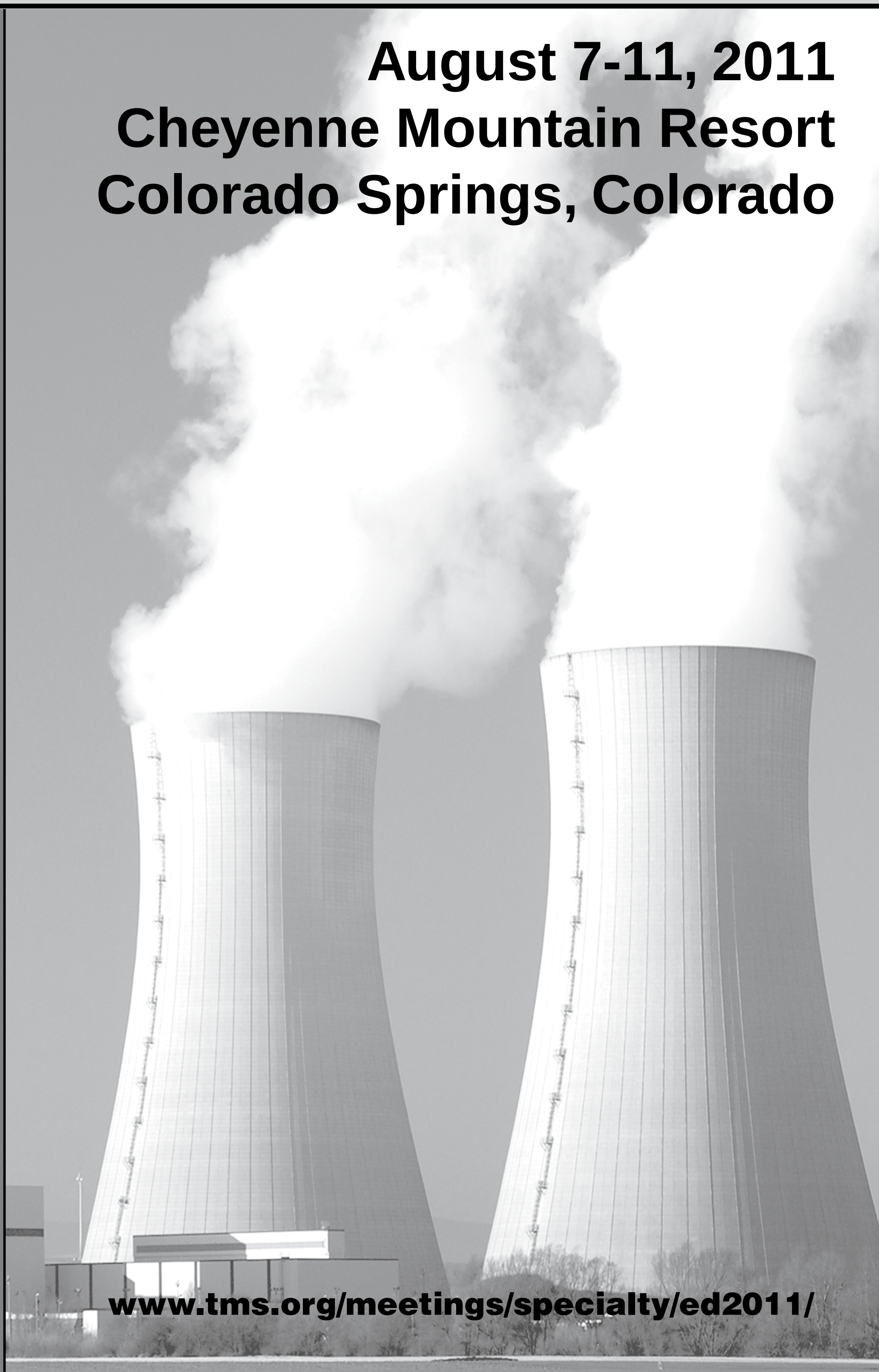
15th International Conference on

ENVIRONMENTAL DEGRADATION

of Materials in Nuclear Power Systems-Water Reactors

August 7-11, 2011
Cheyenne Mountain Resort
Colorado Springs, Colorado

FINAL PROGRAM



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15th International Conference on

ENVIRONMENTAL DEGRADATION

of Materials in Nuclear Power Systems-Water Reactors

WELCOME to the 15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems—Water Reactors!

You are part of a highly beneficial and globally influential forum where new insights into materials problems, new methods, and innovative techniques from an international perspective will be explored.

Your Registration Fee Includes

- Admission to all technical sessions
- Refreshment and coffee breaks
- Single copy of conference proceedings
- All networking and social events

ORGANIZERS

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Jeremy T. Busby
Oak Ridge National Laboratory
Oak Ridge, Tennessee

Technical Program Chair:

Gabriel Ilevbare
Electric Power Research Institute
Palo Alto, California

Asst. Technical Program Chair:

Peter Andresen
General Electric
Schenectady, New York

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ABOUT THE CONFERENCE

Registration Information

The on-site registration desk in the Colorado Lobby on the lower level of the Cheyenne Mountain Resort will be open during the following hours:

Sunday, August 7	4 to 8 p.m.
Monday, August 8	7:30 a.m. to 6:30 p.m.
Tuesday, August 9	7:30 a.m. to 11 a.m. & 6 to 7:30 p.m.
Wednesday, August 10	7:30 a.m. to 5 p.m.
Thursday, August 11	7:30 a.m. to 4 p.m.

Technical Sessions

All conference rooms are located on the lower level of the resort. The general session will be held in the Colorado Ballroom. Concurrent sessions will be presented in Colorado I, Colorado II and Colorado III.

Opening Session
Monday, August 8
Time: 8:15 to 10:00 a.m.

Computer/Network Facilities

Wireless access is complimentary in sleeping rooms and public spaces. There is limited access in the meeting rooms.

Speakers:

Jack Bailey
TVA, USA
8:35 to 9:05 a.m.

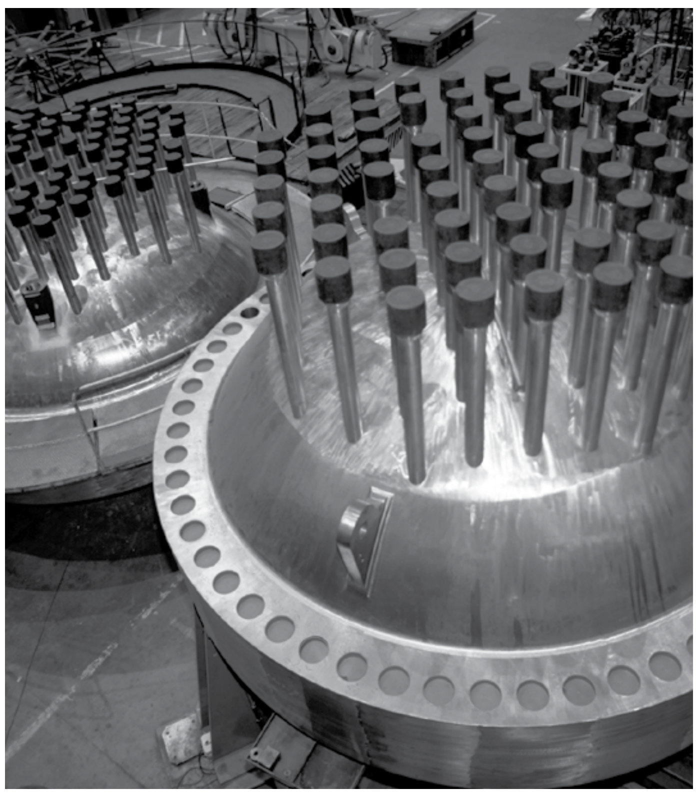
Pete Lyons
Department of Energy - Nuclear Energy, USA
9:05 to 9:35 a.m.

C. E. (Gene) Carpenter
U.S. Nuclear Regulatory Commission, USA
9:35 to 9:55 a.m.

Peter Ekström
Swedish Radiation Safety Authority, Sweden
9:55 to 10:15 a.m.

PROCEEDINGS

The conference proceedings will be published after the meeting as a CD-ROM. One copy of the proceedings will be shipped to each paid registrant when the CD is available. Additional copies of the proceedings can be purchased via the registration form at a cost of $54 each, including shipping and handling.



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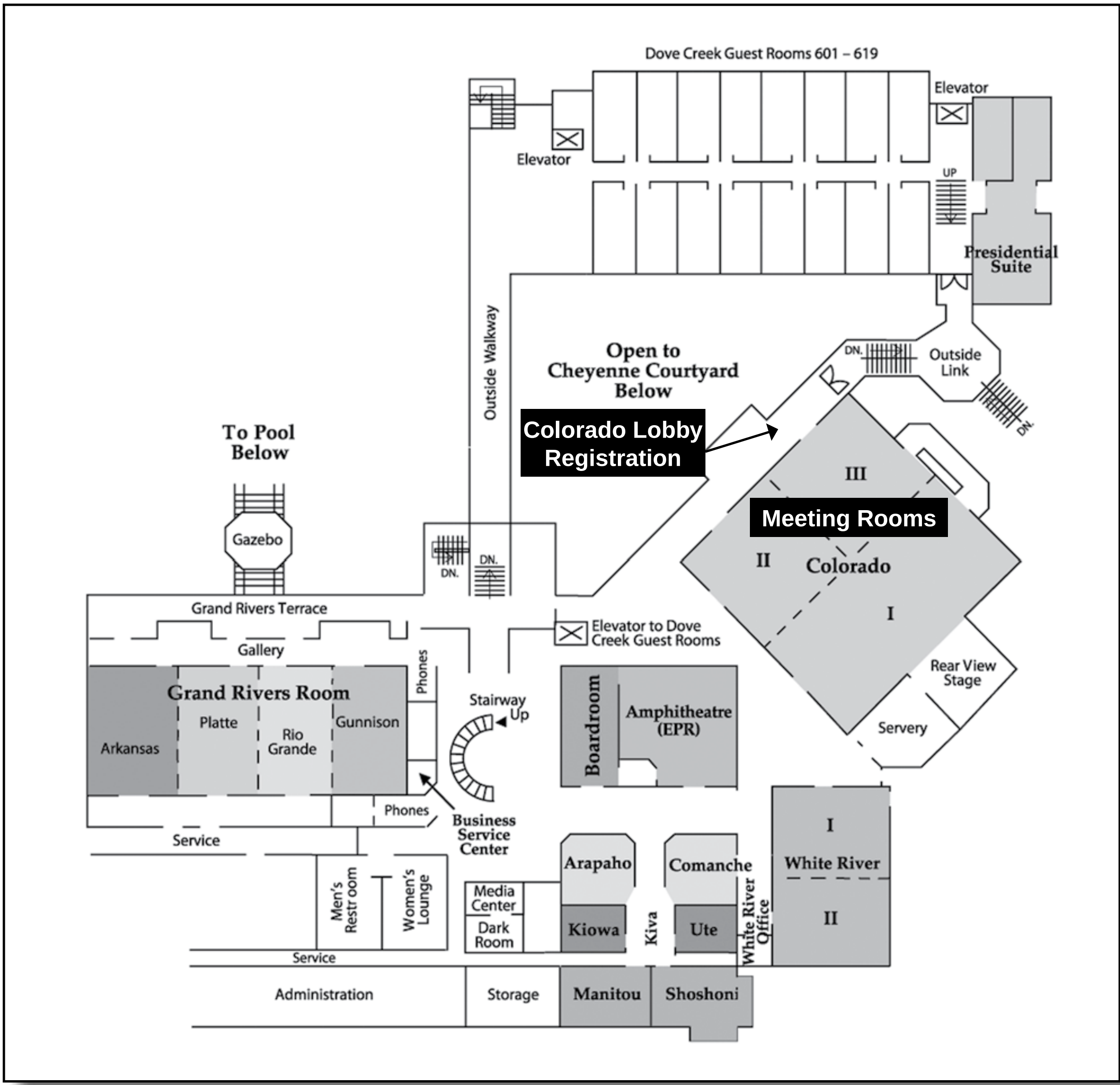

NETWORKING & SOCIAL EVENTS

Sunday, August 7
Welcoming Reception
Cheyenne Mountain Resort 6 to 8 p.m.

Tuesday Night Reception, August 9
Cheyenne Mountain Resort 6:45 to 7:30 p.m.

Wednesday, August 10
Cheyenne Mountain Zoo 6:30 to 8:45 p.m.

Buses leave the Cheyenne Mountain Resort at 6 and 6:15 p.m.

Buses return from the Cheyenne Mountain Zoo at 8:30 and 8:45 p.m.

This relaxed gathering will begin with a reception and giraffe feeding, followed by a dinner buffet. Feel free to experience the zoo at your leisure throughout the evening.

General Information

About the Conference Location

Cheyenne Mountain Resort is surrounded by extraordinary mountain backdrops and picturesque views. Each room has been designed to provide the ultimate escape. The resort features a wide range of year-round activities, including swimming, golf, tennis, basketball and a full service, state-of-the-art fitness center. Jogging, hiking, and biking paths are also available.

Policies

Americans With Disabilities Act

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Opening Session

Monday AM
August 8, 2011

Room: Colorado Ballroom
Location: Cheyenne Mountain Resort

Session Chair: Jeremy Busby, Oak Ridge National Laboratory

8:15 AM Introductory Comments

8:20 AM Opening Plenary Session
Featured Speakers:
Dr. Peter Lyons, Acting Assistant Secretary for Nuclear Energy and former Nuclear Regulatory Commission Chairmen Jack Bailey, Vice President, Nuclear Generation Development, TVA Nuclear

9:20 AM
Overview of NRC Proactive Management of Materials Degradation (PMMD) Program: *C. E. Carpenter*[1]; Amy Hull[1]; Greg Oberson[1]; [1]U.S. Nuclear Regulatory Commission

Materials degradation phenomena, if not appropriately managed, have the potential to adversely impact the design functionality and safety margins of nuclear power plant (NPP) systems, structures and components (SSCs). Therefore, the U.S. Nuclear Regulatory Commission (NRC) has initiated an over-the-horizon multi-year research Proactive Management of Materials Degradation (PMMD) Research Program, which is presently evaluating longer time frames (i.e., 80 or more years) and including passive long-lived SSCs beyond the primary piping and core internals, such as concrete containment and cable insulation. This will allow the NRC to (1) identify significant knowledge gaps and new forms of degradation; (2) capture current knowledge base; and, (3) prioritize materials degradation research needs and directions for future efforts. This effort is being accomplished in collaboration with the U.S. Department of Energy's (DOE) LWR Sustainability (LWRS) program. This presentation will discuss the activities to date, including results, and the path forward.

9:40 AM
Conditions for Long Term Operation of Nuclear Power Plants in Sweden: *Peter Ekström*[1]; Karen Gott[1]; Björn Brickstad[1]; [1]Swedish Radiation Safety Authority

The Swedish reactor fleet consists of 7 BWRs and 3 PWRs which have been operating for up to 38 years and all have plans for long term operation (LTO). SSM has carried out an investigation to identify possible improvements to assessments for safe LTO. The study covered ageing mechanisms for metallic materials and polymers, concrete structures, electrical and I&C equipment, and in service inspection. The paper will concentrate on metallic materials. Most of the degradation mechanisms are controlled satisfactorily by the licensees through the existing inspection and ageing management programs. These should be intensified and reviewed for LTO. The consequences of ageing degradation mechanisms for LTO should be analyzed and reported. Two ageing mechanisms are highlighted, low cycle fatigue in LWR environments and embrittlement of RPV steels, but analyses should be carried out for most of the ageing mechanisms to find early indications of degradation and thus ensure safety.

10:00 AM Break

Alloy 690 and Its Weld Metals I

Monday AM
August 8, 2011

Room: Colorado I
Location: Cheyenne Mountain Resort

Session Chairs: George Young, Knolls Atomic Power Laboratory; Peter L. Andresen, GE Global Research

10:10 AM
Current NRC Perspectives Concerning PWSCC: *David Alley*[1]; Darrell Dunn[1]; [1]U. S. Nuclear Regulatory Commission

Materials currently used in nuclear power plants are reliable and are generally resistant to environmental degradation. However, occurrences of environmental degradation have been observed as the current fleet of reactors age. Primary water stress corrosion cracking (PWSCC) is of particular interest to the US Nuclear Regulatory Commission (NRC). This paper provides a historical assessment of operating experience associated with PWSCC and welding issues associated with PWSCC materials. This paper also provides a brief description of NRC research concerning PWSCC. Finally the paper considers the regulatory issues associated with PWSCC, especially those associated with gaps in the understanding of the behavior of PWSCC resistant material under actual reactor conditions.

10:30 AM
Evaluation of the Susceptibility to SCC Initiation of Alloy 690 in Simulated PWR Primary Water: Thierry Couvant[1]; *Kazuya Tsutsumi*[2]; [1]EDF R&D; [2]Mitsubishi Heavy Industries, Ltd.

Alloy 690 has been widely used to manufacture components of LWR plants as alternative material of Alloy 600 which exhibited a significant susceptibility to PWSCC. However, some authors have reported that Alloy 690 can suffer a significant susceptibility to SCC crack growth when highly cold worked. Despite that most of the recent studies emphasize SCC propagation stage, EDF and its partners consider the material's resistance to SCC initiation. This paper, summarized the current work carried out at EDF MAI, dedicated to SCC initiation stage. The first task addresses the evaluation of the susceptibility to IGSCC depending on material's manufacturing process. The second task is to evaluate the effect of the dissolved hydrogen concentration and ageing on the susceptibility to initiate IGSCC in a pre-industrial Steam Generator tube. The third task is to investigate the effect of cold work (1D cold rolling;13% and 25%) on the susceptibility to SCC initiation.

10:50 AM
Role of Cavity Formation on IGSCC Initiation on CW Carbon Steel and Alloy 690: *Koji Arioka*[1]; Tomoki Miyamoto[1]; Takuyou Yamada[1]; Takumi Terachi[1]; [1]Institute of Nuclear Safety System

Crack initiation behavior was examined at hydrogenated high temperature water and gas environment using blunt notched CT specimen with smooth surface. Then, cavity formation behavior was examined using specimens after test. Based on the results, as one of processes in SCC initiation, the role of cavity formation on IGSCC initiation will be discussed concerning cold worked carbon steel, Alloy 690, and Alloy 600 in hydrogenated high temperature water.

11:10 AM
Stress Corrosion Crack Growth Testing of Cold-Worked Alloy 690 in PWR Primary Coolant: David Tice[1]; Stuart Medway[1]; *Norman Platts*[1]; [1]Serco

Plant experience of Alloy 690 in PWR primary coolant environments has been excellent, both for SG tubing applications and for thick section components. Although most laboratory corrosion studies are consistent with this observation, some recent data indicate that sustained crack propagation due to intergranular stress corrosion cracking can occur under constant applied load in some heats of thick section Alloy 690 which have been subjected to inhomogeneous cold work. Significant directionality in cracking response relative to the direction of cold work is observed for the susceptible heats. The results of SCC growth tests on several heats of cold-worked Alloy 690 in PWR primary water environments will be presented, together with microstructural characterisation studies aimed at elucidating the possible reasons for observed effects of material composition, specimen orientation and microstructural inhomogeneity.

11:30 AM
One-Dimensional Cold Rolling Effects on Stress Corrosion Crack Growth in Alloy 690 Tubing and Plate Materials: *Mychailo Toloczko*[1]; Stephen Bruemmer[1]; [1]Pacific Northwest National Laboratory

Stress corrosion crack-growth experiments have been performed on cold-rolled alloy 690 materials in simulated PWR primary water at 360°C. Extruded alloy 690 CRDM tubing in two conditions, thermally treated (TT) and solution annealed (SA), was unidirectionally cold rolled (CR) to several reductions reaching a maximum of 31% and tested in the S-L orientation. High SCC propagation rates (~8x10-8 mm/s) were observed for the 31%CR alloy 690TT material, while the 31%CR alloy 690SA exhibited 10X lower rates. The difference in SCC susceptibility appears to be related to grain boundary carbide distribution before cold rolling. Tests were also performed on two CR alloy 690 plate heats (reductions of 26% or 20%) and SCC growth rates at 360°C were similar to that measured for the 31%CR alloy 690TT CRDM tubing. Comparisons will be made to other results on CR alloy 690 materials.

BWR Initiation and Oxide Film Characterization I

Monday AM — Room: Colorado II
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Renate Kilian, AREVA NP GmbH; Raul Rebak, GE Global Research

10:10 AM
In-situ and Ex-situ Oxide Characterization by Synchrotron X-ray (SPring-8) in Non-Sensitized 316 Stainless Steel and High Temperature Water Combination: *Toshio Yonezawa*[1]; Masashi Watanabe[1]; Takahisa Shobu[2]; Tetsuo Shoji[1]; [1]Tohoku University; [2]Japan Atomic Energy Agency

The in-situ and ex-situ oxide characterization was studied by synchrotron X-ray (SPring-8), to clarify the mechanism of stress corrosion cracking for a non-sensitized cold worked 316 stainless steel in high temperature water. The refreshed type auto-clave with diamond windows was originally designed and fabricated, to conduct the in-situ measurement by synchrotron X-ray. Notched specimens of a non-sensitized cold worked 316 stainless steel were loaded, exposed in simulated BWR water and measured by synchrotron X-ray under in-situ and ex-situ conditions. The oxide films and their layered structure near exposed surface were identified by fluorescence X-ray spectroscopy and local stress / strain beneath the oxide film were measured by X-ray diffraction.From these experimental results, the oxidation process for the non-sensitized cold worked 316 stainless steel in high temperature water was characterized, and new model about the corrosion process of non-sensitized cold worked 316 stainless steel in high temperature water was discussed.

10:30 AM
High Resolution Electron Microscopy Study on Oxide Films Formed on Nickel-Base Alloys X-750, 182 and 82 in Simulated High Flow Velocity BWR Water Conditions: *Jiaxin Chen*[1]; Fredrik Lindberg[2]; Lyubov Belova[3]; Björn Forssgren[4]; Karen Gott[5]; Johan Lejon[6]; Audrius Jasiulevicius[7]; [1]Studsvik Nuclear AB; [2]Swerea KIMAB AB; [3]Royal Institute of Technology; [4]Ringhals AB; [5]Swedish Iradiation Safety Authority; [6]OKG AB; [7]Vattenfall Nuclear Fuel AB

This work contributes to characterization of the oxide films formed on nickel-base alloys (Alloy X-750, Alloy 82 and Alloy 182) under simulated BWR water environments at ~10 or 18 m/s with or without iron injection. HR SEM/TEM and FIB techniques were applied. The oxide thicknesses on different alloys were substantially different, ranging from 50 nm to 8µm. For Alloy X-750 and Alloy 182 exposed without iron injection, similar oxide phase compositions consisting of sub-micron Fe2O3 and NiFe2O4 grains as well as NiO were formed but with substantially different microstructures. For the corroded Alloy X-750 there was an additional dense layer of possibly Ni1.5Cr0.5O3 in between the NiFe2O4 and NiO layers. On Alloy 82 which contained a relatively low Fe-content only a thin but dense film of Cr1.3Fe0.7O3 was seen. With iron injection the oxide films formed on Alloy 82 were similar to that on the Alloy 182 without iron injection, suggesting that iron injection may play a similar role as if the alloy had an elevated iron content. The implication of the observations for material corrosion behavior in BWRs is elaborated.

10:50 AM
Oxide Film Characterization along Crack Paths in 304 Stainless Steel in Aerated and Deaerated Water Environments: *Elaine West*[1]; David Morton[1]; Nathan Lewis[1]; Michael Burrell[1]; John Sutliff[1]; Joseph Pyle[1]; [1]Knolls Atomic Power Laboratory

Oxide films that formed along crack paths on 304 stainless steel SCC specimens in high temperature aerated and deaerated water (APW and DPW) were characterized through ATEM, EDS, Auger, and FIB/SEM analyses. SCC growth rate experiments were conducted in APW and DPW environments at temperatures from 104-327°C. Crack growth rates were rapid in anion faulted APW (~10 mils/day), while slower growth rates (~0.1 mils/day) occurred in DPW under constant loading conditions with periodic unloads. Measureable crack growth was not discernible in non-cold worked material tested in DPW under pure constant load. The SCC oxides generally exhibited a dual-layered structure with a chromium enriched inner layer (relative to the metal substrate) and an iron rich outer layer. Nickel enrichment was sometimes observed at and ahead of crack tips and along crack flanks at the metal/oxide interface. The nickel enrichment will be discussed in the context of the propensity for crack advance.

11:10 AM
Non-linear Dynamics of Oxide / Metal Interface Morphology of Austenitic Stainless Steels in Simulated BWR Water and Its Implication to SCC Initiation: *Yoichi Takeda*[1]; Takayuki Sato[1]; Daisuke Yamauchi[1]; Tetsuo Shoji[1]; Akio Ohji[1]; [1]Tohoku University

Initiation process of stress corrosion cracking (SCC) usually takes long precursor and incubation period.It can be thought that SCC initiation is a kind of localized oxidation or oxide penetration into metal substrate. In this investigation, apart from fundamental mechanistic studies, oxide / metal interface morphology after the oxidation of austenitic stainless steels were focused on. Oxidation tests were performed using circumferential notched round bar specimens in order to realize the oxidation under high stress state in pure water at 288°C. Based on the oxide / metal interface profiles that were obtained from cross sectional images of the specimens, histogram which can describes how the oxide intrusion develops was demonstrated. In addition, characteristic curve for oxidation localization was proposed and parameters that describe features of oxidation localization were derived. Trend in these localization parameters with oxidation time were shown and its correlation to SCC initiation was discussed.

Fuel and Fuel Related Materials I

Monday AM — Room: Colorado III
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Boching (Bo) Cheng, Electric Power Research Institute; Art Byers, Westinghouse Electric Company

10:10 AM
PWR Fuel Deposit Analysis at a B&W Plant with a 24-Month Fuel Cycle: *Mike Pop*[1]; [1]AREVA NP Inc.

The paper presents the CRUD analysis of a twice burned fuel deposit using the AREVA CRUD sampling method in a B&W Type PWR Plant. The patented AREVA sampling method was applied at the end of cycle (EOC) 16 at Crystal

River Plant (CR) in US. The method allows the separation of deposit flakes that retain all the characteristics of unperturbed CRUD deposition, such as CRUD Flake 531. Radiochemistry data will be presented that suggest that intense boiling was present in Cycle 16 at the location of the Flake 531 collection, producing a CRUD accumulation during Cycle 16 considerably larger then during Cycle 15. The unperturbed deposition 3D analysis of the deposition will be presented, allowing an in depth understanding of the processes of deposition on PWR fuel in conditions of intense localized boiling.

10:30 AM

Effect of DH Concentration on Crud Deposition on Heated Zircaloy-4 in Simulated PWR Primary Water: *Hirotaka Kawamura*[1]; [1]CRIEPI

In order to mitigate PWSCC initiation and propagation of Ni base alloy, Japanese PWR utilities desire to employ optimized dissolved hydrogen (DH) control operation in the near future. Prior to the application of the optimized DH control operation to PWR, the effect of DH concentration on the fuel crud deposition should be clarified. Crud deposition tests were carried out in 1200ppm as B + 2.2ppm as Li + 7 to 25cm^3-STP/kg-H_2O solutions at 325^{o}C under sub-cooled boiling and non-irradiated condition. The corrosion resistance of zircaloy-4 was also investigated.From the test results, it was revealed that the crud layer composed of $NiFe_2O_4$ and NiO was formed on zircaloy-4 fuel cladding. NiO was easy to form in the crud layer under the 7cm^3-STP/kg-H_2O condition. The amounts of deposited crud layer, boron contents incorporated into the crud layer and the corrosion resistances of cladding were not affected by DH concentration.

10:50 AM

Use of AREVA BWR CRUD Model to Study High Zinc Operation at a US Plant: *Mike Pop*[1]; [1]AREVA NP Inc.

The Paper presents the results of applying the AREVA BWR CRUD Model to determine the effect a High Zn operational program applied at a US Plant has on the fuel. The paper explains the CRUD Model capabilities and its role in the AREVA Fuel Risk Assessment Strategy. Simulated CRUD characteristics are analyzed along the 6 years operation comparing them with the CRUD analysis results. Also a number of "What if" scenarios are explored and their results are placed in the context of the fuel operation risk assessment.

11:10 AM

Electrochemical Study of Pre- and Post-Transition Corrosion of Zr Alloys in PWR Coolant: *Jan Macak*[1]; Radek Novotny[2]; Petr Sajdl[1]; Veronika Renciukova[1]; Vera Vrtílková[3]; [1]ICT Prague; [2]JRC Petten; [3]UJP Prague

Corrosion properties of different Zr-Sn and Zr-Nb zirconium alloys were studied in simulated PWR conditions (boric acid, potassium hydroxide, lithium hydroxide) at temperatures up to 340°C and 15MPa using electrochemical impedance spectroscopy (EIS) and polarization measurements. EIS spectra were obtained in a wide range of frequencies. It enabled to gain information of both dielectric properties of oxide layers developing on the Zr-alloys surface and of the kinetics of the corrosion process and the associated charge and mass transfer phenomena. Experiments were run for more than 270 days, thus the study of all the corrosion stages (pre-transition, transition, post-transition) was possible.Experimental impedance spectra were approximated by equivalent circuits based models. Jonscher-type analysis was applied to estimate frequency independent oxide capacitance. A model based on integrated Stern-Geary equation was developed to correlate instant and integral corrosion rates of zirconium alloys.

11:30 AM

AREVA Fuel Condition Index for a Pressurized Water Reactor: *Mike Pop*[1]; [1]AREVA NP Inc.

Three factors are considered paramount in fuel performance. These are heat flux, crud layer, and oxide thickness. Both the crud layer and the oxide thickness may be affected by plant chemistry. AREVA NP has developed a Fuel Condition Index (FCI) for PWR Fuel which provides a method to assign a single numerical value connecting the plant operating chemistry conditions to observed or expected fuel performance. FCI is an AREVA NP fuel damage risk assessment tool. The chemistry parameters and acceptable operating ranges selected for the example calculations to be presented in the paper consider AREVA NP knowledge and Industry consensus. This paper describes the FCI developed by AREVA NP (patent pending) and the results of its use at PWR plants compared with the results of applying at the same plants the Industry accepted High Duty Core Index (HDCI).

Alloy 690 and Its Weld Metals II

Monday PM — Room: Colorado I
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Mychailo Toloczko, Pacific Northwest National Laboratory; Kawaljit (Al) Ahluwalia, Electric Power Research Institute

1:05 PM

SCC Behavior of Alloy 690 HAZ in a PWR Environment: *Bogdan Alexandreanu*[1]; Yiren Chen[1]; Ken Natesan[1]; William Shack[1]; [1]Argonne National Laboratory

The objective of this work is to determine the cyclic and stress corrosion cracking (SCC) crack growth rates (CGRs) in a simulated PWR water environment for Alloy 690 heat affected zone (HAZ). In order to meet the objective, an Alloy 152 J-weld was produced on a piece of Alloy 690 tubing, and the test specimens were aligned with the HAZ. The environmental enhancement of cyclic CGRs for Alloy 690 HAZ was comparable to that measured for the same alloy in the as-received condition. The two Alloy 690 HAZ samples tested exhibited maximum SCC CGR rates of 10-11 m/s in the simulated PWR environment at 320ºC, however, on average, these rates are similar or only slightly higher than those for the as-received alloy.

1:25 PM

Stress Corrosion Crack Growth Rate Testing of Novel Composite Arrest Specimens: *David Morton*[1]; John Mullen[1]; [1]Knolls Atomic Power Laboratory

Stress corrosion crack (SCC) arrest tests have been conducted on composite material specimens to study the SCC susceptibility of highly SCC resistant materials. The "composite arrest" test method entails fabricating composite material specimens consisting of a highly SCC susceptible material welded to a highly SCC resistant material. Specimens are configured such that SCC grows from the susceptible material toward the resistant material. Results from Alloy 690 base metal, Alloy 690 weld metal and stainless steel composite arrest specimen tests showed that non-cold worked Alloy 690, Alloy 690 weld metal and stainless steel are extremely resistant to SCC in deaerated water environments. These results, in conjunction with other test data and outstanding commercial operational performance, indicate that Alloy 690 and stainless steel with regard to SCC are excellent materials for deaerated water environment applications. Results from extensive specimen analytical characterization efforts and SCC mechanistic insight will also be discussed.

1:45 PM

Environmentally Assisted Cracking Growth in Cold Worked Alloy 690TT in Primary Water at Low and High Temperatures: *Qunjia Peng*[1]; Tetsuo Shoji[1]; Juan Hou[1]; Yoichi Takeda[1]; Toshio Yonezawa[1]; [1]Tohoku University

Environmentally assisted cracking (EAC) growth in thermally treated Alloy 690 (Alloy 690TT) cold worked one-dimensionally to the thickness reduction of 25% was investigated by crack growth rate testing in the primary water at low and high temperatures. In the water at 50°C and a dissolved hydrogen (DH) of 2.7-2.9 ppm, crack growth in the alloy was observed under triangular waveform loading but not under trapezoidal waveform loading, indicating the alloy has a high resistance to low temperature crack propagation. Stress corrosion cracking (SCC) growth with a rate of <1×10-11 m/s was observed in the alloy in hydrogenated water at high temperatures of 320°C and 340°C. In water deaerated by N2 without DH, however, an extremely low SCC growth rate was observed. Increasing DH in water from 0 ppm to 0.38 ppm caused an active SCC growth, which was slightly decreased by increasing DH further to 1.4 ppm and 2.6 ppm. The result showed no correlation between the Ni/NiO transition and SCC growth rate. Further, it was suggested that hydrogen could promote SCC growth when increasing DH from a low level. Higher DH levels could mitigate the crack growth while the effect was not strong. Results of the current work confirmed a general high resistance of cold worked Alloy 690TT to EAC growth in the primary water at both low and high temperatures under moderate stress intensity factors.

2:05 PM

SCC of Alloy 690 and Its Weld Metals: *Peter L. Andresen*[1]; Martin Morra[1]; Kawaljit Ahluwalia[2]; [1]GE Global Research; [2]EPRI

Various heats, microstructures and types of cold work of Alloy 690 and Alloy 152/52 weld metals were tested in representative PWR primary water at 290 to 360°C. Intergranular cracking was observed in all materials. Crack growth rates in some Alloy 690 tests were in the range of 1 to 10 x 10–9 mm/s, primarily in orientations other than S-L. Growth rates on homogeneous Alloy 690, including extruded CRDM tubing, often showed growth rates in the range of 2 – 8 x 10–8 mm/s in cold worked condition and an S-L orientation. Growth rates as high as 7 x 10–7 mm/s were observed with microstructural banding and 1-dimension cold rolling tested in the S-L orientation, which aligns the planes of banding, rolling and cracking. However, not all banded material has exhibited such high growth rates. Alloy 152/52 weld metals always exhibited growth rates below 5 x 10–9 mm/s.

2:25 PM

SCC Behavior of Alloy 152 Weld in PWR Environment: *Bogdan Alexandreanu*[1]; Yiren Chen[1]; Yong Yang[2]; Ken Natesan[1]; William Shack[1]; [1]Argonne National Laboratory; [2]University of Wisconsin

The objective of this work is to determine the crack growth rates (CGRs) in a simulated PWR water environment for Alloy 152 weld. In order to meet the objective, specimens made from a laboratory-prepared Alloy 152 double-J weld in the as-welded condition were tested. For the SCC CGR measurements, the specimens were pre-cracked under cyclic loading in a primary water environment, and the cyclic CGRs were monitored to determine the transition from the fatigue transgranular fracture mode to the intergranular SCC fracture mode. The environmental enhancement of cyclic CGRs for Alloy 152 was minimal; nevertheless, the transition from transgranular to intergranular cracking was successful. Weld samples tested from the single heat of Alloy 152 exhibited SCC CGR rates of 10-11 m/s in the simulated PWR environment at 320ºC, which was only about an order of magnitude lower than typical for Alloy 182.

2:45 PM Break

BWR Initiation and Oxide Film Characterization II

Monday PM — Room: Colorado II
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Yoichi Takeda, Tohoku University; Johan Stjärnsäter, Studsvik Nuclear AB

1:05 PM

The Effect of Cold Work on Microstructure and SCC Susceptibility in Simulated BWR Environment for Non-Sensitized Austenitic Stainless Steels: *Yohei Sakakibara*[1]; Guen Nakayama[1]; [1]IHI Corporation

Recently, many researchers have reported the results of SCC test for cold worked stainless steels in high temperature water. In the papers, it is said that cold work has tendency to accelerate the SCC susceptibiltiy. However, there are little report that refers to the relationship between SCC susceptibility and microstructure of cold worked stainless steels. In this paper, we research the SCC susceptibility by CBB(Creviced bent beam) test and microstructure of cold worked stainless steels by EBSD. First, there are siginificant different in the effect of cold work between chemical compositions of materials. Then,we calculate the misorientation in EBSD analysis and reserach the relationship between EBSD parameters and SCC susceptiblities for each sample.

1:25 PM

The Behavior of Stress Corrosion Cracking for Type 316L Stainless Steel with Controlled Distribution of Surface Work Hardened Layer in Simulated Boiling Water Reactors Environment: *Yasufumi Miura*[1]; Yuichi Miyahara[2]; Masaru Sato[1]; Kenji Kako[1]; Jun-ichi Tani[1]; [1]Central Research Institute of Electric Power Industry; [2]Tohoku University

Stress corrosion cracking (SCC) was observed in many L-grade stainless steel components such as core shrouds or primary recirculation piping in boiling water reactors (BWR). In our previous study, we showed that the controlled distribution of surface work hardened layer concerned with enhancement in the SCC susceptibility. In this study, time-dependent creviced bent beam (CBB) test was conducted in a simulated BWR environment in order to evaluate the effect of the controlled layer on SCC initiation behavior. The CBB test specimen was made of type 316L stainless steel. After the CBB test, the SCC was characterized by counting the number of cracks and measuring the crack depth. As a result, it was found that the crack initiation time depended on the distribution of surface work hardened layer.

1:45 PM

Influence of Bulk and Surface Cold Work on Crack Initiation and Crack Growth of Austenitic Stainless Steels under Simulated BWR Environment: *Bastian Devrient*[1]; Renate Kilian[1]; Karin Küster[2]; Martin Widera[3]; [1]AREVA NP GmbH; [2]Vattenfall Europe Nuclear Energy GmbH; [3]RWE Power AG

The influence of surface CW strongly affects the crack initiation behavior of austenitic stainless steels under simulated LWR-environment. Within a parametric study crack initiation and CGR-experiments were performed under simulated BWR-environment to determine critical conditions for plant components which might undergo crack formation and subsequent crack growth. Within this project AISI 347, AISI 316Ti and AISI 316LN were tested in solution annealed and/or several CW conditions.Comprehensive characterization in initial CW condition in comparison to as-tested material condition after e.g. exposure tests for crack initiation studies and CGR-experiments, clearly indicate the influence of localized plastic deformation within the grains of the material on the processes of crack initiation and propagation. With increasing amount of CW the addiction to intergranular cracking seems to increase due to high cumulative strains on active slip paths. Increased strength and reduced plasticity of the material in CW material condition superimpose the localized strengthening effects.

2:05 PM Break

Fuel and Fuel Related Materials II

Monday PM — Room: Colorado III
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Kurt Edsinger, Electric Power Research Institute; Laurent Legras, EDF R&D

1:05 PM

Structure and Thermodynamical Properties of Zirconium Hydrides from First-Principle: *Jakob Blomqvist*[1]; Johan Olofsson[2]; Anna-Maria Alvarez[2]; Christina Bjerken[1]; [1]Malmo University; [2]Studsvik Nuclear AB

Zirconium alloys are used as nuclear fuel cladding material due to their favorable mechanical and material properties, including corrosion resistance, crack resistance and low neutron cross-section. At reactor running conditions, however, hydrogen concentration in the matrix will accumulate and at cooler conditions, for example at reactor stops, zirconium hydride will precipitate in the bulk. These hydrides are a known source of embrittlement, blistering and other unwanted effects and it is important to learn more about their nature. Using phase-field methods it is now possible to model precipitation build-up in metals, for example as a function of hydrogen concentration, temperature and external load, but the technique rely on accurate thermodynamical properties of the system. To that end, we have computed, using density-functional theory

methods, the Gibb's free energy of formation, heat capacity and latent heat of fusion as well as the solve the crystal structure for three zirconium hydride phases: d-ZrH1.6, g-ZrH, and e-ZrH2.

1:25 PM

Hydride Behavior in Zircaloy-4 during Thermomechanical Cycling: *Kimberly Colas*[1]; Arthur Motta[1]; Mark Daymond[2]; Jonathan Almer[3]; Zhonghou Cai[3]; [1]Pennsylvania State University; [2]Queen's University; [3]APS Argonne National Laboratory

Hydrogen ingress into zirconium alloy fuel cladding during operation in nuclear reactors can degrade cladding performance due to formation of brittle hydrides. At temperature and under stress, hydrogen redistribution and reorientation can occur, reducing cladding resistance to failure. Thus, it is crucial to understand the kinetics of hydride dissolution and re-orientation under load and at temperature. High-energy and micro-beam synchrotron diffraction are used to study the kinetics of hydride reorientation and hydride distribution near a crack tip in previously hydrided Zircaloy-4 sheet. Reorientation of hydrides in bulk samples is studied in situ (at temperature and under applied tensile stress). In-situ transmission diffraction data provides unique strain and orientation information on the hydrides. Micro-beam diffraction has been performed on previously cracked compact tension specimens under load. Measurement of the hydride distribution and associated strains can be performed with the micro-beam to determine hydrogen response to an applied strain field.

1:45 PM

Key Mechanisms of Oxidation of Zirconium Alloys Studied by APT and TEM Analysis: *Chris Grovenor*[1]; Na Ni[1]; Sergio Lozano-Perez[1]; Daniel Hudson[1]; George Smith[1]; [1]Oxford University

This presentation will describe recent progress in Oxford in using advanced analytical and microstuctural techniques to study the mechanisms that control the rate of oxidation of commercial zirconium alloys. We have concentrated on analysing the critical changes that occur both in the bulk oxide and at the metal/oxide interface as a function of oxidation time. We will present new results on changes in nanoscale porosity and sub-oxide layers around the first transition in oxidation kinetics, and correlate these obervations with measurements from our colleagues in Manchester University of the electrochemical behaviour of these same samples in the same time interval.

Fuel and Fuel Related Materials III

Monday PM — Room: Colorado III
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: J. Lawrence Nelson, JLN Consulting; Kurt Terrani, University of California, Berkeley

2:05 PM

Development of a Method for Studying the Influence of Stress State on the Iodine-Induced Stress Corrosion Cracking of Zirconium Alloys: *Nathanael Mozzani*[1]; Eric Andrieu[2]; Quentin Auzoux[1]; Christine Blanc[2]; David Leboulch[1]; [1]CEA; [2]Université de Toulouse

Zircaloy-4 is a zirconium alloy widely used as cladding material in Pressurized Water Reactors. During a power transient, a phenomenon known as Pellet-Clad Interaction can lead to cladding failures. The mechanism involved is thought to be iodine-induced stress corrosion cracking (I-SCC). Tensile tests were carried out in an inert atmosphere at room temperature on smooth samples of recrystallized Zircaloy-4, and a viscoplastic anisotropy model was established. In order to study I-SCC initiation, smooth samples were subjected to tensile and bending tests in iodine methanol at room temperature. Notched specimens were used in these tests in order to study the influence of stress biaxiality on I-SCC. Some samples were proton-irradiated in a 3MV electrostatic accelerator in order to quantify the influence of irradiation on mechanical behaviour by tensile testing thin samples and by nano-indentation. Subjecting these irradiated samples to I-SCC tests should yield insights on the role of irradiation I-SCC.

2:25 PM

Wear of Zircaloy-4 Grid Straps Due to Fretting and Periodic Impact with RV Internals Baffle Plates: *Sarah Davidsaver*[1]; Steve Fyfitch[1]; Brian Friend[1]; James Hyres[2]; [1]AREVA; [2]B&W Technical Services Group, Inc.

Physical interaction between fuel assembly Zircaloy-4 grid straps and the reactor vessel (RV) internals Type 304 stainless steel baffle plates has been identified at several U.S. PWR units. Understanding the phenomenon causing the interaction will help prevent further escalation and assist in developing actions to mitigate the issue. This paper summarizes the results of a hot cell evaluation of degraded Zircaloy-4 grid strap material removed from a discharged fuel assembly. Examinations included visual and stereo-visual inspections, scanning electron microscopy (SEM), energy dispersive spectroscopy (EDS), optical metallography, and Vickers microhardness. Comparisons to other relevant wear information reported in the literature are also provided. The laboratory examinations indicated horizontal motion and periodic impacting occurred between the fuel assembly and the baffle plate surface. It is concluded that the degradation mechanism is either fretting wear, adhesive wear, or possibly, a combination of both.

2:45 PM Break

Alloy 690 and Its Weld Metals III

Monday PM — Room: Colorado I
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Raj Pathania, EPRI; Toshio Yonezawa, Tohoku University

3:00 PM

Effect of Hot Cracks on EAC Crack Initiation and Growth in Nickel-Base Alloy Weld Metals: *Hannu Hanninen*[1]; Aki Toivonen[2]; Anssi Brederholm[1]; Tapio Saukkonen[1]; Wade Karlsen[2]; Ulla Ehrnsten[2]; Pertti Aaltonen[2]; [1]Aalto University School of Science and Technology; [2]VTT Technical Research Centre of Finland

Differences in EAC susceptibility between different weld metals were distinguished by doped steam test. Pure weld metals of Alloy 182 and 82 are susceptible to EAC, but Alloy 152 and 52 did not show any crack initiation. Dissimilar metal welds (DMW) with diluted microstructures are less susceptible than pure weld metals of Alloy 182 and 82. No crack initiation/extension from hot cracks occured in any weld metals, which is related to segregated microstructures of hot crack tips. In accelerated doped steam test selective dissolution takes place and metallic Ni forms a continuous layer in the middle of cracks on Cr-rich oxide. Selective dissolution was not observed inside hot cracks. EAC initiation occurred in Alloy 600 base metal of DMWs and selective dissolution inside EAC cracks in Alloy 600 was extensive. Results are discussed based on the selective dissolution creep model of EAC.

3:20 PM

Stress Corrosion Crack Growth of Alloy 52M in Simulated PWR Primary Water: *Mychailo Toloczko*[1]; Stephen Bruemmer[1]; [1]Pacific Northwest National Laboratory

Crack-growth experiments have been performed on six different alloy 52M welds in simulated PWR primary water at 350-360°C. The alloy 52M test matrix included V-groove and narrow-gap welds, an overlay on alloy 182 and an inlay on alloy 82. For the overlay and inlay materials, cracks are propagated from the lower Cr, alloy 182 or 82 weld metal into the higher Cr, alloy 52M weld metal. In one of the narrow gap welds, the crack path was oriented to pass through a distribution of pre-existing weld cracks and their influence on stress-corrosion behavior is evaluated. Intergranular stress corrosion cracking (IGSCC) is observed in several alloy 52M welds, however propagation rates remain below 1x10-8 mm/s in all cases. Comparisons will be made to previous SCC measurements on alloy 152 and 52 welds.

MONDAY PM

3:40 PM

SCC Behavior of Alloy 52M/182 Weld Overlay in a PWR Environment: *Bogdan Alexandreanu*[1]; Yiren Chen[1]; Yong Yang[2]; Ken Natesan[1]; William Shack[1]; [1]Argonne National Laboratory; [2]University of Wisconsin

The objective of this work is to investigate the behavior of a crack initiated in Alloy 182 as it approaches the Alloy 52M WOL interface. Compact tension specimens were fabricated with the notch in Alloy 182 and oriented towards the WOL, and tested in a simulated PWR environment. The first such test revealed that the SCC rates in Alloy 182 were found to decrease by an order of magnitude ahead of the interface, and that the crack advanced from Alloy 182 into Alloy 52M. The post test examination found that crack branching occurred at the interface between the two allows. Growth in Alloy 52M along the interface appears severe, approx. 10-10 m/s. While for the most part (70%) the crack propagated along the interface, SCC cracking was also found to extend into Alloy 52M along the original direction. This cracking is substantial, yielding SCC rates of 10-11 m/s.

4:00 PM

SCC of High Cr Alloys in BWR Environments: *Peter L. Andresen*[1]; Ron Horn[2]; [1]GE Global Research; [2]GE Hitachi Nuclear Energy

Alloy 182 weld metal is known to be susceptible to SCC in high temperature water, and within the last decade both PWRs and BWRs have specified other weld metals. There is increasing evidence that Alloy 82 weld metal, while somewhat more resistant to SCC than Alloy 182, is not sufficiently resistant to be considered adequate for a 60 – 80 year plant life. Indeed, some large studies have shown relatively little difference in crack growth rate response between these weld metals. The data show excellent SCC resistance, and are consistent with PWR data in indicating that a dramatic increase in resistance occurs between ~20 and 26% Cr, with alloy about 26% Cr shown excellent behavior. This provides opportunities to optimize the composition of the weld metal for resistance to ductility dip cracking, hot cracking, etc.

4:20 PM

High-Resolution Characterizations of Grain Boundary Damage and Stress Corrosion Crack Tips in Cold-Rolled Alloy 690: *Stephen Bruemmer*[1]; Matthew Olszta[1]; [1]Pacific Northwest National Laboratory

One-dimensional cold rolling has been shown to promote IGSCC in alloy 690 tested in PWR primary water. Scanning and transmission electron microscopy has been employed to investigate the microstructural reasons for this enhanced susceptibility in two stages, first examining grain boundary damage produced by cold rolling and second by characterizing SCC crack tips. The degree of grain boundary damage from cold rolling was found to depend directly on the initial precipitate distribution. High levels of cold rolling created small IG voids and cracked carbides in alloys with semi-continuous grain boundary carbides. For the same degree of cold rolling, alloys with few IG carbides exhibited much less damage. Crack-tip examinations investigate the interaction between the pre-existing voids/cracks and propagating SCC cracks. Characterizations were performed on alloy 690 CRDM tubing and plate materials to gain insights into IGSCC mechanisms.

4:40 PM Break

BWR Stainless Steels CGR I

Monday PM — Room: Colorado II
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Ron Horn, GE Hitachi Nuclear Energy; Tetsuo Shoji, Tohoku University

3:00 PM

Effect of Nitrogen Addition in 304L Stainless Steel on the IGSCC Crack Growth Rate in Simulated BWR Environment: *Supratik Roychowdhury*[1]; Vivekanand Kain[1]; R. C. Prasad[2]; [1]Bhabha Atomic Research Centre; [2]Dept of Metallurgical Engineering and Materials Science, IIT (Bombay)

Intergranular Stress Corrosion Cracking (IGSCC) in austenitic Stainless Steels (SS) in Boiling Water Reactor (BWR) operating conditions have been reported worldwide. Nitrogen containing Stainless Steel is used in BWRs and it can affect IGSCC behavior. In this investigation type 304L stainless steel with two different levels of nitrogen was used in the sensitized and non-sensitised strain-hardened condition. Experiments were carried out in high temperature water with controlled dissolved oxygen. In the sensitised condition, the Crack Growth Rate (CGR) reduced and in the non-sensitised strain-hardened condition the CGR increased with increase in nitrogen level in SS. Transmission electron microscopic (TEM) investigation of the rolled SS and after tensile testing at 288°C indicated that rolling resulted in higher grain boundary strain which is a possible cause for higher CGR in the SS with higher nitrogen. At 288 C nitrogen did not have a noticeable effect on the deformation mechanism.

3:20 PM

Characterization of Type 304L Stainless Steel: Comparison of ASTM A262 Practice A and Analytical Electron Microscopy Techniques: *Bryan Miller*[1]; M. G. Burke[1]; [1]Bettis Laboratory

The ASTM A262 Practice A test is frequently used to assess whether Type 304/304L austenitic stainless steels are "sensitized". In this study, Type 304L steel containing 18 ppm boron examined in the as-received (nominal mill solution-annealed) condition exhibited a "dual" structure after the Practice A test despite its low C content. Detailed characterization coupled with laboratory sensitization and stabilization heat treatments were performed to assess the precipitation behavior in this steel. Intergranular, Cr-rich, M_2B-type borides, identified by electron diffraction, were observed in the as-received and aged conditions. Samples aged at 700°C ("sensitized") produced fully "ditched" grain boundaries having Cr levels in excess of 14wt.% and concomitant Cr-depleted zones less than 50 nm in extent. Despite exhibiting fully "ditched" grain boundaries, Cr levels of ~18-20wt.%Cr with no Cr-depleted zones were deteced in specimens aged at 900°C (i.e.,"stabilized"). These results show the Practice A test can be mis-used/mis-interpreted for assessing sensitization.

BWR Stainless Steels CGR II

Monday PM — Room: Colorado II
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Darrell Dunn, U. S. Nuclear Regulatory Commission; David Alley, U. S. Nuclear Regulatory Commission

3:40 PM

The Effect of Grain Size on IGSCC in SS 316L in Simulated BWR Environment: *Johan Stjärnsäter*[1]; Bengt Bengtsson[2]; Björn Forssgren[3]; Hannah Johansson[4]; [1]Studsvik Nuclear AB; [2]OKG AB; [3]Ringhals AB; [4]Forsmarks Kraftgrupp AB

Disposition lines are used for flaw tolerance analyses, structural integrity assessment and life prediction of reactor components. One parameter normally not accounted for in the disposition lines is the grain size of the material. However, in actual reactor systems there can be variations in grain size in one component, as well as among different components. The objective of this project was therefore to study if the susceptibility to IGSCC in one heat of Type 316L stainless steel is affected by the grain size. CGR measurements were conducted on sixteen 25 mm CT specimens in simulated BWR environments. Eight specimens were loaded by pull rods (active load), and the remaining specimens were bolt loaded. Three different grain sizes were studied on the same heat of material: 26, 347 and 590 µm. Results on the effect of grain size on the CGR in NWC and HWC will be presented and discussed.

4:00 PM
An Investigation into Stress Corrosion Cracking of Dissimilar Metal Welds with 304L Stainless Steel and Alloy 82 in High Temperature Pure Water: *Tsung-Kuang Yeh*[1]; Guan-Ru Huang[1]; Tai-Ni Yang[1]; Mei-Ya Wang[1]; [1]National Tsing Hua University

For a better understanding to stress corrosion cracking (SCC) in dissimilar metal welds, the SCC growth behavior in the transition regions of weld joints was investigated via slow strain rate tensile (SSRT) tests in 280°C pure water with 300 ppb dissolve oxygen concentration. Prior to the SSRT tests, specimens with dissimilar metal welds were prepared and underwent various pretreatments of post-weld heat treatment, shot peening, solution annealing, and surface polishing. In addition to the SSRT tests, micro-hardness measurements on the transition regions of the metal welds were also conducted. According to the SSRT test results, fracture planes of all specimens were located at the stainless steel sides and were parallel with the fusion lines. Significant amounts of transgranular SCC were observed on the fractured surfaces of the specimens except for those pretreated by shot peening or by solution annealing.

4:20 PM
Deformation Mode and Microstructure on Stress Corrosion Cracking Path and Kinetics in High Temperature Water Environments: *Zhanpeng Lu*[1]; Tetsuo Shoji[1]; Seiya Yamazaki[1]; Fanjiang Meng[1]; Tichun Dan[1]; Yoichi Takeda[1]; Koji Negishi[1]; [1]Tohoku University

Stress corrosion cracking susceptibility of austenitic alloys subjected to various kinds of prior deformation such as one-directionally rolling, two-directionally rolling, three-directionally rolling and weld-shrinkage was investigated by material and microstructural characterization, crack growth rate tests in high temperature water, and crack tip characterization. The active cracking paths and cracking kinetics as functions of types and ratios of grain boundaries, microstructural anisotropy, strength or hardness, local strain in terms of the distribution of misorientation are quantitatively studied. The effects of mesoscopic heterogeneity in microstructure and local deformation on stress corrosion cracking and the interaction with electrochemical conditions are emphasized.

4:40 PM Break

Corrosion Fatigue - BWR, PWR

Monday PM — Room: Colorado III
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Harvey Solomon, Solomon Metallurgical Consulting L.L.c; Hans-Peter Seifert, Paul Scherrer Institute

3:00 PM
Effect of Static Load Hold Periods on the Corrosion Fatigue Behavior of Austenitic Stainless Steels in Simulated BWR Environments: *Hans-Peter Seifert*[1]; Stefan Ritter[1]; Hans Leber[1]; [1]Paul Scherrer Institute

The effect of long static load hold periods on the corrosion fatigue (CF) initiation and crack growth behavior of austenitic stainless steels in BWR NWC and HWC environment was characterized by cyclic fatigue tests with sharply notched and pre-cracked fracture mechanics specimens. With regard to continuous cyclic saw tooth loading, an increase of the genuine CF initiation life was observed with increasing static load hold periods at maximum or mean load, which seemed to saturate for long hold periods above 12 hours. On the other hand, static load hold periods at minimum load had no effect on genuine initiation life. Furthermore, the hold times had very little effect on the subsequent stationary short and long CF crack growth rates in notched and pre-cracked specimens, respectively. No significant effect of static load hold times on the technical corrosion fatigue initiation life is thus expected based on these preliminary results.

3:20 PM
Effects of Material Compositional on Corrosion Fatigue Crack Growth of Austenitic Stainless Steels in High Temperature Water: *Norman Platts*[1]; David Tice[1]; Kevin Mottershead[1]; Laura McIntyre[1]; Fabio Scenini[2]; [1]Serco plc; [2]University of Manchester

Laboratory studies on austenitic stainless steels in PWR primary coolant environments have shown that the ASME XI procedures used to assess fatigue crack growth of reactor components may not always be conservative. Recent work has shown that significant environmental enhancement of growth rates can occur in this environment, especially for some long rise time loading cycles. Although enhancements up to eighty times relative to air data have been observed, under some conditions retardation of the enhanced growth rates can occur, to rates close to the ASME XI air line. Several factors appear to influence retardation, including temperature, water flowrate and material composition. The current study addresses the influence of material composition and it is shown that steels of high sulphur content (>0.02%) are more prone to retardation than low sulfur (<0.01%) steels. Work aimed at elucidating possible mechanisms for this effect is described.

3:40 PM
Fatigue Limit and Hysteresis Behavior of Type 304l SS in Air and PWR Water, at 150°C and 300°C: *Harvey Solomon*[1]; Claude Amzallag[2]; Ronald De Lair[3]; A.J. Vallee[4]; [1]Solomon Metallurgical Consulting L.L.c; [2]Retired from EDF; [3]Retired from GE-GRC; [4]GE-GRC

This is a study of the 10^7 cycle fatigue limit of Type 304L, as measured in fully reversed (R=-1) load-controlled tests, at 150°C and 300°C, in air and PWR water. The staircase method was used to determine the fatigue limit. The tests run here utilized a cycle frequency of 1.818Hz and are compared to other tests from the literature that were run at 30Hz. The fatigue limit measured in the tests run at the high frequency were higher than that measured here. This is explained by measurements of the strain developed during cycling, using the different cycle frequencies. The tests run at the higher frequencies yielded lower strains for a given stress and, as expected, this resulted in higher fatigue limits. Using $10^7$7 cycles to define a run-out also led to a lower fatigue limit.

4:00 PM
NRC Research Activities on Environmentally-Assisted Fatigue: Gary Stevens[1]; *Robert Tregoning*[1]; [1]U.S. Nuclear Regulatory Commission

Over the past ten years, evaluation for license renewal and new reactors has provided significant experience and insight on the use of the environmental fatigue multiplier (Fen) approach, and recognized the need for further refinement of this methodology, as well as its application to other areas. Hence, the NRC has initiated further research work on environmentally assisted fatigue. The objectives of these research activities are as follows: 1. Develop a transient stress evaluation software tool for rapidly determining thermal transient stresses in reactor components. 2. Develop an ASME Code fatigue calculation software tool for estimating fatigue usage factors in reactor components. 3. Develop revised CUF limit criteria for postulated high energy line break locations. 4. Obtain technical support from Argonne National Laboratory to update existing environmental fatigue methodology, develop application techniques for applying the methodology, and revise RG 1.207 accordingly, if appropriate.

Alloy 690 and Its Weld Metals IV

Monday PM — Room: Colorado I
August 8, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Stephen Bruemmer, Pacific Northwest National Laboratory; Bogdan Alexandreanu, Argonne National Laboratory

4:55 PM
Research and Evaluation of Low Temperature Crack Propagation of Ni Base Alloys in Actual Plants: *Kimihisa Sakima*[1]; Harutaka Suzuki[1]; Hideki Fujiwara[2]; [1]Mitsubishi Heavy Industries, Ltd; [2]Shikoku Electric Power Co., Inc.

Many published results have shown that fracture resistance of Ni base alloys is remarkably reduced by intergranular cracking in low temperature (<150°C) hydrogenated water. This phenomenon is called low temperature crack propagation (LTCP). In order to study maintenance assessment of actual components, J-R tests for Ni base alloys (Alloy 690, 600, Alloy 52, 82 and 152 weld metal) were carried out in simulated PWR primary water to investigate susceptibility to LTCP.J-R tests were performed under the conditions of various temperatures and dissolved hydrogen content. As a result, Alloy 690, 600, Alloy 52 and 82 weld metal showed no remarkable susceptibility to LTCP. Alloy 152 weld metal showed susceptibility to LTCP in water with dissolved hydrogen content of over 15ccH2/kgH2O at 50°C. On the other hand, as result of investigation on operating condition of actual Japanese PWRs, it was confirmed that there are no possibility of plant operation in 50°C water with dissolved hydrogen content of over 15ccH2/kgH2O. Thus, it is evaluated there is no possibility that LTCP becomes a threat in actual Japanese PWRs.

5:15 PM
Penetrative Internal Oxidation from Alloy 690 Surfaces and Stress Corrosion Crack Walls during Exposure to PWR Primary Water: *Matthew Olszta*[1]; Stephen Bruemmer[1]; [1]Pacific Northwest National Laboratory

Unexpected penetrative oxidation has been discovered off stress corrosion crack walls up to the leading crack tips in alloy 690 crack growth test samples. High-resolution SEM imaging reveals that the depth of penetrative oxidation increases with exposure time, while TEM elucidates the structure and phases in the nanoscale oxidation. Oxide filaments are on the order of 5 nm in diameter and contain discrete Cr2O3 particles and extremely fine MO-structure nanocrystallites. Atom probe tomography elucidated the distribution of discrete Cr2O3 particles surrounded by a mixed metal oxide in the filaments. In addition, lithium was found to penetrate through the oxidized material from the water environment. Direct comparisons are made to exposed polished surfaces on the same alloy 690 materials and implications on SCC initiation and propagation discussed.

5:35 PM
Predicting IGSCC Resistance of Nickel-Base Alloys: *Youfa Yin*[1]; Feng Zhu[1]; Roy Faulkner[1]; Ed Miller[2]; Paul Moreton[2]; Ian Armson[2]; [1]Loughborough University; [2]Rolls-Royce

There are two so-called grain boundary chromium concentrations which are both important in determining the intergranular stress corrosion cracking (IGSCC) resistance of nickel base alloys. They are the grain boundary chromium concentration at carbide free grain boundary sections and the interfacial chromium concentration at the interface between carbides and the matrix. Existing models for predicting grain boundary chromium depletion either predicts the former or the latter. A Monte Carlo based precipitation kinetics simulation approach has been developed to simulate the evolution of both the grain boundary and interfacial chromium concentration. Application of the method to Alloy 690 yields good agreement with experimental observations with regard to both the chromium depletion evolution and carbide precipitation kinetics. The effect of grain boundary character will also be discussed.

SCC of Alloy 82, 182 Welds I

Tuesday AM — Room: Colorado I
August 9, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Anders Jenssen, Studsvik Nuclear AB; Jean Smith, EPRI

8:15 AM
Interaction of Microstructure, Composition, and Cold Work on the Stress Corrosion Cracking of Alloy 82 Weld Metal: *Denise Paraventi*[1]; William Moshier[1]; [1]BMPC - Bettis Laboratory

Chromium concentration, weld-induced residual stress, and cold work are factors that affect the stress corrosion cracking (SCC) of both wrought Alloy 600 and Alloy 82 weld metal. However, chemical inhomogeneities and weld residual stresses that arise during solidification in welds make it difficult to separate the beneficial influence of chromium in Alloy 82 and the negative impact of weld-induced plastic strains. SCC growth rate tests on a single set of Alloy 82 weld cradles in four conditions were examined, including: as-welded, as-welded and cold worked, fully annealed, and fully annealed and cold worked. Annealed material showed a significant improvement in SCC performance relative to the other conditions, as well as to Alloy 600, with its relatively lower chromium content, in both the annealed and the cold worked conditions. This work shows a fundamental interaction of the alloy microstructure, composition, and cold work in response to SCC performance.

8:35 AM
Stress Corrosion Cracking Behavior of Dissimilar Metal Weldments in High Temperature Water Environment: *Jiunn-Yuan Huang*[1]; Seng-Long Jeng[1]; Jiunn-Shiung Huang[1]; Roang-Ching Kuo[1]; Ming-Fong Chiang[1]; [1]Institute of nuclear energy research

The stress corrosion cracking behavior of dissimilar metal welds, including Alloy 52-A 508 and Alloy 82-A508, under a simulated BWR coolant condition was studied. Effects of postweld heat treatments and specimen sizes on the SCC growth rates of DM welds were evaluated. The crack growth rate for the DM weld heat treated at 621°C for 24 hrs was observed to be faster than those of the as-welded. But the DM weld heat treated at 621°C for 8 hrs and 400°C or 200 hrs shows the better SCC resistance than those of the as-welded. The longer the heat treatment at 621°C, the higher the chromium carbides density along the grain boundary was observed. The SCC growth rate of the 1/2 TCT specimen is faster than those of 1TCT specimen. It could be accounted for by the thinner specimen has the shorter distance for oxygen ions to diffuse to the crack tip.

8:55 AM
SCC Crack Growth Rate of Alloy 82 in PWR Primary Water Conditions – Effect of a Thermal Treatment: Marc Le Calvar[1]; *Catherine Guerre*[2]; [1]IRSN; [2]Commissariat à l'Energie Atomique (CEA)

Stress corrosion cracking (SCC) of wrought alloy 600 and parent weld metals (alloys 182/82) is a significant cause of failure in the pressurized water reactors (PWR). Only a small number of welds fabricated from Alloy 82 is affected by PWSCC. Most of these welds were not thermally heat treated unlike the industrial practice in France. This paper describes constant load crack growth rate (CGR) tests on alloy 82 with and without post weld heat treatment. Metallurgical examination of alloy 82 was carried out using among others Electron. Backscattering Diffraction and Transmission Electron Microscopy. The heat treatment seems to be highly beneficial by decreasing the CGR. This result can be explained by the effect of thermal treatment on the precipitation in alloy 82.

9:15 AM Break

BWR Stainless Steels CGR III

Tuesday AM — Room: Colorado II
August 9, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Ernest Eason, Modeling & Computing Services LLC; Peter Ford, Consultant

8:15 AM
The Effect of Temperature on the Crack Growth Rate in Simulated BWR Environment: *Johan Stjärnsäter*[1]; Anders Jenssen[1]; Christer Jansson[2]; Karen Gott[3]; Bengt Bengtsson[4]; Björn Forssgren[5]; Hannah Johansson[6]; [1]Studsvik Nuclear AB; [2]Vattenfall Power Consultant AB; [3]Swedish Radiation Safety Authority; [4]OKG AB; [5]Ringhals AB; [6]Forsmarks kraftgrupp AB

The effect of temperature on the crack growth rate (CGR) in BWR normal water chemistry (NWC) has been investigated in various studies over the years. However, the effect has not been unambiguously clarified, since in some cases a maximum in the CGR has been observed at ~200°C, while others have reported a monotonic increase of the CGR with temperature. To clarify the effect of temperature, testing has been performed in oxygenated high-purity water (NWC). Crack growth rate measurements were conducted on 25 mm compact tension (CT) specimens in simulated NWC at ~30 MPaµm. Temperatures in the range 288 to 100°C were studied. Results on the effect of temperature on the crack growth rate in high-purity water with oxygen will be presented and discussed.

8:35 AM
Effects of Temperature and Corrosion Potential on SCC: *Peter L. Andresen*[1]; Russell Seeman[2]; [1]GE Global Research; [2]GE Hitachi Nuclear Energy

Hydrogen water chemistry and/or electrocatalysis is used all U.S. BWRs and many overseas BWRs to mitigate SCC. However, hydrogen is currently injected into the feed water lines, where flow occurs only when the BWR is at temperature and steam is flowing to the turbine. Thus, SCC mitigation is unavailable during reactor start up and shut down, when the water chemistry is more oxidizing and often has higher impurity levels, and dynamic strain is present from pressurization, differential thermal expansion, etc. This study was designed to evaluate the response of sensitized stainless steel and Alloy 182 weld metal to changes in corrosion potential at temperatures ranging from 100 to 250°C.

8:55 AM
Effect of Thermal Aging on SCC, Material Properties and Fracture Toughness of Stainless Steel Weld Metals: Timothy Lucas[1]; *Ronald Ballinger*[1]; Hannu Hanninen[1]; Tapio Saukkonen[2]; [1]MIT; [2]Aalto University School of Science and Technology

An experimental program is under way in order to understand how the spinodal decomposition may affect material properties changes in BWR pipe weld metals. Charpy impact, fracture toughness, including in-situ fracture toughness, fatigue and SCC crack growth rates of SS weld metals under simulated BWR conditions are reported. Tensile, microhardness and Charpy-impact energy show an increase in strength and a decrease in impact energy after aging for 1000, 5000 and 10,000 hours at 430 and 400°C. SCC crack growth results indicate an approximately 10X increase in crack growth rate over that of the unaged material. In situ fracture toughness, after greater than 2000 hours exposure @ 288°C, can be significantly reduced-by as much as 40% over that in 288°C air. Tearing resistance after similar exposure but beginning from an SCC crack and in load control has been observed to be reduced by as much as 75%.

9:15 AM Break

PWR Secondary Side/Balance of Plant I

Tuesday AM | Room: Colorado III
August 9, 2011 | Location: Cheyenne Mountain Resort

Session Chairs: Roger Newman, Univ of Toronto; Robert (Bob) Tapping, Atomic Energy of Canada Ltd

8:15 AM
On the Microstructure of Alloy 600 SCC Cracks Observed by TEM on PWR SG Pulled Tubes and on Laboratory Specimens: *Laurent Legras*[1]; Frederic Delabrouille[1]; Salem Miloudi[2]; Elodie Fargeas[2]; Odile De Bouvier[2]; Yannick Thebault[2]; [1]EDF R&D; [2]EDF CEIDRE

Secondary side corrosion cracking of steam generator tubes in Mill Annealed Alloy 600 occurs in flow-restricted areas where impurities get concentrated under heat flux. During spring 2009, eddy current test showed a circumferential indication (tube support plate elevation) in Bugey-3 unit for the first time on French nuclear plants in which very few PbO was detected in deposits in the late 80's after 67000h of service. The corresponding tube was therefore removed. IGSCC cracks and outer oxides layers formed on these pulled tubes were examined by ATEM. The results are compared to previous ones obtained on a tube pulled out from another unit (Dampierre-4) where Pb was detected during ATEM observations and not suspected to be at the origin of IGSCC and to those obtained on specimen tested in laboratory environment all leading to SCC rates comparable to secondary side corrosion cracking rates observed in the field. The oxides formed were compared to identify the typical environment responsible for the degradation observed on the pulled tubes. It appears that the best laboratory environment reproducing oxides morphology observed on the selected pulled tubes was a NaOH with Pb environment even thought Si pollution was sometimes detected in outside oxides layer.

TUESDAY AM

8:35 AM
Balance of Plant Corrosion Issues in Aging Nuclear Power Plants: *George Licina*[1]; [1]Structural Integrity Associates, Inc.

Balance of plant systems in nuclear plants, such as service water systems, are a critical part of the facility's infrastructure. System integrity and performance are vital for plant reliability and essential to achieving a plant life of 40 years and beyond. Corrosion allowances, based upon very simplistic considerations of general corrosion in untreated raw water, were a part of the original design. However, in many such systems localized corrosion phenomena, from microbiologically influenced corrosion, pitting, and underdeposit effects, have compromised system integrity. Because of the complexity and random nature of corrosion processes, it is nearly impossible to develop a mathematically deterministic model that accurately predicts pipe wall loss. However, when statistical distributions are used to describe the various corrosion processes, mathematical algorithms that incorporate all of the distributions, iterated a statistically significant number of times, can be used to forecast the most probable number of leaks.

8:55 AM
Containment Liner Corrosion: *Darrell Dunn*[1]; April Pulvirenti[1]; Paul Klein[1]; [1]US NRC

Of the 104 currently operating nuclear power plants in the U.S, there are 66 plants that have containment buildings constructed with an inner steel liner plate in contact with a thick concrete shell. The steel liner, which is nominally 0.25 to 0.375 inches thick, is designed to function as an essentially leak tight barrier against the release of radiation under accident conditions. Since 1999, there have been several cases of corrosion penetration of containment liners associated with foreign materials that were embedded in the concrete during original plant construction. The objectives of this work were to review plant operating experience, evaluate factors that can affect containment liner corrosion susceptibility, and determine the mechanisms for through-wall corrosion initiated at the concrete/liner interface.

9:15 AM Break

SCC of Alloy 82, 182 Welds II

Tuesday AM | Room: Colorado I
August 9, 2011 | Location: Cheyenne Mountain Resort

Session Chairs: Anders Jenssen, Studsvik Nuclear AB; Jean Smith, EPRI

9:30 AM
Initiation of PWSCC of Weld Alloys 182 and 82: *Thierry Couvant*[1]; Francois Vaillant[1]; [1]EDF R&D

Even if Alloy 182 is usually exhibiting a high susceptibility to cracking in the laboratory, the field experience did not reveal to date a significant susceptibility to PWSCC in PWRs, when the welds have been perfectly stress relieved. However, recently, an increasing number of cracks was reported in USA, Sweden and Japan on Alloy 182 and a few cases on Alloy 82. This paper addresses the work on initiation of PWSCC engaged at EDF R&D (MAI). The main objective is to calibrate an engineering model to predict the time to initiate IGSCC vs. temperature and loading for a weld 182 having a high susceptibility to SCC crack growth. The effect of cyclic loading, strain path and dendrite orientation on initiation was also partly evaluated. Under static loading, initiation was observed down to 350 MPa at 360°C. A limited effect of cyclic loading (R = 0.9, f = 2.8 10-4 Hz) was observed at 360°C for a maximal stress of 350 MPa. The effect of partial periodic loading increased when the temperature decreased and when the stress increased. An empirical model predicting the time to SCC initiation was calibrated.

BWR Low Alloy Steel

Tuesday AM | Room: Colorado II
August 9, 2011 | Location: Cheyenne Mountain Resort

Session Chairs: Ernest Eason, Modeling & Computing Services LLC; Peter Ford, Consultant

9:30 AM
Stress Corrosion Cracking Retardation Behavior near the Fusion Boundary of Dissimilar Weld Joint with Alloy 182 - A533B Low Alloy Steel: *Hiroshi Abe*[1]; Makoto Ishizawa[1]; Yutaka Watanabe[1]; [1]Tohoku University

The stress corrosion cracking (SCC) behavior near the fusion boundary (FB) of a dissimilar weld joint with Alloy 182-A533B low alloy steel (LAS) in high-temperature oxygenated water doped with sulfate has been investigated, with a focus on the relationship between the SCC crack cessation/reinitiation behavior and the microstructural characteristics of the heat-affected zone (HAZ) in LAS adjacent to the FB. Cracks propagated perpendicular to the FB along the dendrite grain boundary in the dilution zone (DZ) of Alloy 182, and then spherical or crack-like oxides were formed in the HAZ of LAS adjacent to the FB; no obvious SCC susceptibility was observed in the non-HAZ region of LAS. The cessation of crack growth occurred when spherical oxides formed in the LAS, while crack-like oxides tended to propagate along the prior austenite grain boundary in the coarse-grained HAZ of LAS. It has been suggested that the microstructure and continuousness between the dendritic grain boundary in the DZ of Alloy 182 and the prior austenite grain boundary in the HAZ of LAS across the FB, that is, the continuousness of the potential crack path, which varied depending on the multiple heat cycles of the welding process, played an important role in the cessation/reinitiation of SCC crack in the FB region.

9:50 AM
Effect of Chloride on General Corrosion and Environmentally Assisted Cracking of Low-Alloy Steel under Oxygenated High-Temperature Water Conditions: *Matthias Herbst*[1]; Armin Roth[1]; Erika Nowak[2]; Ulf Ilg[3]; [1]AREVA NP GmbH; [2]E.ON Kernkraft GmbH; [3]EnBW Kernkraft GmbH

Recent investigations have shown a strong effect of chloride contaminations on the crack growth rate of low-alloy steel (LAS) in oxygenated high-temperature water (HTW). Therefore, a research project was launched to systematically investigate the observed effects. This project focused on investigations on the general corrosion behavior of LAS (German RPV steel 22NiMoCr3-7) without chloride and at different chloride contamination levels up to 50 ppb in oxygenated HTW. Chloride was added either permanently or temporarily to simulate a chloride transient. During these tests, Electrochemical Noise (EN) and Electrochemical Impedance Spectroscopy (EIS) measurements were performed to monitor the electrochemical behavior. After the tests, the specimens were examined macroscopically and microscopically. In addition, the oxide layer thickness was determined using the Focused Ion Beam (FIB) technique.The applied tests clearly revealed a decrease of the oxide layer thickness during permanent chloride contamination. Temporary transients, however, did not cause a long-term memory effect.

PWR Secondary Side/Balance of Plant II

Tuesday AM — Room: Colorado III
August 9, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Carine Mansour, EDF; Samaneh Nouraei, Serco TCS

9:30 AM
Electrochemical Studies of Steam Generator Tube Degradation in the Presence of Thiosulphate: *Lisheng Chi*[1]; Yucheng Lu[1]; [1]Atomic Energy of Canada Limited

Sulphur species in oxidation states between S^{2-} and S^{6+} are known to interfere with the protective oxide films that form on steam generator tubing materials. By assisting in the breakdown of passive films, intermediate oxidation state sulphur species can cause intergranular attack and pitting of steam generator tubing over a wide pH range. Intermediate oxidation state sulphur species have also been observed to induce stress corrosion cracking of sensitized Alloy 600 at low temperatures. This work employed electrochemical methods to investigate the effect of thiosulphate on the degradation of steam generator alloys (Alloy 600, Alloy 690, and Alloy 800). The effect of thiosulphate on steam generator tubing degradation was investigated at 150°C in crevice chemistries containing 0.0015 M sodium thiosulphate simulating the local chemistry environments developed in steam generator crevices or under sludge. The detrimental effect of thiosulphate on the boundary conditions of the recommended ECP/pH zone of Alloy 800 was discussed. Accelerated corrosion tests were also performed at selected potentials to confirm the detrimental effects of thiosulphate on the ECP/pH domain.

9:50 AM
X-Ray Photoelectron Study of the Oxides Formed on Nickel Metal and Nickel-Chromium 20% Alloy Surfaces under Reducing and Oxidizing Potentials in Basic, Neutral and Acidic Solutions: Stewart McIntyre[1]; *Bradley Payne*[1]; [1]University of Western Ontario

The corrosion products produced on polycrystalline Ni metal and Ni-Cr (20%) (NiCr) alloy surfaces exposed to aqueous environments chosen to emulate possible solution conditions in the steam generator (SG) tubing of pressurized water reactors (PWR) were studied using XPS. Additional measurements modelling the distribution of oxidized Ni and Cr species on select alloy specimens were carried out using ToF SIMS. Exposure of Ni metal and NiCr alloy samples to mildly oxidizing potentials in basic solutions resulted in the preferential growth of a ß-Ni(OH)2 phase; driven by the dissolution of metallic Ni at both 25° and 150°C. The presence of ß-Ni(OH)2, Cr(OH)3 and small amounts of a Cr6+-containing oxide on NiCr specimens oxidized under mildly oxidizing conditions at 150°C in neutral solutions suggested that the dissolution of both metallic Ni and Cr followed by the back deposition of the corresponding corrosion products was responsible for oxide growth under these conditions. In acidic media oxide nucleation at 150°C under mildly oxidizing potentials was determined to occur via the dissolution of both Ni and Cr species on NiCr specimens as well. The increased stability of Ni2+ in acidic solution led to a limited precipitation of ß-Ni(OH)2 resulting in the formation of very thin oxides containing higher levels of Cr(OH)3. Reactions on metallic Ni and NiCr surfaces under highly oxidizing potentials resulted in an increase in the NiO content of these films compared to similar exposures carried out at milder oxidation conditions; attributed to accelerated dehydration of the ß-Ni(OH)2 phase. In addition, an increase in the Cr(OH)3 contribution on the alloy surface oxidized at a more oxidative potential suggested a more rapid dissolution of Cr under these conditions; overall, uneven films were formed from these conditions. The composition of the corrosion product formed after an exposure to a highly oxidizing potential was found to be unchanged following a subsequent reaction of equivalent length a much lower oxidizing potential in basic solution.

PWR Oxide Films and Characterization

Tuesday PM — Room: Colorado I
August 9, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Matthew Olszta, Pacific Northwest National Laboratory; En-Hou Han, Institute of Metal Research, Chinese Academy of Sciences

7:30 PM
NRC/EPRI Welding Residual Stress Validation Program (Phase III): Matthew Kerr[1]; Lee Fredette[2]; Howard Rathbun[1]; John Broussard[3]; [1]US Nuclear Regulatory Commission; [2]Battelle Memorial Institute; [3]Dominium Engineering Inc.

The US Nuclear Regulatory Commission (NRC) and the Energy Power Research Institute (EPRI) are working cooperatively under a memorandum of understanding to validate welding residual stress predictions in pressurized water reactor primary cooling loop components containing dissimilar metal (DM) welds. These stresses are of interest as DM welds in pressurized water reactors are susceptible to primary water stress corrosion cracking (PWSCC) and tensile weld residual stresses are one of the primary drivers of this stress corrosion cracking mechanism. The NRC/EPRI weld residual stress (WRS) program currently consists of four phases, with each phase increasing in complexity from lab size specimens to component mock-ups and ex-plant material. This paper discusses Phase III of the WRS characterization program, comparing measured and predicted weld residual stresses profiles through the dissimilar metal weld region of pressurizer safety and relief nozzles removed from the cancelled WNP-3 plant.

7:50 PM
Effect of Dissolved Hydrogen, Surface Conditions and Composition on the Electronic Properties of the Oxide Films Formed on Nickel-Base Alloys in PWR Primary Water: *Abdelhalim Loucif*[1]; Jean-Pierre Petit[1]; Yves Wouters[1]; Pierre Combrade[2]; [1]SIMaP, university of Grenoble; [2]ACXCOR

Nickel based alloys used in primary water of PWR undergo different forms of degradation as cations release and stress crack corrosion. All these phenomena dependent strongly on the properties of the oxide scales developed on the surface of such alloys. In this work, ex-situ Photoelectrochemical technique is used to investigate the influence of dissolved hydrogen, roughness and composition on the electronic properties of oxides formed on nickel alloys in simulated primary water. The experimental results evidence the presence of two semiconducting phases. The main result concerns the effects of dissolved hydrogen on the semiconducting properties of the oxide phase having the highest bandgap. This oxide phase, associated to the protective internal subscale, shifts from n-type at lower content of hydrogen to insulating behavior while the content of hydrogen increases. This new result is discussed and compared to the literature.

8:10 PM
Surface Films Formed on Alloy 600 and Alloy 690 in PWR Primary Water: *Thomas Devine*[1]; [1]University of California, Berkeley

The results of in situ SERS investigations of Alloys' 600 and 690 surface films were combined with the results of a number of ex situ studies conducted by other researchers who used a variety of experimental techniques. Comparing the results of different investigations revealed the films' composition and microstructure were most sensitive to alloy composition and the concentrations of aqueous metal cations (Ni++ and Fe+z). Earlier studies established that saturation concentrations of (Ni++)aq and (Fe+z)aq affect the composition and crystal structure of the films' outer layers. The current investigation indicates that aqueous Ni++ and Fe+z also affect the composition and structure of the films' inner layers. Diffusion Path Analyses is able to qualitatively explain the effects of alloy composition and water chemistry on film microstructure. Our results are relevant to SCC of Alloys 600 and 690 and to cation release from Alloys 600 and 690.

8:30 PM
Characterizing Environmental Degradation in PWRs by 3D FIB Sequential Sectioning: *Sergio Lozano-Perez*[1]; Na Ni[1]; Karen Kruska[1]; Chris Grovenor[1]; Takumi Terachi[2]; Takuyo Yamada[2]; [1]University of Oxford; [2]INSS

Modern dual-column FIB-SEMs are capable of automatically milling and acquiring images that can be used to reconstruct sample volumes in 3D. We will demonstrate that this technique, applied to the environmental degradation of materials in nuclear reactors, is capable of revealing features, phases and/or defects in 3D with nm resolution. In this paper, we have used this technique to characterize surface oxidation and cracking in Zr alloys, the effect of cold work on the oxidation resistance of austenitic steels and crack growth in welded 316L stainless steel.

8:50 PM
High-Resolution Analytical TEM and 3D Atom-Probe Characterization of the Effects of Stress and Cold Work on the Oxidation of 304 Stainless Steel: *Karen Kruska*[1]; Sergio Lozano-Perez[1]; Takumi Terachi[2]; Takuyo Yamada[2]; David Saxey[3]; George Smith[1]; [1]University of Oxford; [2]Institute of Nuclear Safety System, Inc.; [3]The University of Western Australia

Cold-worked 304 stainless steels (SS) can be particularly susceptible to stress corrosion cracking (SCC), but underlying mechanisms are still not properly understood. For this reason, the effects of cold-work and applied stress on the oxidation behaviour of 304SS have been characterized. A set of samples with/without prior cold-work, and with/without stress applied during oxidation, were oxidized in autoclaves under simulated pressurised water reactor (PWR) primary conditions. 3D atom-probe tomography and analytical transmission electron microscopy were used to investigate the local chemistry and microstructure in the different samples tested. Regions containing grain boundaries, twin deformation bands, and matrix material in contact with the environment, were extracted from the coupon specimens with a focused ion beam (FIB) machine. Cavities and hydrogen associated with nickel-rich regions were found ahead of the bulk Cr-rich oxide in some of the samples. The implications of these findings for the understanding of SCC mechanisms will be discussed.

Alloy 718 and X-750

Tuesday PM
August 9, 2011

Room: Colorado II
Location: Cheyenne Mountain Resort

Session Chairs: Bob Carter, EPRI; John Jackson, Idaho National Laboratory

7:30 PM
Microstructure and SCC of Alloy X-750: *Peter L. Andresen*[1]; Martin Morra[1]; Bob Carter[2]; [1]GE Global Research; [2]EPRI

Alloy X-750 is the most common high strength alloy used in light water reactors, although Alloy 718 is also being used in some cases. This study examined the microstructure and SCC response of a core shroud repair component that was delivered to a reactor site, but was never installed. The material was shown to be badly banded, and the crack growth rates in various orientations were shown to be high, with a limited effect of banding or heat treatment condition (e.g., AH, BH or HTH). The observed rates are consistent with prior measurements at low and high corrosion potential on Alloy X-750. Prior data on Alloy 718 using the same conditions and procedures showed that it was also very susceptible to SCC. Initial evaluation of alternative high strength materials show about a two order of magnitude decrease in growth rates.

7:50 PM
Stress Corrosion Cracking and Crack Tip Characterization of Alloy X-750 in Boiling Water Reactor Environments: Jonathan Gibbs[1]; *Ronald Ballinger*[1]; John Jackson[2]; Dieter Isheim[3]; Hannu Hanninen[1]; [1]MIT; [2]Idaho National Laboratory; [3]Northwestern University

The susceptibility of Inconel Alloy X-750, in the HTH heat treated condition, to SCC has been evaluated under simulated BWR aqueous conditions at 288°C. Two chemistry conditions were explored: (1) Normal Water Chemistry (NWC) which consisted of high purity water with approximately 1400 ppb dissolved oxygen and (2) Hydrogen Water Chemistry (HWC) which consisted of high purity water with a hydrogen overpressure. This alloy exhibits reproducible SCC crack growth rate of approximately 1.1x10-7 mm/s in NWC at a stress intensity factor of 27.7 MPaµm and 1.4x10-8 mm/s in HWC in 288°C at a stress intensity factor of 35 MPaµm. This alloy was also tested under HWC conditions at 93°C, but no crack growth was observed for the time and mechanical parameters of the test conditions. Crack and crack tip analysis was conducted using optical, scanning electron and atom probe microscopy.

8:10 PM
SCC Properties of Modified Alloy 718 in BWR Plant: Yoshinori Katayama[1]; *Motoji Tsubota*[1]; Yoshiaki Saito[1]; [1]Toshiba

Modified alloy 718 has been developed as an alternative material to nickel-based alloy X-750 used in the boiling water reactor (BWR). Stress corrosion cracking (SCC) occurred at Jet-Pump beam, one of the components made of alloy X-750, in BWR plant. In order to apply modified alloy 718 to the Jet-pump beam, an evaluation of the SCC initiation time, based on a uni-axial constant load (UCL) test, was carried out. All UCL test specimens in 288C water were not failed after 12,000 hours testing. It was conformed that modified alloy 718 had longer SCC initiation time than that of alloy X-750.

8:30 PM
Influence of Chloride Ions as Contaminants on the Corrosion Behavior of Alloy 718 in Pool Water of Nuclear Power Plants: *Jonathan Hugues*[1]; Christine Blanc[1]; Eric Andrieu[1]; Jean-Marc Cloué[2]; [1]CIRIMAT/ENSIACET/INPT; [2]AREVA, AREVA NP

The electrochemical behavior of alloy 718 in a chloride-containing boric acid (2500ppm) solution was studied to determine the influence of chloride ions as contaminants of pool water of nuclear power plants on the corrosion behavior of the alloy. Experiments were performed at 20°C and 60°C with chloride concentrations from 1.5 to 2900 ppm, using stationary measurements i.e. OCP versus time measurements and plotting of current-potential curves.

After the electrochemical tests, the samples were observed using optical microscopy and scanning electron microscopy. Immersion tests in chloride-containing boric acid solutions were also carried out: samples were immersed for a time as long as 2 months at their corrosion potential and their residual mechanical properties were measured. Results showed that, whatever the chloride concentration, there was no corrosion for samples immersed at their corrosion potential. However, when the samples were polarized, intergranular corrosion might be observed in occluded zones.

Flow Assisted Corrosion

Tuesday PM
August 9, 2011

Room: Colorado III
Location: Cheyenne Mountain Resort

Session Chairs: Derek Lister, University of New Brunswick; George Licina, Structural Integrity Associates, Inc.

7:30 PM
Development of Piping Wall Thinning Screening System and Its Field Application: *Kyung Ha Ryu*[1]; Young Ah Seo[1]; Il Soon Hwang[1]; [1]Seoul National University

A new non-destructive evaluation (NDE) method using direct current potential drop (DCPD) technique has been developed for metal pipes for the detection wall thinning. The method has been showed to be suitable for applications to electric power generation plants where flow accelerated corrosion (FAC) of carbon steel piping is a significant cause of increased maintenance and plant personnel casualty. The wall thinning screening system (WalSS) has been developed in two major phases. In the first phase, the equipotential switching direct current potential drop (ES-DCPD) method was developed for piping wall. In the second phase, in this paper, a quantitative detection criteria was developed. The relative ES-DCPD change of 3.8% has been defined as the screening criteria for wall thinning schematization. The developed WalSS based on ES-DCPD was applied to a moisture separator reheater (MSR) drain line of a commercial nuclear power plant (NPP) during a scheduled overhaul.

7:50 PM
Modelling Material Effects in Flow-Accelerated Corrosion: Pimsiree Phromwong[1]; *Derek Lister*[1]; Shunsuke Uchida[2]; [1]University of New Brunswick; [2]Japan Atomic Energy Agency

The mitigating effects of chromium on flow-accelerated corrosion of carbon steel occur at concentrations in the metal as low as 0.02% and, in some coolant environments, are seen immediately on exposure. We have modeled such effects by including a diffusion barrier at the metal-oxide interface, below the magnetite layer which forms the conventional barrier. This extra barrier is a fixed layer that forms instantaneously with a composition depending upon the chromium concentration. It is very thin, so would be undetectable by normal surface analysis techniques, but has its own properties of porosity, density, etc. The secondary barrier of magnetite behaves as modeled before, in time achieving a steady-state thickness that depends upon its dissolution characteristics and the fluid dynamics of the coolant. By adjusting the properties of the chromium-dependent layer, we have been able to predict the FAC of carbon steel of different chromium contents in typical reactor feed-water environments.

8:10 PM
Flow Accelerated Corrosion of Carbon Steel in the Feedwater System of PWR Plants - Behaviour of Welds and Weld Assemblies: *Carine Mansour*[1]; E.M. Pavageau[1]; J-L. Bretelle[2]; [1]EDF R&D, Materials and Mechanics Department; [2]EDF Power Generation Division

Flow Accelerated Corrosion (FAC) of carbon steel is a phenomenon that has been studied for many years. However, to date, the specific behavior of welds and weld assemblies of carbon steel towards this phenomenon has been scarcely examined. An experimental program of FAC of welds and weld assemblies is conducted by EDF R&D. This paper describes the results obtained on the behavior of weld metal independently of its behavior in a weld assembly as well as the sensitivity to FAC of various weld assembly configurations. Tests are performed in the CIROCO loop which permits to follow the FAC rate by gammametry measurements. Welds are performed by two different methods: Submerged Arc Welding (SAW) and Gas Tungsten Arc Welding (GTAW). The influence of several parameters on FAC of welds is examined: welding method, chromium content and temperature. For weld assemblies, only the impact of chromium content is studied. All the tests are conducted in ammonia medium at pH 9.0 and oxygen concentration lower then 1 ppb. Chemical parameters, as the pH, the conductivity and oxygen concentration, are measured in situ during the test and surface characterizations are performed after the test. The results show that, above 0.15% of chromium, no FAC is detected on the weld metal, which is similar to the base metal behaviour. For the same and lower chromium content, the two types of metal have the same FAC rate. Concerning the temperature effect, for both metals FAC rate decreases with temperature increase. Below 150°C, their behaviour seems to be different. For weld assemblies, the study of different configurations shows that the chromium content is the main parameter affecting the behaviour of the specimens. This paper presents also tests conducted to study the sensitivity to FAC of welded assemblies in presence of an artificial step, in order to study the influence of penetration welds on FAC. Additional tests and modelling studies will be conducted in order to validate the experimental results.

8:30 PM
Parameters Evaluation of Liquid Droplet Impingement Erosion in the Plant Piping System: *Kyung Ha Ryu*[1]; Won Chang Nam[1]; Il Soon Hwang[1]; [1]Seoul National University

Component materials in the steam environment of power generation systems have problems of erosion caused by the impingement of droplets. An evaluation of erosion rate induced by liquid droplet impingement (LDI) is important in maintenance of power plant systems and long term operation. A water-jet peening system was designed to evaluate the erosion incubation period, threshold velocity and the erosion rate. The size and velocity of droplet is controlled and monitored. By impinging droplets with 10~300 meter per second velocity, the erosion rate is examined. Damage depths are evaluated by scanning electron microscope and atomic force microscope. These results are compared with weight loss results and online direct current potential drop (DCPD) results. To evaluate the erosion resistance of materials, fatigue strength coefficients of materials are suggested as well as hardness of materials. These parameters will be used to model the LDI erosion and to design the piping system.

PWR Alloy 600 Oxidation and Mechanisms I

Wednesday AM — Room: Colorado I
August 10, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Catherine Guerre, CEA; Peter Chou, Electric Power Research Institute

8:15 AM
Degradation of Grain Boundary Strength by Oxidation in Alloy 600: *Katsuhiko Fujii*[1]; Terumitsu Miura[1]; Hiromasa Nishioka[1]; Koji Fukuya[1]; [1]Institute of Nuclear Safety System

The degradation of grain boundary strength induced by corrosion is one of the causes of intergranular cracking. The micro tensile testing method for measuring the strength of individual grain boundary was applied to alloy 600 specimens exposed to simulated PWR primary water at 360°C for 2700 h or less. The grain boundaries in the specimens showed oxidation with about 0.1 μm width and over 2 μm depth depending on the exposure time. Specimens of 1×2×4 μm having one grain boundary were made by focused ion beam (FIB) micro-processing and tensioned in an FIB system. The intergranular fracture occurred at 280-350 MPa in the specimens having the oxidized grain boundary while it did not occur at 1000 MPa in the specimen having a non-oxidized grain boundary. It was confirmed that the cracking propagated on the interface of the metal matrix and the oxide by TEM observations.

8:35 AM
Evaluation of the Oxygen Diffusion Coefficient in Nickel-Base Alloys: *Hyo On Nam*[1]; Jae Young Yoon[1]; Il Soon Hwang[1]; Kyu Hwan Lee[2]; [1]Seoul National University; [2]Korea Institute of Science and Technology

Nickel-base alloys such as alloy 600 (Ni-16Cr-9Fe) are known to exhibit intergranular stress corrosion cracking (IGSCC) at pressurized water reactor (PWR) primary water environments. From the microscopic observations, it was found that oxygen plays a role in primary water stress corrosion cracking (PWSCC) of nickel-base alloys and Scott suggests an internal oxidation model. However, it was found that needed oxygen diffusivity to explain the internal oxidation model should be several orders greater than the measured oxygen diffusivity. In this study, oxygen diffusion coefficients in the nickel-base alloys were evaluated by atomistic modeling of oxygen diffusion process based on the proposed vacancy-mediated diffusion model. Density functional theory is used to calculate the energy of a system. Activation barrier energy of diffusion of atomic oxygen is quantified by finding minimum energy path through the most favorable path. Phonon analysis is performed using the direct force-constant method.

8:55 AM
Stress Corrosion Cracking of Alloy 600 in PWR Primary Water: Influence of Chromium, Oxygen and Hydrogen Diffusion: *Catherine Guerre*[1]; Laghoutaris Pierre[1]; Chetroiu Bogdan[1]; Chene Jacques[2]; Marchetti Loic[1]; Molins Régine[3]; Duhamel Cécilie[3]; [1]CEA; [2]CEA - CNRS; [3]Mines Paristech

Alloy 600, a nickel base alloy containing 15 % chromium, is used in primary circuit of pressurized Water Reactor (PWR). This alloy is well known to be susceptible to Stress corrosion cracking (SCC) in PWR primary water. Despite the fact that many laboratory studies have been performed and that many models are proposed in the literature, the mechanisms involved are still not well known. In the proposed model, the transport of species (oxygen, hydrogen and chromium) is considered as playing a key role. Therefore, experiments and calculations are performed in order to study the transport of chromium, hydrogen and oxygen in Alloy 600 and in model alloys with or without strain. The results lead to the conclusion that the transport of oxygen and hydrogen can not be considered as the rate-controlling step. Finally, the influence of species transport in the SCC mechanisms is discussed.

9:15 AM
Grain Boundary Oxidation and Embrittlement Prior to Crack Initiation in Alloy 600 in PWR Primary Water: *Lionel Fournier*[1]; Olivier Calonne[1]; Pierre Combrade[2]; Peter Scott[3]; Peter Chou[4]; [1]AREVA NP; [2]ACXCOR; [3]Peter Scott Corrosion Consultant; [4]EPRI

The influence of temperature, time of exposure, corrosion potential, cold work, and applied stress on intergranular oxidation in Alloy 600 is being investigated. A significant increase in the depths of intergranular oxide penetration was observed with exposure time, the maximum depth increasing from 2.2 µm after 1500h up to 5 µm after 4500h. The extent of embrittlement resulting from intergranular oxidation has been characterized on micro-tensile specimens after oxidation by quantifying the evolution of the percentage of cracked grain boundaries (GB) as a function of strain at room temperature. The corrosion potential was observed to have a very significant influence on the proportion of cracked GB. A maximum in grain boundary embrittlement was observed at 30 kPa of hydrogen partial pressure while little and almost no embritlement was observed respectively at less than 1 kPa and 650 kPa.

9:35 AM Break

IASCC Stainless Steels CGR I

Wednesday AM — Room: Colorado II
August 10, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Karen Gott, Matsafe AB; Yiren Chen, Argonne National Laboratory

8:15 AM
Crack Growth Behavior of Irradiated Type 316 SS in Low Dissolved Oxygen Environment: *Yiren Chen*[1]; Bogdan Alexandreanu[1]; Yong Yang[2]; William Shack[1]; Ken Natesan[1]; Eugene Gruber[1]; Appajosula Rao[3]; [1]Argonne National Laboratory; [2]University of Florida; [3]US Nuclear Regulatory Commission

Cracking susceptibility of austenitic stainless steels is known to be affected by dissolved oxygen (DO) or corrosion potential. In low-DO environments, crack growth rate (CGR) is significantly lower than that in high-DO environment. A strong dependence of CGR on corrosion potential has also been seen in irradiated SSs. While it has been shown that reducing the potential reduced the CGRs of irradiated SSs, some high-dose specimens have shown elevated CGRs even in low potential environments. Thus, it is not clear how irradiation affects the dependence of CGR on corrosion potential. In this study, a disk-shaped compact tension specimen of Type 316 SS was tested in low-DO environment. The specimen was irradiated in the BOR-60 reactor to 5 dpa at 320°C. Post-irradiation CGR and fracture toughness tests were performed. The effect of unloading on crack growth behavior in low-DO environment is discussed.

8:35 AM
Stress Corrosion Crack Initiation Susceptibility of Austenitic Stainless Steels Irradiated in the Bor-60 Fast Breeder Reactor: *Kale Stephenson*[1]; Yugo Ashida[1]; Jeremy Busby[2]; Gary Was[1]; [1]University of Michigan; [2]Oak Ridge National Lab

The susceptibility of austenitic stainless steels irradiated in the Bor-60 reactor to the initiation of irradiation-assisted stress corrosion cracking is assessed. High purity 304 alloys with solute additions of Mo, Si and Hf were strained by constant extension rate testing (CERT) in simulated BWR NWC water chemistry. Samples were deformed at a rate of 3.5 x 10-7/s to 1% strain and then to failure. Analysis of sample surfaces using scanning electron microscopy (SEM) was conducted after both strain interruption and failure. Cracking susceptibility from CERT is compared to that from crack growth rate (CGR) tests on identical alloys in the same environment. Results of CERT tests on neutron-irradiated alloys are also compared with those from proton-irradiated samples of the same alloys tested under the same conditions. The paper will focus on the cracking susceptibility in CERT and the comparison with CGR and CERT tests on proton-irradiated samples.

8:55 AM
Stress Corrosion Cracking Behavior of Type 304 Stainless Steel Irradiated under Different Neutron Dose Rates at JMTR: *Yoshiyuki Kaji*[1]; Keietsu Kondo[1]; Yoshiteru Aoyagi[1]; Yoshiaki Kato[1]; Taketoshi Taguchi[1]; Fumiki Takada[1]; Junichi Nakano[1]; Hirokazu Ugachi[1]; Takashi Tsukada[1]; Kenichi Takakura[2]; Hiroshi Sakamoto[2]; [1]JAEA; [2]JNES

In order to investigate the effect of neutron dose rate on tensile property and irradiation stress corrosion cracking (IASCC) growth behavior, the crack growth rate (CGR) test, tensile test and microstructure observation have been conducted with type 304 stainless steel specimens. The specimens were irradiated in high temperature water simulating the temperature of boiling water reactor (BWR) up to about 1dpa with two different dose rates at the Japan Materials Testing Reactor (JMTR). The radiation hardening increased with the dose rate, but there was little effect on CGR. Increase of the yield strength of specimens irradiated with the low dose rate condition was caused by the increase of number density of Frank loops. Little difference of radiation-induced segregation at grain boundaries was observed in specimens irradiated by different dose rates. Furthermore, there was little effect on local plastic deformation behavior near crack tip in the crystal plasticity simulation.

9:15 AM
In-Pile Tests for IASCC Growth Behavior of Irradiated 316L Stainless Steel under Simulated BWR Condition in JMTR: *Yasuhiro Chimi*[1]; Shigeki Kasahara[1]; Hideo Ise[1]; Yoshihiko Kawaguchi[1]; Junichi Nakano[1]; Yutaka Nishiyama[1]; [1]Japan Atomic Energy Agency

Japan Atomic Energy Agency (JAEA) has a plan of irradiation tests by using Japan Materials Testing Reactor (JMTR), in order to evaluate the effects of change in material properties and water chemistry caused by the neutron/gamma-ray irradiation on SCC growth of stainless steel from the view points of the integrity of reactor core internals for BWR. The difference of SCC growth behavior and its ECP dependence between in-pile and out-of-pile tests is still unknown because of lack of in-pile data which is comparable with out-of-pile database. This paper presents a systematic review on SCC growth data of irradiated stainless steels and the outline of the in-pile test plan for IASCC growth behavior of irradiated 316L SS under simulated BWR condition in the JMTR, together with the development of the in-pile test technique.

9:35 AM
Crack Growth Rates of Irradiated Commercial Stainless Steels in BWR and PWR Environments: *Anders Jenssen*[1]; Johan Stjärnsäter[1]; Raj Pathania[2]; [1]Studsvik Nuclear AB; [2]Electric Power Research Institute

Crack growth rate testing was performed on CT specimens with doses in the range ~10-47.5 dpa. Two specimens of Type 304L (same heat) were tested in BWR and PWR environments, with the objective to compare the CGR behavior of fast reactor irradiation with BWR irradiation. Three specimens of heats tested previously, but at other doses, were tested for assessment of neutron dose and K on IASCC. One specimen of Type 304L was tested in BWR NWC and HWC at two different K levels, while two specimens of cold worked Type 316 were tested at various K levels and temperatures in PWR primary water. To assess the effect of temperature on IASCC, two specimens were tested in either BWR NWC and HWC or PWR primary water at different temperatures. The paper will discuss the effects of fast reactor versus light water reactor irradiation, K, ECP, dose and temperature on the CGR.

9:55 AM Break

PWR Water Chemistry and Mitigation

Wednesday AM — Room: Colorado III
August 10, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Jeffrey Gorman, Dominion Engineering Inc.; Jean-Luc Bretelle, EDF Nuclear Operation Division

8:15 AM
Introduction to a New Real-Time Water Chemistry Measurement System: *Jei-Won Yeon*[1]; Kyuseok Song[1]; Myung-Hee Yun[1]; [1]KAERI

In reactor water chemistry, there are three major chemical factors: pH, redox potential, electrical conductivity. The pH is conventionally evaluated by Li and B concentrations. Instead of redox potential, the concentration of dissolved hydrogen is generally measured to evaluate the redox condition. The conductivity data show the information on impurities. Until now, these methods have been used for measuring chemical factors in NPPs. However, these methods have inherent limitations as real-time monitoring means, because they should be accompanied by the sampling. In order to overcome this limitation, we developed a water chemistry measurement system for real-time monitoring. We adopted a ceramic-based pH electrode as a pH sensor. Pt electrode was used to measure redox potential. And, with potential transient technique, Pt electrode was also used for measuring the concentration of corrosive species. In addition, we enhanced the credibility of high temperature chemical data by confirming the relationship between chemical factors.

8:35 AM
Comparison of DBU, NH3, DMA, ETA, and Morpholine Interactions with Ferrous Chloride Solution and Carbon Steel Surfaces: Seifollah Nasrazadani[1]; Ansel Reid[1]; *Jim Stevens*[2]; Robert Theimer[2]; Billy Fellers[3]; [1]University of North Texas; [2]Luminant; [3]TXU

Objectives of this investigation were to compare the interaction of advanced amines namely DBU, NH3, DMA, ETA, and morpholine with FeCl2 and to characterize oxide morphology on steel exposed to amine containing FeCl2 solution. Aerated solutions containing100 ppm of DMA or ETA showed the highest solution pH. DMA containing solution showed the highest resistance to flocculation as indicated by the longer time required to floc. A solution containing 100 ppm DBU did not show any flocculation. Oxide morphology on steel samples exposed to an acidic 100 ppm DBU containing solutions showed fine round (about 0.1 μm in diameter) oxide particles.

8:55 AM
Role of Dissolved Hydrogen in Water in Corrosion of Alloy 600 in High Temperature Water: *Qunjia Peng*[1]; Tetsuo Shoji[1]; Juan Hou[1]; Kazuhiko Sakaguchi[1]; Yoichi Takeda[1]; [1]Tohoku University

Characterization of the microstructure and chemical composition of oxide films formed on Alloy 600 in high temperature water with various dissolved hydrogen (DH) levels that allowed the Ni/NiO phase transition to occur were conducted. The results showed that increasing DH in water decreased the thickness of the oxide film, in conjunction with the increase of Cr- but decrease of Ni-concentrations in the oxide film. Further, it was found that in the NiO regime the DH deteriorated the stability and protectability of the oxide film. The DH effect was attributed to an oxidative dissolution process of Ni following the reduction of Ni in the oxide by hydrogen, which suggests the corrosion of Ni-base alloys in high temperature hydrogenated water is dominated by a synergic process of dissolution and oxidation. Results of this work contribute to a better understanding of the mechanism of the DH dependence of SCC in Ni-base alloys in high temperature water.

9:15 AM

Quantifying the Benefit of Chemical Mitigation of PWSCC via Zinc Addition or Hydrogen Optimization: *Chuck Marks*[1]; Matthew Dumouchel[1]; Richard Reid[2]; Glenn White[1]; [1]Dominion Engineering, Inc.; [2]Electric Power Research Institute

This EPRI study quantified the benefits of zinc addition and hydrogen optimization to facilitate modification of inspection intervals for PWR pressure boundary components susceptible to PWSCC. Available experimental and plant data on the effects of such chemical mitigation on PWSCC initiation and crack growth were reviewed. After assessing the statistical confidence in these results, the benefits were quantified. Zinc addition was demonstrated to have a strong mitigative effect on PWSCC initiation of Alloy 600. There is some evidence for a benefit of zinc for crack growth of Alloy 600 at low stress intensity factors. Hydrogen optimization was demonstrated to have a strong mitigative effect on crack growth, particularly in Alloy 82 and 182. The study developed expressions for calculating factors of improvement in initiation times and growth rates, and defined equations and parameters necessary for probabilistic modeling of the benefit of chemical mitigation in the context of relevant regulatory frameworks.

9:35 AM Break

PWR Alloy 600 Oxidation and Mechanisms II

Wednesday AM
August 10, 2011

Room: Colorado I
Location: Cheyenne Mountain Resort

Session Chair: François Vaillant, EDF

10:10 AM

Atom Probe Tomography and Transmission Electron Microscopy Characterizations of Grain Boundary Composition and Intergranular Attack in Alloy 600 Materials: *Matthew Olszta*[1]; Stephen Bruemmer[1]; [1]Pacific Northwest National Laboratory

Atom probe tomography (APT) and analytical transmission electron microscopy (ATEM) has been used to investigate grain boundary composition in four heats of alloy 600 tubing with different susceptibility to IGSCC in PWR primary water environments. Distinct differences are observed in grain boundary carbide distributions and chromium depletion along with segregation of boron, silicon and phosphorus. Comparisons are made between IGSCC susceptibility and the atomistic measurements of grain boundary composition. In addition, APT and ATEM examinations are performed evaluating IG attack in several of these same alloy 600 heats after exposure to PWR primary water. Penetrative oxidation along grain boundaries is characterized as well as high-resolution examinations of grain boundaries immediately ahead of the oxidation front. Nanometer-scale sulfide formation is detected at grain boundaries ahead of the IG oxidation in selected materials. Results are assessed with respect to proposed mechanisms of primary water SCC in these alloys.

10:30 AM

Grain Boundary Oxidation and Crack Tip Oxide Structures in Nickel-Base Alloys Strained in High-Temperature Water: *Tyler Moss*[1]; Matthew Olszta[2]; Gary Was[1]; [1]University of Michigan; [2]Pacific Northwest National Lab

The objective of this study is to characterize grain boundary oxidation and crack tip oxides in nickel-base alloys to gain a better understanding of intergranular stress corrosion cracking mechanisms in PWR primary water. Accelerated stress corrosion cracking tests were conducted on alloy 600, alloy 690, and Ni-xCr-9Fe alloys in constant extension rate mode in supercritical water at 400°C and in low pressure steam at 400°C using hydrogen addition to control the corrosion potential. Corrosion coupons were included in both of these conditions to study the oxide structure and the possibility for intergranular oxidation. Hydrogen was added to study the effect of oxide phase stability on cracking and corrosion structures. NiO is stable in deaerated supercritical water and steam; Ni is stable in hydrogenated supercritical water and hydrogenated steam. This paper will discuss the observed grain boundary oxidation and crack tip oxide structures and their impact on stress corrosion cracking mechanisms.

10:50 AM

Origin of the Resistance to SCC of Alloy 800 and Adjacent Compositions in the FeCrNi System: *Roger Newman*[1]; Zoe Coull[1]; Suraj Persaud[1]; [1]University of Toronto

Since 2007, we have promoted the idea that Alloy 800 resists two specific SCC mechanisms - one based on dealloying or a similar selective oxidation mechanism, typical of ordinary stainless steels; the other based on internal oxidation of Cr or Fe, typical of Alloy 600 in PWR primary water. Here we present data on both of these mechanisms, using high-resolution analytical (S)TEM of Fe-rich and Ni-rich alloy surfaces and associated weld metals under high-temperature conditions that promote either dealloying or internal oxidation. To simulate the PW environment, we have used steam/hydrogen mixtures at 400°C (high pressure) and 480°C (low pressure). In such conditions, a beneficial effect of dilution of Ni-based weld metal by carbon steel is observed, despite the reduction in Cr content.

IASCC Stainless Steels CGR II

Wednesday AM
August 10, 2011

Room: Colorado II
Location: Cheyenne Mountain Resort

Session Chairs: Lionel Fournier, AREVA NP; Wade Karlsen, VTT Technical Research Centre of Finland

10:10 AM

The Key Factors Affecting Crack Growth Behavior of Neutron-Irradiated Austenitic Alloys: *Yugo Ashida*[1]; Peter Andresen[2]; Gary Was[1]; [1]University of Michigan; [2]GE Global Research

The key factors affecting crack growth behavior of neutron irradiated stainless steel were investigated in this study. A crack growth rate (CGR) test was conducted on a neutron-irradiated (9.6 dpa) 8 mm RCT specimen of high purity 316L stainless steel with Hf addition in simulated NWC, HWC, and PWR environments. The test was conducted at two temperatures, 320°C and 288°C, and at K levels of 13 and 18 $MPa\cdot m^{1/2}$. The effects of water chemistry, temperature, and K on CGR were examined in detail. In addition, the CGR results from this test were compared with those reported by the previous Cooperative IASCC Research (CIR) program to determine the effect of the addition of Hf and irradiation dose on crack growth rate.

10:30 AM

Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steel WWER Reactor Core Internals: *Anna Hojna*[1]; Miroslava Ernestova[1]; Ossi Hietanen[2]; Ritva Korhonen[2]; Ludmila Hulinova[3]; Ferenc Oszvald[4]; [1]Nuclear Research Institute Rez; [2]Fortum Power and Heat Oy; [3]NPP Dukovany, Czech Power; [4]Paks NPP

The neutron irradiation changes the material's microstructure and mechanical properties, basis of occurrence and increased sensitivity to IASCC. This paper surveys new results regarding IASCC of irradiated austenitic Ti-stabilized stainless steel 08Ch18N10T from WWER 440 Greifswald decommissioned after 15 years in service. Three components LWR irradiated to 2-19 dpa were tested. IASCC was investigated by Slow Strain Rate and Crack Growth Rate tests in simulated water 320°C. IASCC was judged according IG+TG fracture occurrence. Without irradiation components do not suffer SCC in the water. However, areas of mixed IG and TG fracture appeared on specimens. Tests represent different stress strain conditions for IASCC initiation and growth. Effects of SSRT strain rate and CGR test load level were found to be significant for IASCC. Total IG+TG fraction of SSRT ranged 1 to 18% and decreased with decreasing strain rate. Results are compared with previous data on the fast reactor irradiated material.

10:50 AM

Slow Strain Rate Tensile Tests of Irradiated Austenitic Stainless Steels in Simulated PWR Environment: *Yiren Chen*[1]; Bogdan Alexandreanu[1]; William Soppet[1]; William Shack[1]; Ken Natesan[1]; Appajosula Rao[2]; [1]Argonne National Laboratory; [2]US Nuclear Regulatory Commission

Irradiation-assisted stress corrosion cracking is of concern for the safe and economic operation of light water reactors. In this study, cracking susceptibility of austenitic stainless steels was investigated by using slow strain rate tensile (SSRT) tests in a simulated pressurized water reactor (PWR) environment. The specimens were irradiated to 5, 10, and 48 dpa in the BOR60 reactor at 320°C. The SSRT results showed that yield strength was increased significantly in irradiated specimens while ductility and strain hardening capability were decreased. Irradiation hardening was found to be saturated below 10 dpa. The irradiated yield strength of cold-worked specimens was higher than that of solution-annealed specimens. Fractographic examinations were also performed on the tested specimens, and the dominant fracture morphology was ductile dimples. Intergranular cracking was rarely seen on the fracture surface. Transgranular cleavage cracking, however, was found more frequently on the specimen tested in simulated PWR environment.

11:10 AM

Irradiation-Assisted Stress Corrosion Cracking of Stainless Steels in Pressurized Water Reactor (PWR) Environment: *Morgane Le Millier*[1]; Olivier Calonne[2]; Jérôme Crépin[1]; Cécilie Duhamel[1]; Lionel Fournier[3]; Fabrice Gaslain[1]; Eva Héripré[4]; André Pineau[1]; Ovidiu Toader[5]; Yoann Vidalenc[2]; [1]Centre des Matériaux, Mines-ParisTech, CNRS UMR 7633; [2]AREVA NP, Centre Technique; [3]AREVA NP-Tour AREVA; [4]Laboratoire de Mécanique des Solides CNRS UMR 7649; [5]Michigan Ion Beam Laboratory, University of Michigan

This work deals with the study of the irradiation assisted stress corrosion cracking. The purpose is to improve the understanding of irradiation effects on the mechanical behaviour of internal stainless steel structures in PWR environment. The aim is to identify the physical mechanisms responsible for plasticity and damage in relation with the microstructure heterogeneities and to separate the effects induced by the chemical environment to those due to mechanical loading and irradiation. Proton-irradiations are performed in the Michigan Ion Beam Laboratory. Slow Strain Rate Tensile tests are performed both in PWR primary water and under inert atmosphere to study the deformation mechanisms of irradiated samples and their coupling with the environment. To correlate cracking to microstructure, full field analysis are performed, at a microscale, by SEM digital imaging correlation technique coupled with EBSD cartography of the grain orientations after irradiation. Moreover, cracking features are characterized using SEM and TEM.

11:30 AM

A Preliminary Hybrid Model of Irradiation-Assisted Stress Corrosion Cracking of 300 Series Stainless Steels in PWR Primary Environments: *Ernest Eason*[1]; Gabriel Ilevbare[2]; Raj Pathania[2]; [1]Modeling & Computing Services LLC; [2]EPRI

The hybrid model of PWR primary water stress corrosion cracking (PWSCC) in unirradiated Ni alloys presented at the 2009 Environmental Degradation meeting has been extended to irradiated stainless steels in PWR primary environments. The IASCC model is an empirical/theoretical hybrid strain rate model that combines submodels developed by various investigators. The major differences from the PWSCC model include using the Rice-Drugan-Sham (RDS) theoretical expression for strain rate near a growing crack in elastic-perfectly plastic materials and including an empirical dose function. The RDS strain rate expression is appropriate for highly-irradiated materials that show little or no strain hardening. The dose function incorporates many investigators' observations that irradiation has little effect on SCC below some low dose, then an increasing effect as dose increases until there is no further effect above a high-dose saturation level.

Super Critical Water

Wednesday AM | Room: Colorado III
August 10, 2011 | Location: Cheyenne Mountain Resort

Session Chairs: Todd Allen, University of Wisconsin-Madison; Sebastien Teysseyre, Idaho National Laboratory

10:10 AM

Computational Thermodynamics for Interpreting Oxidation of Structural Materials in Supercritical Water: *Lizhen Tan*[1]; Ying Yang[2]; Todd Allen[3]; Jeremy Busby[1]; [1]Oak Ridge National Laboratory; [2]CompuTherm LLC; [3]University of Wisconsin-Madison

Supercritical water-cooled reactor (SCWR) is one of the advanced nuclear reactors being developed to meet the soaring energy demand. The corrosion resistance of structural materials used in SCWR becomes one of the major concerns as the operation conditions being raised up to ~600°C and ~25 MPa. Oxidation has been observed as the major corrosion behavior. To mitigate the oxidation corrosion, stabilities of metals and oxides need to be understood with respect to environmental temperature and oxygen partial pressure. Computational thermodynamics provides a practical approach to assess phase stabilities of such multi-component multi-variable systems. In this study, calculated phase stability diagrams of alloys and corresponding oxides were used to guide the interpretation of oxidation behaviors of SCW-exposed structural materials. Examples include ferritic-martensitic steel, austenitic steels and Ni-base alloy, e.g., HCM12A (Fe-12Cr), D9 (Fe-15Cr-15Ni), 800H (Fe-21Cr-32Ni), and 690 (Ni-30Cr-10Fe). Calculated results are in good overall consistence with the experimental data.

10:30 AM

Stress Corrosion Cracking of Austenitic Alloys in Supercritical Water: *Guoping Cao*[1]; Vahid Firouzdo[1]; Todd Allen[1]; [1]UW-Madison

Tapered uniaxial tensile samples under constant load and bent ring samples under constant strain were used to investigate the stress corrosion cracking (SCC) of Alloy 600 and 690 in supercritical water. To verify the effectiveness of the tapered tensile sample design, the SCC of two 304 stainless steels with high and low carbon content were tested in supercritical water at 400C, 25MPa and 10-15ppb dissolved oxygen. Both testing methods are effective to initiate SCC in 304SS, Alloy 600 and 690. Based on the bent ring SCC testing, the SCC resistance of Alloy 690 in supercritical water is lower than Alloy 600. This unexpected lower SCC resistance is probably due to the much lower tensile ductility (less than 10%) of the tested Alloy 690 as compared with the standard 40-50% ductility and the severe plastic deformation and strain in the bent ring samples. The large TiN particles and their inhomogeneous distribution in the alloy may help facilitate the SCC initiation in this specific Alloy 690 in the bent ring samples. SCC is more likely to initiate from the Electrical Discharge Machined (EDM) or end mill machined surfaces than in ground and polished surfaces.

10:50 AM

Comparison of the Oxidation Behavior of the 14CrODS Alloy in Steam and Supercritical Water: *Jeremy Bischoff*[1]; Arthur Motta[1]; [1]Penn State University

The Supercritical Water Reactor is a GenerationIV design envisioned for its high thermal efficiency and simplified core. One of the major materials issues for the development of this reactor is the corrosion resistance of the cladding and structural materials exposed to supercritical water at a temperature between 500ºC and 600ºC. Ferritic-martensitic steels are promising candidate materials for such an application but their corrosion resistance has to be optimized. We studied the oxide microstructure of these alloys using a combination of micorbeam synchrotron radiation diffraction and fluorescence and electron microscopy. The synchrotron analysis yields both elemental and phase information with remarkable spatial resolution, while electron microscopy enables us to visualize microstructural features observed with the synchrotron.

Furthermore, unique oxide structure information was obtained using TEM EELS-mapping. The combination of these complimentary techniques has shed light on the oxidation mechanism of these steels in SCW, which will be discussed in this paper.

11:10 AM
Grain Boundary Engineering and Air Oxidation Behavior of Alloy 690: *Peng Xu*[1]; Liang Zhao[1]; Eric Schneider[1]; Kumar Sridharan[1]; Todd Allen[1]; [1]University of Wisconsin

Alloy 690 is now recognized as the material of choice for steam generating tubing in LWRs due to its excellent oxidation resistance and ability to resist stress corrosion cracking compared to other structural materials in high temperature oxidative environments. It is also envisioned as an important candidate structural material for Gen IV nuclear reactors. Grain boundary engineering (GBE) was used to modify the grain boundary character distribution of alloy 690. Samples were subject to air oxidation at 650°C and supercritical water oxidation at 500°C for exposure durations of 2 to 12 weeks. Oxidation of the samples was evaluated by weight change measurements. The phase evolution and microstructural developments in the oxide layers were characterized using diffraction, microscopy and spectroscopy techniques such as XRD, SEM, EDS, EBSD, and TEM. The grain orientation-oxidation rate relationships and the effects of GBE on oxidation resistance of Alloy 690 will be discussed.

PWR Alloy 600 SCC I

Wednesday PM — Room: Colorado I
August 10, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: David Morton, Knolls Atomic Power Laboratory

1:05 PM
Strain Path Effect on SCC Initiation and Oxidation of Ni Base Alloys Exposed to PWR Primary Water: *Thierry Couvant*[1]; Laurent Legras[1]; Thierry Ghys[1]; Nicolas Huin[1]; Gabriel Ilevbare[1]; [1]EDF R&D

A detailed understanding of strain path effect on PWSCC and oxidation is of great importance for the prediction of the initiation of SCC of Ni base alloys 600,182 and 690 exposed to the primary water of PWRs. Thus, a cross specimen was used in order to quantify the effect of a change of strain path on the strain localization and increase the understanding of the contribution of the strain hardening and the strain incompatibilities on the precursors for initiation of SCC. Examinations indicated the deleterious effect of the strain localization due to a change of strain path on intergranular SCC susceptibility, especially on weld metal 182. Intergranular cracks initiated where high stress levels due to strain incompatibilities were expected. Oxides were characterized using TEM. The resistance to failure of grain boundaries affected by oxidation was estimated.

1:25 PM
An Experimental Study of Short Stress Corrosion Crack Coalescence in Alloy 600 in PWR Primary Water: O. Calonne[1]; *Lionel Fournier*[1]; P. Combrade[2]; P. Scott[2]; P. Chou[3]; [1]AREVA NP; [2]Consultant; [3]EPRI

An experimental study of short stress corrosion crack initiation, growth and coalescence has been performed on two Alloy 600 heats with different susceptibilities to PWSCC using ovalized tube specimens. The number of cracks initiated was observed to be much higher on the most susceptible heat while cracks were fewer but much deeper on the material with the lower susceptibility. Surface crack growth rates were observed to be much lower than expected from measurements on CT specimens and many cracks tended to become dormant after an initial short crack propagation stage. The coalescence criterion proposed by Parkins was found to be appropriate for the description of PWSC crack coalescence. The implications of these results for the prediction of time to failure of a component will be discussed.

1:45 PM
Mechanistic Study on PWSCC of Ni-Based Alloys Using Hump-SSRT Tests under Dry Hydrogen Gas Environment, Simulated Primary Water and Rapid Straining Electrode Test in Simulated Primary Water: *Nobuo Totsuka*[1]; Takumi Terachi[1]; Kenichi Takakura[2]; [1]Institute of Nuclear Safety System, Incorporated; [2]Japan Nuclear Energy Safety Organization

Hump-SSRT tests of alloy 600 have been carried out under simulated primary water and dry hydrogen gas at the temperature ranging from 360 to 320°C and rapid straining electrode tests of nickel based alloys have been carried out under same water at 320°C with and without dissolved hydrogen (DH). The followings are clarified. 1) Inter Granular (IG) cracking is observed not only under the water but also under hydrogen gas and activation energy of PWSCC is almost same as that of IG cracking under hydrogen gas. 2) In rapid straining tests, repassivation could not be observed and only cathodic current is observed under the condition with DH. These cathodic current reaches some plateau value and the current of alloy 690 is about 1/5 of alloy 600 and 132.According to these results, it is suggested that the basic mechanism of PWSCC is one kind of hydrogen embrittlement.

2:05 PM
Crystallographical Characterization of Initiation of Intergranular Stress Corrosion Cracking of Alloy 600 in PWR Environment: *Shinji Fujimoto*[1]; Masahito Mochizuki[1]; Yuya Morita[1]; Yoshiki Mikami[1]; Hiroaki Tsuchiya[1]; Kazutoshi Nishimoto[1]; [1]Osaka University

Susceptibility of intergranular stress corrosion cracking of Alloy600 in simulated primary water environment of pressurized water reactor were evaluated by means of slow strain rate testing. Annealed and 10 or 20 % cold rolled Alloy600 flat tensile specimens were mechanically polished using SiC papers, then finished with Colloidal Silica. The tensile specimen was elongated for 10 % then left for few tens hours at the constant elongation using SSRT at 360 C with dissolved hydrogen of 2.75 ppm. Many IGSCC cracks were observed on the surface of the tensile specimen. The surface of tensile specimens were characterized by means of EBSD to analyze the crystallographical feature of the inter granular crack initiation. Alloy600 tends to exhibit IGSCC at around 30 degree of misorientation angle. Most of crack initiates at random grain boundaries. On the other hand, coincidence boundaries, such as S3, S9, exhibit no cracks.

2:25 PM
Environmental Effects on PWSCC Initiation and Propagation in Alloy 600: Anders Molander[1]; Kjell Norring[1]; Per-Olof Andersson[2]; *Pål Efsing*[2]; [1]Studsvik Nuclear AB; [2]Ringhals AB

This paper will present new PWSCC results from initiation tests and crack growth measurements on Alloy 600 in primary PWR environment. The effects of hydrogen, lithium and boron have been studied. For hydrogen new initiation data in the higher range of the specified PWR interval (25 to 50 cc/kg) will be presented. For Li initiation tests at contents above 3.5 ppm has been performed. The new data will be compared to existing data at lower hydrogen and lower Li contents. In addition crack growth measurements using different Li/B contents to simulate a reactor cycle and study any variations with the different composition has been performed. The results will be compared with data from beginning of cycle conditions usually used for PWSCC studies. Finally, environmental effects on initiation and crack growth are compared.

2:45 PM Break

Irradiation Effects on Deformation

Wednesday PM — Room: Colorado II
August 10, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Hannu Hanninen, Aalto University School of Science and Technology; Katsuhiko Fujii, Institute of Nuclear Safety System

1:05 PM
Effect of Slip Behavior on Irradiation Assisted Stress Corrosion Cracking in Austenitic Steels: *Michael McMurtrey*[1]; Gary Was[1]; [1]The University of Michigan

Irradiation assisted stress corrosion cracking may be linked to the local slip behavior near grain boundaries that exhibit high susceptibility to cracking. Three austenitic steels with varying degrees of cracking susceptibility were irradiated with 2 MeV protons at 360°C to 5 dpa and strained in 288°C simulated BWR conditions. Both cracking and deformation behavior were then characterized. Deformation behavior was characterized by Taylor and Schmid factors, slip continuity across grain boundaries, the number of activate slip systems, and the angle between dislocation channels and the applied stress. Grain boundary cracking susceptibility was found to correlate strongly with slip continuity, indicating that the two are strongly linked. Higher cracking susceptibility was also found at grain boundaries bounded by grains with low Schmid factor or high Taylor factor. This paper will focus on the role of these deformation behaviors on IASCC susceptibility.

1:25 PM
Cause-and-Effect Relationship between Localized Deformation and IASCC: *Wei-Jen Lai*[1]; Zhijie Jiao[1]; Gary Was[1]; [1]University of Michigan

Recent studies have shown that an excellent correlation exists between localized deformation and IASCC, however, a cause-and-effect relationship has yet to be established. Alloys selected in this study include a commercial purity alloy CP-304 and a high purity alloy, Fe-15Cr-12Ni, both of which are susceptible to IASCC. Samples were irradiated with protons to 5 dpa and strained in argon to 2% to generate localized deformation in dislocation channels and grain boundaries. After characterization of the degree of localized deformation, such as channel height, channel spacing, grain boundary sliding, etc., the samples were strained to an additional 1% in simulated BWR NWC at 200°C to induce cracks. Crack initiation was examined against the magnitude of localized deformation at the same location to determine if localized deformation leads to cracking. This paper will discuss the role of localized deformation in initiating cracks in irradiated stainless steels.

1:45 PM
Influence of Localized Plasticity on IASCC Sensitivity of Austenitic Stainless Steels under PWR Primary Water: *Sarata Cisse*[1]; Eric Andrieu[2]; Benoit Tanguy[1]; Lydia Laffont[2]; Marie Christine Lafont[2]; Catherine Guerre[1]; [1]CEA; [2]CIRIMAT

Although IASCC mechanisms are not well understood, the literature indicates that the effects of irradiation are an important base mechanism of IASCC. Specific effects include radiation induced segregation, radiation induced hardening, and deformation mode changes. Our study focuses on the latter. Deformation heterogeneity is manifested in the form of clear bands, microtwins, and stacking faults ribbons. This work focuses on AISI 304 and A286 steels and explores the effect of localized plasticity on the sensitivity to SCC and the interaction of oxidation mechanisms and localized plastic deformation. The concept is to replicate an irradiated microstructure in each material via proton irradiation and low cycle fatigue treatment. After this step, Slow Strain Rate Tests are performed to characterise the sensitivity of the microstructure to SCC. In parallel, the study aims to enhance comprehension of mechanisms governing oxidation kinetics and considers microstructural modifications developed during localized plastic deformation.

2:05 PM
Deformation Microstructures of 30 dpa AISI 304 Stainless Steel after Monotonic Tensile and Constant Load Autoclave Testing: *Wade Karlsen*[1]; Janne Pakarinen[1]; Aki Toivonen[1]; Ulla Ehrnstén[1]; [1]VTT Technical Research Centre of Finland

Irradiated AISI 304 stainless steel extracted from the Chooz A center filler assembly has been the subject of a number of studies. Previously the results of slow strain rate tensile and constant load autoclave tests of 30 dpa material have been reported. They showed an influence of temperature, strain rate and environment on the fracture behavior of the material. The irradiated microstructure and deformation microstructures of those materials following testing have now been examined by TEM. The findings suggest that substantial channel deformation was associated with the purely ductile fracture following SSRT testing in argon, while the intergranular fractures following SSRT in simulated PWR environment and constant load testing in both simulated PWR and argon environments were associated with very localized deformation primarily exhibiting alpha martensite. This possibility is discussed in the light of literature.

2:25 PM Break

PWR Degradation Management

Wednesday PM — Room: Colorado III
August 10, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: C. E. Carpenter, U.S. Nuclear Regulatory Commission; Ian Armson, Rolls Royce

1:05 PM
Proposed Coordinated U.S. PWR Reactor Vessel Surveillance Program: An Updated Summary Including Program Optimization: *Ryan Hosler*[1]; Sarah Davidsaver[1]; Timothy Hardin[2]; Dennis Weakland[2]; Greg Troyer[1]; [1]AREVA NP; [2]EPRI

Irradiated reactor pressure vessel (RPV) surveillance data is used to predict decreases in RPV fracture toughness due to irradiation embrittlement. A limited amount of data at fluences that many U.S. PWR RPVs will reach in 60 or more years of operation exists today. However, there is a significant amount of test reactor data available at high fluences, which shows higher embrittlement shifts than the power reactor data-based correlations. This has significant implications for plant operation to 60 years. A coordinated plan for withdrawal and testing of the U.S. PWR RPV surveillance capsules is being developed, with the intent of filling high fluence gaps in existing PWR data. This paper summarizes the methodology, optimization strategy, and current results of this coordinated U.S. PWR reactor vessel surveillance program (RVSP). The coordinated RVSP has been optimized to maximize the quantity and quality of high fluence data while minimizing the burden on the industry.

1:25 PM
Developing PWR Aging-Management Strategies for Reactor Vessel Internals: Sarah Davidsaver[1]; Stephen Fyfitch[1]; Hongqing Xu[1]; [1]AREVA NP Inc

Managing materials' aging degradation issues is of high importance to the long-term safety and reliability of major components as current PWRs age. Many U.S. utilities have completed the process of renewing their operating license for an additional twenty years. While doing so, they committed to develop aging-management programs and inspection plans. The U.S. PWR industry is proactively developing generic inspection requirements and standards for reactor vessel (RV) internals. This paper describes AREVA NP's efforts – specifically for B&W-designed units – during the last twenty years, to assist in developing a comprehensive aging-management program for RV internals to fulfill previously made regulatory commitments.

1:45 PM

NRC Research to Support Regulatory Decisions Related to Subsequent License Renewal Periods: *C. E. Carpenter*[1]; [1]U.S. Nuclear Regulatory Commission

The U.S. Nuclear Regulatory Commission (NRC) staff, in collaboration with the U.S. Department of Energy (DOE), the domestic industry, and international partners, is developing an integrated aging management research plan ("Life Beyond 60"), which will focus on those areas covered by 10 CFR Part 54 that may need additional technical information to provide regulatory assurance of the capabilities of the nuclear power plant (NPP) structures, systems, and components (SSCs) and related materials to maintain their safety-related functionality in the second, and subsequent, license renewal periods. This presentation will discuss the activities to date, including results, and the path forward.

2:45 PM

Development of the Extremely Low Probability of Rupture (xLPR) Code: *David Rudland*[1]; Craig Harrington[2]; [1]US NRC; [2]EPRI

10 CFR 50 Appendix A General Design Criteria (GDC) 4 requires that primary piping systems exhibit an extremely low probability of rupture in order to exclude dynamic effects associated with postulated primary pipe ruptures. The Leak-Before-Break (LBB) methodology, as described in NRC Standard Review Plan (SRP) 3.6.3, was developed to meet this goal. Per SRP 3.6.3, active degradation mechanisms are not permitted in systems approved for LBB. However, Pressurized Water Reactors (PWRs) are currently experiencing Primary Water Stress Corrosion Cracking (PWSCC). To resolve this inconsistency for the long term, NRC began a cooperative research program with the Electric Power Research Institute to develop a probabilistic assessment tool (xLPR) that quantitatively assesses the probability of primary piping system rupture and can accommodate such active degradation mechanisms. This paper provides an overview of the xLPR program, focusing on the cooperative structure used for model development and results from the proof-of-concept pilot study.

2:25 PM

Databases of Operationally Induced Damage: *Karen Gott*[1]; [1]Swedish Radiation Safety Authority

The Swedish Radiation Safety Authority set up its database of opetationally induced damage of mechanical components in the early 1990s. The paaper will provide an analysis of the database illustrating among otehr things system and tim edependence of degradation in Swedish nuclear power plants.The Swedish model has been the basis of two similar international databases projects and one planned project orgainesed by the OECD Nuclear Energy Agency. These will be used for comparisons and as examples of the extended potential of such databases as tools for both regulators and utilities.

2:45 PM Break

PWR Alloy 600 SCC II

Wednesday PM	Room: Colorado I
August 10, 2011	Location: Cheyenne Mountain Resort

Session Chairs: Denise Paraventi, BMPC - Bettis Laboratory; Damien Deforge, Electricite de France- EDF

3:05 PM

Probabilistic Environmentally-Assisted Cracking Modeling for Primary Water Stress Corrosion Cracking of Alloy 600: Tae Hyun Lee[1]; *Jae Young Yoon*[2]; Hyo On Nam[2]; Il Soon Hwang[2]; [1]Samsung Engineering Co. Ltd.; [2]Nuclear Material Laboratory, Seoul National University

Environmentally assisted cracking(eac) has been studied intensively during several decades, but its mechanisms are not well defined because of its detection. Therefore it is necessary to predict it using probabilistic estimation. In this study, probabilistic eac(peac) model is developed and it deals with primary water stress corrosion cracking(pwscc), one of the severe eac problems, based on alloy 600. In peac modeling, bayesian parameter updating is applied to decrease uncertainty of parameters using probability of detection. According to this model, it is possible to predict crack growth rate and crack distribution using monte carlos simulation and its results can be comparable with real data. The probability of failure can be estimated from developed failure criteria. Adopting ri-isi in a pwr, peac model results are applicable to estimate the risk reduction.

3:25 PM

The Role of Lattice Curvature on the SCC Susceptibility of Alloy 600: *Fabien Leonard*[1]; Fabio Di Gioacchino[1]; Robert Cottis[1]; Francois Vaillant[2]; Joao Quinta Da Fonseca[1]; Florence Carrette[2]; Gabriel Ilevbare[3]; [1]The University of Manchester; [2]Electricite de France; [3]Electric Power Research Institute

Stress corrosion cracking (SCC) of alloy 600 is regarded as one of the most important challenges to primary water reactor operation worldwide, as most nickel base alloys employed in nuclear power plants are subject to SCC. This study investigates the link between microstructure and stress corrosion cracking susceptibility for three heats of alloy 600 representative of plant components (forged control rod drive mechanism nozzle, rolled divider plate, and rolled divider plate stub). The experimental approach was designed to determine the effect of the manufacturing process (forged vs. rolled materials) and the cold work (as-received vs. cold-worked materials) on SCC of alloy 600. Results showed the carbide distribution to be the main microstructural parameter influencing SCC but a modified crystal misorientation parameter, in synergy with the carbide distribution, has proven to give a better representation of the materials SCC susceptibilities.

3:45 PM

PWSCC Susceptibility in Heat-Affected Zones of Alloy 600: *Thierry Couvant*[1]; Thomas Brossier[1]; Christian Cossange[1]; [1]EDF R&D

The recent field experience and several experimental results have shown the possible deleterious effect of a heat affected zone (HAZ) induced by welding on the susceptibility to the stress corrosion cracking (SCC) of Alloy 600 of bottom penetrations exposed to primary water of PWRs. This work tried to quantify the increasing susceptibility to initiation and crack propagation in 600/182 HAZ. The rolled plate did not exhibit any susceptibility to SCC except for a cold work higher than 10% typically. By contrast, the weld metal was well known for its high susceptibility to SCC. Metallurgical and mechanical characterizations of the HAZ indicated a slight gradient of Vickers micro hardness close to the fusion line (up to few mm) and a lack of intergranular precipitates up to 500µm from the fusion line. SCC tests clearly demonstrated that a non-susceptible plate may exhibit a significant susceptibility to SCC propagation in the HAZ. Results of initiation tests did not allow to observe any SCC in the base metal, due to the high susceptibility to SCC of the weld.

4:05 PM

Quantitative Residual Strain Analyses on Strain Hardened Nickel Based Alloy: *Toshio Yonezawa*[1]; Yuichi Miyahara[1]; Hiroshi Kanasaki[2]; [1]Tohoku University; [2]Mitsubishi Heavy Industries, Ltd.

Many papers have been reported about the effects of strain hardening by cold rolling, grinding, welding, etc. on stress corrosion cracking susceptibility of nickel based alloys and austenitic stainless steels for LWR pipings and components. But, the residual strain due to cold rolling, grinding, welding, etc. is not so quantitatively evaluated. So, authors quantitatively measured and evaluated on the residual strain for strain hardened tensile and torsion specimens of nickel based alloys by X-ray diffraction and hardness measurement to make a calibration curve, at first. Using this curve, the residual strain in the heat affected zone of weld joint was quantitatively measured and that value was evaluated as less than 30% of von Mises equivalent strain. But the residual strain of the 30% cold rolled material was evaluated about 60% of von Mises equivalent strain. 15% cold rolling is thought to be conservatively enough for evaluating the effects of residual strain in the area of strain hardened weld joint on the stress corrosion cracking susceptibility in LWR environment.

4:25 PM
The Study of Stress Corrosion Cracking on Alloy 600 C-Ring Samples by Polychromatic X-Ray Microdiffraction: *Stewart McIntyre*[1]; Jing Chao[1]; Marina Suominen Fuller[1]; Roger Newman[2]; Anatolie Carcea[2]; Martin Kunz[3]; Nobumichi Tamura[3]; [1]University of Western Ontario; [2]University of Toronto; [3]Lawrence Berkeley National Laboratory

Microscopic strains associated with stress corrosion cracks (SCC) have been investigated in stressed C-rings of Alloy 600 boiler tubing. Polychromatic X-ray Microscopy (PXM) was used to measure deviatoric strain tensors and the distribution of dislocations near cracks that had been propagated in electrochemically-accelerated corrosion tests. SCC-generated intergranular cracks were produced in two Alloy 600 specimens after 6h and 18h tests. The diffraction patterns and resultant strain tensors were mapped around the cracked area to a one micron spatial resolution. The strain tensor transverse to the crack growth direction showed tensile strain at the intergranular region just ahead of the crack tip for both specimens. Both cracks were found to follow grain boundary pathways that had the lowest angle of misorientation. Dislocation distributions within each grain were qualitatively obtained from the shapes of the diffraction spots: Where similar dislocations characteristics failed to appear in opposing grains, the intersecting boundary was considered prone to dislocation pileup and render that boundary more brittle to fracture. Generally, grain boundaries that were followed by the cracks had these characteristics.

Irradiation Effects – General I

Wednesday PM — Room: Colorado II
August 10, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Gary Was, University of Michigan; Frank Garner, Radiation Effects Consulting

3:05 PM
Proton Irradiation of 316 Stainless Steel Pre-Oxided in Simulated PWR Environment: *Zhijie Jiao*[1]; Gary Was[1]; [1]University of Michigan

In a PWR, the oxide film evolves under the influence of irradiation. The persistent irradiation damage in the oxide film induced by irradiation may facilitate the transport of ions through the film leading to an overall higher corrosion rate. Furthermore, because of the low ductility of the oxide film, small stresses or strains due to defect accumulation may cause failure of the film. To study irradiation damage in the oxide film, 316 SS samples were exposed in 320ºC primary water for 150 h followed by irradiation to 5 dpa at the same temperature using 2 MeV protons. The morphology and structure of the irradiated oxide film were examined using SEM and cross-sectional TEM and radiation-induced segregation at the grain boundary and the oxide/metal interface were characterized using STEM/EDS. The irradiation effect on the oxide film will be reported and its subsequent influence on oxide film evolution will be discussed.

3:25 PM
Comparison of In-Reactor Creep and Stress Relaxation of Cold Worked 316 and Solution Annealed 304L Stainless Steels in Thermal and Fast Neutron Spectrum Reactors: *John Foster*[1]; Torill Karlsen[1]; [1]OECD Halden Project

In-reactor creep and stress relaxation measurements were performed on CW 316 and SA 304L SS in the Halden Reactor. The initial results were reported at the last conference. All the measurements were performed on-line during irradiation. This is the first time that irradiation stress relaxation measurements have been performed on-line during irradiation. In the case of CW 316 SS N-Lot and SA 304L SS, these materials were irradiated in the fast neutron spectrum test reactor EBR-II. Creep and stress relaxation measurements were performed on the same tube lots in the Halden Reactor. This paper will review the EBR-II test data and update the Halden data. The results show that in-reactor creep and stress relaxation are different in fast and thermal neutron spectrum reactors. Further, the differences are material dependent. These results show that Halden data should be used for LWR applications.

3:45 PM
Recent Insights on the Parametric Dependence of Irradiation Creep of Austenitic Stainless Steels: *Frank Garner*[1]; E. R. Gilbert[2]; Victor Neustroev[3]; [1]Radiation Effects Consulting; [2]Pacific Northwest National Laboratory; [3]Research Institute of Atomic Reactors

Irradiation creep and creep relaxation are important phenomena in determining the performance of structural components not only in light-water reactors but also in heavy water reactors, fast reactors and devices based on fusion or spallation spectra. Development of predictive correlations for irradiation creep requires a full understanding of influential variables be included in the correlation effort. Of particular interest are the dependencies on stress, stress state, stress history, irradiation temperature, dpa rate, neutron spectra, gas generation, composition, metallurgical starting state and void swelling. A review is presented of recent data, analyses, insights and also reinterpretations of earlier data on irradiation creep and swelling-creep interactions. It is shown that when all operating variables are taken into account, and when non-creep strains are separated from creep strains, irradiation creep can be described by a remarkably simple expression that is independent of stress state, irradiation temperature and dpa rate over a very wide range. Correct and consistent definition of dose is very important.

4:05 PM
Cluster Dynamics Prediction of the Microstructure Evolution of 300-series Austenitic Stainless Steel under Irradiation: Influence of Helium: *Mohamed Zouari*[1]; Lionel Fournier[1]; Alain Barbu[2]; Yves Brechet[3]; [1]AREVA NP; [2]Alain BARBU Consulting; [3]SIMAP

It is well established that Helium produced by transmutation stabilizes small cavities. However, this stabilizing effect may either inhibit void growth if small cavities are the dominant sink, or otherwise favor void growth. The evolution of the other elements of the microstructure with dose also contributes to void growth and the understanding of such complex processes clearly requires physics-based model. A new cluster dynamics model considering the mean number of helium atoms in cavities has been specifically developed in order to predict the influence of helium on cavity microstructure evolution and void swelling of austenitic stainless steels up to high dose. The influence of helium on cavity microstructure evolution of Solution Annealed and Cold Worked type 316 stainless steels was predicted under irradiation conditions that are relevant to HFIR (525°C, $1x10^{-6}$ dpa/s, 40-70 appm Helium/dpa) and PWRs (300°C, $5x10^{-8}$ dpa/s, 4-20 appm Helium/dpa). Results are compared with available literature data.

PWR Initiation and CGR Stainless Steels I

Thursday AM — Room: Colorado I
August 11, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Koji Arioka, Institute of Nuclear Safety System; Qunjia Peng, Institute of Metal Research

8:15 AM
Intergranular Cracking in Austenitic Fe-Cr-Ni Alloys during Creep: *Young Suk Kim*[1]; Sung Soo Kim[1]; Dae Whan Kim[1]; [1]Korea Atomic Energy Research Institute

This work aims at understanding the role of creep in intergranular stress corrosion cracking (IGSCC) in austenitic Fe-Cr-Ni alloys. To this end, creep tests were conducted on 316L stainless steel with nitrogen at 550 to 650°C over a stress range of 120 to 360 MPa. Regardless of the test temperature and the nitrogen content, 316L stainless steel showed intergranular (IG) cracking during creep even in air. To understand the effect of creep deformation on a change in the lattice spacing, the lattice spacing of a 40% cold-worked 316L stainless steel were determined using a neutron diffractometer with aging time in a temperature range of 300 to 700°C. The lattice contraction occurred on aging above 300°C with the maximum contraction of 0.17% at 600°C. It shows that the lattice contraction accompanying plastic deformation above 300°C is the cause of not only IG cracking during creep but IGSCC in austenitic Fe-Cr-Ni alloys.

8:35 AM
Understanding the Limits of Lattice Orientation Data Analysis in Environmental Degradation Studies: *Fabio Di Gioacchino*[1]; David Wright[1]; Joao Quinta da Fonseca[1]; Fabio Scenini[1]; [1]The University of Manchester

It is well established that cold work induced plastic strain can significantly increase the susceptibility of metals to environmental assisted cracking (EAC). However, the reasons for this increase susceptibility are still unclear. This is due, in part to the difficulty in quantifying and modeling plastic deformation at the required scale. Here, we use a new experimental procedure to study the local microstructural distribution of strain in 304L stainless steel. Digital image correlation (DIC) was used to map strain at the microstructural level with sub-micron resolution. The results clearly show that that a high degree of strain localization develops within individual grains, in the form of highly localized shear bands and micro-twinning. Electron back scattered diffraction (EBSD) was used to quantify the lattice orientation changes in the same area. Analysis of this data included grain reconstruction and the calculation of kernel average misorientation (KAM) and orientation gradients maps. Comparisons with the DIC data clearly show that in most cases, there is no evidence in the EBSD analysis of the high levels of strain measured using DIC. This has important implications in the use of lattice orientation data in the study of the effects of plastic deformation on environmental assisted cracking.

8:55 AM
A Microstructural Investigation on the Effect of Cold Work on Environmentally Assisted Cracking of Austenitic Stainless Steels: *David Wright*[1]; Fabio Di Gioacchino[1]; João Quinta da Fonseca[1]; Fabio Scenini[1]; [1]University of Manchester

In recent years, cold deformation has been shown to induce some degree of susceptibility to environmentally assisted cracking of austenitic stainless steels in light water reactor environments. These studies have identified a number of interesting relationships between the degradation behaviour and, for example, material parameters and the history of deformation. However, specific dependencies between the cold worked microstructure and the occurrence of cracking are not well understood. It is believed that the heterogeneity of deformation, and the localisation of strain may be significant. In the present work, extensive characterisation of the deformed microstructure of austenitic stainless steel, employing high resolution electron backscattered diffraction has been conducted. The stress corrosion cracking response of the material is discussed with relevance to the microstructure.

9:15 AM
Plastic Strain and Residual Stress Distributions in an AISI 304 Stainless Steel BWR Pipe Weld: *Tapio Saukkonen*[1]; Miikka Aalto[1]; Iikka Virkkunen[2]; Ulla Ehrnsten[3]; Hannu Hanninen[1]; [1]Aalto University School of Science and Technology; [2]Trueflaw Ltd.; [3]VTT Technical Research Centre of Finland

In AISI 304 stainless steel pipe welds weld shrinkage causes large variations in residual plastic strain in different parts of the weld metal and heat-affected zone (HAZ). The amount of strain was analyzed by EBSD quantitatively by comparing the intra-grain misorientations to the calibration curve. Highest degrees of plastic strain (10…20%) were detected in the HAZ close to the root area of a prototypical BWR plant weld. Strain in the weld metal varies in the different directions of solidification, being high in the weld bead boundaries and near the fusion lines. Preliminary studies of the effects of mechanical and elastic anisotropy of the weld metal microstructure on the grain size level were performed by EBSD and nanoindentation. The residual stress distribution in the same weld cross-section was determined by a contour method. The residual strain and stress distributions are superimposed and EAC susceptibility of various areas of the pipe weld is evaluated and discussed.

9:35 AM
Applicability of Lean Grade of Duplex Stainless Steels in Nuclear Power Plants: *Julie Tucker*[1]; Daniel Eno[1]; George Young[1]; [1]Knolls Atomic Power Laboratory

Duplex stainless steels are desirable for use in power generation systems due to their attractive combination of strength, corrosion resistance, and cost. However, thermal embrittlement at intermediate temperatures via α' precipitation limits their applicability. Recently, the development of 'lean grade' alloys may increase allowable service temperatures by delaying the α-α' phase separation. The present work compares the thermal embrittlement kinetics of selected lean grade alloys and their weld filler metals to the most widely used duplex alloy, 2205. Embrittlement of the alloys was assessed via isothermal agings between 500°F and 1000°F for times up to 10,000 hours. The degree of embrittlement was quantified by microhardness and impact toughness testing. Microhardness data were fit to an Avrami-type equation to predict α' formation for times up to 60 years. We find lean grade alloys are much more resistant to thermal embrittlement than alloy 2205, indicating a broader applicability to nuclear power systems.

9:55 AM Break

Irradiation Effects – General II

Thursday AM — Room: Colorado II
August 11, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Aladar Csontos, U.S. Nuclear Regulatory Commission; Richard Reid, EPRI

8:15 AM
Irradiation Microstructure of Austenitic Steels and Cast Steels Irradiated in the BOR-60 Reactor at 320°C: *Yong Yang*[1]; Yiren Chen[2]; Yina Huang[3]; Todd Allen[3]; Appajosula Rao[4]; [1]University of Florida; [2]Argonne National Laboratory; [3]University of Wisconsin-Madison; [4]US Nuclear Regulatory Commission

Reactor internal components are subjected to neutron irradiation in light water reactors, and with the aging of nuclear power plants around the world, irradiation-induced material degradations is of concern for reactor internals. Irradiation defects resulting from displacement damage are critical for irradiation-induced degradations in structural materials. In the present work, irradiation microstructure of austenitic stainless steels and nickel alloys was characterized using a transmission electron microscopy. The specimens were irradiated in the BOR-60 reactor, a fast breeder reactor, up to ~40 dpa at ~320°C. The dose rate was approximately 1E-6 dpa/s. Void swelling and irradiation defects were analyzed for these specimens. The irradiated microstructures

were dominated by a high density of faulted loops. Along with previous TEM results, a dose dependence of defect structure was established at ~320°C. The results of irradiation defect structure were correlated with the changes of yield strength obtained from slow strain rate tensile tests.

8:35 AM
Radiation-Induced Segregation and Precipitation/Transformation at Very High Fluences under Extended Service: *Edward Kenik*[1]; Jeremy Busby[1]; [1]Oak Ridge National Laboratory

Radiation-induced segregation (RIS) to grain boundaries and dislocation loops has been observed in a variety of austenitic alloys under light water reactor (LWR) irradiation conditions. Irradiation-induced changes of alloy microstructure have been shown to lead to embrittlement. At higher temperatures more typical of fast reactor irradiations, RIS contributions to radiation-induced/enhanced precipitation (e.g., γ', G and η phases) and matrix phase instability (γ—>α) are well documented. As higher dose data for LWR conditions becomes available there are indications that phase transformations may also be an important consideration for extended service. This paper will present a comprehensive review of phase transformation observations under LWR-relevant conditions and identify key operating regimes susceptible to this form of degradation. In addition, the results of new analytical characterization of austenitic stainless steels irradiated to high fluence will also be presented.

8:55 AM
Neutron Dose Rate Effect on Radiation Hardening of Type 316L Stainless Steel: Yuji Kitsunai[1]; Tadahiko Torimaru[2]; Shigeaki Tanaka[1]; T. Nakamura[3]; K. Asano[4]; Suguru Ooki[4]; [1]Nippon Nuclear Fuel Development, Co., Ltd; [2]Toshiba Corporation; [3]Hitachi-GE Nuclear Energy, Ltd.; [4]Tokyo Electric Power Company

Radiation hardening behavior of austenite stainless steel has been estimated as a function of dose, since most of data had been obtained under accelerated irradiation in test reactor or in core area of commercial LWR due to limitation of irradiation time. It is important to verify the dose rate influence to radiation hardening for core internal integrity. The materials irradiated up to 0.3-1.4 dpa with the dose rate of 10-9-10-8 dpa/s in commercial BWRs were prepared to investigate the dose rate effect on the radiation hardening of type 316L stainless steel (SS). Micro Vickers hardness tests and microstructure observations using TEM were performed on the materials. These results were compared to reference data of which materials were irradiated with higher dose rate of around 10-7 dpa/s in test reactor. Dose rate effect was found on radiation hardening of type 316L SS in the dose range of 0.3-1.4 dpa. In the dose range of 0.3-1.4 dpa, the amount of radiation hardening in micro Vickers hardness increased with increasing dose rate. Numerical concentration of dot like defect and Frank loop increased with dose rate. Average diameter of dot like defect and Frank loop decreased with dose rate. The radiation hardening of core-internal materials in commercial BWRs is considered to be suppressed comparing with the materials irradiated with higher dose rate in test reactor at the same dose up to the dose of approximately 2 dpa.

9:15 AM
Influence of Neutron Irradiation on Deformation-Induced Martensite Transformation in Western (AISI 304, AISI 316) and Russian (12Cr18Ni10Ti, 08Cr16Ni11Mo3) Stainless Steels: O. P. Maksimkin[1]; S. V. Ruban[1]; M. S. Merezgko[1]; S. V. Rybin[1]; J. T. Busby[2]; *M. N. Gussev*[2]; F. A. Garner[3]; [1]Institute of Nuclear Physics; [2]Oak Ridge National Laboratory; [3]Radiation Effects Consulting

Austenitic stainless steels are prone to martensite formation during deformation, with increasing tendency as the nickel-equivalent decreases and temperature decreases. However, the question of how irradiation impacts the steel's stability is not well-defined. Some recent papers show that neutron irradiation tends to accelerate martensite instability.We used two ways to probe the origins of instability and its influence on mechanical properties. First, we use specimens exposed to different damage doses attained in HFIR, BOR-60, BN-350 and WWR-K reactors. Second, we employ non-irradiated specimens that were severely deformed and then annealed over a wide range of temperatures to produce a variety of starting microstructures. A novel magnitometry method was developed to measure the absolute amount of martensite in small regions of miniature specimens. The critical stresses and critical strains required to produce martensite, and also the kinetics of martensite accumulation, are studied as a function of damage dose, material starting condition, temperature and type of deformation (tensile, compression, indentation).

9:35 AM Break

PWR Initiation and CGR Stainless Steels II

Thursday AM — Room: Colorado I
August 11, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Gabriel Ilevbare, Electric Power Research Institute; Thierry Couvant, EDF R&D

10:10 AM
Effects of Thermo-Mechanical Treatments on Deformation Behavior and IGSCC Susceptibility of Stainless Steels in PWR Primary Water Chemistry: *Samaneh Nouraei*[1]; David Tice[1]; Kevin Mottershead[1]; David Wright[2]; [1]Serco TCS; [2]University of Manchester

Field experience of 300 series stainless steels in primary circuit of PWR has been good. Stress Corrosion Cracking of components has been infrequent and mainly associated with contamination by impurities/oxygen in occluded locations. However, some instances of failures have been observed which cannot be attributed to deviations in the water chemistry. These failures appear to be associated with the presence of cold work produced by surface finishing and/or by welding-induced shrinkage. Recent data indicate that some heats of SS show an increased susceptibility to SCC; relatively high crack growth rates were observed even when the crack growth direction is orthogonal to the cold work direction. SCC of cold worked SS in PWR coolant is therefore determined by a complex interaction of material composition, microstructure, prior cold work and heat treatment. This paper will focus on the interactions between these parameters on both crack initiation and propagation in simulated PWR conditions.

PWR Field Experience I

Thursday AM — Room: Colorado I
August 11, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Anne Demma, EPRI; James Hyres, B&W Technical Services Group, Inc.

10:30 AM
Stress Corrosion Cracking Propagation in a Superficial Cold Work Layer in SG Divider Plates in Alloy 600: *François Vaillant*[1]; Thierry Couvant[1]; Salem Miloudi[1]; Yannick Thebault[1]; Damien Deforge[1]; Emmanuel Lemaire[1]; [1]EDF

Steam Generator Divider Plates (SGDPs) of EDF 900 MWe plants have encountered short SCC cracks (depth less than 2 mm). The objective of the present investigation was to demonstrate that a crack initiated in a superficial cold worked layer (< 2 mm) in a SGDP would stop before it reached the bulk material. Archive materials were susceptible to SCC in laboratory when the pre-strain was higher than 0.07. Additionally, constant displacement tests were performed on notched shot-peened tensile specimens in primary water (4000 h, 360°C) using a crack monitoring. Cracks initiated at the notch and in the smooth parts under the weld spots of the monitoring wires. Depths reached 450-650 µm and hardness at the crack tips 230-250 HV0.1, corresponding to a strain of 0.07-0.08, in accordance with examinations performed on a removed SGDP.It seems reasonable to assume that short cracks in plants were arrested.

10:50 AM

Destructive Examinations on Divider Plates from Decommissioned Steam Generators Affected by Superficial Stress Corrosion Cracks: *Salem Miloudi*[1]; Erwan Firmin[1]; Damien Deforge[1]; Francoise Vaillant[1]; Emmanuel Lemaire[1]; [1]EDF

Nickel Based alloys Stress Corrosion Cracking has been a major concern for all The Nuclear Power Plants utilities since the seventies. From 2002, new cases of Stress Corrosion Cracking (SCC) were reported on Steam Generator Divider Plates; however, no evidence of propagation following the first detection has never been observed. EDF has conducted since 2008 an extensive program of examinations on two Chinon B1 decommissioned SG divider plates affected by SCC. It constitutes a world first. Developed up to 2010 in the EDF hot laboratory, this program permits in particular to reach two main objectives: - Correlate non destructive examinations and representative defects, - Confirm and Characterize SCC damage on SG divider plate (relationship between morphology and microstructure). This work was completed by metallurgical investigations, mechanical tests and SCC tests. All these elements enabled a better understanding of the SCC degradation and contributed to an optimization of maintenance policy.

11:10 AM

Investigations on Core Basket Bolts from a VVER 440 Power Plant: *Ulla Ehrnsten*[1]; Petri Kytömäki[2]; Ossi Hietanen[2]; Alpo Neuvonen[2]; [1]VTT; [2]Fortum Power

NDE investigations using ultrasonic inspections were performed on core basket bolts at two VVER 440 units. Bolts with indications were removed and exchanged. Destructive investigations were performed on most of the removed basket bolts. These investigations comprised of microstructural investigations using optical microscopy, hardness measurements and scanning electron microscopy of fracture surfaces. The bolts are M12 bolt manufactured from solution annealed Ti-stabilised stainless steel. The results from these investigations are reported and the reasons for the observed indications are evaluated.

11:30 AM

Laboratory Analysis of a Reactor Coolant Pump Seal: Michael Sullivan[1]; *James Hyres*[2]; [1]American Electric Power; [2]B&W Technical Services Group, Inc.

This paper describes the results of a laboratory analysis performed on a reactor coolant pump (RCP) seal that experienced elevated leak-off temperatures after three months in service. Analyses included visual examinations, dimensional measurements, scanning electron microscopy (SEM), energy dispersive spectroscopy (EDS), metallography, Vickers microhardness, gamma spectroscopy, and x-ray diffraction. It was determined that the elevated leak-off temperatures were caused by a buildup of orange-brown deposits on the seal surfaces. These deposits contained primarily iron and oxygen along with 8-10% chromium. Sequential gamma spectrographic analysis through the deposit layer indicated the deposits were all approximately the same age, 200+ days, based on the Co-58/Co-60 ratios. The laboratory data indicated deposition of corrosion products, not corrosion/degradation of the seal, was the most plausible mechanism for the deposit buildup. The source of the deposits appeared to be a Type 410 stainless steel component in the system external to the RCP seal.

11:50 AM

PWSCC of Thermocoax Pressurizer Heaters in Austenitic Stainless Steel and Remedial Actions to Preventing SCC: Jacky Champredonde[1]; Yannick Thebault[1]; Philippe Moulart[1]; Thierry Couvant[2]; *Karine Dubourgnoux*[1]; Yves Neau[2]; Jean-Marie Fageon[3]; Denis Lecharpentier[4]; A. Breuil[4]; Viviane Derouet[4]; [1]EDF/CEIDRE; [2]EDF/R&D; [3]EDF/DPN-UNIE; [4]THERMOCOAX

Limited cases of SCC have been observed in French PWRs, in high strain hardened and non-sensitized austenitic stainless steels exposed to primary environment. Intergranular SCC has been observed on several pressurizer heaters. Thus, a R&D program has been associated to hot laboratory investigations, in order to identify the root causes of the degradation, to understand the mechanisms responsible for SCC in nominal primary water, and to improve the resistance of heaters. Based on these results, an experimental program between EDF R&D and CEIDRE in collaboration with heater supplier Thermocoax was made in the development of surface annealing by Induction heating to reduce the cold-working and the residual stresses and therefore, to minimize the susceptibility to PWSCC of sheath material while preserving electrical properties of mineral insulating material. The combination of optimized parameters and process industrialization has produced positive results for the prevention of crack initiation.

BWR Water Chemistry and Mitigation I

Thursday AM — Room: Colorado II
August 11, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Young-Jin Kim, GE Global Research Center; Juan Varela, GE Hitachi Nuclear Energy

10:10 AM

Developments in SCC Mitigation by Electrocatalysis: *Peter L. Andresen*[1]; Young Kim[1]; [1]GE Global Research

Electrocatalysis is a very effective SCC mitigation approach in oxidizing environments, and requires that a stoichiometric excess of reductants over oxidants be present. This paper summarizes the mechanisms and criteria for effective SCC mitigation, with particular focus on the critical location for the catalyst in a crack and recent experimental support for these concepts. Recent results to optimize the process for continuous use using very low levels (<0.01 ppb) of Pt injection are described.

10:30 AM

Use of Noble Metal Nanoparticle for SCC Mitigation in BWRs: *Young-Jin Kim*[1]; Peter Andresen[1]; Sam Hettiarachchi[2]; [1]GE Global Research Center; [2]GE Hitachi-Nuclear

GE developed NMCA technology to mitigate SCC by lowering the ECP through the deposition of noble metal that catalytically recombines hydrogen with oxygen. The current NMCA process uses the Pt salt to form Pt nanoparticle and also introduces the unnecessary ionic species to the reactor water. Recently, new method was developed to eliminate ionic species by injecting noble metal nanoparticle and then disperse them in the water for deposition on the structural surfaces. Nanoparticle possesses a very high surface area and the ability to form colloidal suspensions. Once dispersed, nanoparticles can stay as a colloidal suspension due to Browian motion. Noble metal nanoparticle would also be injected while the reactor is in normal operation.This paper will present the feasibility of noble metal nanoparticles to enhance the surface catalytic property and thus reduce the ECP of structural materials under a simulated BWR condition.

10:50 AM

The Influence of Minor Additions of Platinum Group Metals on Stress Corrosion Cracking in Austenitic Stainless Steels: *Kuveshni Govender*[1]; Fabio Scenini[1]; Stuart Lyon[1]; Ouarda Necib[2]; Andrew Sherry[1]; [1]University of Manchester; [2]Andra

Stress corrosion cracking (SCC) of austenitic stainless steels represents a major challenge to the long-term reliability and integrity of nuclear power plants. This complex phenomenon is often associated with occluded regions which are not easily refreshed at the plant initiation stage. The deployment of SCC-resistant materials is highly desirable. However, whilst new alloys would have notable cost implications and require re-approval, minor changes would allow them to remain within specification and avoid such issues. In this study, the effect of minor additions of platinum group metals (PGMs) on the electrochemical behaviour and SCC resistance of 304 stainless steels was investigated under different environmental conditions relevant to light water reactors. A multi-scale study has been undertaken in order to develop a mechanistic understanding of the factors governing SCC. The results provide strong evidence that minor additions of ruthenium to 304SS are beneficial in terms of SCC resistance and general corrosion behaviour.

11:10 AM

The Effect of On-Line Noble Metal Addition on the Shut Down Dose Rates of Boiling Water Reactors: *Robert Cowan*[1]; Juan Varela[2]; Susan Garcia[3]; [1]EPRI Consultant; [2]GE-Hitachi Nuclear Energy; [3]EPRI

On-line noble metal addition (OLNC) is the third generation of hydrogen water chemistry developed to maintain the ECP of boiling water reactor structural materials in a range that mitigates intergranular stress corrosion cracking. The method utilizes the on-line injection of dilute $Na_2Pt(OH)_6$ into the feedwater over a period of approximately 10 days. The first application of OLNC occurred in July of 2005 and a total of 12 BWRs have applied the technology to date, with many more applications scheduled. It is expected that OLNC will become the de facto standard because it eliminates 60 hours of outage application time and it addresses the crack flanking concerns that can arise under certain conditions. Shut down dose rate data are now available for over 6 plants, several with multiple cycles. This paper will examine this behavior and it's implication for optimizing future OLNC applications and post application operations.

11:30 AM

Effects of H2 on Nickel Alloys in BWRs: *Peter L. Andresen*[1]; Peter Chou[2]; [1]GE Global Research; [2]EPRI

SCC of Alloy 600 and Alloy 182, 132 and 82 weld metals in high temperature water is important because they are structural materials in light water reactors. PWR primary water data show that there is a peak in growth rate vs. H_2 associated with the Ni/NiO phase boundary, and adjusting the H_2 represents a significant opportunity for SCC mitigation. Calculations and recent data show that H_2 play an important role in BWRs, because at lower temperatures the Ni/NiO phase boundary occurs at lower H_2. At the 274°C operating temperature of most structural materials in BWRs, the peak growth rate occurs at ~250 ppb H_2, close to that used by BWRs operating under medium H_2 water chemistry conditions. By contrast, the H_2 level required by electrocatalysis is about 8X lower and can result in about an order of magnitude lower growth rate for nickel alloy weld metals.

11:50 AM

Technical Basis for Water Chemistry Control of IGSCC in Boiling Water Reactors: *Barry Gordon*[1]; Susan Garcia[2]; [1]Structural Integrity Associates, Inc.; [2]Electric Power Research Institute

Boiling water reactors (BWRs) operate with very high purity water. However, even the utilization of near theoretical conductivity water cannot prevent intergranular stress corrosion cracking (IGSCC) of sensitized stainless steel, wrought nickel alloys and nickel weld metals under oxygenated conditions. IGSCC can be further accelerated by the presence of certain impurities dissolved in the coolant. The goal of this paper is to present the technical basis for controlling various impurities under both oxygenated, i.e., normal water chemistry (NWC) and deoxygenated, i.e., hydrogen water chemistry (HWC) conditions for mitigation of IGSCC. More specifically, the effects of typical BWR ionic impurities (e.g., sulfate, chloride, nitrate, phosphate, etc.) plus iron on IGSCC propensities in both NWC and HWC environments will be discussed. The technical basis for zinc addition to the BWR coolant will also provided.

PWR Field Experience II

Thursday PM
August 11, 2011

Room: Colorado I
Location: Cheyenne Mountain Resort

Session Chairs: Ulla Ehrnsten, VTT; In Hyoung Rhee

1:20 PM

Residual Stress Measurement and the Effect of Heat Treatment in Cladded Control Rod Drive Specimens: Ed Kingston[1]; *Ed Kingston*[1]; Jinya Katsuyama[2]; Makoto Udagawa[2]; Kunio Onizawa[2]; [1]VEQTER Ltd.; [2]JAEA

This paper presents results of residual stress measurements and modelling within the cladding and J-groove weld of Control Rod Drive (CRD) specimens in the as-welded and Post Weld Heat Treated (PWHT) states. Knowledge of the residual stresses present in CRD nozzles is critical when modelling the fracture mechanics of failures of nuclear power plant components to dictate inspections intervals and optimise plant downtime The specimens comprised of ferritic steel blocks with 309L stainless steel cladding and a single J-groove weld attaching the 304 stainless steel nozzles. Multiple measurements were made through the thickness of the specimens in order to give biaxial residual stress profiles through all the different fusion boundaries. The results show the effect of PWHT in reducing residual stresses both in the weld and ferritic material. The beneficial use of measurements is highlighted to provide confidence in the modelled results and prevent over conservatism in integrity calculations, costing unnecessary time and money.

1:40 PM

Detailed Root Cause Analysis of SG Tube ODSCC Indications within the Tube Sheets of NPP Biblis Unit A: *Renate Kilian*[1]; Jens Beck[1]; Hermann Lang[1]; Thomas Schönherr[2]; Martin Widera[3]; [1]AREVA NP GmbH; [2]RWE Power, Kraftwerk Biblis; [3]RWE POWER AG

During regular inspections using eddy current testing in 2005 and 2006, a few SG tube indications were detected within the tube sheet between upper and lower mechanical tube expansion. They were limited to the outer tube bundle periphery. Destructive examinations of two pulled tubes revealed axially oriented cracks starting from outer surface (ODSCC). Analysis of debris from the upper expansion area clearly indicated the presence of secondary side water in the volume between tube sheet and tube. This is only possible by a penetration path along the upper expansion.To focus future SG tube inspections, a comprehensive root cause analysis was initiated to clarify in which way such a penetration path can develop. The study includes an assessment of SG manufacturing and operation, an analysis of the deformation behavior in the upper expansion areas and consideration of the secondary side water chemistry data. This paper summarizes the main results.

2:00 PM

Laboratory Investigation of a Leaking Type 316 Socket Weld in a Boron Injection Tank Sampling Line: *Hongqing Xu*[1]; Steve Fyfitch[1]; Ryan Hosler[1]; James Hyres[2]; [1]AREVA NP, Inc.; [2]Babcock & Wilcox

A leak was discovered in a Type 316 stainless steel socket weld joining a boron injection tank sampling line and sampling valve. A section of the pipeline containing the leaking weld was removed for laboratory investigation that included visual and stereovisual inspections, liquid penetrant (PT) testing, metallography, scanning electron microscopy (SEM), energy dispersive spectroscopy (EDS), and ferrite content determinations. The leak path was a through-wall transgranular crack in the socket weld. Cracking initiated along the weld-metal-to-base-metal interface at the tip of the crevice between the socket and pipe. The crevice was exposed to borated water. Shallow intergranular attack (IGA) was found in the exposed base metal inside the crevice. Based on the investigation results, it was concluded that transgranular stress corrosion cracking (TGSCC) is the likely cracking mechanism, even though IGA may have played a role during crack initiation.

2:20 PM

Pressure Tests on SG Pulled Tubes at TSP Level: *Marc Boccanfuso*[1]; Cédric Mathon[1]; [1]EDF

In 2009, a circumferential crack was detected for the first time at the tube support plate level of a Steam Generator tube at one of France's oldest nuclear power plants. In order to identify the involved mechanisms and to characterise the defects, metallurgical examinations were carried out on 12 pulled tubes. In addition, an objective was to evaluate the structural integrity of the tubes through the measurement of the pressure that the tubes could withstand. Leakage/burst pressure tests were analysed and an attempt was made to correlate these results to the shape of the defect. Results suggest that the leakage pressure was mainly influenced by one parameter - the initial depth of the damage.This was carried out firstly using results from previous investigations into pulled tubes, of which several came from a decommissioned Steam Generator and included a part of their tube support plate and, secondly, using analytical modelling.

2:40 PM

Implications of Steam Generator Fouling on the Degradation of Thermal and Material Performance: *Carl Turner*[1]; [1]Atomic Energy of Canada Limited

Fouling of steam generators has a significant negative impact on the thermal and material performance of steam generators in pressurized water reactors. Corrosion products that originate from various components in the steam cycle of a nuclear power plant get pumped forward with the feed water where they deposit on the tube bundle, tube support structure and the tube sheet. Heavy accumulation of deposit within the steam generator has led to some serious operational problems, including loss of thermal performance, under deposit corrosion, steam generator level oscillations, flow accelerated corrosion of carbon steel tube support plates and the failure of steam generator tubes due to high cycle fatigue. This paper will review the mechanisms of steam generator fouling, examine the relationship between fouling and degradation of the thermal and material performance of steam generators and investigate the effectiveness of remedial measures to mitigate fouling.

3:00 PM

Key Issues Related to Corrosion Protection of Brackish Water and Sea Water Bearing Components in Cooling Water Systems: *Erika Nowak*[1]; Bengt Bengtson[2]; Björn Forssgren[3]; Björn Hall[4]; Elisabeth Johansson[5]; [1]E.ON Kernkraft GmbH; [2]OKG Aktiebolag; [3]Ringhals AB; [4]Forsmark Kraftgrupp AB; [5]Outokumpu Stainless AB

Stainless steel components in cooling water systems, using brackish- or seawater as cooling medium, are potentially vulnerable to corrosion. In many cases such corrosion issues are solely ascribed to Microbially Influenced Corrosion (MIC). Within the scope of a study, field tests were performed in different environments (Swedish east coast, Swedish west coast, mouth of the river Elbe in Germany), under conditions favourable to MIC. In some cases even such materials, that are assumed to be resistant, have been affected by corrosion. However, based on the outcomes of the field tests, it is demonstrated in this report, that all results, where corrosion occurred, are best explainable by chloride induced corrosion. Crevice conditions produced from consolidated biofilms on component surfaces or sludge deposits, most probable support the initiation and the progress of this kind of corrosion. Based on the test results, practical recommendations are derived, how to prevent corrosion in brackish- or seawater environments.

BWR Water Chemistry and Mitigation II

Thursday PM — Room: Colorado II
August 11, 2011 — Location: Cheyenne Mountain Resort

Session Chairs: Tiangan (TG) Lian, EPRI; Tsung-Kuang Yeh, National Tsing Hua University

1:20 PM

Water Chemistry in the Primary Coolant Circuit of a Boiling Water Reactor during Startup Operations: *Tsung-Kuang Yeh*[1]; Mei-Ya Wang[1]; [1]National Tsing Hua University

The coolant in a boiling water reactor (BWR) usually contains high levels of dissolved oxygen from exposure to atmospheric air during a cold shutdown, and accordingly the structural materials in the primary coolant circuit (PCC) of a BWR could temporarily experience a strongly oxidizing condition during a subsequent startup operation. In this study, the well developed computer code DEMACE was modified and used to investigate the influence of a startup operation on the water chemistry and the electrochemical corrosion potential (ECP) in the PCC of a local BWR. The simulations were carried out for power levels ranging from 2.5% to 12% of the rated power under normal water chemistry. In summary, a startup operation could induce a strongly oxidizing coolant environment in a BWR and accordingly lead to the initiation of stress corrosion cracking in structural materials.

1:40 PM

Simulation of Water Radiolysis by Sonochemistry: Effects on the Electrochemical Behavior of a Stainless Steel: *Olivier Lavigne*[1]; Yoichi Takeda[1]; Tetsuo Shoji[1]; [1]Tohoku University

Water radiolysis produces both highly oxidizing and reducing species such as: OH•, e-aq, H•, HO2•, H2, H2O2, H+. These one can strongly affect the corrosion kinetics of materials in contact with water. The aim of this work is to develop an experimental set up in order to simulate water radiolysis in laboratory. Here, OH• radical production is thus achieved by water sonolysis. Electrochemical measurements conducted at room temperature (potentiodynamic and potentiostatic curves, EIS) and XPS show a decrease of the corrosion protection properties of the passive film due to the ultrasonic irradiation. The solution conductivity is also an important parameter on the evolution of open circuit potential due to the thickness variation of the diffusion layer which may contain sonolysed species (OH•, H2, H2O2) in different concentrations. The efficiency of the radical production is also studied by means of varying experimental parameters such as dissolved gases species or solution type.

2:00 PM

Protective Insulated Coating for SCC Mitigation in BWRs: *Young-Jin Kim*[1]; Peter Andresen[1]; [1]GE Global Research Center

NMCA has been successfully applied to commercial BWRs worldwide and is a proven technology for mitigating SCC by catalyzing the oxygen and hydrogen recombination and eventually lowering the ECP below the IGSCC protection potential. In order to achieve the catalytic benefit of noble metal, excess hydrogen is essential. However, there may be certain locations where neither noble metal nor hydrogen is present. A novel method for protecting certain areas that are not mitigated by NobleChem has been developed by a protective insulated coating (PIC). PIC was produced by thermal spray methods with coating of Zr or Zr alloys and investigated in high temperature water under various water chemistry conditions. This paper will present recent development of PIC method for achieving the IGSCC mitigation potential, by examining electrochemical kinetics, ECP behavior, microstructure, and durability of a PIC layer in high temperature water.

2:20 PM
Influence of Treating Temperature on the Deposition of TiO_2 on Type 304 Stainless Steels for Corrosion Mitigation in High Temperature Pure Water: *Tsung-Kuang Yeh*[1]; Yu-Jen Huang[1]; Chuen-Horng Tsai[1]; [1]National Tsing Hua University

The effectiveness of titanium dioxide treatment in corrosion mitigation for austenitic stainless steels was investigated in this study. Electrochemical polarization analyses and electrochemical corrosion potential measurements were conducted to investigate the impact of ultraviolet radiation on the electrochemical behavior of oxygen on TiO_2 treated specimens in 288°C pure water. Prior to electrochemical tests, specimens were thermally sensitized and pre-oxidized in high temperature pure water containing 300 ppb dissolved O_2. TiO_2 nanoparticles were then deposited on the specimens by hydrothermal deposition at various temperatures of 90, 150, and 280°C for 96 hrs. Results shown the treating temperature of 90°C led to a poor TiO_2 coating in terms of corrosion reduction. On the other hand, treating temperatures of 150°C and 280°C resulted in relatively better coatings, and no significant differences in ECP reduction and anodic current density were observed between the samples treated at these temperatures.

NOTES

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15th International Conference on
ENVIRONMENTAL DEGRADATION
of Materials in Nuclear Power Systems-Water Reactors

SCHEDULE OF EVENTS

	Colorado Ballroom	Colorado I	Colorado II	Colorado III
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Mon PM		**Alloy 690 and Its Weld Metals II** 1:05-2:45 PM **Alloy 690 and Its Weld Metals III** 3:00-4:40 PM **Alloy 690 and Its Weld Metals IV** 4:55-5:55 PM	**BWR Initiation and Oxide Film Characterization II** 1:05-2:05 PM **BWR Stainless Steels CGR I** 3:00-3:40 PM **BWR Stainless Steels CGR II** 3:40-4:40 PM	**Fuel and Fuel Related Materials II** 1:05-2:05 PM **Fuel and Fuel Related Materials** III PWR-BWR 2:05-2:45 PM **Corrosion Fatigue - BWR, PWR** 3:00-4:20 PM
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Tues PM		**PWR Oxide Films and Characterization** 7:30-9:10 PM	**Alloy 718 and X-750** 7:30-8:50 PM	**Flow Assisted Corrosion** 7:30-8:50 PM
Wed AM		**PWR Alloy 600 Oxidation and Mechanisms I** 8:15-9:35 AM **PWR Alloy 600 Oxidation and Mechanisms II** 10:10-11:10 AM	**IASCC Stainless Steels CGR I** 8:15-9:55 AM **IASCC Stainless Steels CGR II** 10:10-11:50 AM	**PWR Water Chemistry and Mitigation** 8:15-9:35 AM **Super Critical Water** 10:10-11:30 AM
Wed PM		**PWR Alloy 600 SCC I** 1:05-2:45 PM **PWR Alloy 600 SCC II** 3:05-4:45 PM	**Irradiation Effects on Deformation** 1:05-2:25 PM **Irradiation Effects–General I** 3:05-4:25 PM	**PWR Degradation Management** 1:05-2:45 PM
Thurs AM		**PWR Initiation and CGR Stainless Steels I** 8:15-9:55 AM **PWR Initiation and CGR Stainless Steels II** 10:10-10:30 AM **PWR Field Experience I** 10:30 AM-12:10 PM	**Irradiation Effects–General II** 8:15-9:35 AM **BWR Water Chemistry and Mitigation I** 10:10 AM-12:10 AM	
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