



## **WATER CHEMISTRY AND BEHAVIOUR OF MATERIALS IN PWRs AND BWRs**

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### **Abstract**

Water chemistry plays a major role in corrosion and in activity transport in NPP's. Although a full understanding of all mechanisms involved in corrosion does not exist, controlling of the water chemistry has achieved good results in recent years. Water chemistry impacts upon the operational safety of LWR's in two main ways: integrity of pressure boundary materials and, activity transport and out-of-core radiation fields.

This paper will describe application of water chemistry control in operating reactors to prevent corrosion. Some problems experienced in LWR's will be reviewed for the design of the nuclear heating reactors (NHR).

### **1. INTRODUCTION**

The IAEA co-ordinated research program entitled "Investigation on Water Chemistry Control and Coolant Interaction with Fuel and Primary Circuit Materials in Water Cooled Power Reactors (WACOLIN)" was organized and carried out from 1987 to 1991. The reports of this work summarize the present understanding on good coolant chemistry (IAEA Technical Reports Series No. 347 and IAEA-TECDOC-667):

"Good reactor coolant chemistry, corrosion control and minimum of activity build-up are indispensable for the optimum performance of nuclear power plants. Without these the system integrity may be jeopardized and the activity transport may create various problems".

For a nuclear power plant the capability to operate with an optimal chemistry regime is determined by the design including materials, construction and effectiveness of the water purification system. The choice of materials defines water quality requirements and dimensions of water treatment systems. Some features of design like the use of copper-base alloys or Inconels or carbon steels presents limitations for the optimum water chemistry. Therefore recommendations for water chemistry must be established together with the design and material specifications.

Reliable water chemistry specifications have been developed for the existing water cooled reactors taking into account the common material practice for high temperature operation ( $T > 250\text{ }^{\circ}\text{C}$ ); however, there is still room for improvement. Water cooled power reactor experience shows that even under normal operating conditions some undesirable phenomena can occur like stress corrosion cracking and corrosion fatigue, erosion corrosion or deposition of corrosion products on heat transfer surfaces.

## 2 PWR COOLANT SPECIFICATIONS

The goal of recommendations for chemistry in operating PWR's is to limit radioactive transport and thereby reduce out-of-reactor radiation fields. Radiolytic oxygen formation in PWR's is suppressed by maintaining an overpressure of hydrogen. Generally 25 - 30 cc/kg are required to keep oxygen below 5 ppb. Control of pH in PWR primary systems is complicated by the use of boric acid as a chemical shim to control nuclear reactivity, which results in a need to adjust lithium or potassium hydroxide content to avoid a continually changing pH through the fuel cycle.

Typical water chemistry specifications used in PWR's are shown in Tables 1 and 2 as well as in Fig. 1. The three options being consistent with the guidelines are:

- Elevated Li-B-Chemistry
- Modified Li-B-Chemistry
- Co-ordinate Li-B-Chemistry.

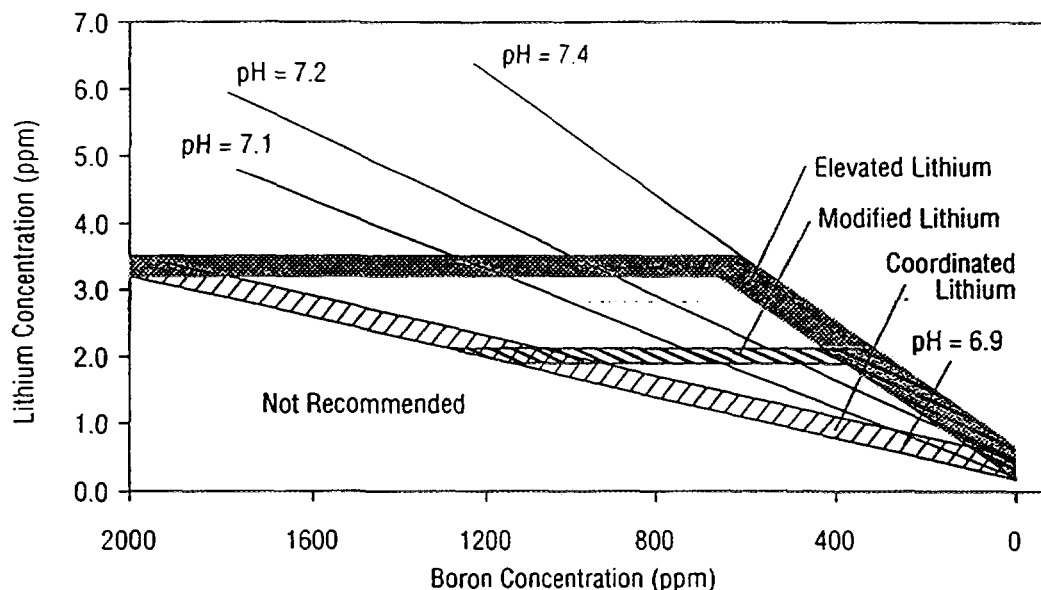


Fig. 1. PWR primary system chemistry control regimes (EPRI NP-7077 1990).

The operational experience of the last years has shown that the modified Li-B-Chemistry should be preferred over the other two options. Corrosion product transport and solubility has been lowest and the estimated relative activity build-up on cold regions of the primary circuit, as a result of precipitation and activation on hot regions and subsequent dissolution and redistribution throughout the system, is minimized (Riess 1993).

PWR primary water chemistry changes greatly at shut-down, when the coolant is oxygenated and borated during cooling period. This change will release activated corrosion products which can be collected by clean-up systems operating at full power.

The guidelines give a possibility to great variation especially concerning pH optimization. This optimization must be done plant-specific. Generic principles for optimization of primary system pH according to EPRI Guidelines (EPRI NP-7077 1990) are presented in order of priority as follows:

1. Operate at or above  $\text{pH}_{t(\text{ave})} = 6.9$  ( $t_{\text{ave}}$  = average of  $t_{\text{inlet}}$  and  $t_{\text{outlet}}$ ) to minimize crud deposition on fuel and enhanced Zircaloy oxidation. If such operation requires greater than 2.2 ppm lithium, the impact on heat exchanger materials should be assessed. If plant-specific considerations require operation below  $\text{pH}_{t(\text{ave})} 6.9$ , a fuel surveillance program should be considered.
2. For operation above 2.2 ppm lithium for the purpose of achieving  $\text{pH} > 6.9$ , plant-specific fuel and materials reviews should be performed, and a fuel surveillance program considered.
3. Once lithium has been reduced to  $2.2 \pm 0.15$  ppm consistent with principles #1 and #2, lithium can either be controlled so as to continue operation at  $\text{pH}_{t(\text{ave})} = 6.9$ , or maintained constant at  $2.2 \pm 0.15$  ppm until a specified  $\text{pH}_{u(\text{ave})}$  between 6.9 and 7.4 has been reached. The plant-specified pH should be selected on the basis-specific impacts on fuel and materials integrity and radiation field control. There are limited data available on which to base operation above  $\text{pH}_{u(\text{ave})} = 7.4$ . Therefore, it is recommended that  $\text{pH}_{u(\text{ave})} = 7.4$  ( $\pm 0.15$  ppm lithium) be considered the upper operating band.
4. Maintain the specific pH at  $\pm 0.15$  ppm lithium until the end of the operating cycle, noting that lithium variations have greater effect on pH at lower boron concentrations.
5. Attempt to minimize pH fluctuations during power operation. However, during power-level changes, some fluctuation in pH changes may be unavoidable.

These guidelines with specified pH values are applicable only for reactors operating at high temperatures and, thus, for nuclear heating reactors (NHR's) the specifications should be reviewed with respect to the fuel cladding failures, stress corrosion and corrosion product release phenomena shown in Fig. 2. The minimum solubility pH of magnetite at temperatures below 200 °C is  $\text{pH}_T > 7.5$ . This can be obtained by having the total Li-content  $< 2.2$  ppm depending on the desired boric acid content needed as a moderator. In Table 3 some examples of the amount of Li needed to obtain the coolant pH providing the minimum solubility of magnetite at different temperatures and boron contents are presented. However, the potential risk of material failures caused by using higher concentrations of Li (i.e., 2.2 ppm) than specified in the guidelines for reactors operating at high temperatures should be evaluated for temperatures to be used in NHR's.

With respect to the NHR's the choice of the BWR water chemistry raises questions concerning general corrosion and following activity levels and SCC problems if the oxygen content is not reduced, e.g., by using hydrogen addition. Anyway, BWR water chemistry has to be optimized also before NHR application as shown in Fig. 3.

Table 1. EPRI guidelines for primary coolant in PWR's (EPRI NP-7077 1990).

Hydrogen (cm <sup>3</sup> ) (STP)/kg H <sub>2</sub> O <sup>a)</sup>	25 - 50
chlorides (mg/kg)	< 0.15
fluorides (mg/kg)	< 0.15
dissolved oxygen (mg/kg)	< 0.01
lithium (mg/kg)	consistent with station lithium program

<sup>a)</sup> STP, standard temperature and pressure (0 °C, atm)

Table 2. Specifications of reactor water quality for PWR's of VVER-440 and VVER-1000 type (Rieß 1993).

Indicator	Values	
	VVER-440	VVER-1000
pH (25°C)	6.0 - 10.2	5.7 - 10.2
K <sup>+</sup> , L <sup>+</sup> , N <sup>+</sup> (mmol/kg) (depending on H <sub>3</sub> BO <sub>3</sub> concentration)	0.05 - 0.45	0.05 - 0.45
NH <sub>3</sub> (mg/kg)	> 5.0	> 5.0
hydrogen (cm <sup>3</sup> /kg)	30 - 60	30 - 60
chlorides and fluorides (µg/kg)	≤ 100	≤ 100
H <sub>3</sub> BO <sub>3</sub> (g/kg)	0 - 9.0	0 - 13.5
oxygen (µg/kg)	≤ 5	≤ 5
copper (ng/kg)	< 20	< 20
iron (ng/kg)	< 200	< 200

Table 3. Required Li concentrations in the coolant with constant boron concentration (1000 ppm as B) needed to obtain the pH of minimum solubility for magnetite at different temperatures.

Li / ppm	B / ppm	pH <sub>T</sub>	pH <sub>T</sub> for solubility minimum
3.8	1000	6.57 (150°C)	8.0
100	1000	7.99 (150°C)	8.0
3.8	1000	6.65 (200°C)	7.5
30	1000	7.51 (200°C)	7.5
3.8	1000	7.21 (300°C)	7.2

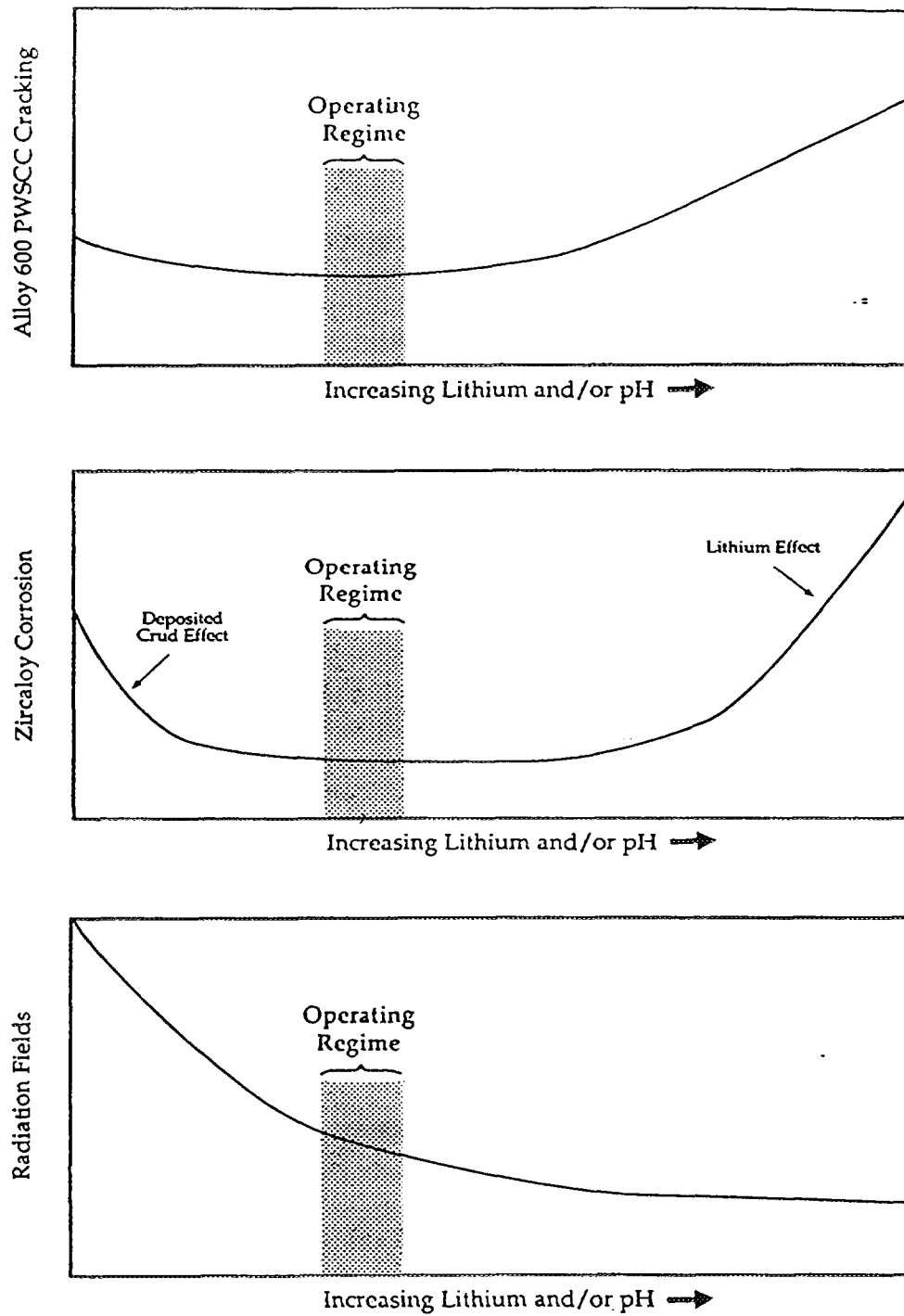


Fig. 2. Schematic presentation of the PWR primary chemistry optimization problem (EPRI NP-7077 1990).

Table 4a. EPRI water chemistry guidelines for BWR's (EPRI NP-4946-SR 1988).

Control parameter	Frequency of measurement	Achievable value	Action levels		
			1	2	3
Reactor water during power operation					
Conductivity (μS/cm at 25 °C)	continuously	≤0.20	>0.30	>1.0	>5.0
Chloride (ppb)	daily	≤15	>20	>100	>200
Sulphate (ppb)	daily	≤15	>20	>100	>200
Diagnostic parameter, silica (ppb)	daily	≤100			

Table 4b. EPRI water chemistry guidelines for BWR's (cont'd) (EPRI NP-4946-SR 1988).

Control parameter	Frequency of measurement	Achievable value	Action levels		
			1	2	3
Reactor feedwater / condensate during power operation					
Feedwater conductivity ( $\mu\text{S}/\text{cm}$ at 25 °C)	continuously	$\leq 0.06$	$>0.07$		
Condensate conductivity	continuously	$\leq 0.08$	$>0.10$		$>10$
Feedwater total copper (ppb)	weekly	$\leq 0.10$	$>0.50$		
	integrated	$\leq 0.30$	$>0.50$		
Feedwater total iron (ppb)	weekly	$\leq 2.0$	$>5.0$		
	integrated				
Feedwater dissolved oxygen (ppb)	continuously	20-50	$<10$		
			$>200$		

## 2.1 BWR COOLANT

BWR's operate under constant oxygen chemistry throughout the fuel cycle and so there is no changes taking place like in PWR's during shut-down affecting the transport of activated corrosion products. Thus, the deposition rate of activity is mainly controlled by the corrosion rate during steady state operation. The aim of the BWR water chemistry is to control the corrosion product input in the feedwater, to reduce crud build-up and to control and minimize radiation field build-up on recirculation piping. Both the above aspects of BWR radiation control require that feedwater and reactor water are kept as pure as possible (EPRI NP-4946-SR 1988).

BWR water chemistry specifications according to the EPRI guidelines are listed in Table 4. These and corresponding guidelines are now worldwide under review. The existing operational problems, specially the IGSCC problem, require more stringent values. For example, it is currently discussed to fix the chlorides and sulphate values of the reactor water for action level 1 at 5 ppb. Also, the iron level of the feedwater seems to be too high. BWR water chemistry has also been modified in order to moderate the SCC problems. Hydrogen water chemistry (HWC) has been used to control intergranular stress corrosion cracking. However, applied HWC has resulted in two radiation effects. The first effect is the increase in N-16 radiation. The second is the Co-60 shut-down radiation field mainly caused by a similar change in the redox environment like in PWR chemistry. Zinc injection has been studied in consideration for reducing cobalt-60 fields. Noble-metal coatings (Pt, Pd) or noble metals as an alloying element have been demonstrated to improve the effects of hydrogen water chemistry. Noble metals even at low concentrations enhance the cathodic reactions and, thus, decrease of corrosion potential takes place faster when hydrogen is induced.

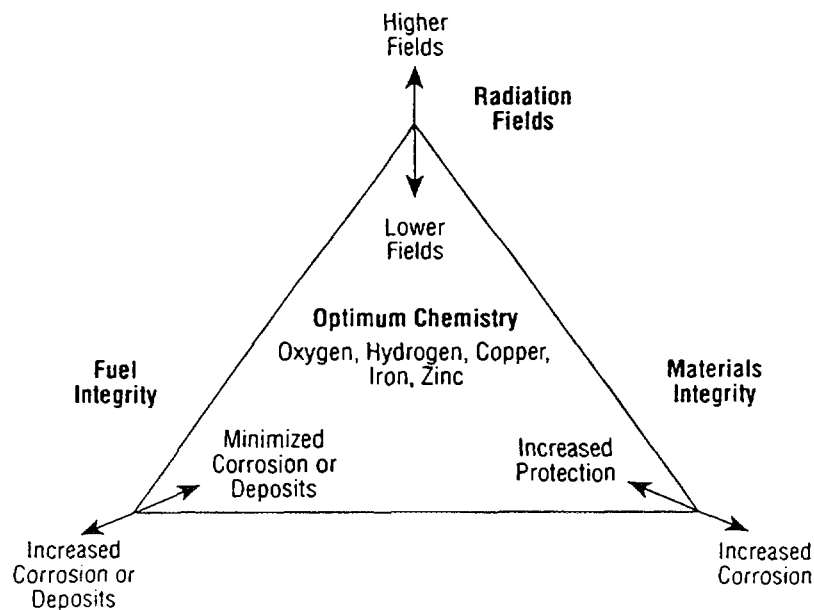
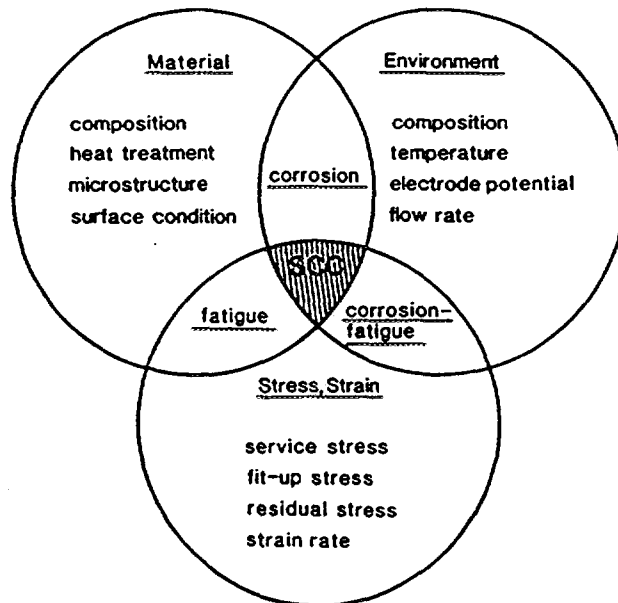


Fig. 3. Problems with optimizing BWR chemistry (EPRI TR-100265 1992).

### 3 ENVIRONMENTALLY ASSISTED CRACKING (EAC) IN LIGHT WATER REACTORS (LWR'S)

Environmentally assisted cracking of LWR pressure boundary components has caused significant outages with occasional safety hazards. Most of the components of nuclear power plants have been affected by corrosion damage. Environment sensitive cracking incidents in pressure vessels, piping and heat exchanger have led to replacements of these major components often after only a small fraction of their design life. Numerous reviews of environment sensitive cracking in light water reactor components have been presented (e.g. Stahlkopf 1982, Berry 1984, Scott 1985, Norring & Rosborg 1984, Hänninen & Aho-Mantila 1985, Hänninen & Aho-Mantila 1986).

The complex interplay of metallurgical, mechanical and environmental factors in environment sensitive cracking is shown in Fig. 4. The number of variables that affect environment sensitive cracking in light water reactor conditions is large and they possibly have a number of synergistic interactions.



*Fig. 4. Factors affecting environment sensitive cracking. Note that specific conditions are required for cracking to occur.*



### 3.1 CARBON STEELS

Corrosion fatigue is the main problem associated with carbon steels in LWR's. The environment sensitive cracking properties of pressure vessel steels such as A533B and A508 have 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. Results obtained in PWR conditions lie usually below the ASME curves (see e.g. Cullen, 1985 & 1986). However, results obtained in pure water containing high dissolved oxygen contents indicate considerably higher crack growth rates than the ASME reference curves, Fig. 5.

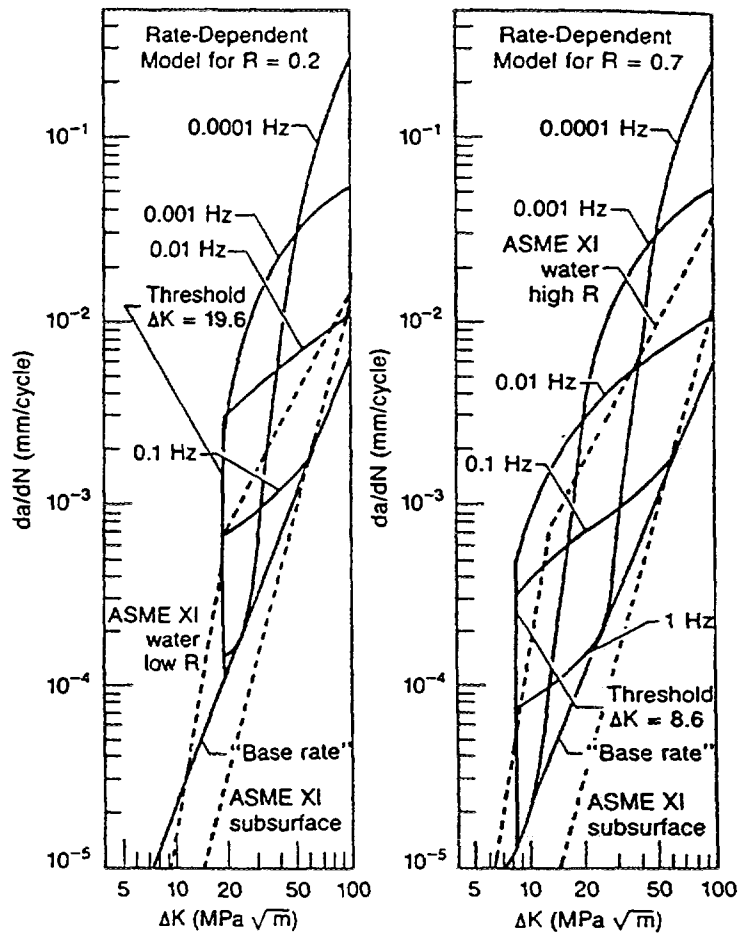


Fig. 5. Predictive curves based on time-dependent corrosion fatigue crack growth model for reactor pressure vessel steels in reactor grade water at 288°C (Gilman 1986).

The importance of metallurgical variables of steels is now clear, based on the laboratory test results. The sulphur content and, especially, the MnS inclusion size, shape and distribution in the steel seem to be responsible for 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. The sulphur species produced from the dissolution ( $\text{H}_2\text{S}$ ,  $\text{HS}^-$  etc.) are known both to enhance hydrogen absorption and to increase anodic dissolution of the steel.

Besides the reactor vessels there has also been cracking in uncladded steam generator shells of PWR's, in steam generator feedwater piping of PWR's, in carbon steel piping of BWR's and in feedwater tanks of nuclear reactors, which are mechanistically relevant to pressure vessel steel problems.

Thermal fatigue is the primary cause of steam generator feedwater pipe cracking incidents induced by thermal stratification during low flow conditions during plant start-up and low power operation. Cracks are oriented circumferentially and located in the base metal outside the weld heat-affected zone. Pitting was associated with the initiation of the cracks. 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 main factor affecting this cracking is thought to be the large number of thermal stratification cycles (0.1-10 Hz); the feedwater chemistry, particularly the oxygen level, was of secondary concern. However, they both contribute significantly to crack initiation.

Strain-induced corrosion cracking (SICC) has caused cracks in the medium-strength, low alloy steel 17MnMoV64 and in the relatively high-strength, fine-grained, structural steel 22NiMoCr37 used in Germany for BWR piping and reactor vessel nozzles. 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 cracking in straight sections of thin-walled piping in German BWRs. SICC refers to those corrosion situations in which the presence of localized, 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. 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.

Erosion-corrosion has reduced pipe wall thickness in several PWR's. An oxide dissolution mechanism is believed to be the mechanism of erosion-corrosion. The interaction of piping design, flow velocity ( $5 \text{ ms}^{-1}$ ), temperature (Fig. 6) and pressure (193°C/25 bar), pH (8.8-9.2), unusually low amounts of alloying elements of the steel, particularly chromium (less than 0.02 %) (Fig. 7) as well as extremely low oxygen content (less than 4 ppb) in the water (Fig. 8), contribute to this type of degradation. Generally, this type of corrosion has been a problem on the secondary loop materials. Erosion-corrosion may become an increasing problem in BWR's, when the water chemistry is changed to a hydrogen water chemistry (HWC). Depending on the type of water chemistry and materials selected for NHR's, the efforts to avoid erosion corrosion should be included in the design.

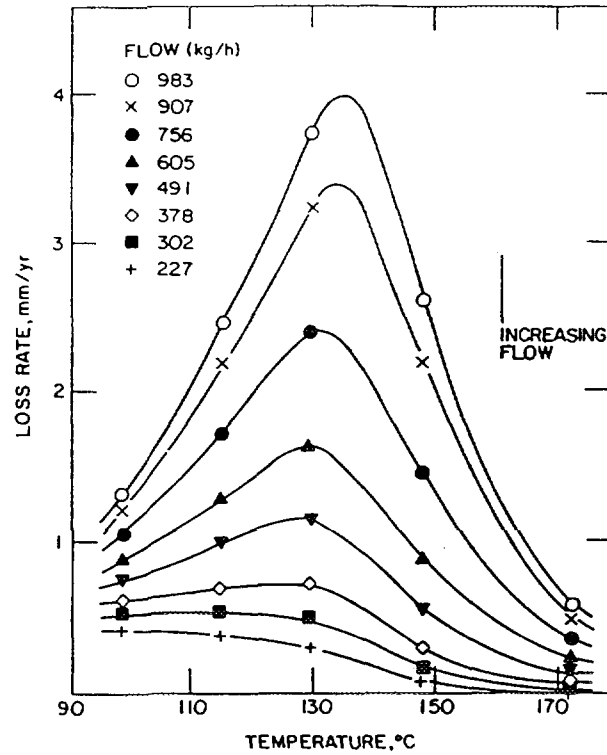


Fig. 6. Temperature dependence of post-orifice erosion-corrosion rates for mild steel in deoxygenated water ( $\text{pH} = 9.05$ ) (Bignold et al. 1980).

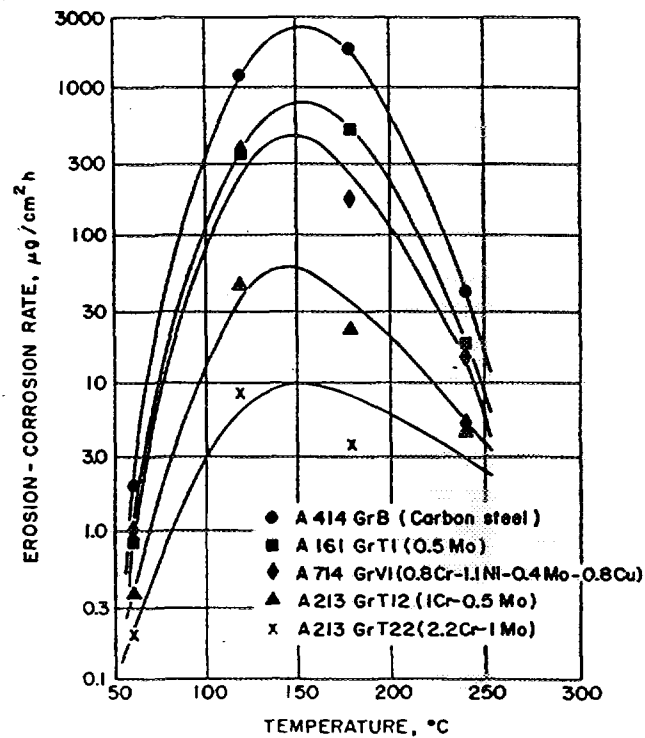


Fig. 7. Erosion-corrosion rate vs. temperature for various steels in water.  $\text{pH} = 7.0$ ;  $p = 40$  bar, flow rate = 35 m/s;  $\text{CO}_2 < 40$  ppb (Heitmann and Schub 1983).

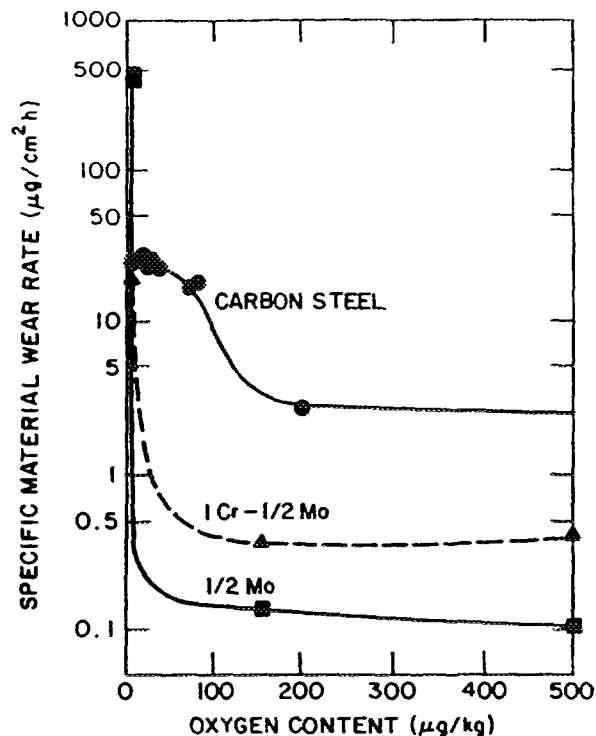


Fig. 8. Effect of oxygen content on material wear rate due to erosion-corrosion for various steels in neutral water for steel denominations (Heitmann and Schub 1983).

### 3.2 STAINLESS STEELS AND NICKEL BASE ALLOYS

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 BWR's. 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 as a function of time is shown in Fig. 9. It can be seen that in about 10 years a marked change in the weld sensitized structure can be expected at reactor operating temperatures.

In addition to the sensitized microstructure and stresses the presence of a certain amount of oxygen in the coolant is necessary for BWR IGSCC. In general, by keeping the amount of oxygen in the coolant low enough IGSCC is inhibited, but the exact level depends on the conductivity of the water. To prevent crack formation in BWR stainless steel piping the electrochemical potential in the cooling water has to be kept more negative than about  $-250 \text{ mV}_{\text{SHE}}$ . Under normal conditions of 100 - 300 ppb dissolved oxygen the electrochemical potential in BWR water varies from plant to plant between  $-100$  and  $+100 \text{ mV}_{\text{SHE}}$  which supports stress corrosion cracking. If the conductivity of the water is below  $0.3 \mu\text{S cm}^{-1}$ , the 20 ppb of oxygen is sufficient low to keep the potential on the right level and to inhibit IGSCC. Therefore, hydrogen addition in conjunction with impurity control has become into widespread use in BWR practice for preventing and mitigating cracking in piping.

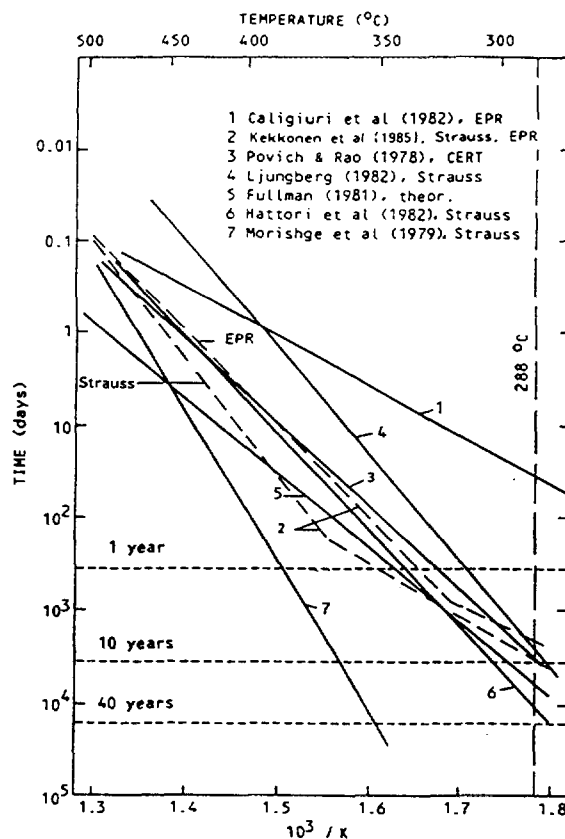


Fig. 9. Several  $(T, t)$  dependencies for LTS of AISI 304 steel showing wide scatter of various tests. The scatter is based on the various starting conditions used as well as test methods (Kekkonen et al. 1985).

Cold work and residual stresses has been attributed to a number of AISI 304 steel pipe cracking incidents in BWR's. Cracking in these cases can be mainly intergranular. An example of this type of cracking has occurred in AISI 304 elbows in the shut-down cooling and clean-up system of the Oskarshamn BWR plant. In cold bending the inner surface deformation (15-20 %) produced some  $\alpha'$ -martensite, which initiated axial cracking in the pipe bend in the absence of sensitization. Extensive cold work should therefore be avoided in austenitic stainless steel primary circuit components.

IGSCC of Inconel 600 has been detected e.g. in a recirculation inlet nozzle safe-end weld at the Duane Arnold plant. The safe-ends were made of Inconel 600 forging and all safe-ends of the plant showed cracking essentially completely around the circumference. Cracking had initiated in the weld HAZ and propagated through the safe-end weldment (Inconel 182). This plant had marked resin intrusions which lowered the pH and increased the conductivity. A large number of recent cracking cases of Inconel 182 weld metal in BWR's emphasizes especially the need for good water quality control. In PWR's similar cracking incidents concerning Inconel 600 and Inconel 182 used in the pressure vessel head penetrations have been observed in many plants.

Thermal fatigue cracks accounts for only a small percentage of the total pipe cracks in stainless steel piping if compared to IGSCC. These cracks are transgranular and can be

prevented by modifying the pipe system design and installing mixers to minimize the thermal gradients. These cracks may be a problem in areas which do not belong to current in-service inspection programs and, thus, for early identification of leaks a good leak detection system is important.

In stagnant borated water of PWR's a number of pipe cracking incidents have been reported. Cracking has occurred in low pressure systems 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 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 °C containing dissolved oxygen while the PWR spent fuel pool environment normally consists of approximately 13000 ppm boric acid at about 65 °C.

Thiosulphate 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 sulphur oxyanions in which thiosulphate and tetrathionate are capable of being reduced to elemental sulphur. A strong synergistic effect exists between thiosulphate and chloride. When mixed together the thiosulphate and chloride produced more pronounced IGSCC than either thiosulphate or chloride separately. This problem was solved by maintaining high-purity water chemistry and using AISI 304L steel to prevent sensitization.

Irradiation assisted stress corrosion cracking (IASCC) is a time-dependent phenomenon, which needs a minimum residence time or threshold fast neutron fluence to occur, Fig. 10. IASCC can occur at very low stresses, but there is an apparent fluence threshold  $> 5 \times 10^{20}$  n cm<sup>-2</sup> (E > 1 MeV) as perceived today. Because the fluence and time dependencies of IASCC are not known precisely at low temperatures, this phenomenon may have great significance in the case of NHR's on in-core component life times.

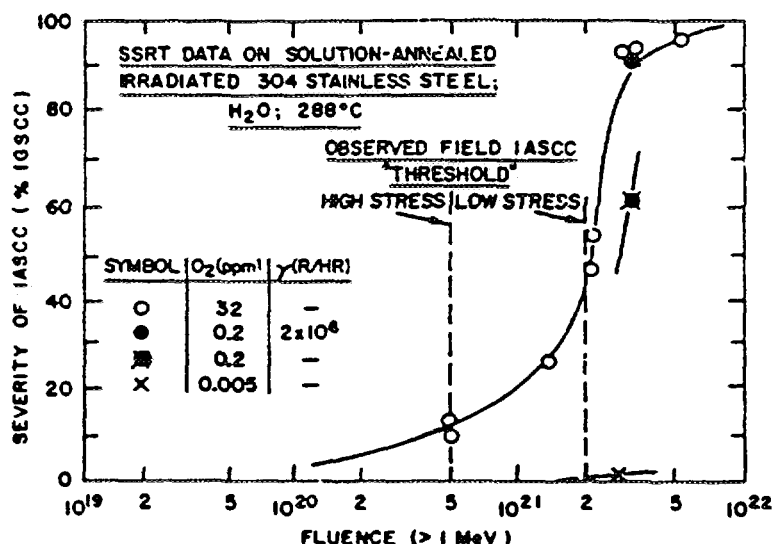


Fig. 10. Relationship between the severity of intergranular stress corrosion cracking and fluence for irradiated type 304 stainless steel under slow strain rate conditions in water at 288°C containing different amounts of oxygen (Andresen, 1989).

### 3.3 STEAM GENERATOR MATERIALS PROBLEMS

Primary water stress corrosion cracking (PWSCC) has been confined to inner row U-bends, severely dented tubes in tube support plate crevices, and roll-expanded areas of the tubesheet region. In general, multiple cracks are observed, suggesting the pattern of residual stresses, e.g., from the rolling operation at the roll transition zone. Dissolved hydrogen in primary side water accelerates cracking.

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 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. The positive effects of intergranular carbides on cracking have been explained on the basis of their effects on local deformation and dislocation structure; grain boundary carbides decrease strain localization near grain boundaries. The factors which have often been assumed to contribute to SCC of Inconel 600 in operating steam generators have been summarized (Paine 1982):

- high hardness, high strength, low carbon content tubing;
- high degree of cold work in the bending or roll expansion;
- ovalization of tubes during bending, i.e., multiaxial stress state;
- residual and operating stresses approaching the yield stress;
- 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 the local environment in order to assess cracking susceptibility. PWSCC is a thermally activated process with a high activation energy. Therefore, the reduced operation temperature may be the remedy for reducing failures.

The tubes and tube supports in steam generators of VVER type reactors are made of titanium-stabilized austenitic stainless steel (AISI 321). The tubes are expanded by rolling and tightness-welded to the primary collectors. So far, experience with the austenitic tubes has been excellent. No denting stress corrosion or wastage corrosion have been observed yet. This is mainly due to the horizontal position of the steam generators, low operation temperature, the austenitic stainless steel tube supports, condensate polishing, effective deaeration and all volatile treatment of the secondary water.

### 3.4 HIGH STRENGTH MATERIALS DEGRADATION

Stress corrosion cracking (SCC) of 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 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 5.

Table 5. Reactor components of age-hardenable austenitic alloys which have experienced cracking.

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:		
	Control rod drive seal	BWR	IGSCC
718	Fuel assembly holddown	PWR	Fatigue
	Fuel assembly finger	BWR	IGSCC
A-286	Springs:		
	Control component holddown	PWR	Fatigue
	Bolts:		
	Thermal shield	PWR	IGSCC/Fatigue
	Fuel assembly	BWR	IGSCC
A-286	Core barrel	PWR	IGSCC
	Beams:		
	Steam separator/dryer/holddown	BWR	IGSCC

The most important applications of the 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 these failures have been caused by erosion-corrosion in PWR primary water leaks. Boric acid is significantly acidic at low temperatures and together with a high velocity jet of leaking primary water severe wastage of ferritic materials can occur.

## 5 CONCLUDING REMARKS

Of various types of corrosion problems in the nuclear industry, stress corrosion and corrosion fatigue have been the most important causes of failures in reactor pressure boundary materials. These environment sensitive cracking incidents in pressure vessels, piping and heat exchanger have given rise to safety concerns and have led to the replacements of some major components after only a small fraction of their design lifetime. Improvements in the material technology may not help any more the existing power plants with their problems. However, improvements in the field of water chemistry control, more stringent guidelines and improvements in the on-line monitoring technology due to new high temperature sensors have excellent changes to prevent cracking incidents in the future.



Many of the above failure modes are time-dependent and, thus, are expected to become more prevalent with ageing power plants. For safe operation in the future various ageing assessment techniques, such as NDE, statistical methods, transient data collection, water chemistry control and operational strategies and predictive models for various forms of failures have to be developed.

NHR's can be designed using the present available failure experiences of existing power plants. In this case also the material selection can be optimized in order to avoid the same problems encountered in existing power plants. Furthermore, the designed low operational temperature provides some additional safety if the corrosion properties of the materials are reviewed and solved in a proper way.

## REFERENCES

Andresen, 1989. In: State of Knowledge of Radiation Effects on Environmental Cracking in Light Water Reactor Core Materials. Cubicciotti, D. (ed.) Proc. of the fourth international symposium on environmental degradation of materials in nuclear power systems - water reactors. Jekyll Island, Georgia, 6 - 10 August 1989. Palo Alto, CA: Electric Power Research Institute, pp. 83-121.

Berry, W. E. 1984. Corrosion problems in light water nuclear reactors. *Materials Performance*, Vol. 23, No. 6, pp. 9-23.

Bignold, G. J., Garbett, K., Garnsey, R. & Woolsey, I. S. 1980. Erosion-corrosion in nuclear steam generators. Proc. 2nd meeting on water chemistry of nuclear reactors. London: British Nuclear Engineering Society, pp. 5-18.

Cullen, W. H. (ed.). 1986. Proc. 2nd IAEA Specialists' meeting on subcritical crack growth. Sendai, Japan 1985. NUREG/CP-0067 MEA-2090, Vol. 1. 485 p.

Cullen, W. H. (ed.). 1986. Proc. 2nd IAEA Specialists' meeting on subcritical crack growth. Sendai, Japan 1985. NUREG/CP-0067 MEA-2090, Vol. 2. 513 p.

Gilman, J. D. 1986. Application of a model for predicting corrosion-assisted fatigue crack growth in LWR environments. Proc. 2nd IAEA Specialists' meeting on subcritical crack growth. Sendai, Japan 1985. NUREG/CP-0067 MEA-2090, Vol. 2. pp. 365-84.

Heitmann, H. G. & Schub, P. 1983. Initial experience gained with a high pH value in the secondary system of PWRs. Proc. 3rd meeting on water chemistry of nuclear reactors. London: British Nuclear Engineering Society. pp. 243-252.

EPRI NP-4946-SR. 1988. BWR Normal Water Chemistry Guidelines: 1986 Revision. Palo Alto, CA: Electric Power Research Institute, September.

EPRI NP-7077. 1990. PWR Primary Water Chemistry Guidelines: Revision 2. Palo Alto, CA: Electric Power Research Institute, November.

EPRI TR-100265. 1992. Radiation-Field Control Manual. 1991 Revision. Palo Alto, CA: Electric Power Research Institute, March.

Hänninen, H. & Aho-Mantila, I. 1985. Environment sensitive cracking in light water reactor pressure boundary materials. Eurotest Conference: Remanent Life: Assessment and Extension. Brussels, 19 - 21 March 1985. 23 p.

Hänninen, H. & Aho-Mantila, I. 1986. Umgebungsinduzierte Rissbildung bei Werkstoffen in druckführenden Bauteilen von Leichtwasserreaktoren. Der Maschinenschaden, Vol. 59, No. 4, pp. 154-65.

Kekkonen, T., Aaltonen, P. & Hänninen, H. 1985. Metallurgical effects on the corrosion resistance of a low temperature sensitized welded AISI 304 stainless steel. Corrosion Science, Vol. 25, No. 8/9, pp. 821-36.

Norring, K. & Rosborg, B. 1984. A compilation of experience of corrosion in Nordic nuclear power stations. Studsvik, Sweden, Report EI-84/150. 24 p. + app.

Paine, J. P. N. 1982. Operating experience and intergranular corrosion of Inconel alloy 600 steam generator tubing. Corrosion 82. Houston, TX: NACE. Paper No. 204.

Rieß, R. 1993. Control of water chemistry in operating reactors. IAEA Meeting on influence of water chemistry on fuel cladding behaviour. Rez, Czech republic, October 4 - 8, 1993.

Scott, P. M. 1985. A review of environment sensitive fracture in water reactor materials. Corrosion Science, Vol. 25, No. 8/9, pp. 583-606.

Stahlkopf, K. E. 1982. Light water reactor pressure boundary components: a critical review of problems. In: Steele, L. E., Stahlkopf, K. E. & Larsson, L. H. (eds.) Structural integrity of light water reactor components. London: Applied Science Publishers, pp. 29 - 54.