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ENVIRONMENT SENSITIVE CRACKING IN LIGHT WATER REACTOR PRESSURE BOUNDARY MATERIALS

H. Hänninen and I. Aho-Mantila

TECHNICAL RESEARCH CENTRE OF FINLAND
Metals Laboratory
Metallimiehenkuja 6
SF-02150 ESPOO
FINLAND

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Hänninen, H. and Aho-Mantila, I.

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ABSTRACT
The purpose of this paper is to review the available methods and the most promising future possibilities of preventive maintenance to counteract the various forms of environment sensitive cracking of pressure boundary materials in light water reactors. Environment sensitive cracking is considered from the metallurgical, mechanical and environmental point of view. The main emphasis is on intergranular stress corrosion cracking (IGSCC) of austenitic stainless steels and high strength Ni-base alloys as well as on corrosion fatigue of low alloy and stainless steels. Finally, some general ideas how to predict, reduce or eliminate environment sensitive cracking in service are presented.

1. INTRODUCTION

Environment sensitive cracking of light water reactor pressure boundary components has caused significant forced outages with associated high cost for repairs and replacement power and occasional safety hazards. Materials related capacity factor losses have been in average 5 % per plant in recent years (Koppe et al., 1984). Most of the components of nuclear power plant have been affected by corrosion damage. In recent years numerous reviews of environment sensitive cracking in light water reactor components have been presented (e.g. Stahlkopf, 1981; Berry, 1984; Scott, 1984; Norring and Rosborg, 1984). This paper provides a review of remedial actions based on stress, environment and material improvements.

Materials related failures in large components do not occur often, but when they occur, outages may extend for months. These low frequency, high impact failures constitute the major causes of forced outages. Table 1 summarizes lost capacity during 1979 due to large component malfunctions (Stahlkopf, 1981). The complex interplay of metallurgical, mechanical and environmental factors in environment sensitive cracking is shown in Fig. 1. It can be seen that the number of variables that affect the environment sensitive cracking in light water reactor conditions is large and they possibly have a number of synergistic interactions.

In extrapolating the laboratory test results for a 30...40 years operating time the test times have to be very long - 6 months or more in many cases. Thus, the material testing in real environments is very expensive and time consuming. This has led to the formation of international joined groups, e.g.

- the Boiling water reactor owners group,
- the Steam generators owners group,
- the International co-operative group on cyclic crack growth rate testing and evaluation (the ICCGR group).

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Table 1. Loss of capacity due to large component malfunction (Stahlkopf, 1981).

Component/Problem	*Reactor Type*	*Average % Capacity Loss (1979)*	*1979 \$ Cost (\$M)*	*Projected 1985–1990 Industry Loss (\$M)*
Steam generator (denting, cracking, IGA)	PWR	3.25	197	2800
Steam turbine (blade pitting and cracking, disc and rotor cracking)	PWR and BWR	1.65	178	2500
RCS piping (intergranular stress corrosion, corrosion fatigue)	PWR	0.10	6.1	83 > 643
	BWR	0.81	38.3	560
Vessel, nozzle pump, etc. (SCC, corrosion fatigue)	PWR	0.21	12.7	173 > 782
	BWR	0.88	41.7	609
Condenser (tube pitting, leakage at rolled joints)	PWR and BWR	0.26	28.0	395

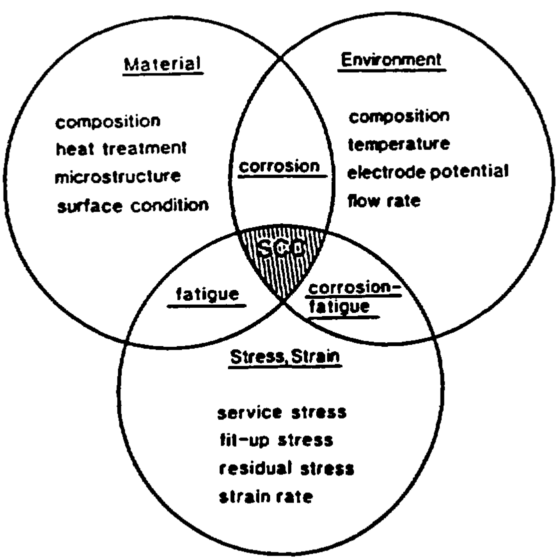


Fig. 1. Factors affecting environment sensitive cracking. Note that specific conditions are required for cracking to occur.

These groups have produced large amounts of data and results, but in the recent years it has also been recognised that it is extremely important to understand as far as possible the mechanisms of environment sensitive cracking; firstly to determine whether it is a relevant phenomenon in the operating conditions, and secondly the mechanisms can provide a reliable, physically-based crack growth prediction procedure over the plant operating time.

2. PRESSURE VESSEL AND CARBON STEEL PROBLEMS

Reactor pressure vessels of PWRs and BWRs are overlay welded with a stainless steel cladding. Stress corrosion cracking of cladding in a Japanese BWR (Kondo et al., 1971) and thermal fatigue cracking in the feedwater nozzle cladding of several US BWRs (Jones, 1984) have occurred. In both cases the cracks had penetrated also into the base metal. Although the causes of these original cracks are now understood, they have led to the need for accurate flaw estimation methods for heavy-section steel components.

The environment sensitive cracking properties of pressure vessel steels such as A533B and A508 have since then been studied to a large extent. Appendix A of Section XI of the ASME Boiler and Pressure Vessel Code presents a procedure for estimating the remaining useful life of a cracked reactor pressure vessel or nozzle. This procedure combines a fatigue crack growth analysis with a maximum allowable flaw size. The fatigue crack growth reference curves of ASME Code are shown in Fig. 2. The linear curve is based on material tests in air and is provided for subsurface flaws. The bilinear curves are based on material tests in simulated reactor water and are provided for surface flaws. The material testing is mainly performed at 288 ^{o}C with a cyclic frequency of 0,017 Hz, ΔK levels of 10...60 MPa$\sqrt{m}$, and R ratios ($R = K_{min}/K_{max}$) of 0.2 and 0.7. The reactor vessel transients involve ΔK levels from near 0 to > 60 MPa$\sqrt{m}$, R ratios ranging from 0 to > 0.95, a very wide spectrum of cyclic frequencies extending well below 0.01 Hz, various water chemistries and temperatures as well as various material compositions and microstructures (Jones, 1984). Because there are so many variables affecting crack growth, the curves should be conservative compared to limited laboratory test results. However, accurate predictions to the component service need in addition to more extensive laboratory test data a mechanistically-based understanding of environment sensitive cracking.

The importance of metallurgical variables of steels is now clear based on laboratory test results. Steel's sulfur content and, especially, the MnS-inclusion size, shape and distribution seem to be responsible to material-to-material and heat-to-heat variability. Large elongated MnS-inclusions generally contribute to rapid crack growth rates, whereas materials containing small spherical MnS-inclusions are less susceptible (Hänninen et al., 1983). The adverse influence of sulfur is thought to result from the dissolution of uncovered MnS-inclusions inside the crack, which creates an aggressive local crack tip environment, Fig. 3. The sulfur species produced (H_2S, HS^- etc.) are known both to enhance hydrogen absorption and to increase anodic dissolution of the steel.

The mechanistic study of corrosion fatigue of pressure vessel steels has covered e.g. fractography, stress corrosion cracking, hydrogen embrittlement, crevice chemistry and electrochemistry, repassivation studies, and surface science, but it is still the general conclusion that the two relevant crack growth mechanisms - film rupture/slip dissolution and hydrogen-induced cracking - are

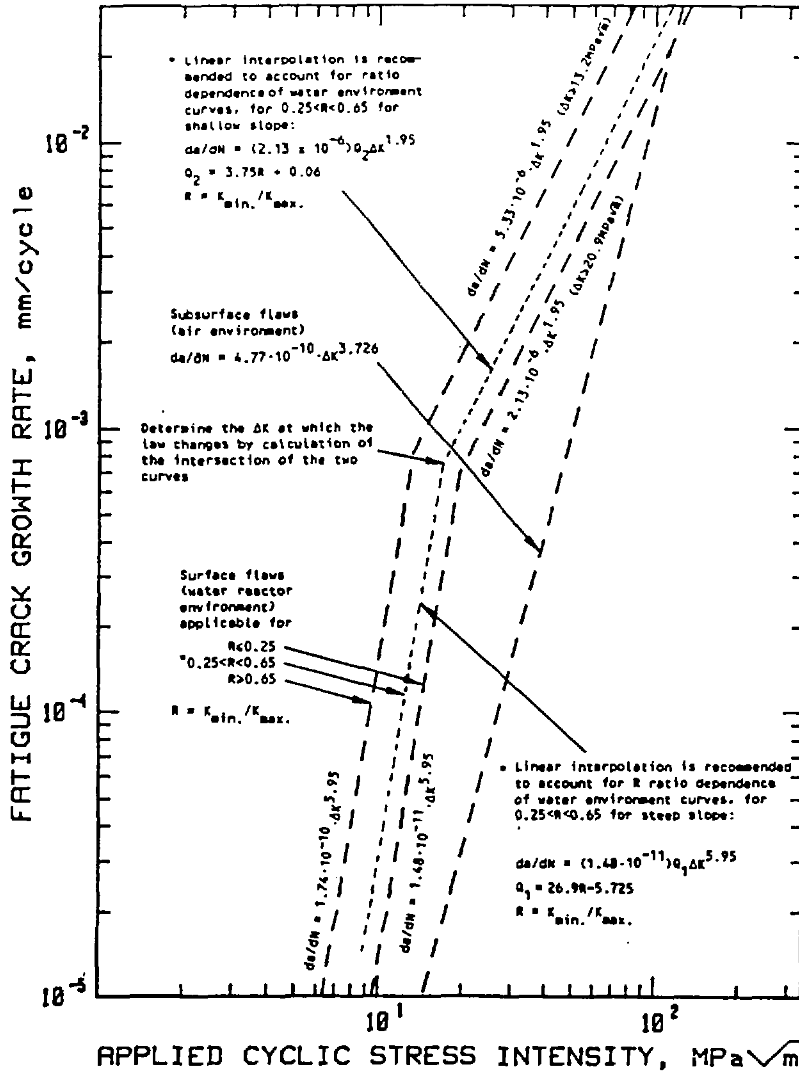


Fig. 2. The ASME Code crack growth rate reference curves for reactor pressure vessel steels.

not easily distinguished since they both depend on the same crack growth rate limiting processes: transport of species in solution, oxide film rupture and repassivation (Ford, 1982). Ford and Emigh (1984) have advanced the anodic dissolution model for the stage where they can calculate the maximum environmental enhancement of crack growth quantitatively. The model has still many unproven assumptions of which the most important is fracture morphology. The hydrogen-induced cracking model awaits an improved understanding of the factors that control the amount of crack growth per cycle.

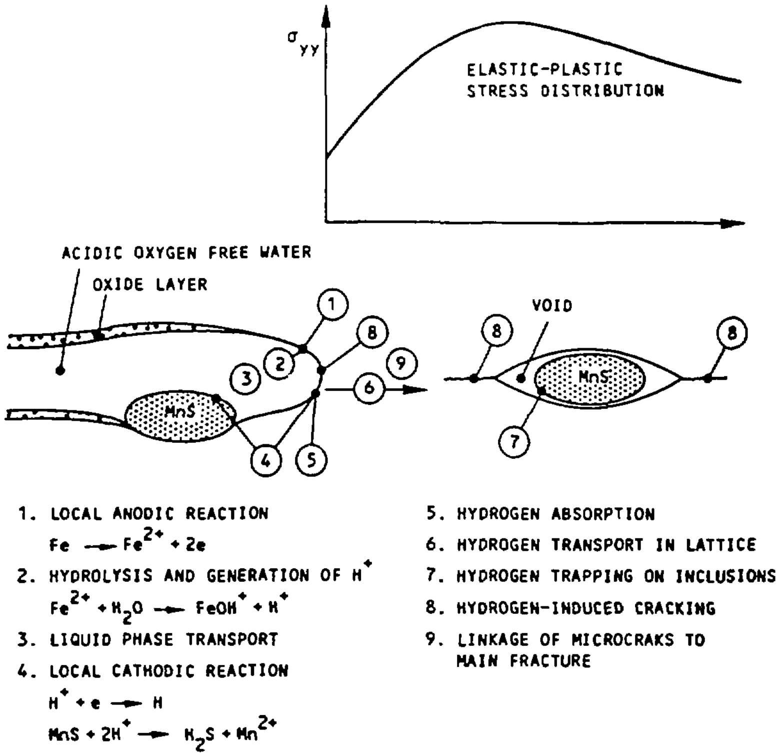


Fig. 3. Schematic illustration of effects of MnS-inclusions of the steel in cyclic crack growth of reactor pressure vessel steels in high-temperature, high-pressure reactor water (Hänninen et al., 1983).

There has also been cracking in an uncladded steam generator shell of a PWR (Czajkowski, 1984), in steam generator feed water piping of PWRs (Enrietto et al., 1981), in carbon steel piping of BWRs (Hickling and Blind, 1984) and in feedwater tanks of both nuclear reactors and conventional power plants (Hickling, 1982), which are mechanistically relevant to pressure vessel steel work.

The primary cause of A106-B or A106-C steam generator feedwater pipe cracking is a thermal fatigue mechanism induced by thermal stratification during low flow conditions during plant start-up. Temperature differences of the order of 120 °C have been measured from feedwater lines between the top and the bottom under low flow conditions. The cracks have had small width to depth ratios and "leak before break" behaviour has been demonstrated for all the cracks encountered (Stahlkopf, 1981). Modifications in geometry of the feedwater lines, modified operating conditions, and installation of thermal diffusers are expected to reduce the susceptibility to cracking.

In the case of steam generator shells and feedwater tanks cracking has occurred mainly at circumferential welds which show environment sensitive cracking also in slow strain rate tests, thus suggesting the stress corrosion cracking mechanism. This cracking is enhanced by oxygen or other oxidizing species in the secondary water from the condenser inleakage. In the case of A302-B PWR steam generator shell cracking the metallurgical structure of inadequately stress relieved weld (Scott, 1984) may have affected cracking. In the case of feed water tank cracking the susceptible material, a MnNiV steel, has been changed to a lower strength 15 MnNi 63 steel or a low carbon manganese steel, when no hardening in the heat affected zones (HAZ) took place, and crack initiation was prevented.

The medium-strength, low alloy steel 17 MnMoV 64 and relatively high-strength, fine-grained, structural steel 22 NiMoCr 37 used in Germany for BWR-piping and reactor vessel nozzles have suffered cracking which is termed strain-induced corrosion cracking (SICC) (Hickling and Blind, 1984). This kind of cracking has caused circumferential cracking in the region of feedwater nozzles and at welds and axial cracking in pipe bends but also in straight sections of thin-walled piping in German BWRs. SICC refers to those corrosion situations, in which the presence of localised, dynamic straining is essential for crack formation to occur, but in which cyclic loading is either absent, or is restricted to a very low number of infrequent events. Since the critical, absolute strain for SICC appears to lie around, or somewhat above, the elastic limit for the ferritic materials considered, it would appear that localised slip process in the metal, and not just the brittle fracture of a protective oxide film are required for cracking. The high content of dissolved oxygen seems to be an important factor. Oxygen leads to the formation of mixed magnetite/hematite oxide films on low-alloy steel surfaces. Hematite is not so ductile as magnetite and may lead to easier crack initiation upon straining. Oxygen can also maintain critical conditions for occluded cell corrosion. Material effects appear to be of less importance than the mechanical and environmental parameters, but the size, distribution and orientation of MnS-inclusions may be more important than the material composition itself.

Strain-induced corrosion cracking has caused a serious guillotine break of a pipe reducing part between pressurizer and main circulation line in a German HDR-plant. On the inside surface of the pipe there initiated 60 corrosion cracks which grew almost entirely through the cross-section of the pipe and the break occurred in the thermal shock test. The thickness of the reducing part was far less than specified and the dissolved oxygen content was high (8 ppm) (MPA, 1984). The mechanical stresses and environment together led to the situation, where the "leak before break" -criterion was not valid.

SICC of low alloy steels can occur in the lines carrying nearly stagnant steam or non-degassed condensate during normal operation, in feedwater nozzles and adjacent sections of horizontal piping, if thermal stratification can occur and in thin-walled piping and pipe bends which can undergo significant, localized mechanical loading. Remedies against SICC are reduced dissolved oxygen levels, drainage and maintenance of low flow rates in stagnant lines and avoiding thermal stratification. Longer term measure can be the use of increased wall thickness of pipes through the use of lower strength steels of higher toughness.

3. STAINLESS STEEL PIPING

Incidents of intergranular stress corrosion cracking (IGSCC) in the weld heat affected zones (HAZ) of AISI 304 and 316 stainless steel reactor pressure vessel nozzle safe-ends and piping have occurred in several BWRs Cracking is predominantly of a circumferential orientation although some axial-oriented cracks have occurred. First the cracking was confined to small diameter pipes of the recirculation loop by-pass lines and core spray lines. In the IGSCC pipe cracking incidents there have been no occurrences of pipe severance. Based on the field data and the stress analysis of the pipe behavior, the concept of "leak before break" was evolved and it continues to be valid. It postulates that leaks should develop in the cracked HAZ before the ductile austenitic stainless steel looses its structural integrity.

In recent years the number of world-wide pipe cracking incidents has increased significantly. Moreover the cracking has extended to large diameter pipes of both AISI 304 and 316 stainless steels. In some plants the cracking has been extensive. Due to generic intergranular stress corrosion cracking problems a BWR in Germany (Grundremmingen) has been decommissioned (Kussmaul, 1984) and at least two BWRs in USA and one in Sweden have replaced the primary recirculation piping with more resistant material AISI 316 NG (0.02 wt. % C and 0.06...0.1 wt. % N). The pipe cracking incidents until June 1983 are presented in Fig. 4 showing the frequency of pipe cracking versus pipe diameter. During last years cracking of large diameter piping has occurred in the pipes > 510 mm, which has raised considerable safety concern.

The BWR pipe cracking has occurred in the sensitized zones of AISI 304 and 316 type stainless steel weldments. After welding the degree of sensitization is generally low, but it can increase during operation due to low temperature sensitization (LTS). The development of a sensitized microstructure in LTS is shown in Fig. 5 (Kekkonen et al., 1984). However, it is not clear how critical parameter sensitization really is since low carbon non-sensitized AISI 316L steels have also suffered IGSCC in USA and Sweden, and since in many cases cracking in the HAZ is occurring in the outer edge of the sensitized zone where the degree of sensitization (DOS) is low but the residual tensile stresses are high. Thus, cracking is not unambiguously locating in the areas of maximum DOS. The grain boundary segregation and crevice conditions may also give rise to IGSCC in the austenitic stainless steels. This is enhanced by irradiation in the case of core components.

The peak tensile stresses in the HAZ of the pipe welds are a sum of operating, fabrication, fit-up and weld residual stresses, which together may exceed the yield stress of the austenitic stainless steel. Stress related pipe cracking remedies are concentrated to decrease the tensile stresses or to produce compressive residual stresses on the pipe inside diameter, Table 2.

The third necessary condition for BWR IGSCC is the presence of a certain amount of oxygen in the coolant. In general, keeping the amount of oxygen in the coolant low enough IGSCC is inhibited, but the exact level depends on the conductivity of the water and also on cyclic stresses. The electrochemical potential depends on both oxygen content and impurities. To prevent crack formation in BWR

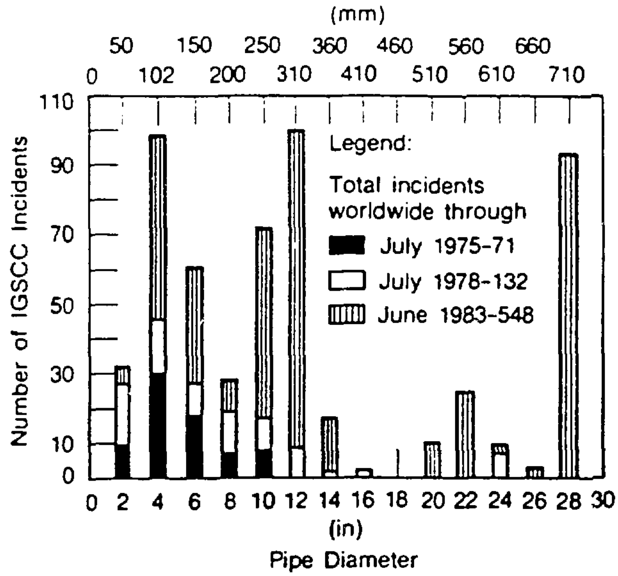


Fig. 4. World-wide frequency of BWR pipe cracking incidents as a function of pipe diameter (Danko, 1984).

stainless steel piping the electrochemical potential in the cooling water has to be kept more negative than about -350 mV_{SHE}, Fig. 6. Under normal conditions of 100...300 ppb dissolved oxygen the electrochemical potential in BWR water is -100...+100 mV_{SHE} which supports stress corrosion cracking. Therefore, start-up deaeration is not considered a pipe cracking remedy. If the conductivity of the water is below 0.3 µmho/cm, the amount of oxygen below 20 ppb is sufficient to keep the potential on the right level and to inhibit IGSCC (Roberts et al., 1984). However, if the water contains powerful oxidative impurities like chromates not even 5 ppb O_2 is low enough to inhibit IGSCC (Ljungberg and Korhonen, 1984), which it otherwise does. The low oxygen levels can be reached with hydrogen additive alternate water chemistry (AWC) which in conjunction with impurity control has the potential to mitigate IGSCC of pipe welds. It seems that hydrogen addition in conjunction with impurity control will become in wide-spread use in BWR practice for preventing and mitigating cracking in piping. In PWRs hydrogen has always been added to water to reduce oxygen.

A number of pipe cracking incidents have also been attributed to cold-work in AISI 304 steel. Cracking can in these cases be mixed or mainly intergranular. An example of this type of cracking is Oskarshamn AISI 304 elbows in shut-down cooling and clean-up system (Danko, 1984). In cold bending the inner surface deformation (15...20 per cent) produced some α'-martensite, which initiated axial cracking in the pipe bend in the absence of sensitization. The crack propagated through the elbow and caused a leak. Extensive cold-work should therefore be avoided in the primary circuit components.

A trough-wall leaking crack was detected in a recirculation-inlet nozzle safe-end weld at the Duane Arnold plant. The safe-ends were made of Inconel 600 and all safe-ends of the plant showed cracking essentially completely around the circumference. This plant had marked resin intrusions which lowered the pH and increased the conductivity. This is the only marked case of Inconel 600 SCC in BWRs and this case emphasizes especially the need for good water quality control.

Compared to IGSCC fatigue cracking accounts for only a small percentage of the total pipe cracks in BWR stainless steel piping. Thermal fatigue cracks have been reported in Sweden and in Finland. The cracks are transgranular and they can be prevented by modifying the pipe system design to minimize the thermal gradients. These cracks may be a problem in areas which do not belong to current inservice inspection programs and thus for early identification of leaks a good leak detection system is important.

Both centrifugally cast CF8A and CF8M and forged pipes of these alloys have been commonly used for large primary piping and other pressure containment applications both in BWRs and PWRs. PWRs use also ferritic steel A516 Grade 70 pipes clad with Type 308 L austenitic stainless weld overlay. IGSCC has not been observed in these pipings but thermal fatigue can be a possible failure mechanism. Inspection of cast stainless steel components is, however, a major problem due to the coarse grain structure.

Table 2. Summary of the BWR pipe cracking remedies and their implementation (Danko, 1984).

REMEDIES	OBJECTIVE
I. SENSITIZATION RELATED	
A. SOLUTION HEAT TREATMENT	REMOVE WELD SENSITIZATION AND RESIDUAL STRESSES
B. CORROSION RESISTANT CLAD	PROTECT THE HAZ FROM ENVIRONMENT
C. ALTERNATE PIPE MATERIAL	ELIMINATE WELD SENSITIZATION
II. STRESS RELATED	
A. HEAT SINK WELD	PROVIDE COMPRESSIVE RESIDUAL STRESSES ON PIPE INSIDE DIAMETER AND PARTIAL THROUGHWALL
B. INDUCTION HEATING STRESS IMPROVEMENT	DITTO
C. LEAST PASS HEAT SINK WELDING	DITTO
D. WELD OVERLAY*	DITTO
E. FLAWED PIPE ANALYSIS*	PERFORM STRESS ANALYSIS TO DOCUMENT REMAINING OPERATING LIFE OF PIPE
III. ENVIRONMENT RELATED	
A. STARTUP DEAERATION	REDUCE DISSOLVED OXYGEN CONTENT IN COOLANT DURING STARTUP
B. HYDROGEN WATER CHEMISTRY	REDUCE STEADY STATE OXYGEN CONTENT IN COOLANT
C. IMPURITIES IN COOLANT	ESTABLISH LIMITS OF IMPURITIES FOR SAFE OPERATION

*FOR INTERIM USE IN OPERATING PLANTS

Table 3 provides a cost estimate of implementation of the three major pipe cracking remedies: alternative water chemistry with hydrogen addition, induction heating stress improvement (IHSI) which produces compressive stresses inside the pipe surface and partially also through wall, and complete piping replacement (Giannuzzi et al., 1984). The replacement power and other implementation costs have to be added to these figures. It can also be seen that replacement operation results in very high radiation exposure to personnel.

Table 3. Estimated costs of three major IGSCC countermeasures (Giannuzzi et al., 1984).

REMEDY	LABOR & MATERIALS	EXTENDED OUTAGE	MAN-REM
Alternative water chemistry with H_2addition	2 M$ + 1 M$/Yr	2 weeks	Low during installation
IHSI	4 M$	5 weeks	100 Man-Rem
Replacement of piping	50 M$	6 months	500 Man-Rem

A number of incidents of pipe cracking has been reported also in low pressure, stagnant borated water systems of pressurized water reactors. Cracking has occurred in the heat affected zones of AISI 304 stainless steel pipe welds. Investigations of the borated water system pipe cracks have shown that cracking is intergranular and it occurs in a similar pattern to pipe cracking in BWRs. The major difference between BWR pipe cracking and PWR spent fuel pool pipe cracking is in the environment. The steady-state BWR environment consists of high purity water at 288 ^{o}C containing dissolved oxygen while the PWR spent fuel pool environment normally consists of approximately 13000 ppm boric acid at about 65 ^{o}C. Sulfur containing species have been shown to be the cause of IGSCC in air saturated boric acid solution (McDonald et al., 1982). Thiosulfate and tetrathionate anions lead to cracking of sensitized AISI 304 stainless steel and the potential range over which IGSCC occurs corresponds to a region for the metastable sulfur oxyanions in which thiosulfate and tetrathionate are capable of being reduced to elemental sulfur. A strong synergistic effect exists between thiosulfate and chloride. When mixed together the thiosulfate and chloride produced more pronounced IGSCC than either thiosulfate or chloride separately. This problem was solved by maintaining high purity water chemistry and using AISI 304 L steel to prevent sensitization.

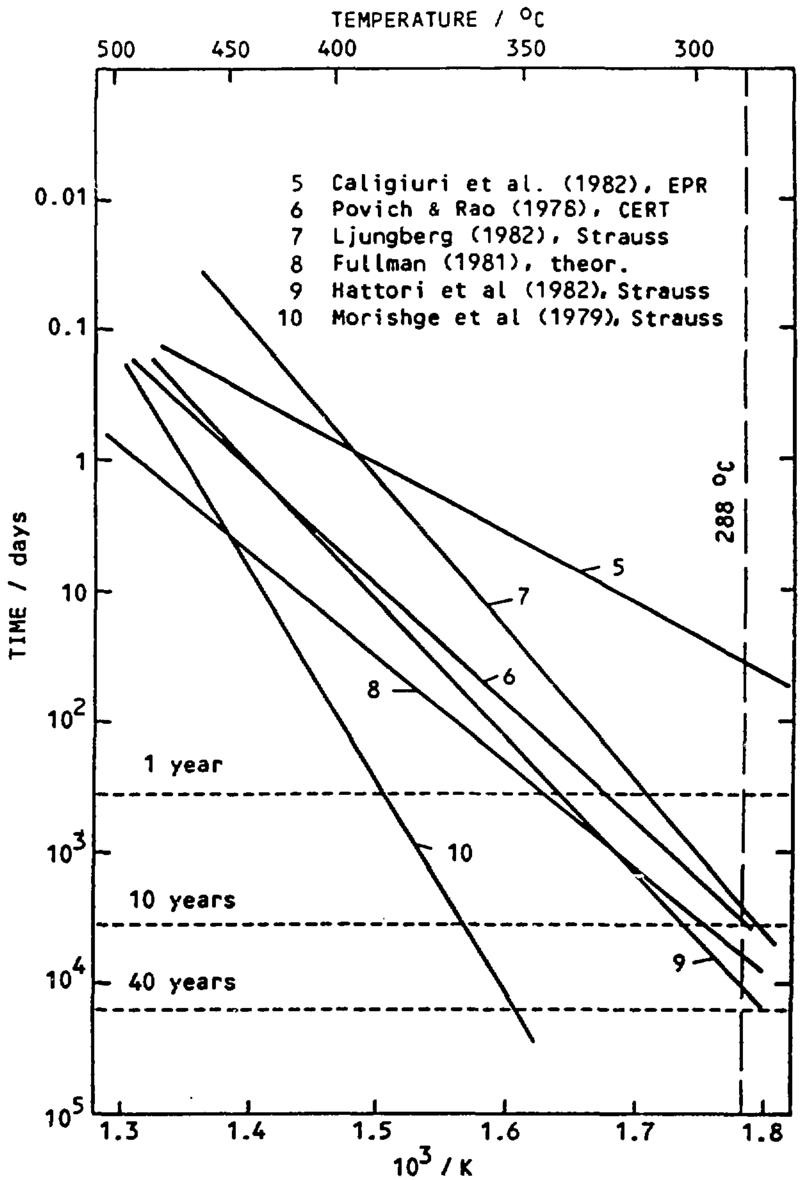


Fig. 5. Several (T, t) dependences for LTS of AISI 304 steel showing wide scatter of various tests. The scatter is based on the various starting conditions used as well as test methods (Kekkonen et al., 1984).

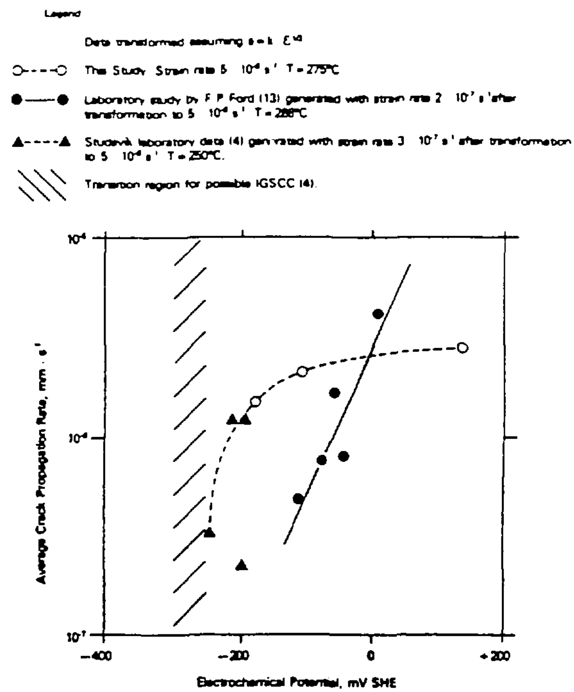


Fig. 6. Variation of average IGSCC crack propagation rate with corrosion potential for sensitized AISI 304 steel indicating a critical potential for IGSCC (Ljungberg and Korhonen, 1984).

4. STEAM GENERATOR MATERIALS PROBLEMS

Steam generators of a typical PWR contain 150...500 km of heat transfer tubing which forms a thin-walled pressure boundary between the primary and secondary systems. The steam generators have an excess heat transfer area of about 20 per cent. If the steam generator tube failure rate is higher than 1 per cent per effective full power year, the reactor with design life of 30 years at a capacity factor of 80 per cent will consume the excess surface area of 20 per cent before the reactor has reached the design life and replacement of steam generators may be required. The major causes of steam generator tube failures since 1972 are summarized in Fig. 7 (Tatone and Pathania, 1985). The diagram shows that considerable progress has been made in arresting steam generator tube degradation arising from some mechanisms, but new mechanisms and locations have emerged. The major causes of tube failures in 1982 were primary side SCC, secondary side SCC and intergranular attack (IGA) and pitting corrosion. Other failure mechanisms include phosphate wastage, thinning, denting, damage by loose parts or foreign objects on the secondary side, corrosion fatigue and fretting wear. By the end of 1982, roughly 2 per cent of the 1,6 million tubes in service have been plugged and of these 76 % are made of Inconel 600, most of them in the mill-annealed condition. Seven reactors have replaced their steam generators and one reactor has recently decided to do so (Ringhals 2, Sweden). The cost for replacement of steam generators approaches 200...400 million $ per plant and the cost of replacement power during outage may run as much as 800.000 $ per day. The steam generator tube leakages raise also safety issues of which multiple tube rupture is the principal one.

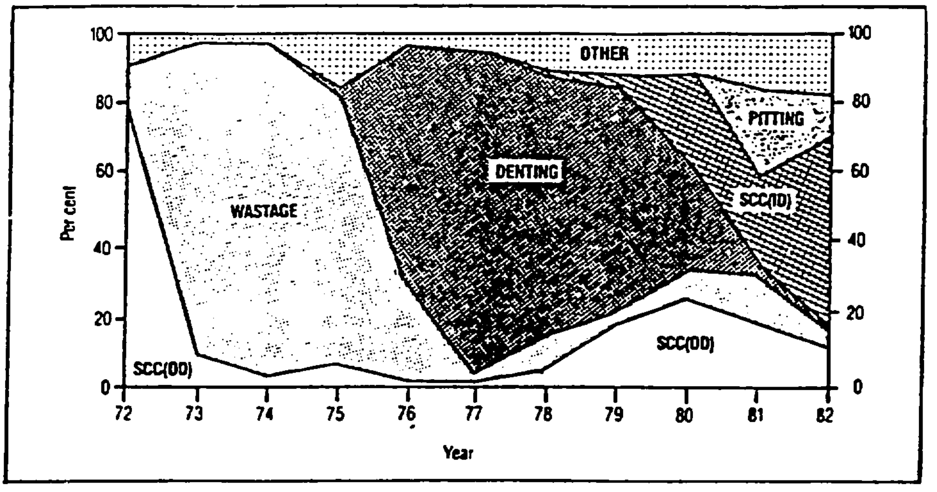


Fig. 7. History of steam generator tube failures. SCC (ID) is primary side stress corrosion cracking, SCC (OD) is secondary side stress corrosion cracking or intergranular attack (Tatone and Pathania, 1985).

Figure 8 shows a typical recirculating steam generator with locations of the problem areas. The sludge pile on the top of the tubesheet is an important area for corrosion. The exact type of corrosion is dependent on the secondary water treatment, contaminating chemicals and the mode of plant operation. The most corrosive regions are in the center of the sludge pile where alternate wetting and drying occurs. This leads to accumulation of corrosive chemicals on the surface of the tubes. The local wastage corrosion occurs due to acid phosphate environment in this area in the reactors where phosphate secondary water chemistry is used. Whereas stress corrosion cracking and intergranular attack from the secondary side are wide-spread problems in the deep tube-to-tube sheet crevices. SCC/IGA is associated with local alkaline environments. It is interesting to note that local, stagnant, acid and alkaline environments may exist even in the different regions of the same steam generator (Tatone and Pathania, 1985). It is not sure if removal of sludge by mechanical or chemical means will arrest these modes of tube damage. A near-term solution to SCC/IGA is in addition to tube plugging to install sleeves within the affected tubes, but their long term integrity remains to be demonstrated. It is clear that the condenser and air inleakages are to be minimized in addition to full-flow condensate polishing in order to avoid this type of steam generator corrosion.

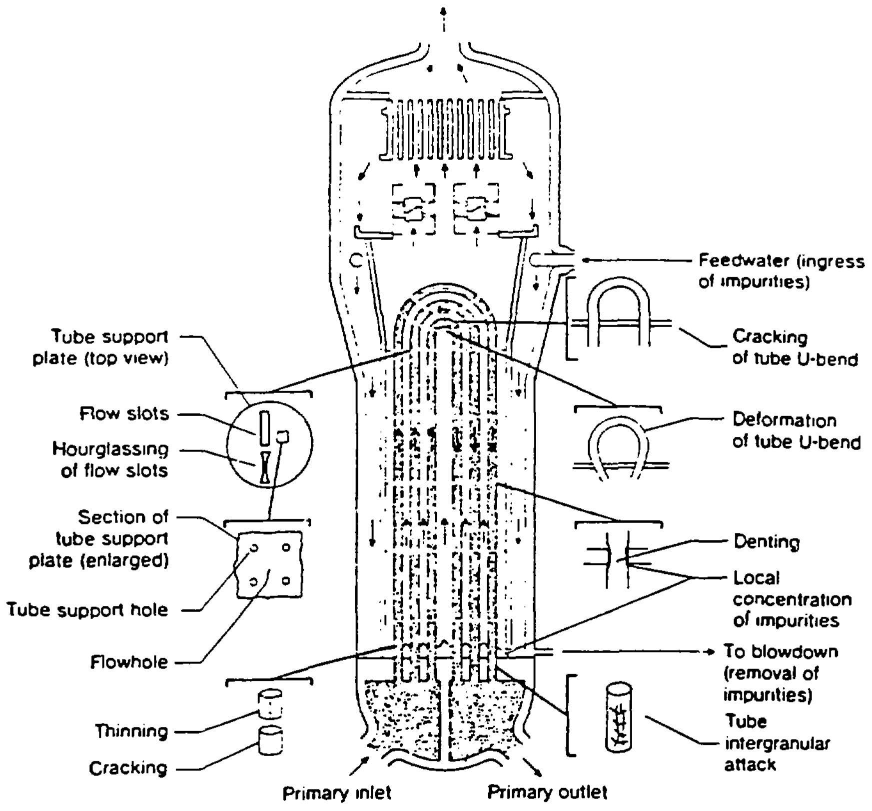


Fig. 8. Schematic picture of a recirculating steam generator with problem areas, where many, but not all units, have had various forms of corrosion (Green and Paine, 1981).

Primary water initiated cracking has been confined to inner row U-bends, severely dented tubes in tube support plate crevices, and in roll expanded areas of the tubesheet crevice. In general, multiple cracks are observed suggesting the pattern of residual stresses e.g. from the rolling operation at the roll transition zone.

In general, it is known that there is a great heat-to-heat variability to SCC in Inconel 600 tubing. Analyses of tubes taken from Ringhals 2 steam generators show that carbide distribution is of great importance in tube cracking: cracking is found in tubes with intragranular carbides while tubes with grain boundary carbides are free of cracking. There was also some indication that low carbon tubes have a better crack resistance (Engström and Norring, 1984), which is against the general opinion. Paine (1982) has summarized the factors which have often been assumed to contribute to SCC of Inconel 600 in operating steam generators:

- high hardness, high strength, low carbon content tubing,
- high degree of cold work in the bending or roll expansion,
- ovalization of tubes during bending,
- residual and operating stresses approaching the yield stress, and
- heat-to-heat factors (e.g., metal composition, grain boundary condition, heat treatment, etc.).

The above five factors must be taken into consideration together with local environment in order to assess cracking susceptibility.

Denting corrosion is caused by corrosion of carbon steel support plates and laminar growth of magnetite (Fe_3O_4) in the tube-to-tube support plate crevice. This corrosion is enhanced by chlorides, copper, and oxygen, i.e. inleaking impurities. The growing oxide is causing plastic deformation and distortions of the Inconel 600 tubes, which often leads to SCC from the inside surface of tubing. Denting can be arrested by minimizing condenser leaks and air inleakage e.g. by replacing copper-bearing components such as condensers with more resistant tube materials like titanium, by chemical additives like boric acid, or calcium hydroxide or by complete replacement of steam generators or tube bundles which involves also the change of support plate material into stainless steel.

Intergranular attack (IGA) occurs in the tube-to-tube sheet crevice region in the presence of caustic environment. Also sulfur species like polythionates, thiosulfates, sulfides etc. can cause IGA of sensitized Inconel 600 tubing very rapidly even at lay-up conditions (Paine, 1982).

5. HIGH STRENGTH MATERIALS DEGRADATION

The age-hardenable nickel-base alloys such as Inconel X-750 and Inconel 718 and stainless steel A-286 were developed originally for high temperature applications, with good mechanical properties and excellent resistance to high temperature oxidation. For Inconel X-750 numerous heat treatments developed for the special high temperature applications have been used also for the different structural parts in LWRs. The experience with Inconel X-750 in reactors has been contradictory. The known cases of SCC failures in LWRs are listed in Table 4 (McIlree, 1983). Some components have cracked without affecting plant operation, while failures of some components have caused shut-downs and have affected plant availability. Loose parts of broken components have affected e.g. the movement of control rods in BWRs, and caused extensive damage to the primary surface of the steam generator tubesheet in PWRs.

The components failed have generally been creviced and highly stressed. They have also contained geometric discontinuities where stresses have intensified. However, it is clear that cracking can occur below yield stress in these alloys and probably a maximum sustained stress level below 0,5 x yield stress or even less may be required to avoid cracking for sure.

Numerous recent studies have identified that a high temperature annealing (1060...1150 °C) plus single aging at about 700 °C improves the SCC resistance of Inconel X-750 in BWR and PWR environments. This heat treatment produces severe Cr-depletion on the grain boundaries, like in the case of thermally treated Inconel 600 (Kekkonen and Hänninen, 1983 and 1984). Especially in France the guide tube support pins having the new heat treatment have been changed in most reactors during normal maintenance shut-downs. Thus, the general trend when replacing an Inconel X-750 component is not alteration of the material but the heat treatment procedure and also redesigning the component to lower stress levels and concentrations (Roberts, 1982).

Inconel 718 has been used much less in reactor internal components and it has not experienced IGSCC, but it has not replaced Inconel X-750 mainly due to its fatigue properties. A-286 has shown considerable susceptibility to IGSCC in BWRs in Sweden and Finland, but no heat treatment improvements have been carried out and the failed components have been changed to AISI 316L steel in case of core grid screws and to a low alloy steel in case of cover beams (Bengtsson and Korhonen, 1983).

Table 4. Reactor components of age-hardenable austenitic alloys which have experienced cracking (McIlree, 1983).

ALLOY	COMPONENT	REACTOR TYPE	INITIATION MODE
X-750	BOLTS:		
	- CORE BAFFLE	PWR	IGSCC
	- FUEL ASSEMBLY	BWR	IGSCC
	PINS:		
	- GUIDE TUBE SUPPORT	PWR	IGSCC
	BEAMS:		
	- JET PUMP	BWR	IGSCC
	SPRINGS:		
	- CRD SEAL	BWR	IGSCC
	- FUEL ASSEMBLY HOLDDOWN	PWR	FATIGUE
	- FUEL ASSEMBLY FINGER	BWR	IGSCC
718	SPRINGS:		
	- CONTROL COMPONENT HOLDDOWN	PWR	FATIGUE
A-286	BOLTS:		
	- THERMAL SHIELD	PWR	IGSCC/FATIGUE
	- FUEL ASSEMBLY	BWR	IGSCC
	- CORE BARREL	PWR	IGSCC
	BEAMS:		
	- STEAM SEPARATOR/DRYER HOLDDOWN	BWR	IGSCC

The most important applications of threaded fasteners are those constituting an integral part of the reactor coolant pressure boundary, such as pressure retaining closures in reactor vessels, pressurizers, reactor coolant pumps, and steam generators. Many of theses failures have been caused by erosion corrosion in PWR primary water leaks. Boric acid is significantly acidic at low temperatures and together with high velocity jet of leaking primary water severe wastage of ferritic materials is occurring (Scott, 1984). The use of sulphur bearing compounds (molybdenum disulfide) in the thread regions may have contributed to the higher service temperature failures, where sulphur-bearing lubricants and moisture can cause highly aggressive compounds (H_2S, H_2SO_4) to be formed. Laboratory experiments have shown a pronounced embrittling effect on carbon and low alloy steels when the material is in contact with molybdenum disulfide in a steam environment. The cracking has also occurred in steam generator manway studs, which were exposed to leaking borated water and Furmanite, a sealing compound containing leachable sulfur, fluorine, and chlorine, which are known promoters of stress corrosion cracking. By proper control for sealant compounds and fasteners lubricants together with avoiding leaks at flanges by improved gasket design fasteners susceptibility to stress corrosion cracking can be minimized (Koo, 1983).

Failures of high strength support boltings have been reported by a number of plants. The bolt failures have primarily occurred in pressurized water reactors in both ambient and elevated temperature environments by SCC, and most of the cracking incidents involved materials with measured hardness levels above the specified. Susceptibility to SCC in high strength maraging steels and low alloy steels (AISI 4340 and 4140) with higher strength levels is generally thought to be due to hydrogen embrittlement (Cipolla et al., 1984). The overly hard condition of supporting bolts was attributed to improper heat treatment and to inadequate quality control. In all failures moisture was present and it was linked to borated coolant leakage. In most failures the bolts were installed with very high preloads. The detailed instructions for heat treatment, removal and cleaning of bolts and for tensioning technique are needed to avoid support bolting degradations.

6. DISCUSSION

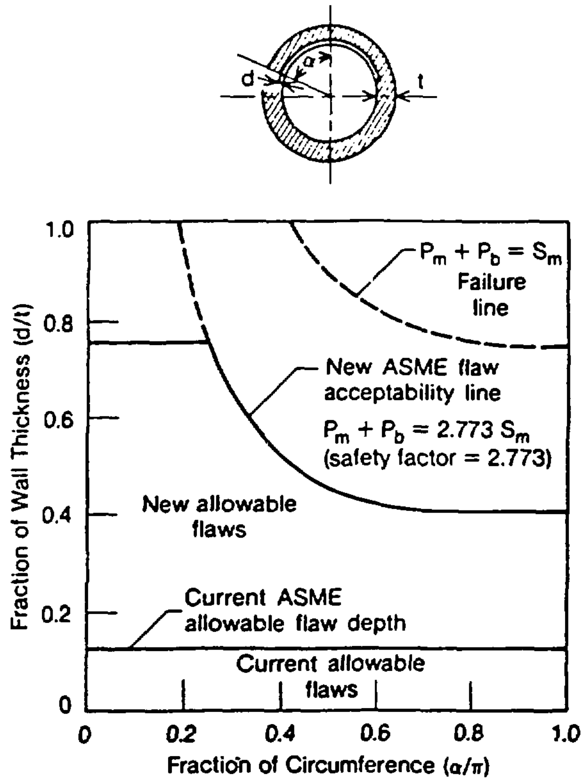
The review of environment sensitive cracking and corrosion phenomena of operating LWRs shows both some important generic failure modes and failures which have been caused by rare incidents when quality control of materials, fabrication or operating conditions have been inadequate. These failures have received very detailed attention and remedial measures have developed, which rely on improvements in materials, environmental control and monitoring as well as nondestructive examination techniques.

IGSCC has been detected by leaks, ultrasonic tests and dye penetrant or radiographic examinations. Since IGSCC cracks are very tight and branched at the crack tip, the cracks are very hard to detect by ultrasonics and it is even more difficult to determine the depth accurately (Weeks, 1983). Most cracks have been found after a through-wall crack was detected in the same system. Most of those leaking cracks have been detected visually rather than trough remote leak detection systems. This is largely due to the tightness of the cracks, which results in very small leaks even for relatively large cracks. This reduces the margin for leak-before-break criterion, which can be improved by more sensitive leak detection systems, like moisture sensitive tapes and acoustic-emission systems, which are still under development. Assurance of leak-before-break in small-diameter lines is good. Because of uneven stress distribution cracks do not generally propagate fully around the cirumference. In large-diameter lines, cirumferential propagation is favoured by residual stress distribution and 360° indications have been observed although only one circumferential through-wall crack has been found (Weeks, 1983), which leaked well before unstable crack growth. However, this casts some doubts for leak-before-break philosophy and the margins for leak-before-break need to be defined.

Until recently, flaws in austenitic stainless steel piping greater than 10 % of the pipe wall thickness were required to be repaired by the ASME Boiler and Pressure Vessel Code. New critical crack sizes are calculated based on net section plastic collapse criterion of the ductile austenitic stainless steel piping. A diagram showing allowable flaw sizes is shown in Fig. 9. The collapse line for the pipe for normal and design conditions, the present acceptable ASME Code Section XI flaw size of 10 %, and proposed acceptance criteria boundary based on a safety factor of 2.77 are shown in this figure. The region of safe operation shows that a 360° circumferential crack with a depth of 40 per cent of wall thickness can be tolerated. In applying the new acceptance criteria, knowledge of the flaw size, especially the depth, the residual stress state, the operating stresses, and the crack growth rate must be available with reasonable accuracy.

Weld overlay involves the application of AISI 308L filler metal over the pipe weld that contains the indication of flaw. Water flowing is maintained during welding inside the pipe in order to prevent sensitization and to produce compressive residual stresses to arrest the existing cracks. Weld overlay is a remedial method for limited time at present (Danko, 1983). This is because it makes the ultrasonic testing difficult and cracks can even propagate in the weld metal with very high stress levels. Weld overlay technique can become a permanent measure, but more demonstration work is needed.

Corrosion of PWR steam generators was shown to result from a complex interaction of water chemistry, thermal and hydraulic design, materials selection, fabrication methods, and secondary plant materials, designs and operations. Even though more resistant materials will be taken into use, the main maintenance and operation task will be prevention of impurity inleakages into secondary water which the availability of full-flow condensate polishing markedly enhances. When the secondary water chemistry is maintained within strict purity limits even stainless steel steam generators are working without corrosion problems like in Finnish PWRs.



P_m = primary membrane stress
S_m = design stress (= the lower of Su/3 or 0.9 x Sy, where Su is the ultimate tensile stress and Sy is the yield stress)
Pb = applied bending stress

Fig. 9. Diagram showing allowable flaw sizes in austenitic stainless steel pipe (Danko, 1984).

New steam generator can incorporate several design improvements that can greatly reduce corrosion problems. These include thermal treatment of the Inconel 600 tubing (e.g. 704 °C for 12...15 h), which however sensitizes Inconel 600 to IGA in acid sulfur-bearing environments, alternate tubing alloys (Inconel 690) with better resistance to cracking, stress relief annealing of small-radius U-bends, alternate support plate materials such as ferritic or austenitic stainless steel, elimination of crevices especially in the tubesheet and stagnant flow areas which decreases sludge formation on the tubesheet (Green and Paine, 1981; Berge and Donati, 1981). Thus, no quick and easy solutions to these corrosion problems are available. Steam generators at some reactors have nevertheless performed very well without failures, which indicates that good performance is possible with good operating practice and in particular with good condenser integrity. However, substantial improvement is required if steam generator tube failures are to be eliminated.

For loose parts of broken internal components specific inspection and monitoring procedures have been developed and implemented to detect degraded components before failure can occur or before broken part is causing extensive damage in the system. The inspection of core components is a difficult task because of high radiation fields. It would be beneficial if the effects of heat treatment, metallurgical structure, irradiation and stress level on cracking susceptibility were known so well that there were no need for recurring inspection. The mechanistic understanding of SCC in age-hardenable nickel-base alloys would be needed for this to be possible.

Most of the bolting failure incidents have been discovered during refueling, scheduled inservice inspections, or maintenance repair outages. About 45 per cent of the reported incidents have involved bolting or threaded fasteners in the reactor coolant pressure boundary. Gross bolting failures possibly could lead to serious malfunctions or even failure of critical components. The possibility of a loss-of-coolant accident (LOCA) cannot be ignored in case of an extensive undetected bolting failure in the pressure boundary. The safety implications are of concern particularly since current ultrasonic testing methods in inservice inspection programs are not sensitive enough to detect hidden cracks in covered bolts. Wastage degradation due to borated water can be detected only by visual inspection of accessible bolts.

The primary focus of the maintenance filosophy is to be directed at maintaining the primary system integrity. The maintenance must be directed at treating symptoms and not at the causes of degradation. In some cases effective solutions would require major changes instead of simple corrective actions like in the case of PWR steam generators. These changes can arise only from fundamental understanding of the underlying mechanisms of degradation. Also the knowledge of the local environmental conditions which cause corrosion damage is needed. The high temperature electrochemical methods have enabled the characterization of conditions in experimental work. Now these electrochemical measurements (potential, pH etc.) can be introduced also to the plant conditions where the knowledge of local conditions is extremely valuable for understanding the occurred failures.

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