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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Secretary
ATTN: Rulemakings and Adjudications Staff
Nuclear Regulatory Commission
Washington, DC 20555-0001

Charles A Tomes
2045 Fawn Lane
Green Bay, WI 54304

Subject: RIN 3150 – AH76, Response to NRC Request for Public Comment to Incorporate ASME Code Case N-729 Revision 1 With Supplemental Requirements into the Code of Federal Regulation

Dear Sir:

References:

1. PWSCC Lifetime Evaluation on Alloy 690, 52, and 152 for PWR Materials, MHI, EPRI PWSCC of Alloy 600 2007 International Conference & Exhibition, June 11 – 14, 2007, Atlanta, GA
2. Crack Growth Response in Simulated PWR Water of Alloy 152 Weld Metal, PNNL, ICG-EAC, Hualien, Taiwan, April 2007
3. PWSCC Growth Rates in Alloy 690 and Its Weld Metal, GE Global Research Center, EPRI PWSCC of Alloy 600 2007 International Conference & Exhibition, June 11 – 14, 2007, Atlanta, GA

The author of this letter wishes to thank the Nuclear Regulatory Commission (NRC) for an opportunity to provide comments on NRC's plans to incorporate ASME Code Case N-729 Revision 1 as amended by NRC supplemental requirements into the Code of Federal Regulation for nondestructive testing of replacement reactor vessel head control rod drive mechanism (CRDM) tubing and j-groove weld metal.

The comments provided herein are based on planning, research, and replacement reactor vessel head activities spanning back to the early 1990's. Following the initial reports of cracking at Bugey Unit 3, the commercial nuclear power industry initiated research projects to develop alternate materials that are highly resistant to PWSCC. Coincident with incidents of CRDM j-groove weld cracking in the USA, utilities initiated plans to replace reactor vessel heads with materials that are highly resistant to primary water stress corrosion cracking (PWSCC).

While employed at the Nuclear Management Company I was involved with development of contracts and oversight activities to fabricate and install replacement reactor vessel heads at five (5) nuclear plants: Kewaunee Power Station, Prairie Island Nuclear Generating Station Units 1 and 2, and Point Beach Nuclear Plant Units 1 and 2. As part of this project, Mitsubishi Heavy Industries fabricated the five (5) replacement reactor vessel heads under contract to Westinghouse Electric Company. To date, all five (5) reactor vessel heads have been replaced with CRDM tubing and j-groove weld metal

Template = SECY-067

SECY-02

fabricated from Alloy 690, 52, and 152 materials. Since completion of this project my employment status has changed from Nuclear Management Company to Dominion Energy Kewaunee, Inc as Dominion purchased the Kewaunee Power Station in Summer 2005. Information provided herein is applicable to Kewaunee Power Station, Prairie Island Nuclear Generating Station 1 and 2, and Point Beach Nuclear Plant Unit 1 and 2.

A primary goal of the project was to build quality into the replacement reactor vessel heads to prevent leakage and reduce the need for detailed inspections during future plant operation. The following enhancements were included in to reduce the likelihood of PSWCC, leakage, and problems encountered during future inspections:

1. Alloy 690, 52, and 152 was used for fabrication of the CRDM tubing and j-groove welds,
2. The grain size from 5 to 7 was selected to optimize PWSCC resistance and also ensure ultrasonic examination,
3. Narrow groove j-groove welds were used to reduce the residual stress,
4. During j-groove welding the ID surface of the Alloy 690 tubing was cooled with water to minimize stresses,
5. The threaded joint on the CRDM latch mechanism was replaced with a butt weld to eliminate the possibility of leakage,
6. Vents on top of the CRDM rod housings were eliminated to reduce the possibility of leakage,
7. The Marmon clamps on the thermocouple ports were replaced with leak free CETNA to reduce the possibility of leakage,
8. Removable insulation with inspection ports was installed,
9. No repairs were permitted on the Alloy 690 tubing,
10. Penetrant testing was performed at pre-defined increments during welding of the j-groove weld,
11. A "PT White" criteria was used on the final surface of the j-groove weld and alloy 690 tubing,
12. The surface of the j-groove welds were polished smooth to permit eddy current testing and penetrant testing,
13. The distance between the thermal sleeve and top of the funnel was increased to better accommodate inspection probes during future inservice inspections,
14. Preservice Inspection (PSI) included bare metal visual (BMV) inspections of the reactor vessel head, eddy current testing of the j-groove weld and alloy 690 tubing above and below the j-groove weld, and ultrasonic inspection of the alloy 690 tubing.

Upon completion of the replacement reactor vessel head projects for the Nuclear Management Company, I am pleased to communicate that the goals and objectives to improve PWSCC resistance, reduce the possibility of leakage above the reactor vessel head, and reduce potential problems with future inspections are successful.

The decision to award a contract to Mitsubishi Heavy Industries was heavily influenced by the knowledge that they had conducted extensive PWSCC crack initiation testing under accelerated PWR water conditions. Up to presently, most of this information was considered proprietary and had not been released to the public.

As part of the replacement reactor vessel head project Nuclear Management Company contracted Mitsubishi Heavy Industries to fabricate eight (8) linear feet of alloy 52 and alloy 152 weld metal to be used for future PWSCC testing. Nuclear Management Company donated this material to the Electric Power Research Institute in order for it to be included in various industry PWSCC testing programs. To date, some of this weld metal has been tested under NRC contract by Pacific Northwest National Laboratory and also by GE Global Research.

References 1 – 3 document PWSCC laboratory test results (applicable to replacement reactor vessel heads at Kewaunee Power Station, Prairie Island Nuclear Generating Station Units 1 and 2, and Point Beach Nuclear Plant Units 1 and 2) from Mitsubishi Heavy Industries, Pacific Northwest National Laboratory, and GE Global Research that have been recently released to the public.

A copy of the presentation made by Mitsubishi Heavy Industries at the EPRI 2007 International PWSCC of Alloy 600 Conference and Exhibit Show, Atlanta, GA, June 11 – 14, 2007 is included in Attachment 1. The Mitsubishi Heavy Industries PWSCC test results show that cracking has not initiated for Alloy 690, 52, and 152 materials in a simulated PWR environment for approximately 73,000 hrs, 84,000 hrs, and 85,000 hours, respectively. All testing was performed at 360 C (680 F). Testing to date confirms no crack initiation has occurred in Alloy 690, 52, and 152 materials. Other materials including Alloy 600, 82, or 182 were included in the test matrix and showed evidence of crack initiation early on during testing consistent with industry experience. One factor for understanding the quality of Alloy 690, 52, and 152 is to make adjustments for testing performed at 680 F to operating temperature. When this adjustment is made, a factor of 6.4 applies for base metal and a factor of 14.9 applies for weld metal. These factors can be multiplied directly to the test duration to adjust for differences in temperature. From this data, the equivalent time without initiation for Alloy 690 base metal is approximately 467,200 hours. Similarly, the equivalent time without crack initiation for weld metal is approximately from 1,251,600 to 1,266,500 hours for Alloy 52 and Alloy 152, respectively. This equivalent time period for base metal and weld metal are 53 years and 142 to 144 years, respectively. It is my understanding that these quality improvements apply to the entire replacement reactor vessel head population supplied by Mitsubishi Heavy Industries to the USA market from 2003 through 2012.

The PWSCC crack growth rates for the Alloy 690, 52, and 152 materials fabricated by Mitsubishi Heavy Industries, donated to the Electrical Power Research Institute by the Nuclear Management Company, and independently tested by Pacific Northwest National Laboratory and GE Global Research are on the order of 10^{-9} mm/s, which are of no engineering consequence. A copy of presentations recently made by Pacific Northwest National Laboratory and GE Global Research is included in Attachment 2 and 3, respectively.

It is with this understanding, after several enhancements and extensive verification that the Alloy 690, 52, and 152 materials are highly resistant to PWSCC that the following comments are made:

1. The NRC requirement to perform both eddy current testing and ultrasonic testing on the wetted surface of the alloy 690 tubing and j-groove weld is too stringent. This requirement being imposed by NRC (and not endorsed by ASME Code) will nearly double the time duration required to conduct the examinations, from 7 days to as much as 14 days, thus

increasing cost and radiation exposure to employees and vendors. Radiation levels are typically on the order of 5 R/hr under the reactor vessel head.

2. The leak path method has provided accurate supplemental information as confirmation of leakage for when assessment is needed of other indications such as eddy current signals and evidence of boric acid crystals observed during visual examinations. The leak path method is considered to be reliable for reactor vessel heads with interference fits such as those fabricated by Combustion Engineering and Mitsubishi Heavy Industries. The supplemental requirements imposed by NRC to perform both eddy current testing and ultrasonic testing is too conservative and may be misdirected as some crack patterns may not be detectable by ultrasonic techniques and must be confirmed by combination of eddy current testing and leak path assessment.
3. The requirement to perform the first NDE examination for replacement reactor vessel heads with Alloy 690, 52, and 152 materials after 10 years is too stringent. The crack initiation and crack growth data discussed herein and attached to this letter verify that the materials are highly resistant to PWSCC and expected to perform inservice for excess of 53 to 142 years without experiencing PWSCC initiation. Alloy 690 steam generator tubing has been inservice in the PWR industry for over 18 years without incident of PWSCC.
4. The requirement to perform successive NDE examination for replacement reactor vessel heads with Alloy 690, 52, and 152 materials after 7 years is too stringent. The crack initiation and crack growth data discussed herein and presented at the 2007 International PWSCC of Alloy 600 Conference in Atlantic, Georgia on June 11 - 14, 2007 verify that the materials are highly resistant to PWSCC and expected to perform inservice for excess of 50 to 140 years without experiencing PWSCC initiation. Alloy 690 steam generator tubing has been inservice in the PWR industry for over 18 years without incident of PWSCC.
5. Utilities performed economic analysis to justify replacement of reactor vessel heads based upon using materials that are highly resistant to PWSCC to reduce or eliminate the need to perform unnecessary NDE underhead examinations on the Alloy 690, 52, 152 tubing and j-groove weld materials. The economic analysis typically includes consideration of radiation exposure to employees and vendors. Adoption of these aggressive NDE testing requirements by NRC will result in unnecessary radiation exposure to nuclear employees and vendors.
6. USA utilities who purchased replacement reactor vessel heads from Mitsubishi Heavy Industries understand that some level of field verification may be needed or desired, by NRC, in the future to confirm the laboratory test results observed by Mitsubishi Heavy Industries, Pacific National Laboratory, and GE Global Research (discussed herein). To this end it may be desirable for USA utilities who purchased replacement reactor vessel heads from Mitsubishi Heavy Industries to propose an integrated replacement underhead NDE inspection program at a frequency of 5 years starting 10 years after installation of the first replacement reactor vessel head as opposed to the continued inservice inspections at predefined durations specified in Code Case 729 Rev 1. Mitsubishi Heavy Industries is in the process of investigating the formation of a USA industry group of Owners that recently purchased replacement reactor vessel heads from Mitsubishi Heavy Industries to formally propose alternative inspection requirements based upon research data discussed herein should the NRC endorse ASME Code Case N729 Rev 1 (along with the cited NRC supplemental requirements).
7. It is recommended that if the NRC endorses requirements of ASME Code Case N729 Revision 1 through adoption into the Code of Federal Regulation it be limited to reactor vessel heads with CRDM tubing and j-groove welds fabricated from Alloy 600, 82, and 182 materials. This

approach will give industry adequate time to formulate and to agree to appropriate NDE requirements for the replaced reactor vessel heads fabricated from Alloy 690, 52, and 152 tubing and j-groove welds with NRC and ASME. The USA commercial nuclear power industry has replaced all of the reactor vessel heads classified as highly susceptibility to date so adequate time exists to reach an agreement with industry and ASME Code.

USA utilities that recently purchased replacement reactor vessel heads from Mitsubishi Heavy Industries include:

- Dominion Generation Kewaunee Power Station
- Dominion Generation Surry Unit 2
- Dominion Generation Millstone Unit 2
- Southern Nuclear Company, Farley Units 1 and 2
- Progress Energy, HB Robinson
- Omaha Public Power District, Fort Calhoun
- Southern California Edison, SONGS Units 2 and 3
- South Texas Project Units 1 and 2
- Nuclear Management Company, Prairie Island Units 1 and 2
- Nuclear Management Company, Point Beach Units 1 and 2

Thank you for considering these comments. Questions regarding the nature of this information may be directed to Mr. Charles Tomes of Dominion Energy Kewaunee, Inc at 920-388-8192 and Mr. Joseph Hutter, Vice Mitsubishi Nuclear Energy Systems, Inc. at 412-374-7395.

Sincerely,

Charles A. Tomes

Charles A Tomes

Attachments

Cc

Leslie Hartz, Vice President – Site Vice President Kewaunee Power Station, Dominion
Jerry Bischof, Vice President Nuclear Engineering, Dominion
Dennis Koehl, Site Vice President Point Beach Nuclear Plant, Nuclear Management Company
Mike Wadely, Site Vice President Prairie Island Nuclear Generating Station, Nuclear Management Company
Joseph E Hutter, Vice President Mitsubishi Nuclear Energy Systems, Inc

Attachment 1

PWSCC Lifetime Evaluation on Alloy 690, 52, and 152 for PWR Materials

Presented by Mitsubishi Heavy Industries

EPRI PWSCC of Alloy 600 2007 International Conference & Exhibition

June 11 – 14, 2007

Atlanta, GA

PWSCC Life Time Evaluation on Alloy 690, 52 and 152 for PWR Materials

EPRI
PWSCC of Alloy 600 2007 International Conference & Exhibition
June 11-14, 2007
Renaissance Waverly Hotel
Atlanta, GA

Seiji Asada, Akira Konishi, Koji Fujimoto
Mitsubishi Heavy Industries Ltd.

Shinro Hirano, Hajime Ito
The Kansai Electric Power Co., INC.

INTRODUCTION

- PWSCCs in Alloys 600, 82 and 182 for Reactor Vessel Head Penetration material and its weld metals were reported in Bugey-3 and other PWR plants.

- In order to evaluate the PWSCC integrity of Alloys 690, 52, and 152 for RV base material and its weld metals, the authors have started uni-axial constant load stress corrosion cracking tests at 360°C in simulated PWR primary water as a Joint Research Program between the Japanese PWR utilities and Mitsubishi Heavy Industries, Ltd. (MHI)

CONTENTS

Experience of PWSCC on SG

ECT indications were found in a large numbers of Steam Generator (SG) Tubes and the root cause was PWSCC

Joint Development Programs on SG Tube Material Data
[Japanese PWR utilities and MHI]

Lessons Learned

- Choice of Material
  - 690 > 600
  - TT > MA

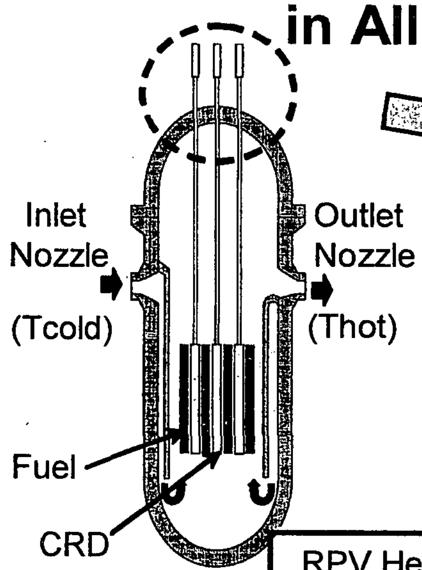
TT690

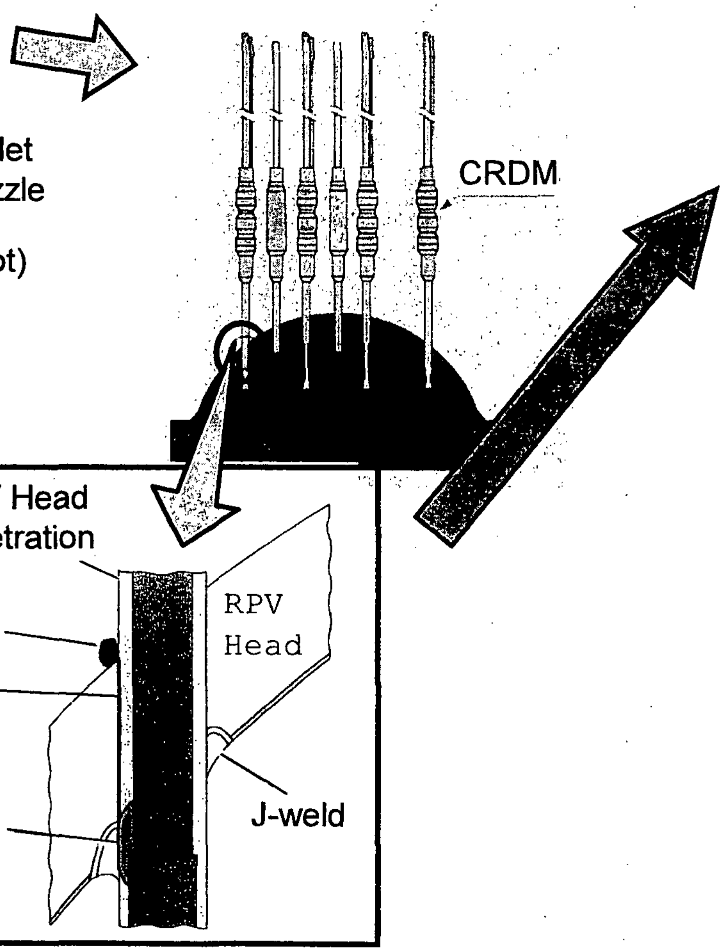
- Establishment of an experimental method on PWSCC for Alloy 600/690

Experience of PWSCC on RPV Head Penetration

Sept. 1991 : Bugey-3 in France

First through wall crack in Alloy 600 RPV head penetration





Lessons Learned

Root cause was PWSCC of Alloy 600 head penetration

↓

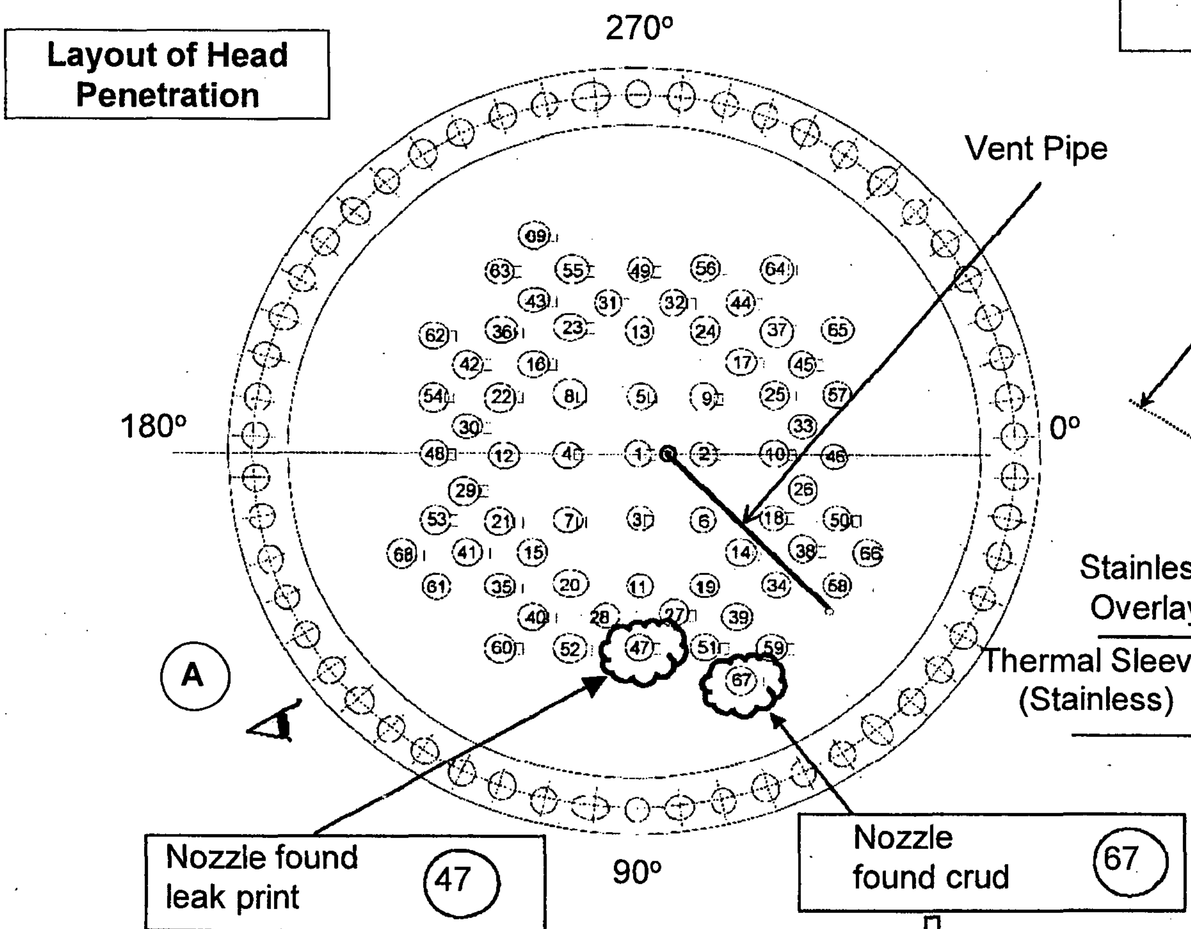
Joint Development Programs on RV Materials

↑

Experimental Method Developed by the Joint Development Programs on SG Tube

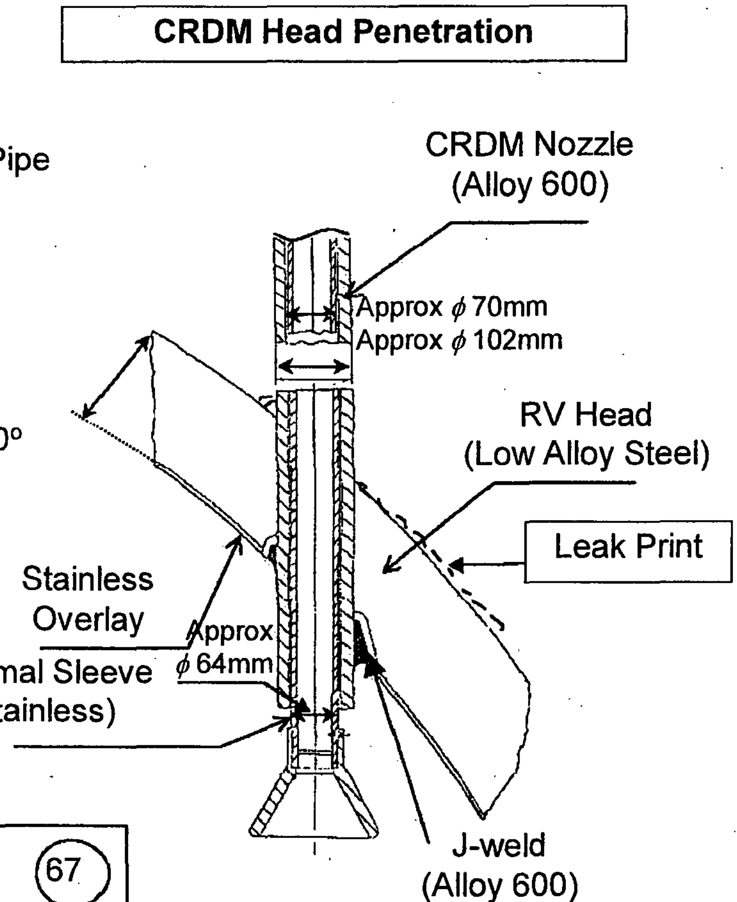
Experience of PWSCC on RPV Head Penetration

May 2004 : Ohi-3 in Japan



↳ **Investigation**

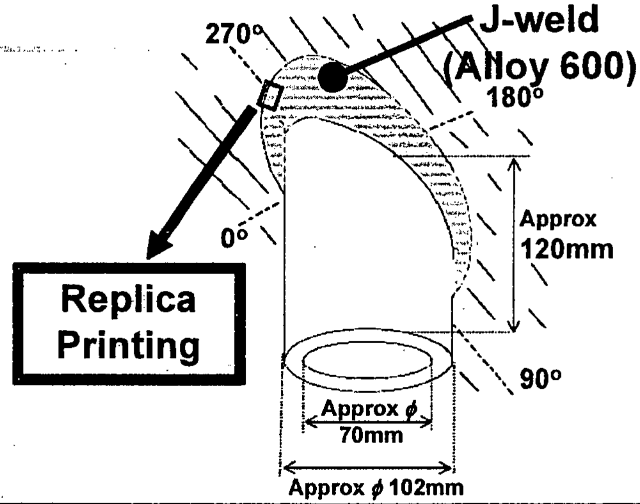
↳ This seems to be a remaining leak print of the leakage from the T/C nozzle seal portion during test operation (1991).



Experience of PWSCC on RPV Head Penetration

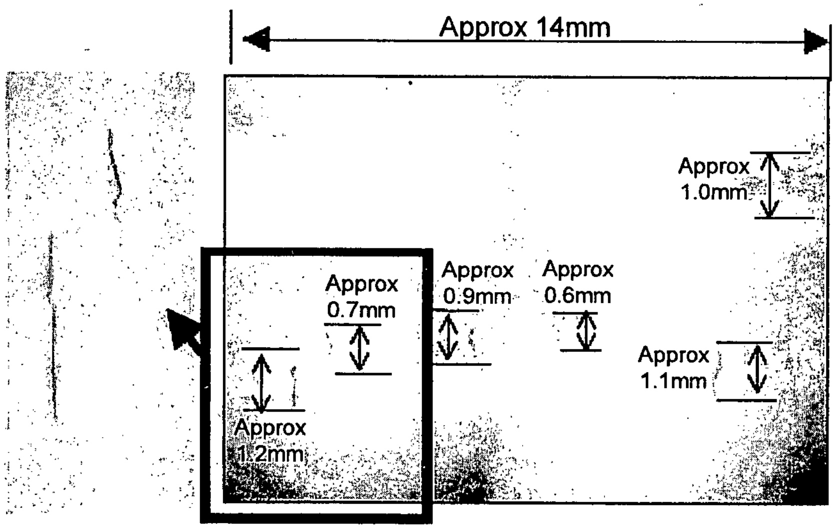
NDE Result

- Location: RV 260° to 280° in the J-weld
- Length : Max. 5mm



(1) 1st Grinding (0.5mm) : Surface

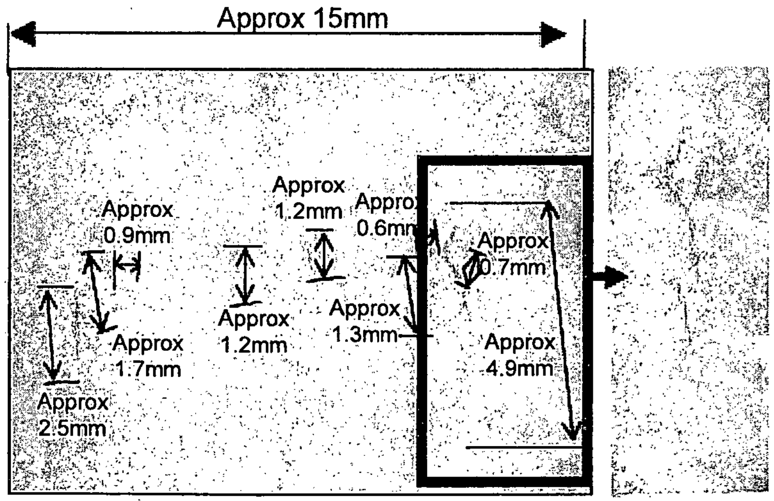
- Linear-like cracks along the dendrite



(2) 3rd Grinding (total 3mm) : Inside

- Branched along the dendrite

- Length becomes longer further inside.

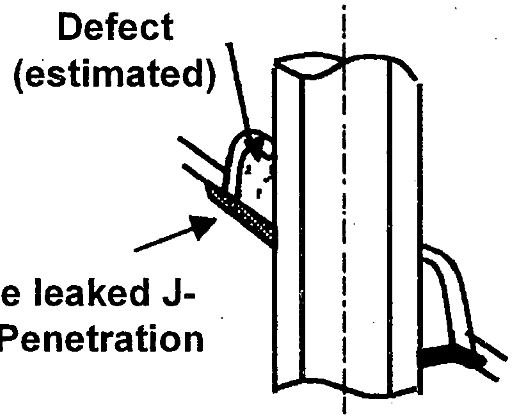


Experience of PWSCC on RPV Head Penetration

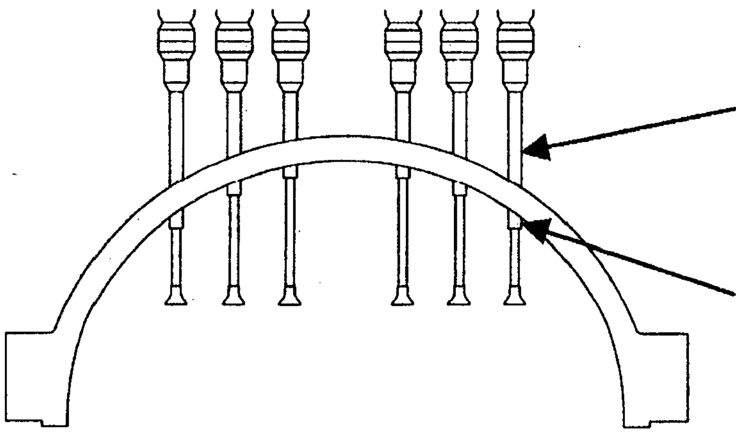
■ Counter Measures : Repair & Replacement

Step 1: Weld repair for #47 J-weld

In order to maintain the integrity of RCS pressure boundary and avoid PWSCC propagation, weld repair was performed by using Alloy 690.



Step 2: RV Head Replacement



Change of material of nozzles
Alloy 600 → Alloy 690
(Improvement of resistance for PWSCC)

Change of material of J-welds
Alloy 600 → Alloy 690
(Improvement of resistance for PWSCC)

Experience of PWSCC on RPV Head Penetration

- It is well-known that Alloy 690 has high resistance against PWSCC and we use Alloy 690 as the counter measure to PWSCC.
- We should obtain PWSCC initiation data for Alloy 690 materials to verify high reliability of Alloy 690.
- The Japanese PWR Utilities and MHI are conducting constant load PWSCC tests for Alloy 690 materials.

Chemical Compositions of Test Materials (Alloys MA600, TT690, 152 & 52)

Alloys		Chemical Composition (mass%)								
		C	Si	Mn	P	S	Ni	Cr	Fe	Cu
MA600	SG Tube (Reference)	0.027	0.35	0.30	0.008	0.001	74.50	15.90	8.51	0.02
TT690	RVH Pene.	0.020	0.35	0.32	0.010	0.001	60.10	30.10	8.65	0.01
	BMI Nozzle	0.021	0.32	0.28	0.008	0.001	60.15	29.70	9.00	0.03

Weld Metals		Chemical Composition (mass%)										
		C	Si	Mn	P	S	Ni	Cr	Cu	Ti	Nb	Al
Alloy152	Weld Joint (SMAW)	0.030	0.46	3.37	0.007	0.007	55.9	28.93	<0.01	0.12	1.62	0.16
Alloy 52	Weld Joint (GTAW)	0.030	0.17	0.24	0.005	<0.001	60.41	28.95	<0.01	0.56	0.01	0.63

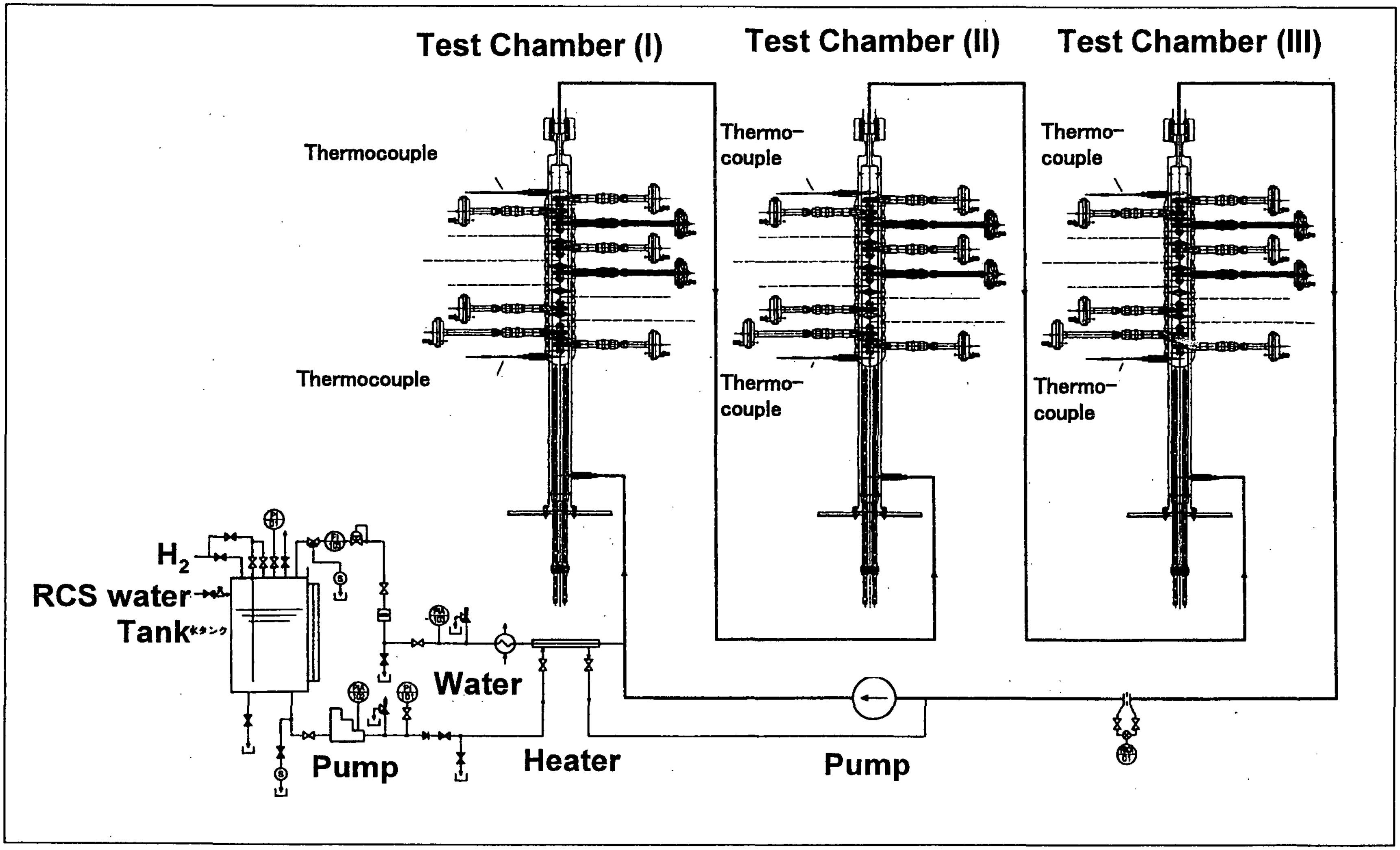
Heat Treatment Condition and Mechanical Properties of Test Materials (Alloys MA600, TT690)

Alloys		Heat Treatment	Mechanical Properties (R.T.)			Grain Size
			Yield Strength	Tensile Strength	Elongation	
		MA	(MPa)	(MPa)	(%)	
MA600	SG Tube (Reference)	975℃	346	680	42	—
TT690	RVH Pene.	1075℃+TT	286	650	50	4.0
	BMI Nozzle	1075℃+TT	284	661	51	5.9

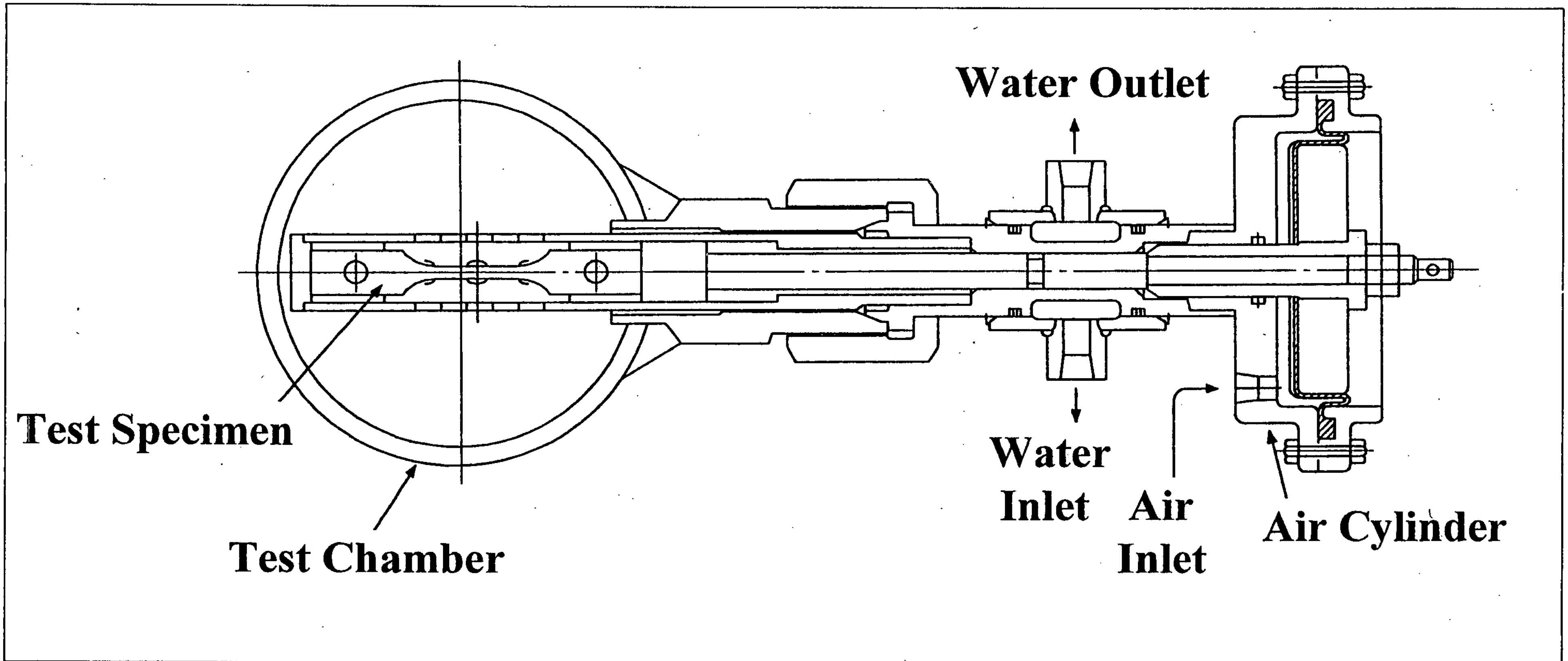
Water Chemistry of Simulated PWR Primary Water (MOC)

Items	Test Conditions
pH (at 25°C)	6～8
Conductivity (μS/cm at 25°C)	5～30
H_3BO_3 (ppm as B)	400～600
LiOH (ppm as Li at 25°C)	0.2～2.2
Dissolved Hydrogen (cc·STP/kg·H_2O)	25～35
Dissolved Oxygen (ppb)	<5
Cl^-(ppm)	<0.05
Temperature (°C)	360

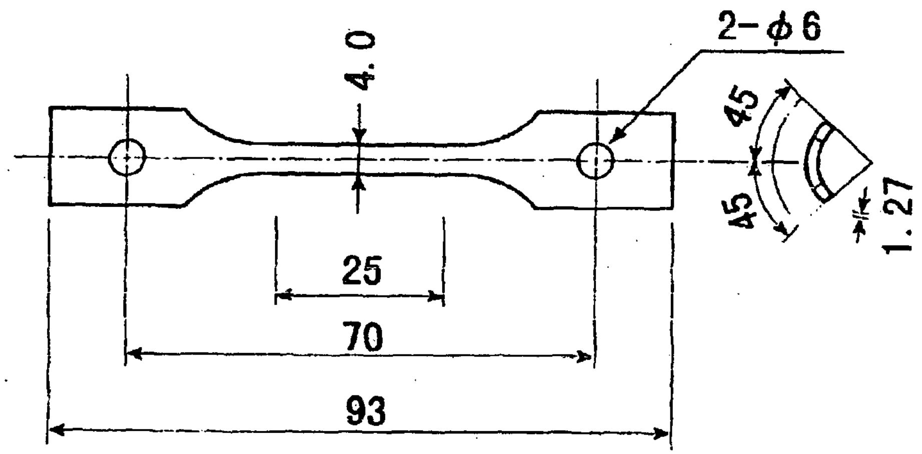
Test Loop for Uni-axial Constant Load Stress Corrosion Cracking Test



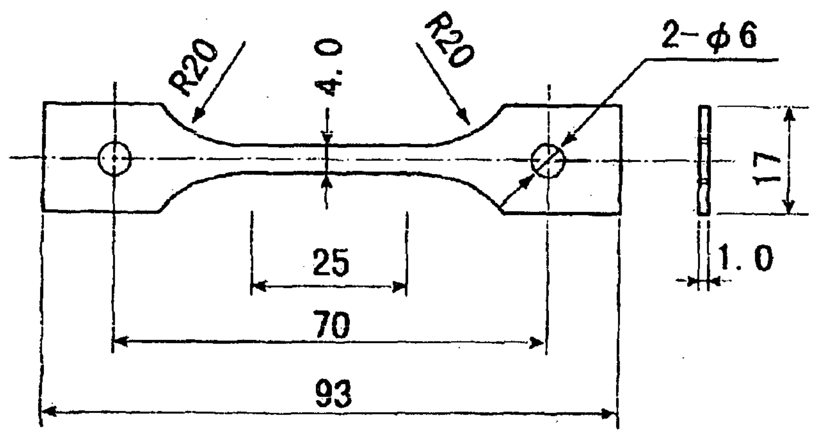
Loading Mechanism of Uni-axial Constant Load Stress Corrosion Cracking Test Instrument



Test Specimens for Uni-axial Constant Load SCC Test (1/2)

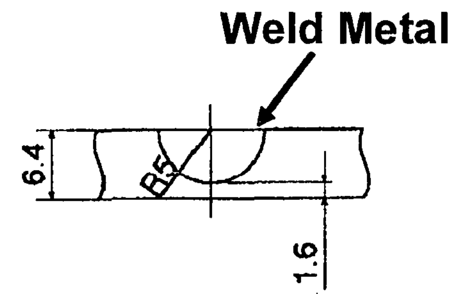


(1) For SG Tube (1/4 Tubular Type)

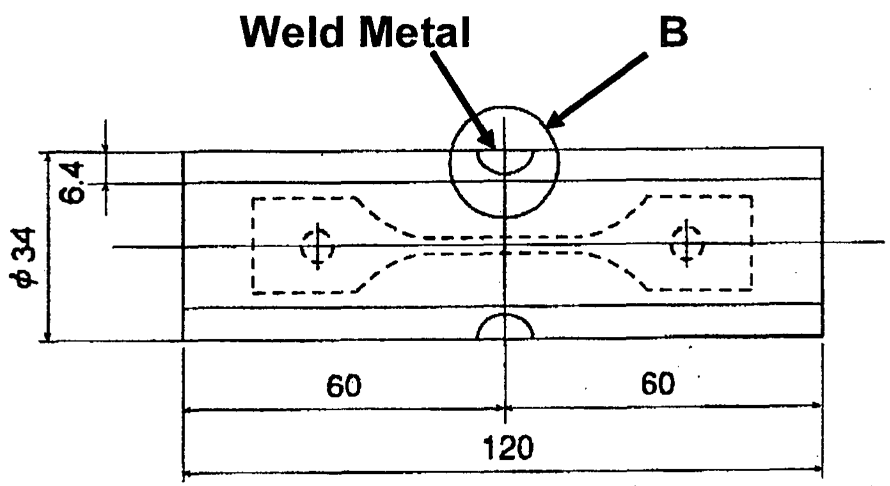


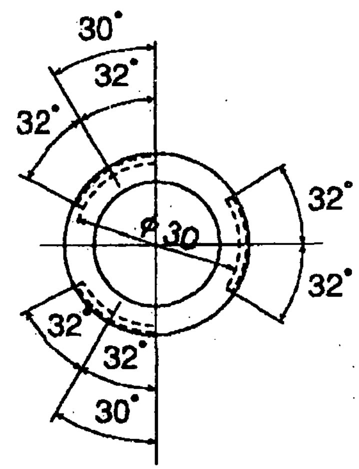
(2) For Alloys (Plate Type)

Test Specimens for Uni-axial Constant Load SCC Test (2/2)



Detail of B



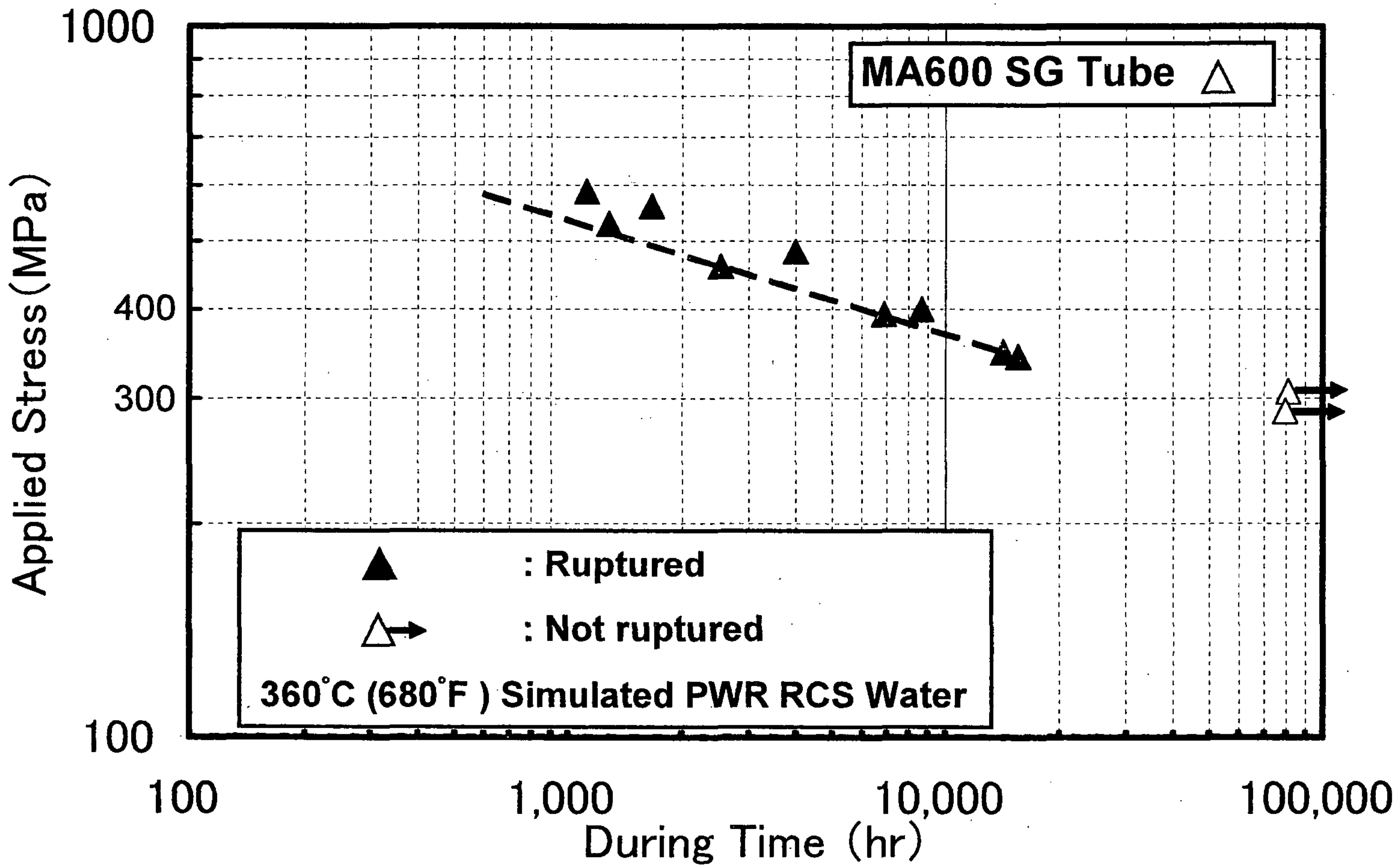


Thickness of Specimens : 1mm

(3) For Weld Metal

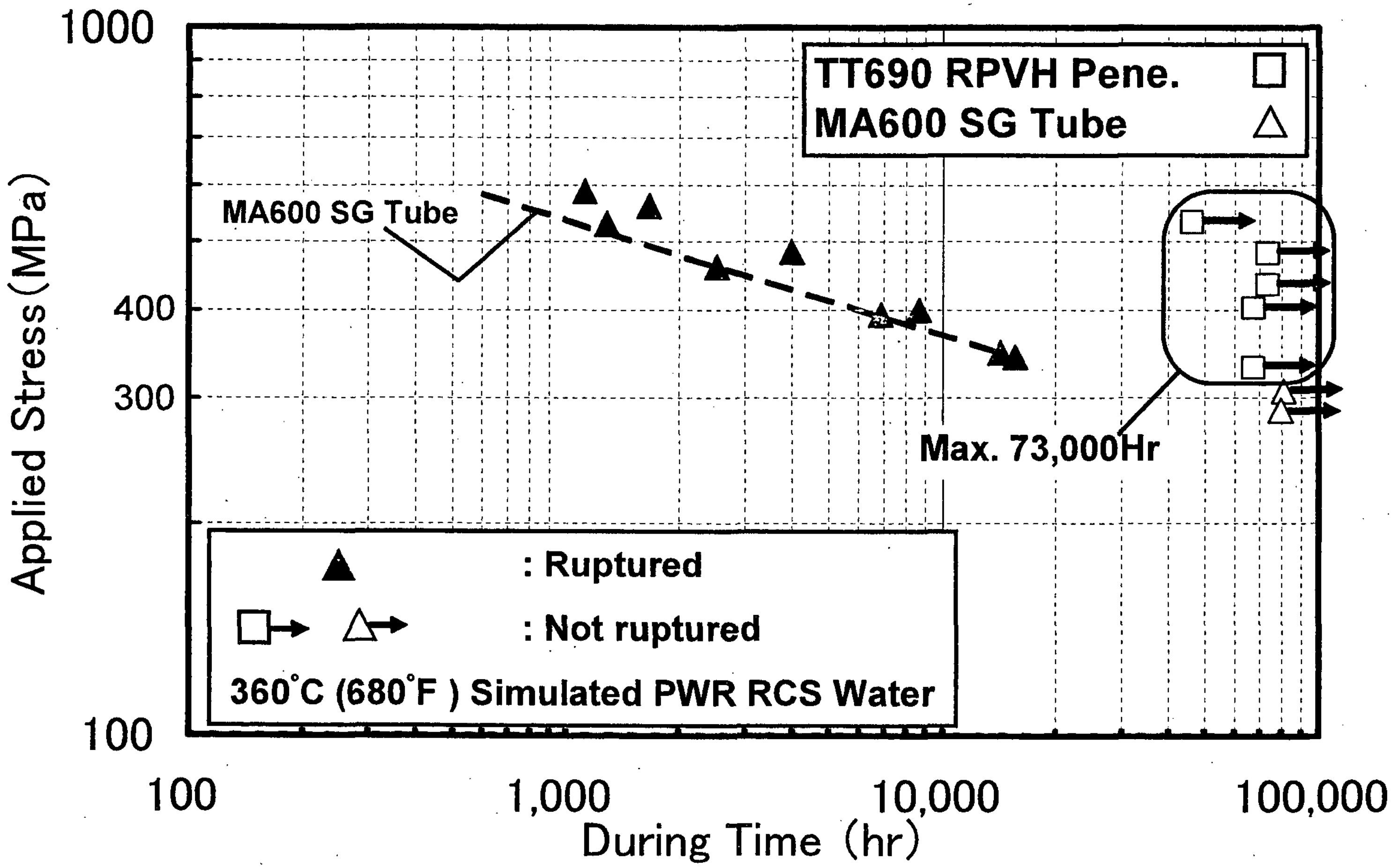
Use of Material Data Base on Alloy 600

Estimation of PWSCC based on Material Data Base on Alloy 600 SG tubes

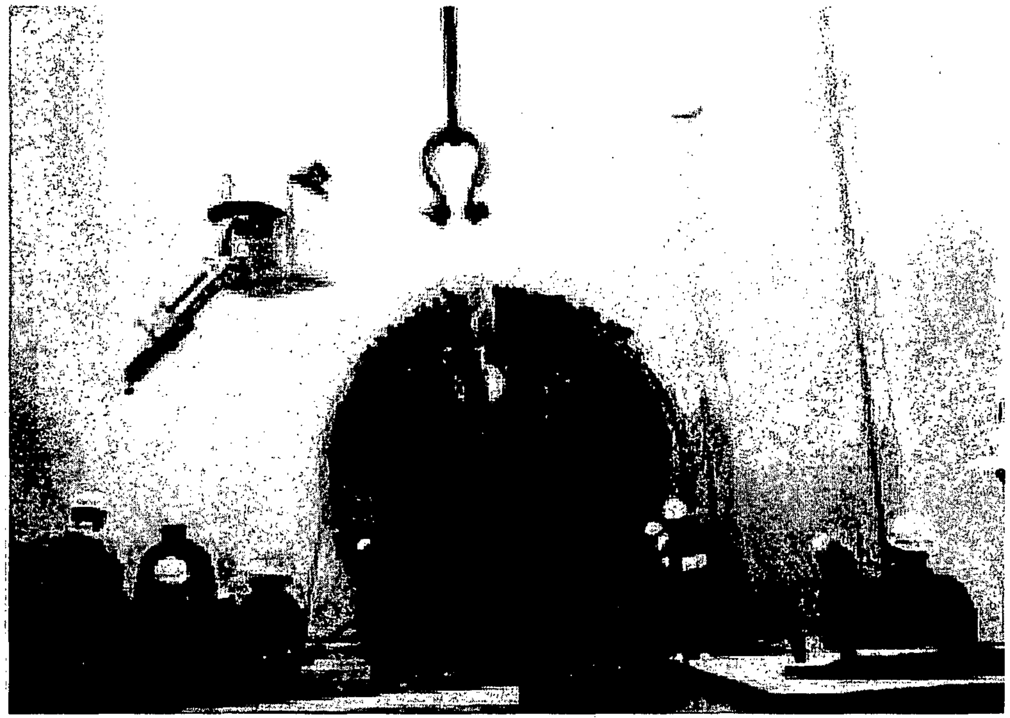


Latest Test Results of Alloy TT690 for RPV Head Penetration

Comparison of PWSCC Initiation Data on TT 690 RPVH Penetration Material with those on Alloy 600 SG tubes



Status of RPV Heads in Japan



(*) Taniguchi,M, Hori, N.,"Maintenance Technology Development for Alloy 600 PWSCC Issue", 12th International Conference on Nuclear Engineering, 2004,

Status of RPV Head Replacement in Japan

Unit	Loop	Utility	RVH R Year
TAKAHAMA 1	3 loops	Kansai	1996
MIHAMA 3	3 loops	Kansai	1997
TAKAHAMA 2	3 loops	Kansai	1997
MIHAMA 2	2 loops	Kansai	1999
OHI 2	4 loops	Kansai	1999
OHI 1	4 loops	Kansai	2000
GENKAI 1	2 loops	Kyushu	2001
GENKAI 2	2 loops	Kyushu	2001
IKATA 1	2 loops	Shikoku	2001
MIHAMA 1	2 loops	Kansai	2001

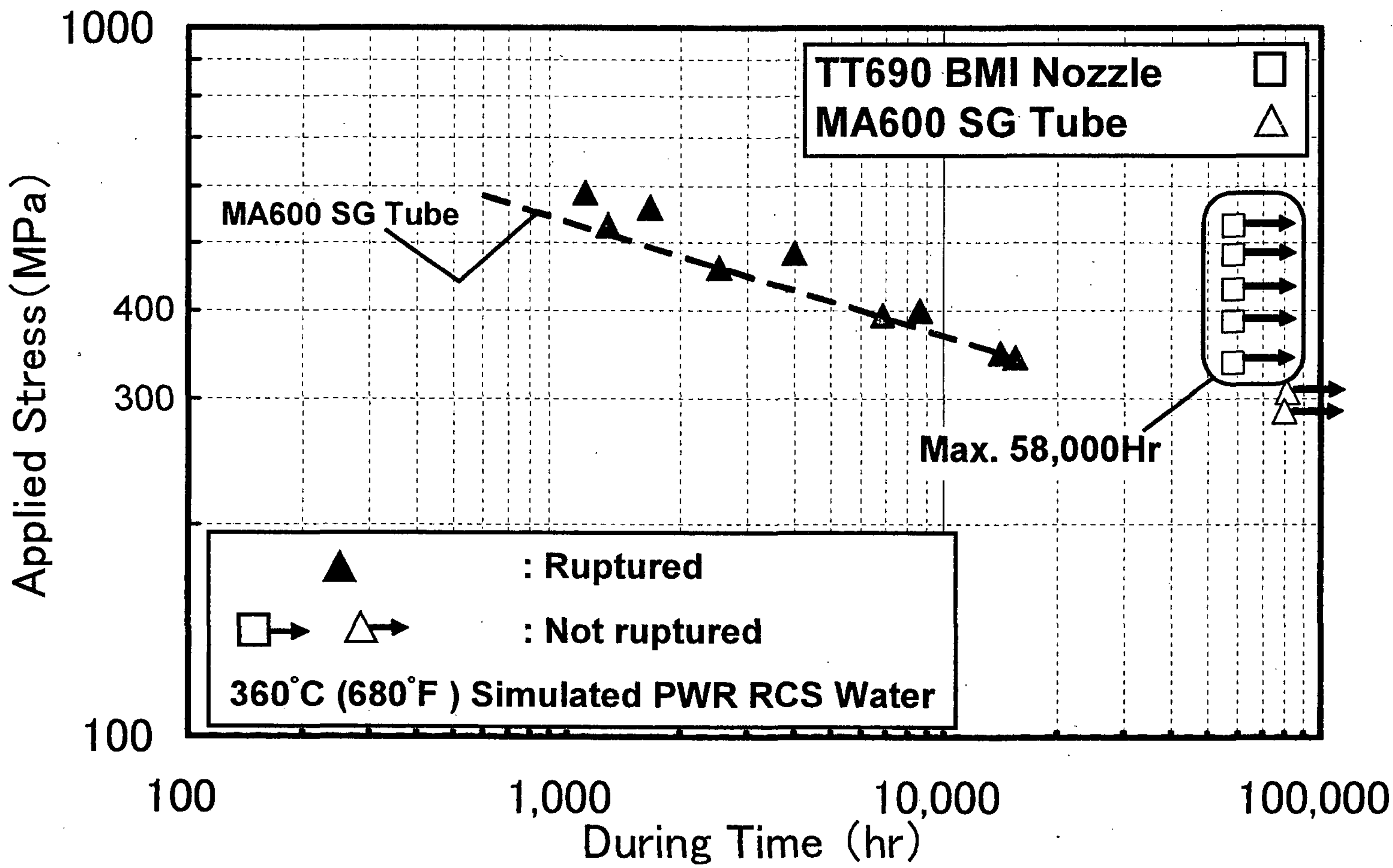
[continued]

Status of RPV Head Replacement in Japan

Unit	Loop	Utility	RVH R Year
IKATA 2	2 loops	Shikoku	2002
OHI 3	4 loops	Kansai	2006
TAKAHAMA 4	3 loops	Kansai	2007 (Plan)
OHI 4	4 loops	Kansai	2007 (Plan)
TSURUGA 2	4 loops	JAPC	2007 (Plan)
TAKAHAMA 3	3 loops	Kansai	2008 (Plan)
TOMARI 2	2 loops	Hokkaido	2009 (Plan)
TOMARI 1	2 loops	Hokkaido	2008 (Plan)
SENDAI 1	3 loops	Kyushu	2008 (Plan)
SENDAI 2	3 loops	Kyushu	2008 (Plan)

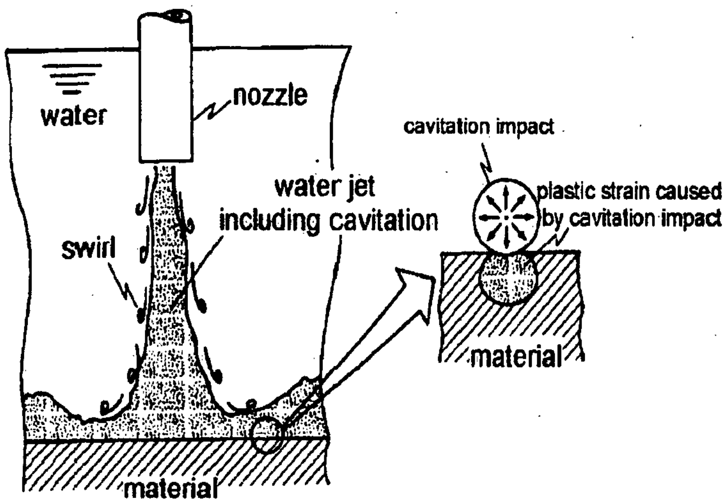
Latest Test Results of Alloy TT690 for BMI Nozzle

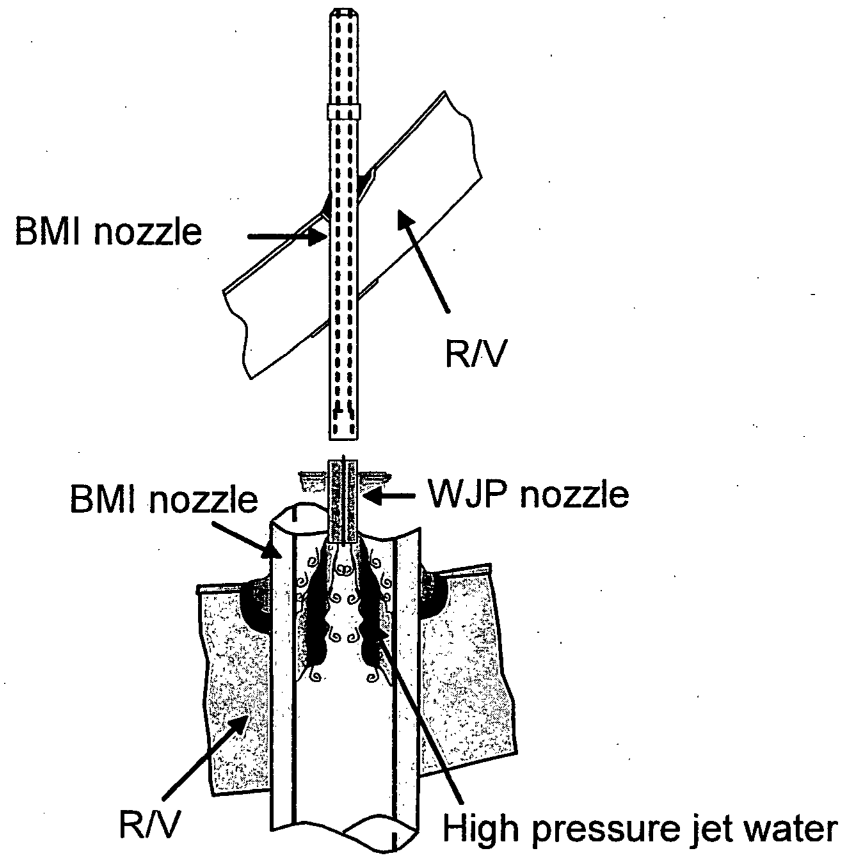
Comparison of PWSCC Initiation Data on TT 690 BMI Nozzle Material with those on Alloy 600 SG tubes



Preventive Maintenance for BMI Nozzle

Water Jet Peening (WJP) is applied to relieve tensile stress.

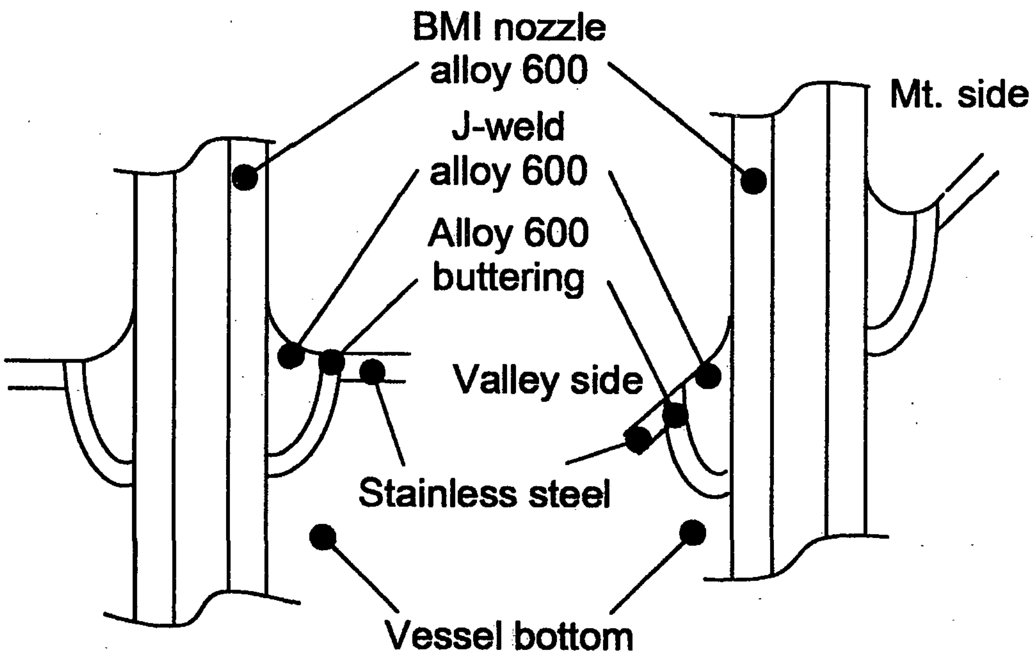




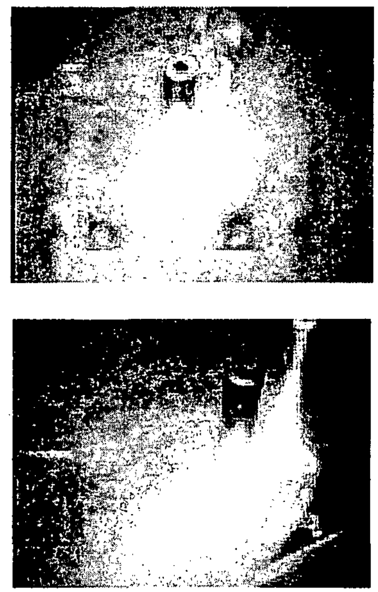
(*) Koji Okimura et al., "Residual Stress Improved By Water Jet Peening Using Cavitation For Small-Diameter Pipe Inner Surfaces", 9th international Conference On Nuclear Engineering, 2001

Preventive Maintenance for BMI Nozzle

Water Jet Peening (WJP) is also applied to J-welds of BMI Nozzles.



BMI nozzle near the center   BMI nozzle near the circumference

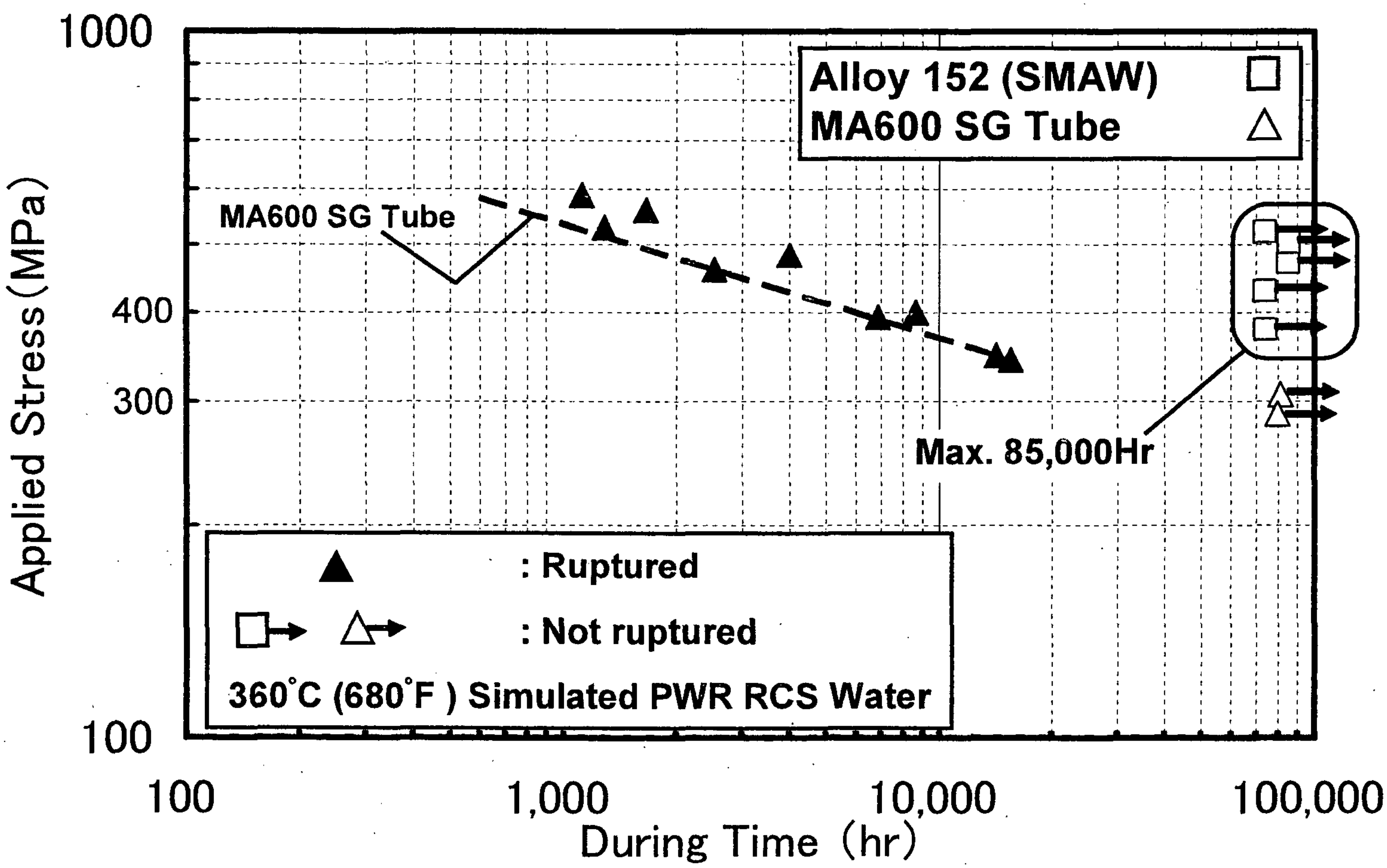


WJP for J-weld

(*) Taniguchi,M, Hori, N.,"Maintenance Technology Development for Alloy 600 PWSCC Issue", 12th International Conference on Nuclear Engineering, 2004

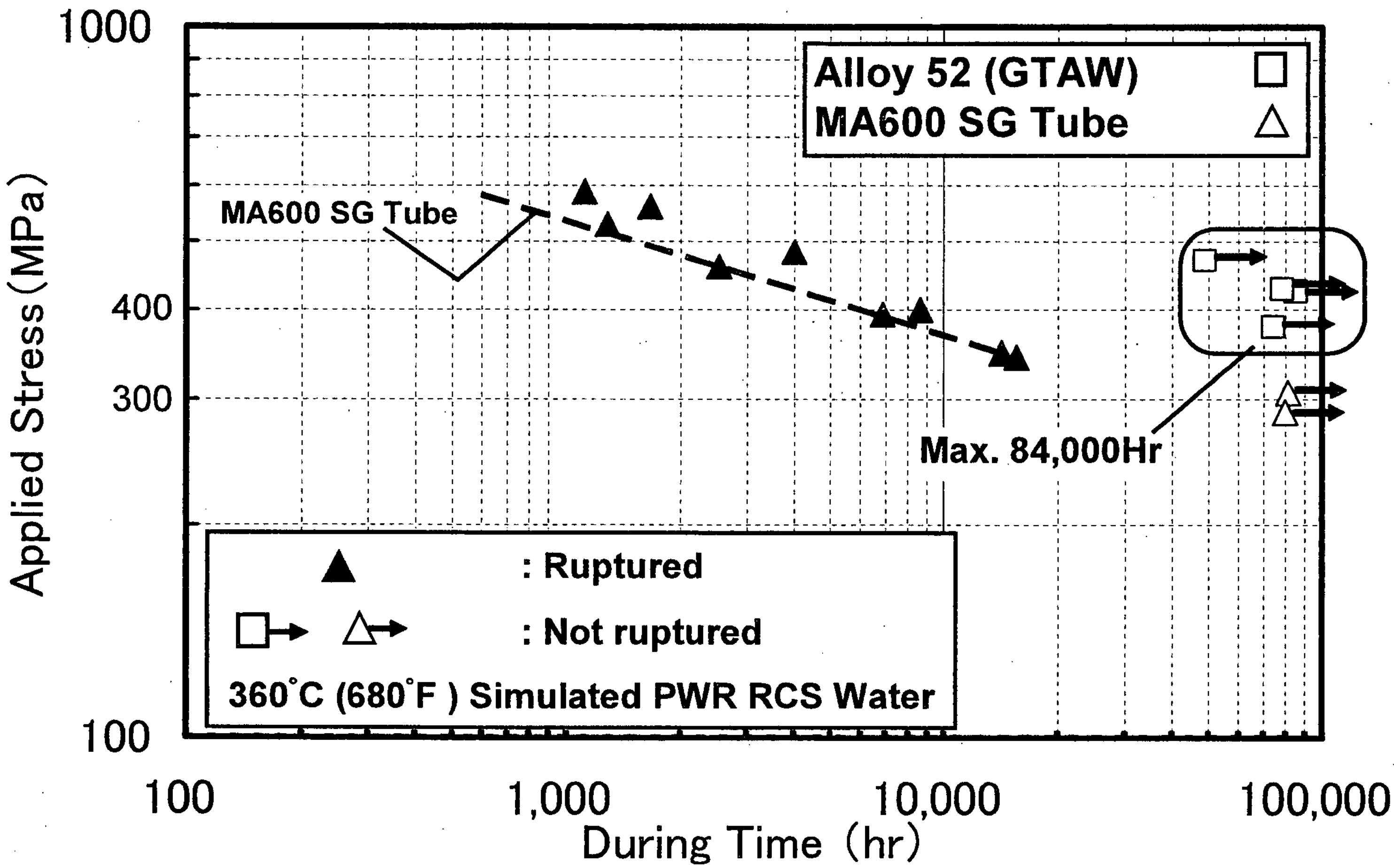
Latest Test Results of Alloy 152 (SMAW)

Comparison of PWSCC Initiation Data on Alloy 152 (SMAW) Material with those on Alloy 600 SG tubes



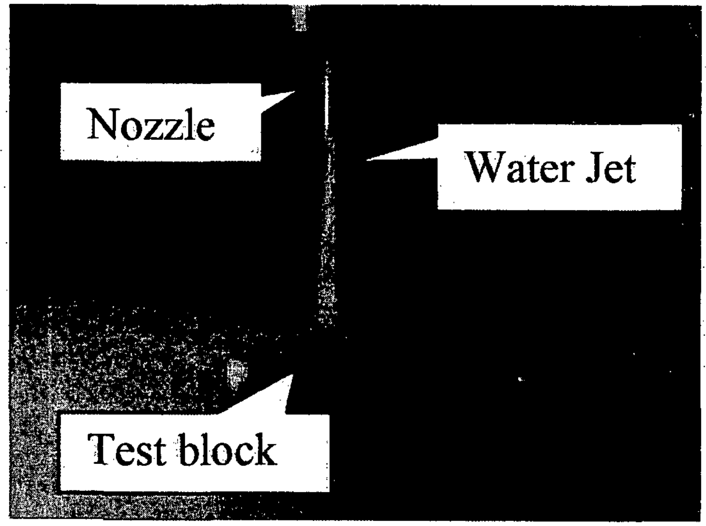
Latest Test Results of Alloy 52 (GTAW)

Comparison of PWSCC Initiation Data on Alloy 52 (GTAW) Material with those on Alloy 600 SG tubes



Preventive Maintenance for Alloy 600 Welds

Water Jet Peening (WJP) is applied to relieve tensile stress.



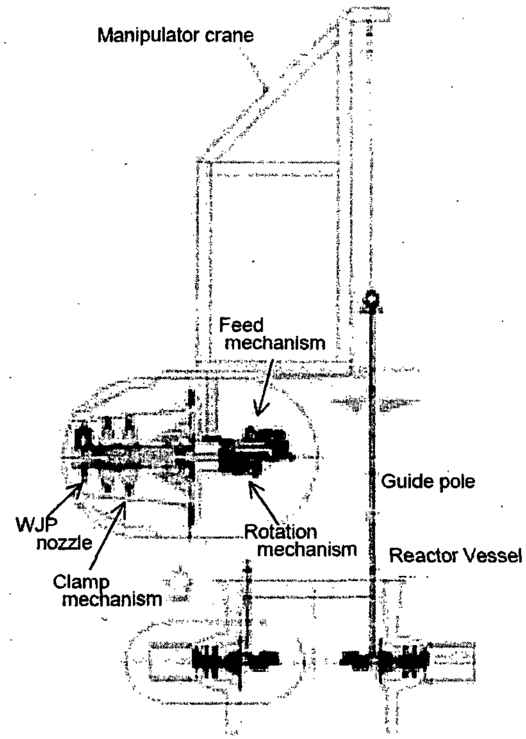
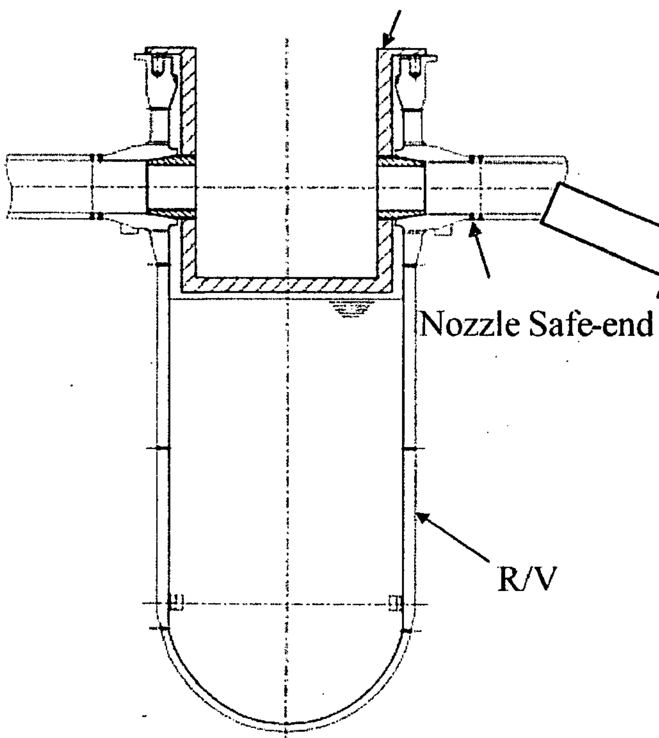


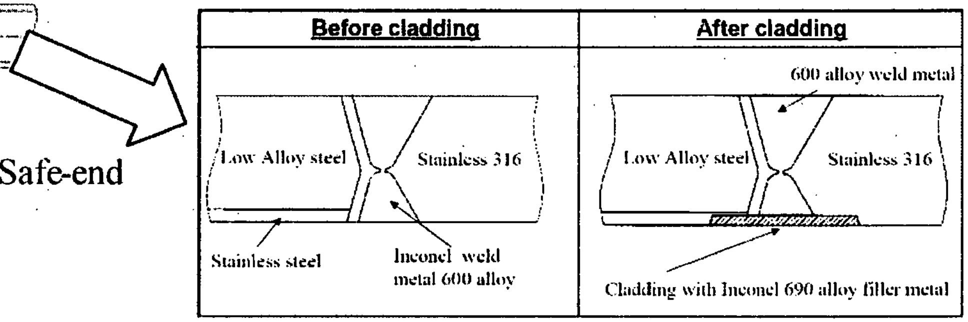
Image of WJP device with guide pole for MCP safe-end

(*) Taniguchi,M, Hori, N.,"Maintenance Technology Development for Alloy 600 PWSCC Issue", 12th International Conference on Nuclear Engineering, 2004,

Preventive Maintenance for Alloy 600 Welds

Alloy 690 Cladding, spool piece replacement , etc. are also preventive maintenance methods.





(*) Taniguchi,M, Hori, N.,"Maintenance Technology Development for Alloy 600 PWSCC Issue", 12th International Conference on Nuclear Engineering, 2004,

CONCLUSIONS

- The Japanese PWR utilities and MHI have been accumulating material data for Ni-based alloys and maintenance for Alloy 600 material in the plants , since ECT indications were found in the SG tubes.
- As the leak in Bugey-3 in 1991 was a turning point, the maintenance strategies for RV Head have been also established based on the estimation for PWSCC initiation by use of the Alloy 600 database, and the Japanese PWR utilities started RVH replacement where new RV heads have Alloy 690 head penetrations.

CONCLUSIONS (continue)

- The constant load PWSCC tests for Alloy 690 materials are being performed and it is ascertained that Alloy 690 materials including weld metals have excellent reliability against PWSCC.
- Preventive maintenance measures, such as WJP, etc. for Alloy 600 portions are also being performed for the Japanese PWR plants.

Attachment 2

Crack Growth Response in Simulated PWR Water of Alloy 152 Weld Metal

Presented by Pacific Northwest National Laboratory

ICG-EAC

April 2007

Huallien, Taiwan

Crack Growth Response in Simulated PWR Water of Alloy 152 Weld Metal

M.B. Toloczko

S.M. Bruemmer

Pacific Northwest National Laboratory

Richland, WA

ICG-EAC Hualien, Taiwan April, 2007

Presentation Outline

- Introduction
- PNNL Crack Growth Systems
- Test Setup
- Alloy 152 CGR Results
- Alloy 152 Fracture Surface
- Summary & Conclusions

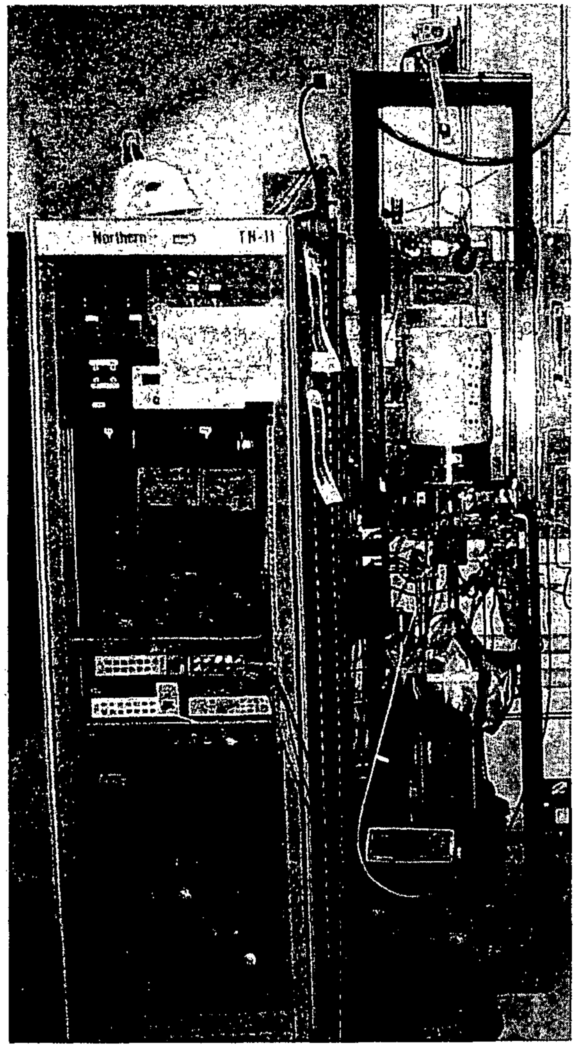
Introduction

- Stress corrosion crack growth testing of alloy 690 CRDM tube heats and prototypic alloy 152 weldments are underway at PNNL in simulated PWR primary water.
- Available information on these materials suggest very low crack growth rates requiring very long tests to achieve measurable crack growth even in the best systems.
- Variations in material (CW, rolling orientation, heat treatment) and environmental (temperature, impurities) are being evaluated.
- Initial data presented on an as-received alloy 152 mockup weld under simulated PWR primary water conditions.

PNNL Crack Growth Systems

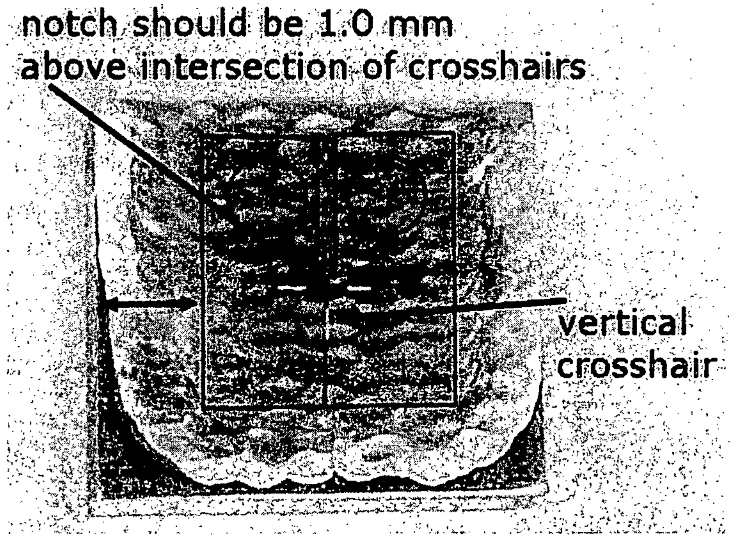
Pacific Northwest
National Laboratory

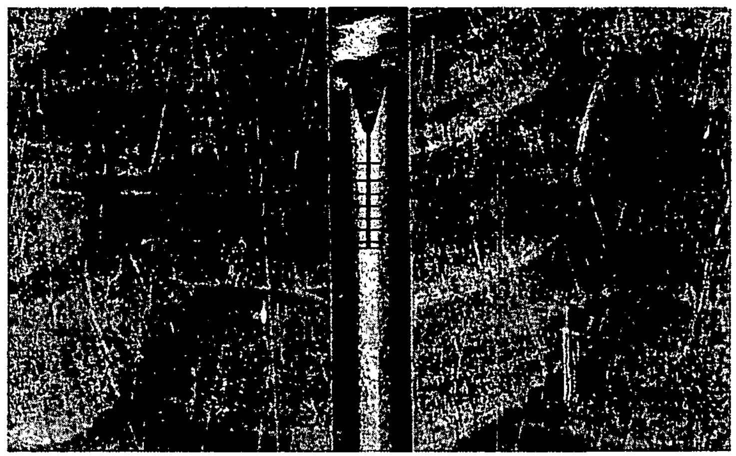
- Outlet conductivity ≤ 0.065 μS/cm under BWR test conditions
- Reversing DCPD, automated K control, autoclave flow rate of 220 cc/min.
- Continuous measurement of load, inlet conductivity, outlet conductivity, DCPD voltage, DCPD current, autoclave water temperature, and other parameters
- Water conductivity in conjunction with manual pH measurement for B/Li determination.



Specimen

- Alloy 152 weld supplied by EPRI NDE Center.
- Weld material is a mockup made by MHI for Kewaunee reactor.
- Sample is composed entirely of weld material.
- Crack root oriented to allow SCC testing in the middle of a weld pass with crack oriented roughly along dendrite direction.





Test Setup

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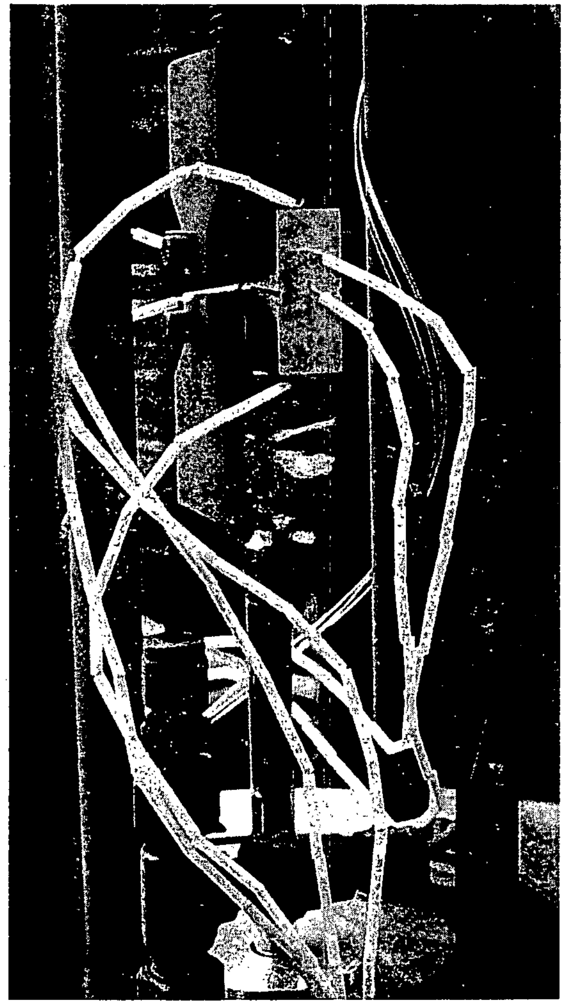
- Conditions: 30 MPa$\sqrt{m}$, 325°C, 1000 ppm B, 2 ppm Li, 29 cc/kg H_2.
- Pre-cracked in-situ using sequence to transition from fatigue to SCC.
- Crack length measurement calculated from DCPD data using reference DCPD potential correction
  - reference DCPD potential taken from probes on back-face of sample
  - reference potential correction algorithm designed for reference probes in this location



Results - CGR Summary

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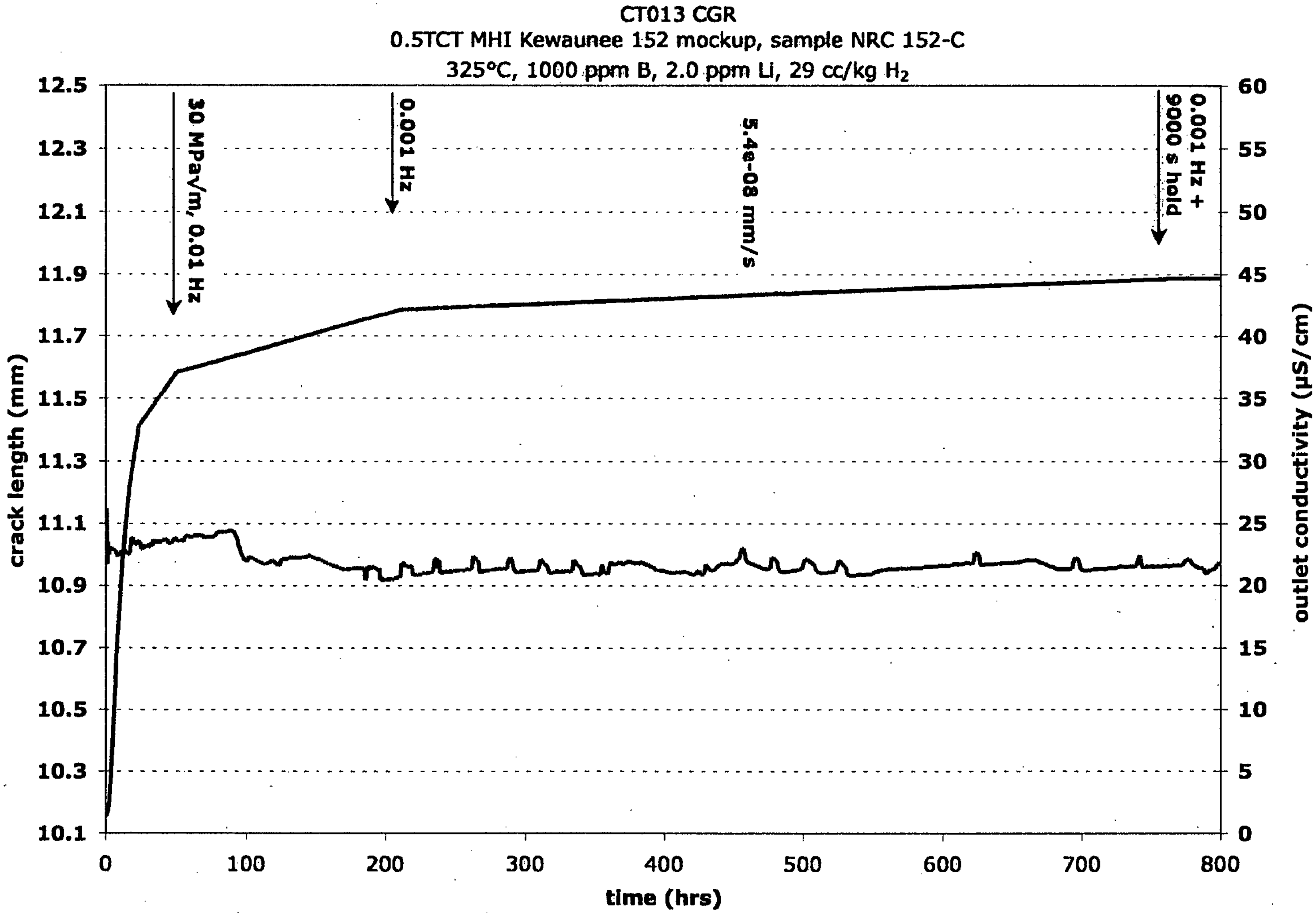
National Laboratory

Expect extremely low or zero SCC CGRs in PWR primary water.
SCC response evlated by approaching constant K through a series of steps with gentle cycling and increasingly longer hold time.

Step	K_{max} (MPa√m)	Load Ratio	Frequency	CGR (mm/s)	length of time (hrs)	crack ext (μm)
1	25	0.3	1 Hz			
2	28	0.5	1 Hz			
3	30	0.6	1 Hz			
4	30	0.7	1 Hz			
5	30	0.7	0.1 Hz			
6	30	0.7	0.01 Hz			
7	30	0.7	0.001 Hz	5.4×10^{-8}	580	100
8	30	0.7	0.001 Hz + 9000 s	6.8×10^{-9}	540	10
9	30	0.7	0.001 Hz + 86,400 s	1.0×10^{-9}	800	3.5
10	30		constant K	*1.0×10^{-10}	2250	1

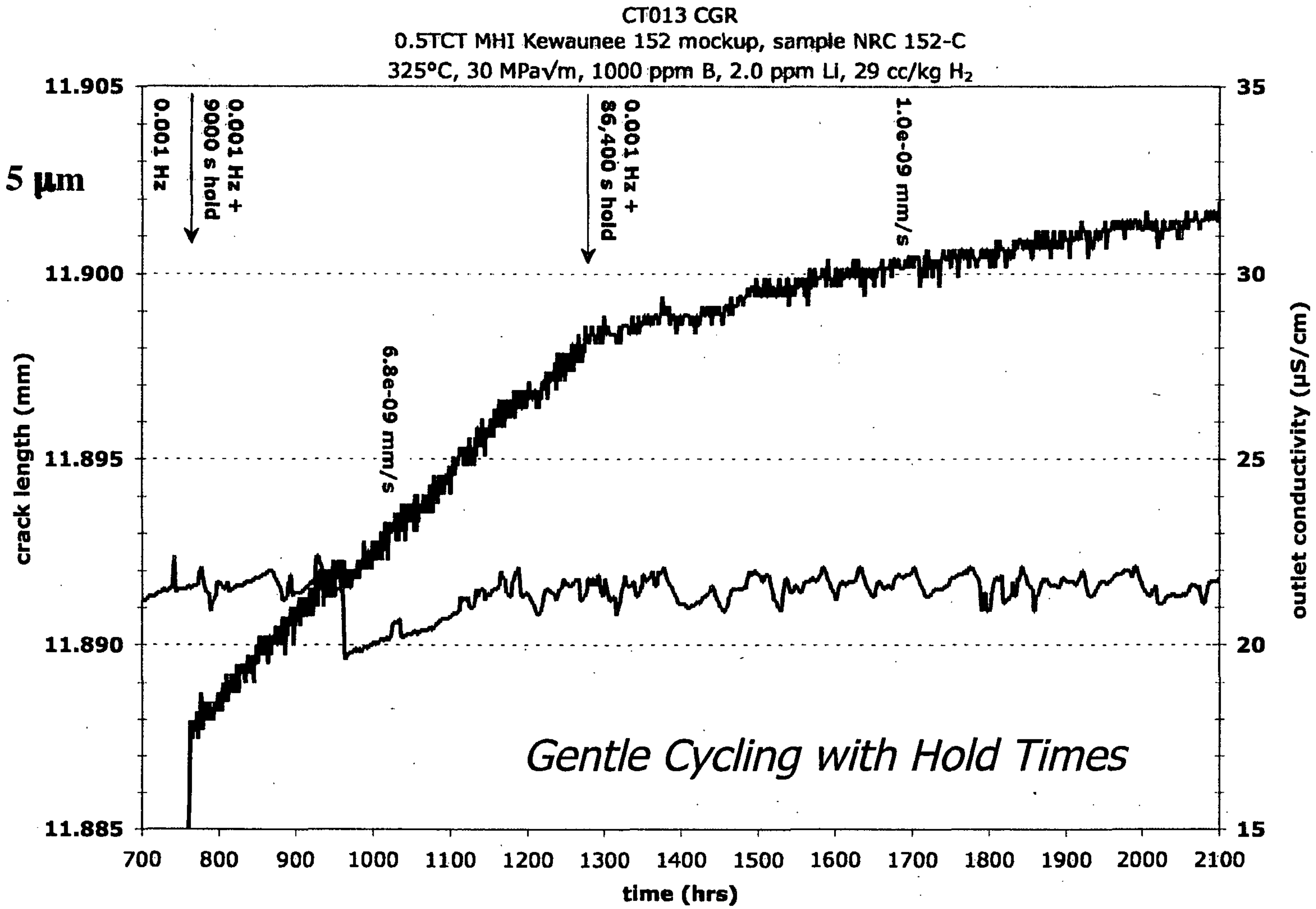
Crack Transitioning: Steps 1-7

Pacific Northwest National Laboratory



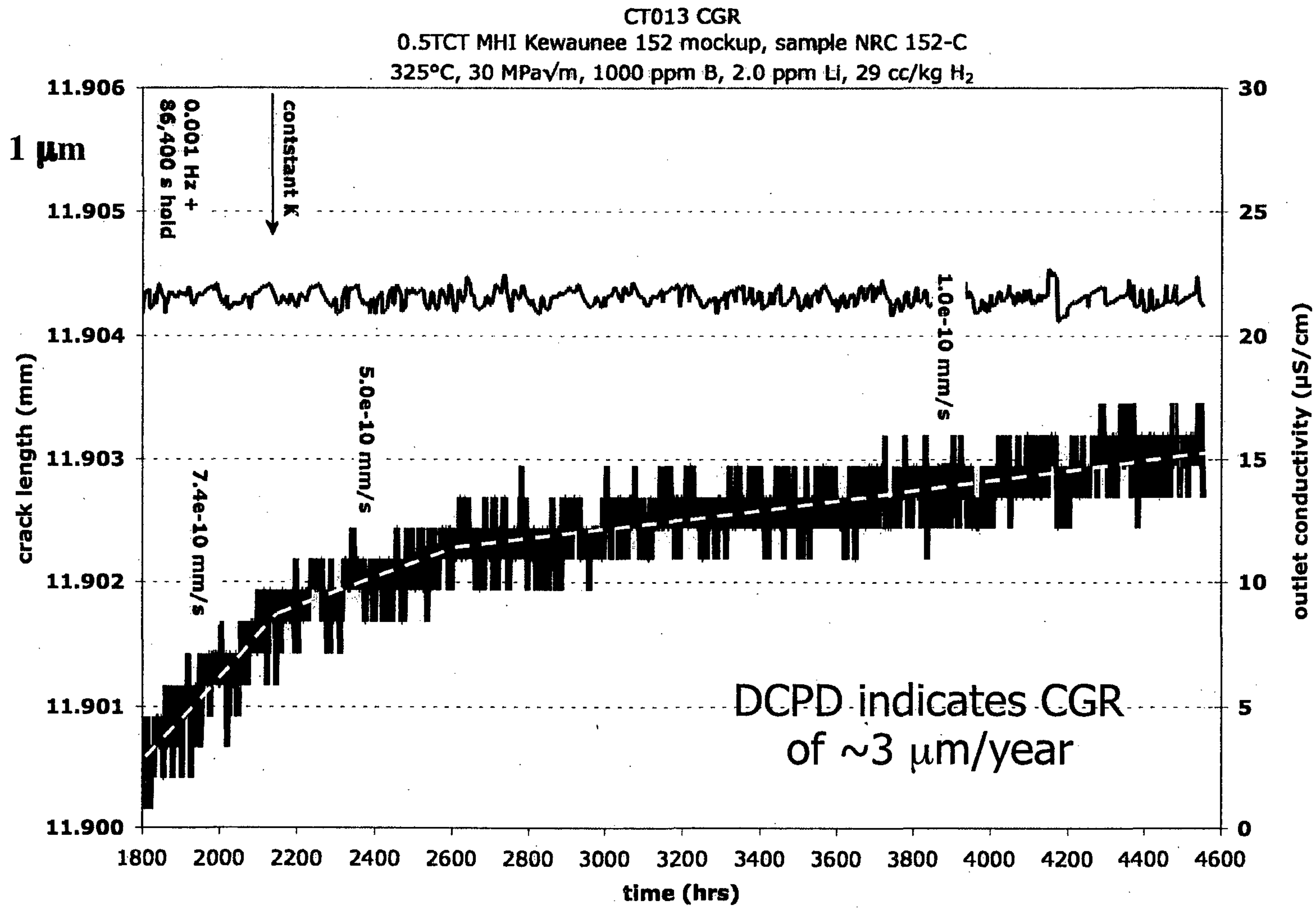
Crack Transitioning: Steps 8-9

Pacific Northwest National Laboratory



Step 10: Constant K

Pacific Northwest National Laboratory



Results - CGR Summary

Pacific Northwest National Laboratory

Step	K_{max} (MPa√m)	Load Ratio	Frequency	CGR (mm/s)	length of time (hrs)	crack ext (μm)
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4	30	0.7	1 Hz			
5	30	0.7	0.1 Hz			
6	30	0.7	0.01 Hz			
7	30	0.7	0.001 Hz	5.4×10^{-8}	580	100
8	30	0.7	0.001 Hz + 2.5 h	6.8×10^{-9}	540	10
9	30	0.7	0.001 Hz + 24 h	1.0×10^{-9}	800	3.5
10	**30**		**constant K**	***1.0×10^{-10}**	**2250**	**1**

Observations

- Low CGRs measured in alloy 152 weld metal decreasing to ~7×10^{-9} and ~1×10^{-9} mm/s during 0.001 Hz cyclic loading with hold times of 2.5 or 24 h, respectively.

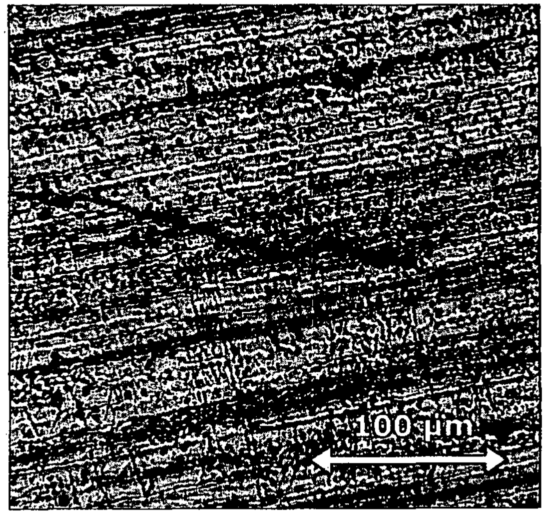
- DCPD measurements suggest consistent crack advance under constant K, but at an extremely low propagation rate where years would be required to obtain sufficient crack extension for a confident assessment.

- CGR under constant K is clearly less than that during the cycle + hold conditions and approaches ~10^{-10} mm/s.

Post-Test Crack Profile

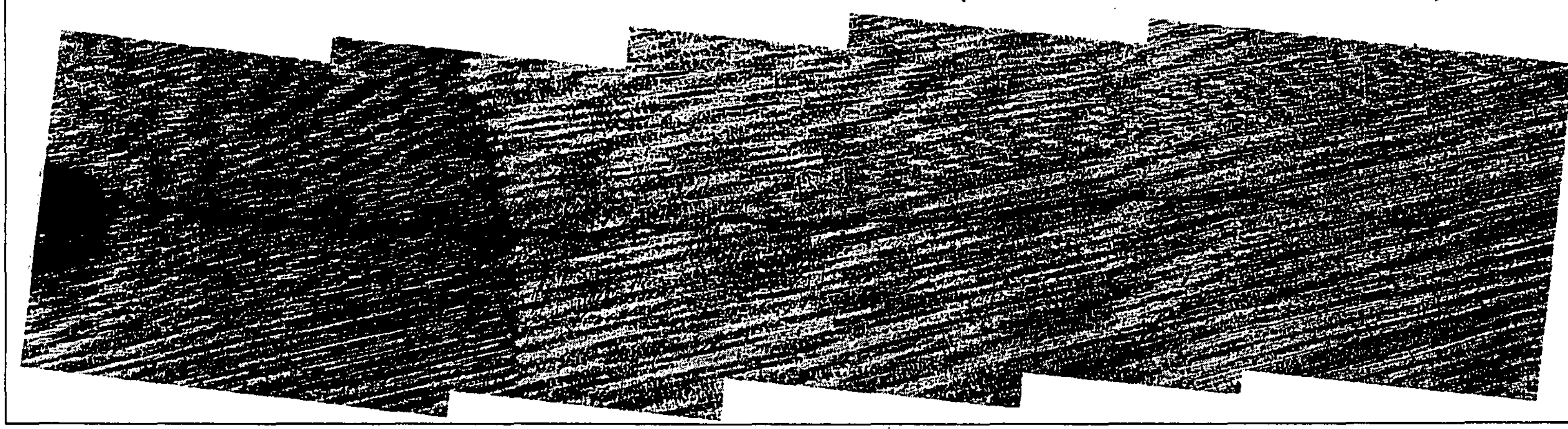
Pacific Northwest
National Laboratory

- Total crack length in this cross-section is about 2.6 mm.
- Fatigue or corrosion fatigue growth does not show significant tendency to flow dendrite boundaries.
- SCC growth limited to final conditions and last few μm.



CT013 Crack Profile

1.0 mm



Post-Test Fracture Surface

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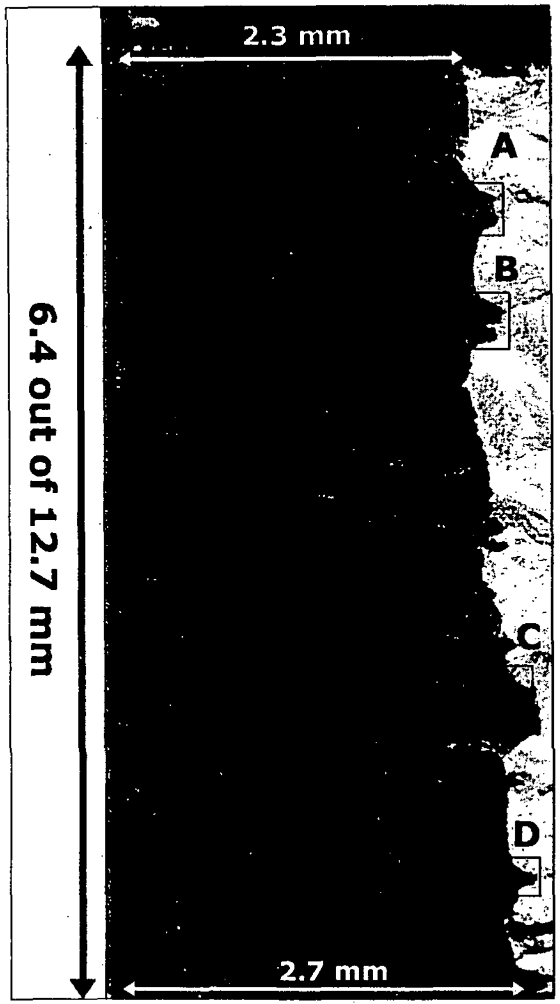
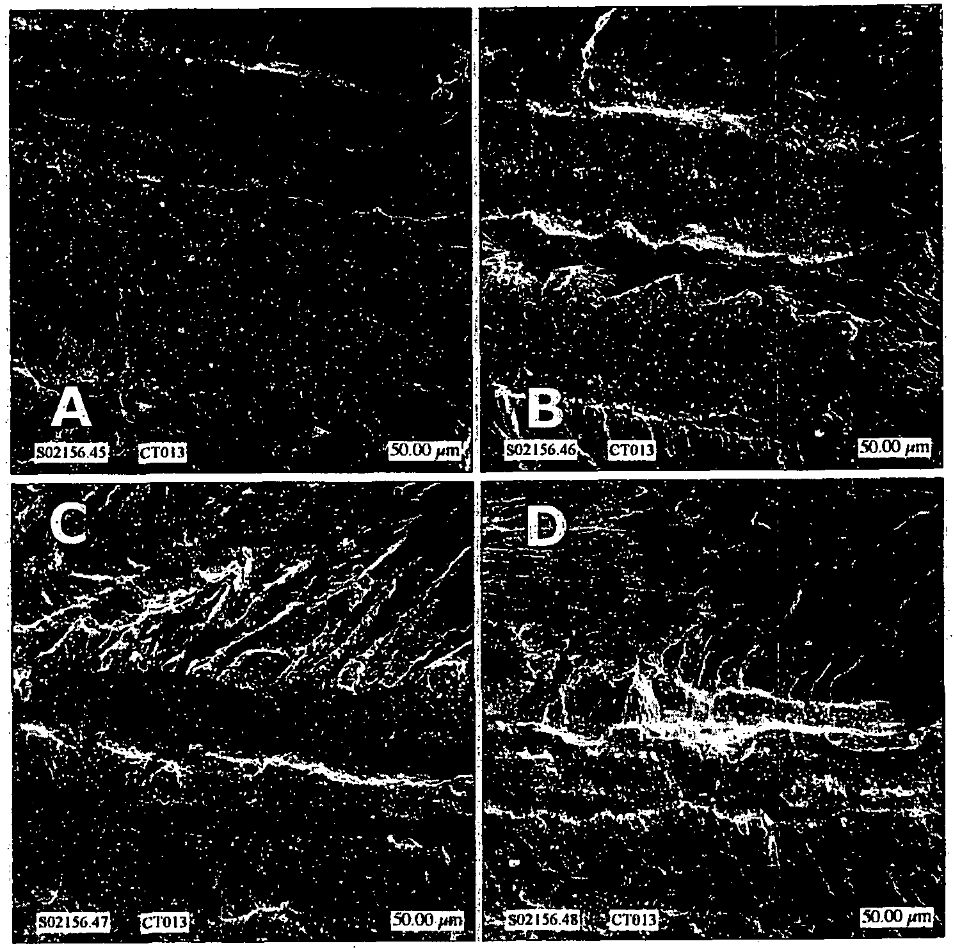
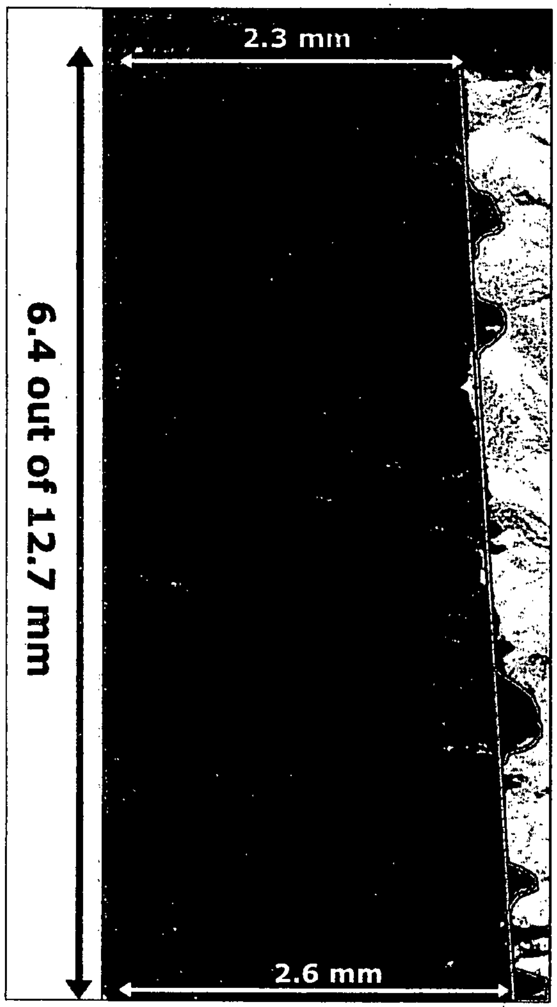


Interdendritic SCC seen at crack front



SCC CGR from Crack Surface



Rough Estimates of CGR in "Local" Regions along Crack Front:

- Assume SCC crack growth limited to regions extending beyond straight line drawn across crack front.
- Calculate area of extensions and divide by width of section analyzed.
- Assume SCC growth begins at onset of first step with hold time (at 800 h).
- Calculated SCC CGR: ~3×10^{-9} mm/s in these "local" regions; consistent with DCPD-measured rates for final steps.

Summary & Conclusions

Pacific Northwest National Laboratory

- Alloy 152 weld metal found to be SCC resistant in simulated primary PWR water at 325°C even when the pre-crack is oriented along dendrite boundaries in a single pass.
- Stable CGR measured at ~5×10^{-8} mm/s during cycling at 0.001 Hz and decreases to ~10^{-9} mm/s during SCC transitioning at 0.001 Hz + hold time of 24 h.
- DCPD suggests consistent crack advance under constant K, but at an extremely low propagation rate where years would be required to obtain sufficient crack extension. CGR under constant K is clearly less than that during the cycle + hold conditions in previous steps and approach ~10^{-10} mm/s.
- Fractography indicates a reasonably straight crack front during cyclic loading with interdendritic SCC during final steps.
- Additional long-term, higher-temperature tests are underway on as-welded and stress-relieved alloy 152 samples in series.

Attachment 3

PWSCC Growth Rates in Alloy 690 and Its Weld Metal, GE Global Research Center

Presented by GE Global Research Center

EPRI PWSCC of Alloy 600 2007 International Conference & Exhibition

June 11 – 14, 2007

Atlanta, GA

PWSCC Growth Rates in Alloy 690 and Its Weld Metals

Peter Andresen, John Hickling, Al Ahluwalia and John Wilson
GE Global Research

The goal of this on-going program is to perform initial evaluations of the environmental crack growth rates on Alloy 690 and Alloys 152 / 52 weld metals. As has been consistently shown for many other SCC-resistant materials, some inherent susceptibility to SCC exists, and the concept of SCC immunity should be replaced with concepts such as adequately low crack growth rates. Thus, while Alloy 690 and 152/52 weld metals have lived up to their good reputation as SCC-resistant materials, stable, sustained SCC growth – albeit at very low growth rates ($2 - 7 \times 10^{-9}$ mm/s) – was observed at constant K in simulated primary water at 340 and 360 °C.

When compared with industry standard estimates for the crack growth rates of Alloy 600 and Alloys 182 weld metal, Alloy 690 and its weld metals exhibited rates ≈ 70 – 400X lower, a very sizeable difference. Note that these approximate factors of improvement must be considered preliminary until more specimens, more conditions, more heats, heat affected zones, etc. are evaluated to provide sufficient confidence in the comparisons being made.

The agreement between dc potential drop and the actual crack length determined from post-test fractography was reasonable (in the range of 4 – 40% error), giving confidence in the reliability of the technique to monitor these very low crack growth rates. Other factors, including statistical measures of linearity of behavior, the magnitude of the resistivity correction, etc. provide a strong basis for confidence in the reported crack growth rate observations.

The crack morphology at (or near) constant K was primarily intergranular in many cases for the base metal, and there was further evidence of intergranular secondary cracking. Some transgranular cracking was also observed, especially in the weld materials, leading to the encouraging conclusion that the grain boundaries, which are usually the weak point in the microstructure from an SCC perspective, possess inherently high resistance to SCC in Alloy 690 and its weld metals.

The CRDM form of Alloy 690 used in these studies is much more homogeneous that the Alloy 690 plate used in prior studies. The plate material, particularly after the 982 °C (1800 °F) final anneal, exhibited compositional and carbide banding, less uniformity in grain size, and a lower density of carbides in the grain boundary. But all forms of Alloy 690 tested to date have exhibited similar, very low crack growth rates.

Recent observations on 1-dimensional cold rolled Alloy 690 in the S-L orientation revealed growth rates elevated by as much as ~50X compared to prior studies on T-L orientation. The relevance of such deformation and orientation is not clear, but such observations must be understood and the nature of deformation during fabrication and weld shrinkage must be characterized.

No effect of pH/B/Li water chemistry parameters was observed on Alloy 690, although only very limited data were obtained. This agrees with a large body of data on Alloy 600 and stainless steel.

While the results of the tests to date are very promising, only a limited range of conditions and microstructures have been evaluated to date. Additional testing, some of which is now in progress, is needed to confirm and better quantify the factor of improvement in PWSCC resistance for Alloy 690 and its weld metals as a function of such key variables as: other heats; different types of cold work and orientation vs. the plane and direction of cracking; the thermo-mechanical and residual strain conditions associated with weld heat affected zones; off-microstructure conditions that might be developed during non-optimal processing; weld dilution effects; variation in H_2 fugacity and test temperature; etc.

PWR SCC Growth Rates of Cold Worked Alloy 690 & Alloys 52/152 Weld Metal

Peter Andresen, Al Ahluwalia[2] & John Hickling[3]

GE Global Research Center

[2]EPRI, [3]CMC

Alloy 600 Conference Atlanta June 2007

Testing Approach

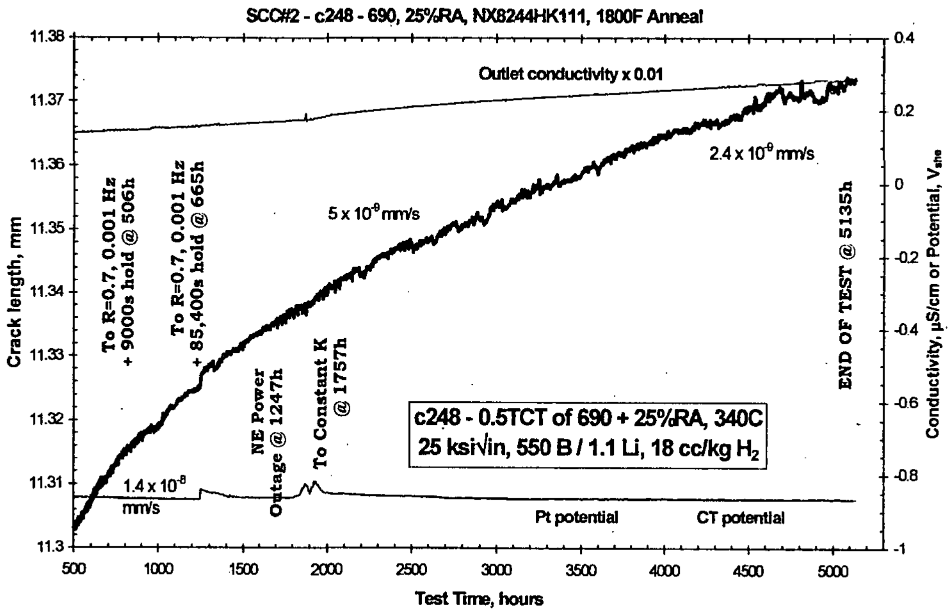
Crack growth rates conditions for alloy 690:

- *cold worked by forging at 25 °C by 20 – 40% (thickness)*
  - *cold work simulates weld residual strain in HAZ*
  - *recent work on 1-dimensional cold rolled (no cross-roll)*
  - *used resistivity coupon for dcpd correction*
- *0.5T CT specimens in 340 & 360 °C PWR primary water*
- *testing at 25 – 35 ksi√in, including "Varying-K" (GE)*
- *18 – 20 cc/kg H_2 to be near Ni/NiO*
- *good water chemistry: ~2 volume exchanges per hour, full-flow demineralization, and active H_2 sparging*
- *measured potentials of 690 & Pt vs. $Cu/Cu_2O/ZrO_2$*

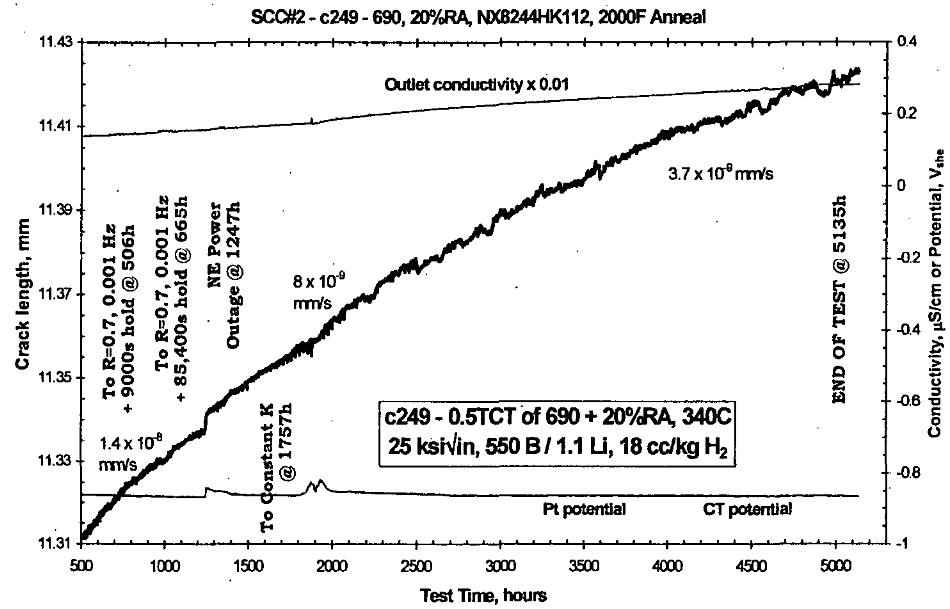
SCC of Alloy 690

1800F Anneal

20% CW Alloy 690



2000F Anneal



Well-behaved, low crack growth rate response during earlier proof-of-concept testing

Alloy 690 CRDM Material

CRDM housing of Alloy 690 (heat WN415)

provided by Duke Power

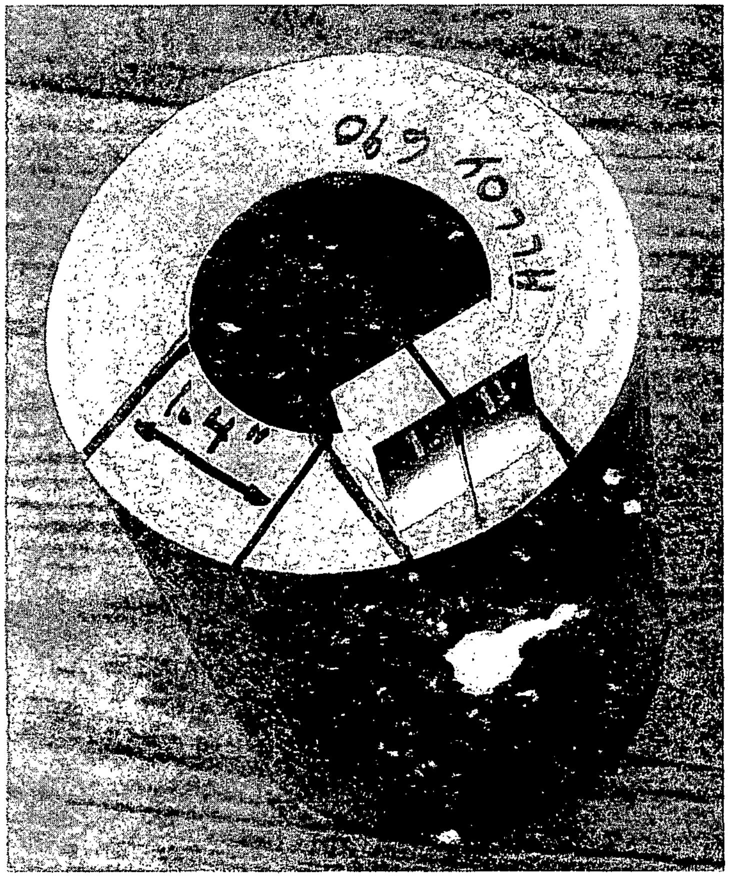
Location	C	Mn	Fe	S	Si	Cu	Ni	Cr	Co
check	0.018	0.31	10.14	0.0007	0.29	0.007	59.67	29.1	0.016
ladle	0.02	0.31	10.1	0.0007	0.28	0.007	59.75	29.04	0.015

Reported average yield strength = 37.7 ksi

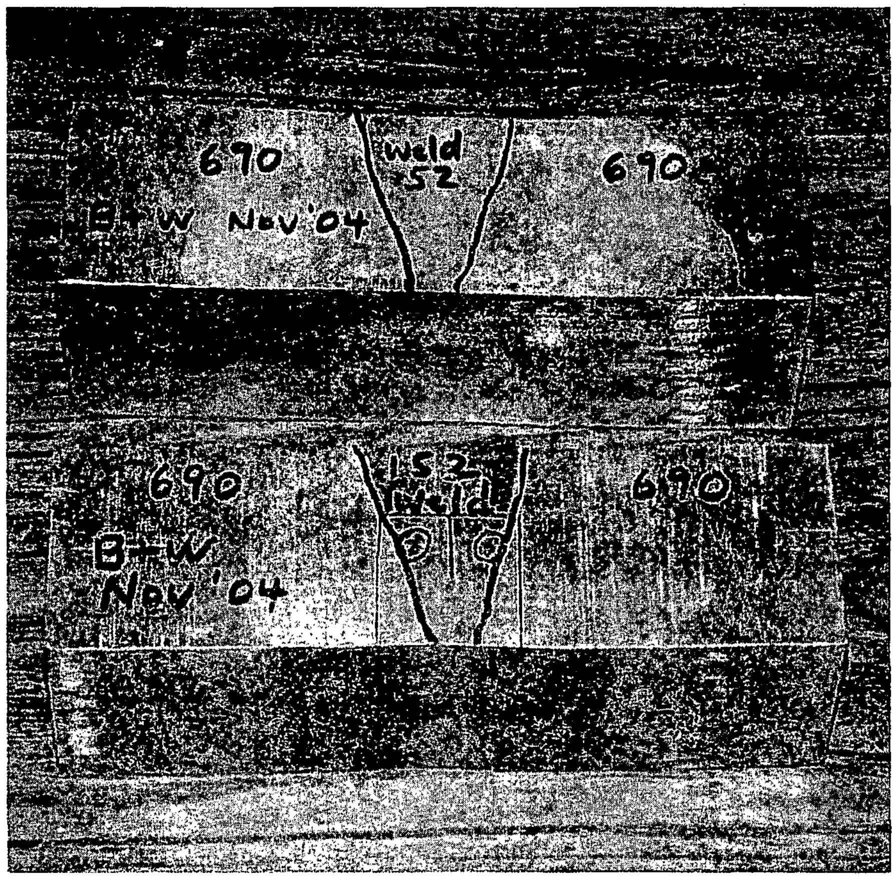
Reported average tensile strength = 89.1 ksi

Annealed at ~721C for ~11 hours

Alloy 690 CRDM & Alloys 52/152

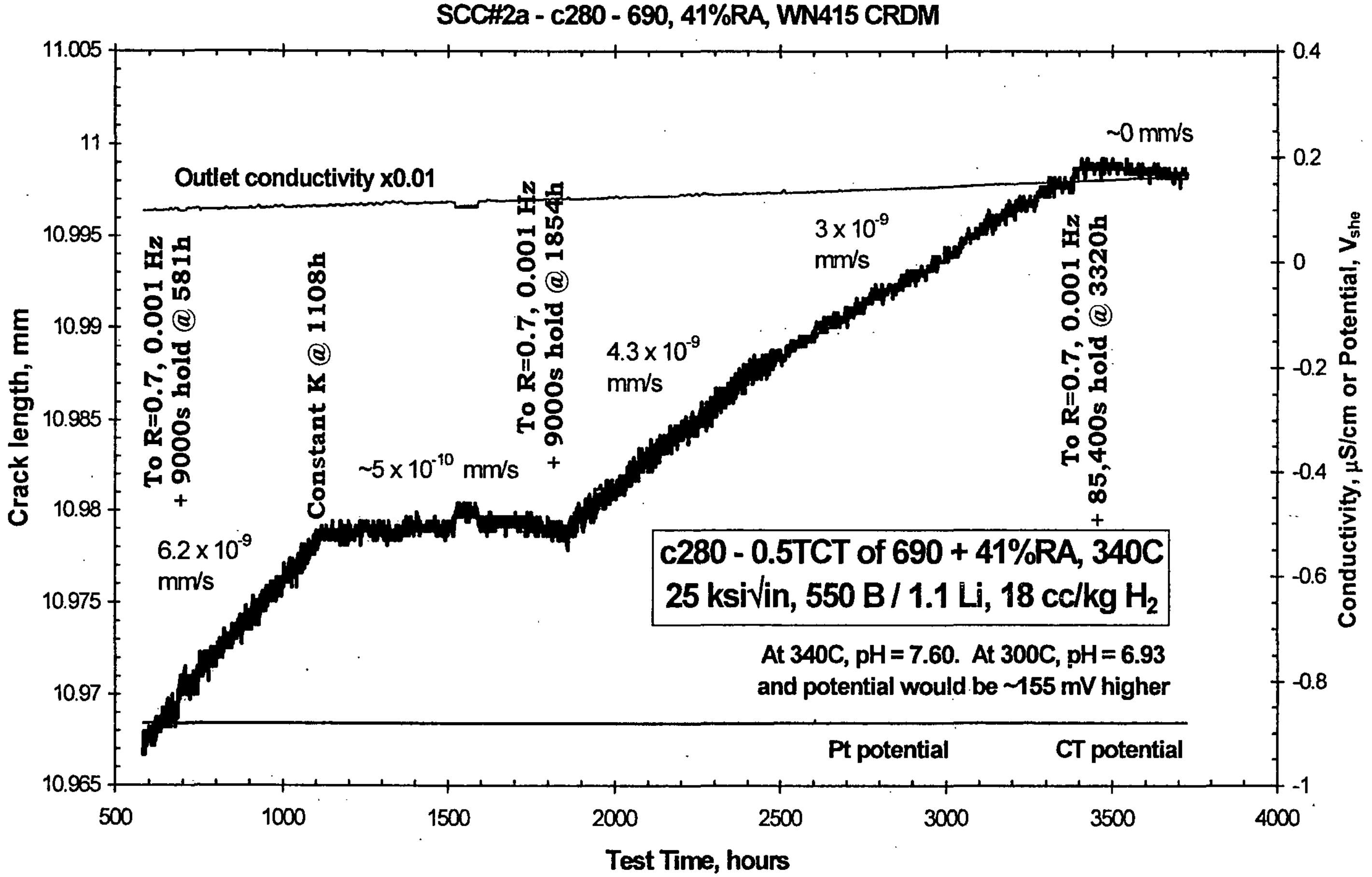


CRDM of Alloy 690
(heat WN415, Duke)



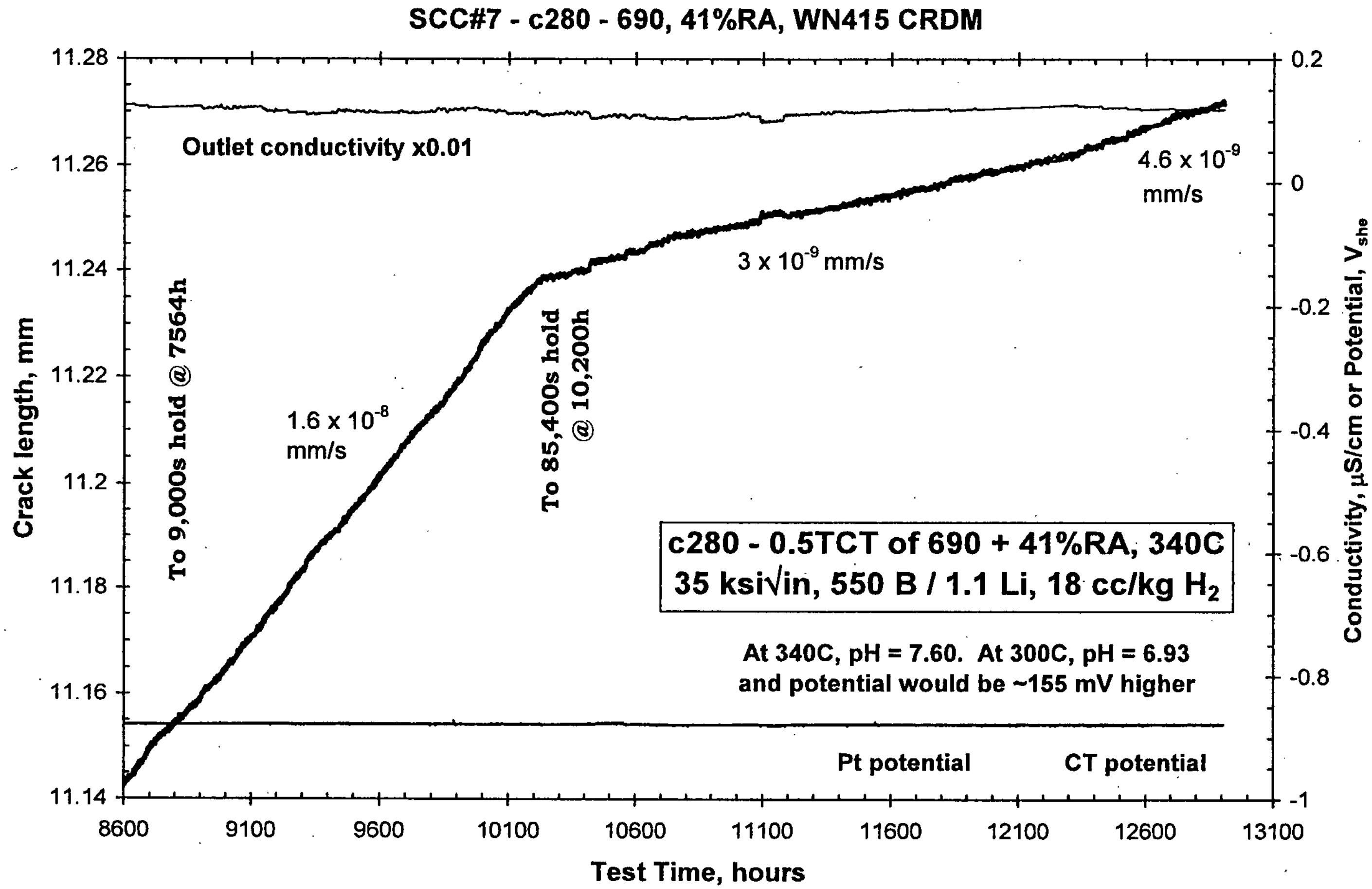
Alloy 52 & 152 weld metal
(from B&W)

41% Cold Work Alloy 690 CRDM



GE tests at Constant & Varying K (dK/da)

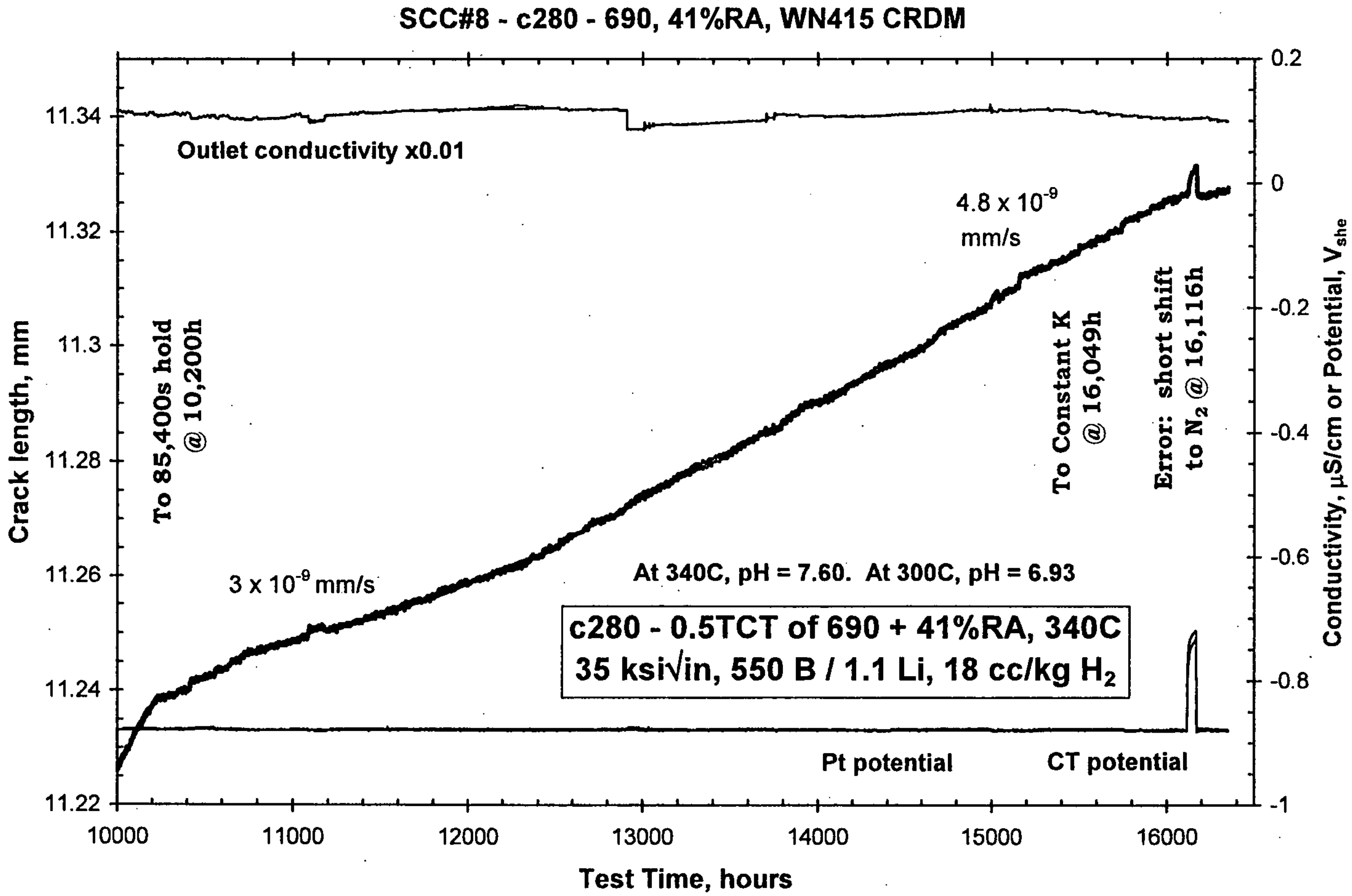
41% Cold Work Alloy 690 CRDM



GE tests at Constant & Varying K (dK/da)

SCC of Alloy 690

41% Cold Work Alloy 690 CRDM



GE tests at Constant & Varying K (dK/da)

SCC of Alloy 690

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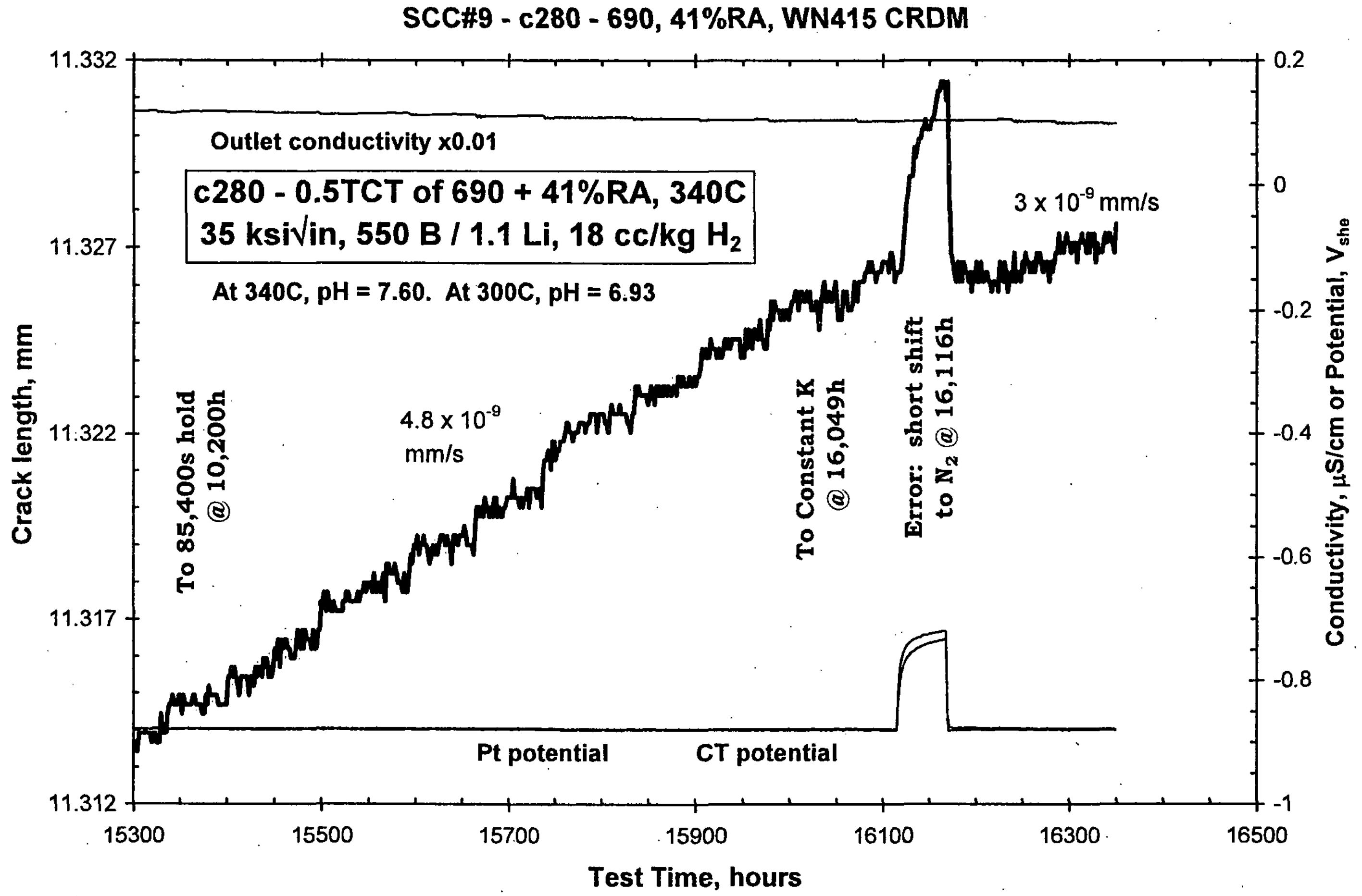
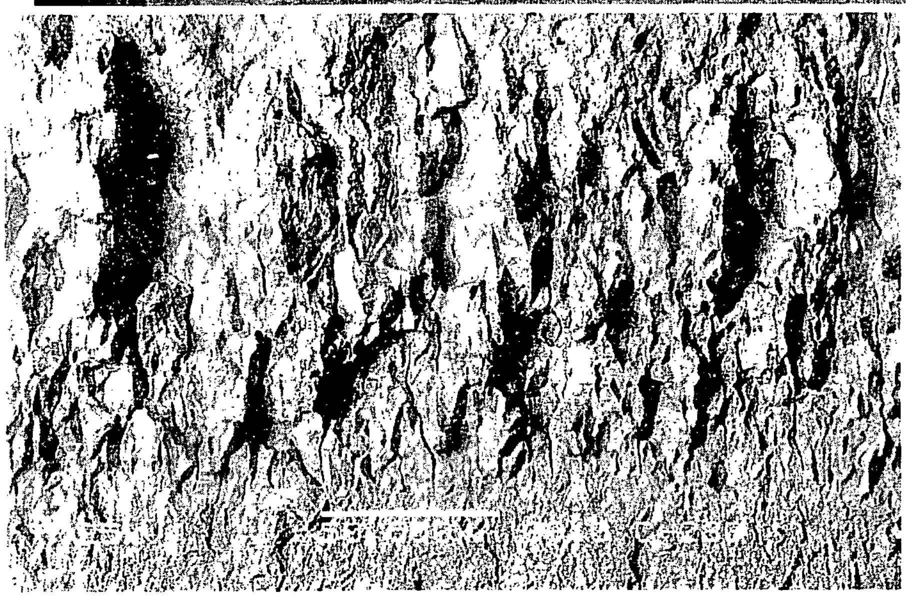
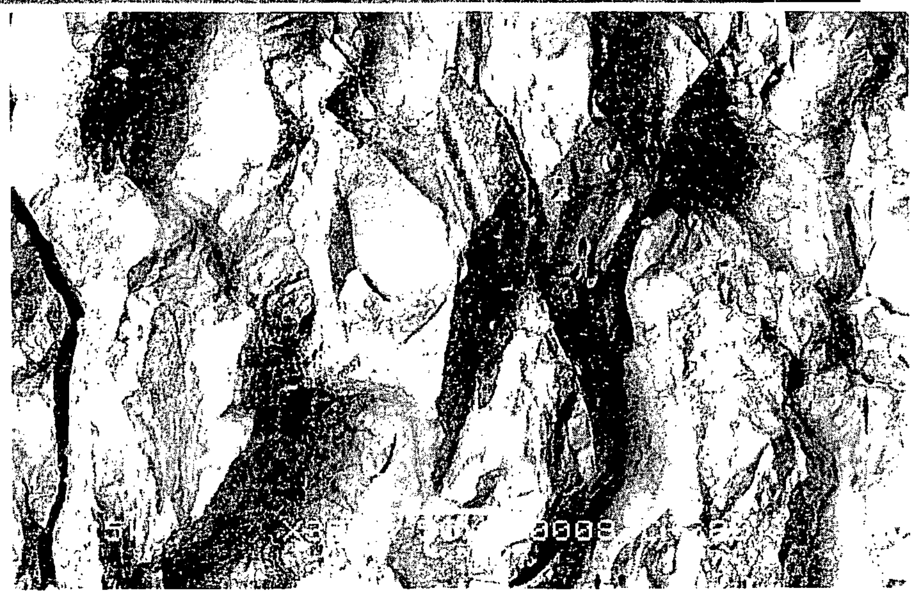


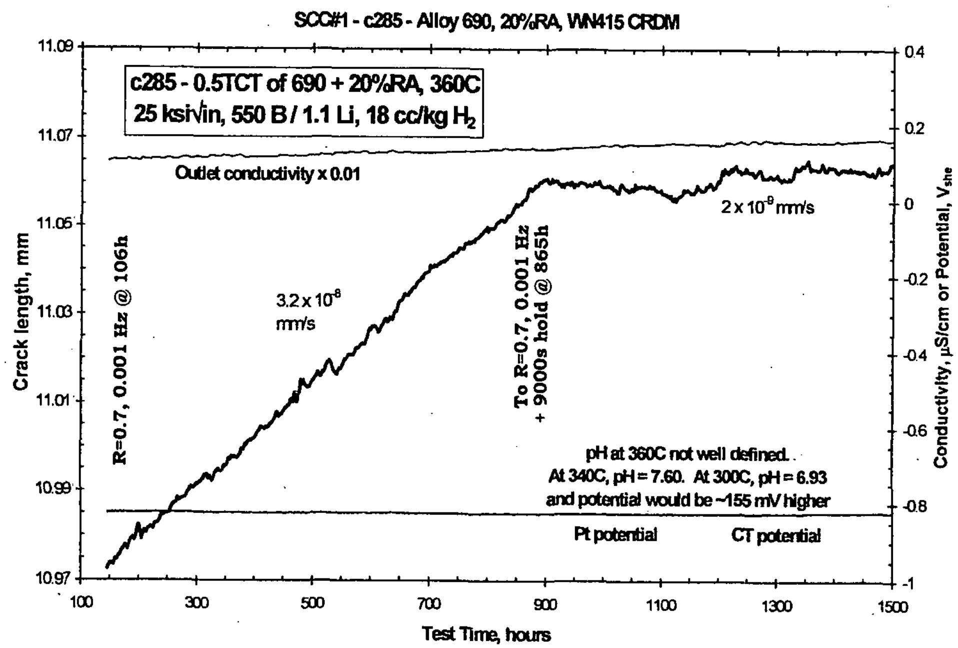
GE tests at Constant & Varying K (dK/da)

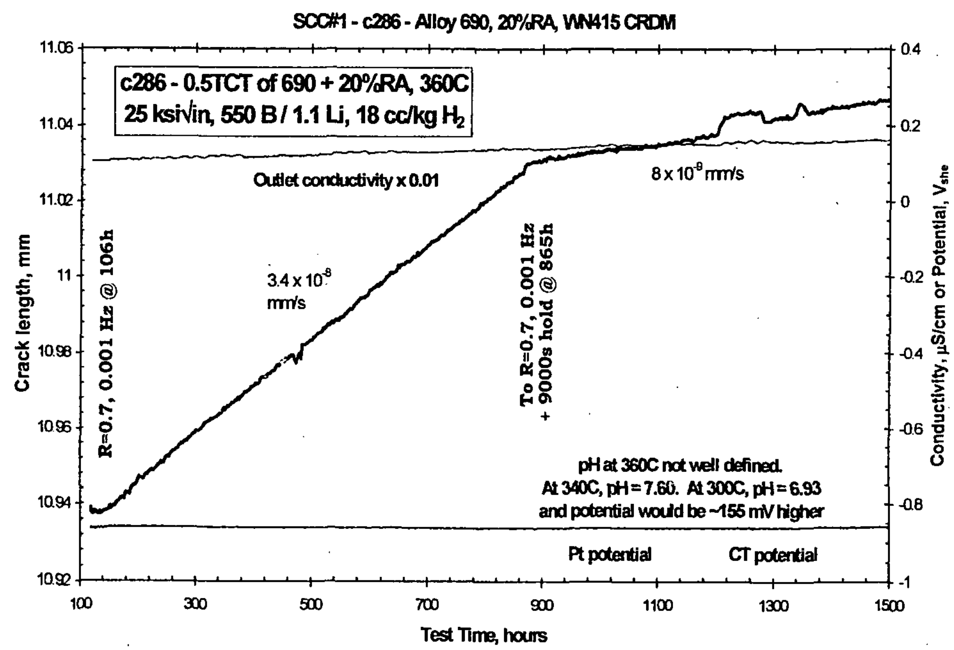
41% Cold Work Alloy 690 CRDM





20% Cold Work Alloy 690 CRDM

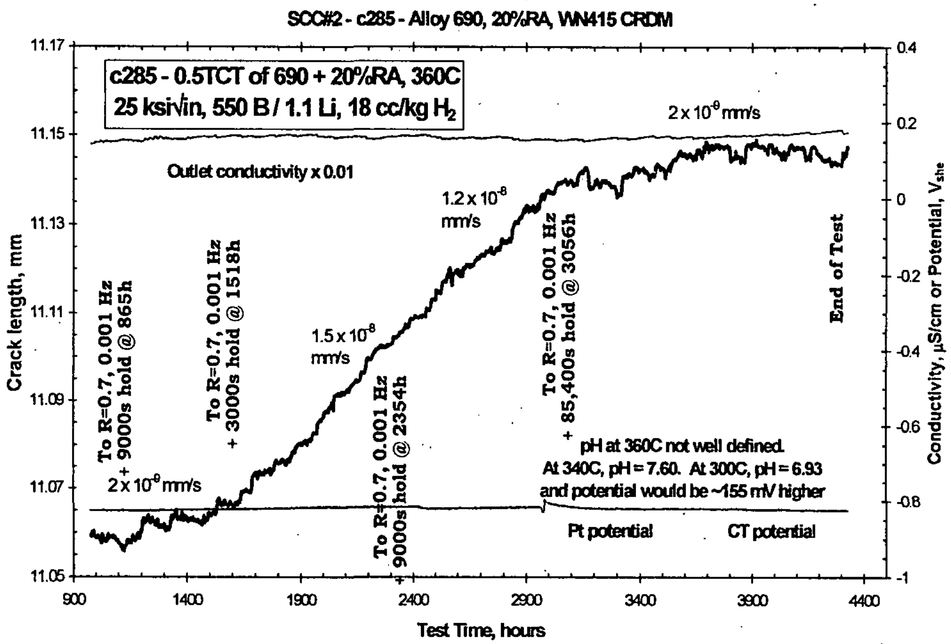


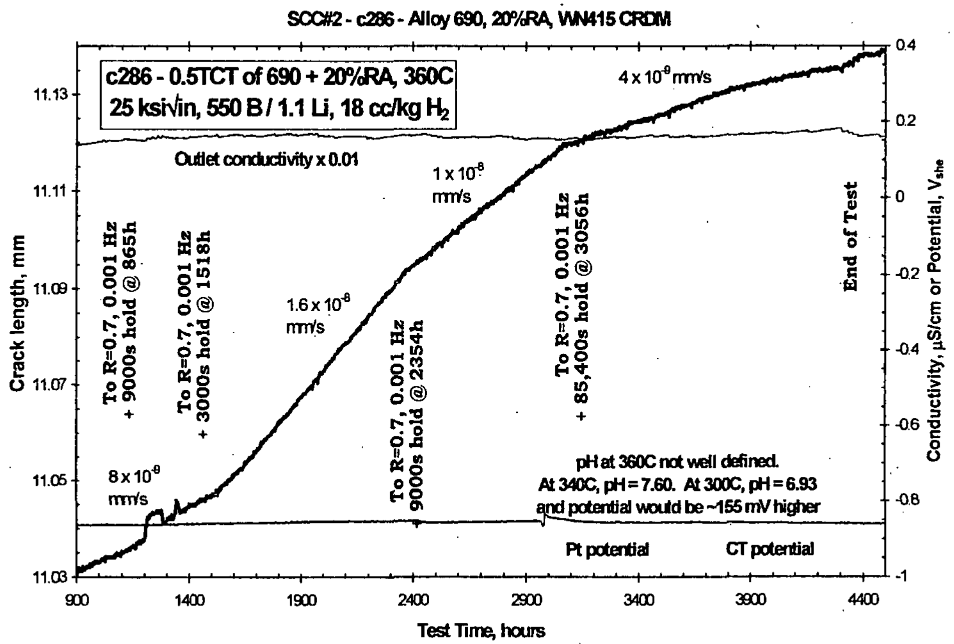


EPRI Program – Constant K_{max}

SCC of Alloy 690

20% Cold Work Alloy 690 CRDM

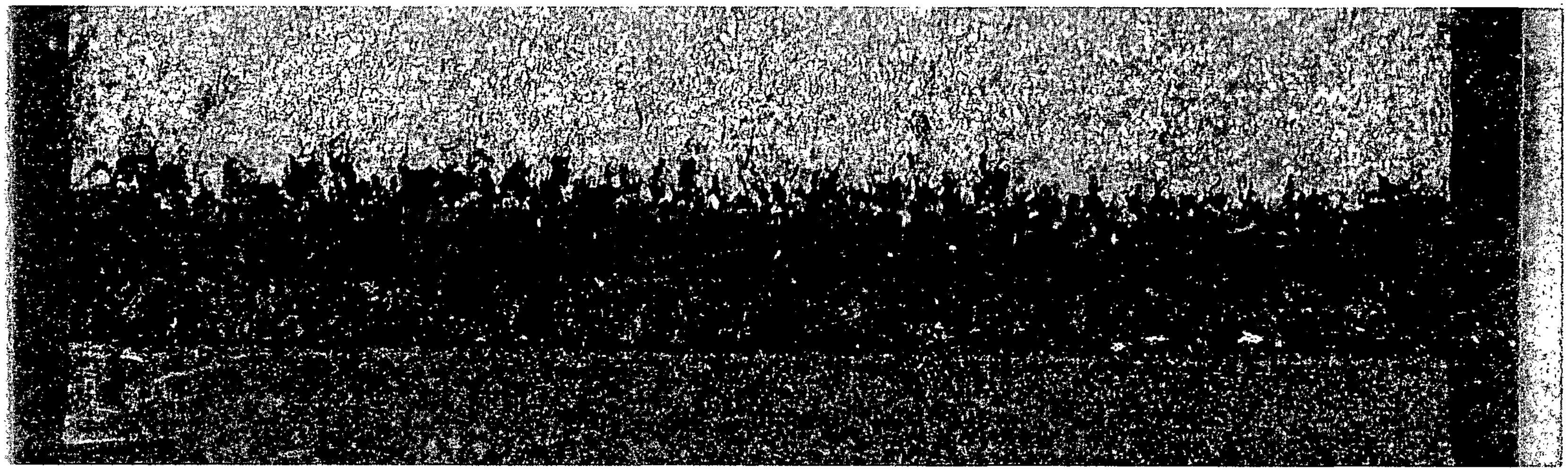




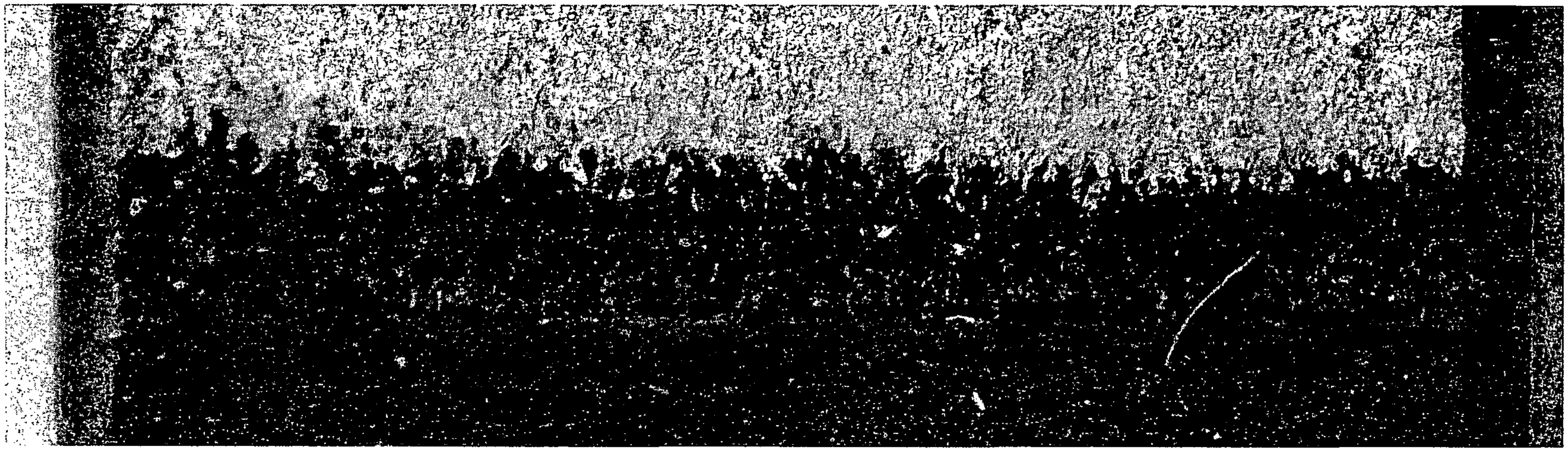
EPRI Program – Constant K_{max}

20% Cold Work Alloy 690 CRDM

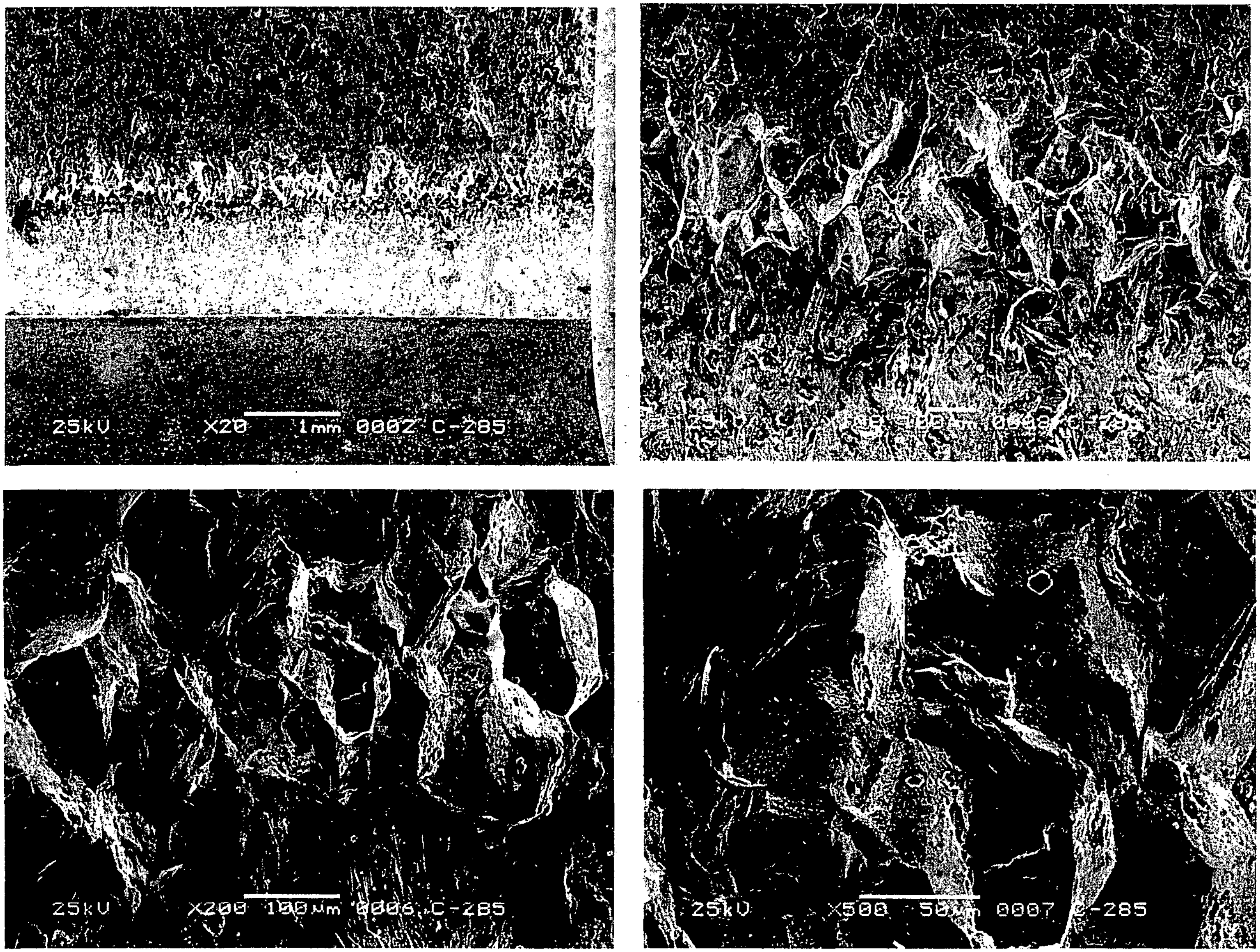
c285



c286

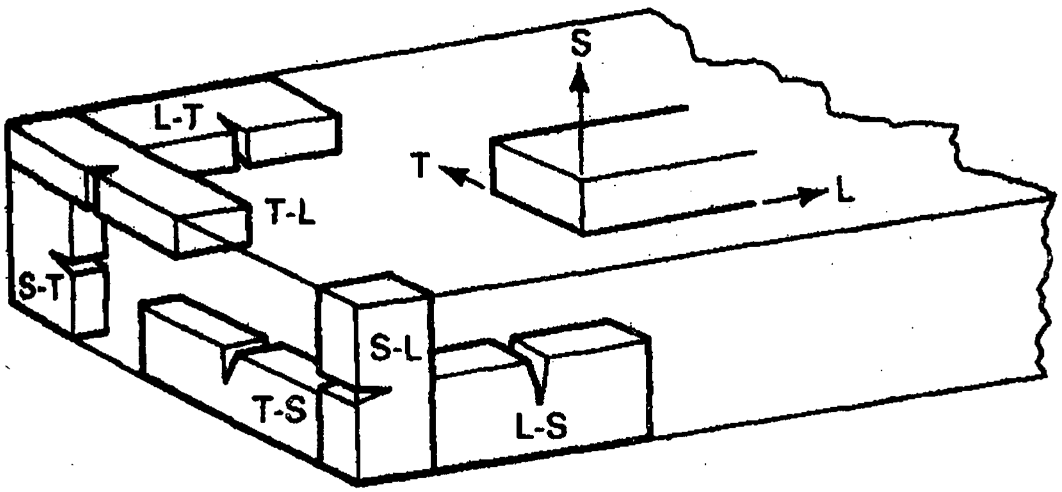


c285/c286 SEM Fractography

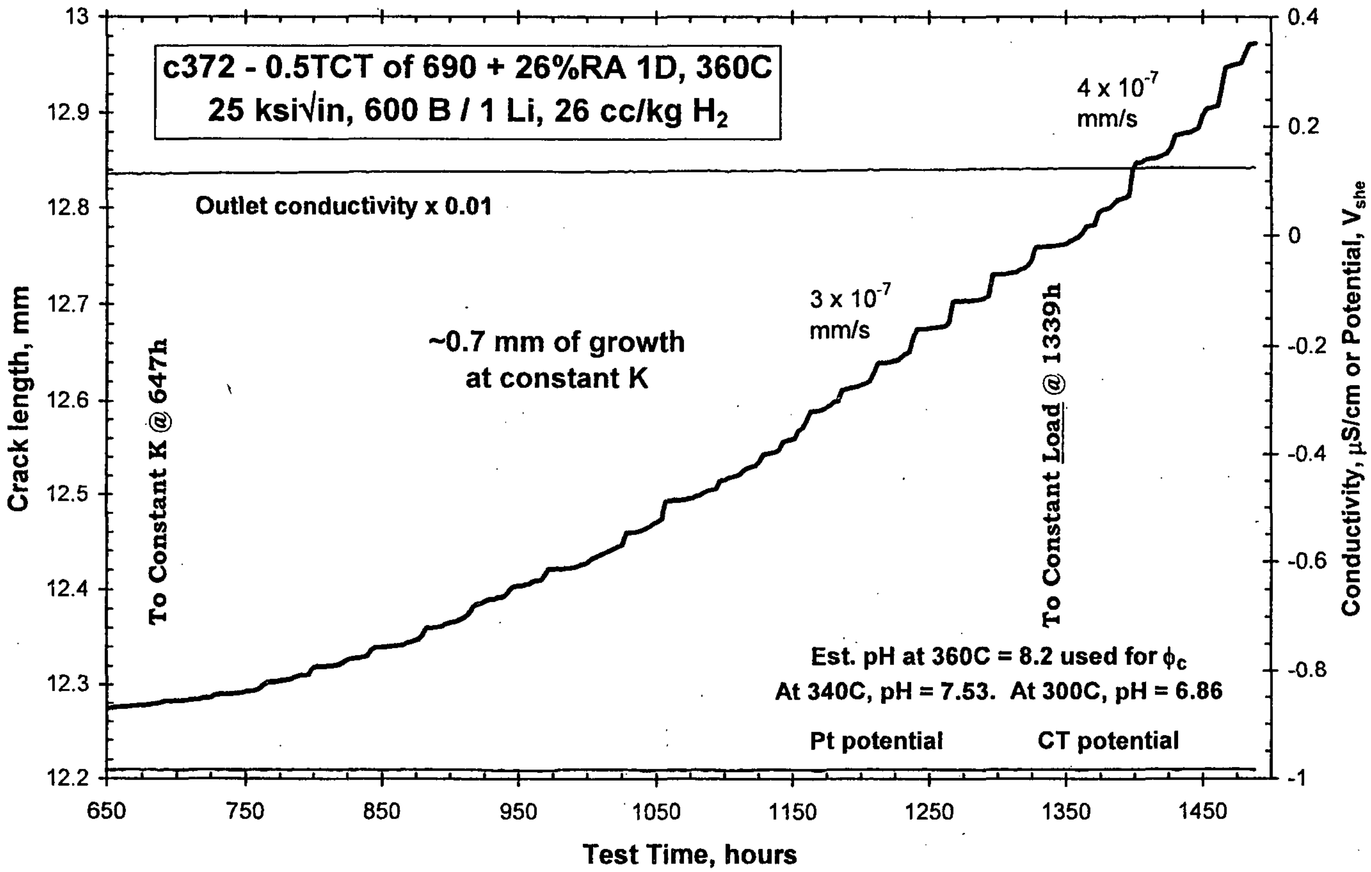


Testing on 1D Cold Rolled 690

- Evaluation of two 0.5T CT specimens of Alloy 690:
  - cold worked alloy 690 by 1D rolling by 20 – 26%
  - use worst S-L orientation: crack plane = rolling plane
  - tested near peak in CGR (near Ni/NiO transition)
  - tested at 360C to accelerate testing
  - used periodic "gentle" cyclic loading to activate SCC
- Observed increased growth rates at constant K

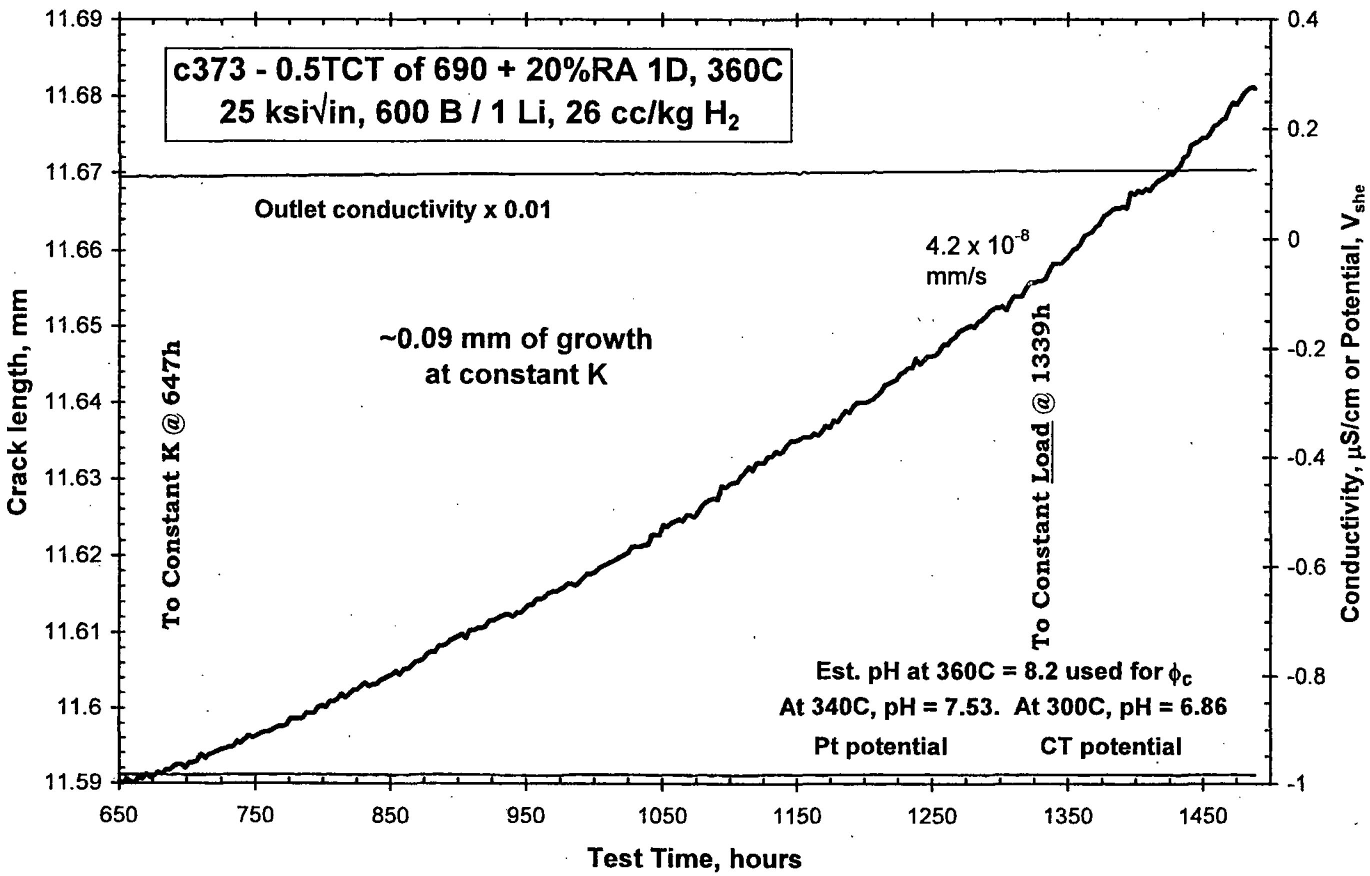


1D, 26% Cold Worked Alloy 690



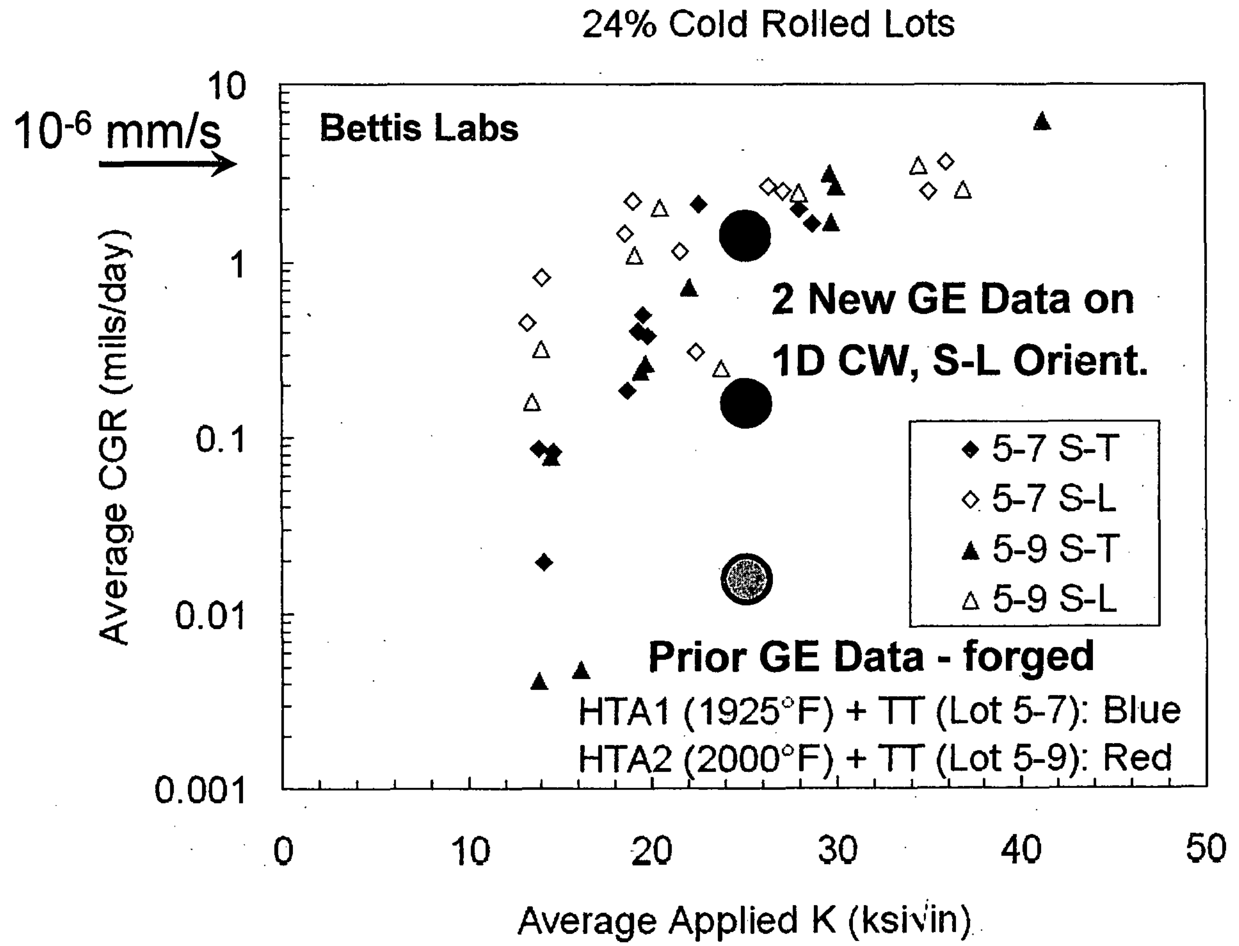
Increased growth rates in S-L orientation

1D, 20% Cold Worked Alloy 690



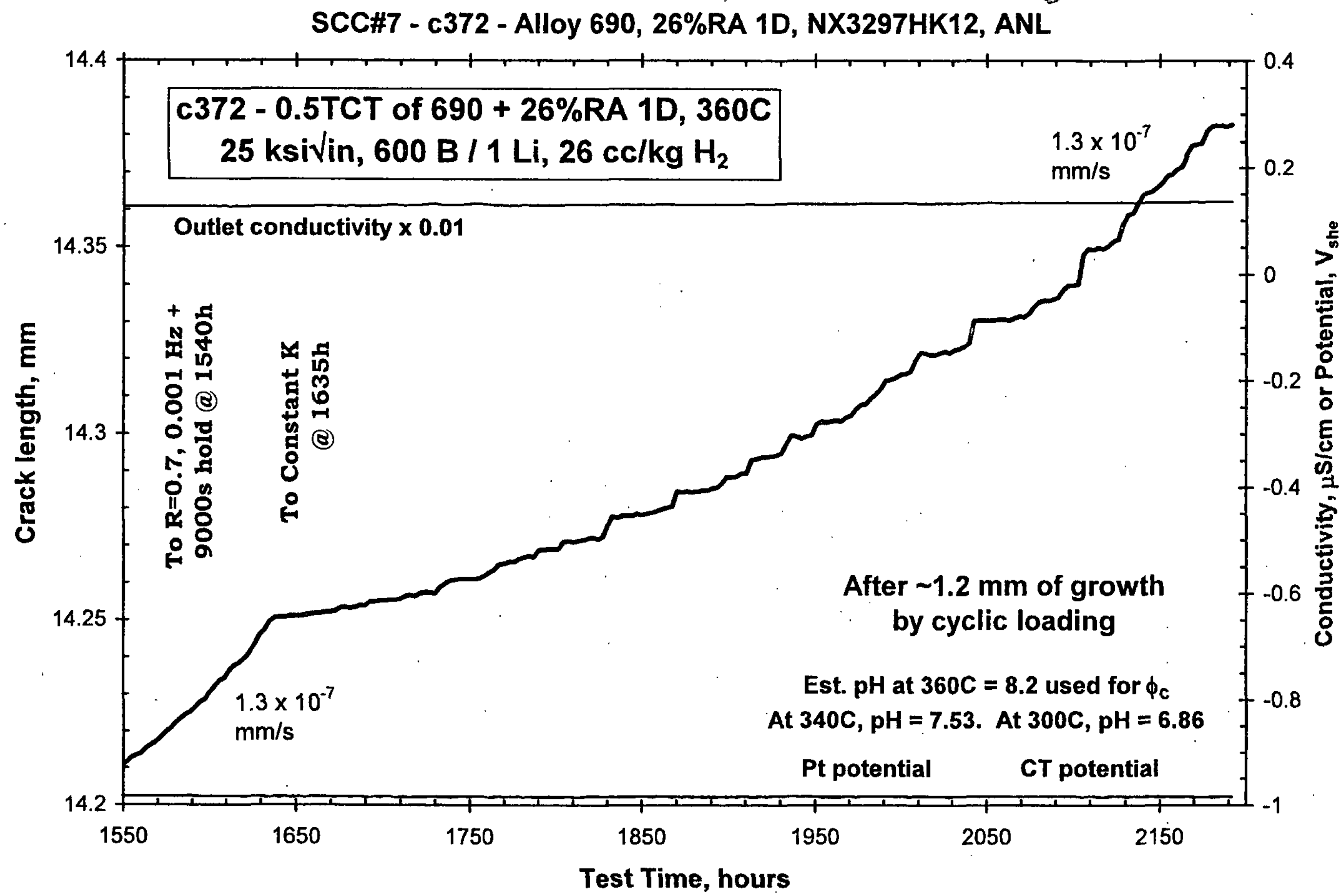
Increased growth rates in S-L orientation

Comparison of GE & Bettis Data



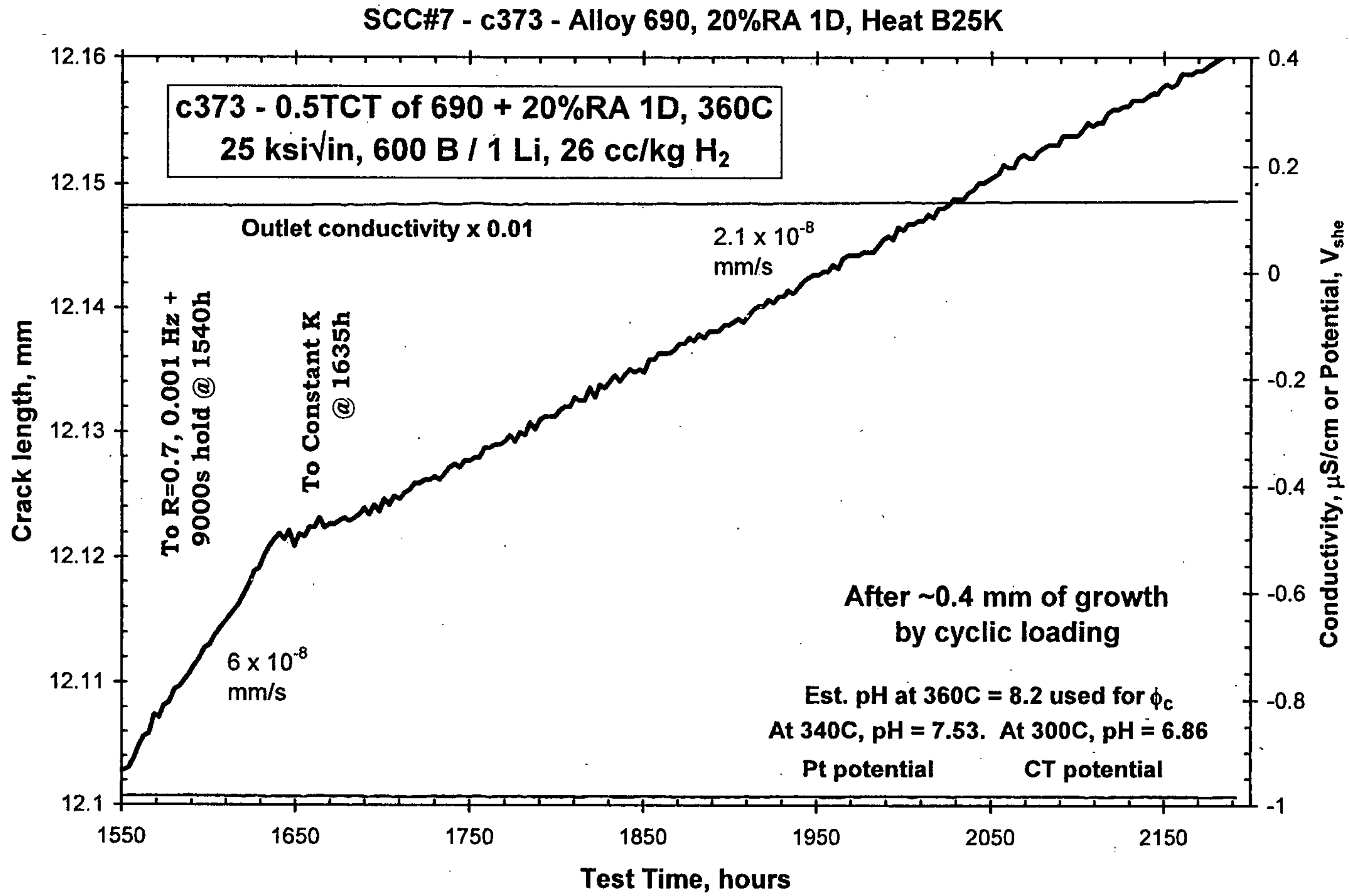
Increased growth rates in S-L orientation

1D, 26% Cold Worked Alloy 690



Somewhat lower CGR after fatigue crack advance

1D, 20% Cold Worked Alloy 690



Somewhat lower CGR after fatigue crack advance

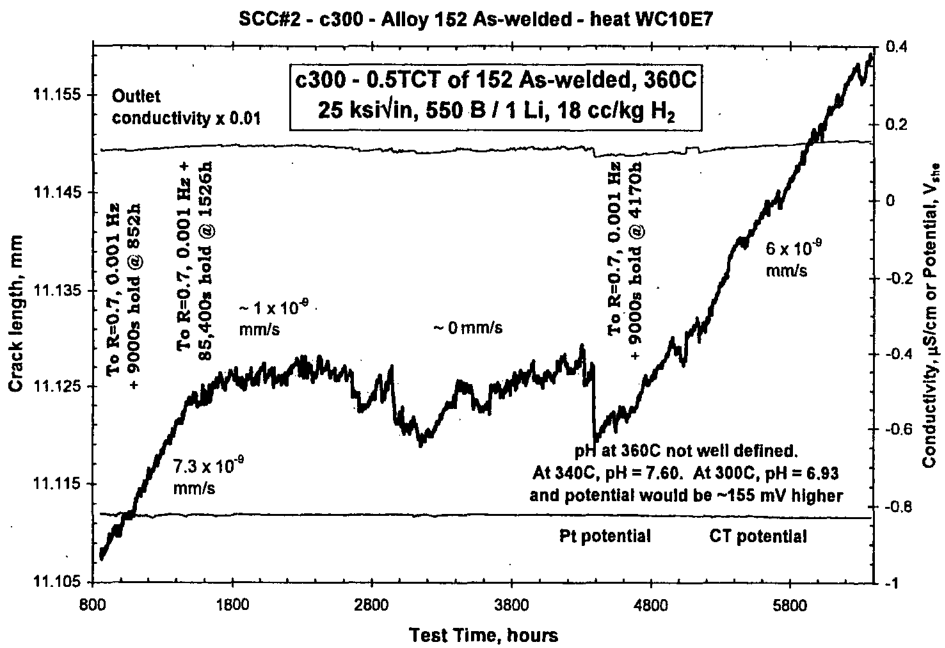
Summary of EPRI Alloy 690 Results

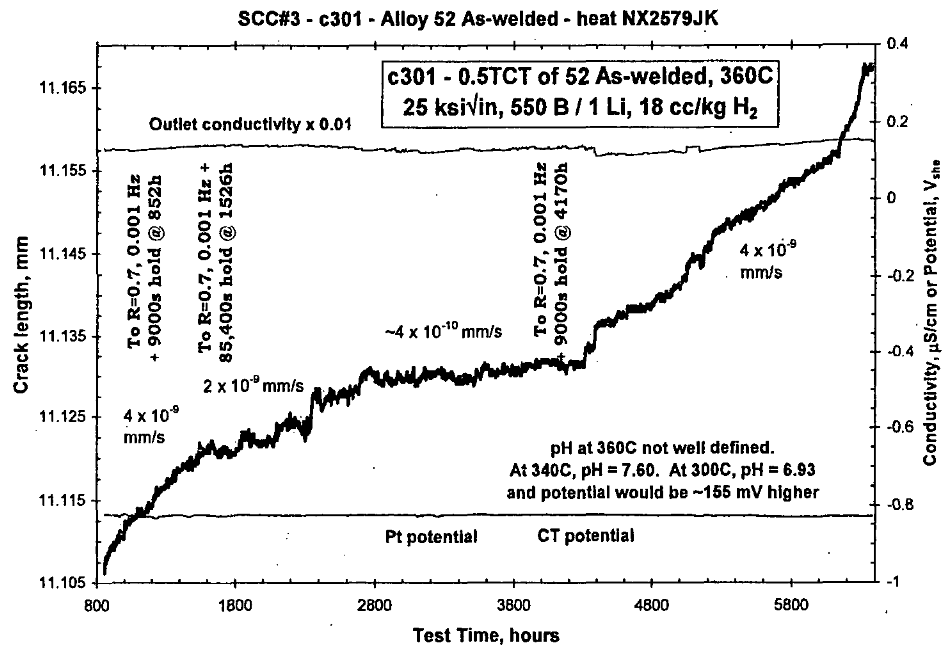
- Crack growth is broadly consistent with other Alloy 690 specimens – some, but slow, SCC growth.
- Much higher growth rates in 1-dimensional cold rolled material with crack plane = rolling plane (S-L orientation)
- Difficulty in sustaining growth at longer hold times.
- SEM exam showed strong evidence of IG cracking.

Summary: Typical heats & microstructures of Alloy 690 are shown to be susceptible to IG SCC growth in primary water, although growth rates are very low.

Vulnerabilities must be probed and understood, including weld heat affected zones, off-microstructures & cold work.

Alloy 152 & 52 Weld Metal

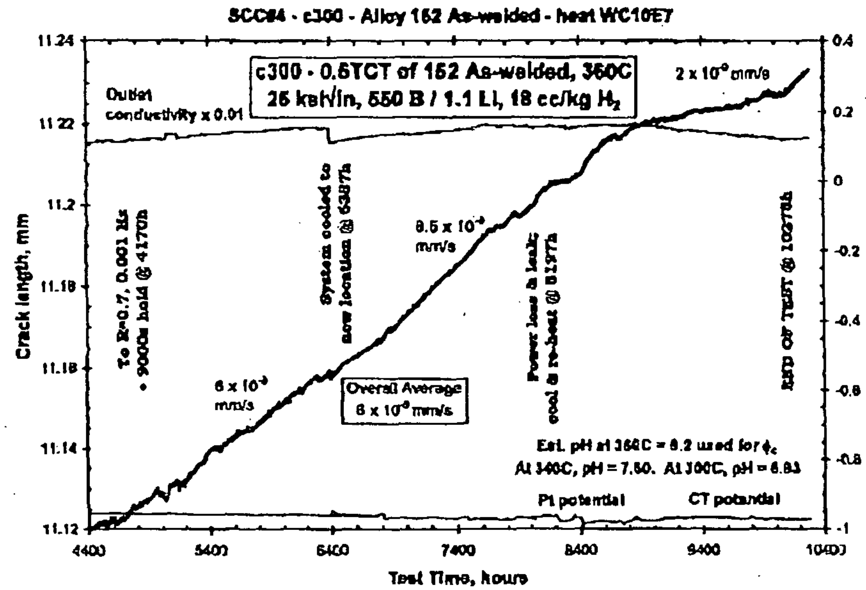


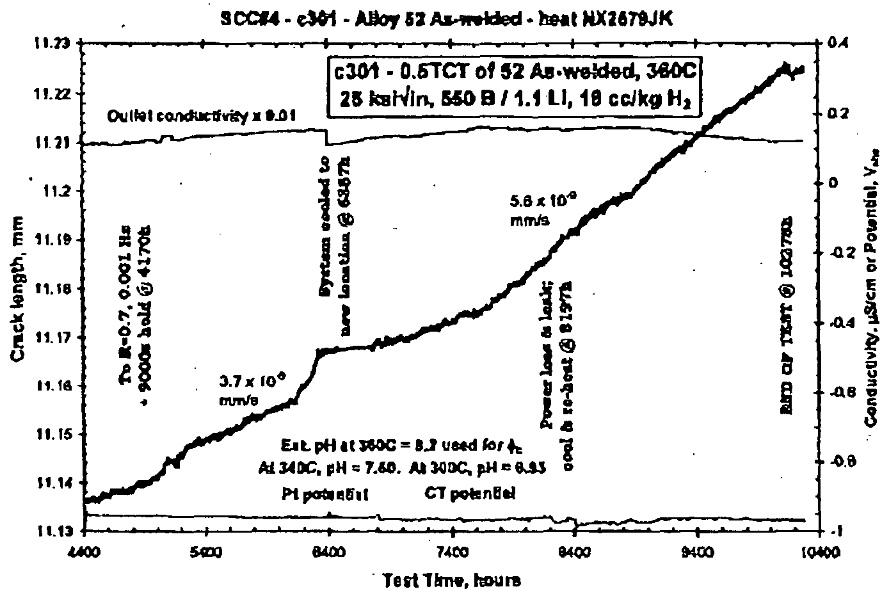


c300 (alloy 152) *c301 (alloy 52)*

EPRI Program – Constant K_{max} + Cycling

Alloy 152 & 52 Weld Metal

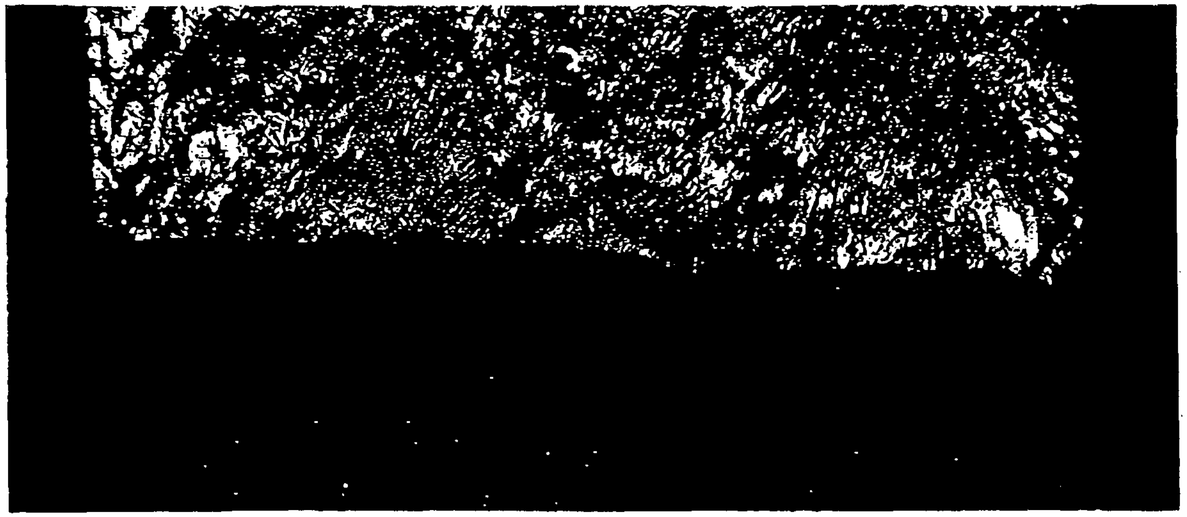




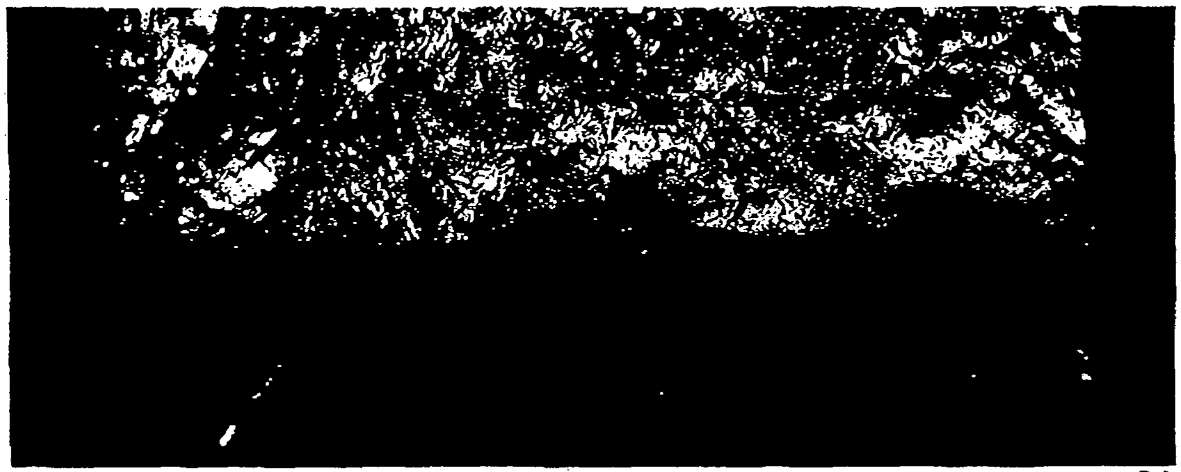
c300 (alloy 152) c301 (alloy 52)

EPRI Program – Constant K_{max} + Cycling

Alloy 152 & 52 Weld Metal

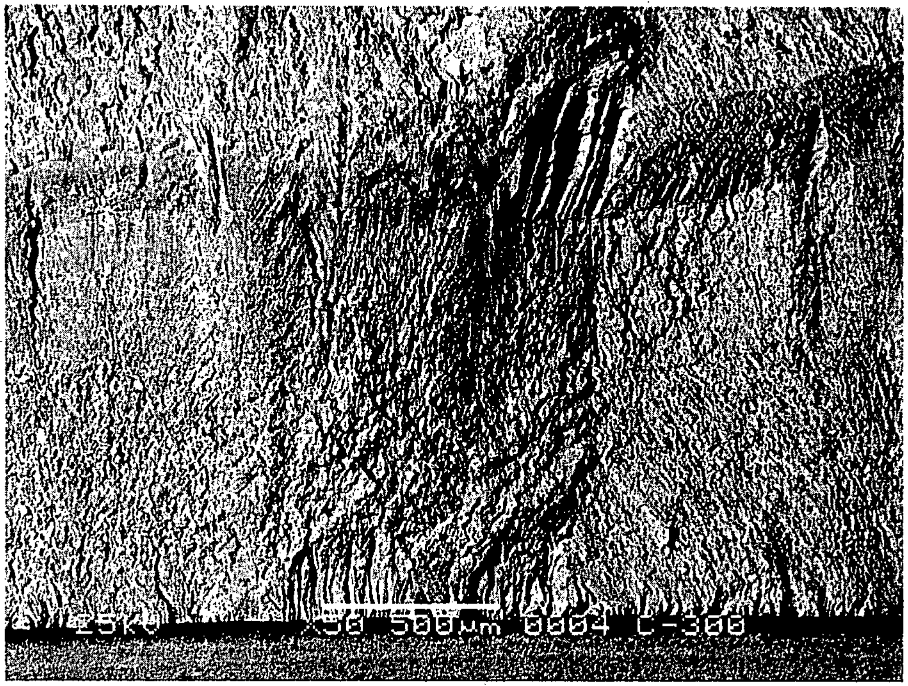


c300 (alloy 152)

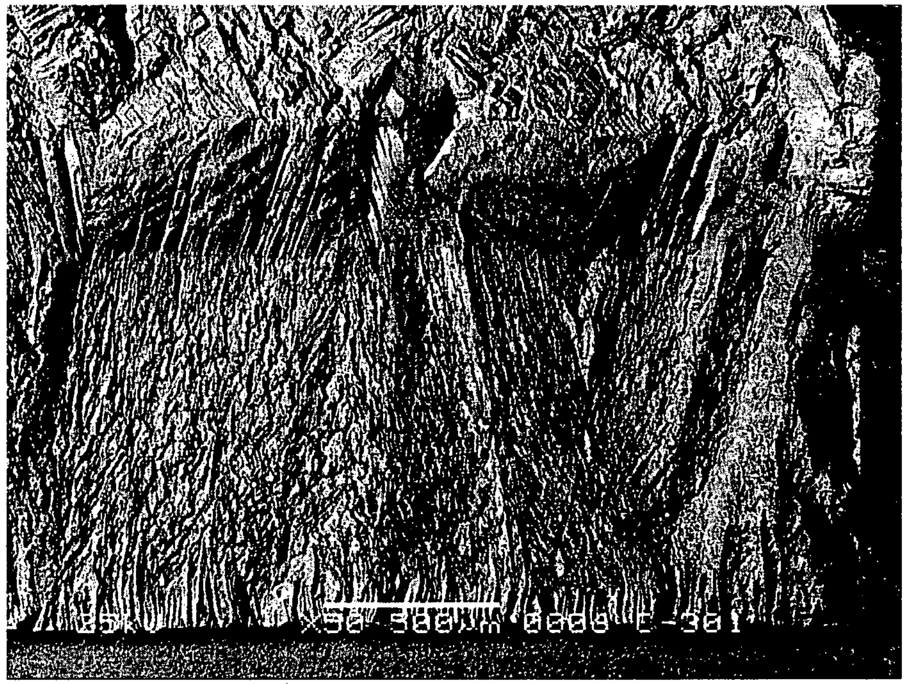


c301 (alloy 52)

Alloy 152 & 52 Weld Metal



c300 (alloy 152)

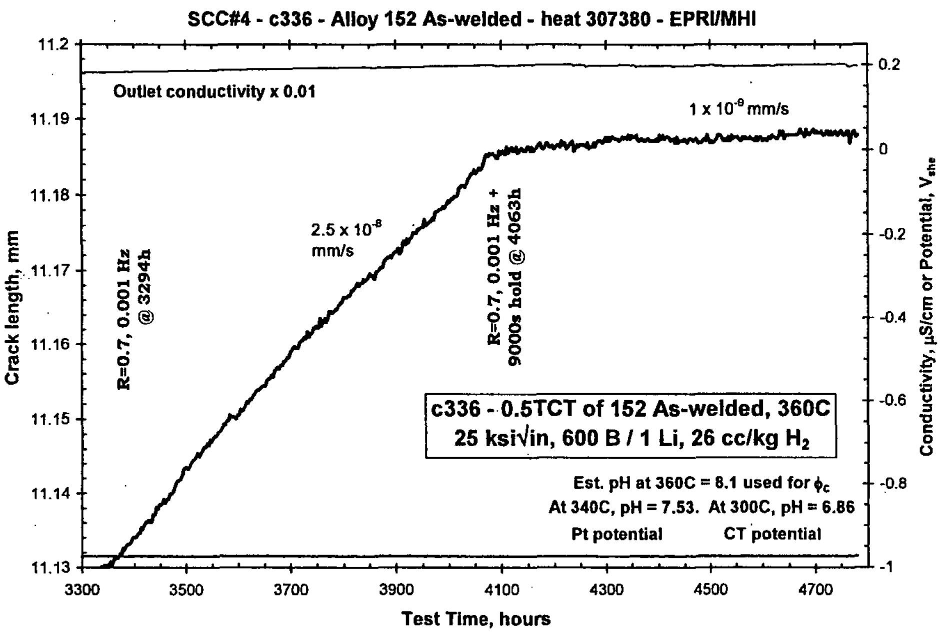


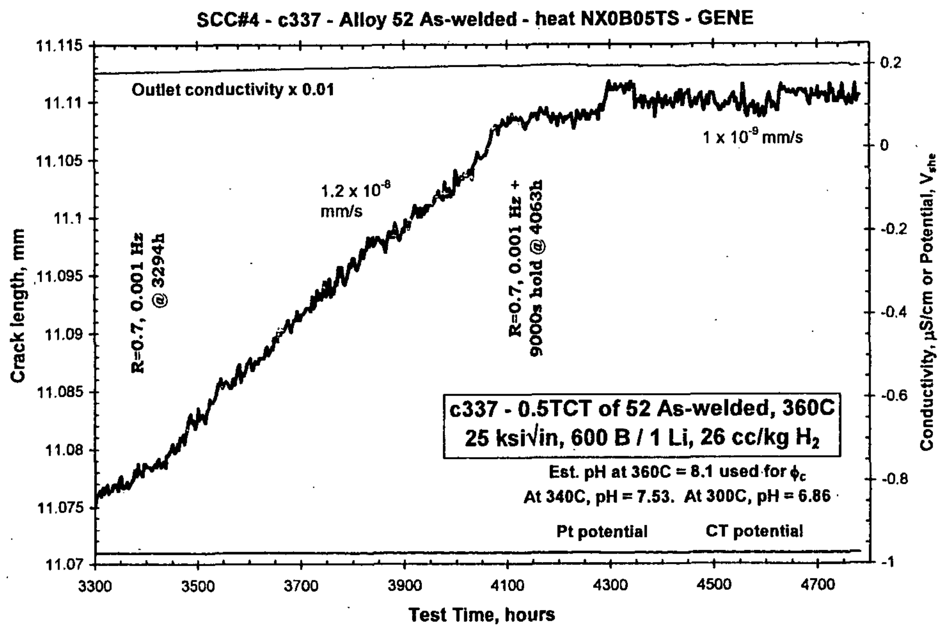
c301 (alloy 52)

EPRI Program – Constant K_{max} + Cycling

Plan to shift to 85,400 s hold & then constant K

Alloy 152 & 52 Weld Metal





c336 – 2nd Alloy 152 c337 = 2nd Alloy 52

Growth rates are very low

Summary for Alloy 152 / 52 Weld Metal

- *Evaluation of Alloy 152 and 52 weld metal indicates similar susceptibility to that observed in Alloy 690:*
  - *prototypical heats and welding processes*
  - *sustained growth is difficult at long hold times*
  - *no major difference between Alloys 152 and 52*
- *Must await post-test fractography to confirm response, cracking morphology, and growth rates.*

Vulnerabilities must be probed and understood, including weld heat affected zones, off-microstructures & cold work.

Conclusions

Results obtained to date under accelerated conditions show:

- *slow crack growth at constant K appears to occur in some (but not all) 2D CW Alloy 690, & Alloys 152/52 welds*
- *increased growth rates at constant K in 1-D cold rolled Alloy 690 with crack plane = rolling plane (S-L orientation)*
- *rising dK/da loading shows somewhat higher CGRs and may be relevant in certain field situations*
- *truly <u>intergranular crack propagation</u> has been demonstrated for Alloy 690 base materials*

Future work should examine:

- *possibility of increased PWSCC susceptibility in HAZ*
- *PWSCC in alternate cold work orientations*
- *effect of "off-microstructures" from material processing*