# AGING OF METAL COMPONENTS IN US NUCLEAR REACTORS

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**Abstract**

This paper presents an overview of the aging of metal components in U.S. Light Water Reactors. The types of degradation being experienced in components such as the pressure vessel, piping, reactor internals, and steam generators, and the programs being implemented to manage the degradation are discussed.

## 1. INTRODUCTION

Commercial Light Water Reactors (LWRs) in the U.S. were designed with explicit consideration given to some forms of aging degradation. The most notable examples are the consideration given to fatigue in piping and piping components, and embrittlement of the reactor pressure vessel. As we gained operating experience, a number of other degradation forms emerged. For example:

- Intergranular Stress Corrosion Cracking (IGSCC) BWR piping systems;
- Thermal embrittlement of cast stainless steels;
- Flow assisted corrosion of some piping systems;
- Environmentally Assisted Cracking of Control Rod Drive Mechanism housings;
- Stress Corrosion Cracking of reactor internals (irradiation assisted in some cases); and
- Degradation of steam generator tubes by a number of mechanisms.

The U.S. Nuclear Regulatory Commission, and the U.S. industry, have implemented a number of programs to address these issues. The following sections of this paper explore the degradation of key primary pressure boundary components and reactor internals, and the ongoing regulatory and research programs to address them.

## 2. AGING DEGRADATION OF KEY PRESSURE BOUNDARY COMPONENTS

### 2.1 Reactor Pressure Vessel Embrittlement

The reactor pressure vessel houses and supports the reactor core, and provides the flow path for the coolant through the core. The pressure vessel is also the only component in the reactor system for which the engineered safety features cannot provide protection in case of a rupture. Thus, assuring its integrity over the life of the nuclear power plant is essential to assuring safe operation of the plant.

The primary degradation mechanism for the reactor pressure vessel is neutron irradiation embrittlement. It has long been recognized that neutrons escaping from the reactor core embrittle the pressure vessel beltline materials, which can limit the safe operating life of a reactor pressure vessel. Vessels fabricated from materials with high levels of the trace elements copper and nickel are particularly susceptible to neutron irradiation embrittlement, but research and analysis of material surveillance data has also shown effects of other trace elements such as phosphorous.

Work in the U.S. to manage embrittlement has progressed on many fronts. For example, detailed statistical analyses of material surveillance data, coupled with research into the mechanisms of embrittlement, has provided improved correlations between material chemistry and embrittlement as measured by the shift in the Charpy 30 ft-lb energy level. These improved correlations are being evaluated as a basis for a potential third revision to the NRC's Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

Concerns over embrittlement have resulted in regulations governing two aspects of plant operation. First, and most widely discussed, are the embrittlement screening criteria used in the U.S. to provide protection against vessel failure during pressurized thermal shock (PTS) transients. The regulation addressing PTS specifies values of the material's reference temperature ($RT_{PTS}$) beyond which the plant cannot be operated without a detailed analysis to demonstrate an acceptable level of safety. The other regulation directly affected by embrittlement governs the acceptable pressure and temperature limits during plant startup and shutdown. As embrittlement levels increase the required temperature for a given pressure increases, which results in more restrictive startup and shutdown limits.

In both cases, there are numerous variables that must be considered, with material chemistry, neutron fluence, unirradiated properties, fracture toughness, potential flaws, and the applied loadings being the most notable. However, recent plant-specific evaluations in the U.S. have demonstrated the importance of variability in the material properties, particularly the material chemistry and unirradiated properties. The concerns over variability apply not only to variability within a specific weld or plate, but also to variability within a population of nominally identical (or at least similar) welds.

The NRC has maintained a database of plant-specific material surveillance data for several years. This database was used in developing the embrittlement correlations published in Revision 2 to Regulatory Guide 1.99, and more recently, provided the initial database used in the statistical evaluations that may lead to Revision 3 of that Regulatory Guide. It has been recognized in those evaluations that the reported surveillance data can show significant variability for the same weld or plate designation.

Recognizing the concern over variability, and the instances where similar materials were used in several pressure vessels but where different chemistry and property values were being reported, the NRC implemented a program where each operating reactor was required to provide detailed information about the materials in the reactor pressure vessel. That information is being compiled in the Reactor Vessel Integrity Database (RVID), and which will be used by the staff in assuring that all appropriate values of chemistry and initial properties are used in evaluations for specific pressure vessels. This effort will improve the accuracy of the chemistry and property values used in pressure vessel evaluations.

The NRC has also been addressing the other key variables through a broad-scope research effort. Significant emphasis has been placed on the fracture analysis methods applicable to shallow cracks subjected to typical loading, fracture toughness formulations and particularly the potential application of the Master Curve approach, and the size and density distributions for fabrication flaws in welds. The intent is provide a firm technical basis for any changes to the regulations and or regulatory guidance relating to pressure vessel safety.

Finally, the NRC has recognized that some pressure vessels may reach unacceptable levels of embrittlement, particularly if it is decided to extend the current operating license of the plant, and that mitigation of the embrittlement must be considered. Thermal annealing is the only known process for effectively relieving the embrittlement damage to the steel's microstructure, thereby restoring the ductility of the material to nearly its unirradiated level. While there has been significant experience in Russia and Eastern Europe with thermal annealing, it has never been accomplished on a commercial reactor pressure vessel in the U.S. Based on the favorable experience in Russia and Eastern Europe, and on extensive research data showing the beneficial effects on material properties, the NRC promulgated a regulation and regulatory guidance on thermal annealing. However, because of lingering doubts about the engineering feasibility of thermal annealing for U.S. designs, the U.S. Department of Energy and a consortium of organizations from the U.S., France, and Japan, undertook an Annealing Demonstration Program. The NRC participated as a observer in these demonstrations. The demonstration program was successful, and has provided a good basis for anticipating that thermal annealing can be successfully accomplished on U.S. design pressure vessels.

Thus, while irradiation embrittlement of the reactor pressure vessel can effectively limit its safe operating life, the combination of NRC and industry activities are providing viable techniques for effectively managing this degradation.

Current interest is focussed on pressure-temperature limits and possible ways to reduce unnecessary conservatism, thereby providing greater flexibility in plant operations and reducing the potential for inadvertent relief valve actuation in PWRs. Areas being considered include flaw orientation and size, use of the reference fracture toughness curve ($K_{IR}$) versus the crack initiation toughness ($K_{Ic}$), the fracture mechanics models used in the analyses, and the implicit and explicit margins used in the analyses.

## 2.2 Environmentally Assisted Cracking of Pressure Boundary Components and Reactor Internals

Environmentally assisted cracking (EAC) continues to be a significant aging degradation mechanisms for numerous components in light water reactors. Six of the eight forms of degradation cited as examples in the Introduction are related to environmentally assisted cracking. For EAC to develop, there must be a confluence of a sufficiently aggressive environment, a susceptible material, and a sufficiently high loading. Experience has shown that these conditions exist in many places in light water reactors.

One of the most notable instances of EAC was the intergranular stress corrosion cracking (IGSCC) of stainless steel piping in BWRs. This problem was initially found in small diameter piping, but fairly quickly was identified in larger diameter piping as well. The relatively high dissolved oxygen content of the BWR environment, coupled with the sensitization of the materials due either to welding processes, heat treatments, or both, and the high residual stresses in these welds, provided the conditions necessary to initiate cracks and for them to propagate through the wall thickness.

This problem was managed by a combination of approaches. Some plants elected to replace the 304 stainless steel piping with a nuclear grade of 316 piping, which provided a much more resistant material. Others have used a combination of full encirclement weld overlay repairs, induction heating stress improvement welding techniques, and much more careful control of the water chemistry to limit or eliminate this problem. It should be noted that many plants have implemented a hydrogen water chemistry program with tighter limits on general water chemistry, to provide a much less aggressive environment. Thus, IGSCC in BWR piping has been effectively managed through changes in materials and environments, and through weld repair techniques.

More recently, IGSCC has resulted in significant cracking in the core shrouds of some BWRs. Again, the combination of a susceptible material, an aggressive environment, and high residual stresses have led to extensive cracking of the core shrouds. However, unlike the piping, the high irradiation levels for the core shroud limit both the inspection and repair options. All of the inspections must be done remotely, and the visual inspection techniques that have been used effectively require cleaning of the surfaces and relatively high magnification. Replacement of the shroud does not appear to be viable for U.S. plants, and weld repair options are extremely limited due to the high irradiation levels in the stainless steel materials. Thus, the industry has devised mechanical clamping schemes to assure the structural integrity of the core shroud, even when large cracks are present. The addition of nobel metals to the coolant is being explored as one option to mitigate the cracking in much the same manner as hydrogen water chemistry was used to mitigate cracking in BWR piping. While this problem does not appear to be completely solved, it does appear that the repair, inspection, and mitigation programs implemented by the industry are effectively managing the degradation.

Stress corrosion cracking is being observed in a number of BWR and PWR components. For example, there has been cracking at the penetrations of the PWR control rod drive mechanism housings through the pressure vessel head. This problem has been observed at a number of non-U.S. reactors, and the U.S. industry has implemented a program to inspect U.S. reactors. There are numerous instances of cracking in other BWR reactor internal components, such as jet pump hold down beams, jet pump riser brackets, and top guides. There also are numerous instances of cracking in high nickel components, such as nozzle safe ends. The industry has developed an extensive set of guidelines for inspection, evaluation, and repair of BWR internals.

In addition to the classic IGSCC for reactor internals, certain highly irradiated components are susceptible to irradiation assisted stress corrosion cracking (IASCC). This degradation mechanism has been observed internationally in both PWRs and BWRs. Perhaps the most notable instance is the emerging problem of cracking of the baffle bolts in PWRs. For this degradation mechanism, the problem of defining a susceptible material, environment, and loading combination is further confounded by the effects of neutron irradiation, and the presence of what appears to be a threshold fluence below which the cracking is not observed.

There are both national and international research programs addressing the general issues of cracking in reactor components. The NRC has ongoing research addressing the broad spectrum of problems, including cracking in high nickel alloys and IASCC. That research is addressing not only the macroscopically observable crack initiation and growth, but includes efforts to explore the mechanisms of cracking.

The industry efforts include research addressing the cracking, mitigation strategies, and potential repair techniques, such as weld repairs. The NRC also is evaluating the merits and limitations of underwater weld repair techniques for irradiated stainless steel components. This is a relatively new line of inquiry in the U.S., at least for light water reactor applications, and it may provide a viable alternative to replacing components, or perhaps an adjunct to other repair techniques such as the mechanical clamps used on the BWR core shrouds.

There are a number of success stories related to EAC. However, this continues to be an area ripe for research inquiries and exploration. Until a better understanding of all of the factors that influence EAC in light water reactors is achieved, and better methods for identifying susceptible materials and components are developed, it appears that the industry and the regulators will have to continue to develop application specific programs in response to emerging cracking problems.

### 2.3 Fatigue and Thermal Embrittlement of Piping Components

The potential for metal fatigue to result in cracking and the failure of piping and piping components has long been recognized. As design codes, such as the ASME Boiler and Pressure Vessel Code, have evolved, the level of rigor required for piping fatigue analyses has increased. Over the last several years, the effects of the light water reactor environments on the fatigue life of typical piping materials has been under evaluation. While the water environment clearly reduces the fatigue life of both stainless and carbon steels, recent generic analyses have suggested that there was sufficient margin in the original design analyses to permit safe operation through the original design life of 40 years. The analyses are continuing to address the potential for operation through a license renewal period of 20 years. This issue is continuing to evolve in the U.S., particularly for those plants considering license renewal.

In the 1980's, there was considerable concern over the potential for thermal embrittlement of cast stainless steel piping and piping components to reduce the fracture toughness of these materials to unacceptable levels. An aggressive research program evaluating both the mechanisms of the embrittlement, and developing models to predict the fracture toughness change due to the embrittlement, provided the tools necessary to evaluate specific applications. In general, although the toughness loss can be nontrivial, the problem is not nearly as large as it once appeared.

### 2.4 Degradation of Steam Generator Tubes

Degradation of steam generator tubes is another aging problem that has evolved over time. Several years ago, the key problems were wastage and denting. As plant operators made changes to the secondary-side water chemistry to address those problems, and with longer service time, new problems emerged, such as secondary and primary stress corrosion cracking. In the early 1990's, outside diameter stress corrosion cracking, Intergranular attack, and primary water stress corrosion cracking emerged as significant degradation mechanisms. More recently, circumferential cracking at the top of the tube sheet, cracking at welds associated with repair sleeves, and mid-span cracking have emerged.

Industry initiatives to implement tube repair strategies, such as sleeving, have been fraught with problems. However, the high cost of replacing steam generators makes replacement an unattractive option for many utilities. Thus, there remains significant interest in a viable repair strategy.

The inspection technology used to detect and quantify tube degradation has made significant strides forward in the last few years. However, it still is not possible to reliably quantify the size of cracking for all applications. This has led to a "plug on detection" approach for many applications.

The NRC has been actively engaged in developing regulatory guidance on the appropriate elements of a program for managing steam generator tube degradation. The approach being promulgated relies on inspection, coupled with predictions of future performance and incorporation of results from the previous operating cycle, to provide a reasonable assessment of tube integrity during the current operating cycle.

Similarly, the industry has been implementing programs to manage the degradation in steam generator tubes, and performance as measured by forced outages due to tube leaks has improved, but the underlying problems have not been solved. Some utilities have elected to replace their steam generators using alloy 690 as the tubing material, which is much more resistant to degradation than the alloy 600 used in the original tubes.

Thus, there are both regulatory programs and operating plant strategies for managing the degradation of steam generator tubes. Nevertheless, continued degradation has the potential to shorten the operating life of PWRs. The experience to date with replacement steam generators has been very favorable, suggesting this is an effective way to handle the problem but may not be economically viable for all utilities.

## 3. SUMMARY AND CONCLUSIONS

Aging degradation can, and in many instances has, affected the integrity of metal components in light water reactors. The types of degradation are nearly as varied as the components, materials, and environments encountered in the reactors. If left unmitigated and unmanaged, aging degradation can limit the safe operating life of key components, such as the pressure vessel and steam generators.

Industry and regulatory programs have been implemented to address aging degradation of the key components. Some programs have resulted in significant successes, such as the management of

IGSCC in BWR piping, and the elimination of thermal embrittlement of cast stainless steels as a major concern. Further, programs underway to address pressure vessel embrittlement, fracture toughness, and the analytical methods used to evaluate pressure vessel safety, the offer the potential for a significant reduction in the concerns for pressure vessel safety for most plants.

Unfortunately, there are some areas where the ability to predict future performance and to understand the degradation phenomena is not as good as we would like. Environmentally assisted cracking, particularly for reactor internals, and degradation of steam generator tubes are two specific areas where the research programs have not provided the tools necessary to effectively mitigate the problems.

In general, the technical community has made great strides forward in understanding aging degradation for metal components. Even in those areas where we are left to react to emerging problems, the technical community has become particularly adept at identifying viable management strategies. Thus, we conclude that (1) there are effective programs for managing aging degradation for most components, (2) new forms of aging degradation are likely to occur for many components, particularly those susceptible to environmentally assisted cracking, and (3) the technical community supporting both plant operators and regulators will be able to address those new forms of degradation in a timely manner.