remain consistent with the approach adopted for the PMDA, the panel utilized the PIRT process in their assessment. The PIRT process was faithfully applied and was expanded to encompass some of the unique characteristics of concrete structures. The panel defined a fourth category matrix, “Structural Significance,” in addition to the original three, for each combination of component and degradation mode that follows. The assessment thus addressed the following.

- The degree of **Susceptibility** to degradation
- **Confidence** of the expert panel in their assessment of susceptibility
- The extent of **Knowledge** of how adequately the relevant dependencies are understood
- The **Structural Significance** of the degradation to the safe operability of the structure

The evaluation of the susceptibility index is based on the operating experience of various industries: nuclear, hydro, and transportation. The benefit of expanding the scope of the research to the mentioned sectors was to investigate the degradation modes of sometimes-older concrete structures (like dams for instance) potentially subjected to a more aggressive environment (e.g., carbonation exposure in industrial and urban areas). Carefully transposing this operating experience to nuclear structures provided a helpful opportunity to extrapolate potential degradation modes. For the specific aging modes in NPPs (borated water attack and irradiation for instance) with limited background data and information, the projection of the effects for long duration operation could only be based on a brainstorming process. As a result, when operating experience exists, the confidence level among the panel is generally higher.

It should be noted here that the EMDA applies a generic process to a wide variety of structural design, outdoor environment, and concrete mix design. The ranking resulting from this approach is intended to provide general trends but does not cover local specificities.

4.2 KEY FINDINGS FOR CONCRETE AND CIVIL STRUCTURES PANEL

Each of the key structural categories was independently analyzed to track the level of knowledge, the significance of the most impactful degradations, and their susceptibilities. The following mechanisms emerged as most important in each category:

- Irradiation for containment concrete emerged as the most important degradation mechanism, mainly driven by insufficient data to improve the level of knowledge about the effects of irradiation on concrete mechanical properties. Alkali-aggregate reaction (AAR), acid attack and creep emerged as secondarily important mechanisms. The biggest surprise in this analysis is the result that susceptibility to fracture emerged as the least important mechanism; this should be interpreted to apply only to concrete cracking of the generally known type which is accounted for in the structural design. There are special forms of concrete damage that potentially evolve with time into discrete fracture under special circumstances involving creep-cracking interaction induced by structural modification or change in loading. These do not qualify as a general aging mechanism, and are addressed separately below.

- Concerning containment steel structures, the corrosion of the liner plate on the concrete side emerged as the mechanism with the highest level of importance, primarily because of being inaccessible. This is followed by the corrosion of reinforcement by chlorides and boric acid, and by the SCC of the pre-stressing tendons. Irradiation effects on steel components, including the liner, emerged as the least important, primarily because of the accumulated low neutron dose levels.

- Concerning the SFP and transfer canal, boric acid attack on concrete in PWRs emerged as the mechanism of highest importance. This is closely followed by SCC of welds in the liner plate and channels. Considering the available field experience, this mechanism was scored highly due to the prevalence of data (high knowledge), and should be considered as important as the boric acid attack.

- Finally, concerning the cooling tower, the corrosion of reinforcement emerged as the most important aging mechanism followed by several mechanisms, which include freeze/thaw, AAR and SCC of prestress tendons in precast elements in the Mechanical Draft cooling tower design. It is important to point out that corrosion of reinforcement, which, while not safety related, is highly important to operate the nuclear power plant economically.

Based on the rankings of important degradation mechanisms for the respective categories of concrete and civil structures, potential knowledge gaps for assessing the integrity of concrete structures for operation up to 80 years were identified as follows.

- The panel identified creep of the post-tensioned concrete containment as a potential knowledge gap. Creep is a long-term process associated with sustained loading and moisture transport that affects the internal stress state and, because it adds to tendon relaxation in causing gradual loss of prestress, which is usually restored by periodic re-tensioning thereby introducing a form of cyclic activation of primary creep, can potentially damage the concrete and lead to tertiary degradation (creep-fracture interaction) under accidental loading.

- Related to the creep mode identified above is the interaction between creep and cracking in post-tensioned containments subjected to repair involving prestress modification during the operational life of the containment. While concrete cracking is a well understood behavior characteristic of concrete structures in general, and is accounted for in the usual manner in the structural design of reinforced containments, it plays a unique role, (usually unaccounted for in design), in post-tensioned containments. Depending upon the position of the tendons relative to the surface of the containment wall, radially oriented dilation damage, eventually leading to discrete split cracking, can form on a lamellar surface parallel to the wall surface, which evolves with time as a creep–cracking interaction mechanism. This mode of cracking can potentially occur during initial pre-stressing, during re-tensioning to repair loss of prestress due to concrete creep and tendon relaxation, or during de-tensioning and re-tensioning operations which may be undertaken as part of life extension reconstruction work. This type of split cracking can be controlled by radial reinforcement, which generally is not part of the initial design, and because such cracking configuration is internal and is not visible on the surface, it can potentially evolve into an undetectable degradation mode.

- The panel also identified the irradiation of concrete as a knowledge gap. This, as mentioned above, is due to a lack of sufficient test data to support a clear evaluation of the significance of such mechanism for long-term operations. As a reminder here, the term “concrete containment” is used in a generic sense to describe any concrete part within the containment building. Irradiation mainly affects the reactor cavity and the biological shield.

- Alkali-silica reactions were also noted by the expert panel. Though this degradation is well documented by the operating experience (for bridges and dams in particular) and scientific literature, its high ranking in the EMDA analysis describes the need to assess its potential consequences on the structural integrity of the containment, considering the recent operating experience at Davis Besse and other plants.

- The next mechanism and potential gap is related to boric acid attack of concrete in the spent fuel pool. The knowledge gaps are essentially related to the kinetics and the extent of the attack (role of the concrete mix design) and their consequences on the structural integrity.

Finally, the panel identified two possible knowledge gaps when assessing the integrity of containment steel components for operation up to 80 years:

- Corrosion and SCC of the tendons, and

- Corrosion of the inaccessible side of the liner. The lack of knowledge here is associated with the absence of a current in-service inspection technique.

These degradation modes and mechanisms have been identified as having the greatest potential effect on preserving the ability of the concrete and civil structures and components to fulfill their safety related functions during long-term NPP operation. This potential effect may be mitigated by improving the overall level of knowledge about the identified degradation modes in order to better predict and mitigate possible consequences; and/or, by identifying and implementing acceptable mitigation strategies (replacement, treatments, etc.). Research will be required in either case and these topics were identified as having the highest priorities for research for concrete and civil structures and components.

5. CABLE AND CABLE INSULATION

A variety of environmental stressors in NPPs, such as temperature, radiation, moisture/humidity, vibration, chemical spray, mechanical stress, and the oxygen present in the surrounding gaseous environment (usually air), can influence the degradation of low and medium electrical power and instrumentation and control (I&C) cables and their insulation. Over time these stressors can lead to degradation, which, if not appropriately managed, could lead to insulation failure of the associated components, and potentially resulting in cables being unable to perform their intended safety function.

In the context of this report, low-voltage cables have ratings below 2,000 volt (V) and generally operate at voltages of 525V alternating current (ac) or below 250 V direct current (dc). Medium-voltage cables are rated at 46 kilovolts (kV) and below. Most in-plant and underground cables are rated at up to 15 kV and are operated at 13 kV or less. Most safety-related medium-voltage cables rated at 5 kV are operated at 4,160 V. Some plants have short lengths of cable with operating voltages between 100 and 230 kV; these are plant-specific cables and are often not insulated with a polymer. As such, unique plant-specific cables are not covered by this report. Furthermore, high-voltage cables are not covered by this report. The details of the assessment of cables and cable insulation are found in Volume 5 of this report.

5.1 SPECIFICS OF PIRT PROCESS FOR CABLE AND CABLE INSULATION PANEL

The cable system expert panel used the PIRT process described above in Section 1.3 to identify safety-relevant phenomena, assess their importance, and identify and prioritize research needs. Five panelists provided scoring on a variety of issues and environments. The PIRT process followed by this specific panel consisted of the following steps:

1. A list of relevant insulation materials was developed, along with a hierarchical identification of the various degradation modes and environments that could affect each of the insulation materials and their performance. A consensus of the issues to be assessed was obtained through discussions among the members of the panel. Crosscutting issues were identified. A total of 44 different scoring categories were considered.

2. A database was developed, containing the independent scoring for each of the above PIRT criteria by each panelist for each insulation material and their related degradation modes. The panel then discussed the individual scoring, and each panelist was provided the opportunity to keep or revise their original scores based on this discussion.

3. Based on the final set of scores, the mean, median, and standard deviation were determined for each potential degradation mode/mechanism.

For I&C cables, the degradation for polymers is highly dependent on the material and the environment. Although the PIRT assessment divides the cable insulations based on the base material, the particular degradation phenomena vary depending on the formulation of the insulation. For example, one XLPO insulation may behave differently from another XLPO insulation, depending on the additives (pigments, plasticizers, anti-oxidants, etc.). Furthermore, the PIRT assessments were performed using the insulation material in a range of environmental conditions in order to assess the insulation in a variety of environments as the insulation material could be used in different areas of a NPP. Since the major stressors to insulation are temperature and radiation, the environmental conditions are considered with a temperature and radiation dose range. For I&C cables, the study did not include wet environments.

5.2 KEY FINDINGS FOR CABLE AND CABLE INSULATION PANEL

The panelists used the PIRT process to prioritize the different material/environmental concerns (the PIRT scores are shown in Appendix A of Volume 5). There are several notable trends in the data. First, the panelists were in agreement as to the present levels of knowledge and overall aging-related susceptibility of cable insulation materials, as demonstrated by the uniformity of the Knowledge and Susceptibility scores. Further, there were very few material/mode combinations where Susceptibility was ranked above “2” with the generic Susceptibility increasing with increasing severity in environment conditions. The Knowledge ranking was either 2 or 3 for all materials, environments, and conditions considered. This is likely a reflection on the 40 years of information on generic aging although this may not extend to specific plant locations/conditions as noted above.

The main area of uncertainty for extending NPP operation beyond 60 years relates to the pre-aging carried out during the equipment quantification (EQ) process and whether it can adequately predict aging over that time scale. However, most concerns are based on the premise that cables will be exposed to the operating and design basis environments (temperature, radiation, humidity, chemical spray, and other environmental factors) that were used in the equipment quantification process. The current understanding, based on general opinion and utility experience, is that most cables are exposed to environments that are considerably less severe than the design environment. Actual environmental conditions should be quantified by measurement and analysis so that the temperatures and dose rates to which different types of cable are exposed are quantified over their qualified life.

Recommendations and conclusions for cable use beyond 60 years are provided below:

1. A reassessment may be made to determine the number of circuits and types of cable that are in the high-radiation zones [i.e., 70 Mrad over 80 years (up to 1 Gy/hr) between 45 and 55 °C (113 and 131 °F)].

2. Measurements of the operating temperatures of cables in plant are needed, particularly for those cable groups that are subjected to EQ, to quantify the actual temperatures to which cables are exposed.

3. If, as expected, environmental information demonstrates that thermal aging is the dominant process for nearly all cables in U.S. NPPs, then it is important that the activation energy for the specific cable materials used, under specific environment, be estimated with increased confidence level. This is because the actual value of activation energy plays a major role in behavior prediction model over time at a given environment. Experiments conducted to estimate activation-energy should be conducted at temperatures close to service temperatures using techniques such as oxygen consumption that have the ability to cover wide temperature ranges. This ability allows one to use the oxygen consumption results to confirm a correlation (same activation energy) with the mechanical properties (e.g., elongation) at the higher temperatures and to use low temperature oxygen consumption

results to probe any changes in activation energy in the low temperature extrapolation region.

4. Inverse temperature effects need to be understood better if semi-crystalline materials, such as some XLPE/XLPO and EPR insulations, are determined from plant assessment (item 1 above) to be exposed to radiation in-plant dose rates that exceed 0.1 Gy/h (10 rad/h). At that level of radiation dose rate, significant degradation may be observed after 60 years for temperatures <50 °C (122 °F).

5. Little is known regarding the consequences of long-term wetting of both low- and medium-voltage cables. Research in that area would enable safety significance assessments of long-term submerged cables.

6. For loss of coolant accident simulations, this research has identified oxygen concentration in the atmosphere during a loss-of-coolant accident to be important, needing a consideration of this aspect in engineering simulations.

6. REFERENCES*

1. NRC, *Expert Panel Report on Proactive Materials Degradation Assessment*, NUREG/CR-6923, U.S. Nuclear Regulatory Commission, February 2007.
2. NRC*, Generic Aging Lessons Learned (GALL) Report*, NUREG 1801, U.S. Nuclear Regulatory Commission, December 2010.
3. R. W. Staehle and J. A. Gorman, “Quantitative Assessment of Submodes of Stress Corrosion Cracking on the Secondary Side of Steam Generator Tubing in Pressurized Water Reactors: Part 2,” *Corrosion* **60**(1), 5–63 (2004).
4. *U.S. Code of Federal Regulations,* Title 10, “Energy,” Part 50 (10 CFR 50), Office of the Federal Register, 2013.
5. ASME, “Rules for In-service Inspection of Nuclear Power Plant Components,” Section XI, Appendix, *Boiler and Pressure Vessel Code (BPVC)*, American Society of Mechanical Engineers, New York, 1995.
6. NRC, *Radiation Embrittlement of Reactor Vessel Materials, Regulatory Guide 1.99*, Revision 2, U.S. Nuclear Regulatory Commission, 1988, ADAMS ML003740284.
7. EPRI, *Materials Degradation Matrix,* Document 3002000628, Electric Power Research Institute, May 2013.
8. EPRI, *Materials Reliability Program: Pressurized Water Reactor Issue Management Tables—Revision 2,* MRP-205, Document 1021024, Electric Power Research Institute, October 2010.
9. EPRI, *BWR Vessel and Internals Project: Boiling Water Reactor Issue Management Tables,* BWRVIP-167NP, Revision 2, Document 1020995, Electric Power Research Institute, August 2010.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

Most nuclear power plants in the United States are currently licensed for up to 60 years of operation. The nuclear industry is assessing the feasibility of operation for up to 80 years. The U.S. Nuclear Regulatory Commission (NRC) and U.S. Department of Energy (DOE) co-sponsored the Expanded Materials Degradation Assessment (EMDA) to identify information gaps and research priorities for aging related degradation of reactor components for up to 80 years. Expert panels were convened to examine four main component groups using the phenomena identification and ranking technique: reactor core internals and piping systems, the reactor pressure vessel, concrete and civil structures, and electrical cables. Panelists included participants from NRC, DOE national laboratories, industry, academia, and international organizations. The EMDA reports include a ranking of degradation scenarios according to the probability of occurrence and level of knowledge, along with a summary of the current state of knowledge for each component group.

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