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NRC / EPRI Steam Generator Task Force Meeting
February 6, 2013

Agenda

8:30 am ***Introductions*** ***NRC and Industry***

Opening Remarks ***NRC and Industry***

EPRI SGMP Steam Generator Task Force Update

- Noise Monitoring
- Anti-Vibration Bar Position Verification
- Divider Plate Cracking
- Tube-to-Tubesheet Weld
- Guidance for Auto Analysis
- Nuclear Safety Advisory Letter 12-01
- Potential Generic Implications of Probe Issue
- Guidance for Performing Inspections Following a Design Basis Event

10:00 am ***Break***

Agenda

10:15 am ***SGMP Steam Generator Task Force Update (continued)***

- Technical Specification Task Force 510
- Upcoming Changes to Industry Documents
- NEI 03-08 Deviations Since February 16, 2012
- ASME Code Changes for Pre-Service Inspection
- Recent Steam Generator Operating Experience

11:00am **Response to *NRC Comments from August 2012 Meeting***

- NRC Comments on ETSS Reports
- Time Dependent Leak Rate
- Industry guidelines pertaining to the use of Nitrogen-16 monitors

11:30 am ***Address Public Questions/Comments (NRC)***

11:45 am ***Adjourn***

Open Technical Issues

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Open Technical Issues - SGTF

- Noise Monitoring
  - Recommendations have been provided to the Examination Guidelines Revision 8 Committee
    - New appendix has been drafted for noise monitoring

Open Technical Issues - SGTF

- AVB Position Verification
  - White paper developed incorporating information from 2012 meeting among SG vendors
    - Available to utilities on EPRI website
  - Recommendations have been provided to the E&R TAC for consideration for inclusion in the next revision of the Integrity Assessment Guidelines
  - Westinghouse published NSAL 12-7, "Insufficient Insertion of Anti Vibration Bars in Alloy 600TT Steam Generators with Quatrefoil Tube Support Plates"
  - SGMP Project began project in 2012 to provide generic input for plant-specific U-bend tube fatigue analysis
  - Follow up survey in 2012 indicates the utilities are familiar with the issue and are taking actions as appropriate

Open Technical Issues - NRC

- Performance Standards
    - EPRI Technical Report 1012984, “Technical Basis for SG Tube Integrity Performance Acceptance Standards”, provided to NRC

General Discussion Items

Investigation into Potential Propagation of Divider Plate Cracking

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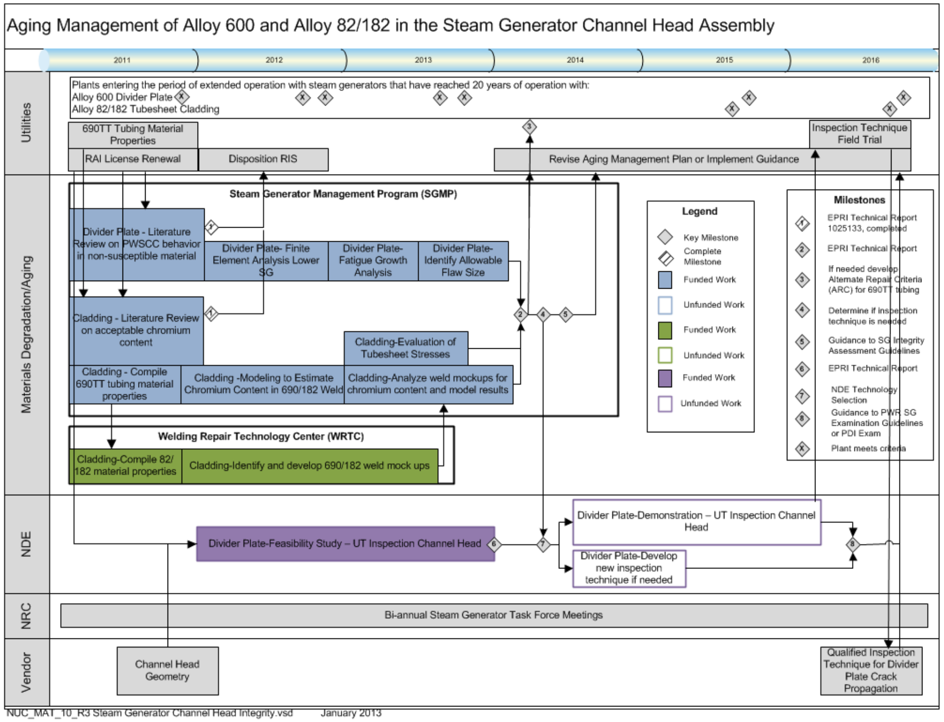
Divider Plate Crack Propagation

- Roadmap developed to address aging management issue of Alloy 600 and Alloy 82/182 in steam generator channel head assembly
  - Divider plate crack propagation into channel head material
  - Tube-to-tubesheet weld susceptibility to PWSCC
  - UT inspection from outside steam generator

Divider Plate Crack Propagation Roadmap



- Roadmap developed to address aging management issue of Alloy 600 and Alloy 82/182 in steam generator channel head assembly
  - Divider plate crack propagation into channel head material
  - Tube-to-tubesheet weld susceptibility to PWSCC
  - UT inspection from outside steam generator

Project to Investigate Potential Propagation of Divider Plate Cracking

- Data and operating experience suggest the divider plate crack propagation into other areas of the steam generator channel head is unlikely
  - Report was published in June, “Assessment of Channel Head Susceptibility to PWSCC,” 1025133
    - NRC staff provided a copy of this report
- Finite element modeling of the stresses in the channel head region is complete
- Tube-to-tubesheet weld mockups have been analyzed for chromium level
  - Initial evaluation supports dilution modeling approach
  - Report will be published in 2013

Finite Element Modeling of the Channel Head

- 3D FEM of the steam generator channel head was developed using ANSYS
  - Includes bottom channel head, the divider plate, stub runner, tubesheet, cladding, and associated welds
  - Model 51 SG dimensions were used as this was determined to be the limiting model in Phase 1 of the project
- A crack is postulated from the divider plate PWSCC susceptible weld and introduced in the model at the triple point. The divider plate is assumed to have a 15-inch long through-wall crack which is considered to be a conservative assumption

Finite Element Modeling of SG Channel Head

- The specified design basis plant conditions for the Model 51 steam generator is used in the flaw tolerance evaluation
- Thermal transients were developed for the steam generator hot leg and cold leg side for the design basis plant conditions
- Results of the FEM stress analyses are post-processed for use in the fracture mechanics analyses
- Several stress paths are defined through the weld attachment region between the stub runner, tubesheet, and channel head wall
  - Both hoop and axial stresses are extracted along these paths
  - Weld residual stresses are also considered in the analyses

Finite Element Modeling of SG Channel Head

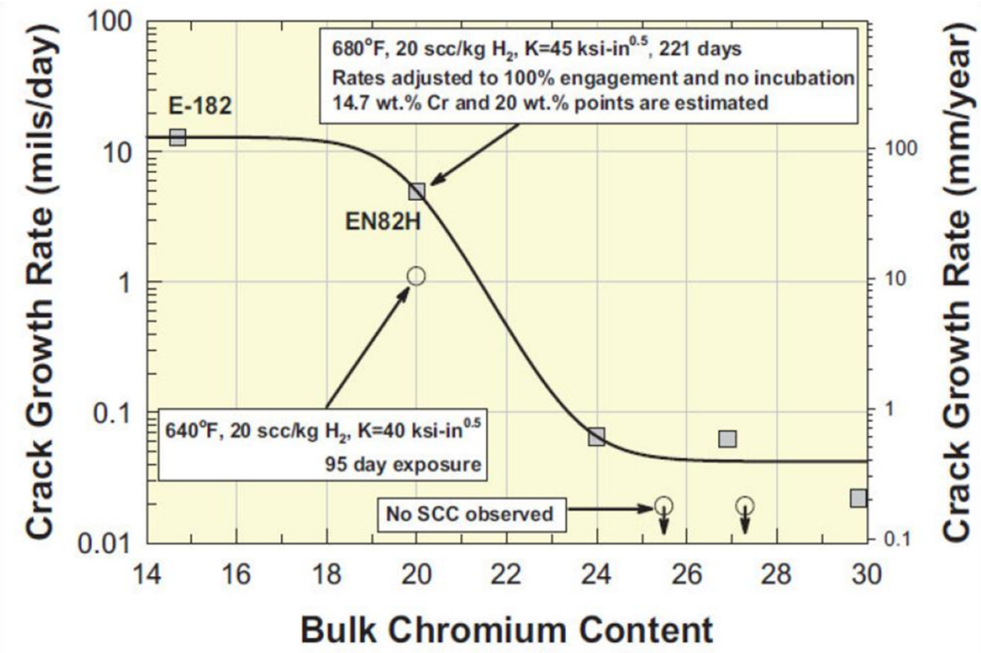
- The allowable flaw depth is determined using ASME Code Section XI, Appendix A methodology and IWB-3600 acceptance criteria
- Fatigue crack growth evaluation is performed using the methodology of ASME Code Section XI, Appendix A and using pc-CRACK software

Conclusions of FEM

- Results show that the maximum stress intensity factor, K, is less than the allowable values for all design basis plant conditions up to 75% of the vessel wall thickness.
  – Allowable flaw depth is 3.9” into the channel head
- A postulated 0.25” initial circumferential flaw would grow to a maximum depth of 0.4 inches in 40 years of operation
- A postulated 0.25” initial deep axial flaw would grow about 1 mil in 40 years of operation
  – Well below the allowable flaw depth
- The model includes several conservatisms
  – An initial flaw in the carbon steel PWSCC resistant channel head material
  – 15” crack in the divider plate

Potential for Tube to Tubesheet Weld Cracking

- Results of literature review and data collection reported at the August 2012 meeting
  - Industry has accepted a 24% chromium content as a conservative threshold for PWSCC initiation
  - Chromium levels down to 20% have excellent resistance to initiation based on testing and operational experience
- High growth rates during lab tests are likely due to cold worked test materials and elevated test temperatures
- Recent tests indicate that PWSCC arrests in resistant material



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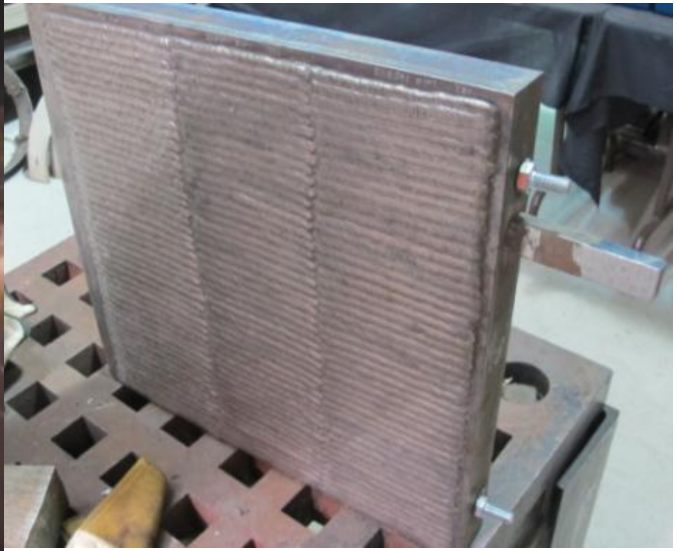
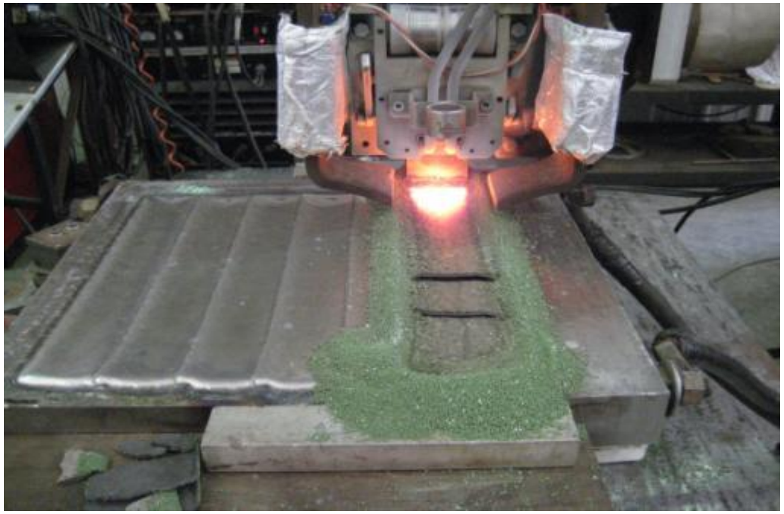
Tube-To-Tubesheet Weld Mockup

EPRI developed prototypical tube-to-tubesheet mockups to analyze chromium levels

182 SMAW process



82 - Electroslag Welding (ESW)



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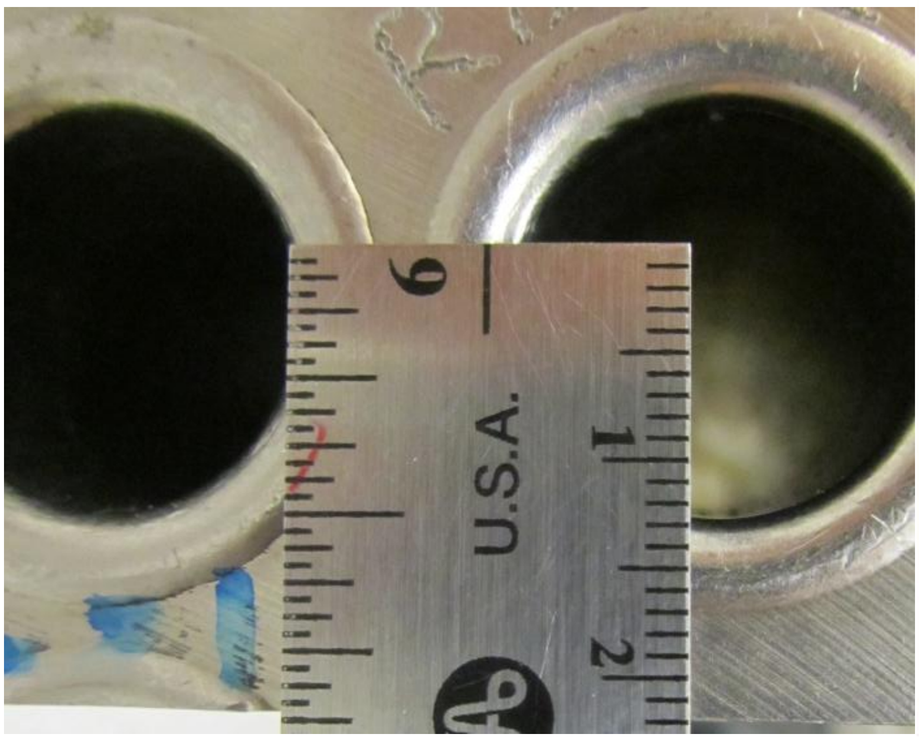
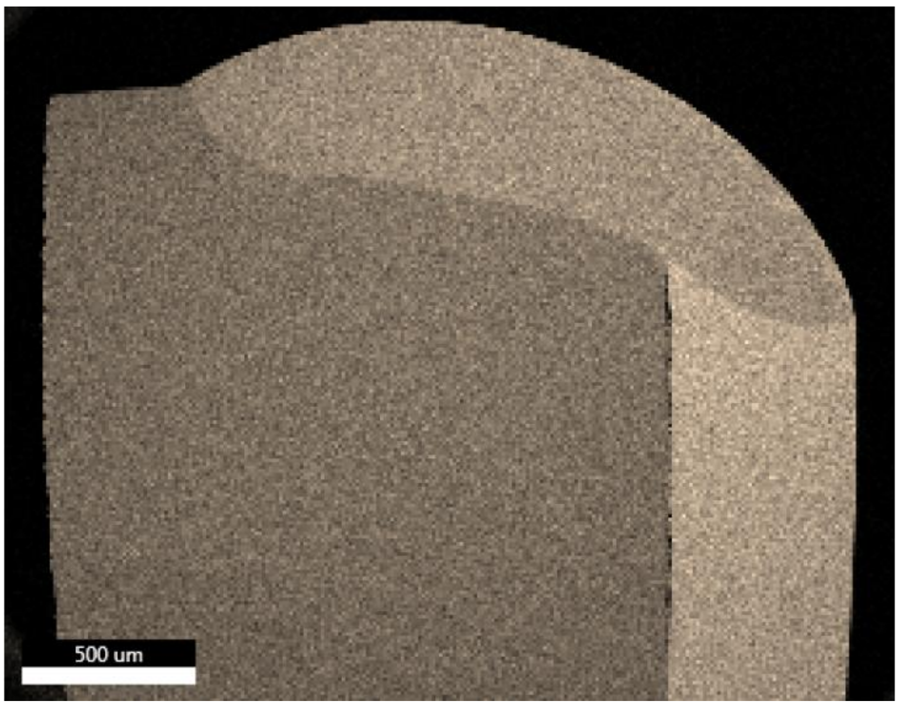
Mockup Evaluation

- Mockup from RSG fabricator with Alloy 82 clad and 690 tube material

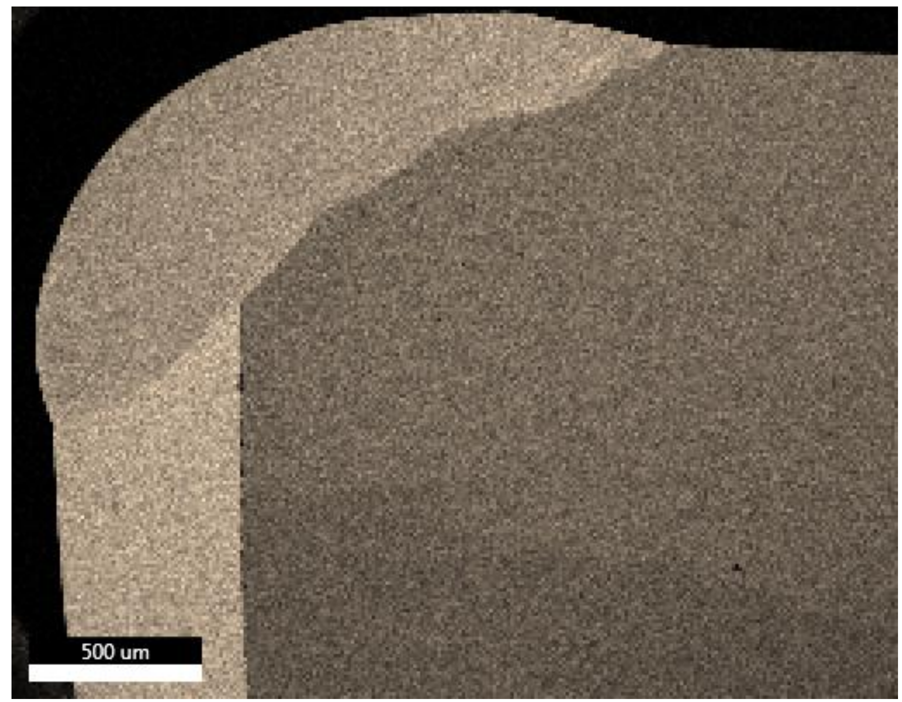


Mockup Weld Geometries Similar

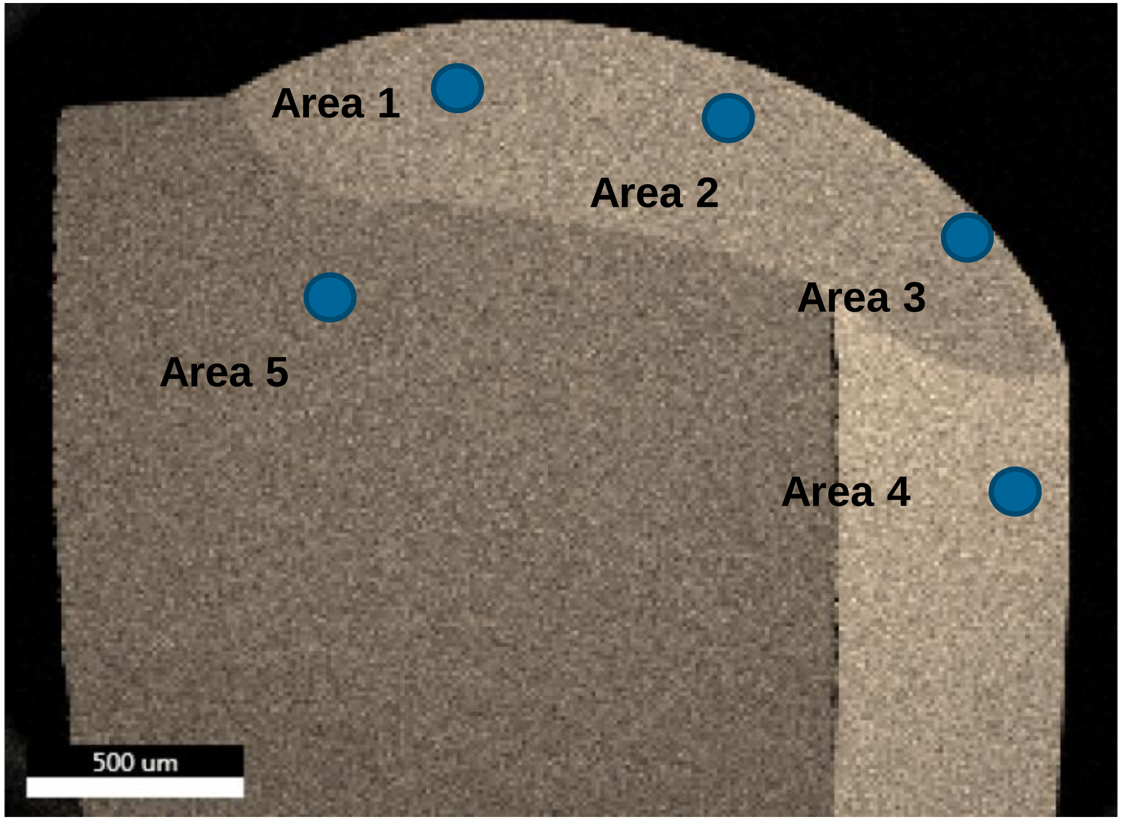
Mockup from RSG Fabricator



EPRI Mockup



Chromium Measurements Taken Confirm the 50% Dilution Algorithm



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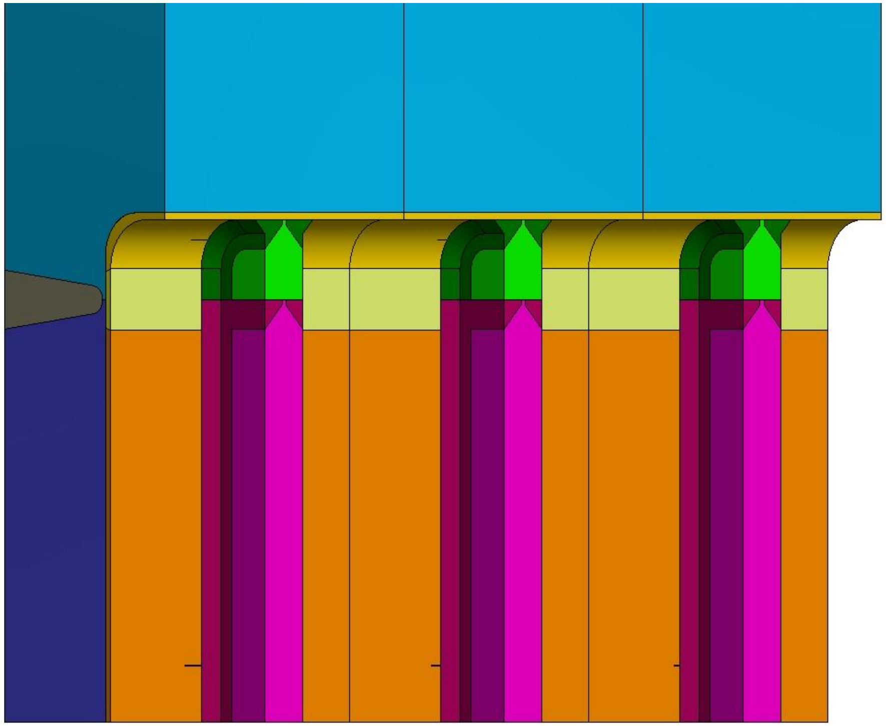
Conclusions from Analysis of Tube-to-Tubesheet Weld Mockup

- EPRI mockups shown to represent field based on the RSG mockup
- Preliminary assessment of the measured chromium content of autogenous welds validates 50% dilution algorithm
- Chrome recovery suggests high resistance to PWSCC
  - 182 weld material tests yield approximately 20% Cr
  - 82 weld material tests yield approximately 24% Cr
- Tests will be documented in a technical report in 2013
- Evaluation of stresses in the tubesheet area will be completed in 2013

Development of a Steam Generator Channel Head Inspection Technique

- Mockups have been designed to simulate the SG channel triple point
  - Demonstrate feasibility of performing UT from the outside of the SG bowl to ensure that cracking has not propagated through the clad and into the channel head material
- ASME Code inspection of the Z-Seam weld cannot be credited to include the triple point



Future Work

- Report on SG channel head FEM work will be published in 2013
  - Technical work complete
  - This report will provide the technical justification for not inspecting steam generator channel head material
- Report on tube-to-tubesheet weld work will be published in 2013
  - Data evaluation ongoing
  - Stress analysis of tubesheet region will be performed in 2013
  - This report is expected to provide the technical justification for not inspecting tube-to-tubesheet welds

Guidance for Automatic Data Analysis

Guidance for Automatic Analysis of SG Tube Eddy Current Inspection Data

- Current industry guidance for automatic data analysis requires two independent data analyses to be performed
  - The results are compared and differences resolved
- Industry guidance allows the independent analyses to be manual and/or automatic
- Two utilities, with replacement SGs, have written a technical justification to deviate from the SG Examination Guidelines requirement to perform two independent data analyses
  - Both utilities performed single pass automatic analysis on the data from their Alloy 690 SG tubes
  - Technical justification for deviation was transmitted to EPRI and NRC in accordance with NEI 03-08

SG Examination Guideline Revision

- A revision of the SG Examination Guidelines is currently ongoing
- As with any guideline revision, the Revision Committee reviews all deviations (since the last revision) and determines if there is a technical basis for changing current guideline requirements
- In addition to the technical justifications written by the utilities that deviated from guideline requirements, other technical information that may be available is also considered
- EPRI has an ongoing project to perform a study that compares the performance of two party manual analysis to single pass automatic data analysis on the same set of field data

EPRI SGMP Project on Single Pass vs Manual Analysis POD

- The initial phase of the ongoing EPRI project includes analysis of bobbin coil data from a field exam of one SG with Alloy 600MA tubing that was in service for ~20 EFPYs
  - The SG was a Model 51 SG and was replaced in 2006
  - Degradation mechanisms include ODSCC, PWSCC, wear and thinning at typical locations
- The field primary, secondary and resolution manual data analysis results will be compared to the results from a single pass auto analysis system
  - Analysis of the field data with the single pass automatic system is ongoing
  - POD information for manual and auto systems will be calculated and binned by eddy current noise
  - Initial phase is scheduled to be completed 1st Qtr. 2013

Nuclear Safety Advisory Letter (NSAL) 12-01 Update

Nuclear Safety Advisory Letter (NSAL) 12-01 Update

- A foreign utility reported base metal degradation in the primary channel head in one of three steam generators
- Unit has been in operation since 1987 and the condition was observed in Fall 2011
- The visually observed degradation is located only on one SG, and only on the cold leg side of the channel head in the vicinity of the bottom bowl drain tube
- Westinghouse communicated to customers via Nuclear Safety Advisory Letter (NSAL) 12-1
- SGMP communicated to members via e-mail

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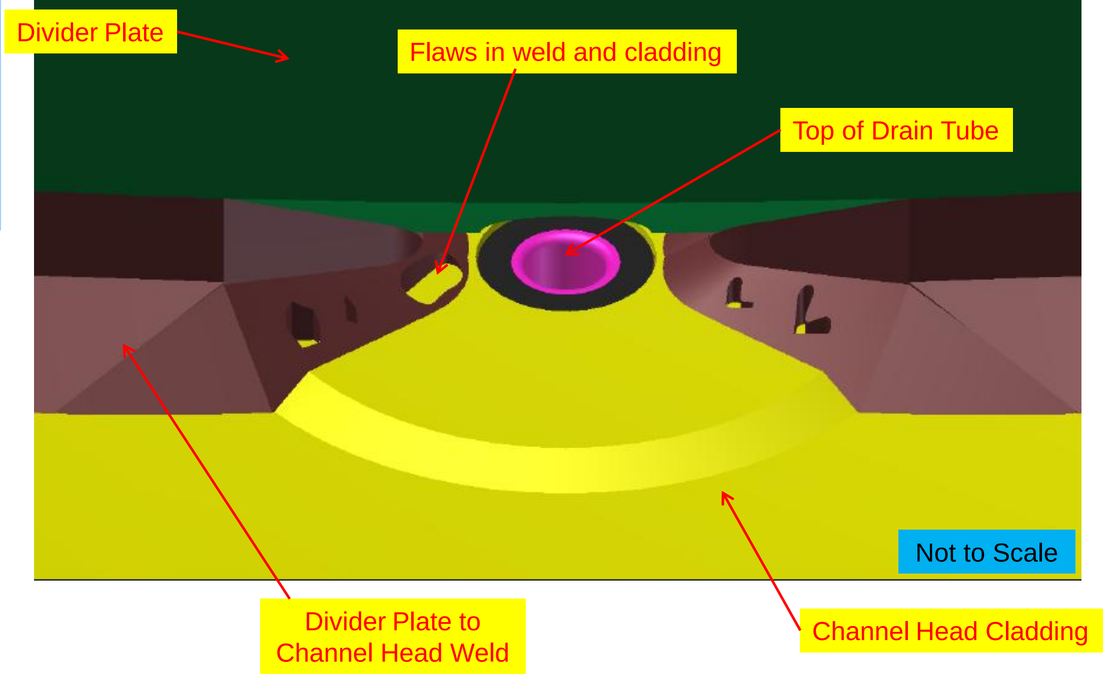
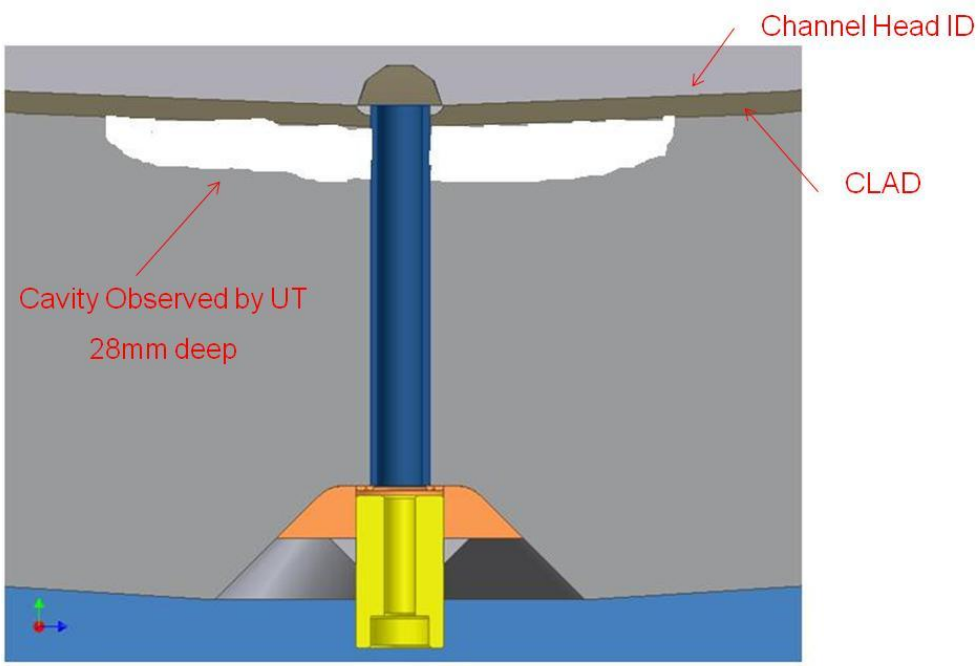


Cladding Defects and Wastage of Channel Head Base Material

Largest defect in the cladding is 7.7 mm (0.3 in) x 14.4 mm (0.6 in) by visual exam



EPRI Failure Modes and Effects Analysis

- Joint evaluation effort by EPRI Steam Generator Management Program, Materials Reliability Program, Chemistry Program, consultants, and vendor and utility personnel knowledgeable in materials and boric acid corrosion
  - Identify the most credible failure modes and applicable degradation mechanisms
  - Draft report complete and industry review is ongoing

EPRI Failure Modes and Effects Analysis Conclusions

- Most likely failure mode is gross defects in stainless steel cladding that resulted in exposure of low-alloy steel to concentrated borated water during outages based on the following assumptions:
  - No knowledge of external channel head leakage during normal operation
    - A leak of sufficient magnitude to cause high corrosion rates would produce ample evidence such that some form of corrective action would be expected
  - Defects present early in the life of the steam generator permitted concentrated borated water solution in the channel head region to contact the low-alloy steel material each outage
- FMEA Report will be published in 2013

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Industry Survey

- 44 of 69 units responded to a recent survey regarding SG channel head configuration and inspections
  - 11 have drain lines similar to the drain line described in the NSAL
    - 8 have active drain lines
  - Most utilities have inspected or are planning to inspect in accordance with the recommendations in the NSAL 12-01 regardless of channel head design
    - Based on information from the NSAL, some designs were not considered susceptible to the degradation
    - Inspection technique is visual
  - No degradation has been reported

Potential Generic Implications of ARC Probe Issue

Potential Generic Implications of ARC Probe Issue

- Bobbin ARC
  - The pulled tube database for bobbin ARC includes data from both probe types
- Bobbin ETSS(s) used by utilities to size volumetric indications
  - The SGMP investigated the ETSSs and found no issue with regard to the problem of using different bobbin probes and evaluating their equivalencies.
    - Exam Guidelines contain guidance to address the issue
- Bobbin leakage screens in the In Situ Pressure Test Guidelines
  - The bobbin voltage screens are for volumetric indications only
  - The screening values are conservatively high and would represent indications that would be of sufficient depth to require in situ testing for structural integrity

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Potential Generic Implications of ARC Probe Issue

- NRC Question: Could bobbin probes manufactured from different suppliers that have variances in voltage amplitude affect the probability of detection or sizing of tube flaws
  - Were the maximum voltage differences considered in SGMP’s investigation of the generic implications

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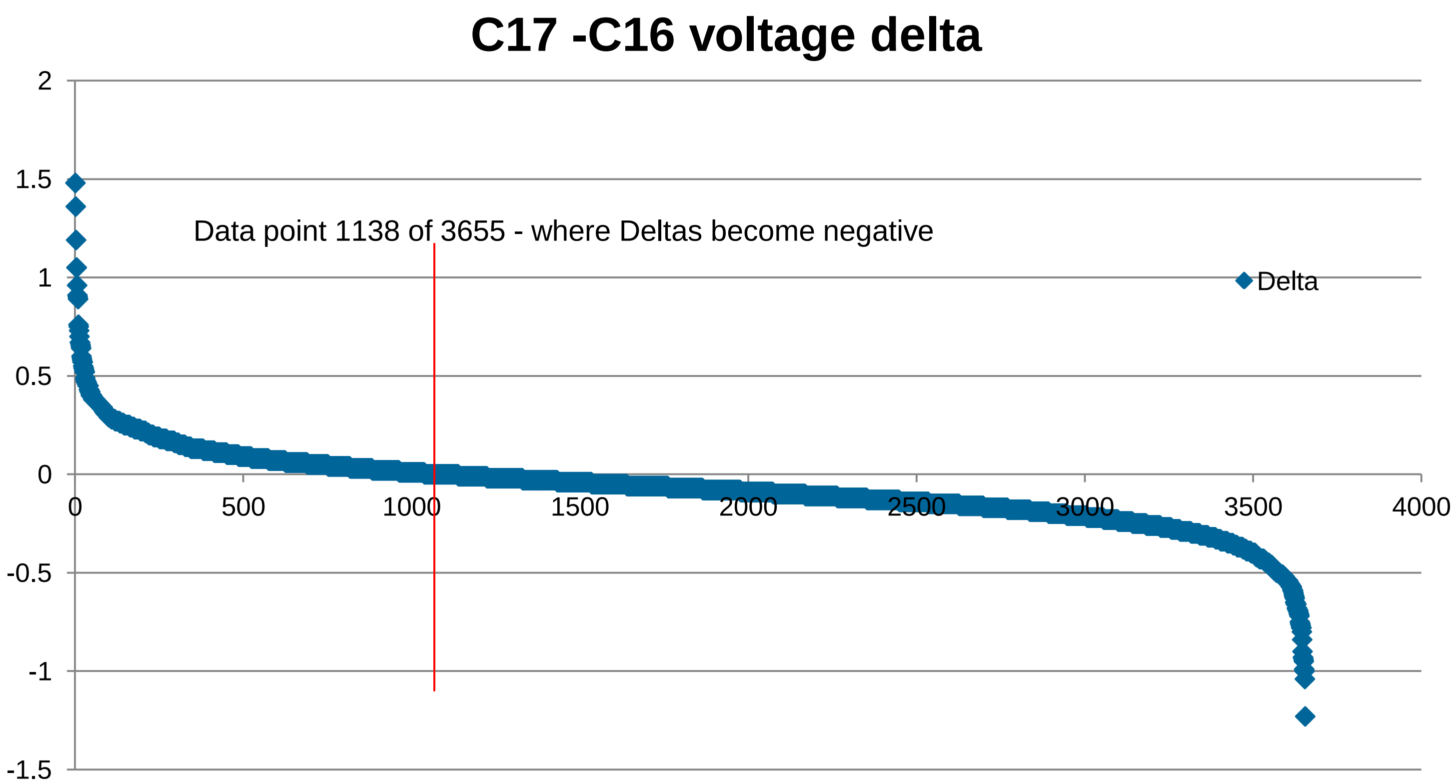
Potential Generic Implications of ARC Probe Issue

- Review of largest voltage delta does not change the conclusions
- There are no generic implications from this experience



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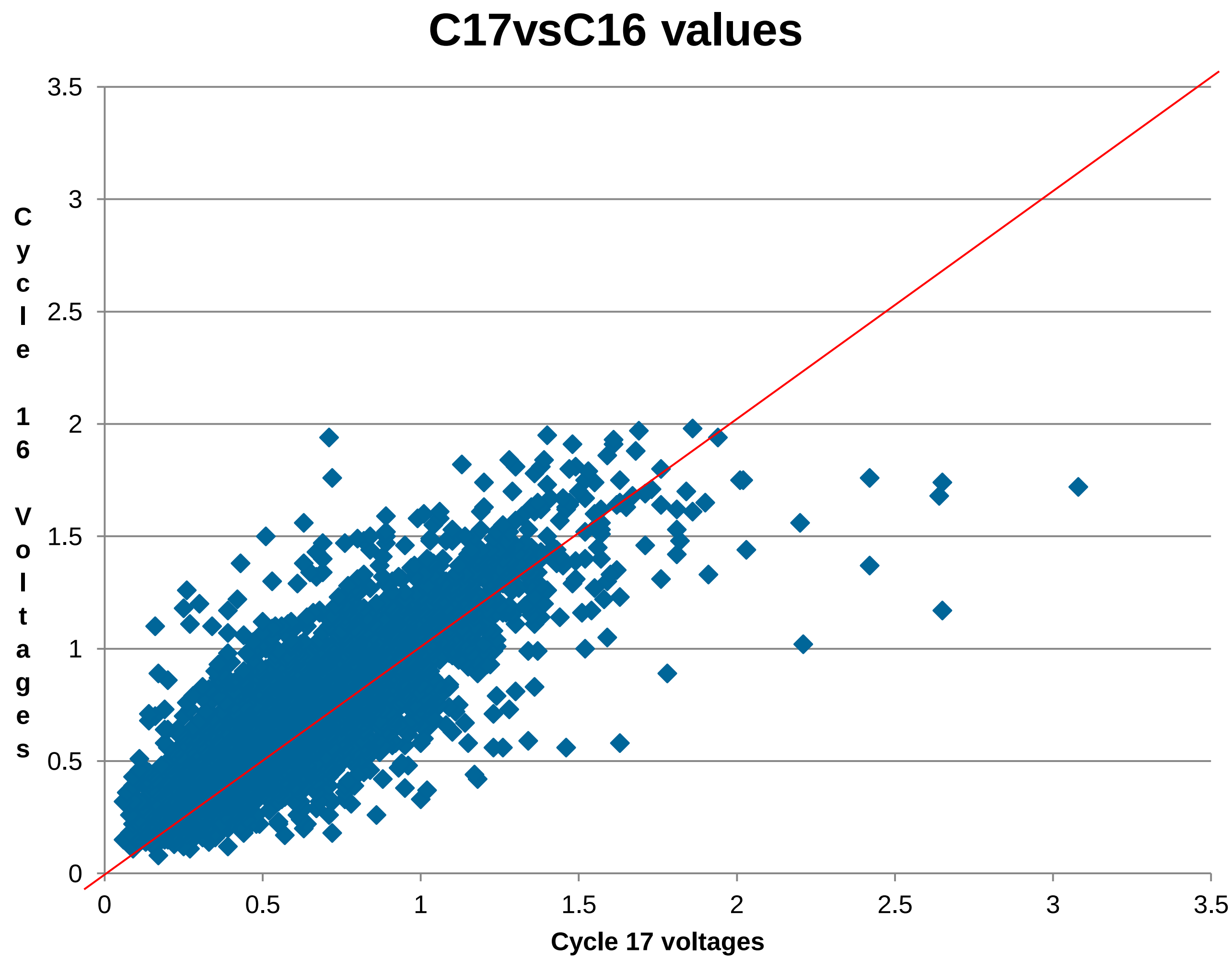
Potential Generic Implications of ARC Probe Issue



Guidance for Inspections Following a Design Basis Accident

Guidance for Inspections Following a Design Basis Accident (DBA)

- Industry Interim Guidance (IG) (SGMP-IG-12-01) was issued on September 26, 2012 to change the prescriptive SG inspection requirements contained in the EPRI PWR SG Examination Guidelines, Revision 7, to performance based inspection requirements
- The need for IG for forced outages following DBA was identified following thorough Industry discussions following a seismic event that exceeded OBE and DBA criteria

Original Prescriptive Guidance in Examination Guidelines

- The original prescriptive guidance contained in Section 3.10 of the Rev 7 G/L was stated as follows:

  “Forced outage examinations shall be performed during plant shutdown subsequent to any of the following conditions:

  a. SG Primary-to-secondary leakage leading to plant shutdown

  b. Seismic occurrence greater than the Operating Basis Earthquake

  c. Loss-of-coolant accident requiring actuation of the engineered safeguards

  d. Main steam line or feedwater line break”

Interim Guidance

- The IG deleted Section 3.10 of the Rev 7 Exam GL entirely
  - Eliminating the prescriptive inspection requirements following DBA
- Modified several sections of the SG Integrity Assessment Guidelines (IAGL) to address performance based inspections and evaluations following DBA:

Interim Guidance

"Prior to a refueling outage that does not have planned SG primary-side and/or secondary-side activities, or when the steam generators have experienced conditions that may not be bounded by assumptions in the Operational Assessment, the information used in projecting steam generator integrity in the OA shall be reviewed. There may have been subsequent plant or industry experience that impacts the information used in the tube integrity assessment process that could impact the planned inspection interval or ability to have an outage without SG inspections. Following design basis accidents that result in a forced outage, the review should consider the need for additional analysis, evaluations, and/or inspections to demonstrate acceptable condition monitoring and operational assessment prior to start-up."

Interim Guidance

"To validate the inspection interval, a review of the DA and OA information used in evaluating tube integrity shall be performed prior to each refueling outage when SG primary and/or secondary-side activities are not scheduled or when the plant has experienced a forced outage following a design basis accident. The review should consider the need for additional analysis, evaluations, and/or inspections to demonstrate acceptable condition monitoring and operational assessment prior to start-up."

Interim Guidance

“Degradation Assessments shall include the secondary side integrity assessment either by directly incorporating the results into the DA or by reference. Refueling outages that do not include secondary side activities (examples: sludge lancing, FOSAR, <u>secondary side integrity inspections) or forced outages following design basis accidents</u> shall be evaluated per section 11.2.4."

Interim Guidance

"A review of the SG integrity assessment documents that justify the planned inspection interval shall be performed prior to each refueling outage when SG primary and/or secondary-side activities are not scheduled, <u>or when the steam generators have experienced conditions that may not be bounded by assumptions in the Operational Assessment.</u> The review <u>should consider</u> industry operational experience, chemistry excursions, plant operating transients, <u>and design basis accidents</u> since the last inspection and/or review. This review should be performed in a timely manner in order to minimize outage impact (scope/budget/planning - critical path), should a change in the planned inspection interval be identified from the review. <u>For reviews following design basis accidents that result in a forced outage, the review should consider the need for additional analysis, evaluations, and/or inspections to demonstrate acceptable condition monitoring and operational assessment prior to start-up".</u>

Basis for Interim Guidance

- Basis for performance based IG for DBA inspections and evaluations:
  - Original prescriptive guidance did not specify the inspection scope, techniques, or criteria and did not tie inspections to SG integrity requirements
  - IG recognizes that each DBA is unique and warrants specific evaluation of the event to determine the appropriate analyses and inspections to demonstrate and ensure SG integrity prior to start-up from the DBA shutdown
  - With the uniqueness of each DBA development of specific prescriptive inspection and evaluation requirements would not be feasible to capture and address every scenario. This may lead licensees to miss key aspects of inspection and SG integrity by relying on prescriptive requirements that may not bound their unique event.

Other Guidelines for Plant Earthquake Response

- EPRI Technical Report NP 6695, "Guidelines for Nuclear Plant Response to an Earthquake." 1989
- Two NRC Regulatory Guides based on NP 6695
  - Regulatory Guides 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions." 1997
  - Regulatory Guides 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event." 1997
- EPRI Report 1025288, Guidelines for Nuclear Plant Response to Earthquake," October 2012 updated previous guidance
  - Included lessons learned from recent seismic events
- IAGL revision committee will incorporate interim guidance and consider new technical report in the revision process

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TSTF 510 Update

Recent Industry Survey Results

- Information received from 60 of 69 units

	TSTF 510 LAR Approved	Submitted LAR (waiting on approval)	Plan to Submit 2013	Plan to Submit 2014	Plan to Submit later	No Current Plan to Submit
Number of Units	**13**	**13**	**18**	**9**	**3**	**4***

* 2 units being retired

* 2 units in extended outage

Standing Issues

SGMP Industry Document Status and Revision Schedule

Guideline Title	Current Rev #	Report #	Last Pub Date	Imple-mentation Date(s)	Interim Guidance	Review Date	Comments
SG Integrity Assessment Guidelines	3	1019038	Nov 2009	9/1/10	SGMP-IG-10-01 SGMP-IG-12-01	2012	Rev 4 in progress
EPRI SG In Situ Pressure Test Guidelines	4	1025132	Oct 2012	10/10/13	None	2015	
PWR SG Examination Guidelines	7	1013706	Oct 2007	9/1/08	SGMP-IG-08-04 SGMP-IG-12-01		Rev 8 in progress
PWR SG Primary-to-Secondary Leakage Guidelines	4	1022832	Sept. 2011	4/11/2012 7/11/2012	None	2014	

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SGMP Industry Document Status and Revision Schedule

Guideline Title	Current Rev #	Report #	Last Pub Date	Imple-mentation Date(s)	Interim Guidance	Review Date	Comments
PWR Primary Water Chemistry Guidelines	6	1014986	Dec 2007	6/17/08 9/17/08	SGMP-IG-09-01 SGMP-IG-11-02		Rev 7 in progress
PWR Secondary Water Chemistry Guidelines	7	1016555	Feb 2009	8/20/09 11/20/09	SGMP-IG-13-01	2013	
Steam Generator Management Program Administrative Procedures	3	1022343	Dec 2010	9/1/11 12/31/11	None	N/A	
Steam Generator Degradation Specific Flaw Handbook	1	1019037	Dec 2009	N/A	None	N/A	Rev 2 in progress

Interim Guidance SGMP-IG-13-01
PWR Secondary Water Chemistry Guidelines

- The Interim Guidance modifies existing "shall" requirements corresponding to when the hydrazine-to-oxygen ratio is less than two in the water feeding the steam generators

- This change resulted from a Review Board Inquiry that identified a gap in guidance for a "loss of feedwater hydrazine event" (i.e., hydrazine-to-oxygen ratio less than two in the water feeding the steam generators) that occurs between low power value (LPV) and mid power value (MPV) for recirculating steam generators (RSG).

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Interim Guidance SGMP-IG-13-01
PWR Secondary Water Chemistry Guidelines

- ***Expansion***: The existing loss of feedwater hydrazine requirement to "commence shutdown" is expanded to mean that the plant must "commence to cold shutdown (RCS ≤ 200°F)".
- ***New Guidance***: The existing guidance (with the above expansion) on "loss of feedwater hydrazine" for the plant at ≥MPV reactor power for RSGs (≥15% reactor power for OTSGs) is extended to all conditions when the plant is at RCS > 200°F by adding guidance .
- **Bases**:
  - Stress Corrosion Cracking (SCC) is thermally driven
  - Hydrazine-to-Oxygen Ratio < 2 causes a significant increase in Electrochemical Potential (ECP), which significantly increases the occurrence and rate of SCC of tubing alloys

NEI 03-08 Deviations

- Four long-term deviations
  - Two Steam Generator Examination Guidelines, R7
    - Single pass auto analysis
  - Steam Generator Secondary Water Chemistry Guidelines, R7
    - Wet lay up steam generator sample frequency
  - Steam Generator Integrity Assessment Guidelines, R3
    - Use of site-specific sizing indices
- Two short term deviations
  - Steam Generator Examination Guidelines, R7
    - PSI prior to hydro
  - Steam Generator Secondary Water Chemistry Guidelines, R7
    - Wet lay up steam generator sample frequency

ASME Code Guidance on Pre-Service Inspection

ASME Code Guidance on Pre-Service Inspection

- NRC Comment regarding Pre-service Examination of SG tubing:
  - IWB-2200(c) of Section XI of the ASME Code indicates that steam generator tube examination shall be governed by the plant Technical Specification.
- ASME Section XI Code action BC 10-129 wording: For Steam Generator tubing a full-length examination on 100% of the tubing shall be performed using methods and techniques that are expected to be employed during future inservice examinations.

ASME Code Guidance on Pre-Service Inspection

- Steam Generator Pre-Service Inspection code requirements are incorporated in an overall steam generator inspection action BC 10-129.
  - Meeting held between Code meetings to determine best action to address qualifications and flaws associated with the steam generator inspections.
  - Working Group PQSVEC will open a separate Code action to change these qualification requirements.
  - Item BC 10-129 is moving forward and will be presented to Subgroup Water Cooled Systems and Subgroup Evaluation and Standards for vote during February 11th Code week.
  - ASME Code Section III action being tracked by Subgroup Industry Experience for New Plants.

Operating Experience

Domestic Plant A – ODSCC in A600TT Tubing

- Event Summary: A domestic plant with A600TT tubing discovered three indications of axial ODSCC in one tube on the hot leg side during the Fall 2012 steam generator inspection. One flaw required in situ pressure testing and passed.
- Pertinent plant operating conditions and design
  - T_{hot} 611^{o}F
  - 21.27 EFPY
  - Stainless steel quatrefoil broach tube support plates
  - The affected tube was a high row tube that potentially contained high residual stress, i.e. -2-sigma tube.

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Domestic Plant A – ODSCC in A600TT Tubing

- Three Indications of axial ODSCC found in one tube high row – high stress tube (-2-Sigma tube)
  - TSP 03H (first TSP above flow distribution baffle)
  - TSP 05H (second TSP above flow distribution baffle)
  - Freespan between TSP 03H-05H (TSP 03H+33.7")

Indication	+PT Max Volts (V_{PP})	Axial Length (inches)	Max Depth (%TW)	Comment
TSP 03H	0.64v	0.58"	69.2%	Contained w/in TSP Land Contact
TSP 05H	0.56v	0.48"	50.0%	Contained w/in TSP Land Contact
Freespan TSP +33.7"	0.39v	0.19"	56.4%	Coincident with a 1.0v Ding

Appendix I Technique I28432 used for sizing

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Domestic Plant A – ODSCC in A600TT Tubing

- The three indications were confirmed by plus-point and the Ghent probes.
- The indications were not aligned axially, but were ~90 degrees apart.

Domestic Plant A – ODSCC in A600TT Tubing

- All indications were screened for In Situ Pressure Testing per Rev 3 of the EPRI SG In Situ Pressure Testing Guidelines
- Indications at TSP 05H and Freespan did not meet criteria for Proof and Leakage Testing
- Indication at TSP 03H met criteria to perform Proof and Leakage Testing
  - Localized pressure testing was performed up to pressures associated with SLB and 3xNOP differential pressure with intermediate pressure tests
  - Proof testing and leak testing passed with no leakage

Domestic Plant A – ODSCC in A600TT Tubing

- Condition Monitoring Assessment
  - TSP 03H: Met CM limits through In Situ Pressure Testing
  - TSP 05H and Freespan: Met CM limits through Flaw Handbook burst pressure requirements (Section 5.1.4)
    - Burst Effective (BE) Length/Depth Less than BE Structural Limits
    - Leakage criteria met – Max Depth Less than Max Depth Threshold for leakage (i.e., no Pop-Through at SLB conditions)
- Operational Assessment
  - All indications met full cycle OA limits through full bundle probabilistic analysis

Domestic Plant B – ODSCC in A600TT Tubing

- Event Summary: A domestic plant with A600TT tubing discovered four indications of axial ODSCC in two tubes during the Fall 2012 steam generator inspection. All indications were on the hot leg side. None of the flaws required in situ pressure testing.

- Pertinent plant operating conditions and design
  - T_{hot} 621°F
  - 18.95 EFPY
  - Stainless steel quatrefoil broach tube support plates
  - The affected tubes did not contain high residual stress

Domestic Plant B – ODSCC in A600TT Tubing

- Three axial indications found in one tube, all located between the flow distribution baffle (FDB) and the first quatrefoil support plate on the hot leg side
- One axial indication found in another tube within a hot leg support plate that contained two dents (11.35v and 8.91v)

Indication	+PT Max Volts (V_{PP})	Axial Length (inches)	Max Depth (%TW)	Comment
Flaw #1 (FS)	0.96	0.52	77%	Coincident with low level ding
FS Flaw #2 (FS)	0.24	0.15	45%	Coincident with low level ding
FS Flaw #3 (FS)	0.38	0.15	56%	Coincident with low level ding
Flaw #4 (TSP)	0.89	0.22	76%	End of Flaw coincident with the edge of a dent

Domestic Plant B – ODSCC in A600TT Tubing

- One of the freespan indications was initially detected by bobbin coil and the two smaller indications were found during the subsequent plus-point diagnostic inspection
  - The three indications were not in same axial plane

Domestic Plant B – ODSCC in A600TT Tubing

- The indication at the dented TSP was found during the dent plus-point inspection program.
  - The dent program scope was expanded upon detection of the axial indication
    - Expansion in SG C: 100% dings/dents >5 volts and 20% sample of >2 and <5 volts on HL plus 100% dings/dents >5 volts and 20% sample of >2 and <5 at TSP 8C.
    - No additional indications found
- All indications were screened for In Situ Pressure Testing per Rev 3 of the EPRI SG In Situ Pressure Testing Guidelines
  - None of the indications required in situ pressure testing

Domestic Plant B – ODSCC in A600TT Tubing

- Condition Monitoring Assessment
  - All indications met CM requirements
    - Burst Effective (BE) Length/Depth Less than BE Structural Limits
    - Leakage criteria met – Max Depth Less than Max Depth Threshold for leakage (i.e., no Pop-Through at SLB conditions)

- Operational Assessment
  - All indications met full cycle OA limits through full bundle probabilistic analysis

US Operating Experience – Alloy 600TT Tubing

- Majority of the 600TT fleet have H* approved
- Cracking continues to be identified in Alloy 600TT tubing

Location	ODSCC		PWSCC	
	Axial	Circ	Axial	Circ
U-Bend			X	
TSP/FDB	X			
TTS/Exp Trans	X	X	X	X
Tubesheet			X	X
Tube End			X	X
Ding*	X			

* Some indications at small voltage dings below reporting threshold

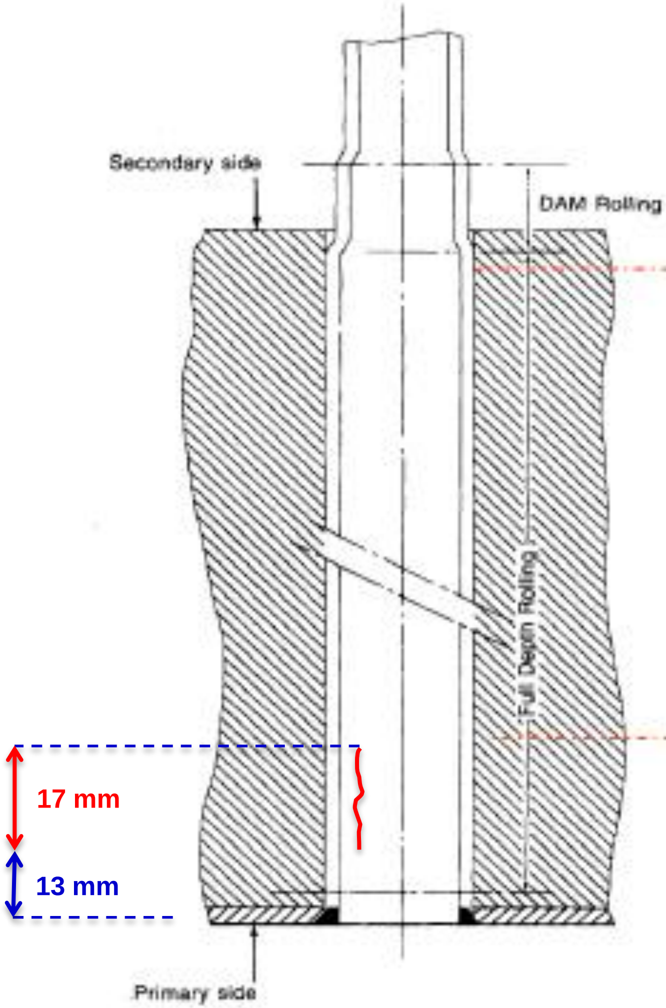
US 690TT Units Affected by Wear

MECHANISM	NUMBER OF DOMESTIC UNITS AFFECTED
Foreign Object Wear	18
U-Bend Support Wear	25
Support Structure Wear	35
Tube-to-Tube Wear	6

Data from Steam Generator Degradation Database as of January 2013

Foreign Plant Tubesheet Leakage Experience

- Tube leakage was detected in 2010 in a tube that was known to have a longitudinal crack within the tubesheet
  - 3-loop 900MWe reactor, with AREVA type 51BI steam generators
  - 7/8 inch diameter Alloy 600MA tubes (manufactured by Vallourec)
  - Tubes were mechanically expanded into the tubesheet by hard rolling and included a kiss roll at the top of the tubesheet
  - ~20 steps to achieve mechanical expansion on the full depth of the tubesheet
  - The leakage was identified in tube R19C43 with a Helium leak test
- Longitudinal crack = 17mm



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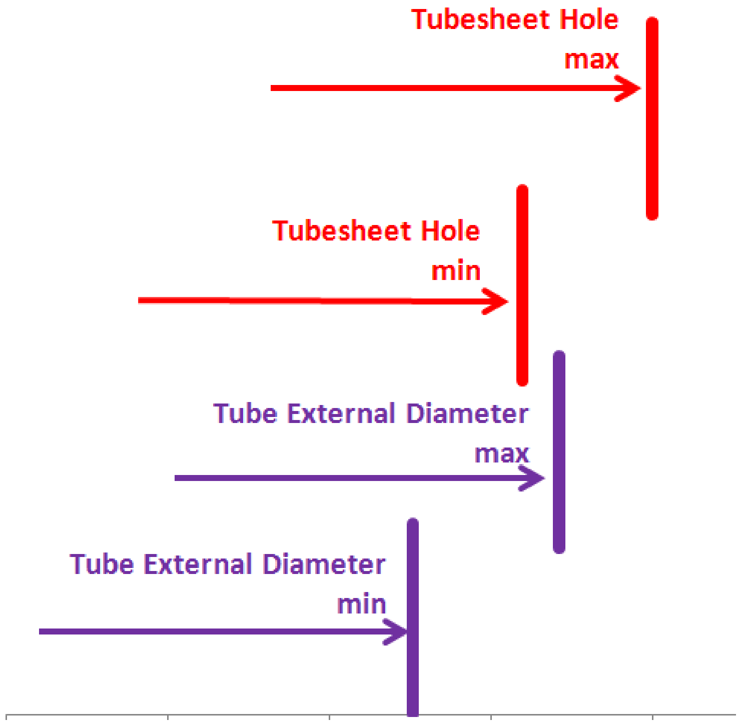
Foreign Plant Tubesheet Leakage Experience

- A root cause evaluation has shown that the tube was manufactured within the tolerances; however a stack up of tolerances could explain the leakage
- Verified that the indication was PWSCC



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Response to *NRC Comments from August 2012 Meeting*

Eddy Current Essential Variable Tolerance

Eddy Current Essential Variable Tolerance

- The EPRI Examination Technique Equivalency Project was planned as a multi-phase, multi-year project to provide a consistent and cost-effective methodology to allow utilities to determine equivalency of eddy current techniques that differed in one or more essential variables from EPRI-qualified techniques

- Three EPRI reports have been issued
  - *Development of a Process for Determining Examination Technique Equivalency*, (1015126), March 2008
  - *Development of Standardized Process for Determining Examination Technique Equivalency*, (1018557), March 2009
  - *Development of Documentation for Examination Technique Equivalencies*, (1020992), September 2010

Eddy Current Essential Variable Tolerance

- The first report documented a method for demonstrating equivalency between EPRI-qualified eddy current inspection techniques for SG tubes and similar inspection techniques with modified essential variables.
  - A master set of calibration tubes was designed.
- The second report documented a review of various EPRI-qualified techniques compared to field techniques with modified essential variables.
  - Tolerances for technique equivalency were established.
- The third report provided information for developing validation documentation for site inspection techniques with modified essential variables. This phase of the project further documented the effect of changes to essential variables on the resulting eddy current signal.

NRC Review of EPRI Examination Technique Equivalency Reports

- April 19, 2012 - All 3 reports were made available for NRC review at EPRI Charlotte office
- August 21, 2012 - NRC provided comments at SGTF meeting
- October 24, 2012 - EPRI met with NRC contractor to review comments

EPRI Plans Going Forward

- Request continued interaction with NRC NDE contractor to receive clarification of initial comments
- Propose a 2014 project to issue a fourth report that:
  - Addresses equivalency topics not addressed in previous reports
  - Considers comments received from utility members, utility contractors and NRC

Time Dependent Leak Rates

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Time Dependent Leak Rates

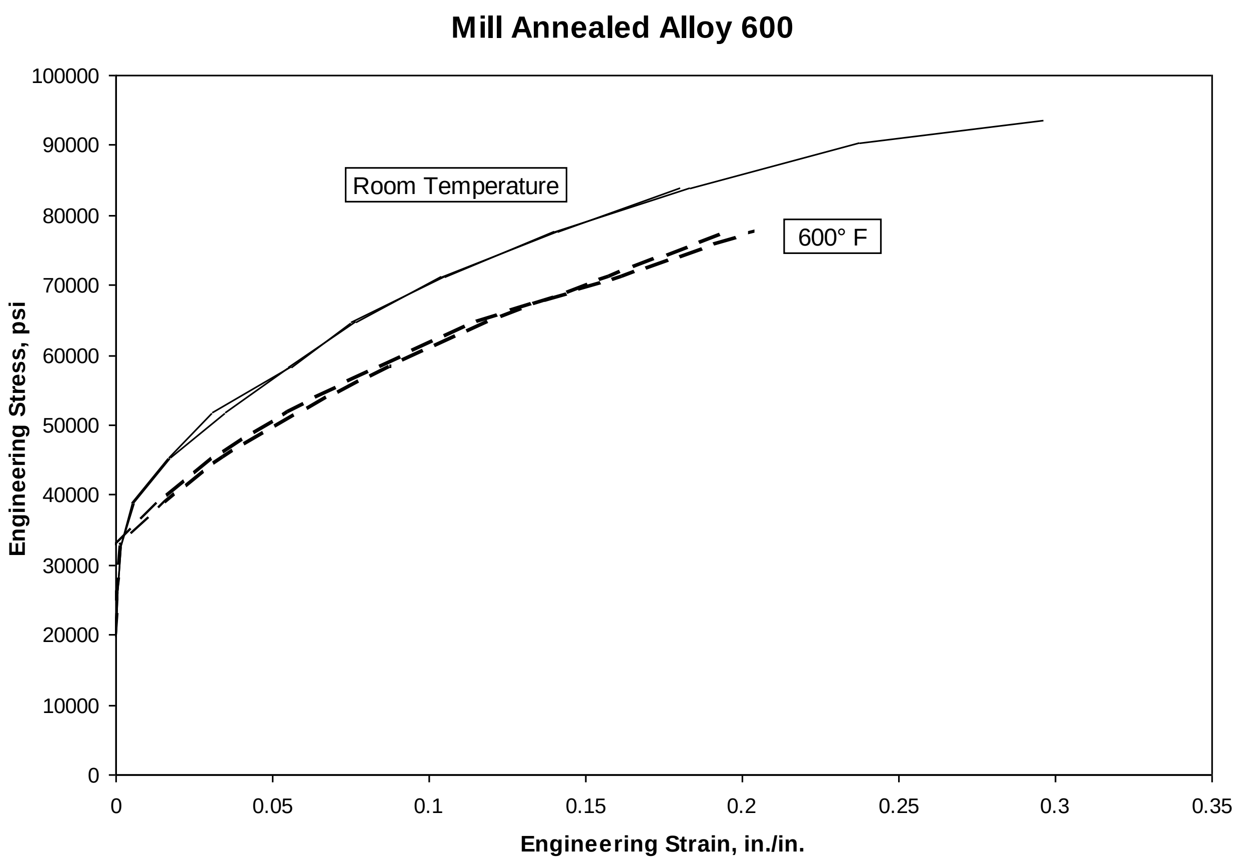
- EPRI Technical Report 1022831, “Onset of Fatigue Cracking in Steam Generator Tubes with Through Wall Flaws”, provided to NRC
  - NRC question: The conclusions in this report are based on room temperature tests. Would similar conclusions be made if the tests were at operating temperatures?
    - Increase in time dependent plasticity at higher temperatures because of higher creep rate?
    - Increase in jet-structure interaction at higher temperatures
- The following slides describe recent test results

Time Dependent Deformation of Alloy 600

- Time dependent deformation can lead to time dependent increases in leak rates of degraded tubing
- Previous testing indicated small levels of time dependent deformation in Alloy 600 at room temperature
- Deformation levels of practical interest were found to be essentially complete at 300 seconds
- Time dependent deformation is approximately equivalent to a 2% decrease in material flow strength
- No adjustments to in situ testing practice required given the conservative test procedure
- Testing recently performed at 600°F to evaluate the possibility of increased levels of time dependent deformation

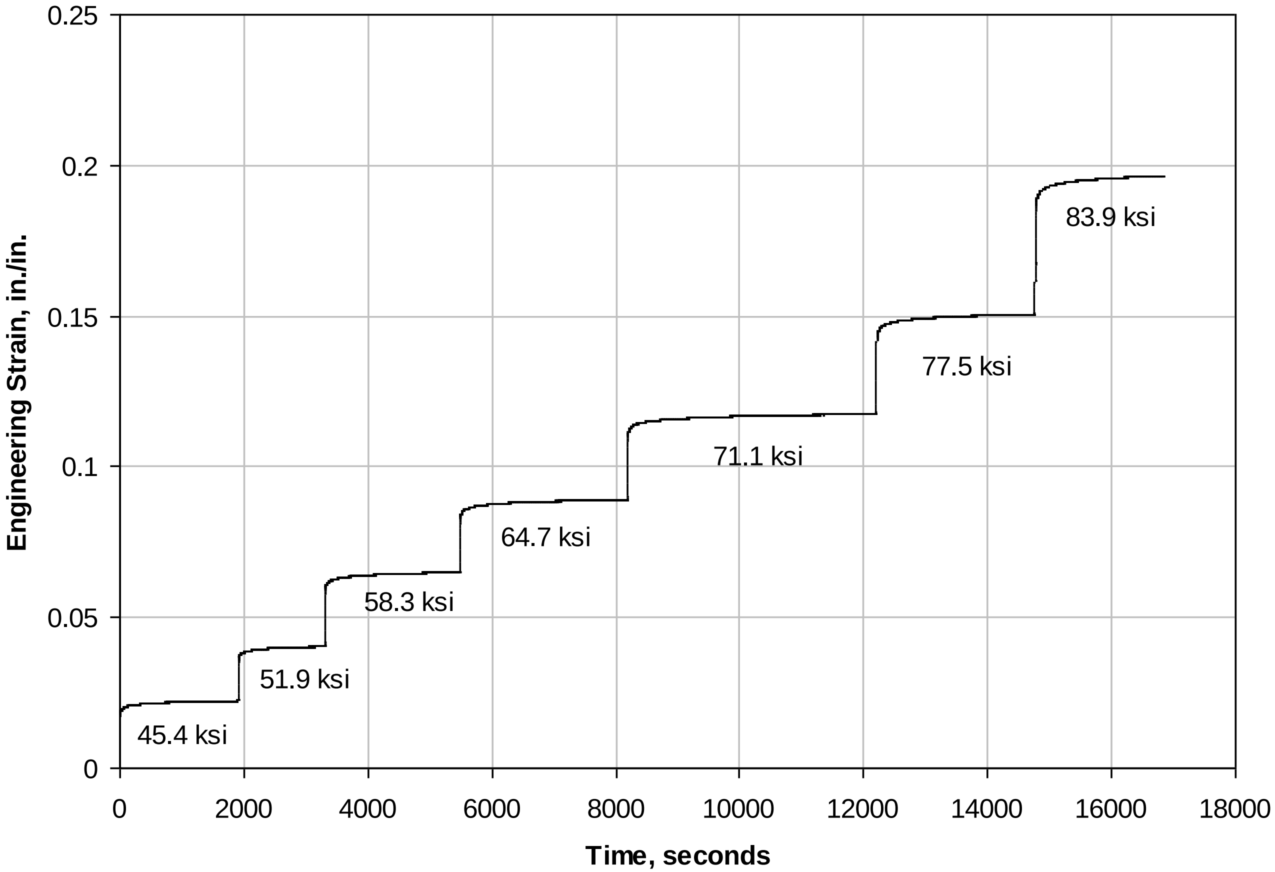
Stress Strain Curves for 600MA Material at Room Temperature and 600°F

- Typical Decrease in Strength is Observed



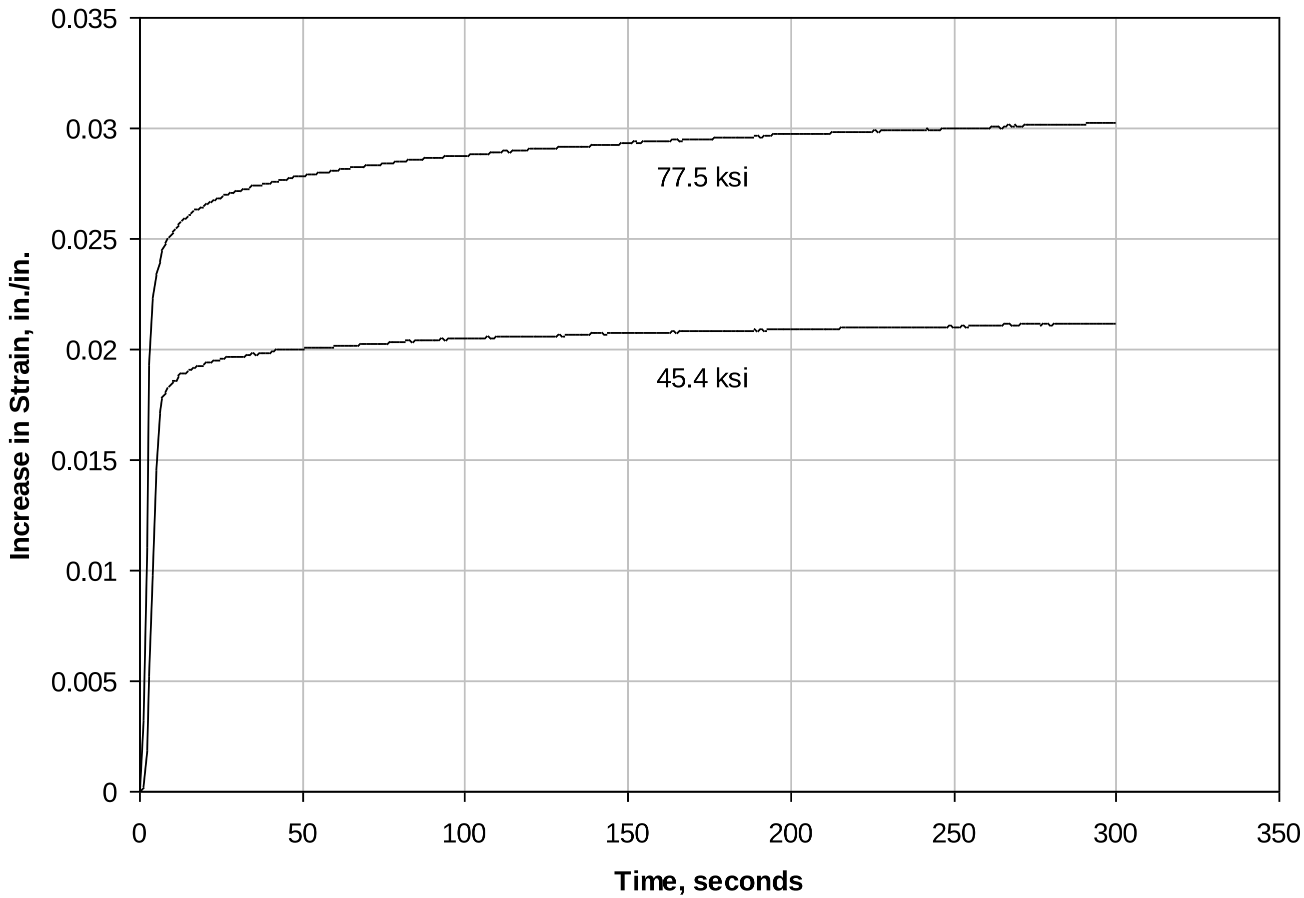
Step Load Test at Room Temperature

- Small Levels of Time Dependent Deformation are Observed



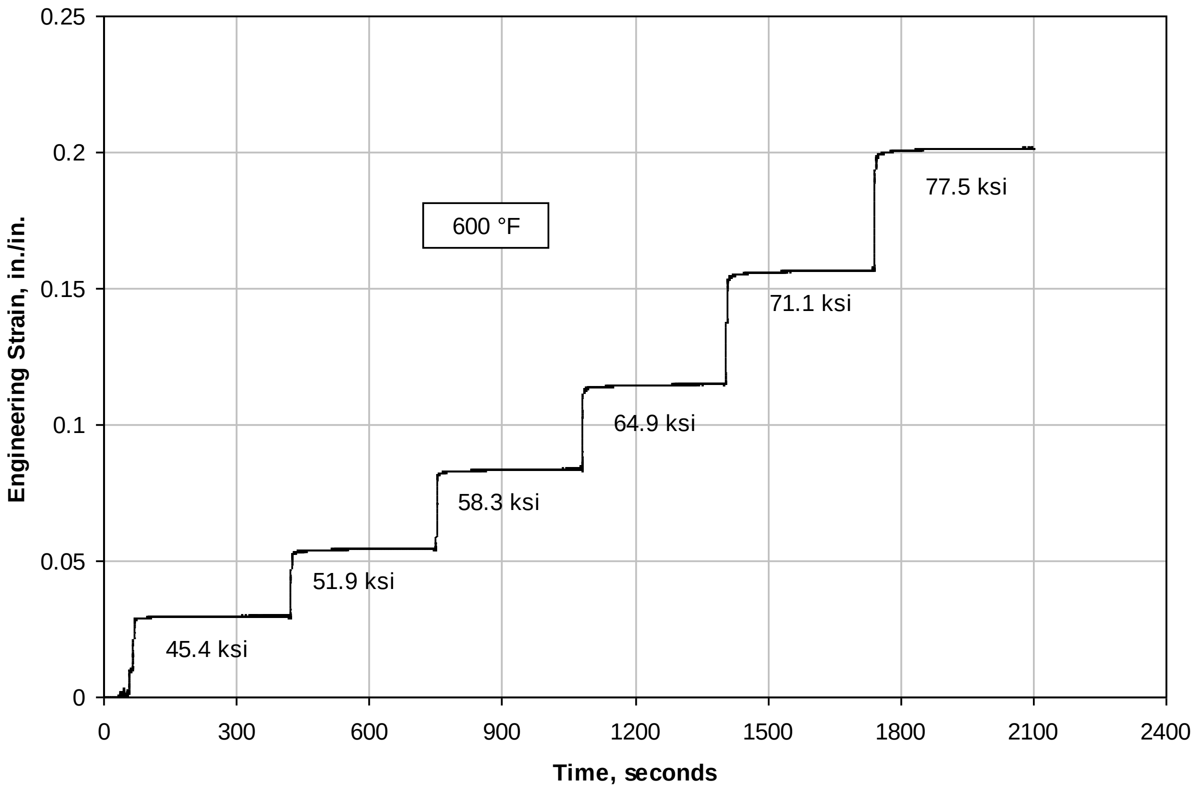
Magnified View of Time Dependent Deformation at Room Temperature

- Deformation is Essentially Complete at 300 Seconds



Step Load Test at 600°F

- Small Levels of Time Dependent Deformation are Observed



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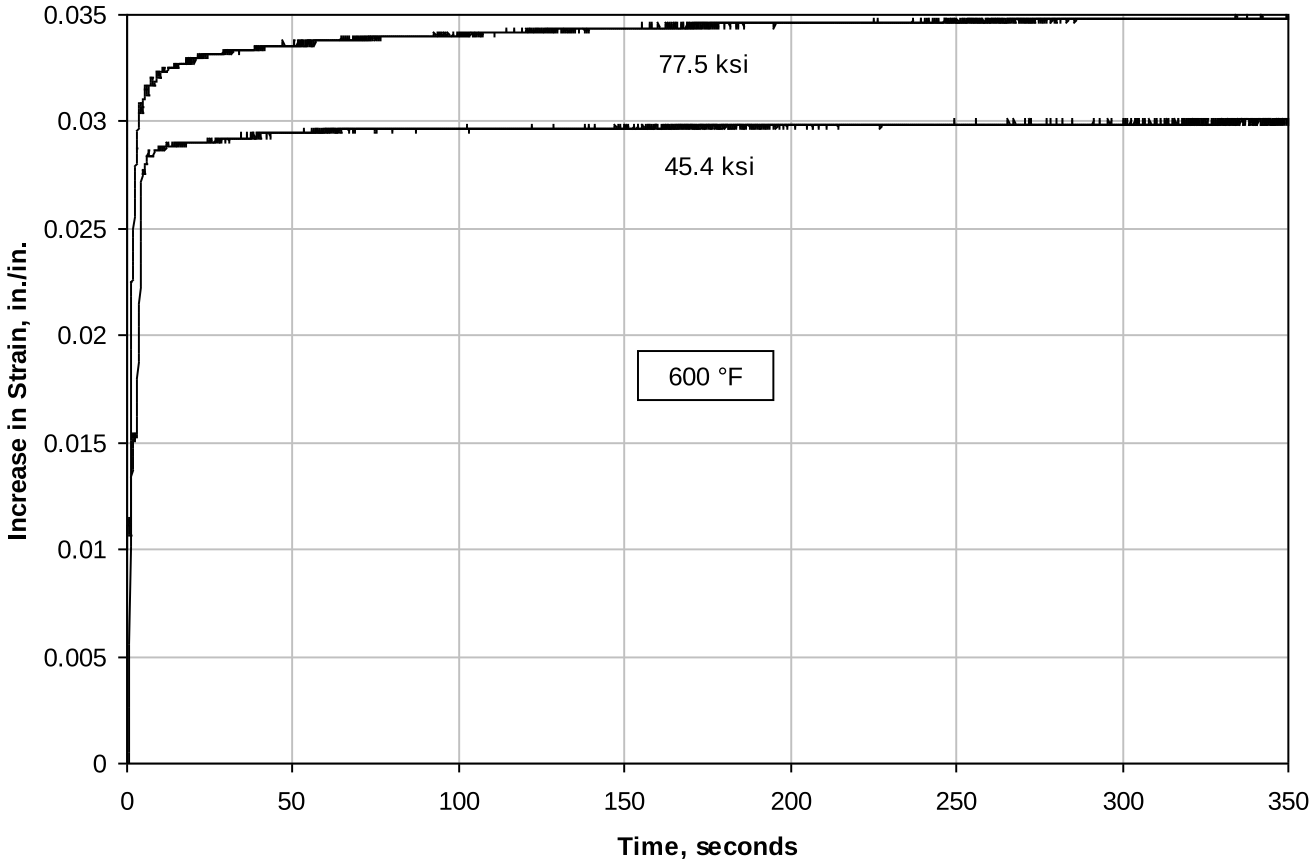
Magnified View of Time Dependent Deformation at 600°F

- Deformation is Essentially Complete at 300 Seconds



Conclusions

- Time Dependent Deformation at a Level of Practical Interest Relative to Time Dependent Leak Rates is the Same at 600 °F as It is at Room Temperature

- No adjustments to in situ test practice are required

- No adjustments to leak rate calculations are required

Industry Guidelines Pertaining to the Use of Nitrogen-16 Monitors

NRC Question

- Do guidelines address the use (or lack of use) of confirmatory measurements when a rapidly increasing leak rate is only detected on the faster response time monitor?

Primary to Secondary Leak Guidelines Requirements

- *Guidelines* allow qualitative confirmation to prevent unnecessary shutdowns
  – N-16 monitors installed on or near steam lines are very good at quickly detecting tube primary-to-secondary leakage, but may not provide an accurate leak rate value.
  – N-16 monitor indications may be an anomalous indication from an instrument issue or EMF interference, etc. which has been experienced in the industry
- *Guidelines* (Action Level 3) require plants without a second radiation monitor shut down based on the indication of one monitor
- *Guidelines* point out response time of radiation monitors
  – Plant personnel evaluate response times and reflect it in plant procedures and training programs so that responses occur within appropriate time frames

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NRC Discussions/Items of Interest

Together…Shaping the Future of Electricity