

Long-Term Operations Program: Assessment of Research and Development Supporting Aging Management Programs for Long Term Operation



Long-Term Operations Program: Assessment of Research and Development Supporting Aging Management Programs for Long-Term Operation

All or a portion of the requirements of the EPRI Nuclear
Quality Assurance Program apply to this product.

YES



EPRI Project Manager
R. Tilley



3420 Hillview Avenue
Palo Alto, CA 94304-1338
USA

PO Box 10412
Palo Alto, CA 94303-0813
USA

800.313.3774
650.855.2121

askepri@epri.com

www.epri.com

3002000576
Final Report, August 2013

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

THIS DOCUMENT WAS PREPARED BY THE ORGANIZATION(S) NAMED BELOW AS AN ACCOUNT OF WORK SPONSORED OR COSPONSORED BY THE ELECTRIC POWER RESEARCH INSTITUTE, INC. (EPRI). NEITHER EPRI, ANY MEMBER OF EPRI, ANY COSPONSOR, THE ORGANIZATION(S) BELOW, NOR ANY PERSON ACTING ON BEHALF OF ANY OF THEM:

(A) MAKES ANY WARRANTY OR REPRESENTATION WHATSOEVER, EXPRESS OR IMPLIED, (I) WITH RESPECT TO THE USE OF ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE, OR (II) THAT SUCH USE DOES NOT INFRINGE ON OR INTERFERE WITH PRIVATELY OWNED RIGHTS, INCLUDING ANY PARTY'S INTELLECTUAL PROPERTY, OR (III) THAT THIS DOCUMENT IS SUITABLE TO ANY PARTICULAR USER'S CIRCUMSTANCE; OR

(B) ASSUMES RESPONSIBILITY FOR ANY DAMAGES OR OTHER LIABILITY WHATSOEVER (INCLUDING ANY CONSEQUENTIAL DAMAGES, EVEN IF EPRI OR ANY EPRI REPRESENTATIVE HAS BEEN ADVISED OF THE POSSIBILITY OF SUCH DAMAGES) RESULTING FROM YOUR SELECTION OR USE OF THIS DOCUMENT OR ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT.

REFERENCE HEREIN TO ANY SPECIFIC COMMERCIAL PRODUCT, PROCESS, OR SERVICE BY ITS TRADE NAME, TRADEMARK, MANUFACTURER, OR OTHERWISE, DOES NOT NECESSARILY CONSTITUTE OR IMPLY ITS ENDORSEMENT, RECOMMENDATION, OR FAVORING BY EPRI.

THE FOLLOWING ORGANIZATION PREPARED THIS REPORT:

Electric Power Research Institute (EPRI)

THE TECHNICAL CONTENTS OF THIS PRODUCT WERE **NOT** PREPARED IN ACCORDANCE WITH THE EPRI QUALITY PROGRAM MANUAL THAT FULFILLS THE REQUIREMENTS OF 10 CFR 50, APPENDIX B. THIS PRODUCT IS **NOT** SUBJECT TO THE REQUIREMENTS OF 10 CFR PART 21.

NOTE

For further information about EPRI, call the EPRI Customer Assistance Center at 800.313.3774 or e-mail askepri@epri.com.

Electric Power Research Institute, EPRI, and TOGETHER...SHAPING THE FUTURE OF ELECTRICITY are registered service marks of the Electric Power Research Institute, Inc.

Copyright © 2013 Electric Power Research Institute, Inc. All rights reserved.

Acknowledgments

The following organization prepared this report:

Electric Power Research Institute (EPRI)
1300 West W.T. Harris Blvd.
Charlotte, NC 28262

Principal Investigators

R. Tilley
S. Bernhoft
R. Dyle

This report describes research sponsored by EPRI.

EPRI would like to acknowledge the wide industry engagement in the process to assess research and development needs to support aging management programs for long-term operation. The results of that assessment effort are summarized in Section 2 of this report, and the extensive work behind the Section 2 summary will be the subject of a separate technical report. Specific acknowledgment is made of the following:

- The EPRI contractor Alliance Engineering, P.C. (W. Lunceford and T. DeWees), which provided the structure and initial assessment for reviewing R&D needs relative to the Section XI Aging Management Programs described in NUREG-1801, Revision 2, the Generic Aging Lessons Learned (GALL) report
- EPRI Nuclear Sector technical staff

This publication is a corporate document that should be cited in the literature in the following manner:

*Long-Term Operations Program:
Assessment of Research and
Development Supporting Aging
Management Programs for Long-
Term Operation.*
EPRI, Palo Alto, CA: 2013.
3002000576.

- The Nuclear Energy Institute, especially the utility staff participating in the NEI License Renewal Task Force, the NEI Mechanical Working Group, the Electrical Working Group, the Civil Structure Working Group, and the License Renewal Implementation Working Group
- The U.S. Department of Energy, especially technical staff of the Light Water Reactor Sustainability Program
- Reactor owners groups



Product Description

Presently, U.S. nuclear utilities and the Nuclear Regulatory Commission (NRC) are discussing a second round of plant license renewals, termed *subsequent license renewal* (SLR). For the U.S. plants this could mean extending a plant's operating license from 60 years to 80. To achieve safe and reliable operation for such an extended period will require a comprehensive technical understanding of aging degradation effects on nuclear plant systems, structures, and components (SSCs). EPRI research projects have long been focused on aging degradation modes, root causes, mitigation options, and repair options. This report presents the results from an assessment of research and development (R&D) projects and plans relative to aging management needs for a period of extended operation from 60 to 80 years.

Background

Aging management programs (AMPs) are used to understand and manage age-related degradation of SSCs in nuclear power plants and are a key element of the U.S. license renewal process to provide reasonable assurance of safe, reliable plant operation during the period(s) of extended operations.

Objectives

The objectives of this project were twofold:

- To identify R&D to address aging management program application through the period of extended operation (to 80 years)
- To map the EPRI-led R&D projects to the existing AMPs to provide a basis to support technical discussions on SLR

Approach

For the mapping of the R&D projects to the aging management program, NRC NUREG-1801, Revision 2, "Generic Aging Lessons Learned Report" (the GALL Report) was used as the baseline. The Section XI AMPs were reviewed by subject matter experts familiar with the current state of technical information on aging degradation.

Each of the 50 GALL Section XI AMPs was then categorized into one of three bins based on R&D needs to support long-term operations:

- R&D support to address knowledge gaps for 60 to 80 years of operation, to better understand and manage materials performance
- R&D in a supporting role where aging degradation is well characterized
- No new R&D role identified

Eight AMPs were assigned to the first bin. The R&D projects for the category 1 AMPs are described in three primary sections of the report.

Results

EPRI has been conducting R&D in materials degradation for several decades, and this report compiles an extensive list of previously published R&D project results that support aging management programs, describes current and planned R&D projects building on this existing research base, and identifies knowledge gaps for which R&D will continue to support long-term operations.

Applications, Value, and Use

Utility-implemented AMPs are a key element in the successful management of aging degradation in nuclear power plants. Advances in technology and technical understanding of degradation will continue to inform the content of AMPs. Importantly, this project represents a key effort to assess R&D needs related to the content of AMPs for an operating period to 80 years. The research as identified in this effort is well along and provides a sound technical basis for continued long-term operation of nuclear power plants.

Keywords

Aging management
Asset management
Materials research
Long-term operations
License renewal

Abstract

The Electric Power Research Institute (EPRI) provides research and development (R&D) leadership and coordination for the electric power industry. This R&D focus includes addressing the aging degradation of systems, structures, and components (SSCs) in operating nuclear power plants (NPPs). For aging degradation, the EPRI research results include identification of degradation mechanisms and their causes, assessment of degradation rates, development of options to mitigate degradation, and tools and processes to support SSC repairs or replacements as needed.

Presently, the U.S. nuclear utilities and the Nuclear Regulatory Commission (NRC) are discussing a second round of plant license renewals, termed *subsequent license renewal* (SLR). For the U.S. plants, this could mean extending a plant's operating license from 60 years to 80. In preparation for the possibility of such long-term operations, EPRI established a Long-Term Operations (LTO) Program. The purpose of the EPRI LTO Program is to ensure a technical basis for safe, reliable plant operation during the extended period of operation. The EPRI LTO Program integrates the R&D projects across EPRI and coordinates with other R&D initiatives, most notably the U.S. Department of Energy (DOE) Light Water Reactor Sustainability (LWRS) Program.

In preparation for 60 to 80 years of operation, EPRI performed a review of the aging management programs currently in NUREG-1801, Revision 2, "Generic Aging Lessons Learned Report" (the GALL Report). The review considered the current state of technical knowledge and research needs for the period of subsequent extended operation. Summarized in this report are the following:

- Extensive existing R&D project results that have been published to support aging management programs (presented as a compiled list in the report's references section)
- Current and planned R&D projects building on this existing research base
- Knowledge gaps for which R&D will continue to support long-term operations

ACRONYMS AND ABBREVIATIONS

Acronym	Meaning
AMP	Aging Management Program
AMR	Aging Management Review
BWRVIP	Boiling Water Reactor Vessel and Internals Program
DOE LWRS Program	Department of Energy Light Water Reactor Sustainability Program
GALL Report	Generic Aging Lessons Learned Report (NUREG-1801, Rev 2)
LTO	long-term operations
MRP	Materials Reliability Program
PSCR	primary system corrosion research
SCC	stress corrosion cracking
SLR	subsequent license renewal
SSC	systems, structures, and components
TAA	time-limited aging analysis

Table of Contents

Section 1: INTRODUCTION	1-1
Objective	1-1
Background.....	1-2
Section 2: AMP REVIEW	2-1
AMP R&D Review Approach	2-1
AMP Review Results.....	2-1
Section 3: MECHANICAL AMPS R&D REVIEW	
RESULTS	3-1
Review Results	3-1
Gap Assessment Summary of Results	3-3
Category 1 AMPs	3-3
Reactor Pressure Vessel Surveillance Program R&D	3-4
Reactor Pressure Vessel Embrittlement Performance	
R&D	3-7
RPV Assessment Tools R&D	3-8
Environmental Stress Corrosion Cracking (SCC) R&D	3-10
Irradiation Assisted Stress Corrosion Cracking	
(IASCC) R&D.....	3-14
Irradiation Induced Void Swelling R&D.....	3-18
Thermal Aging Embrittlement of Cast Austenitic	
Stainless Steel (CASS) R&D	3-19
Category 2 AMPs	3-21
Section 4: STRUCTURAL AMPS R&D REVIEW	
RESULTS	4-1
Review Results	4-1
Gap Assessment Summary of Results	4-2
Category 1 AMPs	4-2
Alkali Silica Reaction (ASR) R&D	4-2
Irradiation Effects on Concrete R&D	4-4
Category 2 AMPs	4-6

Section 5: ELECTRICAL AMPS R&D REVIEW

RESULTS5-1

Review Results5-1

Gap Assessment Summary of Results5-1

Category 1 AMPs5-2

Cable Degradation from Thermal and Radiation

Exposure R&D5-2

Section 6: Conclusions6-1

Section 7: References7-1

Mechanical AMPs7-1

Structural AMPs7-11

Electrical AMPs7-12

List of Tables

Table 2-1 Category 1 AMP R&D Review Results	2-2
Table 2-2 Category 2 AMP R&D Review Results	2-3
Table 2-3 Category 3 AMP R&D Review Results	2-5
Table 3-1 EPRI Program Support for Mechanical Category 2 Aging Management Programs	3-22
Table 4-1 EPRI Program Support for Structural Category 2 Aging Management Programs	4-6



Section 1: INTRODUCTION

Objective

The Electric Power Research Institute (EPRI) provides research and development (R&D) leadership and coordination for the electric power industry. This R&D focus includes addressing the aging degradation of systems, structures, and components (SSCs) in operating nuclear power plants (NPPs). For aging degradation, the EPRI research results include identification of degradation mechanisms and their causes, assessment of degradation rates, development of options to mitigate degradation, and tools and processes to support SSC repairs or replacements as needed.

Aging management programs (AMPs) were initially developed to assist utilities in managing observed component degradation. AMPs have subsequently become central to the review and approval of license renewal applications. The current set of recommendations is contained in Section XI of NUREG-1801, Revision 2, “Generic Aging Lessons Learned Report” (the GALL Report).

The AMPs are based on: the knowledge state of the aging mechanism; degradation rates; inspection and detection techniques, flaw evaluation methodology and remediation and/or repairs. EPRI R&D project technical results provide the inspection and evaluation (I&E) basis for a number of the AMPs. The supporting EPRI reports that provide these I&E technical bases are revised as needed based on R&D results and relevant operating experience from the Industry.

Presently, the US Utilities are considering the possibility of a second round of plant license renewals, termed subsequent license renewals (SLR). For the US plants this could mean extending a plant’s operating license for an additional 20 years, to 80 years total. In preparation for the possibility of such long-term operations, EPRI established a Long Term Operations (LTO) Program. The purpose of the EPRI LTO Program is to ensure the development of technical products to support safe, reliable plant operation during any extended period of nuclear plant operation. The EPRI LTO Program integrates the R&D projects across EPRI, and coordinates with other R&D initiatives, most notably the Department of Energy (DOE) Light Water Reactor Sustainability (LWRS) Program.

To support utility preparation for subsequent license renewal activities, EPRI performed a review of the AMPs and time limited aging analysis (TLAAs) currently in the GALL Report. The purpose of this review was to determine if additional research is needed to assure that the AMPs will continue to provide a sound technical basis for long-term aging management for the period from 60 years to 80 years of operation. The evaluation was performed by subject matter experts, and the findings and results were peer reviewed by EPRI project managers and the Industry via the Nuclear Energy Institute (NEI) License Renewal Working Group membership.

Background

As of 2013, most operating U.S. nuclear plants have applied for, or received, a renewed operating license from the Nuclear Regulatory Commission (NRC) under the 10 CFR 54 license renewal process. This process provides for 20-year operating license renewals (e.g., extending the operating license from 40 years to 60 years). A key aspect of the current license renewal process is a comprehensive aging management review and implementation of aging management programs. The aging management programs prescribe inspections, monitoring, testing, evaluations and related actions to assure that SSCs subject to aging degradation continue to fully satisfy their intended functions. Effective aging management programs require a technical understanding of the aging degradation mechanism, inspection and assessment techniques, mitigation measures, and as needed guidance on repairs or replacements.

Research programs that provide the understanding and management of SSC aging have been in place for several decades:

1. EPRI Programs such as the BWR Vessel and Internals Program (BWRVIP), PWR Materials Reliability Program (MRP), Steam Generator Management Program (SGMP), Primary System Corrosion Research (PSCR), Nondestructive Examination (NDE), Chemistry and Welding Technology have published technical inspection and evaluation guidance reports on RCS materials and aging management. These reports cover the results of R&D projects performed over decades to: gain an understanding of age-related degradation modes; predict propagation rates; design examination and monitoring technologies for metallic components; mitigation strategies; flaw evaluations and assessment techniques, and as needed repair and replacement tools and methodologies.
2. The EPRI Plant Engineering Program has published technical reports on medium and low voltage electrical cable research, provided cable aging management guidelines, and facilitates operating experience exchange through the EPRI Cable Users Group.

3. The EPRI Balance of Plant (BOP) Corrosion Group conducts R&D and has published technical reports for implementation of the Flow-Accelerated Corrosion (FAC) and the Buried Tank and Pipe Management programs. Both of the FAC and Buried piping management programs are implemented via Industry initiatives. These programs continue to evolve through incorporation of operating experience and additional R&D on inspection techniques and corrosion tolerant materials for future replacements.
4. The EPRI NDE Program, in addition to examination technology in item 1), carries out R&D projects on aging management of concrete and civil infrastructures. The areas of research include: developing a mechanistic understanding of the kinetic degradation process; studying the long-term effects of temperature and radiation; investigation of various NDE tools for both the concrete and the embedded liners and developing a platform for concrete repairs and mitigation methods.

These efforts provided the technical bases for the effects of aging and, in select cases, to mitigate age-related degradation. A listing of EPRI R&D publications that provide the technical basis for aging management is included in Section 7. Many of these EPRI Reports are referenced in the GALL AMPs as acceptable approaches or methods for aging management during the license renewal period.

In addition to the EPRI Programs described above the US Department of Energy has created the Light Water Reactor Sustainability (LWRS) Program. The LWRS Program is focused on the following three goals:

1. Developing the fundamental scientific basis to understand, predict, and measure changes in materials and systems, structures, and components (SSCs) as they age in environments associated with continued long-term operations of existing reactors
2. Applying this fundamental knowledge to develop and demonstrate methods and technologies that support safe and economical long-term operation of existing reactors
3. Researching new technologies to address enhanced plant performance, economics, and safety.

Through the LWRS Program, DOE collaborates with industry and the U.S. Nuclear Regulatory Commission (NRC) in appropriate ways to support and conduct the long-term research needed to inform major component refurbishment and replacement strategies, performance enhancements, plant license renewals, and age-related regulatory oversight decisions. The DOE role focuses on fundamental science affecting aging phenomena and issues that require long-term research and are generic to reactor type. Additional information on the DOE LWRS Program can be found on the DOE or Idaho National Laboratory websites.



Section 2: AMP REVIEW

AMP R&D Review Approach

NUREG-1800 Revision 2, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants” (SRP-LR) and NUREG-1801 Revision 2, “Generic Aging Lessons Learned Report” (the GALL Report) are the primary tools used by NRC to support license renewal evaluations. Section 3 of the SRP-LR summarizes the aging management reviews (AMRs) included in the GALL Report and indicates the programs to manage the aging effects identified. Section X of the GALL Report describes generic programs to address time limited aging analyses (TLAAs). Section XI of the GALL Report describes generic aging management programs (AMPs).

This project conducted a review of EPRI R&D efforts for components and systems in scope of the current license renewal rule. GALL, Revision 2, was used as the baseline for this review effort. The purpose of the review was to both confirm that R&D programs are well matched to industry needs and identify potential gaps in those R&D projects that are on-going or planned for the near future.

AMP Review Results

The AMPs were binned into one of three categories based on the R&D supporting efforts for long-term operations. Tables 2-1 through 2-3 provide a summary in three categories:

1. R&D support to address knowledge gaps for 60 to 80 years operation to better understand and manage materials performance.
2. R&D supporting role where aging degradation is well-characterized.
3. No new R&D role identified.

Table 2-1 shows AMPs where R&D support addresses knowledge gaps for 60 to 80 years operation to better understand and manage materials performance. The types of aging factors considered are the impacts of additional time and exposure, e.g., additional thermal and pressure transients, additional neutron fluence, and additional time in the stressor environment. Building on past research efforts

(references in Section 7), EPRI has on-going and planned R&D projects that will continue to provide results to address the knowledge gaps and inform the technical basis of the AMPs. Details on the R&D programs and projects mapped to AMPs are covered in Sections 3, 4 and 5.

Table 2-1
Category 1 AMP R&D Review Results

GALL AMP ID	AMP Name	Potential LTO Impact on AMP	R&D Plan to Address
XI.M9	BWR Vessel Internals	Irradiation and environmental effects on material performance	Sections 3.3.4 and 3.3.5
XI.M11B	Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components	Environmental effects on material performance	Section 3.3.4
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal aging and possible irradiation effects on material performance	Section 3.3.7
XI.M16A	PWR Vessel Internals	Irradiation and environmental effects on material performance	Section 3.3.4, 3.3.5, and 3.3.6
XI.M31	Reactor Vessel Surveillance	Neutron fluence on reactor pressure vessel materials	Sections 3.3.1 3.3.2 and 3.3.3
XI.S6	Structures Monitoring	ASR susceptibility and irradiation effects on material properties	Section 4.3.1
XI.E1	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Combined effects of thermal and radiation exposure	Section 5.3.1
XI.E2	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits	Combined effects of thermal and radiation exposure	Section 5.3.1

Table 2-2 shows AMPs where R&D supporting role addresses opportunities where aging degradation is well-characterized. The R&D focus will be on improving the aging management process. Other AMPs in the category are tied to mitigation processes and system monitoring, i.e., the EPRI Chemistry Program Guidelines. More detailed information on the R&D projects is covered in Sections 3 through 5.

Table 2-2
Category 2 AMP R&D Review Results

GALL AMP ID	AMP Name	Programmatic or Industry Guidance Source
X.M1	Fatigue Monitoring (TLAA)	Plant specific evaluation addressing additional fatigue cycles supported by EPRI Report: Materials Reliability Program: Thermal Fatigue Monitoring Guidelines (MRP-32, Revision 1)
XI.M2	Water Chemistry	EPRI Reports :PWR Primary Water Chemistry Guidelines PWR Secondary Water Chemistry Guidelines; BWRVIP-190: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines - 2008 Revision The Guidelines have an established review and approval cycle and are periodically updated
XI.M3	Reactor Head Closure Stud Bolting	Inspection and assessment guidance in accordance with NRC Regulatory Guide 1.65 supported by on-going EPRI NDE R&D on improved inspection methods
XI.M4	BWR Vessel ID Attachment Welds	Inspection and assessment Guidelines (BWRVIP-48-A: BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines; ASME Code Section XI
XI.M5	BWR Feedwater Nozzle	Inspection and assessment Guidelines in accordance with NUREG-0619, BWRVIP-74-A: BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal, and ASME Code Section XI
XI.M6	BWR Control Rod Drive Return Line	Inspection and assessment Guidelines in accordance with NUREG-0619, BWRVIP-74-A: BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal, and ASME Code Section XI
XI.M7	BWR Stress Corrosion Cracking	Inspection and guidance in accordance with NRC Generic Letter 88-01 and BWRVIP-75-A: Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules
XI.M8	BWR Penetrations	Inspection and guidance in accordance with BWRVIP-27-A, BWRVIP-47-A and BWRVIP-49-A
XI.M10	Boric Acid Corrosion	Inspection and assessment in accordance with NRC Generic Letter 88-05 and NUREG-1823 supported by MRP-199, Reactor Vessel Head Boric Acid Corrosion Testing, and MRP-268, Reactor Vessel Bottom Mounted Nozzle Boric Acid Corrosion Testing: Design and Analysis of Full-Scale BMN Mockups
XI.M17	Flow-Accelerated Corrosion	FAC Program Implementation NSAC-202 "Recommendations for Effective Flow-Accelerated Corrosion Program"
XI.M18	Bolting Integrity	Inspection and assessment per EPRI Report: <i>Nuclear Maintenance Applications Center: Bolted Joint Fundamentals</i>

Table 2-2 (continued)
Category 2 AMP R&D Review Results

GALL AMP ID	AMP Name	Comments
XI.M19	Steam Generators	EPRI Steam Generator Management Program Guidelines for inspections, assessments and repairs; periodically updated – see listing in Section 3.4.3 under SGMP
XI.M21A	Closed Treated Water Systems	Guidance per EPRI Report: Closed Cooling Water Chemistry Guideline, Revision 1
XI.M25	BWR Reactor Water Cleanup System	BWRVIP-190: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines—2008 Revision
XI.M35	One-Time Inspection of ASME Code Class 1 Small Bore Piping	MRP -146 guidelines for location selection criteria for thermal fatigue in addition to ASME Section XI
XI.M37	Flux Thimble Tube Inspection	Inspection and assessment per MRP- 227A: Pressurized Water Reactor Internals Inspection and Evaluation
XI.M38	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Inspection and repair/replace as needed; Guidance in EPRI Report, “Nuclear Maintenance Applications Center: Passive Component Maintenance Guide for Nuclear Power Plant Personnel”
XI.M40	Monitoring of Neutron-Absorbing Materials Other than Boraflex	Monitoring and assessment technology per EPRI Report, Strategy for Managing the Long Term Use of BORAL(R) in Spent Fuel Storage Pools
XI.M41	Buried and Underground Piping and Tanks	NEI Initiative 09-14 “Guideline for the Management of Underground Piping and Tank Integrity”
XI.S8	Protective Coating Monitoring and Maintenance Program	Inspection and assessment in accordance with EPRI Reports, “Plant Support Engineering: Guideline on Nuclear Safety-Related Coatings, Revision 2” and “Field Guide: Coatings Assessment”

Table 2-3 shows AMPs where no new R&D role is identified. The inspection interval is informed by inspection results, plant specific commitments, and relevant OE.

Table 2-3
Category 3 AMP R&D Review Results

GALL AMP ID	AMP Name	Comments
XI.M1	ASME Section XI In-service Inspection	Periodic Code changes subject to NRC review
XI.M20	Open-Cycle Cooling Water System	Periodic inspections and testing. Repair/replace as needed
XI.M22	Boraflex Monitoring	Periodic inspections and testing. Repair/replace as needed
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (corrosion and loss of preload)	Periodic inspections. Repair/replace as needed
XI.M24	Compressed Air Monitoring	Periodic inspections and testing. Repair/replace as needed
XI.M1	ASME Section XI In-service Inspection	Periodic Code changes subject to NRC review
XI.M20	Open-Cycle Cooling Water System	Periodic inspections and testing. Repair/replace as needed
XI.M22	Boraflex Monitoring	Periodic inspections and testing. Repair/replace as needed
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (corrosion and loss of preload)	Periodic inspections. Repair/replace as needed
XI.M24	Compressed Air Monitoring	Periodic inspections and testing. Repair/replace as needed
XI.M26	Fire Protection	Periodic inspections and testing. Repair/replace as needed
XI.M27	Fire Water System	Periodic inspections and testing. Repair/replace as needed
XI.M29	Aboveground Metallic Tanks	Periodic inspections. Repair/replace as needed
XI.M30	Fuel Oil Chemistry	Periodic fuel oil analysis and inspections
XI.M32	One-Time Inspection	One time verification inspection and assessment
XI.M33	Selective Leaching	One time verification inspection and assessment
XI.M36	External Surfaces Monitoring of Mechanical Components	Periodic inspections. Repair/replace as needed
XI.M39	Lube Oil Analysis	Periodic lube oil analysis
XI.S1	ASME Section XI, Subsection IWE	Periodic Code changes subject to NRC review

Table 2-3 (continued)
Category 3 AMP R&D Review Results

GALL AMP ID	AMP Name	Comments
XI.S2	ASME Section XI, Subsection IWL	Periodic Code changes subject to NRC review
XI.S3	ASME Section XI, Subsection IWF	Periodic Code changes subject to NRC review
XI.S4	10 CFR Part 50, Appendix J	Periodic testing assessment
XI.S5	Masonry Walls	Periodic inspections. Repair/replace as needed
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Periodic inspections. Repair/replace as needed
XI.E3	Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Periodic inspections. Repair/replace as needed
XI.E4	Metal Enclosed Bus	Periodic inspections. Repair/replace as needed
XI.E5	Fuse Holders	Periodic inspections. Repair/replace as needed
XI.E6	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Inspection and assessment. Repair/replace as needed

In the following Section 3, summary R&D assessment results are provided for the “mechanical” AMPs as identified by the M designation in GALL Chapter XI. The assessments are subdivided by the three categories described on Tables 2-1, 2-2, and 2-3. Section 4 provides information for the “structural” AMPs as identified by the S designation in GALL Chapter XI. Section 5 provides information for the “electrical” AMPs as identified by the E designation in GALL Chapter XI.



Section 3: MECHANICAL AMPS R&D REVIEW RESULTS

Review Results

The nuclear utility industry through EPRI invests several million dollars each year in R&D projects in the area of primary system metals. The R&D activities are reviewed and prioritized using the EPRI Materials Degradation Matrix (MDM), PWR Issue Management Table (PWR IMT) and BWR Issue Management Table (BWR IMT). The MDM and IMTs are based on an expert elicitation process. A panel consisting of materials experts, industry personnel, and EPRI staff provided the key inputs for the on-going revisions. The expert panel considered relevant operating experience, information from newly published and ongoing research projects worldwide, consequence of failure, and the availability of mitigation strategies, in developing the results.

Importantly, the EPRI effort is well coordinated under the policy provisions of NEI 03-08 in which all US nuclear plant licensees have made a formal commitment to implement this Initiative. The objective of the NEI 03-08 Initiative is to assure safe, reliable and efficient operation of the U.S. nuclear power plants in the management of materials issues. Each licensee has endorsed, supporting and meeting NEI 03-08, "Guideline for the Management of Materials Issues". This initiative became effective on January 2, 2004.

The Initiative has provided:

- A consistent management process
- Prioritization of materials issues
- Proactive approaches
- Integrated and coordinated approaches to materials issues

Actions required by this initiative include:

- Commitment of executive leadership and technical personnel
- Commitment of funds for materials issues within the scope of this Initiative
- Commitment to implement applicable guidance documents
- Provide for oversight of implementation

The systematic approaches implemented under NEI 03-08 to address and prioritize materials issues are the EPRI MDM and IMTs. The MDM and IMTs:

1. Develop a fundamental understanding of the degradation phenomena/mechanisms, and determine materials (and locations) that are known, or can logically be assumed, to be susceptible to aging/degradation phenomena when exposed to the operating environment.
2. Conduct generic operability and safety assessments for the locations of the various materials potentially susceptible to damage/degradation phenomena.
3. Develop inspection and evaluation guidelines and technology for the identified locations, starting with those for which the potential consequences of failure are most severe.
4. Evaluate available mitigation options and, if necessary, develop additional options.
5. Evaluate repair/replace options and, where necessary, encourage/support the development of additional options.
6. Monitor, evaluate and feedback plant operating experience.
7. Obtain regulatory acceptance of the items above and support licensees on plant specific applications as needed.

The MDM focuses on the development of fundamental understanding of the degradation phenomena/mechanisms, based on the materials/environment combination. Expert elicitation, laboratory studies, and field experience were used to identify potential mechanisms by which each of the materials might degrade.

The PWR IMT and BWR IMT are component based evaluation of consequence of failure. This component based approach also emphasizes considerations of mitigation strategies, repair/replacement, inspection and evaluation guidelines, and regulatory requirements. All these considerations will be captured in IMT gaps, which will then be prioritized. The prioritization of IMT gaps provides a basis for industry to prioritize R&D efforts to address materials reliability issues and LTO concerns effectively.

The MDM and IMTs are living documents. In support of the LTO Program an LTO 'flag' was first added to the MDMs during the 2010 revision indicating if there is on-going work, or if additional work is needed to support 60 to 80 years of operation. This flag was retained and updated for the 2013 revision to the MDM report. The objectives of this addition to the MDM were to:

1. Identify applicable degradation mechanisms for operation to 80 years, and assess the extent to which applicable degradation mechanisms are understood
2. Evaluate the state of industry knowledge worldwide associated with mitigation of degradation mechanisms
3. Address any concerns related to regulatory and licensing renewal considerations for the period to 80 years

The primary systems metals projects are managed within EPRI under either the BWR Vessel Integrity Program (BWR VIP), the PWR Materials Reliability Program (MRP), or the Primary System Corrosion Research (PSCR) program. Steam Generator projects are managed by the Steam Generator Reliability (SGMP) program, and inspection projects are managed by the Non-destructive Examination (NDE) program. Supporting the materials projects are the PWR and BWR Primary and Secondary Water Chemistry Programs and the Welding Technology Program.

The EPRI research projects are augmented by coordinated and collaborative research with the US Department of Energy (most notably with the LWR Sustainability Program), reactor vendors, universities and other independent researchers, NSSS Owners Groups, and international organizations supporting related research in aging of nuclear plant structures, systems and components (SSCs).

Gap Assessment Summary of Results

Based on the MDM process and inclusion of the LTO ‘flags’ where R&D gaps were identified as needed for 60 to 80 years, the projects were prioritized within the other projects in that program area. In some cases, such as the need for a repair technique for highly irradiated materials, a new project was started. Throughout this project the findings from the AMPs and TLAA reviews were cross-checked to ensure completeness. The results of both MDM, via the expert solicitation process, and the review of the AMPs and TLAA provided consistent results. All of the findings from the AMP review process where additional R&D would continue to improve the knowledge basis to 80 years of operation were already flagged in the MDM. A number of the projects were already started, such as the further investigation on IASCC and the PWR Coordinated Reactor Vessel Surveillance Program (CRVSP) projects.

The following subsections summarize the key technical issues where R&D work is in progress or planned.

Category 1 AMPs

On-going and planned EPRI led R&D projects are described below that will provide valuable information to address knowledge gaps and provide insights into material performance and aging management for long-term operations. The key driver for research is to envelope the aging factors out to 80 years of operation. These factors include neutron fluence effects, thermal effects, corrosion effects, stress effects, and related or combined factors that are time dependent. Note that the same research topic (e.g., extension of IASCC data to encompass operation to 80 years) impacts the technical basis for multiple AMPs.

Reactor Pressure Vessel Surveillance Program R&D

Issue: Provide adequate surveillance capsule data to confirm embrittlement performance of operating plants through 80 years of service.

Impact on AMP XI.M31 Reactor Vessel Surveillance

For plants considering long-term plant operations, neutron embrittlement of the reactor pressure vessel may pose a challenge. While existing reactor vessel surveillance programs are in place to monitor materials aging and degradation, surveillance data are needed at high fluence levels to make robust embrittlement predictions. Selected previous work that provides the foundation for research on RPV embrittlement issues, including the need for surveillance specimen data is summarized as follow:

Materials Reliability Program: Reactor Pressure Vessel Integrity Primer (MRP-278)

This RPV integrity primer is a reference for reactor vessel embrittlement issues and management. It provides the basic information and tools to design and implement an embrittlement management program and to understand and comply with regulations for maintaining adequate vessel fracture toughness. The information in the handbook can be used in the following ways:

- To obtain background information on issues related to reactor vessel embrittlement and integrity
- To obtain a basic understanding of fracture mechanics as it is applied to vessels
- To train utility engineers and planners who have the responsibility for managing vessel embrittlement and integrity
- To understand current requirements and future changes to U.S. Nuclear Regulatory (NRC) regulations relevant to embrittlement management
- As a reference when developing a plant-specific embrittlement management program

Materials Reliability Program: Developing an Embrittlement Trend Curve Using the Charpy "Master Curve" Transition Reference Temperature (MRP-289)

A more accurate modeling approach was investigated in order to characterize the fracture toughness transition behavior of pressure vessel steels. This report addresses the continued development of strategies that may ultimately lead to direct characterization of fracture toughness of a reactor pressure vessel (RPV) for integrity assessments. The project team normalized CVN energy transition data for U.S. surveillance data to express temperature relative to a reference temperature and calculated the 28J transition index temperatures from the mean tanh curve (T_{28J}-tanh). To remove the influence of upper shelf data on this

attempt to fit a curve to the transition data, the team developed a filtering scheme to select all data exhibiting a fracture appearance of 60% shear or below and calculated a best fit exponential curve to the filtered data. The resulting equation served as the basis of data fits for developing an embrittlement trend curve for predicting degradation in the Charpy-based transition temperature for RPV steels with irradiation.

Materials Reliability Program: Static Tensile Testing of a Pressure Vessel Steel Irradiated to Assess Through-Wall Attenuation of Radiation Embrittlement (MRP-333)

A unique experiment was conducted to quantify the changes in fracture toughness properties of several low-alloyed ferritic reactor pressure vessel (RPV) materials as a result of radiation embrittlement and attenuation of the neutron flux and change in energy spectrum corresponding to different depths in the wall thickness of a nuclear RPV. A capsule consisting of an 18-layer array of 10-mm-thick test specimens was irradiated under carefully controlled conditions at a test reactor. The tensile test results confirmed that the hardening of the JRQ (a moderate copper content A533B-1 steel) irradiated steel decreased with increasing distance from the inner irradiated surface. The trend is consistent with the previous hardness, Charpy V-notch, and Master Curve fracture toughness test results. The through-thickness trend is generally consistent with all of the mechanical property measurements indicating that use of displacements per atom-adjusted fluence is reasonable and generally conservative for assessing the change in mechanical properties.

BWRVIP-86, Revision 1-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan

This report describes the boiling water reactor (BWR) Integrated Surveillance Program (ISP). Based on recommendations from BWR Vessel and Internals Project (BWRVIP) utilities, it was concluded that combining all separate BWR surveillance programs into a single integrated program would be beneficial. In the integrated program, representative materials chosen for a specific reactor pressure vessel (RPV) can be materials from another plant surveillance program or other source that better represents the target vessel materials. The basis for the integrated program was established in BWRVIP-78, and the implementation plan was given in BWRVIP-86. BWRVIP-116 proposed the extension of the ISP into the license renewal period. With the license renewal ISP approved by the Nuclear Regulatory Commission (NRC), BWRVIP-86, Revision 1 merged BWRVIP-86-A and BWRVIP-116 into this single, updated implementation plan for the ISP covering plant operation to 60 years.

BWRVIP-192: BWR Vessel and Internals Project, BWR System Pressure Test Limitations

This report examines the issues and limitations on performing the BWR pressure test at high temperatures and discusses options for managing the issue. The results of the evaluation suggest that pressure tests at higher temperatures (exceeding 212°F) can be performed in BWRs and there are no inherent difficulties that cannot be overcome using available technology.

Building on the above research results and related work, additional R&D effort has been identified and is in progress or planned. Some reactors will experience peak end-of-license fluences as high as 9×10^{19} n/cm², but the embrittlement trend curves in use today are based on little or no data above 5×10^{19} n/cm². Therefore, a need exists to obtain high fluence PWR surveillance data to validate or revise embrittlement trend correlations (ETC) applicable for the high fluence regime. Without the availability of high fluence PWR surveillance data, it may be necessary to use an overly conservative ETC (or an ETC with a high margin at high fluence) that could constrain plant pressure-temperature operating curves, increasing startup and shutdown times and costs.

EPRI's Materials Reliability Program is pursuing two efforts to address this knowledge gap. The first is described in report MRP-326, Coordinated PWR Reactor Vessel Surveillance Program Guidelines (EPRI Product 1022871). This program (CRVSP) identified 13 U.S. PWRs that will defer the withdrawal of selected remaining capsules in order to test the capsules at higher fluences. The program effectively increases the fluence of capsules that are already planned and prevents the testing of capsules below 3.0×10^{19} n/cm² if it is possible for the capsules to achieve that fluence by 2025. This project will be effective for increasing the fluence of future surveillance test data, but no additional capsules will be tested, other than those already in the vessels.

The second initiative, currently in development, is the PWR Supplemental Surveillance Program (PSSP). This program will design, fabricate, irradiate, and test two supplemental capsules containing previously irradiated PWR materials. This program will generate new data. Because the capsules will be populated with previously irradiated PWR surveillance specimens and will add fluence to those specimens, this program will significantly increase the amount of high-fluence PWR surveillance data available for developing embrittlement trend curves for the fluences representative of the end of subsequent license renewal conditions (e.g., to 80 years). The plan is to install the capsules into operating PWRs in 2014-2015 and test them in ~2025, thereby generating ~24 new transition temperature shift (TTS) data points to support embrittlement characterization. This effort will ensure adequate high fluence data is available to develop an embrittlement trend correlation (ETC) applicable to PWR vessels for LTO.

In summary, for PWRs, this issue will be addressed by the MRP projects currently in progress to generate high fluence PWR surveillance data, both by optimizing the withdrawal schedules of remaining plant surveillance capsules and by fabricating and irradiating a supplemental surveillance capsule.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
Coordinated Reactor Vessel Surveillance Program (CRVSP) development						
		CRVSP Capsule Retrieval and Testing (out to 2029)				
PWR Supplemental Surveillance Program (PSSP) development						
		PSSP Capsule irradiation (out to 2024)				
2013	2014	2015	2016	2017	2018	2019

The Boiling Water Reactor Vessel and Internals Program (BWRVIP) manages an Integrated Surveillance Program (ISP) to provide confirmation reactor pressure vessel specimens encompassing the US BWR fleet up to 60 years of operation. This existing program will be evaluated and, as appropriate, updated or modified to extend surveillance specimen coverage to 80 years of operation.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
Develop Specimen Extension to Address Operation to 80 Years BWR Vessel Surveillance						
		Program Development and Approval				
Implement BWR Surveillance Program for LTO						
					BWR Specimen Irradiation, Retrieval, and Testing(out to 2030)	
2013	2014	2015	2016	2017	2018	2019

Reactor Pressure Vessel Embrittlement Performance R&D

Issue: High fluence embrittlement data, including applicability of Test Reactor embrittlement data, and improved characterization of embrittlement phenomena through 80 years of service.

Impact on AMP XI.M31 Reactor Vessel Surveillance

Additional R&D effort was identified to complement the work on surveillance specimen testing. Extensive sets of embrittlement test data have already been generated from both commercial reactor and test reactor irradiation of specimens. The data sets do, however, represent significant variations in neutron energy spectrum and other conditions. To address this item, the Materials Reliability Program (MRP) has acquired archived non-irradiated RPV steels for exposure in Idaho National Laboratory's (INL's) Advanced Test Reactor (ATR) as part of a DOE Light Water Reactor Sustainability Project (LWRS) experiment named ATR-2. This is a joint project with the University of California at Santa Barbara (UCSB) and INL in which the specimens are currently being irradiated. The

objective of this work is to supplement the power reactor data with test reactor data under controlled flux and fluence conditions to develop improved embrittlement models that account for a very wide range of flux. The major task of this joint project is exposing small specimens (e.g., mini-tensile and disc compact tension) of unirradiated archive PWR surveillance materials to a range of fluences under several different fluxes and temperatures. Post-irradiation microstructure examination studies will be performed by UCSB after one year of exposure in the ATR-2. This work will significantly improve the accuracy of using data from test reactor irradiation to predict embrittlement performance of commercial reactor vessels for LTO.

MRP has a project currently underway to develop an Embrittlement Trend Correlation (ETC) using the Charpy Master Curve fit to the U.S. power reactor surveillance database as an interim step to establishing a direct correlation between the Charpy surveillance data-derived ETC and a fracture toughness-based ETC. The primary focus of this project is to develop and refine an ETC using the Charpy Master Curve fit to the U.S. power reactor surveillance database. The research objective is to bridge the gap between the Charpy V-notch energy evaluation approach and the fracture toughness Master Curve (MC) approach. The ultimate goal is to transition away from Charpy-based correlative approaches to assessing fracture safety of RPVs, and toward the direct use of fracture toughness in these assessments. One of the largest challenges to making this transition is that the majority of irradiated data is from Charpy surveillance data, with very little fracture toughness surveillance data available. This work will facilitate development of correlations between embrittlement-related Charpy shift and fracture toughness shift and support a more technically based approach to margins traditionally applied to shift predictions based on Charpy data.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
Complete ATR Irradiations and Post Irradiation Testing						
Irradiation, Testing, and Reporting						
Develop and Refine Charpy Master Curve Embrittlement Trend Correlation						
Updated ETC						
2013	2014	2015	2016	2017	2018	2019

RPV Assessment Tools R&D

Issue: High fluence impact on Equivalent Margin Analyses, Alternate PTS Rule Assessment for PWRs, and Assessment of Beyond Beltline Components and Embrittlement Predictions through 80 years of service.

Impact on AMP XI.M3.1 Reactor Vessel Surveillance

Although current assessment methodology and guidance is available for reactor pressure vessels to address items in the AMP, extension of those assessments to encompass operation to 80 years was identified as a knowledge gap. For these items impacting PWRs, the MRP currently has a project to evaluate and analyze the potential long term irradiation effects for materials outside the traditional RPV beltline – specifically, PWR nozzles. A collaborative effort with CRIEPI is planned for atomic probe tomography (APT) on high fluence PWR specimens. In addition, a collaborative effort for study of a decommissioned PWR inlet/outlet nozzle is proposed with CRIEPI and DOE LWRSP.

A primary objective of this project is to evaluate potential embrittlement of the PWR nozzle shell course region for LTO. Recently, MRP studied the flux/fluence attenuation profile in the inlet/outlet nozzles. Understanding that profile is necessary for ensuring that a plant's pressure-temperature limit curves bound all the portions of the RPV, as required by 10 CFR 50 Appendix G. Analytical results suggest that the exponential attenuation profile specified by Regulatory Guide 1.99, Rev 2 for the beltline wall may not be appropriate for the nozzle. To benchmark the analytical results and improve understanding of embrittlement mechanisms in the extended beltline, MRP and CRIEPI are proposing a collaborative study with DOE LWRSP to assess the through-wall material properties of an inlet or outlet nozzle from the decommissioned Zion nuclear plant.

A second objective of this project is to examine the microstructure of high-fluence U.S. PWR surveillance specimens by APT and other methods in order to characterize embrittlement mechanisms at LTO fluence levels. The objective is to provide forewarning of new embrittlement mechanisms (if any) for PWR materials at high (LTO) fluence levels. This work is being planned in a collaborative effort with CRIEPI (Japan).

For these items impacting BWRs, the BWR Vessel and Internals Program (BWRVIP) plans a project to evaluate several RPV integrity issues for LTO including the risk associated with conduct of leak tests at or near RT_{NDT} ; justification for continued relief from circumferential weld inspection requirements per BWRVIP-05 and BWRVIP-74-A; demonstration that vessel beltline axial weld failure frequency meets risk targets (per the license renewal applicant action item in the NRC Safety Evaluation for BWRVIP-74-A) and evaluation of the equivalent margins analysis (EMA) to assure that all beltline RPV materials including nozzles satisfy 10CFR50 Appendix G.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
PWR Beyond Beltline Evaluations						
		Evaluate Embrittlement of PWR Nozzle Shell Course				
APT examination of PWR surveillance specimens						
BWR Evaluations Beyond RPV Beltline Embrittlement						
		BWR Vessel Integrity Assessment for LTO				
2013	2014	2015	2016	2017	2018	2019

Environmental Stress Corrosion Cracking (SCC) R&D

Issues: Assessment of mitigation needs and timing for addressing Primary Water Stress Corrosion Cracking (PWSCC) uncertainties for T_{cold} reactor heads fabricated from Alloy 600/82/182 materials; Assessment of long-term performance with replacement materials (Alloy 690/52/152); Long-Term Stress Improvement Stability; XM-19 Irradiated Materials Data / SCC Properties; X-750 Irradiated Materials Data / SCC Properties; SCC Mechanistic Understanding and Prediction of SCC Initiation Trends

Impact on AMP XI.M9, BWR Vessel Internals, and M11B, Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components

Stress corrosion cracking (SCC) has long been a significant degradation mechanism impacting metallic components in PWRs and BWRs. For aging management, two key technical bases are mechanistic understanding of crack initiation that serves to identify where and when to inspect components and/or locations on components susceptible to degradation and crack growth rates and behavior that serves to define appropriate inspection intervals for those locations. The existing AMPs that manage SCC have established this level of information on inspection locations and frequency, but a gap has been identified regarding the continuing application of existing guidance for the LTO period.

Selected previous work that provides some of the technical basis for the on-going R&D work is summarized as follows:

BWRVIP-14-A: BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals

This report provides a method for assessment of crack growth in BWR stainless steel shrouds and other stainless steel internal components. Based on extensive crack growth data collected from several sources, an empirical through-wall crack growth correlation has been developed for use in the evaluation of BWR stainless steel internals for fluences $< 5 \times 10^{20}$ n/cm². The correlation is applicable for weld-sensitized components and is bounding for nonsensitized components. The report provides analysis and measurements of residual stresses in core shroud welds and discusses fracture mechanics methods employed in determining stress intensity factors. The report provides three alternative methods for crack growth evaluation. Using conservative ECP and conductivity estimates, results confirm that American Society of Mechanical Engineers (ASME) Section XI safety margins are not compromised by extended operation of core shrouds with IGSCC indications.

BWRVIP-59-A: BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals

This BWRVIP report provides a methodology for assessing crack growth in BWR nickel base alloy shroud support structures and in other nickel base alloy components. The approach used was to determine through-thickness residual stress and stress intensity distributions for core support structure welds representative of the BWR fleet. Both experimental and analytical techniques were used to determine the residual stress distributions with reasonable agreement obtained. These residual stresses were used in a fracture mechanics analysis to determine weld-specific through-wall stress intensity distributions. Crack growth distribution curves obtained from field and laboratory data and the stress intensity factor distributions were used to perform crack growth evaluations for the individual welds.

Stress Corrosion Crack Growth Rate Measurements on Unsensitized Type 304 Stainless Steel in 288-Degree-Celsius Water

This study compares the crack growth response of unsensitized and sensitized Type 304 stainless steel in laboratory simulations of the BWR environment. The results confirmed much of the earlier work performed by EPRI and others, i.e., that unsensitized austenitic stainless steels will exhibit IGSCC in the BWR environment. However, the growth rate is significantly reduced from that observed for sensitized austenitic stainless steels.

Program on Technology Innovation: Environmentally Assisted Cracking in LWR Structural Materials, 2011 PEACE-E Annual Report

This report describes the interim results of PEACE-E, a joint research program. The program addressed environmentally assisted cracking of light water reactor structural materials in pressurized water reactor (PWR) and boiling water reactor (BWR) environments. The project's goal is to develop an improved fundamental understanding of stress corrosion cracking crack initiation and growth processes in stainless steels and nickel-base alloys in BWR and PWR environments and to develop mechanistically based models to predict stress corrosion cracking, to identify more corrosion resistant materials, and to propose effective mitigation methods.

BWRVIP-218: BWRVIP Vessel and Internals Project, Alloy X-750 Characterization Study

This report provides a review of the alloy X-750 components, as well as their properties and fabrication methods, that are present in operating BWRs. This information is used to evaluate the variability of the materials and to establish a basis from which prototypical alloy X-750 microstructures can be identified and/or manufactured for future testing to accurately assess the performance of this material in BWR environments. Compilation and evaluation of available mechanical property data, fabrication procedures, and field experience can provide valuable information to designers to improve SCC resistance for this material.

The research efforts addressing primary system components are seeking to develop a better understanding of the crack initiation and propagation processes, improved predictive models, and effective countermeasures against stress corrosion cracking appropriate to application for LTO.

The objectives of this project include:

- Understand the evolution of metal surface resulting from interaction with LWR environments, and identify the key process leading to cracking
- Understand the metallurgical evolution of component materials under LWR environments, and identify the mechanisms leading to decreased fracture resistance in component materials
- Develop improved prediction models and evaluation methodologies for assessing the reliability of LWR structural materials
- Develop strategies to mitigate the risk of SCC degradation and to extend component life

Considering the complexity of the study and the limitation of experiment capabilities, achieving full understanding on environmental SCC degradation mechanism remains a challenge. As noted, research results over the past several decades have contributed to improved understanding and allowed the industry to

develop effective aging management strategies. This effort will be continued to support LTO. Specifically, the LTO work will be coordinated with both planned work within the PSCR program and work at the Materials Aging Institute. Key research projects include:

- Detailed investigations of the oxide film role in crack initiation have been started and are expected to produce significant results important for LTO. Many advanced characterization capabilities (e.g. atomic probe tomography and synchrotron X-ray measurements) and methodologies have been adopted for film study, including in-situ and ex-situ capabilities. The deeper understanding on oxidation process will help correlate SCC susceptibility with materials and environment variability.
- PSCR shall continue to investigate how the materials inhomogeneity, including those due to fabrication, affects crack initiation and propagation. The fabrication-related materials inhomogeneity includes surface chemical, mechanical, and metallurgical inhomogeneities, as well as local strain/stress inhomogeneity. Investigation on the effect of cold work on the potential risks of SCC initiation will link the practical fabrication processes such as welding (from weld shrinkage strains) and grinding to SCC susceptibility. The study on the damage process prior to crack initiation will help to define the bounding conditions associated with cracking.
- To maximize parametric studies on crack initiation and propagation, EPRI will continue to participate in the internationally cooperative POLIM program. The cooperative nature of POLIM program consists of 7 international organizations and supports full scale parametric studies on SCC crack initiation and propagation.
- Building on both existing and expected research results, EPRI will establish theoretical models for crack initiation and propagation, taking into account material/environment combinations encountered in nuclear plants, realistic surface strains and realistic industrial surface finishes, environmental chemistries, and corrosion potentials. As additional data becomes available from on-going research it will be incorporated to refine the predictive models.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
In-situ surface oxide film characterization and correlation						
Summarize results of in-situ surface oxide film composition and impedance properties as functions of materials/LWR environment combinations, including the effects of cations						
Define Damage Processes Prior to Crack Initiation in Ni Alloys						
Summarize the results of in-situ surface oxide structure and oxidation kinetics						
Establish correlation between oxidation and crack initiation						
Local strain-stress behavior associated with crack						
Results from in-situ synchrotron x-ray stress measurement						
Establish correlation between strain rate and crack growth rate						
Parametric study and development of mitigation strategy						
Summary of parametric experiments on crack growth rate						
Develop and validate mitigation strategies						
Modeling						
Environmental-assisted cracking crack growth prediction model						
Environmental-assisted cracking crack initiation model						
2013	2014	2015	2016	2017	2018	2019

Irradiation Assisted Stress Corrosion Cracking (IASCC) R&D

Issue: Prediction of Irradiation Assisted Stress Corrosion Cracking (IASCC) initiation and growth trends with increasing neutron fluence and exposure to environment

Impact on AMP XI.M9, BWR Vessel Internals, and AMP XI.M16A, PWR Vessel Internals

Irradiation Assisted Stress Corrosion Cracking (IASCC) has been observed in reactor core internal structures fabricated from austenitic stainless steels in both Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR). As in the case for SCC, two key technical bases for effective aging management of this degradation mechanism are mechanistic understanding of crack initiation that serves to identify where and when to inspect components and/or locations on

components susceptible to degradation and crack growth rates and behavior that serves to define appropriate inspection intervals for those locations. The existing AMPs that manage IASCC have established this level of information on inspection locations and frequency, but a gap has been identified regarding the continuing application of existing guidance for the LTO period. IASCC, like all stress corrosion cracking phenomena, requires critical combinations of applied stress or strain, environmental chemistry and metallurgical structure to occur. However, the added feature of IASCC is that neutron irradiation significantly alters the metallurgical microstructure and ionizing (α , β , γ and neutron) radiation can modify the environmental chemistry.

Extensive research efforts have already been completed. Selected summaries of some of that work are provided as follow:

BWRVIP-99-A: BWR Vessel and Internals Project, Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components

This report was prepared to provide a crack growth methodology applicable to irradiated BWR stainless steel internal components for fluence levels from 5×10^{20} n/cm² to 3×10^{21} n/cm². Investigators first reviewed the mechanistic basis for the effects of irradiation on crack growth behavior. Next, they assembled and analyzed data from crack growth measurements on core shrouds in several BWRs. They also summarized the available laboratory data on crack growth in irradiated stainless steels in BWR environments. Their third step was to discuss the significant role of irradiation-induced stress relaxation, which tends to counteract the effect of irradiation on IASCC susceptibility. Based on this collective understanding, investigators developed crack growth curves for the subject fluence range.

BWRVIP-154, Revision 2: BWR Vessel and Internals Project, Fracture Toughness in High Fluence BWR Materials

Over the last several years, the Electric Power Research Institute (EPRI) has funded several projects to assess the effect of irradiation on the fracture toughness of stainless steel. These projects include experimental work to determine the change in toughness due to irradiation and analyses to assess the integrity of irradiated BWR cracked core shrouds. As the fluence of the fleet increases, new fracture toughness data at high fluences are needed to update the database and verify the accuracy of methods for predicting toughness of irradiated stainless steels in order to determine structural integrity and schedule appropriate inspections.

BWRVIP-221: BWR Vessel and Internals Project, Crack Growth in High Fluence BWR Materials

The Boiling Water Reactor Vessel and Internals Project (BWRVIP) has developed a methodology to evaluate crack growth in irradiated stainless steel components in the BWR vessel. The purpose of this study was to generate crack growth data on irradiated stainless steels that can be used to extend the flaw

evaluation methodology to higher neutron doses. This report describes Phase 1 of the work. Work on Phase 2 is in progress: specimens removed from the core shroud weld and heat-affected zone of a retired BWR will be tested in BWR environments. After the completion of Phase 2, CGR data from this project as well as other related projects will be compiled, screened, and used to update the CGR disposition curves described in the EPRI report BWRVIP-99-A.

Materials Reliability Program: Verification of Material Constitutive Model for Irradiated Austenitic Stainless Steels (MRP-259-Rev. 1)

This report summarizes results of the verification of the EPRI material constitutive model for irradiated austenitic stainless steels (USERMAT) that is used for functionality analysis of pressurized water reactor (PWR) internals. The objects of those modeling analyses were austenitic stainless steel reactor internals components that were judged to be susceptible to irradiation-induced degradation in mechanical and/or physical properties. Modeling results were verified by vendors and a contractor.

Building on the current mechanistic understanding of IASCC and data on crack growth rates, additional research will be completed to address the potential gap relative to IASCC behavior under irradiation and environmental conditions expected through 80 years of operation.

Specific LTO Project work on expanding the mechanistic understanding is being jointly executed by EPRI and the US Department of Energy (DOE). The objectives of this LTO Project include:

- Role of solutes in crack initiation
- Role of solutes in crack propagation
- Role of starting microstructure in crack initiation
- Role of starting microstructure in crack propagation
- Effectiveness of proton irradiation in forecasting relative crack growth rate (CGR) behavior
- Comparison of crack initiation following proton and neutron irradiation
- Comparison of crack initiation and crack growth in neutron-irradiated samples as a function of solute addition or starting microstructure
- Structure-property relationship for neutron irradiated alloys
- Effect of alloy, alloy purity, heat and dose on crack growth and crack initiation the IASCC susceptibility in neutron irradiated materials

The combined information from this project and the existing Cooperative IASCC Research (CIR) program will be used to develop a comprehensive understanding of the fundamentals of IASCC for LTO. This improved mechanistic understanding of IASCC will reduce uncertainties, allow more

effective evaluation of component reliability under irradiated environments, support development of predictive models for improving materials aging management, and allow effective mitigation, repair or replacement strategies for BWR and PWR internals.

Crack growth rate tests and complementary microstructure analysis will provide a more complete understanding of IASCC by building on past EPRI-led work for the Cooperative IASCC Research (CIR) group. In the CIR program, austenitic stainless steels with various solute additions were previously irradiated in a test reactor. These neutron-irradiated stainless steels will now be tested in a corrosion testing laboratory at the University of Michigan. The crack growth rate (CGR) tests will be performed on round compact test (RCT) specimens, and the constant extension rate tensile (CERT) tests will be performed on tensile specimens. The CGR test results and CERT test results will be cross compared to evaluate the IASCC susceptibility. In addition to testing the neutron-irradiated stainless steels, the similar stainless steels irradiated to the similar fluence by proton irradiation will be tested by CERT. The cracking susceptibilities associated with neutron irradiation and with proton irradiation will be cross compared. The role of localized deformation on IASCC susceptibility will be investigated. This project is planned for a five year duration.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
Mechanistic Understanding of IASCC						
Report on key factors in IASCC initiation and propagation of austenitic alloys						
Report on improved IASCC crack growth prediction models						
	Development of IASCC growth rate and remaining life models					
Crack Growth Rate Testing and Data						
Experimental work and data development						
		Data evaluation and recommendations				
2013	2014	2015	2016	2017	2018	2019

Irradiation Induced Void Swelling R&D

Issue: High neutron fluence (to bound 80 years of operation) material performance data for void swelling in SS.

Impact on AMP XI.M16APWR Vessel Internals

EPRI work on IASCC, swelling, and phase transformations is coordinated under the Materials Reliability Program for PWRs and under the BWR Vessel and Internals Program for BWRs. Significant work, including international cooperative programs, is directed under these two EPRI Programs. For example, the Gondole Project is a multi-national effort that includes EPRI work that specifically seeks to develop data via test reactor irradiation of prototypical materials to characterize irradiation-induced swelling degradation effects in stainless steels. The current Phase 2 of the project seeks to drive irradiation to doses of 30 dpa. This phase is in progress with completion expected in 2018.

Additional previous work is summarized as follows:

Material Reliability Program Technical Basis Document Concerning Irradiation-Induced Stress Relaxation and Void Swelling in Pressurized Water Reactor Vessel Internals Components (MRP-50)

This report describes current knowledge of these two potential degradation mechanisms, available relevant data and known functional relationships, and a qualitative assessment of these two mechanisms' combined and separate effects on PWR internals components. The project assessed that swelling problems, if they arise in PWR core internals, would be highly localized, occurring in the higher flux and temperature locations. Since these locations are highly localized, swelling, regardless of its magnitude, may not have any effect on long-term functionality of reactor internals components. Stress relaxation, in general, mitigates the effect of swelling.

In Situ NDT Measurements of Irradiation-Induced Swelling in PWR Core Internal Components: Phase 3: Correlation of Void Swelling and Material Properties of Austenitic Steels

This report describes the correlation of swelling with physical material properties (such as electrical resistivity, ultrasound velocity, and elastic constants) of austenitic stainless steels. In the third phase, reported here, the project team performed NDT evaluations on a large set of materials with known swelling to determine the correlations between swelling and material properties. The team also developed analytical expressions for such correlations.

Materials Reliability Program: Cluster Dynamics Prediction of Void Swelling in Austenitic Stainless Steels under PWR Conditions (MRP-321)

This report describes the development of an improved physics-based model for the prediction of void swelling in austenitic stainless steel materials exposed to neutron irradiation under typical PWR operating conditions. This physics-based model is based on the cluster dynamics approach that models the evolution of defect clusters as a function of dose rate, accumulated dose, and temperature, with key model parameters calibrated using laboratory data. The void swelling model developed in this project will be incorporated into the existing global PWR stainless steel constitutive model (MRP-135, Revision 2 and associated IRADSS software). This model is used to determine the effects of irradiation on the rate of degradation in stainless steel used in reactor internal components. Output from the global irradiation effects model is used as input for the component functionality analyses that form the basis for the inspection guidance included in the PWR internals inspection and evaluation guidelines (MRP-227).

As noted previously, there is strong research coordination between EPRI and the DOE LWRS Program efforts. EPRI research will focus on the completion of the Gondole Phase 2 specimen irradiation (to 30 dpa) and testing. This work will be coordinated with specific LWRS Program work tasks on detailed microstructural analysis of swelling in key samples and components (both model alloys and service materials) and development and validation of a phenomenological model of swelling under LWR conditions. A similar set of LWRS Program tasks has been developed for phase transformation understanding.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
Gondole Phase 2						
Gondole Phase 2 Specimen Irradiation and Testing						
					Final Report on Gondole Phase 2 on Void Swelling	
2013	2014	2015	2016	2017	2018	2019

Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) R&D

Issue: Material embrittlement (fracture toughness) behavior with increasing exposure to combined thermal and environmental factors.

Impact on AMP XI.M12 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)

In the area of CASS degradation, the Primary System Corrosion Research (PSCR) Program will investigate fundamental mechanisms of thermal aging in ferritic-austenitic stainless welds and CASS material and at LWR temperatures, as well as effects of thermal aging on mechanical properties and corrosion resistance.

This research will build on previous investigations. Selected prior work is summarized as follows:

BWRVIP-234: BWR Vessel and Internals Project, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals

The purpose of this report is to evaluate the potential synergistic effects of thermal aging and neutron embrittlement of BWR internal components fabricated of cast austenitic stainless steels (CASS). The report also discusses recommendations for determining if augmented inspections are necessary to detect degradation. The project team performed an evaluation of the various CASS components to determine if thermal aging and/or irradiation embrittlement could occur. They developed a screening process that included assessment of ferrite level, fluence, toughness, stress level and results of existing BWRVIP inspections. Given that many components will reach fluence levels where irradiation may have a potential effect on fracture toughness, the assessment includes the stress levels in each component and the predicted fracture toughness properties. The evaluation shows that all the BWR CASS components have ferrite levels below the level for which aging embrittlement is a concern during the initial license renewal period.

Materials Reliability Program: Thermal Aging and Neutron Embrittlement Assessment of Cast Austenitic Stainless Steels and Stainless Steel Welds in PWR Internals (MRP-276)

This report summarizes the results of a review of the component items that could be affected by a synergy between thermal aging and irradiation embrittlement for PWR internals. Existing evaluations of thermal aging in CASS materials and stainless steel welds were supplemented with a summary of available assessment and testing data on both irradiated CASS materials and irradiated austenitic stainless steel welds. A list of reactor internals items that exceed the NRC criterion was developed. After potential requirements were identified for fracture toughness data on CASS and stainless steel welds in PWRs to complete the analysis for an aging management program, any gaps between available test data and analysis requirements were identified and prioritized. Finally, future testing and structural analysis recommendations—including test conditions—were made to close the identified gaps, and structural evaluations for these CASS components and for weld materials were proposed.

In this research area, previously identified gaps (per MRP-276, above) will be further investigated. This work will also be closely coordinated with related work planned for the DOE LWRS Program. Primary EPRI research will focus on predictions for the integrity of the CASS components of LWR NPPs during the extended service life. LWRS Program efforts on mechanical and microstructural data, accelerated aging experiments and computational simulations, and data obtained from operational experience will contribute to this coordinated research.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
Integrity of CASS Components for LTO						
		Evaluation and Modeling				
2013	2014	2015	2016	2017	2018	2019

Category 2 AMPs

This category covers AMPs addressing degradation modes that are well-characterized, or implement established Programs. Specific R&D needs to address uncertainties in degradation modes or rates were not identified, but on-going EPRI and industry efforts are likely to address aging management implementation efficiency and cost issues. Table 3.1 provides a summary mapping of EPRI programs that carry on research and support for AMP areas. Some specific examples of the EPRI R&D programs and projects are summarized in the sections following Table 3.1.

Table 3-1
EPRI Program Support for Mechanical Category 2 Aging Management Programs

GALL AMP ID	AMP Name	EPRI Program Support
XI.M1	ASME Section XI In-service Inspection Fatigue Monitoring (TLAA) and ASME Section XI In-Service Inspection	Multiple Programs, including NDE, MRP, BWRVIP
XI.M2	Water Chemistry	Chemistry Program
XI.M3	Reactor Head Closure Stud Bolting	NDE and Nuclear Maintenance Application Center (NMAC) Programs
XI.M4	BWR Vessel ID Attachment Welds	BWRVIP
XI.M5	BWR Feedwater Nozzle	BWRVIP
XI.M6	BWR Control Rod Drive Return Line	BWRVIP
XI.M7	BWR Stress Corrosion Cracking	BWRVIP
XI.M10	Boric Acid Corrosion	MRP
XI.M17	Flow-Accelerated Corrosion	Balance of Plant Corrosion Program
XI.M18	Bolting Integrity	NDE And NMAC Programs
XI.M19	Steam Generators	Steam Generator Management Program
XI.M21A	Closed Treated Water Systems	Chemistry and PSE Programs
XI.M25	BWR Reactor Water Cleanup System	Chemistry Program
XI.M32	One-Time Inspection	NDE Program
XI.M33	Selective Leaching	NDE Program
XI.M35	One-Time Inspection of ASME Code Class 1 Small Bore Piping	NDE Program
XI.M37	Flux Thimble Tube Inspection	NDE Program
XI.M38	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Balance of Plant Corrosion Program
XI.M40	Monitoring of Neutron-Absorbing Materials Other than Boraflex	Used Fuel Management & High Level Waste Program
XI.M41	Buried and Underground Piping and Tanks	Balance of Plant Corrosion Program

Water Chemistry R&D

R&D Support for XI.M2, Water Chemistry

No specific water chemistry program LTO-related R&D gaps were identified. The benefit of water chemistry technologies is generally time-independent. Although mitigation through chemical means is vital to long-term aging management, any changes to program implementation over time are not likely to be related to time-dependent factors.

Importantly, implementation of the water chemistry program is specifically within the scope of NEI 03-08. A robust industry program exists to ensure that water chemistry guidelines are periodically reviewed and updated and that related R&D gaps are proactively addressed. Opportunities for AMP implementation improvements may be realized from these on-going research efforts.

Key existing EPRI reports include the following:

- BWRVIP-190: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines - 2008 Revision. EPRI, Palo Alto, CA: October 2008.
- Pressurized Water Reactor Primary Water Chemistry Guidelines - Revision 6. EPRI, Palo Alto, CA: December 2007.
- Pressurized Water Reactor Secondary Water Chemistry Guidelines, Revision 7. EPRI, Palo Alto, CA: February 2009

Balance of Plant Corrosion Program R&D

R&D Support for XI.M41, Buried and Underground Piping and Tanks

This research area is under the NEI 09-14 initiative. EPRI buried pipe research is focused on the furthering the state-of-the-art technologies for inspection, analysis, repair and mitigation of ongoing corrosion in buried infrastructure. This includes:

- Development and delivery of appropriate reference documents and training to support broad knowledge awareness for buried and underground piping.
- Development and transfer of new buried pipe inspection technologies, such as remote field NDE inspection robotics.
- Identification and evaluation of existing technologies that may be directly applied or easily adapted for nuclear plant buried piping inspection.
- Improved understanding regarding the use of guided wave acoustic NDE technologies for buried piping inspections.
- Availability of repair and replacement alternatives for buried pipe applications, including high-density polyethylene (HDPE).
- Enhanced buried pipe risk-ranking technologies through updates to existing software.

Research activities are coordinated across EPRI's Plant Engineering and NDE Programs.

The Plant Engineering Program provides buried pipe program owner guidance documents, reference materials, and upgraded risk ranking software (BPWORKS™), and also supports the development of various ASME Code Cases for repair/replacement activities. Training courses are offered for newly assigned Buried Pipe Program owners to help ensure buried pipe management guidance is appropriately deployed in the field. Reference materials on cathodic protection and coatings have been developed and are being used to address buried

and underground pipe program needs. Through the Buried Pipe Integrity Group, EPRI provides a forum for information exchange among nuclear plant personnel, vendors and other stakeholders to identify and transfer best practices for buried pipe inspection and assessment.

The NDE Program is pursuing the identification and assessment of existing robotic and inspection technologies, as well as the development of new robotic inspection technologies using remote field detection technology. Efforts continue to identify, demonstrate, evaluate, and qualify inspection technologies suitable for buried pipe applications, with special emphasis on guided wave ultrasonic technologies.

R&D Support for XI.M17, Flow-Accelerated Corrosion

The flow-accelerated corrosion (FAC) program is based on the EPRI guidelines found in Nuclear Safety Analysis Center (NSAC)-202L which defines the elements for an effective FAC program. The program includes performing (a) an analysis to determine critical locations, (b) limited baseline inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm the predictions, or repairing or replacing components as necessary.

The FAC degradation mechanisms managed by inspection program are well understood and have relatively short development periods, if active. The active R&D effort is focused on inspection technologies and associated assessment processes.

Steam Generator Management Program R&D

R&D Support for XI.M19, Steam Generators

The EPRI SGMP provides guidelines for inspection, repair, monitoring and flaw evaluation of steam generator components and tubing materials. The EPRI Steam Generator Management Program (SGMP) includes aging management activities for the steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator. Program implementation is consistent with the plant technical specifications and includes commitments to NEI 97-06. The nondestructive examination (NDE) techniques used to inspect tubes, plugs, sleeves, and secondary side internals are intended to identify components (e.g., tubes, plugs) with degradation that may need to be removed from service or repaired. The program additionally provides for degradation assessments, condition monitoring (assessment of past performance) and operational assessments (forward-looking assessment of anticipated performance until the next inspection). The steam generator program is based on these six EPRI Guidelines:

PWR Steam Generator Examination Guidelines:

- PWR Primary-to-Secondary Leak Guidelines
- PWR Secondary Water Chemistry Guidelines

- PWR Primary Water Chemistry Guidelines
- Steam Generator Integrity Assessment Guidelines
- Steam Generator In Situ Pressure Test Guidelines

These guideline documents are supported by both evaluation handbooks (e.g., flaw evaluation, foreign object evaluation) and by technical reports that document the results of EPRI research.

Environmentally Assisted Fatigue R&D

R&D Support for XI.M1, ASME Section XI In-Service Inspection and Associated TLAA

NRC provides guidance for fatigue monitoring programs and calculations of fatigue usage factors with adequate conservatism. On-going R&D efforts will investigate the following areas to support long-term operations by refining the application of current guidance for specific plant components and situations:

- Stress limit versus fatigue cycle (S-N) data for Nickel-base Alloys 600/82/182 and 690/52/152
- S-N Data for BWR Hydrogen Water Chemistry (HWC) Conditions
- Testing to Reconcile Experimental S-N Data with Field Experience
- Confirmatory S-N Testing of CASS in PWR Coolant
- Application of F_{en} Correlations to Plant Fatigue Analysis / Evaluation

EPRI has a long-term research project to understand the apparent inconsistencies between predicted environmental affect on fatigue life and industry operating experience. Expansion of the knowledge base on both materials and environments will be useful to optimizing fatigue aging management programs for specific plants. This long term research will address fatigue usage (cycling occurring prior to crack initiation), fatigue crack growth (crack extension after a fatigue crack initiates), and fatigue testing (of specimens and/or components) to provide the additional data needed to address the issue.

Additionally, there is an opportunity to consider fatigue loadings in a probabilistic manner for locations where leak-before-break has been applied. EPRI has developed a research plan more specific to addressing these issues.

The elements of the research plan include the following:

- Continue to track operating experience and inspection results; consider as bases for reduced inspection scope/frequency in the future.
- Convene an expert panel to provide specific direction for future guidance relative to swirl penetration thermal cycling, experts will indicate whether further development of a revised/replacement cycling model is appropriate, or if alternative guidance based on inspection and operating experience can be justified.
- If appropriate, develop and refine predictive model for thermal cycling behavior based on additional analytical and/or experimental work.
- Maintain and upgrade available guidance based on ongoing implementation experience.
- Develop guidance for managing cyclic thermal stratification in BWRs.
- Develop inspection technologies and guidance for inspection of socket welded configurations where thermal cycling is predicted.
- Recommend changes to ASME Section XI and Section III to address additional design loads for consideration in design modifications and new plant designs.
- Follow developments in the area of EAF and apply results to additional locations/loads as appropriate and when necessary to ensure coordinated fatigue guidance.
- Engage in ongoing xLPR effort to assist with inclusion of fatigue degradation mechanisms in this model

The potential products from the on-going R&D efforts on fatigue management of pressure boundary components research will likely include:

- New and updated industry guidelines that reliably address concerns for high-cycle thermal fatigue degradation at all pressure boundary locations.
- Inspection technologies and guidance for inspection of configurations where thermal cycling is predicted (including socket welded configurations).
- Guidance and training for managing fatigue issues throughout plant operating life.
- Interface with related EPRI projects for environmentally assisted fatigue (EAF) and Extremely Low Probability of Rupture (xLPR) to ensure that fatigue issues are addressed appropriately.

Interface with the ASME code body to address impacts to design basis loading definitions.



Section 4: STRUCTURAL AMPS R&D REVIEW RESULTS

Review Results

Aging of civil infrastructure in commercial nuclear power plants is a significant performance and cost consideration for long term operation. To date, the performance of post-tensioned and reinforced concrete structures has been quite good. There are, however, a variety of kinetic processes that can lead to the degradation of civil structures – and these may be accelerated by operating environments specific to nuclear plants (e.g., spent fuel pool leakage). It is important that the industry understand the impact of aging of civil infrastructure, particularly for LTO, as individual utilities will be required to provide both sound technical and economic justifications to continue operation to 80 years.

The goal of the EPRI R&D is to create a concrete structures program that looks at various degradation phenomena being experienced in operating plants. The initial stage of the project was the compilation and publication of an Aging Reference Manual in 2011. This Reference Manual defines the physics of kinetic degradation processes and documents operational issues dealt with by the industry over the past 40+ years. The manual provides a framework for identifying at-risk structures and applicable degradation mechanisms. Building upon this work, a number of individual research projects aimed at further understanding of those degradation mechanisms and structures identified as “at-risk” have been started. The results of the individual studies will be merged into an Aging Management Toolbox Platform, which will be an open-ended tool for plant operators to assess severity of damage and explore repair or mitigation options.

The long term vision for this project is to have a multiple-element toolbox platform made available to utilities to help manage aging concrete infrastructure and make informed O&M decisions as well as provide a technical basis to support long-term operations. Examples of ongoing research are projects that address: 1) radiation damage, 2) alkali-silica reaction (ASR) in concrete, 3) chloride attack of cooling towers, 4) augmented inspection of containment, 5) effects of boric acid on concrete reinforcement (rebar), and 6) containment concrete delamination. Both near-term and long-term efforts are expected to create a robust toolbox to allow effective aging management of structures.

Gap Assessment Summary of Results

Provided in the following subsections is a summary of the key technical issues for structures where R&D work is needed to fully inform the insights for aging management and the aging management program guidance. This information is provided in the three category breakdown as listed in Section 2. Under Category 1, specific attention is paid to the R&D efforts needed to address potential gaps in the technical basis for applying AMPs to the LTO period. As noted, significant R&D work is already in progress to address many of the issues. It is one of the objectives of this project to assist in the prioritization of such research to support a robust technical basis for both the decisions leading to LTO and the safe and reliable operation of the plant within the LTO period.

For Category 2 “S” items, the focus is on the on-going R&D that, although not required for successful application of the existing AMP, is expected to provide efficiency and cost improvements in the execution of the AMP. Category 3 items, as noted in Section 2, are those AMPs not requiring additional, focused R&D efforts.

In the overall assessment process, the evolving character of aging management programs was considered to be an essential factor. Accordingly, beyond the results of research, utility operating experience will continue to frame specific actions and their effectiveness in the AMP processes.

Category 1 AMPs

From Table 2-2.1, two structural AMPs were identified as having potential R&D needs in order to fully inform degradation behavior for LTO. For both AMPs, the R&D issues are the same: long-term concrete performance relative to degradation from alkali-silica reaction (ASR) and long-term concrete performance relative to degradation due to irradiation effects. The following subsections provide details on the approach to addressing such research needs.

Alkali Silica Reaction (ASR) R&D

Issue: Manage potential impact of ASR on concrete mechanical strength and structure functions.

Impact on AMP XI.S2 ASME Section XI, Subsection IWL and AMP XI.S6 Structures Monitoring

Alkali silica reaction (ASR) is a chemical reaction between the silica in the aggregates, the alkalies in cement paste, and water. This reaction generates swelling in concrete due to the formation of an expansive gel. The reaction develops at a slow rate, but can be accelerated by temperature.

The expansion caused by ASR results in cracking when the swelling-induced stress exceeds the tensile strength of the concrete which is low. As the ASR continues, cracks may grow and eventually coalesce, which results in reduced service life of the concrete structure. Note that the lack of shear reinforcement in many structures makes this degradation process potentially severe.

Although design and construction of nuclear power plants sought to prevent ASR from occurring in concrete structures, recent plant experience has shown that ASR susceptibility may still be a factor. Research on this topic will focus on the nature and timing for ASR susceptibility and will build on previous research results as noted below:

Program on Technology Innovation: Nuclear Concrete Structures Aging Reference Manual

The objective of this report is to provide a concrete structure degradation index and research and development gap analysis. It provides a comprehensive discussion of concrete degradation issues and describes how they have or may manifest themselves in nuclear plant environments. This report will be used by plant personnel who are responsible for determining the types and extent of damage occurring in concrete structures. The information will also help operators to make informed technical decisions regarding mitigation and repair of damage occurring in plants. The value to utilities will be to help make informed economic and technical decisions regarding continued operation. In addition, the gap analysis will serve as a framework for additional research and development activities.

The new R&D project will provide utilities with the tools needed to evaluate the risk of having alkali-silica reaction in their structures. Concrete that during construction of the existing fleet was considered as non-reactive may be classified differently today due to improved testing methods that have since been developed to detect the potential for ASR. For those structures that have the potential to develop this degradation, a set of tools for early detection of this pathology will be provided.

The following tasks will be completed:

- Develop a map of potential risk for aggregates with ASR in relation to the location of nuclear plants.
- Comparison of expansion tests based on the existing testing procedure 30 years ago and current testing for at least three types of common aggregates used in nuclear construction in the US.
- Develop a set of tools for early detection of ASR.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
ASR Risk Factors and Management						
Complete a map of types of aggregates used in nuclear plants (will build on LWRS Program Database)						
Catalog reactivity of aggregates used in the existing fleet based on current testing procedures						
Develop a toolbox with inspection procedures for utilities for early detection of ASR						
2013	2014	2015	2016	2017	2018	2019

Irradiation Effects on Concrete R&D

Issue: Neutron irradiation and associated gamma irradiation heating can degrade concrete performance.

Impact on AMP XI.S2 ASME Section XI, Subsection IWL and AMP XI.S6 Structures Monitoring

Radiation can damage concrete in the reactor cavity by creating atomic scale defects (neutron radiation) and radiolysis of water and heating (gamma radiation), which can result in weakening of the concrete. Previous R&D results supporting this effort are summarized as follow:

Effects of Radiation on Concrete: A Literature Survey and Path Forward

The continued operation of existing units is potentially challenged by a change in the properties of concrete structures that are exposed to constant radiation. Concrete structures that are irradiated, such as reactor vessel support structures, are critical to continued operation, and their long-term integrity must be ensured. The information on the interaction of radiation with concrete may be insufficient to adequately determine the condition of irradiated concrete for life beyond 60 years of operation. The purpose of this technical report is to provide a comprehensive review of the available literature as it pertains to long-term irradiation effects on the mechanical properties of concrete structures. This report has been compiled from academic journal articles, national standards and specifications, and government reports. An additional purpose of this report is to provide a “path forward” that will allow the industry to obtain information that will be important to assess the expected life of concrete exposed to radiation.

Aging Identification and Assessment Checklist: Civil and Structural Components

This product provides practical information describing the aging degradation mechanisms that may affect various plant components. It is intended not to make the user an expert in equipment aging, but to promote a questioning attitude when plant personnel observe conditions that might negatively impact plant reliability. The checklists detail basic indicators of aging degradation that might be observed during a typical walkdown. In some cases, the indicators themselves may be a degraded condition that may either cause or accelerate the aging degradation mechanism. This report provides this information in a form that will allow plant personnel to have a ready reference with them as they investigate the condition of plant SSCs.

In this on-going R&D project, the exposure thresholds for irradiation damage will be defined and the effect of damage on mechanical integrity of the biological shielding will be characterized for concrete with up to 80 years of equivalent neutron and gamma exposure. The test plan includes mining existing experimental data, collaborating with teams conducting ongoing experimental irradiation testing, performing irradiation and characterization studies on typical concrete specimens, harvesting irradiated concrete samples from retired plants, as appropriate, and developing a mechanistic understanding and model of concrete aging and degradation when exposed to high levels of radiation long-term. The overall goal of this project is to fully characterize the effects of the radiation on the RPV support pedestals and reactor cavity through a potential extended period of operation to 80 years.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
Radiation Effects on Concrete						
Initial assessment of expected radiation exposure and effects based on available literature data						
	Irradiation testing of concrete specimens					
			Updated assessment of expected radiation exposure and effects			
2013	2014	2015	2016	2017	2018	2019

Category 2 AMPs

This category covers AMPs addressing degradation modes that are well-characterized, or implement established Programs. Specific R&D needs to address uncertainties in degradation modes or rates were not identified, but on-going EPRI and industry efforts will continue to address aging management implementation efficiency and cost issues. Table 4.1 provides a summary mapping of EPRI programs that carry on research and support for the AMP area. Key areas of on-going research are expected to address NDE and coating management techniques and tools.

Table 4-1

EPRI Program Support for Structural Category 2 Aging Management Programs

GALL AMP ID	AMP Name	EPRI Program Support
XI.S8	Protective Coating Monitoring and Maintenance Program	NDE and Plant Engineering Programs



Section 5: ELECTRICAL AMPS R&D REVIEW RESULTS

Review Results

Aging management programs are currently defined for six areas of electrical components consisting of low voltage (instrument and control and power) and medium voltage cables not subject to environmental qualification requirements, metal enclosed bus, fuses, and cable connections. EPRI research has been conducted to assist in establishing inspection and evaluation bases for aging management activities. As was completed for Mechanical and Structural AMPs, the Electrical AMPs were also assessed relative to R&D needs to support application of the current AMPs to a potential LTO period. The three category binning process presented in Section 2 was applied and the results are summarized in the following subsections.

Gap Assessment Summary of Results

Provided in the following subsections is a summary of the key technical issues where R&D work is needed to fully inform the insights for aging management and the aging management program guidance. This information is provided in the three category breakdown as listed in Section 2. Under Category 1, specific attention is paid to the R&D efforts needed to close potential gaps in the technical basis for applying AMPs to the LTO period. As noted, significant R&D work is already in progress to address many of the issues. It is one of the objectives of this project to assist in the prioritization of such research to support a robust technical basis for both the decisions leading to LTO and the safe and reliable operation of the plant within the LTO period.

For electrical AMPs, no Category 2 items were identified. Category 3 items, as noted in Section 2, are those AMPs not requiring additional, focused R&D efforts.

In the overall assessment process, the evolving character of aging management programs was considered to be an essential factor. Accordingly, beyond the results of research, utility operating experience will continue to frame specific actions and their effectiveness in the AMP processes.

Category 1 AMPs

From Table 2-2.1, two electrical AMPs were identified as having potential R&D needs in order to fully inform degradation behavior for LTO. For both AMPs, the R&D issues are the same: long-term cable performance relative to degradation from effects of thermal and radiation exposure.

Cable Degradation from Thermal and Radiation Exposure R&D

Issue: Continuing exposure to environments having high temperature and/or radiation exposure.

Impact on AMP XI.E1, Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements and AMP XI.E2, Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits

As discussed in previous sections, current R&D efforts are extensions of longer standing research efforts on component degradation modes and rates. For electrical components some existing research results are summarized as follow:

Plant Support Engineering: License Renewal Electrical Handbook

As part of the application process for license renewal, applicants must perform an evaluation to confirm that they have adequately considered aging effects that could cause degradation to plant components during the period of extended operation. This report presents a process that can be used by applicants in determining which aging effects are applicable to electrical components and how they can be managed. The electrical component review process includes not only an effort to develop the technical bases for inclusion or exclusion of components from aging management programs but also an awareness that the license renewal process must be complete and robust. The document begins with an overview of the electrical component review process including a description of the encompassing regulatory framework. Additional chapters are dedicated to the technical aspects of performing electrical component reviews.

Natural Versus Artificial Aging of Electrical Components: Interim Report 1996 – 2000

This report describes activities preceding and during the layup period of the Natural Versus Artificial Aging Program. The objective of the program is to compare the natural, in-plant aging of organic materials in typical nuclear plant components with the aging predicted by the accelerated laboratory age conditioning traditionally used to qualify safety-related equipment. To economically increase the amount of information yielded by the program, the removal of specimens was halted until their age had advanced to the point that

larger changes in properties were likely. The main benefit of this work will be to address questions regarding the adequacy of accelerated aging techniques used in equipment qualification testing. As results become available, they can forewarn plant operators of greater-than-expected degradation if any is observed. The data from these studies may be helpful in demonstrating that cable can function satisfactorily for extended periods of time in excess of 40 to 60 years.

Medium Voltage Cable Aging Management Guide, Revision 1

Medium-voltage cables (5- to 35-kV rated cables) have provided reasonable service in nuclear power plants. However, there is a concern that cables that have experienced long periods of wet service might degrade and fail in service. This report has been prepared to provide information that will be of practical use when questions concerning medium-voltage cable longevity in adverse environments and service conditions arise at nuclear plants.

This report provides a wide range of information both on what factors can affect aging of cables and on the effective management of that aging. Guidance is provided on selection of an appropriate test method depending on the cable design and how it is expected to degrade with time. Effects of aging of the insulation and metallic shield system must be considered to allow appropriate selection of the test method.

Plant Support Engineering: Aging Management Program Guidance for Medium-Voltage Cable Systems for Nuclear Power Plants

This report was developed at the direction of utility management and in parallel with the Regulatory Issue Resolution Protocol for Inaccessible or Underground Cable Circuit Performance Issues at Nuclear Power Plants that occurred between the NRC and the industry (through the Nuclear Energy Institute) from mid-2009 into 2010. Implementation of this guide was to form part of the closure process for the protocol, but that did not occur. Instead NEI informed the NRC that the industry would implement aging management programs in accordance with EPRI guidance. This guide was developed to provide a consistent methodology for the industry to follow in developing an aging management program for medium-voltage cable circuits that are subjected to adverse environmental or service conditions that could lead to degradation of the insulation systems.

The report was developed by subjecting drafts to review and revision by a Technical Advisory Group formed of industry cable personnel from nuclear plant organizations, cable manufacturers, and cable test companies. This report describes the scope of the cable circuits to be evaluated, those conditions that are considered to be adverse environments, and the actions to be taken to assess the conditions of the cable circuits subject to adverse conditions. For key test methodology, assessment criteria are described, along with possible corrective actions that could be implemented.

The NRC issued Regulatory Guide 1.218, “Condition-Monitoring Techniques for Electric Cables Used in Nuclear Power Plants” which along with NUREG/CR 7000 provide the NRC view of cable aging management. These documents lay out the techniques that the NRC recommends should be applied to cable aging management and what the scope of that aging management program should be. The NRC and EPRI documents are mainly in agreement on all issues, but not in the level of detail provided for implementation guidance.

Plant Support Engineering: Life Cycle Management Planning Sourcebooks: Medium-Voltage (MV) Cables and Accessories (Terminations and Splices)

EPRI produced a series of Life Cycle Management Planning Sourcebooks, each containing a compilation of industry experience information and data on aging degradation and historical performance for a specific type of system, structure, or component (SSC). This sourcebook provides information and guidance for implementing cost-effective life cycle management (LCM) planning for medium-voltage (MV) cables and accessories (terminations and field splices).

The general objective of EPRI's LCM sourcebook effort is to provide system engineers with generic information, data, and guidance they can use to generate a long-term equipment reliability plan for the plant-specific SSC (aging and obsolescence management plan optimized in terms of plant performance and financial risk). The long-term equipment reliability plan, or LCM plan, for a plant SSC combines industry experience and plant-specific performance data to provide an optimum maintenance plan, schedule, and cost profile throughout the plant's remaining operating life.

Plant Support Engineering: Aging Management Program Development Guidance for AC and DC Low-Voltage Power Cable Systems for Nuclear Power Plants

This report was developed at the direction of utility management and in parallel with the Regulatory Issue Resolution Protocol for Inaccessible or Underground Cable Circuit Performance Issues at Nuclear Power Plants that occurred between the United States Nuclear Regulatory Commission and the industry (through the Nuclear Energy Institute) from mid-2009 into 2010. Implementation of this guide was to form part of the closure process for the protocol. The actual process paralleled that developed for medium voltage cables. This report was developed to provide a consistent methodology for the industry to follow in developing an aging management program for low-voltage power cable circuits that are subjected to adverse environmental or service conditions that could lead to degradation of the insulation systems. This report is a companion to the similar report on medium-voltage cables.

This report describes the scope of the cable circuits to be evaluated, those conditions that are considered to be adverse environments, and the actions to be taken to assess the conditions of the cable circuits subjected to adverse conditions. Applicable test methodology is described, along with possible corrective actions that could be implemented.

Building on the above and other technical reports, on-going research will focus on several key task areas for long term operation. These areas are as follow:

1. Quantify radiation and thermal environments for low voltage cables in containment.
2. Develop additional test methods for medium voltage cables.
3. Develop a methodology to determine the risk and cost exposure for low and medium voltage cable for LTO.

Item 1 is to research the actual environments that must be understood to determine the degree to which thermal and radiation aging will affect low voltage cables. While conservative design data is well known, the conditions where the cables are located within containment are less well known. Having this data available to researchers will allow research programs to be properly designed. Current research is based on the worst case in-containment environments. In this project, the normal environments for inside containment will be obtained at the location of the bulk of the inside containment cables. The cables located in the worst case environments, if any, will be identified to show the conditions for outliers.

Item 2 supports identification of failure mechanisms not yet experienced or detected. Corona damage, partial discharge and tracking in terminations are issues that ultimately may lead to medium voltage cable circuit failures. The best techniques that are practical for nuclear applications will be identified and evaluated for their usefulness.

The third research project is a multi-part plan to combine benchmarking of older or harsher condition boiling and pressurize reactors to identify cable at risk populations. The second part of the project will be to extrapolate that data and create a risk profile out to 80 years of plant operation. Finally, estimates will be made to determine the replacement cost over an 80 year plant life. The overall methodology could be used for other plants to perform evaluations of their exposure.

Additional work is in progress or planned to address submergence qualification for medium voltage pink and brown EPR and rejuvenation of black EPR to extend service life. Significant collaboration with the DOE LWRS Program is a key aspect of the research effort and expected results.

Roadmap Timeline:

2013	2014	2015	2016	2017	2018	2019
Radiation and Thermal Effects on Cables						
Determine containment cable temperature and radiation levels for representative, current NPPs for input into cable aging research						
Assess cable system vulnerability by environment and service condition to estimate replacement needs for long-term operation						
Cable Diagnostics						
Assessment of online continuous differential partial discharge test for medium voltage cable						
2013	2014	2015	2016	2017	2018	2019



Section 6: Conclusions

EPRI performed a review of the aging management programs (AMPs) currently defined in NUREG-1801, Revision 2, “Generic Aging Lessons Learned Report” (GALL Report) to support the utility members planning for long-term operations. AMPs are a key aspect of the current license renewal process. The purpose of the EPRI review was to assess the current state of knowledge and on-going research in progress or planned that builds on the existing technical foundation.

From the assessment, the AMPs were binned into one of three categories based on the level of R&D supporting efforts for long-term operations. As provided in Section 2, the three categories are summarized as follow:

1. R&D support to address knowledge gaps for 60 to 80 years operation to better understand and manage materials performance.
2. R&D supporting role where aging degradation is well-characterized. R&D focus will be on aging management process improvement.
3. No new R&D role identified. The inspection interval is informed by inspection results, plant specific commitments, and relevant OE.

Of the 50 AMPs defined in the GALL Report (Rev 2) it is noted that:

1. 8 AMPs were classified in Category 1
2. 20 were classified in Category 2
3. 22 were classified in Category 3

For Category 1 AMPs additional details were provided in Sections 3 through 5 to identify the R&D gaps and to present initial R&D projects in progress or planned that are expected to address the gaps. Additional R&D will continue to support AMPs in Category 2 but is expected to impact efficiency of AMP implementation. The results of this project effort are planned to be periodically reviewed and updated.

Section 7: References

Mechanical AMPs

1. “PWR Owners Group Materials Subcommittee Interim Strategy for Identifying Outside Diameter Initiated Stress Corrosion Cracking (ODSCC) of Stainless Steel Systems,” 51-9152699-000 NEI 03-08 Good Practice Recommendation and Implementation Date, OG-11-67, PA-MS-0563.
2. 10 CFR 50.55a, Codes and Standards.
3. *2010 Interim Review of the Pressurized Water Reactor Primary Water Chemistry Guidelines--Revision 6*. EPRI, Palo Alto, CA: 2010. 1021133.
4. *2010 Interim Review of the Pressurized Water Reactor Secondary Water Chemistry Guidelines Revision 7*. EPRI, Palo Alto, CA: 2010. 1021132.
5. ACI Standard 318, Building Code Requirements for Reinforced Concrete and Commentary, American Concrete Institute.
6. ANL-LWRS-47, “Report on Assessment of Environmentally-Assisted Fatigue for LWR Extended Service Conditions,” Sept. 2011.
7. ASME Boiler and Pressure Vessel Code, Case N-761, “Fatigue Design Curves for Light Water Reactor (LWR) Environments,” ASME International, September 2010.
8. ASME Boiler and Pressure Vessel Code, Case N-792, “Fatigue Evaluations Including Environmental Effects, Section III, Division 1,” ASME International, NC-Supp. 6.
9. ASME Boiler and Pressure Vessel Code, Section XI, Nonmandatory Appendix L, “Operating Plant Fatigue Assessment.”
10. ASME CC N-755, “Use of Polyethylene (PE) Plastic Pipe for Section III, Division 1, Construction and Section XI Repair/Replacement Activities.”
11. ASME Code Case N722-1, Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated from Alloy 600/82/182 Materials.
12. ASME Code Case N-729-1, Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds.

13. ASME Code Case N-770, Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities.
14. ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components.
15. Assessment of Initial Test Conditions for Experiments to Assess Irradiation-Assisted Stress Corrosion Cracking Mechanisms, J. T. Busby and M. N. Gussev, ORNL, ORNL/TM-2010/346, December 2010.
16. AWWA C203-02, "American Water Works Association Standard for Coal-Tar Protective Coatings and Linings for Steel Water Pipelines-Enamel and Tape-Hot Applied."
17. B. Ter-Ovanessian, J. M. Cloué, J. Deleume and E. Andrieu, "On the stress corrosion cracking behavior of two age-hardenable alloys in PWR primary water", pp. 16-24, Proceedings of 14th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, Virginia Beach, VA, ASM 2009.
18. BWRVIP Correspondence 2009-202, "Interim Guidance for Accelerated Inspections of Jet Pump Riser to Riser Brace Welds and Wedges." June 2009.
19. *BWRVIP-05: BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations*. EPRI, Palo Alto, CA: September 1995. Technical Report TR-105697.
20. *BWRVIP-06, Revision 1-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals*. EPRI, Palo Alto, CA: December 2009. 1019058.
21. *BWRVIP-100, Revision 1: BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds*. EPRI, Palo Alto, CA: October 2010. 1021001.
22. *BWRVIP-135, Revision 2: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP)*. EPRI, Palo Alto, CA: June 2007 Technical Report 1020231.
23. *BWRVIP-139-A: BWR Vessel and Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: July 2009. 1018794.
24. *BWRVIP-14-A: BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals*. EPRI, Palo Alto, CA: 2008. 1016569.
25. *BWRVIP-154, Revision 2: BWR Vessel and Internals Project, Fracture Toughness in High Fluence BWR Materials*. EPRI, Palo Alto, CA: September 2009. 1019077.
26. BWRVIP-155, Evaluation of Thermal Fatigue Susceptibility in BWR Stagnant Branch Lines.

27. *BWRVIP-173-A: BWR Vessel and Internals Project, Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials*. EPRI, Palo Alto, CA: July 2011. Technical Report 1022835.
28. *BWRVIP-18, Revision 1: BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: October 2008. 1016568.
29. *BWRVIP-180: BWR Vessel and Internals Project, Access Hole Cover Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: October 2007. 1013402.
30. *BWRVIP-183: BWR Vessel and Internals Project, Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: December 2007. 1013401.
31. *BWRVIP-190: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines – 2008 Revision*. EPRI, Palo Alto, CA: October 2008. 1016579.
32. *BWRVIP-218: BWR Vessel and Internals Project, Alloy X-750 Characterization Study*. EPRI, Palo Alto, CA: July 2009. 1019070.
33. *BWRVIP-219: BWR Vessel and Internals Project, Technical Basis for On-Line NobleChem™ Mitigation and Effectiveness Criteria for Inspection Relief*. EPRI, Palo Alto, CA: July 2009. 1019071.
34. *BWRVIP-222: BWR Vessel and Internals Project, Accelerated Inspection Program for BWRVIP-75-A Category C Dissimilar Metal Welds Containing Alloy 182*. EPRI, Palo Alto, CA: 2009. 1019055.
35. *BWRVIP-233, Revision 1: BWR Vessel and Internals Project: Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment: Technical Basis for Revisions to BWRVIP-60-A*. EPRI, Palo Alto, CA: November 2011. 1022841.
36. *BWRVIP-234: BWR Vessel and Internals Project, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals*. EPRI, Palo Alto, CA: December 2009. 1019060.
37. *BWRVIP-240: BWR Vessel and Internals Project, Metallurgical Analyses and Macro- and Microstructural Mapping of Alloy X-750 and Alloy XM-19 Plates*. EPRI, Palo Alto, CA: October 2010. 1021003.
38. *BWRVIP-242, BWRVIP Inspection Trends, 2010 Update*. EPRI, Palo Alto, CA: December 2010. 1020996.
39. *BWRVIP-25: BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: December 1996. TR-107284.
40. *BWRVIP-26-A: BWR Vessel and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: November 2004. 1009946.
41. *BWRVIP-27-A, BWR Standby Liquid Control System/Core Plate DP Inspection and Flaw Evaluation Guidelines*, EPRI, Palo Alto, CA: 2003. 1007279.

42. *BWRVIP-38: BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: September 1997. TR-108823.
43. *BWRVIP-41, Revision 3: BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: September 2010. 1021000.
44. *BWRVIP-42, Revision 1: BWR Vessel and Internals Project, LPCI Coupling Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: June 2010. 1020999.
45. *BWRVIP-47-A: BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: November 2004. 1009947.
46. *BWRVIP-48-A: BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines*, EPRI, Palo Alto, CA: 2004. 1009948.
47. *BWRVIP-49-A, Instrument Penetration Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: 2002. 1006602.
48. *BWRVIP-59-A: BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals*. EPRI, Palo Alto, CA: May 2007. 1014874.
49. *BWRVIP-60-A: BWR Vessel and Internals Project, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment*. EPRI, Palo Alto, CA: June 2003. Technical Report 1008871.
50. *BWRVIP-61: Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants*, EPRI, Palo Alto, CA: 1999. TR-112076.
51. *BWRVIP-62, Revision 1: BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection*. EPRI, Palo Alto, CA: December 2011. 1022844.
52. *BWRVIP-74-A: BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal*. EPRI, Palo Alto, CA: June 2003. Technical Report 1008872.
53. *BWRVIP-75-A: Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules*. EPRI, Palo Alto, CA: 2005. 1012621.
54. *BWRVIP-76, Revision 1: BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: November 2009. 1022843.
55. *BWRVIP-94, Revision 2: BWR Vessel and Internals Project Program Implementation Guide*. EPRI, Palo Alto, CA: September 2011. 1024452.
56. *BWRVIP-99-A: BWR Vessel and Internals Project, Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components*. EPRI, Palo Alto, CA: November 2008. 1016566.

57. Computer-Based NDE Training for Thermal Fatigue Cracking, Version 1.1 (MRP-36, Revision 1).
58. *Environmentally Assisted Fatigue Screening: Process and Technical Basis for Identifying EAF Limiting Locations*: EPRI, Palo Alto, CA, CA: 2012. 1024995.
59. EPRI 1003414, *Sixth Inspection – EPRI Boraflex Surveillance Assembly*. EPRI, Palo Alto, CA: 2002.
60. EPRI 1003529, *Review of Dose Rate Effects on RPV Embrittlement*.
61. EPRI 1008055, *Improving Flowable Fills With Coal Ash*, Resource Paper. October 2003.
62. EPRI 1010639, *Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools*, Revision 4.
63. EPRI 1013549, *Nondestructive Evaluation: Seismic Design Criteria for Polyethylene Pipe Replacement Code Case*.
64. EPRI 1016236, *Plant Support Engineering: Flaw Tolerance Evaluation of Thermally Aged Cast Austenitic Stainless Steel Piping*.
65. EPRI 1016687, *Plant Support Engineering: Buried Pipe End-of-Expected-Life Considerations and the Need for Planning*.
66. EPRI 1019037 *Steam Generator Degradation Specific Management Flaw Handbook*, Revision 1.
67. EPRI 1019110, *Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications*.
68. EPRI 1019111, *Nondestructive Evaluation: Update to NDE for Selective Leaching of Gray Cast Iron Components*.
69. EPRI 1020265, *RACKLIFE Version 2.1.1*.
70. EPRI 1020439, *Stress Intensification and Flexibility Factors of High Density Polyethylene Pipe Fittings*. NQA Report.
71. EPRI 1020957, *Program on Technology Innovation: Scoping Study of Low Temperature Crack Propagation for 182 Weld Metal in BWR Environments and for Cast Austenitic Stainless Steel in PWR Environments (Revision 1)*.
72. EPRI 1021024, *Materials Reliability Program: Pressurized Water Reactor Issue Management Tables – Revision 2*.
73. EPRI 1021146, *Nondestructive Evaluation: Cast Austenitic Stainless Steel Study*.
74. EPRI 1021165, *Nondestructive Evaluation: High Density Polyethylene Inspection Technology and Techniques*.
75. EPRI 1021175, *Recommendations for an Effective Program to Control the Degradation of Buried and Underground Piping and Tanks (1016456, Revision 1)*.

76. EPRI 1021451, *Inspection and Testing of Boraflex Surveillance Coupons Nos. 212, 221, and 224 from the Seabrook Nuclear Power Station.*
77. EPRI 1021470, *Balance of Plant Corrosion -- The Buried Pipe Reference Guide.*
78. EPRI 1022565, *Slow Crack Growth Testing of High-Density Polyethylene Pipe: 2011 Update.*
79. EPRI 1022830, *Steam Generator Management Program: Investigation of Steam Generator Secondary-Side Degradation.* September 2011.
80. EPRI 1022876, *Stress-Based Fatigue Monitoring: Methodology for Fatigue Monitoring of Class 1 Nuclear Components in a Reactor Water Environment.*
81. EPRI 1022929, *Nondestructive Evaluation: Guided Wave Status Report.*
82. EPRI 1022930, *Nondestructive Evaluation: Buried Pipe Nondestructive Evaluation Reference Guide -- Revision 1 to Report 1021626.*
83. EPRI 1022931, *Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices.*
84. EPRI 1022941, *Nondestructive Evaluation: Ultrasonic Examination Techniques for High Density Polyethylene Pipes.*
85. EPRI 1023012, *Environmentally Assisted Fatigue Gap Analysis and Roadmap for Future Research.*
86. EPRI 1025204, *Strategy for Managing the Long Term Use of BORAL® in Spent Fuel Storage Pools.*
87. EPRI NDE Program Issue Management Table Report (internal EPRI draft version, dated April, 2012.)
88. EPRI NPC Roadmap, Buried Pipe Integrity.
89. EPRI TR-101986, *Boraflex Test Results and Evaluation*, EPRI, Palo Alto, CA: 1993.
90. EPRI TR-108761, A Synopsis of the Technology Developed to Address the Boraflex Degradation Issue.
91. Evaluation of Conservatisms and Environmental Effects in ASME Code Section III, Class 1 Fatigue Analysis, Deardorff, A. F., Smith, J. K., Contractor report, Sandia National Laboratory, Sandia94-0187, UC-523, 1994.
92. Guide Specification for Controlled Low Strength Materials (CLSM), National Ready Mixed Concrete Association.
93. *Guidelines for Addressing Environmental Effects in Fatigue Usage Calculations.* EPRI, Palo Alto, CA. 2011: 1025823. (not yet published).
94. Holtec Position Paper WS-105, "The Boral Neutron Absorber for Wet Storage Applications," Revision 3.
95. IAEA-TECDOC-1668, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Steam Generators – 2011 Update.

96. *Improved Basis and Requirements for Break Location Postulation*. EPRI, Palo Alto, CA. 2011: 1022873.
97. INL/EXT-11-24173, Baseline Fracture Toughness and CGR Testing of Alloys X-750 and XM-19 (EPRI Phase I).
98. Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 1).
99. Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines – Supplemental Guidance (MRP-146S).
100. *Materials Handbook for Nuclear Plant Pressure Boundary Applications (2010)*. EPRI, Palo Alto, CA. 2010. 1022344.
101. Mitigation of Thermal Fatigue in Unisolable Piping Connected to PWR Reactor Coolant Systems (MRP-29).
102. MRP-115, Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds.
103. MRP-126, Generic Guidance for Alloy 600 Management.
104. MRP-135 Rev. 1, Development of Material Constitutive Model for Irradiated Austenitic Stainless Steels.
105. MRP-139, Revision 1, Primary System Piping Butt Weld Inspection and Evaluation Guideline. EPRI, Palo Alto, CA: 2008. 1015009.
106. MRP-166, Demonstration of Equipment and Procedures for the Inspection of Alloy 600 Bottom Mounted Instrumentation (BMI) Head Penetrations.
107. MRP-169 Rev. 1-A, Technical Basis for Preemptive Weld Overlay for Alloy 82/182 Butt Welds in Pressurized Water Reactors.
108. MRP-175, PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values.
109. MRP-189 Rev. 1, Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items.
110. MRP-191, Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design.
111. MRP-199, Reactor Vessel Head Boric Acid Corrosion Testing.
112. MRP-211, PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data—State of Knowledge.
113. *MRP-218, Materials Reliability Program: Evaluation of Controlling Transient Ramp Times Using Piping Methodologies When Considering Environmental Fatigue (Fen) Effects*. EPRI, Palo Alto, CA: 2007. 1015014.
114. MRP-227-A, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines. EPRI, Palo Alto, CA: 2011. 1022863.
115. MRP-228, Inspection Standard for PWR Internals.

116. MRP-229 Rev. 3, Functionality Analysis for Babcock & Wilcox Representative PWR Internals.
117. MRP-231 Rev. 2, Aging Management Strategies for B&W Pressurized Water Reactor Internals.
118. MRP-232, Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals.
119. MRP-245, Materials Reliability Program: Material Production and Component Fabrication and Installation Practices for Alloy 690 Replacement Components in Pressurized Water Reactor Plants.
120. MRP-259 Rev. 1, Verification of Material Constitutive Model for Irradiated Austenitic Stainless Steels.
121. MRP-267 Rev. 1, Technical Basis for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement.
122. MRP-268, Reactor Vessel Bottom Mounted Nozzle Boric Acid Corrosion Testing: Design and Analysis of Full-Scale BMN Mockups.
123. MRP-273, High-Strength Structural Bolting and Nickel-Base Alloy Components in B&W-Design PWR Internals.
124. MRP-276, Thermal Aging and Neutron Embrittlement Assessment of Cast Austenitic Stainless Steels and Stainless Steel Welds in PWR Internals.
125. MRP-293, Low-Temperature Crack Propagation in PWR Materials.
126. MRP-307, Probabilistic Assessment of Chemical Mitigation of Primary Water Stress Corrosion Cracking in Nickel-Base Alloys.
127. MRP-309, Materials Reliability Program: Primary Water Stress Corrosion Testing of Alloys 690 and Weld Metals.
128. MRP-310, Preliminary Report — Alloy 52M Hot Cracking Study and Prevention Guide.
129. MRP-320, Testing Gap Assessment and Material Identification for PWR Internals.
130. MRP-326: Coordinated PWR Reactor Vessel Surveillance Program Guidelines, EPRI, Palo Alto, CA: 2011.
131. MRP-335, Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement.
132. MRP-47 Revision 1, Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application
133. MRP-55 Rev. 1, Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials.
134. Nanstad and Odette, "Development of a Robust Predictive Embrittlement Model for Reactor Pressure Vessel Lifetime Extension," D.O.E. Light Water Reactor Sustainability Newsletter, Issue 6, Dec 2011.

135. NDE Technology for Detection of Thermal Fatigue Damage in Piping, (MRP-23, Revision 1).
136. NEI 09-14, Revision 2, Guidelines for the Management of Buried Piping Integrity.
137. NEI 97-06, Steam Generator Program Guidelines, Revision 3, 2011.
138. Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs. EPRI, Palo Alto, CA: 2011. 1021467.
139. NP-7139-D, Reactor Pressure Vessel Attachment Welds: Degradation Assessment, EPRI, Palo Alto, CA. 1991.
140. NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," USNRC: August 1988.
141. NRC Generic Letter 88-01, Supplement 1, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," U.S. Nuclear Regulatory Commission: February 1992.
142. NRC Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Components, USNRC, July 1989.
143. NRC Information Notice 2004-08, Reactor Coolant Pressure Boundary Leakage Attributable to Propagation of Cracking in Reactor Vessel Nozzle Welds, April 22, 2004.
144. NRC Information Notice 2009-26, "Degradation of Neutron Absorbing Materials in the Spent Fuel Pool," October 2009.
145. NRC Information Notice 97-79, Potential Inconsistency in the Assessment of the Radiological Consequences of a Main Steam Line Break Associated with the Implementation of Steam Generator Tube Voltage-Based Repair Criteria, U.S. Nuclear Regulatory Commission, November 20, 1997.
146. NRC Interim Staff Guidance (ISG) 2011-05, Ongoing Review of Operating Experience, March 16, 2012.
147. NRC NRR Action Plan, Buried Piping, Sept. 14, 2010 (TAC NO. ME3939).
148. NRC Reg. Guide 1.161, Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb, 1995.
149. NRC Regulatory Guide 1.207, Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components due to the Effects of the Light Water Reactor Environment for New Reactors.
150. *Nuclear Maintenance Application Center: Bolting Guides Consolidation Review*. EPRI, Palo Alto, CA: 2006. 1013550.

151. NUREG/CR-5704, "Effects of Light Water Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," U.S. Nuclear Regulatory Commission: April 1999.
152. NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," U.S. Nuclear Regulatory Commission: March 1995.
153. NUREG/CR-6583, "Effects of Light Water Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," U.S. Nuclear Regulatory Commission: February 1998.
154. NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," U.S. Nuclear Regulatory Commission: February 2007.
155. NUREG/CR-6923, "Expert Panel on Proactive Degradation Assessment," USNRC: February 2007.
156. NUREG-0737, Clarification of TMI Action Plan Requirements, USNRC, November 1980.
157. PNNL-17584, "Preliminary Assessment of NDE Methods on Inspection of HDPE Butt Fusion Piping Joints for Lack of Fusion," May 2008.
158. *Pressurized Water Reactor Generic Tube Degradation Predictions: U.S. Recirculating Steam Generators with Alloy 600TT and Alloy 690TT Tubing.* EPRI, Palo Alto, CA: 2003. 1003589.
159. *Pressurized Water Reactor Primary Water Chemistry Guidelines - Revision 6.* EPRI, Palo Alto, CA: December 2007. 1014986.
160. *Pressurized Water Reactor Secondary Water Chemistry Guidelines, Revision 7.* EPRI, Palo Alto, CA: February 2009. 1016555.
161. *Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 7.* EPRI, Palo Alto, CA: 2007. 1013706.
162. *PWR Primary-to-Secondary Leak Guidelines-Revision 3,* EPRI, Palo Alto, CA: 2004. 1008219.
163. Reactor Pressure Vessel Task of Light Water Reactor Sustainability Program: Milestone Report on Materials and Machining of Specimens for the ATR-2 Experiment, R. K. Nanstad, ORNL, and G. R. Odette, University of California, Santa Barbara, ORNL/LTR-2011/413, January 2011.
164. Regulatory Guide 1.200, Revision 2, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment results for Risk-Informed Activities, USNRC, March 2009.
165. *Resistance of Alloy 600 and Alloy 690 Tubing to Stress Corrosion Cracking in Environments With and Without Lead.* EPRI, Palo Alto, CA: June 2004. Technical Update 1009532.

166. *Steam Generator Degradation Specific Management Flaw Handbook, Revision 1*. EPRI, Palo Alto, CA: 2009. 1019037.
167. *Steam Generator Foreign Object Handbook*. EPRI, Palo Alto, CA: 2007. 1014981.
168. *Steam Generator In-Situ Pressure Test Guidelines – Revision 3*. EPRI, Palo Alto, CA: 2007. 1014983.
169. *Steam Generator Integrity Assessment Guidelines – Revision 3*. EPRI, Palo Alto, CA: 2009. 1019038.
170. *Steam Generator Management Program: Conditions Causing Lead Stress Corrosion Cracking of Steam Generator Tubing; Alloy 690TT*. EPRI, Palo Alto, CA: 2011. 1022829.
171. *Steam Generator Management Program: Electrochemical Evaluation of Lead Stress Corrosion Cracking (PbSCC) Mechanism*, EPRI, Palo Alto, CA: 2010. 1020993.
172. *Steam Generator Management Program: Improvement Factors for Pressurized Water Reactor Steam Generator Tube Materials*. EPRI, Palo Alto, CA: 2009. 1019044.
173. *Steam Generator Management Program: Predicting to Failure of Alloy 600TT in Steam Test Environments*. EPRI, Palo Alto, CA: 2011. 1022640.
174. *Steam Generator Management Program: Steam Generator Deposit Characterization for Steam Generator Tube Degradation Prediction and Management*. EPRI, Palo Alto, CA: 2008. 1018249.
175. Thermal Fatigue Monitoring Guidelines (MRP-32, Revision 1).

Structural AMPs

176. *Plant Support Engineering: Aging Effects for Structures and Structural Components (Structural Tools)*. EPRI, Palo Alto, CA: 2007. 1015078.
177. *Aging Effects for Structures and Structural Components (Structural Tools): B&W Owners Group Generic License Renewal Program, BAW-2279P, 1997*. EPRI, Palo Alto, CA: 2000. TR-114881.
178. *Assessment of Needs for Concrete Research in the Energy Industry*. EPRI, Palo Alto, CA: 2010. 1022373.
179. *Nuclear Concrete Structures Aging Reference Manual*. EPRI, Palo Alto, CA: 2010. 1023035.
180. *Concrete Civil Infrastructure in United States Commercial Nuclear Power Plants*. EPRI, Palo Alto, CA: 2010. 1020932.
181. *Nondestructive Evaluation Inspection of Concrete Structures Subjected to Corrosion*. EPRI, Palo Alto, CA: 2010. 1025627.
182. *Effects of Radiation in Concrete*. EPRI, Palo Alto, CA: 2010. 1025584.
183. *Boric Acid Attack of Concrete and Reinforcing Steel in PWR Fuel Handling Buildings*. EPRI, Palo Alto, CA: 2010. 1025166.

- 184. *Embedded Sensors in Concrete*. EPRI, Palo Alto, CA: 2010. 1023006.
- 185. *Quality Control of Concrete during Construction – Voids Detection*. EPRI, Palo Alto, CA: 2010. 1025300.
- 186. *Nonlinear Ultrasound to Evaluate the Integrity of Thermally Damaged Concrete*. EPRI, Palo Alto, CA: 2010. 1026501.

Electrical AMPs

- 187. *Plant Support Engineering: License Renewal Electrical Handbook*. EPRI, Palo Alto, CA: 2007. 1013475.
- 188. *Natural Versus Artificial Aging of Electrical Components: Interim Report 1996 – 2000*. EPRI, Palo Alto, CA: 2001. 1003062.
- 189. *Medium Voltage Cable Aging Management Guide, Revision 1*. EPRI, Palo Alto, CA: 2010. 1021070.
- 190. *Plant Engineering: Medium-Voltage Cable Failure Mechanism Research, Update 4*. EPRI, Palo Alto, CA: June 2012. 1024894.
- 191. *Plant Support Engineering: Life Cycle Management Planning Sourcebooks: Medium-Voltage (MV) Cables and Accessories (Terminations and Splices)*. EPRI, Palo Alto, CA: 2006. 1013187.
- 192. *Plant Support Engineering: Aging Management Program Guidance for Medium-Voltage Cable Systems for Nuclear Power Plants*. EPRI, Palo Alto, CA: 2010. 1020805.
- 193. *Plant Support Engineering: Aging Management Program Development Guidance for AC and DC Low-Voltage Power Cable Systems for Nuclear Power Plants*. EPRI, Palo Alto, CA: 2010. 1020804.
- 194. *Plant Engineering: Cable Aging Management Program Implementation Guidance*. EPRI, Palo Alto, CA: 2011. 1022968.
- 195. *Plant Support Engineering: Aging Management Program Development Guidance for Instrument and Control Cable Systems for Nuclear Power Plants*. EPRI, Palo Alto, CA: 2010. 1021629.
- 196. *Plant Support Engineering: Nuclear Power Plant Equipment Qualification Reference Manual, Revision 1*. EPRI, Palo Alto, CA: 2010. 1021067.
- 197. *Plant Engineering: Electrical Cable Test Applicability Matrix for Nuclear Power Plants*. EPRI, Palo Alto, CA: 2011. 1022969.
- 198. *Plant Engineering: Evaluation and Insights from Nuclear Power Plant Tan Delta Testing and Data Analysis*. EPRI, Palo Alto, CA: 2012. 1025262.
- 199. *Plant Engineering: Dewatering Effects on Medium-Voltage Ethylene Propylene Rubber Cable*. EPRI, Palo Alto, CA: 2012. 1025263.
- 200. *Long-Term Operations: Ethylene Propylene Rubber (EPR) Insulation Accelerated Aging Methodology Research for Medium-Voltage Cables*. EPRI, Palo Alto, CA: 2012. 1026507.

The Electric Power Research Institute, Inc. (EPRI, www.epri.com) conducts research and development relating to the generation, delivery and use of electricity for the benefit of the public. An independent, nonprofit organization, EPRI brings together its scientists and engineers as well as experts from academia and industry to help address challenges in electricity, including reliability, efficiency, affordability, health, safety and the environment. EPRI also provides technology, policy and economic analyses to drive long-range research and development planning, and supports research in emerging technologies. EPRI's members represent approximately 90 percent of the electricity generated and delivered in the United States, and international participation extends to more than 30 countries. EPRI's principal offices and laboratories are located in Palo Alto, Calif.; Charlotte, N.C.; Knoxville, Tenn.; and Lenox, Mass.

Together...Shaping the Future of Electricity

Program:

Nuclear Power

© 2013 Electric Power Research Institute (EPRI), Inc. All rights reserved. Electric Power Research Institute, EPRI, and TOGETHER...SHAPING THE FUTURE OF ELECTRICITY are registered service marks of the Electric Power Research Institute, Inc.

3002000576

Electric Power Research Institute

3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com