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Proactive Material Degradation Research Subjects for Light Water Reactors

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ABSTRACT

Predictive and preventive maintenance technologies are increasingly of importance for the long term operation (LTO) of Light Water Reactor (LWR) plants. In order for the LTO to be successful, it is essential that aging degradation phenomena should be properly managed by using adequate maintenance programs based on foreseeing the aging phenomena and evaluating their rates of development. In this paper, topics that are judged necessary for managing aging degradation phenomenon in LWR structural materials and which were discussed in the Proactive Aging Management Experts' Panel Meeting of Tohoku-Hokkaido Research Cluster of NISA Project are introduced

KEYWORDS

Proactive materials degradation assessment, Proactive materials degradation management, LWR, BWR, PWR, Plant Aging, Long Term Operation, Stress Corrosion Cracking, Flow Assisted Corrosion

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1. Introduction

A project on Proactive Materials Degradation Management (PMDM) has been carried out at Fracture and Reliability Research Institute, Tohoku University for four years, as a part of the Aging Management Program of the Nuclear Industries Safety Agency (NISA) that was started in 2007. The objectives of this project are to evaluate potential and complex degradation phenomena and their mechanisms in order to identify future risks of component aging in nuclear power plants. The following items are of particular concern in this project. (a) Investigation of potential materials ageing phenomena and corresponding plant issues, and (b) Investigation of the effectiveness of evaluation techniques concerning potential aging phenomena.

Similar to the PMDM approach of the US Nuclear Regulatory Commission (USNRC) [1], a Phenomena Identification and Ranking Table (PIRT) approach was adopted. For this purpose, a panel of international experts on materials degradation in light water reactor (LWR) were nominated and assigned to this task. One major difference between the present PMDM project and the USNRC PMDM is that the preferred approach in the current PMDM project is based upon a more fundamental understanding of the scientific bases of materials ageing. Two approaches are considered to be an essential for proactive aging management. One is a deductive scientifically based method and another is an intuitive one based on a careful analysis of operating experiences. In particularly, the deductive method based on a fundamental scientific understanding of material degradation is paramount for the management of possible latent materials degradation processes which have not yet become obvious in operating plants.

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Many of these fundamental and managerial issues impact on current concerns associated with, for example, flow accelerated corrosion, stress corrosion cracking (SCC) of nickel-base alloys in primary systems of pressurizes water reactors (PWRs), and irradiation assisted SCC (IASCCC) of stainless steels in core structures of boiling water reactors (BWRs) that were initially prioritized in 2007. Later, in 2010, proactive management issues associated with materials aging in LWRs were discussed in terms of suggested research topics that should be undertaken in either the short or long term. Based on these discussions prioritized lists of medium and long term research projects were established for both PWRs and BWRs.

In this paper, the research subjects to be considered for the aging degradation phenomena in LWR structural materials that were discussed at the Proactive Aging Management Experts' Panel Meeting of the NISA Project are introduced.

2. Research Subjects for Basic Research on Environmentally Assisted Degradation of BWR Materials.

2.1 Research subjects for shorter term issues to be completed in ~5 years in BWRs

2.1.1 Definition of the Metal/Environment Interface and Precursor Events

[Background]

It has long been recognized that the environmentally-assisted cracking susceptibility of structural steels and nickel base alloys in BWRs is governed by a significant number of material, stress and environmental parameters and that there are secondary interactions between these parameters. For instance, both the heat input during welding and the fast neutron flux/fluence will affect not only the grain boundary composition in stainless steels, but also the residual stress profile adjacent to welds and the corrosion potential.

It is obvious that, if the kinetics of EAC are to be predicted, then the above materials, environment and stress inputs must be adequately defined, as well as their interactions. Although such system definitions have been developed for many of the critical parameters (for example, the extent of chromium depletion adjacent to grain boundary carbides as a function of heat treatment), this is not the case for all parameters and their interactions.

[Scope]

The scope of this particular RFP focuses on the quantification of critical items (e.g. corrosion potential, anionic activity, stress distribution, material composition) that are central to the prediction of environmentally-assisted degradation in stainless steel, nickel-base alloy and carbon or low alloy steel BWR components. Such quantification involves a basic understanding of, for example, the dislocation morphology and plasticity at crack tips, and the thermodynamics and kinetics of electrochemical reactions and metallurgical transformations. However for practical purposes these system conditions must be redefined in terms of measureable or calculable parameters such as stress, strain, heat treatment, water purity, component geometry, etc.. This is often difficult. For instance, the effect of cold work on the cracking susceptibility of stainless steels and nickel-base alloys in BWRs may be governed by a wide range of interacting physical metallurgical phenomena, and it would be very difficult to formulate these interactions in terms of a fundamentally important parameter such as the strain rate at a crack tip. At the very least, however, it should be possible to “bin” combinations of these engineering conditions that have a similar effect on the fundamentals of crack initiation and

propagation.

2.1.2 Quantification of Crack Initiation for Unirradiated Structural Alloys in BWRs.

[Background]

There are various definitions of "crack initiation" ranging from the initiation of isolated micro-cracks at pits or sites of inter-granular corrosion, to the detection of deeper cracks by a variety of non-destructive testing procedures conducted on the reactor component surface. The interrelationship between these definitions is illustrated in Figure 1. The definition of crack initiation adopted in this RFP is the time, t_O , for a dominant crack to be formed by the coalescence of separated micro-cracks. A fairly extensive number of investigations indicate that, for stainless steels, nickel-base alloys and carbon/low alloy steels in BWR environments, such coalescence is completed when the dominant crack depth, a_o , is in the range 50-300μm . The subsequent crack propagation rate is approximately the same (at the same nominal stress intensity factor) as that associated with propagation rate studies on very much deeper cracks, well in excess of the approx.1000 μm associated with crack detection. Thus, there is an analytical pathway to quantifying lifetime predictions for BWR components.

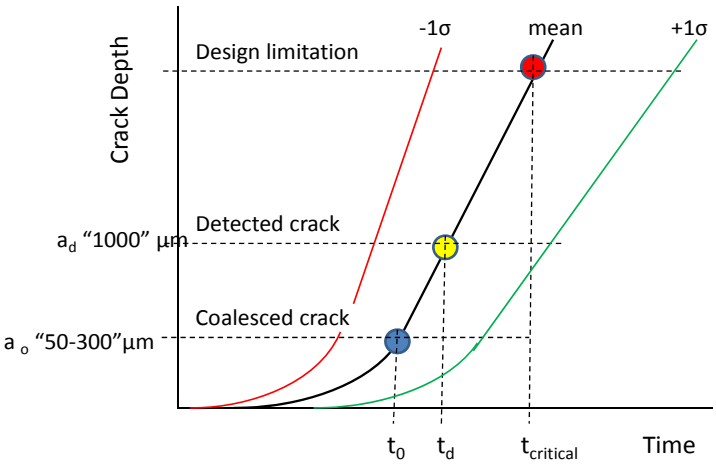

Figure 1. Schematic crack depth vs. time relationships associated with the definitions of crack "initiation" and propagation

It is widely accepted that many of the sub-phenomena that lead to a dominant coalesced crack (such as, inter-granular attack and pitting) are stochastic in nature, and hence, as illustrated in Figure 1, there will be an expected distribution in t_o values associated with the specific a_o value appropriate for a given alloy environment combination. Moreover this dispersion is carried through to evaluations of the "engineering initiation" time, t_d, and to the time when the first crack reaches some design criterion. There have been numerous hypotheses advanced to explain the variations in "initiation time", and these have ranged from the relatively simple explanation of crack initiation due to inter-granular attack of a chromium-depleted grain boundary in sensitized stainless steel, to more involved hypotheses to explain crack initiation on the cold worked surface of non-sensitized stainless steels. These latter hypotheses have concentrated primarily on wrought stainless steels, rather than nickel-base alloys, and have centered on the role of the oxide (composition and structure) and complex changes in metal microstructure immediately below the metal oxide interface (e.g.

recrystallized nano-dimension grain size, slip bands/twins), and strain localization at grain boundaries.

Hypotheses to explain the early micro-crack initiation and coalescence for carbon and low alloy steels have followed the much earlier work in lower temperature/concentrated environments, which focused on micro-crack initiation in pits and the spatial aspects of the pits that enabled crack coalescence. Preliminary observations on such ferritic alloys under fatigue loading in the higher temperature BWR environments indicate that a similar phenomenon is occurring, provided the pits have been initiated at lower temperatures. However this hypothesis for carbon and low alloy steels has yet to be fully quantified. A more recent hypothesis has been advanced for cold worked low alloy steels based on the effect of absorbed hydrogen (produced by flow accelerated corrosion at adjacent geometrical discontinuities) leading to creep crack initiation and growth. This is a distinct possibility for PWR secondary side components operating at higher temperatures, and has yet to be evaluated for BWR components.

[Scope]

The aim of this research subject is to (a) directly measure the values (and dispersion) of the parameters a_0 and t_o shown schematically in Figure 1, and (b) to correlate these observations with micro-structural changes in the surface region.

2.1.3 Development of Qualified Mechanisms-Based Algorithms for EAC Propagation in Unirradiated Structural Alloys.

[Background]

There have been extensive reported investigations over the last 30 years to develop prediction models for environmentally-assisted crack propagation (i.e. stress corrosion cracking and corrosion fatigue) in structural alloys in the BWR RCS. These developments were motivated by a need to understand (and potentially simplify) the interpretation of the extremely complex material, environment and stress interactions. Most attention in this regard has been given to unirradiated stainless and carbon/low alloy steels with further preliminary developments for nickel-base alloys, and irradiated stainless steels.

The mechanism of crack propagation that has been studied the most (primarily at General Electric) is the slip oxidation model. The central hypothesis is that crack advance may be directly related to the oxidation kinetics (i.e. dissolution and oxide formation) at the straining crack tip where the thermodynamically stable oxide is being periodically ruptured. Under such conditions the crack propagation rate (da/dt) may be related to a formulation of the general form;

$$da/dt = A\,(d\varepsilon/dt)^{n}$$

where A = a constant that includes the value n, the fracture strain of the oxide at the crack tip and Faradays constant

$d\varepsilon/dt$ = the crack tip strain rate, which controls the oxide rupture frequency and which may be empirically formulated in terms of calculable engineering parameters, such as stress intensity factor, etc.

and n = parameter that quantifies the oxidation rate on the bared crack tip created by the rupture of the crack tip oxide. This value is a function of the degree of chromium depletion at the grain boundary the corrosion potential at the crack mouth and the water conductivity.

Developments at the Fracture Research Institute (FRI), Tohoku University have expanded on these earlier developments at General Electric in terms of the details of the oxidation process and, in particular, the formulation of the crack tip strain rate at an advancing crack tip. The developments on the former topic are still in the formative state, but the developments in the latter topic have given insight into the reasons for the enhanced cracking susceptibility due to cold work observed in non-sensitized alloys and under more reducing environments.

Very similar predictions have been made using the slip oxidation hypothesis for stress corrosion crack propagation in carbon and low alloy steels and these have been accepted by the USNRC for supporting disposition relationships for pressure vessel steel

[Scope]

In spite of these promising agreements between observation and theory, there remain issues that put to question some of the basic science assumptions behind the General Electric and FRI prediction models. These need to be addressed, and in particular;

(1)The nano-meter scale of the crack tip opening and the effect that this has on the mass transport controlled oxidation reactions at the crack tip. The current crack propagation rate prediction algorithms are based on reaction rates measured in simulated crack tips with unrestricted fluid access.

(2)The (re)formulation of the crack tip strain rate to take into account the yield stress in the *bulk* metal. Such formulations have been developed using strictly empirical (General Electric) and analytical (FRI) approaches. The assumption in this development is that it does not matter how the yield stress is changed (e.g. cold work, irradiation damage, heat treatment). This seems rather over-simplistic and needs more justification.

(3)The formulation of the crack tip strain rate that takes into account *localized* microstructural and plasticity phenomena . Such phenomena include;

(a) the role of stress intensity factor gradients (i.e. dK/da). Positive values of dK/da are observed to give significant increases in the crack propagation rate, which may be relevant during the growth of short cracks.

(b) the role of stress/strain transients on the creep rate

(c) the role of absorbed hydrogen on creep rates.

(d) the role of dislocation morphology and (microscopic) strain localization. The dislocation morphology and how it is changed by such factors as the stacking fault energy, coherent precipitates, grain boundary sliding, etc. has long been recognized as being relevant to stress corrosion cracking. Its effect is understood *qualitatively* in terms of changes in the crack tip strain rate, but there is, as yet, no quantitative correlation.

It is possible that the resolution of these basic science issues will uncover situations where there are compensating errors in the current working hypothesis. However, it is likely that the *basic* hypothesis is correct in principle, since it gives a reasonable prediction of observed SCC propagation susceptibilities for stainless steels (and carbon/low alloy steels) over a wide range of system parameters. If this presumption is correct then this hypothesis should be expanded to cover the prediction of crack propagation in;

- Stainless steels in irradiation environments (e.g. core internals).

- The prediction of crack propagation under changing load and environmental conditions.
- Nickel base alloys (e.g. 600,182,82) that are used in safe ends, attachment welds, access hole covers.
- Stress corrosion crack propagation of *irradiated* low alloy steels.

2.2 Research subjects for Long Term Issues to be completed in 5-10 years in BWRs.

2.2.1 Prediction of Stress Corrosion Cracking of Stainless Steels in BWR Core Internals

[Background]

The SCC susceptibility of the irradiated components in BWRs responds to the various system (i.e. material, stress and environmental) parameters in much the same manner as has been observed for unirradiated stainless steels. For instance, the cracking susceptibility in both cases is dependent on the tensile stress, corrosion potential, anionic impurity concentration and the grain boundary chemistry. Thus, it is reasonable to propose that IASCC is not a new mechanism of cracking, but is merely mirroring irradiation-induced changes to the rate-controlling parameters in the various phases in crack development described earlier in sections 3.1.2. and 3.1.3. Such irradiation-induced changes would be to the corrosion potential, tensile residual stress, grain boundary composition and yield stress. Those changes that are driven by neutron flux (i.e. corrosion potential) will have an effect immediately on the cracking susceptibility, whereas other fluence-driven changes (such as to material composition, stress relaxation and yield strength) will have an effect over an extended time, albeit over periods that are relevant to current BWR operating times.

It has been hypothesized, therefore, that the slip-oxidation mechanism for crack propagation is the relevant cracking model for irradiated stainless steels in BWRs. The task, therefore, is to identify the roles of neutron flux, fluence and gamma irradiation on the corrosion potential, applied stress, the yield stress and the grain boundary chemistry.

Programmatically, the specific effect of irradiation on the fundamental parameters that control crack propagation in stainless steels may be quantified via consideration of the following;

- Corrosion potential and how this changes with radiation flux.
- Irradiation-induced changes in grain boundary composition.
- Irradiation-induced hardening and, for displacement loaded structures, stress relaxation.

The results of preliminary investigations into such irradiation-induced alterations to the slip-oxidation prediction model for crack propagation stainless steels have been reported. For instance, it is apparent that the crack depth is predicted to change with time (fluence), in a manner that reflects the time-dependent reduction in weld residual stress (which decreases the cracking susceptibility) and the increase in irradiation-induced grain boundary chromium depletion (which increases the cracking susceptibility). In this case the total grain boundary chromium depletion is attributed to any initial thermally-induced chromium depletion at the grain boundary due to the welding process plus the additional depletion that occurs with fluence at that particular point in the shroud. In addition, the yield stress will increase with fluence and will increase the cracking susceptibility due to its effect on

the crack tip strain rate. Thus, it is apparent that the crack growth rate is not constant but will vary, mirroring these competing effects of stress relaxation, increasing yield stress and grain boundary sensitization, etc.. (Note that this observation immediately imposes a fluence/time limitation on the quantification of crack propagation rate vs. stress intensity factor disposition relationships)

It is apparent from the discussion above that there is a basis for predicting crack propagation of irradiated stainless steels. However, exactly as with concerns about IASCC in PWR core components, there are potentially extra issues at irradiation fluences circa5 to 10dpa which might be encountered under BWR lives of 60-80 years and which are not experienced at lower fluences. As an example, there is increasing evidence that silicon segregation may be of importance. Silicon enrichment at the grain boundaries increases with fluence and is potentially of concern because many stainless steels containing 0.5 – 1% Si can enrich to >5% at the grain boundary. Indeed, since the compositional measurements are generally made by analytical electron microscopy with a 1 – 2 nm beam size, the actual Si concentration at the grain boundary might approach 50 atomic percent. Crack growth rate measurements on stainless steels with elevated Si levels (e.g., 1.5 – 5% Si) show high growth rates and limited or no effect of stress intensity factor and corrosion potential. This observation may help to explain the loss of the benefit of lowering the corrosion potential at high fluence in some stainless steels, especially since Si enrichment appears to continue after Cr depletion saturates. Unfortunately, there is no *quantitative* life prediction capability for this effect at this time.

[Scope]

The basic science issues, (apart from the more "applied science" concerns associated with the validation of radiolysis/ corrosion potential models), relate to the definition of the grain boundary properties and how these alter with fast neutron fluence. Concerns centre around the fundamental understanding of the role of silicon segregation, hardening, helium bubble and late blooming phase generation on IASCC. This understanding should be linked to the development of the IASCC propagation model described above coupled with developing better models of microstructural development under irradiation. Such microstructural models should take into account interstitial diffusion phenomena and the effects of neutron spectrum on nuclear transformation reactions leading to helium and hydrogen generation. Emphasis in EAC testing should be on both initiation and crack growth testing with attention in the former case as an extension of Section 3.1.2 on the role of irradiation on surface hardening, oxide formation and degradation prior to crack formation. The role of channel formation during deformation and interaction with grain boundaries should also be investigated, with reference to the definition of the crack tip strain rate algorithm.

2.2.2 Life-Prediction Capability for EAC of Structural Alloys in BWR Components

[Background]

Various approaches have been taken in the past to predict the life of a component subject to environmentally-assisted cracking. These have included; (a) analyses of failure times for specific components, (b) statistical approaches for evaluating the role of the different stress, material and environmental "drivers", (c) the evolution of empirically-based prediction algorithms and, (d) in a few

cases, the development of life-prediction algorithms that are based on an understanding of the more fundamental phenomena occurring at the metal/environment interface. All of these approaches have limitations either because they do not recognize the chronology of physical events, such as a precursor time, microcrack initiation at grain boundaries or pits, microcrack propagation and coalescence and, finally, macrocrack propagation or, if they do, the resulting prediction algorithms do not adequately quantify in local and global terms the specifics of these phenomena.

A widely reported approach, which might act as a basis for future lifetime predictions, is that taken for predicting the life of stainless steel piping. In this case it was assumed that an intrinsic defect of depth 50μm was present at the time of reactor commissioning; this assumption is in line with the conservative observation. It was further assumed that the resultant growth of that crack was predicted via the slip oxidation model (section 3.1.3.) using the relevant stress intensity/crack depth relationship for the classification of the piping system and the relevant water conductivity, corrosion potential and sensitization. The range in crack depths noted at one reactor (operating under very (by modern standards) impure water conditions are predicted, with the maximum crack depths corresponding to the predictions for the maximum residual stress for that 28" recirculation piping system. By comparison, as predicted, no cracks were observed (i.e the crack depths did not exceed the NDT detection limit) during the examination period. As expected, the time for the cracks to reach a detectable depth corresponding to a quarter wall penetration was a function of the water purity, with the precise dependency being a function of the residual stress profile and the degree of sensitization. Such a prediction capability leads to the ability to predict future degradation kinetics as a function of, for instance, the planned operating conditions.

[Scope]

The objective is to build on the conclusions arising from Sections 3.1.1, 3.1.2, and 3.1.3, and to develop life prediction algorithms for stainless steels, nickel-base alloys and low alloy steels that are subject to environmentally-assisted cracking in BWRs under a variety of operating modes. The achievement of this objective requires;

- The experimental determination of the mean values of a_0 and t_0 in Figure1 that relate to the achievement of a dominant coalesced crack. There will be definable aleatory uncertainties associated with these parameters, which are inputs to;
- The propagation rate algorithm for the specific alloy/environment that is relevant to a given BWR component. The epistemic uncertainties associated with this sub-objective relate to questions concerning the completeness of the prediction algorithm and the uncertainty in the inputs to that algorithm.

It is emphasized that this overall task can start immediately, using currently available models and model inputs (or assumed ranges of these inputs). Success is therefore quantifiable in terms of predicting the mean and dispersion of laboratory and/or plant data.

2.2.3 Development of an in-situ monitoring /diagnostics and a risk-informed management capability for BWR components subject to EAC.

[Background]

Two developments are technically achievable based on the proposed work described in the previous sections, both of which are compatible with proactive management of materials degradation of structural materials in BWRs. These interelated developments are; (a) a real-time, in-situ monitoring capability and (b) a risk-informed management capability.

[Real-time , in-situ monitoring capabilities]

Real-time, in-situ monitoring capabilities have been developed for other industries including the fossil power generation, aircraft and transport industries. There is no reason in principle that they cannot be developed and used to manage materials degradation issues in BWRs.

This capability requires;

- System definition “tools", associated with; radiolysis and corrosion potential models; models associated with the definition of the material condition that are dependent on the fabrication process; etc.
- System definition monitors placed in accessible positions that act as calibration points to the model. Such monitors include corrosion potential and crack growth rate specimens.
- Life prediction models for specific material / environment / degradation mode combinations.

Elements of all these requirements are already available, or are in development as described in the previous sections.

The use of such a monitoring and diagnostics capability leads to;

- The instantaneous definition of changes in the "frequency" input to a risk analysis that might be associated with , for instance, changes in material degradation due to unplanned changes in the reactor operating conditions
- Assessments of "factors of improvement" (FOI) relating to potential changes in the environment.
- Prioritization of inspection actions based on current degradation predictions

Although preliminary versions of such an "M&D" integrated system may be tested now, there are significant obstacles before "commercial" application would be possible. These include;

- The qualification of radiolysis and corrosion potential models, with appropriate global to local transition capabilities.
- The development of long life, reliable corrosion potential monitors.
- The development of non-destructive, in-situ, residual stress monitors and methods to quantify the local strain condition.

[Risk-informed management capability that address time-dependent materials degradation]

Risk-informed safety analyses have been in place primarily in the US since the 1980s for regulation and operational decisions (e.g. inspection, maintenance, pressurized thermal shock). These methodologies, which combine traditional deterministic safety analyses with probabilistic risk assessment, need to be extended to provide a risk-informed safety margin characterization that takes into account materials degradation that occurs over time.

3. Research Subjects for Basic Research on Environmentally Assisted Degradation of PWR Materials

3.1 Research subjects for shorter term issues to be completed in ~5 years in PWRs

3.1.1 Stress Corrosion Crack Initiation Phenomena, including the Effect of Surface Stress/Strain, Residual Stress, Microstructure and Strain Localization

[Background]

Crack initiation is inherently difficult to study because of the stochastic nature of the processes that usually requires many specimens to be tested under identical conditions in order to determine the mean (and standard deviation) of the response. Nevertheless, with proper attention to specimen preparation with the objective of ensuring that all known observable surface characterization parameters are held constant, significant progress is considered attainable even when testing only modest numbers of replicated specimens, say ~6 to determine mean behavior with a reasonable level of statistical confidence.

The two SCC phenomena where most attention is needed in relation to characterizing initiation response are Primary Water Stress Corrosion Cracking (PWSCC) of nickel base alloys, particularly Alloy 600 and the weld metal Alloys 132/182, and Inter-granular Stress Corrosion Cracking (IGSCC) of cold worked stainless steels, also in nominal PWR primary Water conditions. However, in the second case, the role of water chemistry transients on the process is an additional variable requiring clarification. In addition to these two principal topics, SCC of carbon and low alloy steels (C&LAS) and external surface SCC (mainly of austenitic stainless steels) are additional topics meriting attention in the context of long term plant aging.

[Scope]

Existing research results have clearly revealed the importance of surface condition in relation to the probability of PWSCC initiation in nickel base alloys in addition to the usual parameters of stress, corrosion potential, temperature and the inherent variability in susceptibility of individual heats of material. However, there is dearth of quantitative information relating to the characterization of surface condition and effects on crack initiation probability. Among the parameters that could be assessed are surface residual stress, near-surface microstructure including dislocation density, grain boundary carbide coverage, grain boundary coherency, and strain localization.

The position with SCC initiation in cold worked stainless steels is less clear in that almost no work has been published on IGSCC initiation in PWR primary water other than via the dubious means of slow strain rate tests. Essentially the same parameters as those suggested above for nickel base alloys could be studied with the addition of the effect of transient oxidizing conditions of varying duration that typically occur at lower temperatures than normal operating temperature and are strongly suspected as being implicated in IGSCC initiation. Periodic slow dynamic straining is another parameter requiring fundamental assessment.

In the case of C&LAS, considerable attention has been focused in the past on the influence of transient oxidizing conditions and of transient dynamic straining on SCC initiation (and propagation). However, some recent plant events have drawn attention to the possibility of crack initiation in cold worked materials either at low corrosion potentials or even on external pipe surfaces, possibly due to the flux of hydrogen from internal surface corrosion and flow assisted corrosion. On the other hand, external surface SCC of austenitic stainless steels caused by chloride contamination is a very well characterized phenomenon and it is difficult to see what further fundamental work is required.

3.1.2 Development of Qualified Mechanisms-Based Life Time Models for PWSCC Propagation in Nickel Base Alloys

[Background]

The challenge here is the absence of a consensus on the mechanism of PWSCC propagation as well as an absence of mechanistic explanations for the considerable variability that is observed between different heats of Alloy 600 or different Alloy 132/182 weld metal deposits. Some considerable advances have been made by the Tohoku Fracture group in the application of modern solutions for the stress/strain fields in front of crack tips combined with various hypotheses for the mechanism by which cracks may advance in PWR primary water environments. The latter include the classical application of Faraday's law to metal removal by dissolution/oxidation at a crack tip or embrittlement mechanisms involving oxygen. Similar approaches to hydrogen embrittlement effects have been explored by M. Hall and colleagues.

[Scope]

In the shorter term, it would appear preferable to examine modeling of a single system such as PWSCC growth in Alloy 600 and endeavor to distinguish between the dominance of different potential mechanisms both theoretically and experimentally. The objective, therefore, is to predict quantitatively the effects of all the major known variables of corrosion potential (i.e. hydrogen concentration in PWR primary water), stress, temperature, cold work (including strain path), and carbide morphology. From the existing models incorporating the three proposed rate determining processes at crack tips involving oxidation rates as given by Faraday's law, oxygen diffusion in the crack tip oxide and/or metal, and hydrogen diffusion/accumulation in the crack tip plastic zone, the crack growth rate as a function of K_I should be forecast and then tested methodically in the laboratory. The influence of corrosion potential and microstructure are perhaps the most revealing in terms of likely observable differences between the various mechanistic hypotheses for PWSCC growth. Another important issue is the behavior of long term aged materials, which could include materials removed from operating plants during the course of component replacements.

3.1.3 Characterization of Weld Metals, Dilution at interfaces, and Heat Affected Zones

[Background]

This topic concerns primarily dissimilar metal welds between RPV low alloy steels and stainless steel safe ends (both wrought and CASS) where the weld metals may be Alloys 132/182, 82, 152 or 52 or Type 309/308 stainless steel. The concern is for the toughness and stress corrosion resistance of the various metallurgical discontinuities within dissimilar metal welds and how these vary with welding practices. In particular, the residual stresses and microstructures of interfaces between weld metals and base metals require characterization and ultimately correlation with toughness and resistance to SCC, particularly in PWR primary water in the current context.

[Scope]

For this project, an alliance with a qualified fabricator of nuclear components is essential in order to cover the correct ranges of welding methods, welding parameters, weld profiles and weld constraint. Subsequent metallurgical characterization is by classical optical and electron microscopy coupled with determination of residual stress profiles by X-ray diffraction and hole drilling techniques. An examination of the risks of hot cracking of the weld metal and its dependence on weld metal

composition and other welding parameters could be studied concurrently.

It would also be desirable to correlate this metallurgical information with measurements of fracture toughness of the various interfaces and their resistance to PWSCC. The latter could be achieved by four point bend specimen exposures and crack propagation rate measurements in simulated PWR primary water. The optimum temperature for the maximum acceleration of testing would have to be determined as this may not necessarily be the same for all the materials used for dissimilar metal welds.

3.1.4 Strain Localization, Strain History and Relationship to Cold Work, Microstructure and Compositional Banding

[Background]

Cold work, particularly in one plane due to rolling, is known to provoke PWSCC susceptibility in wrought thick section Alloy 690 and austenitic stainless steels that are otherwise extremely resistant to cracking. The issue is confused by discrepancies between different reports of the importance or not of carbide banding and its interaction with cold work, especially in wrought Alloy 690 plates. There is, therefore, a need to systematically examine the relationship between banding, cold work in the same plane as the banding, and PWSCC propagation rates in that plane. In the case of Alloy 690 there is also a question concerning the differences to be expected for relatively fine grained, steam generator tubing in the mill annealed and thermally treated condition compared to wrought products.

[Scope]

At first sight it would not seem feasible to procure one plate of Alloy 690 with pronounced carbide banding and process it to remove the banding but feasibility could be examined in more detail. Failing that, two separate plates should be procured, with and without pronounced carbide banding, and characterized by classical optical and electron microscopy. In addition, strain localization at various microstructural features should be quantified using EBSD techniques.

Further cold work up to typically 40% cold reduction should be performed on the two types of microstructure and from which CT specimens are subsequently manufactured with the crack plane at different depths through the half-thickness. The aim will then be to carry out PWSCC growth rate measurements and compare the effect of different microstructures.

Material from Alloy 690 steam generator tubing in the mill annealed and thermally treated condition should also be compared in crack growth tests and combined with microstructural characterization and comparisons.

3.2 Research subjects for Long Term Issues to be completed in 5-10 years in PWRs.

3.2.1 Effects of Irradiation Flux and Fluence on Stainless Steels and Nickel Base Alloys and the Effects on EAC

[Background]

Changes in elemental composition adjacent to grain boundaries of austenitic alloys, particularly stainless steels, under neutron irradiation have been well characterized and documented. They typically include depletion of chromium, iron and molybdenum in a band about 5 to 10 nm wide either side of the grain boundary and corresponding enrichment in nickel and silicon. The evolution in grain boundary composition is observed between neutron doses of 1 to 10 dpa and tends to saturate at

neutron doses of ~10 dpa.

Modeling of the vacancy driven inverse Kirkendall mechanism of atom migration under a continuous neutron flux is also well advanced for the so-called over-size atoms like chromium, iron and molybdenum. However, the corresponding grain boundary enrichment mechanism of the so-called under-sized atoms such as nickel and silicon that probably proceeds by an interstitial jump mechanism is much less well developed and there is no known theoretical model in the literature that has been calibrated with the experimental observations.

There are also concerns over the possible differences between fast reactor and thermal reactor irradiated microstructures due to neutron spectrum and flux effects, in particular the generation of large quantities of helium and molecular hydrogen trapped in nanometer scale bubbles observed in thermal reactor irradiated materials and the extent to which grain boundaries may be preferential sites for bubble formation. There is no clear picture of how these processes affect EAC (and mechanical properties) of, in particular, PWR core structural materials. The effect of late blooming phases at higher irradiation doses is another poorly characterized phenomenon in this respect.

Depletion of chromium at grain boundaries is particularly detrimental for IASCC in the more oxidizing environment typical of BWRs on Normal Water Chemistry without any deliberately added hydrogen. However, PWR operating experience also shows IASCC susceptibility of irradiated austenitic stainless steels in the same range of neutron doses even though chromium depletion is of no particular consequence at the low corrosion potentials typical of PWR primary circuit materials and in the aqueous phase of BWRs on hydrogen water chemistry. In PWRs, IASCC susceptibility may continue to increase with even higher neutron doses above 10 dpa characteristic of some parts of PWR internal core support structures and suggestions have been made that silicon segregation in particular may be detrimental. Thus, the mechanism of IASCC in hydrogenated reactor coolants like PWR primary water and BWR hydrogen water chemistry is not properly elucidated. More generally, the interaction between grain boundary elemental segregation and matrix hardening induced by neutron irradiation plus the formation of nanometer scale helium gas bubbles and precipitation of 'late blooming' phases in highly irradiated materials typical of PWR core supports and their effect on IASCC susceptibility remain to be elucidated.

Research work on highly irradiated and activated austenitic alloys like stainless steels and nickel base alloys requires a particularly long term commitment of resources both material and financial. It is also essential for fundamental studies that a university establishes a working alliance with a properly equipped hot laboratory that can devote the resources and manpower required to develop and commission the in-cell equipment required. Links to the fusion reactor community may also be usefully developed, in particular to take advantage of developments in techniques such as co-injection of energetic ionized hydrogen and helium.

[Scope]

The lines of approach suggested at this stage of the development of the knowledge of IASCC in order to improve fundamental understanding are to examine the role of silicon segregation, hardening, helium bubble and late blooming phase generation on IASCC. This should be linked to the development of an IASCC model based, if possible, on a mechanistic foundation and coupled to developing better models of micro-structural development under irradiation taking into account interstitial diffusion phenomena and the effects of neutron spectrum on nuclear transformation

reactions leading to helium and hydrogen generation. Emphasis in EAC testing should be on both initiation and crack growth testing with attention in the former case on the role of irradiation on surface hardening, oxide formation and degradation prior to crack formation. The role of channel formation during deformation and interaction with grain boundaries should also be investigated.

3.2.2 Modeling and Validation of Residual Stress/Strain Profiles in Complex Welded Geometries and how these may Change with Neutron Fluence

[Background]

In many respects, this topic is more an engineering problem of determining residual stresses in complex three dimensional engineering components and how these re-arrange due to the action of irradiation creep and other physical changes induced by neutron irradiation. However, an important element of such assessments concerns the details of weld residual stress/strain profiles on a more local scale in welded joints and relaxation by irradiation creep. It is the irradiation creep aspect of this topic and its interaction with other irradiation induced phenomena such as void swelling that necessitate longer term study. A crucial uncertainty is how effective stress/strain relaxation is in a thick section weld where the dose/flux dependence of irradiation creep varies significantly though the section thickness. There is also a significant overlap with BWR interests where the long term values of residual stress are important for life assessments of core shroud structures.

[Scope]

Just as with the proposed study of residual stress/strain profiles in thick section dissimilar metal welds (PWR Outline Shorter Term Proposal 4), an alliance with a qualified fabricator of nuclear components is essential in order to cover the correct ranges of welding methods, welding parameters, weld profiles and weld constraint for austenitic stainless steel welds. Metallurgical characterization is carried out by classical optical and electron microscopy and coupled with determination of residual stress profiles by X-ray diffraction and hole drilling techniques, in this case before and after neutron irradiation. Since only a few designs of PWR core support internals have welded structures close to the nuclear fuel, the irradiation doses relevant to most welds in austenitic stainless steel are relatively small in most PWRs but nevertheless in a range where relaxation by irradiation creep can be significant. This study could be carried out in tandem with the one described in the preceding outline proposal.

3.2.3 Initiation of SCC in Structural Alloys, Modeling Stochastic Features, Heat-to-Heat Variability, Effects of Long Exposure Periods

[Background]

There is an undoubted dearth of information relating to the effects of long term exposures and stochastic features of crack initiation in systems relevant to PWRs such as PWSCC of Alloy 600 and the weld metals 132/182 and IGSCC of cold worked stainless steels. Such information is required to validate long term predictive models, identify any hitherto unanticipated environmentally induced degradation phenomena, and bridge the gap to shorter term research work that is typically carried out over periods of weeks or months. Another system requiring examination over the long term is IASCC of reactor internals but this subject is dealt with under a separate heading related to neutron irradiated austenitic alloys.

Evaluating SCC in structural alloys in order to quantify the effects of long term exposures and stochastic features of crack initiation automatically entails a significant long term commitment to exposure tests of many specimens under nominally identical test conditions for long periods of time measured in years. It also implies the duplication of significant numbers of test facilities and/or development of test rigs capable of testing many specimens simultaneously. A parallel activity requiring the cooperation of a suitably equipped hot laboratory capable of handling radioactively contaminated artifacts and collaboration with a utility prepared to give access to components removed from service would involve detailed examinations of oxide films and surfaces for signs of environmentally induced deterioration or precursors likely to give rise to crack initiation such as grain boundary oxidation and grooving and pitting.

[Scope]

The main systems considered important to study in the current context of very long term aging in PWRs are PWSCC of Alloys 600, 132, 182 and 82 plus comparisons with Alloys 690, 152 and 52, and IGSCC of cold worked austenitic stainless steels. This should include both the effects of bulk cold work and surface cold work prior to tensile loading in the test facility and the effects of welding defects such as hot cracks. While the mean behavior of any given SCC system under nominally the same conditions can be determined with reasonable statistical confidence from less than 10 nominally identical specimens, evaluation of the standard deviation would require around 30 nominally identical specimens for each test condition.

Given present experimental capabilities identified in the published literature, it is probably feasible to test around 30 small O-ring or C-ring specimens in chains in one experimental autoclave. Monitoring displacements in such chains gives information on time to failure of individual specimens although electrical continuity tests may also be feasible. Another technique based on testing multiple tensile specimens where a small identity ball is displaced on failure of any given specimen is another possible approach. However, tensile tests usually preclude the very high tensile strains/stresses that can develop locally in bending without plastic collapse and which are often the site of crack initiation. The precise experimental strategy adopted would clearly require careful evaluation and prototype testing.

In parallel, it is also proposed to examine any components removed from long term PWR service. The aim would be to perform careful surface and cross-sectional examinations to search for signs of long term deterioration at the oxide/metal interface likely to lead to crack initiation.

4. Research Subjects for Basic Research on Environmentally Assisted Degradation of BWR and PWR Materials in common.

4.1 Research subjects for shorter term projects to be completed in ~5 years.

4.1.1 Effect of Environment on Fatigue and Fracture Resistance in BWRs and PWRs.

[Background]

The influence of LWR aqueous environments on fatigue endurance curves and fatigue crack propagation rates for austenitic stainless steels and carbon and low alloy steels have received considerable attention over the last three decades. The main phenomenological responses to parameters such as cyclic frequency, strain rate during tensile loading, hold times, and corrosion potential (i.e. as established in BWR NWC and HWC and PWR primary water conditions) have been

well characterized for fatigue crack propagation. By contrast, little progress has been made in fundamental understanding of the role of environment on fatigue crack initiation and growth, especially in austenitic stainless steels and nickel base alloys, although somewhat better fundamental understanding exists for carbon and low alloy steels where the role of sulfur based non-metallic inclusions is known to predominate and affect localized dissolution/oxidation at cyclic strain concentrations.

Two hypotheses invoking either anodic dissolution/oxidation at sites of cyclic strain concentration or hydrogen effects for austenitic materials have been proposed and neither gives satisfactory correlations for the effects of corrosion potential. Consequently, empirical algorithms have been imposed by regulatory authorities in Japan and the USA for austenitic stainless steels and nickel base alloys to take into account the effect of the environment on fatigue, particularly for the design process to assess fatigue endurance. These algorithm have uncertainties in their long term application and particularly in their application to complex strain histories associated with plant transients coupled with realistic engineering surface finishes.

By contrast, the assessment of the environment on fracture resistance is a relatively new concern where the basic phenomenology is still being developed for the complete range of LWR structural materials. The topic can be divided into two parts; one is concerned with the apparent drop in J_{Ic} and tearing resistance during J-R tests of nickel base alloys at slow extension rates in PWR primary water compared to conventional tests in air; the second concerns sudden ruptures of fracture mechanics test specimens used for stress corrosion crack growth and corrosion fatigue crack growth testing in PWR primary water at normal operating temperatures between 280 and 345°C and in BWR environments at 288^0C.

In both cases, the strong presumption is that hydrogen absorbed from PWR primary water or BWR HWC is responsible for reductions in fracture resistance. Note that even in oxygenated BWR NWC, a significant fraction of the cathodic reaction is supported by hydrogen evolution (in addition to oxygen reduction) that can be detected diffusing across the walls of thin capsules of stainless steel containing high temperature oxygenated water. The hydrogen embrittlement hypothesis is reasonably well supported by (a) results of J-R type tests conducted with different hydrogen concentration in the water, and (b), the correlation between increased susceptibility to reduced fracture resistance and the presence of interfaces for hydrogen agglomeration such as semi-continuous carbide precipitates on grain boundaries.

[Scope]

The main requirement for further work in both of these areas of LWR environmental effects on mechanical properties of LWR structural materials, especially on fundamental mechanisms, comes from the need to determine the mechanism for the environmental effects on fatigue that will give confidence in the long term reliability of current assessment methods used during the design process, or modify them as appropriate. In the case of fatigue crack initiation and propagation in austenitic stainless steels and nickel base alloys, there is a real need to establish the basic mechanisms of environmental degradation of fatigue resistance. This is particularly the case in BWR HWC as well as PWR primary and secondary conditions where the role of hydrogen concentration and corrosion potential requires particular attention and characterization since these more reducing conditions are known to be more detrimental for fatigue resistance than more oxidizing conditions encountered in

BWR NWC. This concern is highlighted for BWRs when considering that many of these reactors are adopting more reducing environments associated with "hydrogen water chemistry" and noble metal technology.

In the case of fracture resistance, it is necessary to confirm the hydrogen embrittlement hypothesis, to determine whether this is responsible for the sudden specimen ruptures at normal operating temperatures mentioned above, and to determine a quantitative relationship between the concentrations of hydrogen absorbed (since there will be gradients of hydrogen concentration in most components of practical interest), the relevant micro-structural features, and fracture resistance. The proposition that the loss of fracture resistance during J-R testing is primarily due to environmentally assisted cracking at the slow extension rates usually employed also needs to be confirmed since the observation of sudden ruptures argues in favor of an intrinsic effect on mechanical properties due to exposure to the aqueous environment. As well as work on the nickel base alloys 600 and 690 and their weld metals, there is also a need to determine to what extent the phenomenon extends to stainless steels, especially cast stainless steels (CASS) that may be thermally embrittled, and C&LAS.

4.1.2 Flow-Accelerated Corrosion in BWR and PWR Systems

[Background]

Flow-accelerated corrosion (FAC) describes enhanced corrosion that can occur in flowing water or steam under single- or two-phase conditions. It is associated with changes in the mass-transport controlled oxidation and reduction electrochemical reaction rates occurring on an oxidized surface next to the flowing water or steam. FAC is most commonly observed in carbon steels used in the secondary and tertiary circuits of PWRs and in the feedwater systems and steam lines of BWRs. It is often associated with geometrical discontinuities or abrupt changes in flow direction where high turbulence is encountered. Typical metal losses from carbon steels can easily be >1mm/year for single phase flow velocities >6 to 10 m/s at 200°C.

FAC can also occur on copper-base alloys used predominantly in condensers and for some tertiary cooling water system components such as pumps. However, the use of copper-base alloys has been widely phased out in favour of titanium alloys where seawater is used for cooling or stainless steels where river or lake water cooling is used. Thus the scope of this section is confined to carbon steels.

The consequences of FAC include: (a) outright failure of the component due to thinning until the system pressure or stress can no longer be supported, (b) significant contribution to the feed water iron content, (c) an increase in radiation levels in the balance-of-plant of BWRs and, (d) the fouling of flow measurement devices and ion exchange resins by released iron oxides. Extreme examples of the consequences of FAC are the ruptures of an 18 inch diameter carbon steel condensate line on the secondary side of the Surry-2 PWR in 1986 and a 24 inch diameter carbon steel steam line between the low pressure heater and deaerator at the Mihama-3 PWR in 2004.

There are extensive data to indicate that the following system parameters, and their combinations, , are of importance in determining the extent of FAC in carbon steels in water and steam;

- temperature
- oxygen and hydrogen concentration (corrosion potential)
- water chemistry and pH
- chemical composition of the substrate metal

- single or two phase flow and turbulence.

This information has been sufficient to develop mitigation actions based on, for instance; (a)alloying additions (chromium >0.15-0.2wt.%); (b) adding oxygen >20-30ppb to BWR feedwater lines, and (c) controlling the pH(25°C) in PWR secondary systems close to 10.

In addition to developing mitigation actions, this phenomenological knowledge has been used for decisions relating to inspection priorities. Such an instance is the CHECWORKS™ prediction code developed by EPRI. This code was developed after an extensive study of the factors controlling FAC in LWRs. For each of the parameters affecting the phenomenology of FAC described above, empirical predictive algorithms were developed. The code is widely used by LWR operators to predict the most likely areas that may experience FAC, as well as the rate of metal loss in order to fix inspection priorities. CHECWORKS™ is claimed to be accurate to within a factor of 2 when compared with plant measurements, with a confidence level of 98%. A more recent evaluation of this code indicates, however, some discrepancies between observation and prediction.

An empirical approach to determining inspection priorities was also developed by NISA, following the Mihama 3 accident. Thinning rates were examined by measuring the thicknesses of ~20,000 components at 23 PWRs and, with very few exceptions, the guidelines that had been developed as early as 1990, were found to be conservative. BWR data were also examined and found to exhibit lower corrosion rates than PWRs. This was attributed to water chemistry differences. Based on the PWR inspections, the 1990 inspection rules were, nevertheless, revised with the inspection priorities being governed by combinations of steam quality, velocity and temperature.

A similar empirical approach is taken in the COMSY code used in Germany, which is based on all the known experimental and plant data on FAC. This code incorporates detailed modeling of the plant geometry and thermal hydraulic characteristics and then, based on the water chemistry and material compositions, evaluates the zones at risk from FAC. For sub-systems identified as vulnerable to FAC, a detailed analysis is performed to provide life predictions for individual components. An integrated inspection management module enables inspection data to be incorporated as they become available and optimize the inspection scope, locations and intervals.

The BRT-CICERO™ software developed by EDF is based on a physical model of ferrous ion transfer between the boundary layer in equilibrium with magnetite reduction and the ferrous ion concentration in the bulk water. The water chemistry and temperature are taken into account through the equilibrium ferrous ion concentrations in the boundary layer and that in the bulk water as well as the thermal hydraulic parameters via the mass transfer coefficient. Material composition parameters affect primarily the oxide thickness and porosity in the model. The BRT-CICERO model was used experimentally by EDF on a few plants in the early 1990's but became mandatory for all EDF PWRs after the discovery in 2001 of severe FAC on a reducer at the Fessenheim Unit 2 that had been correctly predicted by the model. Since then the model has been tested on over 6000 individual pipe thickness measurements taken over the last 20 years on nearly 4800 different pipe elements in 58 plants. Recent improvements to the thermodynamics database, chemistry calculations (including dissolved oxygen), alloy chemistry (specifically chromium), steam void calculation and many geometrical factors for components have been updated. The agreement between observation (on more than 14,000 measurements) and prediction are good; moreover where discrepancies do occur the predicted metal loss is conservative in 93.3% of the comparisons.

A less well known physical model was originally developed for the British Central Electricity Generating Board to address FAC issues in the once through boilers of Advanced Gas Cooled reactors. It has much the same features as the EDF model but, crucially, also takes into account the affect of flow on the cathodic reaction rate and the lowering of corrosion potential that ensues at high fluid velocities. This gives rise to a distinctly non-linear dependence of FAC kinetics at high mass transfer rates such that, in the specific AGR boiler problem addressed, the FAC rate depended on fluid [velocity]$^{1.4}$. This extra effect of flow on the cathodic reaction seems to be an important effect requiring much more attention.

[Scope]

It is apparent from this background that caution must be given to any prediction capability that is based solely on empirical data, without the support of a mechanistic base. This is especially the case for situations, such as EAC and FAC where the inputs to the phenomenon involve many, interwoven parameters. Such cautions are doubly important when attempting to *extrapolate* a prediction capability beyond the extent of the empirical data base.

4.2 Research subjects for longer term projects to be completed in ~5 to 10 years.

4.2.1 Quantification of Potential Synergistic Effects between Competing Degradation Modes for both PWRs and BWRs.

[Background]

Any study of material degradation mechanisms involving thermal or irradiation induced embrittlement automatically involves a very long term commitment of time and resources. In addition, long term thermal aging cannot be unduly accelerated by using high temperatures since there is a limit of about 400°C above which the physical aging mechanisms change.

The material where potential superposition of more than one aging process could be important is cast austeno-ferritic stainless steel (CASS) and stainless steel weld metals containing delta ferrite. Large quantities of these materials are in use for the primary pipework, valve bodies and core support structures beneath the core of PWRs and BWRs and welded structures. Several variables affecting the degree of thermal aging of CASS and stainless steel weld metals have been extensively studied, such as the percentage ferrite, chromium, molybdenum, and carbon contents, for their effect on fracture toughness. However, very few studies have addressed the potential stress corrosion susceptibility of thermally aged material in PWR or BWR primary water environments and even fewer the combinations with radiation damage.

[Scope]

The objective will be to study at least two heats of CASS with high susceptibility to thermal aging and one with medium susceptibility. These materials would be thermally aged, probably at 400°C, for periods up to those sufficient to simulate end of 40 and 60 operating lives. Note here that the activation energy used for calculating equivalent thermal aging damage at lower temperatures may be lower than commonly accepted, even as low as ~100 kJ/mole, and should be independently verified. Irradiation up to 10 dpa could be done sequentially in a pool type thermal reactor and/or in a fast reactor. It may also be possible to simultaneously thermally age at an elevated temperature up to 400°C and irradiate in a sodium cooled fast reactor with suitable shielding of the specimens to reduce the fast flux to a value consistent with the likely doses in PWRs and BWRs (a few dpa) in the time

required to simulate end-of-life thermal aging.

Following aging with various combinations of thermal and irradiation embrittlement, both fracture toughness and resistance to stress corrosion cracking in simulated PWR and BWR primary coolants would be evaluated.

4.2.2 Quantitative Modeling of Oxidation and EAC based on Fundamental Physical Principles for both PWRs and BWRs

[Background]

A very long term aim of research devoted to EAC of structural materials in light water reactor systems is to develop predictive models based on fundamental atomistic physical principles. In the context of BWRs, the aim is to try and anticipate problems arising in primary circuit materials (and steam generators) from very long term aging associated with plant lifetimes that are currently anticipated to be prolonged from 40 to 60 years and eventually to 80 years and are clearly very difficult to test experimentally without waiting for lead plants to attain such lifetimes.

The FRI, Tohoku University has already taken steps in the direction of more fundamental physically based modeling through its work on the application of modern fracture mechanics solutions for stress/strain fields at crack tips coupled with various hypotheses for crack advance, and in the application of molecular dynamics modeling to hydrogen and oxygen interactions with metals and oxygen. These initiatives respond in part to the desire to extend modeling to *a priori* atomistic predictions of SCC susceptibility.

[Scope]

The following is a précis supplemented by points raised in discussion during the meeting of the Tohoku-Hokkaido cluster held in Miyagi Zao during October 2010.

The surface enclave: smooth surfaces and imperfections; surface reactivity and films growth on metals; misfit strains between oxides and underlying metal; atom removal stresses in films and in the underlying metal.

The crack tip enclave: physical processes in tight nanometer scale wide cracks; vacancies, cavities, dislocations and deformation at crack tips; crack tip chemistry; hydrogen activity and interactions; crack advance processes; short crack coalescence.

Long crack enclave: transition from slow to rapid (fracture mechanics controlled) crack growth; ion migration, impurity hideout and crevice chemistry

Radiation enclave: environmental and metal interactions with neutrons and other ionizing radiations; irradiation damage in metal and protective oxides; physical property changes; IASCC crack advance processes.

SCC phenomenology enclave: This enclave is more component specific. For instance for PWRs the concerns revolve around PWSCC (or LPSCC); Sulfur SCC, Lead SCC; SCC of low alloy steels and turbine alloys, effects of cold work and strain pathway. For the BWRs the concerns are primarily related to IASCC of stainless steel core internal components, in addition to the balance of plant issues identified for the PWRs.

The surface and radiation enclaves were judged in discussion at Miyagi Zao to be perhaps the least well understood at this time.

5. Conclusion

In this paper, research topics considered necessary for a proactive approach to aging degradation phenomena in LWR structural materials which were discussed at the Proactive Aging Management Experts' Panel Meeting of NISA Project, have been introduced. Several short term research subjects (to be completed or to achieve significant progress within 5 years) and several long term research subjects (to be completed in 5-10 years) were outlined for both PWRs and BWRs with a suggested order of the priority. Although some may not necessarily be considered proactive in the sense that they have already been studied extensively by the industry and academia, they are included in the suggested research subjects because of the great concern for the occurrence of the degradation in components that have not yet experienced it. It is also important to continue to promote the systematic elicitation for potential and latent material aging degradation phenomena using the principles developed in the Proactive Aging Management Experts' Panel Meeting.

6. References

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