NYS000393
Submitted: June 19, 2012
EXCERPT

NUCLEAR SECTOR ROADMAPS

JANUARY 2012

TABLE OF CONTENTS

MATERIALS DEGRADATION AND AGING 7

IN USE

BWR and PWR Irradiated Materials Testing and Degradation Models 9

Steam Generator Foreign Object Management...12

Steam Generator Life (Deposit) Management 15

Management of Jet Pump Flow-Induced Vibration 18

Ensuring Reactor Pressure Vessel Integrity through Eighty Years of Operation....................................21

PWR Reactor Internals Aging Management 24

Welding of Irradiated Materials for PWR and BWR Internals 27

Alloy 52 Nickel-Base Filler Metal Weldability Solution 30

High Chromium Weld Metal Development and Cracking Mechanisms 33

Aging Management of Alloy 600 and Alloy 82/182 in the Steam Generator Channel Head Assembly ...36

Mitigation of Boiling Water Reactor Materials Degradation 39

Fatigue Management in Boiling and Pressurized Water Reactor Coolant Systems...............................42

Pipe Rupture Probability Reassessment (XLPR) 45

Primary Water Stress Corrosion Cracking Characterization of Alloy 690 and Weld Metals................48

Advance Welding Process Development in the Nuclear Power Industry 51

Incorporation of Fabrication, Repair and Joining Technologies in Nuclear Codes and Standards.........54

DRAFT

Steam Generator Tube Integrity Assessment 57

FUEL RELIABILITY 61

IN USE

PWR Grid-to-Rod Fretting Fuel Failure Prevention 63

Mitigation of Fuel Failures Caused by Foreign Material 67

Mitigation of PWR Fuel Failures Caused by Crud and Corrosion.. 70

Mitigation of BWR Fuel Failures Caused by Corrosion and Crud 73

This roadmap book is available at http://www.cvent.com/d/6cq8dc/4K.

Loss of Coolant Accident Regulation..........77

Zero Fuel Issues Strategic Matrix..........80

USED FUEL AND HIGH-LEVEL WASTE MANAGEMENT 83

IN USE

Advanced Fuel Cycles..........85

Used Fuel Extended Storage 88

High Burn-Up Fuel Transportation 91

NONDESTRUCTIVE EVALUATION 95

IN USE

Assessment of Cast Stainless Steel 97

Concrete Structure Characterization and Nondestructive Evaluation 100

Remote Visual Examination 103

Nondestructive Evaluation Modeling and Simulation Center 106

Nondestructive Evaluation for New Plants 109

Nondestructive Evaluation Reliability 112

EQUIPMENT RELIABILITY 115

IN USE

Buried Pipe Integrity..........117

Cable Aging Management..........120

Digital Instrumentation & Control Implementation (New/Existing Plants)..........123

Advanced Preventive Maintenance Technologies..........126

Guidance for Maintaining Circuit Cards, Relays, and Other I&C Components 129

DRAFT

Emergency Diesel Generator Systems.......... 132

RISK AND SAFETY MANAGEMENT.......... 135

IN USE

Probabilistic Risk Assessment for Internal Fire 137

This roadmap book is available at http://www.cvent.com/d/6cq8dc/4K.

Probabilistic Risk Assessment for Seismic Events *140*

Methods for Risk Assessment for Low-Power, Shutdown, and Transition Conditions *143*

Phoenix: Advanced, Integrated Software for Nuclear Plant Risk Management *146*

Risk Assessment Methodologies for External Hazards Other than Fire and Seismic *149*

ADVANCED NUCLEAR TECHNOLOGY 153

IN USE

Configuration Management for New Nuclear Plants *155*

Environmentally-Assisted Fatigue *158*

Small Modular Light Water Reactor Requirements *161*

CHEMISTRY, LOW LEVEL WASTE AND RADIATION MANAGEMENT 165

IN USE

Technical Solutions for Reducing Worker Radiation Exposure *167*

Science and Communication of Low-Dose Radiation Risk *170*

Optimized Storage and Disposition of Low and Intermediate Level Waste *173*

Radiological Environmental Protection *176*

Water Chemistry Strategy for Reducing Radiation Fields *179*

Auxiliary System Chemistry Optimization *182*

Development of Chemistry Software Applications *185*

Water Chemistry Guidelines for Advanced Light Water Reactors *188*

Dispersant Application for Fouling Mitigation *191*

LONG-TERM OPERATIONS 195

IN USE

Long-Term Operations *197*

This roadmap book is available at http://www.cvent.com/d/6cq8dc/4K.

MATERIALS AGING AND DEGRADATIONS

Action Plan Roadmap Summary

OBJECTIVES

Metal materials degradation and aging have been problematic for commercial light water reactors since the mid-1970s (e.g., PWR steam generator tube leaks, BWR recirculation pipe cracking, BWR reactor vessel internals issues, PWR reactor pressure vessel head penetration cracking and leaks). Problems associated with actual degradation pose reliability, regulatory and in some cases safety concerns, and as plants age and move intolicense renewal/life-extension, the impact of the radiation environment on the susceptibility to degradation increases.

Materials degradation and aging research at EPRI develops guidelines and technologies to cost-effectively manage component and system degradation and aging and to inform strategic decisions on whether and when to replace, repair, or continue operation of such components and systems. The specific strategic objectives of the program are to:

- Maximize the operating life and reliability of BWR and PWR passive long-lived components;
- Predict component degradation mechanisms and their rate of occurrence to inform decisions on mitigation, repair or replacement options;
- Account for the impact on plant operations associated with implementing materials aging management activities;
- Develop data and physically based predictive models for remaining useful life assessments;
- Identify and disposition degradation mechanism knowledge gaps through fundamental R&D; and
- Conduct research, evaluate and optimize joining, fabrication and repair processes;

CURRENT ISSUES AND PLANNED RESEARCH

Near- and Mid-Term Research

Research in this program for the near and midterm focuses on continued materials testing to better understand the phenomena of crack initiation and growth in light water reactor materials as well as to understand the impact of irradiation on both time-to-initiation and growth rate. This information will then be used to update guidance on activities to periodically inspect and repair or replace impacted components and systems. Additionally there are activities underway to address specific system and component issues.

For PWRs, the ability to monitor and demonstrate the structural integrity of the reactor pressure vessel (RPV) through 80 years of operation is essential. To that end, PWRs will implement EPRI's coordinated reactor vessel surveillance program beginning in 2011, which will generate the high-fluence surveillance data and irradiated materials samples needed to support embrittlement correlations and long-term damage mechanism assessments. For PWR steam generators, recent operating experience demonstrates that knowledge of relevant damage mechanisms may be inadequate for accurately projecting steam generator life. One particular area of concern is the continuing problem of loose parts, which can be introduced to the steam generator during the manufacturing process or from the secondary side of the plant. A new thermal-hydraulic computer code incorporating computing and technological advances over the last 30 years will be instrumental in assessing the various damage mechanisms. Work is underway in several areas – improved management of deposit accumulation, development accumulation computer models, loose parts identification and retrieval, etc. – for developing improved steam generator life management strategies.

For BWRs, a number of plants worldwide are experiencing jet pump degradation associated with flow-induced vibration. Work in this area includes compiling field data (operating experience, repair history, configuration, etc.), sub-scale phenomenological testing, and full scale prototypical jet-pump assembly testing to assess the effectiveness of proposed mitigation solutions.

In the area of BWR and PWR repair technology, high-chromium, nickel-based weld Alloys 52 and 52M, chosen for their superior resistance cracking, are used extensively for repair and mitigation of stress corrosion cracking (SCC) in Alloy 82/182 dissimilar metal welds joining critical reactor coolant system components. Experience shows that the weldability and crack susceptibility of Alloys 52 and 52M vary widely with minor variations in material specification limits. To address this issue until new/improved filler materials can be developed, activities are being undertaken to: 1) perform weldability testing to understand and rank weldability; 2) assess the influence of base metal composition on identified weldability problems; 3) evaluate welding processes and the influence of process parameters; and 4) develop application plans and guidance for welding vendors.

Longer-Term Research

Research in the longer term will focus on BWR and PWR irradiated materials testing and degradation models as well as improved weld repair solutions and mitigation measures. BWR and PWR reactor internals are affected by several irradiation-based degradation mechanisms: irradiation-assisted stress corrosion cracking (IASCC), irradiation embrittlement, creep, stress relaxation and void swelling. A number of knowledge gaps could have a major impact on decisions related to extended plant operations (beyond current design life). Long-term irradiation effects will be characterized by testing materials removed from retired plants as well as using information obtained by continued participation in worldwide programs to develop irradiated specimens in various test reactors. This information will then be used to develop new and improve existing models to predict residual lifetimes of irradiated BWR and PWR reactor internals components.

As noted above, Alloys 52 and 52M weld filler metal have been difficult to use in field applications. There is a need to develop a new high-chromium, nickel-based welding alloy that has the desired mechanical and corrosion resistance properties, but also has significantly improved weldability and superior resistance to weld cracking. To that end, research and laboratory weldability testing will be performed to understand the fundamental issues causing the observed problems, new alloy composition specimens will be developed and laboratory tested, and full-scale mock-up and NDE testing will be performed and validated leading to work with various manufacturers to develop a new weld-metal specification.

Finally, the continued operation of light water reactors will likely require weld repair of certain reactor internals components. Knowledge gaps exist related to the weldability of irradiated nickel alloys and RPV steel and to special welding techniques for repairing high-fluence materials. The initial phase of this work will include: 1) develop a weldability assessment for PWR designs; 2) refine conventional welding models for predicting the weldability of irradiated stainless steel; 3) develop a laser welding predictive model for welding irradiated stainless steel; and (4) develop techniques for applying laser welding to reactor internals repairs. The second phase focuses on development and testing of tools such as models and advanced welding processes for the more highly irradiated materials that will be encountered during life-extension periods.

RISKS

The issues and research plans inherently involve some risks. Both the near- and long-term research plans rely on materials testing that can raise questions regarding applicability of such results to actual field conditions. Obtaining suitable specimens from retired plants or from test reactors poses separate but unique problems: for retired plants, the need to assure that the specimens are representative and can be obtained without interfering with plant activities;; and for test reactors, the need to deal with fluence levels and other scaling factors. Where field trials and tests are needed, there are risks associated with locating and obtaining commitments from suitable volunteer utilities/plants. In the area of weld repairs and weld process development, intellectual property and licensing risks may arise with service vendors.

SUPPORTING OTHER STRATEGIC NEEDS

The Materials Degradation and Aging research programs work across the Nuclear Sector to provide substantial support to both the Advanced Nuclear Technology Program and Long Term Operations programs in the area of materials management and improvement. EPRI's materials research programs also receive substantial support from and interaction with the Nondestructive Evaluation Program, the Low Level Waste and Radiation Management Program, and the Water Chemistry Program.

IN USE: AGING MANAGEMENT OF ALLOY 600 AND ALLOY 82/182 IN THE STEAM GENERATOR CHANNEL HEAD ASSEMBLY

ISSUE STATEMENT

Primary water stress corrosion cracks that initiate in Alloy 600 and associated weld materials in the steam generator channel head could propagate over time to pressure boundaries such as the tube-to-tubesheet weld or the carbon steel materials in the bowl and cause primary-to-secondary leakage. Two scenarios are under consideration.

In the first scenario, a primary water stress corrosion crack in the divider plate assembly (Alloy 600) could reach the channel head, which is a pressure boundary. The channel head is carbon steel and is not susceptible to primary water stress corrosion cracking (PWSCC), but the stresses in this region are unknown and could be sufficient to cause growth via fatigue. This is applicable to U.S. (30 units) and non-U.S. steam generators.

In the second scenario, PWSCC in the tubesheet cladding could propagate over time to the tube-to-tubesheet weld, which is the pressure boundary in some steam generator designs. The applicable steam generator designs have Alloy 690TT tubing and a cladding that is Alloy 600 weld material. The susceptibility of the weld between the tubing and the cladding to PWSCC is unknown. This is applicable to U.S. (25 units) and non-U.S. steam generators

Lack of understanding hinders the ability to make sound decisions regarding monitoring and potential mitigation in the channel head region.

DRIVERS

Aging Management Drivers

PWSCC in susceptible materials could grow over time and reach non-susceptible or less susceptible materials that form the pressure boundary in the channel head assembly. The industry lacks understanding of the impact of such cracks on pressure boundary materials, which is especially important in ensuring safe operation as steam generators age. Research is needed to address this issue in aging management plans.

Regulatory Drivers

Based on operating experience from two utilities in Europe, the U.S. Nuclear Regulatory Commission is requiring plants with Alloy 600 material in the channel head assembly (divider plate, stub runner, tubesheet cladding, and associated welds) to 1) include the material in their aging management plans and 2) commit to inspection after entering the period of extended operation and after the steam generators have reached 20 years of operation.

Inspection and Worker Dose Drivers

There are no qualified techniques to inspect the steam generator channel head. Existing inspection methods used by a utility in Europe to inspect the steam generator divider plates result in significant worker dose. Development of a new, more efficient technique will reduce worker dose.

RESULTS IMPLEMENTATION

Upon completion of this work, it is expected that:

1. Nuclear plants will update aging management plans, and EPRI will update the steam generator guideline documents based on research results related to divider plate crack propagation and cladding crack propagation;
2. Vendors will offer qualified inspection techniques to identify cracking in the steam generator channel head; and
3. Nuclear plants will update steam generator programs and plant procedures to reflect research results and operating experience.

PROJECT PLAN

Divider Plate Crack Propagation

Objectives: To determine the integrity of the steam generator when cracks propagate to the channel head and to develop and demonstrate an inspection technique to determine if cracks exist in the channel head.

Review and Compilation of Existing Information

Other issue programs, such as the Boiling Water Reactor Vessel Internals Project (BWRVIP) and the Materials Reliability Program (MRP), have studied cracking behavior when it comes in contact with material that is not susceptible to PWSCC. This information will be compiled, and existing research results will be investigated to determine the applicability to the divider plate crack propagation issue.

Analytical Modeling

Finite element modeling will be used to determine the maximum stress distributions in a steam generator channel head assembly. This will be used as input to determine a critical flaw size for the channel head material and the allowable flaw size considering factors of safety. Fatigue crack growth analyses will be performed for the channel head to determine the operating period required for the postulated initial flaw to reach the allowable flaw size.

Effective Inspection

Existing technology to inspect the divider plate assembly uses a combination of visual, liquid penetrant, and ultrasonics inspections from inside the steam generator bowl. These methods are slow and dose intensive. To ensure that cracking has not propagated into the pressure boundary base material of the channel head assembly, a more effective solution will be developed. A feasibility study will be conducted to determine if existing ultrasonic methods/transducers can be used from the outside of the bowl to inspect for cracking that propagates through the clad and into the base material of the steam generator bowl. If successful, mockups will be located or developed to demonstrate the inspection technique. If unsuccessful, an investigation will begin to develop a technique to inspect the divider plate by going inside the bowl using phased array ultrasonics.

Steam Generator Guidelines

EPRI will update the Steam Generator Integrity Assessment Guidelines and the Steam Generator Examination Guidelines to incorporate inspection and integrity assessment guidance.

Tubesheet Cladding Crack Propagation

Objectives: To determine the range of potential chromium content in autogenous gas-tungsten-arc welds between Alloy 690 tubing and Alloy 82/182 cladding material and to determine the susceptibility of those welds to PWSCC.

Review and Compilation of Existing Information

Using EPRI's Alloy 82/182 weld material databases and nuclear plant data on 690 tubing material, field tube-to-tubesheet weld compositions will be estimated. A literature search will be conducted to determine the acceptable level of chromium for resistance to primary water stress corrosion cracking.

Analytical Modeling

Weld dilution models will be developed to estimate chromium levels for autogenous gas-tungsten-arc welds between Alloy 690 tubing and Alloy 82/182 cladding. The results of this model in conjunction with the results of the literature review will be used to determine if the tube-to-tubesheet weld is susceptible to PWSCC. Finite element modeling will be used to determine the stresses in the tubesheet area. To determine how a crack in the cladding will propagate, the finite element analysis will be modified to include the initiation of a crack in the cladding. The model will then determine if the crack would ultimately penetrate the weld and lead to a through-wall crack.

Mockup Testing

Test welds from existing mockups or from mockups built by EPRI's Welding and Repair Technology Center will be analyzed for chromium content by measuring across the weld cross-section. The measured chromium distributions will be compared to the distributions predicted using the dilution model to determine the most representative mockups to use.

If the results of the testing indicate that the tube-to-tubesheet weld is susceptible to PWSCC, the industry would develop an alternate repair criteria for 690TT tubing similar to H* for Alloy 600TT tubing that would move the pressure boundary from the tube end weld to some defined distance below the top of the tubesheet.

RISKS

Availability of Information

As-built information about the channel head assembly is needed to build the mockups to demonstrate the inspection technique. Utilities and vendors will need to provide the as-built information. The information may not be easily accessible.

External Stakeholder Participation

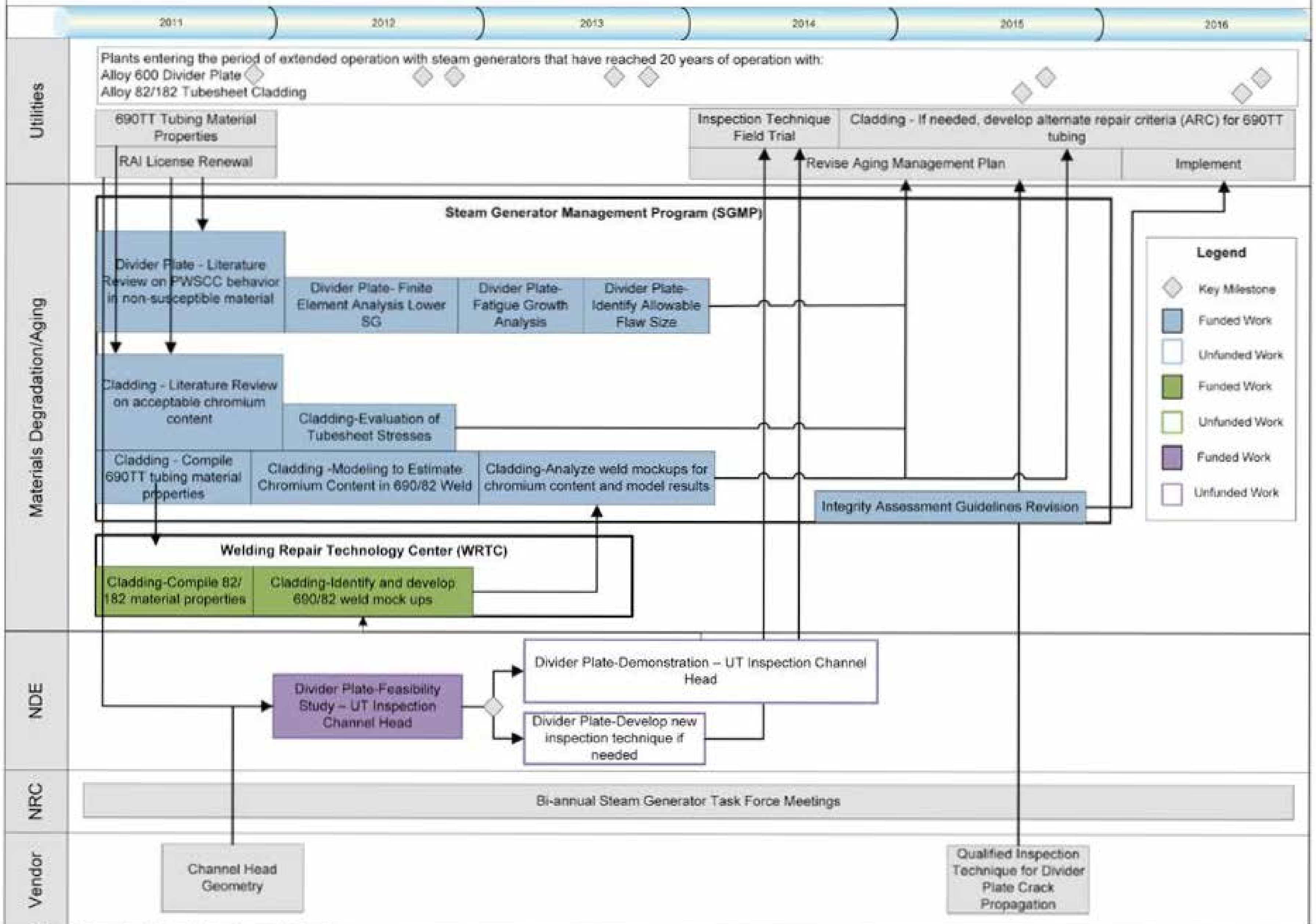
Utility involvement is needed to build the database and develop the mockups for the tube-to-tubesheet welds. If this information is not made available to EPRI in a timely manner, the progress of this project would be affected.

RECORD OF REVISION

This record of revision will provide a high level summary of the major changes in the document and identify the Roadmap Owner.

REVISION	DESCRIPTION OF CHANGE
0	**Original Issue:** August 2011 **Roadmap Owner:** Heather Feldman
1	**Original Issue:** December 2011 **Roadmap Owner:** Heather Feldman **Change:** Flowchart updated. Alloy 82/182 was added to the roadmap title.

Aging Management of Alloy 600 and Alloy 82/182 in the Steam Generator Channel Head Assembly



Steam Generator Channel Head Integrity Rev 1
December 2011