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Stress Corrosion Cracking in Light Water Reactors: Good Practices and Lessons Learned

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STRESS CORROSION CRACKING
IN LIGHT WATER REACTORS:
GOOD PRACTICES AND LESSONS LEARNED

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

IAEA NUCLEAR ENERGY SERIES No. NP-T-3.13

STRESS CORROSION CRACKING IN LIGHT WATER REACTORS: GOOD PRACTICES AND LESSONS LEARNED

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Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
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FOREWORD

One of the IAEAs statutory objectives is to “seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world”. One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Statute Article III, A.6, the IAEA Safety Standards establish “standards of safety for protection of health and minimization of danger to life and property.” The safety standards include the Safety Fundamentals, Safety Requirements, and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the Member States, regulatory bodies and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on and practical application of, nuclear energy for peaceful uses. This includes practical examples to be used by Member States, owners and operators of utilities, implementing organizations, academia, and government officials; among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards.

Stress corrosion cracking (SCC) is a significant ageing degradation mechanism for major components of both pressurized water reactors (PWRs) and boiling water reactors (BWRs). In PWRs, the main problem with SCC has been with Alloy 600 components such as steam generator tubes, pressurizer instrument penetrations and heater sleeves, control rod drive mechanism (CRDM) nozzles, and hot leg penetrations. In BWRs, piping and other components made from austenitic stainless steel or (to a much lesser extent) nickel based alloys have experienced intergranular stress corrosion cracking (IGSCC) and many cases have been reported in BWRs throughout the world..

This report provides general descriptions of damage mechanisms of different types of SCC that are of concern to systems, structures and components (SSCs) in light water reactors. Information on good practical operational experience and practices in Member States for preventing, mitigating and repairing SCC damages as well as information on related international/national R&D programmes are described.

The IAEA initiated work for collecting and sharing information among Member States on good practices to cope with IGSCC or irradiation assisted stress corrosion cracking (IASCC); the results of which are compiled in this report. The IAEA wishes to thank all the participants for their contributions. The IAEA officers responsible for this report were K.S. Kang, and L. Kupca of the Division of Nuclear Power.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The average age of existing nuclear power plants (NPPs) is constantly increasing while the number of new NPP constructions is still limited. In this circumstance, maintaining safety and performance of these ageing NPPs by effectively managing ageing degradations within an acceptable level becomes more and more important for Member States. Stress corrosion cracking (SCC) is one of the significant ageing degradations for major components of both pressurized water reactors (PWRs) and boiling water reactors (BWRs) and is still an important technical issue.

SCC is the term given to crack initiation and sub-critical crack growth of susceptible alloys under the influence of tensile stress and a ‘corrosive’ environment. SCC is a complex phenomenon driven by the synergistic interaction of mechanical, electrochemical and metallurgical factors.

With regard to PWRs, alloy 600 components; such as steam generator tubes, pressurizer instrument penetrations and heater sleeves, control rod drive mechanism (CRDM) nozzles, and hot leg penetrations have experienced primary water stress corrosion cracking (PWSCC) during the last 25 years. As a result, significant research and development efforts have been made to determine the factors affecting PWSCC.

A mechanistic understanding of PWSCC has not yet been established, but an empirical relationship based on field experience and research results has been developed. The results show that PWSCC of alloy 600 components occurs when high tensile stress, a primary water environment, and a susceptible microstructure are simultaneously present. Recent operational experience, such as the Davis–Besse NPP event, shows that PWSCC of the CRDM can lead to boric acid corrosion of the reactor pressure vessel (RPV) head as a result of primary water leaks and therefore can have a significant impact on the plant safety.

For some components in boiling water reactors (BWRs) made of austenitic stainless steel or nickel based alloy; e.g. the recirculation piping, core internals and some parts of the reactor pressure vessel (RPV) such as the in-core monitor (ICM) housings and the control rod drive (CRD) stub tubes, intergranular stress corrosion cracking (IGSCC) has been a significant ageing degradation mechanism. Many cases of IGSCC damage have been reported in BWRs throughout the world. One of the main reasons of such damage was that IGSCC had not been taken into account in the original design of BWRs.

Many research and development (R&D) programmes have been conducted in BWR owner countries. The mechanism of IGSCC of BWR components has been evaluated in detail and various kinds of measures for preventing, mitigating and repairing IGSCC have been established. Nevertheless, the IGSCC problem has not been fully solved and is still a concern for some BWR components.

Exposure to high levels of neutron fluence can also cause stainless steels to become susceptible to SCC. This is a special form of SCC known as irradiation assisted stress corrosion cracking (IASCC) that has occurred in both PWRs and BWRs. IASCC is also characterized by intergranular crack initiation and propagation. However, there are subtle differences between IASCC and IGSCC. Austenitic stainless steels that undergo IASCC need not be thermally sensitized or cold worked. Also, IASCC is highly dependent on neutron fluence exposure level.

Annealed and irradiated austenitic stainless steel becomes susceptible to IASCC when certain criteria (i.e. threshold fluence levels as a function of stress level) are met or exceeded. Both stabilized and non-stabilized stainless steels appear to be equally susceptible to IASCC. IASCC can be a significant ageing degradation mechanism for core internals of both BWRs and PWRs. Although not many core internals of BWRs and PWRs have been affected globally so far, IASCC may be a concern that increases with time (and therefore neutron fluence). Some Member States have initiated R&D programmes to establish measures for mitigating and, if possible, preventing IASCC.

Taking into account the above mentioned status regarding SCC, the IAEA initiated work for collecting and sharing information among Member States on good practices to cope with IGSCC/IASCC; the results are compiled in this report.

1.1.1. Objective

This report provides general descriptions of damage mechanisms of different types of SCC that are of concern to systems, structures and components (SSCs) in light water reactors. Information on good practices for preventing, mitigating and repairing SCC damages as well as information on related international/national R&D programmes are described. Practical operational experience and practices in Member States are also presented.

1.1.2. Scope

This report deals with IGSCC of BWRs, PWSCC of PWRs and IASCC. Transgranular stress corrosion cracking (TGSCC) is dealt with only in relation to IGSCC; e.g. transition cases from TGSCC to IGSCC. This report covers basic mechanisms, contributing factors, prevention, mitigation, analysis and repair methods and international/national R&D projects.

1.1.3. User

This report is intended for use by the staff, researchers, operation and maintenance personnel of organizations involved in material degradation issues including:

- Utilities;
- Material degradation research organizations;
- Technical support organizations;
- Vendors and equipment suppliers.

The report also includes information that may be useful for decision makers, such as regulators, and advisors for plant life management in NPPs.

1.1.4. Structure

The mechanisms of the major contributors to SCC are described in Section 2. SCC is a complex phenomenon driven by the synergistic interaction of mechanical, electrochemical and metallurgical factors. Perhaps the most critical factor concerning SCC is that three preconditions are necessary and must be present simultaneously. The elimination or reduction of any one of these three factors below some threshold level can, in principle, prevent SCC. The three necessary conditions are susceptible material; tensile stress component and aqueous environment.

Section 3 provides a summary of the major operational PWR and BWR service history relevant to ageing degradation by SCC. These incidents offer a perspective on the design bases and their conservatism relative to operating parameters.

Since SCC is a complex phenomenon involving synergistic interactions between metallurgy, chemistry and mechanics, it is necessary to expand knowledge in each technical field and then take actions to reflect the enhanced knowledge in the other fields. In Section 4, the ageing management programme is introduced to reduce the risk of damage due to SCC in nuclear power plants.

In Section 5, the inspection requirements and practices for piping, RPV internal, and vessel heal penetration are described with a description of the current view. Section 6 summarizes the current state of the art in the mitigation and repair techniques that have been developed to counter SCC related ageing degradation in both BWRs and PWRs. These techniques are described in more detail in this section.

The replacement of heavy components is the result of widespread stress corrosion of alloy 600 in the primary system. Component replacement is often the feasible solution to solve the problems associated with PWSCC of alloy 600. Even if mitigation and /or repair were a local solution, replacement offers many advantages when addressing the assortment of potential susceptible parts contained in a major component. In Section 7, the replacement methods are introduced. Section 8 summarized SCC management application. The summary is followed by the appendices.

Appendix 1 discusses the examples of the application of mitigation measures against PWSCC in alloy 600 nickel based alloy and associated weld metals are described.

Appendix 2 addresses the assessment and flaw analysis. Specific guidance is provided for the evaluation of components fabricated from austenitic alloys and affected by IGSCC. The actions needed in the event that plant specific flaw evaluations are required are further listed.

Appendix 3 lists the activities in the past which have been supporting the research on management of stress corrosion cracking.

2. MECHANISMS AND MAJOR CONTRIBUTORS TO STRESS CORROSION CRACKING

2.1. BASIC DAMAGE MECHANISMS OF STRESS CORROSION CRACKING

Stress corrosion cracking is a complex phenomenon driven by the synergistic interaction of mechanical, electrochemical and metallurgical factors. Both BWR and PWR components can suffer from SCC, which may have transgranular (through the grains) or intergranular (along the grain boundaries) morphology.

Sometimes the modes are mixed or the mode switches from one to the other. IGSCC and TGSCC can occur in the same alloy, depending on the environment, the microstructure, or the stress/strain state. SCC usually propagates perpendicular to the principal tensile stress. Cracks can also vary in the degree of branching.

All SCC has a brittle-like appearance, since cracks propagate with little or no macroscopic plastic deformation. An alloy affected by SCC does not usually display abnormal mechanical properties (yield strength and tensile strength) although this may be observed in certain classes of alloys; such as precipitation hardened stainless steels or as a result of irradiation damage. Many alloys are susceptible to SCC in at least one environment. However, SCC does not occur in all environments, nor does an environment that induces SCC in one alloy necessarily induce SCC in another alloy.

SCC is usually divided into an initiation and a propagation phase. The initiation time can vary significantly and can be up to several decades. The propagation phase is often divided into two parts, a ‘slow’ propagation phase and a ‘fast’ propagation phase of which the latter is usually characterized by crack tip stress intensities, K_I, exceeding a characteristic apparent threshold value in pre-cracked fracture mechanics type specimens known as K_{Iscc}.

Perhaps the most critical factor concerning SCC is that three preconditions are necessary and must be present simultaneously. The elimination of any one of these factors or the reduction of one of these three factors below some threshold level can, in principle, prevent SCC. The three necessary preconditions are:

— A susceptible material;
— A tensile stress component;
— An aqueous environment.

Figure 2.1 illustrates the critical factors for stress corrosion cracking.

2.2. MATERIAL ASPECTS

2.2.1. Major contributors to intergranular stress corrosion cracking in boiling water reactor nuclear power plants

There have been two major material factors that have contributed to IGSCC of austenitic alloys in BWR primary coolant systems: thermal sensitization and cold work. Historically, IGSCC of austenitic stainless steels and nickel based alloys occurred first in thermally sensitized materials which were then replaced with low carbon grades or stabilized grades. However, due to cold work of these low carbon and stabilized materials during fabrication, IGSCC has also subsequently occurred in them.

FIG. 2.1. Critical factors for stress corrosion cracking.

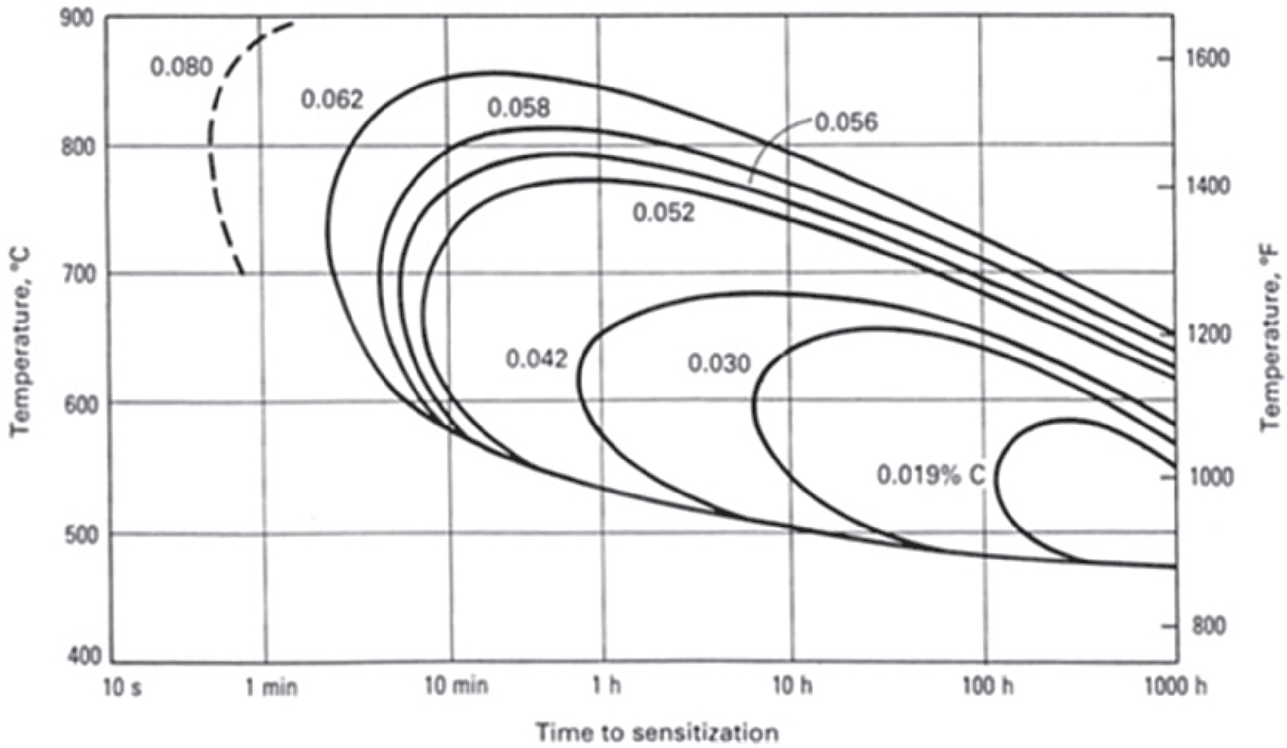


FIG. 2.2. Time-temperature-transformation (TTT) diagram for austenitic stainless steel showing the combinations of time, temperature and carbon content that lead to sensitization.

Thermal sensitization may occur in the heat affected zones of welds in austenitic alloys both during welding and also during stress relief heat treatments given to adjacent low alloy steel components. Austenitic stainless steels are sensitized when subjected to temperatures between approximately 500°C–800°C for times varying between tens of seconds and many hours depending on the carbon content (see FIG. 2.2.) Sensitization is caused by the formation of chromium carbides (e.g. $M_{23}C_6$) on grain boundaries and a concomitant depletion of chromium in the adjacent grains, illustrated in FIG. 2.3. The reduction in chromium concentration adjacent to the grain boundaries, which may be as little as 2% lower than the bulk concentration, gives rise to a reduction in passivity relative to the grains themselves and susceptibility to IGSCC.

Thermal sensitization can be counteracted by using either low carbon grades (L-grades; e.g. type 304L or 316L), in which the bulk carbon content is limited to C ≤0.03%, or by stabilized stainless steels, in which the majority of the

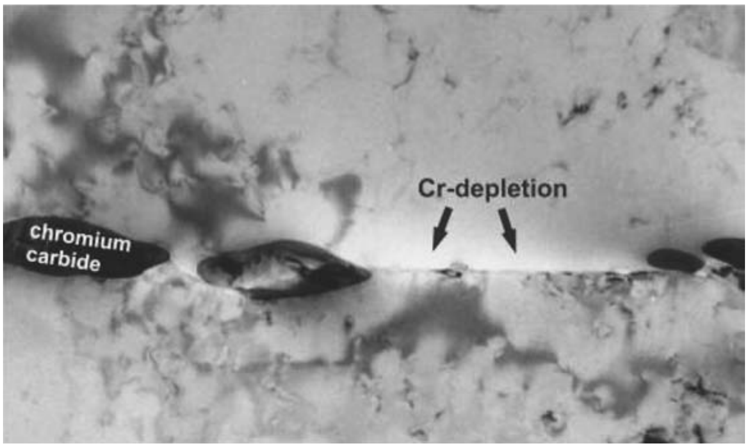


FIG. 2.3. Transmission electron microscope (TEM) image showing chromium carbide precipitates along a grain boundary and the zone of Cr-depletion [2.1].

carbon is bound by appropriate amounts of a strong carbide former such as niobium or titanium (e.g. type 347 and type 321). A combination of both low carbon content, and high stabilization ratio (e.g. Nb/C >13) may enhance the benefits. Similarly, alloy 600 and 182 have been modified with niobium additions to prevent thermal sensitization.

Stainless steel castings and welds, which have a duplex austenitic ferritic structure, are not susceptible to thermal sensitization, because of the high diffusivity of chromium in the ferrite. Consequently, they are not susceptible to IGSCC in BWR systems.

Subsequent to the introduction of both low carbon and stabilized grades of stainless steel, IGSCC occurred in these materials that were clearly not in a sensitized condition. It has been shown that their susceptibility to IGSCC is due to cold work induced during fabrication. Hardness levels involved have been above 300 H_V. In many cases, the initial cracking was found to be transgranular and then changed to an intergranular cracking mode. The initial transgranular cracking is often associated with a surface layer of cold work induced by grinding or other severe surface machining techniques. Failures have also occurred where the occurrence of IGSCC was attributed to the presence of either severe bulk cold worked material (e.g. cold bent piping).

The mechanism by which cold work renders austenitic alloys susceptible to IGSCC is not fully understood and is still being investigated. It is possible that there is an unfavourable interaction between deformation induced martensite, high residual stresses and localized deformation.

For high strength applications, alloy X-750 has been commonly used in one of two heat treatment conditions. X-750 is a precipitation hardened alloy with similar nickel and chromium contents as alloy 600. One heat treatment is known as equalized and aged (EQA), in which the material has a two-step thermal treatment: the first at 885°C for about 24 hours followed by ageing at 704°C for 20 hours. This material condition has been susceptible to IGSCC under BWR conditions. The second heat treatment known as high temperature annealing (HTA), has a single step ageing at 704°C for about 20 hours after solution annealing at 1093°C for one to two hours. The main goal of HTA treatment is to precipitate the strengthening gamma prime phase, $NiAl_3$, together with a fine, dense $M_{23}C_6$ carbide distribution at grain boundaries. In BWRs the second of these treatments is more resistant to IGSCC.

2.2.2. Major contributors to primary water stress corrosion cracking in pressurized water reactor nuclear power plants

In contrast to the IGSCC problems experienced in stainless steels in BWR systems, the same materials used in PWR systems have suffered from relatively few problems and those that have occurred have been mainly attributed to a combination of an inadvertent presence of oxygen trapped in stagnant regions combined with thermal sensitization or cold work [2.2]. Details on the environmental aspects are given in Section 2.3.2. However, nickel based alloys, particularly alloys 600, 132 and 182, have proved to be generically susceptible to intergranular stress corrosion cracking in normal specification PWR primary water systems, commonly known as PWSCC. The high strength analogue of alloy 600, alloy X-750, and to a considerably lesser extent alloy 718 with a somewhat higher chromium content, have also proved to be susceptible to PWSCC. Details are discussed below. WSCC

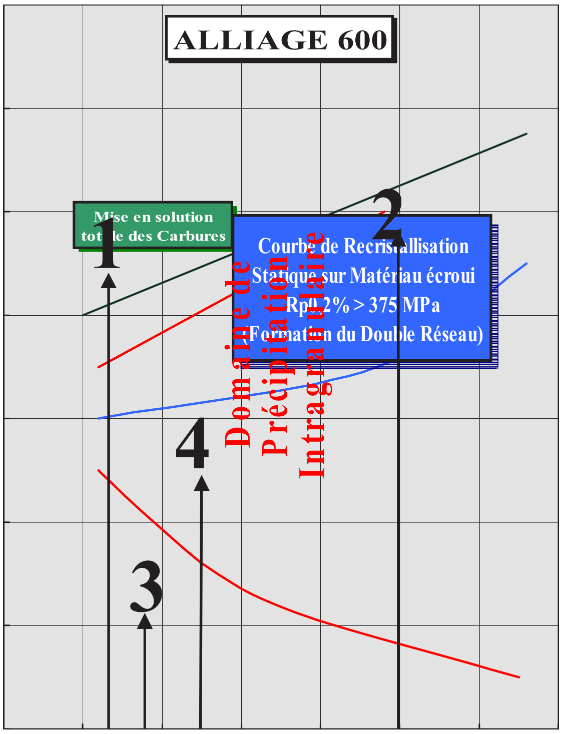


FIG. 2.4. Time-temperature-transformation (TTT) diagram of alloy 600 showing the relationship between carbon content, hot working temperature and time for carbide precipitation.

susceptibility in alloy 600 has been observed to be very dependent on the metallurgical structure, particularly the carbide morphology, and also cold work.

Nickel alloys (especially alloy 600) with many carbides on the grain boundary are found to be more resistant to PWSCC, whereas those with many intragranular carbides are the most susceptible. Since the carbon solubility in nickel alloys is low, carbon in solid solution combines with chromium to form chromium carbides during cooling from hot working temperatures. A time-temperature-transformation (TTT) diagram showing the relationship between carbon content, hot working temperature, and time is shown in Fig. 2.4.

The initial studies of the importance of the effect of carbide morphology on PWSCC susceptibility were carried out in connection with steam generator tube cracking. During the mill annealing process to produce steam generator tubing, the ability to produce a favourable microstructure depends on there being sufficient carbon in solid solution to precipitate as carbides. A high mill anneal temperature favours the precipitation of intergranular carbides to produce the desired microstructure. A low temperature mill anneal results in an insufficient supply of carbon for subsequent intergranular precipitation, so that a high density of intergranular carbides forms. In addition, subsequent thermal treatments (e.g. 705°C for 16 hours) may not significantly modify the microstructure or SCC resistance in this latter case. Another cause of failure of subsequent thermal treatment to give the favourable intergranular carbide morphology is tube straightening after mill annealing which generates dislocations on which carbides may precipitate preferentially. These microstructural considerations are not confined to tubing but also apply to thicker section forgings of alloy 600. A schematic representation of the carbide precipitation process for nickel alloys is shown in FIG. 2.5.

Microstructures that are most resistant to PWSCC have grain boundaries with a semi-continuous decoration of carbides (see FIG. 2.6). Thermally treated alloys (705°C for 16 hours) usually have improved PWSCC resistance for this reason. The mechanism by which PWSCC resistance is improved by grain boundary carbides remains open to debate. One hypothesis for the beneficial effect of intergranular carbides on PWSCC resistance has been proposed by Bruemmer who suggested that IG cracks are blunted by grain boundary carbides. In this mechanism dislocations are preferentially emitted from carbides at the crack tip, thereby reducing the stress concentration around them [2.3]. Another possible explanation arises from considerations of selective oxidation of chromium at the grain boundaries. In this case, carbides act a sink for oxygen and also present a more difficult diffusion path for

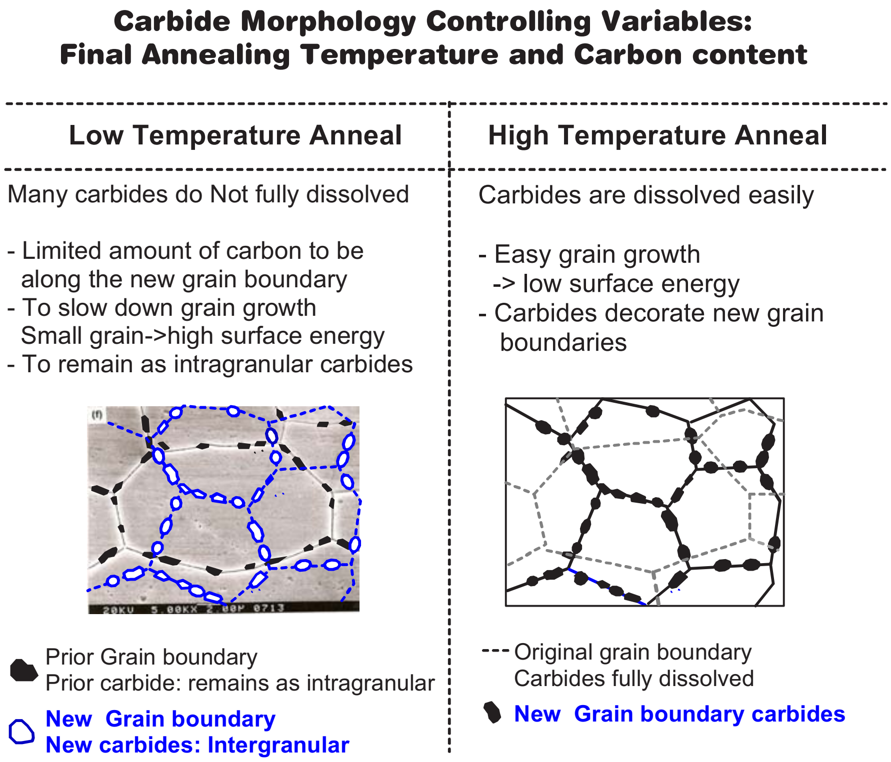


FIG. 2.5. Schematic of the carbide precipitation process.

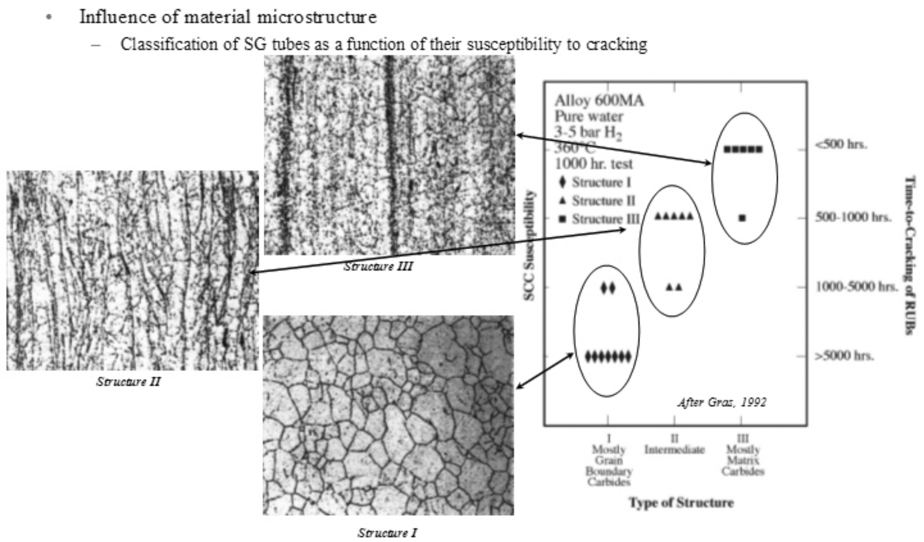


FIG. 2.6. Carbide classification system due to Vaillant.

oxygen. The role of carbon is not so clear in the case of alloy 182 and 82 weld materials, because it is preferentially precipitated as niobium carbide.

The use of alloy 600 for new and replacement components has been discontinued and has been replaced by alloy 690. The corresponding weld metals are alloy 52 and alloy 152. These alloys are characterized by significantly higher chromium content, approximately 30%. No cracking has been reported to date in these materials after up to 20 year service. Nevertheless, laboratory work has shown that PWSCC propagation can occur after at least 10% of unidirectional cold work in the plane of cold work or on steam generator tubes with a degraded microstructure. Many studies have confirmed the very high PWSCC initiation resistance of alloys 690, 152 and 52 at temperatures of up to 360°C.

Alloy X-750 is also used in PWRs for high strength applications. The second of the two HTAs described in Section 2.2.1 produces a microstructure that is also more resistant to PWSCC, presumably due to the carbide morphology. Alloy 718, another precipitation hardened high strength alloy used in PWRs for high strength bolts and springs, is highly resistant to PWSCC initiation, but more susceptible to propagation. A few cases of PWSCC have occurred where oxidation damage of grain boundaries occurred during fabrication, which circumvented the crack initiation phase. Alloy 286 is a gamma prime strengthened stainless steel used in some high strength fastener applications particularly when considerations of thermal expansion coefficients between austenitic components are important. It is, however, rather susceptible to PWSCC, particularly in the higher strength temper, and, when used, the applied stress must be very strictly controlled so as not to exceed the proportional limit even at stress concentrations.

2.2.3. Irradiation effects

Reactor vessel internals of both BWR and PWR plants are mainly fabricated from austenitic stainless steels. Unlike the fuel elements, which are removed after a few years of service, the internals are intended to remain for the full life of the plant and in consequence can be exposed to very high radiation doses, typically 5–10 dpa in a BWR and up to 80 dpa in a PWR (assuming a 40-year life cycle and depending on fuel management). With such high radiation doses, the material microstructure and mechanical properties change considerably; which have a significant impact on the stress corrosion susceptibility in both reactor types. Irradiation assisted stress corrosion cracking (IASCC) is therefore an important potential ageing degradation mechanism affecting the internals. Field experience and experimental work have shown that several austenitic stainless steels, such as types 304, 316L, 316CW and 347, are susceptible to IASCC. However, from a practical point of view, it may be difficult to decide whether cracking in the field is caused by IASCC or by other types of IGSCC [2.4, 2.8].

Neutron irradiation causes atom displacements from their equilibrium crystallographic locations thereby creating atomic scale point defects, i.e. vacancies and interstitials. Neutrons generate large cascades of point defects as energy transfer to the displaced atoms is significant so that the displaced atoms in turn continue and cause even more atom displacements. Subsequent diffusion of point defects to various sinks such as grain boundaries, dislocations and surfaces, leads to significant changes in microstructure and mechanical properties in metallic materials. Neutron irradiation effects are primarily athermal. However, in the case of thick section components, significantly higher temperatures than the surrounding aqueous coolant can be generated within the material by gamma heating. Such higher temperatures can have a significant effect on the likelihood of void swelling occurring. In addition, neutron capture reactions induce transmutation reactions and hence changes in chemical composition.

From a materials viewpoint the following radiation induced changes should be considered in relation to IASCC:

Microstructure

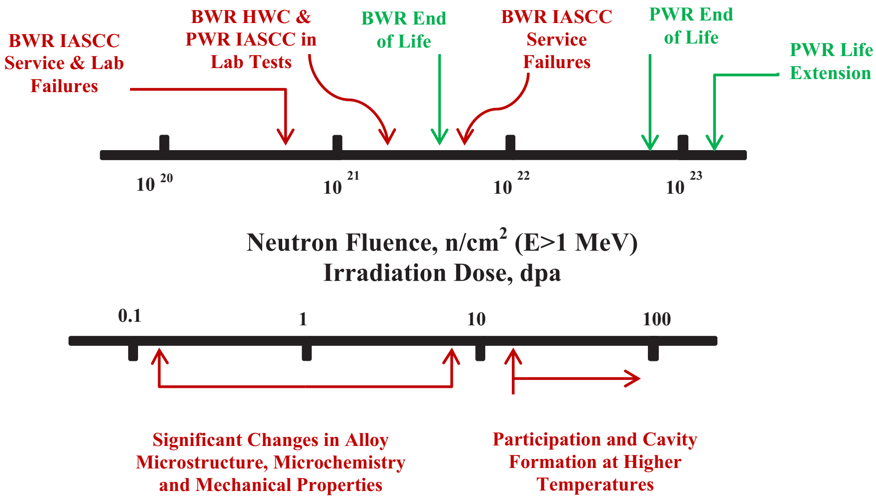
— High irradiation induced Frank dislocation loop density;
— Cavities (bubbles and voids).

Mechanical properties with saturation between 5 and 10 dpa

— Increased tensile properties (e.g. yield and ultimate tensile strengths up to approximately 1000 MPa);
— Decreased uniform and total elongation (e.g. <1% uniform elongation);
— Increased hardness;
— Decreased fracture toughness (e.g. down to ~45 MPa m1/2).

Chemical composition

— Radiation induced segregation (RIS) at grain boundaries (mainly, Cr, Mo and Fe depletion and Ni and Si enrichment).



***Note:** ~15 dpa ≡ 10^{22} n/cm^2 E ≥ 1 MeV (for PWR and BWR neutron spectra)*
~7 dpa ≡ 10^{22} n/cm^2 E ≥ 0.1 MeV (for PWR and BWR neutron spectra)

FIG. 2.7. Relationship between the irradiation effects and fluence.

Others

— Swelling as a result of cavity formation (at very high levels of neutron fluence);
— Radiation induced creep leading to stress relaxation.

Hardening and RIS are considered to be major factors likely causing IASCC susceptibility. The increase in tensile strength and hardening, based on recent knowledge of other types SCC in cold worked materials, is also seen to be an important parameter affecting IASCC susceptibility. In addition, it has been recently reported that helium bubbles (which also contain hydrogen) may concentrate preferentially on grain boundaries and may, therefore, play a significant role in IASCC [2.7].

RIS has a potentially significant impact on IASCC susceptibility particularly due to chromium depletion at grain boundaries and silicon enrichment. However, chromium depletion will only be significant in the more oxidising environment of BWR with normal water chemistry (NWC) but has no effect on grain boundary cracking susceptibility in hydrogenated PWR primary environments. On the other hand, silicon enrichment may give rise to silica-rich oxide films on the affected grain boundaries, which is unstable in both BWR and PWR environments.

The metallurgical consequences of neutron irradiation as a function of dose and their effect on IASCC susceptibility are shown in FIG. 2.7. It is noted that there are approximate thresholds of neutron fluence leading to IASCC susceptibility, which are conservatively considered to be about 1 dpa for austenitic stainless steels in BWR plants and about 3 dpa in PWR plants.

2.3. ENVIRONMENTAL ASPECTS

2.3.1. Boiling water reactors

IGSCC can occur and propagate in all grades and conditions of stainless steel and nickel alloys if the environment and electrochemical corrosion potential (ECP) are conducive to SCC. However, from a practical point of view, the commonly agreed threshold for the possible occurrence of IGSCC in BWRs, based on laboratory results and field experience, is an ECP ≥ -230 mV_{SHE} [2.9, 2.10].

Under BWR normal water chemistry (NWC) conditions, the ECP is mainly influenced by the presence of oxidizing radiolysis products, O_2 and H_2O_2, dissolved in the high temperature water. These environmental

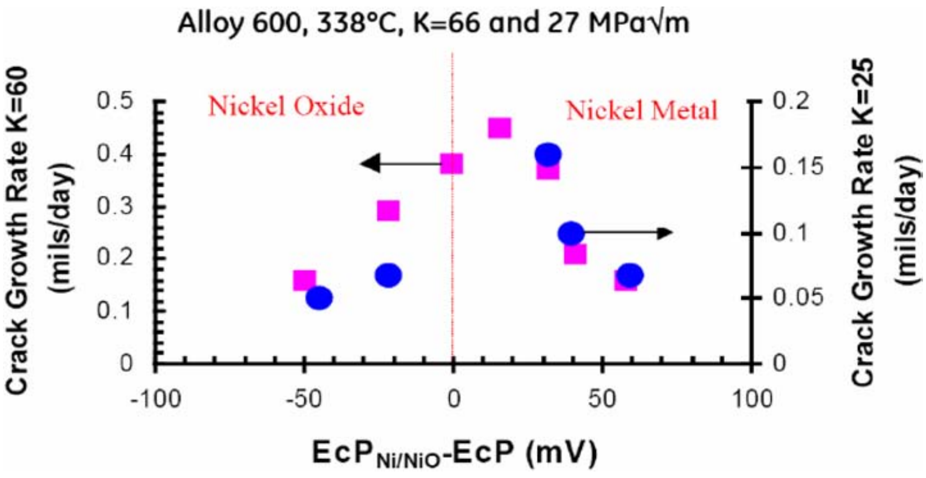


FIG. 2.8. Crack growth rate for alloy 600 in simulated PWR environment at 338°C [2.11].

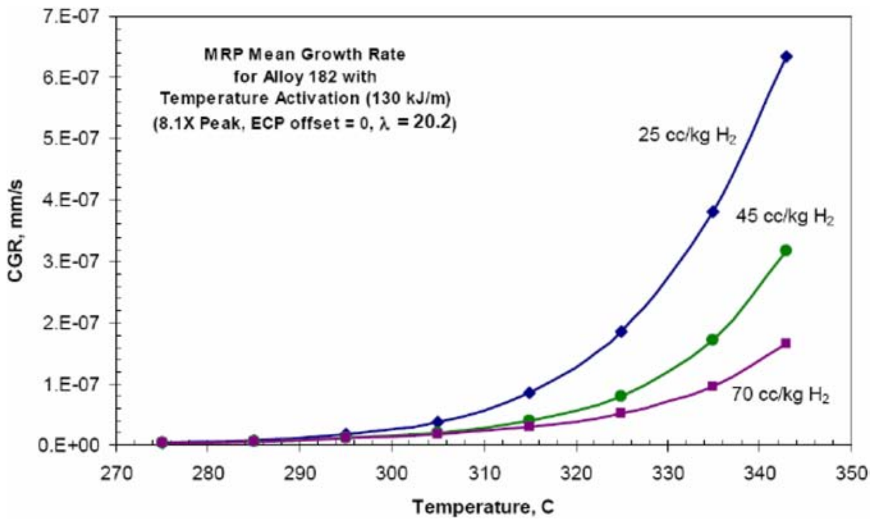


FIG. 2.9. Effect of temperature on PWSCC crack growth rate for alloy 182 [2.11].

conditions are potentially suitable for the occurrence of IGSCC among other contributing factors; notably metallurgical factors such as grain boundary sensitization or cold work, and a high stress. The main oxidant is hydrogen peroxide which exists in transiently in direct cycle BWR systems, but decomposes quite rapidly to oxygen on out of core surfaces away from the radiation field.

Other environmental conditions such as flow conditions, temperature and presence of impurities such as sulphate and chloride also influence the occurrence of IGSCC.

2.3.2. Pressurized water reactors

The main environmental parameters in PWR primary systems influencing the SCC are the temperature and the hydrogen concentration, and to a much lesser extent the Li-content, interior related pH-value, and the presence of zinc.

The effect of hydrogen on the crack growth rate in alloy 600 and its weld metals has been extensively studied during the last few years. It has been shown that the crack growth has a weak maximum in alloy 600, larger in the case of the weld metal alloys 182 and 132, at a hydrogen concentration approximately corresponding to the Ni/NiO equilibrium potential, as illustrated in FIG. 2.8.

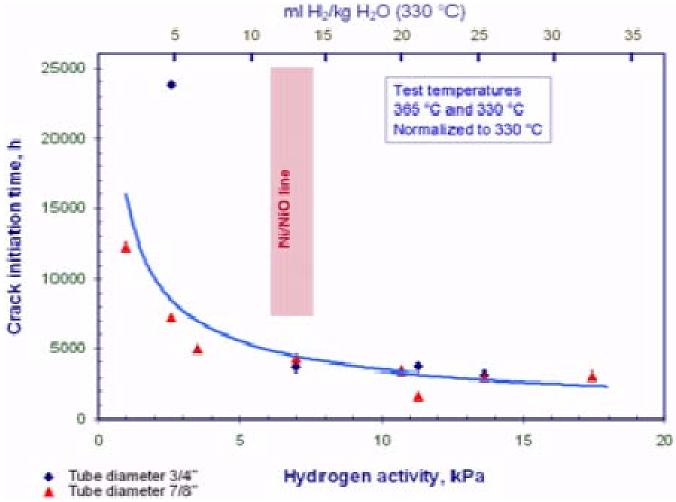


FIG. 2.10. The effect of hydrogen on initiation time. The data includes measurements at different temperatures normalized to the hydrogen partial pressure at 330°C [2.12].

In addition, the temperature has a considerable effect on the PWSCC crack growth rate as illustrated in FIG. 2.9 for alloy 182 for three different hydrogen concentrations. The temperature varies in the primary system with the lowest temperature in the cold leg pipework and the highest in the pressurizer. The figure shows that the crack growth rate increases significantly with temperature; it also shows that the effect of hydrogen is considerable and that at high hydrogen concentrations corresponding to potentials less than the NiNiO equilibrium potential the crack growth rate decreases (as in FIG. 2.8).

Studies of the effect of hydrogen on the initiation of PWSCC do not show a peak in susceptibility as a function of hydrogen concentration but resistance to cracking continuously increases with decreasing hydrogen content (see FIG. 2.10). However, in the normal range of hydrogen applied in operating PWRs (25–50 cc/kg), the influence of hydrogen on crack initiation is relatively small. Note that hydrogen partial pressures below 5 kPa (corresponding to 7–8 ml H_2/kg H_2O at 330°C) would be required to obtain significant benefits from reduced hydrogen concentrations. This is outside the present hydrogen specification and such low hydrogen contents have not yet been used in any operating PWR.

In summary, it would appear that there are certain differences between initiation and propagation data as regards the effect of hydrogen partial pressure. As illustrated in FIG. 2.11, the propagation data shows a weak maximum whereas certain initiation data a continuous decrease of the initiation time with increasing hydrogen. The apparently strong effect of corrosion potential close to Ni/NiO oxide stability has not been explained. Other oxides have been identified in long term experiments which might influence the cracking tendency.

The effect of lithium on PWSCC has drawn some attention in recent years, as extended fuel cycles imply operation at higher Li contents (and pH) during the beginning of a fuel cycle. Earlier crack initiation data indicate that moderate increases in Li content decrease somewhat the initiation time. The effect of higher Li content (>7 ppm) on the initiation of PWSCC has not been investigated. Regarding the effect of Li on the crack growth rate, data generated so far show no or little influence. EDF has observed a small detrimental influence of Li 3.5 ppm/ 2 ppm on crack growth rates (CGRs) (×2.6) but no influence on initiation (for times greater than 18 000 h).

Zinc is being added to some PWR primary loops mainly to reduce the activity buildup. The zinc concentration used is around 5–10 ppb. However, zinc may also have a small but significant beneficial effect on the initiation of PWSCC. Zinc additions have also been applied in plants for such reasons. The additions are larger and up to 40 ppb have been used (see Section 6.2.7). Regarding crack growth, the available data are inconclusive, but seem to indicate that there is little or no effect of Zn.

The effect of any future changes in PWR chemistry should be carefully considered with respect to PWSCC of alloy 600, alloy 182 and other structural materials due to the unresolved contradictions between initiation and propagation data. Utilities are recommended to carefully study and follow ongoing research. It should also be pointed out that a change of hydrogen can also influence activity transport and buildup as well as AOA conditions. These questions have not yet been resolved.

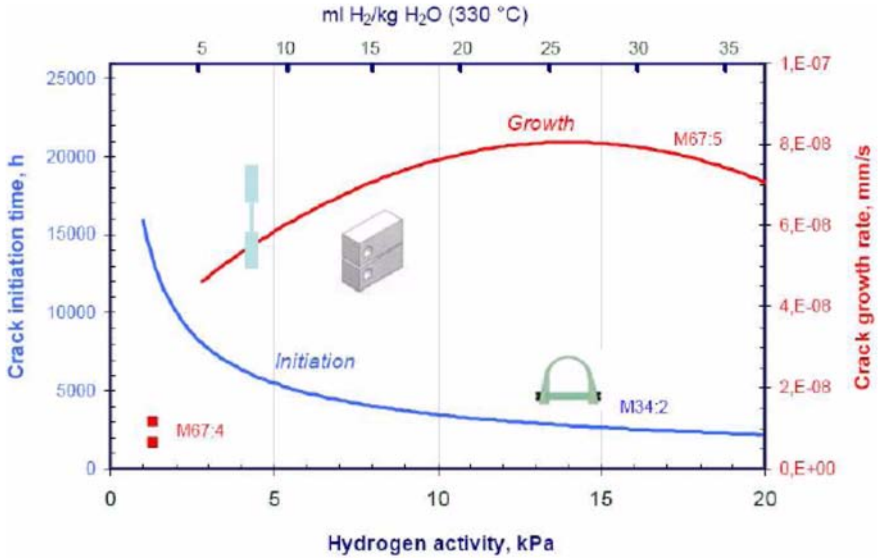


FIG. 2.11. Comparison of the effect of hydrogen on initiation and propagation of PWSCC in alloy 600 MA [2.12].

2.3.3. Irradiation effects

Irradiation effects must be considered for their possible influence on IASCC for type 304ss or 316CWss in PWR core internals. In PWRs, the production of radical oxidants due to neutron and gamma irradiation is suppressed by adding 25–35 cc/kg STP·H_2O hydrogen in the primary water. Subsequently, there is a report that dissolved hydrogen concentration could play a role in IASCC susceptibility in PWR primary water environment, presumably due to hydrogen embrittlement. It also has to be noted that for highly irradiated austenitic stainless steel, intergranular cracking has been observed during slow strain rate testing in an inert environment. Thus, the aqueous environment appears to accelerate initiation and propagation but is not a necessary condition according to this research result [2.4].

2.4. STRESS ASPECTS

2.4.1. Boiling water reactors

From a theoretical point of view, the role of tensile stresses is important for damaging or rupturing protective oxide films during both initiation and propagation of cracks. In the field, almost all SCC cases in BWR components occur in the vicinity of welds where the level of residual stresses produced by weld shrinkage is a very important factor having an impact on both crack initiation and crack propagation. For austenitic steels, SCC cracks propagate mainly through the heat affected zone (HAZ) of the base metal. For dissimilar welds, cracks are observed mainly in the weld metal alloy 182 (in some case propagating a short distance into the base material). Not only residual stresses but also cold work of the material can have pronounced effects on crack initiation and crack propagation.

The stress level at the surface triggers the process of crack initiation and these stresses may be either applied or residual. Flaws and other surface imperfection can act as stress raisers increasing local stresses at the (near) surface. Thus, corrosion attack (pitting or intergranular corrosion) or fabrication/welding defects/imperfections can act as a starting point for SCC. Very high surface stresses may also result from cold work introduced by fabrication processes such as machining, grinding or other surface finishing operations. Cold work can allow local near surface stresses to remarkably exceed the original yield strength of the bulk material.

In contrast to crack initiation, the stress dependence of crack growth can be more easily quantified. Crack growth is driven by the sum of stresses due to operational and residual manufacturing stresses and SCC growth rates can be correlated with the local stress intensity factors K_I. Compared to the base metal, the calculation of the stress-state in welds is more complex and residual stress measurements are much more difficult because of

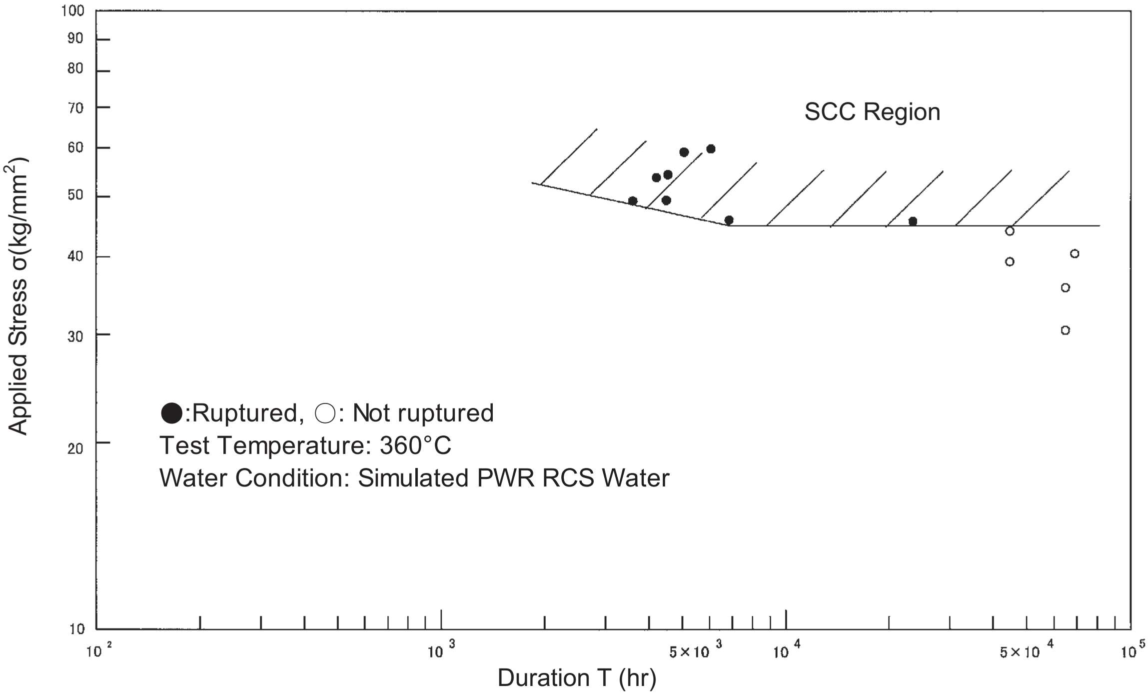


FIG. 2.12. Results of constant load PWSCC tests for alloy 600 base metal.

anisotropy of the weld metal microstructure. Nevertheless, as described in Section 6, with a given stress profile the time-dependent crack propagation can be derived for the purpose of flaw analysis. For mitigation of SCC by stress improvement, different processes have been developed to reduce surface stresses or introduce compressive stresses at the surfaces exposed to BWR coolant; e.g. improved welding techniques, post weld heat treatment and peening (for more details see Section 6.1).

2.4.2. Pressurized water reactors

Constant load PWSCC tests have been performed on alloy 600 base metal, alloy 690 base metal and alloy 132/182/82 weld metals. Some of these results are shown in FIG. 2.12–2.14. As a result, alloy 600 and alloy 132/82 have identifiable threshold stresses but for alloy 690 and the associated high Cr weld metals (152/52), no threshold stress has been measured in PWSCC tests since no cracking occurred [2.13].

In general, few instances of IGSCC in stainless steels have been observed in PWRs. There are however many SCC issues with alloy 600 and other Ni-based alloys. However, contrary to BWR experience, few PWR SCC issues are associated with weld HAZs. Most cracking has been observed in wrought materials remote from HAZs or in weld metals. The surface residual stress due to grinding or other surface finishing operations plays an important part in all cases and strain reversals in cold worked material are always involved.

2.5. IRRADIATION-ASSISTED STRESS CORROSION CRACKING

For BWRs two thresholds values for the onset of irradiation assisted stress corrosion cracking (IASCC) have been reported depending on the stress level of the component. For components with high tensile stresses the threshold is ~5×10^{20} n/cm^2 (E>1MeV); with lower tensile stresses the threshold is ~2×10^{21} n/cm^2. Under BWR conditions, stress relaxation by irradiation creep can be expected at welds of near-core components.

Many experiments (including post irradiated examination) have been carried out worldwide to get a better understanding of and to establish a database for IASCC of austenitic stainless steels. In 2007, two relationships

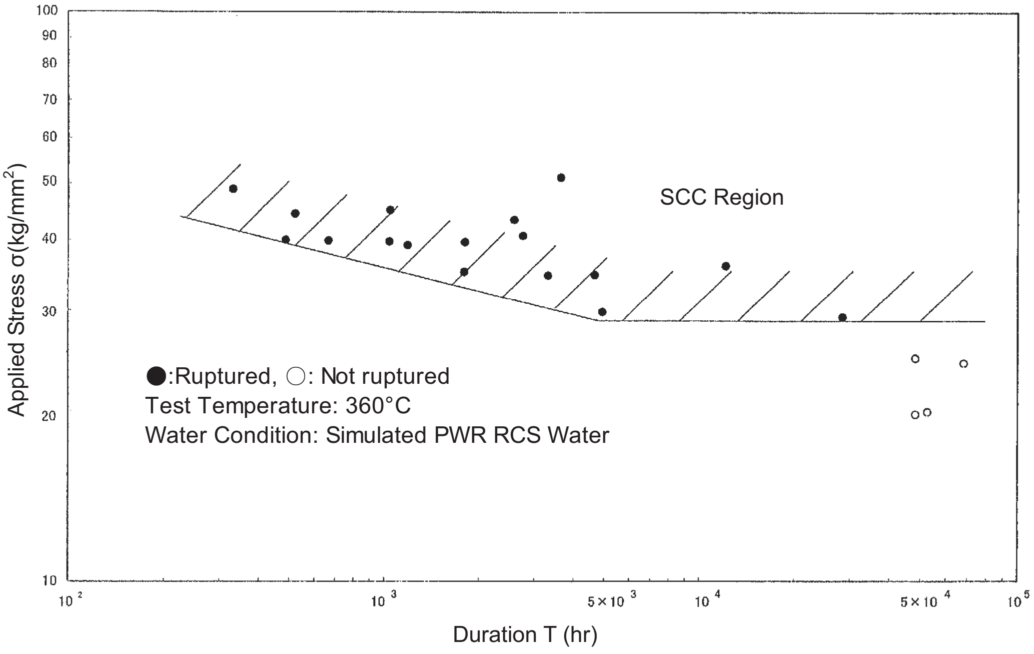


FIG. 2.13. Results of constant load PWSCC tests for alloy 132/82 weld metal.

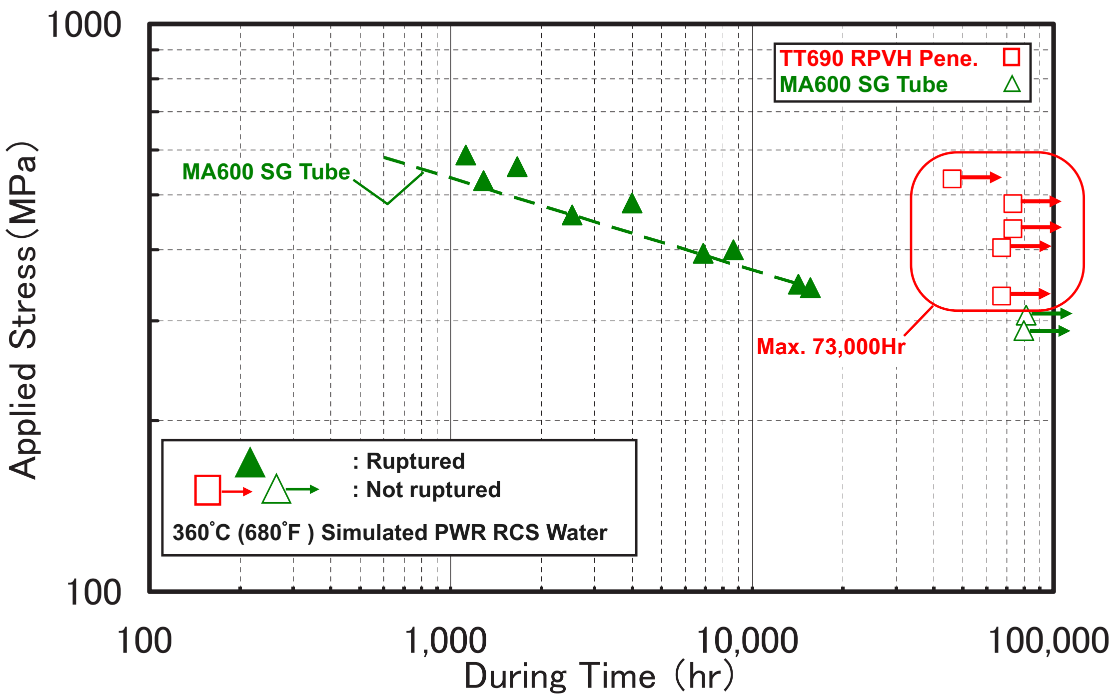


FIG. 2.14. Results of PWSCC tests on alloy 690 base metal and Alloy 600 base metal showing no cracking detected in alloy 690 after test times up to ~75 000 hours.

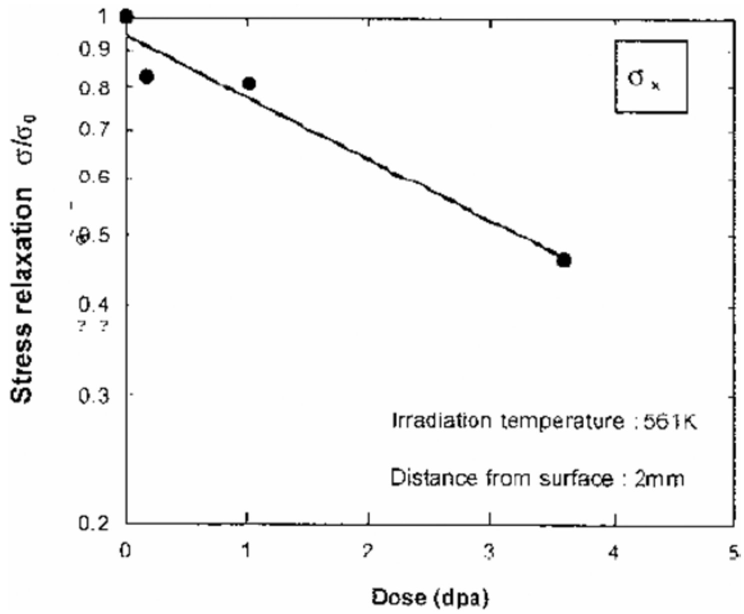


FIG. 2.15. Radiation induced stress relaxation of type 304 [2.14].

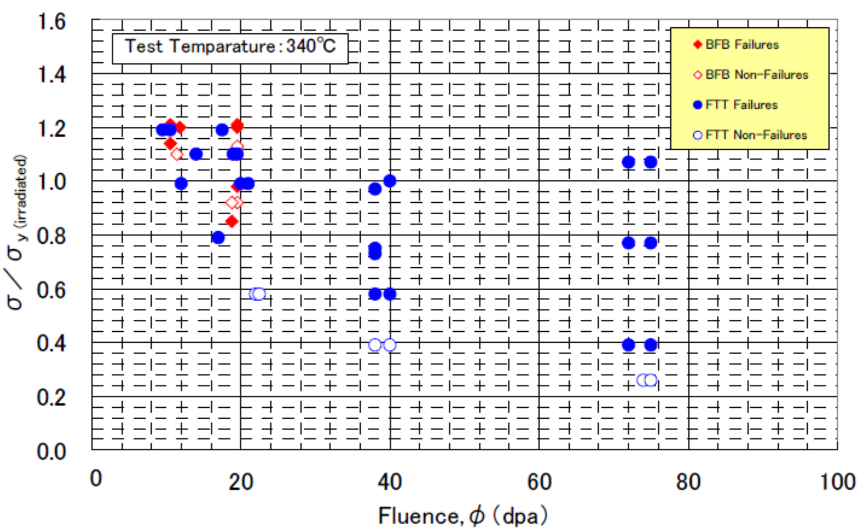


FIG. 2.16. SCC test results as stress/σ_y vs. fluence in dpa (at 340°C) [2.8].

between IASCC susceptibility of 316CWss and applied stress were published that were based on experimental results in simulated PWR primary water using irradiated samples removed from several operating plants. Both studies reached the same conclusion that IASCC in 316CWss was characterized by a threshold stress, as shown in FIG. 2.16 and FIG. 2.17. Although there are some differences in the threshold stress reported in both papers, the threshold stress for IASCC initiation clearly decreases with increasing fluence. Highly irradiated 316CWss at a neutron fluence of more than ~30 dpa showed IASCC susceptibility above relatively low stresses between ~0.4 σ_y to 0.6 σ_y (where σ_y is the as-irradiated yield stress of typically ~1000 MPa at such high fluence).

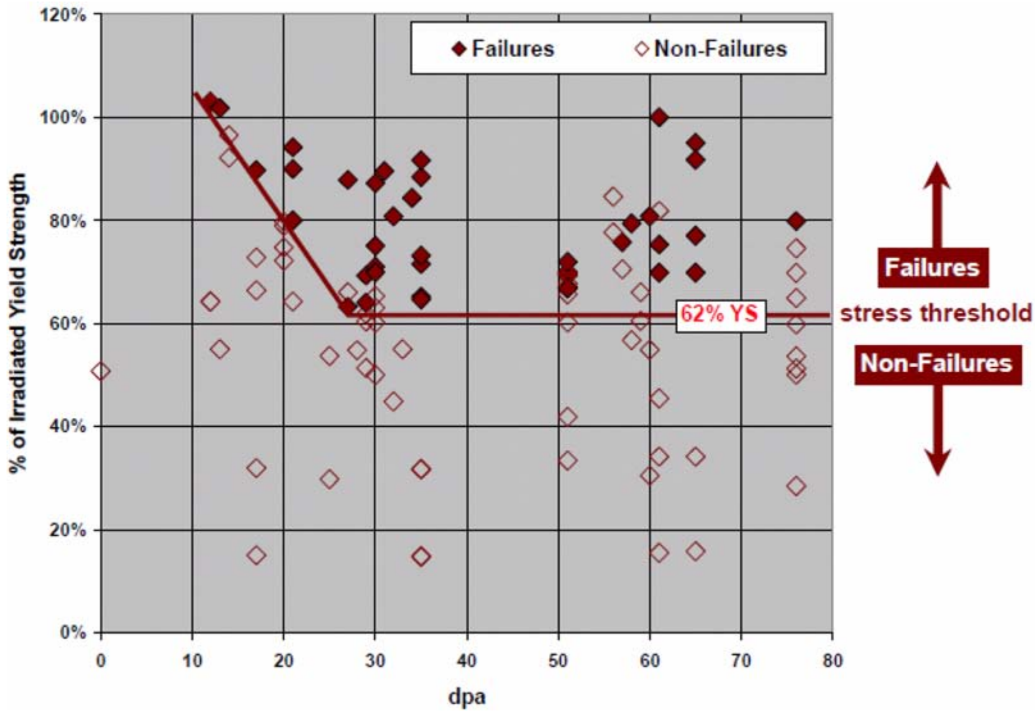


FIG. 2.17. SCC test results as stress/σ_y vs.fluence in dpa (at 340°C) [2.15].

2.6. TRANSGRANULAR STRESS CORROSION CRACKING

A SCC issue common to both types of light water reactors is transgranular stress corrosion cracking (TGSCC) of austenitic stainless steels, which is primarily due to chloride contamination although other halide anions such as fluoride can also cause TGSCC. It is generally a problem that initiates on the outside surfaces of austenitic stainless steel components mainly due to lack of attention to adequate cleanliness. Wetting due to condensation or nearby water leaks can allow an aqueous environment to form that leads to TGSCC that is usually accompanied by pitting or crevice corrosion. The stress required for chloride induced TGSCC is relatively modest, the threshold being close to the proportional limit of solution annealed austenitic stainless steels. Implementation of known procedures that ensure adequate surface cleanliness is a continuing necessity that requires careful management attention at all stages of construction and operation of nuclear power plants.

One issue having an impact on the risks of chloride induced TGSCC of austenitic stainless steels is the choice and specification of thermal insulation materials. Fibreglass thermal insulation has been used predominantly in the past and has the advantage of having large concentrations of soluble silicate which have a favourable buffering action in the presence of chloride contamination of external surfaces of austenitic stainless steel. The allowable limits for surface chloride contamination in combination with the soluble silicate content of thermal insulation are encapsulated by the Karne’s diagram (see FIG. 2.19). In newer plants, mineral wool insulation is tending to replace fibreglass in many countries because of concerns about clogging of reactor building sump pump filters during major loss of coolant accidents caused by debris from glass fibre shredding. Mineral wool insulation is less prone to clogging such filters but has the disadvantage that it has much less soluble silicate and is therefore much less tolerant of surface chloride contamination with obvious consequences for management of surface cleanliness.

Chloride induced TGSCC can also occur from internal surfaces, generally in dead legs and stagnant regions due to the simultaneous presence of chloride contamination and oxygen. The combinations of chloride contamination and oxygen concentration leading to SCC in both solution annealed and sensitized austenitic stainless steels are shown in Fig. 2.19. One location that has been rather frequently affected in PWRs is in canopy seals that assure the pressure boundary of threaded connections in the control rod drive housings that are located above the reactor pressure vessel upper head. Such leaks have caused serious boric acid corrosion of the upper head low alloy steel.

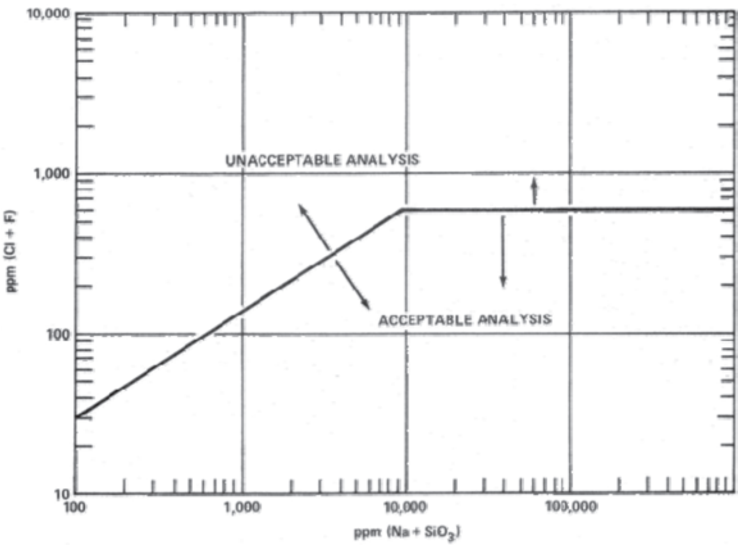


FIG. 2.18. Karnes ASME (USNRC Reg. Guide 1.36) showing the safe and unsafe areas for chloride and soluble silicate in insulation material.

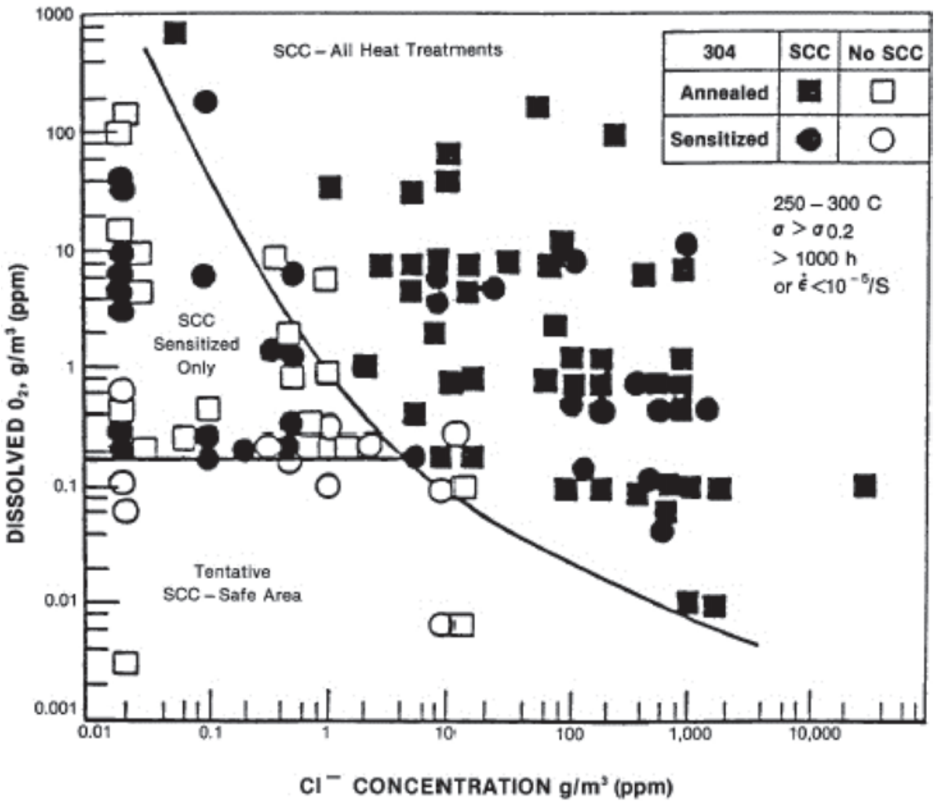


FIG. 2.19. Effect of chloride concentration on the critical concentration of dissolved oxygen for SCC in high temperature water [2.17].

The origin of the problem of cracking of canopy seals is air bubbles that are trapped during refuelling when the reactor pressure vessel is open to air followed by inadequate de-oxygenation procedures capable of removing the air bubbles from such locations with a very complicated pathway to the reactor vessel itself. Procedures during plant startup for eliminating these air pockets vary between operators but the one acknowledged reliable method involves completing the final fill of the primary circuit after a vacuum pump has been connected to a penetration in the upper head.

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3. OPERATING EXPERIENCES

This section provides a summary of the major operational PWR and BWR service history relevant to ageing degradation by SCC. These incidents offer a perspective on the design bases and their conservatism relative to operating parameters. It is particularly noteworthy that each has been resolved by qualified repair programmes. Nozzle cracking, stub tube cracking, safe end cracking and closure stud cracking are all age related degradation mechanisms; which have been effectively managed. The OECD/NEA SCAP event database will provide details of previous SCC events.

3.1. BOILING WATER REACTORS

For SCC, as already noted, all three prerequisites; material condition, environment chemistry and stress must be fulfilled. In the original design of LWRs, SCC phenomena were not explicitly considered until, beginning in the mid-seventies, the worldwide BWR fleet began to suffer from a sequence of IGSCC incidents.

The ensuing damage resulted in substantial economic losses for utilities, especially in the eighties. A tremendous amount of effort was devoted during the ensuing years to mitigate IGSCC and, in particular, to improve the water chemistry. Due to these efforts, plant availability has increased and, in addition, radiation buildup has been effectively

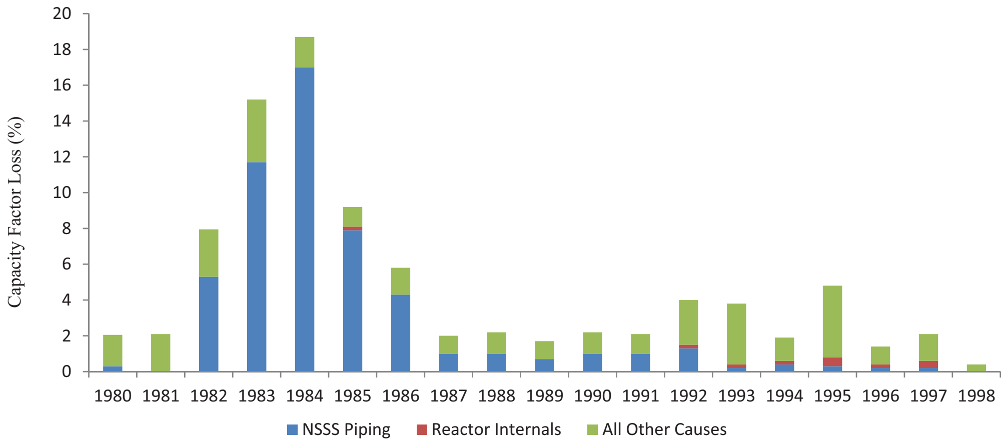

FIG. 3.1. Corrosion related capacity factor losses due to corrosion in BWRs [3.1].

mitigated. The evolution of capacity factor losses, as of 1980, according to FIG. 3.1, reflects that early mistakes have been corrected over time; partly by improving water chemistry but also through component replacements.

The dominating early failure type in BWRs was IGSCC of sensitized stainless steel and more recently of cold worked stainless steels; e.g. type 316L. In PWRs, steam generator tube cracking issues were dominant. However, unpredicted SCC attacks still occur and influence plant performance and availability.

3.1.1. Piping

In 1965, the first SCC incident in the heat affected zone (HAZ) of non-stabilized austenitic stainless steel was reported at Dresden-1 (USA) in a by-pass line of a recirculation loop manufactured from type 304 stainless steel. The phenomenon was initially considered as plant specific. After Dresden-2 (USA), further recirculation lines in other plants suffered from SCC in HAZs and in 1974 the SCC phenomenon was recognized as a generic issue for type 304 stainless steel in BWRs. As a consequence of these occurrences, specific recommendations were given in NUREG-0313 Rev.2 that provided the technical bases regarding actions that could be taken to ensure that the integrity and reliability of BWR piping be maintained.

From this point in time, an increasing number of stainless steel pipes suffering from SCC were found. Failure analyses revealed that welding related thermal sensitization of non-stabilized stainless steels was the causative factor.

In 1992, IGSCC was found in the HAZ of pipe welds in titanium stabilized stainless steels (type 321) components in the German plants Würgassen and Brunsbüttel. The resultant increased inspection programme for all other German BWRs initiated by these occurrences revealed further cracking in systems containing hot reactor water at operating temperatures above 200 C. The systems predominantly affected were the so-called reactor water cleanup and pressurized bearing water systems.

Failure analyses revealed that the predominant root cause was welding related thermal sensitization of stabilized steels that showed relatively high carbon content and a low stabilization ratio (Ti/C). However, around the same time, IGSCC was also found in non-sensitized stabilized stainless steels (mostly type 321 and in some cases type 347). This cracking was related to either severe cold work present on the surface and in other cases to moderate cold work in combination with crevice conditions formed by excess penetration and shrinkage in the root area of the welds. FIG. 3.2 shows intergranular cracking that was attributed to the latter kind of scenario [3.2].

In 1997/1998, Forsmark 2/1 reported IGSCC in the HAZ of a low carbon austenitic stainless steel of a type 316NG pipe bend after ten years of operation. The material was characterized using several methods. Intergranular cracking occurred very close to the fusion line, within 0.3 mm in this case, and weld strain measurements revealed very high residual strains of up to 20% in this zone. No sensitization of the material was detected. Cracking was concluded to be caused by SCC, which had occurred in deformed, non-sensitized steel. Deformation of the HAZ was mainly due to weld shrinkage [3.3].

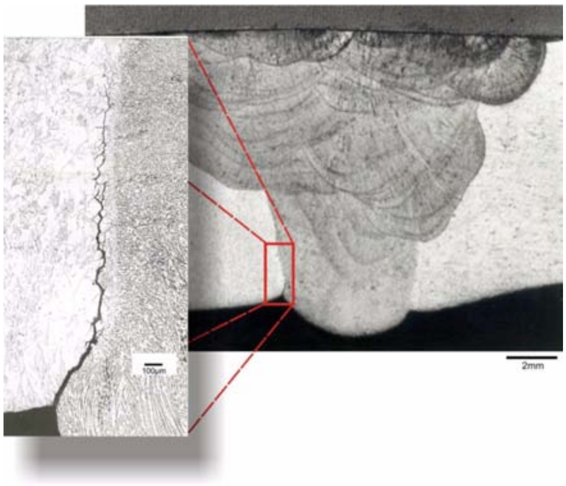

FIG. 3.2. IGSCC initiated at a notch root as a result of moderate cold work and a crevice condition.

Starting in 2002, SCC of non-sensitized type 316(NG) recirculation piping was found in many Japanese BWRs. Cracks were circumferential and also close to the weld fusion line. Cracks did propagate towards the weld metal and in most cases arrested at the weld metal interface. In the remaining cases, the cracks propagated into the weld metal but stopped after crack growth of less than 4 mm. Most cracks were less than 10 mm in depth (i.e. less than 25% of wall thickness*)* regardless of the operating time or the pipe diameter. The cracking rate was highest in the main pipes with a nominal pipe size of 600 mm. Surface cold work resulting from machining prior to welding and plastic strain caused by weld shrinkage were considered to be the causative factors.

3.1.2. Vessel penetrations and nozzles

Several GE BWRs have experienced control rod drive (CRD) stub tube SCC since the start of operation. All stub tubes in these BWRs were originally fabricated from type 304 furnace sensitized stainless steel. Furnace sensitization of type 304 stainless steel was eliminated from later BWR vessels by changing the manufacturing sequence. High residual stress in the sensitized weld material was attributed as the root cause [3.4].

The first SCC in nozzles was observed at alloy 600 recirculation inlet nozzle safe ends of Duane Arnold (USA) in 1978. Cracking occurred in the HAZs at thermal sleeve to safe end welds of all eight nozzles and one of them was through-wall thus leading to leakage of reactor coolant. Root causes were identified as high residual stresses and crevice conditions due to thermal sleeve weld design [3.5]. SCC of alloy 182 weld metal was first observed in the weld butters of recirculation inlet/outlet nozzle safe ends at Pilgrim (USA) in 1984. Similar alloy 182 cracking in nozzle safe end welds has been found in several plants.

In 1988, SCC was found in type 304 stainless steel of an in-core monitor (ICM) housing of Hamaoka-1 (Japan). Cracking was due to IGSCC in the HAZ of the housing attachment weld to RPV and was through-wall causing leakage. It was considered that leak path was made possible by an unusual HAZ profile due to excessive weld heat input of the ICM housing. Cracking of type 304 stainless steel ICM housings has been found in three other Japanese BWRs.

In 2001, leakage from a CRD stub tube due to SCC of alloy 182 weld metal was found in Hamaoka-2 (Japan). High residual stress due to the welding procedure specific to stub tubes at peripheral locations (see FIG. 3.3. was considered to increase SCC susceptibility. No other cracking of alloy 182 at CRD stub tubes has been reported.

3.1.3. Reactor pressure vessel internals

The first reported case was found in October 1978 in the core spray sparger piping at a BWR facility. Subsequent examinations at the same plant in January 1980 identified additional cracking indications. Also, in January 1980, a second plant identified cracking in both the upper and lower core spray spargers. At both plants, the cracking was hypothesized to be initiated and propagated by IGSCC due to possible sensitization and cold work during the fabrication and installation process.

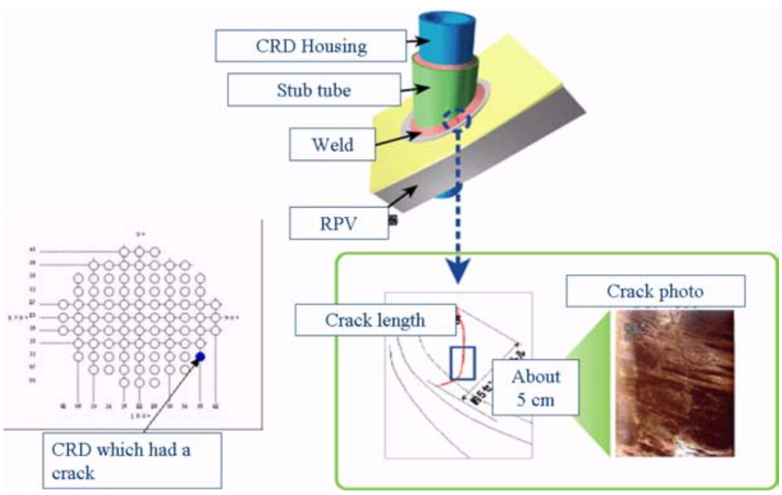

FIG. 3.3. CRD stub tube leak at Hamaoka NPP.

Responding to these instances of cracking in core spray spargers, requirements for augmented inspection programmes were implemented and addressed the performance of core spray piping and sparger examinations at an increased frequency using improved inspection techniques.

Most BWR utilities have been routinely performing non-destructive examination of core spray piping and spargers in accordance with the augmented inspection programmes. As a result of these routine examinations, additional cracking in the above mentioned locations and cracking in new locations have been observed. Additional information on this matter can be found in IAEA-TECDOC-1471.

The first documented incident of cracking in a core shroud was reported in August, 1990 at the Kernkraftwerk Mühleberg BWR (GE-type BWR/4). A metallurgical sample in 1992 [3.6–3.8] confirmed that the cracking was intergranular and affected both the weld structure and the base metal. The vendor`s diagnosis of the root cause of the cracking was IASCC promoted by weld residual stresses and possible corrosion induced oxide wedging stresses [3.8].

In 1994, IGSCC was found in the core shroud of the German plant Würgassen. In this case, the top guide and core plate manufactured from niobium stabilized type 347 suffered from cracking. The root cause of this failure was found to be a sensitized microstructure as a result of a post weld furnace heat treatment (PWHT) with the principle aim of reducing the residual stress state. The top guide and core plate affected were manufactured from one identical heat of type 347 that unfortunately exhibited a high carbon content in combination with a low stabilization ratio, thus leading to sensitization during the aforementioned furnace anneal. Residual stress from adjacent welds not subjected to PWHT contributed to the stress leading to IGSCC.

In 1991, a through-wall crack was observed in the top guide of a GE BWR/2 in the USA. An inspection at the plant’s next refuelling outage revealed two more cracks similar to the first. The cracks were located in an un-notched area of the beam. The material of the beam was type 304 stainless steel but no cold work or stress risers associated with the cracked regions were found. In addition, there was no indication of an overload condition. The most likely cause of cracking in this case was thought to be IASCC.

Cracking of L-grade stainless steel was reported in 1994 for the first time in BWRs [3.6]. Cracks were found in the upper core shroud weld areas of two GE BWR/4s in material certified as type 304L stainless steel [3.6, 3.9]. Previously, cracking had been found only in type 304 stainless steel, which has higher carbon content. Hot operating times for these ‘L-grade’ plants were 10 and 11.3 years. In both of these plants, indications were initially reported in the weld HAZ. Low carbon material does not normally sensitize during welding and the neutron fluence in this location was believed to be below the established threshold for IASCC [3.9]. Since 1994, the number of affected plants has continued to rise and in 2002 a total of 13 cases of cracked L-grade core shroud were reported [3.10].

In 2001, SCC of non-sensitized type 316L was observed in the HAZ of core shroud welds at Fukushima Daini-3 (Japan). Cracking initiated by a transgranular mode in the surface cold work layer from machining and then propagated by an intergranular mode. Since then, SCC of type 316L core shrouds has been found in many Japanese BWRs. Cold work by machining and hard grinding were thought to accelerate SCC initiation.

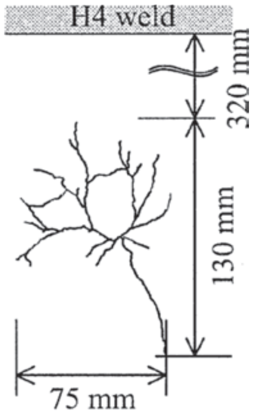

FIG. 3.4. An example of radial shaped SCC crack on a core shroud middle shell remote from any weld.

At lower core shroud ring welds such as H6a, cracking was circumferential and propagated along almost full length of the weld in several plants. At the middle shell welds such as H3 and H4, cracks were radial and some were observed at hundreds of mm away from any weld, as shown in FIG. 3.4. In most cases of this kind, cracking was found to be very shallow.

Over the last twenty five years there have been several instances of cracking in replaceable alloy X-750 jet pump hold-down beams, some resulting in failure. The first two jet pump beam failures occurred in 1979 and 1980. The beams were of the BWR/3 design and were supplied in the equalized and aged (EQA) condition. IGSCC was determined to be the failure mechanism. Subsequent inspections at other plants revealed crack indications in additional jet pump beams. The observed IGSCC cracks initiated in the thread region and then propagated across the beam essentially following the high stress trajectory (perpendicular to the beam axis). The cracking occurred on both sides of the beam bolt hole.

Design improvements were made to extend the service life of these jet pump beams including a modified heat treatment, lower preloads, and a larger cross section to lower the stress.

The second region of cracking in alloy X-750 was associated with the loss of a jet pump inlet mixer in a BWR/6 in September 1993. The inspection confirmed that beam failure was responsible for the event. The beam was also supplied in the EQA heat treat condition. Subsequent metallurgical analysis established that the cracking mechanism was IGSCC, consistent with past failures. Cracking was found to have been present in the ‘ear’ location at the end of the beam. In May 1994, a second failure in the transition region occurred in another plant after several years of operation. A failure analysis confirmed that the cracking mechanism was IGSCC. In contrast to all earlier failures, this particular beam had received the improved high temperature, single step ageing treatment (HTA).

In January 2002, a jet pump beam failure occurred at Quad Cities 1 due to IGSCC in a beam region that had not previously experienced cracks and that is not normally evaluated during in-service inspection. The beam that failed was an original component that had been in service for approximately 30 years. The failure location was about midway down the transition region between the thick centre part of the beam and the thinner ends. This occurrence emphasizes the need to replace older beams with beams of newer design and improved heat treatment.

Alloy X-750 in the EQA condition is known to be susceptible to IGSCC initiation and growth in the BWR environment. As previously described, there have been several failures of beams in the EQA condition, including the recent Quad Cities 1 failure. The HTA condition was found to be more resistant to IGSCC initiation than the EQA condition under the same loading conditions. However, the test data indicate that even the HTA material condition will eventually fail by IGSCC; but at higher stress levels than in the case of the EQA heat treatment condition. Furthermore, as the beam age increases, the probability of IGSCC initiation increases.

In 2006, during an in-vessel visual inspection, it was discovered that two of the X-750 tie rod upper supports at Hatch Unit 1 had experienced cracking. The cause for the cracking was determined to be IGSCC, likely caused by large sustained stresses in the alloy X-750 material during normal operation. Alloy X-750 material is susceptible to IGSCC if subjected to large sustained tensile stress conditions.

Jet pump BWRs are designed with access holes in the shroud support plate; which is located at the bottom of the annulus between the core shroud and the reactor vessel wall. The access hole cover of these GE-BWRs is made from alloy 600 material [3.6]. In January 1988, intermittent short cracks were found in the weld heat affected zone around the entire circumference of the alloy 600 covers at Peach Bottom Unit 3.

The cracking was probably caused by high residual stresses resulting from welding, together with a possible crevice geometry, when combined with less than ideal water quality. This combination presents a condition conducive to IGSCC [3.17]. GE SIL 462 S1 [3.18] provides recommendations concerning management of cracks in access hole cover.

Cracking of shroud head bolts (SHB) was observed at several BWR/4 and BWR/3 at the beginning of the 1980s. The first incidences of cracking occurred in the alloy 600 shaft of the SHB in a creviced region formed by a type 304 SS sleeve welded to the bolt shaft. The cause of failure was confirmed to be crevice accelerated IGSCC. Since then, several cracked SHBs of the original design have been found in pre-BWR/6 plants.

In 1993, a complete failure of the shaft of one of the original design SHBs occurred at a BWR/4. The failure location was different from the one previously mentioned. The failed bolt separated approximately 68 inches above the bottom of the bolt at the weld connection between the lower portion of the alloy 600 rod and the type 304 SS stud. Although the cause of the cracking was not positively identified, IGSCC was suspected.

In 1999, about 300 SCC cracks in alloy 182 weld metal were found in the shroud support welds of Tsuruga-1 (Japan) (see IAEA-TECDOC-1471 page 66). Most cracks were at the horizontal weld to the RPV and perpendicular to the welds. SCC of shroud support has been found various alloy 182 welds in several BWRs.

In 1999, IASCC in a type 316L SS control rod handle was observed in Tokai-II (Japan). Cracks were initiated at the roller pin structure. A crevice environment with radiation was thought to have accelerated cracking. IASCC of control rod handle has been found in several plants. Some cracks were not at the roller pin but initiated from the sheath weld to handle. In 2004, IASCC of type 316L sheath and tie rods was found in control rods using hafnium plate neutron absorber in Fukushima Daiichi-6. A crevice between the sheath and hafnium plate was considered to accelerate IASCC. Tensile stress generated by irradiation induced deformation of hafnium might have assisted extensive cracking. Similar cracking of control rods has been found in several Japanese BWRs.

In 1986, visual inspections revealed cracks or cracking indications in dry tubes at several BWR plants. The observed branched cracks were found in creviced areas of the upper two feet of in-core dry tube assemblies adjacent either to the weld between the tube and guide plug or the weld between the tube and the primary pressure boundary [3.20]. Cracking was attributed to IGSCC with features characteristic of IASCC [3.21].

In 1992, stainless steel cracking was found at locations different to those previously described [3.21]. This new cracking was observed either at the bottom of the collar or approximately half-inch below the base of the collar. Neutron fluence here was fairly high (5 to 7×10^{21} n/cm^2) and none of these indications were associated with welds. No branched cracking was found. Due to these observations, it is not clear whether or not this new cracking is due to IASCC.

Incidents of IASCC were in fact first reported in the early 1960s and involved intergranular cracking of type 304 stainless steel fuel cladding that was used in early BWR designs. Neutron irradiation fluence levels at the time of failure were estimated to be on the order of 10^{22} n/cm^2 (>1 MeV). Since these early incidents, several failures of highly irradiated reactor internals failures have been attributed to IASCC; e.g.:

— Neutron source holders;
— Fuel bundle cap screws;
— Instrument dry tubes;
— Control rod absorber tubes;
— Control rod handles;
— Shroud head bolts.

The core shroud and the top guide can also potentially undergo IASCC because they are locations that receive the highest neutron flux. Otherwise, from field experience it is not yet possible to conclude whether some of the observed intergranular cracking in these components were really caused by IASCC or not [3.22, 3.23].

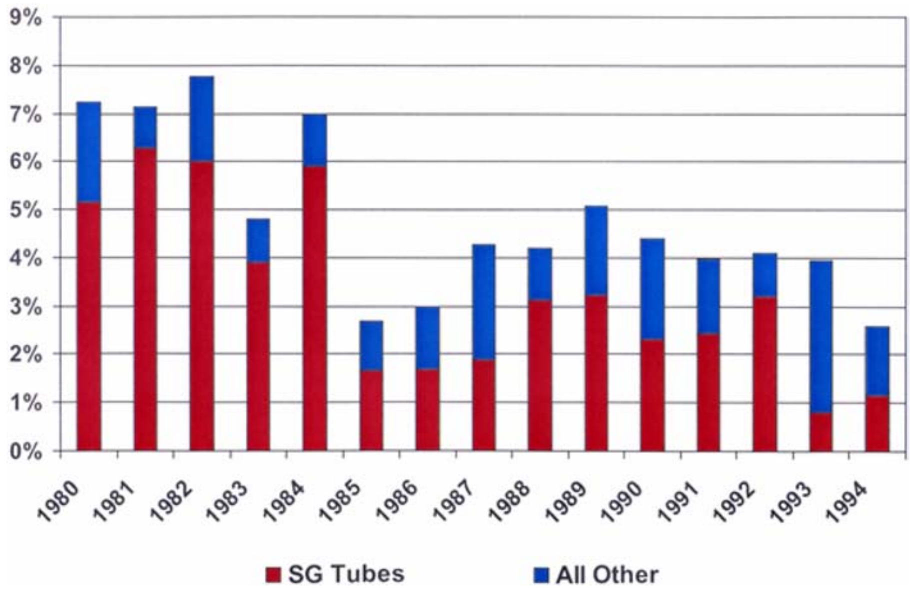

FIG. 3.5. Corrosion related capacity factor losses due to corrosion in PWRs [3.1].

3.2. PRESSURIZED WATER REACTORS

In the original design of NPPs, SCC phenomena were not explicitly considered. Beginning in the mid-1970s, the worldwide PWR fleet suffered from a series of SCC incidents that were mostly confined to alloy 600 steam generator tubing, initially from the secondary side (ODSCC) then from the primary side (PWSCC). The ensuing tube damage resulted in substantial economic loss for utilities in the 1980s and premature replacement SGs in the USA and elsewhere. In the early 80s, alloy X-750 GT support pins also began to suffer from PWSCC and many have been replaced. Subsequently, PWSCC extended in the 1990s to wrought alloy 600 components, most notably reactor vessel upper head nozzle penetrations for CRDMs. ODSCC of mill annealed alloy 600 steam generator tubing has also continued to the present day and led to many SGs being replaced. ODSCC of thermally treated alloy 600 SG tubing has also been observed.

3.2.1. Piping

As noted earlier, stainless steel piping has had a remarkably good record in PWR primary service and the relatively rare incidents of SCC have been attributed to either external chloride contamination or, for internally initiated SCC, the simultaneous presence of chloride and oxygen from air bubbles trapped in occluded crevices such as canopy seals in CRDM housings.

Some designs of PWR primary piping have used carbon and low alloy steel (C&LAS) with an internal weld overlay clad layer of stainless steel. Instrument penetrations into these pipes have often been fabricated from alloy 600 and welded in place with an internal J-groove weld of alloy 182. These alloy 600 penetrations have also often been roll expanded into the hole in the C&LAS piping. Many of these alloy 600 penetrations and their J-groove welds have had to be replaced or repaired due to PWSCC.

3.2.2. Vessel penetrations and nozzles

Reactor bessel head

Bugey 3 NPP (France)

In France, in September 1991, a leak occurred on the Bugey 3 T54 CRDM vessel head penetration. The leak was detected by acoustic emission during the first ten yearly hydrotest and was estimated to be 1 liter/h. The leaking nozzle was removed for destructive examination and a mainly longitudinal, through-wall crack that had initiated from inside the penetration was found. NDE based on the initial non- destructive examinations using dye penetrant testing, eddy currents and ultrasonic testing revealed internal longitudinal cracks at the level of the J-groove weld between the CDRDM penetration and the upper head. The metallurgical analysis concluded that cracking was due to PWSCC.

A large NDE programme was launched by EDF using eddy current, ultrasonic and visual examinations. At the end of 1992, the first NDE results showed at that five 900 MW and four 1300 MW vessel heads had cracked CRDM penetrations (after 30 000–40 000 operating hours). Due to the similarity of construction of the EDF PWR fleet of vessel heads and RPVs, EDF decided at the beginning of 1993 to replace all vessel heads that were originally equipped with alloy 600 penetrations (54 vessel heads out of 58, last four ones being equipped with alloy 690 penetrations). At the end of 2009, all of the original 50 VH had been replaced with new upper heads equipped with alloy 690 CRDM penetrations.

The importance of carbide morphology for the resistance of alloy 600 to PWSCC that had been first established for SG tubing was applied to upper head penetrations. Analysis was based on carbon content, temperature at the end of forging or rolling operations, yield strength after hot-working. Three classes of PWSCC susceptibility were determined (Class A with mainly intergranular carbide precipitates, Class B for re-crystallized material with carbides mainly on a prior grain boundary network, and Class C for re-crystallized material with randomized intergranular carbides as well as carbides on prior grain boundaries). Modelling the probability of cracking was achieved by attributing a material PWSCC susceptibility index taking into account the susceptibility of the different classes of material and determining the likely residual fabrication stress from the angle of penetration relative to the upper head and the localization of weld induced deformation (near the weld or opposite across the diameter). The influence of cold work, especially of the internal surface from machining operations was also established.

Davis–Besse NPP (USA)

Near through-wall corrosion of the RPV closure head occurred at the Davis–Besse nuclear power station in March 2002. Davis–Besse is a PWR, manufactured by Babcock and Wilcox with a licensed thermal power output of 2772 MW. The plant began commercial operation in August 1978 and is currently licensed to operate until April 2017. The RPV has an operating pressure of 2155 psig (151.50 kg/cm^2) and a design pressure of 2500 psig (175.75 kg/ cm^2). Davis–Besse had accumulated 15.8 effective full power years (EFPY) of operation when the plant shut down for its thirteenth refuelling outage on February 16, 2002. During that refuelling outage, while performing RPV vessel closure head inspections required by the US NRC, workers discovered a large cavity in the 6 inch (15.24 cm) thick low alloy carbon steel RPV head material. The cavity was about 6.6 inches (16.76 cm) long and 4–5 inches (10.16–12.70 cm) at the widest point extending down to the 0.25 inch (0.635 cm) thick internal surface type 308 stainless steel cladding.

The technical root cause analysis determined that the corrosion was the result of boric acid interaction with the carbon steel on the RPV head. The source of the boric acid was a primary water leak via a through-wall crack in a CRDM nozzle (see FIG. 3.7 (a)). This crack was initiated as a result of PWSCC.

The Davis–Besse utility had believed that the boron accumulated on the RPV head was due to leaking CRDM flanges above the RPV head (that are specific to the B and W design) and that such accumulation would not cause extensive corrosion due to the elevated temperatures at that location. This boron accumulation on the top of the RPV head was not fully removed during refuelling outages and it masked the typical ‘popcorn’ boron indications that are observed from a CRDM nozzle containing a crack (see FIG. 3.7 (b)). Accordingly, the boric acid leaking through the nozzle crack was allowed to corrode the carbon steel head creating a cavity.

Ohi 3 NPP (Japan)

During a periodic inspection in 2004 at Ohi Unit 3 after a plant operating time of about 100 000 hours, leakage from #47 CRDM head penetration was found during a bare metal visual inspection, as shown in FIG. 3.8. It should be noted that the Ohi Unit 3 RV head reactor coolant temperature was modified to 289°C from 310°C in 1997.

In order to identify the leaking portion of the CRDM penetration, the thermal sleeve below the vessel head was cut off and helium leak testing (HLT), eddy current testing (ECT), dye penetrant testing (PT), and ultrasonic

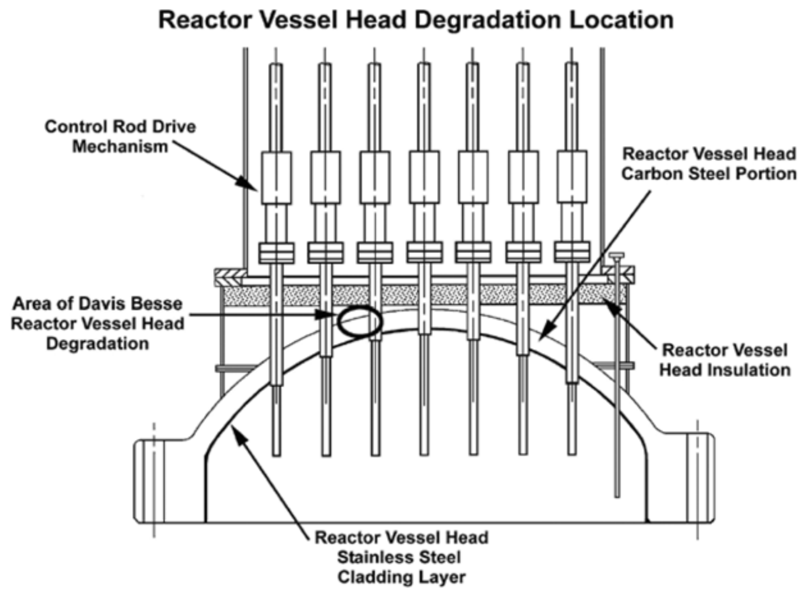

FIG. 3.6. PWR vessel head penetration cracking of alloy 600 allowed leakage of borated coolant to occur. This corroded the C&LAS RPV head externally down to the internal surface stainless steel cladding.

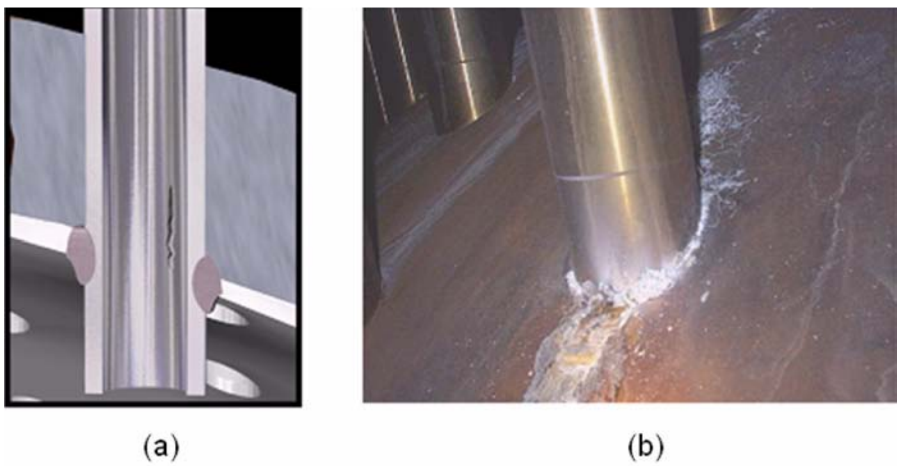

FIG. 3.7. (a) Depiction of a through-wall nozzle crack, (b) Typical appearance of 'popcorn' boron indications from leakage due to through-wall cracking.

testing (UT), etc. were performed both on the nozzle and the J-groove weld of #47 CRDM head penetration. HLT found a leak on the J-weld and ECT found indications on the J-weld. After surface grinding, PT and replica printing were performed and radial, linear-like, cracks were observed on a portion of the PT indication located along grain boundaries. Additional grinding found longer cracks and that some cracks had coalesced. Those cracks had branches along the weld dendrites. Based on the above observation, it was deduced that the radial crack had propagated through the J-weld and caused RCS water leakage. The RV head was replaced in 2007 and has alloy 690 head penetration nozzles and compatible J-welds.

There are currently 23 operating PWR plants in Japan. At present, 14 RVHs have been replaced and 7 additional RVHs are planned to be replaced in the near future with alloy 690 TT CRDM penetrations. One plant already had CRDM head penetration made of alloy 690 thermal treated since plant construction. Other Japanese PWR utilities have addressed the issue of CRDM head penetration cracking by modifying the component temperature to that of the cold leg. Upper heads that operate at the cold leg temperature are considered to have lower susceptibility to PWSCC but, nevertheless, some of these Japanese plants have decided to replace the RVHs equipped with alloy 690 CRDM penetrations.

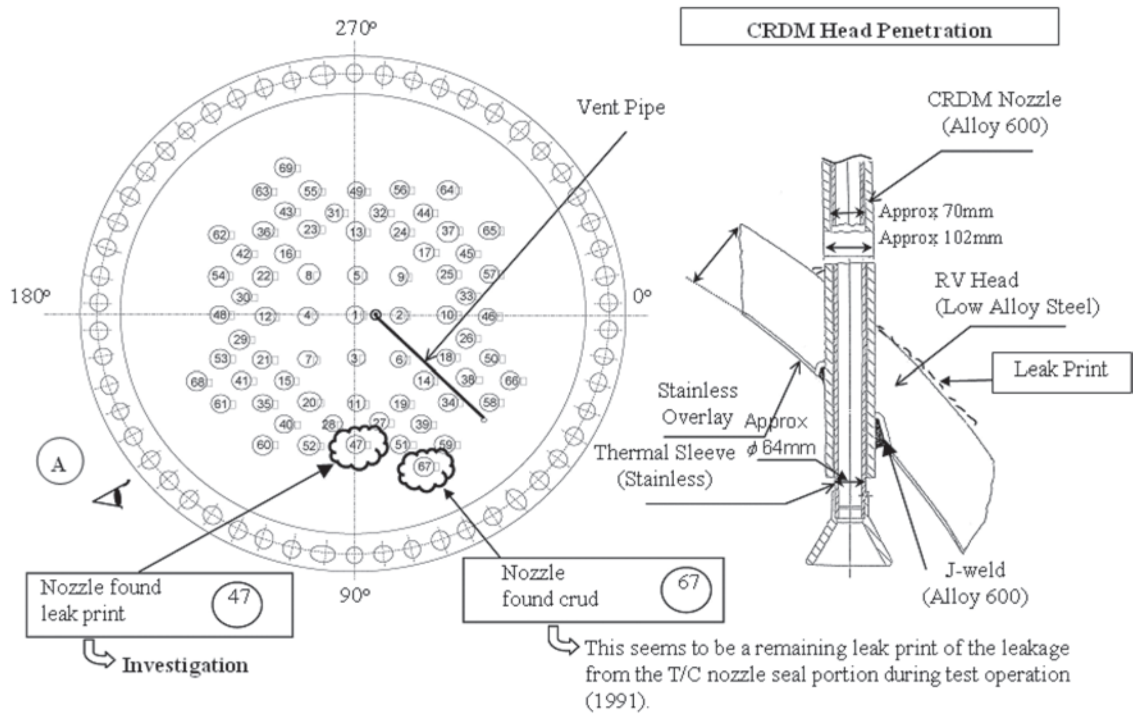

FIG. 3.8. Location of CRDM head penetrations with leakage [3.24].

BMI

S. Texas 1 NPP (USA)

In April 2003, small boron deposits around two of the 58 BMI penetrations (penetrations 1 and 46) were identified in South Texas Project Unit 1 (STP Unit 1). This is the only evidence of BMI nozzle penetration leakage reported by a US facility to date. The STP Unit 1 BMI penetrations were constructed from drilled alloy 600 bar stock and connected to the reactor vessel lower head by an alloy 82/182 J-groove weld.

The South Texas event prompted a coordinated US industry response. As of August 2007, >800 penetrations had been inspected. No additional indications have been found. Inspections are scheduled to continue.

Takahama 1 NPP (Japan)

In January 2003, one small indication was detected at inner surface of one BMI penetration nozzle shown in FIG. 3.9 following eddy current inspections of 50 BMI penetrations at Takahama Unit 1. The indication was within the acceptance criteria (≤3 mm depth). Nevertheless, the utility concluded that there is some possibility that it was an indication of the initiation stage of PWSCC. The utility removed the indication by drilling and then performed water jet peening on the inner surfaces of the BMI penetration nozzles as a preventive measure. Laser and water jet peening have also been applied as mitigation measures at other Japanese PWRs. Additionally, these peening treatments have been performed on the J-groove welds at these locations.

PZR nozzles

Tsuruga 2 NPP (Japan)

During the 13th periodic inspection of Tsuruga Unit 2 in September 2003, a crack was found in the pressurizer relief line nozzle stub weld, as shown in FIG. 3.10. This was first noticed when boric acid precipitation was observed at that location. Ultrasonic tests on the relief line stub showed two indications that were located in repair welds. Ultrasonic testing of other nozzle stubs revealed an indication on the safety valve (A) while others showed

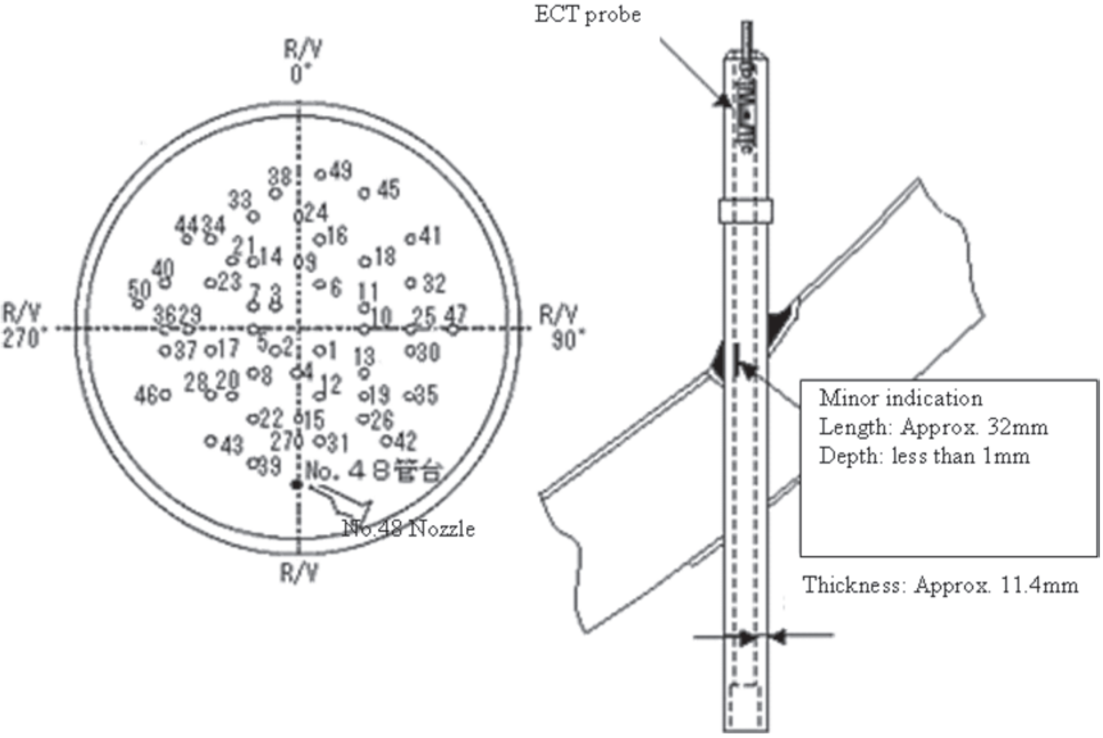

FIG. 3.9. Location of the BMI penetration nozzle Takahama Unit 1 with a small indication [3.24].

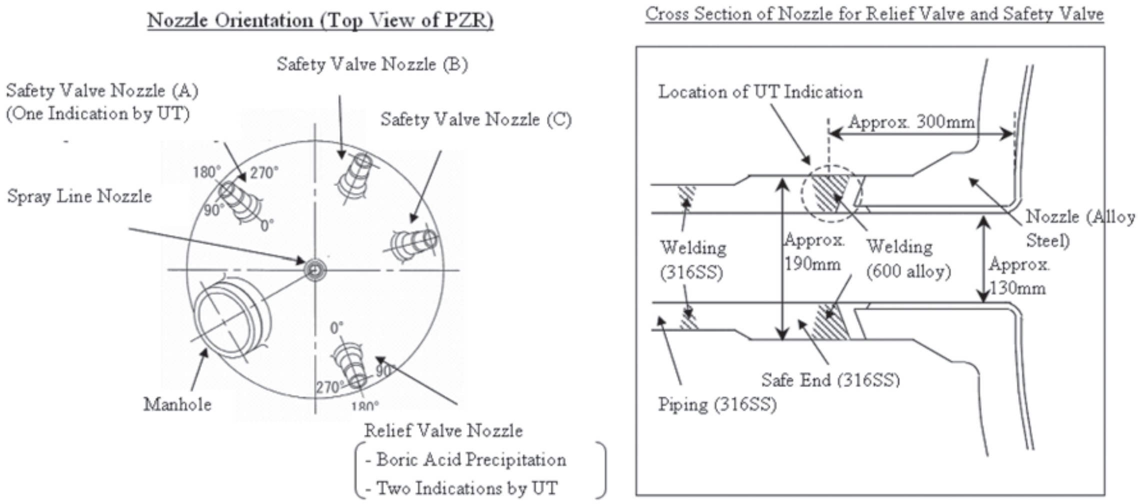

FIG. 3.10. Location of the nozzle with indication at Tsuruga Unit 2 [3.25].

no indications. It was deduced from these observations that the cracks remained in only the weld metal and that the fracture surfaces were along the columnar grains. The cause was recognized to be stress corrosion cracking (PWSCC) in the nickel based weld metal (alloy 600 type). Regarding weld portions of the piping nozzle stub of the pressurizer relief valve, the piping nozzle stub for pressurizer safety valve (A) and the safe end, the welding metal materials were changed to nickel based alloy 690 type which has good resistance to this kind of stress corrosion cracking.

RPV nozzles

Cracking in alloys 182, 132 and 82 was not observed in operating PWR plants until the year 2000, when several incidents occurred. The first was in the outlet nozzle to pipe safe end weld of Ringhals Unit 4, in July of

2000. Several small axial cracks were found, and removed in a boat sample by EDM. The first cracking event was actually found in June of that year, in Ringhals Unit 3, but the indications were thought to be shallow artefacts and the plant was allowed to remain in service without repair. In both cases, the welds were alloy 182, and the cracks were axial. The Ringhals 3 and 4 cracks were machined out and repaired using a weld inlay process with alloy 52M material in 2003/2004.

The next major incident occurred in October 2000, when the VC Summer Plant was found to have a through-wall flaw in the same region as the Ringhals plants, the reactor vessel outlet nozzle to pipe safe end weld [3.26]. In this case, the weld was a field weld, which had experienced multiple repairs during the construction process. .

Ultrasonic tests performed on the pipe from the inside surface initially revealed a single axial flaw near the top of the pipe. Follow-up exams conducted in the Spring of 2002 revealed that there were several flaws, of which all but one were axial, and that the largest axial flaw was through-wall. The flawed region was removed, and a new spool piece welded in place, so restoring this region to its original condition. The VC Summer outlet nozzle to pipe weld was repaired with alloy 52, for a portion of the thickness, and the remainder of the weld was filled with alloy 82. The other VC Summer outlet nozzles were later mitigated using MSIP.

3.2.3. Steam generators (primary water stress corrosion cracking)

PWSCC experience in SG tubes worldwide

PWSCC of alloy 600 SG tubes at Obrigheim was reported for the first time in 1971 [3.33]. Similar degradation was observed worldwide until such time as the tubes were replaced with thermally treated alloy 690. Significant damaged areas were the tight row 1, 2 and 3 U-bends, roll transitions, and some tube support regions that had a high tensile stress due to secondary side induced tube denting.

Most of the cracks were in the hot leg side, but there were also some cases of cold leg cracking. The cracks were mostly axial but some plants showed circumferential cracks.

PWSCC has never been observed in alloy 800 SG tubing or in replaced steam generators with alloy 690 tubing [3.34].

PWSCC in Korean NPPs

PWSCC has been reported in steam generator tubing in Korean NPPs. As shown above, mill annealed alloy 600 tubes of plant A have suffered from secondary side pitting and ODSCC. When PWSCC was detected for the first time during ISI after the chemical cleaning in 1990, 22 tubes in SG A, and 26 tubes in SG B had to be sleeved. In order to characterize the behaviour of the PWSCC defects and verify the performance of the new motorized rotating pancake coil (MRPC) inspection method, which has been applied since 1992, three tubes were pulled in 1992.

Multiple circumferential cracks, which had penetrated up to 56% of the wall thickness, were found inside the tubes. Also, multiple axial cracks up to 6.8 mm long, and 100% through-wall were observed that had initiated from the inner wall of the tube.

In alloy 600, as noted earlier, the best microstructure that is resistant to most forms of IGSCC is reported to have a semi-continuous carbide decoration of the grain boundaries [3.27, 3.28]. In this case, many carbides in the microstructure of the pulled tubes were within the grains (i.e. intergranular) and this microstructure seemed to render these tubes susceptible to PWSCC. After this failure analysis, a shot peening and a reduction in primary water operating temperature were recommended as countermeasures in order to reduce the risk of a PWSCC.

The steam generators of plant C have been operating since 1988. The tubing material is thermally treated alloy 600, and the tubes were mechanically full depth rolled to the top of the tube sheet and kiss rolled a short distance above it. The kiss roll length of the tubes was shot peened after the 5th cycle of operation in 1994. 793 tubes from the higher temperature region of the hot leg side were examined by MRPC in 1993 and many axial cracks inside the tubes were detected at the roll transitions. Two tubes were extracted from SG C in 1994 and examined by EDF [3.29]. The objectives of the examinations were to determine the failure mode and to predict the crack propagation rate. To achieve these goals, the failure behaviour was compared with French experience using similar material. In addition, the chemical composition, grain size and mechanical properties were determined based on the heat number of each tube. Finally, the apparent growth of the longest cracks in each tube inspected

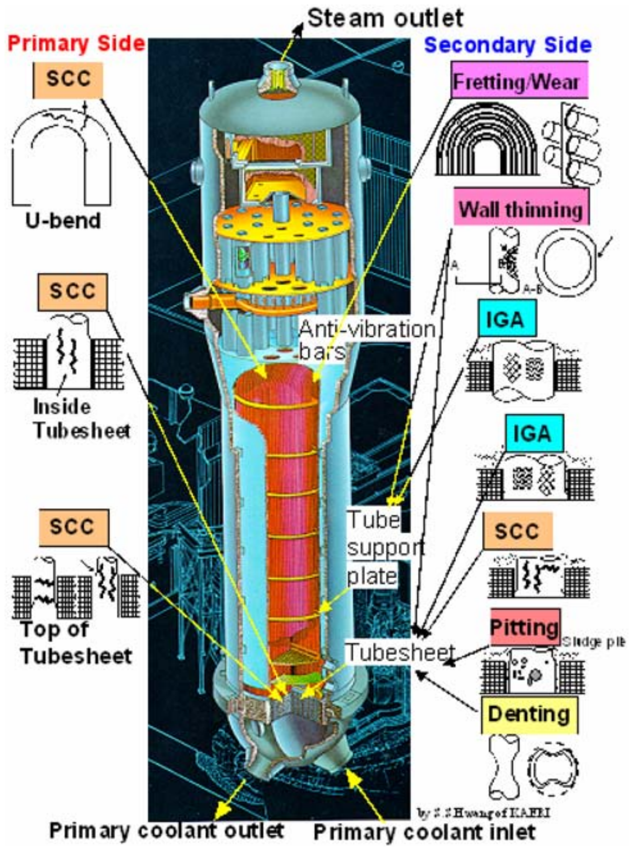

FIG. 3.11. Degradation mechanisms of various steam generator components.

between the 1993 and 1994 was calculated. The crack length and depth determined from the MRPC data and destructive analysis data obtained after a burst test were also compared.

The average grain size was consistent with ASTM number 9 for both tubes indicating that the grain size is rather small. Others have reported that tubes with a small grain size (>ASTM 8) were susceptible to a PWSCC [3.30, 3.31]. High tensile strength (>717 MPa), high carbon (>0.018%), high silicon and low chromium contents were other indicators of potentially high cracking susceptibility [3.31]. Average crack growth rates in plant C were estimated to be 0.5 mm/a to 1.3 mm/a depending on the tube heat identity, which were on the lower boundary of those observed previously for mill annealed tubes [3.29].

There were no ODSCC indications detected by ECT on the outer surface of the two tubes examined. Nevertheless, one tube had 2 longitudinal cracks of which the maximum length and depth were 6.86 mm and 99% through-wall respectively. This tube had a leak pressure of 42.2 MPa and a burst pressure of 65.4 MPa. The other tube had 3 longitudinal cracks of which the maximum length and depth were 5.88 mm and 93% through-wall respectively, and it showed a leak pressure of 31.5 MPa and a burst pressure of 59.2 MPa.

Remedial measures were suggested such as defining a plugging criterion based on crack length, nickel plating, preventive sleeving, and a primary water temperature reduction.

Despite the shot peening carried out in plant C in 1994, the ECT voltage from indications in the tubes continued to increase and primary to secondary coolant leakage was reported from SG B and SG C in 1997. A total of 2067 defect signals were observed from the three steam generators of this plant in 1998. Of these, 986 tubes were sleeved, and two defective tubes plus a ‘sound’ tube (i.e. one with no reportable ECT indication) were extracted and examined in 1999 [3.32]. In this analysis, the crack lengths and depths were compared with the ECT data. The ECT history of the two defective tubes was studied and the effect of shot peening was evaluated based on the ECT results at each ISI. The microstructure in terms of the carbide morphology was also studied using TEM.

The crack depths in the defective tubes were 80–100% through-wall regardless of the defect length. This means that the shot peening had prevented any further crack length propagation, but did not suppress the through wall propagation.

The alloy 600 tubes of unit C had been mill annealed for 2 minutes at 960°C–1000°C, and thermally treated for 12 hours at 700°C–730°C to develop carbides at the grain boundaries. However, the microstructural analysis showed that many carbides were within the grains rather than at the grain boundaries. This microstructure was classified as type II or type III, as suggested by EDF [3.23], and is considered to be rather susceptible to a PWSCC. The carbide microstructure seemed to be related to excessively high carbon content (0.035%) and did not show any beneficial effect of the 700°C thermal treatment. That is to say, the earlier mill annealing temperature had been too low to dissolve the total carbon into solid solution prior to the thermal treatment.

Inside diameter cracks 2.5 mm–6 mm long were located at the top of the tube sheet and penetrated by 72% to 100% of through-wall. Another tube, which was considered as a ‘sound’ tube from the ISI, had a crack 2.1 mm long and 84% through-wall penetration. This means that there were likely to be many undetectable cracks on the tubes of the SG at the Kiss roll transition.

The main cause of PWSCC in plant C was a susceptible microstructure. Consequently, the recommendations suggested after this analysis were as follows. Tubes having large increases in their EC voltage should be repaired. Detailed inspection was required for the PWSCC sensitive regions depending on the particular steam generator. An allowable leak rate limit of 10 l/min for plant C was also recommended.

PWSCC experience of channel head drain line attachment welds

In Ringhals, cracks in channel head alloy 82 welds were discovered during a refuelling outage in 2004. Boron deposits were found in two positions during visual inspection of drainage pipes from the channel head manhole covers. The drainage pipes were made from stabilized stainless steels and connected to the low alloy steel lower dome by a nickel based dissimilar J-groove weld of alloy 82 equivalent weld metal. Metallographic examination of boat samples showed that the cracks were most likely service induced (PWSCC) degradation of the alloy 82, but manufacturing defects were observed [3.35].

PWSCC of SG divider plates

In France, the area of concern is the weld between the stub and the divider plate in the channel head. This weld (made with alloy 182) is the last weld to be made during SG fabrication — it is made manually and is not followed by any thermal treatment. The stubs and divider plates for most of steam generators are made from forged plates of alloy 600 whose final thermal treatment occurred at the end of their forging sequence. Alloy 690 has only been used for the most recent SGs. An in-service inspection programme started in France in 1999.

At the end of 2007, 72 SGs had been inspected including eleven 1300 MW/4 loop SGs. Ten SGs from 900 MW(e)/3 loop units were affected by indications on hot leg side that were located in the base metal quite close to the welds in the stub The alloy 182 welds were not affected. Cracking was superficial except for two SGs. No SGs have been affected from 1450 MW(e)/4 loop units and from 900 MW(e)/3 loop units that are equipped with replacement SGs. The various inspections have highlighted the fact that these defects are located in the stub of the hot leg, with no signs of significant evolution either by fatigue or corrosion.

A superficial layer, which is cold worked during the fabrication process, is suspected to be responsible for the initiation of these defects. The most influential parameters identified for PWSCC initiation are the material PWSCC susceptibility index (only specific heats of material have been affected), stubs with low yield stress associated with a high difference in yield stress between the stub and attached partition plate, hammering by loose parts (PWSCC in the cold worked area), manufacturing parameters such as the stub/partition plate alignment during final assembly, and partition plate thickness (34 mm for 3 loop units and 60 mm for 4 loop units).

PWSCC of alloy 600 MA tubing of Kansai NPP

In 1976, a leak occurred at a row 1 U-bend of a SG in Takahama Unit 1. It was thought that the crack was due to PWSCC influenced by localized plastic deformations of the tube. The deformed area was located between the U-bend and straight tube and was generated by passing a ball mandrel during bending operations in the fabrication process of U-bends in the tubes. This area was believed to have high residual stresses. PWSCC at U-bends also occurred at OHI Unit 1 and Mihama Unit 2. Another leak occurred in small U-bends by a similar mode of PWSCC at OHI Unit 2 in 1994 where the area of varying ovality was relatively large.

Since 1982, PWSCC in the tubesheet region has been detected in many units. PWSCC in both the hard rolled area and expansion transitions with full depth expansion in tubesheet were detected at Mihama Unit 3, OHI Unit 1 and 2. PWSCC at hard rolled area with full depth expansion in the tubesheet was detected at Genkai Unit 2, IKATA Unit 1 and 2. PWSCC at the expansion transitions with partial depth rolling in tubesheet was detected at Takahama Unit 1 and Mihama Unit 2. PWSCC at tube expansion transitions occurred due to high residual stress generated in the transition area by mechanical rolling. PWSCC in the hard rolled area occurred due to high residual stress caused by insufficient expansion during mechanical rolling in irregular shaped drilled holes. PWSCC of 600 MA tubing was eliminated by SG replacement with 690 TT tubing.

PWSCC of 600 TT tubing of Kansai NPP

Since 1999, some indications have been detected by ECT at the expanded area of alloy 600 TT tubing in the tubesheets in three plants. Investigations were performed on pulled tubes with indications and the results showed the cracking was due to PWSCC. During manufacturing of the SGs for these three plants (Sendai Unit 1, Takahama Unit 3 and 4), full depth mechanical roll expansions were made after full depth hydraulic expansions. Cracking was found at the end of the top mechanical roll (at transitions with the hydraulic expansion area) or in overlapped areas of adjacent mechanical rolls; i.e. cracking was located in the mechanical roll expansion area. Cracking did not occur in the hydraulic expansion transition region. From the investigation at Takahama Unit 4, it was also found that the tube hole had a locally slightly oversized diameter. It was thought that such an oval shape was made by eccentric polishing of tube hole during manufacturing. The cracking had occurred in the area with an oversized diameter. From mock-up tests, high residual stress was observed at the end of mechanical roll or overlapped area of adjacent mechanical rolls in tube holes with such an irregular shape. The cause of PWSCC was deduced to be high residual stress associated with mechanical rolls on such irregular shaped tube holes.

PWSCC of primary inlet nozzles of Kansai NPP

In 2007, it was planned to apply shot peening to the primary inlet nozzles of the steam generators in Mihama Unit 2 in order to mitigate the possibility of PWSCC of alloy 600 type welds. Equivalent weld metals are applied to the SG nozzles in that plant so that it is inappropriate to mention registered trademark materials such as 132/182/82. In the planned programme, ECT was performed before applying shot peening in order to inspect for any existing cracks. Small cracks were found by ECT on the inner surface of alloy 600 type weld in a primary inlet nozzle, as shown in FIG. 3.12. Similar cracks were also found in Tsuruga Unit 2, Takahama Unit 2, Genkai Unit 1, and Takahama Unit 3 by ECT. No cracks have been found in the primary outlet nozzles of any plants.

During microscopic observations in most of the above mentioned plants, it was confirmed that the cracks propagated along the dendrite crystals of the alloy 600 type weld metal. These are characteristics of PWSCC of alloy 600. As an example, the machined surface was observed around the cracks at Mihama Unit 2. It was confirmed that such machined surfaces can have high residual tensile stresses larger than the threshold for PWSCC of alloy 600, as found by mock-up testing. In the investigations for some other plants, traces of grinding were found as possible causes of high residual stress.

The investigation at Mihama Unit 2 revealed some additional information concerning very shallow minor cracks that were found using a special replica technique on the inner surface of a type 316 stainless steel safe end in a primary inlet (higher temperature) nozzle. Crack propagation was observed by microscopic observation to occur along the grain boundaries. Further root cause investigation is ongoing.

3.2.4. Reactor pressure vessel internals

Guide tube support pins

A failure of a guide tube support pin made of alloy 750 occurred at Mihama Unit 3 in 1978, during the second cycle of operations; the first case observed in the world. A part of the broken support pin migrated as a loose part and was found in the bottom channel head area of a SG. Typical locations where cracks occurred on the support pins are shown in FIG. 3.13. After this incident, all such pins used in other plants in Japan were inspected using an UT technique and significant indications were confirmed. The support pins are located at the bottom of guide tubes and

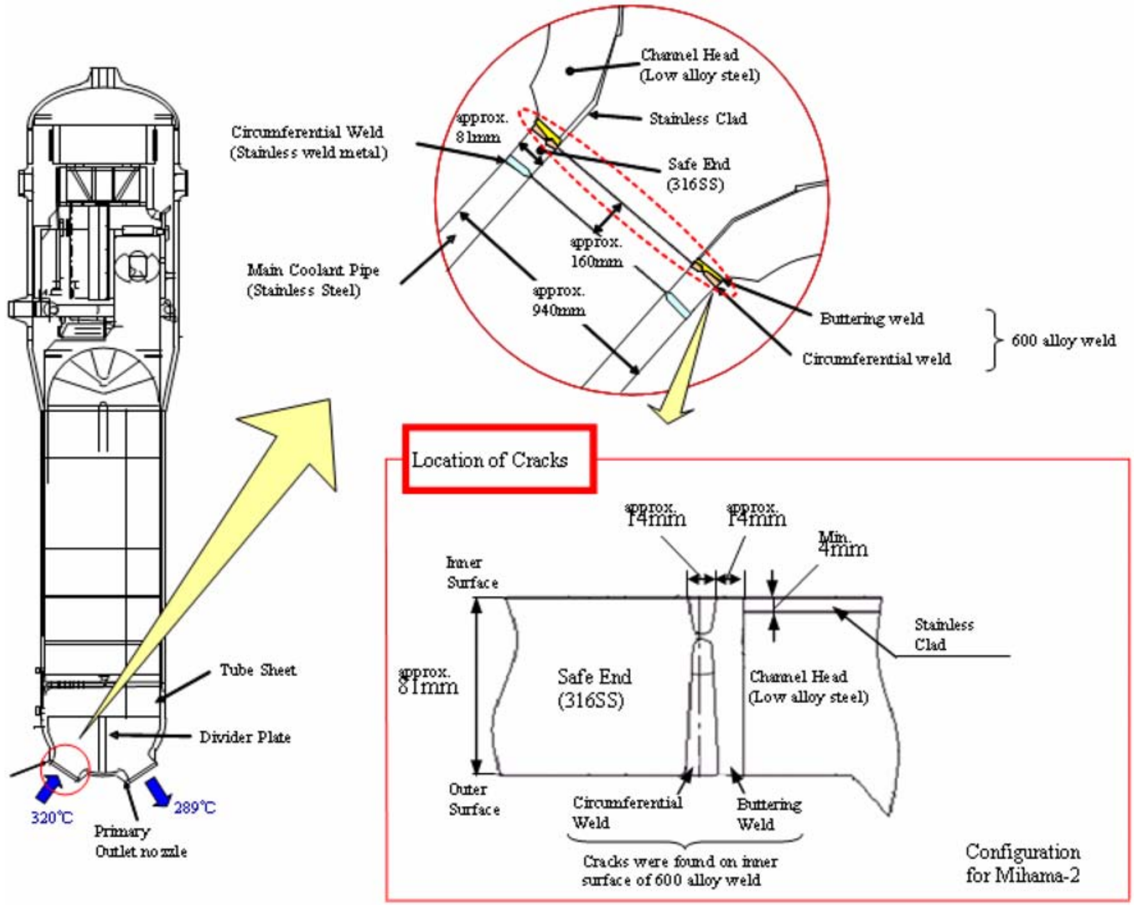

FIG. 3.12. Location of the primary inlet nozzles with indication.

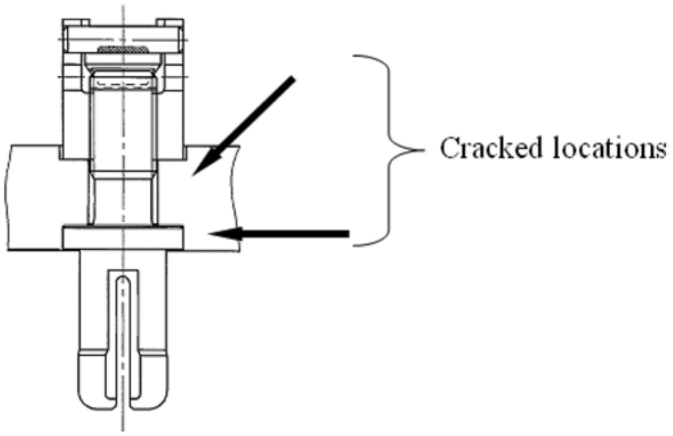

FIG. 3.13. Typical cracked location.

serve to position and attach the lower part of the guide tubes to the upper core plate. Thus, the pin was used under conditions of high stress and high temperature in the PWR primary water.

Many research studies including SCC experiments were carried out for more than five years to understand this phenomenon and to develop countermeasures. Throughout these studies, the results showed that the alloy X-750 used for the pins was susceptible to PWSCC and that it was strongly influenced by the heat treatment condition. Based on the experimental results, the heat treatment condition for alloy X-750 was optimized and verified as an alternative material with better PWSCC resistance.

Between 1982 and 1984, all pins made of alloy X-750 with uncertain heat treatments and susceptibility to PWSCC were replaced in Japanese PWRs. The new pins for replacements were designed with the following two modifications: applying the improved material heat treatment to reduce PWSCC susceptibility and modifying the mechanical design to reduce applied stresses during plant operation.

So far, there are no reports of failures of the new replacement pins in Japan.

Similar experiences were observed in other countries from the early 1980s. Pins have been replaced with lower stress/better microstructure materials. The SCC experience of this second generation of pins has generally been good. In the 1990s, guide tube support pins made from CW316 SS were also offered to PWR utilities that preferred not to use nickel based alloys in this application. Currently both materials are offered.

Baffle former bolts

In the 1980s, following observations of flow induced vibration of fuel rods in fuel elements on the periphery of the core caused by water jetting through gaps between baffle plates, inspections of some first generation French 900 MW(e) plants indicated that baffle former bolts were cracking in PWR core support structures. This cracking is a concern and made necessary the development of ultrasonic methods for the non-destructive examination of the bolts. The bolts are made of cold worked (typically 10–30%) type 316 stainless steel. Destructive examinations of removed bolts with indications showed that they failed by intergranular cracking. Normally, type 316 steel is not prone to IGSCC in the hydrogenated PWR primary water environment. However, all the bolts that cracked were located in the second and third rows from the bottom of the active core, which correspond to the highest neutron irradiation flux. This indicates that neutron irradiation is a significant feature for this cracking and further work has concluded that IASCC is the likely cause.

To date, baffle bolt cracking has been observed mainly in those plants with the so-called ‘down flow’ design (indicating the direction of water flow between the core baffle plates and the core barrel). Most of such plants have been modified to be ‘up flow’, which reduces the risk of water jetting to the peripheral fuel rods and results in lower temperatures in the baffle former bolts even after taking gamma heating into account. Up flow plants have so far been little affected by baffle former bolt cracking but other factors such as improved bolts with reduced stress concentration factors between the bolt heads and shanks have also probably played a part.

French experience

In the 1980s, baffle jet failures of peripheral fuel rods occurred in the older, first generation CP0 French reactors (six 3-loop plants at Fessenheim and Bugey). In 1988, the first UT inspection of baffle former bolts was carried out and cracks were detected. Between 1989 and 1993, all the CP0 plants were converted to up flow instead of down flow, and an inspection plan was defined. Some baffle bolts were extracted for metallurgical investigation. IASCC was confirmed as the degradation mechanism. Subsequently, three units with higher fractions of baffle former bolts with indications changed about 1/3 of their baffle bolts between 2000 and 2003.

After baffle bolt cracking was first detected by non-destructive examination, some CP0 baffle bolts in cold worked type 316 stainless steel with indications were extracted for metallurgical examinations. Some metallurgical examinations were performed on irradiated baffle bolts with neutron doses 10–25 dpa that had been removed after 10 and 20 years of operation. Other examinations were performed on type 304 austenitic stainless steel from the decommissioned Chooz A unit. A baffle corner from this unit was removed for metallurgical examination and mechanical properties and IASCC testing with neutron doses up to 36 dpa (i.e. corresponding to the maximum value at approximately mid-life of currently operating reactors).

Fracture surface examination of removed baffle former bolts revealed very similar, primarily intergranular cracking (with very small transgranular and ductile areas) which was characterized as IASCC. The cracks initiated in the head to shank area and extended across the baffle bolt below the head. Hardening of the material and radiation induced segregation at grain boundaries were observed. From hardness profile measurements, hardening saturation occurred between approximately 5–10 dpa. No evidence of void swelling was found.

The removal torque values of sound baffle bolts were recorded in those plants where bolts were replaced. It was established from irradiation induced creep laws obtained in the R&D programme that the torque values observed were consistent with a consequence of irradiation induced creep. Moreover, it was also established from plant experience that the evolution of baffle bolt preload due to irradiation induced creep is a beneficial feature for

IASCC but not detrimental in terms of fatigue. Studies have shown the equalizing effect of the redistribution of baffle bolt loads from high flux areas to lower flux exposure locations, thus maintaining the mechanical and geometrical stability of the core baffle structure.

The rate of cracked baffle bolts has increased slowly with the increasing neutron dose. By analyzing the results from every inspected baffle bolt from every ISI campaign on every CP0 unit, a dose threshold of about 3–4 dpa for baffle former bolt cracking is clearly observed in service. Higher cracking rates are observed following the threshold and there is a slow evolution of the number of cracked bolts with increasing neutron dose.

Japanese experience

As proactive countermeasure against baffle former bolts issues experienced in France, USA, etc., two types of approach were applied since 1988 to Japanese PWR plants. The first approach was to replace the baffle former bolts themselves. All the original bolts made of type 347 stainless steel at Mihama Units 1 and 2 were replaced between 2001–2002 with new ones made of type 316CW stainless steel.

The second approach was to replace complete reactor vessel internals. In this case, both the lower internals including baffle former bolts and upper internals were replaced at the same time. Complete replacements of reactor vessel internals have already been applied to three PWR plants in Japan since 2004. Consequently, the above mentioned proactive countermeasures have been applied at some 2-LOOP PWR plants constructed in the early stages of nuclear power development in Japan in the 1970s.

Current research on baffle former bolts in Japan

Regarding research studies of IASCC in irradiated austenitic stainless steels like those used for baffle former bolts in PWRs, a Japanese national project has been conducted since 2000 in order to get better understanding of the IASCC phenomenon. Throughout the project, more data have been collected and have shown that IASCC initiation depends on a combination of applied stress and neutron fluence. In other words, the threshold stress for initiating IASCC depends on the neutron fluence; the threshold stress tends to decrease with increasing neutron fluence. The data collection project was scheduled to continue through 2008. At the completion of these tests, a detailed analysis will be performed and recommendations made. Japanese utilities and manufacturers have also established a guideline in 2002 to deal with maintenance actions such as inspection of baffle former bolts in operating plants. The guideline will be modified to reflect the latest knowledge obtained through the Japanese national project in the coming years.

Korean experience

Concerns with potential IASCC in Korean power plants started in 1999 with the inspection of the Kori-1 baffle former bolts. The baffle former bolts were evaluated by the calculations on the basis of US research results in order to identify those that exceeded an IASCC threshold (5×10^{21} n/cm^2, E>1.0 MeV). The UT inspection was performed in accordance with ASME Code Section XI. The inspected parts were 728 baffle former bolts and 176 baffle edge bolts. All bolts were made of CW 316 stainless steels. The UT inspections in 1999 indicated 2 defective baffle former bolts and 6 baffle former bolts with uninterpretable UT signals.

Further inspections were carried out by Korea Plant Service & Engineering (KPS) in 2006 [3.36]. One bolt was had an uninterpretable UT signal, probably because this bolt was inaccessible for acquiring a normal signal. All bolts with indications or uninterpretable signals were identified as safe in 2006.

In addition, the control rod guide tube pins were identified as potentially affected by a neutron exposure over the IASCC threshold. The original control rod guide tube pins made from alloy X-750 Rev. B were replaced in 2007 with CW type 316 stainless steel pins. IAEA-TECDOC–1557 (Table 13) shows similar figures (from France, Japan, USA etc.).

TABLE 3.1. KORI 1 NPP INSPECTION RESULTS OF BAFFLE FORMER BOLTS

Inspection Section	Kori 1 NPP inspected by Korean Power Service (2006)	
	Baffle former bolts	Baffle edge bolts
Defective bolts	0/728	0/176
Un-interpretable bolts	1/728	0/176
Intact bolts	727/728	176/176

TABLE 3.2. EXPERIENCES OF INSPECTION OF RPV WELD INTERNALS

Country	Long t Unit	EFPY	Last inspection	Result
Germany	KKE	A	2005.06(?)	OK
Republic of Korea	Yonggwang-3	11.09	2006.04	OK
USA	Z	C		OK

US experience

The Westinghouse Owners Group (WOG) provided an assessment of the cracking of the baffle former bolts identified in foreign PWRs, including the potential impact of cracking on US domestic Westinghouse plants, and provided information on its current and planned activities. The WOG stated that because of the large number of baffle former bolts in the baffle assembly, the failure of a few bolts should not have a significant safety impact. The WOG activities include:

— Development of analytical methods and acceptance criteria for bolt analysis;
— Performance of risk-informed evaluations.

In the USA, the Nuclear Energy Institute (NEI) formed a Materials Technical Advisory Group (MTAG) made up of representatives from the utilities. The MTAG requested that EPRI support efforts to prepare guidelines/recommendations for the ISI of RVI components that are critical to the continued and safe operation of nuclear power plants. In preparing the inspection guidelines, degradation of reactor internal components due to irradiation, fatigue, wear, and corrosion were taken into consideration. Westinghouse and AREVA prepared a report as part of the EPRI MRP to identify critical reactor internal components using screening based on the various degradation mechanisms. The guidelines were discussed with the USNRC prior to the end of 2006 or 2007 because under the USA License Renewal schedule, two plants were scheduled for ISI before the end of 2009.

3.2.5. Reactor pressure vessel weld internals

Welded internals are used in the Siemens Konvoi (sp) plants, some CE reactor vessel internals designs and in the later Korean standard PWRs. These reactor internals have undergone inspections, as shown in Table 3.2. No cracking in RPV welded internals has been observed to date.

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4. AGEING MANAGEMENT APPLICATION ON STRESS CORROSION CRACKING

Since SCC is a complex phenomenon involving synergistic interactions between metallurgy, chemistry and mechanics, it is necessary to expand knowledge in each technical field and then take action to reflect the enhanced knowledge in the other fields. For example, topics such as SCC mechanisms, inspection, mitigation and repair need to take into account the expanded knowledge coming from the latest field experiences. The most important thing to be considered is how to apply an ageing management programme to reduce the risk of damage due to SCC in nuclear power plants.

4.1. BOILING WATER REACTORS

4.1.1. Scope of the ageing management programme based on understanding ageing

BWR components are potentially susceptible to two predominant forms of SCC. They are: (a) IGSCC and (b) IASCC. IGSCC, especially, has been significant for some components in BWRs made of austenitic stainless steel or nickel based alloys. Examples of such components are the recirculation piping, core internals and some parts of the RPV such as the in-core monitor (ICM) housing and the control rod drive (CRD) stub tubes.

4.1.2. Preventative actions to minimize and control ageing degradation

Material aspects

Preventive actions in this context consist of the selection of SCC resistant materials, which include low carbon grades of austenitic SS and weld metals with a maximum carbon content of 0.035 wt.%, and a minimum ferrite content of 7.5% in weld metals and cast austenitic stainless steels (CASS). For nickel alloys, alloy 82 is the only commonly used nickel base weld metal considered to be practically resistant to SCC since no failures have occurred in service to date. Other nickel alloys, such as alloy 600, need to be evaluated for SCC resistance on an individual basis.

Stress aspects

Preventive actions include specially developed processes to relax residual tensile stresses, such as solution heat treatment (SHT), heat sink welding (HSW), induction heating stress improvement (IHSI), and mechanical stress improvement (MSIP). These processes are also designed to leave a compressive residual stress on the surface in contact with the reactor coolant.

Environmental aspects

Mitigation by water chemistry includes measures such as hydrogen injection (i.e. hydrogen water chemistry or HWC), noble metal technologies like NMCA, and TiO_2 injections. These are effective preventive measures for a BWR service environment. However, in order to detect any adverse effects on fuel performance and integrity or on radiation exposure resulting from changes to water chemistry, it is necessary to evaluate the latest observations and operating experience before implementation. Also, methods that can be used to control and monitor any adverse effects must be established.

4.1.3. Monitoring and trending of ageing effects

Monitoring and trending of SCC induced ageing includes detecting and sizing cracks, detecting reactor coolant leakage, and analyzing plant data as it accumulates for any trends with the objective of validating component integrity. When one or more cracks are found, it is important to inspect additional similar components to the same extent as the original inspection. In addition, measuring electrochemical corrosion potential (ECP) is an effective monitoring and trending method, as well as monitoring and trending the main water chemistry parameters.

The equipment, personnel, and details of inspection are based on the codes and standards of each country. Inspection can reveal cracking and leakage of reactor coolant. The extent and frequency of inspections are based on the as-fabricated conditions of each weld (e.g. whether the weldments were made from IGSCC-resistant material, whether a stress improvement process was applied to reduce residual stresses, and how the weld was repaired if it had cracked).

4.1.4. Acceptance criteria

Detected flaws are evaluated with the appropriate code and/or standard. Effects of preventive action and mitigation may be credited in the evaluation only if its validity has been verified. In the case that the cracks are detected by the microscopic inspection, crack growth and fracture evaluation should be conducted to confirm whether structural integrity can be maintained during further plant operations and for how long.

4.2. PRESSURIZED WATER REACTORS

4.2.1. Scope of the ageing management programme based on understanding ageing

The scope of the ageing management programme against PWSCC is limited to alloy 600 series nickel based alloys and similar weld metals such as those used in the reactor vessel (RV) upper head penetrations and associated J-groove welds, adjacent RV upper head nozzles, RV hot leg/cold leg nozzles, SG inlet/outlet nozzles, and PRZ nozzles.

4.2.2. Preventive actions to minimize and control ageing degradation

The preventive measures to mitigate PWSCC are intended to improve the three synergistic aspects of SCC relating to material, stress and environment.

Material aspects

Wrought alloy 690 and similar nickel based alloy weld metals, for which PWSCC resistance is enhanced by increasing the chromium content to ~30%, are used for the wetted sections. PWSCC resistant alloy 690 can be used for replacing nozzles or for overlay cladding existing alloy 600 type nickel based alloys and weld metals.

Stress aspects

Shot peening can be applied usually after isolating the component from the reactor coolant while water jet peening can be applied under water to improve residual stresses and leave compressive stresses on the wetted surfaces. If it is difficult to access a nozzle from the ID, a high power laser beam (L-SIP technique) can be used to rapidly heat it, which results in a temperature difference across the wall causing thermal expansion strains and on cooling an improvement of residual stresses on the ID wetted surface due to the generation of compressive stresses.

In addition, recent findings have shown that PWSCC has a high probability of initiation where high residual stresses and a hardened surface layer exist due to the effects of welding and surface finishing. To address these concerns, surface finishing methods; such as buffing to remove the hardened layer are an effective preventive maintenance measure against the possibility of PWSCC initiation.

Environmental aspects

Temperature reductions can be possible in some components; for example the RV head. The temperature inside the RVH can be reduced to close to that of the cold leg (T-COLD technique) by increasing the bypass flow to

RVH within an acceptable amount, while not resulting in any influence on reactor performance and safety. It is effective for reducing susceptibility to PWSCC because of the high temperature dependence of PWSCC. In addition, it is essential to accumulate knowledge of other environmental conditions impacting susceptibility to PWSCC.

4.2.3. Monitoring and trending of ageing effects

It is necessary to perform leak tests to check whether there are any indications of primary coolant leakage due to through-wall cracking, which might exist, for example, in upper vessel head penetration (VHP) nozzles, BMI nozzles and associated partial penetration J-groove welds, in order to confirm the structural integrity of the components before any loss of function as a pressure boundary.

Bare metal visual inspections can also confirm indications of primary coolant leaks in the form of boric acid residues on the component outer surface, or corrosion products (rust) that are generated from low alloy steel components.

For locations in RV, SG and PRZ nozzles, surface inspections of the ID by dye penetrant testing (PT), eddy current testing (ECT), or ultrasonic testing (straight beam and longitudinal wave angle beam UT) are applied to confirm whether or not there is any significant cracking. However, UT may not be capable of detecting a clear crack tip echo for certain types of cracks with a longer crack depth than the surface crack length.

In addition, knowledge about the incidents of PWSCC needs to be consistently updated so that corrective actions and preventive measures taken at operating plants can be applied to other plants with common factors in their surface finishing condition.

4.2.4. Acceptance criteria

If cracks are detected by microscopic inspection, crack growth and fracture evaluation should be conducted to confirm whether structural integrity can be maintained during plant operation. Even if the detected cracks are determined to be axial or radial cracks rather than circumferential, it is recommended that a conservative evaluation is performed with an assumption of the presence of circumferential cracks in order to ensure integrity.

If there is no established code for the crack growth and fracture evaluations of nickel based alloys, integrity can be assured in a rather conservative manner by completely removing the cracks and then applying an appropriate surface finish. The aim should be to achieve a smooth surface where excessive tensile stresses would not concentrate while ensuring a minimum wall thickness compatible with the design requirement. It is also recommended that stress improvement measures are taken after the removal of cracks.

5. INSPECTION

5.1. BOILING WATER REACTORS

5.1.1. Piping

Requirements and practices in the USA

NPP Class 1, 2, and 3 piping in the USA is subject to the requirements of Section XI of the ASME Code, as required by the Federal Regulations [10 CFR 50.55a(g)(4)]. Section XI contains the minimum in-service inspection requirements which are typically written around four 10-year inspection intervals (Inspection Programme B of Section XI) to cover a 40-year operating life. This requirement includes a pre-service inspection (PSI) and four in-service inspections (ISI) at 10-year intervals during the 40-year operating life of a nuclear power plant. The specific edition of Section XI required by the Regulations is based on the start of each 10-year inspection interval. In accordance with the Regulations, the examination of components must comply with the latest edition and addenda incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 10-year inspection interval. Further details about these requirements can be found in IAEA-TECDOC-1361.

IGSCC in BWR piping was observed in small bore piping in the early 1970s and in large bore piping in 1982. The BWR Owners Group for IGSCC research was formed in 1979 to address IGSCC in a safe and cost effective manner and developed options to control or eliminate IGSCC including pipe replacement with improved materials, stress improvement, and water chemistry controls.

The US Nuclear Regulatory Commission (NRC) initially responded to this problem by issuing generic communications addressing pipe replacement and weld inspections. In 1984; the NRC developed a long range plan that was documented in SECY 84-301. In 1988, the NRC published an updated position on IGSCC in Generic Letter 88-01, "NRC position on IGSCC in BWR austenitic piping." The generic letter included NRC staff positions on material categorization and associated inspections, mitigation options, repair methods, and flaw evaluation. The BWR industry developed special examination procedures, standardized repair methods, and improved mitigation schemes including industry wide water chemistry improvements. All these enhancements have resulted in reliable inspections and a significant reduction in IGSCC initiation and growth. NUREG-0313, Rev.2, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, contained the technical bases for staff positions. In 1992, the NRC issued a supplement to GL 88-01 that modified some staff positions in the original generic letter.

Special examination methods were developed in response to the need for improved detection and sizing of IGSCC. The BWR Owners Group for IGSCC Research and EPRI developed a programme, the 3-party NDE Coordination Plan, which the NRC endorsed as part of GL 88-01. This programme remained in place until March 1996. At that time, a transition from the NDE Coordination Plan to a new qualification programme was initiated. This new programme was agreed to by the NRC, the GE BWR Owners Group (BWROG), and the Performance Demonstration Initiative (PDI) Steering Committee. This new qualification programme brought the IGSCC examination qualification process into alignment with the PDI programme for satisfying the rules of Appendix VIII of ASME Section XI as amended by the document Performance Demonstration Initiative (PDI), Performance Description, Revision 1, Change 1, December 1996.

The Improved Water Chemistry Committee within the BWROG developed criteria for determining acceptable performance of HWC systems for the purpose of extending inspections schedules for BWR piping weldments. In parallel, the BWR Vessel and Internals Project (BWRVIP), which was formed in 1994, developed a revision to the schedule of piping inspection frequencies in GL 88-01. The final report provides technical bases to support a reduction in the number of welds to be examined in some categories or an increase in the time between inspections in other categories.

Requirements and practices in Spain

The Spanish regulation requires that Spanish plants comply with the rules of the country where the plants were designed. In Spain, there are two BWRs, both of which were designed in the USA and therefore must follow US inspection requirements. Spanish regulators may add additional requirements.

Requirements and practices in Switzerland

In Switzerland, the requirements on in-service inspection are stipulated in the Swiss regulation NE-14. In many cases, NE-14 follows the requirements given in Section XI of the ASME Code. Additional inspections are performed if technical reasons appear or service experience indicates some reason to increase the frequency of inspection.

Requirements and practices in Sweden

Sweden has applied a qualitative risk based approach to in-service inspection for many years. It is based upon a matrix combining potential risk of damage in a given system with the consequences of failure, and the anticipated severity of subsequent radioactive release. The matrix is illustrated in Table 5.1.

The consequence index expresses in a qualitative manner the likelihood that a crack or other degradation process will result in fuel damage, discharge of large amounts of radioactive substances, or other forms of damage

TABLE 5.1. MATRIX USED FOR A QUALITATIVE RISK BASED IN-SERVICE INSPECTION

Damage index	Consequence index		
	High (I)	Medium (II)	Low (III)
High (1)	A	A	B
Medium (2)	A	B	C
Low (3)	B	C	C

which could lead to health problems or an accident. In NPPs, the consequence index is determined mainly by the margin to such consequences as the result of a break or malfunction of the specific component or part of a system. Two aspects are important when determining the assignment of the consequence index:

— System margins — how many system circuits are essential in relation to the number available;
— Thermal margins — how much the fuel can be heated up in relation to acceptable margins.

The damage index expresses in a qualitative manner the likelihood for crack formation or other degradation process occurring in the specific component or system part. The damage index is determined by the loading, environment and material in relation to the dimensions of the component. Components or parts that may be exposed to loads or other conditions, which experience has shown can result in damage or degradation, should be assigned the highest damage index. Components that experience has shown are not expected to be subjected to loads or other conditions which will result in damage are assigned damage index II, and components exposed to minimal loads or other benign operational conditions are assigned damage index III.

In systems in which stress corrosion cracking cannot be excluded, the following conditions are considered reason for assigning damage index I to a component:

— High temperatures (>150°C) and high carbon content (>0.04%) in stainless steel, including stabilized austenitic stainless steels;
— High temperatures (>150°C) and cold worked stainless steel which has not had a subsequent heat treatment;
— High temperatures (>150°C) and nickel based alloys such as alloy 182, alloys 600 and X-750 with compositions and heat treatments which experience has shown are sensitive to stress corrosion cracking;
— High neutron fluence (>5.1020 n/cm2, E >1 MeV).

The above system was first introduced in the SKI regulations SKIFS 1994:1 concerning mechanical components in nuclear facilities and they have been applied since that date. In the more recent versions of the regulations, the specific conditions listed above are no longer included as part of the regulations but are included in the utilities documentation which has been approved by SKI.

In the current version of the regulations concerning mechanical components (SKIFS 2005:2), the use of quantitative risk based inspection programmes is permitted. One of the utilities has been granted permission to apply a modified version of the Westinghouse Owners Group methodology, but did not have any stress corrosion cracking sensitive materials to be considered.

The inspection interval is determined on the assumption that a crack exists in the component or system of a size for which the applicable inspection method has been qualified. The inspection interval is then calculated as the time required for the crack to grow to a critical size using crack propagation data for the specific material/environment combination. The same crack propagation data is used when a crack is found in a component to permit the component being kept in service until a planned repair or replacement can be performed. Qualification of inspection techniques has to be performed using realistic (not geometric) cracks in test blocks made from typical material.

Requirements and practices in Germany

According to KTA 3201.4, the outer and inner diameter and the zone near to the surface of welds of piping above 50 mm nominal diameter have to be examined using PT, UT or RT. The NDT interval is in general 5 years. A quota of welds being tested is determined between 10% and 40% that depends on nominal diameter and operational temperature (above or below 200°C).

Requirements and practices in Japan

Basic inspections are conducted based on JSME S NA1-2002. The NISA regulatory requirements, NISA-161a-03-01, issued in 2003 require more frequent UT than the JSME code for recirculation piping fabricated from type 316(NG) stainless steel, taking account recent SCC experience.

5.1.2. Vessel penetrations and nozzles

Requirements and practices in the USA

Vessel penetrations and nozzles are part of the RPV, and therefore the inspection requirements for these components are contained in IAEA-TECDOC-1470.

Requirements and practices in Spain

The Spanish regulation requires that Spanish plants comply with the rules of the country where the plants were designed. In Spain, there are two BWRs, both of which were designed in the USA and therefore must follow the US inspection requirements. Spanish regulators may add additional requirements.

Requirements and practices in Switzerland

For nozzles and penetrations, basic inspections are conducted based on SVTI NE-14. Welds between safe ends and nozzles follow the same requirements as for piping welds. UT is performed during the inspection of nozzles.

Requirements and practices in Germany

Basic inspections are conducted based on KTA 3201.4 and plant specific regulations. According to KTA 3201.4, the outer and inner diameter and the zone near to the surface of dissimilar welds of piping nozzles above 200 mm nominal diameter have to be examined using PT, UT or RT for flaws in the circumferential direction. 100% of nozzle dissimilar metal welds have to be examined within 5 years. For dissimilar welds, inspection must be aimed at both axial and circumferential flaws according to RSK recommendations.

Requirements and practices in Japan

Basic inspections are conducted based on JSME S NA1-2002. UT is performed during the inspections, except for penetrations. For penetrations, VT is requested.

5.1.3. Reactor pressure vessel internals

Requirements and practices in the USA

Inspection requirements for RPV internals are discussed in IAEA-TECDOC-1471. It should be noted that in the meantime, US plants have committed to comply with the BWRVIP guidelines.

Requirements and practices in Spain

The Spanish regulation requires that the Spanish plants comply with the rules of the country where the plants were designed. In Spain, there are two BWRs, both of which were designed in the USA and therefore must follow US inspection requirements. However, although Spanish regulators may add additional requirements, Spanish utilities have not committed to complying with the BWRVIP guidelines.

Requirements and practices in Switzerland

Basic inspections are performed based on SVTI NE-14 and plant specific regulations. IVVI is performed for the inspection. UT is used for sizing of existing flaws at Mühleberg NPP.

Requirements and practices in Germany

Basic inspections are conducted based on KTA 3204 and plant specific regulations. VT is performed for the inspection.

Requirements and practices in Japan

For the core shroud and shroud support, basic inspections are conducted based on JSME S NA1-2002. For other internals, JSME codes (2004 version) are presently undergoing the endorsement process by NISA. NISA regulatory requirements, NISA-161a-03-01, issued in 2003 require more frequent and extensive VT than the JSME code for core shrouds; taking account recent SCC operating experience. VT is performed during the inspection. ET will substitute VT in the near future.

5.2. PRESSURIZED WATER REACTORS

5.2.1. Piping

Requirements and practices in Germany

Basic inspections are conducted based on KTA 3201.4 and plant specific regulations. According to KTA 3201.4, the outer and inner diameter and the zone near to the surface of welds of piping above 50 mm nominal diameter have to be examined using PT, UT or RT. The NDT interval is in general 5 years. A quota of welds to being tested is determined between 10% and 20% depending on nominal diameter.

Requirements and practices in Switzerland

The requirements for in-service inspection are stipulated in Swiss regulation NE-14. In many cases, NE-14 follows the requirements given in Section XI of the ASME Code. Additional inspections are performed if technical reasons appear or service experience indicates some reason to increase the frequency of inspection.

5.2.2. Vessel head penetrations and nozzles

Requirements and practices in the USA

ASME Code Case 694 will require each licensee to determine the required inspection(s) for each refuelling outage at their facility based on the susceptibility category of each reactor vessel head to PWSCC related degradation; as represented by a value of EDY at the end of each operating cycle (See IAEA-TECDOC-1556 pages 109–110). A number of plants have elected to replace RPV heads rather than inspect.

Requirements and practices in Japan

The basic inspection requirements are given in JEAC-4205, the Japan Electric Association Code for ISI of light water cooled nuclear power plant components and also in the JSME Code on Fitness-for-Service for Nuclear Power Plants, JSME S NA1-2002. The requirements in both are same. Examination Categories B-A to B-D, B-F to B-H, B-J, B-O, B-P and G-P (Section IB, Class 1 Components and Section IG, Reactor Vessel Internals) prescribe the methods, inspection area and frequencies for the RPV ISI. The basic examination required is a periodic volumetric examination of the reactor pressure vessel weld lines.

Requirements and practices in France

The requirements for the French ISI programmes are published in RSE-M. The Code requires periodic hydrotests with acoustic emission monitoring for leaks during the hydrotests, NDE during the outages, a material surveillance programme, loose parts (noise) monitoring during operation, leak detection during operation, and fatigue monitoring. The Code specifies a complete programme including both the utility and regulatory agency required inspections. Areas of the RPV that must be inspected include the beltline region of the shell, all the welds, the top and bottom heads, the nozzle and safe end welds, the penetrations, the control rod drive housings, the studs, the threaded holes, and the supports (See IAEA-TECDOC-1556 page 116).

At the end of 2009, the current schedule is for all alloy 600 RPV heads in France to be replaced with alloy 690 materials and compatible high Cr welds.

Requirements and practices in Germany

ISI in Germany dates back to the late 1960s when a large research and development programme, funded by the Federal Ministry for Research and Technology, was launched. In 1972, a draft version for the In-service Inspection Guidelines of the Reactor Safety Commission was published and this document remained almost unchanged in subsequent issues. This became the basis for the formulation of the German KTA 3201.4 Code, which today specifies the NDE requirements for ISI (See IAEA-TECDOC-1556 page 112).

Basic inspections are conducted based on KTA 3201.4 and plant specific regulations.

Vessel head penetrations of German RPVs are attached with fully ferritic welds. The inner surfaces of circumferential welds of control rod guide tubes (austenitic welds and dissimilar metal welds) are examined by PT, RT or UT. In general, 10% of these welds have to be examined within 10 years.

According to KTA 3201.4, the outer and inner diameter and the zone near to the surface of dissimilar welds of piping nozzles above 200 mm nominal diameter have to be examined using PT, or RT for flaws in the circumferential direction. Four of eight of these nozzle dissimilar welds have to be examined within 5 years.

Considering dissimilar metal welds, inspection must be aimed at both axial and circumferential flaws according to RSK recommendations.

Requirements and practices in Brazil

NRC practice is applied (Order EA-03-009). At each refuelling outage, the EDY parameter is calculated. Its value determines the category for susceptibility to stress corrosion cracking. Angra 1 susceptibility is considered moderate (EDY~10). This susceptibility level requires either NDT or visual inspection at each refuelling outage. In Angra 1, the 40 penetrations of the RPV head were inspected in 1994 by ECT. As of 2000, they were visually inspected at each refuelling outage. In February 2008, all the 40 penetrations were inspected visually and by UT for the first time. No indication was detected. An inspection programme for the welds of vessel head penetrations and nozzles is being prepared based on EPRI document MRP-139. Ultrasonic inspections of Inconel welds of the RPV nozzles (hot and cold legs) were carried out in 2006 in Angra 1. Visual inspection was undertaken of all of Inconel welds of pressurizer nozzles (relief, safe, surge and spray). Up to the present time, no indication has been detected.

Requirements and practices in Switzerland

The requirements for in-service inspection are stipulated in the Swiss regulation NE-14. In many cases, NE-14 follows the requirements given in Section XI of the ASME Code. Additional inspections are performed if technical reasons appear or service experience indicate some reason to increase the frequency of inspection; e.g. based on technical analyses of the UT inspections of the dissimilar vessel head penetrations welds at NPP Beznau, which are performed every 4 years.

5.2.3. Steam generators (primary water stress corrosion cracking)

Requirements and practices in Brazil

The two steam generators of Angra 1 were replaced in the first semester of 2009. The inspection programme used on the replaced Angra 1 steam generators was:

— 100% of the tubes using a bobbin coil probe (the whole extent of the tubes);
— 100% of the tubes on the hot leg side at the top of the tubesheet using a plus point probe (3 inches above and 2 inches below);
— 100% of the tubes at intersections with the tube support plates using a pancake probe, when the bobbin coil signal exceeds 1.0 volt;
— 100% of the tubes at intersections with the baffle plate (01H plate) using a plus point probe, when the bobbin coil probe signal shows a DSI (distorted signal indication), NQI (non-quantified indication) or copper deposits greater than 2.0 volts or a DNI (dent) when the signal is greater than 4.0 volts;
— Approximately 1000 dented intersections using a pancake probe (the selection is made randomly);
— 100% of sleeves using a plus point probe.

Angra 1 adopted a repair on detection criteria. This means that all tubes with any sign of degradation due to SCC were repaired using a plug or sleeve. Angra 2 steam generator tubes are made of alloy 800 and less extensive inspections are planned based on the good SCC operating experience to date. Every 4 years, a 10% sample of tubes in each SG is inspected.

Requirements and practices in Spain

All six Spanish PWR plants in operation have requirements for inspection of the steam generator tubes in their technical specifications and also in their In-Service Inspection Programme. Additionally, all plants have limits and requirements for monitoring primary-to-secondary leaks in the technical specifications.

The Spanish Regulator (CSN – Consejo de Seguridad Nuclear) has requested that the eddy current inspection techniques used in steam generators be demonstrated according to a national methodology (CEX-120), based on the ENIC methodology.

After issuance of the US NRC GL 2006-01, Steam Generator Tube Integrity, all Spanish PWR plants adopted the proposed inspection intervals of TSTF-449 Rev. 4, Steam Generator Tube Integrity, in their In-Service Inspection Programme. The inspection techniques and the sample sizes are reviewed, following a CSN request, in the document Degradation Assessment. This revision is based on EPRI guidance for the evaluation of tube integrity according to the degradation mechanisms identified by operating experience.

The only Spanish plant with alloy 600TT steam generator tubes has totally changed its technical specifications according to TSTF-449 Rev. 4. The other plants with alloy 800 modified tubes have not changed their technical specifications, which are based on the 1976 US NRC RG 1.83, In-service Inspection of Pressurized Water Reactor Steam Generator Tubes. Some plants have special amendments in their technical specifications due to historical degradation of SG tubes or other special features in their licensing basis.

The tube plugging criterion for the Spanish US designed plants is 40% through-wall, as required by ASME XI. This criterion in the only German design plant is established by US NRC RG 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes.

Requirements and practices in Germany

Basic inspections are conducted based on KTA 3201.4 and plant specific regulations. According to KTA 3201.4, the outer and inner surface and wall thickness of the SG tubing has to be examined using ET. The NDT has to cover the tube in the area exposed to the secondary water above the upper tube expansion in the tube sheet. For each SG, 10% of the tubes have to be examined within 5 years.

Requirements and practices in Japan

The basic inspection requirements are given in the JSME Code on Fitness-for-Service for Nuclear Power Plants, JSME S NAI-2002. Examination Categories B–Q prescribes the methods, inspection area and frequencies of examination for steam generator tubing ISI. The basic examination required is a volumetric examination using ECT and the frequency of examination varies according to the tube material. 100% inspection of alloy 600 tubes are performed using an array type ECT probe (MHI intelligent probe) every outage. 100% inspection of alloy 690 tubes are performed using a bobbin type ECT probe every two outages.

Requirements and practices in the Republic of Korea

Inspection guidelines are described in Bulletin number 2004-13 of the Ministry of Education, Science and Technology in the Republic of Korea, Official Regulations for In Service Inspection of Nuclear Facilities. The regulations describe inspection tools, procedures, technical specifications, sampling criteria, inspection intervals, etc. The reference regulation for Bulletin 2004-13 is the USNRC R. G. 1.83 (In-service Inspection of Pressurized Water Reactor Steam Generator Tubes). Defects with over 40% of tube wall penetration should be repaired. Independent of the regulation, all crack-like defects are recommended to be plugged, except for two units.

Requirements and practices in Switzerland

At NPP Beznau, problems with alloy 600 TT tubing began only one year after commercial operation and the steam generators have experienced different types of tube defects caused by both IGSCC and fretting. Therefore, the two steam generators in Unit 1 of the Beznau NPP were replaced in 1993. The steam generators in Unit 2 were replaced in 1999. The steam generators are now equipped with thermally treated alloy 690 tubes.

The requirements for in-service inspection are stipulated in the Swiss regulation NE-14. In many cases, NE-14 follows the requirements given in Section XI of the ASME Code. Additional inspections are performed if technical reasons appear or service experience indicate some reason to increase the frequency of inspection

5.2.4. Reactor pressure vessel internals — baffle former bolts

Ultrasonic examination is a useful, accurate and reliable technique for the evaluation of reactor internals components such as baffle former bolts where detection of indications is an essential part of reactor internals ageing management. Ultrasonic examination techniques must be customized for specific geometrical configurations of baffle former bolts; e.g. the presence of locking devices on the fastener heads, head type (internal or external hexagonal bolt heads), and/or accessibility restrictions.

Requirements and practices in the USA

ASME Section XI in-service inspection requires visual inspection of RVI. The required visual inspections of RVI are carried out within a given 10 year period. Currently, enhanced visual inspection of RVI is under development and is expected to be an ASME XI requirement. In addition, it is expected that in the near future, volumetric inspection of critical RVI such as baffle former bolts will be required.

Requirements and practices in France

The basic inspection technique for RVI at all plants is a televisual examination, which is carried out every 10 years. For the baffle former bolts, televisual inspection of the 900 MW plant series is applied to the 3 rows of bolts at the bottom of the lower core internals and to the upper row. For the 4-loop plant series, televisual inspection is extended to the two upper rows of bolts. Televisual inspections are also performed on the upper baffle plates. Complementary inspections using ultrasonic testing are applied periodically (every 2 to 10 fuel cycles) on some specific 900 MW plants where baffle former bolt cracking was detected (in particular the oldest CP0 plants). Moreover, monitoring of the RVI is conducted using techniques such as the neutron ex-core monitors in the CP0 plant series. Neutronic measurements allow the amplitude and spectrum of the vibrations of RVI to be monitored. Accelerometer sensors are also used for monitoring loose parts in the primary circuit. Repair operations are based on the global extent of cracking and specific location of cracked bolts in any one plant (See IAEA-TECDOC-1557 page 46).

Requirements and practices in Germany

In Germany, after the replacement of alloy X-750 Baffle-Former bolts with austenitic stainless steel 1.4571, intensive ultrasonic testing was performed by the utilities. After an extended period without any indications being detected, the standard requirements as per KTA 3204 were applied once again; i.e. visual examination of the RVI. The scope of inspection is defined specifically for each NPP. According to KTA 3204, it can be conducted over 4 or 5 refuelling outages, resulting in a scope of 20–25% of the baffle former bolts per refuelling outage (See IAEA-TECDOC-1557 page 44).

Requirements and practices in Japan

Although Japanese utilities have carried out UT inspections for baffle former bolts at several operating plants, those inspections did not show any indications of cracking. However, the Japanese PWR utilities and manufacturer established guidelines for inspection and evaluation of baffle former bolts taking into account the service experience of foreign PWRs. This is considered as a proactive maintenance approach to maintain and improve the reliability of RVI components. The guidelines were published by the Thermal and Nuclear Power Engineering Society in March 2002, and have been already incorporated into the JSME Code on Fitness-for-Service for Nuclear Power Plants.

The main part of the guideline is used to predict when and how many bolts will fail taking into consideration plant specific operational conditions; such as operational times, temperature, neutron fluence, and applied stress including the effects of void swelling and irradiation creep. The initial inspection times and inspection intervals are determined based on the relationship between an acceptable number of damaged bolts and the predicted number of damaged bolts.

Requirements and practices in the Republic of Korea

In service inspection according to ASME Section XI has been applied to manage reactor internals in the Republic of Korea. The UT system called KOBIS (Korea baffle former bolt inspection system) inspects baffle former bolts and baffle edge bolts. Primary water chemistry control is based on the plant specific operational guideline Chemistry 0-6-001. A loose part monitoring system (LPMS) has been operating in order to protect the reactor internals from any metallic loose parts. An improved inspection technique is being considered to ensure the integrity of RPV internals.

Requirements and practices in Brazil

RVIs are inspected every 10 years. For Angra 1, the RVIs were visually inspected during the 14th outage (2006).

REFERENCES TO SECTION 5

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6. MITIGATION AND REPAIR METHODS

Table 6.1 summarizes the mitigation and repair techniques that have been developed to counter SCC related ageing degradation in both BWRs and PWRs. These techniques are described in more detail in the following section.

6.1. BOILING WATER REACTOR TECHNIQUES

6.1.1. Material changes

German experience

The damage found in the weld regions of German BWR piping fabricated from stabilized stainless steels was predominantly caused by thermal sensitization of the HAZ during welding, see Section 3.1.1.

Extensive post-service examinations of German BWR piping plus R&D work have lead to the conclusion that the titanium stabilized grades (1.4541, equivalent to type 321) have lower resistance against thermal weld sensitization compared to the niobium stabilized grades (1.4550, equivalent to type 347). The root cause is the lower thermal stability of titanium carbide compared to niobium carbide. This leads to a higher amount of free carbon in solid solution after welding heat input, which then promotes secondary chromium carbide formation during heat input from subsequent weld passes.

Consequently, niobium stabilized low carbon grade piping (1.4550 S, equivalent to type 347NG) has been chosen since the 1990s for all replacements of hot reactor water (T >200°C) piping in German BWRs [6.1]. In the German code KTA 3201.1, the carbon content has been accordingly limited for this purpose to a maximum of 0.03% and the minimum stabilization ratio, Nb/C, was raised to 13, with a maximum of 0.65% Nb in order to prevent hot cracking. In addition, optimized manufacturing and welding methods have been used for the replacement work. So far, after two full NDE cycles (representing a total of 8 or 10 years), no indications of SCC cracking have been detected in the replaced piping [6.2].

The RPV internals of all operating German BWRs are made of niobium stabilized stainless steels with controlled low C contents and sufficiently high stabilization ratios (Nb/C) similar to the above mentioned piping grades; see also German code KTA 3204. The same materials could be used for replacement work, but due to the excellent operational behaviour of the RPV internals in all operating German BWRs, no replacement work has been needed.

TABLE 6.1. MITIGATION AND REPAIR METHODS FOR BWRs AND PWRs

Technique	BWR	PWR
Material changes	— From 321 to 347 NG — From 304 to 316 L/NG — Improved heat treatment of X-750 — Alloy 600: Mod. Alloy 600 (Nb added) — XM19	— SG: from 600/CS to 690, 4XX SS (TD981) — Instrument tubes — Pressurizer nozzles for instrumentation alloy 600 to type 316 — Pzr sleeves 600 to 690 — RV nozzle spool piece (VC Summer) — X-750 AH to improved version X-750 — X-750 to CW316 SS
Isolation techniques	— Corrosion resistant cladding	— 690 SG sleeving — Plating (Ni) — Coating (cold spray)
Weld material changes	— From alloy 182 to Nb controlled 182, 82	— From 82/182 to 52/152 — Inlays
Design changes	— Reducing number of welds by using induction bent pipes or forged materials	— SG: form drilled support plates to broached quatrefoil holes or egg crate TD981 — Down flow to up flow RPV internals conversion — Bolting design changes e.g. head-shank radii, thread details — Cooling holes in baffle formers
Weld overlays	— External overlays to generate compressive stresses on internal surface	— Pressurizer external overlays to generate compressive stresses on internal surface and replace structural function of original weld
Stress improvements	— Induction heat stress improvement — Peenings (LP, WJP, SP, USP) — MSIP — Polishing — PWHT — SHT — Improved weld preparation (including narrow gap welding, heat sink welding)	— MSIP — Shot peening — Laser peening — Cavitation/waterJet — L-SIP — In situ heat treatment
Environment improvement	— HWC — NMCA, NMCL — Improved NWC — Low leakage fuel loading	— Improved secondary side chemistry — Inhibitor application — Zinc on primary side — Optimized hydrogen — Temperature reduction — Low leakage fuel loading
Mechanical repair	— Tie rods — Clamps — Brackets — Roll repair	— Half nozzle repair — Tie rod repair
Component replacement/removal	— Core shroud — Jet-pump, etc. — Piping	— Steam generator — RPVH — Guide tube support pins 1st time/2nd time — RPV Internals 1 upper USA/4 U/L in Japan by end of 2008 — Pressurizers (4)

Japanese experience

During the early 1980s, many cases of SCC of type 304 stainless steel were reported in the heat affected zones (HAZ) of recirculation piping. One of the root causes of the SCC was identified to be the increased susceptibility to IGSCC caused by thermal sensitization during the welding process. The degree of thermal sensitization was mainly related to the extent of Cr-depleted zones on the grain boundaries, which depends on the carbon content and time at the sensitization temperatures during welding.

Consequently, the heat input during the welding process was decreased and low carbon grade s of stainless steel such as type 304L and type 316L were chosen to avoid the thermal sensitization. These low carbon grade stainless steels have carbon contents of less than 0.03%. It is, therefore, difficult to form carbides on grain boundaries and thermal sensitization does not occur during the welding process.

Since low carbon grade stainless steels such as type 304L and type 316L tend to have less strength than type 304 due to reduced carbon content, type 316NG (nuclear grade) was developed to compensate for this strength reduction by adding nitrogen. In type 316NG, the carbon content is less than 0.02% and the nitrogen plus carbon content is less than 0.12%.

These low carbon grade stainless steels have shown in many laboratory tests a much higher IGSCC resistance and much lower crack growth rates than those of type 304 stainless steel. Since the 1980s, type 316NG stainless steel has been used for recirculation piping and type 316L has been mainly used for in-reactor components. This material change has significantly improved plant performance by reducing the occurrence of IGSCC.

X-750

Alloy X-750 in the EQA condition (equalized and aged heat treatment) is known to be susceptible to IGSCC initiation and growth in the BWR environment. For GE-BWRs there have been several failures of replaced alloy X-750 hold-down beams in the EQA condition, including the recent Quad Cities 1 failure. Design improvements were made to extend the service life of these beams, including modified heat treatment, lower preloads and larger cross-sections to lower stresses [6.3].

Regarding the heat treatment, for the ‘old design’ EQA condition, the closed die forged beam was equalized at 885ºC for about 24 hours followed by an ageing treatment at 704ºC for 20 h. As a result of the observed failures of hold down beams, the heat treatment was revised to the so-called ‘high temperature annealing (HTA)’ [6.4]. After closed die forging, the material is solution annealed at 1093ºC for about 1 to 2 hours, followed by a water quench and then by ageing at 704ºC far ca. 20 h. The main goal of the HTA treatment is to precipitate the strengthening phase γ’, $NiAl_3$; another advantage is a fine, dense $M_{23}C_6$ carbide distribution at grain boundaries [6.5]. The change to the less susceptible HTA material condition was combined with a reduction of the preload of hold down beams because both heat treatment and loading strongly affect susceptibility to IGSCC.

6.1.2. Isolation techniques

Corrosion resistant cladding

Corrosion resistant cladding (CRC) is a SCC mitigation technology that relies on a corrosion resistant metal deposit such as type 308LSS or alloy 82 on the surface of SCC susceptible regions of components.

For CRC, welding is most often used but there are also other methods available for in-core monitor housing (ICMH) such as melting on a thin plate using a TIG torch or by melting a sprayed layer of metal particles using laser irradiation [6.6].

CRC can also be performed with weld metal containing a noble metal. The process is then called Noble metal cladding (NMCL). The advantage of this technology is that in combination with hydrogen water chemistry, an additional assurance of SCC mitigation can be achieved.

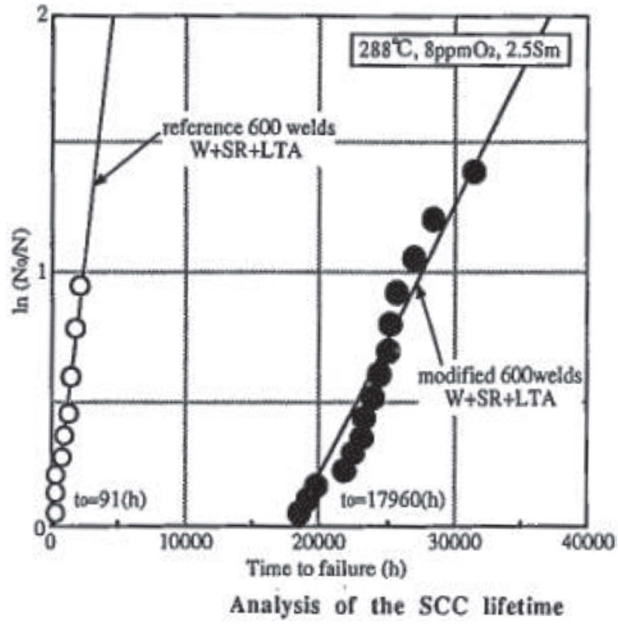


FIG. 6.1. SCC lifetime of conventional and modified Ni based alloys (alloy 600 welded with alloy 182) [6.7].

6.1.3. Weld materials changes

Nb controlled alloy 182 and 82

Similar to austenitic stainless steel, Cr-depletion adjacent to grain boundaries caused by Cr carbide precipitation can lead to IGSCC of alloy 182. In contrast to austenitic stainless steel, a decrease of carbon content is not effective for improving SCC resistance because Cr carbides are formed even at small carbon contents due to negligible carbon solubility in this alloy. Therefore, carbon stabilization by Nb addition was adopted for SCC resistant alloy 182 in the same way as for stabilized stainless steel. Nb content and C/Nb ratio are controlled in the improved alloy. FIG. 6.1 shows SCC test results of conventional and improved alloy 182 Indicating that the IGSCC initiation time of improved alloy is much longer than for conventional alloy.

Alloy 82 contains higher Cr and Nb than alloy 182 and has proved to be SCC resistant in BWRs. Recently constructed Japanese BWRs use alloy 82 or Nb-controlled alloy 182.

6.1.4. Design changes

Reducing the number of welds by induction bent elbows

During the replacement of hot reactor water (T >200°C) piping in German BWRs in the 1990s, piping induction bends were used instead of elbows. The primary goal was to reduce substantially the total number of welds (by 50% and more), thus minimizing future NDE needs for these pipes. As an additional beneficial effect, the number of potential initiation sites for IGSCC by both mechanisms i.e. with and without thermal sensitization, were also minimized.

Reducing the number of welds by using forged material

The overall structure of the replaced core shroud in Fukushima-Daiichi Unit 3 of the Tokyo Electric Power Company in Japan is shown in FIG. 6.2. The left hand side shows the cross-section of the new shroud and the right hand side shows the old core shroud design. The number of welds was very significantly reduced because it directly influences the probability of cracking. The new core shroud consists of three forged cylinders (type 316L SS) with no vertical welds, and a lower ring (alloy 600) with two vertical welds. In all, seven circumferential welds were reduced to four circumferential welds, and 24 vertical welds were reduced to two vertical welds [6.8].

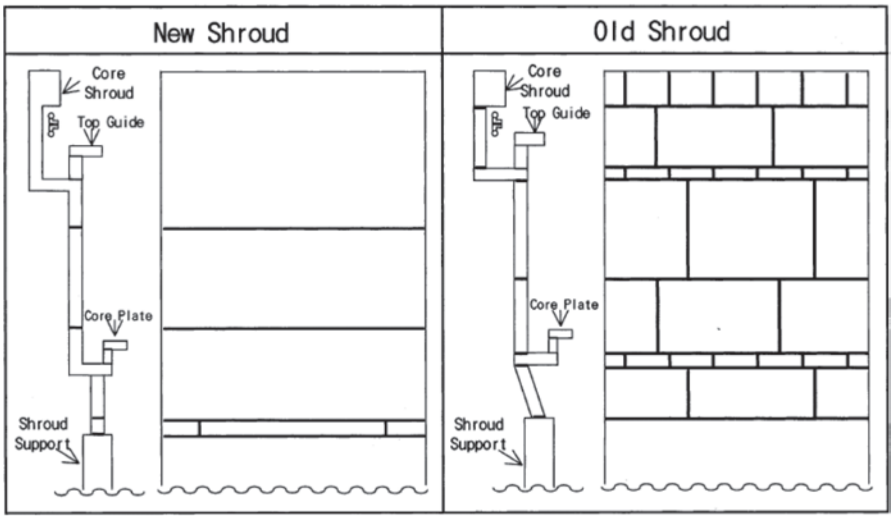


FIG. 6.2. Comparison between the old and new core shroud structures.

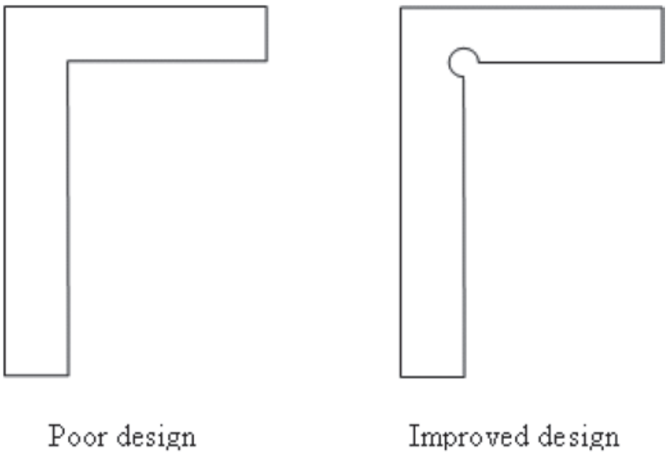


FIG. 6.3. Sketch of the tie rod upper support designs.

Load stress reduction by design change

Over the last years there have been several instances of cracking, with some resulting in failure, in replaced alloy X-750 jet pump hold down beams. The observed cracks initiated in high stressed regions, the thread region and the end region, and then propagated across the beam, following the highest stress trajectory (perpendicular to the beam axis).

Apart from modifying the heat treatment of the material, two design modifications were introduced to lower the stress level in the component in order to extend the life of these hold-down beams. First the applied preload on the beam was reduced and second, the geometry of the component was optimized with a higher cross-section to lower the stress.

Another issue related to the use of alloy X-750 is tie rod upper support cracking that occurred in Hatch-1 in 2006. Cracking initiated in a highly stressed region caused by a poor design; a sharp corner leading to stress concentration. As a consequence, the configuration of the upper support was modified, allowing for higher radius corners in order to reduce stress concentration, see FIG. 6.3.

6.1.5. Weld modifications (overlays)

US experience

The weld overlay (WOL) repair provides a practical technique for repairing piping components. WOL can restore the load carrying capability and integrity of the repaired location through the application of external weld material. WOLs entitle two mitigation methods for redundancy: a resistant material corrosion barrier and a compressive residual stresses on the inner diameter of the piping to arrest or prevent further cracking. A crack growth analysis is performed for a pre-emptive full structural WOL by assuming a flaw having a depth of 75% of the wall thickness over the full circumference of 360º. For pre-service examination, the final weld overlay and surrounding area are surface examined using PT and/or MT and ultrasonically examined in accordance with the ASME Code and applicable Code Cases. If any flaws are identified in the upper 25% of the underlying material, then the as-found flaw has to be used in the crack growth analysis.

WOLs for BWRs pressure boundary piping applications are currently applied routinely in the US. WOLs are also applied in other countries such as Spain, Switzerland (for temporary repair) or Taiwan. WOLs are considered in the USA as long term repairs and do not require specific approval from the US NRC. For austenitic stainless steel piping, WOL sizing (i.e. thickness and length) is determined through guidance provided in ASME Code Case N-504. An experimental confirmation of the load carrying capacity of the WOLs was provided in a US NRC sponsored report, NUREG/CR-4877.

A recent ASME Code Case N-740 provides the design guidance for full structural WOLs using alloy 52 (or alloy 52 M) to reinforce alloy 600 base material and alloy 82/182 welds, which is applicable to pressure boundary piping of both BWRs and PWRs. The non-mandatory Appendix Q of ASME Section XI also provides additional design, examination and inspection guidance for stainless steel WOLs. In the 1980s, the EPRI NDE Center conducted a comprehensive programme, documented in EPRI NP-4720-LD, to develop procedures and to demonstrate the effectiveness of NDE techniques and the conditions under which successful WOL inspections are possible. The report also included a survey of the BWR recirculation piping system WOLs then in-service. The WOL repair technique development in the USA is summarized in the ASME guide book [6.9].

6.1.6. Stress improvements

Improved welding techniques

Manufacturing processes including welding have a large influence on the stress corrosion behaviour of components since they affect both material susceptibility and the residual stress state. Both are important contributors to SCC. Therefore, improved welding techniques can help minimize SCC. In that context, the following aspects are addressed in the following sub-paragraphs:

— Weld edge preparation;
— Narrow gap welding;
— Heat sink welding.

Weld edge preparation

Sound weldments with minimal risk of SCC begin with appropriate and careful weld edge preparation to ensure reproducible conditions during welding. Whenever possible this should be performed by machining with parameters that minimize cold work and the resulting relatively high levels of hardness and residual stress. By limiting cutting speeds and using specific HSS (high speed steel) tooling, surface related cold work can be kept to reasonable depths (<150 μm) and hardness levels (<250 H_V). Limiting such parameters has a beneficial effect on re-crystallization and grain growth in the HAZ that also influence later SCC behaviour. For example, it has been reported that low carbon stainless steels can be susceptible to SCC when the hardness exceeds 300H_V [6.10].

For further reduction of cold work of surfaces exposed to the hot water reactor coolant, polishing of the inner diameter of the weld edge region can be used before welding.

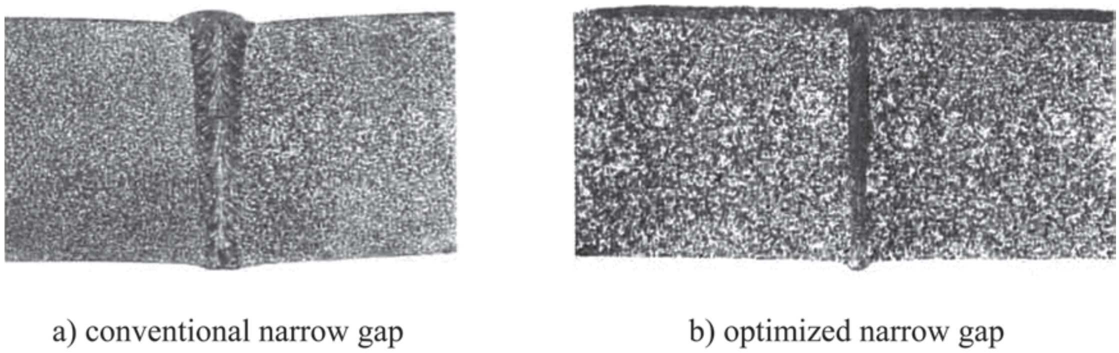


FIG. 6.4. Narrow gap welds in 30 mm austenitic stainless steel; the positive effect of the further reduced gap width on distortion can readily be seen.

Narrow gap welding

Due to its geometry, the narrow gap welding technique has the general benefit of a reduced weld volume and consequently less heat input. This aspect positively affects shrinkage and distortion as well as the width of the weld heat affected zone. This welding technique has been successfully employed for many years.

A process variation is provided by the so-called ‘modified’ narrow gap technique. In comparison to ‘conventional’ narrow gap techniques, the weld volume is further reduced by using welding gaps as narrow as approximately 5 mm. This technique deposits many very thin stringer beads with very low heat input and low interpass temperatures. This results in relatively short dwell times within the sensitizing time-at-temperature regimes and, furthermore, leads to less shrinkage and distortion. Also, there is a beneficially influence on the HAZ microstructure with this technique.

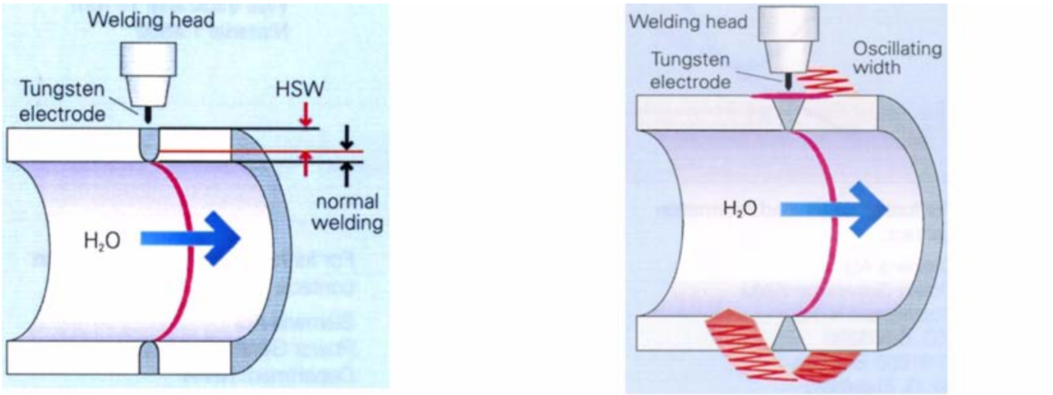
The modification additionally employs a wide top layer comprising numerous stringer beads that result in a shift of residual stresses and leaves the root pass area in a compressive state. Fabrication experience has shown that to take full advantage of this technique, fully automated GTAW orbital welding techniques are recommended. FIG. 6.4 shows images from a 30 mm narrow gap joint welded by the conventional and modified techniques. The positive effect on shrinkage and distortion can readily be seen.

Heat sink welding

Heat sink welding (HSW) involves water cooling on the inside surface of pipes during all weld passes that are deposited after the root pass or the first two layers. The cooling effect can be obtained by using either a slow or turbulent water flow. This technique can be readily adapted to on-site welding without impacting joint design and later inspectability. FIG. 6.5 displays the principle features during the application of HSW.

HSW has two main beneficial effects. First, the cooling effect of the water minimizes the time-at-temperature in regimes where thermal sensitization could occur. However, more importantly HSW results in a steep temperature gradient through the pipe wall during welding, which causes thermal tensile stresses on the pipe internal diameter to exceed the material yield strength. After completion of welding and cooling down of the entire weldment, a stress transformation occurs after which the weld root area is under compressive stress for a reasonable depth into the pipe section. This, in combination with a reduced risk of thermal sensitization, provides significant resistance to SCC. It should be mentioned that due to its nature, this welding process is more appropriate for replacement campaigns since its beneficial effects can be better utilized in newly welded joints.

Similar to the principles of HSW, a process variation called last pass heat sink welding (LPHSW) can be applied. LPHSW represents a post weld measure that can be applied to existing welds with the aim of producing compressive residual stresses in the root pass. The basic principle is identical to HSW with a water flow inside the pipe being used to generate a heat sink during welding operations, except that in this case only the last passes are welded with water cooling inside the pipe. The capping of such welds is performed with a relatively wide weave bead (with a width of approximately 3 times the pipe wall thickness) that is deposited under high heat input. Mechanized processes such as GTAW and GMAW are preferred but other processes such as SMAW could also be used.



(a) Heat sink welding (HSW) (b) Last pass heat sink welding (LPHSW)

FIG. 6.5. HSW and LPHSW schematically shown for a GTAW application.

Induction heat stress improvement (IHSI)

IHSI (induction heat stress improvement) can improve the residual stress state at the pipe ID by applying a temperature difference across the pipe thickness by induction heating the pipe OD and water cooling the pipe ID. IHSI has been applied to type 304 piping welds in many BWRs in Japan and in the USA. For application to type 304, the maximum temperature is limited to 550°C to avoid thermal sensitization; in this case the post-IHSI stress at the pipe ID remains slightly tensile. Recently, a modified IHSI has been developed for type 316(NG) piping. For modified IHSI, the maximum temperature at the pipe OD can be increased to 650°C due to the high sensitization resistance of the base material so that it can also improve the stress state to be fully compressive at the pipe ID. It has been applied to Type 316(NG) recirculation piping of Japanese BWRs.

Peening (WJP, LP, SP, etc.)

Techniques which introduce a compressive surface residual stress have been shown to be effective for mitigating SCC. Peening introduces a compressive stress on the surface by local energetic impacts by different means. A compressive residual stress of several hundred MPa to a depth of 300–1000 μm can be obtained. SCC susceptibility can be significantly reduced or eliminated by such peening processes and whose effectiveness for austenitic alloys has been demonstrated by laboratory testing.

Water jet peening (WJP) utilizes shock pressure waves derived from collapsing cavitation bubbles generated by a high pressure water jet [6.7.1–6.7.3]. Laser peening ~~-~~(LP) utilizes green light of a frequency-doubled high energy pulse Nd:YAG laser delivered with an optical fibre that penetrates water and generates a high pressure plasma of several GPa on the surface.

Shot peening (SP) utilizes spherical type 304 stainless steel shots (diameter <2mm) that are hardened during the production process so as to have Vickers hardness about 500. The shot are projected by highly pressurized water (~1MPa) or air onto the surface.

Ultrasonic shot peening (USP) utilizes ultrasonic vibrations of a piezo transducer to drive shots larger than conventional SP to the surface being treated. USP is deployed in a closed system with a chamber.

WJP, LP and SP have been remotely applied under water to core shroud welds of several Japanese BWRs. SP has been performed in dry condition for core shroud replacements. WJP and LP have been applied to BWR CRD stub tube attachment welds in Japan.

Polishing

Polishing of a weld can be used not only for removing surface cold work layers but also for improving the surface residual stress. However, the depth profile of compressive stress is generally shallower than in the case of peening. The degree of cold work layer removal and stress improvement depends on the polishing tool and process.

Polishing technology for core internals welds has been applied in Japanese BWRs to improve the surface residual stress state to compressive over a depth of about 50 µm.

Recently a new polishing technology has been developed in the USA [6.11]. This polishing technology removes a rather deep surface layer (~0.2 mm) to eliminate the region degraded during plant operation and introduces compressive stress on the renewed surface. These characteristics are especially suitable for application to components of alloy 182 which may have a degraded surface layer containing micro fissures or defects.

6.1.7. Environment improvements

Water chemistry controls for BWRs

IAEA-TECDOC-1471 [6.3] describes several SCC mitigation methods by coolant chemistry control that offer significant potential to mitigate SCC. Commonly used approaches are:

- Very clean water achieved by lowering the reactor water chloride and sulphate concentrations and conductivity to very low values;
- Lowering the ECP in the bulk water by hydrogen addition.

In addition, the protective effect of the oxide layer on the metal surface can be improved by adding zinc.

Internationally accepted recommendations on water chemistry control can be found in the VGB guidelines R401J 2006 revision [6.12] and the EPRI BWR Water Chemistry Guidelines –2004 Revision [6.13].

The method currently being used to mitigate IGSCC/IASCC in BWR internals is lowering ECP by HWC, with or without noble metal technologies. Additional information about experience with these mitigation methods and their effects on IGSCC/IASCC, radiation dose and fuel integrity is provided in the BWR Water Chemistry Guidelines — 2004 Revision. These Guidelines are an industry consensus document and are updated periodically. The mitigation methods are summarized in the following sub-sections.

Hydrogen water chemistry

Hydrogen water chemistry (HWC) was first tested in Sweden in the late 1970s at the Oskarhamn NPP and in the early 1980s at Dresden Unit 2 in the USA. Hydrogen was introduced into the feedwater in order to reduce highly oxidizing species (O_2 and H_2O_2) to low concentrations in the recirculation water and produce a much less oxidizing environment thereby mitigating IGSCC in recirculation piping.

Under normal water chemistry (NWC), the ECP is about +200 mV (SHE). Laboratory and in-reactor tests have shown that initiation and propagation of IGSCC is effectively mitigated when the ECP is below -230 mV (SHE). However, even at higher ECP, a beneficial mitigation effect can be achieved (see FIG. 6.6).

Depending on the concentration of hydrogen and on the plant design, the effect of HWC varies according to location in the internals (see FIG. 6.7). The concentration of feedwater hydrogen commonly used to mitigate IGSCC in BWRs varies from 0.3 to 2.5 ppm. One of the drawbacks of HWC is an increase in the main steam line radiation levels caused by N-16 as volatile ammonia instead of nitrate. FIG. 6.8 shows a more detailed description of the ECP distribution for BWR internals as a function of the hydrogen dosage.

It should also be pointed out that the required amount of hydrogen needed to reach a certain ECP depends on the core loading, which determines the dose rate in the downcomer and in turn determines the hydrogen/oxidizing species recombination kinetics and hence the oxidant concentrations. Also, the required amount of hydrogen varies over the reactor fuel cycle as a consequence of burnup.

As a summary of the environmental effects on IGSCC, FIG. 6.9 shows the strong effect of environment expected on the behaviour of existing cracks. Even though the figure shows a principle and is based on predictions, it demonstrates the very important effect of environment with respect to mitigating SCC. The importance of both water purity and HWC is illustrated [6.16].

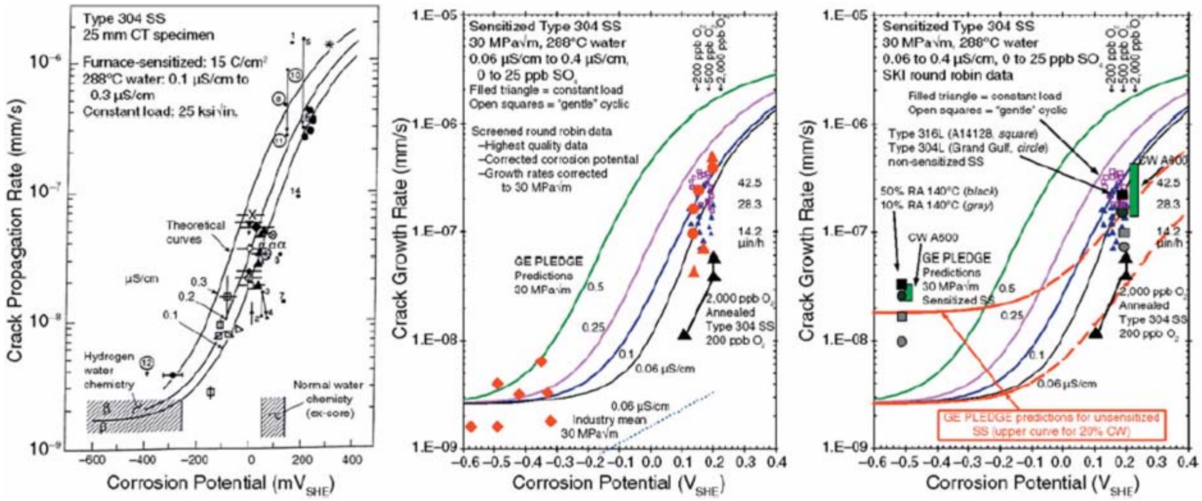


FIG. 6.6. Relationship between crack propagation rate and corrosion potential for sensitized Type 304 stainless steel in 288°C water under constant load and conductivity between 0.1 to 0.3 μS/cm. The curves are the predicted relationships based on the slip-oxidation mechanism of crack propagation [6.14].

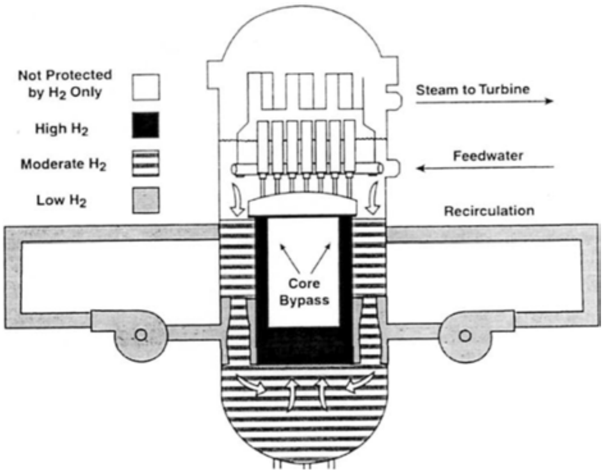


FIG. 6.7. Hydrogen water chemistry in reactor internals [6.15].

Noble metal technologies

One of the important drawbacks of HWC is an increase in the main steam line radiation levels caused by N-16 in the chemical form of volatile ammonia. In order to reduce the dose effect caused by HWC, three noble metal technologies were developed; i.e. (a) noble metal chemical addition (NMCA), (b) noble metal coating and (c) on-line noble metal chemical addition. The objective of these technologies is to deposit noble metal particles on the surfaces in contact with the reactor water.

NMCA involves injecting platinum and rhodium compounds into the reactor water during an outage. The platinum and rhodium particles incorporated into the surface oxide film catalyse the hydrogen recombination effect that in turn reduces the ECP. This leads to a reduction in ECP to values to below -230 mV at feedwater hydrogen concentrations of 0.2–0.4 ppm, in contrast to 1–2 ppm required with HWC without noble metal. A cooperative effort to demonstrate NMCA at the Duane Arnold Energy Center was undertaken by GE, IES Utilities, BWRVIP and EPRI in 1996. The demonstration showed that NMCA treated piping and reactor internals in lower and upper core could be protected at a feedwater hydrogen concentration of 0.25 ppm without significantly increasing the main steam line radiation levels. The BWRVIP conducted an extensive surveillance programme of NMCA

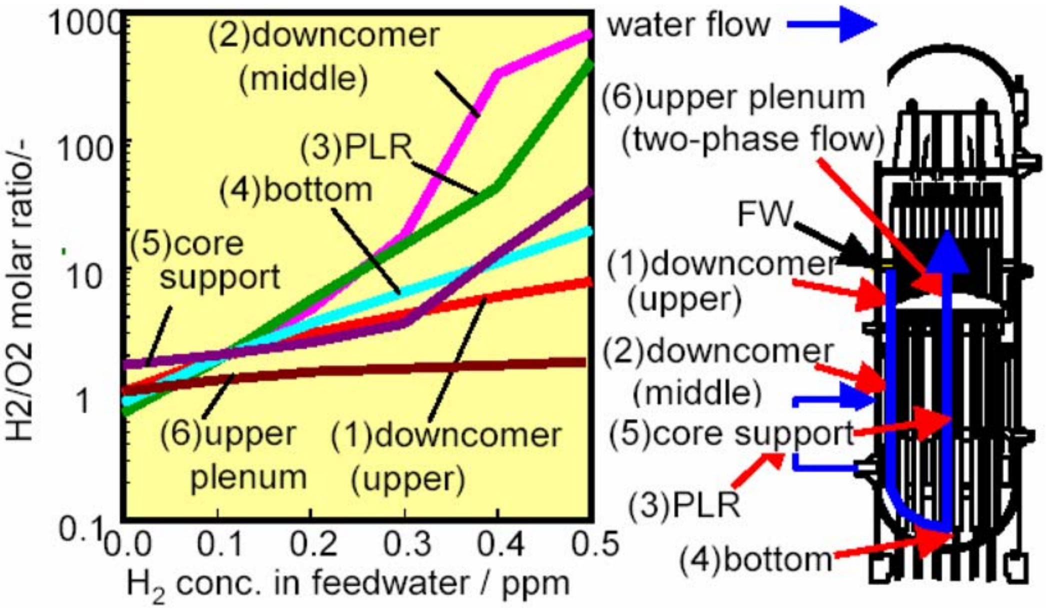


FIG. 6.8. ECP distribution for BWR internals.

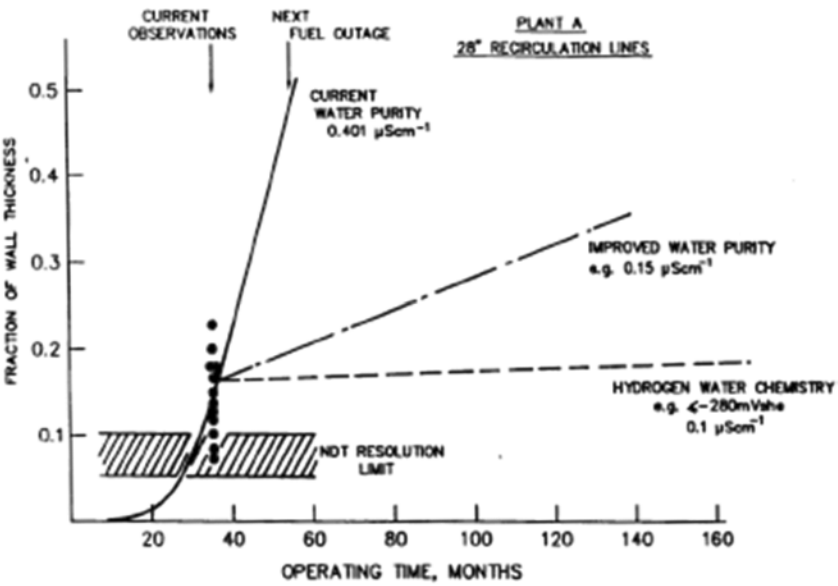


FIG. 6.9. Predicted response of defective piping for defined changes in water chemistry in BWR plant. The data points are the observed crack depth for type 304 stainless steel 28" recirculation piping in a given reactor.

effectiveness over two cycles and fuel surveillance over 3 cycles. The results of this demonstration were documented in a series of BWRVIP reports. After the successful demonstration at Duane Arnold, the NMCA process was applied at many BWRs in the USA and also a few in Europe and Japan (about 30 in total).

During operation, however, there is a depletion of noble metal from the reactor internal and piping surfaces. Consequently, every 3 to 5 years a re-application of NMCA is necessary. The proof of effectiveness of NMCA is to date based on laboratory results using crack growth specimens combined with in-reactor corrosion potential measurements and noble metal deposition measurements. The demonstration of effectiveness to mitigate crack propagation using in-reactor UT crack size measurements is ongoing. The BWRVIP is also evaluating data from core shroud re-inspections to assess the effectiveness of NMCA and HWC in mitigating cracking.

However, there are observations of continued crack growth rate in a plant which applied NMCA. Significant crack growth was found one year after the NMCA application and during subsequent inspections the crack was found to show continued crack growth in spite of the performed NMCA treatment. This has been explained as being caused by oxidants penetrating behind the NMCA treated crack surfaces, especially during reactor startup or during interruptions of the hydrogen dosage. Continued crack growth after NMCA treatment has been called ‘crack flanking’. To overcome this problem, the ‘On-Line NMCA’, OLNC, approach was developed. During such a treatment, a continuous noble metal addition is performed during an operating period of a few weeks after the outage. Table 6.2 outlines the major differences between NMCA and OLNC. The advantage of OLNC is said to be that the cracks are more open during plant operation, thus allowing enhanced penetration of the Pt catalyst inside the cracks compared to previous NMCA treatments.

Noble metal coating and cladding

Noble metals can also be deposited on the reactor internal surfaces by utilizing welding procedures or by coating technologies. Noble metal cladding (NMCL) involves weld cladding with filler wire containing noble metal. Noble metal coating (NMC) can be applied under water using the plasma spray coating process. This application is particularly suitable for components such as the core shroud. NMC has been applied to core shroud welds of a Japanese BWR. An underwater welding process has also been developed to apply noble metal cladding; results from preliminary test programmes show high quality and uniform application.

Titanium-oxide injection

Titanium-oxide injection has been developed as new environmental mitigation technology for BWRs. It utilizes a photoelectrical effect of irradiated TiO_2 to reduce ECP in the reactor water. Micro particles of TiO_2 are injected into the reactor water to form a deposit on the surface of reactor internals and recirculation piping. Cherenkov radiation in the reactor core region is the light source for photo-excitation of TiO_2. Thus, TiO_2 injection is thought to be an effective mitigation technique for reactor internals and vessel penetrations without any hydrogen addition. The first plant application is planned in Japan in the near future [6.21, 6.22].

6.1.8. Mechanical repair

Core shroud tie rods

Tie rod assembles are installed outside the core shroud and comprise a bar shaped tie rod and radial restraint. The tie rod connects the top of the core shroud to the shroud support plate. The radial restraint provides lateral support of the core shroud for lateral loads which are transmitted from the core shroud to the reactor pressure vessel.

TABLE 6.2. CRITICAL DIFFERENCES BETWEEN NMCA AND OLNC

NMCA	OLNC
During hot shutdown	During reactor operation
Application temperature >135°C	Application temperature >280°C
Noble metals, Pt and Rh	Noble metal, Pt only
Sodium and nitrate ions 100s to 1000s ppb in reactor water during application	Sodium ions <20 ppb, and no nitrate ions added during the application
Reactor water cleanup in operation	Reactor water cleanup in operation
Reactor water cleaned up prior to plant startup	Reactor water cleaned up during plant operation
Core flow — NA	Core flow >85% (>75% for MELA plants)
Hydrogen injection off	Hydrogen injection on
Zinc injection off	Zinc injection on
Application period — 48 hours	Application period — 1–3 weeks

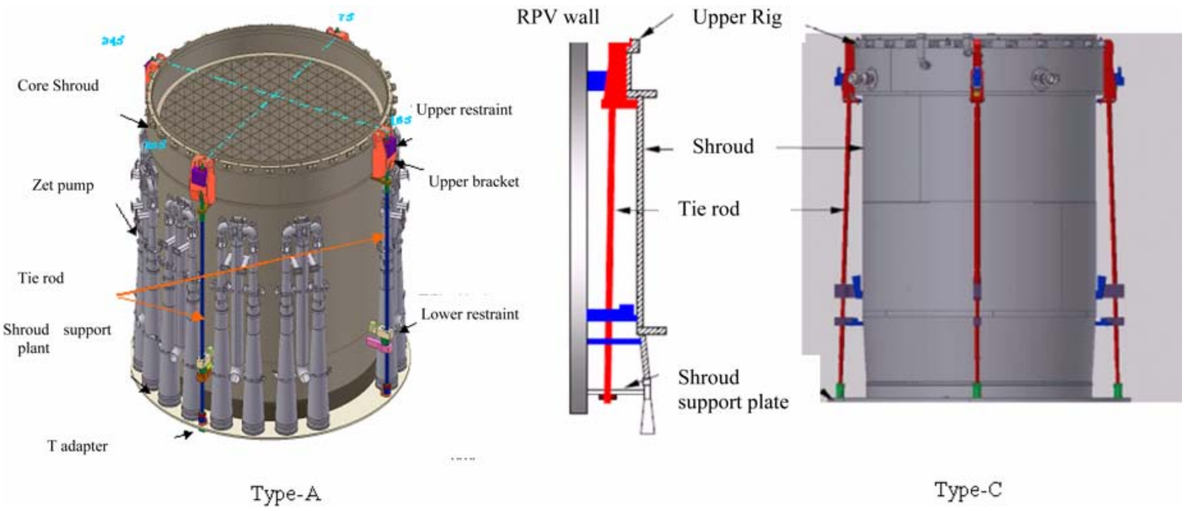


FIG. 6.10. Core shroud tie rods.

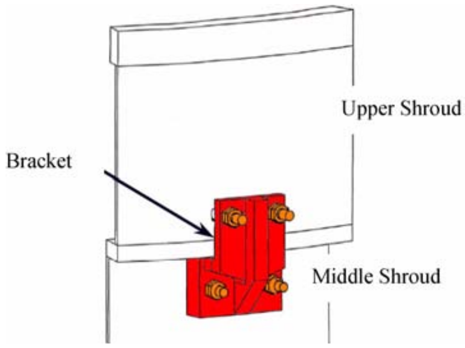


FIG. 6.11. Brackets repair technology.

The functions of tie rod assemblies are as follows; (a) Prevention of upward movement of the core shroud caused by differential pressure between the core shroud top and bottom during normal operation, (b) Prevention of overturning and support for lateral stresses on the core shroud caused by seismic loading. The tie rod assemblies can maintain structural integrity and function of the core shroud, regardless of any decrease in structural strength of the shroud caused by cracked circumferential welds. As a result, inspection of the circumferential weld of core shroud is not required.

Brackets

Brackets repair technology will be applied to circumferential welds of the core shroud. The upper side and lower side shell of the core shroud is connected by bolting through the bracket. Even if a through-wall 360° crack exists at the circumferential welds, the structural integrity of core shroud is maintained when the bracket is installed across the cracking area.

Roll repair

In some BWR plants, control rod drive (CRD) and in-core instrument penetrations have leaked due to SCC cracking of CRD penetration stub tubes or of the in-core monitor housing (ICMH) to vessel welds. In these plants, a roll/expansion repair (‘roll repair’) is used to eliminate leakage from the penetration.

The roll repair process for CRD and in-core penetrations is done from underneath the reactor vessel lower head using a rolling tool. Remote inspection, measuring, and rolling equipment are inserted in sequence into the penetration piping at the flange connections in order to accomplish the repair process. Two different conditions have to be fulfilled for a successful roll repair [6.23]:

— The CRD housing should be expanded by the tool such that the housing to vessel bore interface contact pressure under operating conditions is about three-to-five times the system pressure inside the reactor vessel. This is achieved by plastically expanding the CRD housing using the rolling tool;
— A continuous contact between the housing and the vessel is necessary. To achieve this objective, a target wall thinning of about four to six per cent is required.

As described in BWRVIP-146 [6.24], roll repairs maintain all functional requirements of the penetrations and cause no material damage to the housing or vessel. In addition, it requires less outage time, less development cost and less radiation exposure than alternative repair methods such as welding. For BWR CRDs, the roll repair method is approved by the NRC and the ASME Code Case N-730 provides specific criteria for its application [6.25].

6.2. PRESSURIZED WATER REACTOR TECHNIQUES

6.2.1. Material changes

SG: from alloy 600/carbon steel (CS) to alloy 690, 4XX ferritic SS (TD981)

Units with original steam generators incorporating the Inconel 600 mill annealed alloy tubing and drilled carbon steel support plates are almost certain to face environmentally induced degradation problems. More corrosion resistant materials are now being used. The US industry’s consensus on the best steam generator tube material is thermally treated alloy 690, which is also being used in France, Japan and elsewhere. Alloy 800M tubing is being used in Belgium, Canada, Germany and Spain. Alloy 690 thermally treated tube has proven to be at least 9 to 10 times more resistant to secondary loop cracking than alloy 600 mill annealed while alloy 800M has proved very reliable in service. The tube support structures in new steam generators are now being fabricated with 12% chromium ferritic stainless steels such as types 409, 410 or 405 with broached quatrefoil tube holes to preclude impurity hideout and denting [6.26].

Sleeves

Sleeves for tube repair are normally made of a material having better corrosion resistance than the original tube material, such as thermally treated alloy 600 or alloy 690 [6.27].

Pressurizer instrumentation nozzles alloy 600 to type 316SS

Cracking of alloy 600 instrumentation nozzles of pressurizers was discovered in France in 1989. It rapidly became clear that the cracking was a generic problem for 14 French 1300 MW units equipped with alloy 600 instrumentation nozzles [6.28]. Due to the high temperature of the environment and the high level of residual stresses from manufacturing (roll-expansion of the nozzles into the wall of the pressurizer and welding to an overlay on the pressurizer cladding), both water and steam phase nozzles and their welds were replaced as rapidly as possible. The choice of austenitic stainless steel for replacement was made regarding the good field experience with stainless steel nozzles and weldments on pressurizers in French 900 MW units and elsewhere. All the repair processes were developed and qualified and personnel trained before the field operations. Several non-destructive tests were used to verify the soundness of the repair welds (dye penetrant testing, radiography, helium leak test).

Alloy X-750 AH to improved version alloy X-750

Stress corrosion cracking of first generation alloy X-750 guide tube pins were encountered in France in 1982. SCC was detected from 1987 on second generation pins. Investigations including surface examinations, NDE, fractographic and metallographic examinations showed clearly that the cracking occurring in several different locations (collar, leaves, and less frequently, at the threads) was due to PWSCC. Based on periodic ISI results, different campaigns of replacement have been undertaken and pins have been continuously improved up to the latest New Generation in 1999.

All the inspected pins from this last improved pin generation have been shown to be sound. This is the result of an extensive analysis of the field experience and a testing programme showing that the surface finish in the highly stressed areas is of primary importance. In addition, some changes to the assembly method were made in order to decrease the stress level in the most sensitive areas. A complete review of conception was also carried out. Improvements relied on alloy X-750 heat treatment and boron content adjustments, and manufacturing and conception modifications (such as a reduction in the length of the thread, slight increases in diameter of the shank, and adding water circulation holes in the retaining nut) [6.29].

6.2.2. Isolation techniques

Alloy 690 SG sleeving

A short tube or sleeve is inserted into the base of steam generator tubes to bridge any degraded area. The sleeve is then welded inside the tube to isolate the degraded section of the tube and effectively seals the leak from the secondary-loop water. They are designed to take the full load that the original tubing was designed for, i.e. the sleeve replaces the tube as a structural element from its top joint to its lower joint. Most of the currently available sleeving techniques are designed to cover the inside surfaces of PWR steam generator tubes in the region from the bottom of the tubesheet to slightly above the sludge piles on the tubesheet.

Another location for sleeving is at the tube support plate intersections where sleeves are used to repair IGA/IGSCC occurring at the tube support plate to tube crevice. Sleeves at tube support plates have only been used on a very limited basis in the USA but they are now being extensively used in some Japanese plants as well as others. Sleeves can also be installed into previously plugged steam generator tubes in order to restore plant capacity if the plugs can be successfully removed.

Sleeving is used only for steam generator tubes with cracks penetrating no more than 40 per cent of the tube wall; more serious cracking requires the tube to be plugged. The anticipated performance of a sleeve (lifetime) depends on the nature of the sleeve repair (mechanical without stress relief versus a fusion weld with or without thermal stress relief, etc.) the location of the degradation, whether the degradation is PWSCC or ODSCC, the resistance of the parent tube to stress corrosion, the extent of restraint at nearby tube support plates, operating temperature, etc. Consequently, the lifetime of a sleeved tube could be as little as two fuel cycles when the parent tube has low resistance to stress corrosion cracking, the joints are mechanical and not stress relieved, and the steam generator is operating at high inlet temperatures (e.g. 330°C). On the other hand, the lifetime of a sleeved tube could be as long as 20 years for thermally stress relieved laser welded sleeves in a steam generator with a low inlet temperature (e.g. 315°C).

Plating (Ni)

Ni plating technique to repair tubes affected by PWSCC has been studied and applied in a few plants. However, in service inspection by eddy currents inside the SG tubes after their repair is no longer possible and is considered to be a limitation for application of this technique.

Cold spraying

One method that is being considered to protect PWR components made from alloys 600/182/82 is to cover these components with a corrosion resistant coating that is deposited on the metal surface at relatively low temperatures [6.30]. A fine metal powder is deposited onto the surface using a supersonic heated gas stream. A

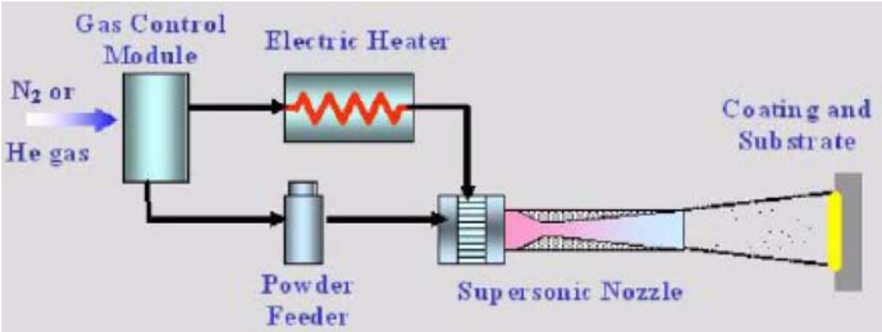


FIG. 6.12. Sketch of the cold spray corrosion resistant coating process.

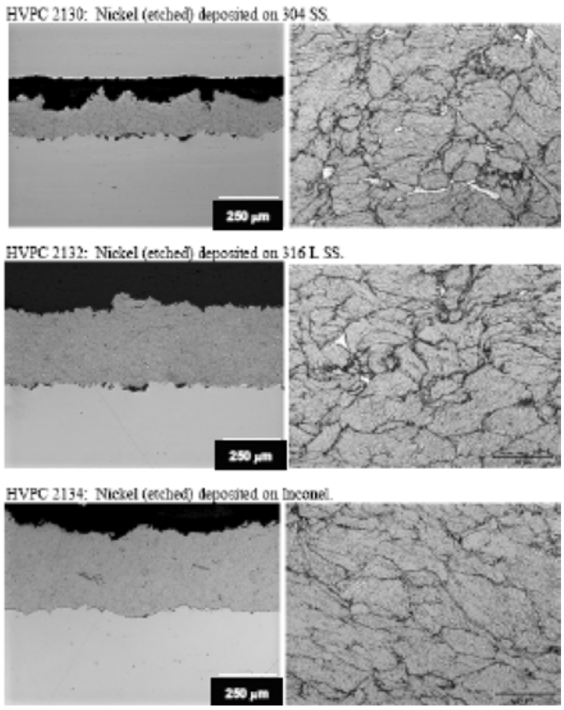


FIG. 6.13. Typical appearance of cold spray coatings in cross-section and on the surface. Both nickel and stainless steels coatings have been applied to type 304 SS, type 316 SS and alloy 600 substrates.

sketch of the process is shown in FIG. 6.12. Individual, micron range sized particles are welded kinetically to the substrate to buildup a corrosion resistant layer without introducing any of the thermal stresses associated with welding. The process produces a diffusion bond with the substrate, and there is typically no dilution layer that can reduce corrosion resistance. Other advantages of this technique include the possibility to create high density coatings with wrought like metal microstructures together with compressive stresses that are typically introduced by the cold spray process. FIG. 6.13 shows the microstructural appearance of the coating.

Laboratory stress corrosion testing has shown that these layers can isolate cracked surfaces from the corrosion environment and prevent further crack propagation. The coating has proved to be effective in preventing further crack initiation. Demonstration testing is continuing prior to the first implementation in the field.

6.2.3. Weld material changes

Cladding with alloy 690 weld metals like alloy 52 is an effective mitigation technique for operating plants that covers existing alloy 600/182/132 components susceptible to PWSCC. The process eliminates exposure of alloy 600/182/132 to PWR primary water. A groove is first machined in the surface in order to maintain the same

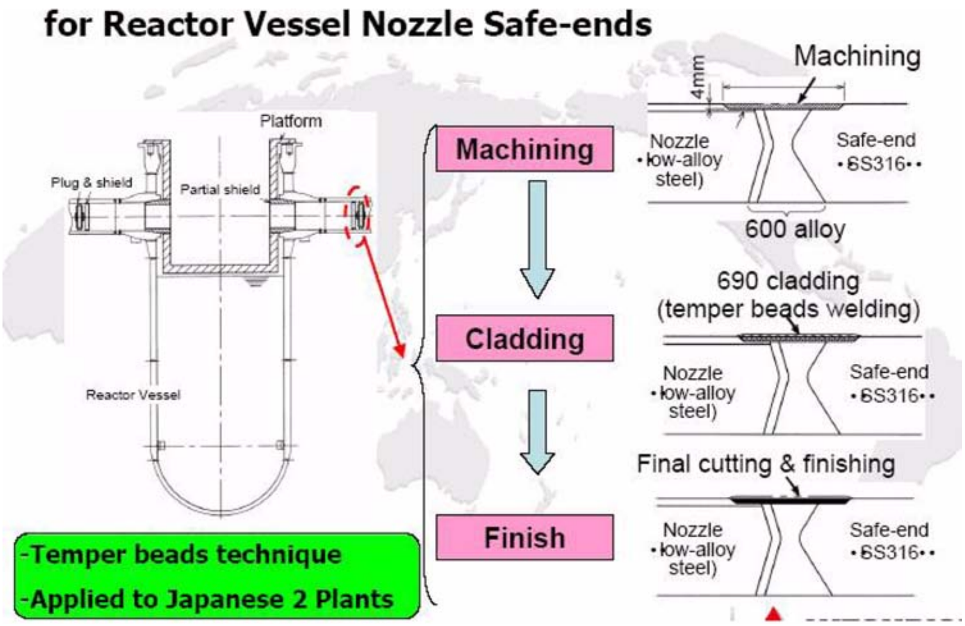


FIG. 6.14. 3-layer temper bead weld cladding technology.

dimensions of the flow passage after cladding and to facilitate inspection capability after repair. Alloy 690 weld metals such as alloy 52 are introduced into the pre-machined groove using temper bead welding in order to avoid any need for PWHT (post weld heat treatment). The surface is then machined to its final surface finish specification. This technique using automatic 3-layer temper bead welding without PWHT has been already applied to the dissimilar metal welds of reactor vessel nozzles in two Japanese PWRs.

6.2.4. Design changes

Core internals — Upflow conversion

During the 1980s, baffle jetting of peripheral fuel rods occurred in French CP0 reactors. In 1988, the first ultrasonic inspection of baffle bolts was carried out and some cracked bolts were identified. At that time, EDF decided to convert all the CP0 reactors from downflow to upflow in the cavity between the baffle plates and the core barrel. For the five units under consideration, the conversion was completed between 1989 and 1993. The main objective was to reduce the differential pressure between the core barrel/baffle plate inter-space and the core itself. The pressure differential between the cavity and the core during normal operation are reduced by a factor of 10 and, consequently, the static pressure loadings on the baffle plates are lower [6.31].

TABLE 6.3. EDFs PLANT UNITS — DIFFERENCES BETWEEN CP0 UNITS AND OTHER UNITS

Series name	Number of reactors	Nuclear power	First startup	Flow circulation	Cooling of bolts
CP0	6	900 MW	1977	Downflow initially (upflow since 90s)	No
CPY	28	900 MW	1978	Upflow	Yes
P4, P’4	20	1300 MW	1984		
N4	4	1450 MW	1997		

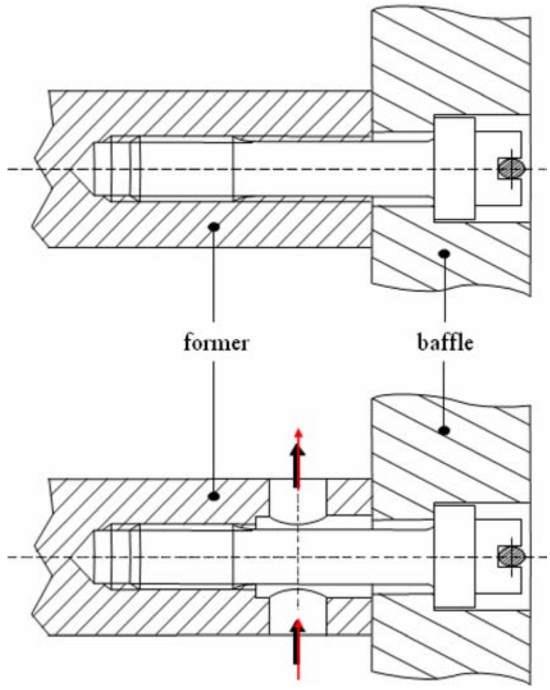


FIG. 6.15. Holes for water flow through formers to cool baffle bolts.

Holes for water cooling of baffle bolts through holes in the former

Reducing the temperature of baffle former bolt was achieved by drilling holes for water passage through the former close to the baffles at the level of the baffle bolt shank. Providing a cooling flow in this zone limits the effects of IASCC. Holes also eliminate any possibility of environment modification in the crevice between the baffle bolt shank and the former plate by allowing local water circulation.

Improving the shape of the shank to head stress concentration of baffle bolts

A parabolic shaped connection between the baffle bolt shank and bolt head was adopted rather than the earlier circular design in order to reduce the stress concentration factor.

6.2.5. Weld overlays

Since 2004, the nuclear industry in the USA has been committed to self-management of materials issues through the Nuclear Energy Institute (NEI) utilizing the Materials Executive Oversight Group (MEOG) and Materials Technical Advisory Group (MTAG). One of the outcomes of this process was development of Butt Weld Inspection and Examination Guidelines described in report MRP-139. This guideline commits plants to mandatory inspections of butt welds. Processes are underway to insert these inspection requirements into the ASME code and code cases are being currently developed.

Most utilities in the USA are choosing to plan mitigation because inspections of alloy 182/82 welds of PWRs continue to identify indications. Utilities in the USA are currently focused on butt welds for inspection and mitigation, particularly butt welds located in the pressurizer as a first priority because of the high service temperature. Many of the pressurizer locations are difficult to inspect because of the geometry of the nozzle region and the large grained microstructure present in the welds. The use of a weld overlay process is a suitable repair technique since the location can be made more amenable to NDE inspection techniques and high Cr weld metals can be used that are more resistant to PWSCC. FIG. 6.16. Generic weld overlay configuration for a PWR pressurizer nozzle.Figure 6.16 shows a typical geometry of a weld overlay on a pressurizer nozzle. The overlay is often designed to be structurally sufficient to maintain a safe condition even if deep cracks are present in the original weld.

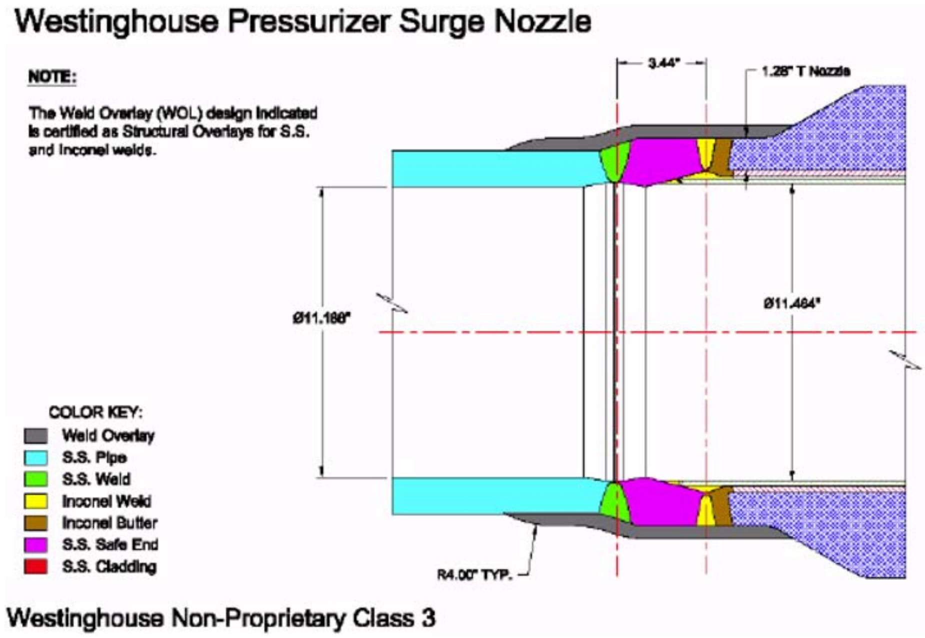


FIG. 6.16. Generic weld overlay configuration for a PWR pressurizer nozzle.

The weld overlays are typically designed using ASME Code Case N-504-2. The process usually assumes that a 360º, through-wall defect is present. The structural weld overlay replaces the pipe function and the overlay thickness is sized to serve as a structural replacement. The overlay filler material is either alloy 52 or alloy 52M and contains approximately 28–31% Cr.

6.2.6. Stress improvements

Mechanical Stress Improvement Process (MSIP™)Inelastic finite element studies were performed to simulate the application of the Mechanical Stress Improvement Process (MSIP™) using representative geometries and materials properties. Following these studies, selective geometries were fabricated for tooling qualification and process definition. The analyses and testing confirmed that MSIP generates compressive stresses on the inner weld surface. This technique has been used to remediate PWSCC in pressurizer welds [6.32].

Use of the MSIP™ has been an effective method of PWSCC remediation of sensitive locations in PWRs. Among the locations susceptible to PWSCC, the pressurizer to safe-end dissimilar metal welds using alloys 82/182 rank near the top of the priority list. MSIP™ is a stress related mitigation method that produces a favourable stress pattern by removing 'as-welded' tensile residual stresses and generating compressive residual stresses at the ID regions of the pipe. MSIP™ is accepted by the US NRC. It has also been successfully used for mitigating stress corrosion cracking in BWRs including over 1300 welds at over 30 plants since 1986. A sketch showing the MSIP™ concept is shown in FIG. 6.17.

Although the weld filler materials, alloys 82/182, are the same in both BWRs and PWRs, the wall thicknesses are significantly greater in PWRs and the axial lengths of the safe-ends are shorter. A programme was sponsored by the Westinghouse Owners Group to verify that the MSIP™ technique could be applied for mitigating PWSCC in pressurizer surge, safety relief and spray nozzles. Inelastic finite element studies were performed to simulate the application of MSIP™ using representative geometries and materials properties. A sketch of a typical application for pressurizer surge and spray nozzles is shown in FIG. 6.18. Following these studies, selective geometries were fabricated for tooling qualification and process definition. The analyses and testing confirmed that MSIP generates compressive stresses on the inner weld surface.

MSIP is often considered a cost effective mitigation process when compared to weld repair and weld replacement techniques. Since it is a one-time application process that can be completed within 1–2 outage shifts, this process reduces critical path time during a reactor outage and personnel radiation exposure. It has been shown to prevent further crack initiation and arrests existing cracks within the compressive zone.

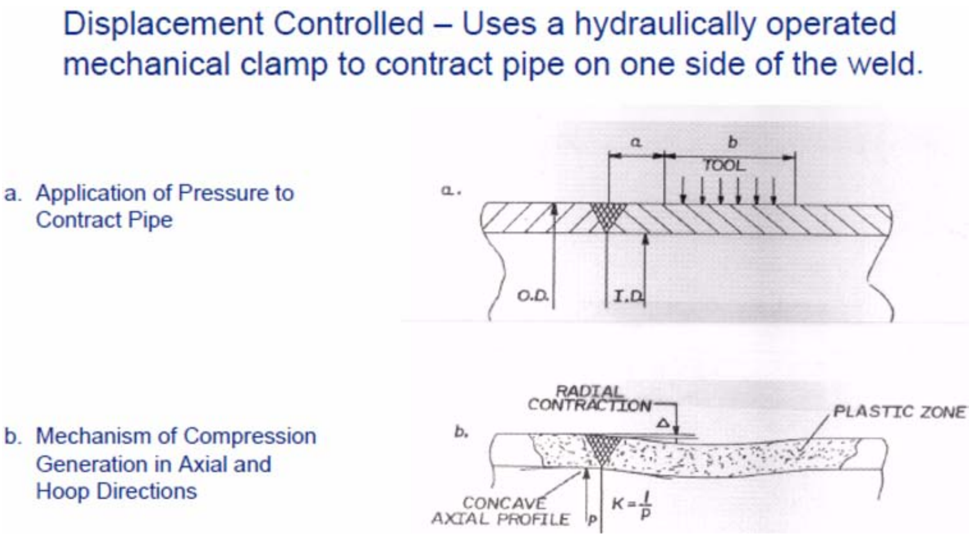


FIG. 6.17. Basic concept of MSIP: a clamp is applied on one side of a pipe weld to plastically deform the weld and in the process generates compressive stresses in both the axial and hoop directions.

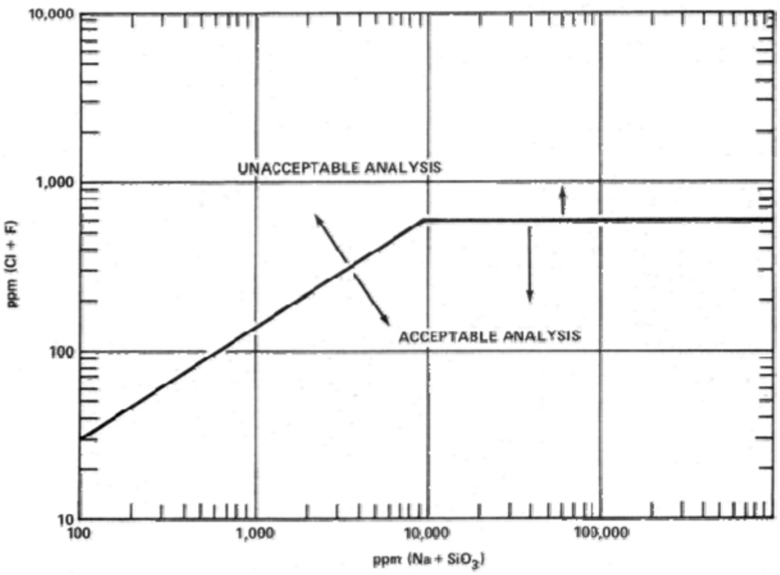


FIG. 6.18. Sketch showing position of MSIP clamps on PWR pressurizer spray and safety nozzle dissimilar metal welds.

Cavitation/water jet peening

Water jet peening (WJP) can be used to generate compressive stresses at water-wetted surfaces in the vicinity of PWSCC susceptible weld metals; like alloy 600/182/132. This mitigation technique makes use of the energy of collapsing bubbles generated by high pressure jet water near the targeted surface, as shown in FIG. 6.19.

WJP can improve the stress condition at both the inner and outer surfaces including weld metals, such as BMI nozzles (inner surface, outer surface and J-weld) and inlet/outlet nozzles of reactor vessels. This mitigation technique has already been applied to 14 Japanese PWRs since 2001.

In addition, there are several other types of peening such as shot peening (SP), ultrasonic shot peening (USP) and LP. Shot peening has been applied to steam generator heat transfer tubes, BMI nozzles, vessel head penetrations, and pressurizer nozzles of Japanese PWRs. USP has been applied to vessel head penetration J-welds and steam generator nozzles in Japanese PWRs.

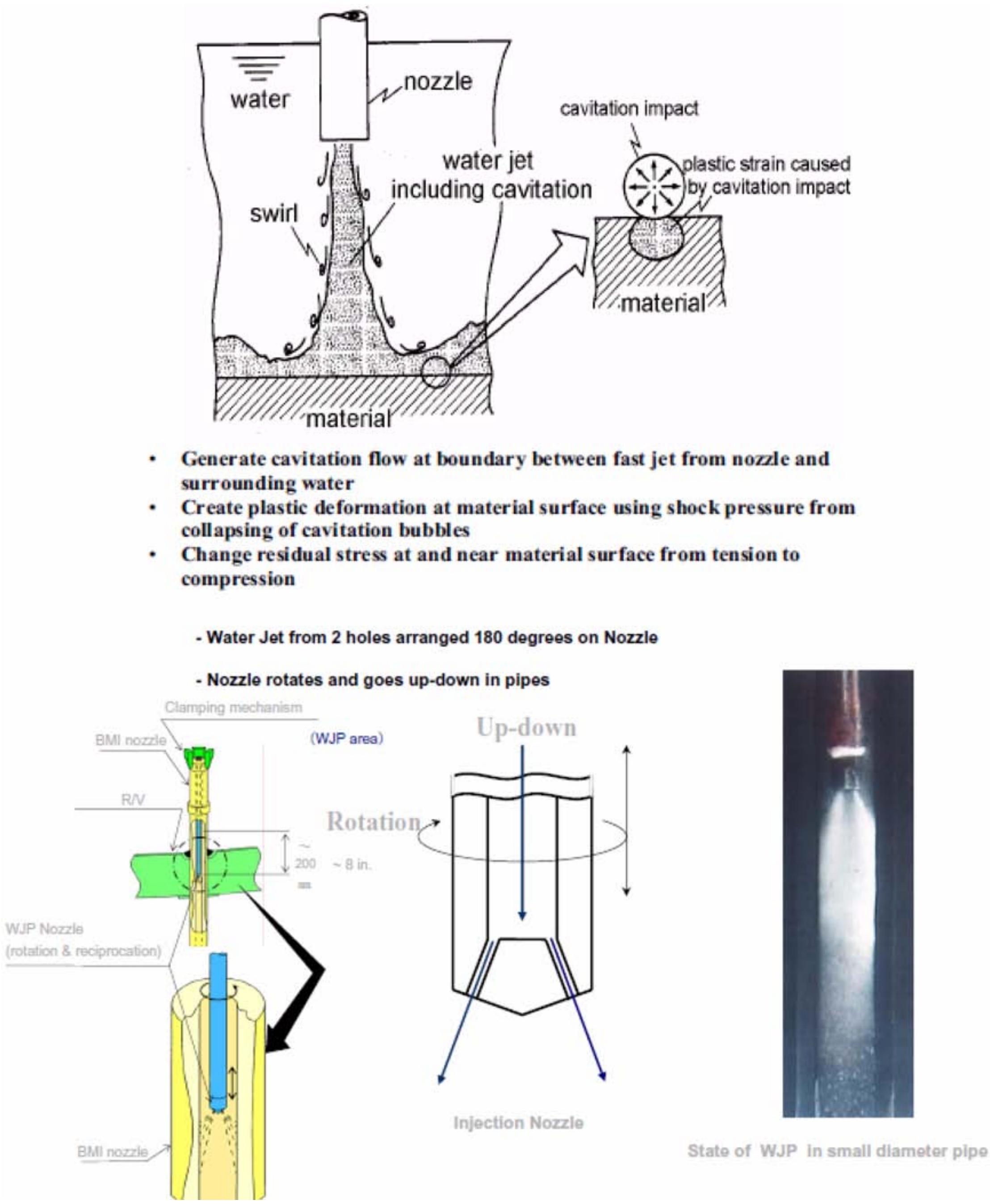


FIG. 6.19. Water jet peening process at the inner surface of a BMI nozzle.

Laser-stress improvement process)

Laser-stress improvement process (L-SIP; i.e. outer surface irradiated laser stress improvement process) introduces compressive stress on the inner surface of a pipe by irradiating the outer surface with a laser rotating around the pipe. Laser irradiation generates a temperature difference between inside and outside of pipe that changes the residual stress on inside surface into compressive after cooling, as shown in FIG. 6.20. High density laser beams can heat up the outside of a pipe very rapidly to more than 500°C, so that water cooling for inside can be omitted, especially for smaller pipes. During laser irradiation, the optical heads rotate around the pipe so that the complete outside surface of the pipe is not irradiated simultaneously. This means that the irradiated and heated portion is confined to a limited area and the required power for L-SIP is reduced. Moreover, concerning the energy transmission line, the thinness and flexibility of fibre cable makes cable installation in the field easier than with large capacity electric cables. L-SIP has been applied to pressurizer nozzles in a Japanese PWR.

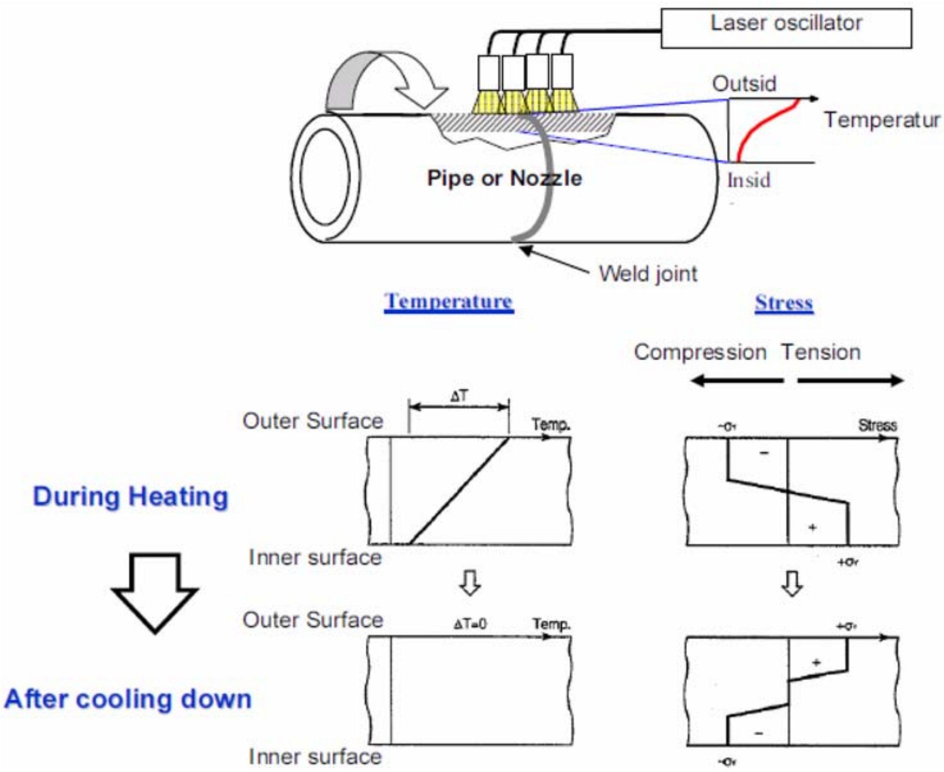


FIG. 6.20. Principle of L-SIP.

6.2.7. Environment improvement

Improved secondary side chemistry

Most secondary side intergranular attack/intergranular stress corrosion cracking (IGA/IGSCC) of alloy 600 tubing, especially in the mill annealed condition, has been attributed to either strongly caustic or strongly acidic solutions that build up in superheated crevices on the secondary side of the steam generator tubes by secondary water impurity 'hideout'. Consequently, environmental control strategies have concentrated on limiting the ingress of the water impurities that concentrate in superheated crevices, notably sodium, chloride and sulphate; see for example, the current secondary water guidelines issued by EPRI that set very low limits on such impurity concentrations in secondary side feedwater as well as restrictions on the length of periods of operation that can be allowed during impurity transients.

More recently, increasing attention has focused on lead (Pb) as a potentially more important cause of secondary side IGA/IGSCC than had been thought hitherto. Lead (Pb) is known as a very aggressive chemical specie for SCC of Ni base alloys used in nuclear steam generators [6.33]. Sludge piles in contact with the cracked areas often contain a high concentration of lead. Chemical analyses have shown that lead also concentrates in superheated secondary side crevices in SG even when the source term in the feedwater is extremely low in the tens of ppt range [6.34]. However, some or even most of the lead may be tied up in insoluble mineral species and probably unavailable in solution to cause cracking. Results from laboratory studies indicate that steam generator (SG) tubing materials are very susceptible to Pb-induced SCC (PbSCC). Alloy 690 TT can also be attacked by the lead in high caustic solution and explains why much attention is currently directed to understanding the role of lead in secondary side SCC.

Sulphur in the form of sulphate also attacks alloy 600 SG tubes in a mixed mode of IASCC and IGA (intergranular attack). Reduced sulphur species are also known to easily attack Cr depleted grain boundary (i.e. sensitized material) in oxidizing conditions. Thus, improved secondary water chemistry guidance is required concerning the amounts of lead and sulphur species in the secondary system in order to suppress the occurrence of SCC of Ni alloys [6.35].

Inhibitor application

In order to suppress secondary side SCC of the alloy 600 SG tubes, some inhibitors such as boric acid, cerium oxide and titanium oxide had been studied since the mid 1990s. TiO_2 had been applied in some operating plants (Ringhals-3, Oconee, Kori-1, etc.). Although the role of cracking retardation in laboratory tests was proven, the effectiveness of the inhibitor was not well demonstrated in the operating plants.

Zinc addition

The addition of soluble zinc additives to PWR primary coolant leads to incorporation of zinc in the nickel substituted ferrite films and the inner chromite layers that form on nickel based alloys exposed to primary water. The main goals of using zinc injection are (a) reduction in plant radiation fields and (b) mitigation of PWSCC.

Westinghouse studied zinc addition for PWSCC mitigation since 1990. Initial studies indicated that zinc injection into primary water was successful in delaying PWSCC initiation and that the effect is related to the zinc injection concentration. Subsequent analysis has indicated that the effectiveness of zinc injection as a PWSCC inhibition agent is related to the integrated quantity of zinc that ends up in the corrosion film of PWSCC susceptible nickel based alloys and welds. These laboratory studies resulted in the Farley 2 PWR being the lead zinc injection plant that started to inject zinc in 1994. As of 2008, there are over 24 of 69 PWRs in the USA injecting zinc and the number of additional plants that are injecting zinc has been increasing as PWSCC becomes a more significant ageing degradation mechanism.

The amount of zinc in the films is related to the average zinc concentration present in the coolant and the time present. As such, the integrated zinc concentration, as defined by the ‘ppb-mo’ integrated exposure, is thought to be a good indication of the effectiveness of zinc for infiltrating the corrosion product film. Based on initial SCC experiments, it appeared that approximately 300 ppb-mo of zinc exposure is necessary to provide reasonable mitigation of PWSCC. This 300 ppb-mo integrated exposure is similar to the exposure needed to significantly reduce corrosion product release from alloy 600, as shown in FIG. 6.21. This makes sense since both the corrosion product release and PWSCC remediation depend on developing a more protective corrosion product film.

The effectiveness of zinc addition on mitigating PWSCC initiation was demonstrated by the performance of alloy 600 material installed in PWR reactor heads. As PWSCC initiation was detected in PWR reactor vessel heads in the USA, a specific alloy 600 heat dependence was indicated and one of the most susceptible heats with respect to PWSCC initiation was alloy 600 heat M3935. This heat had a less than optimum annealing heat treatment and was used in the CRDM penetrations of five PWR plants, including Farley 2, Davis–Besse, Oconee 3, Beaver Valley 1 and ANO 1. PWSCC initiation was observed in 4 of 5 PWR heads as indicated in Table 6.4.

The Farley 2 plant injected zinc since 1994 and had an integrated zinc injection of 1800 ppb-mo when the reactor vessel head was retired in 2004. No cracking was observed at Farley 2 although PWSCC initiation was detected in each of the other four PWRs where this heat was installed. Segments of this heat M3935 were removed from both the Farley 2 and Davis–Besse PWRs and PWSCC initiation tests are currently being performed. The testing to date suggests that the microstructure and cracking susceptibility of alloy 600 removed from Farley 2 is as susceptible to PWSCC as the material removed from Davis–Besse and that the good PWSCC initiation behaviour at Farley 2 is related to the presence of zinc in the corrosion product films.

Optimized hydrogen

The influence of hydrogen partial pressure and optimization of the hydrogen concentration has been discussed for many years. Based on the crack propagation studies presented in Section 2, there are suggestions to increase the hydrogen concentration in US PWRs. FIG. 6.22 summarizes the benefits with respect to crack growth of an increase of the hydrogen content at 325°C for both alloy 600 and alloy 182.

The maximum crack growth rate obtained at corrosion potentials equivalent to the Ni/NiO stability line shifts to lower hydrogen concentrations when the temperature decreases, as shown in FIG. 6.23. Using such data, there is now a trend in the USA and other countries, following their water chemistry practices, to increase the hydrogen content in the primary system. Many utilities are already operating closer to the higher specification limit than before.

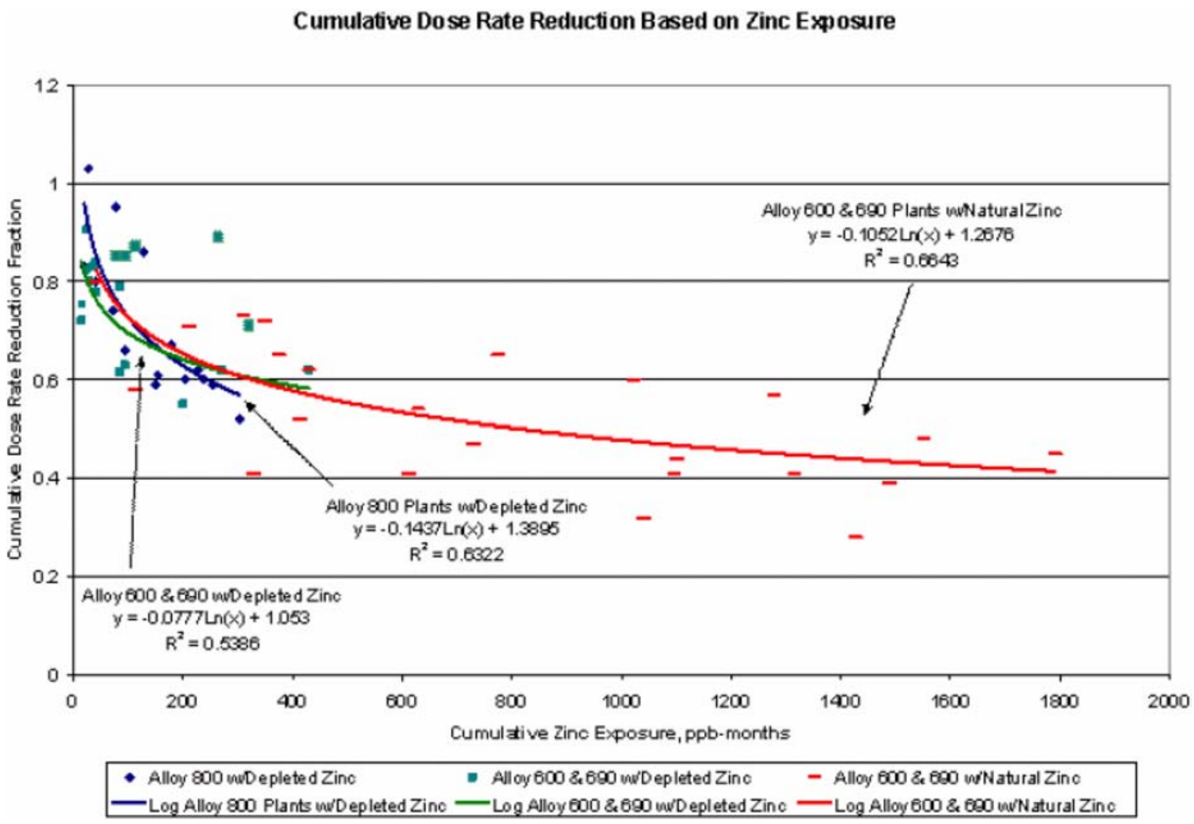


FIG. 6.21. Trends observed at US PWRs adding zinc, dose rate reduction vs. cumulative zinc exposure (ppb-mo). A significant fraction of the decrease in dose due to less corrosion product release is observed after ~300 ppb-mo of zinc exposure. Similar effects are anticipated for reduction in PWSCC [6.39].

TABLE 6.4. INDUSTRY EXPERIENCE WITH PWSCC OF ALLOY 600 HEAT M3935

Plant name	# of nozzles heat no. M3935	% in industry heat no. M3935	# inspected by UT	# required repair	% of M3935 in RV head with defect
Oconee 3	68	49%	68	14	20%
Davis–Besse	5	4%	5	4	80%
ANO 1	1	<1%	1	1	100%
Beaver Valley 1	4	3%	4	4	100%
Farley 2	61	44%	61	0	0%
Total	139		139	23	
%			100%	17%	

There are also suggestions based on the initiation data (see Section 2.3.2) to apply decreased hydrogen contents and such studies are also ongoing.

Temperature reduction

Since SCC in the systems relevant to PWRs is a thermally activated process, reduction of hot leg side temperature is an effective measure to suppress, in particular, PWSCC of Ni alloys. A lower reactor power enables the hot leg temperature to decrease and a relatively small reduction of temperature causes a large increase in time to observable cracking. There is also a favourable effect on secondary side IGA and IGSCC in that the superheat in secondary side crevices is also reduced and with it the driving force for concentration of secondary side impurities.

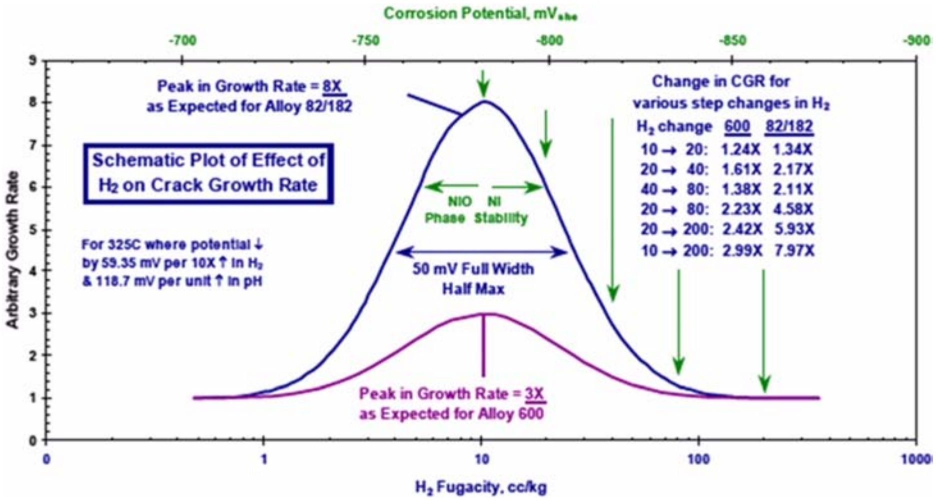


FIG. 6.22. Modelled crack growth rate for alloy 600 and alloy 182 for different hydrogen concentrations at 325°C.

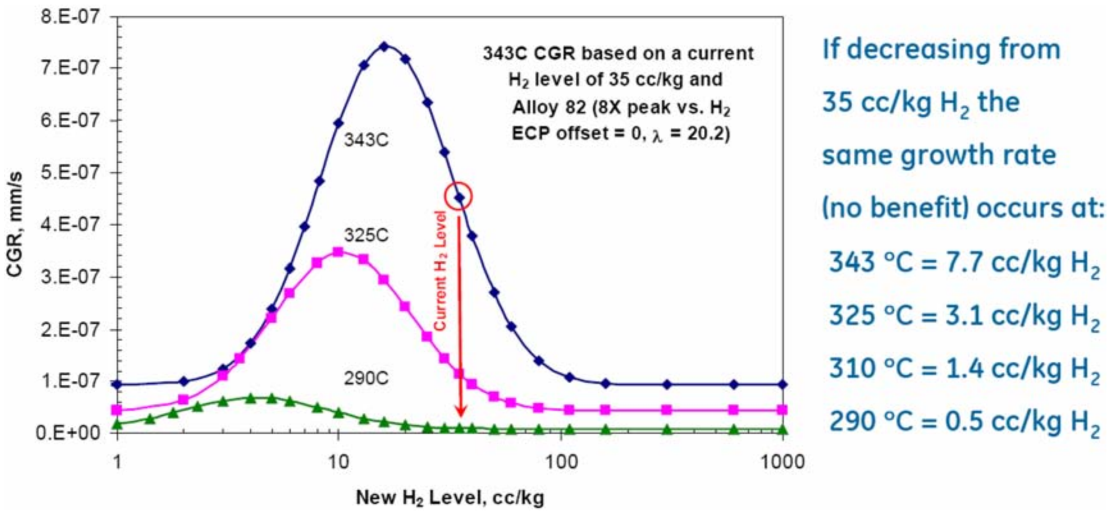


FIG. 6.23. Modelled crack growth rates for Inconel 82 showing the effect of hydrogen at different temperatures.

6.2.8. Mechanical repair

Mechanical nozzle seal assembly

The mechanical nozzle seal assembly (MNSA) is a mechanical device that provides sealing and structural support for small bore nozzle connections. It was developed starting in 1993 as an alternative to weld repair for leaks in J-groove welds at PWR instrumentation nozzles. The MNSA is installed from the outside of the pressure boundary and can be installed on a leaking nozzle. They have been installed on PWR pressurizers and hot leg nozzles without having to remove fuel from the reactor or drain the primary system. One current design is specifically designed to seal against leakage from the annulus between the PWR bottom mounted instrument (BMI) nozzles and reactor vessel caused by cracking initiated in the nozzle material and propagating through the nozzle to vessel weld. The BMI MNSA is a modified design that seals on a machined counterbore on the outside of the reactor vessel and replaces the weld and performs two functions:

— Acts as the primary pressure boundary for the primary water;
— Structurally replaces the weld to prevent the nozzle from being ejected from the reactor vessel.

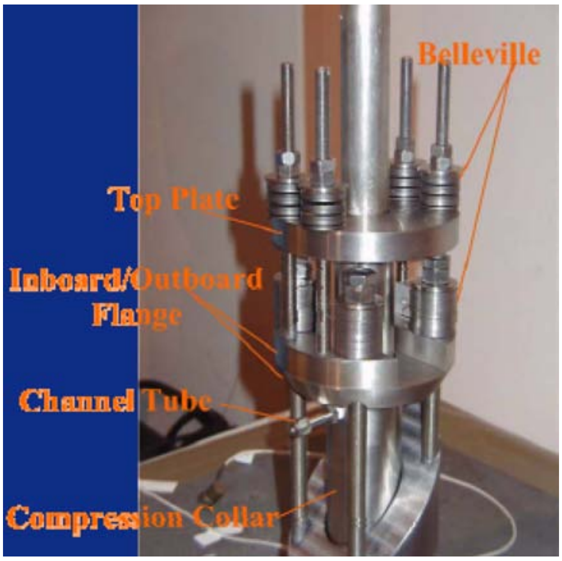


FIG. 6.24. MNSA device on a test location.

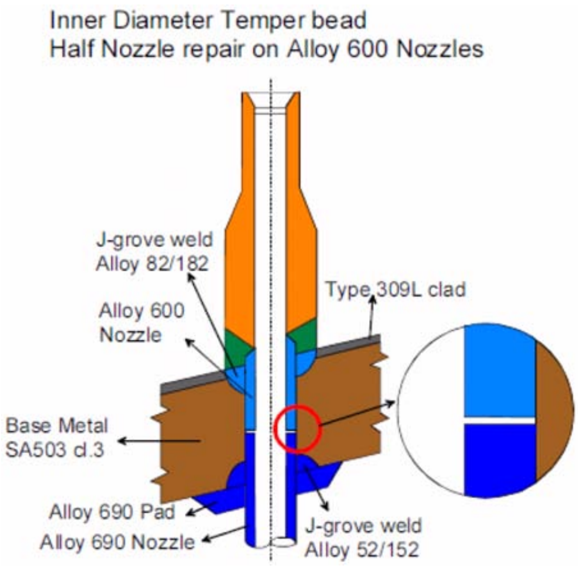


FIG. 6.25. Schematic of a crevice formed by the half nozzle repair.

An image showing a MNSA device on a test location is shown in FIG. 6.24. These mechanical nozzle seal assemblies repair techniques have been implemented on many PWR pressurizers and RCS instrument nozzles. This repair technique has been applied 43 times at 10 PWRs in the USA between 1995 and 2005.

Half nozzle repair

Since alloy 600 and its welds, alloys 132/182/82, have shown PWSCC in reactor vessel head CRDM nozzles as well as nozzles in pressurizers and SGs, modification or replacement of the degraded area can be necessary. A half nozzle repair for a small bore thick wall penetration like a steam generator drain nozzle or instrumentation nozzles with SCC affecting the weld region could be considered as one possible repair option [6.42]. It has been demonstrated that the crevice corrosion behaviour of the low alloy steel (LAS) thereby exposed to primary water in the form of an annulus near the mid-thickness of the low alloy steel is not so significant because of the very low concentration of oxygen during normal operation [6.43]. However, a question remains concerning the corrosion rate of the low alloy steel inside the crevice (see FIG. 6.25) in the high oxygen, air saturated environment during refuelling outages [6.42]. It seems likely that the protective oxide film built up at high temperature offers protection against crevice corrosion for some considerable time at low temperature during refuelling, but that remains to be demonstrated experimentally.

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7. COMPONENT REPLACEMENT INCLUDING PREVENTION METHODS FOR NEW SYSTEMS AND COMPONENTS

7.1. BOILING WATER REACTORS

7.1.1. Strategy for integrated reactor internal components replacement

SCC of the core shrouds of BWRs were first observed in many countries in the 1990s (Germany, Sweden, USA, Japan, etc.). Since the shroud is typically welded to the reactor pressure vessel and other parts of core internals, mechanical repairs using brackets or tie rod devices have been applied to reinforce the cracked shrouds. This has been done with regulatory approval. A method of replacement of a core shroud together with other internal components such as the jet pumps was developed under the joint project by the Japanese BWR owners and fabricators and applied to the following six plants:

— Fukushima daiichi Unit 3 (TEPCO, performed in 1998);
— Fukushima daiichi Unit 2 (TEPCO, performed in 1999);
— Tsuruga Unit 1 (JAPO, performed in 2000);
— Fukushima daiichi Unit 5 (TEPCO, performed in 2000);
— Shimane Unit 1 (Chugoku Electric Power Company, performed in 2001);
— Fukushima daiichi Unit 1 (TEPCO, performed in 2001).

In Europe, there is experience of BWR reactor internals replacement work at the Swedish Oskarshamn unit 1 in 1997. The replacement work in this case was different from that at current Japanese plants because the majority of the internal components of Oskarshamn unit 1 were bolted together in the structure. This differs from the welded structures that are the major type in Japanese and US BWR plants.

In the case of replacement work for welded reactor internal components, there are several basic criteria to be fulfilled in order to conduct the work efficiently.

— Staff should be able to approach and work on the reactor bottom during certain periods of the replacement work where it enables complex and difficult work near the bottom of the reactor to be performed;
— In order to get access to the bottom of the reactor, chemical decontamination should be performed and effective shields should be installed during certain periods of the replacement work;
— Removal of old reactor components should be performed remotely in order to reduce the radiation exposure;

— The new core shroud weld edge should be machined to allow narrow gap welding in order to reduce residual stress and minimize the welding time.

7.1.2. Recirculation piping replacement

US experience

In 1962, the first recorded SCC crack occurred at the GE supplied Vallecitos NPP on weld sensitized type 304 stainless steel reactor recirculation piping. This event was followed by an observation of SCC at the Dresden-1 NPP in 1965, again on the weld sensitized type 304 stainless steel reactor recirculation piping. Since then, many instances with similar characteristics occurred.

The next historically significant SCC event took place in 1974, when several through-wall cracks on the Dresden-2 NPP reactor recirculation bypass line were found. This event triggered discoveries of similar observations at Quad Cities 2 and Millstone 1, and then further cracking at 15 locations including 5 through-wall cracks at six other plants. In all, examinations were performed at 21 BWR plants. Piping of reactor recirculation and residual heat removal systems were either partly or completely replaced with type 316NG or 316L stainless steel at 12 BWR plants through 1999.

The weld overlay technique was developed and utilized in lieu of piping replacement at 13 BWR plants through 1983. It continued to be applied to over 260 locations at more than 20 plants through 1999. The effectiveness of this technique was proved by over 1000 follow-up examinations with no crack propagation being observed.

German experience

Stabilized austenitic stainless steel has been used for all German LWRs with only one exception, Gundremmingen-A, which used type 304 stainless and was permanently shutdown in 1980. As far as piping material is concerned, titanium-stabilized stainless steel similar to type 321 was chosen for all BWR plants with only one exception; i.e. the very first German BWR, which is not operating any more, where niobium stabilized type 347 stainless steel was used.

Unlike BWRs in the USA, all BWRs in Germany are equipped with reactor internal pumps with no reactor recirculation piping outside the pressure vessels. Type 347 stainless steel is used for reactor internal components such as core shrouds. Also unlike US BWRs, all BWR plants in Germany continue to operate with normal water chemistry.

The first crack in type 347 stainless steel piping was found in 1991. The cause of cracking was attributed to fabrication defects. The first crack with type 321 stainless steel was reported in 1992, when a GRS notice was issued that required all BWR plants with ~100 000 operating hours to implement an enhanced NDE programme.

In response to this instruction, over 3100 welds of type 321 stainless steel piping greater than 2 inches in diameter (DN 50) were examined at 6 BWR plants and a total of 58 welds were found to exhibit cracking until the end of 1995. The cracks were mostly confined within heat affected zone. However, there were some instances where chromium depletion along grain boundaries was not evident. All cracked type 321 stainless steel pipes were replaced with type 347 stainless steel pipes, section by section. So far, no crack initiation on replaced sections has been detected.

Japanese experience

As observed internationally, BWR plants in Japan, whose exploitation stemmed from the technology brought from the USA through General Electric, began to exhibit SCC on the type 304 stainless steel piping from the mid 1970s.

Typical corrective actions taken in the mid 1970s included replacement of sections of piping while reducing sensitization using water cooled welding, elimination of components such as reactor recirculation bypass lines, which only provided extra sites for SCC initiation with no specific functions, and application of induction heat stress improvement (IHSI) on welded joints where replacement was not needed.

These conventional corrective actions were replaced by a new approach in early 1990s when low carbon stainless steel was introduced for new plant construction. Type 316NG stainless steel was selected for replacing existing piping as a permanent remedy for the then operating plants. In conjunction with such pipe replacement, new fabrication technologies became available to produce large diameter seamless pipes, single elbow-nozzle forgings integrated into T-joints, and heat induction bent pipes composed of elbows and straight sections. These new products significantly reduced the number of field-weld joints and contributed to the reduction of person-hours and man-rem associated with pipe replacement work at BWR plant site.

Methods adopted to reduce susceptibility of low carbon stainless steel piping to SCC include the following, of which all have experience of practical application in Japan:

— Improvement of weld residual stress profile by heat sink welding (welding process applied to piping with cooling water inside), IHSI, solution heat treatment, and narrow gap welding;
— Cladding the inside wall of pipe heat affected zones;
— Moderate hydrogen water chemistry (HCW).

In 2003, a flaw evaluation guideline was made available for nuclear power plants in Japan. However, the only case where it has ever been used so far was for the reactor recirculation piping at Unit 3 of the Kashiwazaki-Kariwa NPP. Pipe replacement of degraded sections has been performed for all other cases when cracks were detected by UT during scheduled outages.

7.2. PRESSURIZED WATER REACTORS

7.2.1. Strategy for steam generator replacement

For SG replacements, tasks of special importance are design and associated calculations. These activities apply to all temporary equipment (such as SG rigging, piping devices, etc.) as well as permanent plant equipment including all related modifications (such as rerouting of piping, thermal insulation, openings in the steel liner of the containment building). Piping design covers rerouting of feedwater and auxiliary feedwater piping, adaptation of instrumentation piping, main steam lines, blow-down lines, reactor temperature detection piping, drain lines, and sampling lines. Design calculations basically cover analyses of structural, seismic and fluid dynamic data. In parallel with basic engineering, the safety of all activities leading to any modification of plant equipment or activities and introducing a specific risk, such as handling, rigging, transportation, waste handling, is evaluated for review by the licensing authorities.

A steam generator replacement project includes all of the following activities:

— Fabrication of replacement steam generators;
— Installation of new steam generators;
— Licensing of steam generator replacement components.

7.2.2. Strategy for reactor vessel internals replacement

Until 2004, there were no known reports of replacement of whole reactor vessel internal (RVI) comprising both the upper and lower internals. However, an upper internals module had been replaced at Prairie Island in 1986. The objectives of internals replacements in Japan for the 2 loop plants that were originally designed and manufactured for the early generation of PWR plants in the 1970s are as follows:

— Apply proactive and preventive countermeasures against the potential ageing degradation by irradiation assisted stress corrosion cracking (IASCC) of baffle former bolts;
— Add four more guide tubes (drive lines) to the upper internals in order to keep enough shut down margin in preparation for applying high burn-up fuels.

Since 2005, replacements of whole RVI have been completed in the following PWR plants in Japan listed in Table 7.1.

TABLE 7.1.LIST OF RVI REPLACEMENTS IN PWRs

Country	Unit	Supplier	Replacement
Japan	IKATA 1	MHI	2005
Japan	GENKAI 1	MHI	2005
Japan	IKATA 2	MHI	2006

* MHI: Mitsubishi Heavy Industry (MHI)

7.2.3. Strategy for reactor vessel head replacement

In 1991, a leakage in one control rod drive mechanism (CRDM) head penetration (fabricated from alloy 600) was discovered during 10 year hydro test at the Bugey NPP (France). This was the first experience in the world of leakage of a CRDM penetration caused by PWSCC. Longitudinal (axial) cracks propagated from the inside surface of the penetration. Leakage was also found by visual inspection in RVH in the USA and Japan. Some leakage came from cracks in J-groove welds, which then propagated to outer surfaces of the head penetrations. The cracks were caused by PWSCC, exacerbated by the high residual stress in the J-groove welds. In 1994, a decision was made to replace all vessel heads of 900 MW(e) and 1300 MW(e) in France. Subsequently, replacement of vessel heads started in the USA and Japan.

To prevent cracks in J-groove welds and head penetrations, RVH and head penetrations should be replaced as a minimum option. It may then be decided to reuse or to replace the following components during RVH replacement (RVHR):

- — Control rod drive mechanisms including pressure housings and latch mechanisms;
- — Coil stacks;
- — Control rod position indicators;
- — Upgrading packages.

The CRDM may be with or without Canopy seals. In case of reusing CRDMs, the weld between the CRDM housing and head penetration flange must then be performed at site.

7.2.4. Strategy for pressurizer replacement

As discussed earlier, several pressurizer (PZR) components fabricated from alloy 600 are subject to PWSCC. As such, several repair options exist to remedy or mitigate potential as well as known degraded locations. Additionally, PZR replacement, similar to steam generator replacement, exists as an option for resolution of this issue. Decisions on repair/replacement generally evolve around the economics and technical aspects of repair versus replacement. Four (4) CE design PZRs have been replaced to date in the USA.

In the USA, plants with less than 40 heater sleeves have consistently elected to repair or mitigate by either welded pad half-nozzle repairs or inner diameter half nozzle weld repairs. In either approach, the pressure boundary is relocated and re-established with A52/152 material. The remaining small bore instrument nozzles are also repaired using half nozzle repairs. Large bore dissimilar metal welds are normally mitigated using mechanical stress improvement or by structural weld overlay.

For those US plants with PZRs containing greater than 40 heater sleeves, replacement has been the chosen approach. This is based upon cost savings for reduced outage duration compared to repair/mitigation of many sleeves. This comparison also assumes a dedicated containment opening does not have to be constructed in the reactor containment building for the replacement. Often, replacements are timed to be concurrent with other major component replacements such as reactor vessel heads and steam generators. This provides a substantial reduction in overall outage impact and optimizes the typically specialty contract resources necessary for rigging, radiological controls, special welding processes, etc.

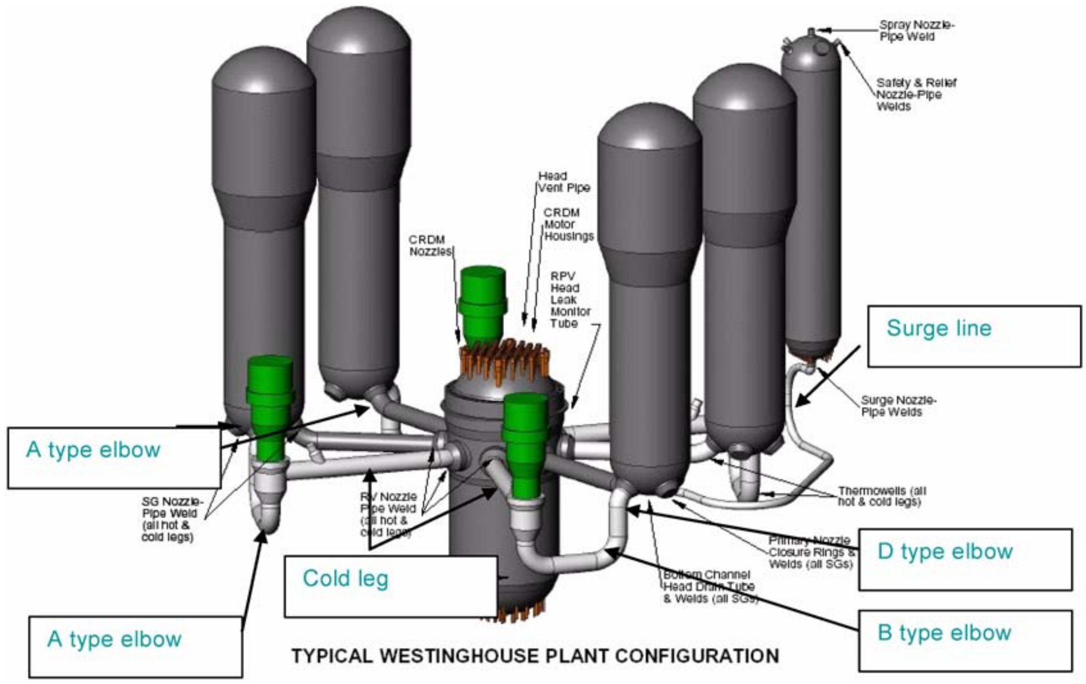

FIG. 7.1. Location of replaced RCL parts and elbows.

7.2.5. Strategy for reactor coolant loops

Replaced parts are typically elbows (attached to steam generator, for example) or parts of the cold or hot legs or surge line. The referenced elbows are shown in Fig. 7.1. The replacement of elbows and/or parts of the primary piping is due to following degradation phenomena:

— Thermal fatigue due to a large number of thermal cycles (which can be the case for mixing areas of water of significantly different temperatures), with the consequence of possible surface cracking;
— Thermal ageing of cast stainless steel, with a resulting embrittlement and loss of fracture toughness.

In the first case, replacement is an alternative to repair, particularly when anticipating an extension of the plant life. It also offers the opportunity to bring some design improvements to prevent new occurrences of thermal fatigue. In the second case, replacement is the only presently foreseen solution. New reactor coolant loops elbows can be made from forged stainless steel that shows no sensitivity to thermal ageing.

8. SUMMARY ON MANAGING STRESS CORROSION CRACKING

Stress corrosion cracking is a significant ageing degradation mechanism for major components of both PWRs and BWRs.

In PWRs, the main problem with SCC has been with alloy 600 components such as steam generator tubes, pressurizer instrument penetrations and heater sleeves, control rod drive mechanism (CRDM) nozzles, and hot leg penetrations. The phenomenon is known as PWSCC and as well as affecting wrought and mill annealed alloy 600 has also affected the compatible weld metals, alloys 132/182/82 that are found throughout the PWR primary water circuit. In addition, the secondary side of alloy 600 steam generator tubes has suffered from intergranular attack

(IGA) and IGSCC due to the accumulation of impurities by 'hideout' in superheated crevices on the secondary side. By contrast, austenitic stainless steels have experienced relatively little SCC in PWRs, although irradiation effects due to high neutron fluence on core internals components can render these materials susceptible to IASCC.

In BWRs, piping and other components made from austenitic stainless steel or (to a much lesser extent) nickel based alloys have experienced IGSCC and many cases have been reported from BWRs throughout the world. Both thermally sensitized (i.e. those materials with chromium depleted grain boundaries) and cold worked materials have been affected. Exposure to neutron irradiation (albeit an order of magnitude less intense than in PWR internals) can also exacerbate IGSCC of stainless steels in BWRs.

Another form of SCC common to both PWRs and BWRs is caused by internal or external contamination by halides; notably chloride.

As in all cases of stress corrosion cracking, remedial measures are directed at alleviating one or more of the three critical parameters (environmental chemistry, metallurgy or stress) that in combination are responsible for susceptibility to cracking.

In BWRs, the focus environmental chemistry improvement has been on better water quality in order to minimize impurity concentration in cracks and crevices and on lowering the electrochemical corrosion potential (ECP) of austenitic components exposed to the reactor coolant. The latter can be achieved by adding hydrogen to scavenge the radiolytic decomposition products of water, notably oxygen and hydrogen peroxide, and thereby lower the ECP below ~-230 mV (SHE), which is considered to be the effective threshold for IGSCC of sensitized austenitic alloys. This process is more efficient and requires less hydrogen when used in combination with trace noble metal additions. A recent new development to reduce the ECP is to exploit a photoelectron electron effect of Cherenkov radiation on thin TiO_2 deposits on the component surfaces and should have improved efficiency compared to other methods in the two phase flow region in the reactor pressure vessel.

The scope for environmental chemistry changes in PWR primary loops is, however, limited. Soluble zinc additions appear to be effective in significantly hindering PWSCC initiation in alloy 600 and its compatible weld metals. Optimization of hydrogen partial pressure is another strategy under review, but difficulties related to the different response of PWSCC initiation and propagation to hydrogen partial pressure complicate the analysis of potential benefits. A third environmental change that has been used successfully to significantly slow PWSCC initiation is reduction of the hot leg temperature; but this carries a penalty of reduced power output.

On the secondary side of PWR steam generators, mitigation has first relied on improving secondary feedwater management to minimize the formation of concentrated solutions of impurities in superheated crevices next to the tubes. Damaged tubes can be plugged or in some cases recovered by fitting internal sleeves over the affected tube length. However, steam generators are being replaced in many PWRs using much more resistant alloy 690 or alloy 800 tubing together with many design changes to reduce the risk of impurity hideout.

For both BWRs and PWRs, constant vigilance is necessary to avoid external surface contamination of austenitic stainless steel components, in particular by chlorides, which can accumulate over long periods of time from airborne aerosols, accidental wetting of inappropriate thermal insulation materials, and leaks from cooling and fire fighting systems that carry waters that are much less pure than reactor coolants.

Remedial measures based on improved metallurgy depend in BWRs on avoiding or ameliorating the fabrication conditions that lead to thermal sensitization (i.e. reducing the time at temperature where sensitization can occur) and using replacement materials with better resistance to thermal sensitization; such as low carbon L grade stainless steels, with or without nitrogen strengthening (LN or NG grades). Fabrication techniques to avoid, limit or remove cold work have also been adopted.

In PWRs, metallurgical improvements mainly involve replacing alloy 600 and its compatible weld metals alloys 132/182/82 in contact with primary water with alloy 690 and 152/52, respectively. Replacement components (upper heads, pressurizers) and other weld repair processes now systematically use alloy 52/152 as in, for example, the so-called 'half-nozzle repair' which transfers the pressure boundary from the original internal alloy 132/182/82 J-groove welds to new ones on the external surface made with alloy 52/152. The 'mechanical nozzle seal assembly' is also an effective alternative repair method for J-groove welds of small bore nozzles. Other techniques involving the deployment of PWSCC resistant coatings are internal surface overlay weld cladding with alloy 52/52M, inlay weld layers of alloy 52/52M for butt welds, and cold spray coatings of a PWSCC resistant material.

The effects of neutron irradiation on metallurgical microstructure and susceptibility to IASCC in both BWRs and PWRs is the subject of much investigation worldwide as plants age and neutron doses to internal core support structures increase, essentially linearly, with operating time. Presently, there are no materials that appear to offer significant advantages over the austenitic stainless steels used to date. Mitigation strategies are based on reducing neutron exposure where possible, for example by adopting low leakage cores, and replacing cracked components such as core shrouds in BWRs and baffle former bolts in PWRs.

Reduction of the stress driving SCC in both BWRs and PWRs can be achieved by various methods. One technique applied to the external surface is MSIP and results in a compressive stress on the inside surface. The weld overlay process is also applied to the external surface with the same objective and can be designed to be a full structural overlay that replaces the mechanical duty of the original nozzle and weld. Another similar stress improvement technique that has been developed is the Outer Surface Laser Improvement Process. Various welding techniques have also been developed that minimize residual tensile stresses and can leave the internal surface stress in compression.

Improvements to internal surface roughness and particularly its stress state have also been achieved by polishing, buffing, water jet peening, shot peening, ultrasonic peening and laser peening. These processes eliminate cold worked surface layers and in many cases generate a compressive stress in the water wetted surface.

Appendix I

STRESS CORROSION CRACKING MANAGEMENT APPLICATION

In this section, the examples of the application of mitigation measures against PWSCC in alloy 600, nickel based alloy and associated weld metals are described. The components affected are nozzles and penetrations in the RV upper head penetrations for control rod drive mechanisms, thermocouple nozzles, in-core instrumentation nozzles, RV upper head nozzles including exhaust line nozzles, associated J-groove welds and adjacent RV upper head nozzles, RV hot leg/cold leg nozzles, SG inlet/outlet nozzles, and PZR nozzles, in Japanese PWRs.

I.1. PREVENTIVE ACTIONS

Preventive measures to mitigate PWSCC are intended to improve the three synergistic parameters of material, stress and environment that combine to cause SCC. For the material aspect, alloy 690 nickel based alloy and its compatible weld metals, for which PWSCC resistance is notably enhanced by increasing the chromium content, are used for primary water wetted sections. PWSCC resistant alloy 690 can be used for replacing nozzles or cladding existing alloy 600 type nickel based alloy weld metals.

In order to improve the stress aspect of PWSCC, shot peening can be applied in the air while water jet peening can be applied under water to generate compressive residual stresses on the wetted surfaces. If it is difficult to gain access from the nozzle ID, high power laser beam (L-SIP technique) can irradiate the OD in order to rapidly heat it so that the resulting temperature difference across the wall thickness causes thermal expansion strains that generate a compressive residual stresses on the ID wetted surface. In addition, recent findings have shown that PWSCC is likely to be initiated where high residual stresses and a hardened layer exist on the surface due to the effects of welding and surface finishing. To address these concerns, a surface finishing method to remove the hardened layer may be applied; such as by buffing, which is an effective preventive maintenance measure to reduce the probability of PWSCC occurring.

For the environmental aspect, temperature reduction may be possible for limited portions of the primary circuit, for example, in the RV head where the temperature inside the RVH can be reduced to that of the cold leg; i.e. T-COLD by increasing the bypass flow to the RVH within an acceptable amount so as not to have any effect on reactor performance and safety. It is effective for reducing susceptibility to PWSCC because of the high temperature dependence of PWSCC. In addition, it is essential to accumulate knowledge of environmental conditions impacting susceptibility to PWSCC other than temperature condition, for example hydrogen concentration in primary water.

I.2. MONITORING

In order to confirm the structural integrity of the components before loss of component function as a pressure boundary, it is necessary to perform the periodic leak tests to check whether there are any indications of primary coolant leakage due to through-wall cracking, which might, for example, exist in upper VHP nozzles, BMI nozzles and associated partial penetration J-groove welds. Bare metal inspection can also confirm whether there are any indications of primary coolant leakage in the form of boric acid residues on external component surfaces or rust coloured corrosion products generated from low alloy steel components. It should be noted that the timing of PWSCC initiation at operating units can differ from that indicated by laboratory data, depending on the surface finish condition of the operating units.

For locations in the RV, SG and PRZ nozzles where surface cracking might be observed by inspection of the ID (penetrant testing (PT) or by eddy current testing (ECT), ultrasonic testing (straight beam and longitudinal wave angle beam UT) are applied to confirm whether cracking is significant or not. However, UT may not be capable of detecting a clear crack tip echo for certain types of cracking, particularly when the crack depth is greater than the surface length.

The following are examples of monitoring actions and follow-up actions in Japanese PWRs for alloy 600 type nickel based alloy weld metals, such as SG nozzles, VHP nozzles and BMI nozzles. In the case of SG inlet/outlet nozzles, the locations where cracking is detected by surface inspections (PT or eddy current testing (ECT)) are then examined by UT from the ID to identify the shape and size of the cracks. The priority of inspections should be placed on SG inlet nozzles rather than outlet nozzles since former are more sensitive to PWSCC due to a higher operating temperature.

To identify the cause of cracking, it is important to observe the morphologies (i.e. location, orientation and shape) of surface cracks by conducting visual inspection, ECT, replicas and SUMP observations, and to confirm whether they are similar to others observed in past incidents. When performing replica or SUMP observations, representative sections can be selected, due to the similarity of cracks, in order to minimize workers' radiation doses.

In case of VHP and BMI nozzles, bare metal inspections are conducted according to the regulations in addition to leak tests conducted at every refuelling outage. Bare metal inspection required by the regulatory authority have the objective of detecting primary coolant leaks from VHP nozzles and associated J-groove welds to confirm any possible loss of low alloy steel resulting from wastage due to concentrated boric acid. In particular, the schedule and frequencies of VHP nozzle inspections depend on the sensitivity classifications ('low', 'mid', and 'high' or 'replacement' according to the calculation of total effective degradation years) for each plant. For the purpose of classifying the operating plants according to the sensitivity to PWSCC, it should be noted that the timing of PWSCC initiation can differ from that indicated by laboratory data depending on the surface finish condition of the operating units.

In addition, knowledge of the incidents of PWSCC needs to be consistently updated so that corrective actions and preventive measures taken at the operating plants can be applied to other plants having common factors, such as surface finish condition and/or period of manufacture.

I.3. ACCEPTANCE CRITERIA

If cracks are detected by microscopic inspection of VHP nozzles, crack growth and fracture evaluations should be conducted to confirm whether structural integrity can be maintained during plant operation. Even if the detected cracks are determined to be axial or radial rather than circumferential, it is recommended that the evaluation is conservatively performed with an assumption of the presence of circumferential cracks, in order to be sure of structural integrity.

If there is no established code for evaluating crack growth and fracture of nickel based alloys, integrity can be assured in a rather conservative manner by completely removing any cracks and then applying a surface finishing method. The aim is to achieve a smooth surface where excessive stresses cannot concentrate while assuring that the minimum wall thickness of the design and construction phases is respected. It is also recommended that stress improvement measures are taken after the removal of the cracks.

I.4. CORRECTIVE ACTIONS

The measures described above as preventive actions can also be applied, in principle, as corrective actions. The surface condition after applying a repair should be appropriate for allowing adequate inspection capability. Isolation of the affected parts from the aqueous environment by corrective measures, such as the cladding, enables further continuous plant operation while leaving the cracks under the cladding material (and within the structural material). When cracks are relatively shallow and a repair weld is not applied after removing the cracks, it is necessary to finish the surface appropriately to ensure adequate resistance to stress corrosion cracking. In order to prevent PWSCC initiation, any hardened layer should be removed or other appropriate measures taken to improve residual stresses.

I.5. FEEDBACK FROM R&D RESULTS

For the cladding repair technique, it should be noted that there are greater difficulties with applying alloy 690 series nickel based alloy weld metals than alloy 600 series weld metals due to susceptibility to hot cracking, insufficient weld pool fluidity, and oxidation. When applying alloy 690 series nickel based alloy weld metals, special consideration should be given to the susceptibility to micro-cracking due to local (micro) residual stresses in repair welds.

For the sizing of cracks, UT may not be capable of detecting a clear tip echo in case of the cracks, which have greater crack depth than their surface length. In this regard, there is an increasing demand for the improvement and verification of sizing accuracy by using the advanced UT techniques, such as phased array UT.

For assessing the impact of environmental improvements, changes in water chemistry may have adverse effects on the performance and integrity of fuel. Occupational radiation exposure should also be evaluated before its implementation based on the latest operational experience and knowledge. Methodologies should be also established for controlling and monitoring the adverse effects of any water chemistry changes. The effectiveness of water chemistry measures for prevention and/or mitigation of PWSCC should be reviewed, particularly if high burnup fuel and power up rating are to be introduced since they could effect environmental parameters, such as radiolysis, thermal hydraulics, etc.

Finally, it is important to utilize the knowledge collected in OECD/NEA SCAP activities.

Appendix II

ASSESSMENT AND FLAW ANALYSIS

General guidance on this subject can be found in TECDOCs applicable to LWR RPVs (IAEA-TECDOC-1470 and IAEA-TECDOC-1556), LWR internals (IAEA-TECDOC-1471, and IAEA-TECDOC-1557) and PWR primary piping (IAEA-TECDOC-1361). In this section, specific guidance is provided for the evaluation of components fabricated from austenitic alloys and effected by IGSCC.

The first consideration that has to be taken into account when performing structural evaluations of components, cracked or uncracked, is that it should always comply with the Design Basis, Codes, Standards and Regulations applicable to the plant. These documents are different depending on the country where the respective plants were designed and built. Again, the above mentioned publications contain a detailed list of the Design Bases, Codes, Standards and Regulations applicable in different countries for LWR RPVs, LWR internals and PWR primary piping.

The actions needed in the event that plant specific flaw evaluations are required are listed in the following paragraphs.

II.1. LOADING

This section contains a brief description of the various loading and the load combinations that need to be considered to determine the primary and secondary stress levels appropriate for various operating conditions.

II.2. APPLIED LOADS

Typical applied loads on austenitic alloys components consist of the following: deadweight, mechanical, thermal, seismic and accident loads. In some cases, other types of loading can also be present; e.g. hydraulic, fluid drag, acoustic, vibration.

Deadweight consists of the weight of the component. For flaw evaluation purposes, the stress from this load is treated as primary.

Mechanical loads are those caused by the interaction with the adjacent systems/components. These loads are classified as primary.

The anchor points of the components thermally expand vertically and horizontally at different rates. Also, these displacements are expected to vary during transients. The loads produced by these thermal anchor displacements and thermal expansions are treated as secondary.

Seismic inertia consists of horizontal and vertical inertia forces acting on the component due to seismic excitation. For flaw evaluation purposes, the stresses from the seismic inertia loading are treated as primary.

Seismic anchor displacements are applied at the attachment points of the component. These displacements are obtained from the plant seismic analysis report. For flaw evaluation purposes, the stresses from the seismic anchor displacement loading are treated as secondary.

Two types of earthquakes can be considered: operating basis earthquake (OBE) and safe shutdown (or design basis) earthquake (SSE).

The occurrence of an accident, for instance a loss of coolant accident (LOCA) in the pressure boundary, can result in loading applied to the analyzed component. These loads can be primary and secondary.

Hydraulic loads (primary) arise from the pressure and fluid momentum forces. Fluid drag loads (primary) consist of the forces resulting from fluid flow past the component. The pressure shock wave load (primary) is a momentary shock load after a postulated double ended break of a system line (LOCA). Vibrations (primary) can have different sources depending on the component subject to assessment.

This section has given a typical list of existing loads. However, for each specific evaluation more sources of loading (defined in the design stress report) can be present. It should be carefully checked that all loads are taken into account.

II.3. LOAD COMBINATIONS

The load combinations used in the evaluation should be consistent with the requirements of the plant licensing basis documentation. Load combinations are classified in different categories: normal, upset, emergency and faulted.

Some plants are not required to combine LOCA plus SSE in their licensing basis. For those plants that combine LOCA plus SSE, a square root of the sum of the squares (SRSS) method of combining SSE and LOCA loads may be used.

II.4. ALLOWABLE FLAW SIZE DETERMINATION

This section presents methodologies for evaluating the effect of flaws in austenitic alloy components. Due to the different nature of the various components, detailed flaw evaluation methods are not presented for every potential failure location. Rather, general guidance is provided describing the important elements which should be addressed in performing analyses.

The allowable flaw size (a_{allow}) required in order to operate for 'n' years prior to the next inspection is given as:

$$a_{allow} = a_{lim} - (CGR \times n)$$

where: a_{lim} = limiting flaw size
n = inspection interval (years)
CGR = crack growth rate

In evaluating cracking observed with VT or other surface techniques it should be assumed that the cracking is through-wall. For volumetric measurements, the measured depth may be used in evaluations. Also, an assumption must be made regarding the condition of uninspected regions.

If multiple indications are detected during the inspection at any location, then the interactions, if any, between these indications must be accounted for in the structural margin evaluation.

II.5. FAILURE CRITERIA

To account for the effect of embrittlement (neutron or thermal) three different failure criteria are considered to calculate the allowable flaw size or to determine whether a given measured crack satisfies the structural margins. The three criteria are: (a) limit load, (b) linear elastic fracture mechanics (LEFM) and (c) elastic-plastic fracture mechanics (EPFM).

Austenitic alloys are inherently ductile and therefore, in most cases, the structural integrity analysis can be performed entirely on the basis of limit load. The only case for the use of other techniques such as LEFM or EPFM would be when irradiation or thermally induced changes in the material fracture toughness properties are judged to be significant. Properties relevant to material fracture toughness include yield and ultimate tensile strengths, uniform elongation and upper-shelf Charpy energy.

Thermal embrittlement affects mainly cast austenitic stainless steels. This material is generally considered to be not susceptible to IGSCC, although there is an absence of an adequate database on thermally aged material. This ageing mechanism will not be considered in this evaluation.

Alternatively, the mechanical properties of austenitic alloys are a function of the neutron fluence they have received; therefore it is necessary to determine a fluence value above which the use of LEFM or EPFM techniques would be required.

Reference values of the applicable fluence ranges (E > 1 MeV) for the various fracture criteria are the following:

Limit load only:	(*fluence*) $<3 \times 10^{20}$ n/cm^2
LEFM or EPFM with limit load:	3×10^{20} n/cm^2<(*fluence*) $<10^{21}$ n/cm^2
LEFM with limit load:	(*fluence*) $>10^{21}$ n/cm^2

The limit load methodology described in Appendix C of ASME Section XI is one of the approaches that may be used to determine the critical and allowable flaw lengths for circular sections. Alternative methods may also be used if justified. This criterion can be expressed as:

$$\sigma_{app} < \sigma_f/SF$$

where: σ_{app} = applied stress at the subject location
SF = safety factor appropriate for the operating condition being evaluated
σ_f = material flow stress

The safety factor varies for the different conditions; for instance in the ASME Code, it takes a value of 2.77 for normal/upset conditions and 1.39 for emergency/faulted conditions.

The LEFM failure criterion is expressed as:

$$K < K_{Ic}/SF$$

where: K = applied stress intensity factor at the subject location
K_{Ic} = material fracture toughness

A value of 150 ksi√in is considered a conservative estimation of K_{Ic} for unirradiated stainless steel.

The EPFM based concepts can be used in lieu of the conservative LEFM approach in which only crack initiation is considered. The EPFM approach considers ductile crack extension in determining the load carrying capability of a cracked structure. This methodology is usually formulated in terms of the J-integral, which characterizes the intensity of the plastic stress-strain field surrounding the crack tip. A tearing stability analysis is then performed to examine the stability of ductile crack growth. The following conditions must be satisfied:

(1) The crack driving force must be shown to be less than the material toughness as given below:

$$J_{app} < J_{Ic}$$

where, J_{app} is the J-integral value calculated for the postulated flaw. The parameter J_{Ic} is the J-integral characteristic of the material resistance to ductile tearing.

(2) The flaw must be stable against ductile crack growth as given below:

$$(d\, J_{app} /da) < (d\, J_{material} /da)$$

where, $J_{material}$ represents the resistance versus crack growth curve (J-R curve) of the material under evaluation.

II.6. CRACK GROWTH

Although this publication is devoted to analyzing mainly IGSCC damage of austenitic alloys, in general, when performing structural assessments and crack growth estimations in particular, all potential ageing mechanisms must be taken into account.

Apart from IGSCC, the most common ageing mechanism that promotes crack growth is fatigue. As in preceding sections, crack growth rate estimations for IGSCC and fatigue can be found for the different type of components and materials in the applicable TECDOCs.

Sections 6.2 and 6.3 of IAEA-TECDOC-1470 provide fatigue and IGSCC assessment, methodologies, respectively, for BWR RPVs. These same issues are covered in Sections 7.1 and 7.2 of IAEA-TECDOC-1471 for BWR internals. PWR RPVs are addressed in IAEA-TECDOC-1556: Section 6.3 covers fatigue assessment procedures in different countries. Section 6.4 deals with PWSCC assessment of alloy 600, again in several countries, and finally Section 6.5 addresses assessment methods for stress corrosion cracking of RPV closure head studs. Section 5 of IAEA-TECDOC-1361 covers all types of assessment methods for primary piping in PWRs including thermal fatigue, vibratory fatigue and PWSCC of alloy 600.

Crack growth calculations may be based on the estimated hot-operating hours in each year, rather than on the total number of hours in a year.

II.7. OTHER CONSIDERATIONS

Although this report deals with intergranular stress corrosion cracking (IGSCC), degradation of austenitic alloys and fatigue have also been considered in relation to crack growth evaluations. Some other failure mechanisms that may cause loss of material in the components reducing their wall thickness and their mechanical strength, thus penalizing their structural integrity.

Neutron irradiation leads to embrittlement of austenitic alloys; i.e. a reduction of ductility and fracture toughness, as previously mentioned. This phenomenon also promotes the IGSCC damage when IASCC crack initiation and growth are favoured.

Austenitic alloys have good resistance to general corrosion mechanisms, including flow assisted corrosion.

As it has already been mentioned, cast stainless steel and, to a lesser extent weld metal, are susceptible to thermal ageing. However, embrittlement does not directly cause cracking and these materials have not been affected so far by IGSCC.

Mechanical wear has been identified as a degradation mechanism at specific locations of BWR and PWR internals. In PWR RPV, one location concerned is the bolted flange, and the degradation can be detected long before the effects of wear begin to compromise the RPV structural integrity.

Radiation induced creep in austenitic alloys is a function of stress level, temperature and time at temperature. Some creep/relaxation of baffle bolts has been observed during testing and replacement of baffle bolts in the USA, France, Japan and Belgium.

There is not credible evidence that void swelling can cause cracking in austenitic stainless steel although there is some concern that helium bubble generation from neutron transformation reactions, notably of nickel, could eventually influence cracking.

Finally, other considerations that have to be taken into account when performing flaw analyses are:

- Structural integrity in not the only acceptance criterion because in some cases excessive leakage or excessive deformation can be more limiting;
- Leak before break is a possible analysis procedure for evaluating piping integrity;
- The data needed for the assessment may be different depending on the condition of the evaluated component, that is, depending on whether the component is degraded or not.

Appendix III

INTERNATIONAL/NATIONAL RESEARCH ACTIVITIES TO MANAGE STRESS CORROSION CRACKING

III.1. INTERNATIONAL RESEARCH ACTIVITIES

Organization	Activity name	Objective (additional information)
EC	Nuclear plant life prediction (NULIFE)	— Create a single organizational structure capable of working at the European level to provide harmonised R&D in the area of lifetime evaluation methods for structural components for the nuclear power industry and the relevant safety authorities. (http://nulife.vtt.fi)
	Sixth framework	**National AUSTOS project in the context of sixth framework** With respect to irradiation assisted stress corrosion cracking, cold worked material conditions are widely used to simulate irradiation induced embrittlement for exposure experiments focused on crack initiation and crack propagation. Corrosion issues of internals were particularly investigated in this project focusing on crack initiation and crack growth of cold worked but not (thermally) sensitized austenitic stainless steels (type 347, type 316NG and type 316Ti) exposed to simulated light water reactor conditions (BWR and PWR).—Different degrees of cold work were realized by cross rolling. — The influence of surface preparation was investigated using different surface finishes simulating normal and worst case scenarios. — Crack initiation in BWR and PWR environments was investigated by using flat bar specimen, U-bend specimens, passive and active loaded 4-point-bending specimens, tapered specimens and finally custom designed crevice cubes. — For crack propagation experiments, standard C(T)-specimen were used. — Material characterization was performed using standard metallography complemented by highly sophisticated methods such as the nano-indentation technique, EBSD-measurements and TEM investigations.
EPRI	CIR, MRP, BWR VIP, SGMP	**MRP[a]:** The EPRI MRP was formed in 1998 to identify and address issues that could affect operability of major components in PWRs. Major activities are coordinated with the nuclear steam supply system (NSSS) vendors, the vendor owner's groups, The Nuclear Energy Institute (NEI), and the NRC. The MRP provides for a unified industry approach to the resolution of technical and regulatory issues related to PWR materials degradation. **BWRVIP[b]:** In the Summer of 1994, the BWR Owner's Group formed the BWRVIP to address reactor vessel and internals cracking issues. The objective of the BWRVIP was four fold: — Generic resolution of reactor vessel and internals integrity and operability issues. — Development of generic, cost effective strategies. — Focal point for regulatory interface. Information sharing. Five subcommittees were formed to meet the objectives, and dealt with mitigation, inspection, assessment, repair, and integration.
OECD/NEA	SCC Project (SCAP)	— Establish a complete database with regard to major ageing phenomena for SCC and degradation of cable insulation through collaboration by OECD/NEA members. — Establish a knowledge base in these areas by compiling and systematically evaluating the collected data and information. — Perform an assessment of the data and identify the basis for commendable practices which would help regulators and operators to enhance ageing management. (http://home.nea.fr/html/jointproj/scap.html)l

Organization	Activity name	Objective (additional information)
OECD/NEA	Halden Reactor Project	The plant lifetime assessment programme[c] is aimed at studying the potential degradation of reactor vessel internals due to irradiation effects as the age of operating nuclear power plants increases, such as stress relaxation and the irradiation assisted stress corrosion cracking (IASCC) susceptibility of core component structural materials. The experimental programme on IASCC is aimed at generating data that provide a fundamental mechanistic understanding of IASCC, predicting future behaviour, in particular the cracking response of irradiated materials, assessing possible countermeasures and determining the limits of operation for existing materials. The majority of the IASCC investigations are performed in loops able to simulate light water reactor operating and water chemistry conditions, while the stress relaxation and embrittlement studies are performed in inert environments.

[a] Stan T. Rosinski, Robert O. Hardies, Pressurized thermal shock screening criteria re-evaluation effort — US industry activities, International Journal of Pressure Vessels and Piping; Volume 78, Issues 2–3, February 2001, Pages 147–153.

[b] Taken from: http://www.structint.com/tekbrefs/sib96138/SIB96138r2.htm

[c] Taken from report HP-1217, Halden Project proposal 2007, Institutt for energiteknikk

III.2. NATIONAL R&D PROGRAMMES

Country	Activity name	Objective (additional information)
Belgium and Brazil	Agreement between Belgium (SCK.CEN) and Brazil (CDTN/CNEN) for Scientific, Technological and Industrial Cooperation — Project Agreement I: Collaboration on characterization of the stress corrosion cracking of nickel based alloys for structural applications in nuclear reactors	— Characterize the metallurgical condition of an industrial weld retrieved from the cancelled PWR reactor of the Lemoniz plant in Spain; — Identify critical conditions for the occurrence of SCC in primary water conditions relevant to the operation of a pressurised water reactor; — Quantify the rate of propagation of a stress corrosion crack in relevant materials as a function of temperature, stress intensity and water chemistry (hydrogen content and pH); — Correlate the crack propagation velocity observed with the metallurgical state of the material and the electrochemical interaction between material and environment.
	Detection, assessment and mitigation of PWSCC — Applied to Angra 1. CDTN/CNEN-ETN-IPEN (participation of Belgian utilities is being negotiated)	— Develop detection (NDT) and assessment methods for stress corrosion cracks, including development of theoretical and experimental models. — Study the use of weld overlay techniques on pressuriser nozzle to safe-end welds, including microstructural and mechanical analysis of mock-ups and characterisation of their SCC behaviour.

Country	Activity name	Objective (additional information)
France	EDF	1 – SCC Development of models mainly for *initiation* but also for propagation of EAC. The investigated materials are *Ni-alloys* 600, 690 and their weld metals 182, 82 and 152/52, and type 304L, 316L, 304, 316 *stainless steels (SS)* together with weld metal 308L, all in a *PWR environment*. Four (4) work packages (WPs), covering both classes of materials (Ni-alloys and stainless steels) will contribute to the development of the physical modelling of EAC: — Two WPs will investigate SCC and corrosion fatigue, including the development of knowledge and laboratory techniques: they will result in semi-empirical (engineering) models based on laboratory results. In the SCC WP, the relationship between the conditions of manufacture for components in alloy 600 and their SCC behaviour will be proposed, in order to improve the engineering model for initiation. Engineering models will be also developed for initiation of SCC for alloy 690 and stainless steels. Initiation and propagation of *corrosion fatigue in SS* will be started in RHRS conditions. — Two WPs will develop specific contributions to the corrosion process : growth kinetics and mechanical failure of *oxides*, *hydrogen/material interactions*. The 5th WP will be the physical modelling itself for SCC initiation, based on the mechanism of cracking, mainly for *alloy 600 and cold worked stainless steels in nominal PWR primary environment*. A coupling of oxidation kinetics with the plastic behaviour of the material will be attempted, validation being based on laboratory results from the engineering approach. 2 – IASCC The specific objectives of the project in progress are: — To understand materials degradation mechanisms with irradiation to help formulate a quantitative modelling of IASCC and void swelling; — To define a predictive model for the cracking of baffle/former bolts, validated on the field experience of CP0 bolts inspections and extend use to CPY and 1300 MW(e) bolts. — To develop justification of internals lifetime depending on operating conditions.
Korea	Korea Atomic Energy Research Institute (KAERI)	National R&D programmes in the Republic of Korea: Ten year nuclear R&D programme was finished as a first phase in 2006. Another five year (2007–2011) R&D programme has been implemented as a second phase. An evaluation of PWSCC of nozzles and penetrations and of ODSCC of steam generator tubes, IASCC research for reactor internals. Water chemistry in PWRs are also corrosion related research activities in the national research laboratory(KAERI)
Japan	Japan Nuclear Energy Safety Organization (JNES)	Japan Nuclear Energy Safety Organization (JNES) has conducted many investigations, testing and research to ensure the safety of nuclear installations and associated with safety regulations. The activities include safety research on stress corrosion cracking covering the fields of integrity evaluation of components, maintenance and repair, and inspection and monitoring. Major safety research activities on SCC are as follows:
	Integrity assessment of flawed components with structural discontinuity (IAF)	— One of the objectives of the IAF projects is to establish an evaluation method of weld residual stress necessary to predict crack growth and a crack growth evaluation method for nickel based alloy welds such as vessel penetrations, nozzle safe-ends and shroud supports, where the SCC has recently occurred. In parallel, an evaluation method for fatigue crack growth is to be established for elbows and tees of pipe joints. — The programme includes tasks on residual stress evaluation for Ni-based alloy welds and stress intensity factor solutions for Ni-based alloy welds. — The residual stress profile for the H10 weld on BWR core shroud supports was obtained by calculation from the elastic strain released by cutting mock-up test pieces. — Project periods: 2001JFY–2007JFY — Conducting organization: JNES

Country	Activity name	Objective (additional information)
	Validation of fracture evaluation method for Ni-base alloy weld (NFA)	— Objectives of the project are to (a) obtain and compile the material data necessary for the fracture evaluation of nickel based alloy welds; such as the J-groove welds of reactor vessel penetrations, reactor coolant out/inlet nozzle welds and shroud support welds etc., (b) establish a fracture evaluation method for nickel based alloys based on the results of crack growth behaviour and fracture loads observed in fracture tests of test specimens simulating welds of operating plants, and (c) prepare judgment criteria and a technical basis for structural integrity evaluation. — The programme includes tasks (a) material tests for the nickel based alloy weld metal and base metal, (b) fracture tests for basic studies to understand the fracture behavior of Ni base alloys, to establish an evaluation method for verifying the adequacy and conservatism of the proposed fracture evaluation method, and (c) overall evaluation. — Project periods: 2005JFY–2009JFY — Conducting organization: JNES
	Evaluation methodology for crack growth rate assessment for Ni-based alloys (NiSCC)	— The objectives of the project are to obtain sufficient CGR data for nickel based alloys (base metals and weld metals) in PWR and BWR environments for and to prepare ‘CGR vs. K’ curves for evaluating the integrity of plants. — The SCC growth tests for Ni base alloy base and weld metals are being conducted systematically in simulated PWR and BWR water under constant load using CT specimens, in order to derive clear t relations between the CGR and stress intensity factor. — Regarding SCC growth evaluation in nickel based alloys, SCC growth data are being obtained and the SCC growth rate diagram developed for various nickel based weld and wrought alloys in both BWR and PWR conditions. — Project periods: 2000JFY–2005JFY — Conducting organization: JNES
	Verification of evaluation technology for stress corrosion crack growth rate in Nickel-base alloy s(NSC)	— The objectives of the programme are to verify SCC growth evaluation technology for welds of nickel based alloys based on state of the art knowledge, to establish an adequate method for evaluating integrity as a standard, and to make recommendation for academic and association standards, if necessary. — The project includes SCC growth tests for weld metal taking into account residual stress distribution and full scale verification testing under PWR conditions. — The results show that the SCC growth rate under K decreasing-type conditions tends to be smaller than the rate measured under K increasing-type conditions. — Project periods: 2005JFY–2009JFY — Conducting organization: JNES
	Evaluation of irradiation assisted stress corrosion cracking (IASCC)	— Objectives of the project are to (a) obtain the data to characterize IASCC susceptibility and crack growth based mainly on post-irradiation tests, (b) establish an IASCC database for structural integrity evaluations of ageing plants (envisaging operation up to 60 years), and (c) propose an IASCC evaluation guide which can be used by the regulatory authorities. — For the BWR study, the crack length is calculated for detected cracks in BWR core internals. The main data to be obtained are crack growth rates and fracture toughness to determine the lifetime. Specimens with the same chemical composition as the operating plant materials of the core shroud are irradiated in the Japan material test reactor (JMTR) followed by SSRT and then fracture toughness tests of the irradiated materials. — For the PWR study, since baffle former bolts are considered to lose their function when cracks initiate in them, crack initiation data are the main objective. The tests are performed using irradiated materials removed from operating plants, in order to obtain highly irradiated materials in the short term. — Project periods: 2000FY–2008FY — Conducting organization: JAPEIC/JNES(-2003.10)

Country	Activity name	Objective (additional information)
	Intergranular stress corrosion cracking of nuclear grade stainless steels (IGSCC)	— The objectives of this programme are to (a) obtain a reliable crack growth rate database applicable to integrity evaluations of primary loop recirculation piping and core internals made of low carbon stainless steels and (b) prepare a guide for evaluating crack growth, which the regulatory authorities can use for investigations of structural integrity. — The project has the following task items: (a) manufacturing of mock-ups simulating operating plants as far as possible in respect of materials, welding method and size in order to assure equivalency, (b) crack growth tests, (c) verification tests of crack growth and (d) evaluation. — The results of SCC crack growth tests show that CGRs increase with hardness above about 200 HV regardless of welding technique. It was confirmed that the SCC CGRs obtained for the HAZ were small compared to those for sensitized SUS304 described in the Code on Fitness for Service of the Japan Society of Mechanical Engineers (JSME). — Project periods: 2003FY–2007FY — Conducting organization: JAPEIC/JNES(-2003.10)
	Evaluation of neutron irradiation effects on SCC crack growth of L-grade stainless steel (ENI)	— Objectives of the project are to (a) obtain SCC crack growth data for low carbon stainless steels irradiated up to the fluence threshold for IASCC initiation susceptibility, (b) to prepare a SCC CGR diagram for stainless steels subjected to weld hardening and neutron irradiation, (c) identify the fluence threshold at which accelerated CGRs are observed, and (d) establish an improved crack growth evaluation of core internals. — The project includes the following tasks (a) manufacturing mock-ups, (b) pre-irradiation tests, (c) neutron irradiation, (d) post-irradiation tests, (e) basic tests, and (f) overall evaluation. — Project periods: 2007FY–2010FY — Conducting organization: JNES
	Evaluation of low crack growth rates in L-grade stainless steel (ELC)	— The project started in 2008. The project includes tasks on (a) SCC growth tests for weld hardened zones of low carbon stainless steels focusing on low stress intensity factors and (b) SCC growth tests using large scale pipes to verify the applicability of SCC growth equations to welding hardened zones of low carbon stainless steels for integrity evaluations of operating plants. — Project periods: 2008FY–2010FY — Conducting organization: JNES
	Repair welding technology for irradiated materials (WIM)	— Objectives of the project are to (a) develop repair welding techniques for neutron irradiated materials such as austenitic stainless steels and low alloy steels, (b) qualify the techniques for core internals and reactor (pressure) vessels, and (c) recommend updated repair welding techniques for the technical rules and standards. — The project includes tasks on (a) technical survey of the current status, (b) preparation of samples and neutron irradiation, (c) basic studies using un-irradiated materials, (d) welding tests on irradiated materials, (e) evaluation of weld joints and (f) overall evaluation — The results of the tests in the project have identified a relationship between helium contents, weld heat input and occurrence of cracks in irradiated stainless steels for BWR reactor core shrouds. Based on the results of tests using temper bead welding, proposed technical guidelines for selecting repair welding processes for reactor (pressure) vessels were proposed and made public as a JNES safety standard report, JNES-SS-0501. — Project periods: 1997FY–2004FY — Conducting organization: JAPEIC/JNES (-2003.10)
	Integrity assessment of repair technologies for irradiated materials (RWIM)	— The objectives of the project are to obtain data for integrity evaluations such as ageing characteristics in repair welds and to develop evaluation guidelines regarding repair welds in irradiated materials. — The project includes tasks on (a) design and fabrication of test specimens for evaluating the integrity of irradiated weld joints, (b) material tests and welding tests for irradiated materials, (c) study detailed implementation plans of integrity evaluation methods for irradiated low alloy steel welds, test conditions, detailed specifications, and test matrix and (d) design and fabrication of capsules for neutron irradiation tests. — In 2006FY, an H3 welded joint simulating a BWR core shroud was fabricated for weld simulation tests. — Project periods: 2006FY–2011FY — Conducting organization: JNES

Country	Activity name	Objective (additional information)
	Nuclear power plant material improvement technology (PMT)	— The objectives of the project are to verify the effectiveness of surface treatment processes, such as laser treatment, to improve stress corrosion resistance. — Verification items include tests on (a) surface modification technologies for reactor (pressure) vessel internals to confirm their applicability for RPV internals of Japanese domestic plants, (b) surface modification technologies for primary coolant pressure boundary equipment to verify their applicability (for example, BMI nozzle), etc., and (c) overall evaluation of surface modification technology — Based on the results of verification tests in the project, essential parameters for repair technologies, including laser cladding inside bottom mounted instrument nozzles for PWRs and laser surface treatment for irradiated SUS 304 and SUS 316L for BWR core shrouds were established. The project was completed in 2004. — Project periods: 1996FY–2003FY — Conducting organization: JAPEIC/JNES(-2003.10)
	Non-destructive inspection technologies for core shroud integrity assessment (NSA)	— The objectives of the project are (a) to verify ultrasonic test (UT) and eddy current test (ECT) techniques for detection and sizing of stress corrosion cracking (SCC) which occurs on core shrouds and primary loop recirculation (PLR) piping made of low-carbon stainless steels for BWR and (b) to prepare draft inspection guidelines for components. — The programme includes test items on (a) design and manufacturing test specimens simulating representative parts of the core shroud and PLR piping, (b) basic tests to obtain ultrasonic and electromagnetic characteristics of low-carbon stainless steels, (c) verification tests including flaw detection tests, primary evaluation, destructive verifications and secondary evaluation of detectability and sizing capability, in order to evaluate detectability and sizing capability of the various inspection methods, (d) analysis evaluation by simulation, (e) overall evaluation and (f) incorporation into draft inspection guidelines for low-carbon stainless steels. — In 2006FY, evaluations were performed regarding detectability and sizing accuracy of crack length and depth for each series of test specimens simulating the PLR piping and core shroud. Based on the results of the overall evaluation, draft flaw inspection guidelines were developed; draft guideline for ultrasonic testing (PLR piping and core shroud) and draft guideline for eddy current test (ECT) (core shroud) — Project periods: 2003FY–2006FY — Conducting organization: JAPEIC/JNES(-2003.10)
	Nondestructive inspection technologies of Ni-based alloy weld joints (NNW)	— The purpose of the project is to conduct investigations and tests on advanced UT and ECT inspections for nickel based alloy welds and to prepare a draft inspection guideline. — The programme includes tasks on design and manufacturing of test specimens and equipment, preliminary tests to select inspection methods, basic tests including flaw inspection of electric discharged notches, verification tests to confirm flaw detestability, sizing accuracy of UT and ECT for test specimens with SCC, overall evaluation and incorporation into guidelines. — For verification, blind tests using a test specimen of a BMI nozzle weld with SCC were conducted in 2006FY. A preliminary draft proposal for the recommended method for inspection of reactor vessel penetrations was prepared as an inspection guideline for flaw detection, length sizing and depth sizing in 2006FY. — Project periods: 2002FY–2008FY — Conducting organization: JAPEIC/JNES(-2003.10)
	Nondestructive inspection technologies for the narrow penetrations on reactor vessel (NPV)	— The objectives are to perform verification tests for detection and quantitative evaluation of depth and length of flaws in the narrow parts and to develop draft guidelines for non-destructive inspection of narrow penetrations. — The programme includes tasks on design and manufacturing of test specimens, basic tests using UT and ECT for flaw detection in narrow parts, verification tests to simulate measurements on operating plants with both artificial flaws and SCC, analysis evaluation by simulation, overall evaluation and incorporation into guidelines. — For verification tests, results for specimens simulating in-core instrumentation tube nozzles in 2006FY showed that it was difficult to detect EDM notches by UT but the depth of the detectable EDM notches could be sized with high accuracy. On the other hand, ECT was able to detect all EDM notches in the tests. — Project periods: 2005FY–2008FY — Conducting organization: JNES

Country	Activity name	Objective (additional information)
Spain	ENDURO	Effect of hardening on IGSCC of austenitic stainless steels. Implications for IASCC processes. Carried out by CIEMAT between 2001 and 2003. Funded by Spanish Regulators (CSN) and UNESA (Association of Spanish Utilities).
	He-X 750	Alloy X-750 IASCC susceptibility under BWR conditions. Carried out by CIEMAT between 2004 and 2006. Funded by Spanish Regulators (CSN) and UNESA (Association of Spanish Utilities).
	ES-18	Alloy 690TT resistance to PWSCC in PWR type reactors (1st phase) Carried out by CIEMAT in collaboration with EPRI between 2004 and 2007. Funded by Spanish Regulators (CSN) and UNESA (Association of Spanish Utilities).
Sweden	SKC - Swedish Centre for Nuclear Technology http://www.swedishnuclear.se/index.php	Materials and chemistry The materials area includes work on: — Materials that are used or could be used for barriers against spreading of radioactivity from a nuclear power plant; — Materials that are used or could be used for important structures in a nuclear power plant. These structures include internal components of the reactor pressure vessel; — Materials that are used in component, structures or cables that are important to the safety of the plant.
	Work programme at Studsvik:	— Crack growth rate measurements for BWR and PWR applications; — Measurements of mechanical properties, in particular, fracture toughness tests; — Water chemistry studies on corrosion and mitigation of activity buildup in reactor systems, — Studies on shadow corrosion, crevice corrosion, fuel crud, crud transport, AOA, Stellite corrosion, effect of zinc etc; — ECP studies and ECP modelling tools; — Post-irradiation examinations of fuel rods; — Mechanical testing of irradiated materials, such as fuel cladding and samples from core components; — Spent fuel leaching experiments in hot cells.
Switzerland	KORA project	Within the KORA-I and –II projects, the following tasks are being investigated in the Laboratory for Nuclear Materials at the Paul Scherrer Institute in the field of environmentally assisted cracking of austenitic LWR structural materials: — SCC crack growth behaviour in the weld metal fusion line region and heat affected zones of Inconel 182/82 dissimilar metal welds under BWR and PWR conditions. — Environmental effects on fatigue initiation and crack growth in austenitic stainless steels and Ni-alloys under PWR and BWR/HWC conditions. — Evaluation of the electrochemical noise measurement technique for the detection of SCC initiation in stainless steels under BWR conditions. This research is partially funded by the Swiss Regulator HSK

CONTRIBUTORS TO DRAFTING AND REVIEW

Fujimori, H.	Hitachi-GE Nuclear Energy Ltd, Japan
Gorrochategui Sanchez, I.	Nuclenor S.A.; Santa Maria de Garona NPP, Spain
Gott, K.	Swedish Radiation Safety Authority, Sweden
Heldt, J.	Kernkraftwerk Leibstadt, Switzerland
Hwang, S.S.	Korea Atomic Energy Research Institute, Republic of Korea
Jacko Richard, J.	Westinghouse Electric Company, USA
Kang, K.S.	International Atomic Energy Agency
Kupca, L.	International Atomic Energy Agency
Molander, A.	Studsvik Nuclear AB, Sweden
Scott, P.	Consultant, France
Takamori, K.	Tokyo Electric Power Company, Japan
Tanaka, H.	The Kansai Electric Power Co., Inc. Japan
Widera, M.	RWE Power AG, Germany
Winter, M.	AREVA NP GmbH, Germany
Yaguchi, S.	Mitsubishi Heavy Industries Ltd, Japan
Yamamoto, A.	OECD Nuclear Energy Agency
Yamashita, N.	Tokyo Electric Power Company, Japan

Consultants Meetings

Vienna, Austria: 24–27 September 2007
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Key

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