Name	Organization; Country	Title of presentation	Abstract
Dr. AKASHI, Masatsune	Consultant; Japan	Propagation sustainability of localized corrosion of stainless steels in 1F spent fuel pool environments	The chloride ion concentration in 1F spent fuel pool water was high enough for the initiation of localized corrosion for Type 304 stainless steel. During the purification treatments, the chloride ion concentration has decreased below the critical concentration for the initiation of localized corrosion. This paper discussed the possibility of the localized corrosion initiated and propagated in higher concentration sustain the propagation even in the condition below the critical concentration for initiation.
		Water chemistry estimation of 1F spent fuel pools for the discussion of carbon steel corrosion	The corrosion tendency of carbon steel piping in natural fresh water environments changes on the water chemistry condition, especially on the bicarbonate, chloride, and sulfate ion concentrations. On the other hand, the actual water chemistry information for 1F spent fuel pools is limited for chloride ion concentration and pH. This paper estimates the water chemistry in detail, based on the fundamental ionics.
Dr. ANDRESEN, Peter L	General Electric Global Research Center (GE-GRC); USA	A perspective on the realities of SCC	SCC has proven complex and elusive to measure, understand and predict, and this talk distills some realities of SCC that must be acknowledged if we are to make progress. These include technical issues (immunity, measurement, iniitiation), material degradation management (where systematized delineation of the issues has failed to lead to projects to address the issues within the next few centuries), and sustaining expertise and capability and funding to address issues in the future.
		SCC of Alloy 800, high Cr Alloy 800 and Alloy 825 in high temperature water	SCC growth rate experiments were performed on Alloy 800, high Cr variants of Alloy 800 and Alloy 825 in high temperature water to evaluate their suitability and attractiveness as alternative materials.
Dr. ARIOKA, Koji	Institute of Nuclear Safety System, Inc. (INSS); Japan	Ni concentration dependence on the rate of SCC growth in PWR primary water.	Influence of Ni concentration on the rate of SCC growth was examined in PWR primary water using 20%CW Ni-Cr-Fe alloys between 11%Ni to 70%Ni. The effect of Ni concentration and the temperature dependence on the rate of SCC growth will be presented.

Dr. BRUEMMER, Stephen M	Pacific Northwest National Laboratory (PNNL); USA	Is there evidence for grain boundary internal oxidation of Alloy 600 in PWR primary water?	Numerous analytical transmission electron microscopy (ATEM) examinations of intergranular attack (IGA) and stress corrosion crack tips in alloy 600 have been performed on materials removed from PWR primary water service or on samples tested in the laboratory. Selective oxidation clearly occurs at the IG corrosion-oxidation front, but conclusive evidence for grain boundary internal oxidation has been lacking. New research will be presented combining ATEM with atom probe tomography (APT) to resolve atomistic compositions at the IGA oxidation front and in the leading grain boundaries. Results confirm the significant depletion of certain elements from the grain boundary ahead of the oxidation front, but there is again no evidence for the formation of discrete oxide precipitates or oxygen-rich clusters in the metal to support an internal oxidation mechanism.
		Stress corrosion cracking susceptibility of Alloy 690 in PWR primary water	Current understanding of stress corrosion cracking (SCC) susceptibility for as-received and cold-worked alloy 690 plate, bar and tubing heats will be reviewed based on detailed material characterizations and crack-growth-rate testing. Correlations will be presented among general microstructure, carbide distributions, hardness, electron backscatter diffraction measurements of strain and SCC propagation rates. New results evaluating material condition and grain boundary microstructure effects on SCC susceptibility will be discussed.
Mr. CHOI, Kyoung Joon	Ulsan National Institute of Science and Technology (UNIST); South Korea	A study of Thermal Aging Effects on Microstructures of Dissimilar Metal Weldment	In this study, the advanced instrumental analysis has been performed to investigate the effect of long-term thermal aging on the fusion boundary region between weld metal and low alloy steel in dissimilar metal welds. A representative dissimilar weld mock-up made of Alloy 690-Alloy 152-A533 Gr. B was fabricated and aged at 450oC for 2,750 and 5500 hours. The nano-scale characterization was mainly conducted near in weld root region using scanning electron microscope, transmission electron microscope, secondary ion mass spectrometry, and 3 dimensional atom probe tomography. It was found that the weld root region was generally divided into several regions including unmixed zone in the Ni-base alloy, fusion boundary, and heat-affected zone in the low alloy steel. The result showed the non-homogeneous distribution of elements with higher Fe but lower Mn, Ni and Cr in A533 Gr. B compared with Alloy 152, and the precipitation of carbides along and near the fusion boundary of as-welded and aged dissimilar metal joints. Also, it was found that the precipitation of carbides was enhanced in the aged dissimilar metal welds comparing to that in the as-welded weld material.

Mr. COUVANT, Thierry	Electricité de France (EdF)	Modeling IGSCC of austenitic stainless steels exposed to primary water	Stress corrosion cracking (SCC) has been affecting nickel-based alloys of primary and secondary circuits of pressurized water reactors (PWRs) for more than 20 years. More recently, cases of SCC were also reported for strain hardened austenitic stainless steels exposed to the primary environment. Quantifying and understanding of the involved mechanisms are therefore a key issue, given the costs of controls and replacements of affected components. Many SCC models have been developed so far for alloy 600: • Quantitative models, trying to predict initiation and crack growth rate of the alloy 600 exposed to primary water. These empirical models do not describe physical mechanism and suffer a lack of accuracy: they do not consider the deleterious effect of reverse loading on the susceptibility to cracking, and they are not parameterized by the limiting microstructural factors driving corrosion (diffusion…). • Qualitative models, describing the possible physical mechanisms responsible for degradation: selective oxidation of grain in the case of Alloy 600 exposed to middle primary PWR, assisted by corrosion plasticity in the case of stainless steel exposed to magnesium chloride. Empirical models have the advantage of being operational, at least for the ranking of material susceptibility to SCC. In return, their area of validity is extremely small, most often to the few conditions studied in detail in the laboratory. The weakness of these models is based on the fact that they do not describe the synergy between the physical mechanisms at the root of the cracking. Mechanism-based models allow partly to understand the possible operating phenomena and their interactions. Unfortunately, they do not allow to make any interesting prediction from an industrial point of view: assessment of the time needed to initiate cracks detectable by a non-destructive testing, estimating of the crack growth rate. The current study proposes to fit phenomenological-based descriptions of SCC with quantitative modeling, through the use of experimentation and simulation, in order to make progress in the prediction of the engineering models for initiation and propagation. It is assumed that the simulation of SCC could help to identify and to rank the most relevant physical parameters governing SCC to be considered in engineering models.
Dr. de CURIERES, Ian	Institut de Radioprotection et de Surete Nucleaire (IRSN); France	Corrosion issues in PWR : a TSO perspective	This presentation will provide with the views on current and future issues dealing with corrosion from a TSO perspective

Dr. EASON, Ernest D	Modeling & Computing Services; USA	Modeling IASCC of stainless steels in LWR environments	(Co-author: Raj Pathania, EPRI) EPRI is developing crack growth rate models from a large set of IASCC experimental data on 300-series stainless steels tested in LWR environments. The data were collected in an EPRI IASCC Database and reviewed by an expert panel, as described at previous ICG-EAC meetings. A brief update on the overall effort will be presented, describing data selection for modeling based on expert panel screening and rankings, followed by a discussion of two different irradiation terms currently being considered for the crack growth rate models. The two terms provide similar fits to the data, as will be shown by selected results. The two types of irradiation terms are a fluence term and an irradiated yield stress term. It is well known that above a threshold fluence and below a saturation fluence, yield stress increases with increasing fluence. Other irradiation effects also increase with increasing fluence in that range, including radiation induced segregation (RIS) and microstructural changes, which causes irradiated yield stress, RIS, microstructural changes, and other radiation effects to be interrelated. The rationale for using a function of yield stress instead of a function of fluence to model effects of irradiation is as follows: • irradiated yield stress is a relatively easily-measured variable that can serve as a surrogate for fluence and various fluence-caused effects that are not as simply measured or modeled; • irradiated yield stress can be easily estimated from fluence using appropriate correlations if direct measurements are not feasible; • higher yield stress is associated with higher crack growth rate in both IASCC and unirradiated SCC test data, so yield stress is a useful variable for modeling CGR over a broad range of SCC phenomena, including unirradiated and irradiated conditions; • irradiated yield stress takes into account the different strength levels of types 304 and 316 material, initially cold worked or not and at various levels of fluence, so it can help combine IASCC data on various types of stainless steel with different processing histories, without using additional variables; • and a yield stress model can be simpler.

			The preliminary fluence term is a hyperbolic tangent function of fluence, which has an appropriate form to model the threshold and saturation of the fluence effect, while the preliminary yield stress term is a simple power law. When calibrated to the same NWC data set, the two irradiation terms provide equally good fits as indicated by standard error and residual plots. The flat residual plot for fluence (which is not an explicit modeling variable when yield stress is used) shows that the stronger effects of fluence, including the known effects of RIS and yield stress on CGR, are adequately represented by the model based on yield stress. In low-ECP environments, the two terms also provide similar fits when calibrated to the same data set, though the yield stress term has statistical advantages over the fluence term in low-ECP environments. The exponents on the yield stress terms for IASCC data are similar to exponents fitted to unirradiated yield stress vs. SCC data in NWC and low-ECP environments, supporting the theory that yield stress plays a similar role in both unirradiated and irradiated stress corrosion crack growth rate in various LWR environments.
Mrs. ERNESTOVA, Miroslava	Nuclear Research Institute Rez; Czech Republic	Structure of a WWER dissimilar metal weld	The DMW studied in the R&D project represents a typical safe-end weld of an eastern pressure water reactors (WWER). The ferritic low alloy steel of the hot collector of steam generator and the austenitic stainless steel of the pipes meet at the joint creating a DMW. This aged joint has been found to be susceptible to failures. Therefore, the R&D project focused on accelerated ageing of DMW have been made to ensure WWER long term operation.
Mrs. FIGUEIREDO, Celia A	Centre of Nuclear Technology Development (CDTN); Brazil	Corrosion behavior of Alloy 182 weld in PWR primary water environment	The influence of dissolved hydrogen (DH) on the susceptibility to stress corrosion cracking (SCC) of nickel-base Alloy 182 in simulated pressurized water reactor (PWR) primary water was studied using electrochemical tests and the slow strain rate technique. Two levels of DH were used: 25 and 50 ml H2/kg H2O. The chosen values are the specified range limits of DH used in PWR´s. Analyses of the oxide film formed on surface were performed and in-situ electrochemical measurements were carried out to help understanding the corrosion behavior of the material. The results suggest that the susceptibility of Alloy 182 weld to SCC can be mitigated adjusting the DH concentration in the primary water.

Mr. FOUCAULT, Marc	Areva NP; France	Development of alternating current potential drop (ACPD) method for the detection of primary water stress corrosion cracking initiation in stainless steels and nickel based alloys	This study deals with the development of an ACPD-based method to detect Primary Water Stress Corrosion Cracking initiation time on Stainless Steels and Nickel based alloys. Often restricted to post mortem visual inspections after testing materials in autoclaves, finer examinations are required to investigate SCC crack initiation mechanisms in Primary Water conditions. First tests are performed on Four Point Bent specimens in 316L Stainless Steel and exposed to a boiling Chloride Magnesium solution at 154°C. Complementary tests are carried out with these conditions on several SCC specimens. The ACPD device is a high frequency model CGM-7 (up to 500Hz) supplied by Matelect. A calibration of this apparatus is achieved at different temperatures on micro-machined notched specimens. Comprehensive analytical work is supplied to understand better effects on potential drop of leads positioning, the environment conductivity and the crack configuration. Then, crack monitoring is applied to Primary Water conditions at 360°C using an original Slow Strain Rate Tensile test machine. Application of ACPD is validated by measuring crack initiation on reference Reversed U-Bend specimens (Alloy 600). An in situ Four Point Bending jig is used to test proton irradiated Stainless Steels for IASCC purpose.
		The use of EBSD technique for evaluating cold work	(Co-authors: B. Brugier, O. Calonne) Strain measurement can be performed by numerous techniques, of which EBSD is particularly suitable. This latter offers numerous possibilities of qualitative representations, but also makes possible strain quantification. This work is based on both a rigorous methodology and the assumption of dislocations production (especially geometrically necessary dislocations). This production eminently depends on experimental conditions (mechanical tests), and its analysis, depends on EBSD parametering. Effectively, the “fabrication” of calibration curves requires both exactness and caution at each of the three following stages. The first one is about mechanical tests, the second one concerns the acquisition parameters of the EBSD system and the last one, which is crucial, tackles the post-treatment. The first stage needs the experimental conditions of mechanical tests achievement, and if possible, the plastic deformation mechanisms of the given material. The second stage requires the mastery of acquisition parameters and their fitting to microstructural data (in particular grain size). The last stage, which is maybe the most important source of mistakes, deals with the post-treatment involving the exact knowledge of parametering.

			The accurate quantification of plastic strain requires the calibration curve of the given material et the use of suitable procedures. Numerous applications can be considered, they deal with the study of HAZ, cold work layers and SCC/CF specimens. Erroneous quantifications can be supplied due to wrong applications of calibration methods
Dr. FOURNIER, Lionel	Areva NP; France	Cluster dynamics prediction of the microstructure of austenitic stainless steels under irradiation: application to proton irradiation and the understanding of IASCC	A cluster dynamics model has been developed in order to both understand the important physical variables that influence the evolution of the microstructure of austenitic stainless steel under irradiation, such as temperature, dose rate, and helium generation, and to provide some predictive capability on the basis of a sound grasp of the metal physics involved. The cluster dynamics model has been calibrated in the past against Type 304 low carbon austenitic stainless steel data on Frank loops and SFTs after irradiation in the Bor60 fast reactor at 320°C and 9.4×10-7 dpa/s. The calibration was revisited here considering new results on the microstructure of Type 304 low carbon austenitic stainless steel (particularly on voids) proton irradiated to 10 dpa at 360°C and 1×10-5 dpa/s. The cluster dynamics model was then appplied to post irradiation annealing heat treatment on proton irradiated specimens in order to contribute to the understanding of the mechanisms controlling IASCC.
Dr. FUJIMOTO, Koji	Mitubishi Heavy Industries, Ltd. (MHI); Japan	Corrosion behavior of ion irradiated 316 stainless steel for dissolved hydrogen in PWR	Irradiation (@ 340 Ž, max.10dpa) by using a Ni ion 3MeV, after corrosion test of 1,000 hours, the effect of the texture coating concentrations DH different simulated environment the primary system PWR stainless steel cold-worked 316 to irradiation hardening I have examined. In Japan, there is a directional density reduction DH, from the results of this, from the viewpoint of optimizing the concentration of DH coolant primary system PWR, tends to impair IASCC resistance also reduces the concentration of DH was not observed, especially.
Dr. FUKUYA, Koji	Institute of Nuclear Safety System, Inc. (INSS); Japan	Solute clustering under irradiation in stainless steels	Solute redistribution under irradiation was examined using atom probe tomography in austenitic and duplex stainless steels for general understanding of solute clustering. Clusters with Si, Ni, Mn, Cu, P were identified. The type and formation processes of clusters will be discussed.

Prof. HAN, En-Hou	Institute of Metal Research (IMR); China	Zn injection against corrosion of 316 SS in primary water of PWR	Zn injection against corrosion of 316 SS in primary water of PWR
Dr. HELIE, Max G	CEA; France	Parametric study of PWSCC initiation on cold worked austenitic stainless steels	Experiments were carried out in order to reproduce and characterize the crack initiation on cold worked stainless steels in primary medium. Parametric studies were performed in order to evaluate the impact of different parameters on alloy 316L PWSCC susceptibility: temperature, stress and constant vs cyclic load. The results showed mixed trans/intergranular cracking with a limited role of the temperature in the range 290 to 340°C. Comparison with constant load tests showed that cyclic stresses were more likely to promote crack initiation.
Dr. HUANG, Jiunn-Yuan	Institute of Nuclear Energy Research (INER); Taiwan	Dynamic strain aging behavior of Alloy 600 in a simulated BWR coolant environment	The slow strain rate tests were conducted on cold-worked Alloy 600 at a nominal strain rate 1¡Ñ10-6/sec in simulated BWR coolant environments. The dynamic strain aging phenomenon at 200 oC, 250 oC, 275 oC and 300 oC was studied. The jerky flow on the stress-strain curves for all the cold-worked specimens at 200 oC showed a combination of A- and B-type serrations, but only B type serrations were observed for those tested at the other temperatures. The elongation of the cold-worked specimens increased with increasing the test temperature, but by contrast, the reduction of area decreased with increasing the test temperature. It could be accounted for by the localized deformation induced by dynamic strain aging. The strain hardening exponent decreased with an increase of the cold-worked level and showed little or no dependence on the test temperature.

Mr. ISHIBASHI, Ryo	Hitachi Ltd.; Japan	Effect of carbon concentration on intergranular corrosion and precipitates at grain boundaries of Alloy 82 weld metals	The intergranular corrosion (IGC) behavior of post-weld-heat-treated Alloy 82 weld metals with carbon concentrations of 0.002 to 0.048 mass% was investigated using a modified ASTM G28 test. Significant inhomogeneous IGC behavior due to thermal sensitization was observed at the surface of the higher carbon-containing weld metal in a multi-layer welding. Precipitates at the heat-affected zone in the weld metals were finer than those distant from a fusion line in the weld metals. This result is considered to be caused by inhomogeneous precipitation derived from the welding procedures. M23C6 and MC carbides were detected and observed as major precipitates, whereas MN nitrides were the major precipitates in the 0.002 mass% carbon containing weld metals. The precipitates became progressively smaller as the carbon concentration decreased. Slight shallow IG penetrations were observed although Cr depletion at the grain boundaries was observed. This is noteworthy because the width of Cr depletion and the Cr concentration at the grain boundaries are dominant factors contributing to the IGC.
Mr. ITABASHI, Yu	IHI Corporation; Japan	Evaluation of SCC behavior in BWR water by elasto-plastic parameter	Applicability of elasto-plastic parameter for evaluating stress corrosion cracking (SCC) behavior was examined. Currently linear fracture mechanical approach has been applied for crack evaluation; even though some SCC is found in plastic deformation zone near welding areas where linear elastic fracture mechanics no longer is applicable. Authors have proposed elastic-plastic approach gequivalent stress intensity factor KJ h for evaluating SCC growth rate based on J-integral value, which is valid in both elastic and plastic stress fields. In order to verify the applicability of evaluation by KJ, SCC propagation test in compact tension specimens with varying specimen size and stress conditions were conducted in boiling water reactor environment. As a specimen, sensitized type304 stainless steel was used. When a crack growth rate (CGR) of SCC was evaluated by stress intensity factor K, linear relationship between K and CGR does not exist in the high K value, where CT specimens are out of small scale yielding condition. CGR increases exponentially according to increasing stress intensity factor K to exceed regulation by JSME S NA1. On the other hand when CGR was evaluated by elasto-plastic parameter KJ, linear correlation between KJ and CGR was confirmed regardless specimen size and stress condition. This suggests that by applying elasto-plastic parameter KJ for SCC behavior, CGR at high K value could be estimated easily with using a smaller specimen.

Prof. JANG, Changheui	Korea Advanced Institute of Science and Technology (KAIST); South Korea	Environmental fatigue behaviors of Ni-base alloys in PWR primary water environments	The low cycle fatigue (LCF) behaviors of Alloy 690 and 52M weld, were investigated. The LCF tests were performed in strain control mode with a fully reversed (R=-1) triangular waveform in simulated PWR environments. The fatigue lives in primary water environments are shorter than those in RT air condition, and the degree of decreased fatigue life was smaller than austenitic stainless steels. In primary water, Alloy 690 and 52M weld showed a continuous hardening up to fatigue life, while austenitic stainless steels showed a hardening and softening behavior. Such differences were discussed in view of dynamic strain aging (DSA) responses of the alloys. Also, the roles of corrosion-produced hydrogen on DSA response and LCF life were discussed.
Dr. KAJI, Yoshiyuki	Japan Atomic Energy Agency (JAEA); Japan	Neutron iradiation dose rate effect on deformation microstructure in austenitic strainless steels	Neutron irradiation dose rate effect is one of the concerns in predicting the degradation and the consequent IASCC behavior of structural materials in nuclear power plants using data experimental obtained by the accelerated irradiation tests. In this study, in order to investigate the dose rate effect on mechanical properties of austenitic stainless steels, tensile tests were conducted on type 304SS specimens irradiated at Japan Material Testing Reactor (JMTR) with two different irradiation dose rates. Selected irradiation dose rate levels are about 6x10-17 n/m2/s for high dose rate and about 9x10-16 n/m2/s for low dose rate, while irradiation dose level of these specimens were set the same, about 3.5x1024 n/m2. Tensile tests revealed that increase in 0.2% proof stress was enhanced at high dose rate irradiation, even though the same irradiation dose levels. According to our previous study, it is supposed to be due to the difference in the microstructural evolution caused by the irradiation dose rate. Furthermore, observation of the local misorientation within the grains of deformed specimen was examined by SEM/EBSD technique after tensile tests interrupted at the strain levels 5%, 10% and 15%. Kernel average misorientaiton (KAM) map showed that lattice misorientation tended to localized in the vicinity of grain boundaries in high dose rate irradiated specimens, while spread broader within the grains in the case of low dose rate specimens. Considering results of EBSD analysis on deformed specimens, it is speculated that the irradiation dose rate influence the irradiation microstructural evolution and the following dislocation movement during the deformation.

Ms. KARLSEN, Torill M.	Institutt for Energiteknikk; Norway	Preliminary results from an in-pile crack growth rate study on 1 - 7.7 dpa 304 and 304L stainless steel CTs in BWR conditions	This presentation describes the preliminary results that have been obtained for an in-pile crack growth rate study that is being performed in simulated BWR conditions in the Halden reactor. The test matrix comprises six ¨ùT Compact Tension (CT) specimens that were machined from irradiated core component materials. Crack growth rate data have been obtained for CTs prepared from material from the heat affected zone of a ~1 dpa 304 SS core shroud, a 5.9 dpa 304L SS control blade and a 6.2 dpa 304SS top guide. In addition, Post Irradiation Annealing (PIA) treatments of 500 ¡ÆC / 25 hours and 550 ¡ÆC / 25 hours were applied to two CTs prepared from a 7.7 dpa 304L SS control blade. Crack growth rates have been measured on the samples as a function of applied stress intensity level (~10-15 MPa¡îm), temperature (280 versus 320 ¡ÆC), and chemistry (5 ppm O2 versus 2 ppm H2). The cracking response of the two annealed 7.7 dpa samples are compared with the cracking behavior of a non-treated 7.7 dpa 304L SS CT from an earlier investigation.
Mr. KIM, Young Suk	Korea Atomic Energy Research Institute (KAERI); South Korea	Governing factor for intergranular stress corrosion cracking of Alloy 600 in primary water of pressurized water reactors	The aim of this work is to understand the intergranular stress corrosion cracking (IGSCC) mechanism of Alloy 600, leading to failures of steam generator tubes in pressurized water reactors. Our original idea is that IGSCC of austenitic Fe-Cr-Ni alloys is dictated by the amount of lattice contraction arising from atomic ordering. To this end, Alloy 600 tube has been subjected to different cooling rates after solution annealing at 1095oC for 0.5h to make Alloy 600 tubes with different degree of atomic ordering. The specific heats measured by differential scanning calorimeter (DSC) show that the water quenched (WQ) Alloy 600 is in fully disorder while the air cooled (AC) and furnace-cooled (FC) one is in short range order. Neutron diffraction was applied to determine a change of lattice spacing after isothermal annealing of Alloy 600 at 400oC: the WQ showed the largest amount of lattice contraction when compared to the FC or AC after aging for more than 2000h. Given our idea, the WQ-Alloy 600 with the largest amount of lattice contraction would be the most susceptible to IGSCC. Experimental evidence is provided by citing Yonezawa¡¯s observations that the WQ-Alloy 600 showed 100% failure and the FC- or AC-Alloy 600 zero or much less failure during SCC tests on U-bend specimens in simulating primary water at 332oC for 23,939h. Considering that all Alloy 600 is exposed to the same environment during SCC tests, the IGSCC behavior of Alloy 600 with cooling rate indicates that IGSCC of 600 is governed by the degree of lattice contraction due to atomic ordering. Given this fact, we can account for the early failure of Alloy 600 steam generator tubes only in a Korean nuclear power plant which were made with the highest cooling rate after high temperature mill anneal.

Mr. KIM, Jong Jin	Ulsan National Institute of Science and Technology (UNIST); South Korea	In-situ analysis of aging effect on surface oxide films on Ni-base alloy/LAS dissimilar metal weld interfaces using Raman spectroscopy	In-situ Raman Spectroscopy has been applied to analyze the surface oxide film formed on the dissimilar metal weld (DMW) interfaces of nickel-base alloy/low alloy steel in primary water conditions of pressurized water reactors (PWRs). For the analysis of oxide films in high temperature/pressure aqueous conditions, an in-situ Raman spectroscopy system was developed by constructing a hydrothermal cell where the laser directly contacted the specimen by immersion optics. In-situ Raman spectra were collected for interfaces of as-welded and aged DMW in simulated PWR water conditions at different temperatures up to 300 ¡ÆC. In measured Raman spectra, the peaks of NiCr2O4 were observed in the as-welded DMW, while the peaks of FexCr3-xO4 and NiFe2O4 were observed in aged DMW. It is considered that the change in the oxide chemistry could be originated from the diffusion-induced chemistry redistribution by a thermal aging process.
Dr. KITSUNAI, Yuji	Nippon Nuclear Fuel Development Co. Ltd.; Japan	Typical resluts of IASCC related properties obtained by newly developed test technique	In order to evaluate the IASCC related properties of stainless steels, newly techniques were apilied as follows, -residual stress measurment by XRD with reduced gamma ray effect -chemical analysis of surface oxide by GD-OES -chemical analysis of nano size precipitates by FE-STEM/SDD-EDS -true stress - true strain curves and youngs modulus measurment by using clip type strain gauge The obtained typical results will be presented.
Mr. KOSHIISHI, Masato	Nippon Nuclear Fuel Development Co. Ltd.; Japan	Evaluation of the influence of irradiation dose rate on SCC behavior	Crack growth rates (CGR) obtained from the CT specimens irradiated in JMTR are faster than those evaluated from re-inspection data on core shrouds in BWRs . Dose rates in JMTR are two orders of magnitude higher than those in BWR. To evaluate the influence of dose rate on IASCC crack growth rate, followings are studied. □E Correlations between CGR and hardening, between CGR and RIS □E Dose rate dependence of hardening and RIS □E CGR calculation using slip dissolution/oxidation model

Dr. KUNIYA, Jiro	Tohoku University; Japan	Evaluation of proactive management issues associated with materials aging in light water reactors	A Proactive Materials Degradation Management (PMDM) project has been carried out at the Fracture Research Institute (FRI), Tohoku University for 5 years, as a part of a Nuclear Industries Safety Agency (NISA) project that was formed in 2007 to define an Aging Management Program that addresses unexpected structural material failures in Light Water Reactors (LWRs). Such a program required, therefore, the development of a life prediction capability for specific combinations of degradation modes, structural materials, and reactor componenst.In this presentation, PIRT results for the aging degradation phenomena in LWR structural materials are introduced.
Mr. LIM, Yun Soo	Korea Atomic Energy Research Institute (KAERI); South Korea	Investigation on stress corrosion crack tips of Alloy 600 CRDM tubing material in PWR primary water environment	Primary water stress corrosion cracking (PWSCC) in reactor pressure vessel head penetration nozzles, their welded parts, and steam generator tubes at pressurized water reactors have been found throughout the world. Several models have been reported for the PWSCC phenomena however, the failure mechanisms have not been fully understood up to now. In the present study, the PWSCC cracking properties of Alloy 600 used as the CRDM nozzle material were characterized using microscopic equipment. PWSCC tests were performed at 325 ¡É in a simulated primary water environment of a pressurized water reactor. The microstructural and chemical changes around crack tips were studied using TEM specimens fabricated by a focused ion beam method. It was found that oxygen diffused into the grain boundaries just ahead of the crack tips from the external primary water. As a result of oxygen penetration, Cr oxides were precipitated on the crack tips and the attacked grain boundaries. The oxide layer in the crack interior was revealed to consist of double (inner and outer) layers. Cr oxides were found in the inner layer, with NiO and (Ni,Cr) spinels in the outer layer. Cr depletion (or Ni enrichment) zones were created in the attacked grain boundary, the crack tip, and the interface between the crack and matrix, which means that the formation of Cr oxides was due to the Cr diffusion from the surrounding matrix. The oxygen penetration and resultant metallurgical changes around the crack tip are believed to be significant factors affecting the PWSCC initiation and growth behaviors of Alloy 600.

Dr. LOU, Xiaoyuan	General Electric Global Research Center (GE-GRC); USA	Recent studies on the effect of ppb levels of chloride on the stress corrosion cracking of pressure vessel steel	This work updates the on-going efforts of evaluating the stress corrosion cracking (SCC) behavior of low-alloy pressure vessel steel (A533B) in BWR water. The main focus of this work is to quantify the effect of very low levels of chloride (< 5ppb) on the SCC of low alloy steel. 5 ppb chloride is Action Level 1 in the EPRI water chemistry guideline. Data in this and other programs show that low alloy steel may exhibit an increasing SCC susceptibility at chloride levels below 5 ppb. The crack growth of low alloy steel was studied using periodic partial unload (with long holding time) and constant load tests. The effects of chloride (1 to 5ppb), stress intensity factor (K) and specimen orientation were evaluated. The results suggest that chloride levels at least as low as 3 ¨C 4 ppb can increase the SCC susceptibility of low alloy steel in BWR water.
Prof. LU, Zhanpeng	Shanghai University (SHU); China	Material anisotropy and SCC growth in high temperature water environments	Anisotropic materials properties of nuclear power plant components can be caused by many factors such as manufacturing and fabrication. These properties would affect the SCC path and dynamics. Several SCC systems related to material anisotropy have been investigated. The SCC mode of one-directionally warm-rolled 304NG stainless steel in 288 oC pure water is mainly intergranular in the Transverse-longitudinal (T-L) orientation specimen, while it is occasionally intergranular in the Longitudinal-transverse (L-T) orientation specimen. The crack growth rates in the T-L orientation specimen are higher than those in the L-T orientation in both oxygenated and deoxygenated environments. SCC growth of a 3-directionally cold-rolled 316NG stainless steel in 288 oC pure water were investigated. Crack branching and intergranular stress corrosion cracking along random grain boundaries were found. The cracking paths in the T-L and L-T orientation specimen tend to be along the final rolling direction (L direction). SCC growth rates of Alloy 182 weld metals in T-S and T-L orientations in 288 oC pure water were measured. Extensive inter-dendritic SCC paths have been observed on the side surfaces and fracture surfaces of both T-S and T-L orientation specimens. CGRs of the T-S specimen are significantly higher than those of the T-L specimen under the same test conditions. Crack growth path in the T-S orientation specimen is apparently perpendicular to the loading direction, which appears to be parallel to the loading direction in the T-L specimen. These results showed the strong effect of material anisotropy. Other reference systems are discussed as well.
Mr. MAEGUCHI, Takaharu	Mitubishi Heavy Industries, Ltd. (MHI); Japan	Evaluation of PWSCC initiation lifetime for Alloy 690 PWR components	Constant load PWSCC tests have been conducted for alloy 690, 52 and 152 in a simulated PWR environment at 360 deg.C to assess PWSCC lifetime of PWR plant components. Activation energy for PWSCC was evaluagted by crack growth rate in a simulated PWR primary water environment at 290, 315 and 360 deg.C.

			Based on the constant load PWSCC test result and the obtained activation energy, PWSCC liftime of the plant component was assessed. The assessment shows that alloy 690, 52 and 152 are immune to PWSCC initiation at least for its design lifetime, at operating temperature of 289 to 325deg.C.
Dr. MAENG, Wan Young	Korea Atomic Energy Research Institute (KAERI); South Korea	Influence of the dispersant (PAA) injection on the corrosion behavior of carbon steel	Corrosion products induce the material degradation in the secondary system and depress the thermal performance of the nuclear power plants. Dispersant injection is considered to reduce the accumulation of the corrosion products in the secondary system of PWRs. Polyacrylic acid (PAA) as dispersant is being injected in some power plants. This study is to evaluate the influence of PAA injection on the corrosion behavior of carbon steel in the secondary system of PWR. General corrosion tests were carried out to study the effects of PAA injection on the corrosion resistance of the carbon steels(SA-106 Gr.B et al.) in water at 40~90¡É. Corrosion tests were carried out in water solution of 0~100 ppm PAA concentrations for 14 days. The pH was controlled at 9.7~9.8 by using ETA. Corrosion rate increases as the dispersant concentration increases in the tested environment. While there is no significant increase of corrosion rate up to 10 ppm PAA, the injection of 100 ppm PAA increases the growth rate of oxide layer rapidly and deteriorates the formation of protective oxide on the carbon steels. It is considered that the dispersant influences the growth mechanism of the oxide layer on the steels. The injection of high concentration dispersant should be applied after a careful consideration of the impact on the corrosion integrity of the power plants

Dr. MARTIN, Oliver	EU Joint Research Centre Petten; Netherlands	Stress corrosion cracking testing in super-critical water	The main target of the HPLWR Phase 2 follow-up FP7 project “Fuel Qualification Test for Super Critical Water Reactor” is to make significant progress towards the design, analysis and licensing of a fuel assembly cooled with supercritical water (SCW) in a research reactor which also includes the material research on those in-core materials which could be licensed in near future. In frame of WP4 - Pre-qualification, stress corrosion cracking (SCC) resistance of three selected austenitic stainless steels 08Cr18Ni10Ti (equivalent of 321), 347H and 316L was evaluated. Presented work summarizes in the first part preliminary results of slow strain rate tensile tests (SSRT), performed using a step-motor controlled loading device in an autoclave at 550°C SCW. Susceptibility to SCC was examined by SSRT tests with constant elongation rate of 5.2·10-7s-1 in an autoclave connected to recirculation loop allowing continual water chemistry control during the test. In the second part, the development work of JRC IET and VTT on a new type of loading devices which are expected to decrease these costs and at the same time guarantee enough reliability and flexibility for both SCC and future irradiation assisted stress corrosion cracking (IASCC) testing to be performed in SCW environments, is summarized. Obtained stress-strain curves didn’t indicate any significant change of mechanical properties of the examined materials in 550°C SCW compare to those measured in inert atmosphere. Nevertheless, post-test SEM fractographic analysis showed evidences of SCC initiation areas on main fracture surface and elevated number of secondary cracks along gage section of the tensile specimen. Finally, this paper summarizes the first results of SCC crack growth rate measurements in SCW by using a pneumatic bellows based loading device. Crack growth rates in SCW (550oC, 8000 ppb of dosed oxygen) are compared to those measured by using the same loading device in 288oC subcritical water.
Dr. MEDWAY, Stuart L	AMEC; UK	Effect of cold work on crack growth rates for nickel based alloys A690 and A625 in PWR primary environments	The presentation describes the results of crack growth rate tests on Alloy 690 and Alloy 625 in simulated good quality PWR primary coolant. Alloy 690 crack growth data measured on a number of heats will be compared with those reported by other labs and the variables discussed. For Alloy 625 the data presented will focus on the effect of different levels of cold work on crack growth rates. The presentation will conclude with possible implications of cold work for nickel based alloy disposition curves

Dr. MIYASAKA, Matsuho	Ebara Corporation; Japan	Galvanic reaction between aluminum alloy fuel rack and stainless steel liner of spent fuel pool - simulation by boundary element method	Seawater injection to the spent fuel pool has raised concerns over galvanic corrosion of the fuel racks made of aluminum alloy. For spent fuel pools, both the stainless steel pool liner serving as a boundary and aluminum alloy fuel racks supporting fuel assemblies must be protected from corrosion. Electrical conduction between the liner and the racks may inhibit corrosion of the pool liner with galvanic effects between the materials; at the same time, galvanic corrosion may lead to accelerated corrosion of the fuel racks. Boundary element analysis was performed to identify the risk of galvanic corrosion of the racks caused by a macro-cell formation between the racks and the liner. With the highest conductivity and microbiological effects considered, a maximum corrosion rate of several mm/y was predicted for the ruck edge protruding toward the liner. However, lower conductivity significantly reduces galvanic corrosion. Thus, it has been verified that decreased conductivity and microbiological control through water purification are effective in maintaining the soundness of the racks.
Dr. MORRA, Martin M	General Electric Global Research Center (GE-GRC); USA	Process Parameters Affecting Inhomogeneity of Material Microstructure, Material Quality Then and Now	A study was conducted to examine whether material quality, in terms of microstructure homogeneity, had degraded over time. Changes in the way austenitic stainless steels are processed from the 1960s to present day were documented and correlated with microstructure uniformity. SCC growth rate data will be presented for some of the more common microstructure defects seen in commercial products.
Mr. OGAWA, Takuya	Toshiba Corp.; Japan	IASCC Crack Growth Analysis for BWR Reactor Internals with Distribution of Neutron Flux	Recently, structural integrity assessment method for IASCC has been prepared as gIASCC evaluation guide for BWR core internalsh. However, since the results of structural integrity assessment of BWR core shrouds would be so conservative relative to the actual experiences of IASCC crack growth, the rationalization of IASCC crack growth analysis is desirable. In this study, IASCC crack growth analysis methods considering the residual stress relaxation caused by neutron irradiation and the distribution of neutron flux in core shroud were investigated. And it was found that the methods were effective for the rationalization of IASCC crack growth analysis.
Mr. PEROSANZ, Francisco Javier J	CIEMAT; Spain	On influence of EBSD parameters in plastic strain assessment	Recent studies have shown that crack growth rate of 690 alloy increases at least an order of magnitude over certain degree of plastic strain. Therefore, it is necessary to assess the plastic deformation retained in the material. Among Instrumental techniques conventionally used for this purpose, electron backscatter diffraction technique (EBSD) allows to obtain spatial information about strain location in an easy and quick way. Although the qualitative interpretation of results is immediate, its quantification has still some important open issues.

			In this work the influence of various parameters, some related to the measurement process, and others regarding to the material deformation degree has been studied, in order to understand the relationship between misorientation and plastic deformation. For this purpose two of the most commonly used materials in the nuclear industry, an alloy 690 and 316L stainless steel has been tested. To evaluate the influence of the deformation mode, an alloy 690 was strained using two procedures, tensile and rolling, up to a comparable cold work value around 20%. Regarding to the measurement procedure used in this work, (Local misorientation measures), step size has a strong influence on the shape of the curves misorientation versus strain. Nevertheless it should be noted that different measurement procedures lead to different misorientation values, thus the inter-labs comparison exercise is not easy. Focusing on deformation mode, micro-structure and strength values of the material like yield strength, UTS and elongation show a remarkable difference between the two procedures, while values estimated by EBSD (average misorientation) are only slightly different. A similar behaviour was found for CGR values in primary water.
Mr. QUIRK, Graham P	EDF Energy Nuclear Generation; UK	Environmentally assisted cracking identified in the AISI 316 upper tubeplate radius of a superheater header in the boiler of an advanced gas cooled reactor	(Co-authors: D. Crowle, M. Hobson) During the 2012 statutory outage of one of EDF Energy's Advanced Gas Cooled Reactors (AGR) routine remote visual internal inspection of a AISI 316 superheater header revealed crack-like indications on the upper tubeplate radius. The indications were almost fully circumferential around the tubeplate radius and four representative areas were metallographically prepared and replicated. A montage image of cracking from the replicas clearly showed a predominantly transgranular crack with significant levels of branching. The more central section of the crack appeared to be quite gaping and the crack did appear to be quite oxidised. The mechanism appears to be environmentally assisted cracking (EAC) with a high degree of stress dominance and probably dynamic straining.

During normal steady state operation, the superheater outlet header of the boiler will be exposed to dry steam with a high degree of superheat and is not highly stressed and it is considered that there is no credible mechanism for EAC to initiate or propagate. The 2mm wall boiler tubes also experience the same environment and there have been no on-load failures, supporting this judgement. During operation header creep-fatigue may be possible but this is relatively well understood in terms of crack growth rates and the metallographic evidence does not provide convincing evidence for it. During reactor shutdown conditions the superheater header tubeplate region is considered to be in compression.

However, during a relatively short period during reactor shutdown, it is understood that the superheater header tubeplate region is under high stress, to the extent of dynamic straining, as the boiler cools and materials contract. In addition to this factor, immediately following reactor shutdown, the boilers are flooded forward to flush out residual salts (mainly sodium sulphate) that become deposited from steam within the evaporator and superheater sections. This operation is to remove the risk of localised corrosion of the austenitic superheater under off-load conditions when the boilers are at low temperature, can be water solid with possibly no feedwater polishing capacity nor deaeration, or can be drained with high humidity under nitrogen but there is the risk of air ingress. It has been considered that, under the transient washout conditions at shutdown, effective deaeration of the feedwater was maintained as at full load; however, recent plant monitoring has indicated that short term oxygen excursions can coincide with boiler washout and so the environment becomes supportive of EAC.

Considerable work has been performed in the areas of plant environment characterisation, detailed inspections and header structural analyses. The conclusion is that continued operation is justifiable based upon maximum EAC depth relative to relative to critical defect size (the header is very thick at this location), projected crack growth rates, stress distributions (the stress becomes compressive within the header wall) and time at risk. Nevertheless it may be necessary to operate the reactor with known cracks and a detailed maintenance programme to control this risk has been adopted. This presentation is made with the objective of stimulating discussion from other ICG-EAC members with experience in managing plant with known EAC, for example, boiling water reactors.

[This paper may be considered by the ICG-EAC committee for oral presentation or as a poster.]

Mr. RITTER, Stefan	ICG-EAC Administration; Switzerland	2009, 2010 and 2011 EC minutes	The final, approved version of the EC minutes for the above years.
		2012 EC minutes	Second draft version (incorporating comments from JLN) for approval at 2013 EC meeting
		Pt deposition behaviour on stainless steel under BWR conditions - part I: lab results	On-line NobleChem is a technology developed by General Electric-Hitachi to mitigate stress corrosion cracking in reactor internals and recirculation pipes of boiling water reactors (BWRs) without the negative side-effects of the classical hydrogen water chemistry. For a more efficient reduction of the electrochemical corrosion potential noble metals (e.g., Pt) are injected into the feed water during power operation. They are claimed to deposit as very fine metallic particles on all water-wetted surfaces and to stay electrocatalytic over long periods. To investigate the deposition and distribution behaviour of Pt in BWRs systematic tests were performed in a sophisticated high-temperature water loop at PSI, in which stainless steel specimens were exposed to simulated BWR water. During the tests Pt solution (Na2Pt(OH)6) was injected into the loop and Pt was deposited on the specimens. To study the Pt deposition behaviour and to assess the effectiveness of the OLNC technology under real plant conditions, specimens were also exposed at two locations (mitigation monitoring system and reactor water sample line) in nuclear power plant Leibstadt (KKL, Switzerland). The specimens were then analysed by Laser Ablation-Inductively Coupled Plasma-Mass Spectrometry (LA-ICP-MS) and electron microscopy after the exposure. The current presentation highlights some selected results from the lab tests.
		Pt deposition behaviour on stainless steel under BWR conditions - part II: results from specimens exposed in KKL	On-line NobleChem is a technology developed by General Electric-Hitachi to mitigate stress corrosion cracking in reactor internals and recirculation pipes of boiling water reactors (BWRs) without the negative side-effects of the classical hydrogen water chemistry. For a more efficient reduction of the electrochemical corrosion potential noble metals (e.g., Pt) are injected into the feed water during power operation. They are claimed to deposit as very fine metallic particles on all water-wetted surfaces and to stay electrocatalytic over long periods.

			To investigate the deposition and distribution behaviour of Pt in BWRs systematic tests were performed in a sophisticated high-temperature water loop at PSI, in which stainless steel specimens were exposed to simulated BWR water. During the tests Pt solution ($Na_2Pt(OH)_6$) was injected into the loop and Pt was deposited on the specimens. To study the Pt deposition behaviour and to assess the effectiveness of the OLNC technology under real plant conditions, specimens were also exposed at two locations (mitigation monitoring system and reactor water sample line) in nuclear power plant Leibstadt (KKL, Switzerland). The specimens were then analysed by Laser Ablation-Inductively Coupled Plasma-Mass Spectrometry (LA-ICP-MS) and electron microscopy after the exposure. The current presentation highlights some selected results from specimens exposed in KKL.
Dr. ROTH, Armin	Areva NP GmbH; Germany	Effect of Chloride on General Corrosion and EAC of Low-Alloy Steels in BWR-relevant Environment - Update of Ongoing Study	AREVA GmbH and VGB launched a conjoint experimental program in order to examine the principle effects of permanent or temporary chloride contaminations in reactor cooling water on environmentally assisted cracking (EAC) of a German low-alloy pressure vessel steel in BWR environment. Three project phases aimed to study the effects of chloride on general corrosion using exposure tests (Phase I), on crack initiation by EAC using constantly strained C-ring specimens (Phase I), on crack initiation by EAC during SSRT tests (Phase II) and on crack advance by EAC using pre-fatigued CT-specimens (Phase III). An update of the project progress will be provided in this contribution. Results of Phases I and II will be briefly summarized. New results from crack advance tests will also be shown and discussed.
Mr. SETO, Hitoshi	Nippon Nuclear Fuel Development Co. Ltd.; Japan	Evaluation of prediction method for IASCC susceptibility of BWR core internals	In Japan, the values of neutron fluence are used as the thresholds of the IASCC initiation. However, it is preferred to consider neutron flux for the evaluation of IASCC susceptibility because irradiation hardening and radiation induced segregation, which are both thought to be major factors of IASCC, depend on neutron flux. In this study, the prediction method for IASCC susceptibility on various conditions of neutron fluence and neutron flux was evaluated by combining the prediction method of irradiation hardening and radiation induced segregation, and the data of Slow Strain Rate Tensile test.

Prof. STAIRMAND, John W	AMEC; UK	Studies of environmental fatigue endurance and propagation of stainless steel and carbon steel	The presentation will provide an update on fatigue endurance measurements of stainless steel hour glass specimens with modified surface conditions in high temperature water and in air. In addition, blunt notch compact tension specimen data will be presented for carbon steel and for stainless steel.
Ms. SUMIYA, Rie	Toshiba Corp.; Japan	Effect of Specimen Size on Stress Corrosion Crack Growth Rate at Low K Region	In SCC growth test, a specimen of various size is used depending on a test condition. Specimen size is determined by size of test piece, small scale yielding condition depending on K and plane strain condition depending on K. Few study has been conducted at low K region where the conditions of small scale yielding and plane strain are satisfied enough. In order to evaluate effect of specimen size on SCC growth rate at low K region, SCC growth test and stress analysis were conducted using CT specimens of various size at low K region. As a result, the clear difference of SCC growth rate depending on a specimen size was not obtained at this K level and the clear difference in K distribution depending on a specimen size was not obtained. It is confirmed that it is possible to obtain an appropriate SCC growth rate using various specimen size treated in this study at low K region.
Dr. SUZUKI, Shunichi	The Tokyo Electric Power Co., Inc. (TEPCO); Japan	Current status of 1F and future decommissioning plan	I will present an overview of the current status of Fuku,“hima Daiichi and future decommissioning plan.
Mr. TAKAMORI, Kenro	The Tokyo Electric Power Co., Inc. (TEPCO); Japan	Corrosion problems and mitigation in Fukushima Dai-ichi power station	TBD
Dr. TAKEDA, Kiyoko	Nippon Steel & Sumitomo Metal Corporation (NSSMC); Japan	Effect of microstructure on SCC growth of Alloy 690	To investigate the effect of microstructure on CGR, TT690 CRDM and plate which is controlled the microstructure are studied. The CGRs are estimated under K=25-50 MPam^1/2.
Mr. TAKESHIMA, Kikuo	Japan Nuclear Energy Safety Organization (JNES); Japan	Verification of crack growth rate diagram of low carbon stainless steels	SCC growth rate diagram at hardened heat affected zone (HAZ) for low carbon stainless steels has been developed based on SCC growth rate data obtained from CT specimens. We JNES verified the applicability of SCC growth rate diagram to integrity evaluation of actual plant by comparing SCC growth rate data obtained from the large scales piping.

Dr. TOLOCZKO, Mychailo	Pacific Northwest National Laboratory (PNNL); USA	Comparison of surface microstructures between Alloy 600 and Alloy 690 crack initiation specimens in PWR primary water	Significantly differing crack initiation times are being observed in firsts tests of primary water stress corrosion crack (PWSCC) initiation response of alloy 600 and alloy 690. Whereas crack initiation occurred in 1800-2000 hours in two ~18% cold-tensile strained alloy 600 specimens loaded to their yield stress at 360°C, cold-rolled alloy 690 that exhibited high SCC crack growth rates of 0.6-1x10-7 mm/s has not undergone crack initiation after 7000 hours at the yield stress in the same environment used for SCC crack growth rate testing. Microstructural examination of the alloy 600 specimens in cross section revealed not only the initiated crack but also intergranular attack and short cracks while early observations of cold tensile strained alloy 690 exposed for 1400 hours at its yield stress show no indications of oxidation on the grain boundaries that intersect the surface. Instead, a stable Cr2O3-rich oxide layer forms over the grain boundary at the point where it intersects the surface.
		Stress corrosion crack growth response of Alloy 152/52-carbon steel dissimilar metal welds in PWR primary water	Primary water stress corrosion cracking (PWSCC) tests have been conducted on alloy 152/52-carbon steels welds with cracks positioned either on the fusion line or in the weld near the fusion line with the goal of assessing dilution and fusion line effects on SCC susceptibility. No increased crack growth rate was found when SCC testing a 20% Cr dilution zone in alloy 152M joined to carbon steel that had not undergone a post-weld heat treatment (PWHT). However, high SCC crack growth rates were observed when the crack reached the fusion line of that material where it propagated both on the fusion line and in the heat affected zone of the carbon steel. An alloy 52M-carbon steel fusion line specimen that exhibited similarly high SCC response was given a prototypic low alloy steel PHWT. Constant K SCC crack growth rates subsequently dropped down to ~5x10-9 mm/s showing that the PWHT has a significant effect on SCC susceptibility.
Dr. TOMOZAWA, Masanari	CRIEPI; Japan	Solute segregation on ƒ°3 and random grain boundaries in type 316L stainless steel	In this study, we quantitatively evaluate grain boundary character and misorientation dependence of grain boundary segregation in an unirradiated austenitic stainless steel by means of atom probe tomography. Mo P, B, and C segregated on all the examined random grain boundaries while no segregation or depletion was observed on ƒ°3 boundaries. Degree of segregation on random grain boundaries was not affected by grain boundary misorientation. Large variation of concentration of segregated elements within a grain boundary plane was observed. No co-segregation was observed.

Ms. TOTH, Rebecca	Electricité de France (EdF); France	Water jet peening tests on EDF mock-up	Mitigation processes as Water Jet Peening are studied by EDF to prevent PWSCC on BMI. EDF ran tests on a BMI mock-up treated by Water Jet Peening by MHI to get a description of WJP effect on a representative component. Even though compressive stresses were already observed before WJP due to the stress relieving treatment performed on French reactor vessel, XRD measurements on the outer diameter of the BMI mock-up present bigger compressive and more homogeneous stresses than before treatment. No strain hardening was observed. NRD measurements provided a stress profile in depth and showed how they are balanced. These results raised an issue about the effect of treatment on potential big defects already present in the BMI, since EDF's strategy would be to keep NDT currently used. That issue will be the following study of EDF. NDT Tests were also performed to check the controllability of the component after WJP.
Mr. VANKEERBERGHEN, Marc	SCK–CEN; Belgium	Why not re-visit re-passivation?	In the early days of crack growth rate modelling the film-rupture/dissolution/re-passivation model was developed by Peter Ford for application within the nuclear reactor community. It is well accepted for crack growth in austenitic stainless steel in a BWR environment. It also seems to be plausible for crack growth in austenitic stainless steel in a PWR environment. It is however not accepted as a model for nickel-based alloys. Should it? Here, it will be shown that maybe it could also be a plausible model for crack growth in nickel-based alloys … by making the re-passivation constant a function of corrosion potential. The film-rupture/dissolution/re-passivation model has been modified accordingly and is able to match the peak in crack growth rate observed at low corrosion potential for some nickel-based alloys. Furthermore, a 'corrosion potential'-dependent re-passivation rate might shed light on the issue of enhanced environmentally-assisted fatigue in low corrosion potential environments, PWR and BWR-HWC, with respect to the higher corrosion potential environment, BWR-NWC. And … maybe an 'environment'-dependent re-passivation rate could shed light on initiation time differences between the various LWR coolant chemistries? So, do we re-visit re-passivation?

Prof. WANG, Jianqiu	Institute of Metal Research (IMR); China	Comparison of corrosion resistance of Alloy 690 and Alloy 800	Corrosion behaviour of Alloy 690 and Alloy 800 in simulated primary water containing 2 ppm and 100 ppb dissolved oxygen (DO) are studied by open-circuit potential (OCP) measurements, electrochemical impedance spectroscopy (EIS), scanning electron microscopy (SEM) and X-ray photoelectron spectroscopy (XPS), respectively. The surface films after immersion are characterized by transmission electron microscopy (TEM). Results show that corrosion potentials of Alloy 690 are gradually higher than Alloy 800 with the increase of immersion time, while the electrochemical impedance at low frequency of Alloy 690 gradually decreases and becomes smaller than that of Alloy 800 with the increase of immersion time. On Alloy 690, needle-like oxides with spinel structure are formed in the outer layer and porous NiO oxides are formed in the inner layer. On Alloy 800, the outer layer is granulated spinel oxides, and the inner layer is Cr-containing oxides with mixed crystalline and amorphous spinel structure. Alloy 800 shows a better corrosion resistance than Alloy 690 with the increase of immersion time in the simulated primary water containing 100 ppb and 2 ppm dissolved oxygen.
Prof. WAS, Gary S	University of Michigan; USA	Micromechanics of dislocation channeling and IGSCC in irradiated stainless steels	This talk will focus on characterization and modeling of the micromechanical processes involved in IGSCC in irradiated stainless steel. The emphasis will be on the role of dislocation channels, the nature of their interaction with grain boundaries, and how these interactions may lead to IGSCC.
Prof. WATANABE, Yutaka	Tohoku University; Japan	Overview of CRIEPI committee on anti-corrosion measures for Fukushima Daiichi NPP	Unit 1 to Unit 4 of Fukushima Daiichi Nuclear Power Plant lost all the electric power due to tsunami and cooling function of the spent fuel pools was also lost. Seawater was pumped into SFP to make up water level as part of the emergency action, then, switched to river water. Temporary cooling facilities were installed and cooling operation was started a few months later. CRIEPI Committee on Anti-Corrosion Measures for Fukushima Daiichi NPP was set up to evaluate possible corrosion problems and potential corrosion rate and also to discuss available countermeasures. The major corrosion issues of concern were; - Metal lining of SFP (Type 304 SS)▯| localized corrosion (crevice corr., pitting, SCC) - Fuel storage rack (aluminum alloys)▯| galvanic corrosion - Fuel cladding (Zircalloy 2)▯| localized corrosion (crevice corr., pitting, SCC) - Pipes (carbon steel)▯| uniform corrosion

		Synchrotron X-ray measurements under in-situ environment on the oxide film of type 316 austenitic stainless steel in simulated PWR primary water	In-situ and ex-situ synchrotron X-ray diffraction (XRD) and X-ray fluorescent (XRF) measurement techniques have been developed and used for non-destructive characterization of surface oxide films on Type 316L austenitic stainless steels which were exposed to simulated PWR primary water environments. The effect of DH contents (5 - 30 cc/kg(H2O)) on the layer structures of the spinel oxides for Type 316L stainless steel were obtained under in-situ condition in the simulated PWR primary water.
Dr. YAMAMOTO, Masahiro	Japan Atomic Energy Agency (JAEA); Japan	Corrosion on zircaloy and radiation effect on dissolved oxygen mitigation by hydrazine in Fukushima Dai-ichi spent fuel pool	Seawater was poured into the spent fuel pool (SFP)s of Fukushima dai-ichi Nuclear Power Plant (1F) for emergency cooling. It was concerned about corrosion of metallic materials in SFP. We focused in Zircaloy, fuel cladding materials, to localized corrosion. Measurements of electrochemical procedures showed very low possibility of occurrence the localized corrosion. At 1F SFPs, hydrazine was added to mitigate corrosion. Hydrazine is well-known to decrease dissolved oxygen (DO) concentration in water at high temperature, but this effect weakens at room temperature. It has been reported that radiolysis product of hydrazine react with oxygen molecules to reduce DO in water. Gamma-ray irradiation experiments using 1F SFP modified water was conducted. The results indicated that DO concentration drastically reduced by gamma-ray irradiation. This reaction was not affected by sea water content. So, hydrazine injection to 1F SFP water mitigates corrosion by decreasing DO concentration.

Prof. YEH, Tsung-Kuang (T-K)	National Tsing Hua University; Taiwan	Durability study on zirconium coatings hydrothermally deposited on type 304 stainless steels in high temperature water	A durability study on the integrity of zirconium dioxide coatings hydrothermally developed on Type 304 stainless steels and exposed to 288 oC pure water at various flow rates was conducted. A number of Type 304 SS samples in the shape of a square disk were prepared. They were pre-oxidized in oxygenated pure water at 288 oC for 7 d. The samples were then coated with ZrO2 nanoparticles by hydrothermal deposition at 90 oC for 4 d in pure water with a ZrO2 concentration of 100 ppm. ZrO2-treated samples were later exposed to oxygenated pure water at 288 oC for various periods of time at different flow rates of 30, 60, and 150 c.c./min. Electrochemical polarization analyses were conducted to investigate changes in the electrochemical parameters of the ZrO2-treated samples after the exposure tests. In addition, surface morphologies of the samples before and after the exposure tests were examined by scanning electron microscopy (SEM) and energy dispersive X-ray spectroscopy (EDX). Electrochemical impedance spectroscopy analyses in a selected electrolyte at ambient temperature were also performed to evaluate the charge transfer resistances of the treated samples before and after the exposure tests. It was found that the longest exposure time of 28 d and the greatest coolant flow rate of 150 c.c./min selected did not yield any significant changes in corrosion current density, surface morphology, and charge transfer resistance of the treated samples.
Prof. YONEZAWA, Toshio	Tohoku University; Japan	Stress corrosion cracking susceptibility of strain hardened 316 stainless steel in simulated PWR primary water	In order to clarify the characterization of the IGSCC for strain hardened 316 stainless steel, the IGSCC initiation and propagation tests for strain hardened 316 stainless steel were conducted in simulated PWR primary water under static loading condition with / without pre-crack.
		The effects of metallurgical factors on PWSCC crack growth rate for highly cold worked TT Alloy 690 in simulated PWR primary water	Recently, some papers reported that the PWSCC crack growth rate of highly cold worked TT Alloy 690 was widely changed in heat by heat. In this study, the basic metallurgical characteristics and PWSCC propagation characteristics for highly cold worked TT Alloy 690 are investigated, to confirm the metallurgical factors on PWSCC crack growth rate for highly cold worked TT Alloy 690.

Mr. YOON, Jae Young	Seoul National University; South Korea	Development of equipotential switching array probe direct current potential drop for online monitoring of nuclear piping weldments	The equipotential switching array probed direct current potential drop (ESAP-DCPD) system has been developed to monitor on-line the initiation and growth of a crack in piping weldment. The new technology of equipotential establishment at all borderlines of a monitoring segment eliminates any significant current leakage to outside systems. The thermal fluctuation of DCPD signal has been suppressed by using an already established switching current method. The arrangement of an array probe surrounding the weldment was optimized by performing finite element analysis (FEA) of DCPD in such ways to improve its detectability for circumferential cracks. In order to verify the reliability and detectability of piping weld monitoring, artificial cracks introduced into a full-scale weldment mockup were monitored by using ESAP-DCPD in laboratory environments. Then experimental results were compared with FEA results for the mockup to show a good agreement. Therefore FEA can be used to design a field application system. In order to pursue a first-of-a-kind application in a nuclear power plant, a preliminary licensing basis has been developed through development of an initial 10 CFR 50.59 analysis, as described herein.
Ms. ZHAI, Ziqing	Tohoku University; Japan	Atom probe characterization of segregation behavior in HAZ of RPV steel multipass weld for BWR core shroud support after PWHT and step cooling	The segregation behavior of various microstructures formed in heat affected zone (HAZ) in a model A533B during multipass welding to A182 was investigated by APFIM. Characterization was carried out for as-weld and post-weld heat treated material, as well as the material went through an extra step-cooling sequence. Although it is still too early to make any general conclusion considering the limited size of dataset, segregation of phosphorous was observed at various lath and grain boundaries in different morphology (planar, thin film, etc.) with large fluctuation in amount from site to site. In addition, segregation of carbon seems more prevalent than that of phosphorous, and the two elements often segregate to the same location, interfering a closer look at the interaction between them. A summary on current results of the undergoing characterization will be given in this presentation, and the difficulties as well as the corresponding countermeasures in specimen preparation and data analysis will be discussed.