Section 5:
Session 5: Mitigation of Nickel-base Alloy Degradation and Foreign Experience

Belgian Activities on Alloys 600 and 182 Issues

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Abstract

Inspections have been carried out on the RPV head penetrations of the 7 Belgian PWR units since 1992, after cracking of these components was first discovered in France. This led to the detection of significant cracking in the base material Inconel 600 of several head penetrations of a same heat in the Tihange 1 unit in 1998. Crack growth analysis made it possible to justify the operation for one more 17-month cycle, and in this period a new head was purchased and installed at the next outage in 1999. Inspections in other units have shown some indications but to a lesser extent and with relatively slow growth. The situation is monitored by regular inspections.

The cracking events in VC Summer and Ringhals in the hot leg nozzle to piping Inconel welds gave rise to the concern that a similar situation might occur in the Belgian units, which also have Inconel 182 welds in the reactor pressure vessel and pressurizer nozzle to safe-ends welds. A detailed review of the fabrication procedures was performed, as well as stress analyses to evaluate the operating stress level and fracture mechanics analyses to evaluate the corrosion crack growth. Flaw stability and leak rate through a postulated through-wall flaw were evaluated in a defense-in-depth perspective. These studies concluded that the situation was relatively favorable for the RPV nozzle to safe-end welds, but that the pressurizer welds could be more sensitive and required more frequent inspections.

Introduction

7 nuclear units are operated in Belgium by the Utility Electrabel, all of the PWR type, which represent a total capacity of more than 5700 MW. In 2002 they generated 57.6% of the country's electricity. The main characteristics of these units are summarized in Table 1. The present paper summarizes the activities carried out in Belgium in relation with the Inconel 600 and 182 issues (inspections, analyses, general strategy).

Unit	Netto capcacity(Mw)	First operation	NSSS designer
Doel 1	392.5	1974	Westinghouse
Doel 2	392.5	1975	Westinghouse
Doel 3	1006	1982	Framatome
Doel 4	985	1985	Westinghouse
Tihange 1 (1)	962	1975	Framatome
Tihange 2	1008	1983	Framatome
Tihange 3	1015	1985	Westinghouse

(1):owned jointly with EDF (50/50)

Table 1: Characteristics of the Belgian units

Inconel zones in Belgian RPVs

The location of the Inconel zones in Belgian reactor pressure vessels and pressurizers are summarized in table 2. We have:

- Inconel 600 RPV head penetration and bottom penetrations with Inconel 182 buttering and welds.
- Inconel 182 buttering and welds at the dissimilar metal welds between the low alloy steel (SA508 Cl.3) of the RPV nozzles and the SA182F316L stainless steel safe-ends. These welds were stress relieved with the vessels.
- Inconel 182 buttering and welds at the dissimilar metal welds on the pressurizers (surge line, spray line, safety valves). The buttering was stress relieved with the pressurizers but the weld itself was made after the final heat treatment.
- On Doel 1-2 reactor pressure vessels (2-loop units), we have in addition two Inconel 600 safety injection nozzles welded on the vessel with Inconel 182 welds.

Component	Part	Base material	Weld	Buttering	Cladding	Units
Pressurizer	Safe ends		X	X		D3/T2/D4/T3
Steam generator	Partition plate		X	X		D1/D2
	Tube	X	X			D1/D2
	Tubesheet				X	D3/D4/T3
	Drain					D4/T3
	Safe-ends welds		X (82)	X (82)		T1
Reactor pressure vessel	Safe ends		X	X		D3/T2/D4/T3
	Bottom penetrations	X	X	X		ALL
	Core support blocks	X	X	X		ALL
	Safety Injection nozzles	X	X	X		D1/D2
RPV head	Penetrations	X	X	X		ALL except T1

Table 2: location of the Inconel zones in the primary circuit of Belgian units
(D stands for Doel , T for Tihange)

RPV head inspection status (base material)

Inspections have been carried out on the RPV head penetrations of the 7 Belgian PWR units since 1992, after cracking of these components was first discovered in France. The inspections carried out on the different units are summarized in table 3, which also gives the number of operating hours at the time of inspection as well as the operating temperature under the RPV head. The only case of significant cracking was in Tihange 1, which is described in more details later and led to the RPV head replacement. Some small cracks were also detected in Doel 1 and 2 RPV heads, and are followed by regular inspections, but the growth of these cracks is slow, particularly in the depth direction. No indications were detected in the other units, except one indication in Tihange 3 which cannot be confirmed as a crack and did not evolve in 3 successive inspections. In total 21 RPV head inspections by ET (in some cases combined with UT for in-depth dimensioning) have been carried out since 1992. Recently high resolution televisual inspections of the J-groove welds were added to the program.

Inspect. date	Unit	Age *	EFP* (hours)	T° (°C)	Inspector	Inspected penetr.	Inspect. method	Results
Oct. 92	Tihange 1	17	123 000	318	ABB-R	65	ET	no crack (1 indication)
March 98	Tihange 1	23	163 000	318	FRA	65 + 1	ET + UT	Several cracks (1 heat)
Oct. 99	Tihange 1	24			RPV head replaced			
Oct. 93	Tihange 2	10	81 000	287	ABB-R	65 + 1	ET	no crack
March 00	Tihange 2	17	131 000	287	ABB-R	65 + 1	ET	no crack
March 93	Tihange 3	8	59 000	318	ABB-R	65 + 2 + 3	ET	no crack
Dec. 96	Tihange 3	11	87 000	318	ABB-R	65 + 2 + 3	ET	1 indication
June 98	Tihange 3	13	99 000	318	ABB-R	65 + 2 + 3	ET	1 indication
Sept. 01	Tihange 3	16	124 000	318	W-R(3)	65 + 2 + 3	ET	1 Indication No Propagation
Sept. 93	Doel 1	19	137 000	307	W	49	ET	no crack (1 indication)
Aug. 98	Doel 1	24	174 000	307	ABB-R	49 + 1	ET + UT	Several indications (small cracks)
Aug. 99	Doel 1	25	182 000	307	ABB-R	11	ET + UT	small cracks
Aug. 01	Doel 1	27	198 000	307	W-R(3)	36	ET + 4UT	small cracks propagation
Sept.03	Doel 1	29	215 000	307	FRA	49	UT +visual (welds)	
May 94	Doel 2	19	125 000	307	W	49	ET + UT	no crack(2)
May 00	Doel 2	25	170 000	307	W-R	49 + 1	ET + UT	small cracks
June 03	Doel 2	28	194 000	307	FRA	49	UT +visual (welds)	small scratches (not considered as cracks)
June 93	Doel 3	10	82 000	287	ABB-R	65 + 1	ET	no crack
April 00	Doel 3	17	134 000	287	ABB-R	65 + 1	ET	no crack
April 94	Doel 4	9	66 000	314	ABB-R	65 + 2 +3	ET	no crack
June 99	Doel 4	14	100 000	318	ABB-R	65 + 2 + 3	ET	no crack

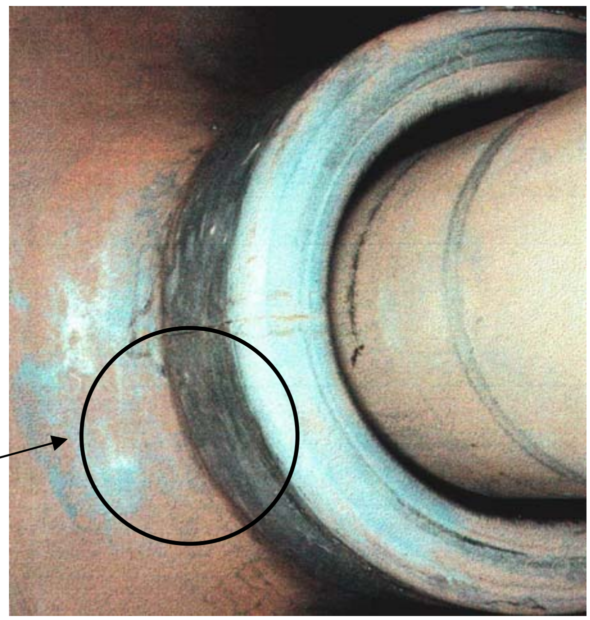
* At the inspection date
(1) Intercontrôle (IC) and AIB-Vinçotte
(2) Indications of original defects in the weld due to lack of fusion
(3) ex ABB-R

Table 3: RPV head inspection status (base material)

Tihange 1 RPV head penetration cracking

The Tihange 1 RPV head exhibited significant cracking in one penetration, with an axial crack extending below the weld all the way to the bottom of the penetration (figure 1). There was however a ligament of more than 25 mm remaining between the upper crack tip and the triple point of the weld (above which a leak would be possible). Other penetrations of the same heat were affected to a lesser extent. Fracture mechanics and crack growth analyses demonstrated that the crack would not grow beyond the triple point (which would result in a leak) in the next 17-month cycle and the unit was allowed to restart for one cycle. An inspection of the retired head in 1999 confirmed that the crack growth was lower than the prediction and there remained a 20 mm ligament between the crack tip and the triple point. A new head with Inconel 690 penetrations was installed at the next outage in October 1999.

Figure 1



Axial through-wall crack extending to the bottom of the penetration

Stress Corrosion crack growth in RPV head penetrations

Based on the few instances where a same crack was monitored in successive inspections in the Belgian units, quite different crack growth rates were sometimes seen in the depth and length directions. The length sizing is however rather less reliable than the depth sizing, the extremities of the flaw being generally very shallow.

The measured crack propagation was compared to crack propagation equation generally used in the U.S. for the base material Inconel 600 [5]:

$$da/dt = 2.89\ 10^{-12}\ (K_I-9)^{1.16}\ m/s\ \ (at\ \ 325°C)$$

This equation can be adapted to other temperatures by means of an Arrhenius law with an activation energy of 130 kJ/mole. Compared to the measures propagation, this law is conservative for the crack growth in the depth direction, but it underestimates the growth in length. This evaluation is based on a simple analysis with a constant stress level, but the difference is too big to be explained by a non uniform stress distribution (especially since stresses could be expected to decrease rapidly away from the weld). Another point that could be discussed regarding the crack growth equation is the 1.16 exponent, which is probably overconservative at high K_I, where experimental results indicate a plateau-like behavior.

Bare metal visual inspections

5 units were inspected in 2002 and the last one in 2003. (Tihange 1 will not be inspected in the near future since the head was replaced in 1999). Even though the accessibility was not always as good as on figure 2 and it was not always possible to inspect visually 100% of the surface, the heads were generally clean with locally some rusty streaks or small boric acid crystals attributed to previous canopy seals leaks. No evidence of RPV head corrosion was seen.
The heads that could not be visually inspected at 100%, or where some of boric acid traces from canopy leaks made the interpretation more difficult, will be reinspected at the next outage.

Figure 2

Equivalent degradation years

The Belgian units were evaluated in terms of “Equivalent degradation years” in accordance with the MRP procedure (by converting the Equivalent Full Power Years at the effective RPV head operating temperature to “Equivalent Degradation Years” at a reference temperature of 316°C, by means of an Arrhenius law with an activation energy for crack initiation of 209 kJ/mole). Based on the results, the units can be ranked by susceptibility. The results of the evaluation are given in table 4.

Even though the EDY were quite high for Tihange 1 when the RPV head was replaced, there was no leak at that time and the ligament between the crack tip and the triple point of the weld was still more than 25 mm, which means that even with very conservative crack growth rates the RPV head could have operated for at least 3 additional cycles before a leak would have occurred.

Unit	Temperature (°C)	EDY
Tihange2 – Doel 3	287	1.9
Doel 1-2	307	12.7-11.6
Tihange 3 – Doel 4	318	18.3-17.6
Tihange 1 (head replaced in1999)	318	24
Tihange 1 (new head)	318	4

Table 4: Equivalent degradation years for the Belgian units

Other components in Inconel 600

Inspections were also carried out on other Inconel 600/182 components, namely the safety injection nozzles and the bottom penetrations in Doel 1 and 2. Since these components are at the cold leg temperature (287°C), the risk of cracking and the crack propagation rate in case of cracking are significantly reduced. Regarding the bottom penetrations, there were some indications interpreted as fabrication defects (lack of fusion) at the weld/base metal interface.

There are some small surface indications in the safety injection welds, but they are not critical due to the relatively low stress level and low temperature. In addition, there are fabrication defects of the “lack of fusion” type at the weld/base metal interface. The inspection plans for the Bottom Mounted Instrumentation penetrations are currently being re-evaluated, following the discovery of leaking penetrations at South Texas unit 1. The details of the inspections are summarized in table 5 .

Inspect. date	Unit	Age *	EFP* hours	T° (°C)	Inspecto r	Inspected penetr.	Inspect. technique	Results
BMI Penetrations								
May 95	Doel 2	20	132 000	287	TRC	13	ET + UT	no crack(2)
May 00	Doel 2	25	170 000	287	W-TRC	13	ET + UT	no crack(2)
SI Nozzles								
Oct. 95	Doel 1	21	152 000	287	(1)	2	UT	no crack(2)
August 99	Doel 1	25	182 000	287	IC	2	ET + UT	small surface indication + (2)
May 00	Doel 2	25	170 000	287	IC	2	ET + UT	small surface indication + (2)

* At the inspection date
(1) Intercontrôle (IC) and AIB-Vinçotte
(2) Indications of original defects in the weld due to lack of fusion

Table 5: Inspection status of other components in Inconel 600

Alloy 182 safe-ends welds

The cracking events at VC Summer and Ringhals in the hot leg nozzle to piping Inconel welds gave rise to the concern that a similar situation might occur in the Belgian units, which also have Inconel 182 welds in the reactor pressure vessel and pressurizer nozzle to safe-ends welds in the Doel 3 – Tihange 2 and Doel 4 – Tihange 3 units (the older units Doel 1-2 and Tihange 1 have stainless steel buttering and welds). The new Tihange 1 steam generators, installed in 1995, also have Inconel welds but in this case they are alloy 82, expected to be less sensitive to cracking than alloy 182.

A detailed review of the fabrication procedures was performed, in order to identify weld repairs and heat treatments carried out in fabrication. This review showed that for the reactor pressure vessels, all the welds were made prior to the final heat treatment and stress-relieved with the vessels. All repairs were made before the final heat treatment, except in one specific case, which is fortunately on a cold leg.

For the pressurizer welds, the buttering was made before the final heat treatment and stress relieved, but the weld itself was performed after and is not stress relieved. Combined with the higher temperature, this obviously makes the pressurizer welds a more sensitive location which deserves enhanced inspections.

Finite element stress analyses were performed for all dissimilar metal welds with alloy 182, in order to evaluate the normal operating stresses. Residual stress solutions available from the literature for such relatively simple configurations were added to the operating stresses. A more precise analysis including residual stress determination by welding process modeling is under way. The stress analyses show very clearly that the hoop stresses are much larger than the axial stresses, which is in agreement with the axial orientation of the defects detected in VC Summer and Ringhals. This is also reassuring, since it makes a circumferential defect less likely than an axial defect and does not invalidate the Leak Before Break analyses performed on the primary loop piping.

Based on the calculated stresses, on the operating temperature, on the existence of weld repairs and of course on the operating time, a semi-empirical ranking of the different locations was made. Not surprisingly, this exercise identified the pressurizer to surge line welds as the most critical locations, especially in Tihange 2 were a repair was made in fabrication. Even though this repair was made from the outside and did not affect the inside surface, it could influence the residual stress distribution in an unpredictable way, and it was considered conservatively that the effect could be detrimental. The stress level in these welds reaches 300 to 350 MPa at the inside surface, which is close to the 350 MPa generally considered as a threshold for the initiation of cracking in Inconel 182.

The second most critical locations are the Inconel 182 welds on the 4” and 6” lines (spray and safety valves) on the pressurizer upper head. The risk of cracking is basically identical to the surge line weld, but the potential consequences of a failure are smaller due to the smaller diameter. The next position in the ranking is occupied by the RPV hot leg nozzle to piping welds, and we included also in this group the cold led weld where a repair was made after the final heat treatment. In these welds, the hoop stress level is of the order of 220 MPa at the inside surface, which is sufficiently below the 350 MPa threshold to consider a crack initiation unlikely. In the last group, we find the other RPV cold leg welds, where the hoop stresses are comparable or lower and where the lower temperature makes initiation even more unlikely.

Defect Tolerance Analyses were performed for the RPV outlet nozzle and Pressurizer nozzles welds in order to evaluate the size of acceptable defects as a function of the inspection interval, and to define the qualification objectives for the inspection procedure. This raised the question of the crack growth equation to be used. A review of the information available in the open literature quickly showed that the experimental data on Inconel 182 cracking is rather scarce, which reflects the generally good service experience with this material up to now. In addition the crack growth data that is available covers only a limited range in K_I. Although the databases used for evaluations in the U.S, in Sweden and in France contain to a significant extent the same data, it can be interpreted in very different ways.

In the U.S., the mathematical form of the Scott’s equation (established initially for Inconel 600) was used, with a K_I threshold of 9 MPa$\sqrt{m}$ and an exponent of 1.16. The justification for that is very limited, and this choice seems mainly motivated by the fact that there is not enough Inconel 182 data to define a specific expression for this material. In the justification for VC Summer [1], a coefficient of 1.4 10^{-11} was proposed, but in the Safety Evaluation Report [2] the NRC required to increase this coefficient by 1.5, leading to the following expression:

$$da/dt = 2.1\ 10^{-11}\ (K_I-9)^{1.16}\ \text{m/s}\ \ (\text{at } 323°\text{C})$$

A different expression was used in Sweden for the justifications performed for Ringhals [4]. The expression is given for a temperature of 320°C (these value must be increased by a factor 1.14 to convert of a temperature of 323°C, based on an Arrhenius law with an activation energy of 130 kJ/mole):

$$da/dt = 5.70\ 10^{-20}\ K_I^{\ 9.3}\ mm/s \text{ for } K_I < 25\ MPa\sqrt{m}$$

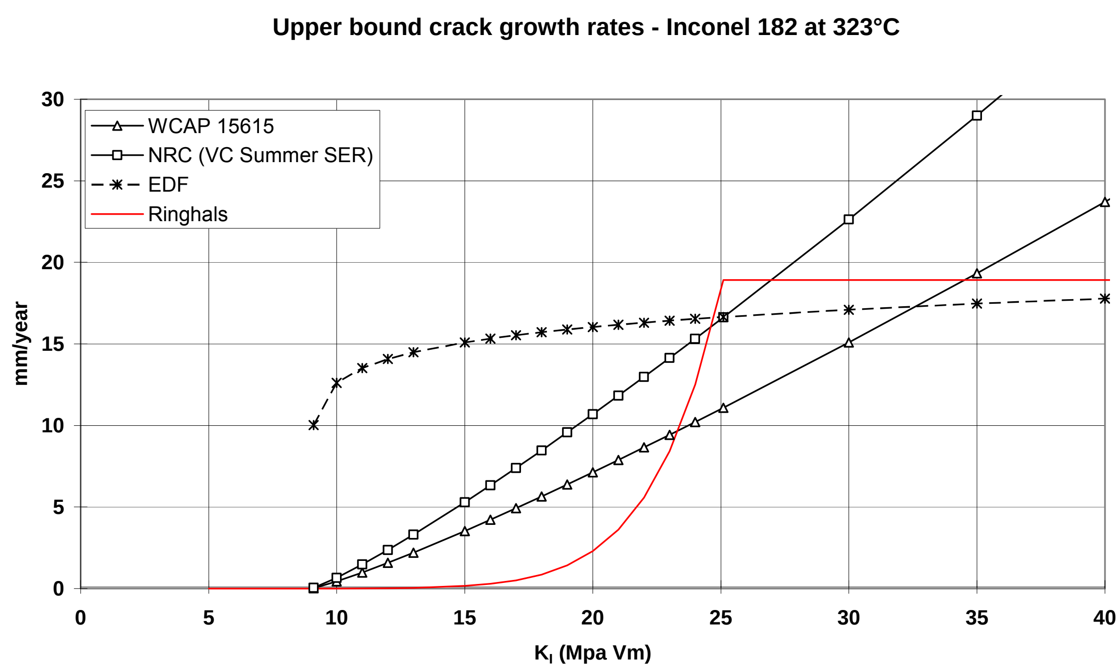
$$da/dt = 6.00\ 10^{-7}\ mm/s \text{ for } K_I > 25\ MPa\sqrt{m}$$

EDF proposed in [3] a more complex expression, taking into account the cold work, stress-relieved or non stress-relieved condition, orientation and type of loading (cyclic or constant loading). For a non stress-relieved weld with cyclic loading and orientation parallel to the dendrites (most penalizing combination), this leads for a temperature of 323°C to the following expression:

$$da/dt = 4.\ 10^{-10}\ (K_I\text{-}9)^{0.1}\ m/s \quad \text{(at 323°C)}$$

The different equations are compared on figure 3, which shows very clearly that the US equation leads to very penalizing results for high K_I, while both the French and the Swedish laws indicate a plateau-like behavior and lead to results that are in good agreement above 25 MPa√m, but differ considerably below.

Figure 3



Even for the RPV outlet nozzle weld, where the stresses and temperatures are moderate, the analysis based on the crack growth equation recommended by the US NRC for the VC Summer case results in small acceptable defects and fast crack growth rate once a defect is initiated. In the absence of a defect, the situation can be considered as relatively safe since the initiation in Inconel 182 is unlikely for the stress level present at the inside surface. The axial stresses are much lower that the hoop stresses, which is in agreement with the axial orientation of the cracks found in Ringhals and VC Summer.

For the pressurizer welds, the temperature is close to 345°C, which leads to an acceleration factor of 2.54 as compared to the equations established at 323°C. Combined with a rather high hoop stress level (due mainly to the contribution of the residual stresses, since the weld is not stress relieved), this leads to acceptable crack sizes that are extremely small for the axial cracks. The situation is much better for the circumferential cracks due to the much lower stresses in the axial direction.

Any axial crack of more that a few mm in depth leads to very fast crack growth and reaches the maximum acceptable crack size in a few months, especially with the "NRC" equation with its continuous increase in crack growth rates. This results from the very conservative nature of these equations, which are designed to be an absolute upper bound. In reality, the scatter in da/dt is such that the real growth rates could very well be a factor 100 lower. Postulating an existing flaw is also very detrimental since the material is very resistant to initiation for stresses below 350 MPa. This means that it would be totally unrealistic to define an inspection schedule based on these results. It shows on the other hand that it is also not acceptable to stick with the ASME XI inspection frequencies and that an enhanced inspection (in terms of frequency as well as performance) of the pressurizer welds is necessary.

As a result of this evaluation, the inspection schedule of the Belgian units was modified to inspect the pressurizer welds earlier than initially foreseen, using a qualified automated UT technique. This led in Tihange 2 to the discovery in October 2002 of a small axial flaw at the inside surface, of 4x26 mm approximate size. The flaw is located close to the repair made in fabrication, and might well be a fabrication defect that was undetected at the time. On the other hand, we cannot exclude the possibility of a PWSCC crack. In both cases, calculating the crack growth using the different equations currently available leads to very penalizing results and the crack is predicted to go through-wall in a matter of months.

The flaw stability of a postulated through-wall axial flaw in an RPV outlet nozzle to safe-end weld (limited to the Inconel 182 metal) was evaluated in a defense-in-depth perspective, as well as the leak rate through this crack (figures 4 and 5). These studies concluded that the through-wall crack was stable with large margins for a crack in the outlet or inlet RPV nozzle to safe-end weld, but that the leak rate would be very small. Parametric studies performed for different number of IGSCC crack turns and surface roughness yielded leak rates of an order of 2-5 kg/hour. The analysis was extrapolated to the pressurizer surge nozzle to safe-end weld. Although the stability of the through-wall crack was still ensured, the margins were lower than for the RPV nozzles. On the other hand the leak rate is about 5 to 6 times larger and could be at the limit of detectability by the existing systems (at least for a sufficiently long observation time).

On the basis of these analyses, the Safety Authorities gave the green light for the restart of the unit for six months, an inspection being scheduled at mid-cycle in May 2003. In the meantime, the Mechanical Stress Improvement Process (MSIP, AEA patented process to mechanically contract the pipe on one side of the weld, replacing the residual tensile stresses with compressive stresses) was validated for application on the pressurizer to nozzle weld. Repair techniques are also investigated.

The weld was reinspected in May 2003 and no evolution was detected. It will be inspected again in October 2003.

Figure 4

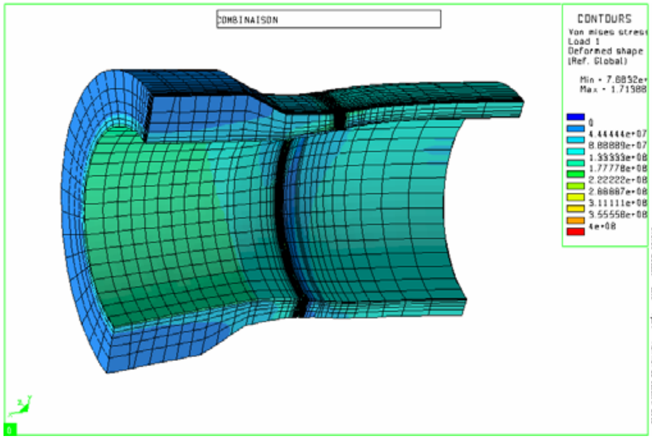
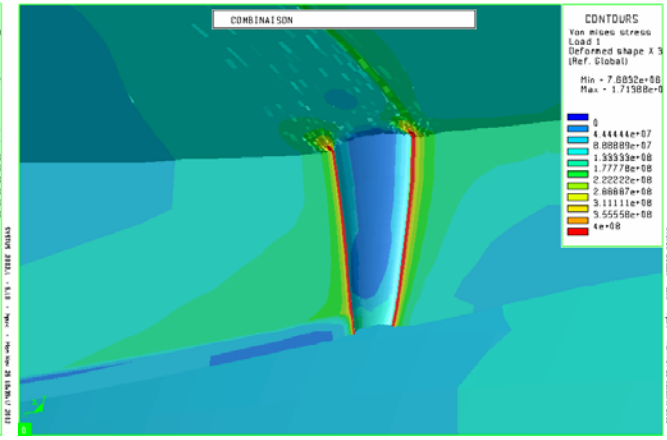


Figure 5



Strategy

- For the reactor pressure vessel heads, the only case with extensive cracking was replaced in 1999 in Tihange 1. For the other units, inspections are carried out at frequent intervals to detect and follow any crack growth. No systematic preventive replacement is contemplated for the present time, but the repair/replacement options are being evaluated. For the J-groove welds, high resolution visual inspections are implemented since the 2003 inspections.
- For the BMI, the current policy was up to now to follow the ASME ISI programs. This is presently being reevaluated following the South Texas event
- For the reactor pressure vessel and pressurizer dissimilar metal welds, detailed stress and defect tolerance analyses were performed. A ranking of the most sensitive locations was made and the inspections program was enhanced accordingly. More precise evaluations of the welding residual stresses are planned or under way.
- For the other locations, the ASME ISI program is followed.

Conclusions

In conclusion, a RPV head penetration inspection program is in place since 1992 in the Belgian units and 21 inspections have been performed to date. Due to the significant cracking detected in Tihange 1 in 1998, the head was replaced in 1999. Small indications in other units are followed by regular inspections but the growth of these indications (if any) is slow.

A significant program on Inconel 182 safe-end welds started after VC Summer event and involves fracture mechanics analyses, a ranking of the most critical locations and increased inspections. These analyses show that the pressurizer safe end welds are the most critical location since they are not stress relieved, and combine high residual stresses and high operating temperature.

Increased inspections were implemented on these welds, which led to the detection of a small indication in Tihange 2 surge line weld in October 2002. Defence-in-depth analysis of through-wall axial crack shows it would remain stable. A new inspection six months later did not show any evolution, and this indication might very well be a fabrication defect present since the beginning. The weld will be inspected again in October 2003. The Mechanical Stress Improvement process was qualified for application on this weld, by extensive stress analysis and validation on a mock-up, and the necessary equipment is available. Possible repair techniques are being evaluated and the necessary materials were purchased preventively.

This proactive inspection and analysis program made it possible to detect defects early on, well before they could lead to a leak.

Aknowledgements

The authors thank Electrabel for the permission to publish these results.

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German Experience with RPV Head Penetrations

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Summary

The 357 MW nuclear power plant Obrigheim (KWO) has been operated successfully and safely for the last 35 years. The design of its reactor pressure vessel head penetrations is comparable to those plants in which failures have occurred in the same area. Since the first occurrence of this type of failure in 1991 in France many investigations, inspections and measurements have been performed in Obrigheim to maintain and confirm the integrity of these components. The material quality, manufacturing process, design of the nozzle flanges, and low in-service loads are the contributors to the failure-free operation of the Alloy 600 nozzles. Two special local leakage indication systems have been installed on the vessel head and in the area of the nozzle flanges to complement the global leak detection system. In addition, the unrestricted access to the nozzles during outages as well as the non-destructive testing techniques used on the vessel head and the nozzles enable a continuous confirmation of the integrity of these components. Over the past 20 years only a few instances of very small flange leaks have occurred, and to date no crack indications have been detected in the nozzles or the vessel head.

With the exception of the NPP Obrigheim, the Siemens/KWU PWR designed plants exhibit fundamental differences with respect to design, materials and manufacturing/fabrication of head penetration nozzles in comparison to plants of other NPP suppliers. The primary difference in the choice of material is the use of extruded compound piping comprising a carbon steel tube with a Type 347 cladding for all surfaces exposed to the coolant. Structural differences primarily include a threaded rather than a shrink fitted design for the head penetrations. None of the inspections of Siemens/KWU plants have detected any indications.

I INTRODUCTION

As one of the first units in Germany, the nuclear power plant Obrigheim (KWO, see Figure 1) is a Westinghouse licensed 2-loop plant that began commercial operation in 1968. It has been running for more than 35 years with high availability.

Although the head penetration design is basically comparable to designs of plants that have experienced leakages at those locations in recent years, similar failures have not been observed at KWO. The following paragraphs summarize the investigations and actions taken to confirm the structural integrity of the RPV head penetrations at KWO. The time frame covers a period from the initial occurrence of CRDM nozzle cracking at Bugey 3 in 1991 until the KWO outage performed in 2002.

The in-service experience with other Siemens/KWU PWR plants since NPP Stade which have a different design for the CRDM nozzles, is outlined as well.

PWR plant	***Country***	***Year of Start up***	***MWe Gross/Net***
Obrigheim	*Germany*	*1968*	*357 / 340*
Stade	*Germany*	*1972*	*672 / 640*
Biblis A	*Germany*	*1974*	*1204/ 1146*
Biblis B	*Germany*	*1976*	*1300 / 1240*
Neckar 1	*Germany*	*1976*	*840 / 785*
Unterweser	*Germany*	*1978*	*1300 / 1230*
Grafenrheinfeld	*Germany*	*1981*	*1300 / 1235*
Grohnde	*Germany*	*1984*	*1395 / 1325*
Philippsburg 2	*Germany*	*1984*	*1349 / 1268*
Brokdorf	*Germany*	*1986*	*1395 / 1326*
Isar 2	*Germany*	*1988*	*1400 / 1320*
Emsland	*Germany*	*1988*	*1341 / 1270*
Neckar 2	*Germany*	*1989*	*1360 / 1225*
Borssele	*Netherlands*	*1973*	*481 / 449*
Atucha 1	*Argentina*	*1974*	*357 / 335*
Gösgen	*Switzerland*	*1979*	*1020 / 970*
Trillo	*Spain*	*1988*	*1066 / 1000*
Angra 2	*Brazil*	*2000*	*1309 / 1229*

Figure 1 NPPs with PWRs and Year of Start-Ups built by Siemens/KWU

II RVP HEAD PENETRATION NOZZLES AT KWO

II.A Materials and Manufacturing Process

The total of 49 head penetrations were manufactured from Alloy 600 with an OD of 85 mm and a wall thickness of 10 mm. These dimensions are significantly smaller compared to similar plants. The tubing manufacturing process included a seamless extrusion (hot-forming) followed by solution annealing at 1050°C. Thus the manufacturing-related residual stresses and SCC susceptibility were minimized.

The welding of head penetrations into the RPV head was performed without buttering, using an Alloy 82/182 consumable, which is comparable to other plants of similar design. The substantial effort, and what is believed to be the most decisive counter-measure to reduce SCC susceptibility, was a stress relief annealing of the entire head for 10 hours at 600 °C. This post weld heat treatment (PWHT) was performed after all penetrations were welded and therefore greatly reduced residual stresses.

Due to the angular position of head penetrations at the very outer circumference compared to the inner RPV surface the welds are relatively non-symmetrical (Figure 2). This resulted in non-symmetrical shrinkage and ovality of penetration tubing up to 2mm. This in turn generated hoop stresses, that reached levels well beyond the yield strength if not properly stress relieved annealed. Thus, in the early 1990's the first cracking in plants Bugey 3, Bugey 4 and Fessenheim occurred exclusively in penetrations located at the outer circumference of the RPV head. At KWO these penetration locations are to date free of cracking.

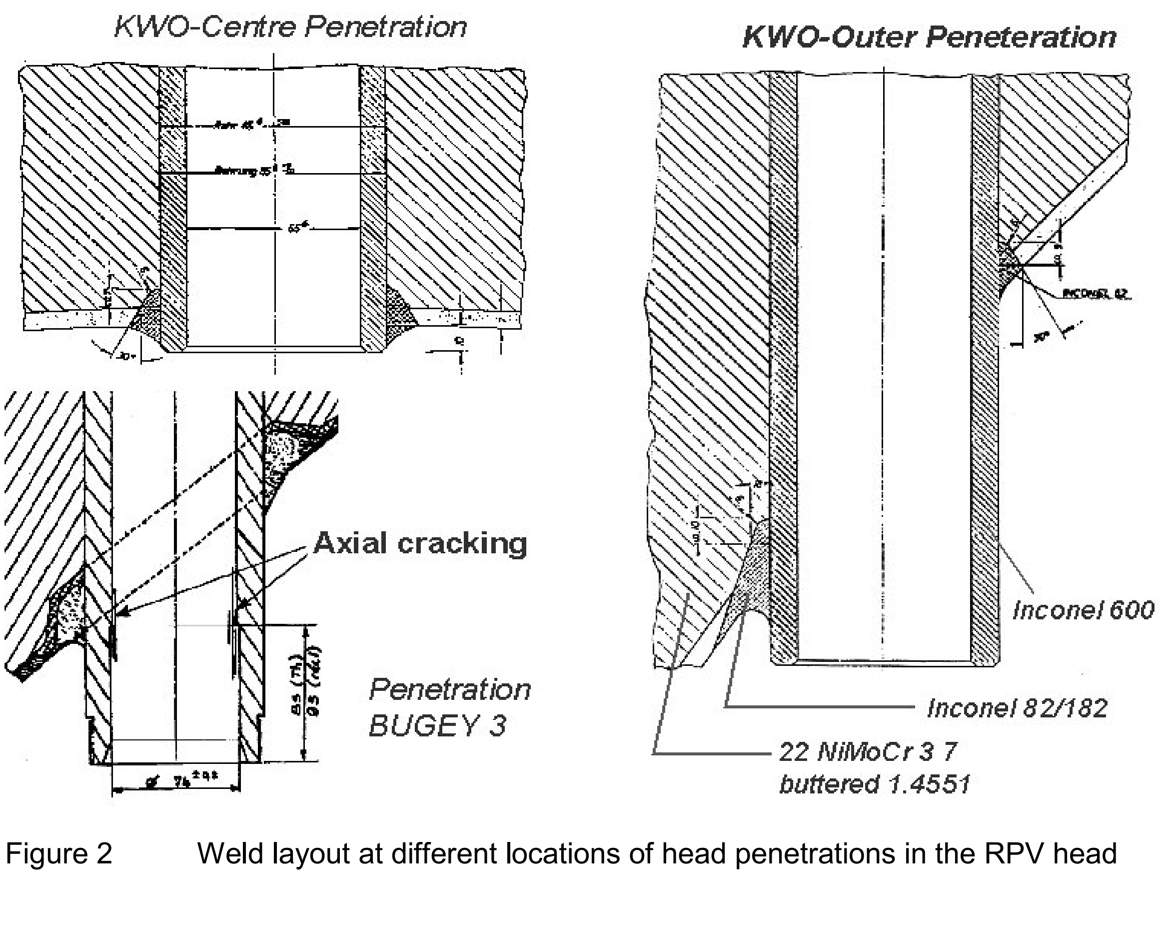

Figure 2 Weld layout at different locations of head penetrations in the RPV head

II.B Operational Loading

In combination with a detailed finite element analysis (FEA) an operating temperature of 295°C of the weld area at the penetrations was calculated. In comparison to plants that have suffered from cracking the operating temperature at KWO is approximately 20 K lower resulting in a significantly decreased probability of stress corrosion cracking.

The combined effects of pressure testing, normal operation and the state of assembly were taken into account in order to determine operational hoop stresses. These were calculated to be at levels of approximately 200 MPa (located at weld areas). This corresponds to approximately 75 % of the material's yield strength in the solution-annealed condition.

Fatigue analysis of the penetration weld area for a 40 year service life showed a 24% utilization of the component's fatigue life. The calculation included conservative assumptions with respect to the type and frequency of load transients. In total the mechanical loading of penetration tubing is relatively low.

II.C Assessment of PWSCC Behavior

It is well known that Alloy 600 is susceptible to PWSCC. One of the most important factors affecting the PWSCC behavior is the microstructure. Mill annealing at sufficiently high temperatures (> 1000°C) maintains enough carbon in solid solution resulting in low transgranular precipitation and with preferential grain-boundary carbide precipitation. A subsequent heat treatment at 700 °C for several hours (thermally treated) results in semicontinuous grain-boundary precipitations and improves PWSCC resistance.

In KWO the susceptibility of Alloy 600 as well as of Alloy 182 to PWSCC is reduced by PWHT. The magnitude of residual stresses is reduced down to a level that is considered not high enough for initiating PWSCC (or only after a very long time). Looking at the issue from a materials point of view, the low stress level at KWO head penetrations is one of the decisive criteria for the resistance to stress corrosion cracking at this location.

Due to carbide precipitation during mill annealing at 1050°C and during heat treatment at 600°C the PWSCC resistance of Alloy 600 is increased (roughly TT condition versus MA condition). This effect is confirmed by several investigations on "thermally treated" Alloy 600 and 690 material ("RUB" - split tube reverse U-bend-specimens in high temperature water), see e. g. [1].

According to published literature, potential crack propagation due to SCC in heat treated Alloy 600 in typical PWR coolant at a temperature of 310°C showed rates of approximately 1 mm/year [2].

Similar behavior can be stated for Alloy 182 weldments. The magnitude of residual stresses is also reduced by PWHT. A stress relief heat treatment seems also to have beneficial effects on reducing crack growth rates (CGR) [2], as shown in Figure 3. Considering the operation temperature of KWO (295 °C), CGRs in the range of 1 mm/year can be estimated for heat treated Alloy 182.

Considering the wall thickness of 10 mm for the nozzle and the frequency of KWO inspections, a crack will always be detected before becoming a through-wall defect causing leakage.

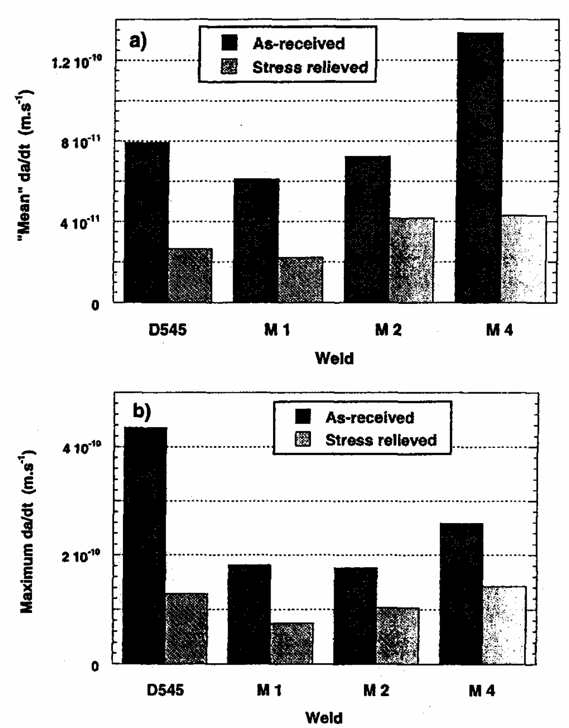

Figure 3 Effects of heat treatment on SCC susceptibility of Alloy 182 [2], results of CGR tests carried out in primary water at 330 °C and an initial K_I of 20 MPaKm

II.D Leakage Detection

The only leakage that has been observed in the upper head area at KWO did not originate from the CRDM nozzle tubing but from the flange gasket located at the upper end of the penetration. This gasket, made from martensitic stainless steel exhibited corrosion resulting in four minimal leakages up to approximately 20 l/day (which corresponds to 0.004gpm). In 1996 all conical gaskets were replaced and no further leakages have been detected since.

At KWO the sensitivity for leakages originating from the primary circuit is very high and several global and local detection systems are available. On a global scale the plant environment is monitored with respect to inert gas activity, temperature and humidity/condensate levels in air recirculation coolers. Additionally, in the case of a suspected leakage, sampling and detailed examination of the plant environment is performed up to three times a day.

For local leakage monitoring of penetration flanges, the very sensitive system BLISS (Bartec Leakage Indication Sensor System) is installed that can detect volumes of water as low as 1 cm^3 (corresponding to 0.00026 gal), see Figure 4. In addition, another very sensitive system FLÜS (Feuchte Leckage Überwachungs System - Humidity Leakage Monitoring System) is installed on the RPV head amongst the protruding penetrations, see Figure 5. This system allows both the detection of actual leakage rates and location of the problem area. This system detects leak rates as low as 1 liter/hour (corresponds to 0.0042 gpm).

Figure 4 BLISS Sensor Cable installed on RPV Head

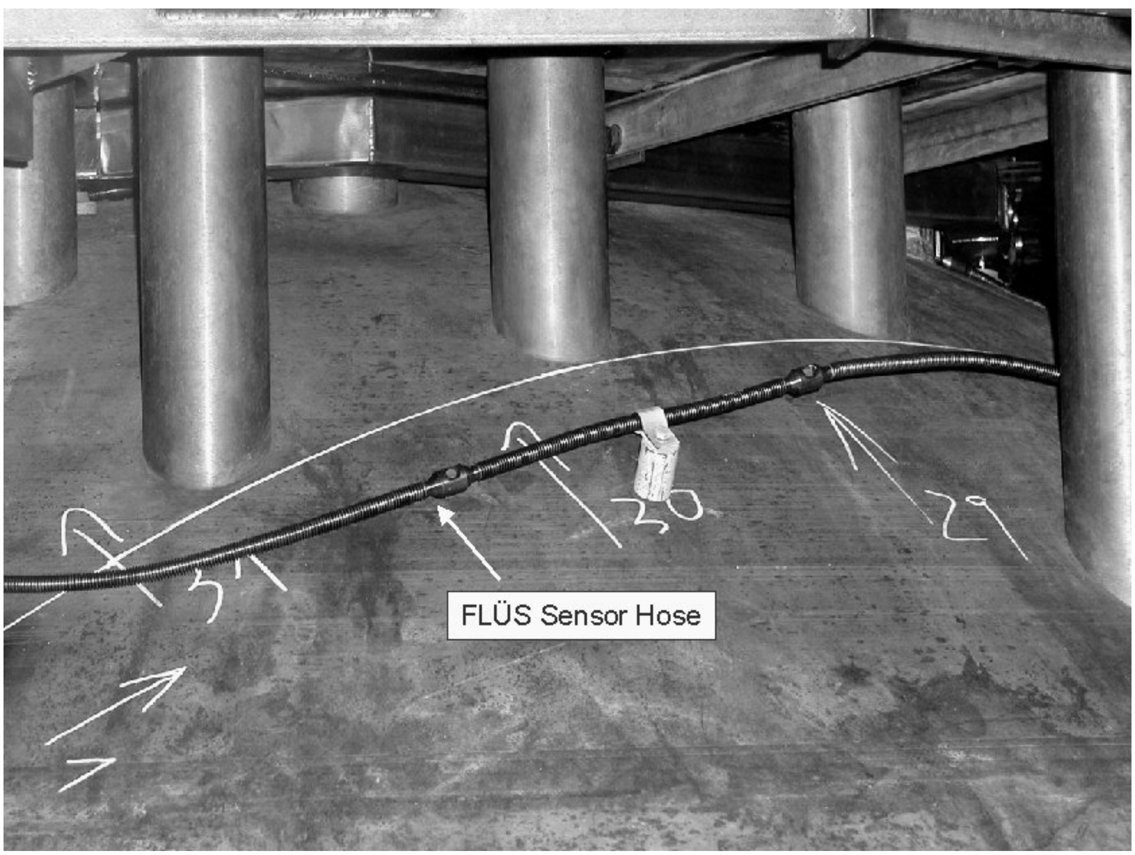

Figure 5 FLÜSS Sensor Hose installed on RPV head of KWO

II.E Inspection and Testing

The RPV head is equipped with a removable insulation cover allowing thorough visual inspection of all components with respect to leak-tightness and boric acid deposits during the annual plant outage.

In the years 1994 and 2000 all penetration nozzles were inspected using eddy current techniques and no indications were detected. Closure head ligaments were inspected using ultrasonic testing (phased array technique of 100 % volume). The primary focus during this inspection was on failures possibly initiating from the bores of the nozzles. This type of testing has been performed at four year intervals since 1976. No indications of failures were found.

III SIEMENS/KWU PWR PLANTS OTHER THAN KWO

With respect to stress corrosion cracking of Alloy 600 head penetrations all other German PWR plants have been considered, as well.

There are fundamental differences in design, choice of material and thus corrosion behavior between these PWRs and those that have experienced cracking.

III.A Materials and Manufacturing Process

In German PWR plants all head penetrations are manufactured from extruded compound tubes with a load bearing substrate equivalent to carbon steel, German designation St 52-4, in the quenched and tempered condition. The cladding bonded by an extrusion process has a thickness of 2,5 mm and is manufactured from Nb-stabilized austenitic stainless steel, German material No. 1.4550 (corresponding to Type AISI 347 stainless steel). The head penetration ends are fully buttered with the Nb-stabilized austenitic stainless steel (welding consumable for Type 347 stainless steel), see Figure 6. After manufacturing/fabrication the penetration nozzles are stress relief annealed.

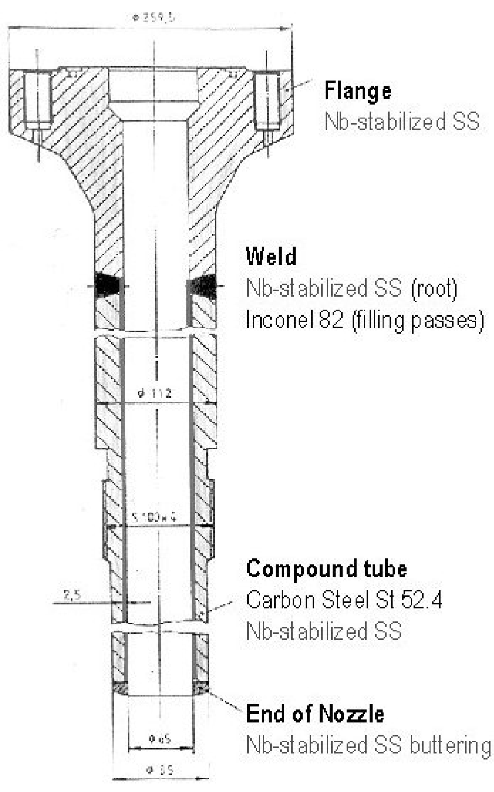

Figure 6 Head Penetration Nozzles: Compound tube St 52.4/1.4550; end of nozzle buttered with Nb-stabilized austenitic SS

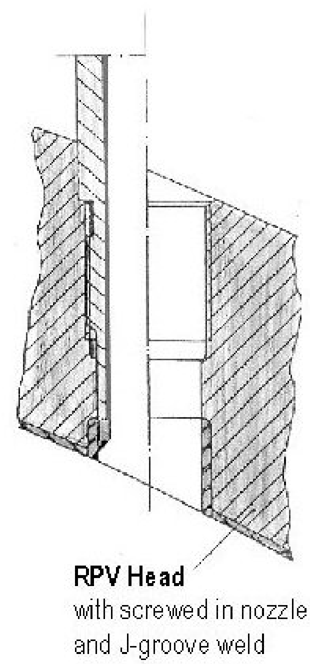

Figure 7 Section of RPV head with screwed-in and seal welded nozzle end

The essential difference in design lies in the fact that the penetrations are not shrink fitted but threaded and seal welded into the RPV head, see Figure 7. The weld preparation of the internally cladded RPV head includes buttering with Nb-stabilized stainless steel followed by the actual welding which was also performed with a Nb-stabilized welding consumable.

This results in a clear separation of tasks. The threaded portion of the penetration tubing bears operational loads whereas the weld is solely responsible for sealing purposes. This particular design and the manufacturing sequence allows the joining of individual stress relieved, ready-machined components with only minor residual stresses resulting from fabrication.

With respect to the failure mechanism for the Alloy 600 penetrations, an essential design feature is the cladding of all coolant exposed surfaces with the Nb-stabilized stainless steel. This material exhibits SCC resistance under PWR primary circuit conditions.

III.B Flange Connection

The flange connections of control rod and core instrumentation nozzles are equipped with a double gasket (inner and outer gasket) manufactured from the Ti-stabilized austenitic stainless steel German material No. 1.4541 (corresponding to Type AISI 321 stainless steel) as shown in Figure 8. Every flange connections is equipped with a testing line enabling leak-tightness testing with vacuum techniques (see Figure 9).

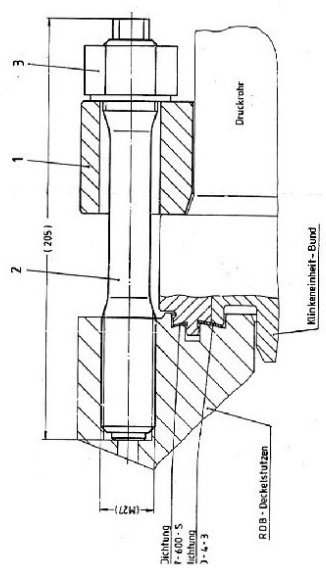

Figure 8 Flange design with double gasket

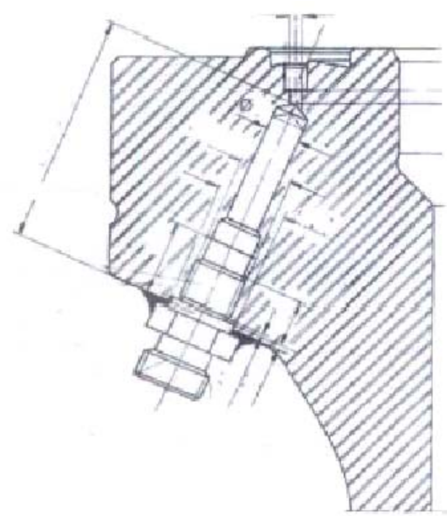

Figure 9 testing line connection for leak-tightness testing

To date, leakage at the core instrumentation nozzles has been detected once during a pressure increase during leak-tightness testing of the RPV head. The actual detection was the result of aerosol activity and repeated starting of the sump pump. The reason for this malfunction was an improper assembly of the flange connection. Correct re-assembly (opening, cleaning and gasket replacement) led to satisfactory results. No damage of the carbon steel components caused by this “single event” wetting were recorded.

III.C Inspections and Findings

In the context of evaluating the RPV head a comprehensive annual visual inspection is conducted during the outage. As can be seen in Figure 10 and Figure 11, there are no constraints from encasings or insulation material. Boric acid deposits, even very small ones, would have been detected with certainty.

As stipulated in KTA 3201.4 non-destructive testing is conducted at intervals of 4 and 5 years, respectively. Since 1989 ultrasonic testing (phased array technique) is being utilized to inspect closure head ligaments of the entire internal and external RPV head surfaces.

No inspections conducted at Siemens/KWU plants have revealed any reportable indications.

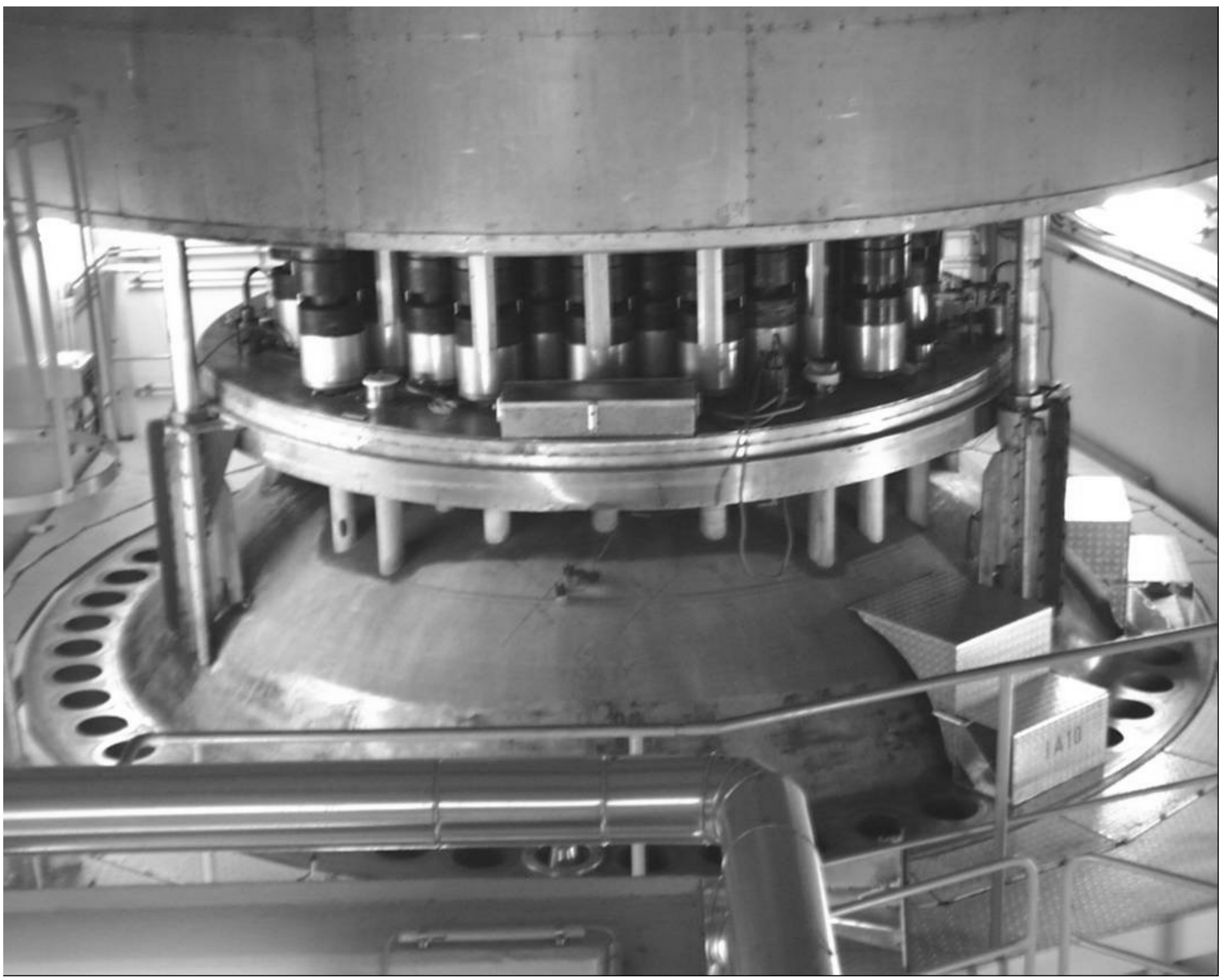

Figure 10 RPV head on its storage location during outage

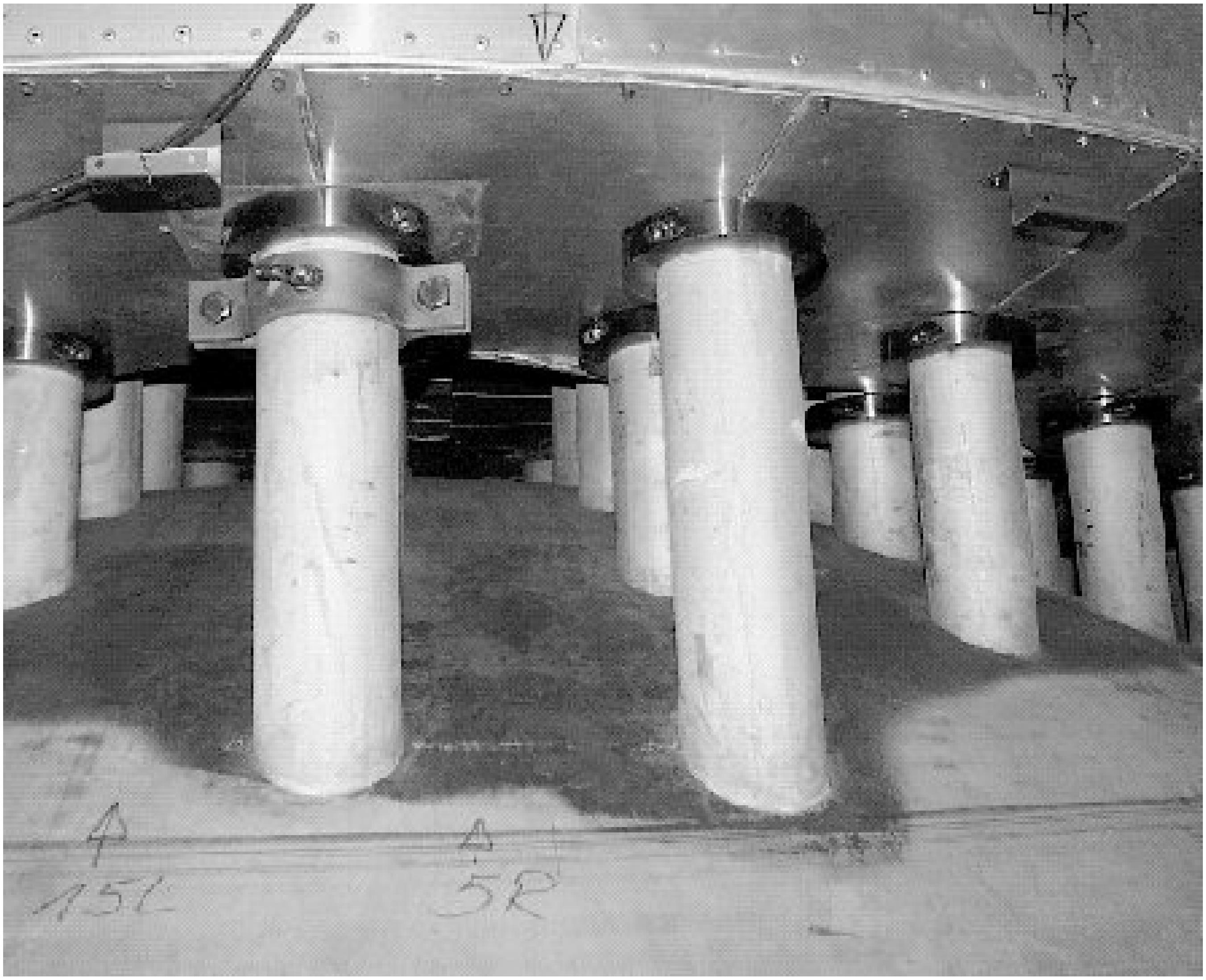

Figure 11 Good accessibility to RPV head penetrations in Siemens/KWU plants

IV SUMMARY

The structural integrity of head penetrations at the NPP Obrigheim is considered sound because of the following facts:

- thermal history of penetration material included a solution and stress relief annealing resulting in a low level of residual stresses and less susceptible microstructure
- good accessibility allowing regular visual and comprehensive non-destructive inspections
- sensitive leakage detection systems on a local and global scale.

Therefore it can be concluded that cracking of Alloy 600 CRDM nozzles, as encountered in other plants of similar design is very unlikely to occur in the NPP Obrigheim.

All other German PWR plants exhibit fundamental differences in upper head design and choice of material. Cladding of all surfaces exposed to the coolant with Nb-stabilized stainless steel material No. 1.4550 greatly reduces the SCC susceptibility. Individual stress reliefing of penetrations, in combination with a threaded versus a shrink-fitted design has minimized residual stresses. Based on the current state of knowledge susceptibility to IGSCC in head penetrations and leaks in those areas cannot be envisaged.

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Activities on Alloy 600 Cracking of PWRs in Japan

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Abstract

This paper introduces an assessment of and maintenance programs on PWSCC of Alloy 600 for reactor vessel head (RVH) penetrations and bottom mounted instrumentation (BMI) nozzles in Japan.

[RVH Penetration Nozzles]

RVHs for 11 plants have been replaced and T-cold conversion has been adopted in the 11 other plants as a mitigation measurement.
Non-destructive inspections (ECT for nozzles) for 6 plants in which T-cold conversion has been adopted were performed in 1993 and 1994, and no indication was detected. Some of these plants are scheduled to implement inspections of vessel head penetration nozzles and the J-welds in the future.

[BMI Nozzles]

It has been assessed that BMI nozzles for Japanese PWR plants are potentially susceptible to PWSCC and mitigation of PWSCC is needed to maintain integrity until the end of plant life. ECT inspection and water jet peening technology for BMI nozzles have been performed as a mitigation method for preventing PWSCC before the estimated initiation time for PWSCC. Moreover, an emergency repair method and a replacement technology have been developed in case leakage from the BMI nozzles is encountered.

[Hot Leg Nozzles]

Following the observation of cracking at V.C.Summer , UT from the inside or outside of the hot leg nozzles has been performed and no indication has been reported to date. In case a need for emergency repairs arises, a spool piece replacement technology and installation equipment have been developed. Water jet peening technology as a mitigation measurement is also under development.

1. Introduction

PWSCC of alloy 600 became an issue in the beginning of 1986 in the small-diameter tubes of pressurizers designed by Combustion Engineering (after first appearing in the heat transfer tubes of steam generators). Then in 1991, a small leak was found on the outer surface of the upper head of Bugey Unit 3 in France, which turned out to have been caused by cracking of an upper head nozzle by PWSCC. Domestically, the PWSCC event at Bugey Unit 3 caused much concern and therefore countermeasures were taken to replace the upper heads of some reactors or lower the operating temperature of others.

In the context of upper head nozzle PWSCC, alloy 600 (including weld metal) was studied further and the data obtained showed that the weld metal is more susceptible to cracking than the base metal. However, cracks in the weld metal of operating plants had not been found until PWSCC cracks were identified at Ringhals Units 3 and 4 and at the V. C. Summer reactor vessel outlet nozzle safe end in 2000.

PWSCC is affected not only by weld residual stress but also by surface finish (buffing, machining process, etc), and its evaluation has not been clarified sufficiently. Since V. C. Summer, however, PWSCC cracks of alloy 600 weld metal have been reported in the upper head J-welds of some U.S. plants, and it seems evident that it is indispensable to maintain the integrity of alloy 600 components which is not only base metal but also weld metal.

In this review, an introduction relative to the current status of PWSCC evaluations and preventive maintenance is given for the reactor vessel pressure boundary locations using alloy 600.

2. PWSCC susceptibility evaluation and maintenance program for components using alloy 600.

Sections of the reactor vessel pressure boundary using alloy 600 are as shown in Fig. 1. Table 1 summarizes our evaluation of the PWSCC susceptibility and required maintenance measures for each of these sections.

2.1 Upper head nozzle

In the U. S. and Europe, a number of cracks caused by PWSCC of upper head nozzles, including some leading to leakage, have been reported, while cracks at J-welds have also been observed. Domestically, on the other hand, some slight roughness has been observed in the base metal of nozzle in an investigation of the old retired upper head at Takahama Unit 2.

In the stress evaluation of the relevant sections, the base metal of the nozzle and the J-weld are both under a tensile stress exceeding the yield stress. The surface of the J-weld, however, is thought to be under a compressive stress due to pre-service buffing. The operating temperature has also been reduced to T-cold. In the PWSCC sensitivity evaluation, the base metal is deemed sensitive while the J-weld has low sensitivity. As regards countermeasures taken so far, 11 plants have been replaced with the upper heads employing alloy 690 nozzles. Newer plants have implemented T-cold conversion. Also in the period 1993 through 1994, ECT of the nozzle base metal was carried out at 6 plants, with no indication being detected. There has been no request so far for inspection from Japanese regulator At several T-cold plants, however, inspections of upper head nozzles and J-welds are planned.

2.2 BMI nozzle

Domestically following ECT of the BMI nozzle base metal, a small indication was detected at Takahama Unit 1. In the U. S., leaks were found in two nozzles at South Texas Unit 1and other axial defects were detected in the base metal.

According to a stress evaluation, the nozzle base metal and J-welds are both under a tensile stress exceeding the yield stress. The surface of the J-weld, however, is thought to be under a compressive stress due to pre-service buffing and the temperature is T-cold. As regards the evaluation of PWSCC sensitivity, the base metal is judged to be sensitive while the J-weld is considered to have low sensitivity.

Concerning inspection and preventive maintenance measures taken so far, ECT from the internal surface and Water Jet Peening (WJP) technology have been developed for the BMI nozzle base metal[1], and have been applied to 7 plants so far. Fig. 2 shows the concept of WJP applied to the internal surface of the nozzle and the application system. Fig. 3 shows that residual stress down to a depth of more than 0.5 mm from surface layer can be improved into compression by WJP. The inspection technology and WJP for the J-welds are now under development.

For the emergency repair method in case any defect is found, a cap repair technique from the outer vessel surface has been developed. Fig. 4 shows the concept of the cap repair method. In this case, a Type 316SS cap is welded to the BMI using existing Inconel cladding on the outer surface side of the BMI vessel penetration, so that a new pressure boundary may be formed. A BMI replacement technology has also been developed.

2.3 Outlet nozzle safe end

Cracks due to PWSCC were found in the outlet nozzle safe end welds at V. C. Summer in the U. S. and at Ringhals Units 3 & 4 in Sweden. Domestically, on the other hand, UT from the nozzle inner/outer surfaces is being used and so far no indications have been detected.

As to the stress evaluation, the stress seems to be tensile and close to the yield stress while the temperature is T-hot. Evaluation of the PWSCC susceptibility has indicated that this component is potentially sensitive to cracking.

For possible maintenance measures devised so far, a spool piece replacement technology and its installation methodology have been developed as an emergency repair in the event that any defect is found. WJP technology from the inner surface of nozzles is now under development as a mitigation measurement. Fig. 5 shows the concept of the WJP technology for the inner surface of the outlet/inlet nozzle safe end.

3. Summary

In this review, the current situation in Japan relating to the evaluation of PWSCC and maintenance measures for alloy 600 have been introduced and described, in particular, for the pressure boundary locations of the reactor vessel.

In many cases, it takes a long time to restore public confidence and carry out repairs if component cracking occurs in Japan. It is important, therefore, to be able to predict service life and to take preventive maintenance measures. PWSCC of alloy 600 is, above all, a generic issue for PWR plants and it is essential, therefore, to discuss and prepare preventive measures for maintenance based on an evaluation of the risk of PWSCC occurring. Evaluation of PWSCC service life has been attempted for alloy 600 base metal and mitigation measures have been partially applied to actual plants. In the future, it is believed to be a very important issue as to how to evaluate PWSCC and the mitigation techniques to be applied to the J-welds and safe end welds.

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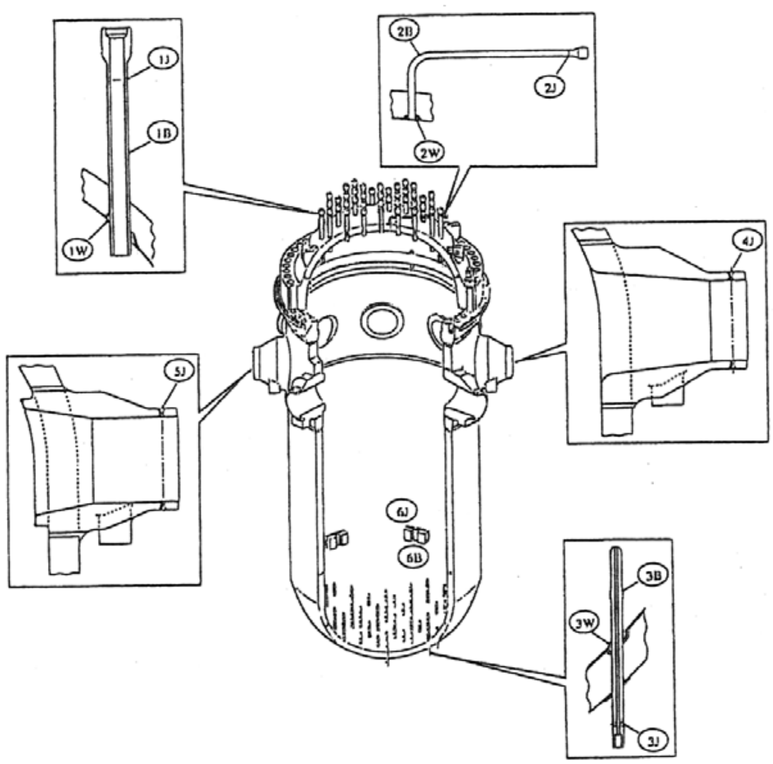

Fig.1 Alloy 600 components of the RV pressure boundary

Table 1 PWSCC susceptibility evaluation and maintenance measures

: Ready for field application : Under development		RVH Penetration Nozzle		BMI Nozzle		Hot Leg Nozzle
		Base metal	J - weld	Base metal	J - weld	
Cracking	Japan	Takahama #2 (<1mm)	No Indication in Inspected	Takahama #1 (≤1mm?)	No Indication in Inspected	No Indication in Inspected
	US & Europe	Many	More than 10 plants	South Texas 1 (under investigation)	No report	V.C.Summer Ringhals
PWSCC Assessment	Stress	Tension (> σ_y)	Tension (> σ_y) (Surface: Compression)	Tension (> σ_y)	Tension (> σ_y) (Surface : Compression)	Nearly equal to σ_y
	Temperature	Converted from T-hot to T-cold		T-cold		T-hot
	Susceptibility	Moderate	Low	Moderate	Low	Moderate
Inspection	Technique	ECT UT	VT ECT	ECT UT	VT ECT	UT ECT
	Application	17 plants	Not yet	6 plants	Not yet	5 plants
Mitigation		Done (T-cold conversion)		WJP (5plants)	WJP	WJP
Counter Measure or Contingency		RVH Replacement (11plants)		Emergency Repair Method BMI Nozzle Replacement		690 Cladding Spool-piece Replacement

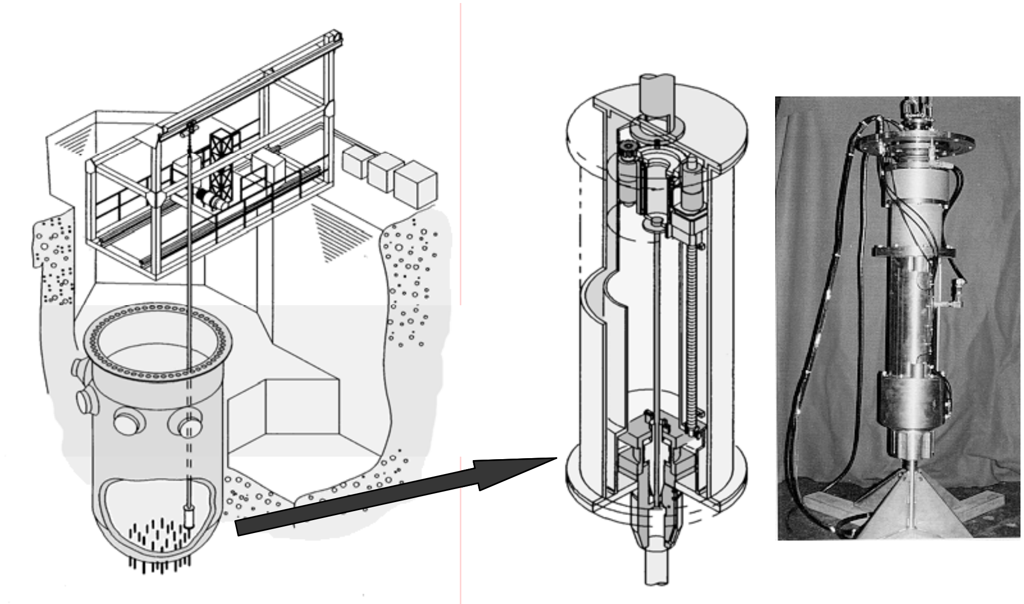

Fig.2 Concept of WJP for BMI nozzle and the system

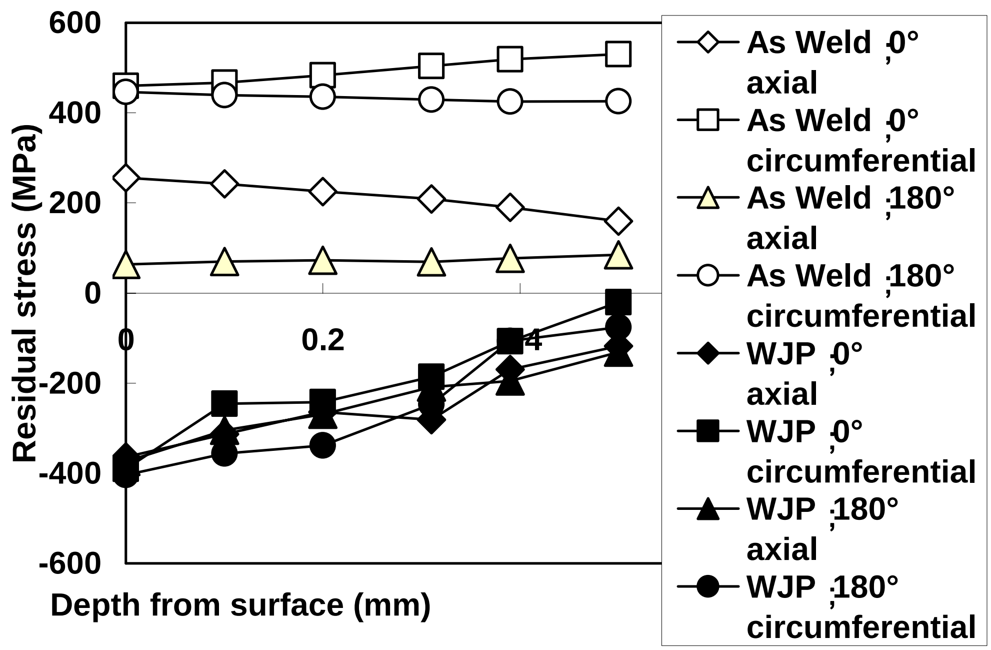

Fig.3 Results of residual stress measurement before and after WJP

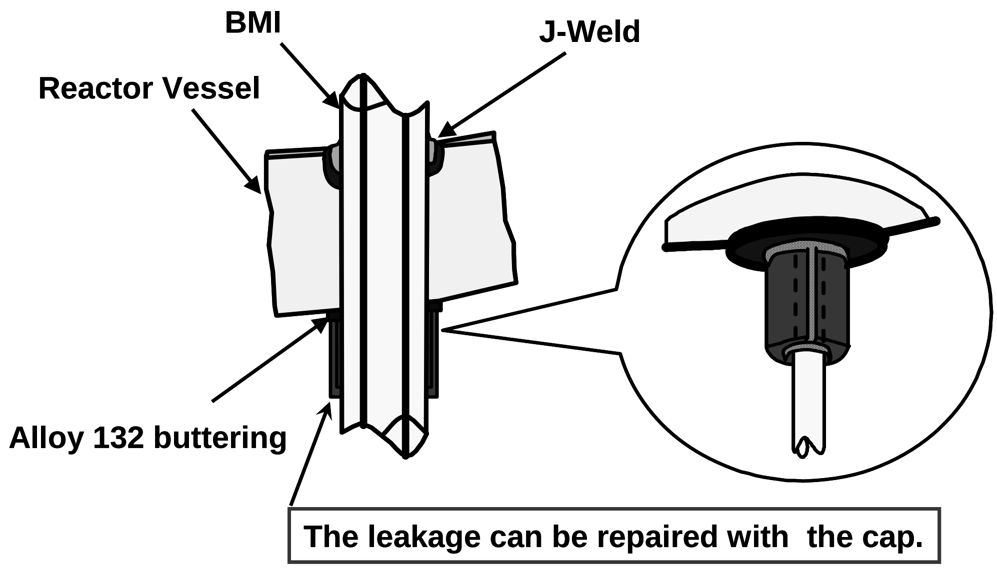

Fig.4 Concept of cap repair method for BMI nozzles

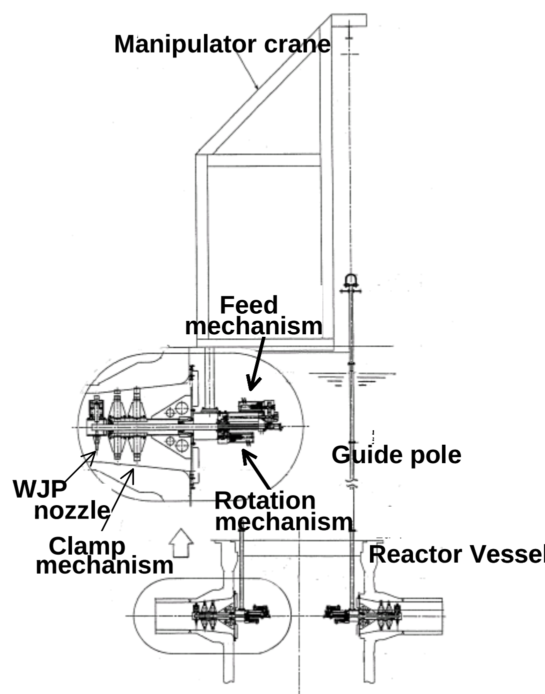

Fig.5 Concept of WJP for outlet/inlet nozzle safe ends

RPV Heads: Inspection versus Replacement Strategies for Ringhals PWR Units 3 and 4

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Abstract

During the last decade, shallow defects has been monitored in the Alloy 600 vessel head penetrations of the most modern Swedish PWR stations, Ringhals units 3 and 4. Within the control order that was applicable until the issuance of the regulatory guide SKIFs 1994:1, the interval of these control areas was inspection every third year. As part of the current control order, a more rigid system regarding both the inspections methodology, i.e. ”qualified inspection systems” and the reasoning behind the inspection program, i.e. defect tolerance analyses has been introduced.

A major part of any defect tolerance analysis is a crack growth rate assessment with respect to elapsed time. A large effort has been made to enable reasonable inspection interval with respect to crack growth in both Alloy 600 and its weld metal Alloy 182. The evaluation of the database and its interpretation and transformation into a growth rate law is shown in the presentation. The accredited third party body in Sweden has scrutinized this work. Some key issues had to be solved in cooperation with the Swedish Nuclear Power Inspectorate, the SKi, since the third party body at that time did not have the necessary competence in some key issues.

Due to the result of the analysis, small detection targets was to be applied as a result of the relatively rapid crack growth. This, together with large difficulties connected with a complex qualification process, led to the decision to exchange the vessel heads. The reasoning behind this decision is primarily due to costs of inspections since the inspection interval has been drastically reduced under the new control order, and the estimated residual lifetime of the power plants. The replacement heads are fitted with Alloy 690 penetrations welded with Alloy 52. In addition to this the head are plastically formed out of one piece, thus there are no butt weld at the vessel head, further decreasing the inspection demands.

INTRODUCTION

During the refueling outage of 1993, defects were detected in the J-groove weld of some reactor vessel head penetrations of Ringhals 2. The weld metal was Inconel 182, connecting the stainless steel cladded low alloy vessel head with the penetrations made of Inconel 600 forgings. The primary defect of concern was determined by NDE to be a 180 mm long surface breaking defect oriented circumferentially around penetration #62 in the J-groove weld. The vessel head had at that time been in service for 18 years, or approximately 140000 EFPH. Eventually the finding led to the replacement of the vessel head of Ringhals 2 in the middle of the 1990's. The discovery of the defects also led to an increased interest to investigate the other two vessel heads in Ringhals, those of the slightly more recently commissioned units 3 and 4. After extensive inspections during several campaigns using inspection methods that are qualified for this specific purpose, a number of shallow defects have been identified and monitored in these two plants in the Alloy 600 vessel head penetrations during the last decade. Within the control order that was applicable until the issuance of the regulatory guide SKIFs 1994:1, the interval of these control areas was inspection every third year with an inspection program based on a validated inspection system. This inspection order was in power until 1996 after which the utilities were supposed to have implemented the new regulatory guide in the quality assurance systems. As part of the new control order, a more rigid system regarding both the inspections methodology, i.e. "qualified inspection systems" and the reasoning behind the inspection program, i.e. defect tolerance analyses has been introduced in order to verify the need for specific inspection capability as well as the inspection intervals. The current valid regulatory guide SKIFs 2000:2 is more general in its appearance and rely more heavily on the power companies to suggest guidance for the control order and to subject those suggestions to review of the regulatory body, the SKi, for acceptance. Within this control order, risk informed inspection may be included if this is found appropriate by all involved parties.

INSPECTION QUALIFICATION AND RESULTS

The qualification methodology can from a materials science perspective essentially be divided into three steps. The first step is the Defect and Degradation mechanism analysis, the DOS-analysis. The purpose of this study is to pin-point active and relevant degradation mechanisms and to quantify the relative risk both with respect to reactor safety and likelihood of the mechanisms. It includes taking into account any experience from other related areas and to ensure the correct inspection means. One basic function of the control order is to make certain that areas where there are definite causes for inspections, are included. Thus the phrase: "Inspection by Cause" can be introduced as the guiding light. The analysis sums up all kind of likely, probable and unlikely defects that can be identified, service induced and manufacturing induced, and ranks them from a relevance, a significance and a probability point of view. The next step is to investigate the most intriguing defect type that may occur in he inspection area. This step is generally called WCD, or Worst Case Defect, analysis. This step includes not only the most difficult type of defect that can be expected from a service induced degradation point of view, but also any kind of manufacturing defects that may have been missed during pre-operation inspections or previous In-service Inspections. This also implies that by this time, a suitable technique for inspecting the area has to be proposed.

When both these two steps have been taken it is possible to assess the defect tolerance of the inspection area. When qualifying an inspection system there is a great need for an accurate, not overly conservative and complete defect tolerance analysis. In Sweden the currently used methodology is based on the safety evaluation system of the ASME XI code coupled with the R6-analysis methodology[1]. The analysis is performed in several steps, of which the acceptance step is the first. When an acceptable defect size has been established an inversed crack growth analysis can be performed. If the capability of the inspection technique is fairly well known, an inspection interval can be derived from this analysis, figure 1. If the inspection technique is less well known, a desired inspection interval can be used to evaluate the demands on the inspection technique instead. The entire scope of analysis is then review and, if judged to be: complete-correct-concise-transparent-traceable, C_3T_2, accepted by the accredited third party body. When the scope is complete, the qualification process begins at the accredited qualification body, SQC, that eventually releases a qualification report.

When establishing the control program there are thus several parameters to assess. If these parameters are reviewed carefully it can be seen that most of them are fairly well described and/or known or code and practice dependent. This includes service induced stresses, residual stresses, acceptable defect sizes, detection targets and tolerances etc. In fact there is within the current Swedish practice virtually only one parameter that still is under discussion that can be said to have major influence on the inspection interval: the crack growth rates. The assessment of valid and accurate is thus a critical step to ensure safe and economic operation of the power plants.

The assessment process utilized in Sweden have been described in detail elsewhere[2], therefore only a very short review is included here. The basis is an accepted data base where all data have undergone an aggressive screening in order to verify that only good quality data is used in the Crack Growth Rate Assessment. The screening criteria, or filters, include test procedure requirements, environmental issues and mechanical issues. In figure 2, an example is given for the assessment of a crack growth rate law for Alloy 600 is presented. Generally an upper bound of the resulting acceptable database is utilized, not implying that this is an experimentally and statistically determined upper bound. In figure 3 the proposed crack growth rate law, CGR for the weld metal Inconel 182 is presented using the same screening criteria as above. In summary the screening gives by hand that we have ample evidence to suggest a stress intensity level, K, dependent part of the CGR at low to medium high K-levels. However, depending on how the data is treated it may be difficult to demonstrate and prove the CGR behavior at low stress intensity levels, as well as the behavior at high stress intensity levels. Recent investigations performed in this area indicates that there is plateau value, close to $K=30\text{MPa}\sqrt{\text{m}}$ above which the crack growth appears to be relatively independent of increasing K. No threshold value with respect to observable crack growth has been established or even suggested. Even though from an engineering point of view it is obvious that when the crack growth rates below 10^{-11} mm/s, or less than 0.5 mm/operating years are suggested as being of no structural concern. These data however indicate the need to enhance the knowledge and data base at both high and low K levels to give higher accuracy to the assessment. It is however a general trend regarding environmentally assisted cracking that there can be found three different areas of interest, see figure 4. In the first stage, stage I, the stress level is the determining property, In stage II, the local chemical properties such as surface and/or interface properties replaces the mechanical properties as rate determining, whereas in stage III, the mechanical over loading of the specimen again replaces the surface and interface properties as rate determining. During the review process several issues regarding acceptance criteria of specific data points as well a the general acceptance of temperature transfer regarding the question whether SCC may be regarded as a thermally activated process thus allowing the transfer from one test temperature to another using an Arrhenius equation. Most of these issues was accepted by the regulatory body, with the exception of a general acceptance of a plateau level, something that was regarded to be judged from case to case.

During the inspections, a total of 26 surface breaking defects have been found in the vessel head penetrations of the two units during the most recent inspection campaign 2000 and 2002. These inspections have in common that have been carried out using qualified inspection systems. From the inspection reports it is suggested that all of the defects are less than 2 mm in through wall extension, TWE. This is derived from the fact that the defect can be detected but not accurately sized and the detection target in TWE is 2 mm in depth. The length of the defects is varying from less than 4 mm up to some 18 mm. In 16 of the penetrations, defect clusters similar to that of the “crackled surface” often associated with thermal fatigue has been detected.

ANALYSIS

There has been a considerable effort made to evaluate and establish a comprehensive control program for the vessel head penetration welds of the PWR-plants. The situation can be summarized as:

- Rapid crack growth
- Small detection targets
- Complicated qualification process due to small detection target and
- Short inspection intervals

With this in hand it is easy to draw the conclusion that as long as Environmentally Assisted Degradation is active and an issue, it may be efficient to consider replacement o the vessel heads instead of continued operation if degradation can be excluded during operation. The current technology is based on penetrations manufactured in Alloy 600 and welded onto the vessel head with Inconel 182 weld material. Both these are known to be susceptible to stress corrosion cracking in PWR primary water conditions. As replacement materials the materials used are Alloy 690 forgings and Inconel 52 or 52M weld material. These materials have been suggested to have a greater resistance to EAC than the previously used materials and combinations. It is also possible with modern technology to produce a vessel head that has no butt welds. This will further decrease the need for In-Service Inspections and increase the probabilistic safety of the power plants.

With these data in hand, the next step is to estimate the costs for the remaining expected technical life. The costs of inspections, including the qualification costs are fairly well known and quantifiable. So is the cost of a replacement vessel heads. The largest individual cost however is that of prolonged or forced outages. This cost is dependent on what kind of contracts the power plants have regarding sales of the power. In this case not only the inspection and/or repair costs are included but also loss of revenue and maybe also costs of for the company. When these costs are summed it was seen that the best way of defending both the availability and the cost efficiency of the power plant was to recommend the units to replace the vessel heads. The replacement head have been ordered for delivery in 2004 and 2005 respectively. The heads are made from a single forging, with penetration in Alloy 690 welded onto the head with Inconel 52 weld metal.

This does however put us in a familiar situation; there is still a need for a comprehensive inspection program despite using more resistant materials. Even though the materials used in the current design are known to have larger resistance to EAC there still exist a need for hard data to support this statement. Thus the replacement causes a great deal of engineering work to be performed, due to the fact that we are leaving the well known situation of Alloy 600 and the derivatives and enter a state were insufficient good quality crack growth data is present in the community. The number of reactor operation years with these types of materials is also limited. It is however so that the resistance to crack initiation is well documented as is the fracture toughness of the material. The field experience, especially from steam generator tubing is also excellent as is the experience from the previously replaced vessel heads, among these, the vessel head of Ringhals unit 2. It should however be noted that simply by exchanging the materials, a complete solution isn't found. If there, for example are thermal fluctuations in the flow this portion may be a determining part of crack initiation. This may place us in the same situation within a shorter or longer time period. Among the residual work to be carried out as a follow up of the replacement is data collection from other RPVHs as well as focused crack growth rate experiments in order to validate a new inspection program. It is judged that the maximum allowable inspection interval of 10 years can be reachable within the coming inspection period. So far however, an interim 3 years inspection interval has been accepted by the third party body.

CONCLUSIONS

With respect to the data presented above and taking into account the cost situation it was an obvious decision taken by Ringhals to replace the vessel heads of the two units. This since it , at that time, estimated that the inspection costs single handed would be larger than the cost of the replacement. In addition to this the replacement is an important step towards Plant Life Extension up to 60 years of operation.

ACKNOWLEDGEMENTS

The authors are grateful to the personnel at the unit-engineering groups at Ringhals unit 3 and 4 for allowing this reasoning to be presented and published as well as commenting the presentation and the final paper.

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2 Pål Efsing and Christer Jansson, Screening of crack growth data and the relevance from an end-users perspective, To appear in Proceedings from 11th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, Stevenson, Wa. 2003.

FIGURES

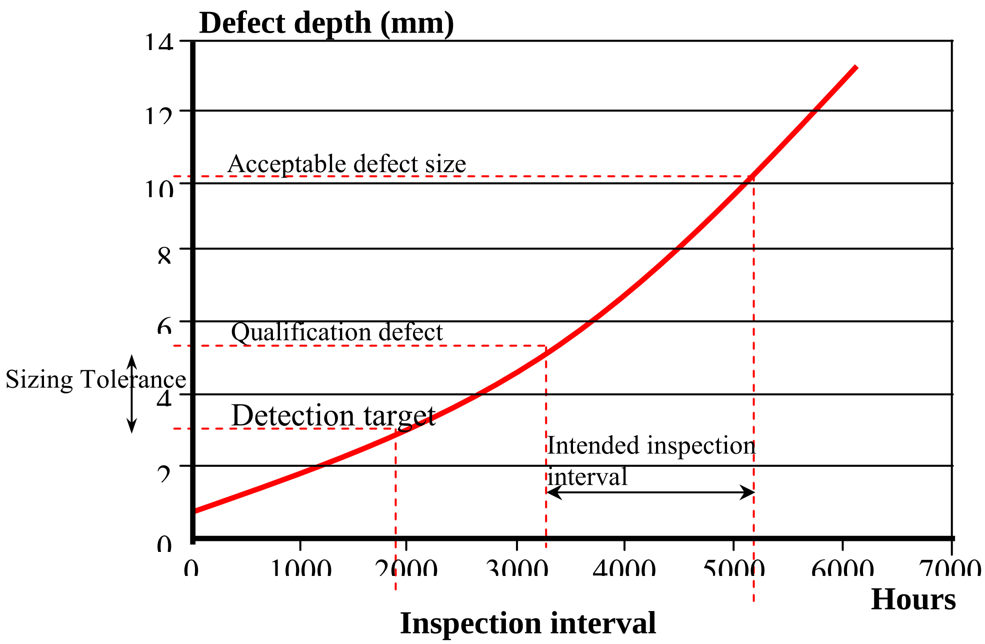

Figure 1, Visualizing the defect tolerance analysis step with regard to the ISI-program.

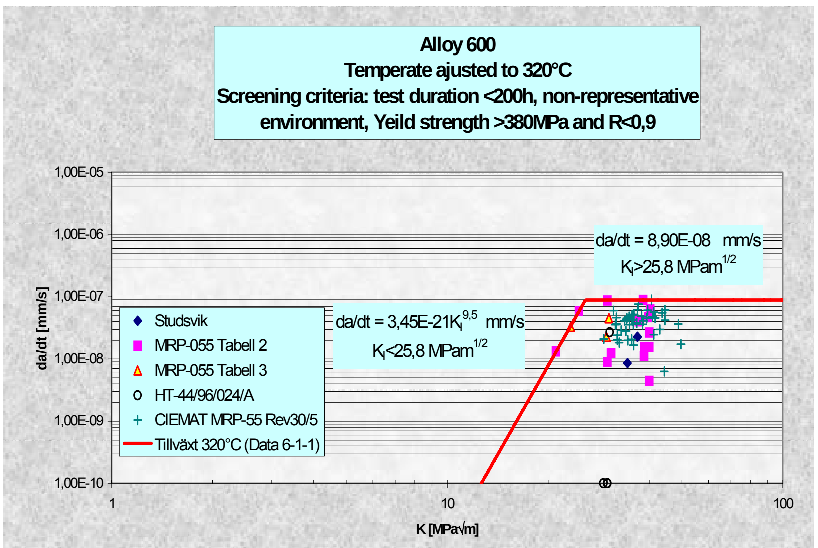

Figure 2, The final results after screening the Alloy 600 database with regards to the criteria presented in [2].

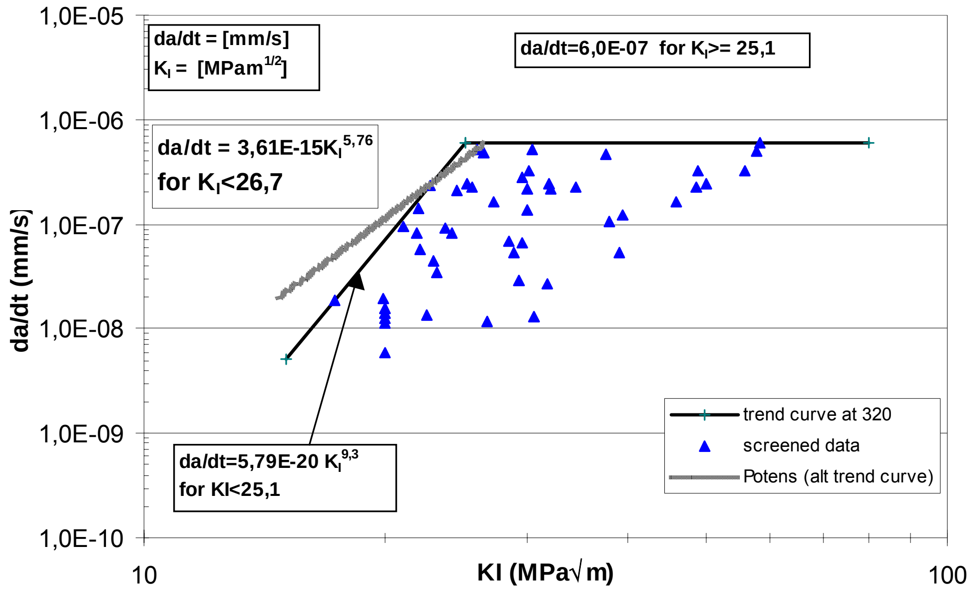

Figure 3, The outcome after screening the database of Inconel 182.

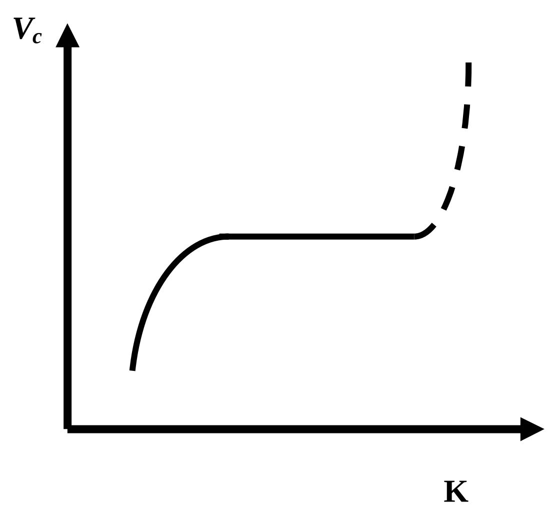

Figure 4, Sketch of the general appearance of an environmentally assisted cracking process with respect to some crack driving force, in this case the stress intensity level K.

Alloy 600 PWSCC Mitigation: Past, Present, and Future

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ABSTRACT

Primary water stress corrosion cracking (PWSCC) of wrought Alloy 600 material and its weld metals has become a major equipment reliability challenge for owners of pressurized water reactors (PWRs). Notable examples of equipment failures include extended outages and emergent repairs or replacements at Calvert Cliffs, V.C. Summer, Oconee Nuclear Station, Davis-Besse, and North Anna. PWSCC requires three key factors present simultaneously: an aggressive environment, susceptible material, and significant, prolonged tensile stress. Eliminating any one of these three factors will mitigate cracking. A number of techniques have been employed in the past and a few techniques are currently being applied to mitigate PWSCC. Examples include shot peening, nickel plating, surface flapping, zinc injection, and abrasive water jet conditioning. This paper reviews the application of the available techniques and provides an insight into a few potential techniques for future application.

INTRODUCTION

Alloy 600 component items were used in pressurized water reactors (PWRs) due to the material's inherent resistance to general corrosion in a number of aggressive environments and because of a coefficient of thermal expansion that is very close to that of low alloy steel. Over the last thirty years, primary water stress corrosion cracking (PWSCC) has been observed in Alloy 600 component items such as steam generator tubes and plugs, pressurizer heater nozzles, pressurizer instrument nozzles, reactor vessel closure head nozzles, and more recently in an Alloy 182 weld attaching an instrumentation nozzle to the lower reactor vessel head. Table 1 provides a synopsis of the Alloy 600 PWSCC experience in commercial PWRs. This table identifies the first commercially observed occurrence of PWSCC for each particular component item in a PWR and lists the approximate service life (in calendar years) at the time PWSCC was identified at that particular location. It seems clear that cracking was first observed at very highly stressed locations in the hot leg of steam generators. Pressurizer nozzles, which operate at the highest temperature in PWRs, were the next locations to have leakage and failures identified. Given more operating time, PWSCC has now spread to the somewhat lower temperatures of the reactor vessel closure heads and hot legs, has also been observed in the cold legs of some steam generators, and has also been observed at high stressed locations in other cold leg locations within the reactor coolant system.

Table 1. Alloy 600 PWSCC Experience in Commercial PWRs

Component Item	Date PWSCC Initially Observed	Service Life[a] (Calendar Years)
Steam Generator Hot Leg Tubes and Plugs	~1973	~2
Pressurizer Instrument Nozzles	1986	2
Steam Generator Cold Leg Tubes	1986	18
Pressurizer Heaters and Sleeves	1987	5
Steam Generator Channel Head Drain Pipes	1988	1
Control Rod Drive Mechanism Nozzles	1991	12
Hot Leg Instrument Nozzles	1991	5
Power Operated Relief Valve Safe End	1993	22
Pressurizer Nozzle Welds	1994	1
Cold Leg Piping Instrument Nozzles[b]	1997	13
Reactor Vessel Hot Leg Nozzle Buttering/Piping Welds	2000	17
Control Rod Drive Mechanism Nozzle/RV Head Welds	2000	27
Surge Line Nozzle Welds[c]	2002	21
Reactor Vessel Lower Head In-Core Instrumentation Nozzles/Welds	2003	14

[a]This listing identifies the first reported occurrence of identified cracking for each component item. Leakage has occurred in some component items in less than one year of service life and in other component items after nearly 30 years of service.
[b]One plant identified "suspect" visual evidence of boric acid leakage; nozzles were preventively repaired without investigating whether leakage had in fact occurred.
[c]Crack-like flaw indications have not been confirmed as PWSCC at this time.

The occurrence of PWSCC has been responsible for significant downtime and replacement power costs at PWRs. Notable examples of equipment failures include extended outages and emergent repairs or replacements at Calvert Cliffs, V.C. Summer, Oconee Nuclear Station, Davis-Besse, and North Anna. Repairs and replacements have generally utilized wrought Alloy 690 material and its weld metals (Alloy 152 and Alloy 52), which have been shown to be considerably less susceptible to PWSCC. To avoid the costly and time consuming component item repairs or replacements associated with PWSCC, a number of preventative measures have been or are currently being applied to mitigate PWSCC. Examples include shot peening, nickel plating, surface flapping, zinc injection, and abrasive water jet conditioning. This paper reviews the application of the available techniques and provides an insight into a few potential techniques for future application.

CAUSES OF ALLOY 600 PWSCC

Stress corrosion cracking of metals and alloys is caused by the synergistic effects of environment, material condition, and stress. In a PWR primary water environment, intergranular stress corrosion cracking of wrought Alloy 600 material and its weld metals (Alloy 182 and Alloy 82) is commonly referred to as PWSCC. The occurrence of stress corrosion cracking of Alloy 600 in high-purity water has been extensively studied since the first reported observation of cracking in laboratory tests by Coriou, et al.[1] in 1959. The mechanism of this cracking phenomenon is not completely understood, and prediction of crack initiation time has proven to be extremely difficult, if not impossible, due to the uncertainty of numerous contributory variables including heat treatment, cold work, and residual stress.

The primary water environment and existing coolant chemistry controls (particularly in the range of dissolved hydrogen used, 25-50 cc H_2/kg H_2O) seem to be sufficient to promote PWSCC. Resin or sulfur intrusions, by themselves, will not produce PWSCC in Alloy 600 material. However, sulfate will promote intergranular attack and intergranular stress corrosion cracking.

In addition, PWSCC is a thermally-activated mechanism that can be correlated with a Svante Arrhenius relationship (exponential) and is very temperature dependent. This is evidenced by the fact that the vast majority of PWSCC of steam generator roll expansion transitions has occurred on the hot leg side of the tube sheet. The 50-70°F (27-38°C) temperature differences between hot and cold legs are enough to significantly influence the time to initiation and subsequent crack growth rate. However, failures of Alloy 600 material have also occurred in reactor vessel upper head nozzle material at a temperature of approximately 554°F (290°C)[2] and in at least one other occasion on a component item at a significantly lower water temperature of 423°F (217°C).[3]

The susceptibility of Alloy 600 material depends on several factors including the chemical composition, heat treatment during manufacture of the material, heat treatment during fabrication of the component item, and operating parameters of the component item.[4] The carbon and chromium contents of the material appear to be the most important chemical composition variables. These, in turn, affect the carbide precipitation. Microstructural conditions such as grain size and location and degree of carbide precipitation also are important variables that determine the susceptibility of a material to PWSCC. And finally, fabrication parameters and heat treatment of the material determine the overall yield strength and degree of cold work associated with the material. Apparently, a material that has been low temperature mill-annealed, with poorly decorated grain boundaries, and has relatively high yield strength (due to some degree of remaining cold work) is the most susceptible to PWSCC.

Tensile stresses, resulting from both residual and operating stresses, can be significant for some Alloy 600 component items. A stress magnitude close to the material yield strength is generally necessary for PWSCC to occur. Operating stresses are produced from mechanical and thermal loading, while residual stresses are generated as a result of fabrication, installation, and welding processes. Residual stresses are more difficult to quantify than operating stresses and, in many instances, are of a higher magnitude than operating stresses.

In summary, PWSCC requires three key factors present simultaneously: an aggressive environment, susceptible material, and significant, prolonged tensile stress. Eliminating any one of these three factors will mitigate cracking, although in practice it is prudent to attack all of these factors at once, when feasible.

MITIGATION TECHNIQUES

A number of techniques have been evaluated and are available to delay or eliminate the occurrence of PWSCC in PWRs. In general, these techniques fall into three main categories:

1. Mechanical surface enhancement (MSE)
2. Environmental barriers or coatings
3. Chemical or electrochemical corrosion potential (ECP) control

MSE techniques represent processes that reduce surface tensile residual stresses or induce compressive surface stresses on a component item or weld. Examples of MSE techniques include shot peening and electropolishing. Environmental barrier or coating techniques represent processes that protect the material surface from an aggressive environment. Coating examples include nickel plating and weld deposit overlays. Chemical or ECP control techniques represent changes to the environment that alter the corrosion process or produce corrosion potentials outside the critical range for PWSCC. Examples of chemical or ECP control include zinc additions to the primary water and modified primary water chemistry (e.g., dissolved hydrogen levels, lithium concentrations, and boron concentrations).

Previous Mitigation Techniques

When PWSCC began to occur in steam generator tubing and plugs, a number of mitigation techniques were developed. Included, were a variety of sleeving techniques using Alloy 690 material (e.g., mechanical and welded sleeves), which provided a structural barrier at the flaw(s) and also a way to keep the tube location in-service. Following the identification of PWSCC in pressurizer instrumentation nozzles, a few additional mitigation techniques were evaluated. Nearly all of the previous mitigation techniques were designed to reduce residual tensile stresses and hence fell into the MSE technique category. These include shot peening, flapper wheel grinding, electrical-discharge machining, electropolishing, and stress relief heat treatment.

Shot peening works very well on a surface that has not previously been in-service.[5,6,7] Steam generators shot peened prior to commercial operation have seen little or no PWSCC to date.[8,9] Stress relieving prior to service has also shown excellent behavior to date. Only one B&W-design pressurizer instrumentation nozzle, which was stress relieved before commercial operation, has leaked as a result of PWSCC.[10] However, stress relief heat treatment obviously does not make Alloy 600 material immune to PWSCC as evidenced by the recent observations of primary side cracking in once-through steam generator tubing.[11] In-situ stress relief heat treatment of tight U-bends of steam generator tubing[12] has had some success in extending the useful life of steam generators, but it also has not made it resistant to PWSCC.

Component item temperature reduction, which would fall into the chemical or ECP control category techniques, has also been used. Many utilities have reduced the hot leg temperature of operating PWRs to increase the PWSCC initiation time for steam generator tubing and plugs. As with in-situ stress relief, an extension of the useful life of the steam generator has been obtained, but nearly all utilities planning for license renewal will replace their steam generators. The other techniques listed above also have had limited success in mitigation of PWSCC.

Present Mitigation Techniques

Once PWSCC became fairly widespread throughout the primary water system, a number of other mitigation techniques were evaluated and qualified for use. Present day PWSCC mitigation techniques span all three categories.

MSE category techniques used today include abrasive water jet (AWJ) conditioning and the mechanical stress improvement process (MSIP). Both of these techniques will reduce the component items overall tensile stress and induce compressive residual stress on the exposed surface.

The AWJ conditioning process[13,14,15] utilizes high-pressure water that, after passing through small diameter orifices, draws the abrasive into the high velocity fluid stream in the mixing chamber and delivers the abrasive particles through a focusing tube onto the component item surface (Figure 1). The process can be used to remove flaws detected by non-destructive examination (NDE) and is also qualified for use in surface remediation to remove flaws that are too small for detection. This process has been successfully used in the repair and remediation of CRDM nozzles at a number of PWRs. An alternative process, known as water jet peening, does not employ an abrasive, but instead depends on cavitation impact generated by the water jet.[16]

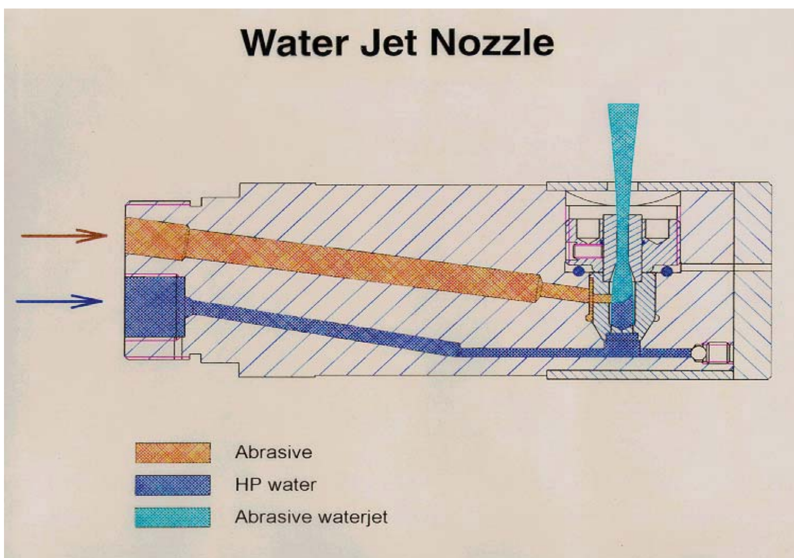

Figure 1. Typical Configuration of an Abrasive Waterjet Conditioning Nozzle.

The MSIP technique[17,18,19] works by using a simple hydraulically operated clamp, which slightly contracts a pipe on one side of a weldment. The permanent contraction under the tool generates a concave contour at the weld location and results in a slight corresponding reduction in pipe circumference (Figure 2). Once the tool has been removed, the weldment remains in axial compression through about half of the wall thickness and is protected by a layer of compressive hoop stress, which extends more than halfway through the wall thickness. MSIP has been accepted by the U.S. Nuclear Regulatory Commission (NRC)[20] as a stress improvement process for the mitigation of SCC in BWRs and has been successfully applied to over 1300 welds at BWRs worldwide. Most recently, it has been applied to hot leg nozzle-to-pipe welds at V.C. Summer.[21]

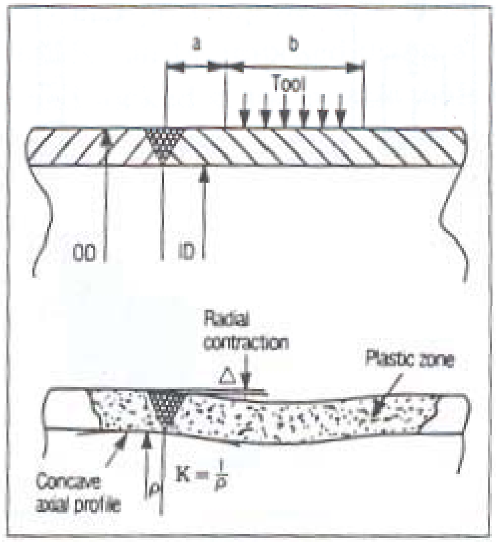

FIGURE 2 TOOL LOCATION AND DEFORMATION CONTOUR AFTER APPLYING MSIP

Environmental barrier or coating techniques available for use today include nickel plating, electroless nickel plating, weld deposit overlay, and laser weld deposit/cladding. Each of these techniques provides a barrier, which eliminates the environment that may promote PWSCC, and stops existing cracks from propagating.

Nickel plating[22,23] consists of plating the inside surface of the Alloy 600 component item with pure nickel using an end effector tool (Figure 3). The process steps include cleaning, pre-plating with a "strike" solution, depositing the nickel plating with a circulating bath and a soluble anode, and final cleaning. The nickel plating technique has been used most extensively in steam generator tubing[24] and has also been qualified and used in pressurizer heater nozzles.[25,26,27]

The electroless nickel plating process deposits a nickel-phosphorous plating onto the inside surface of the component item. It has been used in numerous applications outside of the nuclear industry, especially at oil, gas, and chemical process facilities[28] and could be adapted for PWR applications, although its current brittleness is not favored. The weld deposit overlay and laser weld deposit or cladding processes also provide corrosion resistant deposits on the inside surface of the component item. Weld deposit overlays of Alloy 690 weld material recently have been qualified and applied to CRDM nozzle repairs[29] and they have also been applied to bimetallic welds between the reactor vessel and the primary piping in Sweden. The laser weld deposit or cladding processes have been evaluated and tested in laboratory specimens, but have yet to be applied to in-service component items.[30,31,32,33,34]

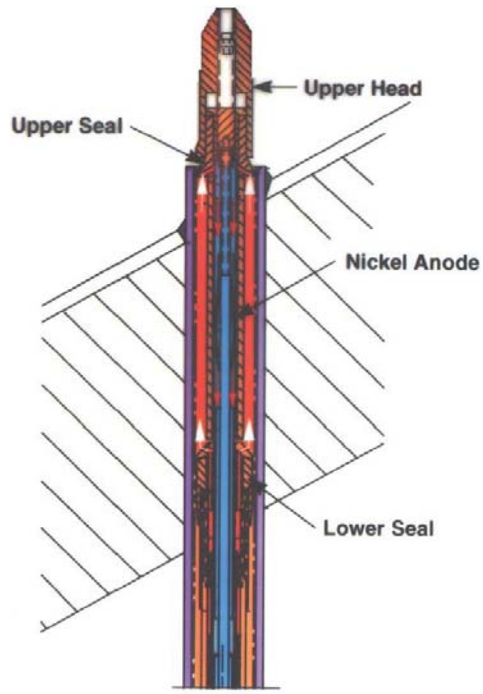

Figure 3. Typical End Effector Tool for Nickel Plating Pressurizer Heater Nozzles.

The final present day mitigation technique is the addition of zinc to the primary water system,[35,36,37,38,39,40] which is considered a chemical or ECP controlling process, which improves the protective oxide integrity. General Electric first instituted the injection of zinc into the BWR reactor coolant system in the mid 1980's to reduce plant dose rates with a process known as "GE-ZIP." In the late 1980's Westinghouse began investigation of the potential use of zinc injection for the same purpose. Since that time, they also investigated the ability of zinc to reduce both the PWSCC initiation time and crack propagation rate for Alloy 600 component items. Zinc addition has been employed at five PWR units in the U.S. and four additional units worldwide.[29] However, although PWSCC plugging trends for steam generator tubing appear to be decreasing, direct evidence of PWSCC mitigation in the field may require more cycles of zinc addition.

Future Mitigation Techniques

A number of promising mitigation techniques have been utilized in other industries or in BWRs to mitigate SCC of stainless steel piping and welds. Potential MSE techniques include laser shock peening, low plasticity burnishing, heat sink welding, and induction heating stress improvement (IHSI).

Laser shock peening is a process in which a laser beam is pulsed upon a metallic surface, producing a planar shockwave that travels through the work piece and plastically deforms a layer of material. Figure 4 shows the basic process. Laser shock peening is similar to shot peening, but the compressive layer can be deeper with less surface cold work.[41] An example of the residual stress distributions for both a laser peened and unpeened welded plate made from Alloy 22 (UNS N06022), a nickel-based material, is given in Figure 5.[42] Laser shock peening has recently been used on BWR core shroud weld HAZs at the Hamaoka Unit 1 reactor in Japan.[43] Figure 6 shows a comparison of the compressive stress as a function of the depth from the surface for several of the MSE techniques identified in this paper.

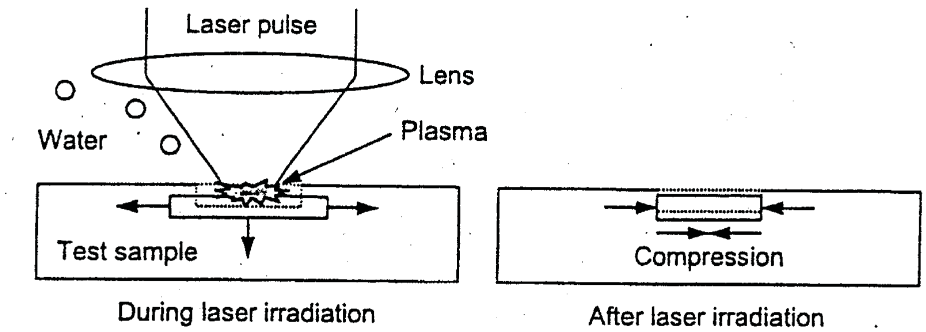

Figure 4. Typical Laser Peening Process.

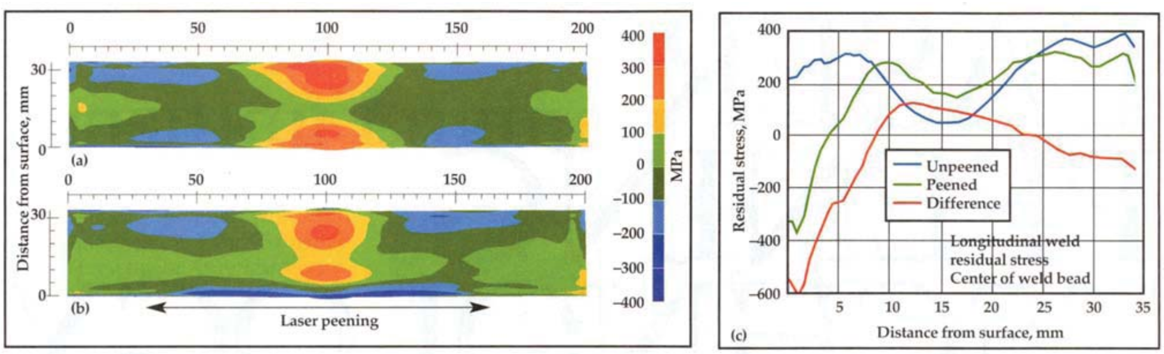

Figure 5. Residual Stress in a 33-mm Thick Alloy 22 Welded Plate: (a) Map of Residual Stress in Unpeened Weld; (b) Map of Residual Stress in Laser Peened Weld; (c) Line Plot of Residual Stress Versus Depth at the Center of the Weld Bead,

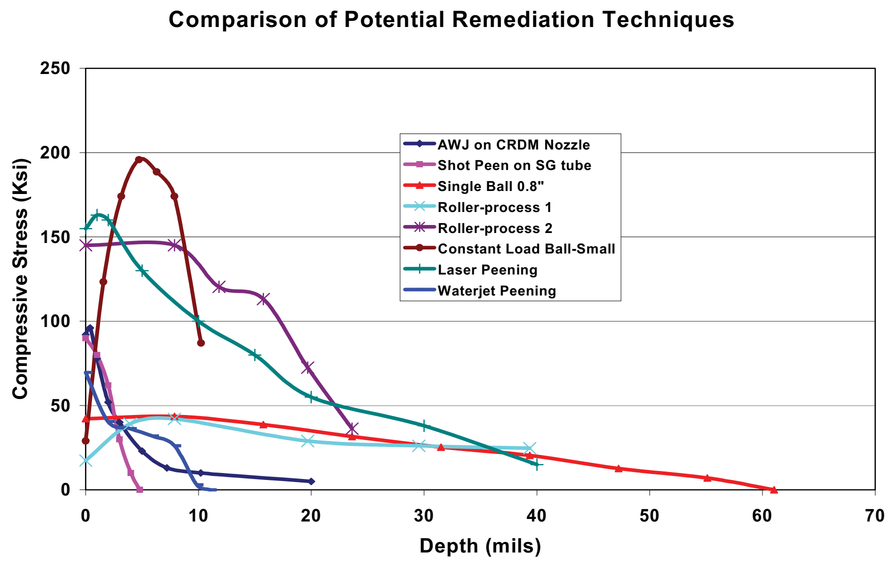

Figure 6. Typical Compressive Stress Distribution
Comparison of Several Remediation Techniques.

Roller burnishing has been used for years by the automotive industry, which leaves residual compressive stresses on the surface of a component item. Low plasticity burnishing (LPB) is a method of controlled burnishing and will provide a deep stable surface compression for improved fatigue and stress corrosion performance. A smooth, free-rolling spherical ball is pressed against and rolled along the surface of the area to be burnished (Figure 7). LPB produces minimal cold work and hence greater resistance to thermal relaxation at high temperatures.[44]

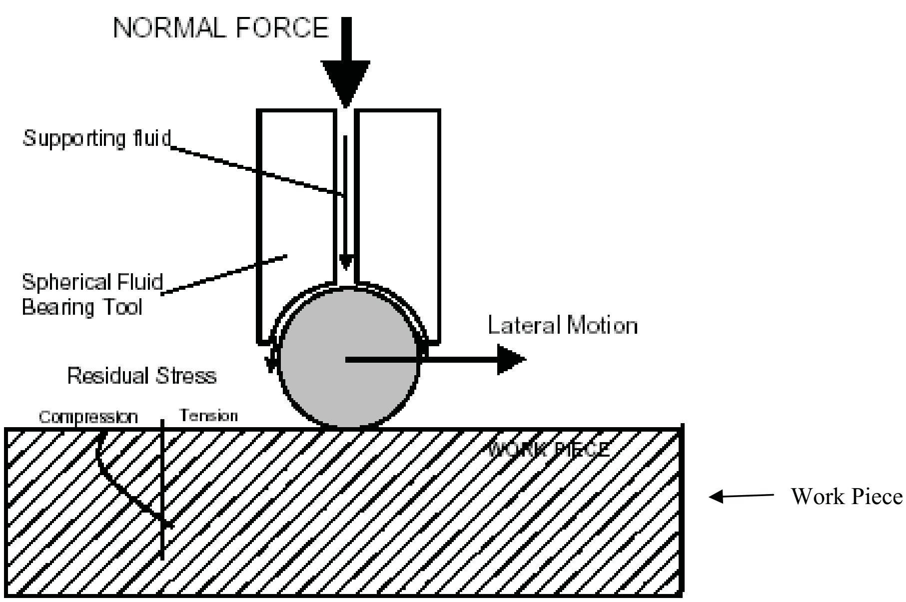

Figure 7. Low Plasticity Burnishing Schematic.

Heat sink welding and IHSI were both developed for BWR applications of weld SCC failures.[45] The basic idea is the same in both cases: an additional bending moment is applied to the area in which residual welding stresses prevail in such a manner that the tensile stresses in the weld area on the inside surface are either lowered or converted to compressive stresses. Heat sink welding uses a TIG arc as the heat source to heat the outer surface while depositing filler metal. During the process, the inside of the pipe is flushed with water, thus cooling it and establishing a thermal differential. The IHSI process utilizes an induction coil to heat the outside of the pipe.

Potential environmental barrier or coating techniques include thermal spraying and ion implantation or ion beam enhanced coatings. Thermal spraying is a generic term used to define a group of processes that deposit finely divided metallic or non-metallic materials onto a prepared substrate to form a coating. The coating material may be in powder, rod, or wire form. The thermal spray gun uses a plasma arc, combustible gases, or an electric arc to generate the heat necessary to melt the coating material. Particles are transferred to the component item and built up to form a coating.[46,47] Thermal spray is not a new technology. It began soon after the introduction of the oxyacetylene torch, between the years of 1890 and 1910 and has grown exponentially, particularly as the demands in jet engines advanced.[48]

Ion implantation is a process by which virtually any element (e.g., chromium) can be introduced into the surface of a solid material to selected depths and concentrations by means of a beam of high-velocity ions striking a target mounted in a vacuum chamber.[49,50] Ion beam enhanced deposition (IBED) processing is used to deposit high performance coatings on precision engineered components and manufacturing tooling.[51] IBED coatings exhibit superior adhesion and optimum physical properties when compared to coatings deposited by alternative processes because the coating atoms first penetrate into the substrate to form a case layer in the surface, and then are grown out from this case layer as a thick coating. Driven in kinetically instead of thermally, coatings are "ballistically bonded" to the substrate thus forming a metallurgical bond that is much stronger than a mechanical or Van der Waals bond. One drawback to the use of these techniques is that they currently must be performed in a vacuum.

Possible chemical or ECP control techniques include modified primary water chemistry control, anodic protection, and other potential chemical additions (besides zinc). Changes in primary water chemistry through the use of reduced boric acid (enriched boric acid), lower lithium, and optimized hydrogen control may also reduce PWSCC concerns.

Titanium compounds (TiO_2, TiB),[52] cerium boride (CeB),[53] etc. may possibly be employed to change the ECP on the materials surface. Finally, anodic protection is based on the formation of a protective film on metals by externally applied anodic currents.[54] If carefully controlled anodic currents are applied to a nickel alloy, it may maintain a passive surface layer and the rate of metal dissolution and PWSCC may decrease.

SUMMARY AND CONCLUSIONS

Over the last thirty years, PWSCC has been observed in a number of Alloy 600 component items. PWSCC requires three key factors present simultaneously: an aggressive environment, susceptible material, and significant, prolonged tensile stress. Eliminating any one of these three factors will mitigate cracking, although in practice it is prudent to attack all of these factors at once, when feasible. Component item replacement with a more PWSCC resistant material is an obvious alternative, but it is not considered a mitigative technique.

A number of practical techniques have been employed in the past and a few techniques are currently being applied to delay or eliminate the occurrence of PWSCC in PWRs. In general, these techniques fall into three main mitigation categories: (1) mechanical surface enhancement, (2) environmental barriers or coatings, and (3) chemical or electrochemical corrosion potential control. Most mitigative techniques are very geometry dependent and many are not effective if a crack has already initiated. The ability to perform NDE during future outages must also be considered. Techniques in use today include shot peening, nickel plating, abrasive water jet conditioning, mechanical stress improvement, and zinc addition to the primary water system.

A number of promising techniques, which have been used in other industries, are potentially available for application in PWRs. These include laser shock peening, roller burnishing, heat sink welding, thermal spraying, and ion implantation. However, most will require further developmental efforts before they are available for commercial use.

Utilities need to evaluate a variety of techniques since one mitigative technique will most likely not be applicable to all locations. Additional cost-benefit analyses need to be performed prior to any decisions regarding the use of a particular mitigation technique. It is also recommended that utilities consider developing site-specific Alloy 600 aging management programs both for current license attainment and for license renewals.[55]

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A Reactor Vessel Upper Head Temperature Reduction Program to Prevent and Mitigate Alloy 600 Cracking

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Alloy 600 primary water stress corrosion cracking (PWSCC) is very sensitive to temperature. Lowering temperatures can be an effective way to increase the time for crack initiation, reduce the rate of crack propagation and to extend the useful life of a reactor vessel head. The reactor vessel Upper Head Temperature Reduction (UHTR) program is a method of lowering the reactor vessel head temperature by reducing the bulk fluid temperature in the upper head region. This modification has been performed at a number of plants within the Westinghouse fleet and additional plants have scheduled this modification for future plant outages. Extension of a significant number of years to the original reactor vessel head life and a reduction in the number of required head inspections may be realized with the implementation of a UHTR program. A UHTR program should be considered as part of each plant's overall Alloy 600 management program.

Introduction

Recent plant operating experience with nickel based alloys (i.e., Alloy 600) indicates this material is, in general, susceptible to primary water stress corrosion cracking (PWSCC) when exposed to operating temperatures in excess of 500°F (260°C). Alloy 600 was typically used for pressure boundary components because of its thermal compatibility with carbon steel, superior resistance to chloride attack, and higher strength than the austenitic stainless steels.

Recently, cracking and leakage have been observed in reactor vessel head penetrations, which have been attributed to PWSCC. The vessel head penetrations form an integral part of the reactor coolant system pressure boundary. Penetration cracks and subsequent leakage provide a significant challenge to plant availability and personnel radiation exposure limits. The magnitude of the upper head region fluid temperature has therefore become increasingly important since it is one of the significant boundary conditions for the determination of temperatures in the reactor vessel head penetrations.

In this paper, a method to lower the temperature of the reactor vessel upper head region, which is called upper head temperature reduction (UHTR), is presented. The effects of temperature on cracking and the impact on the plant's effective degradation years (EDY) are discussed to show the benefits in lowering head temperature. Actual UHTR implementation designs and Westinghouse UHTR field modification experience are also presented.

Westinghouse Plant Designs

There are two basic categories with regard to reactor vessel upper head region bulk fluid temperatures in pressurized water reactors of Westinghouse design.

1) In "T_{cold}" plants, the bulk fluid temperature in all regions of the upper head is at the cold leg temperature. This is accomplished by providing a path for a significant amount of flow to go from the downcomer region directly into the upper head.

2) In "T_{hot}" plants, the bulk fluid temperature in the upper head is between T_{hot} and T_{cold}. The temperature magnitude between T_{hot} and T_{cold} is dependent upon the hydraulic characteristics of the spray nozzles and the upper internals package components. Since present Appendix K nuclear safety analyses can only assume either T_{hot} or T_{cold} in the upper head region, plants that have upper head temperatures that are not at T_{cold} are assumed to be at "T_{hot}".

In general, the older plants in the Westinghouse fleet are "T_{hot}" plants. Thus, conceptually, the bulk fluid temperature in the upper head region can vary between the core inlet temperature and the maximum core exit temperature. The extent that the bulk fluid temperature approaches the maximum core exit temperature is a function of the hydraulic paths between the downcomer and the upper head region, the plant specific/cycle specific core power distributions and the hydraulic performance of the fuel assemblies. For these T_{hot} plants, the temperature of the bulk fluid is dependent on the relative contributions of the hotter core exit flow, which proceeds up through the various control rod guide tubes, and the head cooling flow, which is diverted directly from the reactor inlet flow through spray nozzles, as shown in Figure 1. Due to the presence of a radial pressure distribution in the upper plenum region, the pressure in the guide tubes varies from location to location. These guide tube pressure variations create the potential for flow to either enter or exit the upper head region via the guide tubes. One of the design functions of the spray nozzles is to allow a small amount of colder fluid from the downcomer to flow into the upper head region, thus maintaining a smooth temperature transition between the closure head and the reactor vessel. In general, for these older design plants, the head cooling flow from the downcomer (at the cold leg temperature) was kept relatively low to minimize core bypass flow.

When the loss of coolant accident (LOCA) benefits of a T_{cold} upper head region became apparent and newer core designs supported lower core flow rates, a series of design changes were implemented in the upper support plate in the upper internals package and in the core barrel flange of the lower internals package to allow T_{cold} to be achieved in the upper head region. Approximately thirty plants in the Westinghouse fleet reflect this more recent design philosophy. The changes that were made to these newer plants to achieve T_{cold} include increasing the spray nozzle flow areas to provide the higher cooling flow rates, and modifying the internals flanges to accommodate the spray nozzle modifications.

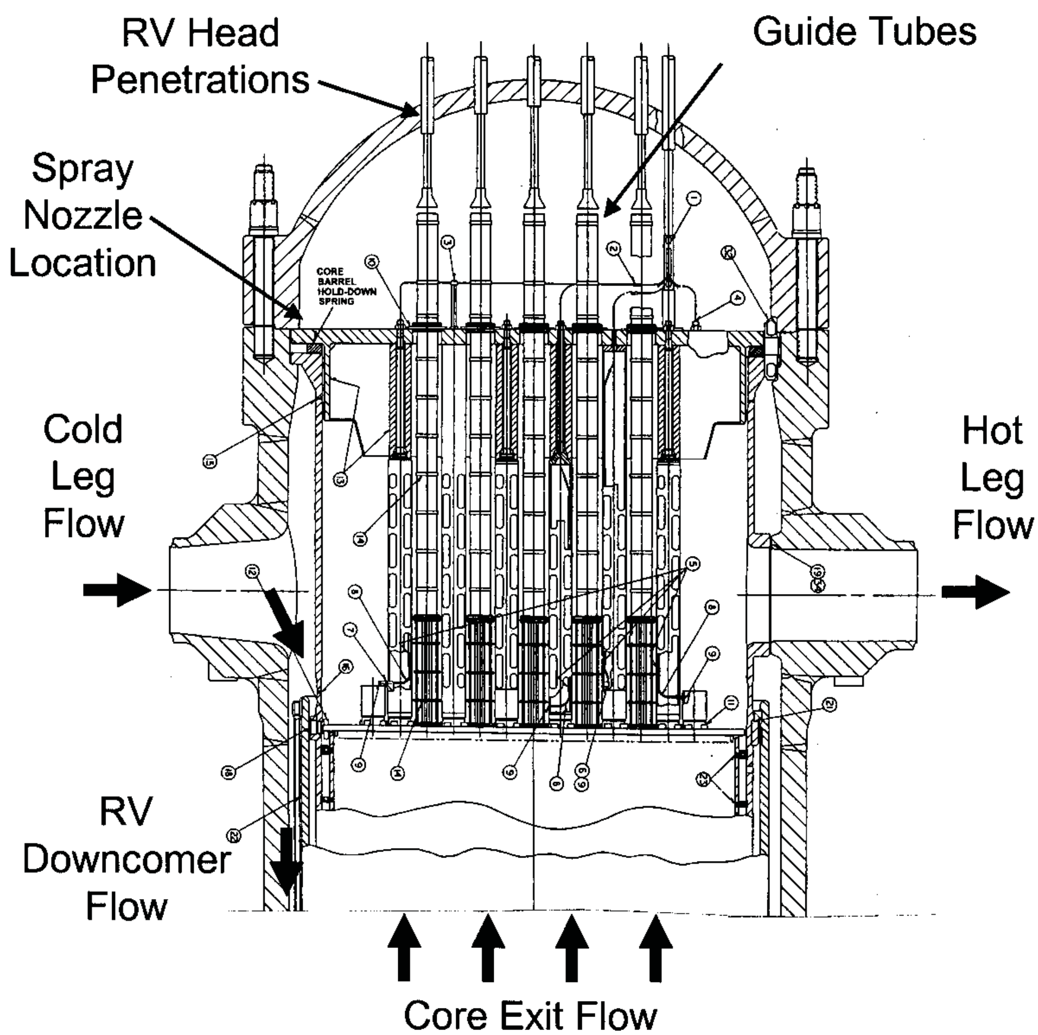

Figure 1: Upper Internals Components and Reactor Coolant Flow Paths

Upper Head Temperature Reduction

An upper head temperature reduction (UHTR) program is a field modification performed on the reactor vessel internals structures at the "T_{hot}" plants which will:

1) Increase the bypass flow from the reactor vessel downcomer region (which is at cold leg temperature) into the upper head region

2) Lower the bulk fluid average temperature underneath the closure head to the cold leg temperature, resulting in lower reactor vessel head penetration temperatures.

A schematic of the reactor internals components where modifications are made for UHTR is shown in Figure 2. The flow that enters the reactor vessel head via the paths made between the core barrel flange and the upper support plate flange has a temperature corresponding to cold leg temperature (i.e., T_{cold}) since these paths connect the reactor vessel downcomer region with the upper head region.

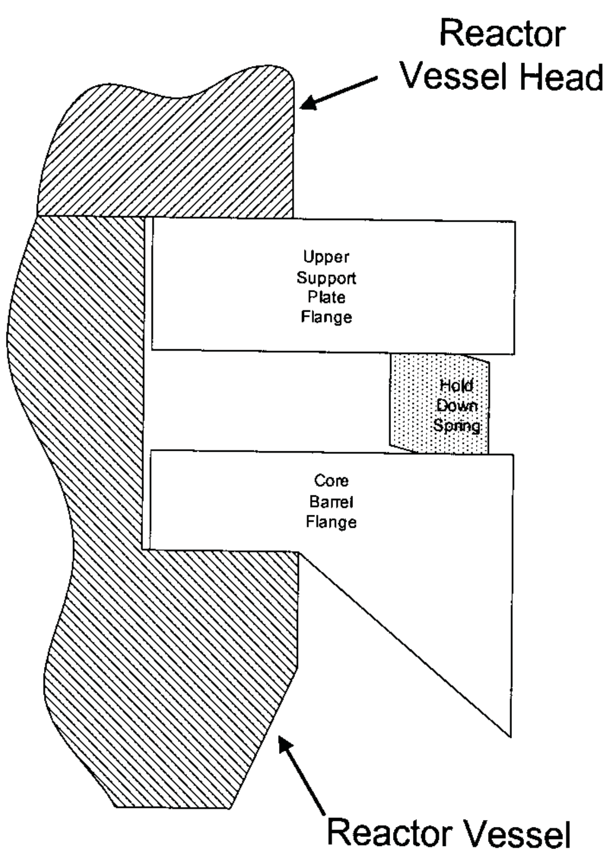

Figure 2: Detail of Components where UHTR Modification is Implemented

Effect of Temperature on Crack Initiation and Crack Growth Rate

PWSCC of Alloy 600 is sensitive to temperature. A simplified model for assessing the potential for cracking based only on operating time and bulk head temperature was developed in Reference [1]. This model does not address other factors such as stress, microstructure, surface cold work, and head fabrication practices, and therefore has limitations, such as large uncertainties and minimum predictive capability. Therefore, use of such data beyond ranking, and determining susceptibility sub-populations for inspection purposes may not be appropriate. However, Figure 3 (extracted from Reference [2]) does support the argument that time at temperature is a strong indicator of cracking potential.

The Reference [1] susceptibility model calculates the operating time normalized to a reference temperature of 600°F (1059.67°R, or 315.6°C). The standard Arrhenius activation energy dependence on temperature is applied to each time period with a distinct head temperature:

$$EDY_{600°F} = \sum_{j=1}^{n} \left\{ \Delta EFPY_j \exp\left[-\frac{Q_i}{R} \left(\frac{1}{T_{head,j}} - \frac{1}{T_{ref}} \right) \right] \right\}$$

<u>where:</u>

EDY_{600F} = total effective degradation years normalized to reference temperature of 600°F

$\Delta EFPY_j$ = effective full power years accumulated during time period j

n = number of time periods with distinct 100% power head temperatures

Q_i = activation energy for crack initiation (50 kcal/mole)

R = universal gas constant (1.103×10^{-3} kcal/mole-°R)

$T_{head,j}$ = 100% power head temperature during time period *j* (°R = °F + 459.67)

T_{ref} = arbitrary reference temperature (1059.67°R)

The exponential term in the above equation is plotted in Figure 4, and shows the relative effect of temperature on degradation, as measured by crack initiation normalized to 600°F. At 550°F, crack initiation is predicted to take about 8.3 times longer than at 600°F. That is, the model predicts that one year of operation at 550°F is equivalent to 0.12 years of operation at 600°F with regard to PWSCC initiation of Alloy 600.

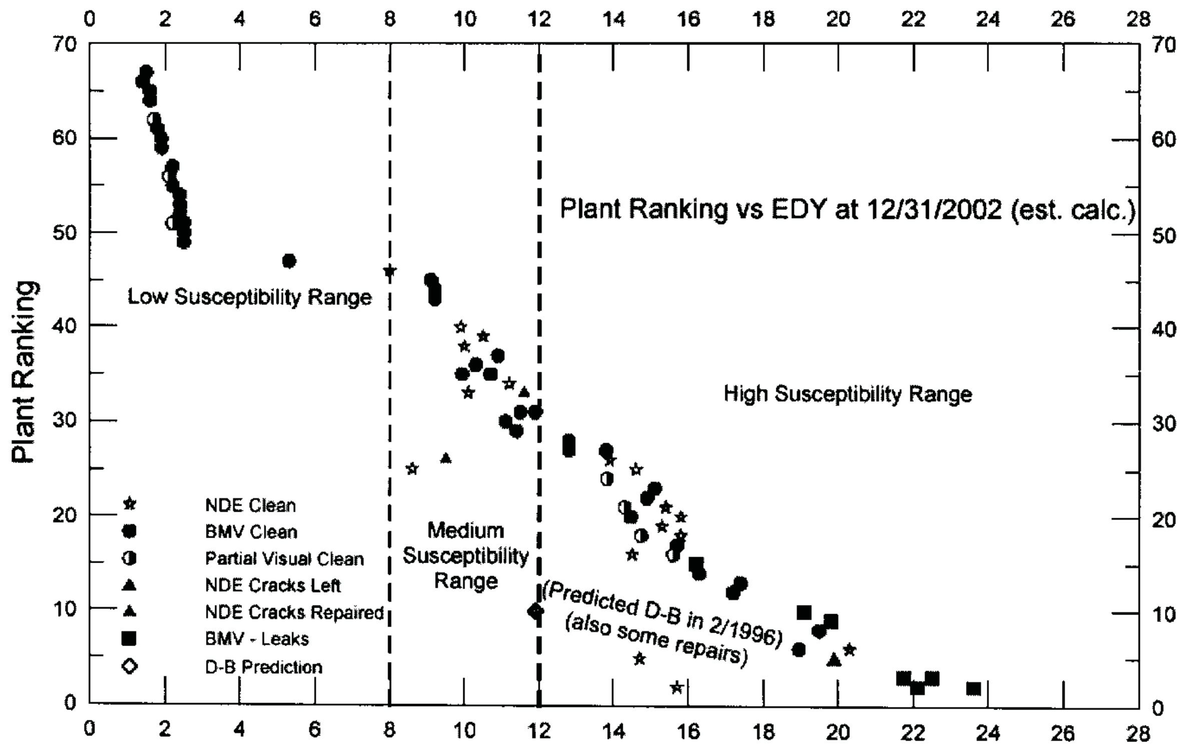

Figure 1. Ranking of domestic plants according to the EDY formula, showing results of inspections, evidence of leakage, and repairs. Many plants are shown with multiple symbols, indicating a "clean" inspection at inspection opportunity, followed by a different finding at a subsequent inspection (e.g., Oconee 2: clean NDE @ EDY=15.7, leaks and circ. flaws @ 22.1)

Figure 3: Domestic Plant Ranking vs. EDY (Reference [2])

Crack initiation is a more important factor than crack growth for assessing units since the time to crack initiation is longer than the time for crack growth. However, a reduction in the upper head temperature significantly reduces the crack growth rate. Reference [3] provides the following formulation to evaluate the growth of SCC flaws in thick-wall components fabricated from Alloy 600 materials:

$$\dot{a} = \exp\left[-\frac{Q_g}{R}\left(\frac{1}{T}-\frac{1}{T_{ref}}\right)\right]\alpha(K-K_{th})^{\beta}$$

<u>where:</u>

$\dot{a}$	=	crack growth rate at temperature T (inches/year)
Q_g	=	thermal activation energy for crack growth (31 kcal/mole)
R	=	universal gas constant (1.103×10^{-3} kcal/mole-°R)
T	=	absolute operating temperature at location of crack (°R)
T_{ref}	=	absolute reference temperature used to normalize data
α	=	crack growth amplitude (3.69×10^{-3} at 617°F)
K	=	crack tip stress intensity factor (ksi $\sqrt{\text{inch}}$)
K_{th}	=	crack tip stress intensity factor threshold (8.19 ksi $\sqrt{\text{inch}}$)
β	=	exponent (1.16)

The exponential term in the crack growth equation above is plotted in Figure 4, and shows the relative effect of temperature on degradation, as measured by crack growth normalized to 600°F. At 550°F, crack growth is predicted to take about 3.7 times longer than at 600°F. That is, the model predicts that one year of operation at 550°F is equivalent to 0.27 years of operation at 600°F with regard to PWSCC growth of Alloy 600.

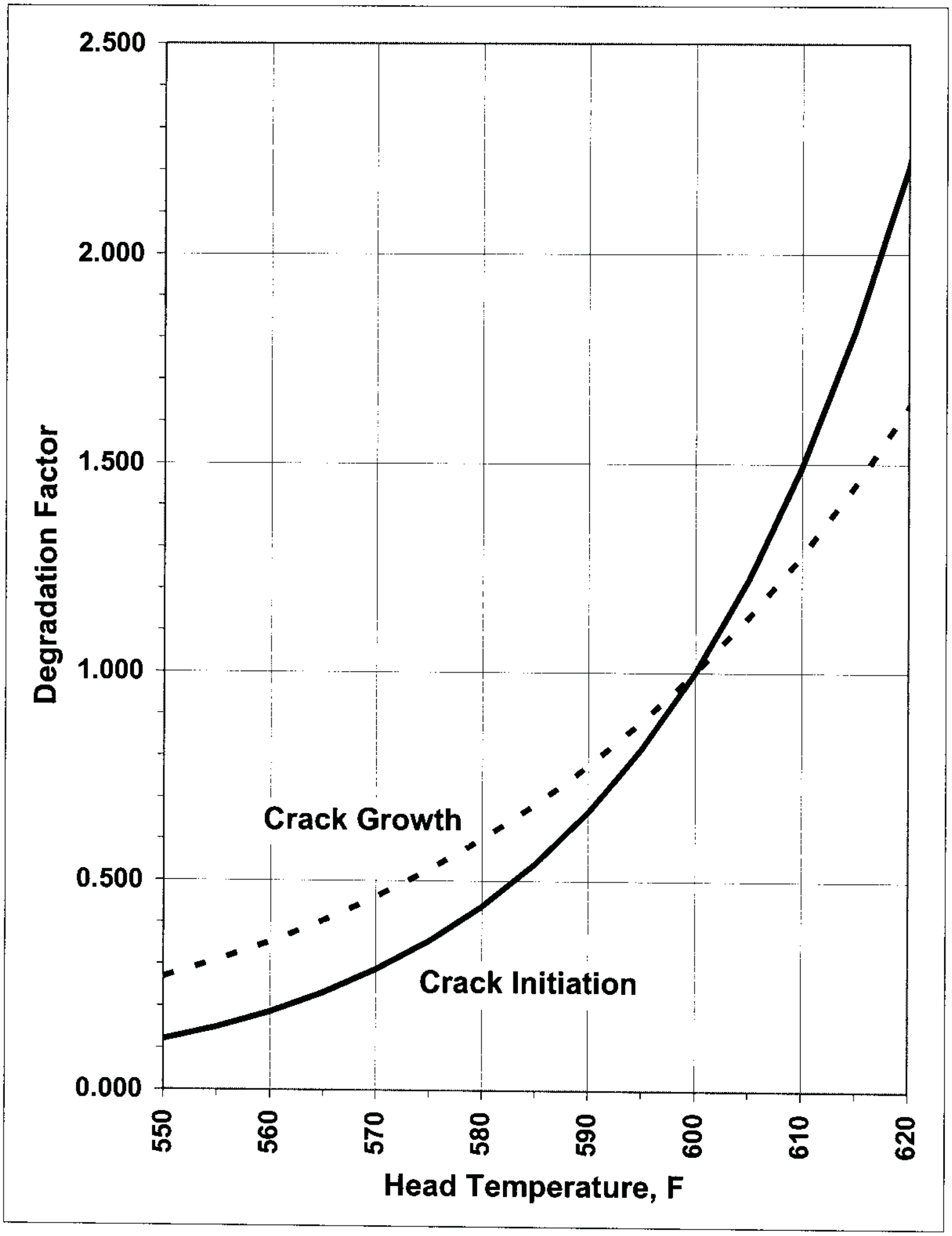

Figure 4: Degradation Factor as a Function of Temperature

Example for Plant Implementing UHTR

The impact of implementing a UHTR program is illustrated in Figure 5. In this example, it is assumed that a plant begins commercial operation with a bulk head temperature of 600°F. Without a UHTR program, the EDYs increase at a rate of 1.00 times the operating time, as the degradation factor is 1.00 at 600°F. The plant would therefore reach the NRC criteria of 8 and 12 EDYs after 8 and 12 EFPYs, respectively.

To extend the useful life of the reactor vessel head, the utility decides to implement a UHTR program at 10 EFPYs, reducing the head temperature to 550°F. This significantly reduces the degradation rate, as the degradation factor is 0.12 at 550°F. The plant would then reach the NRC criterion of 12 EDYs in about 17 EFPYs from the time of the UHTR (or 27 total EFPYs), rather than 2 EFPYs (or 12 total EFPYs) if UHTR is not implemented. This is about a 15 EFPY improvement over not implementing the UHTR.

Similar benefits may be realized for plants that are below 8 EDYs. Obviously, the earlier that a UHTR program is implemented, and the larger the temperature reduction, the greater the benefit with regard to cracking susceptibility and NRC Order inspection requirements. Should a utility decide to replace the reactor vessel head concurrent with UHTR, the plant would not reach the NRC Bulletin 2002-02 criteria of 8 EDY until 66 EFPYs, assuming a temperature reduction to 550°F.

Benefits of UHTR

The benefits of performing a UHTR program are as follows:

1) A significant reduction in the effective degradation years (EDY) accumulation rate for the plant will be realized (see Figure 5). This can lead to a reduced number of required inspections as specified in United States Nuclear Regulatory Commission Order EA-03-009 (Reference [4]) for either a new closure head or an existing closure head.

2) Reduces the risk associated with head penetration cracking by significantly increasing the time for crack initiation, and significantly reducing the rate of crack propagation.

3) Significantly increases the peak clad temperature (PCT) margin for large break LOCA, which can be utilized to support an increase in power output.

4) Slightly increases the reactor coolant system flow rate, which increases the steam generator tube plugging margin.

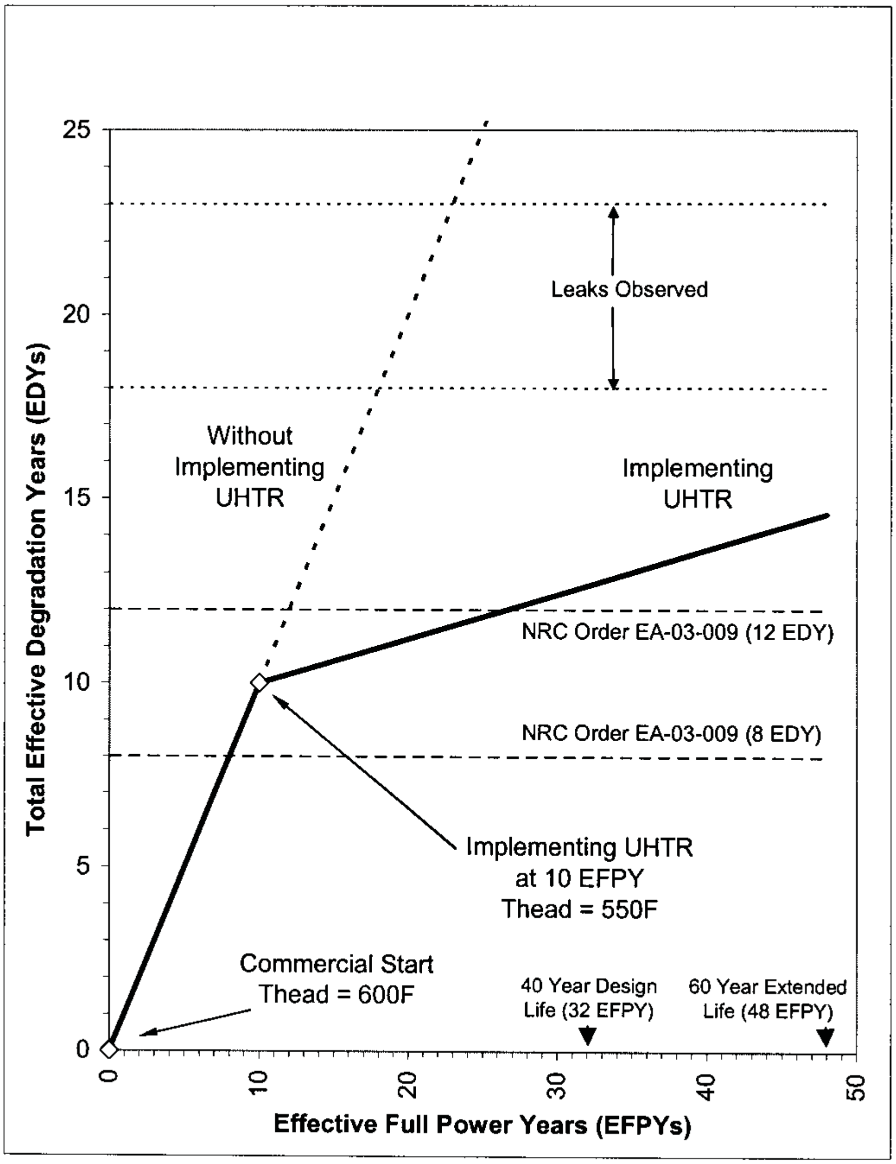

Figure 5: Example of UHTR Effectiveness for a Unit Approaching the 12 EDY Limit (with Respect to NRC Order EA-03-009)

Westinghouse Fleet Status/ UHTR Field Modification Experience

Approximately one-half of the plants within the Westinghouse fleet were designed and have operated with upper head region fluid temperatures at the reactor vessel cold leg temperature. For the remaining units within the fleet, Westinghouse has developed and implemented a process to lower the fluid temperatures in the upper head region. This process requires no machining in the core barrel flange and does not require the removal of the lower internals package from the reactor vessel. Moreover, the modifications can be performed off-critical path, resulting in no impact on the refueling outage times.

In 1992, Westinghouse performed a UHTR program at a European plant. The main benefit, at that time, was to maintain the large break LOCA peak clad temperature (PCT) margin. In 2000 and 2002, UHTR programs at two domestic units were performed to alleviate and reduce the impact of increased core flow due to new replacement steam generators, as well as to lower the susceptibility to PWSCC in the closure head penetrations. A UHTR program was recently performed in 2003 at a second European plant to lower the risk associated with head penetration cracking and to provide margin to support an increase in power output. A third European unit is scheduled to perform a UHTR modification in early 2004.

For the four units at which a UHTR program has been performed, three units required the removal of specified plugs from existing spray nozzles on the core barrel flange that protrude through holes on the upper support plate flange, as shown in Figure 6. The total time, which was off the refueling critical path, to implement this modification at each plant was 10 hours, with a total radiation exposure of 10 mR. For the fourth plant, a selected number of irradiation specimen access plugs, located in the core barrel flange, were removed and large holes in the upper support plate (directly above the exposed specimen access holes) were machined utilizing an Electric Discharge Machine (EDM), as shown in Figure 7. The total time to implement this field modification, again off the refueling critical path, was 4 days. The total radiation exposure was 637 mR.

The UHTR modifications at these 4 plants have been successfully completed and the units have returned to full power operation. No adverse impacts on plant availability due to the modifications have been reported.

Conclusions

Recent cracking incidents in the reactor vessel upper head penetrations have focused regulatory and industry attention on the issue of Alloy 600 cracking. Because Alloy 600 cracking is sensitive to temperature, lowering temperatures at Alloy 600 locations can be an effective approach to mitigate primary water stress corrosion cracking and extend the useful life of a component.

The reactor vessel Upper Head Temperature Reduction (UHTR) program is a method of lowering the reactor vessel head temperature by reducing the bulk fluid temperature in the upper head region. This proven method has been performed at a number of plants within the Westinghouse fleet and additional plants have scheduled this modification for future plant outages. Extension of a significant number of years to the original reactor vessel head life may be realized by implementing a UHTR program. Additionally, the onset of additional inspection requirements may be delayed for either an existing head or a replacement head. A UHTR program should be considered as part of each plant's overall Alloy 600 management program.

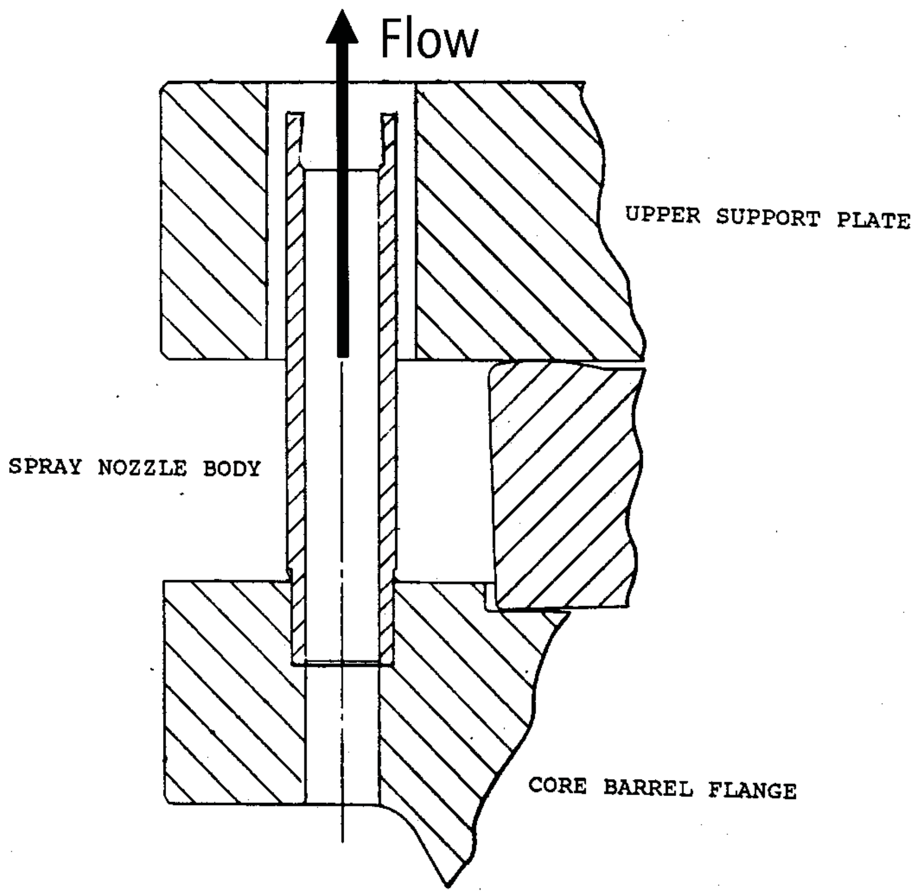

Figure 6: Westinghouse UHTR Implementation Design (3 Units)

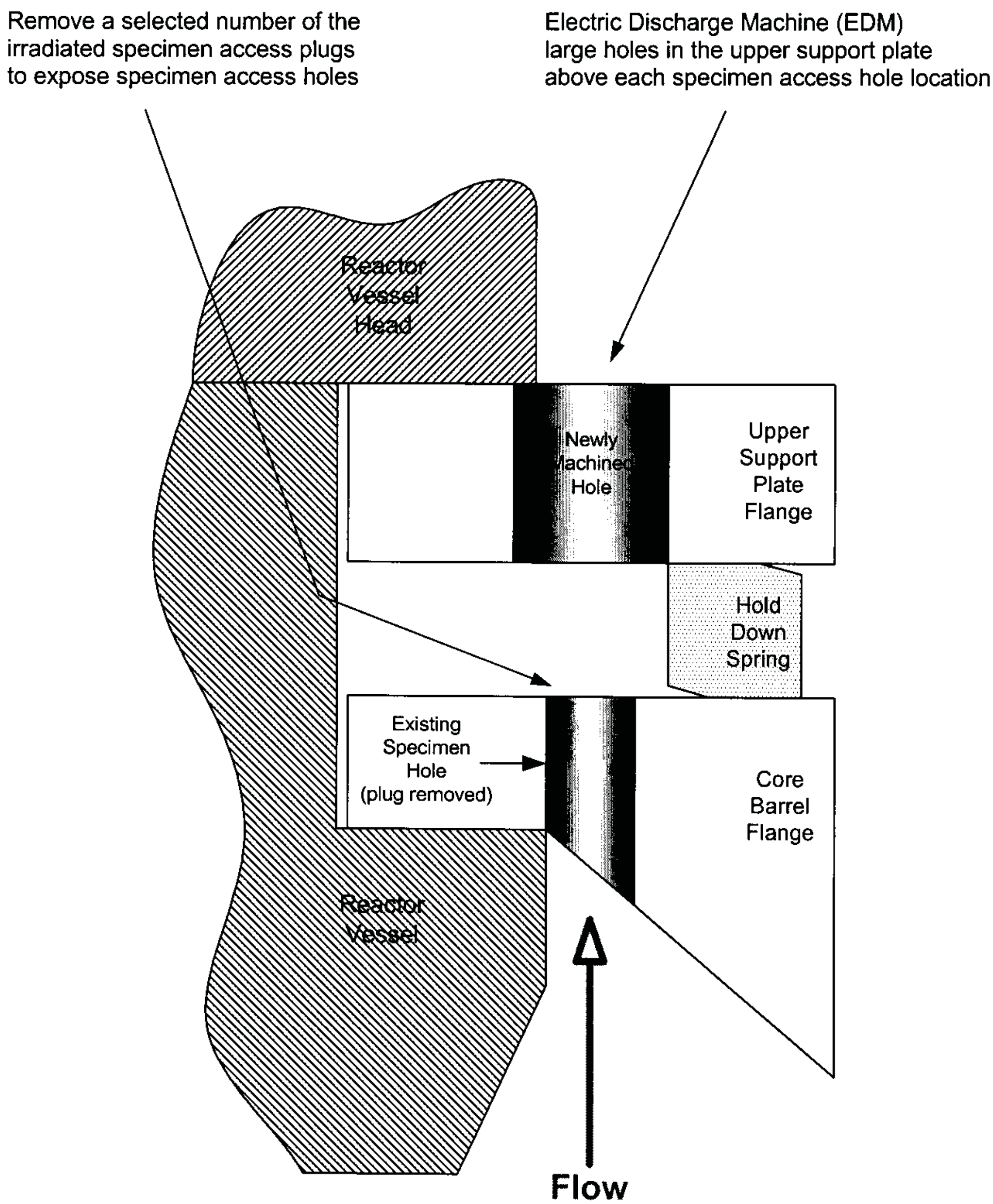

Figure 7: Westinghouse UHTR Implementation Design (1 Unit)

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Status Report on the Effect of Zinc Additions on Mitigation of Primary Water Stress Corrosion Cracking in Alloy 600

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Summary

Addition of zinc to the reactor coolant system of PWRs has been used in the US since mid-1994, primarily as a means to achieve radiation dose rate reductions, but also as a possible approach to mitigate the occurrence or severity of primary water stress corrosion cracking (PWSCC) of Alloy 600. The basis for this activity was the experience with zinc additions in BWRs, complemented by laboratory experiments in simulated PWR environments. The effectiveness of zinc in contributing to reduced radiation fields in PWRs is now well established, with positive results observed in both domestic and German PWRs. A more elusive issue has been the effectiveness of zinc in mitigating PWSCC. The objective of the evaluation reported in the present paper was to perform a review of the literature, and to consider the expanding operating plant experience, in order to develop a more coherent understanding of the role of zinc with regard to PWSCC of nickel-base alloys.

The mechanism by which zinc affects the corrosion of austenitic nickel-base alloys is by incorporation of zinc into the spinel oxide corrosion films. Reduction of general corrosion leads to reduced metal release rates and an associated dose rate reduction in operating steam generators by modifying the corrosion source term. Nearly without exception, the results of laboratory testing also indicate a benefit of zinc injection in mitigating the <u>initiation</u> of PWSCC in Alloy 600. Early laboratory data suggest this benefit may vary with the concentration of zinc in the RCS.

Data for a beneficial effect on crack <u>propagation</u> are more mixed. The laboratory results vary from a substantial reduction in crack growth rates to no effect at all. Interpretation of these differences based on the nature of the crack tip oxides is currently inconclusive. Field data over portions of several fuel cycles with zinc injection in two units of a US plant appear to show reductions in the initiation and propagation of primary-side cracks in SG tubing. Attributing these effects solely to the addition of zinc is complicated, however, by the fact that changes have occurred in eddy current inspection equipment and scope, and in plugging criteria, over the same period.

The Mitigation Working Group of the EPRI Materials Reliability Program is currently planning additional testing work to clarify the extent to which zinc addition can be qualified as a practical measure to retard the growth of pre-existing PWSCC cracks in thick-section Alloy 600/182/82 materials.

1. Introduction

Zinc additions have been widely used in boiling water reactors both to control radiation fields and also to help inhibit the occurrence of intergranular stress corrosion cracking (mainly of stainless steels). Laboratory data and German experience indicated similar potential benefits with regard to radiation fields for pressurized water reactors (PWRs). As a result, EPRI and Southern Nuclear cosponsored the initial field demonstration of zinc addition at Farley Unit 2 in 1994-95. The results of this demonstration confirmed the beneficial effect of zinc in mitigating radiation fields, which is now well established with positive results observed in several domestic plants (see examples in Fig. 1 and 2).

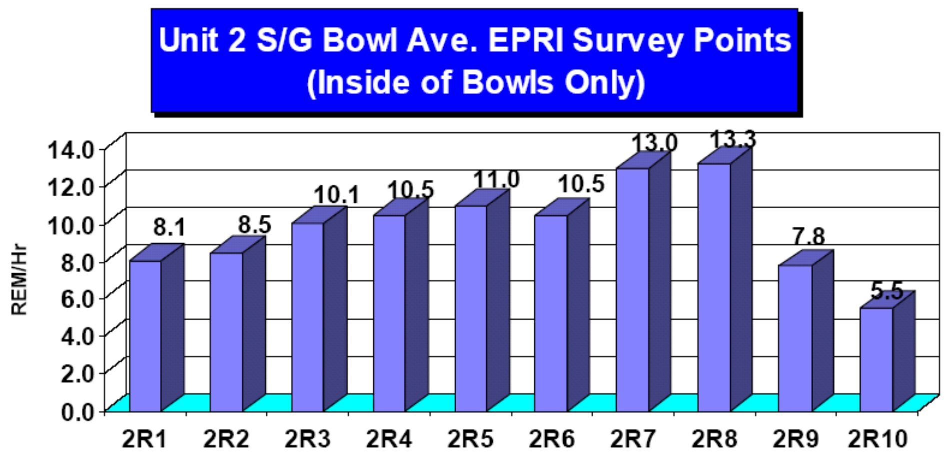

Figure 1 **Post zinc dose rate trends for Diablo Canyon Unit 2**
(first additions during cycle 9; range 15 to 30 ppb)

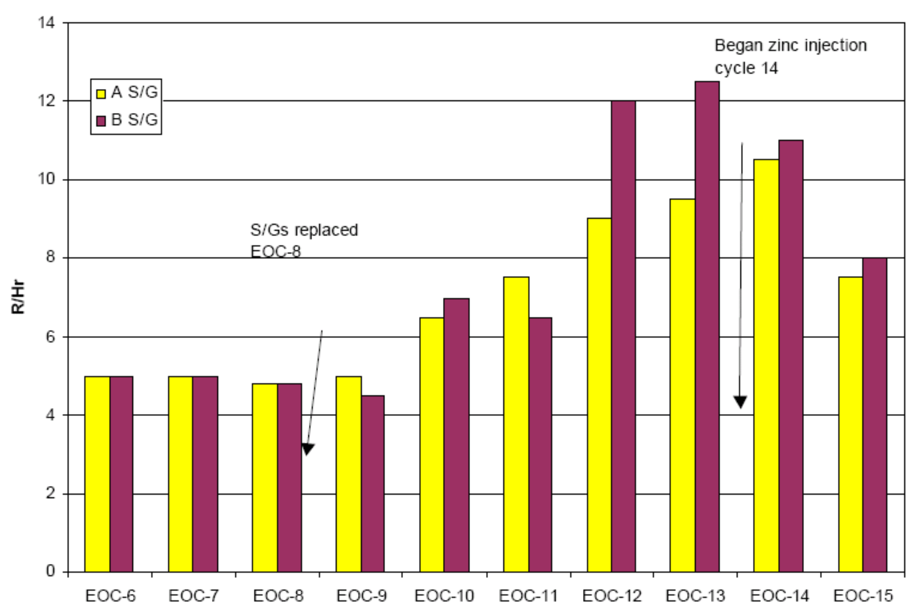

Figure 2 **Post zinc dose rate trends for Palisades channel head**
(levels up to 10 ppb)

A more elusive issue has been the effectiveness of zinc in mitigating primary water stress corrosion cracking (PWSCC). Although most laboratory testing has indicated a beneficial effect on mitigating crack initiation in Alloy 600, data for crack propagation are mixed. With additional information on laboratory and field experience of zinc addition becoming available recently, an update on the effects of zinc on PWSCC has been carried out, including a thorough review of both the literature and operating plant experience, in order to develop a more coherent understanding of the role of zinc with regard to PWSCC in nickel-based alloys.

2. Interaction of zinc with corrosion films on nickel-base alloys in PWR primary water

The long-term exposure of austenitic nickel-base alloys in PWR primary water leads to the development of a duplex oxide film. The inner layer that forms on materials such as Alloy 600 and Alloy 690 is a chromite spinel of the form $(Fe,Ni,Co,)Cr_2O_4$, in which the Cr and Fe levels are enriched relative to their concentrations in the base metal. This layer is formed by corrosion of the alloy at the metal-oxide interface. After several thousand hours of exposure at about 315 ^{o}C (600°F), this layer is typically on the order of 200 to 400 nm thick.

The outer film consists of an irregular layer of particles that is best described as solution-grown or hydrothermally deposited. These particles range in size from about 0.5 to 2 μm. This solution-grown film forms at the oxide-solution interface and, for Alloys 600 and 690, is typically on the order of the particle size. The composition is controlled by corrosion product release from the underlying alloy, as well as the species in (super)saturation in the fluid boundary layer adjacent to the corroding metal, and typically consists of ferrites such as $NiFe_2O_4$ or $CoFe_2O_4$, which have an inverse spinel structure.

Because of its higher site preference energy in the normal spinel oxides, the addition of zinc to the primary coolant leads to exchange reactions whereby zinc displaces Co, Ni, and other cations into the coolant stream where they can be removed by the cleanup systems. This principle has been used for over fifteen years by operators of boiling water reactors (BWRs) to reduce primary system radiation buildup

In the case of stainless steel, the situation is much the same as for the nickel-base alloys except that the relative thickness and composition of the chromite and ferrite layers are slightly different. However, the basic reactions, and effect on corrosion rate, are the same (see Fig. 3) .

Examination by Auger electron spectroscopy of the surfaces of Alloy 600 and stainless steel that have been exposed to simulated primary water containing zinc clearly indicates efficient incorporation of zinc into the oxide corrosion films. At regions nearest the surface, the zinc concentrations range from 16% to greater than 32% and suggest that a large fraction of the tetrahedral sites are occupied by zinc. Both the concentration at the surface and the depth to which zinc has penetrated appear to correlate with the total exposure times, rather than with the concentration of zinc in the coolant, although the specific tests referred to here involved exposures on the order of 2000 hours with zinc concentrations up to greater than 100 ppb.

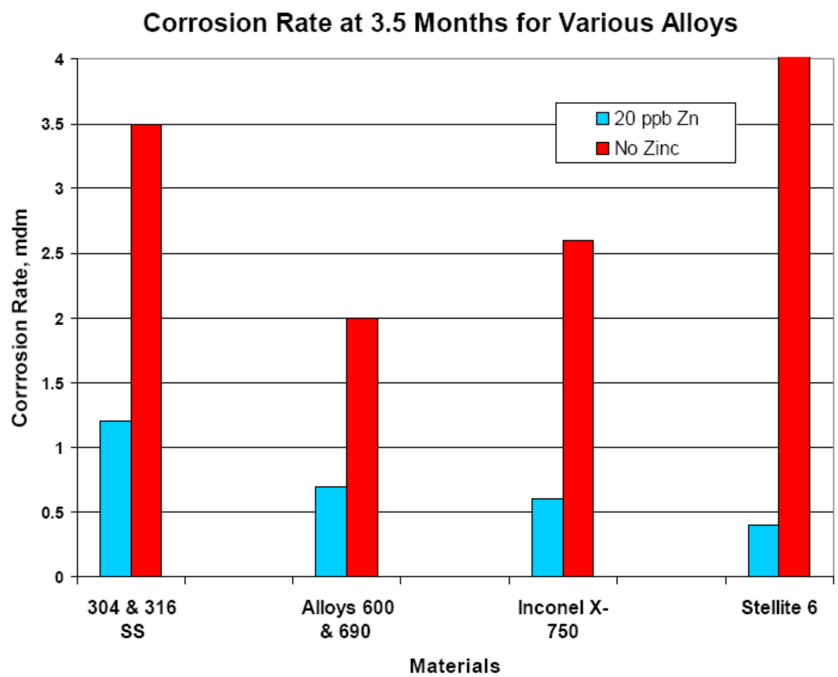

Figure 3 **Effect of zinc on corrosion rates of various alloys in laboratory tests (after Esposito et al. [1])**

Auger examination of a tube pulled from Farley Unit 2 at the end of Cycle 10, following approximately nine months of zinc addition to the primary coolant at a concentration of approximately 40 ppb, gave results very similar to those seen in laboratory research. Zinc concentrations at the surface were as high as 25%, although the depth of penetration was less than that found in the laboratory specimens.

The field experience is clearly consistent with expectations based on laboratory tests and rational interpretation of the phenomena. Zinc leads to significant releases of iron, nickel and cobalt from the tetrahedral sites. Almost immediately upon the introduction of zinc, the RCS concentrations of, in particular, nickel and its daughter isotope ^{58}Co increase significantly and generally remain high throughout the period of zinc injection. In time, ultimate exhaustion of the source term for these releases should lead to stable concentrations, reflecting their removal by the cleanup system demineralizers.

3. Review of experience with respect to zinc and PWSCC

3.1 Laboratory data

It is well known that the main factors affecting PWSCC of nickel-base alloys are materials, material condition, stress and temperature [2]. Water chemistry within the normal operating range has a relatively small – though not insignificant – effect: e.g., dissolved hydrogen levels have been shown to have an important influence, particularly on crack propagation rates [3]. The available literature reporting studies of the effects of zinc on PWSCC was recently reviewed for EPRI by Gold [4].

The data with regard to crack initiation show that zinc additions can have a marked, beneficial effect. An example of this is shown in Fig. 4 for both mill-annealed and thermally treated Alloy 600 tubing. Overall, reported factors of improvement (in terms of the time to initiate cracking) have ranged from about 2 (for addition of 20 ppb Zn) to greater than 10 (for 120 ppb Zn).

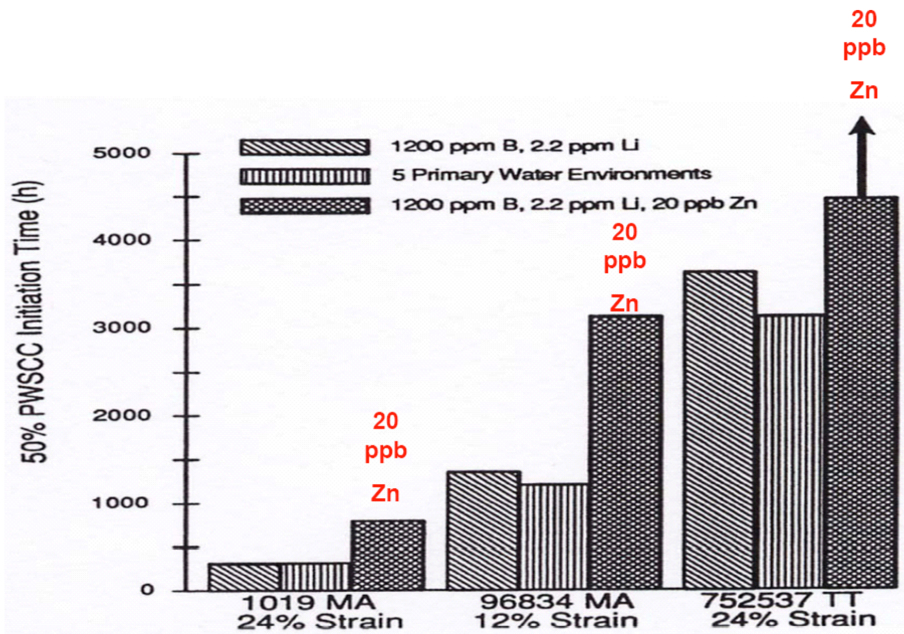

Figure 4 **Example for the effect of zinc on time to initiate PWSCC in laboratory tests (after Esposito et al. [1])**

Some laboratory data has also indicated a positive effect of zinc additions (even at the 10 ppb level) on crack growth through PWSCC. Figure 5 shows results obtained in 1998 by Kawamura et al. [5] using DCB specimens from plate material.

The same authors also reported a beneficial effect of Zn additions (factor of ~ 2x) on the crack depth measured in slow strain rate testing (SSRT). It is noteworthy that the measured crack growth rates (CGR) were relatively low in this work.

Other PWSCC data on CGR, as measured in fracture mechanics specimens in the laboratory, is less conclusive. In a test at 330 ^{o}C, Gold et al. [4] found that 20 ppb Zn lowered the CGR of a pre-cracked SG tube specimen under dead-weight loading at high stress intensity by a factor of around 3x. Further tests, this time on thick-section Alloy 600 material, at lower stress intensities showed no crack growth at all when Zn was present at levels of 50 –250 ppb. However Airey et al. [6] found no apparent effect of 40 ppb Zn in extended tests at high K levels using both WOL and CT specimens. Later tests in this program gave the same result at lower stress intensities [7]. Tests by Morton et al. [8] also failed to show a beneficial effect of Zn, even when added at high levels (~ 100 ppb).

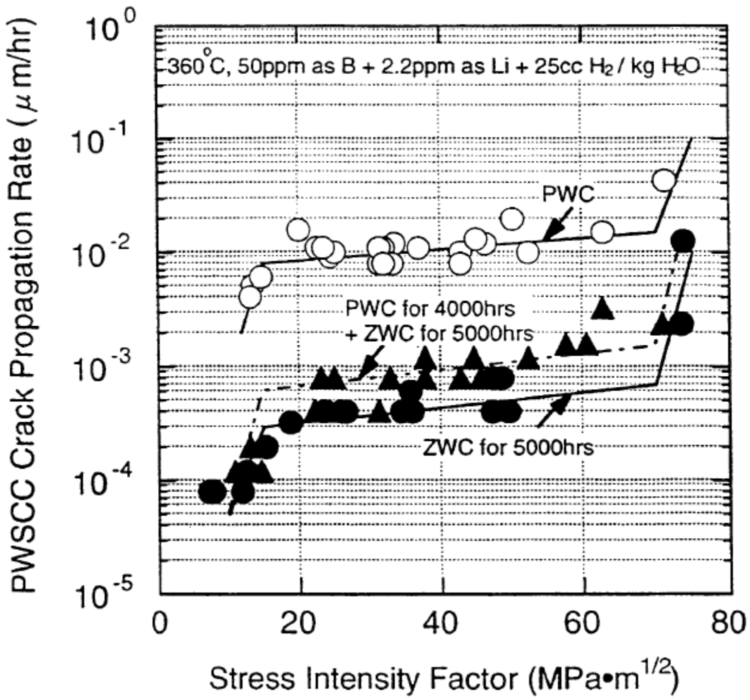

Figure 5 **Example for the beneficial effect of zinc on PWSCC CGR in laboratory tests using DCB specimens: PWC = normal primary water chemistry; ZWC = PWC + 10 ppb Zn (after Kawamura et al. [5])**

A review of the major results from the crack propagation tests in simulated PWR environments reported in the literature shows that they include the following variables:

- Material form: tubing, plate, archived CRDM nozzle
- Test specimen: dead-wt. loaded tube, wedge-loaded 1/2T-CT, bolt-loaded WOL 1/2T-CT, active load 1T-CT, wedge-loaded double cantilevered beam
- Applied stress intensity: 20 to 60 MPa√m
- Monitoring or inspection: active by DC potential drop, or post-test fractography
- Test temperature: 316 to 360°C
- Variations in water chemistry and level/form of zinc addition (ZnO, acetate, borate)

Perhaps not surprisingly, the conclusions of the various research teams range from partial or total inhibition of cracking to no effect whatsoever. In no case have negative effects been observed.

The research reporting a distinct benefit may be influenced by the apparent effectiveness of zinc in mitigating crack initiation (particularly in SSRT) and the observed incorporation of zinc into the chromite spinel corrosion films, as reported, e.g., by Kawamura et al. [5]. In contrast, two of the research groups reporting no influence of zinc interpret their results in terms of the lack of spinel oxide stability at the crack tips. Thus, if only cubic nickel oxide is present, there is no reason to judge that zinc, which acts by modifying the structure and perhaps morphology of the spinel oxide, would play a role in retarding crack propagation. Recent, high-resolution ATEM studies indicate that the crack-tip oxides in Ni-based alloys are complex and may vary according to quite subtle differences in the environmental conditions under which they form.

It may be that the discussion offered by Andresen et al. [9] pointing out the role of the electrochemical potential (ECP) offers some hope in unraveling this conundrum. This latter argument points out that, in the higher potential range (i.e. ≥ 150 mV), the potential gradient reduces the zinc concentration at the crack tip by driving anions into the crack tip and cations (such as Zn^{2+}) out of the crack. The ECP on the primary side of PWRs is much more negative (essentially at the hydrogen equilibrium value), so that this effect would no longer be expected. However, Andresen also points out that adequate time is required for incorporation of zinc at the tip of a growing crack. It is notable from the literature data that, with perhaps one exception, it was for the lowest crack growth rates that an apparent benefit of zinc was reported.

The EPRI Materials Reliability Program (MRP) is starting additional CGR measurements, using advanced testing techniques, in the near future to examine these issues and hopefully resolve some of the apparently contradictory results from previous laboratory testing.

3.2 Field Experience

The only relevant experience with the use of zinc as an agent to mitigate the occurrence and consequences of PWSCC in operating SGs is that reported for the Diablo Canyon Units 1 and 2, and this suggests that zinc may, in fact, be having a beneficial effect (see Fig. 6 and 7). However, it will be necessary to continue monitoring the steam generator experience with PWSCC to confirm this benefit, since the current data are ambiguous because of a number of factors, such as changes in inspection practice and equipment, limited and intermittent periods of zinc injection, modified plugging criteria, etc.. Nevertheless, it is worthy of note, that the crack propagation rates found at the dented tube-tube support plate intersections may be quite low, thereby enhancing the probability that zinc could be effective.

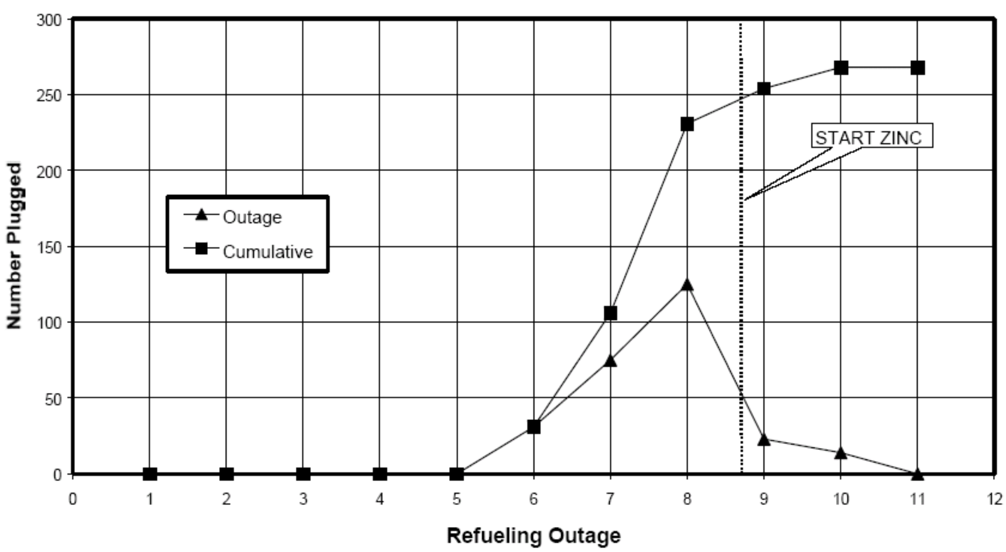

Figure 6 **Example of field data from Diablo Canyon Unit 1: plugging for PWSCC at tube-support-plate (TSP) locations**

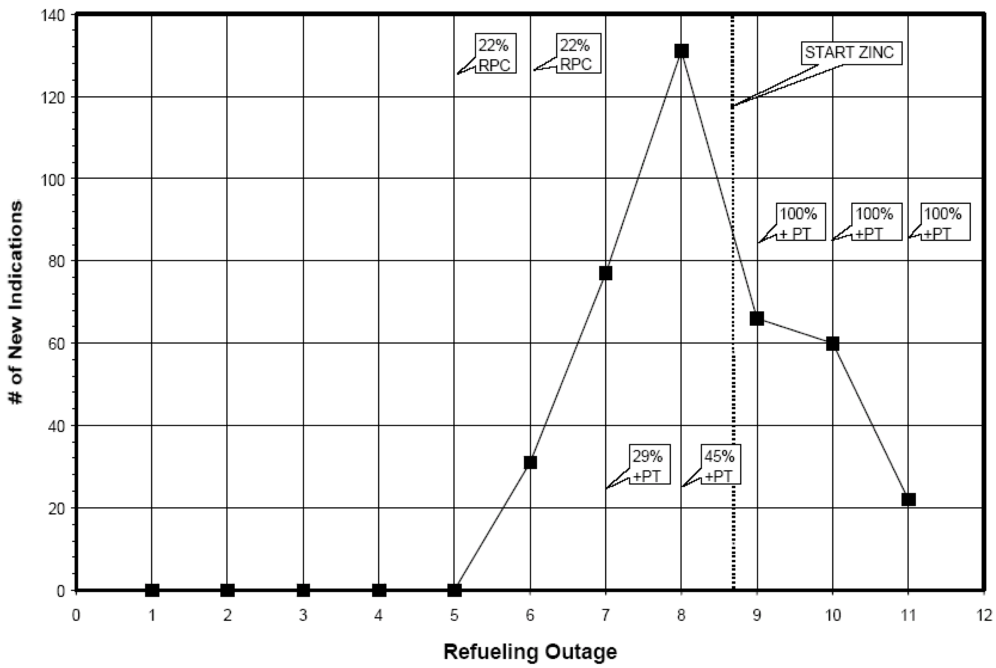

Figure 7 **Example of field data from Diablo Canyon Unit 1: new indications of PWSCC at tube-support-plate (TSP) locations**

4. Fuels issues when adding zinc

Consideration of possible effects on Zircaloy fuel cladding performance is always a dominant issue when changes are made to primary water chemistry. Thus an increase in oxidation observed at Farley Unit 2 after cycle 10 aroused concern as to a possible negative effect of zinc additions. However, the detailed root cause analysis showed that the increased oxidation was caused not by this, but by higher thermal duty and a measurement bias. Subsequent oxide measurements at this plant showed lower values on both once- and twice-burned fuel, confirming that there had been no effect of zinc.

As a further, positive indication, recent fuel cladding scrapes at Palisades have shown no evidence of zinc silicate formation, despite prolonged operation with Zn concentrations of ~ 9 ppb and silicate levels up to 2.5 ppm.

As zinc addition – for whatever reason – becomes more widespread, potential issues associated with fuel performance in high duty plants will require careful monitoring. To prepare for this, the latest EPRI Primary Water Chemistry Guidelines [3] recommend that plants should consider implementing additions of 5 – 10 ppb Zn, both to reduce radiation buildup and to prepare them for PWSCC mitigation benefits (probably requiring higher Zn levels), if these are confirmed to be effective.

5. Conclusions

Based on this review of the current laboratory and operating plant experience with zinc injection in PWR primary water, the following conclusions are offered.

- The mechanism by which zinc affects the corrosion of austenitic nickel-base alloys is by incorporation of zinc into the inner spinel oxide films formed by corrosion of the underlying metal.
- The surface concentration and penetration depth appear to correlate more with exposure time than with the coolant Zn level.
- Reduction of general corrosion leads to reduced metal release rates and an associated dose rate reduction in operating steam generators by modifying the corrosion source term.
- Essentially without exception, the results of laboratory testing indicate a benefit of zinc injection in mitigating the initiation of PWSCC in Alloy 600. Early laboratory data suggest that the extent of this benefit may vary with the concentration of zinc in the RCS.
- Laboratory data for a beneficial effect on crack propagation are mixed and currently inconclusive. The data vary from a substantial reduction in crack growth rates to no effect at all. In no case have negative effects been observed. Further testing is required.
- The only substantial operating plant data are those from Diablo Canyon Units 1 and 2, where zinc has been injected for portions of three fuel cycles in each plant. Significant reductions in the initiation and propagation of cracks at both tube-sheet and tube-support-plate locations have been observed in outages since zinc injection was adopted. However, attributing these effects solely to the addition of zinc is complicated by the fact that concurrent changes have occurred in eddy current inspection equipment and scope, and in the plugging criteria. More inspection data is needed.

6. Acknowledgements

Thanks are due to Bob Gold (Westinghouse) for carrying out the detailed review, documented in [4], upon which much of this paper is based. Appreciation is expressed to John Wilson (Exelon), Rick Eaker (Duke Power) and various members of the MRP Expert Panel on PWSCC for their assistance in designing a new test program on chemical mitigation (including zinc) to be funded by the MRP Alloy 600 ITG Mitigation WG.

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An Assessment of MSIP for the Mitigation of PWSCC in PWR Alloy 82/182 Thick Wall Piping Welds by Instrumented Full Scale Mockup Testing

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AEA Technology

1. INTRODUCTION

Nickel Chromium Alloy 600 base material and Alloy 82/182 welds in Pressurized Water Reactors (PWRs) have been the subject of Primary Water Stress Corrosion Cracking (PWSCC) over the past decade (Ref. 1). Recently, PWSCC cracking has been reported in the Reactor Vessel Outlet Nozzle and Pressurizer Surge Nozzle Safe End (Alloy 82/182) Welds (Refs. 2 and 3). Among the factors contributing to the cracking, residual tensile stresses associated with the welds are considered a significant contributor. Mechanical Stress Improvement Process (MSIP) was devised by AEA Technology to alleviate the IGSCC in BWR piping welds by introducing plastic compressive strains on the ID surface near the welds by mechanically squeezing (Ref. 4). The technology was successfully applied to BWR piping welds to eliminate IGSCC. The nozzle piping and welds in the PWRs are significantly thicker (approximately by 65%) compared to the BWR piping and welds. This paper describes the testing and the results of an instrumented full scale mockup test program conducted to demonstrate the generation of adequate inside diameter (ID) surface compressive residual stresses from the proposed MSIP treatment of the Tihange Unit 2 thick wall pressurizer surge nozzle Alloy 182/82 weld (Ref. 5). The purpose of the testing is to ensure that neither crack initiation, nor additional growth of existing crack occurs at the Alloy 182/82-nozzle weld under continued service conditions.

2. OBJECTIVES

The objectives of the instrumented mockup testing of the Tihange 2 pressurizer surge nozzle are:

- To validate the analysis and confirm applicability of MSIP for thick wall PWR piping.
- Verify that the desired MSIP load locations and range of radial contractions achieve permanent compressive residual stresses at the nozzle ID surface at the nozzle-to-safe-end weld.
- To verify post-MSIP OD surface profile satisfies inspection acceptability.

3. MECHANICAL STRESS IMPROVEMENT PROCESS (MSIP)

MSIP was devised by AEA Technology as a method to eliminate susceptibility of piping and weldments to stress corrosion cracking by alleviating weld-induced tensile stresses in the vicinity of circumferential welds (Ref. 4). In general terms, MSIP consists of squeezing a pipe plastically near the weld using a specifically designed set of rings that grip a short length of pipe. The squeezing is continued until the tensile residual stresses along the inner region of the weld are replaced by compressive strains. An analytical study of the residual stress patterns resulting from MSIP is available in Reference 6. A schematic representation of the pipe wall displacements from the MSIP loading is illustrated in Figure 1.

4. EQUIPMENT AND NOZZLE MOCKUP

The mockup was made of a material that has approximately the same yield strength as the Tihange 2 pressurizer surge nozzle safe-end. The pipe-to-safe-end weld and the safe-end-to-surge nozzle weld were reproduced as much as possible with similar weld preparation geometry. The weld was performed using standard welding techniques for the size and geometry of the weld preparation. The equipment and training mockup also represented approximately the same space envelope as the Tihange 2 pressurizer compartment boundaries and heater assemblies. Figure 2 illustrates the mockup test setup with MISP tool in place.

5. QUALIFICATION PROCEDURES

Detailed qualification procedures were developed for the MSIP tooling, calibration, testing, data recording, and documentation. Qualification procedures were also developed for strain gauge instrumented monitoring of the MSIP testing. These included strain gauge positioning and mounting procedures, strain gauge calibration and data recording, and documentation of the strains during the MSIP test. The qualification procedures are documented in Reference 7.

6. TEST METHODS

6.1 MSIP Load Tests

The MSIP loading (pipe squeezing) of the mockup nozzle weld region was accomplished by the application of hydraulic pressure through the MSIP tool clamp rings, mounted on the OD surface at predetermined positions (Fig. 2).

A planned sequence of three pipe squeezes were applied to obtain the desired data and accomplish the desired radial displacements and the ID surface strains.

- The maximum pressure generated by the MSIP tool was increased to obtain minimum specified contraction and generate acceptable residual stress redistribution in the nozzle-to-safe-end weld region upon unloading.

- Additional loading of the tool was applied to reach the maximum specified contraction and achieve the desired condition in the weld region following removal of the tool.

Post-MSIP OD profile was measured. A summary of loading and pipe diameters associated with the three MSIP treatments is provided in Table 1.

6.2 Strain Gauging and Strain Monitoring

Single electrical resistance strain gauges were applied at each of the eight measurement locations situated 45 degrees apart on the inside diameter surface near the weld. All strain gauge (center line) locations were 17 inches from the surface of the plate on the inside diameter of the tube. Bi-axial strain gauges were applied to the 0, 90, 180 and 270 degree positions with the grids aligned in the hoop and axial directions. Strain gauge rosettes were applied to the 45, 135, 225, and 315 degree positions with the number one grid aligned in the hoop direction. Figure 3 illustrates the mockup and the position of the eight strain gauges on the ID surface. The total strain resulting from squeezing the mockup on the outside diameter was recorded at various load increments. Figures 4(a) and 4(b) illustrate the appearance of the mounted strain gauges and the strain measuring equipment looking from the nozzle end.

7. RESULTS AND DISCUSSION

The strains developed during active loading and the permanent residual strains after the release of loading were recorded for the three consecutive load cycles. Strains were recorded along the hoop and the axial directions at all eight locations (45 degrees apart) on the ID while strains were recorded along 45 degrees direction at the four tri-axial gauges mounted 90 degrees apart. The results are illustrated graphically in Figures 5 through 7. The outside diameter changes of the safe-end with each of the squeeze is illustrated in Figure 5. Figures 6 and 7 illustrate polar plots of the axial strains developed in the weld. Strain results were fairly axisymmetric except for the two (2) axial gauges at the 135 degree and 270 degree positions, which indicated significantly higher magnitudes of compressive strain than at the other positions. Variations in material properties, non-uniform piping wall thickness, initial piping ovality, strain gauges being located at or near structural transitions, and non-axisymmetric as-welded residual stresses and strains are considered among the potential factors that may have contributed to the non-axisymmetric strain readings.

The biggest variant may well be the last item above. It is well known from studies performed, that residual stresses and strains that result from welding processes used for piping are often very highly non-axisymmetric in the weld and surrounding regions. MSIP tooling applies a sufficient displacement controlled radial contraction such that compressive plastic hoop and axial strains imposed in the weld region overcome the residual tensile strains due to welding, resulting in compressive hoop and axial residual stresses and strains. During redistribution of residual stresses and strains in the weld region due to the application of MSIP, non-axisymmetric material plastic flow would take place to overcome any initial non-axisymmetric state of residual stress and strain. In the weld region, final residual stresses would be fairly axisymmetric, but the strains applied by MSIP and the final residual strains could be somewhat non-axisymmetric.

Figure 8 illustrates the expected through-wall stress distribution at three locations in the weld from the finite element analysis (Ref. 6).

Observing strain gauge readings for Tihange Mockup Squeeze No. 1 for 15,000 psi hydraulic pressure, at which point only a small degree of plasticity has been reached, the measured axial and hoop strains are fairly axisymmetric. However, at 18,000 psi, when significant stress and strain redistribution has been achieved, some non-axisymmetric axial strains become evident, and remained so for Squeezes 2 and 3.

8. CONCLUSIONS

- Instrumented mockup-test results of the Tihange 2 pressurizer surge nozzle weld demonstrated that significant residual compressive strains (stresses) are developed at the weld ID surface due to the MSIP treatment.

- The mockup test results demonstrated that MSIP is effective in generating compressive residual stresses at the weld in PWR thick wall piping.

- Post MSIP OD surface profiles are acceptable for in-service inspection.

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3. “Mechanical Stress Improvement Process (MSIP) for Tihange Unit 2,” Christian Malékian, Westinghouse Alloy 600 Workshop, August 13-14, 2003, Pittsburgh, PA.

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6. AEA Technology Document “Analytical Verification of the MSIP for Pressurizer Surge Nozzle to Safe-end Weld for Electrabel Tihange Unit 2 – TSP,” April 2003.

7. “Westinghouse Qualification Procedure for Application of the Mechanical Stress Improvement Process (MSIP) for the Pressurizer Surge Nozzle at Tihange Unit 2,” Document No. MRS-SSP-1491, Rev. 0, dated April 23, 2003.

Table 1 Summary of MSIP Treatment Loads and Pipe Diameters Corresponding to the Three Pipe Squeezes

MSIP Application No.	Safe-End Average Diameter				Change in Safe-End Average Diameter			Hydraulic Pressure (psi)/ Box Press Force (lbs.)
	Pre-Squeeze		Post-Squeeze					
	Inches	mm	Inches	mm	Inches	mm	%	
Squeeze No. 1	14.774	375.26	14.598	370.79	0.176	4.47	1.19	18,000/1.276x10^6
Squeeze No. 2	14.598	370.79	14.569	370.05	0.205	5.21	1.39	18,500[(1)]/1.311x10^6
Squeeze No. 3 (Resqueeze No. 2)	14.569	370-05	14.533	369.14	0.241	6.12	1.63	19,000/1.347x10^6

Note:
1. Hydraulic pressure value is an estimate.

MSIP TECHNOLOGY

- Mechanically contracts the pipe on one side of weldment
- Replaces residual tensile stresses with compressive stresses

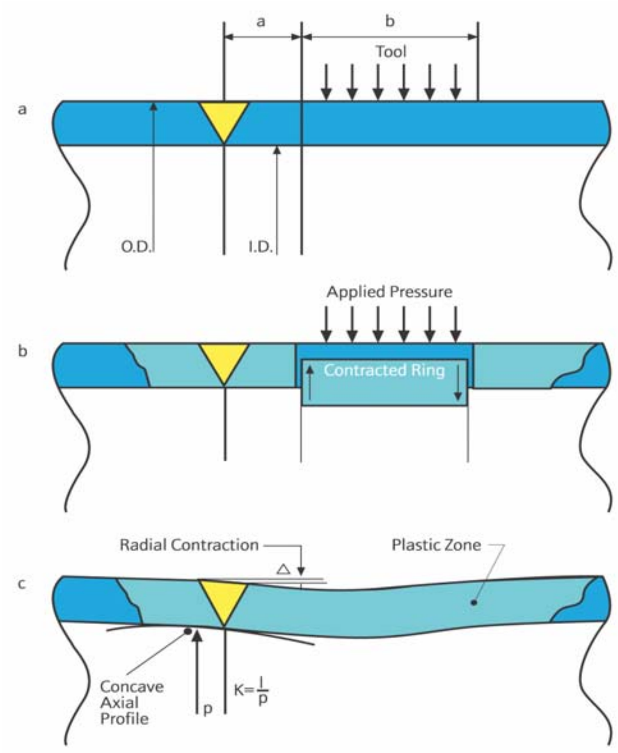

Figure 1 – Schematic Illustration of MSIP Process

Figure 2 – Photograph Illustrating the Mockup Test Setup and MSIP Tool in Place

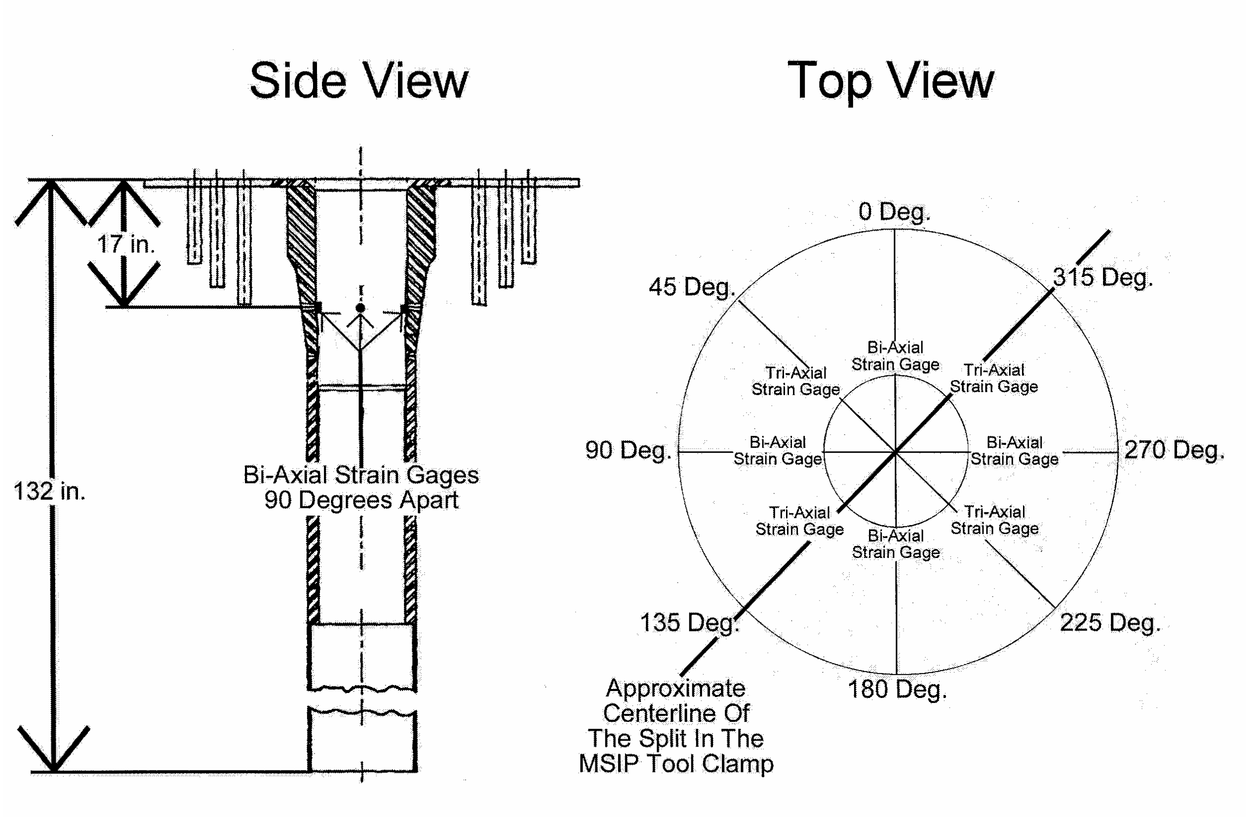

Figure 3 – Schematic Illustration of the Positions of Bi-axial and Tri-axial (Rosette) Strain Gauges near the Weld Region at the ID Surface

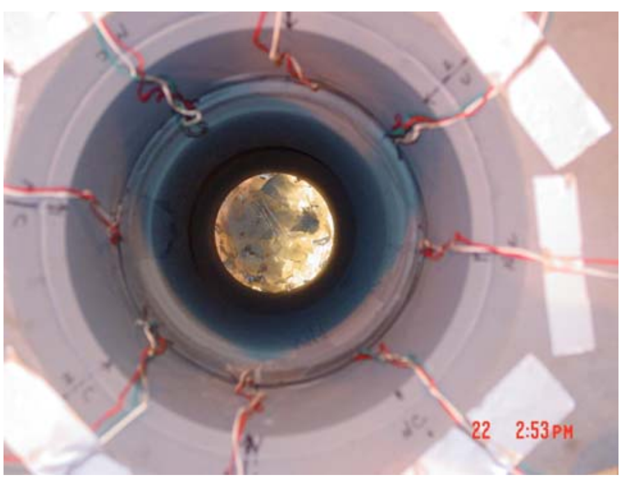

Figure 4(a) Illustration of the Instrumented Strain Gauges Installed on the ID Surface Near the Weld Location of the Mockup Pressurizer Nozzle

Figure 4(b) Illustration of the Installed Strain Gauges and the Strain Measuring Instrumentation at the Girard Site, Prior to MSIP Load Test

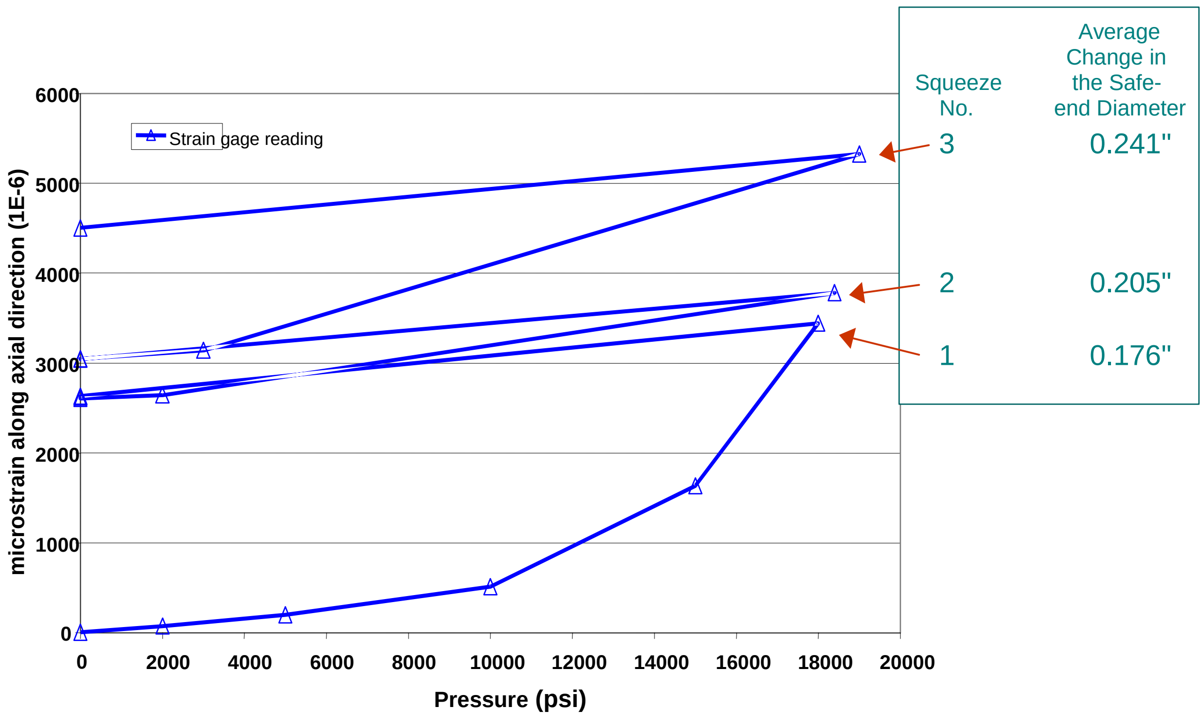

Figure 5 – Safe-end Diameter Changes Associated with the Three MSIP Squeezes

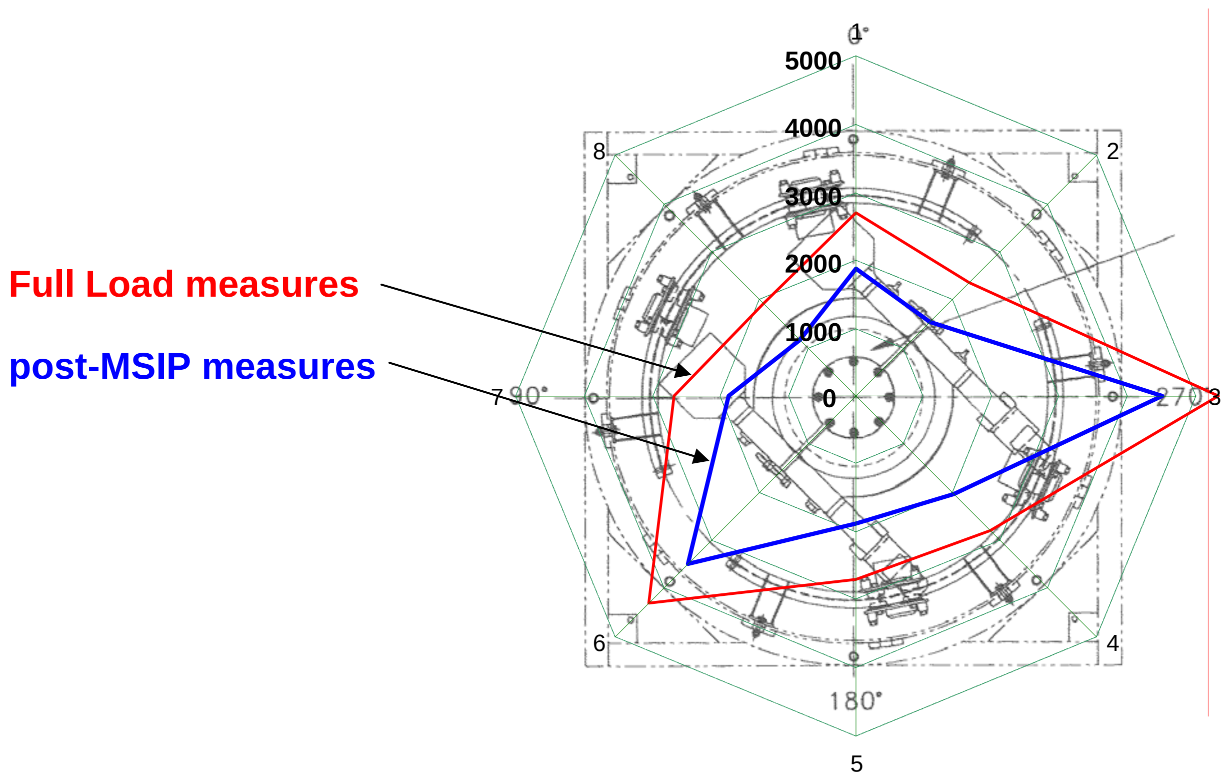

Figure 6 – Polar Plot of Axial Strains (Micro-Inches per Inch)

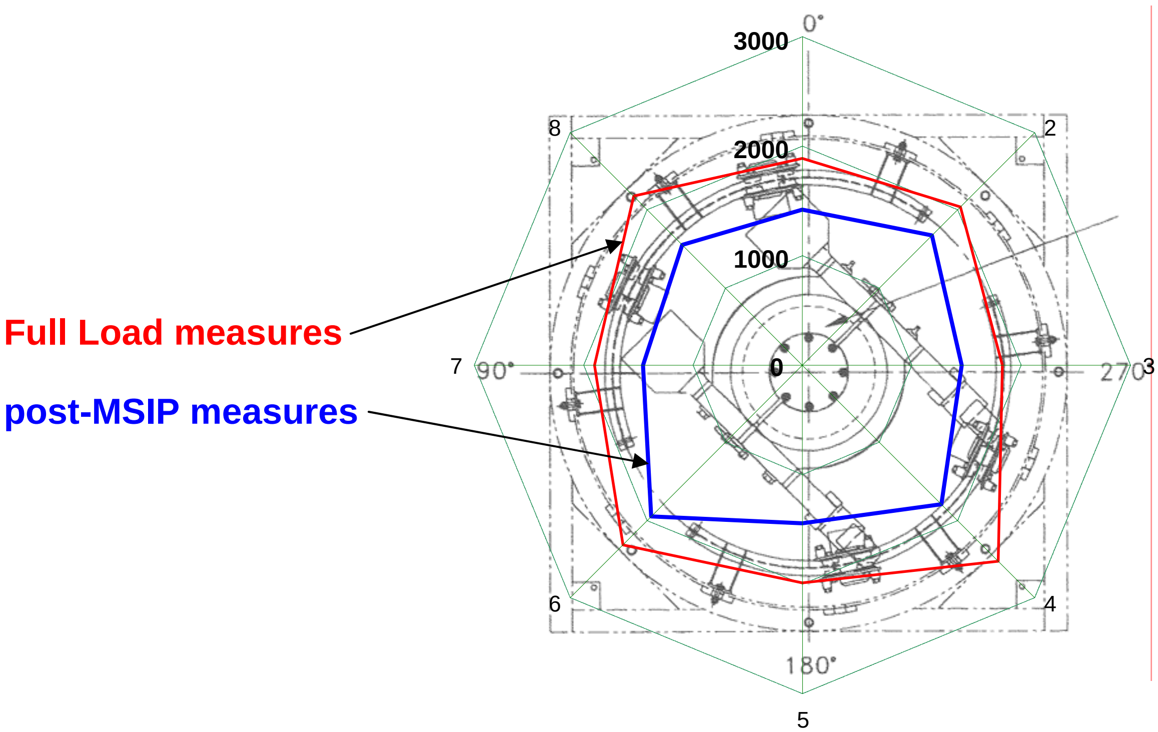

Figure 7 – Polar Plot of Hoop Strains (Micro-Inches per Inch)

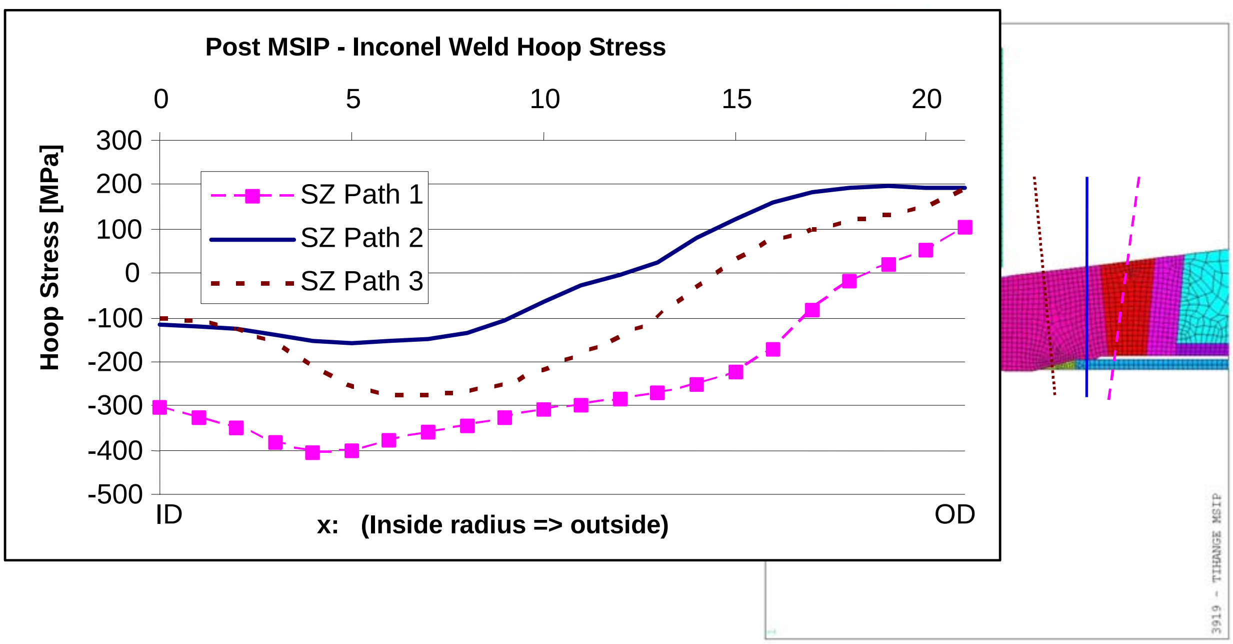

Figure 8 – Predicted Distribution of Through-Wall Strains at the Weld Joint (Ref. 2)

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11. ABSTRACT *(200 words or less)*

These two volumes of proceedings contain the visual projections (in Volume I), and the contributed manuscripts (in Volume II) from the Conference on Vessel Head Penetration, Crack Growth and Repair, held at the Gaithersburg Marriott at Washingtonian Center on September 29 - October 2, 2003. The conference was co-sponsored by the U. S. Nuclear Regulatory Commission and Argonne National Laboratory. Over two hundred attendees were provided with 45 presentations, divided into five sessions: (I) Inspection Techniques, Results, and Future Developments, (II) Continued Plant Operation, (III) Structural Analysis and Fracture Mechanics Issues, (IV) Crack Growth Rate Studies for the Disposition of Flaws, and (V) Mitigation of Nickel-Base Alloy Degradation and Foreign Experience. The conference opened with a plenary session including presentations giving the overview from the NRC Office of Regulatory Research, and an overview of nickel-base alloy cracking issues worldwide. The conference closed with a panel session consisting of industry representatives and NRC management discussing the prognosis for future issues in this area of concern.

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