Ninth International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors

Newport Beach Marriott Hotel	Monday AM	Monday PM	Tuesday AM	Tuesday PM	Wednesday AM	Wednesday PM	Thursday AM	Thursday PM
Pacific Ballroom C	PWR Primary Session I - Mechanisms (8:00AM)	PWR Primary Session II Chemistry and Failure Analysis (1:30PM)	PWR Primary Session III Hydrogen Effects & Microstucture (8:00AM)	Regulation Aspects (6:00PM)	PWR Primary Session IV Crack Growth & Creep (8:00AM)	PWR Secondary Session I - System Definition (1:30PM)	PWR Seconday Session II - Cracking Response (8:00AM)	PWR Seconday Session III - Mechanisms (1:30PM)
Pacific Ballroom D	BWR Session I - Cracking Response (8:00AM)	BWR Session II - Mechanism/ Life Extension (1:30PM)	BWR Session III - SCC Data Quality & System Definition (8:00AM)	Welding/ Processing (6:00PM)	BWR Session IV - Mitigation/ Life Extension (8:00AM)	Radiation Effects Session I - Radiation Effects on Stress Corrosion Cracking (1:30PM)	Radiation Effects Session II - Radiation Effects on Deformation and Swelling (8:00AM)	Radiation Effects Session III - Radiation Effects on Microstructure and Microchemistry (1:30PM)
Pacific Ballroom E&F					Low Alloy Steel Session I - Embrittle-ment (8:00AM)	Low Alloy Steel Session II - EAC & Deformation (1:30PM)		Zircaloy (1:30PM)

BWR: Session I - Cracking Response

Monday AM Room: Pacific Ballroom D
August 2, 1999 Location: Marriott Hotel

Session Chairs: P. L. Anderson, GE CRD USA; C. Janssen, Vittenfall, Sweden

8:00 AM
IGSCC in Stabilized Stainless Steels under BWR Conditions, Lab and Field Experience: *J. Hickling*[1]; P. L. Andresen[2]; H. Hoffman[3]; U. Ilg[4]; V. Maier[5]; O. Wachter[6]; M. Widera[7]; [1]CMC Corrosion & Materials Consultancy, Hirtenweg 16, Taufkindnen D-82024 Germany; [2]General Electric Research & Development, One Research Circle, Bldg. K-1 3A39, Schenectady, NY 12309 USA; [3]VBG; [4]EnBW; [5]BAG; [6]PE; [7]RWE
The research described addresses the SCC response of Ti and Nb stabilized stainless steels, (German designations 1.4541 and 1.4550, respectively) under water chemistry conditions that range from a worst-case scenario (e.g., high 02 and high SO_4) to one that closely simulates the conditions in operating BWRs. It involved crack growth measurements on fracture mechanics specimens of non-sensitized base material and was performed with very careful control and monitoring of water chemistry, corrosion potential, stress intensity, etc., using state-of-the-art experimental techniques. Different pre-cracking procedures were used and these may have affected the ease of SCC initiation, but they produced relatively small differences in subsequent SCC response. Crack growth through SCC began almost immediately on loading to stress intensities of 20 or 35 MpaÖm, although in none of the specimens tested did intergranular SCC initiate along the entire transgranular fatigue pre-crack front. IGSCC occurred in all specimens, whether at high (0.5mS/cm, K_2SO_4) or low (very high purity water) conductivity; high (35 MpaÖm) or lower (20 MpaÖm) stress intensity; high (8000 ppb) or lower (400 ppb) dissolved O_2; titanium (1.4541) or niobium (1.4550) stabilized stainless steel, etc. The crack growth rates measured in various O_2 containing solutions ranged from 0.2 x 10^8 mm/s to 5.9 x $l0^{-8}$ mm/s. On changing to simulated hydrogen water chemistry, which produced a very low potential (~500 mÖ$_{she}$), cracking arrested immediately and no further growth was observed. In O_2-containing solutions, after exposure to high sulfate concentrations (139 ppb SQ with K_2SO_4), it appears that higher SCC rates persisted for a long period of time after the sulfate level was reduced. Under otherwise comparable conditions of oxygen and sulfate content, crack growth rates were higher by a factor of 1.4 to 3.2 at a stress intensity of 35 MPaÖm than at 20 MpaÖm. In general, it appears that crack propagation behavior is fairly similar for both base materials. Because of the small, overall increments in intergranular crack growth observed (average values of only 80 to 200 mm, despite extended testing times), some residual uncertainty remains as to the applicability of the measured crack growth rates to BWR plants. The above observations are discussed in the broader context of both laboratory and field data on intergranular SCC of sensitized and non-sensitized stainless steels.

8:30 AM
Intergranular Stress Corrosion Cracking Unsensitized Stainless Steels in BWR Environments: *T. M. Angeliu*[1]; P. L. Andresen[1]; J. A. Sutliff[1]; R. M. Horn[1]; [1]General Electric, CR&D, Bldg. K-1, Rm. 3A-51, 1 River Rd., Schenectady, NY 12301 USA
lntergranular stress corrosion cracking (IGSCC) of austenitic iron and nickel base alloys continues to affect the availability of boiling water reactors (BWR) and represents an important consideration when selecting materials for advanced BWRs. The objectives of this program are to better understand why L-grade stainless steels are apparently more susceptible to IGSCC in core internal applications than expected and to guide recommendations for remedial measures if needed. Microstructural characterization indicates that various L-grade materials are not thermally sensitized, but slight Cr depletion due to thermal nonequilibrium segregation may be present. With the possibility that sensitization is not dominant, other fundamental aspects controlling the cracking behavior, such as deformation and the crack tip strain rate, are being studied. Laboratory experiments on deformed 304L SS specimens have produced intergranular cracking behavior equivalent to that associated with thermally sensitized 304 SS. Electron back scattered pattern techniques have characterized the level of strain in a weld HAZ which has been correlated with the cracking behavior of bulk worked specimens. Experiments and modeling focused on the effects of alloy composition and deformation on the crack tip strain rate are in progress.

9:00 AM
Failed Components from the Ringhals 1 (BWR Steam Dryer): *Kjell Norring*[1]; Jan Lagerstrom[2]; Lars Storm[3]; Kurt Norrgard[1]; Goran Embring[2]; Mats Olmeby[2]; [1]Studsvik Material AB, Nykoping SE-61182 Sweden; [2]Vattenfall AB Ringhals,Varobacka SE-43022 Sweden; [3]ABB Atom AB, Vasteras SE-72163 Sweden
During the refueling outage in 1998 severely damaged components were found in the reactor vessel moisture separators at Ringhals 1, which is a BWR. The components that had failed were moisture separator tighteners and condensate collector including their nuts and casings. The condensate collector showed complex failures. Erosion and wear (metal to metal) had been acting together. Most probably the course of events started by wear. Vibrations may then have forced the water in small crevices caused by the wear to move with a sufficiently high speed to cause erosion. In a few cases large defects occurred when the entire condensate collector came loose. The materials of the tighteners are Inconel X-750 and Inconel 600. The cracks had initiated from crevices in the Inconel X-750 part. The crevices were closed by welding and should thus have been dry. However, cracks and/or pores in the welds made it possible for moisture to penetrate into the crevices and thereby initiate IGSCC. A repair procedure was established. The failed tighteners were cut, electrolytically decontaminated and rewelded with a modified end nut of stainless steel, instead of Inconel X-750/600. The whole repair procedure was performed as an integrated project.

9:30 AM
Stress Corrosion Cracking of Stabilized Austenitic Stainless Steels in Various Types of Nuclear Power Systems: *Markus O. Speidel*[1]; Ruth Magdowski[1]; [1]Swiss Federal Institute of Technology ETH, Institute of Metall., Zurich CH-8092 Switzerland
Intergranular stress corrosion cracking of stabilized austenitic stainless steels (types 321 and 347) have been observed in various types of light water nuclear reactors: BWR, RMBK, and WWER. The present paper describes the occurrence of such stress corrosion cracking and relates it to the known quantitative information on stress corrosion crack growth rate data from laboratory experiments. It is shown that electrochemical potential, conductivity, cold work, and stress intensity are major influential variables, as are microstructure and heat affected zones near the welds. An upper and a lower limit of crack growth rates and therefore cracking component life times likely to be encountered in component service can be estimated.

10:00 AM Break

10:30 AM
Initiation of Stress Corrosion Cracking in Alloys 600 and 182: *Anders Jenssen*[1]; Margareta Stigenberg[1]; Lars G. Ljungberg[1]; [1]ABB Atom, Nuclear Sys. Div., Västerås SE-721 63 Sweden
Wrought alloy 600 and weldments of alloy 182 were tested for initiation of stress corrosion cracking, intergranular (IGSCC) in the wrought material, and interdendritic (IDSCC) in the weld alloy. For the tests a modified bolt loaded compact tension (CT) specimen was developed, in which a U-notch replaced the conventional crack notch. Various stress states close to the notches were achieved by providing the specimens with different U-notch radii. All specimens were loaded to the same

displacement at the bolt. A few control specimens with conventional crack notches were included. Testing in most cases was performed under enhanced crevice conditions. The specimens were exposed to BWR normal water chemistry (NWC), or hydrogen water chemistry (HWC), in two Swedish BWRs. IDSCC initiated in many of the alloy 182 specimens tested in NWC, while the frequency of crack initiation was much less in HWC. As for alloy 600, very little crack initiation occurred in NWC, and none was detected in HWC. Two types of attack were found, thumbnail shaped cracks located perpendicular to the principal stress, and intergranular or interdendritic penetrations with no preferred orientation. The latter indicates that local chemical attack was the major factor causing initiation of cracks in these materials.

11:00 AM

The Electrochemical Corrosion Potential and Stress Corrosion Cracking of 304 Stainless Steel under Low Hydrogen Peroxide Concentrations: *Yoichi Wada*[1]; Masahiko Tachibana[1]; Atsushi Watanabe[1]; Naohito Uetake[1]; Shunsuke Uchida[1]; [1]Hitachi, Ltd., Power & Indust. Sys. R&D Div., 7-2-1 Omikia, Hitachi, Ibaraki 319-1221 Japan

The stress corrosion cracking (SCC) propagation of structural materials used in the boiling water reactors has been studied as functions of hydrogen peroxide concentration and the electrochemical corrosion potential (ECP). Crack growth rate (CGR) at low hydrogen peroxide concentrations around 10ppb, are especially concerned, which is taken as corrosion environment formed in hydrogen water chemistry. The 1/4 inch compact tension specimen with employing potential drop method is used for measurement of the CGR of sensitized 304 stainless steel under high temperature and high purity water. Since hydrogen peroxide is easy to decompose thermally, a PTFE lined autoclave was used for minimizing its decomposition on the surface of the autoclave. The CGR in hydrogen peroxide condition was different from that in oxygen condition even though the ECP of the both conditions were the same. The data implies that the ECP would not be only an environmental deterministic parameter for SCC behavior.

11:30 AM

Crack Growth of Stabilized Steel in O_2-containing High Temperature Water; Influence of Environment and Materials Conditions: *R. Kilian*[1]; U. Eberie[2]; G. Bruemmer[3]; H. Hoffman[4]; U. Ilg[5]; V. Maier[6]; M. Wachter[7]; [1]Siemens AG. KWU. NBTW, Kembrennstoff-Kreislauf, Hammerbacherstr 12&14, Postfach 3220, Erlangen D-91058 Germany; [2]Siemens AG; [3]HeW; [4]VGB; [5]ENBW; [6]BAG; [7]RWE

During plant shutdown of several German BWRs non-destructive testing has been carried out on the stainless steels (SS) piping systems. Since 1992 in some cases intergranular cracks were found in the heat affected zone (HAZ) of the weld joints. Based on numerous detailed investigations it can be stated that the mechanism for crack initiation and crack growth was intergranular stress corrosion cracking (IGSCC) as a result of sensitization during welding. The database of the IGSCC behavior of stabilized SS is limited compared to those of unstabilized 55 like AISI type 304 or 316. In Germany extensive research programs therefore were started with the purpose of a systematic characterization of the correlation between material condition of stabilized SS and the IGSCC behavior in 0_2-containing high temperature water. The first programs were focused on sensitization behavior and the general susceptibility of stabilized 55 to IGSCC depending on degree of sensitization which was demonstrated by slow strain rate tensile (SSRT) tests. The main emphasis of the following research programs was respectively concentrated to crack growth rate measurements to show the influence of material and environmental conditions. Therefore 1T CT fracture mechanics specimens of different heats of stabilized SS with various material conditions were tested in a refreshing autoclave observing the test characteristics which are in discussion worldwide. The results of the crack growth rate measurements will be shown and discussed.

PWR Primary: Session I - Mechanisms

Monday AM — Room: Pacific Ballroom C
August 2, 1999 — Location: Marriott Hotel

Session Chairs: G. Airey, British Energy UK; C. Thompson, KAPL, USA

8:00 AM

An Overview of Internal Oxidation as a Possible Explanation of Intergranular Stress Corrosion Cracking of Alloy 600 in PWRs: *P. M. Scott*[1]; [1]Framatome, Tour Fiat, 1 Place de la Coupole, Paris - La Defense, Cedex 16 92804 France

Internal oxidation was first proposed as a plausible mechanism of intergranular stress corrosion cracking (IGSCC) in hydrogenated PWR primary water by Scott and Le Calvar in 1993(1). Since then several experimental studies have been undertaken to test the hypothesis. These were last reviewed by the present author in 1996(2). More recently, some detailed microscopical examinations of both primary and secondary cracks in Alloy 600 using Secondary Ion Mass Spectrometry (SIMS) and Analytical Transmission Electron Microscopy (ATEM) have been carried out and preliminary results have already been published(3). These investigations will be described in detail in other papers in this conference (4,5). The first purpose of the present paper is to address some of the points of criticism which have arisen as to the applicability of the internal oxidation mechanism at typical PWR operating temperatures and corrosion potentials. This is the specific problem of reconciling the apparent rate of intergranular diffusion of oxygen in nickel base alloys with the observed rates of cracking and the thermodynamic requirement for internal oxidation that the corrosion potential be at or below the Ni/NiO redox potential. The latter point is of particular concern if this mechanism is invoked to explain the secondary side steam generator tube IGA/IGSCC. The second objective is to describe the morphology of known cases of intergranular internal oxidation cracking observed at higher temperatures in order to provide a point of reference for other papers in this conference concerned with detailed examinations of cracks in alloy 600 produced under actual or prototypical PWR conditions.

8:30 AM

Methodology to Understand the Mechanisms of PWSCC: *T. Yonezawa*[1]; [1]Mitsubishi Heavy Industries, Ltd., Takasago R&D Center, 2-1-1, Shinhama, Arai, Takasago, Hyogo Pref. 676 Japan

Many papers were published about the mechanisms on the PWSCC. Most papers persisted the effect of hydrogen on the cracking as PWSCC, but recently, internal oxidation was proposed as the mechanisms of PWSCC. However, evidence will be needed to clarify the mechanisms of PWSCC. This paper describes the methodological approach to understand the mechanisms of PWSCC, based upon the previous studies of authors. The following methodological approach was recommended: (1) Study on the effect of nickel, chromium and silicon content on oxidation at the grain boundary and hydrogen trapping by SIMS analysis after exposure in simulated PWR primary water included O^{18} and deuterium, and micro autoradiography after exposure in simulated PWR primary water included tritium water, respectively; (2) Study on the effect of stress on the oxidation at the grain boundary and the hydrogen trapping by the same technique as (1); (3) Study on the effect of dissolved hydrogen and oxygen on the oxidation at the grain boundary and the hydrogen trapping by the same technique as (1). From the analysis of the above results and the comparison with the present literature on PWS CC, the true mechanisms of PWSCC shall be understood clearly.

9:00 AM

Hydrogen Effects on PW SCC Mechanisms in Monocrystalline and Polycrystalline Alloy 600: *Thierry Magnin*[1]; [1]Ecole des Mines de St. Etienne, Centre SMS URA CNRS 1884, 158 Cours Fauriel, 42023 St. Etienne, Cedex 02 France

Slow strain rate tests on alloy 600 in PWR environment shows two main results: 1) Cathodic polarization (from -3OmV to - 320 mV) on polycrystals clearly shows an acceleration by a factor 5 of the crack velocity in comparison to free potential; 2) single crystals (axis <100> parallel to the tensile axis) do not exhibit any crack initiation on smooth specimen but are sensitive to SCC propagation on pre-cracked specimens. Differences between crack initiation and crack propagation mechanisms are emphasized. The role of grain boundary chemistry is underlined. The two main results reported above are then discussed in terms of SCC mechanisms through hydrogen effects. New modelings of hydrogen-plasticity interactions are presented and then lead to proposed macroscopic tests on single crystals which experimentally show a hydrogen induced plasticity. Consequences on the cracking processes are discussed, mainly for crack propagation, through the corrosion enhanced plasticity model.

9:30 AM

Insights into Crack Growth Mechanisms from Analytical Transmission Electron Microscopy of SCC Crack Tips: *L. E. Thomas*[1]; S. M. Bruemmer[2]; [1]Washington State University, School of Mech. and Mats. Eng., Pullman, WA 99164-2920 USA; [2]Battelle Pacific Northwest National Laboratory, P.O. Box 158, Mail Bin 26, Richland, WA 99352 USA

The application of analytical transmission electron microscopy (ATEM) to characterize microstructures and microchemistries at resolutions approaching the atomic levels offers great promise for improved mechanistic understanding of stress corrosion cracking (SCC). The ability to perform this work arises from two developments: 1. Methods for preparing cross-sectional samples with electron-transparent areas at crack tips or other specific sample locations; 2. Gun TEMs with capabilities for in microstructural, crystallographic, and compositional analysis at sub-nm spatial resolutions. Recent examinations of stress-corrosion cracked, alloy 600 steam generator tubing will be used to illustrate the strengths and complications of this characterization approach. Unique insights into the crack and crack-tip electrochemistry and into possible solid-state reactions occurring at grain boundaries ahead of open cracks will be demonstrated. ATEM measurements have revealed fine details of primary- and secondary-side environmental degradation by the characterization of crack walls corrosion films, crack tips and attacked grain boundaries off the open cracks. Significant differences are observed among the various primary- and secondary-side cracks that have been examined, but many exhibit narrow (10-30 nm wide), deeply attacked regions along former high-energy grain boundaries in the wake of the primary SCC crack. Given their small dimensions, these structures have not been identified in studies by most other analytical methods. The extent of this fine-scale attack at most grain boundaries suggests a strong active-path process driving intergranular degradation and that internal oxidation may play a role during the intergranular attack and SCC of steam generator tubing.

10:00 AM Break

10:30 AM

Measurement of the Fundamental Parameters for the Film-Rupture/Oxidation Mechanism: *J. S. Fish*[1]; S. A. Attanasio[1]; D. S. Morton[1]; P. M. Rosecrans[1]; W. W. Wilkening[1]; T. M. Angeliu[1]; [1]Lockheed Martin Corporation, P.O. Box 1072, Schenectady, NY 12301 USA

The identification of the mechanism(s) of primary water stress corrosion cracking (PWSCC) in nickel-base alloys is a highly controversial topic. Numerous mechanisms, including the film-rupture/oxidation (i.e., slip-oxidation) mechanism and various hydrogen-related mechanisms, have been proposed to explain PWSCC. According to the film-rupture/oxidation mechanism, the observed sensitivity of PWSCC to material and environmental factors may be explained by the combined effects of oxide fracture strain, repassivation kinetics, and crack tip strain rate. Previous research has shown that the chromium content of Ni-9%Fe-Cr and the coolant hydrogen concentration strongly influence the PWSCC susceptibility of nickel-base alloys. Consequently, measurements of these fundamental parameters were performed in high temperature water as a function of chromium content and coolant hydrogen concentration. This paper illustrates how these three parameters must be considered concurrently to explain the observed sensitivities of PWSCC in nickel-base alloys.

11:00 AM

Comparison of Hydrogen Effects on Alloy 600 and 690: *H. Hanninen*[1]; [1]Helsinki University of Technology, Lab. of Eng. Mats., P.O. Box 4200, TKK FIN-02015 Finland

A comparative study of hydrogen interaction with lnconel-600 and lnconel-690 alloys was performed. Hydrogen distribution over interstitial positions and diffusivity in these alloys was studied by internal friction and electrochemical measurements of the chemical potential of hydrogen. Snoek-type internal friction peak caused by hydrogen in lnconel-690 was observed at 220 K, which is 30 K higher than the temperature of the corresponding peak in lnconel-600. This indicates lower diffusivity of hydrogen in Alloy 690 as compared to Alloy 600. A study of the hydride formation and accompanying internal stresses was performed for both alloys by X-ray diffraction and the method based on the bending of a thin strip sample being hydrogen charged from one side only. The obtained results are discussed in terms of IGSCC in nickel-based alloys used in the nuclear power industry.

11:30 AM

Comments on a Proposed Mechanism of Internal Oxidation for Alloy 600 as Applied to Low Potential SCC: *R. W. Staehle*[1]; Z. Fang[1]; [1]University of Minnesota, Dept. of Chem. Eng. and Mats. Sci., Corrosion Center

Abstract Text Not Available

12:00 PM

SIMS Evaluation of Steam Generator Tubing for Evidence of Internal Oxidation: *T. S. Gendron*[1]; S. J. Bushby[1]; I. J. Muir[1]; [1]Atomic Energy of Canada, Ltd., Station 61, Chalk River Laboratories, Chalk River, Ontario K0J1J0 Canada

Internal grain boundary oxidation has been proposed to play a role in both primary and secondary side intergranular stress-corrosion cracking of alloy 600 steam generator tubing[1]. To evaluate the validity of this proposal, Secondary Ion Mass Spectrometry (SIMS) has been used to examine grain boundaries of steam generator tubing removed from service. Sections of tubing containing stress corrosion cracks that were carefully removed from Ringhals-2, Oconee-3, Tihange-1, and Doel-4 have been examined by SIMS to determine whether internal oxidation was responsible for SCC. Our primary interest was whether grain boundaries were oxidized without penetration by primary or secondary environments. A Cameca IMS 6F SIMS instrument, capable of surveying relatively large areas (£250 mm) for oxidized grain boundaries and trace amounts of coolant constituents (e.g. Na), was used in this study. Others [2] have examined the structure and chemistry of similar cracks using high resolution transmission electron microscopy. It is debatable whether electrochemical conditions for internal oxidation are met on the secondary side, and therefore other species may assist through degradation of the protective external oxide or by crack initiation. Species of particular interest were sodium, sulfur, and lead in tubing from Oconee, Tihange, and Doel, respectively. Where relevant, results have been compared with previous laboratory studies of primary water cracking which have revealed evidence of internal oxidation.

BWR: Session II - Mechanism/Life Extension

Monday PM Room: Pacific Ballroom D
August 2, 1999 Location: Marriott Hotel

Session Chairs: R. N. Horn, GE Nuclear, USA; G. Bruemmer, HeW, Germany

1:30 PM
Effect of Stress on SCC Growth Rate in Oxygenated High Temperature Water: *Shuichi Suzuki*[1]; [1]Tokyo Electric Power Company, Mats. Eng. Group
It is well known that SCC growth rates taken at different laboratories in a wide range, which have been mainly explained by the different crack tip water chemistry. However, they have used the different sizes of specimens and there is a possibility of different crack tip strain rates among them. In this paper, to verify the effect of stress initially on SCC growth rate, SCC tests using CT specimens made of sensitized SUS304 with a variety of thicknesses were conducted in oxygenated 288 pure water. As a result, above 100mVSHH, SCC growth rate doesn't depend on the size of CT specimens below 1" thickness regardless of their crack shapes, wide below 50 mVSHE, it depends on the size of specimens below 2" thickness, which suggest the importance of crack tip under different crack tip water chemistry.

2:00 PM
Prediction of Environmentally Assisted Cracking and its Relevance to Life Management/Relicensing of BWRs: *F. P. Ford*[1]; P. L. Andresen[1]; T. M. Angeliu[1]; H. D. Solomon[1]; R. N. Horn[2]; R. Cowan[2]; [1]General Electric Research Development Center, P.O. Box 8, Schenectady 12301 NY; [2]General Electric Nuclear Energy
The prediction and mitigation of environmentally assisted cracking of structural materials in BWRs is analyzed on two time scales. The first time scale is that of the fuel cycle where the main concern is that once a crack is discovered, how far will it propagate during the next fuel cycle under the existing, or new operating conditions. The resolution of this concern has been the motivation for numerous data collection and mechanistic studies and the qualification of mitigation actions based on controlling relevant material, environment and stress parameters. The second, and increasingly important time scale is that of relicensing where no cracks have been observed, or are expected with the current "best practice" operating conditions in 40 years. The concern is how to predict the likelihood that cracks might be observed say 60 years, and if so, how will they be managed. This paper reviews the first time scale of the fuel cycle in terms of the growing quantitative prediction capability for crack propagation in increasingly complex material environmental BWR systems, associated with irradiation, subtle changes in grain boundary chemistry and variable loading conditions. These predictive capabilities guide the formulation of disposition relationships for life management purposes. The second time scale of relicensing is addressed by extending the deterministic prediction capability for crack propagation to the probabilistic aspects of crack initiation for predicting and extending the life of difficult to replace, critical components.

2:30 PM
Prediction and Mitigation of Cracking in BWR Core Components: *Peter L. Andresen*[1]; F. Peter Ford[1]; R. N. Horn[1]; T. M. Angeliu[1]; [1]GE Corporation, Research & Dev., One Research Circle, Rm. 3A39, Bldg. K-1 3A39, Schenectady, NY 12309 USA
By replacing external piping, I reduced the incidence and impact of stress corrosion cracking in the BWR component, the greatest historical concern. However, in the last decade the onset of stress corrosion cracking in core internals has created another, more complex challenge in which the effects of radiation damage and more aggressive, radiolytic water chemistry must be understood and mitigated. This paper discusses the incidence of stress corrosion cracking in internals, summarizes the complex interdependencies that control its kinetics, computes predictions for laboratory and field observations in irradiated materials, and describes techniques and benefits of various mitigation approaches.

3:00 PM Break

3:30 PM
Stress Corrosion Cracking Initiation in Austenitic Stainless Steel in High Temperature Water: *Karel Matocha*[1]; J. Wozniak[1]; [1]Vitkovice, J.S.C., Res. and Dev. Div., Pohranicni 31, Ostrava CR
The effect of ionic impurities (Cl, Na+), dissolved oxygen content, pH and applied tensile stress on nucleation of pits and microcracks on a smooth surface in water environment at 275°C was studied. Axially loaded, quantitatively stressed standard tension test specimens, 3mm in diameter were used for investigating the resistance of austenitic stainless steel of type AISI 321 to stress corrosion cracking. On the basis of experimental results obtained, a mathematical model of stress corrosion cracking initiation was proposed expressing the time to initiation as a function of applied tensile stress, concentration of chloride ions, corrosion potential and critical potential for microcrack nucleation.

4:00 PM
Effects of Acceleration Factors on the Probability Distribution of Stress-Corrosion Crack Initiation Life for Alloys 600, 182, and 82 in High-Temperature, Water Environment: *Masatsune Akashi*[1]; Guen Nakayama[1]; [1]Ishikawajima-harima Heavy Industries Co. Ltd., IHI/Research Institute, 3-1-15 Toyosu, Koloku, Tokyo 135-8732 Japan
A series of constant loaded SCC life tests were conducted for Alloys 600, 182, and 82 specimens in a high temperature, high-purity water environment. Failure time data was analyzed by means of the exponential distribution model. Effects of applied stress, electrode potential, and alloy chemistry were discussed on the probability distribution of the SCC initiation life.

4:30 PM
Modeling Noble Metal Coatings for Hydrogen Water Chemistry in BWRs: *D. D. Macdonald*[1]; I. I. Balachov[1]; [1]SRI International, Pure and Applied Phys. Sci. Div., Menio Park, CA 94205 USA
SCC degradation of reactor in-vessel and ex-vessel represents a significant obstacle for safe and economical operation of boiling water reactors (BWR). Easily applied, in-site methods for inhibiting crack growth are under development and testing. The electrochemical theories of SCC predict that crack growth will be mitigated by catalyzing the redox reactors that occur on the metal surface, provided that excess hydrogen (molar ratio> 1) is present (Noble Metal chemical Additions, NMCA), or by inhibiting the reactions under unrestricted conditions (Dielectric Coatings). However, the extent of catalytic effect of noble metals on the corrosion potential and crack growth rate in SS304 components of an operating BWR. The accumulated damage, calculated over ten years of operation of a BWR taking into account changes in plant operating parameters (temperature, flow rate, chemistry, etc., simulating outages, startups, shutdowns, and variations in reactor power during fuel cycles), is compared to the operation of normal hydrogen, and noble metal enhanced water chemistry conditions.

5:00 PM
The Prediction of Integral Damage Functions of IGSCC in BWR Primary Circuits: *D. D. MacDonald*[1]; [1]SRI International
Abstract Text Not Available

PWR Primary: Session II - Chemistry and Failure Analysis

Monday PM — Room: Pacific Ballroom C
August 2, 1999 — Location: Marriott Hotel

Session Chairs: P. M. Scott, Framatome Tour Fiat, Paris - La Defense, Cedex 16 92804 France; T. Yonezawa, Mitsubishi Heavy Industries, Takasago R&D Center, Takasago, Japan

1:30 PM
The Effect of Primary Coolant Zinc Additions on the SCC Behavior of Alloy 600 and 690: *M. G. Angell*[1]; S. J. Allen[1]; G. P. Airey[2]; [1]British Energy, Barnett Way, Barnwood, Glouster GL4 7RS UK; [2]British Energy
It has been postulated that the addition of Zinc to the primary coolant of pressurized water reactors (PWRs) can reduce the radiological dose and reduce the stress corrosion cracking (SCC) susceptibility of Alloy 600 components. Tests have been carried out to assess the effect of a Zinc addition on SCC initiation and growth on Alloy 600, reactor pres-

sure vessel (RPV) penetrations and Alloy 600 and Alloy 690 steam generator (SG) tubing. Crack initiation was assessed in SG tubing reverse U bends (RUBs) and bent beams fabricated from RPV penetrations, while crack growth was measured in compact wedge open loading specimens. For alloy 600, zinc addition inhibited crack initiation but had no effect on crack growth. There was no crack initiation in Alloy 690, SG tubing or RUBs in any test environment. The test data was correlated with surface oxide measurements.

2:00 PM

PWSCC of Alloy 600: A Parametric Study: *C. Soustelle*[1]; M. Foucault[1]; A. Gelpi[1]; P. Combrade[1]; T. Magnin[2]; [1]Framatome, Tour Fiat, 1 Place de la Coupole, Paris -La Defense, Cedex 16 92084 France; [2]Ecole des Mines de Saint Etienne France

The effects of temperature (from 290 to 360°C) and hydrogen partial pressure (from <0.01 to 20 bar) on PWSCC of Alloy 600 in primary PWR water have been studied. In particular, $K_{I\text{-}SCC}$ and plateau crack growth rates on CT specimens and crack initiation time on smooth surfaces were determined at 360°C in presence of different partial pressures of hydrogen selected to promote very different sensitivities to PWSCC. The observed cracking behavior has been correlated with the characteristics of surface oxide films such as thickness, composition, structure and electrical properties. Comparison with oxide films grown on Alloy 690 in the same conditions was also performed. The results obtained in this study, along with other recent results of this literature, are discussed in the scope of the different mechanisms which are currently proposed to explain PWSCC of nickel base alloy.

2:30 PM

Modeling of the Stress Corrosion Cracking Initiation on Alloy 600 in Primary Water: *Son Le Hong*[1]; Claude Amzallag[2]; Angel Gelpi[3]; [1]Electricité de France, R&D Div., Centre des Renardières, Moret Sur Loing 77818 France; [2]Electricité de France, Septen, 12-14 Avenue Dutrièvoz, Villeurbanne Cedex 69628 France; [3]Framatome, Cedex 16, Paris La Défense 92084 France

A model is proposed for the evaluation of the stress corrosion cracking (SCC) initiation. This model has been developed with the results obtained by several laboratories on steam generator tubes in Alloy 600. The model is based on the observation of three different stages of the SCC: an incubation, a slow propagation and a rapid propagation. The initiation time is defined as the necessary time to reach the rapid propagation stage, assuming that the transition to the rapid propagation is determined by a critical crack length or a threshold of the stress intensity factor KISCC. The initiation time is therefore mainly influenced by the slow propagation of the crack in the surface layers of the material and we have observed indeed that the time to failure can be lower for specimens with a cold worked surface layer than without. The surface layer has been then characterised in terms of cold work, residual stress, mechanical behaviour and sensitivity to the SCC. This model particularly takes into account the influence of the surface condition on the initiation time. It shows that the initiation time does not only depend on the stress – even very high – at the surface of the material, but also on the depth of the cold worked layer and on the mechanical loading state of the bulk material. This model has been validated by the comparison with available results on specimens whose surface condition has been characterized and it can be extended to other materials in Alloy 600.

3:00 PM Break

3:30 PM

Effect of Water Chemistry on Environmentally Assisted Cracking in Alloy 600 in Simulated Primary Side PWR Environments: *Per Lidar*[1]; Martin König[1]; Jan Engström[2]; Karen Gott[3]; [1]Studsvik Material AB, Corrosion and Crack Growth, Nykoping SE-611 82 Sweden; [2]Vattenfall Energisystem AB, P.O. Box 528, Vällingby SE-162 16 Sweden; [3]SKI, Stockholm SE-106 58 Sweden

The crack growth in Alloy 600 in simulated primary side PWR environments has been studied. The purpose of this work was to quantify the effects of water chemistry (Li, B and H2 concentrations, and the pH-value of the water by adding KOH) on the crack growth rate, da/dt. A factorial test design was used including 16 different environments. The da/dt was measured on 12.5mm wide Compact Tensions using on-line crack measurements with the DC Potential Drop technique. A once-through autoclave loop was used at 330°C. The Alloy 600 spare pipe came from a heat used for reactor vessel head penetrations. The measured da/dt varied with more than a factor of 30. Cracking due to intergranular stress corrosion cracking was dominant in the specimens. Multivariate analysis was used on the results. A partial least squares (PLS) regression was developed and interaction effects between the four variables were found. The PLS regression predicts observed da/dt very well.

4:00 PM

Unique Primary Side Initiated Degradation in the Vicinity of the Upper Roll Transition in Once Through Steam Generators from Oconee Unite 1: *J. P. Molkenthin*[1]; T. P. Magee[1]; J. F. Hall[1]; G. C. Fink[1]; Dewey Rochester[2]; Al McIlree[3]; [1]ABB Combustion Engineering Nuclear Operations, CEP 9459-1209, P.O. Box 500, 100 Prospect Hill Rd., Windsor, CT 06095 USA; [2]Duke Power Company, Charlotte, NC USA; [3]Electric Power Research Corporation, Palo Alto, CA USA

At the end of the cycle 17 (EOC17) outage (Fall 1997), Duke Power Company (DPC) removed sections of tubing from the upper tubesheet region of five steam generator tubes from a Oconee Unit I Once Through Steam Generator (OTSG) for laboratory non-destructive and destructive examinations. Eddy current testing (ECT) of the steam generator tubes during the outage indicated each tube had indications in the vicinity of the roll transition zone (RTZ). The emphasis of the laboratory activities was to investigate the nature of these indications in the vicinity of the RTZ and to determine the causes of these phenomena. In specific, were these indications typical of PWSCC, patch like intergranular attack (IGA), wear, some other type of degradation or possibly some type of manufacturing or installation artifact. The examinations conducted included: visual and dimensional examinations to document the as-received condition of the tubes; laboratory ECT of the tube sections using bobbin coil, 2 and 3-coil motorized rotating pancake coil (MRPC), MICA, CECCO and Delta coils; helium leak testing using the Mass Spectrometer Leak Detector (MSLD) to detect any helium leakage through the walls of selected tube sections; dye penetrant testing and silastic molding; light optical microscope characterization of the defects.

BWR: Session III - SCC Data Quality & System Definition

Tuesday AM Room: Pacific Ballroom D
August 3, 1999 Location: Marriott Hotel

Session Chair: R. Pathania, EPRI, Palo Alto, CA USA; P. Ford, GE R&D Center, USA

8:00 AM

SCC Testing and Data Quality Consideration: *Peter L. Andresen*[1]; [1]G.E. Corporation Research & Development, One Research Circle, K1, -3A37 Room 3A39, P.O. Box 8, Schenectady, NY 12309 USA

Continuous measurement of crack growth response provides the most statistically reliable and efficient approach for characterizing stress corrosion cracking and its dependencies. Additionally, in many systems "initiation" and growth occur by the same mechanism, and cracks on a very small scale (e.g., <20 um) possess many common characteristics of larger cracks. However, collations of crack growth data – even for a limited range of well-defined conditions – show very large scatter. Many extrinsic (i.e., experimental) and intrinsic factors are responsible for this scatter, and the accuracy and reproducibility can be dramatically improved when these factors are identified and optionally controlled. Compared to determinations of strength, hardness, Charpy, fracture toughness, fatigue crack growth, etc., SCC is a vastly more complex phenomena with numerous interdependent controlling parameters, and yet no useful standards exist to help ensure high quality data. This paper highlights the most critical extrinsic and intrinsic factors in SCC testing, and proposes guidelines for achieving optimal results.

8:30 AM
Stress Corrosion Cracking of Sensitized Type 304 Stainless Steel in 288°C Water: A Five Laboratory Round Robin: *Peter L. Andresen*[1]; Karen Gott[2]; J. Larry Nelson[3]; [1]GE Corporation Research & Development, One Research Circle, Rm. 3A39, Schenectady, NY 12309 USA; [2]Swedish Nuclear Power Inspection, Klarabergsviadukten 90, Stockholm S-10658 Sweden; [3]EPRI, 3412 Hillview Ave., Palo Alto, CA 94303 USA

The international stress corrosion crack growth data on structural materials exhibits a large scatter, making it difficult to determine dependencies and even establish an overall bounding disposition response with confidence. It was suspected that this problem was strongly related to differences in testing and measurement techniques more so, e.g., than material variability. This program selected five of the world's best laboratories, and provided them with identical, fatigue pre-cracked CT specimens and specified identical, well-controlled test conditions. The results showed that many factors have a very large effect on the observed crack growth rate, and that there were substantial differences in laboratory practice and measurements. When common techniques were employed, similar growth rate data could be obtained, although achieving well-behaved growth rates is not reliably achieved simply by following standard "recipes". An overview of the program, crack growth results, recommendations for testing methods, and implications is given.

9:00 AM
First Lower Plenum ECP Measurement in an Operating BWR: *S. Hettiarachchi*[1]; D. A. Hale[1]; R. Burrill[1]; S. Suzuki[2]; [1]GE Nuclear Energy, 6705 Vallecitos Rd., Surol, CA 94586 USA; [2]Tokyo Electric Power Company, Yokohama, Japan

It is well known that good control of electrochemical corrosion potential (ECP) is an absolute necessity to maintain the intergranular stress corrosion crack (IGSCC) mitigation of the boiling water reactor (BWR) internals. Hydrogen Water Chemistry (HWC) has been recognized to be the most effective method of achieving such IGSCC mitigation of reactor internals. However, increased hydrogen addition increases the main steam line operating dose rate which is a major side effect of the HWC operation of BWRs. To control operating dose rate, but still achieve ECP of internals below the HWC specification potential of -230 mV(SHE), it is important to inject the minimum hydrogen levels into the feedwater of the BWR. In the past, this has been achieved by monitoring ECP of BWR internals surfaces at numerous accessible locations such as the lower core, upper core, recirculation line and the bottom head drain line either by using modified local power range monitors (LPRMs) or by using ECP flanges. However, these measurements do not provide information on the ECP behavior of the lower plenum to feedwater hydrogen addition, which is critical for IGSCC mitigation of lower plenum internals. This paper describes the first ECP measurement and the ECP mapping of the lower plenum of a BWR. The measurements resulted in significant findings that helped to improve the understanding of the ECP response of the lower plenum to hydrogen addition.

9:30 AM
The Role of H2O2 on BCP Under Different Flow Rate and Different Temperature: *M. Samborgi*[1]; [1]Tokyo Electric Power Company, Mats. Eng. Lab.

Recently the numbers of SCC of in-line components is increasing in BWR. It is well known that the in-environment is pretty different from the one of environment because of the evidence of H2O2 generated by water. In this paper, the relation between electrochemical oxidation potential (BCP) and oxidants such as H2O2,O2 is evaluated as a function of the rate in 288°C pure water with and without pre. Besides the oxidation between BCP and as well as impurities (SO3, OO3) is also evident, varying the amount of H2O2 to stimulate the BWR condition such as start-up. As a result, small amount of H2O2 below 5ppb is found to affect BCP and its technology becomes more remediable under higher flow rate. Finally, the correlation between actual in core BCP data and the effect of is decreased.

10:00 AM Break

10:30 AM
Corrosion Potential Monitoring in Swedish BWRs on Hydrogen Water Chemistry: *Anders Molander*[1]; Gosta Karlberg[2]; [1]Studsvik Material, AB, SE-611 82, Nykoping SE-611 82 Sweden; [2]Barseback Kraft AB, P.O. Box 524, Loddekopinge SE-246 25 Sweden

A compilation of corrosion potential monitoring results from Swedish BWRs with external recirculation lines and on hydrogen water chemistry (HWC) has been performed. The compilation comprises measurements in the main recirculation system and in the water clean up system and is focused on the period 1994-1997. The needed amount of hydrogen to suppress the corrosion potential varies with operational parameters and with time. To illustrate the conditions, a so-called operation line for HWC has been defined and used for further data interpretation. The operation line relates the corrosion potential to the calculated downcomer hydrogen concentration and main recirculation flow. The operation line concept is used to compare and analyze results between the plants and explain differences between reactor cycles. Many of the differences between the plants with respect to HWC are explained by plant specific circumstances but also from the different approaches to control HWC adopted by the plants. Finally the results are compared to laboratory data and to modeling results. The BwrChem model has been adopted for calculation of environmental conditions and is in this work extended to corrosion potential calculations.

11:00 AM
ECP Suppression Mechanism and ECP Simulation for a Small-Area Noble Metal Deposition under Hydrogen Water Chemistry Conditions: *Masanori Sakai*[1]; [1]Hitachi, Ltd., Hitachi Research Laboratory, 1-1, Oimka-cho 7 Chome, Hitachi-shi, Ibaraki-ken 319-12 Japan

Noble metal chemical addition has been shown, recently, to be an effective technique for hydrogen water chemistry (HWC) with regard to IGSCC mitigation in BWR plants. Deposition of a small amount of noble metal is now known to suppress ECP of BWR structural materials under HWC conditions. ECP suppression of substrate materials has been experimentally verified for deposited noble metal amounts less than micrograms per cm^2. However, because of the small ratio of the deposited noble metal area to that of the substrate, the substrate ECP suppression can not be demonstrated quantitatively under HWC conditions if only hydrogen electrode reversibility is considered for the deposited noble metal. In the author's laboratory, ECP suppression mechanism and ECP simulation for a small-area noble metal deposition under HWC conditions has been studied. In this paper, an ECP suppression model for a small area noble metal deposition is introduced and ECP simulation performed by this model under BWR conditions is presented.

PWR Primary: Session III - Hydrogen Effects & Microstructure

Tuesday AM	Room: Pacific Ballroom C
August 3, 1999	Location: Marriott Hotel

Session Chairs: W. J. Mills, Bettis Atomic Power Laboratory, West Mifflin, PA 15122-0079 USA; P. Doherty, Babcock & Wilcox, Canada

8:00 AM
On the Possible Existence of the Ordered Phase Ni_2 Cr in Alloy 690: *J. O. Nilsson*[1]; T. Larsson[1]; J. Frodigh[2]; [1]Sandvik Steel, Res. and Dev. Cen., Dept. of Phys. Metall., Sandviken, Sweden S-811 81; [2]Dept. of Tube Research

There is a concern that the ordered phase Ni_2Cr precipitates in Alloy 690 at service temperatures in the interval 300-400°C thereby possibly causing embrittlement and premature IGSCC failures. Although no direct proof of the existence of Ni_2Cr in Alloy 690 has been presented, observations of a hardness increase after long term service have been interpreted as an indication of a phase transformation. This hardness increase is tentatively explained in terms of the precipitation of Ni_2Cr. The problem is one where the proof of non-existence of Ni_2Cr can be of crucial importance. In the present, work results from new and carefully designed experiments will be presented and critically examined. Alloy 690 was isothermally aged for 3000 h at 300, 400 and 500°C, subsequently tested with respect to hardness and examined in the transmission electron microscope (TEM). Certain specimens were intentionally pre-strained to 2% and 5% before microstructural investigation to allow dislocation-precipitate interaction to be investigated. Differential scanning calorimetry (DSC) and differential thermal analysis (DTA) were used as a complement to TEM to study phase transformations. To facilitate interpretation of microstructures in Alloy 690 similar investigations were performed on Hastelloy C-276, which is an alloy well documented for its formation of Ni_2Cr. The precipitation of Ni_2Cr in Hastelloy C-276 was unambiguously confirmed by electron diffraction patterns on zone axes <001>, <011>, <111> and <112>. Indirect evidence of Ni2Cr in deformed specimens was found in terms of triple dislocations. Hardness testing of Alloy 690 showed no significant aging-induced hardness increase. No evidence of the presence of Ni_2Cr in Alloy 690 was found in electron diffraction patterns on zone axes <00l>, <011>, <111> and (112). This in combination with the total absence of triple dislocations show that if Ni_2Cr at all exists in Alloy 690 the volume fraction must be extremely small and presumably without practical significance. Results from DSC and DTA support this view.

8:30 AM

Hydrogen Embrittlement of Ph 13-08 Mo Stainless Steel in PWR Environment: Effects of Microstructure: *J. M. Cloue*[1]; M. Foucault[1]; E. Andrieu[2]; [1]Framatome Technical Center, Porte Magenta BP 181, I-20-DAW, Lyon Cedex 6 France; [2]ENSC Toulouse, 118 Route de Narbonne, Toulouse, Cedex 04 31077 France

The hydrogen embrittlement of PH 13-08 Mo in PWR primary water was studied on flat and round products with different aging heat treatments. The study was focused on the effects of microstructure and more particularly the reverted austenite on material behavior in PWR primary water in comparison with alloy behavior in gaseous hydrogen environment. Some corrosion tests as SSRT tests at different deformation rates and U bent tests were performed. PH 13-8 Mo is sensitive to hydrogen embrittlement in both gaseous and PWR primary water conditions. Few ppm of absorbed hydrogen are sufficient to induce severe embrittlement during SSRT tests. The rupture occurs by quasi-clivage and the degree of embrittlement is dependent on material microstructure. The proportion of reverted austenite appears to be the most effective parameter on hydrogen effects and controls the rupture mode.

9:00 AM

Combined Effect of Special Grain Boundaries and Grain Boundary Carbides on IGSCC of Ni-16Cr-9Fe-xC: *Brent M. Capell*[1]; Bogdan Alexandreanu[1]; Gary S. Was[1]; [1]University of Michigan, Dept. of Nucl. Eng. and Radiol. Sci., 2940 Cooley Bldg., 2355 Bonisteel Blvd., Ann Arbor, MI 48109-2104 USA

Susceptibility to intergranular stress corrosion cracking in Alloy 600 and in Ni-16Cr-9Fe-xC alloys has been shown to be reduced for alloys with either an increased fraction of special grain boundaries, i.e. coincident lattice site boundaries (CLSB) and low angle boundaries, or grain boundary carbides. The cracking of alloys containing both microstructure features was investigated using interrupted constant extension rate tensile tests in a primary water environment (360°C, 0.001M LiOH + 0.01M H_3BO_3 solution with less than 5 ppb O_2 content). Heat treatment and deformation were used to increase the fraction of special boundaries. Lower temperature heat treatments were also developed which resulted in preferential formation of carbides along high angle grain boundaries. Orientation Imaging Microscopy was used to determine the relative grain misorientations and scanning electron microscopy was used to quantify grain boundary carbide locations prior to testing. After each strain increment, the same areas in each sample were examined. Results taken from CSLB-enhanced materials show that cracks develop preferentially on high angle boundaries and that samples with higher CSLB fractions have fewer cracks. Results will be presented on the effectiveness of these microstructure features, individually and in combination, in reducing IGSCC susceptibility.

9:30 AM

Environmental Cracking Behavior of Nickel-Based Alloys in Low Temperature H_2O: *W. J. Mills*[1]; C. M. Brown[1]; M. G. Burke[1]; [1]Bettis Atomic Power Laboratory, P.O. Box 79-ZAP 03N, West Mifflin, PA 15122-0079 USA

The cracking resistance of Alloy 600, Alloy 690 and their welds, EN82H and EN52, was characterized by conducting J_{IC} tests in low temperature water with elevated hydrogen levels. Alloy 690 and the two welds were found to be susceptible to low temperature embrittlement, as J_{IC} values in 54°C water with 150 cc H_2kg H_2O were typically 70% to 90% lower than their air counterparts. Tearing moduli in hydrogenated water were reduced by almost two orders of magnitude. The toughness degradation was associated with a fracture mechanism transition from microvoid coalescence to intergranular fracture. Comparison of the cracking response in water with that for hydrogen-precharged specimens tested in air demonstrated that susceptibility to low temperature cracking is due to hydrogen embrittlement of grain boundaries. Decreasing the hydrogen content of the water from 150 to 15 cc H_2lkg H_2O improved the cracking resistance of Alloy 690 but had little or no effect on the weld response. Susceptibility to low temperature embrittlement disappeared when the test temperature was increased above 300°F. Unlike the other materials, Alloy 600 is not susceptible to low temperature cracking as the toughness in 54°C water remained high and a microvoid coalescence mechanism was operative in both air and water.

10:00 AM Break

10:30 AM

Hydrogen-Assisted Failure of Alloys X-750 and 625 Under Slow Strain-Rate Conditions: *Robert S. Daum*[1]; Arthur T. Motta[1]; Donald A. Koss[2]; Digby D. Macdonald[2]; [1]Penn State University, Dept. of Mech. and Nuclear Eng., 231 Sackett Bldg., University Park, PA 16802-1408 USA; [2]Penn State University, Dept. of Mats. Sci. and Eng., 202 Steidle Bldg., University Park, PA 16802-1408 USA

The hydrogen embrittlement (HE) susceptibility of nickel-based alloys X-750 and 625 was evaluated using a slow strain rate test in a pure water solution. Tests were conducted at 26°C and 288°C in aqueous environments with two levels of hydrogen in solution (0 and 60 cc hydrogen/kg water STP). HE effects were determined by measuring the time and elongation to failure, reduction in area at failure, amount of intergranular fracture, and surface crack characteristics. No clear evidence of HE was observed at 288°C in either alloy. At room temperature, both alloys experienced a severe loss of ductility (by about 50%) and transition from a predominantly ductile to a brittle, intergranular fracture, as the hydrogen level increased. Fractography of the room temperature specimens showed a crack growth transition from a mixed mode slip band decohesion in the low hydrogen environment to a mode I intergranular fracture path in the high hydrogen environment.

11:00 AM

An Experimental Study of the Hydrogen Embrittlement of Alloy 718 in PWR Primary Water: *O. Brucelle*[1]; J. G. Spilmont[1]; J. Cloue[1]; M. Foucault[1]; E. Andrieu[2]; [1]Framatome Fuel Division, Tour Fiat, 10, Rue, 1 Place de la Coupole, Paris - La Defense, Cedex 16 92084 France; [2]ENSC, Toulouse

Hydrogen embrittlement of alloy 718 in relation with surface reactivity aspects (kinetics and corrosion products) was studied at 80°C and 300°C by means of SSRTT. Thus, materials with different prestraining were used together with samples with different thickness. Conventional tensile and low cycle fatigue mechanical tests have also been carried out under air testing conditions on pre-exposed specimens in order to assess the effect of degassing on the mechanical properties of this alloy. A detailed examination of fractured surfaces allowed to define microstructural characteristics associated with the hydrogen embrittlement of this alloy. An effect of hydrogen on both brittle and ductile fracture process is shown. A partial reversibility of hydrogen embrittlement was also evidenced and finally, conclusions can be drawn on the mechanism of hydrogen ingress of the material during service conditions.

11:30 AM

Stress Corrosion Crack Propagation Rate of Alloy 600 in Primary Water of PWR, Influence of a Cold Worked Layer: *O.*

Raquet[1]; [1]Service de Corrosion, d'Electrochime et de Chimie des Fluides, Gif Sur Yvette, 0 91191 France

Constant elongation rate tests were carried out in 360°C primary water on electropolished Alloy 600 specimens without a superficial cold worked layer. It is shown that crack propagation begins with a slow rate, and that, for the largest cracks this first stage is followed by a fast-rate propagation stage. This result demonstrates that a cold worked layer is not necessary for a slow rate-fast rate transition to occur; however, it is liable to influence this transition. In order to investigate the influence of the depth of a cold worked layer, constant elongation rate tests and constant load tests were performed on shot peened specimens with the same surface stress level. It is shown that a large depth of cold worked layer increases strongly the average propagation rate of the latest cracks. Complementary tests are performed to investigate the effect of cold work on the slow-rate propagation stage, the fast-rate propagation stage and on the depth at which this rate transition occurs.

Regulation Aspects

Tuesday PM — Room: Pacific Ballroom C
August 3, 1999 — Location: Marriott Hotel

Session Chairs: F. P. Ford, GE CRD, USA; K. Gott, SKI, Sweden

6:00 PM

EPRI R&D for Safe and Economic Long-term Nuclear Plant Operation: *R. Jones*[1]; J. Carey[1]; C. Wood[1]; [1]EORI, Mats. and Corrosion, 3512 Hillview Ave., Palto Alto, CA 94303 USA

EPRI research initiated over 20 years ago identified the potential for safe and economic operation of nuclear power plants well beyond the initial licensed term of 40 years. Early studies of specific plants identified critical plant components and structures important to long-term plant operation and significant aging effects requiring cost-effective aging management. These studies of plant aging and life extension by EPRI, the U.S. Nuclear Regulatory Commission and other industry organizations have evolved to the point where the regulatory requirements for license renewal have been defined and two utilities have recently applied for license renewal for their nuclear units. The efforts to establish the license renewal process, has focused on the management of detrimental aging effects on important plant equipment. This has led to a systematic review and assessment of aging mechanisms including the full range of potential effects and the adequacy of monitoring, inservice examination, evaluation, repair and replacement programs. Many of these formal license renewal activities have been incorporated into operating plant system, structure, and component life cycle management efforts and this trend is expected to continue. As a result, research and development (R&D) work in support of license renewal is gradually being adapted to the management of aging in support of cost-effective operation of U.S. commercial nuclear power plants. In this paper, five areas of EPRI R&D in support of license renewal are described, and the transitional adaptation to life cycle management is outlined. The first area is the license renewal evaluation of fatigue crack initiation and growth, which has been dominated by considerations of design-basis versus actual operating transients and by potential reactor water environmental effects. The use of on-line cyclic monitoring has been able to show that such considerations are compensated for by conservatism in the fatigue design process. In addition, the EPRI R&D results related to on-line monitoring provide essential information needed to make run, repair, replace decisions as a part of the life cycle management process. Four other areas will also be covered in this paper: (1) long-range R&D in support of the assessment of neutron irradiation embrittlement effects on austenitic stainless steel reactor internals, including potential reductions in ductility, reductions in preload for bolts, pins, and fasteners, and stress corrosion cracking; (2) continuation of a long-term R&D program on neutron irradiation embrittlement of low-alloy steel pressure vessel materials; (3) additional R&D in support of inservice inspection procedures, flaw evaluation methodology, and repair strategies for critical plant components subject to stress corrosion cracking erosion-corrosion, and corrosion-assisted fatigue; and (4) R&D to develop practical and cost-effective condition monitoring program elements for managing long-term degradation of steel and concrete containments and other structures, with an emphasis on areas that are nominally inaccessible and which may be subject to aggressive environments."

6:30 PM

The History of Cracking the RCPB of Swedish BWR Plants: *Karen Gott*[1]; [1]Swedish Nuclear Power Inspectorate, S-106 58, Stockholm S-106 58 Sweden

The nuclear power plants in Sweden are required to report all cracks to the Swedish Nuclear Power Inspectorate (SKI). This rule applies to all the systems covered by SKIs regulations concerning the structural integrity of mechanical components. As a result SKI has over the years gathered extensive information concerning the history of the various degradation mechanisms which have been observed in Swedish plants. In the last couple of years this information has been put into a database set up specifically for the purpose. The information in the base includes details of when and how the cracks were detected, their dimensions and cause, as well as system and component details The database also has a comprehensive reference list of all the related documentation associated with a crack or group of cracks. The database will be described and its use illustrated with the trends found in the RCBP of the Swedish BWR.

7:00 PM

Plant Life Management Program Preparation for VVER-440/v-213C Units in Czech Republic: *Milan Brumovsky*[1]; Martin Ruscak[1]; Jiri Zdarek[1]; [1]Nuclear Research Institute, Rez, Div. of Integrity and Mats., Vltavska 1, Rez 250 68 Czech Republic

Plant life and life management program for VVER-440/V-213C type units in Czech republic has been under preparation for a long time. Analysis of potential degradating mechanisms in main reactor components have been performed, categorization of components has been the following step. Quantitative analysis of degradating consequences together with their stressors sources have been used for a preparation of procedures for continuous/periodic assessment of degradation rate. This rate is compared with a design one to be able to set necessary improving measures. Finally, for the most important component, a set of procedures has been finalized to define all necessary parameters to be measured, their allowed values and degradation rates. In the same time, for the most critical parts, mitigation measures have been also proposed.

7:30 PM

Regulatory Perspective of Industry's Response to GL 97-06, "Degradation of Steam Generator Internals": *Stephanie M. Coffin*[1]; [1]Nuclear Regulatory Commission, NRR/Div. of Eng., Mail Stop O-7-D4, Washington, DC 20555 USA

On December 30, 1997, the NRC staff issued Generic Letter 97-06, "Degradation of Steam Generator Internals" to communicate findings of damage to steam generator internals at foreign and U.S. pressurized water reactor facilities. The reported foreign and U.S. experience highlighted the potential for degradation mechanisms that may lead to tube support and tube bundle wrapper damage. The steam generator tube brace supports the tubes against lateral displacement, vibration and minimizes bending moments in the tubes in the event of an accident. Tube support damage can impair the support's ability to perform this function and, thus, could potentially lead to the impairment of tube integrity. Vibration-induced fatigue could present a potential problem if tube supports lose integrity, particularly in areas of high secondary side cross flows. The staff also emphasized in the generic letter the importance of performing comprehensive examinations of steam generator internals to ensure steam generator tube structural integrity is maintained. The staff reviewed the industry's response to the generic letter to ensure that the condition of steam generator internals comply with and conform to the current licensing bases.

8:00 PM

Status of Review of Issues and Applications for License Renewal by Materials Engineering Staff at the US Nuclear Regu-

latory Commission: *M. Banic*[1]; [1]US Nuclear Regulatory Commission, Mats. and Chem. Eng. Branch, O7D4, Washington, DC 20555 USA
This paper provides an update of license renewal at the US Nuclear Regulatory' Commission (NRC) with emphasis on the review by the staff of the Materials and Chemical Engineering Branch. The paper covers the rules governing license renewal, the applications received, the schedule, the approach, and the technical issues. The NRC has a tight schedule of 30-36 months to renew a license. Expecting many applicants for license renewal, the staff has taken steps to address the public's concern that the effects of aging will be adequately managed and the industry's concern that the reviews will be timely, efficient, and uniform. These steps include identifying aging effects and making the results available in a report and computerized database, approving topical reports and aging management programs for generic use, and reviewing aging management programs according to specific criteria. Members of the Materials Engineering staff have a major role because many of the aging issues are materials related.

Welding/Processing

Tuesday PM — Room: Pacific Ballroom D
August 3, 1999 — Location: Marriott Hotel

Session Chair: N. Iyer, Westinghouse, USA

6:00 PM
Development of Compressive Residual Stresses in Underwater PTA Welds: *A. Fang*[1]; [1]EWI
The repair of nuclear reactor internals can generally best be done underwater, to take advantage of the shielding that the water provides, and to eliminate the need to remove the fuel. These advantages prompted a study of underwater plasma transfer are (PTA) welding. It was found that there was actually a benefit to performing the welding underwater. Rather than the typical tensile residual stresses that are developed in the weld and HAZ (which promotes SCC), the underwater welds exhibited compressive stresses, or at least greatly reduced tensile stresses. This paper aims at explaining why this is the case. Welding under water produces a significant difference in the temperature distribution. In conventional welding, the poor heat conduction out from the face of the plate being welded, results in the surface being hot relative to the interior of the plate. Welding underwater reverses this gradient because of the good heat conduction into the water. This temperature distribution difference is modeled using FEA and used to demonstrate, through FEA, why these compressive stresses develop.

6:30 PM
Development of Repair-Welding Technology for Irradiated Materials of LWR: *Kiyotomo Nakata*[1]; Hiroyuki Takeda[1]; Shigeki Kasahara[1]; Masayuki Oishi[1]; [1]Japan Power Engineering and Inspection Corporation, Tokyo Research & Development Center, Business Court Shin-urayasu Bldg., 9-2, Mihama 1-Chome, Urayasu, Chiba 279-0011 Japan
Repair of irradiation-degraded components will be necessary for plant life extension. Welding in neutron-irradiated stainless steels is not always successful, while welding is the primary technique for repairing core components. Helium cavitation on grain boundaries in weld heat-affected zones is one of the causes of weld cracking in irradiated steels. Helium concentrations, generated by thermal neutron irradiation, were calculated to be about 20 appm in a core shroud and about 8 appm in a reactor vessel after the 40- and 60-year operation, respectively, which were significantly higher than the limiting helium concentration of about 1 appm for sound welding by the conventional TIG welding. In order to suppress the helium cavitation, the welding techniques with low heat inputs and high cooling rates were beneficial. A new national project on the development of repair-welding technology for core internals and reactor (pressure) vessel has been started in Japan. Stainless steels and low alloy steel specimens were irradiated up to 2X10^25 and lX10^24 n/m^2 (>1 MeV), respectively. The YAG laser welding with heat inputs of less than 0.4 MJ/m was applied for buildup welding in the irradiated specimens.

7:00 PM
Use of Abrasive Waterjet for Machining and Remediation of Nuclear Components: *J. B. Hall*[1]; K. B. Stuckey[1]; S. Fyfitch[1]; [1]ABB Combustion Engineering, CEP 9459-1209, P.O. Box 500, 100 Prospect Hill Rd., Windsor, CT 06095 USA
Various cracking mechanisms seen in the nuclear reactor coolant system such as IASCC, PWSCC, and IGSCC require three factors to be present simultaneously: aggressive environment, susceptible material, and significant prolonged tensile stress. Elimination of any one of the three required factors will mitigate cracking. An abrasive waterjet process has been qualified, and used to remove flaws and to remediate the surface. This process removes both flaws detected by non-destructive examination and flaws that are too small for detection. In addition, the remediation process eliminates the residual tensile stresses and leaves a residual compressive stress thereby mitigating the aforementioned cracking mechanisms. Alloy 600 tubes of the same nominal dimensions of a B&W-designed Control Rod Drive Mechanism (CRDM) nozzle were used as representative mockups for abrasive water jet (AWJ) surface remediation process qualifications. A process was also qualified to remove a detected flaw using AWJ, filling of the cavity by welding, and surface remediation of the weld and surrounding area. Residual stress measurement, metallurgical examination, and embedded particle analysis were performed on the AWJ remediated surface in order to qualify the process. These results are presented and discussed.

7:30 PM
Surface Modification for PWSCC Prevention of Alloy 600 by a Laser Cladding Technique: *Shigeki Kasahara*[1]; Masatoshi Sato[1]; Toshizo Ohya[2]; Masaya Kanikawa[3]; Hiroshi Kanasaki[1]; [1]Japan Power Engineering and Inspection Corporation, Tokyo R&D Center, Business Court Shin-urayasu 4F, 1-9-2 Mihama, Urayaasu, Chiba 279-0011 Japan; [2]Mitsubishi Heavy Industries, Ltd., KOBE Shipyard & Machinary Works, PlantService Engineering Sect., 1-1-1 Wadasaki, Hyogo-ku, Kobe, Hyogo 652-8585 Japan; [3]Mitsubishi Heavy Industries, Ltd., Takasago R&D Center, 2-1-1 Arai-cho, Takasago, Hyogo 676-8686 Japan
Primary water stress corrosion cracking (PWSCC) in Ni base alloy is one of the serious degradation phenomena in components of the primary side of PWRs. At the viewpoints of PWSCC prevention for the components which are difficult to be replaced, application of surface modification by a laser heat input method is studied. The modified surface layer, which is alloyed into the chemical compositions equivalent to Alloy 690 by a laser cladding technique, is promising for the mitigation of PWSCC susceptibility in Alloy 600. The optimization of the laser cladding process is carried out on the inner surface of the pipe specimens drilled from Alloy 600 rods. PWSCC susceptibility of the modified surface is also examined in a simulated primary water condition.

8:00 PM
Welding as a Repair Option for BWR In-Vessel Components: *A. L. Lund*[1]; L. E. Willertz[2]; R. C. Thomas[3]; R. L. Dyle[4]; [1]U.S. Nuclear Regulatory Commission, 10E10 TWF, Washington, DC 20555 USA; [2]Pennsylvania Power and Light; [3]Electric Power Research Institute; [4]Inservice Engineering
The US Nuclear Regulatory Commission and Electric Power Research Institute/Boiling Water Reactor Vessel and Internals Project (EPRI/BWRVIP) are currently involved in a cooperative effort to evaluate the feasibility of welding on highly irradiated stainless steels in the boiling water reactors. Use of weld repair for highly irradiated components within the reactor vessel is subject to large uncertainties due to the effects of neutron bombardment on the component material, leading to helium-induced cracking resulting from the welding operation. Weld repair options are desirable because some components are in regions or are configured such that mechanical repairs are not possible or not economical. The program will attempt to determine: locations for which repair is feasible, the level of helium that eliminates welding as a repair option, welding techniques that can be employed and the range of helium concentrations for which each technique is applicable.

BWR: Session IV - Mitigation/Life Extension

Wednesday AM — Room: Pacific Ballroom D
August 4, 1999 — Location: Marriott Hotel

Session Chairs: L. Nelson, EPRI, USA; J. Hickling, CMC, Germany

8:00 AM
Electrochemical Validity of Noble Metal Technology for BWR Application: *Young-Jin Kim*[1]; [1]GE Corporate Research & Development Center, One Research Circle, K1-3A37, Schenectady, NY 12301 USA

The important influence of the ECP on the intergranular stress corrosion cracking (IGSCC) susceptibility of structural materials in BWRs is well established. The most efficient way to reduce ECF is to use noble metals to catalyze the recombination of H_2 and O_2 on the metal surface and thereby mitigate the crack growth rate with minimal negative impact on BWR operation. A wide variety of approaches (noble metal alloy: NMA, thermal spray coating of noble metal alloy powders: NMC, and noble metal chemical addition: NMCA) to apply noble metals to BWR components have been developed for creating a catalytically active surface. Recently, in-situ NMCA was applied to a commercial US BWR and it has been successfully demonstrated that NMCA plus low hydrogen was effective in mitigating ICSCC. The objective was to shed further light on the electrochemical catalytic kinetic behavior of noble metal-doped 304 SS in high temperature water. It is evident that the incorporation of small amounts of noble metals dramatically improves the electrochemical kinetics, particularly the hydrogen oxidation rate and thus enhances the catalytic recombination efficiency of H_2 to O_2 to form H_2O. The improvement in the catalytic property on 304 SS surface by doping noble metals results in a thermodynamically lowest ECP ($<$-500 mV_{SHE}) when a stoichiometric or higher amount of hydrogen is present in the water.

8:30 AM
Full Cycle Performance of a Noblechemô Treated BWR: *S. Hettiarachchi*[1]; R. J. Law[1]; W. D. Miller[1]; T. P. Diaz[1]; R. L. Cowan[1]; [1]GE Nuclear Energy, 6705 Vallecitos Rd., Surol, CA 94586 USA

Hydrogen Water Chemistry (HWC) has been successfully employed to mitigate the intergranular stress corrosion cracking (IGSCC) of boiling water reactor (BWR) internals over the past decade. However, the use of elevated levels of feed water hydrogen in the BWR results in high operating dose rates due to N^{16} partitioning into the main steam. Application of noble metals to BWR internal surfaces allows the use of low H (e.g. 0.4 ppm or less in the feedwater), with little or no increase in operating dose rates, but enables achieving the HWC specification criteria. A simple method of applying noble metal on to reactor internals involve the addition of a noble metal compound into reactor water to cause deposition of noble metal from solution onto surfaces. The noble metal chemical addition (NMCA) technology has been successfully used in numerous laboratory tests and in an operating BWR to produce a "noble metal like" surface on major structural materials. The NMCA technology (NobleChemô) has successfully decreased the ECP of surfaces below -230 mV(SHE), in stoichiometric excess hydrogen in real BWR environments. This paper describes the full cycle performance of the first NobleChemô treated BWR and the early results from the second NobleChemô treated BWR.

9:00 AM
Effect of Corrosion Potential on the SCC Initiation Lifetime of Alloy 182 Weld Metal: *Norihisa Saito*[1]; Shigeaki Tanaka[2]; Hiroshi Sakamoto[2]; [1]Toshiba, Met. & Cera. Tech. Gr., PIC, 8, Shinsugita-cho, Isogo-ku, Yokohama, Kanagawa 235-8523 Japan; [2]Toshiba, Appl. Metall. & Chem. Dept., Nuclear Energy Div., 8, Shinsugita-cho, Isogo-ku, Yokohama, Kanagawa 235-8523 Japan

To evaluate the effect of hydrogen water chemistry (HWC) on the SCC initiation life time of Alloy 182, periodically unloading uniaxial constant load (PU-UCL) tests were conducted, simulating the BWR water chemistry condition. Utilizing the exponential distribution model, the lifetime improvement factor was evaluated comparing the location parameter of the HWC condition to that of normal water chemistry (NWC) condition, which parameters were obtained as the lower limits of SCC lifetime in PU-UCL tests. The improvement factor of ten was expected for the moderate HWC simulate condition, that was represented by corrosion potential at 0 mV(SHE) where radiation dose rate did not increase in main steam, as compared with the NWC simulated condition that was represented by corrosion potential at +190 mV(SHE). It was found that the moderate HWC condition is beneficial to mitigate the SCC initiation of Alloy 182 weld metal in BWR environment.

9:30 AM
Effects of Mixed Metal Addition on Surface Film and Corrosion Prevention of Stainless Steel in BWR Water: *Takeshi Sakai*[1]; Yoshiyuki Saitoh[2]; Yuuji Midorikawa[2]; Teruchika Kikuchi[1]; [1]Nuclear Fuel Industries, Ltd., Eng. Svc. Dept., 950 Ohaza-Noda, Kumatori-Cho, Sennan-Gun, Osaka 590-0451 Japan; [2]Tohoku Electric Power Co., Inc., R&D Center, 2-17-Chome Nakayama, Aoba-Ku, Sendai, Miyagi 981-0952 Japan

As an alternative technique to zinc injection into BWR water, mixed metal addition has been developed. Mixed metal addition uses a compound of manganese, nickel, magnesium and small amount of natural zinc. Zinc concentration is decreased to the permissible limit that does not increase the radiation buildup of activated Zn-64, i.e., less than 1ppb according to the analysis of reactor water at Onagawa-1, and a synergistic effect of mixed metal elements to reduce corrosion and radiation buildup on stainless steel surfaces is expected. In the current study, high temperature autoclave testing was performed to investigate the effects of mixed metal addition on the oxide film characteristics and the corrosion of stainless steel in the simulated BWR environments. The results suggest that the mixed metal addition could be an alternative technique to zinc single injection.

10:00 AM Break

10:30 AM
The Predictive Effectiveness of NMCA at the Chinshan BWR: *Tsung-Kuang Yeh*[1]; Y. C. Lin[1]; C-H Tsai[1]; J. Chang[2]; F. Chu[2]; [1]National Tsing-Hua University; [2]Taipower

The technique of noble metal chemical addition (NMCA), accompanied by a low level hydrogen water chemistry (HWC), is being employed by a U.S. nuclear power plant for mitigating intergranular stress corrosion cracking the vessel internals of its boiling water reactor (BWR). Since the Chinshan Nuclear Power Plant will be operated under HWC by the year 2000, the NMCA technique is rarely the next candidate to adopt. Before NMCA is actually put into practice in the Chinshan BWR, a computer model named as DEMACE is employed to theoretically predict the effectiveness of NMCA throughout the primary heat transport circuit (PHTC) of this BWR. The effectiveness of NMCA is justified by the electrochemical corrosion potential (ECP) prediction around the entire PHTC. Before the modeling work started, the Mixed Potential Model, which is benchmarked based upon the recently published literature data for NMCA treated Type 304 stainless steels. It is found that the effectiveness of NMCA in the PHTC of the Chinshan BWR varies from region to region. In particular, NMCA may have negative impact in regions near the core outlet.

11:00 AM
Embrittlement of CF-8 Stainless Steel - A Non-Destructive Evaluation: *L. N. Liezan*[1]; [1]North Carolina State University, Raleigh, NC 27695-7909 USA

CF-8 stainless steel (CF8-SS) is used for several components such as primary coolant piping, and pump and valve bodies in light water reactor. These components are subject to thermal aging at the reactor operating temperatures (280 - 325°C). Cast CF-8 has a duplex microstructure consisting of austenite and about 10-20% delta-ferrite. Low temperature aging results in spinodal decomposition of the delta-ferrite leading to increased strength and decreased toughness. As part of an investigation on non-destructive assessment of aging induced embrittlement of reactor components, tensile and fracture propitiation of CF-8 SS were determined using automated ball indentation (ABI) technique after aging at 400° from 6, 12 and 18 months of duration.

Aging up to 12 months resulted in an increase in strength and a decrease in indentation energy to fracture. Prolonged aging showed a tendency towards saturation in tensile strength and toughness. Independence correlation with data from conventional tensile and impact tests are also included in this paper. This work is supported by the project funded by the INEEL University Research Consortium. The INEEL is managed by Lockheed Martin Idaho Technologies Company for the U.S. Department of Energy, Idaho Operation Office, under the Contract No. DB-AC-7 941D13223.

11:30 AM
Corrosion Inhibition Using Aerobic Bacteria in Service Water Systems: *James C. Earthman*[1]; Khaled M. Ismail[1]; Peggy J. Arps[2]; Kirsten C. Trandem[1]; [1]University of California, Irvine, Chem. and Biochem. Eng. and Mats. Sci., 916 Engineering Tower, Irvine, CA 92697 USA; [2]University of Nevada-Reno, Dept. of Chem. and Metall. Eng., Reno, NV 89557-0136 USA
Field and laboratory "side-stream" systems have been developed to study microbiologically influenced corrosion (MIC) under realistic service water conditions. The primary objectives of the present work are twofold: (1) to characterize MIC of different piping materials and (2) to investigate corrosion inhibition by genetically engineered biofilms under service water conditions. The present side stream systems are capable of combined testing approaches involving microbiology, electrochemistry, and surface chemistry which are being used to provide insight into complex interactions between biofilms and metal surfaces. For example, microbiological cultures, biochemical assays and genetic probes are being used to investigate the presence of specific species of bacteria. Electrochemical testing, including electrochemical impedance spectroscopy (EIS), electrochemical noise analysis (ENA) and DC techniques are also being performed to analyze MIC for four alloys in service water environments. Results for the first year of this ongoing research program will be discussed.

Low Alloy Steel: Session I - Embrittlement

Wednesday AM August 4, 1999 — Room: Pacific Ballroom E&F, Location: Marriott Hotel

Session Chairs: G. Lucas, University of California, Santa Barbara, CA, USA; E. Simonen, PNNL, USA

8:00 AM
Observations on Sensitivity of RPV Integrity Probabilistic Fracture Mechanics Evaluations to Input Parameters: *Allen L. Hiser*[1]; Simon C. F. Sheng[1]; Shah N. Malik[2]; [1]U.S. Nuclear Regulatory Commission, Mats. and Chem. Eng. Branch, Mailstop O-7D4, Washington, DC 20555 USA; [2]Electrical, Materials and Mechanical Engineering Branch
Probabilistic fracture mechanics (PFM) evaluations of reactor pressure vessel (RPV) integrity are used as a tool for understanding the vulnerability of the RPV to postulated transient conditions. In these evaluations, Monte Carlo simulations are used to sample critical material toughness and flaw characteristic parameters, with the specified loading history (pressure and temperature as a function of time) applied to the RPVs to determine the structural integrity of the vessel under the sampled condition. The parameters which are sampled to provide the material toughness include the initial properties [RTNDT(U)1, the chemical composition (copper and nickel contents) and the neutron fluence. The flaw characteristic parameters include the flaw size distribution and the flaw density. In particular, the flaw characteristic parameters have a key role in affecting the results of the PFM evaluation, but, in real life, the determination of them is associated with uncertainty. This study uses the results from recent PFM calculations for boiling water reactors (BWR). These evaluations assessed a variety of flaw size distribution and density parameters, including, in some cases, the effect of in-service inspection (IS I) on these parameters. These results are examined from the following perspectives: Effect of different flaw size distributions and densities; Impact of lSl; Differences between results for axial and circumferential welds; Effect of different loading histories (i.e., temperature and pressure time histories).

8:30 AM
Irradiation Behavior of Electricite De France PWR Vessel Steel: *C. Pichon*[1]; Y. Grandjean[2]; S. Saillet[2]; *G. Bezdikian*[2]; J. M. Frund[3]; [1]SEPTEN, Direction de I'Equipement, Electricite de France, 12-14 Avenue Dutrevoz, Villeurbanne Cedex 69628 France; [2]Electricite de France - Exploitation du Parc Nucleaire, Group des Laboratoires, Service d'Expertises, de Chinon, BP23, Avoine 37420 France; [3]Electricite de France - Direction des Etudes et Recherches, Departement EMA, Les Renardieres route de Sens, Ecuelles, Moret Sur Loing 77818 France
This paper draws up a synthesis of knowledge in France on irradiation embrittlement of Electricite De France (EDF) Pressurised Water Reactor vessel steel. At first, radiation effect surveillance programs on low alloy Mn_3Ni, Mo vessel steel including base metal, heat affected zone and weld are presented and results are discussed based on operation of more than 100 capsules removed from reactors at fluences up to 7 10^{19} n/cm^2(E> 1 MeV). The studies concerning in particular metallurgical and mechanic irradiation behavior fields are described in relation with the component life and Fast Rupture Margin files. This allows to identity the main parameters governing the aging phenomenon and to confirm the used approach in France concerning the component material embrittlement assessment.

9:00 AM
Effects of Copper, Phosphorus and Nickel on Radiation Damage in ASTM A 533-B Type Steel: *Milan Brumovsky*[1]; [1]Nuclear Research Institute Rez, Div. of Integ. and Mats., Vltavska 1, Rez 250 68 Czech Republic
Radiation damage is the most degrading mechanism in reactor pressure vessel steels. Steel of ASTM A 533-B type is mostly used for reactor pressure vessels of PWR and BWR type reactors. In last several years, the International Atomic Energy Agency Co-ordinated Program on irradiation embrittlement of reactor pressure vessels steels was performed and analyzed. This paper will show results from the study of the influence of the three most important elements – phosphorus, copper and nickel – on radiation embirittlement of this type of steel. All three elements were found as strongly affecting the resulting embrittlement, in single as well as in synergistic effect.

9:30 AM
Understanding the Role of Defect Production in Radiation Embrittlement of Reactor Pressure Vessels: *Dale E. Alexander*[1]; P. R. Jemian[2]; L. E. Rehn[1]; B. J. Kestel[1]; G. R. Odette[3]; G. E. Lucas[3]; D. Klingensmith[3]; D. Gragg[3]; [1]Argonne National Laboratory, Mats. Sci. Div., Bldg. 212, Rm. E-206, 9700 South Cass Ave., Argonne, IL 60439 USA; [2]University of Illinois at Urbana-Champaign, Materials Research Laboratory, UNICAT, Argonne, IL 60439 USA; [3]University of California-Santa Barbara, Dept. of Mech. and Environ. Eng., Santa Barbara, CA 93106 USA
Experiments were performed to evaluate mechanical property and microstructural changes induced by fast neutron and 10 MeV electron irradiation in a model pressure vessel alloy (Fe-0.9 wt. % Cu-1.0 wt. % Mn) and pure iron. Comparison of changes resulting from neutron irradiation, in which nascent defect clusters form in dense cascades, with electron irradiation, where nascent cluster formation is minimized, can provide insight into the role that the in-cascade clusters have on precipitation kinetics and embrittlement. Miniature tensile specimens and sheet coupons were irradiated to damage levels £4 mdpa under well-controlled conditions at 300°C. Large (> 60% increase over unirradiated samples) yield strengthening was observed for the alloy electron irradiated to low damage doses (<0.6 mpda) while minimal effect was observed in pure Fe. The alloy material was examined microstructurally using the synchrotron-based technique of anomalous small angle x-ray scattering (ASAXS) which permits element-specific characterization of scattering centers in a multicomponent alloy.

10:00 AM Break

10:30 AM
The Effect of Neutron Irradiation on Positron Lifetime and Micro-Vickers Hardness of Fe-Cu Model Alloys and Reactor Pressure Vessel Steel: Angela Hempel[1]; Masayuki Hasegawa[1]; *Gerhard Brauer*[2]; Masaaki Saneyasu[1]; Fernando Plazaola[3]; Sadae Yamaguchi[1]; [1]Tohoku University, Institute of Materials Research, Katahira 2-1-1, Sendai-chi, Aoba-Ku, Sendai 980-8577 Japan; [2]Research Center Rossendorf, Inc., Institute for Ion Beam Physics and Materials Research, P.O. Box 510119, Dresden 01314 Germany; [3]Elektrika & Elektronika Saila, UPV/EHU Zientzi Fakultatea, 644 P.K., Bilbo 48080 Spain
Cu impurities are known to be a dominant embrittlement feature in low-alloy reactor pressure vessel (RPV) steels. Clarification of effects on the formation of irradiation-induced defects of these impurities and thereby changes in mechanical properties of RPV steel is of high technological and scientific importance. This study was undertaken to investigate the effect of varying copper content and heat treatment on the annealing behavior of Fe-Cu model alloys (0-1.0 wt. % Cu) and A533B RPV steels (0.02-0.14 wt.% Cu). Positron lifetime and micro-Vickers hardness has been measured after neutron irradiation to fluences in the range $1x10^{18}$ ~ $1x10^{19}$ n/cm^2 at irradiation temperatures varying from 60 to 288°C. After neutron irradiation below 150°C longer positron lifetimes ranging from 230 to 420 ps have been observed in the Fe-Cu alloys as well as in the RPV steel, indicating microvoid formation. The longer lifetime and hence microvoid size are found to depend both on the Cu content, heat treatment before irradiation and irradiation temperature. To estimate the microvoid sizes, superimposed-atom model calculation of positron lifetimes of microvoids, consisting 1 (V_1) to 66 (V_{66}) vacancies have been performed. Using the results of these lifetime calculations, the formation and post-irradiation annealing behavior of microvoids is discussed. In addition, irradiation-induced hardening has been monitored by micro-Vickers hardness.

11:00 AM
An Evaluation of Temper Embrittlement in A508 Class 4 Steel: *D. B. Knorr*[1]; [1]Lockheed-Martin Corporation, P.O. Box 1072, Schenectady, NY 12301 USA
A508 Class 4 is a 3.3Ni-1.75Cr-0.5Mo bainitic steel, which demonstrates susceptibility to temper embrittlement when exposed in the temperature range 325-500°C. Both continuous slow cooling and isothermal exposure induce degradation that is measured by both a shift in the transition temperature and an increase in the extent of intergranular fracture features along prior austenite grain boundaries. The effect of processing history on the transition temperature has been demonstrated, particularly the effect of cooling rate as it impacts the development of temper embrittlement. Isothermal exposure in the temperature range 343-427°C increases transition temperature that follows a Öt relationship, except where the onset of saturation reduces the rate of temper embrittlement. The extent of intergranular crack propagation has been quantified according to processing/exposure history. The extent of temper embrittlement damage is related to phosphorus segregation, where higher bulk phosphorus levels are more damaging.

PWR Primary: Session IV - Crack Growth & Creep

Wednesday AM August 4, 1999
Room: Pacific Ballroom C
Location: Marriott Hotel

Session Chairs: R. Tapping, AECL, Canada; J. Hall, ABB Combustion, USA

8:00 AM
Stress Corrosion Crack Growth Rate Measurements on Ni Alloys in Primary and Caustic Environments: *Thierry Cassagne*[1]; D. Caron[1]; J. Daret[1]; A. Proust[2]; G. Turluer[3]; D. Boulanger[3]; [1]CEA/CEREM/LETC, Etablissement de La Hague, 50444 Beaumont, Hague, Cedex, France; [2]EPA; [3]IPSN-DES
This publication presents recent work on the investigation of the crack growth rate behavior of Alloy 600 and 182 weld materials used in the primary circuit of pressurized water reactors. Tests are conducted in a high pressure, high temperature loop where compact tension specimens can simultaneously be tested at three different temperatures under constant load. Crack growth rate measurements have been performed in primary water on three heats of Alloy 600 material at temperatures of 330, 310 and 290°C. Four 182 weld materials having different carbon and silicon compositions have also been tested at 330°C in the as-welded and stress relieved conditions. All materials have been characterized in terms of microstructure. Crack growth monitoring has been performed using DC potential drop and acoustic emission techniques. The results are discussed in terms of material microstructure and composition and compared with literature data. For Alloy 600, the influence of primary water temperature is also discussed.

8:30 AM
Stress Corrosion Life Assessment of Alloy 600 PWR Components: *C. Amzallag*[1]; S. Le Hong[1]; F. Vaillant[1]; C. Pages[1]; [1]Electricite de France, Exploitatin du Parc Nucleaire, Dept. Maintenance, 1 Placy Pleyel, Saint-Denis, Cedex 93207 France; [1]Electricite de France - Direction de l'Equipement, SEPTEN-Division Reaceur, 12-14, Avenue Dutrievoz, Villeurbanne, Cedex 69628 France
This paper presents the methodology used to assess the stress corrosion life of Alloy 600 PWR components. Crack initiation prediction is based on indexes on stress, temperature and material susceptibility. Validation of the indexes has been obtained from both field experience and laboratory data. The method has been used to rank the risk of crack initiation on the Alloy 600 materials used in the primary loop components in French PWR power plants. Crack propagation laws used for flaw evaluation in Alloy 600 components have been derived from the analysis of crack propagation rates measured in service on reactor vessel head penetrations and of crack propagation rates measured in laboratory on tests conducted on representative materials.

9:00 AM
Influence of Chromium Content and Microstructure on Creep and PWSCC Resistance of Nickel Base Alloys: *F. Vaillant*[1]; J. D. Mithieux[1]; O. De Bouvier[1]; D. Vancon[1]; Y. Brechet[2]; F. Louchet[2]; [1]EDF, R&D Div. Mats. Dept., Les Renardieres, Moret Sur Loing 77818 France; [2]ENSEEG, Institut National Polytechnique de Grenoble France
Alloy 600 (15% chromium) for steam generator tubings is known) to be susceptible to Stress Corrosion Cracking (SCC) in PWR primary water. It is now replaced by alloy 690 (30% Chromium) which has proved to be an excellent SCC-resistance in laboratory tests. The understanding of this chromium effect would definitively validate the choice of alloys with 30% chromium. The initiation of the cracking should require at least a competition between the destabilization of the protective film by creep and its recovery by a repassivation process. So the respective roles of the chromium content and of the microstructure on the repassivation kinetics and on the creep behavior have been checked in relation to SCC resistance. Tests were performed on tubes in alloys 600 and 690 with various types of microstructure. Parametric laws have been established for creep and for Grain Boundary Sliding (GBS) of alloys 600 and 690. The creep rate decreased when the chromium content or intergranular precipitation increased. But thermal treatment (700°C), which produced intergranular precipitation and improved SCC resistance of alloy 600, enhanced the creep rate by depleting soluble carbon in the matrix. On the opposite, GBS – which decreased with intergranular precipitation and increased with the creep rate – was correlated with SCC of alloy 600. On the other hand, the chromium content was slightly beneficial on repassivation kinetics in primary water. With regard to SCC, a beneficial effect of the chromium content was found on the initiation (no cracking on RUBs in alloy 690). But some significant cracking was observed on some tubes in alloy 690 with intragranular carbides during the early stage (below the stress intensity

threshold KSCC) of the propagation with Constant Extension Rate Tests (CERTs). Considering both alloys, Crack Growth Rates (CGRs) during CERTs depended mainly on the microstructure and on the creep rate but not basically on the chromium content. A correlation has been established between CGRs and GBS, indicating that GBS was at least a tracer of the required plasticity for cracking. The good resistance to initiation of SCC of alloy 690 could be probably explained by a low dissolution rate leading to a low generation of hydrogen then to a reduced hydrogen/plasticity interaction.

9:30 AM
A Simplified Correlation for SCC Susceptibility and An Evaluation Method for Effects of Cold Work Layer and Strength Characteristics on the Susceptibility: *Y. S. Garud*[1]; R. S. Pathania[2]; [1]APTECH Engineering Services, Inc., Sunnyvale, CA USA; [2]EPRI, Palo Alto, CA USA
The effect of cold work on SCC susceptibility has been well documented for many alloys. The related and often confounding effect of higher yield strength has also been reported by many, especially in describing the stress dependence of SCC. In this paper, we propose a quantitative model that accounts for these effects leading to a basis for indexing the SCC susceptibility from heat-to-heat variation and cold work characteristics. The model is based on the original concepts of strain-rate damage incorporated in a simplified model described at the 7th International Symposium on Environmental Degradation of Materials. This model is further developed to show some useful correlation between model parameters and material properties data. The proposed model and correlation are shown to be in good agreement with test data on Alloy 600 in high purity water environments; the data was obtained from various sources covering a wide range of mechanical properties, cold work levels, and material conditions. The data requirements are described for further verification and its application illustrated by the case of control rod drive mechanism (CRDM) nozzle cracking assessment. Also, the significance of depth of cold work layer-as opposed to the peak surface residual stress-is demonstrated on the basis of application of this simplified model by incorporating a crack growth rate law and the expected gradients in the acting stress and material properties across the component thickness.

10:00 AM Break

10:30 AM
Creep of Nickel-Base Alloys in High Temperature Water: *Gary S. Was*[1]; John Cookson[1]; Yongsun Yi[1]; John S. Fish[2]; S. A. Attanasio[2]; H. T. Krasodomski[2]; W. W. Wilkening[2]; [1]University of Michigan, Nuclear Eng. and Radiol. Sci., 1911 Cooley Bldg., 2355 Bonisteel Blvd., Ann Arbor, MI 48109-2104 USA; [2]Lockheed Martin Corporation, Schenectady, NY 12301 USA
Creep of nickel-base alloys in high temperature water is important in light water reactors because of its potential role in the SCC process. Creep may affect SCC either by direct degradation of the grain boundaries in the alloy or by causing rupture of the protective oxide film. For this reason, it is critical that the stress, temperature and composition dependences of creep are understood. In this work, the stress dependence of Alloy 600 in the mill-annealed (MA) condition was determined at 640°F and 680°F in deaerated primary water containing 40-60 cc/kg hydrogen. Experiments were performed to compare the creep behavior of commercial Alloy, Alloy 600 containing low carbon (LC) and the precipitation strengthened alloys 625 and X-750 in the AH and HTH conditions. The creep results were compared to the known SCC performance of these alloys. The creep rates of the LC alloy are higher on average than those with nominal carbon level. Intergranular cracking occurred only in the X-750 AH alloy and the precipitation treated LC A600 alloy. These results support earlier work that showed that low carbon alloys are more susceptible to creep and IG cracking than are high carbon alloys. However, these results also show a smaller influence of a water environment on the creep rate of commercial, creep-resistant alloys compared to high purity alloys.

11:00 AM
Static Load Crack Growth of Alloy 182 in Simulated PWR Environment: *W. Bamford*[1]; J. Foster[1]; R. Pathania[2]; [1]Westinghouse; [2]EPRI
Abstract Text Not Available

11:30 AM
Accelerated SCC Cracks Initiation Susceptibility Testing of Alloy 600 Reactor Vessel Head Penetration Materials: *Gutti V. Rao*[1]; [1]Westinghouse Electric Company, Pittsburgh, PA 15230 USA
This report summarizes the results of an industry supported program on the SCC susceptibility testing of reactor vessel head penatration materials. This program is sponsored by the Electric Power Research Institute (EPRI) and the utility owners groups, namely, the Westinghouse Owners Group (WOG), Combustion Engineering Owners Group (CBOG), and the Babcock & Wilcox Owners Group (B&WOG). The objective of this program is to establish the relative susceptibility to primary water stress corrosion crack (PWSCC) initiation of several commercial heats of Alloy 600 material representative of the vendor fabrication practices employed in the commercial production of the reactor vessel head penetration (RVHP). The materials have been obtained from various vendors and utility organizations to represent the RVHP materials currently in service. The initial test samples consisted of full ring sections machined from the RVHP tube materials ovalized by controlled loading to represent the residual stresses introduced due to weld thermal cycles during fabrication. The initial SCC initiation testing was conducted on ovalized ring samples fabricated from approximately half the archive materials procured using accelerated 400° steam plus hydrogen corrosion tests. After 800 hours of exposure during the initial testing, limited SCC initiation was observed in some of the test materials. However, the initiation sites were not located, where intuition and measurements indicated the highest strain levels. A re-examination of the specimen loading results prior to completing the exposures, revealed load relaxation in the specimens during the test. This was further confirmed by comparing with the results of an independent finite element elastic-plastic areas analysis of the ovilzation loading. Based on a refined assessment of the situation, modifications in the loading program.

Low Alloy Steel: Session II - EAC & Deformation

Wednesday PM   Room: Pacific Ballroom E&F
August 4, 1999   Location: Marriott Hotel

Session Chairs: H. Hanninen, U Helsinki, Finland; H. P. Seifert, PSI, Switzerland

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Stress Corrosion Cracking of Low Alloy Steels under BWR Conditions: Assessment of Possible Crack Growth Rates and Prediction of Component Behavior: *F. P. Ford*[1]; R. M. Horn[2]; J. Hickling[3]; G. Bruemmer[4]; R. Pathania[5]; [1]GE Research and Development Center, P.O. Box 8, Schenectady, NY 12301 USA; [2]GE Nuclear Energy; [3]Corrosion and Materials Consultancy; [4]Hamburgische Electricitaetswerke; [5]EPRI
The stress corrosion cracking (SCC) behavior of low alloy pressure vessel steels in high temperature water has been a subject of controversial discussion for many years, in part because the term stress corrosion can be understood either in a narrow sense (environmentally assisted cracking (EAC) under purely static, mechanical loading) or as part of the broader spectrum of EAC (i.e., including the transition to strain-induced corrosion cracking (SICC) and low-cycle corrosion fatigue (LCF)). Many of the older laboratory experiments on this topic are now seen to be inadequate, both for this and other reasons, and relatively few data have been generated in recent years. Even if crack initiation is ignored and only crack growth is considered, the available results with regard to measured SCC propagation rates unfortunately exhibit extreme scatter. This presents a problem when formulating a crack propagation rate vs. stress intensity disposition relationship that is valid for the specific loading, environmental and material conditions relevant to Boiling Water Reactor (BWR) operation. There are two synergistic approaches to resolving this problem. First, to evaluate the quality of all the existing

stress corrosion data and "filter" the database to include only that which is relevant to BWR operating conditions. Second, to derive a quantitative, mechanisms-based model of the cracking process which allows logical extrapolation of the prediction of crack behavior beyond that covered experimentally and which must be validated against the filtered database. Both of these approaches are being addressed in various ongoing studies. This paper covers the present status of work and compares the previously described GE theoretical crack propagation rate/stress intensity relationships with a preliminary filtered database. On the basis of both theory and observation, it is concluded that it is difficult to maintain active crack growth in currently used BWR pressure vessel steels in oxygenated 288°C water, provided the following criteria are met: Truly constant load; Plain-strain stress conditions at the crack tip; Bulk water conductivities £ 0.25 mS/cm. Under these specific conditions, the crack propagation rate will decrease rapidly with exposure time and reach steady-state rates which are of no practical significance. However, if these stress and environment conditions are violated (e.g., through transients in mechanical loading or water purity), the crack propagation rate may increase and attain a value that will be a function of the material yield strength and sulphide morphology, as well as the corrosion potential, water flow rate and temperature. The crack propagation rate/stress intensity algorithms for such "upset" conditions are not yet known, but are likely to be bounded by the previously derived, GE "Low Sulfur" disposition relationship.

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LCF Crack Initiation in WB36 in High Temperature Water: *H. D. Solomon*[1]; R. E. Delair[1]; E. Tolksdorf[3]; [1]GE R&D Center, P.O. Box 8, Room K1-3A51, Schenectady, NY 12301 USA; [3]VGB

WB36 is a German LAS. It differs from A533B primarily by its Cu precipitation hardening, which raises the strength without decreasing the ductility. This paper, the first of several which will describe the behavior of this alloy, describes the correlation of the observation of small cracks and the resultant load drop in tests run in strain control, with total strain limits. These tests were run with an extensometer which operated in oxygenated (0.2 or 8.0 PPM $)_2$ water at elevated temperatures (220°C or 232°C). The operation of this extensometer, and its use with either an attachment to the gage length or shoulder of the specimen will be described and contrasted. The alloy exhibits a small inverse strain rate dependence of the flow stress, (i.e., it gets stronger as the strain rate is reduced). It also cyclically strain hardens more at low strain rates than at higher ones. As with most LAS's the fatigue life in high temperature water is reduced as the strain rate is lowered. It differs, however, in that this effect is also observed in high temperature air tests, although the effect is not as pronounced as is observed in water tests at the same temperature. The results of all of the air tests and some water tests will be described and contrasted.

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Program of Stress Corrosion Cracking Sensitiviy of Irradiated Ferritic Steel: *Martin Ruscak*[1]; Jan Kysela[3]; Anna Brozova[1]; Oldrich Erben[2]; Marek Postler[1]; Gunter Brummer[2]; Harald Hoffmann[5]; Ulrich Ilg[6]; W. Ruhle[8]; [1]Nuclear Research Institute Rez Plc., Div. of Integ. and Mats., Vltavska 1, Rez 250 68 Czech Republic; [2]HEW AG, Uberseering 12, Hamburg 22286 Germany; [3]Nuclear Research Institute, Reac. Svcs. Div., Vltavska 1, Rez 250 68 Czech Republic; [5]VGB e.v., Essen Germany; [6]Badenwerk AG, Philipsburg, Germany

Stress corrosion cracking of low alloyed steel 22NiMoCr37 is evaluated with the goal to determine crack growth rate in irradiated steel under closely simulated conditions of BWR RPV under operation. Testing with both unirradiated and irradiated specimens gives a full experimental matrix that enables us to separate particular influences on the stress corrosion cracking in base metal and heat affected zone. For the experiment, in a pile of BWR an experimental loop has been built. Special attention was devoted to H2O2 determination in the vicinity of the samples. To achieve these results, three items were needed – technical installations, measurements and modeling of radiolysis. Special sampling lines for H2O2 measurements and ECP electrodes in pipes were used. Hydrogen peroxide was measured in a different node of the reactor and loop operation. Comparison of modeling and measurement enables us to calculate H2O2 along the loop and near testing specimens. 2T CT specimens have been chosen in order to achieve high level of loading. The irradiation rig has been built for pre-irradiation of specimens to the fluence of 1.5x1018 n/cm2 (E>1MeV). During the experiment, specimens are loaded by cyclic and constant load. Crack growth is monitored by means of potential drop measurement and COD. The first results obtained from the in pile and out of pile preliminary tests of unirradiated specimens will be presented in this paper together with the fractographic analysis of crack increments.

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Determination of SCC Threshold in Ferritic RPV Steel by Constant Load and Constant Strain Rate Methods: *Anna Brozova*[1]; Martin Ruscak[1]; Wolfgang Dietzel[2]; [1]Nuclear Research Institute Rez Plc., Div. of Integ./Corr. and Microst. D, Rez, Czech Republic 250 68; [2]GKSS-Forschungszentrum Geesthacht GmbH, Max-Planck Strasse, Geesthacht, Schleswig-Holstein D-21502 Germany

This paper deals with stress corrosion cracking (SCC) of Russian type reactor pressure vessel steel 15Ch2NMFA in high oxygen hot pressurized water. The steel serves as model material for the study of SCC process at high temperature water. Two methods of testing for stress corrosion cracking were chosen, the standard constant loading method (CL) and the new one – rising displacement test (RDT) now under evaluation procedure for acceptance as a standard ISO 7539-9. Experiments were carried out in a refreshed autoclave loop system at 320°C. Pre-fatigue CT specimens were used and the crack length was measured during tests by the potential drop method. Fractography of all specimens were carried out after tests. Testing results showed a good agreement of both methods in threshold values of KISCC, but some differences were observed in the crack growth rate measurements.

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Evaluation of Crack Tip Solution Chemistry of Low Alloy Steels in Oxygenated High Temperature Water: *Yunju Lee*[1]; Tetsu Shoji[1]; Raja Krishnan Selva[1]; [1]Tohoku University, Research Institute of Fracture Technology, Aobayama 01, Aobaku, Sendai 980-28579 Japan

The corrosion fatigue crack growth behavior of several low alloy steels with different sulfur content and MnS morphology has been evaluated in oxygenated high temperature water. The crack tip solution has been sampled out by micro-sampling technique during corrosion fatigue tests. Aridity was analyzed by the capillary electrophoresis method, used to determine the chemistry prevailing at the crack tip. Sulfate ions were detected in the crack tip solution and the amount of sulfur ion is affected by the bulk sulfur content of the material arid by the crack growth rate. In this study, a new terminology, normalized sulfur ion content, has been introduced to account for the difference in micro-sampling events. The normalized sulfur ion content quantifies the sulfur ions produced per unit increment of the crack length. A linear relation was observed between normalized sulfur ion content arid the crack increment for both medium sulfur (0.013%) and low sulfur (0.003%) containing steels. However, the relation between the crack growth rate and the normalized sulfur ion content was not found to be linear in case of low sulfur content material and the behavior was different from that of medium sulfur content. The sulfur ion content at the crack tip was observed to vary with the loading frequency and the MnS morphology. This paper discusses the reasons for the difference in crack growth rates of the materials having different sulfur content and different morphologies of the inclusions by using the quantified data obtained through the micro-sampling method.

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The Stress Corrosion Cracking of Reactor Pressure Vessel Steel under Boiling Water Reactor Conditions: *J. Heldt*[1]; H. P. Seifert[1]; [1]Paul Scherrer Institute, Laboratory for Safety and Accident Research, OVGA/120, Villigen, Kanton Aargau CH-5232 Switzerland

Reliable quantitative crack growth data is the basis of structural integrity assessments of LWR pressure boundary components. With qualified tests performed during the last years we generated a large amount of data (> 400 specimen tested) concerning the SCC of low alloy RPV steel under simulated BWR conditions. All tests were performed with precracked specimen. The cracking susceptibility was determined by constant load/constant displacement tests. Additionally, crack initiation and growth was studied by a systematic variation of initial loading rate and subsequent holding period of constant load. Stable sustained SCC only occurred if small scale yielding conditions were clearly exceeded. For this case environmental margins for cracking susceptibility can be given. Crack growth was triggered by initial slow loading. Under subsequent static loading the crack advance significantly slowed down. Even fast crack growth (>10-8 m/s) found in high sulphur steel could not be maintained during subsequent constant load period. The obtained results can be rationalized by the time dependence of the crack tip strain rate due to low temperature creep.

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European Round Robin Test on Environmentally Assisted Cracking of Low Alloy Ferritic Steel under BWR Conditions: A. Wunsche[1]; D. Blind[1]; F. Huttner[1]; *K. Kuster*[2]; P. Karjalainen-Roikonen[3]; U. Ehrnsten[3]; A. Roth[4]; H. P. Seifert[5]; J. Heldt[5]; [1]MPA; [2]HEW; [3]VTT; [4]Siemens KWU; [5]PSI Switzerland

Internationally, a large scatter in data exists regarding environmentally assisted cracking of low alloy ferritic steel under BWR conditions. This causes a divergence in opinions about the cracking behavior and, thus, uncertainty in life time predictions. Constant load tests were performed in 4 European laboratories under nominal identical experimental conditions to obtain comparable results. The tests were performed using 25 mm C(T) specimens of 20 MnMoNi 5 5 (sim. A 508 CI.3) at 288°C in high purity water containing 0.4 ppm oxygen. During the tests, the crack growth was monitored over 1000 h at stress intensity factor levels of 25-75 MpaÖm. The crack growth increment was measured along the crack front after the tests in connection to detailed fractographic analysis. The results of the different laboratories show a good agreement. Fractographic investigations indicated local features resembling EAC crack growth, especially in the vicinity of MnS-inclusions. The average crack growth rates are below 1.10^{-8} mm/s in the stress intensity factor range of 25-55 MpaÖm and below 2.10^{-8} mm/s in the range of 66-75 MpaÖm.

PWR Secondary: Session I - System Definition

Wednesday PM — Room: Pacific Ballroom C
August 4, 1999 — Location: Marriott Hotel

Session Chairs: A. McIlree, EPRI, Palo Alto, CA USA; R. Staehle, University of Minnesota, USA

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Experimental Simulation of Boiling Crevice Chemistry: *Chi Bum Bahn*[1]; Il Soon Hwang[1]; In Hyoung Rhee[2]; Uh Chul Kim[3]; Jung Won Na[3]; [1]Seoul National University, Dept. of Nuclear Eng., 56-1, Shinlim-dong, Gwanak-gu, Seoul 151-742 Korea; [2]Sun Chun Hyang University, Dept. of Chem. Eng., 53-1, Umnae-ri, Sinchang-myun, Asan, Chungnam Korea; [3]Korea Atomic Energy Research Institute, 150, Dukjin-dong, Yusong-gu, Taejon 305-353 Korea

In a locally restricted steam generator geometry, impurities in the bulk water can be concentrated by a boiling process to an extreme pH that may then accelerate the corrosion of tubing and adjacent materials. To mitigate the corrosion, Molar Ratio Control technique is widely implemented with the EPRI initiative. In order to maximize its beneficial effect, the understanding of crevice processes must be advanced. For direct observation of the processes, low temperature simulation experiments have been conducted for both tubular and planar crevices with transparent windows. A simulation apparatus is also equipped with microelectrodes for the measurement of electrochemical potential, pH and conductivity and the provision for local solution sampling. Evidences of competition between boiling concentration and flushing effect are observed. In qualitative agreement with a simple physico-chemical model, the amount of wall superheat and the gap size are found to be important factors.

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Heated Crevice Investigation of Effects of Hydrazine and Sodium/Chloride Ratio: *J. Lumsden*[1]; G. Pollock[1]; C. Fauchon[2]; P. Millet[2]; H. Takamatsu[3]; [1]Rockwell Science Center; [2]EPRI; [3]Kansai Electric

Abstract Text Not Available

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Secondary Side Corrosion of French PWR Steam Generator Tubing: Contribution of Surface Analyses to the Understanding of the Degradation Process: *J. M. Boursier*[1]; M. Dupin[2]; P. Gosset[1]; Y. Rouillon[1]; [1]Electricite de France, Generating and Transmission Div., Chinon Hot Laboratory, Avoine, BP 23 37420 France

The secondary side corrosion (OD corrosion) of mill annealed alloy 600 steam generator tubing occurred in France later than the Primary Water Stress Corrosion Cracking (PWSCC). The first tubes to be pulled out for secondary side corrosion have been removed from Fessenheim Nuclear Power Plant unit 1 in 1986. Before that, 97 tubes had already been pulled out only due to a PWSCC concern. But over the last few years, the number of tubes plugged due to O.D. Corrosion has rapidly increased, and in 1995 EIectricite de France decided to replace for the first time a steam generator due to secondary side degradation at Saint-Laurent B unit 1. For several years, Electricite de France has undertaken numerous studies relating to the description and the understanding of the secondary side steam generator degradation. But up to now, the chemical media, root causes of O.D. Corrosion, are not very well known. One attempt among others to better know these media is the characterization of deposits in the crevice areas, and the determination of tile nature and chemical structure of oxide layers on the pulled tube surfaces that can provide significant information necessary to understand the secondary side degradation mechanism. This paper focuses on the French feedback from pulled tubes in terms of cracking location, degradation parameters, tube deposits (more than 250 analyses), characterization of the scale in the crevice areas of some Dampierre unit 1 Tube/TSP intersections which are located at various elevations in the steam generator. In addition, some Glow Discharge Optical Emission Spectroscopy analyses have been performed on steam generator tubes withdrawn from various nuclear power plants in order to try to correlate the oxide layer composition versus the pH value of the medium. Finally, some comparisons between Nuclear Power plant feedback and fundamental R&D studies are discussed, particularly in the scope of trying to understand what could be the secondary side degradation medium.

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Evidence for the Reduction of Sulfates under Representative SG Secondary Side Conditions, and for the Role of Reduced Sulfates on Alloy 600 Tubing Degradation: *Jacques Daret*[1]; Th. Cassagne[1]; Y. Lefevre[1]; T. Tran[1]; R. Benoit[2]; R. Erre[2]; [1]CEA CEREM/LETC, Establissement de La Hague, 50444 Beaumont Hague, Cedex France; [2]CNRS Orleans

A large program is being conducted in model boilers to elucidate the cause(s) of secondary side degradation of alloy 600 steam generator tubes in creviced areas. Attention is presently focused on the reduction of sulfates under representative SG conditions, and the effect of reduced species on the tubing degradation. Concerning reduction, the major findings are that the percentage of reduced sulfates does not depend on the concentration of hydrazine in the make-up water, and that the reduced species preferentially concentrate in the steam phase. This last finding sustains the possibility of a corrosion mechanism in a polluted steam environment that can be present in a steam-blanketed area. In the neutral pH range, tube surface analysis using EDS, AES and XPS showed that tubing degradation is associated with the existence of a poorly adherent, chromium rich oxide film, but also necessitate the presence of reduced sulfur species.

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Study of Deposits and Corrosion Products in the Secondary Side of Steam Generators by Fourier Transform Infra Red Spectroscopy: *S. Chevalier*[1]; M. Organista[1]; B. Sala[1]; R. Erre[2]; A. Gelpi[3]; [1]Framatome, Technical Center, Tour Fiat, 1 Place de la Couple, La

Creusot 92084 France; [2]CNRS/CRMD, Orleans France; [3]Framatome, Paris La Defense France
Laboratory studies were performed to simulate the surface reaction leading to the formation of aluninosilicate rich deposit and non protective oxide layers observed on corroded area of MA Alloy 600 tubes pulled from plants. Surface analyses associated with electrochemical studies by impedance showed the different interactions between both with silica and phosphates and particularly the role of organic compounds in the corrosion process in the secondary side of PWR steam generators. A new surface analysis approach was initiated by FTIR (Fourier transform Infra-Red Spectros-copy) associated with XPS/AES and SIMS. We detected organic functions on surfaces of pulled tubes and on laboratory test coupons. Organo-metallic complexes as well as the stages resulting in the formation of complexes were identified. In particular, the decomposition of organic compounds at high temperature can induce the formation of amine or amide compounds and, under certain conditions, nitril functions. All these functions are able to combine with metallic elements from the matrix.

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Electrochemical Study of the Corrosion Process of the Secondary Side of the Steam Generators: *B. Sala*[1]; S. Chevalier[1]; A. Gelpi[1]; H. Takenouti[2]; M. Keddam[2]; [1]Technical Center of Framatome Porge Magenta, BP 181- 71205 Le Creusot, Cedex France; [2]LP15 du CNRS, Physique des Liquides et Electrochimie, Tour 22, 4 Place Jussieu 75252, Paris, Cedex 02 France
Previous work on SG tubes pulled from French plants and laboratory simulations had shown that corrosion in the secondary side of the steam generators occurs under alumino-silicate or aluminate deposits: the analyses of surface reactions showed that iron and nickel migrate from inside the alumino-silicate deposit while chromium is located as hydroxide in a non protective layer below the deposits. In the presence of carboxylic compounds, chromium can also migrate inside this alumino-silicate layer. After the study of the surface degradation, the objective of this work is to study the intergranular corrosion process. Laboratory experiments with impedance measurements (EIS) confirmed the fact that two media, silica with organic species and phosphate with organic species are able to produce IGA especially under polarization. EIS technique can detect the IGA formation in situ, in aqueous medium at high temperature due to change in diffusion response. Corrosion at the grain boundaries is also studied by analyzing the corrosion products and the matrix with different techniques.

Radiation Effects: Session I - Radiation Effects on Stress Corrosion Cracking

Wednesday PM Room: Pacific Ballroom D
August 4, 1999 Location: Marriott Hotel

Session Chairs: S. Bruemmer, PNNL, USA; T. Shoji, Tohoku University, Japan

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Effect of Pre-irradiation Grain Boundary Chemistry on IASCC: *Mitsuhiro Kodama*[1]; Yoshihide Ishiyama[1]; Shunichi Suzuki[2]; Satoshi Namatame[2]; Koji Fukuya[3]; Hiroshi Sakamoto[3]; Kiyotomo Nakata[4]; Takahiko Kato[4]; [1]Nippon Nuclear Fuel Development Co., Ltd., Research, 2163, Narita-cho, Oarai-machi, Higashi-Ibaraki-gun, Ibaraki-ken 311-1313 Japan; [2]The Tokyo Electric Power Co., Ltd., Egasaki-cho, Tsurumi-ku, Yokohama, Kanagawa-ken 230-8510 Japan; [3]Toshiba Corporation, Shinsugita, Isogo-ku, Yokohama, Kanagawa-ken 235-8523 Japan; [4]Hitachi, Ltd., Saiwai-cho, Hitachi-shi, Ibaraki-ken 317-8511 Japan
Grain boundary analyses of pre-irradiated and irradiated stainless steels (Type 316 series and Type 304) and slow strain rate tests of irradiated stainless steels were conducted to investigate grain boundary chemistry and IASCC susceptibility. The effect of segregation before irradiation on radiation induced segregation (RIS), and the relationship between RIS and IASCC susceptibility were studied. The significant enrichment of Cr and Mo was detected at the grain boundary of several pre-irradiated stainless steels out of all tested ones. The difference of pre-irradiation grain boundary chemistry between stainless steels related to mill-annealing condition. The pre-irradiation enrichment of Cr and Mo retarded the radiation induced depletion of Cr and Mo. The stainless steels of higher Cr and Mo concentration at grain boundaries showed lower IASCC susceptibility. This result suggested that the higher enrichment of Cr and Mo before irradiation delayed IASCC to higher neutron fluence.

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Irradiation-Assisted Stress Corrosion Cracking of Model Austenitic Stainless Steels: *H. M. Chung*[1]; W. E. Ruther[1]; R. V. Strain[1]; W. J. Shack[1]; T. M. Karlsen[2]; [1]Argonne National Laboratory, 9700 S Cass Ave., Argonne, IL 60439 USA; [2]OECD Halden Reactor Project, Halden, Norway
Hot-cell stress corrosion testing and microstructural characterization were performed to investigate irradiation-assisted stress corrosion cracking (IASCC) of model austenitic stainless steels (SSs) and provide a better understanding of irradiation-induced degradation of boiling water reactor (BWR) core internal components. Irradiation test matrix consisting of 27 model austenitic stainless steel alloys was constructed according to Taguchi's standard orthogonal array, which is an optimized matrix designed to determine systematically the effects of compositional variables (i.e., bulk material concentrations of Cr, Ni, Si, P, 5, Mn, C, and N) at three concentration levels. In addition, limited number of alloys were included to determine the effects of Mo, Nb, O, B, and commercial vs. laboratory fabrication of the steels. After irradiation at 288°C in helium in the Halden boiling heavy-water reactor to AO.3 x 1021 nxcmD2 (E> 1 MeV) and AO.9 x 1021 nxcmD2, slow-strain-rate tensile (SSRT) tests were conducted in simulated BWR water that contains A8 ppm dissolved oxygen. Fractographic analysis by scanning electron microscopy was also conducted to determine susceptibility to IASCC manifested by the degree of transgranular and intergranular fracture surface morphology. Heat-to-heat variations in susceptibilities to intergranular and transgranular stress corrosion cracking (IGSCC and TGSCC) were very significant. A high-purity heat of Type 316L SS exhibited the highest susceptibility to IASCC (manifested by highest percent IGSCC) at AO.3 x 1021 nxcmD2 (E> 1 MeV). A Type 304 SS heat that contained an unusually high level of oxygen exhibited poor work-hardening capability, low ductility, and significant susceptibility to TGSCC even in the nonirradiated state, and after irradiation to AO.9 x 1021 nxcmD2 (E> 1 MeV), the heat exhibited relatively higher susceptibility to IGSCC than other heats. At AO.3 x 1021 nxcmD2 (E> 1 MeV), total elongation and susceptibility to transgranular and intergranular cracking (i.e., percent TGSCC plus percent IGSCC) of the alloys could be correlated surprisingly well with nitrogen and silicon concentrations of the alloys. Irrespective of the carbon and other impurity contents, all alloys that contained low levels of nitrogen (<50 wppm) and silicon (<1.0 wt.%) exhibited low ductility and high susceptibility to TGSCC, and in some heats, high susceptibility to IGSCC. All alloys that contained either high silicon (>1.0 wt.%) or nitrogen >100 wppm exhibited high ductility, low percent TGSCC, and negligible percent IGSCC. This observation indicates that alloys that contain >100 wppm nitrogen and >1.0 wt.% silicon are effective in suppressing the onset of and susceptibility to IASCC. Because most commercial heats of austenitic stainless steels contain >100 wppm nitrogen, this implies that silicon concentration higher than the ASTM-specfled maximum of 1 wt.% is significanfly beneficial from the standpoint of IASCC. Test results also indicate that when fluence increased from AO.3 x 1021 n cm-2 (E> 1 MeV) to AO.9 x 1021 n cm-2 in the low-nitrogen and low-silicon alloys, susceptibility to TGSCC decreased while susceptibility to IGSCC increased at the expense of TGSCC. Results of systematic correlation of other compositional variations with susceptibilities to TGSCC and IGSCC will be presented for 16 alloys irradiated to AO.3 x 1021 nxcmD2 (E> 1 MeV) and 24 alloys irradiated to AO.9 x 1021 nxcmD2.

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Intergranular Cracking of an Irradiated Ti-stabilized Stainless Steel Spacer Grid Sleeve from a VVER-440 Reactor: *Ulla M. Ehrnstén*[1]; Pertti Aaltonen[1]; Pertti Nenonen[1]; Risto Teräsvirta[2]; Ossi Hietanen[2]; [1]VTT Manufacturing Technology, P.O. Box 1704, Espoo, VTT 02044 Finland; [2]IVO, Power Engineering Ltd, Vantaa, IVO 01019 Finland

A failure of an irradiated spacer grid sleeve of a fuel assembly, observed after three years in a core of a VVER-440 reactor, is reported. The spacer grid sleeve, holding the spacer grid on the central tube, is made of solution annealed Ti-stabilized austenitic stainless steel and the estimated dose after three years in reator is 1.9 E22 n/cm2. The intergranular cracking had occurred without any observed plastic deformation of the sleeve. The hardness of the irradiated material was high, 350 HV1, and the grain size small, corresponding to ASTM 9. The cause of the failure is estimated to be irradiation assisted stress corrosion cracking. In addition to the failure analysis results, results of FEGSTEM investigations on irradiated as well as non-irradiated sleeve material will be reported in the paper.

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Water Chemistry and Stress Intensity Effects on the Cracking Behavior of Irradiated Austenitic Stainless Steels: *Elisabeth Hauso*[1]; Torrill Karlsen[1]; [1]OECD Halden Reactor Project, P.O. Box 173, N-1751, Halden, Norway

An Irradiation Assisted Stress Corrosion Cracking (TASCO) program was initiated by the Halden Project in 1991, with the aim of addressing the IASCC susceptibility of core component materials such as austenitic stress steels, and to achieve a better understanding of factors affecting the phenomenon. Successful qualification of instrumentation and loading techniques required for in-pile monitoring of cracking behavior during exposure to representative LWR (typically BWR) environments, and the use of 1/4T Compact Tension (CT) specimens prepared from irradiated stainless steels, have been performed. Results from on-going experiments, which clearly demonstrate the benefits of lASCC mitigation measures such as Hydrogen Water Chemistry (HWC), in addition to crack growth rates recorded for 304, 316 and 347 stainless steel as a function of water chemistry, stress intensity (15 - 20 $MPam^{1/2}$) and varying levels of fluence (1 - 9 X $l0^{21}$ n/cm_2), are presented. Crack initiation studies designed to assess the stress-neutron fluence boundary for crack initiation in pressurised stainless steel tube specimens exposed to BWR environments are described.

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Neutron Irradiation-Tnduced Changes in Percent IGSCC of Thermally-Sensitized Type 304 Stainless Steels: *Takeo Onchi*[1]; Koichiro Hide[1]; Masami Mayuzumi[1]; Taiji Hoshiya[2]; [1]CRIEPI, Komae Research Laboratory, 2-11-1, Iwado Kita, Komae-shi, Tokyo 201-8511 Japan; [2]JAERI, Narita-cho, Oarai-Machi, Ibaraki-ken 311-13 Japan

Percent intergranular stress corrosion cracking (%IGSCC) of neutron-irradiated thermally-sensitized type 304 stainless steels was examined by slow strain rate tensile (SSRT) testing in 290℃ water of 0.2 ppm dissolved oxygen concentration. Tensile properties were also determined by the SSRT in inert gas. Neutron fluences ranged between 5.4E+19 and 3.0E+21 n/cm2 (E>1MeV). The %IGSCC increased with increasing neutron fluences of up to 1.0E+20 n/cm2 and decreased at 3.0E+21 n/cm2 anomalously down to values far smaller than those for the unirradiated thermally-sensitized materials. The ductility also gradually reduced with fluences of up to 1.0E+20 n/cm2 and further decreased at 3.0E+21 n/cm2, although tensile strength increased steadily increased in the range of neutron fluences examined. The anomaly in the behavior of %IGSCC for the highly irradiated material would not be due to a large reduction in IGSCC susceptibility but attributable to the significantly large loss of ductility. It is deducible that the %IGSCC as an index of the IGSCC susceptibility is less appropriate, although commonly used as well, at higher neutron fluences.

4:30 PM

Stress Corrosion Cracking of Type 304L Stainless Steel Core Shroud Welds: *H. M. Chung*[1]; J. H. Park[1]; J. E. Sanecki[1]; N. J. Zaluzec[1]; T. T. Yang[2]; M. S. Yu[2]; [1]Argonne National Laboratory, Argonne, IL 60439 USA; [2]Institute for Nuclear Energy Research, Lungtan, Tiawan ROC

Failure of welded core internal components fabricated from Type 304 and 304L austenitic stainless steels (SSs) has increased significantly in recent years; especially, cracking of core shroud welds has been prevalent in boiling water reactors (BWRs). Although BWR core shrouds are subject to relatively low neutron fluence, many vertical and horizontal welds crack by the time they accumulate relatively low fluences of A2 x 1019 n cm-2 to A5 x 1020 n cm-2. At these low fluences nonwelded base-metal components fabricated from Type 304 and 304L SSs would not be considered susceptible to irradiation-assisted stress corrosion cracking, which typically occurs after a threshold fluence of A5 x 1020 n cm-2. Partly because of this, most cases of core shroud cracking have been attributed to classical intergranular stress corrosion cracking (IGSCC) of thermally sensitized SS, in which significant thermal sensitization by grain-boundary carbide precipitation has been assumed to occur in the heat-affected zone (HAZ) of a core shroud during the welding. However, an increasing number of cracking incidents observed in core shrouds fabricated from Type 304L SS is difficult to explain on the basis of classical IGSCC, because thermal sensitization by grain-boundary carbide precipitation would not be expected to occur in low-carbon SSs. Most large core internal components, such as BWR core shrouds are welded by shielded-metal-arc (SMA) or submerged-arc (SA) welds, in which Type 308 SS electrodes coated with fluoride-containing flux material are used to join arc-melted thick plates in air. To better understand the cracking of low-carbon Type 304L SS core shroud welds, microstructural characteristics of a Type 304L BWR core shroud and simulated SMA and gas-tungsten-arc (GTA) welds of Type 304 and 304L SSs were investigated by chemical analysis, scanning electron microscopy, Auger electron spectroscopy, secondary-ion mass spectroscopy, and field-emission-gun advanced analytical electron microscopy. At the same time, stress corrosion behavior of the simulated welds were also investigated by slow-strain-rate tensile and bend-beam testing in simulated BWR water. The heat-affected zone of the unirradiated SMA welds were significantly contaminated by 0 and F, whereas contamination in the fluxless GTA welds was negligible. This finding shows that the contamination is a result of exposure to air and the F-containing weld fume. Oxygen contamination promoted F contamination, and F was higher in the 0-rich region beneath the weld scale, on grain boundaries, and in oxides. Similar contamination was also observed in the field-cracked Type 304L SS core shroud weld. Grain boundaries of the heat-affected zone of the Type 304L SS shroud weld were characterized by fine oxide particles and high F, but grain-boundary carbides, Cr depletion, or martensite film were absent. Several previous works have demonstrated strong influences of weld fumes and weld scale on intergranular stress corrosion. These works and the present observations suggest that oxygen and fluorine contamination and oxyfluorine-assisted stress corrosion play a major role in the cracking of flux-welded Type 304L SS core shroud welds in the absence of thermal sensitization. A cracking model based on oxyfluorine-assisted stress corrosion is proposed.

5:00 PM

Development of Comprehensive Material Performance Database (JMPD) and Analyses of Irradiation Assisted Stress Corrosion Cracking Data: *Yoshiyuki Kaji*[1]; Takashi Tsukada[1]; Hirokazu Tsuji[1]; Hajime Nakajima[1]; [1]Japan Atomic Energy Research Institute, Dept. of Nuclear Energy Sys., Shirakata-Shirane 2-4, Tokai-mura, Naka-gun, Ibaraki 319-1195 Japan

A comprehensive material performance database for nuclear applications, which was named JAERI Material Performance Database (JMPD), has been developed since 1986 at JAERI with a view to utilizing various kinds of characteristic data of nuclear materials efficiently. The data stored in the JMPD are mainly fatigue crack growth data on low alloy steels, creep data on superalloys, tensile data on aluminum alloys and stress corrosion cracking data (Slow Strain Rate Testing (SSRT), crack growth rate, etc.) on austenitic stainless steels. Irradiation Assisted Stress Corrosion Cracking (IASCC) of austenitic stainless steels in high temperature water has been considered as a degradation phenomenon potential not only in the light water reactors (LWRs) but rather common in systems where the materials are exposed simultaneously to radiation and water environments. This paper will describe the present status of the JMPD, which is available through the internet partially. Furthermore, some trials of sophisticated utilization of the system focused on the issues relating to IASCC will be mentioned.

5:30 PM

Program of SCC Sensitivity of Irradiated Ferritic Steel: *Martin Ruscak*[1]; Miroslav Zamboch[1]; Oldrich Erben[2]; [1]Nuclear Research Institute Rez, Div. of Integ. and Mats., Vltavska 1, Rez 250 68 Czech Republic; [2]Nuclear Research Institute, Rez, Div. of Nucl. Svcs., Vltavska 1, Rez 250 68 Czech Republic

The Program of RPV internals assessment of WWER NPPs has been started to develop model of changes of mechanical and corrosion mechanical behaviors under operation conditions for stainless steel 08Kh18N10T, weldments and high strength steels used in these structures. The goal is to initiate the lifetime assessment of internals. The paper will describe the calculations of fluences, stresses and deformations, pre-irradiation of specimens to the fluences up to 20 dpa and program of experimental measurements of changes of material properties due to irradiation. The pre-irradiation of specimens was performed at the BOR-60 fast reactor in Dimitrovgrad, Russia. A mock up experiment was performed in order to determine the neutron field and condition of irradiation. During the irradiation the fluences were monitored with sets of activation neutron fluence monitors. The results from tensile tests in argon and SSRTs in WWER environment will be shown for materials irradiated between 0 and 10 dpa. The program of inspections based on the instrumented hardness measurement will be presented.

PWR Secondary: Session II - Cracking Response

Thursday AM — Room: Pacific Ballroom C
August 5, 1999 — Location: Marriott Hotel

Session Chairs: P. Millet, EPRI, Palo Alto, CA USA; F. DeKeroulas, EDF, France

8:00 AM

Roll of Grain Boundary Characteristics in Caustic IGA/SCC Resistance of Thermally Treated Alloy 690 and Shot Peened Alloy 800: *H. Kawamura*[1]; H. Hirano[1]; S. Shirai[2]; H. Takamatsu[3]; M. Matsunaga[3]; K. Yamaoka[4]; K. Oshinden[5]; H. Takiguchi[6]; [1]Komae Research Laboratory, Central Research Institute of Electric Power Industry, Iwatokita, Komae-shi, Tokyo 201 Japan; [2]The Hokkaido Electric Power, Co., Inc., Facilities Management Sec., Nuclear Power Dept., Higashi 1-2, Ohdori, Chuo-ku, Sapporo 060-91 Japan; [3]The Kansai Electric Power Co., Inc., General Office of Nuclear and Fossil Poer Production, 3-3-22, Nakanoshima, Kita-ku, Osaka 530-70 Japan; [4]The Shikoku Electric Power Co., Inc., Nuclear Power Operating Management Sec., Nuclear Power Dept., 2-5, Marunouchi, Takamatsu, Kagawa 760-91 Japan; [5]The Kyushu Electric Power Co., Inc., Nuclear Operation Dept., 2-1-82, Watanabe-Dori, Chuo-ku, Fukuoka 810-91 Japan; [6]The Japan Atomic Power Co., Inc., Plant Management Headquarters, Plant Eng. Dept., 1-6-1, Otemachi, Chiyoda-ku, Tokyo 100 Japan

Intergranular Attack and Stress Corrosion Cracking (IGA/SCC) initiation and crack growth behaviors of thermally treated alloy 690 (690TT) and shot peened alloy 800 (800SP) were examined in deaerated $NaOH+N_{a2}CO_3$ solutions at 320℃ to 350℃, using C-ring and Double Cantilever Beam (DCB) tests, comparing with those of alloy 600. The grain boundary characteristics were also examined using Auger Electron Spectroscope (AES) and Transmission Electron Microscope equipped with Field Emission Gun (FEG-TEM). The C-ring test results showed that the ranking for caustic IGA/SCC resistance was alloy 600 < alloy 800SP < alloy 690TT. The DCB test results showed that the crack growth rate of alloy 800SP was lowest. AES and FEG-TEM studies showed that boron and phosphor were segregated to the grain boundaries of alloy 690TT and alloy 800SP, and that chromium carbide was precipitated semicontinuously at the grain boundary of alloy 690TT, whereas intragranular precipitates was dominantly in alloy 800SP. When the alloys were subjected to plastic deformation, slips of dislocation in alloy 690TT were canceling each other by intricate dislocations, whereas the slips of alloy 800SP were increased. Based on the above test results, it was judged that the chromium carbide precipitation and the slipping behavior of dislocation could be a contribution factor of caustic IGA/SCC of alloy 690TT and 800SP.

8:30 AM

Causes and Mitigation of OD Stress Corrosion Cracking: *Allen Baum*[1]; P. J. Prabhu[2]; Peter Kuchirka[2]; [1]Westinghouse Electric Company, 1006 Macon Ave., Pittsburgh, PA 15218 USA; [2]Westinghouse Electric Corporation, P.O. Box 158, Madison, PA 15168 USA

This paper relates model boiler and isothermal laboratory test data to steam generator operating experience in order to formulate a degradation scenario which is believed to be most compatible with the observed plant corrosion. Comparison criteria include the local variability in the extent of corrosion and the magnitude of the bobbin probe eddy current voltages. It is concluded that initial SCC propagation is most likely caused by at least moderately alkaline concentrated solutions. More neutral solutions, acidic solutions, and superheated steam are discounted either because they are incapable of producing the high degree of variability characteristic of the plant data or eddy current voltages are inconsistent with plant experience. Examinations of pulled tubes from a variety of plants found no correlation between silica salt concentrations and the extent of corrosion. These salts are sometimes mentioned as a cause of SCC. The comparison also suggests that longer term crack propagation kinetics may be governed more by a superheated steam environment than by a concentrated liquid solution. This is because there is little evidence of an alkaline chemistry in crack face oxide film evaluations, plant eddy current voltages are sometimes lower than those produced by laboratory tests conducted with concentrated solutes, and longer term crack growth rates are not consistent with the rates produced by concentrated solutes. Consequently, it is concluded that most plant SCC reflects contributions from both concentrated solutions and superheated steam.

9:00 AM

Top of Tubesheet Cracking Bruce A NGS Steam Generator Tubing - An Assessment of Bruce a Versus Bruce b Tubing Material: *M. Mirzai*[1]; M. Clark[2]; O. Lepik[2]; I. Thompson[2]; [1]Ontario Hydro Nuclear, 700 University Ave., Toronto, Ontario M5G 1X6 Canada; [2]Ontario Hydro Technologies, 800 Kipling Ave., Toronto, Canada

Steam generators at Bruce A Units 1 and 4 have recently been found to exhibit circumferentially oriented degradation at the top of tubesheet (TTS). In Unit 1 removed tubes show the degradation to be OD initiated IGAIIGSCC. In Unit 4, the tubes removed have exhibited circumferential cracking initiated from both the primary side (PWSCC), and the secondary side (ODSCC). Steam generator tubes at Bruce A (BA) and Bruce B (BB) are made of Alloy 600. World experience shows that all alloy 600 tubed SGs are vulnerable to PWSCC and ODSCC. In light of observed PWSCC/ ODSCC at $BA_1$the susceptibility of BB steam generator tubing material to PWSCC and ODSCC, compared to BA steam generator tubing was assessed. In this paper the preliminary results of this comparison will be presented.

9:30 AM

ATEM and SIMS Examinations of Oconee Nuclear Station Steam Generator Pulled Tubes: *Dewey Paul Rochester*[1]; [1]Duke Power Company, Nuclear Chemistry, 526 S. Church St., EC07D, Charlotte, NC 28202 USA

Since 1994 tubes have been removed from the Oconee Nuclear Station once through steam generators to investigate the freespan axially oriented intergranular corrosion. The tubes were subjected to a variety of traditional destructive and non-destructive examination techniques such as scanning electron microscopy (SEM), X-ray photoelectron spectroscopy (XPS) and Auger electron spectroscopy (AES). The results of these analyses were inconclusive in determining the specific damage mechanism involved. To obtain additional information the tubes were analyzed by secondary ion mass spectroscopy (SIMS) and analytical transmission electron microscopy (ATEM). The results of these analyses will be compared to the traditional examination techniques to provide additional insight into the possible damage mechanisms.

10:00 AM Break

10:30 AM

Corrosion Control and Lay-up of the Crystal River-3 Steam Generators and Secondary Plant during an Extended Outage: *Rocky H. Thompson*[1]; Bill Kassen[2]; [1]Florida Power Corporation, 15760 W. Power Line St., NR-1A, Crystal River, FL 34428 USA; [2]NWT Corporation, 7015 Real Dr., San Jose, CA 95119 USA

The Crystal River-3 nuclear plant was shut down for over 17 months (September 1996 to February 1998) to perform upgrades on safety systems. When the plant initially shut down, the total duration of the outage was unknown. Therefore, a phased approach was taken to lay-up the secondary plant for corrosion control. As the duration of the outage became better defined, the mode of lay-up of the balance of plant was modified. For 11 months of the shutdown, the secondary balance of plant was idle and placed in some form of layup. The once through steam generators were placed in wet lay-up and recirculated every other day for the entire length of the outage. Procedural guidance existed for the lay up of the steam generators, however, none existed for lay-up of the balance of plant. The literature was reviewed to obtain guidance for lay up of the balance of plant. Limited information was found. The literature contained no quantitative correlations between lay-up techniques and associated corrosion rates. However, an overview of lay-up techniques was found in Electric Power Research Institute (EPRI) Sourcebook for Plant Lay-Up and Equipment Preservation published in 1992. This source book was used to develop a lay-up strategy for the balance of plant. This paper summarizes the developed lay-up strategy and the lay-up techniques employed in both the steam generators and the balance of plant to minimize corrosion during the long shutdown. Plant start up data is summarized and compared to similar start up data from outages of much shorter durations in order to assess the effectiveness of the lay-up techniques in controlling corrosion. The mass of impurities removed form the steam generators by leaching out into the wet lay-up water during the long shutdown is compared to that removed during hideout return during shutdowns. Lay-up of the secondary balance of plant was proceduralized based on the lessons learned from this outage, the EPRI sourcebook, and a review of over a dozen lay-up procedures form other nuclear plants. The basic approach of the lay-up procedure is summarized.

11:00 AM

Lead Induced SCC Propagation Rates in Alloy 600: *Michael D. Wright*[1]; [1]AECL, Chalk River Laboratories, Chalk River, Ontario KOJ 1JO Canada

Lead induced stress corrosion cracking has been a significant degradation mode in Alloy 600 steam generator (SG) tubing. Laboratory tests have been performed to determine crack growth rates using compact tension fracture mechanics specimens under constant load conditions. Crack growth rates were measured in simulated all volatile treatment (AVT), secondary side water, with two levels of lead contamination. Tests were also performed in environments simulating lead contaminated SG crevice chemistries, covering a pH range of 3.3 to 8.9. The results of these tests have been compared with crack growth rates inferred from other laboratory tests and plant data. Crack growth rates for the heavily lead contaminated SGs at the Bruce “A” plant are thought to be 0.01 * 0.006 *m/h. Rates measured in the fracture mechanics tests at 300℃ in AVT plus PbO suggest a higher growth rate of 0.3*m/h at 270℃. In contrast, the concentrated crevice environments suppress lead induced cracking and no crack growth was detected in the fracture mechanics tests. Compilentary constant extension rate tests confirm that cracking is suppressed but does still occur in the crevice environment. A lead induced SCC growth rate for the crevice environment based, on crack extension detection limit, supports the continued use of crack growth rates inferred from station data for fitness for service purposes.

Radiation Effects: Session II - Radiation Effects on Deformation and Swelling

Thursday AM — Room: Pacific Ballroom D
August 5, 1999 — Location: Marriott Hotel

Session Chairs: G. Was, University of Michigan, USA; K. Pettersson, Royal Institute, Stockholm, Sweden

8:00 AM

Radiation-Induced Microstructure Effects on the Hardening and Deformation Behavior of Austenitic Stainless Steel: *D. J. Edwards*[1]; E. P. Simonen[1]; S. M. Bruemmer[1]; [1]Pacific Northwest National Laboratory, P.O. Box 999, MSC P8-15, Richland, WA 99352 USA

Austenitic stainless steels used for structural components in light-water-reactor (LWR) core internals experience considerable hardening after low dose irradiation exposure. Tensile yield strengths can increase by more than four times the annealed value as fluence levels reach about $2x10^{21}$ n/cm^2 or three displacements per atom (dpa) and strengths tend to saturate at around 5-10 dpa. Over this same dose range, tensile ductility decreases along with the resistance to stress corrosion cracking. It is clear that a significant fraction of the hardening occurs below 1 dpa, a regime where the primary microstructural feature is a high density ($>10^{23}$ m^{-3}) of very small Frank loops (< 5 nm diameter). However, there have been limited microstructural observations on LWR irradiated steels to determine how the small Frank loops (or other defects) influence hardening or deformation behavior. Radiation hardening in metals and alloys has been commonly described based on the dispersed barrier model with the radiation-induced defect microstructure (small clusters and dislocation loops) act as barriers to the motion of dislocations. More recently, a cascade-induced source hardening model was introduced that attributes the hardening observed in pure metals to the blocking of dislocation sources, that is, the inability to generate new dislocations controls the behavior rather than the inhibition of dislocation movement. To provide further insight into the behavior of the irradiated stainless steels, these models are used to assess relationships between measured defect microstructures and hardening response for several 300-series stainless steels. In addition, in-situ and ex-situ experiments were conducted using transmission electron microscopy to isolate defect microstructural effects on deformation characteristics. Deformation behavior in low- and intermediate-dose samples has been examined along with microstructures altered by post-irradiation annealing. Mechanistic issues related to radiation hardening and localized deformation of LWR-irradiated stainless steels will be discussed.

8:30 AM

Optimized Chemical Composition and Heat Treatment Conditions of 316CW and High Chromium Austenitic Stainless Steels for PWR Baffle Former Bolts: *T. Yonerawa*[1]; [1]Mitsubishi Heavy Industries, Ltd., Ftakasago R & D Center

Authors have clarified that the intergranular cracking baffle former bolts made of austenitic stainless for PWR after long term operation shall be caused by the PWSCC, similar to nickel base alloys, due to grain boundary chemical composition change as radiation induced segregation, in previous studies. And also, authors have found that the PWSCC resistance of simulated austenitic steels of which chemical composition is simulated to the grain boundary chemical composition after irradiation of 2x1022 n/cm2 (E>0.1MeV), increases with increasing chromium content and precipitation of the grain boundary M23C carbides, and with decreasing silicon content, by SSRT tests in simulated PWR primary water. In order to develop the irradiation assisted stress corrosion cracking resistant austenitic stainless steels, authors have studied the optimized chemical composition and heat treatment conditions of 316CW and high chromium austenitic stainless steels for PWR baffle former bolts. As optimized chemical composition of the 316CW stainless steel, ultra low impurity elements, up to 30 weight percent chromium control to increase SCC resistance, up to 30 weight percent nickel content to be stable austenitic phase and to close thermal expansion coefficient of that material as same as one of 300 series steel have been recommended. Heating at 700-725 for 40-50h was selected as suitable aging condition after solution treatment in order to precipitate coherent M23C6 carbides with materials along the grain boundaries and to recover the sensitization. Ten-fifteen percent cold working after aging was selected to satisfy the mechanical properties for bolting materials.

9:00 AM

Irradiation Creep Behavior of High-Purity Stainless Steels and Ni Base-Alloys: *F. Garzarolli*[1]; P. Dewes[1]; S. Trapp Pritsching[1]; J. L. Nelson[2]; [1]Siemens AG. KWU. NBTW, Kembrennstoff-Kreislauf, Hammerbacherstr 12&14, Postfach 3220, Elangen, FRG D-91050 Germany; [2]EPRI, P.O. Box 10412, Palo Alto, CA 94303 USA

Austenitic stainless steels and Ni-base alloys are used in the core of light water reactors. The experience with these materials has, in general, been good. However, not much is known of their long term deformability under irradiation. In the high flux region of BWR's and PWR's both austenitic stainless steels and Ni-base alloys have been observed to suffer under irradiation assisted stress corrosion cracking (IASCC). The objective of this work was to determine the in-pile creep behavior of the materials which were tested in the course of an extensive program on IASCC-resistance of austenitic stainless steels and Ni-base alloys. Most of the IASCC-samples were internally stressed tubular specimens. However, some specimens or parts of specimens had no swelling body. These were used for irradiation creep evaluation. The tangential stress of these samples ranged from 118 to 138 N/mm^2, the maximum neutron fluence reached $1.5*10^{22}$ n cm^{-2}. The results of this study indicate an almost constant creep rate in the fluence range examined. The creep rate varied between the different materials up to a factor of 2. The IASCC-resistant austenitic high purity variants revealed a lower creep strength. The difference, however, is too small to be considered as the primary cause of IASCC resistance.

9:30 AM

The Effect of Low Dose Rate Irradiation on the Swelling of: *Todd R. Allen*[1]; Hanchung Tsai[2]; Shigeharu Ukai[3]; Shunji Mizuta[3]; Tsunemitsu Yoshitake[3]; James I. Cole[1]; [1]Argonne National Laboratory-West, Eng. Div., P.O. Box 2528, Idaho Falls, ID 83403-2528 USA; [2]Argonne National Laboratory, Engy. Tech., 9700 South Cass Ave., Argonne, IL 60439-4832 USA; [3]Power Reactor & Nuclear Fuel Development Corporation, Fuel and Mat. Div., 4002 Narita, Ibaraki-ken, Oarai-machi 311-13 Japan

In pressurized water reactors (PWRs), stainless steel components are irradiated at temperatures that may reach 400°C due to gamma heating. If large amounts of swelling (>10%) occur in these reactor internals, significant swelling related embrittlement may occur. Although density measurements from fast reactor studies indicate that swelling should be insignificant at PWR temperatures, the low dose rate conditions experienced by PWR components may possibly lead to significant swelling. To address these issues, JNC and ANL have collaborated to analyze swelling in 316 stainless steel, irradiated in the EBR-II reactor at temperatures from 376-444°C, at dose rates between 4.9x10-8 and 5.8x10-7 dpa/s, to doses up to 56 dpa. For these irradiation conditions, the swelling decreases markedly at temperatures less than 386°C, with the extrapolated swelling at 100 dpa being only 2.8%. For a factor of two differences in dose rate, no statistically significant effect of dose rate on swelling was seen. The swelling measurements of this study do not support statistically significant swelling of 316 stainless steel in PWRs.

10:00 AM Break

10:30 AM

Recent Data on Void Formation in US & Russian Stainless Steels at PWR-Relevant Conditions: *F. A. Garner*[1]; G. L. Bond[2]; B. A. Gurovich[3]; S. I. Porollo[4]; [1]Pacific Northwest National Laboratory; [2]New Mexico Institute; [3]Kurchatov Institute of Atomic Energy; [4]Institute Physics & Power Engineering

In the last five years there has developed a growing realization that void swelling may develop in some components of PWR baffle-former assemblies, with annealed 304L components probably swelling earlier than cold-worked 316 components based on earlier studies conducted in U.S. fast reactors. Evidence continues to accumulate to support this possibility, but to date the majority of the data at PWR-relevant dose rates have been derived from Russian steels irradiated in various reactors in the former Soviet Union. Two new sets of U.S. data are presented in this paper on the swelling of annealed MSI 304 stainless steel at PWR-relevant displacement rates in the EBR-ll fast reactor. In the first series, five specimens were irradiated at 378-385°C and rates of 0.7 to 1.5 x 10^{-8} dpa/sec to exposures of 4-10 dpa. In the second set, three specimens were irradiated at 278-380°C at rates of 6.5 to 16.0 x $l0^{-8}$ dpa/sec to 14-16 dpa. Swelling was observed by both microscopy and density change in all specimens, ranging from 0.3 to 1.2%, increasing with dose and also increasing with decreasing displacement rate. These swelling levels are much larger than observed earlier at displacement rates of 0.5 to 1.0 x 10^{-6} dpa/sec in EBR-II.

11:00 AM

Measurement of Hydrogen Generation, Accumulation & Release in Irradiated Fe and Ni Alloys during Irradiation: *F. A. Garner*[1]; B. M. Oliver[1]; B. A. Gurovich[2]; [1]Pacific Northwest National Laboratory; [2]Kurchatov Institute of Atomic Energy

This work addresses two perceptions concerning irradiation of stainless steels in PWRs. The first perception is that no void swelling will occur, and second is that hydrogen plays no role in radiation-induced microstructural development. This study used four Russian stainless steels (AEI-844BU, OXl8H10T AEP-753 and AEP-337U with nickel contents ranging from 10.8 to 40%, that were irradiated at low displacement rates to 1-8 dpa in a defense-related PWR or a related materials test reactor. Swelling was determined by microscopy. At these low dose levels, cavity swelling extended down to 340-350°C in the low nickel steels and down to 270°C in the high nickel steels. The lower temperature limit for high nickel steels is thought to be related to the role of nickel to produce most of the hydrogen and helium. Post-irradiation annealing for one hour at each of 400, 500, 600, 700, 800, 900, and l000°C was carried out on two alloys. Changes in hardness and microstructure were determined and most importantly, the evolution of hydrogen gas was measured during annealing. It was found that the hydrogen was released primarily when the voids dissolved, and not earlier when black spots and loops dissolved. Annealing to higher temperatures led to formation of large, previously unobserved helium bubbles. The conclusion drawn is that hydrogen contributes strongly to the stabilization of cavity nuclei at PWR-relevant temperatures, and that most helium resides in the matrix and not in visible cavities.

11:30 AM

Effects of Neutron Irradiation on Mechanical Behavior of Nickel-Based Fastener Alloys: *R. Bajaj*[1]; W. J. Mills[1]; B. F. Kammenzind[1]; M. G. Burke[1]; [1]Bettis Atomic Power Laboratory, P.O. Box 79-ZAP 03N, West Mifflin, PA 15122-0079 USA

This paper presents the effects of neutron irradiation on the tensile properties, fracture behavior and deformation microstructure of high strength nickel-based alloy fastener materials, Alloy X-750 and Alloy 625. Four heats of Alloy X-750 in the HTH heat treatment, and three heats of Alloy 625 in the direct-age heat treatment were irradiated to a fluence of $2.1x10^{20} n/cm^2$ (E > 1 MeV) at 680°F in the advanced test reactor (ATR). Post-irradiation tensile tests were carried out at 150°F and 550°. Fracture surfaces of samples taken to failure were examined in a scanning electron microscope, and deformation structures at the 2% and 4% strain levels were examined in a transmission electron microscope. Alloy X-750 showed irradiation hardening and embrittlement with a yield strength increase of about 40% and a ductility degradation to as low as 2% total elongation. In contrast, Alloy 625 showed irradiation softening and a concomitant increase in ductility after irradiation. Alloy X-750 failed in a faceted intergranular mode and Alloy 625 exhibited a transgranular ductile fracture mode. The deformation microstructures of the two alloys were also different. Alloy X-750 deformed by planar slip with fine microcracks forming at the intersections of slip bands with grain boundaries. Alloy 625 showed much more homogeneous deformation with fine closely spaced slip bands and an absence of microcracks. The role of irradiation hardening and deformation microstructures in the mechanism of irradiation assisted stress corrosion cracking (IASCC) of these materials will be discussed.

PWR Secondary: Session III - Mechanisms

Thursday PM Room: Pacific Ballroom C
August 5, 1999 Location: Marriott Hotel

Session Chair: P. Lichtenberger, Ontario Hydro, Canada; M. Clark, Ontario Hydro, Canada

1:30 PM
Modeling the Secondary Side Corrosion of Tubings: A Help to the Maintenance Policy of PWRs Steam Generators: *F. Vaillant*[1]; E. M. Pavageau[1]; M. Bouchacourt[2]; J. M. Boursier[3]; P. Lemaire[3]; [1]EDF, R&D Div.; [2]Engineering and Construction Division
Steam generator (SG) tubings in mill-annealed (MA) Alloy 600 suffer Intergranular Attack (IGA) and Intergranular Stress Corrosion Cracking (IGSCC) in flow-restricted locations where pollutants in the secondary water could concentrate. During the first ten years of operation the crevice environment was thought to be strongly alkaline. In spite of improvements in the operating conditions, corrosion degradations continue to grow in weakly basic crevices. Therefore the development of a model was undertaken in order to provide help to the maintenance policy and to predict the life duration of SGs. The first run of the model is based on caustic environments. Analytical relationships have been developed in order to account for IGA and IGSCC behaviors of Alloys 600 MA and TT in the laboratory: IGA depth was a function of time, temperature, stress and pH; IGSCC occurred when the stress level was higher than a threshold related to the yield stress (YS) of the tubes. The slow crack growth rates (CGRs) depended on pH, temperature and material properties. After a thermomechanical analysis of SG tubes and an approach of the local environment during operation, the stress, the temperature and the assumed pH inside the crevices were introduced into the corrosion laws. The occurrence, orientation and location of IGSCC in SGs were well described by the model and the calculated CGRs were in reasonable agreement with the CGRs deduced from pulled tubes examinations. Then a suitable maintenance policy (definition of frequencies for NDE, life prediction for replacement) based on calculated CGRs seemed possible for tubings in Alloys 600 MA remaining in operation and mostly in Alloy 600 TT. The margins with Alloys 690 and 800 were also evaluated. A future development of the model will incorporate SCC results obtained in neutral to midly basic environments.

2:00 PM
Comments on a Proposed Mechanism of Internal Oxidation for Alloy 600 as applied to Low Potential SCC: *R. W. Staehle*[1]; Z. Fang[1]; [1]University of Minnesota, Dept. of Chem. Eng. and Mats. Sci., Corrosion Center
The SCC parameter is derived from comparing fast and slow polarization scans. The result from this comparison produces a parameter called the SCC parameter, P_{SCC}, which can be determined as a function of potential. Peaks in this parameter identify potentials where SCC may occur depending on the height and width of these peaks. While SCC has been predicted for environment-material combinations near regions of obvious instability, the possibility of SCC has been identified in the mid-range of potential according. The P_{SCC}, is compared with the occurrence of SCC for both chloride and sulfur environments.

2:30 PM
Nickel Alloy Stress Corrosion Cracking in Neutral and Lightly Alkaline Sulfate Environments: *O. DeBourvier*[1]; [1]EDF, R&D Division, Mats. Studies Dept., Les Renardleras, Moret sur Loing Cedex 77818 France
Since 1986, EDF plant operators was concerned with steam generator secondary side corrosion. For steam generators with drilled tube support plates (TSP) and mill-annealed (MA) alloy 600 tubes, cracking appears in flow-restricted locations (top of the tube sheet and tube support plate elevation) where species concentration can be very high. The studies, dealing with the effects of different parameters and the predictive model are confronted with the unknown of the crevice that it is used to consider as a strong alkaline environment. However, hideout return results and recent surface analyses of pulled tubes have shown that the environments are complex and strongly alkaline. In this work, the process is to study the influence of various pollutants identified with the hideout return analyses. In a first approach neutral and lightly alkaline sulfate environments were investigated because of their very high concentration during the hideout return. Results of stress corrosion cracking tests (C-ring samples) at 320°C in sulfate medium (pHT 5 to 9.5) have shown that IGSCC for alloy 600 MA is high for pH 5 and is decreasing with increasing pH. No major influence of the stress level on crack growth rate is observed beyond a stress threshold. Thermal treatment (16 h at 700°C) decreases by a factor of two, the crack growth rate. With regard to the replacement materials, 800 alloy is susceptible to cracking only in neutral environment and alloy 690 is not affected. This pH effect is not sufficient to explain the corrosion cracking of 600 alloy which depends also on the sulfate concentration and probably on the electrochemical potential. So, an electrochemical potential interpretation of the results is suggested.

3:00 PM Break

3:30 PM
Pb SCC of Alloy 690: *M. J. Psaila-Dombrowski*[1]; F. H. Hua[1]; P. E. Doherty[3]; [1]Babcock & Wilcox, Res. & Dev. Div., 531, Coronation Blvd., Cambridge, Ontario N1R 5V3 Canada; [2]Babcock & Wilcox International; [3]Dominion Engineering
Lead (Pb) is a common contaminant in operating steam generators (SGs). Pb concentrations in the order of 1000's of ppm have been measured on pulled tubes. Lead has been identified as a stress corrosion cracking initiator (PbSCC). PbSCC has been identified as a cause of Alloy 600 tubing failure in operating SGs. Laboratory evidence indicates that Alloy 690 is also susceptible to PbSCC. In an order to obtain a better understanding of Alloy 690 PbSCC tests were performed. Two possible mechanisms were evaluated: electrochemical polarization and hydrogen embrittlement. Electrochemical tests were performed on thermally treated (TT) Alloy 690 and Alloy 600 SG tubing in NaOH solutions with and without PbO. Corrosion potential was measured as a function of time, temperature and NaOH concentration. Large differences in open circuit potentials were observed at room temperature indicate some surface effect. Testing at elevated temperatures indicate that the presence of PbO may polarize the nickel based alloys as much as 150 mV under specific conditions. CERT tests performed under conditions were polarization occurred did not produce SCC. Polarization scans were also performed under these test conditions. Lead acts as a hydrogen recombination poison and promotes the entry of hydrogen into alloys at room temperature. Tests were performed to evaluate the effect of hydrogen on Alloy 690. C-ring specimens were exposed in two canisters containing 10%NaOH (one canister was saturated with PbO). Hydrogen (H_2) content of specimens was evaluated after 2000 hours exposure. All specimens indicated an increased H_2 content. The presence of lead had no impact on hydrogen content of the specimens. The Alloy 690 specimens had twice as much hydrogen compared to the Alloy 600 specimens exposed to the same solutions. Constant Extension Rate (CERT) tests were performed at room temperature and elevated temperatures with and without cathodic polarization. These tests were conducted in arsenic contaminated acidic and NaOH solutions. Specimens exhibited a significant degradation in properties with hydrogen charging. Maximum load and time to failure were dramatically decreased. Results are summarized in the paper.

Radiation Effects: Session III - Radiation Effects on Microstructure and Microchemistry

Thursday PM Room: Pacific Ballroom D
August 5, 1999 Location: Marriott Hotel

Session Chairs: S. Suzuki, TEPCO R&D, Japan; P. Spellward, Magnox Electric, UK

1:30 PM
Microstructural, Microchemical and Hardening Evolution in LWR-Irradiated Austenitic Stainless Steel: *S. M. Bruemmer*[1]; B. W. Arey[1]; L. A. Charlot[1]; D. J. Edwards[1]; [1]Battelle Pacific Northwest National Laboratory, P.O. Box 158, Mail Bin 26, Richland, WA 99352 USA
A comprehensive characterization of pressurized-water-reactor (PWR) irradiated, 348 and 304 stainless steels has been performed documenting radiation dose effects on material microstructure, microchemistry and hardness. Individual heats were selected based on their susceptibility to irradiation assisted stress corrosion cracking (IASCC) during in-core swelling mandrel experiments. Characterization of irradiated and non-irradiated materials included defect microstructures by transmission electron microscopy (TEM), grain boundary composition by analytical TEM and by scanning Auger microscopy, helium transmutation by isotope-dilution mass spectrometry and strength determinations by hot hardness. Detailed comparisons were made between IASCC-susceptible and IASCO-resistant heats in order to assess radiation-induced material changes that may influence cracking response. Differences in radiation-induced microstructures and microchemistries were observed among the 348SS heats to rationalized IASCC behavior, but this was not true for the 304SS heats. Current understanding of microstructural and microchemical effects on IASCC will be summarized and the paramount need for controlled irradiations and quantitative testing to isolate variables promoting IASCC will be discussed.

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Microchemistry and Microstructure Evolution in Proton-Irradiated Austenitic Stainless Steels: *Jeremy T. Busby*[1]; Jian Gan[1]; Matthew Daniels[2]; Steve Bruemmer[3]; E. A. Kenik[4]; G. S. Was[1]; [1]University of Michigan, Dept. of Nuclear Eng. and Radiological Sci., 2940 Cooley Bldg., 2355 Bonisteel Blvd., Ann Arbor, MI 48109-2104 USA; [2]University of Michigan, Dept. of Mats. Sci. and Eng., 2940 Cooley Bldg., 2355 Bonisteel Blvd., Ann Arbor, MI 48109-2104 USA; [3]Batteile Pacific Northwest National Laboratory, P.O. Box 158, Mail Bin 26, Richland, WA 99352 USA; [4]Oak Ridge National Laboratory, Metals and Ceramics Div., Bldg. 5500 MS 6376, P.O. Box 2008, Oak Ridge, TN 37831 USA
The evolution of microchemistry and microstructure is examined in Fe-Cr-Ni alloys irradiated with 3.2 MeV protons at a dose rate of 7 x 10-6 dpa/s over a temperature range of 200°C to 600°C and up to doses of 3.0 dpa. Measurements are compared to either neutron-irradiation data (where available) or to model prediction. Grain boundary composition was measured using scanning transmission electron microscopy with energy dispersive x-ray spectrometry (STEM-EDS) and dislocation loop microstructure was characterized using transmission electron microscopy (TEM) (both bright and dark field imaging). Measurements of grain boundary segregation in Fe-20Cr-24Ni between 200°C and 600°C revealed that segregation peaks at intermediate temperatures, in agreement with model prediction. Dislocation loop density decreased, and loop size increased with temperature. The dose dependence of microchemical and microstructural changes was investigated at a temperature of 360°C for 304 and 316 stainless steel alloys. A large amount of Cr enrichment was measured in unirradiated 316 SS, which evolved into a distinct "W"-shaped segregation profile at 1.0 dpa and distinct depletion by 3.0 dpa. Unirradiated 304 SS showed a smaller amount of Cr segregation than in 316 SS, and did not develop the distinct "W"-shaped profile during irradiation. Grain boundary composition measurement and segregation profiles from both heats are in excellent agreement (magnitude and profile shape) with those taken from the same heats irradiated with neutrons to comparable doses. Loop densities and loop size distributions for both heats are comparable or below those from neutron irradiation. Radiation hardening, measured using Vickers indentation, tracks hardening induced by neutron irradiation very closely.

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Comparison of Radiation Induced Degradation in Several Austenitic Stainless Steels Used for Core Internals in LWR: *T. Fukuda*[1]; T. Fukuda[2]; M. Sagisaka[1]; Y. Isobe[1]; A. Hasegawa[2]; M. Sato[2]; K. Abe[2]; K. Matsueda[3]; Y. Nishida[3]; Y. Kaneshima[3]; [1]Nuclear Fuel Industries, Ltd., 950 Ohaza-Noda, Kumatori-cho, Sennan-gun, Osaka Japan; [2]Tohoku University, Aramaki-Aza-Aoba, Aoba-ku, Sendai, Miyagi Japan; [3]Kansai Electric Power, Co., Inc., 3-3-22, Nakanoshima, Kita-ku, Osaka Japan
In austenitic steels used for core internals in LWR, neutron irradiation induces irradiation hardening and irradiation assisted stress corrosion cracking (IASCC) etc. Besides, more attention has been paid these days for void swelling problem. This study presents microchemical and microstructural changes in existing and alternative austenitic stainless steels used for core internals (SUS304, SUS347, SUS310+Nb and XM-19), which were irradiated by light ions. The specimens were implanted with helium and irradiated with 2 MeV H2+ ion to about 1 dpa at 300 and chemical analysis was performed using field emission-gun transmission electron microscope (FE-TEM) and energy dispersive spectroscopy (EDS). The results showed that the depletion of chromium near grain boundaries and strongly depended on both chemical compositions and helium contents. Observation by TEM also indicated that small cavities generated in all irradiated steels. These results were discussed focusing on effects of chemical compositions and helium contents on radiation damages.

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Local Evolution of Microstructure and Microchemistry Near Irradiated Grain Boundaries in Austenitic Stainless Steels: *E. P. Simonen*[1]; D. J. Edwards[1]; S. M. Bruemmer[1]; [1]Battelle Pacific Northwest National Laboratory, P.O. Box 999, MSIN P8-15, Richland, WA 99352 USA
Grain boundary microchemistry is often examined in relation to irradiation-assisted stress corrosion cracking (IASCC) of light-water reactor (LWR) core internals, but there has been little discussion of near boundary microstructures. Radiation-induced segregation (RIS) measurements clearly establish that grain boundaries are effective sinks for point defects in austenitic stainless steels. The depletion of point defects near grain boundaries affects the expected development of microstructure that in turn affects the development of RIS profiles. Rate theory calculations predict that damage microstructures will impact defect flow to grain boundaries when the matrix sink density (i.e., dislocation loops) exceeds a critical value. These densities can be exceeded during LWR irradiation of austenitic stainless steels. At lesser sink densities, mutual recombination of vacancies and interstitial controls the fate of point defects near grain boundaries and microstructural development should have a lesser impact on the grain boundary condition. In this paper, interactions between microstructural evolution and RIS are evaluated to better define the grain boundary condition over the range of LWR irradiation temperatures and doses.

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Effect of Minor Elements on EAC of Austenitic Simulated Steels in PWR Primary Water and Implication to IASCC: *Guangfu Li*[1]; *Yoshiari Kaneshima*[2]; Testsuo Shoji[1]; [1]Tohoku University, Res. Insti. for Fract. Tech., Graduate School of Eng., 01 Aza-Aoba, Aramaki, Aoba-ku, Sendai 980-8579 Japan; [2]The Kansai Electric Power Co., Inc., Office of Nucl. and Fossil Power Prod., 3-3-22, Nakanoshima, Kita-Ku, Osaka 530-8270 Japan
The susceptibilities of four austenitic steels with different Si, P and S contents to environmentally assisted cracking (EAC) in PWR primary water at 325°C were assessed by using trapezoidal wave cyclic loading

testing. The basic chemical composition of the steels was 12%Cr-28%Ni, simulating the grain boundary composition of 304 stainless steel with radiation induced segregation (RIS). In solution-annealed condition, intergranular cracking at relatively high rates occurred in the Si-doped steels. The higher the Si content, the higher the crack growth rate was observed. Crack morphology showed that the grain boundaries of the high Si (2.74%) steel was very brittle in the deoxygenated water. Only transgranular cracking at low rates took place in the steels with negligible Si but doped with P and S respectively. A heat treatment at 650°C for 20 hours (TT treatment) decreased crack growth rate but could not eliminate intergranular cracking in the high Si steel. The implication of the results to irradiation assisted stress corrosion cracking in a PWR was discussed.

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Studies on Surface Oxide Films of Stainless Steels Having Simulated Post-Irradiated Grain Boundary Chemistries: *K. S. Raja*[1]; T. Shoji[1]; [1]Tohoku University, Research Institute of Fracture Technology, Aobayama 01, Aobaku, Sendai 980-28579 Japan

In order to understand the mechanism of initiation of irradiation assisted stress corrosion cracks (IASCC) in austenitic stainless steels, samples of special materials were prepared whose bulk chemistry simulates the grain boundary of 304 type stainless steel. The basic alloy was Fe28Nil2Cr and the contents of Si,P and S were varied to study the effect of radiation induced segregation of those elements individually. This study reports a part of an on-going project and is concerned only with the surface oxide films formed on those un-stressed samples in the simulated BWR conditions at two different temperatures (200 and 290°C) under controlled electrochemical potentials. The passive films of the samples were characterized by measuring the electroresistance of the films by using the Contact Electric Resistance (CER) method which was proposed by Marichev et.al. The sensitivity of the CER method enables one to differentiate the passivating capability of alloys with different chemistries. It was observed that the material with higher Si content showed higher electroresistance (implying higher passivity) than other materials. Anodic polarization increased the electroresistance of the surface films of high phosphorus material monotonically, whereas for other materials an 'increase and decrease behavior was observed which was electrochemical potential dependent. Increase in test temperature increased the electroresistance of the films. The test temperature affected the semi-conducting property of the passive films during anodic polarization which could be associated with the change in composition of the films. Further studies are being carried out to characterize the passive films ex-situ by using secondary ion mass spectroscopy (SIMS) and atomic force microscopy (AFM) to augment the in-Situ results of CER method.

Zircaloy: Session I

Thursday PM — Room: Pacific Ballroom E&F
August 5, 1999 — Location: Marriott Hotel

Session Chairs: R. Adamson, GE Nuclear, USA; F. Garzarolli, Seimens, Germany

1:30 PM

The Use of Impedance Spectroscopy to Follow Oxidation of Zirconium Alloy in Situ at High Temperature: *B. Albinet*[1]; B. Sala[1]; A. Frichet[2]; [1]Framatome, Tour Fiat, 1 Place de la Coupole, Paris, LaDefense, Cedex 16 92084 France; [2]Framatome, Lyon

In order to increase the fuel lifetime, sensitivity of zirconium to lithium has to be studied. Thus, the evolution of zircaloy-4 coupons preoxidized in pure water was studied out at 300°C, in a deaerated medium containing 300 ppm of lithium. Electrochemical Impedance Spectroscopy (EIS) was successfully used to follow in situ the evolution of the oxide film for several hundreds of hours. We found a good correlation between the oxide thickness calculated from Nyquist and Cole-Cole diagrams, and the measured thickness obtained by post-test metallographic examinations, whatever the oxide thickness. The electrical resistance calculated from the high frequencies arc of the Nyquist diagram, allowed to follow the degradation of the protective character of the oxide film in lithia containing media. The results showed a preliminary step followed by a large decrease of the resistance after several hours of testing. The thickness of the film, measured by the Cole-Cole technique was constant during the experiment and corresponded to the final metallographic examination. This work showed that EIS is a valuable technique to study the behavior of zirconium alloys in severe environments.

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AC Impedance Characteristics of Oxide Films on Zircaloy-4: *Jeong Youn Lim*[1]; Il Soon Hwang[1]; [1]Seoul National University, Dept. of Nuclear Eng., 56-1, Shinlim-dong, Gwanak-gu, Seoul 151-742 Korea

Anodic oxide films grown on pure polycrystalline zirconium and Zircaloy-4 in high temperature water have been characterized by ac impedance techniques in a PWR primary water environment. This technique has been used as an in situ monitor of the thickness and resistance of the film during corrosion process. The thickness of anodically grown oxide is also estimated from a set of coulometric analysis on the oxide formation. Both ac impedance and potentiostatic polarization behaviors are obtained as function of pH and temperature. The results are compared with the structure and the morphology of oxides. Through the fitting of impedance data, parallel RC circuit with a phase shift is found to represent the oxide behavior. The impedance measurement results at anodically applied potential suggest the corrosion process possibly be controlled by diffusion in the oxide films. Since the anodically formed zirconium oxide films are reported to have p-type semiconducting behavior, it is postulated that oxygen vacancy diffusion is controlling the process. From the Warburg impedance value for the oxide film, the diffusion coefficient of oxygen can be estimated.

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Amorphization of Laves-Phase Precipitates in Zircaloy-4 by Neutron Irradiation: *Dale F. Taylor*[1]; H. Richard Peters[1]; Walter J. S. Yang[1]; [1]Lockheed Martin Corporation, MS-089, P.O. Box 1072, Bldg. C-1, Rm 126, Schenectady, NY 12301-0172 USA

Examination of corrosion coupons by transmission electron microscopy after their exposure in the Idaho Advanced Test Reactor (ATR) has broadened the Zircaloy-4 precipitate-amorphization database and validated a new kinetic model for previously unavailable values of temperature and fast-neutron flux. The model describes the amorphization of $Zr(Fe,Cr)_2$ intermetallic precipitates in zirconium alloys as a dynamic competition between radiation damage and thermal annealing that leaves some Fe atoms available for flux-assisted diffusion to the alloy matrix. It predicts the width of the amorphous zone as a function of neutron flux (E>l MeV), temperature and time. In its simplest form, the model treats the crystalline/amorphous and precipitate/matrix interfaces as parallel planes, and its accuracy decreases for small precipitates and high fluence as the amorphous-zone width approaches precipitate dimensions. The simplest form of the model also considers diffusion to be rate-determining. This is an accurate approximation for steady-state conditions or slow changes in flux and temperature, but inappropriate for the analysis of faster transients. The paper addresses several difficulties inherent in measuring amorphous-zone width, and utilizes the expanded database to evaluate the improvements in predictive accuracy available through both conversion of the model to spherical coordinates and extension of its time dependency.

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Characteristics of Axial Splits in Failed BWR Fuel Rods: *Gunnar Lysell*[1]; V. Grigoriev[1]; K. Pettersson[1]; [1]Royal Institute, Stockholm; [1]Studvik Nuclear AB, Nykoping SE-611 82 Sweden

Secondary cladding defects in BWR fuel sometimes have the shape of long axial cracks or "splits". Due to the large open U02 surfaces exposed to the water fission product and U02 release to the water can reach excessive levels leading to forced shut downs to remove the failed fuel rods. A number of such fuel rods have been examined in Studsvik over the last 10 years. This paper describes a number of observations from the PIE of long cracks and discusses the driving force of the cracks. Details such as starting cracks, macro and microscopic fracture surface appearance, cross sections of cracks, hydride precipitates, location and degree of plastic deformation are given. Observations show that the axial splitting can be treated as a process of delayed cracking with hydrogen participation. Some considerations on the controlling factors and probable mechanisms are discussed.

4:00 PM

The Fracture of Zircaloy-2 Plate and Cladding in High Pressure Hydrogen Gas: *R. G. Rowe*[1]; S. T. Mahmood[1]; S. B. Wisner[1]; R. B. Adamson[1]; [1]General Electric CRD, P.O. Box 8, Schenectady, NY 12301 USA

Brittle fracture of unirradiated Zircaloy-2 plate and cladding tubing has been observed at 300°C to 325°C in a gaseous environment of high pressure hydrogen gas. Testing under the same conditions in air or an inert environment (Ar) resulted in ductile necking to high plastic strain. Fracture in hydrogen gas environment occurred by an aggressive and rapid brittle fracture mode that led to crack propagation rates as high as 1 cm/s. The mechanism of embrittlement was the formation and fracture of a thick layer of zirconium hydride at the crack tip. Hydride layer thickness at the fracture surface ranged from 15 to 35 μm. A threshold stress intensity as low as 20 Mpa√m was observed at 325°C. Thick compact tension (CT) specimens and thin walled cladding tubing exhibited similar brittle fracture features. Details of the test specimens, testing techniques, and the resulting fractography will be discussed.

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On the Mechanism of Axial Splits in Failed BWR Fuel Rods: *K. Edsinger*[1]; S. Vaidyanthan[1]; R. B. Adamson[1]; [1]General Electric Nuclear Energy, Vallecitos Nuclear Center

Some boiling water reactors (BWRs) have experienced axial cracks in the fuel cladding often accompanied by large releases of fission products to the coolant. This type of fuel degradation has been the focus of a number of studies over the last several years. Post-irradiation examination (PIE) of degraded fuel cladding by several facilities has identified many of the characteristics of these degraded fuel rods. However, the exact mechanism of crack propagation has remained in question. Recently, cracking with the same characteristic features of in-reactor failures has been achieved in 300°C water with high concentrations of lithium hydroxide. The role of the lithium hydroxide is apparently to supply hydrogen to the cladding through an accelerated corrosion reaction. The hydrogen then precipitates ahead of the crack tip and cracks when the local fracture resistance is sufficiently reduced. The similarity of features in these tests with results from PIE suggest that the in-reactor mechanism of split propagation is also related to hydrogen. The key remaining question is what route the hydrogen takes to the crack tip. Presumably, the hydrogen comes either from crack tip corrosion (similar to stress corrosion cracking) or from corrosion of the inside cladding surface. In the later case, hydrogen migration is driven by the crack tip stress field and the thermal gradient (which attracts hydrogen to the cooler outer surface). A series of laboratory tests, which independently control corrosion/hydrogen pickup at the crack tip and at cladding surfaces, have been conducted to establish the route of hydrogen which leads to crack propagation. These results have implications on the effectiveness cladding remedies for degradation resistance.

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Studies on Delayed Hydride Cracking of Zircaloy Cladding: *Kjell R. Pettersson*[1]; Pål Efsing[2]; Kwadwo Kese[1]; [1]KTH (Royal Institute of Technology), Dept. of Mats. Sci. and Eng., Mech. Metall., Stockholm SE-100 44 Sweden; [2]Barsebäck Kraft AB, P.O. Box 524, Löddeköpinge SE-246 25 Sweden

Occasionally Zircaloy fuel cladding will fail by forming long axial splits which lead to substantial releases of fission products. Post-irradiation examinations of failed fuel suggested that Delayed Hydride Cracking (DHC) could be the mechanism. Tests on unirradiated cladding were used in order to develop a test method for irradiated cladding. These tests showed that unirradiated cladding was sensitive to DHC and that the threshold stress intensity factor for crack growth decreased and the maximum crack growth rate increased with increasing yield strength. The subsequent tests on irradiated cladding showed that it had about the same properties as unirradiated cladding with a similar yield strength. The cracking mechanism was studied with metallographic and fractographic examinations. The observations suggest that the cause of cracking is re-orientation of hydride plates in the vicinity of the crack tip from a plane perpendicular to the plane of the crack to an orientation parallel with the crack. This reduces the local fracture toughness to a value below the applied K and the crack grows until it is arrested in material where the re-orientation has not yet taken place. The driving force for re-orientation is the stress field at the crack tip, and as expected from such a mechanism the observed temperature dependence of cracking is consistent with the combined activation energies for hydrogen diffusion and solubility in Zircaloy. A recent study of the effect of hydride plates oriented perpendicular to the stress in uniaxial tests indicates however that these have no effect on ductility at temperatures above 100°C, while the DHC tests were conducted at 200 and 300°C. The exact mechanism of cracking is therefore still something of a mystery.

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ADDENDUM

Session: PWR Primary: Session III-Hydrogen Effects & Microstructure
Tuesday AM, August 3, 1999
Marriott-Hotell, Pacific Ballroom C

Session Chairs: **W. J. Mills, Bettis Labs, USA** **P. Doherty, Dominion Engineering**

Session: Regulation Aspects
Tuesday PM, August 3, 1999
Marriott-Hotell, Pacific Ballroom C

Session Chairs: **S. Suzuki, Tepco R &D** **K. Gott, SKI, Sweden**

Session: PWR Primary: Session IV - Crack Growth & Creep
Wednesday AM, August 4, 1999
Marriott Hotel, Pacific Ballroom C

Session Chairs: **M. Mirzai, Ontario Power Generation** **J. Hall, ABB Combustion, USA**

Session: Radiation Effects: Session III - Radiation Effects on Microstructure and Microchemistry
Thursday PM, August 5, 1999
Marriott Hotel, Pacific Ballroom D

Session Chair: **Peter Ford - GRCRD**

Session: PWR Secondary: Session II - Cracking Response
Thursday, August 1, 1999
Marriott Hotel, Pacific Ballroom C
11:30 AM

Characterization of Stabilized Stainless Steels Welds of PWR Piping Systems

M. Widera[1]; R. Kilian[2]; R. Bartsch[3]; H. Hoffman[4]; G. Konig[5]; O. Wachter[6]; [1]RWE; [2]Simens; [3]KKO; [4]VGB; [5]KKN; [6]Preussen Electra

During plant shutdown of several German BWRs non-destructive testing has been carried out on the stainless steel (SS) piping systems in those regions which are non-isolable from the pressure retaining boundary within the containment. From l992 to 1994 in some cases intergranular cracks were found in the heat affected zone (HAZ) of weld joints. The material for the piping was the titanium stabilized SS with German material no. 14541 corresponding to AISI 321. The weld filler material was niobium-stabilized SS with German material no. 1.4551 (AISI 347). Crack initiation predominantly occurred in the root notch and the cracks grow along adjacent to the fusion line. Based on numerous detailed investigations it can be stated that the mechanism for crack initiation and crack growth was intergranular stress corrosion cracking (IGSCC) as a result of sensitization during welding. All affected piping systems were replaced by an optimized Nb stabilized austenitic SS with low carbon content and a high Nb/C ratio. Well controlled manufacturing and welding procedures with a good root weld quality were realized. Independent of the reactor type - BWR or PWR - stabilized SS are used for pipings in Germany. The Stress Corrosion Cracking (SCC) phenomenon requires the simultaneous action of material condition, stress state and corrosive medium. The factors medium and stress state are specific of reactor type respectively of component. The material condition depends on many factors like material selection, chemical composition, welding procedure and so on. The occurrence of cracks in the HAZ of stabilized austenitic SS pipes was the major reason for basic examination of austenitic weld joints, which have been in operation for several years, from PWR plants which have been erected in the same time period as the affected BWR plants. In the present study the material conditions respectively the degree of sensitization in the HAZ of welded Ti- and Nb-stabilized SS are mainly considered. The purpose of this work is to show the influence of operating temperature and time as well as the influence of PWR primary water chemistry conditions on the materials behaviour. Therefore the HAZ is characterized by hardness measurements, determination of ferrite content, oxalic acid aging, EPR tests and microstructure examinations of selected welds using TEM.