October 17, 2007

MEMORANDUM TO: Sean E. Peters, Acting Chief
Corrosion and Metallurgy Branch
Division of Engineering
Office of Nuclear Regulatory Research

FROM: Samantha Crane, Materials Engineer ***/RA/ C.E. Carpenter for***
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SUBJECT: SUMMARY OF MEETING BETWEEN THE NRC STAFF, ARGONNE NATIONAL LABORATORY PERSONNEL, AND INDUSTRY REPRESENTATIVES, REGARDING THE ENVIRONMENTALLY ASSISTED CRACKING PROGRAM

On September 25–26, 2007, staff representatives of the U.S. Nuclear Regulatory Commission (NRC) met with personnel from Argonne National Laboratory (ANL) and the Electric Power Research Institute (EPRI), as well as other industry representatives, in a public forum at ANL facilities in Argonne, Illinois. The enclosure to this memorandum provides a list of meeting attendees.

The purpose of this meeting was to coordinate and kick-off a research program to study environmentally assisted cracking (EAC) of light-water reactor structural materials. In so doing, the intention was to avoid unnecessary overlap between NRC and industry programs, while focusing on the most important and timely topics, and sharing information regarding recent, germane results or redirection. The primary objective of this meeting was to present and discuss the recent progress in the NRC-sponsored EAC program at ANL and industry programs related to EAC mitigation, as well as future testing, experiments, and examinations. The focus was to identify and address the most significant gaps, in order to either initiate or further the research in those areas.

The meeting addressed the following topics:

- causes and mechanisms of irradiation-assisted stress-corrosion cracking (IASCC) in boiling-water reactors (BWR)
- causes and mechanisms of IASCC of austenitic stainless steels (SSs) in pressurized-water reactors (PWRs)
- cracking of nickel (Ni) alloys and welds

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Day 1: September 25, 2006

Causes and Mechanisms of IASCC in BWRs

The meeting began with an overview of prior EAC research programs conducted at ANL and the current NRC-sponsored EAC program that began in September of this year. The presentation continued with an overview of prior research efforts regarding the causes and mechanisms of IASCC in BWRs. Topics of discussion included regulatory issues with EAC, current concerns regarding cracking of core internals, short- and long-term objectives, working with industry, and the scope of the research program through 2011.

The discussion proceeded with a review of research performed to date at ANL, which has addressed IASCC susceptibility, as well as crack growth rate (CGR) and the fracture toughness behavior of irradiated SSs and Alloy 690. These studies also investigated the effects of alloying and impurities, grain boundary engineering (GBE), and dose on IASCC. The data were analyzed to determine the threshold and saturation values of materials properties with respect to irradiation.

That discussion was followed by a more detailed presentation regarding IASCC susceptibility of the materials irradiated in the Halden reactor. That research involved evaluating the influence of GBE treatment, alloy composition, and the effects of irradiation dose by performing slow strain rate tensile (SSRT) tests under the normal BWR water chemistry conditions, which include high-dissolved oxygen (high-DO) water at 289 °C (552 °F). The materials examined included Types 304, 304L, 316, and 316LN SSs, as well as Alloy 690. The fracture surfaces were then analyzed to determine the percent of intergranular (IG) fracture, which is used as a measure of IASCC susceptibility. The results of this study indicated that the GBE process did not provide any beneficial effect for Types 304 and 304L SSs. However, Alloy 690 demonstrated a smaller reduction of uniform elongation (UE) at 2 displacements per atom (dpa), thereby reducing its IASCC susceptibility in high-DO water. Type 316 SS was generally more resistant to IG cracking than Type 304 SS, possibly because of the presence of molybnium or a higher Ni concentration. By contrast, a higher oxygen content contributes to IASCC susceptibility by enhancing the initiation of IG cracking, while UE increases linearly with carbon content bellow 0.06 wt-%, showing greater susceptibility to IASCC in low-carbon SSs.

Following the discussion regarding IASCC susceptibility, ANL presented the results of prior studies of crack growth and fracture toughness of irradiated SSs. The objectives of those studies were to better understand threshold fluence, determine disposition curves with respect to fluence levels, and assess the significance of specimen size criteria. The results showed that threshold fluence appeared to be significant for irradiation effects at values around 0.45 dpa, while CGR of SSs irradiated to 4 dpa was 6–8 times the disposition curve presented in NUREG-0313, “Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping.” This presentation appeared to inspire the most discussion regarding the issue of the K/size criterion for irradiated SSs. The expected decrease in CGR was not observed when the DO level was decreased (from 400 ppm to 20 ppm), and loading conditions seemed to have no effect until the DO level was decreased. In addition, the results (including no change in fracture plane) revealed the need to further assess the adequacy of the K/size criterion for irradiated SSs. Those present at the meeting agreed that further studies should be performed.

ANL also tested the fracture toughness in both air and water environments. In particular, those studies showed that neutron irradiation decreased the fracture toughness of austenitic SSs. Also, for irradiated SSs, the toughness of cast SS proved to be lower than that of the heat-affected zone (HAZ) materials, while the HAZ materials showed lower toughness than the sensitized SSs. In addition, both sensitized and cast SS indicated a possible effect of the water environment, which will need further examination. These studies also helped to define a fracture toughness trend curve, which bounds existing data, in terms of J_{IC} vs. neutron dose.

Following the ANL presentation, EPRI representatives discussed their research on crack growth and fracture toughness of irradiated SSs, which included the Corrosion Research Program, BWR Vessel and Internals Project (BWR-VIP), and Materials Reliability Program (MRP). EPRI has also created issue management tables, which will help to classify as many components as possible and, ideally, be able to identify areas where additional research would be beneficial to PWR/BWR operability.

EPRI continued with a discussion of the cooperative IASCC research (CIR) program, in which the NRC is a participating member, and the results of the program's research on irradiated SSs. The primary focuses of the CIR program are to develop a mechanistic understanding of IASCC, derive a predictive model, and identify its possible countermeasures. Some of the findings include the role of initial cold work in retarding IASCC, susceptibility of different alloys as a result of localized plastic deformation in terms of stacking fault energy, and the significant influence of an increase in hardness and tensile strength on IASCC. Other topics of discussion included tensile test and fracture toughness results, and the effects of orientation on fracture toughness. The scope of work has been extended through 2009.

ANL concluded the presentations regarding the causes and mechanisms of IASCC in BWRs with an overview of the scope of its future research. In particular, this research will bring closure to any lingering requirements of the 5-year EAC Program, complete CGR and fracture toughness testing on wrought and weld HAZ materials, and characterize the fracture morphology of CGR and J-R curve specimens. ANL will prepare a draft topical report by August 2008, followed by the final report by April 2009.

Causes and Mechanisms of IASCC of Austenitic SSs in PWRs

To begin the discussion of the causes and mechanisms of IASCC of austenitic SSs in PWRs, ANL presented an overview of its prior research in this area. The purpose of this task was to investigate the modes of degradation of austenitic SS core internals in PWRs, as a function of fluence, material chemistry, and cold work. This research involved testing materials from irradiations conducted in the BOR-60 reactor, including solution-annealed and cold-worked Types 304, 304L, 316, 316LN, and 347 SS; GBE-treated Types 304 and 316 SS and Alloy 690; CF-3 and CF-8 cast SSs; several commercial and lab heats of SSs with low or high sulfur (S) or oxygen (O) content; and Type 304 SS components from EBR-II, which were irradiated in sodium (Na) to approximately 50 dpa at 370 °C (698 °F).

Next, ANL presented the preliminary results of its IASCC study on BOR-60 materials. The idea was to promote further understanding of the behavior of IASCC in PWRs, including the effects of fluence, alloying elements, grain boundary structure, cold-work, and irradiation conditions. Both SSRT and crack growth tests were performed in a PWR environment, and are currently undergoing Transmission Electron Microscopy (TEM) examination. In addition, ANL is analyzing the influence of various irradiation conditions on IASCC behavior by comparing SSRT results of common materials irradiated to comparable dose levels at both the Halden reactor (which has a light-water reactor-type spectrum) and the BOR-60 fast reactor.

Preliminary research indicated that the SSRT testing in high-DO water produced similar results among different materials irradiated in both the Halden and BOR-60 reactors. IG cracking in high-DO water tests for most materials was also consistent for both Halden and BOR-60 irradiations; however, BOR-60 irradiations appeared to be less damaging than Halden irradiations. Overall, the results obtained to date could not conclusively identify whether the temperature effect or the irradiation spectrum effect dominates the difference in the Halden and BOR-60 irradiations. EPRI representatives suggested performing additional tests of Halden and BOR-60 IG cracking susceptibility for quantitative and comparative purposes.

The next ANL presentation addressed the microstructural examination of austenitic SSs and Alloy 690 irradiated to 25 dpa under PWR conditions. These tests were performed under PWR-relevant temperatures and doses, with the primary focus on examining void swelling and the characterization of microstructures. No voids were observed in irradiated SSs, although minimal voids were observed in the base and GBE-treated Alloy 690. A literature review showed that the results of this study are consistent with previous reported studies with respect to the density and size distribution of dislocation loops in SSs and Alloy 690. Future work will include further examination of void swelling in higher doses and the establishment of dose dependence of dislocation loop characteristics.

The EPRI representatives then presented their research on the causes and mechanisms of IASCC in PWR internals materials. This research focused on characterizing the properties of internals materials, including mechanical properties; IASCC susceptibility, initiation, and growth; fracture toughness; microstructure; void swelling; thermal aging; and stress relaxation. Currently, EPRI is conducing six testing projects covering a variety of aspects of IASCC susceptibility. As a result, EPRI has extensively studied a number of mechanisms, and the data obtained (and still to be obtained) will support the development of degradation models and establish the technical basis for numerous screening/evaluation curves for PWR internals components.

ANL concluded this topic with a presentation of the scope of future work. This presentation included the materials and evaluation criteria to be used, with the primary focus areas of IASCC susceptibility of PWR internals materials, void swelling behavior, fracture toughness, and the effectiveness of mitigative measures. In addition, the presentation specified the irradiated material to be used, task scope, and significant milestones.

Day 2: September 26, 2007

Cracking of Ni Alloys and Welds

To begin the discussion of cracking of Ni alloys and welds, ANL presented information regarding its prior research efforts in this area. The task objectives consisted of providing technical data and analytical methods on cracking, compiling and reviewing available data, developing laboratory crack growth data for confirmation purposes, and providing a validated methodology for predicting CGRs in Ni alloys and welds.

The discussion continued with a presentation on metallographic examinations, performed using chemical analysis, optical and scanning electron microscopy (SEM), and orientation imaging microscopy (OIM). This study investigated the Alloy 600 control rod drive mechanism (CRDM) nozzle 3 and Alloy 182 J-groove weld from nozzle 11 at Davis-Besse Nuclear Power Station, as well as the Alloy 82/182 hot-leg nozzle-to-pipe weld from Virgil C. Summer Nuclear Station. For this study, ANL prepared additional Alloy 182 laboratory welds to determine the effects of grain boundaries and orientation on crack propagation. This examination revealed that the grain size for Alloy 600 is in the range of 25–200 µm (~1–8x10^{-3} inches) [compared to the average size of 75 µm (~3 x10^{-3} inches) identified by the American Society for Testing and Materials (ASTM)], with grain boundary carbide coverage of 50–60 percent. Titanium- (Ti-) rich precipitates were observed in both grain boundaries and the matrix, with average yield strength of 280 MPa (40,610 psi) and ultimate tensile stress of 548 MPa (79,480 psi). Alloy 82/182 and laboratory-prepared Alloy 182 welds showed typical dendritic structures with niobium- (Nb-) rich precipitates in the grain boundaries and matrix. In addition, Ti-rich precipitates were observed in both the grain boundaries and the matrix of the ANL laboratory welds.

The meeting continued with a presentation regarding CGRs of Ni alloy welds in primary water. The materials used in this study were the same as those for the previous study (above), with the addition of laboratory-prepared Alloy 152. Using those samples, ANL performed CGR tests in a representative PWR environment to facilitate the transition from transgranular fatigue cracking to intergranular SCC before performing the constant load test. Results for Alloy 182 showed that SCC CGRs are consistent with other laboratory data, with activation energies of 252 KJ/mol for double-J welds and 189 KJ/mol for deep-groove welds. However, field welds appeared to have lower CGRs than laboratory-prepared welds, and the SCC CGR results for Alloy 152 were as high as 5.4 x 10^{-11} m/s for Kmax = 30.2 MPa-m$^{1/2}$.

Next, ANL presented a discussion regarding CGRs of Alloys 600 and 690 in PWR water. In this study, Alloy 600 (taken from Davis-Besse nozzle 3) showed very high CGRs with IG fracture mode during pre-cracking. Those CGR results were unexpected because of the alloy's average strength and grain boundary carbide coverage (50–60%). By contrast, the Alloy 690 tests demonstrated uniform fracture surfaces for both specimens. In addition, the cyclic CGRs of both Alloy 600 and Alloy 690 appeared to be environmentally enhanced in PWR water at temperatures around 316 °C (600 °F).

EPRI/MRP representatives then presented an overview of their Mitigation and Testing Integrated Task Group Research and Development (R&D) Programs for PWR Reactor Coolant System (RCS) Materials. In particular, this overview included Alloy 690/52/152 CGR testing, CGR "Low-K" testing, and Alloy 600/690 HAZ CGR testing. The primary objectives of these programs were to quantify all margins of improvement, and document the additional results in a revision of MRP-111, “Material Reliability Program: Resistance to Primary Water Stress-Corrosion Cracking [PWSCC] of Alloys 690, 52, and 152 in Pressurized-Water Reactors.” Overall, the Phase 1 results revealed that neither Alloy 690 nor its weld metals are immune to crack growth through SCC in PWR primary water of nominal composition. Crack morphology in base metal is predominantly intergranular; while weld metal fracture surfaces are highly unusual (with the SCC region very flat and featureless). Phase 2 included further testing of the same materials, with the addition of Alloy 152 and 52 welds. As a result of the large body of sometimes inconsistent CGR data that exists for the Alloy 690/52/152 system, EPRI is currently evaluating the data and will summarize the results in a white paper.

Currently, the Boiler and Pressure Vessel Code promulgated by the American Society of Mechanical Engineers (ASME) implies a CGR value of zero for K factor values <9 MPa√m because of the unavailability of adequate supporting data. This has sparked interest in further understanding of PWSCC CGRs at low K values. In particular, EPRI learned that related work was already being performed in laboratories outside the United States, and contacted those organizations to assess the potential for cooperation. Specifically, the testing partners are Studsvik of Sweden and AREVA of Germany. The EPRI representatives then briefly presented information regarding Alloy 600/690 HAZ CGR, addressing the concerns attributable to localized deformation close to fusion lines. The recent findings in this area could potentially challenge the CGR disposition curve presented in MRP-55, “Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress-Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials.”

Finally, ANL concluded this topic with a presentation of the scope of future work related to cracking of Ni alloys and welds. In that regard, the primary objective is to provide the NRC with technical data and analytical methods on the cracking of Ni alloy components and welds. Such data and methods would facilitate independent estimates of CGRs in reactor components for regulatory determinations of residual life, inspection intervals, repair criteria, and effective countermeasures for reactor internal components. Alloys to be tested will include those previously used (i.e., Alloy 690/152 welds and 690 HAZ), with the addition of CRDM Alloy 690TT and Alloy 152 welds on CRDM Alloy 690TT. In addition, ANL presented the scope of work and testing schedule through 2011. ANL and the NRC are in the process of discussing the possibility of including several variations of Alloy 52 in this study.

The meeting adjourned at 4:00 p.m. on September 26, 2007.

Enclosure:
As stated

in a revision of MRP-111, "Material Reliability Program: Resistance to Primary Water Stress-Corrosion Cracking [PWSCC] of Alloys 690, 52, and 152 in Pressurized-Water Reactors." Overall, the Phase 1 results revealed that neither Alloy 690 nor its weld metals are immune to crack growth through SCC in PWR primary water of nominal composition. Crack morphology in base metal is predominantly intergranular; while weld metal fracture surfaces are highly unusual (with the SCC region very flat and featureless). Phase 2 included further testing of the same materials, with the addition of Alloy 152 and 52 welds. As a result of the large body of sometimes inconsistent CGR data that exists for the Alloy 690/52/142 system, EPRI is currently evaluating the data and will summarize the results in a white paper.

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Enclosure:
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NAME	I. Anchondo	S. Crane	S. Crane	S. Peters	S. Peters
DATE	10/ 17 /07	10/ 17/07	10/ 17 /07	10/17 /07	10/ 17 /07

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LIST OF ATTENDEES:

Kick-Off Meeting
for the ANL Program on Environmentally Assisted Cracking of LWR Structural Materials

September 25–26, 2007

Argonne National Laboratory
Room DL-208, Building 212

Argonne National Laboratory

Omesh Chopra	EAC program
Bogdan Alexandreanu	EAC program
Yiren Chen	EAC program
William Soppet	EAC program
Bill Shack	EAC program
Ken Natesan	Section Manager, Corrosion & Mechanics of Materials
Chris Grandy	Department Manager, Engineering Development & Appications
Hussein Khalil	Director, Nuclear Engineering Division

U.S. Nuclear Regulatory Commission

Samantha Crane
Robert Tregoning
Gene Carpenter
Hipolito Gonzalez
Al Csontos
Isaac Anchondo
Mauricio Gutierrez
Jay Collins

Electric Power Research Institute

Al Ahluwalia
Raj Pathania
H.T. Tang

Industry Representatives

Jean Smith	Exelon Corp.
John Wilson	Exelon Corp.
Michael Burke	Westinghouse
Robin Dyle	Southern Nuclear Operating Co.
Duane Snyder	GE Hitachi Nuclear Energy
Joe Rashid	Anatech
George Theus	Dominion Engineering
Dave Rudland	Engineering Mechanics Corp.
Alexander Butcavage	Constellation Energy
Steve Fyfitch	AREVA
Michael Wright	Atomic Energy of Canada Limited (AECL), Chalk River
Charles Moreau	AECL, Chalk River