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 issued a Regulatory Information Summary (RIS 2011-05) that requires 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 designed 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. If the engineering analysis concludes that inspections are not necessary and the NRC issues a Safety Evaluation Report documenting their concurrence with this conclusion, mockups will not be built and inspection techniques will not be developed.

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.

<table>
<thead>
<tr>
<th>REVISION</th>
<th>DESCRIPTION OF CHANGE</th>
</tr>
</thead>
</table>
| 0 | Original Issue: August 2011  
Roadmap Owner: Heather Feldman |
| 1 | Revision Issued: December 2011  
Roadmap Owner: Heather Feldman  
Changes: Flowchart updated. Alloy 82/182 was added to the roadmap title. |
| 2 | Revision Issued: August 2012  
Roadmap Owner: Heather Feldman  
Changes: Added RIS reference to Regulatory Driver. Updated flowchart to include milestones. Moved NDE technology development decision from 2013 to 2014 to be aligned with the conclusion of the engineering analysis. |
| 3 | Revision Issued: December 2012  
Roadmap Owner: Heather Feldman  
Changes: Updated flowchart. |
| 4 | Revision Issued: August 2013  
Roadmap Owner: Helen Cothron  
Changes: Updated Flowchart. |
| 5 | Revision Issued: August 2014  
Roadmap Owner: Randy Stark  
Changes: Updated flowchart and section on effective inspection |
| 5 | Revision Issued: August 2015  
Roadmap Owner: Helen Cothron |
| 6 | Revision Issued: August 2016  
Roadmap Owner: Helen Cothron  
Changes: Updated Flowchart and closed roadmap |
Plants entering the period of extended operation with steam generators that have reached 20 years of operation with:

Alloy 600 Divider Plate
Alloy 62/182 Tubesheet Cladding
690TT Tubing Material Properties

Legend
- Key Milestone
- Complete Milestone
- Funded Work
- Unfunded Work
- NDE Technology not needed
- NRC ISG Letter 2016-01

Milestones
- EPRI Technical Report 1025133
- EPRI Technical Report 30200200411
- EPRI Technical Report 30200202850
- SG Integrity Assessment Guidelines, Rev 4 302007571
- NDE Technology not needed
- NRC ISG Letter 2016-01

Bi-annual Steam Generator Task Force Meetings
Review Final Report for Possible Generic SER