Materials Degradation in Light Water Reactors: Life After 60

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1. Introduction

Nuclear reactors present a very harsh environment for components service. Components within a reactor core must tolerate high temperature water, stress, vibration, and an intense neutron field. Degradation of materials in this environment can lead to reduced performance, and in some cases, sudden failure. A recent EPRI-led study interviewed 47 US nuclear utility executives to gauge perspectives on long-term operation of nuclear reactors. Nearly 90% indicated that extensions of reactor lifetimes to beyond 60 years were likely. When polled on the most challenging issues facing further life extension, two-thirds cited plant reliability as the key issue with materials aging and cable/piping as the top concerns for plant reliability.

Materials degradation within a nuclear power plant is very complex. There are many different types of materials within the reactor itself: over 25 different metal alloys can be found with can be found within the primary and secondary systems, not to mention the concrete containment vessel, instrumentation and control, and other support facilities. When this diverse set of materials is placed in the complex and harsh environment coupled with load, degradation over an extended life is indeed quite complicated. To address this issue, the USNRC has developed a Progressive Materials Degradation Approach (*NUREG/CR-6923)*. This approach is intended to develop a foundation for appropriate actions to keep materials degradation from adversely impacting component integrity and safety and identify materials and locations where degradation can reasonably be expected in the future.

Clearly, materials degradation will impact reactor reliability, availability, and potentially, safe operation. Routine surveillance and component replacement can mitigate these factors, although failures still occur. With reactor life extensions to 60 years or beyond or power uprates, many components must tolerate the reactor environment for even longer times. This may increase susceptibility for most components and may introduce new degradation modes. While all components (except perhaps the reactor vessel) can be replaced, it may not be economically favorable. Therefore, understanding, controlling, and mitigating materials degradation processes are key priorities for reactor operation, power uprate considerations, and life extensions.

This document is written to give an overview of some of the materials degradation issues that may be key for extend reactor service life. A detailed description of all the possible forms of degradation is beyond the scope of this short paper and has already been described in other documents (for example, the *NUREG/CR-6923).* The intent of this document is to present an overview of current materials issues in the existing reactor fleet and a brief analysis of the potential impact of extending life beyond 60 years. Discussion is presented in five distinct areas:

- -Reactor pressure vessel
- -Reactor core and primary systems
- -Reactor secondary systems
- -Weldments
- -Concrete
- -Modeling and simulations

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Following each of these areas, some research thrust directions to help identify and mitigate lifetime extension issues are proposed. Note that while piping and cabling are important for extended service, these components are discussed in more depth in a separate paper. Further, the materials degradation issues associated with fuel cladding and fuel assemblies are not discussed in this section as these components are replaced periodically and will not influence the overall lifetime of the reactor.

2. Reactor Pressure Vessel

This section provides a brief summary of major technical issues regarding aging of reactor pressure vessels (RPV), and identifies those issues that require further research to provide information not currently available or to enhance existing information with a view towards reducing the associated uncertainties. Current regulations require RPV steels to maintain conservative margins of fracture toughness so that postulated flaws do not threaten the integrity of the RPV during either normal operation and maintenance cycles or under accident transients, like pressurized thermal shock.

The last few decades have seen remarkable progress in developing a mechanistic understanding of irradiation embrittlement. This understanding has been exploited in formulating robust, physically-based and statistically-calibrated models of Charpy V-notch (CVN)-indexed transition temperature shifts. However, these models and our present understanding of radiation damage are not fully quantitative, and do not treat all potentially significant variables and issues. Similarly, developments in fracture mechanics have led to a number of consensus standards and codes for determining the fracture toughness parameters needed for development of databases that are useful for statistical analysis and establishment of uncertainties. The CVN toughness, however, is a qualitative measure that must be correlated with the fracture toughness and crack-arrest toughness properties necessary for structural integrity evaluations. Direct measurements of the fracture toughness properties are desirable to reduce the uncertainties associated with correlations.

The progress notwithstanding, however, there are still significant technical issues that need to be addressed to reduce the uncertainties in regulatory application. The issues regarding irradiation effects, briefly summarized in this section, are those identified by a cross-section of researchers in the international community. Of the many significant issues discussed, those deemed to have the most impact on the current regulatory process and life extension are listed below and include both experimental and modeling needs. Moreover, the combination of irradiation experiments with modeling and microstructural studies provides an essential element in aging evaluations of RPVs.

2.1. High fluence, long irradiation times, and flux effects

Sparse or nonexistent data at high fluences, for long times, and for high embrittlement create large uncertainties for embrittlement predictions. This issue is directly related to life extension with the number of plants requesting license extension to 60y and those expected to request 80y. Simply stated, extending operation from 40y to 80y will result in a doubling of the neutron exposure for the RPV. Moreover, because the recent PTS re-evaluation project has resulted in lower average failure probabilities for PWRs, many plants are increasing their operating power level which will further increase the fluence. To obtain data at the high fluences for life extension will require the use of test reactor experiments that use high neutron fluxes. Substantial research is needed to enable application of data obtained at high flux to RPV conditions of low flux and high fluence. Moreover, there is now experimental evidence that so-called "late blooming phases"(rich in nickel and manganese) could produce an effect that could have serious implications to RPV life extension.

2.2. *Material variability and surrogate materials*

The subject of material variability has experienced increasing attention in recent years as additional research programs began to focus on the development of statistically viable databases. With the development of the master curve approach for fracture toughness and the potential use of elastic-plastic fracture-toughness data for direct application to the RPV, attention has focused on the issue of surrogate materials. Many surveillance programs contain CVN specimens of a different heat of base metal or different weld than that in the RPV. This issue has received attention within the industry and is under evaluation by the NRC. Application of the master curve methodology to RPVs is not likely to occur without resolution of this issue, including development and acceptance of the associated uncertainties.

2.3. *High-nickel materials*

It is well known that increasing nickel content, all other factors being equal, causes increasing embrittlement in RPV steels. Moreover, the subject of high-nickel content is applicable to concerns of "late-blooming phases" at high fluences, potential effects on the development of copper-rich precipitates, and effects on post-weld heat treatments. The strong synergistic interactions between copper, nickel and manganese are understood at low to intermediate fluence. The behavior at higher fluence, in both low and higher copper steels, needs to be established. Similarly the basic role of phosphorous is established. However, potential interactions with copper and phosphorous effects at high fluence have not been quantified.

2.4. *Fracture toughness master curve*

The master curve was identified by almost every source as a recommended subject for continued research. The issues most identified were, 1) the shape of the master curve at high levels of embrittlement and at high fluence, 2) specimen size (see issue on precracked Charpy), 3) dynamic loading (including crack-arrest), 4) the effects of intergranular fracture, and 5) the technical underpinning for the universal shape of the curve.

2.5. *Precracked charpy and smaller specimens*

The precracked Charpy V-notch (PCVN) specimen and smaller specimens are identified as a separate issue because of the important link to RPV surveillance programs. Various test programs have identified a non-conservative bias in determination of the fracture toughness reference temperature with the PCVN specimen that ranges from a few degrees to as much as 40°C, when compared to larger specimens. Further evaluation of specimen size effects is needed to fully understand the limits of applicability and associated uncertainties. The International Atomic Energy Agency (IAEA) has recently conducted two Coordinated Research Projects relative to this subject.

2.6. *Attenuation*

There is still some controversy over the way in which embrittlement variations through the RPV wall arising from attenuation of the neutron flux should be estimated. The current methodology is based on neutron fluence greater than 1 MeV, but the use of displacements per atom (dpa) is more technically sound. There are several types of research that are needed to better resolve both the issue of the proper dose unit and to provide a proper framework for assessing attenuation. Development of the attenuation model can be accomplished through test reactor experiments (such as that recently sponsored by the IAEA in a Russian test reactor) or through direct examination of a decommissioned RPV.

2.7. *Modeling and microstructural analysis*

This issue is identified separately because it represents the core of research in irradiation effects. Prediction of irradiation-induced embrittlement for any given RPV steel is the goal. The issues and relevance of modeling activities to nuclear plant life extension are discussed elsewhere in this white paper.

2.8. Annealing and reirradiation

Post-irradiation annealing is still an approach that is of international interest for mitigating embrittlement, especially given the potential doubling or more of neutron exposure to be experienced with life extension to 80y. The NRC has issued a regulatory guide on thermal annealing of RPVs, but the nuclear industry has apparently been reluctant to adopt the procedure for non-technical reasons. Given operation of some very radiation-sensitive RPVs to 80y, and considering the unknown factors discussed in this paper, it is likely that thermal annealing may be seriously considered in the future. Thus, there is a need for additional data on re-irradiation behavior of annealed RPV materials.

2.9. Thermal aging

Thermal aging effects on RPV materials is deemed a technical issue for evaluation because of the extended time in service and the lack of adequate information with which to judge thermal effects to be insignificant. There are only sparse data available on long term thermal aging effects; a particular well-known study by B&W (now AREVA) evaluated a few different RPV base and weld materials for over 200,000 h. Although the results indicated no significant effects of thermal aging, the exposure temperature was only 260°C, well below the average of about 288°C for U.S. PWRs and even further below the typical RPV temperature of about 300°C for B&W reactors. There are other international thermal aging studies on similar steels showing increasing effects of thermal aging with temperature and dependent on the specific material, with chemical composition (e.g., copper) and heat treatment being two other important variables.

2.10. Specific structural integrity issues

Regarding application of fracture mechanics to RPV structural integrity, it is now widely acknowledged that the regulatory products developed over the past few decades incorporated large implicit and explicit conservatisms. Since that time, considerable progress has been made in many areas of fracture characterization and assessment methodologies, and development of deterministic and probabilistic computer codes that integrated all of those findings (including those from irradiation effects studies) into a unified predictive tool for RPVs.

Beyond these state-of-knowledge advancements in the technical areas of mechanical metallurgy and fracture mechanics, a key enabling advance that has occurred concurrently is the order-of-magnitude increases in computational ability that has occurred over the last two decades. The overall state of understanding has evolved to the point that the NRC is now in a position to begin updating the various applicable technical bases to reduce the associated conservatisms while retaining adequate safety margins.

Over the past several decades, ORNL and other contractors in the nuclear community (national and international) have performed assessments of the failure probability for a number of different pressure bearing components in NPPs [e.g., the reactor pressure vessel (RPV) when subjected to PTS, piping when subjected to earthquake loading, steam generator tubes subjected to severe accident loading]. Their probabilistic analyses have been performed using custom-built software that cannot be easily scaled or adapted to the solution of other similar problems. This *ad hoc* approach to model development has led to considerable repetition of work and to difficulties and delays in using probabilistic insights to inform regulatory decision-making. The aim of a new project at ORNL is therefore to develop a configuration-controlled, modular-designed probabilistic computer code that can be used to assess the structural integrity of any passive pressure-bearing component in a nuclear power plant on a risk-informed basis.

3. Reactor Core and Primary Systems

As noted in the introduction, the reactor core is a very adverse environment, combining the effects of stress, corrosion, and irradiation. Components in this environment are also often the must critical for safe and reliable operation as the failure of a core internal component may have very severe consequences. In general, service life beyond 60 years will increase time at temperature and neutron fluence, leading to increased susceptibility and severity for known degradation mechanisms (although new mechanisms are also possible). Therefore, understanding the materials performance and degradation mechanisms is key. As noted above, *NUREG/CR-6923* presents a more detailed discussion of all the forms of degradation that have been observed in the current LWR fleet. The issues described below represent those that may warrant additional attention in life beyond 60 years and are grouped into three key areas: thermal aging and fatigue, irradiation-induced effects, and corrosion.

3.1. Thermal aging and fatigue

The effects of elevated temperature service in metal alloys have been examined for many years. Possible effects include phase transformations that can adversely impact mechanical properties. Extended time at elevated temperature may permit even very slow phase transformations to occur. This is of particular concern for cast stainless steel components where the formation of a brittle alpha-phase can result in a loss of fracture toughness and lead to brittle failure. The effects of aging on other components are also of concern and should be examined. The effort required for identifying possible problems can be reduced, though, by using modern materials science modeling techniques and experience in other industries.

Fatigue refers to an aging degradation mechanism where components undergo cyclic stress. Typically, these are either low-load, high frequency stresses or high-load, low frequency stresses generated by thermal cycling, vibration, seismic events, or loading transients. Environmental factors may accelerate fatigue and eventually may result in a component failure. In a light water reactor, components such as the pressure vessel, pressurizer, steam generator shells, steam separators, pumps, and piping are among the components that may be affected. *NUREG/CR-6923* identified fatigue as an issue for a number of different components and subsystems for both PWR and BWR's.

Due to the potential for thermal aging and fatigue damage during extended lifetimes, the design methodology for core internal structures should also be examined. During the initial plant design, each component was designed with a load to expected and specific lifetimes and operating conditions using established guidelines (typically ASME). An 80-year reactor lifetime corresponds to over 600,000 hours of service (at a 90% service factor) while most creep data used in design comes from tests operating much less than 100,000 hours, leading to non-conservative design. For example, creep and creep-fatigue over such a long-life may result in component failure. The extension of lifetimes beyond these initial design considerations should be carefully examined.

3.2. Irradiation-induced effects

Over the forty-year lifetime of a light water reactor, internal structural components may expect to see up $\sim 10^{22}$ n/cm^2/s in a BWR and $\sim 10^{23}$ n/cm^2/s in a PWR (E > 1 MeV), corresponding to ~7 dpa and 70 dpa, respectively. Extending the service life of a reactor will increase the total neutron fluence to each component. Fortunately, radiation effects in stainless steels (the most common core constituent) is are also the most examined as these materials are also of interest in fast-spectrum fission and fusion reactors where higher fluences are encountered.

The neutron irradiation field can produce large property and dimensional changes in materials. This occurs primarily via one of five radiation damage processes: Radiation-induced hardening and embrittlement, phase instabilities from radiation-induced or -enhanced segregation and precipitation, irradiation creep due to unbalanced absorption of interstitials vs. vacancies at dislocations, volumetric swelling from cavity formation, and high temperature helium embrittlement due to formation of helium-filled cavities on grain boundaries. For light water reactor systems, high temperature embrittlement and creep are not common problems due to the lower reactor temperature. However, radiation embrittlement, phase transformation, segregation, and swelling have all been observed in reactor components.

3.2.1. Radiation-induced segregation and phase transformations

Under irradiation, the large concentrations of radiation-induced defects will diffuse to defect sinks such as grain boundaries and free surfaces. These concentrations are in far excess of thermal-equilibrium values and can lead to coupled-diffusion with particular atoms. In engineering metals such as stainless steel, this results in radiation-induced segregation of elements within the steel. For example, in 316 stainless steel, chromium (important for corrosion resistance) can be depleted at areas while elements like nickel and silicon are enriched to levels well above the starting, homogenous composition. While radiation-induced segregation does not directly cause component failure, it can influence corrosion behavior in a water environment. Further, this form of degradation can accelerate the thermally-driven phase transformations mentioned above and also result in phase transformations that are not favorable under thermal aging (such as gamma or gamma-prime phases observed in stainless steels). Additional fluence may exacerbate radiation-induced phase transformations and should be considered. The wealth of data generated for fast-breeder reactor studies and more recently in LWR-related analysis will be beneficial in this effort.

3.2.2. Radiation-induced swelling and creep

The diffusion of radiation-induced defects can also result in the clustering of vacancies, creating voids. If gas atoms such as He enter the void, it becomes a bubble. While swelling is typically a greater concern for fast reactor applications where it can be life-limiting, voids have recently been observed in LWR components such as baffle bolts. The motion of vacancies can also greatly accelerate creep rates, resulting in stress relaxation and deformation. Irradiation-induced swelling and creep effects can be synergistic and their combined influence must be considered. Longer reactor component lifetimes may increase the need for a more thorough evaluation of swelling as a limiting factor in LWR operation. As above, data, theory, and simulations generated for fast reactor and fusion applications can be used to help identify potentially problematic components.

3.2.3. Radiation-induced embrittlement

Radiation embrittlement results in an increase in the yield and ultimate tensile strength of the material. This increase in strength comes with a corresponding decrease in ductility. This hardening can be caused the changes in the alloy’s microstructure including radiation-induced segregation, phase transformations, and swelling. Ultimately, hardening and loss of ductility will result in reduced fracture toughness and resistance to crack growth. Extended reactor lifetimes may lead to increased embrittlement issues.

3.3. Corrosion and stress corrosion cracking

In addition to elevated temperatures, intense neutron fields, and stress, components must also be able to withstand a corrosive environment. Temperatures typically range from 288ºC in a BWR up to 360ºC in a PWR, although other water chemistry variables differ more significantly between the BWR and PWR’s.

Corrosion is a complex form of degradation that is strongly dependent on temperature, material condition, material composition, water purity, water pH, water impurities, and gas concentrations. The operating corrosion mechanism will vary from location to location within the reactor core and a number of different mechanisms may be operating at the same time. These may include general corrosion mechanisms such as uniform corrosion, boric acid corrosion (BAC) , flow accelerated corrosion (FAC), and/or erosion corrosion which will occur over a reasonably large area of material in a fairly homogenous manner. Localized corrosion modes occur over much smaller areas, but at much higher rates than general corrosion and include crevice corrosion, pitting, galvanic corrosion, and microbiologically influenced (MBI) corrosion. Finally, environmentally assisted cracking (EAC) includes other forms of degradation, which are closely related to localized or general corrosion with the added contribution of stress. In a LWR, a number of different environmentally assisted cracking mechanisms are observed: intergranular stress-corrosion cracking (IGSCC), transgranular stress corrosion cracking (TGSCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC) and low-temperature crack propagation (LTCP).

While all forms of corrosion are important in managing a nuclear reactor, IASCC has received considerable attention over the last four decades due both to its severity and unpredictability. Despite over thirty years of international study, the underlying mechanism of IASCC is still unknown, although more recent work led by groups such as the Cooperative IASCC Research Group has identified other possible causes.

Clearly, unmonitored corrosion (in all the various forms) is not acceptable for the safe and reliable operation of a reactor. With extended lifetimes, components must resist additional time in contact with the coolant. As a result, the various corrosion mechanisms must all be understood and evaluated for different reactor component and sub-systems to ensure extended service. The *NUREG/CR-6923* report provides a clear and in-depth review of all these mechanisms for the current LWR fleet. Additional effort is required to evaluate the importance of each with further lifetime extensions.

4. Reactor Secondary Systems

Components in the secondary (steam generator) side of a nuclear reactor power plant are also subject to degradation. While the secondary side of the reactor does not have the added complications of an intense neutron irradiation field, the combined action of corrosion and stress can create many different forms of failure. The majority steam generator systems in US power plants today originally used Alloy 600 (a Ni-Cr-Fe alloy), although service experience showed many failures in tubes through the 1970's. In the last 20 years, most steam generators have been replaced with Alloy 690, which shows more resistance to stress-corrosion cracking. In addition to the base material, there are weldments, joints, and varying water chemistry conditions leading to a very complex component. Indeed, the array of modes of degradation varies with location. In a single steam generator examined by Staehle and Gorman in *Corrosion (2004)*, twenty-five different modes of corrosion degradation were identified. Stress-corrosion cracking is found in several different forms and may be the limiting factor for extended service. The integrity of these components is critical for reliable power generation in extended lifetimes, and as a result, understanding and mitigating these forms of degradation is very important. Adding additional service life to these components will allow more time for corrosion to occur. The various forms of corrosion must be evaluated as in *NUREG/CR-6923* with a special attention to those that may be life limiting in extended service.

5. Weldments

Welding is extensively used in construction of nuclear reactor components and subsystems. The performance of weldments (including both weld metal and the adjacent heat affected zone) is critical to the safe and efficient operation of the nuclear reactor. EPRI's recent strategic plan for long-term nuclear reactor operation identifies two critical long-standing welding related technical challenges requiring further research and development.

5.1. Advanced welding simulation tools

As noted above, corrosion and irradiation create a difficult environment for service. As weldments often have a different microstructure from the base metal, they are often more susceptible. Certain weld regions have been found to be vulnerable to stress corrosion cracking and other environmental degradation. These include the Alloy82/182 weldments throughout the primary system and type 304/316 stainless steel heat affected zones (HAZ) at moderate fluences. Examples of such problems include the cracking and erosion in connection welds and more recent nozzle cracking in dissimilar metal welds of the pressurizer. In a recent NRC study, it was found that the predictions from the component integrity analysis were highly dependent upon welding residual stress profiles. However, there is currently a lack of weld residual stress simulation models in the industry capable of accurately and reliably predicting the residual stress distribution in the weld region.

Today, most welding simulation models include only the thermal and mechanical aspects and are simplified using two-dimensional formulations. The next generation of weld residual stress prediction models must integrate all the essential fundamental phenomena governing the formation of weld residual stress: thermal, mechanical, and microstructural aspects. Specifically, the next generation, advanced welding simulation tool will need to have the following key capabilities: (1) weld microstructure simulation capability to simulate the non-equilibrium phase transformations during welding and to fully couple the effects of phase transformation in weld residual stress prediction; (2) efficient three-dimensional simulation algorithms to deal with the complex weld geometry in vessel structures; and (3) improved high-temperature material deformation constitutive laws. Additionally, improvements in residual stress measurement for actual and or geometrically complex mock-up weldments need to be developed. Finally, efficient computational algorithms are needed to incorporate the vast amount of high-fidelity weld residual stress simulation information from the advanced weld model to the lifetime prediction model.

5.2. Development of new welding technologies for repair and upgrades

Today, welding is widely used for repair, maintenance and upgrade of LWR components. These repair welds need to have improved resistance to SCC and to other long-term degradation. New and improved welding techniques (processes and techniques) are needed to avoid and/or reduce any deleterious effects associated with the traditional welding fabrication practices. There have been significant advances in welding technology; both process technology and knowledge of welding residual stress control, in the past decades and some are candidates for further development. Specifically, the following areas should be evaluated: (1) proactive weld residual stress control and mitigation techniques through welding process innovation and/or post-weld treatment; (2) welding technology to repair irradiated reactor internals to avoid helium induced cracking during welding repair; (3) improved weld metal development; and (4) new solid-state joining process such as friction stir welding and high-energy welding such as laser welding for microstructure and residual stress benefits. Development of new and improved welding technology for weld residual stress and microstructure control will require better understanding and predictive capability.

6. Concrete Structures

As concrete ages, changes in its properties will occur as a result of continuing microstructural changes (e.g., slow hydration, crystallization of amorphous constituents, and reactions between cement paste and aggregates), as well as environmental influences. These changes do not have to be detrimental to the point that the concrete will not be able to meet its functional and performance requirements. Concrete, however, can suffer undesirable changes with time because of improper specifications, a violation of specifications, or adverse performance of its cement paste matrix or aggregate constituents under environmental influences (e.g., physical or chemical attack). Changes to embedded steel reinforcement as well as its interaction with concrete can also be detrimental to concrete's service life.

In general, the performance of reinforced concrete structures in nuclear power plants has been very good. Incidents of degradation initially reported generally occurred early in the life of the structures and primarily have been attributed to construction/design deficiencies or improper material selection. Although the vast majority of these structures will continue to meet their functional or performance requirements during the current and any future licensing periods, it is reasonable to assume that there will be isolated examples where as a result primarily of environmental effects the structures may not exhibit the desired durability (e.g., water-intake structures and freezing/thawing damage of containments) without some form of intervention.

Although several activities have addressed aging of nuclear power plant structures (e.g., Nuclear Regulatory Commission, Nuclear Energy Agency, and International Atomic Energy Agency) there are still several areas where additional structures-related research is desired to demonstrate that the structures will continue to meet functional and performance requirements (e.g., maintain structural margins). Structural research topics include: (1) compilation of material property data for long-term performance and trending, evaluation of environmental effects, and assessment and validation of NDE methods; (2) evaluation of long-term effects of elevated temperature and radiation; (3) improved damage models and acceptance criteria for use in assessments of the current as well as estimating the future condition of the structures: (3) improved constitutive models and analytical methods for use in determining nonlinear structural response (e.g., accident conditions); (4) non-intrusive methods for inspection of thick, heavily-reinforced concrete structures and basemats; (5) global inspection methods for metallic pressure boundary components (i.e., liners of concrete containments and steel containments) including inaccessible areas and backside of liner; (6) data on application and performance (e.g., durability) of repair materials and techniques; (7) utilization of structural reliability theory incorporating uncertainties to address time-dependent changes to structures to assure minimum accepted performance requirements are exceeded and to estimate on-going component degradation to estimate end-of-life; and (8) application of probabilistic modeling of component performance to provide risk-based criteria to evaluate how aging affects structural capacity.

7. Modeling and Simulations

Theory and modeling of radiation effects can make a significant contribution to the ongoing assessment of material and component reliability in nuclear power plants. In-service property changes can not be assessed solely by experimental means because of the long times involved. So-called accelerated test techniques that involve higher exposure temperatures or higher damage rates can provide valuable information to help anticipate material property degradation, but they can also be misleading without a sound, model-based analysis of the effect of temperature and damage rate on the mechanisms responsible for the property changes. In addition, a completely representative experimental assessment is precluded by the combinatorial nature of the exposure environment. Even a single component may be exposed at a range of temperatures, radiation damage rates, and mechanical loading conditions. Modeling may be used here to both interpolate and extrapolate the available data to the full range of conditions required.

For example, the reactor pressure vessel is exposed to a relatively flux of fast neutrons leading to atomic displacement rates in the range of 10^{-11} to 10^{-10} dpa/s. The vessel properties are monitored through specimens mounted inside the vessel at locations in which the damage rate is typically 2 to 5 times higher. The objective of these surveillance programs was to obtain mechanical property data at end-of-life conditions within about ten effective full power years (EFPY). Based on the understanding of radiation damage mechanisms at the time the reactors went into service, it was believed that a factor of five increase in damage rate would not compromise the data obtained. The issue of accelerated damage rates is being revisited in the radiation damage community at this time. More importantly for life extension programs, commercial reactors generally do have surveillance specimens in place to account for the higher doses that will be reached beyond 30 to 40 EFPY. The primary alternative to obtain the required dose is irradiation in materials test reactors at much higher rates, accelerated by more than a factor of 10. Current experimental and modeling results indicate that such data may be non-conservative with respect to low dose rate in-service conditions. Material response to changes in dose rate has been shown to be sensitive minor composition variations which determine which microstructural features are primarily responsible for embrittlement in different regimes of dose and dose rate (and temperature). In addition, thermodynamic modeling has identified the potential for additional embrittling phases to form at higher doses that have not been observed in commercial materials to date. Only a well integrated program of critical experiments and modeling can provide the assurance needed to continue safe operation of the current reactor fleet and avoid unexpected, and unwanted, technical surprises in the years ahead.

8. Summary of Nuclear Reactor Materials Degradation

Components serving in a nuclear reactor plant must withstand a very harsh environment including extended time at temperature, neutron irradiation, stress, and/or corrosive media. The many modes of degradation are complex and vary depending on location and material. However, understanding and managing materials degradation is a key for the continued safe and reliable operation of nuclear power plants.

Extending reactor service to beyond 60 years will increase the demands on materials and components. Therefore, an early evaluation of the possible effects of extended lifetime is critical. The recent *NUREG/CR-6923* gives a detailed assessment of many of the key issues in today's reactor fleet and provides a starting point for evaluating those degradation forms particularly important for consideration in extended lifetimes. While life beyond 60 will add additional time and neutron fluence, the primary impact will be increased susceptibility (although new mechanisms are possible).

For reactor pressure vessels, a number of significant issues have been identified for recommended attention in future research activities. Sparse or nonexistent data at high fluences, for long times, and for high embrittlement create large uncertainties for embrittlement predictions. The use of test reactors at high fluxes to obtain high fluence data is problematic for representation of the low flux conditions in RPVs. Late-blooming phases, especially for high nickel welds, have been observed and additional experimental data are needed in the high fluence regime where they are expected. Other discussed issues include specific needs regarding application of the fracture toughness master curve, data on long term thermal aging, attenuation of embrittlement through the RPV wall, and development of a embrittlement trend curve based on fracture toughness.

For the reactor core and primary systems, several key areas have been identified. Thermo-mechanical considerations such as aging and fatigue must be examined. Irradiation-induced processes must also be considered for higher fluences, particulary the influence of RIS, swelling, and/or precipitation on embrittlement. Corrosion takes many forms within the reactor core, although IASCC and PWSCC are of high interest in extended life scenarios. Research in these areas can build upon other ongoing programs in the LWR industry as well as other reactor materials programs (such as fusion and fast reactors) to help resolve these issues for extended LWR life.

In the secondary systems, corrosion is extremely complex. Understanding the various modes of corrosion and identifying mitigation strategies is an important step for long-term service.

In the area of welding technology, two critical long-standing welding related technical challenges requiring further research and development (both fundamental and applied). The first is the need for an advanced weld simulation tool to support component life extension and reliable lifetime prediction, especially as related to the issue of residual stresses as a primary driving force for stress corrosion cracking. The second challenge is the development of new welding technologies for reactor repair and upgrade.

Concrete structures can also suffer undesirable changes with time because of improper specifications, a violation of specifications, or adverse performance of its cement paste matrix or aggregate constituents under environmental influences (e.g., physical or chemical attack). Changes to embedded steel reinforcement as well as its interaction with concrete can also be detrimental to concrete's service life. A number of areas of research are needed to assure the long-term integrity of the reactor concrete structures.

Finally, modeling and simulation will play an important role in lifetime extensions. In many cases, experimental data to support a 60-80 year service life are not available and it is not currently practical or possible as such experiments should have been started 20 years ago. However, modeling and simulation can be useful to interpolate and extrapolate available data to the full range of conditions required.

9. Research Thrust Areas

There are many forms of materials degradation in a nuclear power reactor. Many of these are highly dependent upon a number of different variables, creating a complex scenario for evaluating lifetime extensions. Nonetheless, many of the diverse topics and needs described earlier can be organized into a few research thrust areas. These could include mechanisms of degradation, mitigation strategies, modeling and simulations, monitoring, and management.

Mechanisms of degradation: Basic research to understand the underlying mechanisms of selected degradation modes can lead to better prediction and mitigation. For example, research on IASCC and PWSCC would be very beneficial for extended lifetimes and could build on other existing programs within EPRI and NRC. Other forms of degradation such as swelling and embrittlement are better understood and mechanistic studies are not needed.

Mitigation strategies: While some forms of degradation have been well-researched, there are few options in mitigating their effects. Techniques such as post-irradiation annealing have been demonstrated to be very effective in reducing hardening of entire pressure vessels. Annealing may be effective in mitigating IASCC, based on initial studies. Water chemistry techniques such as NobelChem have been very effective in reducing some corrosion problems. Additional research in these areas may provide other alternatives to component replacement.

Modeling and simulation: Improved modeling and simulation efforts have great potential in reducing the experimental burden for life extension studies. These methods can help interpolate and extrapolate data trends for extended life. Simulations predicting phase transformations, radiation embrittlement, and swelling over component lifetimes would be extremely beneficial to licensing and regulation in extended service.

Monitoring: While understanding and predicting failures are extremely valuable tools for the management of reactor components, these tools must be supplements to active monitoring. Improved monitoring techniques will help characterize degradation of core components. For example, improved crack detection techniques will be invaluable. New non-destructive examination techniques may also permit new means of monitoring pressure vessel embrittlement or swelling of core internals.

Management: Extended life and materials degradation will require sound management practices such as the Proactive Materials Degradation Approach. New strategies to deal with advances in component monitoring (and subsequent regulation) and mitigation strategies will be needed.