SPECIALIST MEETING ON

STEAM GENERATORS

STOCKHOLM, SWEDEN 1–5 OCTOBER 1984

HOSTED BY
SWEDISH NUCLEAR POWER INSPECTORATE, SKI
AND
SWEDISH STATE POWER BOARD, SSPB
NEA/CSNI - UNIPEDE SPECIALIST MEETING ON
STEAM GENERATORS

STOCKHOLM, SWEDEN
1st - 5th OCTOBER 1984

PROCEEDINGS

HOSTED BY
SWEDISH NUCLEAR POWER INSPECTORATE (SKI)
AND
SWEDISH STATE POWER BOARD (SSPB)
The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and nuclear licensing. The Committee was set up in 1973 to develop and co-ordinate the Nuclear Energy Agency's work in nuclear safety matters, replacing the former Committee on Reactor Safety Technology (CREST) with its more limited scope.

The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries. This is done in a number of ways. Full use is made of the traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences. Some of these arrangements are of immediate benefit to Member countries, for example by enriching the data base available to national regulatory authorities and to the scientific community at large. Other questions may be taken up by the Committee itself with the aim of achieving an international consensus wherever possible. The traditional approach to co-operation is increasingly being reinforced by the creation of co-operative (international) research projects, such as PISC and LOFT, and by a novel form of collaboration known as the international standard problem exercise, for testing the performance of computer codes, test methods, etc. used in safety assessments. These exercises are now being conducted in most sectors of the nuclear safety programme.

The greater part of the CSNI co-operative programme is concerned with safety technology for water reactors. The principal areas covered are operating experience and the human factor, reactor system response during abnormal transients, various aspects of primary circuit integrity, the phenomenology of radioactive releases in reactor accidents, and risk assessment. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on power plant incidents.

The Committee has set up a sub-Committee on Licensing which examines a variety of nuclear regulatory problems, provides a forum for the free discussion of licensing questions and reviews the regulatory impact of the conclusions reached by CSNI.

* * * * * * * * * *

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SESSION 4

ANALYSIS OF

DETERIORATION MECHANISMS

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PROBLEMS
SUMMARY OF SESSION 4 - ANALYSIS OF DETERIORATION MECHANISMS AND STRUCTURAL INTEGRITY PROBLEMS

Session Chairman: Mr. F. Nilsson

Session 4 mostly deals with the deterioration mechanisms that can cause loss of structural integrity of steam generator tubes and tube supports. The majority of the papers (4.2-4.5) presented deals with observations on tubing that has experienced different kinds of deterioration. In these papers a number of general conclusions are drawn.

- Crevice chemistry is strongly correlated to the propensity of secondary side cracking. The total amount of impurities seems to be an important factor. If, however, the tube material contains grain boundary carbides cracking does not occur. The latter also applies to primary side cracking.

- The details of the chemical mechanisms of IGC are not yet clear. Sulfur attack is in 4.5 held as the most important mechanism.

- The strain in the rolling transition zone is an important parameter for primary side cracking.

- Small cracks (<.5 mm) are not detected by eddy current testing (4.2.1).

Since crevice conditions are of such importance it is obviously of interest to get a deeper understanding of the processes involved. An attempt at simulation of solute concentrations is reported in 4.1.

For safety decisions one is interested in the sizes of critical flaws. Using plastic collapse theory critical flaw sizes are calculated in 4.6. It is furthermore concluded that leak-before-break is likely to occur although one cannot exclude sudden breaks.

The general conclusion from the papers presented is that the understanding of the different damage mechanisms has increased, but that especially as regards chemistry much remains to do. Good material control (i.e. heat and mechanical treatment) is seemingly effective in preventing cracking and corrosion.
NEA/CSNI-UNIPEDE SPECIALIST MEETING
ON STEAM GENERATOR PROBLEMS
Stockholm, October 1984

SESSION 4.1
KINETICS AND PROCESSES OF SOLUTE CONCENTRATION
AT CREVICES IN STEAM GENERATORS

by

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SESSION 4.1: KINETICS AND PROCESSES OF SOLUTE CONCENTRATION AT CREVICES IN STEAM GENERATORS

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1. INTRODUCTION

When a solution is boiled, non-volatile solutes tend to concentrate at the heat transfer surface. The ratio of the surface to bulk concentration (concentration factor) however, remains near unity unless the surface contains a crevice or is covered with a porous deposit. The concentration factor may then be orders of magnitude higher and the concentrated solution often leads to corrosion of the adjacent surfaces. In earlier designs of PWR steam-generators, the crevice between the tube and tube-support plate acts as a concentration site which can lead to corrosion of the tube and support plate. The rate of concentration, the magnitude of the concentration factor and the rate of release of solute when conditions change are important factors when devising strategies to minimise corrosion. Values of these parameters for sodium chloride were therefore measured in a laboratory simulation of the crevice and are reported as a function of the heat transfer rate and the bulk concentration. The rate of salt release was measured following changes in heat transfer and system pressure.

2. EXPERIMENTAL

A diagram of the heated crevice is shown in Fig. 1. The alloy 600 tube (13 mm O.D.) was heated electrically over a length of 25 mm. The space between the heater sheath and tube was filled with a low melting-point alloy (Wood's metal), and contained five thermocouples. In the first series of experiments the annular gap between the tube and the low-carbon
FIG. 1 DIAGRAM OF HEATED CREVICE

Steel ring was 0.3 mm wide and 18 mm long. For a second series of experiments, four thermocouples sheathed in stainless steel were brazed and machined flush with the surface of the tube in the positions shown in Fig. 1. The tube was then nickel plated giving an annular gap of 0.23 mm. Initial experiments showed that little or no salt concentrated in an empty crevice. For this work therefore, the crevice was packed with carbon fibre in the form of a braided tube which was slipped over the alloy 600 tube before assembly to give a porosity of 56% and 49% in the crevice for the two series of experiments. Carbon fibre is inert under
steam-generator conditions and simulates the packing of the crevice by magnetite which occurs in operation.

The pressure in the autoclave was adjusted to give a boiling temperature of 280°C and experiments were conducted at constant heater power or at a constant temperature; for the latter experiments, the heater was thyristor-controlled to give a constant temperature at the centre thermocouple in the Wood's metal.

Deoxygenated solutions of NaCl (7 μg/kg Cl to 70 mg/kg Cl) with NH₃ added to pH 9.0-9.3 were pre-heated to near saturation temperature and pumped through the autoclave for various periods at a rate of 2.2 dm³/hour. The steam-water mixture leaving the autoclave was condensed and analysed continuously for sodium and chloride. When the crevice heater was switched off at the end of an experiment (maintaining pressure and preheat temperature) a surge in concentration was recorded. The whole of the effluent was collected during this stage; its volume and concentration gave the weight of salt released from the crevice.

3. RESULTS

3.1 Salt concentration at constant power

Results obtained with the first crevice using three different NaCl concentrations at the autoclave inlet and three levels of heater power are shown in Fig. 2. The data suggested that salt accumulated at a constant rate until an equilibrium value was reached and lines of slope 1 and 0 have been drawn through the data points to illustrate this. The temperature of the Wood's metal tended to rise as salt accumulated in the crevice, and plots derived from the temperature observations confirmed the sharp break shown in the curves of Fig. 2.

From the measured weight of salt and the known free volume of the crevice, the average concentration of salt in the crevice can be calculated. The equilibrium concentrations measured ranged from 8.5 to 270 g/dm³ NaCl, giving average concentration factors of 700 to 36000.
FIG. 2 ACCUMULATION OF NaCl IN A HEATED CREVICE AT DIFFERENT COMBINATIONS OF CONCENTRATION / HEAT FLUX (mg kg⁻¹ NaCl)/(kW m⁻²)

Equilibrium established previously at 120/150

3.2 Salt concentration at constant superheat

Results using the second crevice are shown in Fig. 3. A preliminary series of tests was conducted at a constant heat flux of 125 kW/m² with a bulk concentration of 12 mg/kg NaCl. As before, salt accumulated at a constant rate until an equilibrium was reached, though the limiting weight was greater under comparable conditions than in the first crevice.

Subsequent tests were carried out with the Wood's metal controlled at a constant temperature of 325°C. The saturation temperature was at 280°C throughout, giving a maximum available superheat of 45 K. Results for four different NaCl concentrations are shown in Fig. 3. The highest average concentration factor was $8 \times 10^5$ at the end of the 300 hour tests with 12 and 120 μg/kg NaCl solutions.
FIG. 3  ACCUMULATION OF NaCl IN A HEATED CREVICE AT DIFFERENT BOILER-WATER CONCENTRATIONS

\[ X \quad : \text{CONSTANT HEAT FLUX 125 kW/m}^2 \]

\[ \text{OTHERS} \quad : \text{CONSTANT } \Delta T \text{ 45 K} \]

The temperatures of the tube surface showed a consistent pattern of change during these tests. An example is shown in Fig. 4 for a bulk concentration of 12 mg/kg NaCl and an available \( \Delta T \) of 45 K. All three thermocouples within the crevice showed a distinct temperature rise at different times. With a bulk concentration of 1.2 mg/kg NaCl, the same temperature changes were seen, but at periods approximately ten times greater. Fig. 5 compares the tube surface temperature changes in the two tests plotted on different timescales.
FIG. 4  SUPERHEAT AT TUBE SURFACE FOR BOILER-WATER
CONCENTRATION OF 12 mg/kg NaCl. CONTROLLED PRIMARY ΔT: 45K

FIG. 5  SUPERHEAT AT TUBE SURFACE FOR TWO
DIFFERENT BOILER WATER CONCENTRATIONS:
—— 12 mg/kg; ——— 1.2 mg/kg NaCl
3.3 Salt release from the crevice

In every experiment shown in Figs. 2 and 3, the salt was released from the crevice within three hours of switching off the crevice heater. Experiments were also carried out to determine the effect of a controlled power reduction on salt release. A typical result is shown in Fig. 6.

![Graph showing salt concentration over time](image)

**FIG. 6 SALT CONCENTRATION AT AUTOCLAVE OUTLET WHILE REDUCING POWER APPLIED TO CREVICE**

About 2.5 mg NaCl were accumulated in the crevice by heating for 12 hours with a bulk solution of 1.2 mg/kg NaCl. At time zero, the concentration entering the autoclave was reduced to 12 μg/kg NaCl and when the outlet concentration was close to this level, power was reduced by adjusting the controlled ΔT downwards every hour. The figure shows the resulting power generated by the heater. It can be seen that no appreciable salt was lost from the crevice until the power was reduced to about 20% of its original value; in other experiments power was reduced to about 10% of the original value before significant salt release was observed.

In some of the experiments it was observed that a change of pressure in the autoclave when the crevice was nearly at the equilibrium level of salt tended to release salt from the crevice. Tests were therefore carried out...
with deliberately imposed pressure changes of up to 8 bar using a crevice containing about 2.5 mg NaCl accumulated as before. Each pressure change was completed within about 30 seconds but the weight of salt released by each change was less than 1 µg, or 0.05% of the total weight of salt in the crevice.

4. DISCUSSION

4.1 Concentration model

The linearity of the accumulation curves at constant heat flux (Figs. 2 and 3) shows that the rate of escape of salt from the crevice does not increase with the accumulated weight until the limiting weight of salt is reached. This suggests that salt accumulates in the crevice according to the model shown in Fig. 7. It is proposed that salt accumulates initially deep within the crevice, and that the transition from concentrated to dilute solutions occurs over a small distance (Fig. 7A). The release of salt due to diffusion or convection is therefore very low and continued boiling leads to an increase in the volume of the concentrated solution (Fig. 7B) without an increase in the release rate. As the salt continues to accumulate however, the boundary between dilute and concentrated solution approaches the mouth of the crevice. At this stage a small
increase in the weight of accumulated salt leads to a rapid rise in outward transport due to diffusion and convection, and hence to the establishment of equilibrium (Fig. 7C).

It is further proposed that in the absence of effective salt removal from deep within the crevice, the salt concentration rises to the equilibrium value determined by the available superheat within the crevice or to saturation. The transition from concentrated to dilute solution shown in Fig. 7 is therefore a transition from a non-boiling to a boiling solution. As the boiling boundary crosses any location, a temperature rise should be seen because of the fall in heat-transfer coefficient as boiling ceases. This is amply confirmed by Figs. 4 and 5. The similarity of the curves in Fig. 5 shows that salt accumulated in the same way from the two different solutions, but at a rate proportional to the bulk concentration.

In all cases, the temperature rise was seen first at the upper two thermocouples, suggesting that the equilibrium salt concentrations were formed first in the upper half of the crevice rather than at the centre as shown in Fig. 7.

4.2 Rate of salt accumulation

The maximum possible rate of accumulation is given by $QC/L$, where $Q$ is the heat flux into the crevice, $L$ is the enthalpy of evaporation and $C$ is the bulk concentration. From the accumulation rates of Figs. 2 and 3, the efficiency of the concentrating process can be calculated as a fraction of the maximum. For Fig. 2 (from left to right) the efficiencies were 0.52, 0.88, 0.81 and 0.52, and for the single constant-power test of Fig. 3 the efficiency was 0.88. Fig. 2 shows that of the two lowest results, one is based on only two experimental points, while the data for the other are scattered, mainly because of the short time scales involved. It is concluded that at constant power the measured salt accumulation rates are between 80-90% of the theoretical maximum.

At constant superheat, the model suggests that the power required in the crevice will diminish as the crevice fills with concentrate, because of the diminution in the heat transfer coefficient when boiling ceases. Such a power decrease was observed for the two upper curves of Fig. 3. The
decrease in the measured accumulation rate corresponds to the reduction in the rate of boiling.

The rate of accumulation at constant superheat can be predicted from the model by assuming that the power required for boiling is proportional to the length of crevice not filled with concentrate, and that concentration takes place at 100% efficiency. The mass of salt in the crevice \( m \) is then given as a function of time \( t \) by:

\[
m = m_e [1 - \exp (-Q_0 C t / L m_e)]
\]

where \( Q_0 \) is the initial heat flux and \( m_e \) is the equilibrium mass of salt in the crevice. This equation is used to plot the family of curves shown in Fig. 3 using only one empirical constant \( m_e = 14 \) mg. The good fit with the data supports the concentration model, and provides further evidence that concentration occurs at near 100% efficiency.

4.3 Equilibrium crevice concentration

The model predicts that the equilibrium crevice concentration is determined by the superheat within the crevice. It was not possible to test this prediction accurately, because of the limited array of thermocouples around the crevice. In one experiment with 12 mg/kg NaCl and a controlled primary \( \Delta T \) of 45 K, the average superheat at the tube surface was estimated as 15 K. This corresponds to an average concentration of 250 g/l in the crevice compared with a measured average of 164 g/l. Since no allowance was made for the temperature gradient between tube and support plate, the agreement is satisfactory.

The available superheat in the crevice will depend on the heat flux and Fig. 2 confirms the strong effect of the heat flux on the equilibrium weight of salt. Figs. 2 and 3 also show that the effect of the bulk concentration on the equilibrium weight is relatively small.

4.4 Salt release from the crevice

The rate of release of salt from the crevice after switching off the crevice heater agreed well with values calculated by assuming that the salt was released by diffusion from the packed crevice. Fig. 6 shows that little salt was released until the power was at a low level, in agreement
with plant experience of hide-out return. This is expected if the concentration process is under diffusion control, since little diffusion against the flow into the crevice will occur until the rate of flow (controlled by the boiling rate) is low.

Pressure disturbances led to a relatively small loss of salt when the crevice contained about 20% of the equilibrium weight, in contrast to bigger effects seen when the crevice was full. This confirms again that salt accumulates initially deep within the crevice.

5. CONCLUSIONS

For bulk water concentrations between 7 μg/kg and 70 mg/kg Cl salt accumulated in an annular packed crevice at a rate close to the maximum given by the rate of evaporation in the crevice multiplied by the bulk concentration. Reducing the bulk concentration to levels at least as low as 7 μg/kg Cl therefore reduced the rate of salt accumulation in the crevice, but did not lead to salt removal.

Salt continued to accumulate in the crevice until an equilibrium weight was reached. The equilibrium was determined primarily by the available superheat within the crevice, and was only weakly dependent on bulk concentration. Concentrated solutions formed first in the upper half of the crevice, and then increased in volume until the crevice was full.

The salt was released by diffusion from the crevice within a few hours of switching off the crevice heater while maintaining bulk water temperature. Very little salt was released however until the power was reduced to between 10 and 20% of its original value. Pressure fluctuations of up to 8 bar also failed to release significant weights of salt from a crevice, one-fifth full of salt concentrate.

Acknowledgements. This work was carried out at the Central Electricity Research Laboratories U.K. as part of EPRI Contract 1171-3 and is published by permission of CEGB and EPRI.

REFERENCES

1. G.N.W. Mann and R. Castle, Salt Concentration in Heated Crevices and Simulated Scale, EPRI report NP-3050 (October 1983)
SESSION 4.2.1

INTERGRANULAR STRESS CORROSION CRACKING OF RINGHALS 2 STEAM GENERATOR TUBES

Jan Engström
Swedish State Power Board
Kjell Norring
Studsvik Energiteknik AB

1. INTRODUCTION

Ringhals 2 is a PWR of Westinghouse design, operated by the Swedish State Power Board. It was put into operation in 1974. The unit has three steam generators with mill annealed Alloy 600 tubing. During the last fuel cycles a number of unscheduled shutdowns have occurred due to escalating cracking of steam generator tubes in the tube-to-tubesheet crevice region.

To identify the cause and extent of failures sections of steam generator tubes have been removed. One was removed in 1981\(^1\) and a further two in 1982\(^2\).

To achieve a better understanding of the overall bundle condition in the tubesheet area, for purpose of strategic planning, for correlation with nondestructive eddy current (EC) testing, and for better understanding of the crevice chemistry, it was decided to remove another fifteen tubes during the refuelling shutdown in May 1983.

Three tubes with EC-indications were selected for EC-correlation, while the other twelve tubes were chosen among the tubes without any sign of EC-indications. They were taken from various positions in the bundle.
As soon as practically possible after removal the tubes were soaked with demineralized water on earlier defined zones, each having an area of about 100 cm². The filtrate was analysed for species which were soluble in water during soaking (20°C). The particles collected on filters and the cotton wads used were also analysed.

To capture the complete crack situation the outer surfaces of the tubes, within the tubesheet area, were examined with fluorescent penetrant flaw detection.

After removal from the steam generator the tubes were EC-tested once more. Various types of probes ("Bobbin", "Array probes" and others) were used, but in no case was there any response except for the three tubes previously known to have EC-indications.

The tube sections were then transported to Studsvik where the bottom 100 mm, containing the roll transition zone, was cut from each tube. This section was split into two halves and the inner surfaces were examined by penetrant flaw detection using the same technique as for the earlier inspection.

Guided by the results from the above-mentioned examinations an extensive metallographic examination was performed.

2. CRACKS FROM THE OUTSIDE OF THE TUBE WALL IN THE TUBESHEET CREVICE REGION

The occurrence of cracks from the outside of the tube wall was first assessed by penetrant flaw detection. An example is shown in Figure 1.

The three tubes having EC-indications all show long axial indications (single cracks or bands of cracks). Seven out of the twelve tubes without EC-indications showed indications during penetrant flaw detection. One tube could not be evaluated due to heavy scratching and four tubes were free of indications.

From the extensive metallographic examination of cross-sections it can be concluded that only the three tubes having EC-indications have cracks with depths exceeding .5 mm (40 % of the tube wall thickness). However one of the tubes removed in 1982 contained a .56 mm deep crack (45 %) which was not
detected by EC-testing\(^2\). This examination shows that the EC-testing does not detect cracks with a depth less than .5 mm.

![Graph showing OD and ID defects](image)

Fig. 1  Cracks on a tube removed because of EC-indications

It can be concluded that out of the eleven tubes without EC-indications that have been metallographically examined:

- two tubes were free from OD-cracks
- three tubes have cracks less than .20 mm deep
- four tubes have cracks with depths of .20 - .35 mm
two tubes have cracks with depths of .36 - .50 mm.

All tubes removed because of EC-indications have cracks at least .9 mm deep (70 % of the wall thickness).

As illustrated in Figure 2 the carbide distribution is an indication of great importance as to the extent of cracking. The severe cracking is found in tubes with intragranular carbides, while tubes with grain boundary carbides are almost crackfree.

---

Figure 2 Influence of microstructure on the extent of cracking from the outside in the tubesheet crevice region
3. CREVICE ENVIRONMENT

The crevice chemistries have been assessed by soaking specific areas on the tubes immediately after removal. Both filtrates and the unsoluble particles remaining on filters etc were analysed\(^4\). The results from areas located deep in the crevice area are given in Appendices 1 and 2.

If it is assumed that the crevice is completely filled with water the following average concentrations of soluble species would be obtained\(^4\) (g/l):

<table>
<thead>
<tr>
<th></th>
<th>Na</th>
<th>K</th>
<th>PO(_4)</th>
<th>Cl</th>
<th>SO(_4)</th>
<th>F</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1.4</td>
<td>.05</td>
<td>.9</td>
<td>.3</td>
<td>.05</td>
<td>.05</td>
</tr>
</tbody>
</table>

An average crevice also contains the following amounts of unsoluble substances (mg):

<table>
<thead>
<tr>
<th></th>
<th>Na</th>
<th>Ca</th>
<th>Fe</th>
<th>PO(_4)</th>
<th>SO(_4)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>3.8</td>
<td>1.9</td>
<td>264</td>
<td>91</td>
<td>.9</td>
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</table>

In these calculations only the impurities on the tube were considered. The tubesheet may contribute and in that case the figures would be a bit higher.

In good agreement with data from the crevice flushing in 1982 and 1983\(^4\) it was found that there is a good correlation between sodium and phosphate among the water-soluble species. The Na/PO\(_4\) molar ratio was as high as 6 - 8. The ratio for the sodium phosphate that was used during a few months in 1974 was about 2.5.

A computer analysis aiming at, among others, to get a better understanding of the influence of crevice chemistry on the extent of cracking is not completed yet, but a few indications are worth mentioning.

Two tubes deviating for the better (crackfree) also revealed differences in the crevice chemistry. For these two tubes almost all sulphate dissolved in water during soaking, while on the other tubes this portion was less than about 20%. Some hints also appear if the ternary system iron - phosphate - alkali metals is regarded (Figure 3).
Fig. 3  Schematic representation of "good" and "bad" tubes in the diagram Fe - PO₄ - (Na + K + 2Ca)

At first it can be mentioned that there are no clear correlations between for instance Na (aq), PO₄ (aq) or conductivity on the extent of cracking. Much better relations are found if PO₄ (tot) and Fe (tot) is regarded. Among others this indicates that an environment probably containing iron-phosphate accelerates the intergranular stress corrosion cracking under these circumstances.

If the total environment is taken into account there is a rather good, positive, correlation between the total amount of impurities deep in the crevice and the extent of cracking. A correlation of the same magnitude, but negative, is obtained if the ratio between total amount of soluble species and the total amount of impurities within the lower part of the crevice is regarded (Figure 4). On the other hand the correlation between soluble species and extent of cracking is poorer (negative) than the other two.
Fig. 4  The influence of soluble part of crevice content on the maximum crack depth

4. CRACKS FROM THE INSIDE OF THE TUBE WALL AT THE ROLL TRANSITION ZONE

It was found that ten to eleven, out of fourteen examined tubes, reveal stress corrosion cracking from the primary side at the roll transition zone. This cracking will be discussed in detail in another paper presented at this conference.
5. SUMMARY

To achieve a better understanding of the overall bundle condition in the tubesheet area, for purpose of strategic planning, for correlation with nondestructive testing, and for better understanding of the crevice chemistry, it was decided to remove fifteen tubes from steam generator 1 at Ringhals 2 during the refuelling shutdown in May 1983.

At least eleven tubes show secondary side intergranular cracking in the tube sheet crevice region. Cracks below .5 mm is not detected by eddy current testing.

Clear differences in crevice chemistry were found. The results indicate a correlation between crevice chemistry and tube cracking.

At least ten tubes show primary side intergranular cracking in the roll transition zone. Its shape has a decisive influence.

Tubes with grain boundary carbides are crack free.

REFERENCES


5. K Norring. "Influence of Roll Transition Shape on Primary side IGSCC of Steam Generator Tubes". This conference.
| Tube   | Conductivity µS/cm | Analysis (ppm) |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |
| R3-C31 | 13.5               | 8.32           | 2.8| 0.30| <0.05| 0.01| 0.09| 0.100| 0.006| <0.05| 0.23| <0.01| 0.75| 0.02| 0.6| <0.3| 0.25| 4.8| <1 |
| R7-C38 | 40.5               | 8.77           | 8.6| 0.25| <0.05| 0.01| 0.09| 0.042| <0.001| <0.05| 0.03| <0.01| 0.01| 0.05| 0.9| 5.2| <0.1| 12.0| <1 |
| R10-C58 | 8.6               | 8.84           | 1.7| 0.12| <0.05| 0.01| 0.09| 0.050| <0.001| 0.05| 0.02| <0.01| 0.01| 0.01| 0.5| 0.8| <0.1| 4.3| <1 |
| R14-C45 | 7.9               | 7.89           | 1.1| 0.09| 0.1| 0.06| 0.07| 0.053| 0.003| <0.05| 0.16| 0.01| 0.36| 0.06| 0.5| <0.3| <0.1| 1.8| <1 |
| R15-C29 | 17.8              | 8.18           | 3.6| 0.62| 0.1| 0.01| 0.03| 0.025| 0.003| <0.05| 0.03| <0.01| 0.05| 0.10| 0.8| 2.0| 0.5| 7.2| <1 |
| R15-C47 | 20.5              | 8.40           | 4.0| 0.18| <0.05| 0.01| 0.06| 0.032| 0.001| <0.05| 0.02| <0.01| 0.01| 0.05| 1.8| <0.3| <0.1| 4.9| <1 |
| R18-C64 | 87.0              | 10.07          | 16.8| 0.24| <0.05| 0.01| 0.04| 0.085| 0.004| 0.05| 0.10| <0.01| 0.03| 0.23| 4.3| 12| 0.15| 9.8| 6.5 |
| R22-C22 | 4.3               | 7.11           | 1.6| 0.13| <0.05| 0.01| 0.03| 0.018| <0.001| <0.05| 0.02| <0.01| 0.01| 0.13| 0.3| 0.9| <0.1| 4.3| <1 |
| R24-C23 | 5.4               | 7.51           | 0.30| 0.34| 0.15| 0.19| 0.04| 0.034| 0.001| 0.06| 0.62| 0.02| 1.40| 0.26| 1.1| <0.3| 0.15| 2.9| <1 |
| R24-C47 | 72.0              | 9.88           | 14.0| 0.33| <0.05| 0.01| 0.09| 0.105| 0.001| 0.13| 0.17| <0.01| 0.01| 0.19| 1.8| 12| 0.8| 12.9| 3.4 |
| R26-C70 | 9.2               | 7.34           | 1.45| 0.32| 0.15| 0.01| 0.03| 0.062| <0.001| <0.05| 0.05| 0.01| 0.01| 0.53| 0.6| 0.8| <0.1| 4.0| <1 |
| R27-C17 | 53.0              | 9.70           | 10.4| 0.25| <0.05| 0.01| 0.05| 0.047| 0.001| <0.05| 0.02| <0.01| 0.01| 0.27| 0.2| 7.0| 0.15| 17.8| <1 |
| R32-C75 | 27.0              | 8.70           | 5.8| 0.17| <0.05| 0.01| 0.03| 0.014| <0.001| <0.05| 0.01| 0.01| 0.1| 0.6| 3.6| 0.3| 11.4| <1 |
| R34-C67 | 26.0              | 8.67           | 5.6| 0.19| <0.05| 0.01| 0.03| 0.034| 0.003| <0.05| 0.03| 0.01| 0.07| 0.26| 0.7| 3.3| <0.1| 8.4| <1 |
| R42-C47* | 290              | 10.78          | 54| 0.67| 0.05| <0.01| 0.08| 0.119| <0.001| 0.05| 0.75| 0.01| 0.01| 0.4| 4.5| 30| 2.1| 20.4| 15.3 |
| Ref A   | 1.2               | 7.0-| <0.05| 0.05| <0.05| 0.01| 0.01| 0.010| <0.001| <0.05| 0.01| 0.01| 0.01| 0.01| 1.3| <0.3| <0.1| <1 |
| Ref B   | 1.35              | 7.0-| <0.05| 0.09| 0.1| <0.01| 0.01| 0.006| <0.001| <0.05| 0.02| <0.01| 0.01| 0.01| 1.1| <0.3| <0.1| 1.7| <1 |
| Ref C   | 1.0               | 7.0-| <0.05| 0.12| <0.05| 0.01| 0.01| 0.003| <0.0003| <0.05| 0.01| <0.01| 0.01| 0.01| 1.1| <0.3| <0.1| 2.2| <1 |

* Calculated: OH⁻ 3.1 ppm
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<th>Ca (mg)</th>
<th>Li (μg)</th>
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<th>Fe (mg)</th>
<th>Cu (mg)</th>
<th>Zn (mg)</th>
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<th>Mg (mg)</th>
<th>B (μg)</th>
<th>F (μg)</th>
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* The solution for analysis of all elements except Li, Na and Cl was partially lost due to a damaged Pt-crucible.
SESSION 4.2.2

INFLUENCE OF ROLL TRANSITION SHAPE ON PRIMARY SIDE IGSCC OF STEAM GENERATOR TUBES

Kjell Norring
Studsvik Energiteknik AB

1. INTRODUCTION

Ringhals 2 is a PWR of Westinghouse design, operated by the Swedish State Power Board. It was put into service in 1974. The unit has three steam generators with mill annealed Alloy 600 tubing. Within the tubesheet only the bottom 50-60 mm of each tube is expanded.

In 1981 intergranular stress corrosion cracking (IGSCC) was detected from both the outside, at the tubesheet crevice region, and the inside of the tube wall, at the roll transition zone. To identify the cause and extent of tube failures sections of eighteen tubes have been removed from the hot leg side of the steam generators (1981, 1982, and 1983)\(^1\)-\(^4\). The intergranular stress corrosion cracking from the primary side at the roll transition zone was carefully examined.

Literature data indicates that cold work, decreases the time to failure of primary side IGSCC. Presented results\(^5\) show that the time of initiation decreases with increasing amounts of cold work (0-45 %) and that the
state of stress is important. Triaxial stresses being more detrimental. Cold work also increases the propagation rate of an existing crack. It is reported that small amounts of cold work (5%) may have an influence of the same magnitude as a larger amount of cold work (20%).

2. APPEARANCE OF RINGHALS 2 TUBES

Cracks from the primary side at the roll transition zone have been found on eleven to twelve out of sixteen examined tubes. A typical situation at a cracked area is shown in Fig. 1. This crack pattern has been visualized by light microscopy and stepwise (0.5 mm in each step) grinding and polishing of one tube half.

![Diagram](image)

Fig. 1 A typical crack pattern at a roll transition zone. (ID-surface, cross section, and shape of roll transition)
The shape of nonpenetrating cracks is a sometimes somewhat deformed semiellipse, having an a/c-ratio in the order of .6 to .9 (a = crack depth, 2c = crack length).

The shape of the roll transition zones differ considerably among the removed Ringhals 2 tubes. The radial expansion may differ with a factor of four (Fig. 2, right). The variation between two generatrixes on the same tube is also evident, at least a factor of two (Fig. 2, left).

![Graphs showing radial expansion and axial position](image)

**Fig. 2** Measured profiles of most and least deformed area.
- **Left:** on one single tube.
- **Right:** all tubes included.

3. **INFLUENCE OF MICROSTRUCTURE**

The carbide distribution is an indication of great importance to whether an Alloy 600 steam generator tube will crack or not due to IGSCC from the primary side. The influence on the extent of cracking is shown in Fig. 3.
Fig. 3 Influence of microstructure on the number of cracks

The cracks are found in tubes with intragranular carbides while tubes with grain boundary carbides are crackfree.

4. INFLUENCE OF ROLL TRANSITION SHAPE

For tubes with a crack sensitive microstructure i.e. intragranular carbides the shape of the roll transition zone is an important parameter in determining the extent of cracking. Two different parameters have been regarded: the radial expansion (r) and the length of the transition zone (l). All tubes had the same nominal diameter. Thus discussing r-values or r/r₀-values is equivalent. Another interesting parameter in this connection is the ratio r/l which reflects the gradient at the roll transition zone.

The radial expansion (r) for crackfree tubes with a crack sensitive microstructure, is found to be .09-.24 mm. The corresponding values for cracked tubes are considerably higher, .18-.34 mm. Although it seems to be a clear difference the correlation between r-values and number of cracks close to
(90° segment) the evaluated line is not obvious. The correlation coefficient is .40. The correlation coefficient for crack length versus number of cracks is in the same order, but of course negative.

A much better correlation (correlation coefficient .89) is obtained if the gradient (r/l) is considered. (Fig. 4).

![Graph showing the influence of r/l-ratio on number of cracks in the roll transition zone. Tubes with intragranular carbides.](image)

Fig. 4 The influence of r/l-ratio on number of cracks in the roll transition zone. Tubes with intragranular carbides

This diagram shows that cracks in the crack sensitive tubes are located to those roll transition zones, or parts of them, which have the most severe geometry (high r/l-values) while tubes with smoother transitions remain crackfree. Tubes with grain boundary carbides remain crackfree regardless of the shape of the roll transition zone.
5. CONCLUSIONS

The performed destructive examination of steam generator tubes removed from Ringhals 2 demonstrates that:

- cracks are confined to the rolling transition zone
- the shape of the transition zones differ considerably between tubes and even on the same tube
- tubes with grain boundary carbides are crackfree
- The shape, or more precise the gradient - r/l, has a decisive influence on the extent of cracking.

This evaluation of results from Ringhals 2 clearly shows that great attention must be paid to tube and hole dimensions and the rolling operation in order to avoid primary side IGSCC in the roll transition zone.

REFERENCES

1. J. Engström, K. Norring, "Intergranular Stress Corrosion Cracking of Ringhals 2 Steam Generator Tubes", This conference.


SESSION 4.3: RESULTS OF RESEARCH UTILIZING
A RETIRED FROM SERVICE PWR STEAM GENERATOR

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INTRODUCTION

Beginning in 1976 the United States Nuclear Regulatory Commission (U.S. NRC) sponsored research on steam generator integrity at Battelle Pacific Northwest Laboratory. Initial work investigated remaining integrity of Inconel 600 tubing with laboratory induced defect simulations of field experience. This investigation empirically established constitutive equations relating defect type and severity to tube burst or collapse pressure. While the program progressed, several new generator degradation mechanisms emerged. First indication of these new mechanisms was often in the form of tube failures resulting in leakage of primary coolant. Standard nondestructive testing procedures were not consistent in detecting previously undiscovered degradation mechanisms. At about this time, severe degradation of steam generators resulted in the first replacements in operating units. These removed from service units provided the opportunity to fully characterize actual steam generator conditions. They represented an inventory of defects that could be utilized for confirmatory research on remaining integrity of service defected tubing. Most importantly, an opportunity existed to validate the techniques for in-service nondestructive inspection of steam generators, through subsequent destructive assay.
The U.S. NRC with the cooperation of Virginia Electric and Power Company obtained the 'A' generator removed from the Surry 2 nuclear station. This unit was subsequently transported via barge to the Hanford Reservation (Richland, Washington), where it was later placed in a facility specifically designed and constructed for nondestructive and destructive characterization of the generator. Recognizing the difficulty and expense of research on a large radioactive component, NRC invited world-wide cooperative participation in this unique opportunity. The NRC was joined by consortiums from France, Japan, Italy, and the Electric Power Research Institute in funding a five year multifaceted research effort on the retired steam generator. The Steam Generator Group Project (SGGP) initiated research efforts in January 1982.

**Research Program**

The overall program task structure is illustrated in Figure 1. Principle research objectives deal with continuing generator integrity and safety throughout its service life. Three interrelated aspects are generator inspection (frequency/extent), nondestructive inspectability (accuracy/reliability), and actual remaining integrity of service induced defects. The SGGP will validate the reliability of defect detection and accuracy of defect sizing provided by current state-of-the-art eddy current steam generator tube inspection. This will be accomplished by establishing variability in data acquisition between teams and between differing equipment, and seeking to define the variability in data analysis. Validation will be through destructive assay of statistically significant numbers of removed-from-the-generator specimens. Actual remaining tube integrity will be established via burst and leak rate testing conducted at operating temperature on removed specimens. The NDE reliability data will be coupled with tube integrity determinations and an overall assessment of steam generator tube defect locations as inputs into a model of steam generator inspection. The basis inspection model will be that developed by Easterling.¹
FIGURE 1.

Several additional results are expected from the program. The usefulness of secondary side inspection as a compliment to primary side NDE will be explored. The effects of candidate primary side decontamination methods and possible secondary side chemical cleaning techniques on generator integrity will be sought. We hope to establish if advanced NDE techniques offer significant inspection improvement. The out-of-service generator also provides a realistic test bed for application of advanced maintenance and repair techniques. Subsequent destructive assay can confirm a technique's effectiveness.

Steam Generator Test Bed

The SGGP recognizes that a single removed-from-service generator cannot provide definitive information on all steam generator problems. However, the access to service induced defects for methodical study has been so limited that even a single generator offers great potential for advances in understanding aspects that affect all generators, such as NDE accuracy/reliability and decontamination. The Surry 2A generator is
a Westinghouse model 51D. The unit has a creviced tube sheet, drilled hole carbon steel support plates, and nonstress relieved inner row U-bends. The Surry stations received condenser cooling water from the brackish James River. Condensers were tubed in 90:10 Cu:Ni and had a history of leakage. Initial secondary side water conditioning utilized sodium phosphate additions. After a little more than a year in service, the unit was converted to all volatile treatment (AVT) chemistry.

During the slightly less than six years this unit was in service, ~22% of the tubes were plugged because of defects or engineering judgement. Known defects include wastage, inner row U-bend cracking and severe denting. The SGGP has obtained records on operation and in-service nondestructive testing.

Results

Channel Head Decontamination.² An important preparatory step to primary side research activities was reduction of the radiation field in the steam generator channel head. Though not a programmatic research objective, it was judged beneficial to explore the use of dilute reagent chemical decontamination techniques. These techniques presented potential for reduced personnel exposure and reduced secondary radwaste generation over currently used abrasive blasting techniques. Two dilute chemical reagent decontamination techniques were tested, LOMI (low oxidation state metal ion) and Can-Decon, one on either side of the channel head. The decontamination factors (DF's) achieved are illustrated in Figures 2 and 3. An overall field reduction within the channel head of ~10 was achieved by each technique. Stainless steel components had DF's in excess of 20, and Inconel 600 exhibited DF's of 4 to 5. Both techniques were limited to decontamination of the channel head bowl and underside of the tube sheet. Solution entrance into the steam generator tubes was avoided to prevent possible damage to the generator (i.e., through a leak to the secondary side) which could
compromise later experimental work. Thus, considerable shine from the tubes is experienced. Real time metal removal and radioactive species removal during each process was measured via corrosometer and frequent grab samples. In general, the duration of exposure to chemicals could be reduced based on these findings. Numerous corrosion coupons, stressed U-bend coupons, metal couple coupons and specimens of tubing from the generator were exposed to the decontamination processes. In no case was significant degradation, corrosion or cracking observed. A primary problem with the decontamination efforts was uniformity of solution agitation. This affected film removal and decontamination uniformity. This problem could be overcome with improved design of the fluid nozzles. In addition, a smeared film remained on the channel head surface. High pressure water lancing readily removed this, without further decontamination resulting.

**Tube Unplugging.** The majority of defects of interest for NDE validation work were in tubes that had been removed from service by plugging. An explosive Inconel 600 plug had been used. Using a remotely operated drilling technique, 969 plugs were removed in 20 days. Production rates in excess of 50 plug removals per 20 hour
operating day were readily achieved. Total personnel exposure associated with this task was 60.6 Rem. Figure 4 illustrates the drilling equipment in position. Of particular interest was the discovery of water in 82 tubes. Subsequent analysis indicated most tubes contained secondary side water, but other tubes had primary side or a mixture of primary and secondary side water, as determined by Boron content, Boron/Lithium ratio and indications of copper and phosphorus. Additionally, 28 plugs had sludge behind them, ranging from a few centimeters to over half a meter in depth. Sludge was expected in tubes, portions of which had been removed during service. However, a number of tubes that had been preventatively plugged and were not identified as leakers at the time of plugging, also contained sludge. This suggests continuing degradation of these tubes after plugging. Figure 5 is a map showing locations of water and sludge behind removed plugs.

FIGURE 4.
Yellow = Contained Water
Red = Contained Sludge
Black = Plug Locations

2A STEAM GENERATOR

Figure 5.

Post-Service Eddy Current Baseline Examinations. Two examinations were conducted of all accessible tubes after the unplugging operation. One examination utilized Zetec MIZ12/DDA4 technology and the other Intercontrole IC3FA/IC4AN technology. The results of these examinations are summarized in Table 1.

TABLE 1. POST-SERVICE BASELINE EDDY CURRENT INSPECTION

<table>
<thead>
<tr>
<th>Defect Size</th>
<th>&gt;40%</th>
<th>20-40%</th>
<th>&lt;20%</th>
</tr>
</thead>
<tbody>
<tr>
<td>Team A</td>
<td>207</td>
<td>329</td>
<td>175</td>
</tr>
<tr>
<td>Team B</td>
<td>337</td>
<td>200</td>
<td>520</td>
</tr>
</tbody>
</table>

42% match within 10%
Initial poor correlation between the two team's results was greatly improved when each team was informed of the others results. In the great majority of cases, both teams showed indications at similar locations on their raw data tapes. Discrepancies, however, remain due to differences in signal interpretation. The most common difference is one team indicating a copper deposit and the other team indicating a defect. However, even where both teams agree a defect exists, in excess of 27% of the defect sizing calls differ by more than 10%. One particular feature of interest was the location of a number of bulged tubes. All of these tubes had previously been plugged and a number still contained water at the time of plug removal. This over pressure phenomena correlates with secondary side inspection observations of tube ruptures in tubes that were not leakers when preventatively plugged.

Secondary Side Inspections. Extensive secondary side characterization of the steam generator was conducted. This characterization serves as a supplement to and a check on information generated by primary side inspections. Additionally, these inspections allow a comprehensive evaluation of support structure condition within the generator. The greatest difficulty in secondary side inspections is accessing the interior of the bundle, as is necessary for support plate characterization, sludge and corrosion product determination, and identification of loose parts. The project developed a miniaturized pin hole camera technology (Figure 6) to address this issue. We have successfully obtained access to all portions of the tube bundle from the secondary side using pin hole cameras, fiberscopes or borescopes. Figure 7 shows an example of an inner row U-bend crack. The top of an inner row U-bend has been severed, as shown in Figure 8. This potentially could be a result of the same pressurization that causes bulging in plugged tubes with leaky plugs. An example of support plate cracking and tube deformation is provided in Figure 9. Very extensive support plate deformation, almost exclusively from the hot leg side, is evident
in all but the uppermost support plate (Figure 10). A number of loose parts have been located by secondary side inspection; to date, these parts have not been consistently reported by primary side eddy current tests. Figure 11 shows support plate fragments that have accumulated at the tube sheet upper surface, and Figure 12 shows a small diameter wire (probably welding rod) that has been lost in the generator. A photograph that we believe shows wastage at the sludge pile is provided in Figure 13.
Figure 11.

Figure 12.
Current Efforts

The program is currently (at this writing) conducting round robin eddy current examinations of selected tubing samples. We are seeking data sets to aid in the statistical evaluation of eddy current detection reliability and sizing accuracy. The next activity will be looking at advanced or defect specific primary side characterization techniques, such as rotating eddy current point probe ultrasonics, and profilometry. Following these activities the generator will be sectioned for metallographic characterization and NDE validation. Other removed sections will serve as specimens for integrity testing and decontamination experiments. Sections of the tube sheet are scheduled for removal and characterization.

Conclusions

After less than six years operation the Surry 2 generators were replaced. A combination of design shortcomings and poor operating practice led to this early failure. This research program has documented massive structural deformation of the secondary side, which of itself could
potentially compromise the primary system boundary. No major unexpected
defect types have been found, with the exception of subsequent deforma-
tion and bursting of plugged tubes. Preliminary evaluation of eddy
current post-service baseline and round robin test data indicate some
discrepancies in detection of indications. However, there were consi-
derable team to team variations in analysis of indications, defect
versus deposit, and in defect sizing.

REFERENCES

1. R. G. Easterling, Exponential Responses with Double Exponential
Measurement Error - A Model for Steam Generator Inspection, Sandia
Laboratories (November 1978).

2. R. P. Allen, R. L. Clark, W. D. Reece, Surry Steam Generator
Channel Head Decontamination Task 6, Battelle Pacific Northwest
Laboratory report PNL-4712 (March 1983).

3. K. R. Wheeler, P. G. Doctor, L. K. Fetrow, M. Lewis, Surry Steam
Generator Selective Tube Unplugging Steam Generator Group Project
Task 8, Battelle Pacific Northwest Laboratory report PNL-4876
(September 1983).

Report Task 10, Battelle Pacific Northwest Laboratory report
PNL-5033 (October 1983).
SESSION 4.4: EXAMINATION OF TUBE SUPPORT AND TUBE SECTIONS REMOVED FROM AN OPERATING NUCLEAR STEAM GENERATOR

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1. INTRODUCTION

In 1975 a phenomenon which became known as steam generator tube denting was discovered at several nuclear power stations in the United States. Denting is the process of heat transfer tube deformation caused by corrosion of the tube supports as a result of concentration of impurities at local tube/support structure intersections.

During 1977, Millstone Unit No. 2 shut down for main condenser retubing. During this period eddy current and visual examinations were performed on the two steam generators. These inspections revealed that tube denting was occurring at the tube support plates. Based on these results, Combustion Engineering removed and examined sections of tubes and tube support structures in early 1978 to characterize the condition of the units. This paper describes some of the results from this investigation as well as a companion effort conducted at Battelle Columbus Laboratories.(1)

The Millstone Point Unit No. 2 steam generators are of the recirculating U-tubed design with mill annealed Ni-Cr-Fe Alloy 600 heat transfer tubing and
carbon steel tube support structures. Horizontal support of the tubes consists of nine levels of the eggcrate design (interlocking strips forming an open grid system) and two partial drilled support plates at the uppermost locations. These plates, which support approximately 25% of the tubes, consist of 1 inch thick carbon steel with 3/4 inch tube holes in a triangular arrangement and 1/4 inch flow holes at ligament intersections. The outermost heat transfer tubes are formed into a double 90° configuration with a horizontal section. This horizontal portion of the tube is supported by interlocking grids of carbon steel strips (Figure 1).

2. PLANT HISTORY

2.1 Operational

Millstone 2 went into commercial operation in December 1975. In May 1977, after only 18 months of operation, denting was detected in both steam generators.

During the initial 18 months of operation, maintenance of the all volatile steam generator secondary side water chemistry was hampered by numerous condenser leaks. Average blowdown chloride levels were near 0.6 ppm for this time period.

Two major equipment changes were implemented by Northeast Utilities in order to improve secondary water capabilities and thereby reduce corrosion effects. The original aluminum brass condenser was retubed with 70-30 Cu-Ni in May 1977, and full flow condensate polishing was implemented in April 1978. These modifications significantly reduced blowdown impurity levels.

Blowdown chloride and sodium prior and subsequent to these modifications are shown in Table 1. Cumulative chloride exposure experienced by the steam generators for the first 1000 days is shown in Figure 2. Although copper transport rates into the steam generators are not available for operation prior to May 1977, the Millstone 2 polishers have been shown to have a removal efficiency of 82 percent for oxidized copper species in solution.
Figure 1
Millstone Unit No. 2
Steam Generator

Figure 2 Cumulative Chloride Exposure
TABLE 1:
CHLORIDE AND SODIUM BLOWDOWN

<table>
<thead>
<tr>
<th>Blowdown (ppm)</th>
<th>Chloride To 4/77</th>
<th>6/77 - 1/81</th>
<th>Sodium To 4/77</th>
<th>6/77 - 1/81</th>
</tr>
</thead>
<tbody>
<tr>
<td>0-0.1(1)</td>
<td>2</td>
<td>96</td>
<td>39</td>
<td>98</td>
</tr>
<tr>
<td>0.1-0.5(1)</td>
<td>45</td>
<td>3</td>
<td>51</td>
<td>1</td>
</tr>
<tr>
<td>0.5-1.0(1)</td>
<td>30</td>
<td>&lt; 1</td>
<td>10</td>
<td>&lt;1</td>
</tr>
<tr>
<td>1.0-10.0(1)</td>
<td>23</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

(1) less than but not equal to

The improvements in bulk chemistry due to the modifications were not sufficient to stop the denting progression, but denting rates were reduced substantially.

2.2 Eddy current and visual inspection

Eddy current inspection of the steam generators revealed that a large percentage of the tubes passing through the partial drilled support plates had experienced some degree of denting. The most severe instances were present at the periphery of the tube bundle where some tubes would not allow passage of a .540 inch (14 mm) diameter eddy current probe. Visual examination of the uppermost support plate also indicated that a crack ran through a number of the tube holes in this region.

Based on these data, it was decided to remove a peripheral section of the upper support plate containing restricted tubes as well as those containing dents ranging from 0.005 to 0.014 inch (0.13 to 0.36 mm). Support strip sections associated with the horizontal run of some of these tubes were also removed.
3. SUPPORT STRUCTURE EXAMINATION

The drilled support plate and horizontal tube run support strips were subjected to visual and metallographic examination as well as various chemical analyses of the corrosion products. Of particular interest was the extent and composition of corrosion products, correlation of tube hole corrosion product thickness with dent measurements, and the morphology of the drilled support plate cracks.

3.1 Drilled support plate

Upon removal from the unit, the section of tube support plate was found to have numerous cracks in the tube hole to flow hole ligaments. The reconstructed plate section, and associated tube segments, are shown in Figure 3. The I.D. of all tube holes contained a substantial amount of hard scale and all flow holes contained some corrosion products. Figure 4 depicts a cross section from a typical portion of the plate. Note the complete corrosion of one of the tube hole/flow hole ligaments. A difference can also be seen in the appearance of the tube hole and flow hole corrosion product. That found in the tube hole was very dense and compressed with no observable porosity. Conversely the flow hole material was granular and less dense.

Measurements of the tube hole corrosion product indicated that it was essentially uniform in thickness at circumferential locations. In the axial direction, the thickest area was found to be approximately 1/8 to 1/4 inch (3-6 mm) in from either (upper or lower) plate surface. The average radial thickness in this region was 0.100 inch (2.5 mm) with a maximum measured of 0.140 inch (3.6 mm).

An attempt was made to correlate these measurements with eddy current dent size data. In general the amount of support plate corrosion was greater than that predicted by ECT. This was due primarily to the averaging of dent size by the eddy current technique and the asymmetric shape of the actual dent as a result of support plate deformation.
Figure 5 depicts the morphology of tube hole corrosion product. It was grayish in color, exhibited a lamellar structure and was very densely packed with no observable porosity. Extensive cracking was observed in the outer edge near the tube interface. Three distinct layers were evident: (1) a thin, light gray indigenous layer at the steel interface, 0.005 to 0.010 inch (0.13 to 0.26 mm) thick; (this layer was not present in every case); (2) a lamellar bulk layer and (3) a granular outer layer adjacent to the tube. This third layer varied from 0.005 to 0.015 inch (0.13 to 0.39 mm) thick and existed only in the center of the plate thickness.

Figure 5 also shows the expected composition of the tube hole corrosion product as determined by X-Ray diffraction, Mossbauer spectroscopy and X-Ray dot maps.

In the bulk sample, the corrosion products consisted mainly of two spinels \( \text{Ni}_x\text{Fe}_{1-x}\text{Fe}_2\text{O}_4 \) and \( \text{Zn}_y\text{Fe}_{1-y}\text{Fe}_2\text{O}_4 \) with copper, manganese, silicon, magnesium and sulfur in amounts greater than 0.100 weight percent and other trace elements present at less than 0.100 weight percent.

The inner layer adjacent to the steel principally contained Fe, Cl and Mn. The Cl concentration ranged from 1200 ppm to 3400 ppm, with a median value of 1600 ppm.

The middle layer contained Fe and Mn plus traces of Al, Si and S.

The outer layer contained Fe, Mn, Al, Si, Cu, Mg plus traces of Ca, S, Cl, Zn and Ni. The magnesium and silicon were uniformly distributed in this outer layer while copper was restricted to a narrow band on the inboard side of the layer. These species are typical of feedtrain corrosion products and seawater salts.

Figure 6 depicts the flow hole corrosion product which was granular, less dense, dark in color and contained large amounts of metallic copper particles. A thin light gray indigenous layer, similar to the one seen on the tube hole, was observed at the steel interface. This layer was also 0.005 to 0.010 inch (0.13 to 0.26 mm) thick but was not present in all cases.
Figure 3: Tube Support Plate Segment

Figure 4: Tube and Flow Hole Corrosion Product
Figure 5: Montage of Tube Hole Corrosion Product
In bulk sample, $\text{Fe}_3\text{O}_4$ and metallic copper were found with zinc, manganese, titanium, phosphorus and aluminum present in amounts greater than 0.100 weight percent. Fe, Al, Cu and Zn were uniformly distributed throughout the corrosion products. These species are typical of feedtrain corrosion products and seawater salts.

In assessing the condition of the steam generator, it was of interest to determine whether cracking of the tube support plate was due to mechanical means alone or whether other mechanisms particularly hydrogen damage were involved. Metallographic examination of partial tube/flow hole ligament cracks indicated they were transgranular and exhibited no decarburization or fissuring as would be indicative of hydrogen damage.

3.2 Horizontal tube run supports

Examination of the horizontal tube run support strips revealed that they also experienced accelerated corrosion at tube contact areas. Typically, an accumulation of corrosion products formed a wedge on either side of the tube/support contact point. This deposit exhibited a granular morphology and consisted of magnetite with high concentrations of copper and zinc oxides. Immediately under the tube, a thicker layer of magnetite was formed. This localized corrosion resulted in an observed support strip thickness reduction to 0.013 inch (.33 mm), and pits to 0.015 inch (0.38 mm) in depth. The tube side of this indigenous corrosion product also contained zinc, manganese, nickel, magnesium, silicon (oxides) and metallic copper.

4. TUBE SEGMENTS FROM SUPPORT PLATE REGIONS

Three tube sections exhibiting major and minor amounts of plastic deformation were selected for metallographic examination. Three distinct O.D. corrosion phenomena-pitting, localized etching, and intergranular penetrations - were observed on surfaces located within the tube support plate-tube crevice.

The pits were of a low population (2-5 pits/cm$^2$), fairly broad mouthed and shallow, with a maximum depth of 0.001 inch (0.02 mm). The width to depth ra-
tio varied from 4 to 20. Scanning electron microscopy indicated a morphology similar to that found in test boilers operated with sea water faulted volatile chemistry. (2)

The locally etched areas suggested some degree of metal loss as indicated by obliteration of belt polishing marks, however, no measurable wall thinning could be detected.

The third condition consisted of axially aligned O.D. intergranular penetrations to 0.0016 inch (0.04 mm) deep. These penetrations were found only in the area of maximum strain, on the tube exhibiting the largest dent (Figures 7 and 8). Mechanical deformation of unaffected sections of the same tube confirmed that these penetrations were a result of intergranular attack and not simply mechanical deformation or stress corrosion cracking. X-Ray maps indicated the penetrations were filled with material predominately consisting of zinc and sulfur.

No cracking or penetrations were observed on the I.D. surfaces of any of the tubes examined.

5. SUMMARY AND CONCLUSIONS

Localized corrosion of tube support plate and horizontal tube run support strips in the Millstone Unit No. 2 steam generators involved non-protective magnetite growth. The corrosion product morphology, chemical analyses, and distribution are consistent with the plant operating history and the accepted process of non-protective magnetite growth in acid chloride environments, as reproduced in experimental work confirming that process (2, 3, 4).

Fouling of tube hole annuli by feedtrain corrosion products most likely provided conditions in which contaminants present in the bulk water could concentrate. Unexpectedly, fouling and non-protective magnetite growth occurred in less restricted areas, such as flow holes and grid type supports, with lower heat fluxes.

Heat transfer tubing, while experiencing superficial O.D. corrosion, did not exhibit evidence of stress corrosion cracking under the strains imposed.
Figure 7: Dented Tube Cross Section

Figure 8: O.D Surface of Figure 7 Tube at Location A
REFERENCES


ACKNOWLEDGEMENT

The authors would like to thank the many people who contributed to this effort in particular, G. V. Forrest and R. Morin, for metallography, and P. E. C. Bryant, J. J. Koziol and K. R. Craig for their guidance and support.
SESSION 4.5

REVIEW OF RECENT STEAM GENERATOR TUBE EXAMINATIONS AT BABCOCK & WILCOX


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ABSTRACT

Portions of nuclear steam generator tubes have been removed and examined at the Babcock & Wilcox Company Lynchburg Research Center (LRC). These examinations have revealed several mechanisms responsible for degradation on the secondary surfaces of the Alloy-600 tubing. Pitting corrosion was observed in the sludge pile region of tubes removed from the Millstone Point Unit 2 steam generators and was attributed to the reduction of copper oxides to copper metal in an acid environment. Tubes from the R. E. Ginna, Arkansas Nuclear One Unit 1, and Palisades plants have experienced intergranular corrosion (IGC) within either the tubesheet or tube support crevice regions. The damage in the ANO-1 and Palisades plants is thought to have been caused by sulfur bearing solutions active at low temperatures in an acidic environment. The environment within the Ginna steam generator was most likely alkaline.
INTRODUCTION

Nuclear steam generators are basically shell-and-tube heat exchangers through which water passes to produce steam. Pressurized primary water containing lithium hydroxide and boric acid is heated to about 600°F by the reactor core and then passes through several thousand Alloy-600 tubes. Slightly alkaline boiler water at a lower pressure on the outside (secondary side) of the tubes boils to produce saturated steam to drive the turbines and generate electricity. Primary and secondary water chemistry control has recently become an item of high priority to plant owners in an attempt to minimize problems within steam generators. One of the main problems has been the prevention of corrosion-type problems in the Alloy-600 tubing.

Tube sections removed from nuclear steam generators are routinely examined at the Babcock & Wilcox Company's Lynchburg Research Center (LRC) near Lynchburg, Virginia. Most of this work is performed in the laboratories adjacent to the modern hot cell facility, which are equipped to perform nondestructive and destructive inspections. These include visual and photographic documentation, scanning electron microscopy and metallography on radioactive specimens. Tube sections recently examined include those removed from the Arkansas Nuclear One Unit 1, R. E. Ginna, Millstone Point Unit 2, and Palisades nuclear plants. Table 1 lists information specific to the design and operation of each of these plants. Tubing used in Babcock & Wilcox plants is sensitized due to the post-manufacture stress relief of the steam generators, while the Westinghouse and Combustion Engineering plants primarily use mill-annealed tubing. This paper describes the types of degradation observed on the tubes and discusses the most probable mechanisms responsible for the damage.

EXAMINATION RESULTS

Pitting Corrosion

To date, only two domestic nuclear plants have experienced severe pitting corrosion problems on the outside surface of steam generator tubes. In both plants, the majority of the pits have occurred within the sludge pile region.
which was present on top of the tubesheet and have been associated with areas of wastage, or general corrosion.

In late 1981, pitting corrosion was identified on the cold leg plenum in the steam generators at Millstone Point Unit 2, at which time a number of tubes were plugged. Pitting was again detected in mid-1983 along with an unidentified eddy current signal. To characterize this signal, portions of several tubes were removed from the hot leg plenum of steam generator #1 for laboratory analysis. Sludge and water samples were collected for chemical analysis during the sludge lancing operation performed following tube removal.

Examination results showed pitting and wastage corrosion present on the tubes just above the tubesheet secondary face elevation, near the base of the sludge pile. The majority of the pitting observed was present on one tube as single and overlapping penetrations to a maximum of 50 percent of the tubewall. Figure 1 shows the appearance of these pits in transverse cross section. On the tube surfaces adjacent to the pitting, wastage-type corrosion was observed as scalloped tube material loss and is shown in Figure 2.

Microchemical analyses of the tube deposits and corrosion products detected species such as silica, elemental copper and copper oxides, sulfides, and chlorides, in addition to oxides of the Alloy-600 constituents (Cr, Fe, Ni). Sludge samples were comprised primarily of oxides of copper and iron, with lesser quantities of those of zinc, manganese, magnesium, and silicon.(1)

**Intergranular Corrosion**

Of the many steam generator tubes examined at B&W, intergranular corrosion (IGC) is the most common type of damage observed. Crevice regions tend to be the most severely affected due to concentration of aggressive species.

As part of a project to define the mechanisms responsible for tubesheet crevice corrosion,(2) tube samples were removed from the R. E. Ginna Nuclear
Plant and subjected to extensive laboratory investigations to obtain metal-
lurgical and deposit chemistry data. In-situ eddy current testing (ECT) had indicated that IGC and possibly wall thinning was present on the tubes in the tubesheet region. Upon destructive examination of the tubes, uniform IGC was observed on the outer surface to a maximum depth of about 20 percent wall thickness, with a significant number of penetrations up to 75 percent. The damage was confined to the upper 12-inch portion of the tubesheet region of the tubes, with no damage observed above the tubesheet. Typical photomicrographs of the damage taken using the scanning electron microscope is shown in Figure 3. Auger electron spectroscopy (AES) identified C, O, N, Cl, Cu, Na, and S in corrosion deposit layers adjacent to regions of IGC. The tube material microstructure was determined using scanning transmission electron microscopy (STEM) to be partially sensitized, indicating slight chromium depletion of regions adjacent to grain boundaries.

Similar damage was observed on tube sections removed from a steam generator at Arkansas Nuclear One Unit 1 during the late 1982 refueling outage. In-
situ ECT had indicated defects in the upper tubesheet (UTS) region and upper tube support plate (TSP) elevations. To verify the type of damage and determine its extent, two tubes were removed and shipped to B&W for exami-
nation. One tube had multiple defect indications within the UTS ranging in depth from 36 to 84 percent wall thickness. Laboratory destructive exami-
nations showed the region of the tube corresponding to the lower 6-inch region of the UTS to be severely damaged due to IGC. A total of five circumferential cracks were present within this region, four of which had penetrated the tubewall. Each crack was filled with iron-rich deposits, which apparently had prevented primary coolant leakage during reactor operation. The bowl-like network of IGC associated with two partial-wall penetration cracks which became "linked" together is shown in longitudinal section in Figure 4. The corrosion product within a region of IGC was analyzed using electron spectroscopy for compound analysis (ESCA) which showed sulfur present as sulfide. The second tube examined showed a region of 20 percent IGC at the uppermost TSP location, where ECT had indicated a 63 percent defect. In addition, shallow (<10 percent) IGC was observed on
the outer surface of both tubes in regions where no indications were observed by ECT.

During the 1983 refueling outage at the Palisades nuclear plant, defect indications were observed during eddy current testing at the upper (11 to 14) TSP elevations of both steam generators (SG). To confirm and aid in interpreting the various depth indications, 73 tube sections from 58 individual tubes were removed and shipped to B&W for laboratory examinations. Results of these examinations showed regions of IGC penetrating from 8 to 100% of the wall thickness in tubes from the "A" steam generator. Nearly all defects were associated with TSP denting. The majority of the tubes examined from the "B" steam generator were free from defects. Huey tests showed all tubes tested from the "A" SG were heavily sensitized, while those from the "B" SG ranged from non-sensitized to mildly sensitized.\(^5,6\)

Electron microprobe analyses of the defect areas showed the deteriorated grain boundaries to contain iron, oxygen, nickel, and sulfur. Figure 5 shows an SEM photomicrograph and associated X-ray dot maps of these elements, which indicate their form was probably iron oxides and nickel sulfide. ESCA revealed NiO, Ni, Fe\(_3\)O\(_4\), PO\(_4\), SO\(_3\) and SO\(_2\) present in the surface scale adjacent to the IGC.\(^6\)

**DISCUSSION AND CONCLUSIONS**

**Pitting Corrosion**

The chemistry data generated during the examination of tubes from Millstone Point were used to postulate a likely mechanism responsible for the pitting corrosion. Pourbaix potential-pH diagrams for Cr, Fe, and Ni in aqueous solutions containing S and Cu were referenced to infer local chemical conditions under which the pitting may have occurred.\(^7\) It is postulated that copper oxides accumulated in the sludge pile due to copper feedtrain component corrosion. Localized boiling near the tubesheet concentrated acid-forming anions such as chlorides and sulfates in the sludge. This caused the local pH to drop below approximately 4, breaking down areas of the passive oxide film on the tube surfaces. Copper oxides were then
reduced to copper ions and involved in a dissolution reaction with the more chemically active Alloy-600 constituents. Variations in the electrochemical potential along the tube caused damage at different sites, namely the pitting and wastage corrosion.

The IGC in the Ginna tubes was confined to the tubesheet crevice region above the mechanical roll. Outside surface deposits contained Na and K, which indicates a caustic-like environment\(^4\) at elevated temperatures. Carbonate, silicate and unidentified compounds containing sulfur and chloride were also present in the deposits. Laboratory corrosion testing\(^2\) has shown that a combination of electrolytes containing NaOH and Na\(_2\)CO\(_3\) tested at 650°F in a crevice environment produces IGC in mill-annealed Alloy-600 tubing. Further testing produced significant amounts of IGC at elevated temperatures without the carbonate addition.

Similar crevice corrosion has been observed on tubes from the hot leg plenum by other investigators.\(^8\) The damage was attributed to caustic concentration within the tubesheet crevices as a consequence of temperature gradients. Concentration of carbonates and sulfates within crevices has been shown to accelerate IGC, while some of the silicates and chlorides apparently inhibit the corrosion in a caustic environment. The actual mechanism responsible for the tube damage at Ginna is complex and has not been completely verified by the laboratory testing to date.

Intergranular corrosion was also observed in crevices in the ANO-1 and Palisades steam generators. Most of the damage in the ANO-1 plant was confined to the upper tubesheet region, while the Palisades damage was within tube support plate crevices and normally associated with denting. Chloride and sulfur species were identified in corrosion products on tubes from both plants. All tubing in the ANO-1 steam generators is in the sensitized condition, while most of the tubes which experienced damage at Palisades were also sensitized to some degree. Sensitized microstructures are generally susceptible to IGC in certain sulfur bearing acid environments such as polythionic acid and sodium tetrathionate.\(^9\) Intergranular
corrosion of Alloy-600 in the mill-annealed condition has generally been associated with caustic environments. The branched IGC networks observed in both ANO-1 and Palisades tubing are representative of those caused by acid solutions, while that observed during the Ginna tube examinations indicated caustic-induced corrosion.

The most probable cause of the IGC damage in the ANO-1 and Palisades tubes was sulfur attack of grain boundary chromium depleted regions. Reduced sulfur species were "hiding-out" in crevices during wet lay-up conditions. An oxidizing crevice environment was created in the upper regions of the steam generators due to air ingress producing oxygenated water. Reduced sulfur species present were then converted to compounds such as thiosulfate, which attack chromium depleted regions. Upon reactor start-up and hydrazine addition, thiosulfate (unstable at elevated temperatures) was reduced to non-aggressive sulfur species.

Laboratory testing has shown denting to result from acid chloride solutions in tube support plate crevice environments. (10) Since denting, chloride, and sulfides of Cu, Fe, and Ni were detected in corrosion products on the Palisades tubes, this is strong evidence in support of an acid environment. In 1974, a laboratory examination (11) of tubes from Palisades did not show support plate denting. Since phosphate treatment secondary chemistry was used before late 1974, this suggests that the denting occurred using AVT chemistry.

ACKNOWLEDGMENTS

The authors would like to thank those persons from which permission was granted to publish the information in this paper: Mr. P. R. Habicht of Northeast Utilities, Dr. B. L. Dow of Arkansas Power & Light, Mr. A. E. Curtis of Rochester Gas & Electric, J. R. Schepers of Consumers Power Company, and Dr. J. P. N. Paine and Mrs. M. J. Angwin of Electric Power Research Institute.
REFERENCES

1. S. C. Inman, "Examination of Steam Generator Tube Sections From the Millstone Point Unit 2 Nuclear Power Plant," Project S304-6, EPRI, Palo Alto, California (to be published).


5. G. O. Hayner and T. J. Zeh, "Examination of Steam Generator Tube Sections From the Palisades Nuclear Power Plant," Project S304-1, EPRI, Palo Alto, California (to be published).


<table>
<thead>
<tr>
<th></th>
<th>Arkansas Nuclear One-1</th>
<th>R. E. Ginna</th>
<th>Millstone Point-2</th>
<th>Palisades</th>
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<tr>
<td>Steam generator design, type</td>
<td>B&amp;W-177, OTSG</td>
<td>W - 44, RSG</td>
<td>CE-67, RSG</td>
<td>CE - original commercial, RSG</td>
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<td>Tubing - Vendor</td>
<td>B&amp;W - TPD</td>
<td>Huntington</td>
<td>Pacific Tube</td>
<td>Huntington(2)</td>
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<td>Heat treatment</td>
<td>Sensitized, 1100-1150°F</td>
<td>Mill annealed, 1800-1850°F</td>
<td>Mill annealed, 1875 + °F</td>
<td>Variable</td>
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<tr>
<td>Secondary Water Chemistry - Full flow condensate polishers</td>
<td>Yes</td>
<td>Since January 1978</td>
<td>Since April 1978</td>
<td>(3)</td>
</tr>
<tr>
<td>All volatile treatment (AVT)</td>
<td>Yes</td>
<td>Since November 1974</td>
<td>Yes</td>
<td>Since October 1974</td>
</tr>
<tr>
<td>pH range</td>
<td>8.8 - 9.2</td>
<td>8.7 - 9.3</td>
<td>8.8 - 9.2</td>
<td>8.5 - 9.5</td>
</tr>
</tbody>
</table>

(1) Abbreviations: B&W - The Babcock & Wilcox Company  
W - Westinghouse Electric Corporation  
CE - Combustion Engineering Corporation  
TPD - Tubular Products Division  
OTSG - Once through steam generator  
RSG - Recirculating steam generator

(2) Probable, but may have been supplied by two vendors.

(3) Used 20 days in 1979.
Figure 1. Transverse cross section of pitting corrosion just above the tubesheet secondary face of a Millstone Point Unit 2 steam generator tube.

Figure 2. Transverse cross section of wastage-type corrosion just above the tubesheet secondary face of a Millstone Point Unit 2 steam generator tube.

Figure 3. Scanning electron photomicrographs of intergranular corrosion in the tubesheet region of an R. E. Ginna steam generator tube.
Figure 4. Longitudinal cross section of intergranular corrosion within the crevice region near the tubesheet secondary face of an Arkansas Nuclear One Unit 1 steam generator tube. (slightly stressed)

Figure 5. Scanning electron photomicrograph and X-ray dot maps of an area of intergranular corrosion in a tube support plate crevice of a Palisades steam generator tube.
SESSION 4.6 : CRITICAL FLAW SIZES
IN STEAM GENERATOR TUBING

P. Hernalsteen
Tractioneel

1. INTRODUCTION

Flaws detected through PWR steam generator tubing inspection have to be evaluated against acceptance criteria (such as proposed by USNRC R.G. 1.121-Ref. 1). This requires knowledge of the critical sizes of various types of defects.

Available theories allow prediction of the ductile failure of pipes and tubes, as a function of
- The shell parameter \( \lambda = k \frac{c}{\sqrt{Rt}} \)
  
  \( (c = \text{crack half length}/R = \text{pipe mean radius}/t = \text{pipe thickness}) \)
- The material "flow stress" \( \sigma_0 \) \( (YS < \sigma_0 < UTS) \)

However there is a lack of data for small diameters \( (< 3") \), large values of \( \lambda (> 8) \), specific inconel properties and dependance of \( \sigma_0 \) on conventional tensile properties (YS, UTS).

An experimental program has been conducted from 1980 to 1983 to produce such data on tubes in the range of 3/4" to 7/8" diameter from inconel 600 (several heats) and similar ductile materials (austenitic or ferritic).
A similar program was conducted, independently and at about the same time, by Battelle PNL on behalf of the USNRC (Ref. 2); as there is but little redundancy between these two programs, they can be considered as complementary.

2. DESCRIPTION OF TEST PROGRAM

Materials included in the test program are:
- Inconel 600 (2 sizes x 2 heats)
- Austenitic stainless steel SA 176 TP 304 (2 sizes)
- Austenitic alloy Al-6 X-HT (25% Ni/20% Cr/6% Mo)
- Ferritic alloy SEA CURE (2.5% Ni/26% Cr/3% Mo/.5% Ti).

Dimensions and mechanical properties (measured from flattened transverse tensile tests) are given in Table 1. Materials were not limited to Inconel in order to cover a wider range of mechanical properties and provide a broader theoretical basis for practical extrapolations.

Test specimens (tube sections with Swagelock end fittings) were provided with various defect types (from 1 to 5 defects/specimen)
- Through thickness axial electrodischarged machined (EDM) slits or natural (fatigue) cracks
- Partial penetration axial EDM slits
- Wall thinning (machined flat) in the axial direction
- Through thickness circumferential EDM slits

An inside plastic hose (backed up by a thin copper patch in some cases) was used to restore leaktightness of specimen with through-thickness flaws.

Each specimen was inserted into a cold water test loop and pressurized up to unstable flaw extension. Specimens provided with multiple defects were repressurized after cutting out the ruptured section resulting from a first test.
About 65 bursting tests were performed; a number of geometrical measurements (COD, bulging, flaw and tube deformations) were recorded during test and after bursting.

3. THICKNESS AXIAL FLAWS

3.1 Failure mode

For EDM flaws no slow (stable) crack growth was observed to precede the bursting pressure.

Failure was characterized by 45° inclined shear lips associated with axial cracking from each end of EDM slit; some deviation in the circumferential direction was observed for the SS-304, Al-6X-HT and, mainly, SEA CURE materials.

The failed specimens showed extensive "fishmouthing" and a marked ovalization at both ends of the crack (minor diameter in the flaw radial plane). Large ductile deformations were observed when the pressure exceeded about one half of the bursting pressure, as illustrated by

- Fig. 1 for crack opening displacement (COD) at both end of the EDM slit
- Fig. 2 for flaw opening, measured at mid length of the EDM slit
- Fig. 3 for "bulging" of the tube.

The COD at rupture ranged from .5 mm (Al-6X-HT) to 2 mm (SS 304); for inconel it was in the range of 1 to 1.2 mm.

3.2 Validation of collapse load theories

Collapse load theories (Ref. 3) predict failure when the circumferential stress reaches a critical value defined by

\[ \sigma_{Cr} = \frac{\sigma}{m} \]

where

- \( m \) is the "bulging factor" (also called FOLIAS factor), a function of \( \lambda \), for which various proposed approximations are given by Fig. 4.
- \( \sigma \) is the "flow stress", assumed to be constant for a given material and somewhat ill defined between YS and UTS.
The inconel (Ø 7/8") test results correlated very well with this model (see Fig. 5) for values of the shell parameter up to \( \lambda = 18 \); the value of \( \bar{\sigma} \) was selected to obtain a best fit adjustment.

3.3 Dependence of "flow stress" on YS and UTS

Based on the good correlation obtained hereabove, the "flow stress" was calculated by \( \bar{\sigma} = m \times \sigma_{cr} \) for the other materials and tube sizes (the flaw length was adjusted to keep an almost constant value of \( m = 3 \)).

The obtained \( \bar{\sigma} \) values were then compared to be usual room temperature tensile properties (in the transverse direction) using several tentative correlation criteria as summarized in Table 1.

The best correlation was obtained by

\[
\bar{\sigma} = k (\text{UTS} + \text{YS}) \\
\text{with } k = 0.57
\]

Each of the listed values of \( \bar{\sigma} \), YS and UTS is the mean of about 4 experimental results. It should be noted that, while the data scatter band was usually less than 10% for \( \bar{\sigma} \) and UTS, it was somewhat larger for YS (influence of the tensile specimen flattening?).

The proposed \( k \) value (0.57) is applicable for very ductile materials in the considered range of diameters and schedules; for other conditions, \( k \) might prove to be dependant on the tube geometry and/or material fracture toughness.

3.4 Parametric variations

Initial tube ovality (out of roundness of about 11%) did not significantly influence bursting pressure. Parallel flaws (from 5 mm to 90° apart) reduced slightly the bursting pressure (from 0 to 20%).
The effect of notch acuity was investigated by two ways:

- Repressurizing an already ruptured specimen (with relatively small crack extension) gave the expected bursting pressure (from the "new" flaw length); however, no COD was observed before unstability.
- Pressurizing a flawed specimen, with fatigue crack extension on both ends of the EDM slit, resulted in a somewhat lower bursting pressure (20%).

A non significantly reduced COD was observed together with stable crack growth starting at about 90% of maximum pressure; therefore the bursting pressure was probably not significantly reduced when related to the increased crack length.

As these data are based on a number of tests too small for statistical significance, their quantitative value should thus be assessed with some care.

4. OTHER AXIAL FLAWS

4.1 Partial penetration flaws (EDM slits)

Only relatively long defects of this type (λ ≈ 12) were investigated.
COD variation, at the center of the flaw, was measured by "clip-on gages" and is typically illustrated by Fig. 6.

For such large values of λ (λ ≥ 12), the critical stress is given by

\[ \sigma^- = UTS \left(1 - \frac{a}{t}\right) \]

where \( a \) is the flaw depth.

as should be expected for the case of uniform wall thinning.

4.2 Axial wall thinning (EDM slits)

For "long" defects, failure is predicted by the same formula as hereabove.
For "short defects" (λ < 12), the best correlation of experimental data was found, with Battelle formula (Ref. 2) relative to uniform thinning 
(of same length)

\[ \sigma^- = UTS \left(1 - \frac{a}{t}\right)^{\alpha}, \quad \alpha = 1 - e^{-0.26 \frac{c}{R(t-a)}} \]
5. THROUGH THICKNESS CIRCUMFERENTIAL FLAWS

For the tested inconel specimens, the COD at bursting pressure was similar (about 1 mm) to that for through-thickness axial flaws; flaws lengths were varied from 120 to 180°.

The bursting pressures could be correlated by using the "net section collapse criterion" (Ref. 4).

\[ \sigma = \bar{\sigma} \frac{2}{n} \left[ \cos^{-1}\left(\frac{\sin \frac{\alpha}{2}}{2}\right) - \frac{\alpha}{2} \right] \]

where 
- \( \alpha \) is the half crack angle
- \( \bar{\sigma} \) is a flow stress

\( \bar{\sigma} \) was taken equal to .57 (UTS + YS) with UTS and YS measured in the axial direction.

The number of tests was too small to assess the statistical significance of this conclusion.

6. CRITICAL FLAW SIZES UNDER SERVICE CONDITIONS

6.1 Service loading

The only effective load (resulting in ASME "primary stresses") acting on the S.G. tube bundle, in the tubesheet vicinity, is the pressure differential \( \Delta p \). The normal operation value (about 100 bar) is but slightly exceeded during postulated accidents (such as LOCA), yielding a circumferential stress of about 85 MPa.

6.2 Flow stress at temperature

The minimum flow stress \( \sigma \) may be evaluated as follows for Inconel 600 at operating temperature (600°F = 316°C).
From ASME code case N^20 1484 (Ref.5), the following minimum properties are applicable.

<table>
<thead>
<tr>
<th>T (°C)</th>
<th>YS (MPa)</th>
<th>UTS (MPa)</th>
</tr>
</thead>
<tbody>
<tr>
<td>20</td>
<td>276</td>
<td>552</td>
</tr>
<tr>
<td>316 (600°F)</td>
<td>244</td>
<td>552</td>
</tr>
</tbody>
</table>

From extensive data at 343°C (650°F) available from the Doel4/Tihange 3 inconel tubing, the YS temperature decrease is confirmed (89%) but the UTS appears to be also influenced (96%).

From limited data on correlation between longitudinal and transverse properties, a reduction factor of 0.9 is also assumed appropriate.

This leads to a minimum value of
- \( \sigma = 0.57 \times (244 + 0.9 \times 552) = 440 \) MPa in the axial direction
- \( \sigma = 0.9 \times 440 = 400 \) MPa in the transverse direction.

6.3 Critical flaw sizes

Based upon the hereabove assumptions, the minimum critical sizes can be established for the various types of defects under consideration; the applicable values for through-tickness flaws are listed in Table 2.

6.3 Allowable flaw sizes

Allowable sizes are dependant on the particular safety margins requested by the applicable regulations. If R.G. 1.121 (Ref. 1) is considered, a factor of 3 must be applied to the service stresses leading to the "allowable values" listed in Table 2. It can be seen that the resulting safety factor on the crack length is larger than 3 for axial flaws and smaller than 3 for circumferential flaws.

6.4 Tubes partially rolled in tubesheet

When tubes are only rolled in the lower part of the tubesheet, it can be shown that no axial defect can be critical in the crevice area, because of the extensive precritical tube bulging (Fig. 3).
6.5 Leak before break behaviour

Through thickness cracks lead to large leak paths (much larger than predicted from fracture mechanics evaluation) before a critical size is reached. Fig. 1 and 2, for instance, show that early warning should be available through easily detectable leaks.

However this does not demonstrate that a partial penetration crack could not reach a length such that it would be critical when going through; such ruptures, without any prior leak, did indeed occur in tube bends; a particular case was the SGTR event of the Doel 2 unit in June 1979.

REFERENCES

1. USNRC Regulatory Guide 1.121
   "Basis for Plugging Degraded PWR Steam Generator Tubes" (1976).


3. Erdogan F. "Ductile Fracture Theories for Pressurized Pipes and Containers".

4. Ruiz and Corran "Practical Application of Extremal Elastic - Ideally Plastic Solutions for the Assessment of the Severity of Cracks"

5. ASME Code Case N 20 (1484-3)
   "SB-163 Ni-Cr-Fe Tubing (Alloys 600 and 690) and Ni-Fe-Cr Alloy 800 at a Specified Minimum Yield Strength of 40 ksi/Section III, Div. 1, Class 1" (1979).
Fig. 1

COD (AXIAL FLAWS) AS A FUNCTION OF PRESSURE

Fig. 6

AXIAL, PARTIAL PENETRATION FLAWS COD AS A FUNCTION OF PRESSURE
INCONEL Ø 7/8"
Fig. 2

TRough Thickness Axial Flaws
Flaw (Central) Opening as a Function of Pressure

Fig. 3

TRough Thickness Axial Flaws
Bulging as a Function of Pressure
Fig. 4

BULGING FACTOR

\[ \sqrt{1 + 0.5 \lambda} \]
\[ \sqrt{1 + 0.5 \lambda} \]
\[ Q_{1/4} = 0.365 F \lambda, 0.081 \lambda \]

+ NUMERICAL VALUES (KRENK)

Fig. 5

BURSTING PRESSURE

THEORETICAL CURVE

\[ P = \frac{1}{8} \frac{\pi}{m} \]

WITH \( \sigma_c = 605 \text{ MPa} \)

Experimental points

Inconel \( \phi 7/8'' \)
Table 1 - CORRELATION BETWEEN FLOW STRESS $\tilde{\sigma}$ AND TRANSVERSE PROPERTIES YS AND UTS
(Trough-thickness axial flaws)

<table>
<thead>
<tr>
<th>Material</th>
<th>YS</th>
<th>UTS</th>
<th>R</th>
<th>t</th>
<th>2c $\sqrt{Rt}$</th>
<th>m</th>
<th>$P_{cr}$</th>
<th>$\sigma_{cr}$</th>
<th>$\tilde{\sigma}$</th>
<th>UTS-$\tilde{\sigma}$</th>
<th>$\tilde{\sigma}$-YS</th>
<th>$\tilde{\sigma}$+YS</th>
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<tbody>
<tr>
<td>Inconel 600(Ø7/8&quot;)</td>
<td>385</td>
<td>700</td>
<td>10.5</td>
<td>1.27</td>
<td>VARIOUS</td>
<td>-</td>
<td>-</td>
<td>95</td>
<td>.86</td>
<td>.70</td>
<td>.56</td>
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<td>Inconel 600(Ø7/8&quot;)</td>
<td>335</td>
<td>610</td>
<td>10.4</td>
<td>1.35</td>
<td>20</td>
<td>2.67</td>
<td>2.94</td>
<td>23.5</td>
<td>181</td>
<td>530</td>
<td>.87</td>
<td>.71</td>
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<td>675</td>
<td>8.95</td>
<td>1.15</td>
<td>16.5</td>
<td>2.57</td>
<td>2.89</td>
<td>26.8</td>
<td>209</td>
<td>605</td>
<td>.90</td>
<td>.76</td>
</tr>
<tr>
<td>AISI 304 (Ø 3/4&quot;)</td>
<td>335</td>
<td>635</td>
<td>8.95</td>
<td>1.20</td>
<td>16.5</td>
<td>2.52</td>
<td>2.87</td>
<td>27.7</td>
<td>207</td>
<td>595</td>
<td>.94</td>
<td>.87</td>
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<tr>
<td>A1-6X-HT (Ø 20)</td>
<td>605</td>
<td>760</td>
<td>9.7</td>
<td>0.77</td>
<td>14.5</td>
<td>2.66</td>
<td>2.93</td>
<td>21.3</td>
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<td>SEA CURE (Ø 20)</td>
<td>590</td>
<td>685</td>
<td>9.6</td>
<td>0.71</td>
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<td>2.77</td>
<td>2.98</td>
<td>18.0</td>
<td>245</td>
<td>730</td>
<td>-45</td>
<td>1.07</td>
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Table 2

CRITICAL AND ALLOWABLE LENGHTS FOR TROUGH-TICKNESS FLAWS

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<tr>
<th>Flaw direction</th>
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<th>Circumferential</th>
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<td>Tube diameter</td>
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<td>Allowable size</td>
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<td>7 mm</td>
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</table>
SESSION 4.7
CHARACTERISTICS OF ROLL TRANSITION CRACKS IN STEAM GENERATOR TUBES.

Ph. BERGE, F. CATTANT, P. CALLE, J.P. HUTIN.

The operating experience of the steam generators of Electricité de France's P.W.R. has been described in another presentation in this meeting (1). The presence of numerous cracks in the tubing due to Stress Corrosion by the primary coolant has been indicated. Due to the high stress level and the higher temperature, the most susceptible area for this phenomenon is the upper end of the portion of the tubes mechanically expanded in the tube sheet. The morphology of the cracks and their possible consequences for the safety aspect have received a special attention from E.D.F. This presentation describes the characteristics of the cracks and the tests performed to evaluate the validity of the "leak-before-break" aspect for this type of defects.

1) Tubes extracted from Steam Generators

Despite the extremely low primary-secondary leaks induced by this type of cracks, portion of tubes, selected after Eddy Currents examination, were extracted to be examined in E.D.F.'s hot cells in Chinon. Since 1981, 17 tubes (22.2 mm diameter, 1.27 mm thick) have been extracted from 5 steam generators of different plants.

Radiographic examination

An X-Ray generator (under 100 KV - 5 mA) has been used to radiograph the tubes and detect the location, length and orientation of the cracks. The films are placed inside the tube. 36 shots corresponding to 36 tube revolutions of $10^\circ$ each, give a good image of the cracking. Fig. 6, 7, 8 represent a summary of the angular position and length of the cracks.
Metallographic examination

Characterisation of the alloy, research of traces of impurities on the tubes or inside the cracks are made by the different physical and chemical analyses.

The Nickel Alloy (Alloy 600 for ASTM) has always been found within the specification ranges for composition and hardness. The structure shown on Fig. 2 is typical for this alloy, when the tubing has received a mild annealing at relatively low temperature (∼ 900° C). The grain size is small (ASTM 8-9). This structure is known now as making this alloy very susceptible to pure water stress corrosion-cracking as described by Coriou in 1959 (1).

The cracks are initiated from inside. They are intergranular as shown on Fig. 2. Minor general intergranular penetration, less than one grain deep, has been noticed in some cases. The presence of small amount of impurities (such as sulfur, phophorus, zinc), have been sometimes identified, but not always. A possible contamination might have accelerated the phenomenon, which can nevertheless occur in pure water on clean surfaces.

Except in one plant, where mechanical rolling has not been applied to the top (in this case rolling on the lower part of the tube sheet has been followed by expansion by explosion) and for a few tubes, where the procedure has not been applied correctly, the cracks are purely longitudinal as shown on Fig. 1. The cracks are numerous, up to 12 on the circumference, and limited to the transition of the expansion. This location and the orientation of the cracks can be understood from previous work on residual stresses evaluation in steam generator tubing (2).

Several of these short cracks are through-wall despite the absence of measurable primary-to-secondary leakage. The length of the cracks found on internal diameter, where initiated, is usually 3-8 mm (11 mm on the I.D. in one case). The length of the cracks have been detected with a very good accuracy by the rotating Eddy Current Probe (3) and the correlation between the X-Ray detections on the extracted tube and the micrographic examination is excellent.
2) Laboratory reproduction of longitudinal stress corrosion cracks

The susceptibility of Alloy 600 and austenitic stainless steels to intergranular stress corrosion, at 20-40° C in 0.1 M Sodium tetraphionate solution \( (4) \) has been used to reproduce cracks in the laboratory. In order to be susceptible to S.C.C. in this solution, the alloy with a carbon content of 0.041 % is sensitized by a solution annealing temperature \( (4 \text{ minutes at } 1075° \text{ C}) \) followed by a chromium carbide precipitation treatment \( (1 \text{ hr at } 700° \text{ C}) \).

Mock-up with tubes mechanically expanded in a tube sheet can reproduce the residual stresses giving cracks similar to the ones described above. In order to measure leak rates versus the length of longitudinal cracks, we introduced cracks with a device shown on Fig. 3. The device represented on the upper part of the figure makes it possible to obtain single longitudinal cracks after protection of the half surface by a protective painting. On the lower part of Fig. 3, the device set up to obtain several cracks \( (6 \text{ simultaneously or more after a second run}) \) is represented. Fig. 4 shows a section with 8 longitudinal intergranular cracks.

3) Leak rate measurement

The criterion of "leak before break" needs a good evaluation of leak rate for S.C.C. longitudinal cracks of different lengths. Tests have been performed under 10 MPa internal pressure in R.T. water, and in 120° C water in order to obtain boiling on the outside of the tubing. It is recognized that the hydraulic condition inside the crack might be different from the actual conditions of primary-secondary leak, but comparison with data points obtained elsewhere \( (5) \) in an autoclave at 288° C under the same \( \Delta P \) shows an acceptable correlation. Above 12 mm on the outside, the leak becomes higher than 100 l/hr.
4) Burst tests

Experimental procedures

In order to obtain the pressure by hydraulic pressurisation, a 0,1 mm thick stainless steel sleeve is introduced inside the tube which is filled with a visqueous paste to prevent leaking during the test when raising the pressure. For single cracks, 19 MPa pressure (corresponding pressure in the tube without the sleeve of 17,2 MPa) is applied instantaneously.

For multiple cracks (8 cracks), after X-Ray determination of the length and exact position, the pressure is raised to reach the burst of the stainless steel sleeve and the opening of the cracks.

Fig. 6 shows a typical tube with 8 longitudinal cracks of length between 5,5 and 10 mm. After burst of the sleeve (for a pressure of 30 MPa) it is noticed that, if the cracking has circumferential branches, the burst pressure is lowered (Fig. 7). Burst tests have been performed on two tubes extracted from an operating steam generator. The shape of the cracks obtained by X-Ray is given on Fig. 8.

Results and discussion

Numerous burst tests have been previously performed on tubes with machined cracks. All these tests led to the conclusion that the ruin mechanism was not unstable crack growth, but plastic instability. This is the reason why a criterion based on the flow stress concept was found more appropriate. For longitudinal through-wall cracks, this criterion is :

\[ \frac{M \cdot \sigma_e}{\sigma_f} = \frac{\sigma_{eq}}{\sigma_f} \]

: Hoop stress

\( M \) : Bulging corrective factor

\( \sigma_{eq} \) : Flow stress of the material.

The M factor is given by :

\[ M = (1 + 1,61 \frac{c^2}{Rt})^{\frac{1}{3}} \]

\( c \) : Half crack length

\( R \) : Tube radius

\( t \) : Tube thickness
The flow stress value was reached by two separate approaches: one based on the theoretical Walmsley theory, the other derived from tests on uncracked tubes. In both cases, the flow-stress was estimated to be around 650 MPa (for an ultimate strength of about 750 MPa). In the case of tubes with stress corrosion cracks obtained in laboratory, the flow stress value must be corrected to take into account the sensitization heat treatment which decreases the yield strength by about 30% and the ultimate strength by 10%. This leads to a "sensitized" flow stress of 550 MPa.

Another difficulty arises from the fact that, when validating the failure criterion on machined cracks, the length of the cracks is well defined. In stress corrosion cracks, the internal length is different from the external one. It was felt that considering the average length was more appropriate.

**Tests on laboratory S.C. cracks**

The pressure was instantaneously increased to 17.2 MPa. Five tubes having cracks of respective average length: 10.35 mm (tube A), 20.25 mm - 28.25 mm (O.D. initiated) did not fail. According to the above mentioned criterion, the critical crack size was 22.5 mm (see Fig. 9).

**Tests on tubes extracted from the operating steam generator**

In this case, the pressure was increased progressively. One of the tubes failed at 60 MPa with a crack of 7 mm on the inside and 4 mm on the outside. For this pressure, the predicted critical size is 5.3 mm. The second tube failed at 47 MPa. The crack length was 11 mm inside and 5 mm outside. The predicted critical size is 8.2 mm (see Fig. 8).

It is worth noticing that the criterion predictions are applicable in the case of multiple cracking and after several years operation.
5) Conclusion

The presence of numerous cracks have been observed in the steam generator tube, in the area of the top of the tube sheet. The cracks are due to stress corrosion of the Alloy 600 by the primary coolant under high stresses taking into account the residual stresses related to the expansion of the tube in the tube sheet. In the normal manufacturing procedure, the cracks are longitudinal, limited to a few millimeters and tight enough to induce no significant primary-secondary leak.

Burst test applied to two tubes extracted from a steam generator presenting multiple cracks have shown that the criterion used to predict the flow stress is applicable even if 11 parallel cracks were formed on the same circumference.

Due to the evidence of possible evolution of the cracks in the future, either in length, where the DAM (Dudgeonnage Amélioré Mécaniquement) (2-6) has been applied, or circumferentially if not. Studies are undertaken in order to develop modifications of the tensile stresses in this area of the tubes in steam generators in operation.
REFERENCES


LONGITUDINAL CRACKS AT ROLL TRANSITION

Cracks

Tubesheet

Roll transition zone

Fig 1
EXPERIMENTAL DEVICES TO FORM CORROSION CRACKS IN SODIUM TETRATHIONATE

a) Single crack

\[ F = 600 \text{ daN} \times \text{nb of tubes} \]

b) Multiple cracks

![Diagram showing multiple cracks and the solution of tetrathionate](image)

Fig. 3
Multiple Corrosion Cracks

*4.8

*100

Fig 4
LEAK RATE UNDER 10 MPa = \Delta P

LEAK RATE (Liters/H)

ROOM TEMPERATURE TESTS

× 1st test
● 2nd test
○ Cracks O D initiated
× Test at 12 MPa

TESTS AT 120°C

+ 1st test
○ 2nd test

○ Test at 288°C (Ref 5)

CRACK LENGTH (mm)

Fig 5
Tube R7_163

L: ID, crack length in mm

Burst Pressure 30 MPa

Fig 6
TUBES EXTRACTED FROM OPERATING STEAM GENERATOR

L: I.D. CRACK LENGTH IN mm
BURST PRESSURE 68 MPa

L: II. CRACK LENGTH IN mm
BURST PRESSURE 47 MPa
\[
\frac{\sigma_\theta}{\sigma_t}
\]

- **Bugey Tube**
- **Failure**
- **Laboratory S.C.C. Tubes**
- **No Failure**
- \[ \sqrt{1 + 1.61 \frac{C^2}{Rt}} \]

*Fig. 9*
SESSION 5

PREVENTIVE AND

CORRECTIVE ACTIONS
SUMMARY OF SESSION 5 - PREVENTIVE AND CORRECTIVE ACTIONS

Session Chairman: Mr. G. Frédéric

The current corrective actions for operating SG such as plugging, sleeving, rerolling are discussed in two papers (5.6 and 5.9) only but a separate session is entirely devoted to these techniques.

Another paper (5.7) presents generic preventive actions for mitigating MA Inconel 600 susceptibility to pure water SCC (tube sheet global heat treatment and peening techniques).

The other papers describe mainly the actions which are taken at the early stage of SG and/or plant design to cope with the different problems found on SG.

These actions result from operating experience at the first PWR units and also from R & D programs.

Early tube material choices are justified and new materials providing additional resistance to the different mode of corrosion (5.1, 5.2, 5.3, 5.5) are presented.

Design modifications are introduced to better control the flow velocities, to improve the blowdown system, to solve the problems associated with the tube/tubesheet joints and with the interface between tube and tube support devices (5.3, 5.5).

The importance of good secondary chemistry is enhanced and the improvements of secondary equipment (leak tight condenser, heaters, condensate polisher, deaerator) are discussed (5.4, 5.5, 5.8).
SESSION 5.1: PREVENTIVE AND CORRECTIVE ACTIONS
THE PRIMARY SIDE STRESS CORROSION CRACKING
PERFORMANCE OF MILL ANNEALED INCONEL - 600
TUBING IN C-E DESIGNED STEAM GENERATORS

C. M. Owens
Combustion Engineering, Inc.

1. INTRODUCTION

Combustion Engineering (C-E) designed steam generators have not experienced primary side stress corrosion cracking (the "Coriou" Syndrome) of its steam generator tubing. This outstanding performance is attributed to the specification of low yield strength Alloy-600 tubing. Low yield strength product requires a high temperature final anneal in tube manufacturing that produces a medium to coarse grain size and results in a distribution of carbon/carbides distinctly different from that associated with "Coriou Prone" tubing.

A highly "Coriou" susceptible product, Process Stabilized Inconel (PSI), has been developed in the laboratory at C-E. The strength and microstructure of this tubing appears to be very similar to that characterized in tubing that has cracked from the primary side in field service in a number of non C-E designed units around the world.

Additional laboratory experiments at C-E have documented the key role that the final anneal has, both on heat-up rate and ultimate temperature, in determining the microstructure and ultimately the "Coriou" performance of the finished tube.
2. STEAM GENERATOR TUBING FIELD SERVICE EXPERIENCE

2.1 Non C-E designed steam generators

Primary side stress corrosion cracking (SCC) has been identified in a number of non C-E plants around the world. Some of these units have experienced cracked tubing at support plates and tight radius U-bends as a result of denting. However, there have also been plants that have developed primary side SCC in the absence of denting. These latter plants have experienced tube cracking in tight radius U-bends (first two rows) in addition to the more serious problem of roll transition cracking in the tube to tube sheet joint.

2.2 Combustion Engineering designed steam generators

There are 17 C-E designed steam generators with commercial operating experience beyond five years as part of C-E Nuclear Steam Supply Systems. A tabulation of these plants, including steaming times, is shown in Table 1 where a steaming day is defined as operation for at least 12 hours on a given day. Three of the listed plants, Palisades, Fort Calhoun and Maine Yankee have been operating for more than ten years. None of the plants listed in the table have ever experienced primary side initiated SCC. It is important to note that tubes have been essentially crushed in Millstone II as the result of support plate induced denting but tube cracking did not occur.
Table 1
Steaming Days for C-E Designed Steam Generators
with Five or More Years of Commercial Operation

<table>
<thead>
<tr>
<th>Plant</th>
<th>Commercial Operation</th>
<th>Number of Generators</th>
<th>Tubes Per Generator</th>
<th>Total Steaming Days*</th>
</tr>
</thead>
<tbody>
<tr>
<td>Palisades</td>
<td>1/72</td>
<td>2</td>
<td>8,519</td>
<td>2,513</td>
</tr>
<tr>
<td>Fort Calhoun</td>
<td>8/73</td>
<td>2</td>
<td>5,005</td>
<td>3,013</td>
</tr>
<tr>
<td>Maine Yankee</td>
<td>11/72</td>
<td>3</td>
<td>5,703</td>
<td>3,383</td>
</tr>
<tr>
<td>Calvert Cliffs I</td>
<td>1/75</td>
<td>2</td>
<td>8,519</td>
<td>2,692</td>
</tr>
<tr>
<td>Calvert Cliffs II</td>
<td>12/76</td>
<td>2</td>
<td>8,519</td>
<td>2,311</td>
</tr>
<tr>
<td>Millstone II</td>
<td>10/75</td>
<td>2</td>
<td>8,519</td>
<td>2,227</td>
</tr>
<tr>
<td>St. Lucie I</td>
<td>4/76</td>
<td>2</td>
<td>8,485</td>
<td>1,937</td>
</tr>
<tr>
<td>Arkansas</td>
<td>6/79</td>
<td>2</td>
<td>8,511</td>
<td>1,203</td>
</tr>
</tbody>
</table>

*As of 6/4/84

3. METALLURGICAL VARIABLES IN PRIMARY SIDE (SCC)

3.1 Highly susceptible tubing

There are two common denominators that are characteristic of "Coriou" prone tubing. Cracked tubing removed from plants such as Obrigheim\textsuperscript{1}, Doel II, Ringhals II\textsuperscript{2} and Trojan have exhibited a characteristic microstructure and yield strength. The high yield strength (~60KSI, 413MPa) is set by the fine grain size (ASTM - 9 to 11). The fine grained microstructure is the result of a low temperature (~1700°F, 926°C) final anneal which also dictates a very specific carbon inventory. "Coriou" prone microstructures typically exhibit a preponderance of intragranular carbides, few if any intergranular carbides and little or no solid solution carbon.
The absence of grain boundary carbides has been shown to be undesirable for resistance to "Coriou" type SCC. This undesirable microstructure, combined with the high yield strength where elastic stresses can build and persist, apparently above the threshold required for crack initiation, constitutes the ultimate metallurgical condition for "Coriou" susceptibility. This condition develops as a direct result of employing low temperature at final anneal.

3.2 Highly resistant tubing

C-E tubing is purchased to a specification which aims at obtaining a yield strength of 45KSI (310MPa) and is restricted to a maximum of 55KSI (379MPa). This control precludes the use of low temperature annealing at final tube size and avoids the high yield strength and carbide free boundaries that contribute to "Coriou" susceptibility. Tubing annealed at higher temperatures (~1900°F, 1037°C) will have a yield strength less than 55KSI (379MPa) and will exhibit a metallurgical condition where both inter- and intragranular carbides are present as well as a significant amount of solid solution carbon. In fact, this product exhibits the microstructural feature, grain boundary carbides, that constitutes the principal objective of the Thermal Treatment Process (1300°F, (704°C); 15 hours). Based on field experience, low yield strength combined with grain boundary carbides has proven to be a metallurgical condition highly resistant to primary side SCC in spite of the fact that the tubing is not stress relieved. This material routinely contains 10-15KSI (69-103MPa) tensile residual hoop stress and has experienced damage from support plate denting. Stress relieving this product would not be recommended because of the available solid solution carbon and the resulting secondary side concerns over a sensitized microstructure. What is more important is that field service experience indicates that stress relieving is unnecessary.
4. STEAM GENERATOR FABRICATION VARIABLES IN PRIMARY SIDE SCC

4.1 Non C-E designed steam generators

Field failures of steam generator tubing from the primary side have been associated with large plastic strain gradients in susceptible product. Non-uniform plastic strain, by definition, generates elastic residual stress, which has contributed to failures in non-dented steam generators at transition locations in tight radius U-bends (Trojan) and partial rolled tube to tube sheet joints (Doel II and Ringhals II). Cracking has occurred in the transition zone (between the plastic strain and the as-manufactured material condition) which is the location of maximum residual stress. The elastic residual stress can build (and persist) to a magnitude equal to the proportional limit which is 60KSI (413MPa) for the material in question.

Fabrication transitions created in the mini sleeve repair work at Doel II have also experienced primary side SCC. Cracking occurred in about 14 months, which is an indication of this material's sensitivity.

4.2 Combustion Engineering designed steam generators

The success C-E has experienced in field service with its tight radius U-bends representing the innermost tube rows is attributable to a number of factors. Low yield strength, larger minimum bend radius, lower OD to wall ratio and attention to ovality and bend profile all contribute to minimizing the formation of abrupt contour changes (transitions) that would contain high residual stress. In addition, the full depth tube to tube sheet expansion joint is controlled by an explosive charge. Axial location of the charge and charge length are controlled to yield little or no bulging at the secondary face of the tube sheet.
The design and fabrication techniques used at C-E in U-tube preparation and installation are geared to minimizing deformation and abrupt contour changes which ultimately keeps the residual stresses to a minimum. Primary side SCC in tight radius U-bends and tube to tube sheet joints has not been a problem in C-E units.

5. LABORATORY DEVELOPMENT PROGRAMS

5.1 Process stabilized Inconel (PSI) tubing

Approximately seven years ago, as the result of increasing concerns over residual stress in steam generator tubing U-bends, an objective was established to develop a tubing process to accommodate a stress relief heat treatment. The goal was to manufacture tubing and eliminate the residual stresses without introducing sensitization such that secondary side performance would not be compromised.

The problem was to stabilize the carbon in the material in some way to avoid grain boundary precipitation, with its attendant chromium depletion, during the stress relief heat treatment. It seemed reasonable to try a processing sequence in the tube mill that would intentionally precipitate all available solid solution carbon at some early upstream heat treatment. The process would work if all subsequent heat treatments were performed at a low temperature to avoid dissolving the previously formed carbides. In addition, because of the recrystallization reaction subsequent to the cold working to final size, all the carbides could be located at intragranular sites. Hence, grain boundaries would be carbide free and heat treatment in the sensitization range for stress relief would not produce new boundary particles.
The process was initially demonstrated on a low (0.02) and a high (0.04) carbon heat. Heavily cold worked "Final Intermediate" (1 1/4" x 0.100", 31.8mm x 2.5mm) product was diverted from the standard tubing process to vacuum anneal. This material was heat treated for two hours at 1300°F (704°C) (Precipitation) followed by two hours at 1450°F (788°C) (chromium healing).

This material, identified as Process Stabilized Inconel (PSI), was subsequently cold worked to final size (3/4" x 0.042", 19mm x 1mm) and bright hydrogen annealed in a belt furnace at 1485°F (807°C). Final anneal was designed to accomplish recrystallization, and obtain specification properties, without disturbing the previously precipitated carbides.

Metallographic examination of the resulting microstructures along with modified Huey test results, as manufactured and after four hours at 1200°F (649°C), indicated that the objectives of the experiment had been achieved. The tubing met all the requirements of ASME Code Specification SB-163 and it could be stress relieved without fear of sensitization. Process stabilization had been demonstrated in a tube mill production environment.

Long term "Coriou" corrosion susceptibility tests were initiated on triplicate reverse U-bend specimens of the 0.02 carbon material (Heat NX-0471) in an autoclave at 690°F (365°C). Samples of the PSI condition and the standard C-E mill annealed condition from the same heat were on test for 6,500 hours when the first crack was detected. One of the PSI specimens had cracked in an intergranular manner. This represented the very first "Coriou" type crack ever produced at C-E. The remaining two specimens of PSI material were found to be cracked as well during the subsequent 7,500 hour inspection. The mill annealed specimens did not crack and the test was terminated at 10,000 hours.

Although the program objectives were met in the development of PSI tubing, it was obvious that we had produced a metallurgical condition that was highly susceptible to "Coriou" type IGSCC. It is worth noting that heat NX-0471 is not a susceptible heat of material but susceptibility to the "Coriou" syndrome is a function of the processing employed.
It became apparent with time that the PSI metallurgical condition was very similar to that observed in primary side field failures. The PSI yield strength in the annealed condition was 60-68KSI (413 to 468MPa) and the grain size was typically at ASTM 10-11. The carbon inventory, as planned, showed little if any solid solution carbon and the complete absence of grain boundary carbides. As evidenced in Figure 1, all the carbon was in intragranular carbide form. The yield strength and carbon inventory characteristic of PSI tubing is very similar to that which is failing in field service in non C-E units around the world via "Coriou" IGSCC.

Subsequent to the development phase of the PSI effort, a report\(^3\) was reviewed that described a very similar processing technique developed by the Mannesmann Company in West Germany during 1971. Their effort centered on Alloy-800 and their observations and conclusions agreed with what was learned in the development of the PSI tubing process at C-E.

### 5.2 Importance of final anneal in primary side scc

#### 5.2.1 Annealing temperature

The ultimate temperature at final anneal controls the grain size (strength) and significantly influences carbon inventory in Alloy-600 steam generator tubing. "Coriou" susceptible tubing is probably final annealed in the range of 1500 to 1700°F (815 to 926°C). A fine grained microstructure results, and almost all the carbon is present as intragranular carbides.
500X  10% Oxalic Acid  39517
PSI Tubing
As Final Annealed - Longitudinal Section
Showing Fine Grained Microstructure and
Intragranular Carbides

20,000X  6307
PSI Tubing
After Stability Test - Heat Treated at 1200°F (649°C)
for 4 Hours - Note Absence of Grain Boundary Carbides

Figure 1. Stabilized Tubing Microstructures
"Coriou" resistant tubing is annealed in the temperature range of 1850 to 1950°F (1010 to 1060°C). A medium to coarse grain size results with a corresponding yield strength of from 35 to 55KSI (241 to 379MPa). Carbon is distributed as solid solution carbon and both inter- and intragranular carbides.

5.2.2 Heat-up rate. In the development of PSI tubing at C-E, it was observed that carbide precipitation occurs during heat-up of cold worked material containing solid solution carbon. This precipitation occurs before recrystallization and the carbide habit is the existing cold worked grain boundaries as well as the slip planes. These observations can be seen in Figure 2 where the dual etch microstructure of the vacuum annealed condition is depicted. This sequence of events is peculiar to a low heat-up rate, viz., about 25°F (14°C)/minute.

Precipitation during heat-up and prior to recrystallization was unanticipated and somewhat surprising. However, this information is very useful in understanding the tubing manufacturing process. A process, for example, that employs low temperature annealing (for carbon stabilization), can take advantage of this phenomenon. The tube reduced extrusion (TREX) and initial cold worked intermediates are massive and will heat-up slowly in the annealing furnace. Precipitation followed by recrystallization will automatically take place and carbon stabilization becomes simple. If carbide dissolution temperatures are avoided, Process Stabilized Inconel (PSI) tubing will result and the long vacuum anneal employed by C-E in earlier PSI development is obviously unnecessary.
Figure 2: Dual etch metallography of vacuum annealed condition.
(longitudinal view, 500X)
6. SUMMARY AND CONCLUSIONS

Primary side IGSCC field failures of Alloy-600 steam generator tubing have occurred in non C-E designed units in susceptible material adjacent to fabrication and operationally induced plastic deformation. This susceptible tubing is characterized by high yield strength and the absence of grain boundary carbides. Tubing characterized by low strength and the presence of grain boundary carbides, as employed in C-E designed units, has performed well without an identified failure to date from the primary side.

Laboratory studies have duplicated the "Coriou" prone metallurgical condition that is failing in the field, and information was obtained indicating the ease in which this undesirable condition can be achieved. Low temperature final annealing promotes high strength and intragranular carbides, the principal characteristics of a "Coriou" prone product.

REFERENCES


2. J. Engstrom and K. Norrig, Primary and Secondary Cracking at Ringhals 2, EPRI Workshop on Primary and Secondary Side SCC of PWR Steam Generator Tubing, Clearwater Beach, Florida (November 1983)

3. F. Schnabel, Incoloy-800 with Controlled Carbide Precipitation, Mannesmann Colloquium on Reactor Pipes, Duseldorf, West Germany (November 1971)
SESSION 5.2

PERFORMANCE OF THERMALLY TREATED
INCONEL 600 AND INCONEL 690
STEAM GENERATOR TUBING ALLOYS

NEA/CSNI - UNIPEDE SPECIALIST MEETING
ON STEAM GENERATOR PROBLEMS

STOCKHOLM, SWEDEN

OCTOBER 1-5, 1984

Westinghouse Electric Corporation
Steam Generator Technology
Division
Pittsburgh, Pa 15230
SESSION 5.2: PERFORMANCE OF THERMALLY TREATED INCONEL 600 and INCONEL 690 STEAM GENERATOR TUBING ALLOYS

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Westinghouse Electric Corporation
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ABSTRACT

The corrosion and intergranular stress corrosion cracking (SCC) performance of thermally treated Inconel 600 and 690 (I-600 and I-690) were evaluated in both secondary side and primary side environments, and results compared with those from mill annealed I-600. Results are presented for exposures to deaerated sodium hydroxide, sodium hydroxide plus oxidizing species, sodium sulfate and primary water environments. Primary water metal release data for both materials are presented. Results indicate that both thermally treated I-600 and I-690 provide enhanced SCC resistance under both secondary and primary side environments when compared to mill annealed I-600 and that I-690 offers additional benefit in relevant environments.
INTRODUCTION

Inconel Alloy 600 and Inconel Alloy 690 (I-600 and I-690) are austenitic nickel-base alloys. I-600 has been extensively used, originally in the mill annealed condition, as steam generator tubing in pressurized water reactors.

In recent years, attempts to enhance the IGSCC resistance of I-600 have focused on the application of a thermal treatment in the carbide precipitation temperature range (593°C to 760°C). Microstructural modifications have concentrated on the grain boundary region since the SCC morphology in I-600 is predominantly intergranular. The maximum enhancement in caustic and primary water SCC performance was correlated with the presence of a semicontinuous grain-boundary carbide precipitate. There have been a number of reports on the added stress corrosion cracking resistance (SCC) of thermally treated I-600 of which three are referenced.¹,²,³

Currently, I-690, which contains a higher chromium content (30 percent) than I-600, is being qualified for application in advanced model and replacement steam generators.⁴ An extensive data base has been generated by Westinghouse to characterize the mechanical, physical, and, most specifically, the corrosion resistance properties of I-690.

EXPERIMENTAL PROCEDURES

Up to eight heats of I-600 and five heats of I-690 tubing were employed in the various corrosion tests in both the mill annealed and thermally treated conditions. All material was processed under production conditions using current commercial practices. The chemical composition and mechanical properties met the requirements of ASME SB163 and Code Case N-20. In the mill annealed condition, straightening and polishing were the final processing steps, while in the thermally treated condition the heat treatment (nominally 704°C for 15 hours) followed the straightening and polishing operations.
C-ring specimens were used for the SCC evaluation under secondary side contaminants, while reverse U-bends (RUBs) were employed for the primary water SCC evaluation. The two stress levels for the C-ring specimens were expressed as 150 percent of the room temperature yield strength and TLT (till the legs touched). These two deflections corresponded to 0.3 and =14 percent strain, respectively.

The primary water corrosion release rates were studied in simulated flowing reactor coolant waters. The test specimens were tubing sections mechanically joined together to form a loop. Weight change before and after stripping was used to measure the amount of ID corrosion. The test conditions for measurement of the metal release rates were as follows: 327°C temperature, 15.5 MPa differential pressure, 9.3 x 10^{-3} m/s flow rate, 1000-2000 ppm B as H₂BO₃, 1 ppm Li as LiOH, <0.15 ppm Cl, <0.15 ppm F, <0.1 ppm O₂ and no hydrogen overpressure.

Mill annealed and thermally treated I-600 and thermally treated I-690 reverse U-bends (RUBs) have been exposed to primary water environments at 360°C for 13000 hours. Beginning and end of fuel cycle water chemistries (i.e. 1200 ppm B as H₂BO₃ and 2.0 ppm Li as LiOH and 200 ppm B as H₃BO₃ and 0.5 ppm Li as LiOH respectively) were evaluated. Dissolved H₂ gas was included in both solutions at equilibrium concentrations of 3.8 ppm (240 KPa overpressure).

RESULTS

The caustic SCC performances of mill annealed and thermally-treated I-600 and thermally treated I-690 were evaluated in a 10 percent NaOH solution as a function of temperature from 288°C to 343°C. Since the test data were obtained over various exposure intervals ranging from 2000 to 8000 hours, the test data were normalized in terms of average crack growth rate determined from destructive examination of the C-ring test specimens. No attempt was made to distinguish between initiation and propagation rates.

As shown in Figure 1, the performance of the two alloys in the thermally treated condition is approximately equal at temperatures of 316°C and
FIGURE 1  SCC GROWTH RATE FOR C-RINGS (150 PERCENT YS AND TLT) OF I-600 AND I-690 IN 10 PERCENT NaOH. FILLED SYMBOLS PRESENT AVERAGE VALUES AND OPEN SYMBOLS REPRESENT THE EXTREMES.
below. At 332°C and 343°C, the additional SCC resistance of thermally-treated Inconel 690 is observed. In all instances the SCC morphology was intergranular in nature.

Testing in 10 percent NaOH solution at 332°C was performed to index the relative intergranular attack (IGA) resistance of I-600 and I-690. Comparison of the IGA morphology for I-600 and I-690 rings stressed to 150 percent of the 0.2 percent yield strength is presented in Figure 2. Mill annealed I-600 is characterized by branching intergranular SCC extending from a 200μ front of uniform IGA. Thermally treated I-600 exhibited less SCC and an IGA front limited to less than a few grains deep. Thermally treated I-690 exhibited no SCC and only occasional areas of intergranular oxide penetrations, limited to less than a grain deep.

![Inconel 600 MA](image1)

![Inconel 600 TT](image2)

![Inconel 690TT](image3)

**FIGURE 2** IGA MORPHOLOGY FOR INCONEL 600 AND 690 C-RINGS (150 PERCENT OF YIELD STRENGTH) AFTER 5000 HOURS EXPOSURE TO 10 PERCENT NaOH AT 332°C (200X)

Results of various autoclave tests to determine the effect of oxidizing species on the caustic SCC performance of thermally treated I-600 and I-690 1443c/0193c/080884:5 5
are summarized in Table 1. The addition of 10 percent copper oxide to the 10 percent sodium hydroxide solution has a deleterious effect on the SCC resistance of both thermally treated I-600 and I-690. Both intergranular and transgranular SCC was observed.

Table 1
EFFECT OF OXIDIZING SPECIES ON THE SCC SUSCEPTIBILITY
OF THERMALLY TREATED I-600 and I-690 C-RINGS IN DEAERATED CAUSTIC

<table>
<thead>
<tr>
<th>Environment</th>
<th>Temperature (°C)</th>
<th>Exposure Time (Hrs)</th>
<th>I-600 TT</th>
<th>I-690 TT</th>
</tr>
</thead>
<tbody>
<tr>
<td>10 Percent NaOH + 10 Percent CuO</td>
<td>316</td>
<td>4000</td>
<td>Increased Susceptibility*</td>
<td>Increased Susceptibility*</td>
</tr>
<tr>
<td>10 Percent NaOH + 1 Percent CuO</td>
<td>332</td>
<td>2000</td>
<td>No effect</td>
<td>No effect</td>
</tr>
<tr>
<td>1 Percent NaOH + 10 Percent CuO</td>
<td>332</td>
<td>4000</td>
<td>No effect</td>
<td>No effect</td>
</tr>
<tr>
<td>10 Percent NaOH + 10 Percent Fe₂O₄</td>
<td>316</td>
<td>4000</td>
<td>No effect</td>
<td>No effect</td>
</tr>
<tr>
<td>10 Percent NaOH + 10 Percent SiO₂</td>
<td>316</td>
<td>4000</td>
<td>No effect</td>
<td>No effect</td>
</tr>
</tbody>
</table>

*Intergranular and transgranular SCC.

Mill annealed and thermally-treated I-600 and I-690 were also evaluated in a number of 8 percent sodium sulfate environments. The room temperature pH value, at the beginning of the test, was adjusted using sulfuric acid and ammonia. Test results are presented in Figure 3. As the pH is lowered, decreased SCC resistance for mill annealed and thermally-treated I-600 is observed, but thermally treated I-690 material did not crack even at a pH of 2, the lowest tested.

The primary water SCC test data are presented in Figure 4. For the beginning of fuel cycle water chemistries, 10 of 10 specimens of mill annealed I-600 exhibited SCC, while 1 of 10 specimens of thermally-treated I-600 had cracked. In the end of the fuel cycle water chemistries, 7 of 10 specimens of mill annealed I-600 exhibited SCC, while 3 of 10 specimens of
**FIGURE 3** MAXIMUM SCC DEPTH FOR TLT C-RINGS EXPOSED TO SULFATE at 332°C

**FIGURE 4** FRACTION OF REVERSE U-BEND SPECIMENS OF I-600 and I-690 THAT EXHIBITED SCC DURING EXPOSURE TO PRIMARY WATER AT 360°C
thermally-treated I-600 had cracked. After 13,000 hours of testing, no SCC has been observed in the mill annealed or thermally-treated I-690 specimens in either test environment.

Metal release measurements for I-600 and I-690 in simulated reactor coolant are compared in Table 2. The metal release exhibited after 8.3 months for I-690 TT was appreciably less than that of I-600 TT. Those trends are generally consistent with the measurements of the relative metal release data for I-600 and I-690 reported elsewhere. From measurements of the metal release rates in primary water, the system activity was estimated using a semi-empirical Westinghouse computer code. The activity build up within the primary circuit of a Westinghouse PWR utilizing I-690 tubing was estimated to be 50-60 percent of that expected for an I-600 plant.

<table>
<thead>
<tr>
<th>Materials</th>
<th>Metal Release (mg/dm²)</th>
<th>Film Growth (mg/dm²)</th>
<th>Total Corrosion (mg/dm²)</th>
<th>Time (Mos)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inconel 600 TT</td>
<td>15-27</td>
<td>36.5-48</td>
<td>40-60.5</td>
<td>11</td>
</tr>
<tr>
<td>Inconel 690 TT</td>
<td>5-6</td>
<td>10.5-13</td>
<td>12.5-15</td>
<td>8.3</td>
</tr>
</tbody>
</table>

DISCUSSION

The crack growth rates presented in Figure 1 indicate that thermally treated I-600 and I-690 have enhanced caustic SCC resistance compared to that of the mill annealed condition. The performance of thermally treated I-600 and I-690 are approximately equal at temperatures of 316°C and below. The superior performance of thermally treated I-690 at higher temperatures is a result of a lesser temperature dependency.

The enhancement in IGA resistance can be attributed to two factors; heat treatment and alloy composition. A characteristic of mill annealed I-600 C-rings exposed to deaerated sodium hydroxide environment is the presence 1443c/0193c/080884:5 8
of intergranular SCC along with uniform grain boundary corrosion referred
to as intergranular attack (IGA). The relationship between SCC and IGA is
not well established but it does appear that IGA occurs at low or
intermediate stress levels and at electrochemical potentials where the
general corrosion resistance of the grain boundary area is a controlling
factor. Thermal treatment of I-600 provides additional grain boundary
corrosion resistance along with additional SCC resistance. In the case of
I-690, the composition presumably provides an additional margin of
resistance to IGA and the thermal treatment enhances the SCC resistance.

The addition of oxidizing species to deaerated sodium hydroxide
environments results in either a deleterious effect or no effect on the SCC
resistance of thermally treated I-600 and I-690 depending on the specific
oxidizing specie and concentration (Table 1). The addition of 10 percent
copper oxide to 10 percent sodium hydroxide decreases the SCC resistance of
thermally treated I-600 and I-690, and also modifies the SCC morphology
with the presence of transgranular cracks. The exact mechanism responsible
for these changes is not well understood, but it is believed to be related
to an increase in the specimen potential, corresponding to a transpassive
potential, which results in an alternate cracking regime. The specific
oxidizing specie and the ratio of oxidizing specie to sodium hydroxide
concentration appear to play an important role. By lowering the copper
oxide or sodium hydroxide concentration, the apparent deleterious effect on
SCC resistance is eliminated.

Continuing investigation of the SCC resistance of I-600 and I-690 in
primary water environments has shown mill annealed I-600 to be susceptible
to cracking at high levels of strain and/or stress. Thermal treatment of
I-600 in the carbide precipitation region greatly improves its SCC
resistance. The performance of I-690, both mill annealed and thermally
treated, demonstrates highly desirable primary water SCC resistance,
presumably due to alloy composition.

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CONCLUSIONS

1. Thermally treated Inconel 600 tubing exhibits enhanced SCC and IGA resistance in both secondary-side and primary-side environments when compared to the mill annealed condition.

2. Thermally treated Inconel 690 tubing exhibits additional SCC resistance compared to thermal treated Inconel 600 in caustic, acid sulfate, and primary water environments.

3. The alloy composition of Inconel 690 along with a thermal treatment provides additional resistance to caustic induced IGA.

4. The addition of 10 percent CuO to a 10 percent deaerated NaOH environment reduces the SCC resistance of both thermal treated I-600 and I-690. Lower concentrations of either CuO or NaOH had no effect, nor did additions of Fe$_3$O$_4$ and SiO$_2$.

5. Metal released in flowing primary water was lower for Inconel 690 compared to Inconel 600. A reduction in plant activity is predicted for PWR plants with I-690 steam generators.

REFERENCES


ACKNOWLEDGEMENTS

Many of the test programs described herein were initiated by G. P. Airey, presently with Central Electricity Generating Board. His contributions are gratefully acknowledged.

This paper contains information generated in accordance with provisions of the research and development agreement between Commissariat a' l' Energie Atomique, Electricite' de France, Framatome and Westinghouse Electric Research and Engineering for Atomic Systems, incorporated.
SESSION 5.3: INFLUENCE OF OPERATING EXPERIENCE ON STEAM GENERATOR DESIGN IMPROVEMENTS WITH RESPECT TO CORROSION RESISTANCE

Buchalet Christian
Framatome Paris

1. INTRODUCTION

The steam generators of PWR nuclear reactors are among the primary components most affected by corrosion problems. Corrosion of the steam generator tubes, which assure heat transfer between the primary and secondary circuits, have been observed on a large number of operating steam generators, especially in the United States. According to an NRC survey, as of November 1981, forty PWR units with steam generators of the recirculation type were in operation in the U.S. Of these, 32 have been found to have one or more forms of tube degradation.

Construction of the French PWR nuclear program started in the early 70s, at the time a number of operating plants in the U.S were being affected by the first corrosion problems. Since, at that time, its construction program was in an early stage, FRAMATOME was able to make modifications on the first units to improve the steam generator resistance to corrosion. For instance, full depth expansion of the tubes in the tubesheet using an explosive process (Westex) was performed on Fessenheim 1 steam generators already installed on site.

Later on, continuous operating experience was being obtained in the U.S, before startup of the French units. This allowed FRAMATOME to react rapidly and take immediate corrective actions at the design stage, during fabrication and sometimes even on site in order to mitigate the risk of corrosion in the steam generators.
In parallel to the operating experience feedback, results from an extensive Framatome R & D program were obtained on thermohydraulic aspects, corrosion, manufacturing technology, resulting in further design improvements.

More recently, operating experience of PWR steam generators throughout the world has confirmed the continuation of some of the early corrosion problems and has also revealed new corrosion phenomena. Design of the latest Framatome steam generator models have benefited from the more recent operating experience, as well as from the latest results of its development program on new materials.

Framatome is confident that the present design of its steam generator models, including a large number of major improvements is adequate to prevent major corrosion problems to occur during operation. However, the company is pursuing an important development program to further improve the corrosion resistance and thereby, the reliability of its steam generators. This program includes studies on new tube to tubesheet joints, mechanical treatment of tubes to reduce primary side residual stresses as well as on local thermohydraulic flow, tube vibration and wear etc...

2. THE VARIOUS CORROSION PHENOMENA AFFECTING STEAM GENERATORS TUBES.

The main corrosion phenomena observed, to date, on steam generator tubes are schematically presented on figure 1.

2.1. Denting

Denting consists in plastic deformation of the tubes at tube support plate intersections, due to squeezing of the tubes by buildup of corrosion products in the gaps between tubes and tube support plates. The plastic deformation of the tubes induces high residual tensile stresses at the inside surface of the tube which may result in stress corrosion induced cracking.
Also, the buildup of corrosion products in the annuli between tubes and tube support plates leads to support plate deformation, especially of the upper support plate, which may induce bending and ovality of the tubes, at the apex of the inner row U-Bends. This ovality may produce high residual stresses at the inside of the tube and increase the risk of stress corrosion cracking.

2.2. Wastage

The wastage phenomenon is a secondary side localized corrosion of inconel 600 tubes caused by a chemical attack from acid phosphate residues, concentrated in low flow velocity areas above the tubesheet.

2.3. Stress Corrosion Cracking (SCC)

Stress Corrosion Cracking occurs in areas where high stresses exist. The corrosion agent is not always identified.

Caustic Stress Corrosion Cracking (CSCC) refers to SCC where the aggressive agent has been identified as caustic. This form of degradation has been observed on the secondary side in the rolling transition zone of the tubes, in the crevice between tubes and tubesheet in the case of partial rolling, at the tubesheet surface level in case of full depth expansion.

When the aggressive agent cannot be identified, the phenomenon is referred to Secondary side Stress Corrosion Cracking (SSCC). SSCC has been observed in the same areas as CSCC.

Primary Side Stress Corrosion Cracking (PSCC) designates a corrosion of the tubes on the primary side, without a specific chemical agent. This type of corrosion has been observed in the tube rolling transition zone, in the tubesheet, at the apex of the small U-bends in case of denting and also at the transition between the straight and bent portions of the tubes (presence of residual stresses).
2.4. Intergranular Attack (IGA)

IGA refers to the corrosive attack of inconel 600 grain boundaries, without any stress related preferential orientation. This form of corrosion has been observed on the secondary side, at the tubesheet level, in the sludge pile area and/or in the tube-tubesheet crevice, in case of a partial expansion.

2.5. Pitting

This type of corrosion consists in a localized attack of the tubes. Pitting has been observed on the secondary side, at several locations on the tubes, above the tubesheet.

3. STEAM GENERATOR TUBE EXPERIENCE

The largest and earliest experience of operating PWR steam generators has been obtained in the United States. The first evidence of steam generator tubes affected by corrosion problems appeared in the U.S in early 70s and included localized thinning (wastage) and Caustic Stress Corrosion Cracking (CSCC) on the secondary side. During this period, the predominant method of controlling the secondary water chemistry was by phosphate control. These early problems have been attributed to difficulties in adequately controlling phosphate concentration and to impurities carried into the steam generators by feedwater. The establishment of all volatile treatment (AVT) control in the mid 70s succeeded in arresting any significant wastage by phosphates but caustic SCC has continued to be a concern.

Also, in 1975, a new corrosion phenomenon referred to as denting appeared on a number of plants which had shifted from phosphate water chemistry control to AVT. To date, approximately 24 PWR units in the U.S have reported denting, including 8 plants extensively affected. Later, other corrosion phenomena have
occurred such as primary side SCC in U-bends and, in 1977, intergranular attack (IGA) and SCC of steam generator tubes in the crevice between the tubes and the tubesheets. Tubeshell crevice corrosion has occurred in at least 7 of the 17 U.S plants where the tubes were not expanded the full depth of the tubesheet. For example, the degradation at San Onofre unit was quite extensive, necessitating sleeving and plugging repairs of approximately 7000 tubes.

More recently, other tube degradation phenomena have occurred such as pitting of the tubes above the tubesheet and fretting between the tubes and the anti-vibration bars (AVB) located in the U-bend region.

Recent tube failures were observed on Westinghouse preheat type steam generators at Ringhals 3 in Sweden and Almaraz 1 in Spain. The tube degreadations occurred within the preheater sections and were attributed to flow induced vibrations, resulting in tube wear by rubbing against the baffle plates.

Early operating experience on PWR steam generators in some European Countries and in Japan revealed the same problems as in the U.S.

In France where the operating experience is more recent (≤7 years), almost none of the corrosion phenomena mentioned previously have been observed. As described later, the first S.G models constructed by Framatome already included design modifications to avoid the major corrosion problems identified at that time. In addition, all the French PWR units have been operating on AVT from the beginning, with very stringent chemistry requirements. Also, all units operating on sea water have condensers with titanium tubing.

Recently, primary side stress corrosion cracking have been observed in some French (Fessenheim 1, Bugey 5, Dampierre 1) and Belgian (Doel, Tihange) plants, in areas of high residual stresses, typically, in areas of defective expansion operation in the tubesheet. This phenomenon has also been observed recently in other countries.
4. DESIGN IMPROVEMENTS

Today, the Framatome nuclear construction program includes 60 units.

In France:
- Fessenheim: 2 x 900 MWe units (FSH 1, 2)
- Bugey: 4 x 900 MWe units (BG 2, 3, 4, 5)
- Contrat Program: 28 x 900 MWe units (CP1 to CP28)
- 1300 MWe Program: 18 x 1300 MWe units (PQ1 to PQ18)
- 1450 MWe Program (N4): 2 x 1450 MWe units (Chooz B1, B2)

Export:
- Belgium: 2 x 900 MWe units (Doel 3, Tihange 2)
- South Africa: 2 x 900 MWe units (KB 1, 2)
- Corea: 2 x 900 MWe units (KU 9, 10)

Figure 2 presents the stretch-out in time of this program.


During this period of time, the S.G models constructed by Framatome have benefited principally from the operating experience in the U.S which can be summarized as follows:
- localized tube wear in U-bends at intersection with cylindrical Carbon steel anti-vibration bars
- secondary side corrosion in the tube to tubesheet crevice.
- secondary side corrosion of tubes in areas of sludge accumulation on the tubesheet hot leg, due to low flow velocities.

The first S.G model constructed by Framatome is the 51 A model. This model is derived from the Westinghouse reference model 51 but already includes design modifications:

the hydraulic flow above the tubesheet was improved in order to minimize areas of low flow velocity where sludge deposition on the tubesheet is predominant and corrosion phenomena enhanced.
Early design improvements include the following:
- increase of the circulation ratio (C.R) by removing the pressure drop adjustment flanges in the down-comer
- reduction of the tube bundle by-pass flow by introducing a tube lane blockage device.
- reequilibration of boiling between cold ans hot legs (80%-20% feedwater offset)
- introduction of inconel 600 chromium coated, square anti-vibration bars
- closure of the tube to tubesheet crevice by full depth expansion of the tubes in the tubesheet on all steam generators, including those already on site (Fessenheim).

These improvements were implemented on Framatome steam generators as early as for Fessenheim and Bugey steam generators.

In model 51M, Framatome continued to improve the thermohydraulic characteristics of the flow above the tubesheet by reducing the opening at the base of the wrapper, which increases the fluid momentum at penetration of the tube bundle, by introducing a flow distribution baffle, which directs the flow towards the center of the tubesheet and reduces upwards flow velocities, and by improving the performance of the tubesheet blowdown system with "L" shape blowdown tubes having 2/3 of the holes located under the central hole in the flow distribution baffle. Figure 3 shows schematically, the design improvements related to flow distribution above the tubesheet. The Fessenheim and Bugey plants are equipped with models 51A. The first 15 units of EDF 900 MWe "Contrat Programme" and the two Doel 3 and Tihange 2 plants are equipped with 51M models.

4.2. Continuous Design Improvements (1975 - 1979)

Continuing operating experience in the United States revealed two new phenomena, particularly in plants with significant periods of phosphate operation before conversion to AVT:
- Denting of the tubes at the tube/tube support plate intersections and the resulting deformation of the small radius U-bends,
- Caustic Stress Corrosion Cracking (CSCC) and intergranular attack (IGA) of the tubes, on the secondary side.
In 1975, when the first evidence of denting appeared on a number of steam generators in the U.S, Framatome adopted broached quatrefoil tube support plates on all steam generators from CP16 unit, and ferritic stainless steel with 13% Cr for the support plate material from CP19 unit.

Figure 4 shows the initial drilled tube support plate design and figure 5 shows the improved broached quatrefoil design. In the broached quatrefoil design, the fluid is forced to flow around the tubes (sweeping) thereby reducing the risk of dry-out and preventing accumulation of corrosion products. Broached quatrefoil design and 13% Cr support plate material should eliminate the risk of denting.

In 1977, corrosion of the tubes in the crevice between tube and tubesheet appeared on a number of operating steam generators in the U.S. In fact, as early as 1975, Framatome decided to perform full depth expansion of the tubes in the tubesheet on all steam generators including those already on site. From CP1, Framatome improved further the efficiency of the full depth expansion by performing a complementary expansion operation (Kiss Roll) in the transition region, which reduced the residual stresses on the secondary side of the tubes.

Figure 6 shows the tube geometries at the secondary face of the tubesheet, after full depth mechanical expansion and after the complementary rolling operation.

More recently, from CP21, Framatome used inconel 600 tubes, thermally treated at 720°C to decrease residual stresses and to improve the resistance to corrosion of the tubes.

At the beginning of this period, Framatome and EDF started the design studies of the 4 loop, 1300 MWe plant. At that time, the decision was taken not to follow the Westinghouse steam generator model E2 (with preheater) of the South Texas reference plant, but to develop the 68/19 model, a steam generator of the boiler type, having the same power than the E2 model.

The results of the R & D program carried on to develop the 68/19 model were also utilized to design a new 3 loop model, the 51B model.
4.3. Today's Steam Generator Models and Latest Improvements (1979 - 1984)

During this time period, steam generator operating experience in the world confirmed the previously identified problems and revealed some new ones:
- Primary Side SCC in the small U-bends at the tangent points between the bend and the straight parts of the tube,
- Primary Side SCC in the roll-transition zone. In France, this type of corrosion has been observed at Bugey 5 and Dampierre 1.

The S.G models proposed today by Framatome are as follows:
- 55/19 : 950 MWe S.G of the boiler type, derived directly from model 68/19 (Paluel)
- 73/19 : 1100 MWe S.G with axial mixing preheater.

These models benefit from the results of an extensive development program on tube material and can utilize a new material, totally qualified today: inconel 690. As described in the next paragraph, this material has been selected for its improved resistance to all forms of corrosion (1).

The continuous major design improvements of the various Framatome steam generator models with respect to corrosion are summarized on table 1.

Since a great improvement in the tube resistance to primary and secondary side corrosion is expected from the use of thermally treated inconel 690, it is worth developing the bases upon which the selection of inconel 690 for tube material was made

(1) It is worth noting, however, that thermally treated inconel 600, used in S.G model 51 B and 68/19 greatly improves the tube resistance to PSICC.
4.4. Choice of inconel 690 for tube material

In the mid-seventies, in the frame of a quadripartite agreement between Electricité de France (EDF), Commissariat à l'Energie Atomique (CEA), Westinghouse (W) and Framatome (F), an extensive research and development program was launched to evaluate materials exhibiting a better corrosion resistance than mill-annealed inconel 600. At first, thermally treated inconel 600 was recognized to have improved behavior and was introduced in our fabrication since 1979.

However, research work was pursued with the objective of finding, testing and qualifying an even better alternate material. Austenitic and dual phase alloys were tested. Due to the joint effort of the four parties (EDF/CEA/W/F) an extensive testing program was performed to obtain the relevant properties needed for selecting the most performing alloys:
- mechanical and physical properties
- structural stability (in service behavior)
- behavior of tube/tubesheet joints performed by mechanical expansion
- corrosion resistance.

Of course, this last property was particularly studied and various forms of corrosion phenomena, susceptible to occur during normal and incidental secondary and primary water chemistry events, were investigated. A large amount of data was obtained on the following aspects:
- general corrosion (release rate)
- pitting susceptibility
- stress corrosion in chlorides, caustics, pure water and primary coolant.

Based on the test results, the conclusion with regard to corrosion resistance, was that thermally treated inconel 690 out-performs each of the other alloys for one or more types of corrosion and is never less resistant for the other types.
In particular, T.T inconel 690 appears not to be susceptible to PSCC. Therefore, a fabrication preseries of thermally treated inconel 690 tubing was manufactured, using three different heats from two different suppliers. Since the results were satisfactory, the decision was taken with EDF to utilize this alloy in the fabrication of the steam generators of the new 4 loop reactor N4.

5. CONCLUSION

Due to the fact that the U.S nuclear construction program started about 10 years before the construction of the first French PWR units, Framatome was able to benefit at the beginning of its program from operating experience in the U.S and to make modifications on the first steam generators, to improve their resistance to corrosion.

Later on, continuing experience and results from an important R & D program allowed to improve on a continuous way the design of the various S.G models.

The S.G models proposed today by Framatome include all the improvements implemented from the first models and can, in addition, utilize T.T inconel 690 for tube material. This material exhibits an excellent resistance to primary and secondary corrosion.

Although, the operating experience of Framatome S.G is still relatively young (Fessenheim has been operating since 1977), it can be considered to date as satisfactory. A detailed status of this operating experience is given in the presentation by Electricité de France.
FIG. 1 - TYPICAL CORROSION PROBLEM AREAS IN PWR STEAM GENERATORS

- U-bend
- Denting
- Support plate tube holes
- Denting-induced deformation of support plate flow slots
- Thinning
- Sludge Pile
- Cracking
- Tube-to-Tubesheet Crevice
- IGA
- Fe₃O₄ (magnetite)
FIG. 2 - THE FRENCH PWR PROGRAM
FIG. 3 - FLOW DISTRIBUTION IMPROVEMENTS
FIG. 5 - QUATREFOIL TUBE SUPPORT PLATE DESIGN
FIG. 6 - TUBE GEOMETRY AFTER

a) FULL DEPTH MECHANICAL EXPANSION
b) COMPLEMENTARY ROLLING OPERATION
<table>
<thead>
<tr>
<th>FRAMATOME S.G MODEL</th>
<th>IMPROVED FLOW CIRCULATION ABOVE TUBE-SHEET</th>
<th>FULL DEPTH EXPANSION</th>
<th>KISS ROLL</th>
<th>DISTRIBUTION BAFFLE</th>
<th>IMPROVED BLOW-DOWN</th>
<th>BROACHED SUPPORT-PLATES</th>
<th>13 % CR SUPPORT PLATE MATERIAL</th>
<th>FINAL HEAT TREATMENT OF TUBES</th>
<th>INCONEL 690 TUBE MATERIAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>51 A</td>
<td></td>
<td></td>
<td></td>
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<td></td>
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SESSION 5.4

CHEMICAL AND ENVIRONMENTAL CONTROLS - A KEY FACTOR IN EXTENDING STEAM GENERATOR LIFE

by

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INTRODUCTION

The integrity of steam generator tubing is a critical aspect of the overall availability of a nuclear power plant. Experience has made the nuclear industry well aware of the susceptibility of steam generator tubing to a variety of degradation mechanisms - general wastage, intergranular attack (IGA), stress corrosion cracking (SCC), pitting, tube denting, and heat transfer and pressure losses. Such conditions can lead to unreliable operation, plant deratings, and even steam generator replacement. The major sources of these problems are an improper chemical environment on the secondary side of the steam generators and a lack of environmental controls during all phases of plant operation and maintenance.

The overall health and longevity of a steam generator depend on vigilance in nine areas, all of which are discussed in this paper.

- Knowing the impact of potential contaminants
- Monitoring the chemistry of the system
- Maintaining the proper parameters
- Correcting out-of-specification conditions
- Controlling chemical processes and equipment
- Knowing the sources of and preventing contamination and recontamination
- Implementing controls during maintenance, repair, and modification work
- Indoctrinating and training personnel
- Using quality assurance and quality control programs

KNOWING THE IMPORTANCE OF POTENTIAL CONTAMINANTS

Many problems that have affected power operation in nuclear plants and have resulted in many shutdown periods to make repairs and perform maintenance stem from the chemical environments on the secondary side of the steam generators. The chemical environment is related to all forms of corrosion general wastage, intergranular attack (IGA), stress corrosion cracking (SCC) and pitting as well as, tube denting, and heat transfer and pressure losses. These, in turn, can be related to deposit formation, underdeposit environments, crevice environments and stagnant conditions.
The chemical species responsible for the aforementioned have been associated with the anomalies include:

- Iron (denting and deposit formation)
- Copper (denting)
- Silica (deposits)
- Chloride (denting and pitting)
- Sodium (IGA, SCC, and pitting)
- Sulfur (IGA and pitting)
- Oxygen (IGA, SCC, general corrosion, and pitting)

The concentration of total solids and pH of the system are also contributors to steam generator problems. Large buildup of total solids can lead to denting and heat transfer and pressure drop losses, while improper pH contributes to IGA, SCC, and pitting. The potential impact of these parameters, makes it important to develop and maintain procedures for controlling the chemical environments in the steam generators. To control the chemical environment, the importance of maintaining correct parameters must be recognized, understood, and enforced by all parties concerned with power operations, maintenance, and ancillary activities associated with the plant.

The significance of maintaining good steam generator chemistry has been recognized by two groups: The Electric Power Research Institute/Steam Generator Owners Group (EPRI/SGOG) and The Institute of Nuclear Power Operations (INPO). The Electric Power Research Institute (EPRI)/Steam Generator Owners Group (SGOG) have recently issued guidelines that recognize the impact of chemistry parameters and environments on the performance and reliability of nuclear steam generators. In addition the Institute of Nuclear Power Operations have issued guidelines for the qualification of chemistry personnel in nuclear power plants.

**MONITORING THE CHEMISTRY OF THE SYSTEM**

Monitoring the chemistry of the steam system is a key item in environmental considerations. Without proper monitoring, the chemical environment cannot be adequately assessed to forewarn when a potential problem may occur or take the proper corrective actions. Further, it makes it difficult to assess the situation once the problem has occurred.
The present monitoring schemes for steam generators mainly comprise a few on-line monitors and grab sample analyses of the feedwater for the once-through steam generators (OTSGs) and analysis of the feedwater and blowdown water for the recirculating steam generators (RSGs). However, this type of monitoring does not present a true picture of the chemistry of the material remaining in the steam generator. Some chemical constituents are soluble or volatile in the steam (e.g., sodium compounds and silica) and are thus carried over to the turbine. Additionally, some of the material (e.g., sodium compounds) goes with the water in the moisture separator reheater (MSR) and is routed back to the feedwater via the MSR drains. Thus, just monitoring the feedwater does not indicate the amount of material going into and settling in the steam generator and remaining there.

Another problem with the present-day monitoring is that there is usually a significant delay between taking the sample and reporting the analytical results. Compounding this delay are often the weeks or more before the data are plotted, trended, and completely evaluated. Further, the long-term storage of chemistry data is often such that it is difficult to retrieve when the need may arise to develop the historical data for evaluating a problem. Another drawback is that the chemistry monitoring does not factor in data for flows and other conditions that impact determining how much material has been carried into the steam generator and remains there.

Computer technology has made possible a chemistry monitoring system that can avoid these drawbacks. By using on-line monitors at strategic locations (e.g., feedwater line, steam line, MSR drain line, blowdown line, condenser hotwell, condensate system) key chemistry parameter, along with other pertinent data, such as flows, temperatures, and pressures, can be fed into a computer system. The system allows the data to be continuously available for displaying, trending, long-term and short-term storage, identification of out-of-specification conditions, and most any other conceivable method for assessing the immediate, short-term, or long-term impact of the parameters. The computer output can also be routed to a remote location (Central Engineering Office) for other purposes such as reviews and evaluations by a utility wide chemistry staff - either immediately or in the future.
MAINTAINING THE PROPER PARAMETERS

The specified or control values for chemical parameters should be meaningful, achievable, and maintainable. For example, there are some cases where specified values that are much higher than those that operating experience have shown to be normally achievable and maintainable. In such cases, the specified or control values should be reduced to reflect this experience. The EPRI/SGOG guidelines(1) follow this philosophy. For example, B&W originally specified a value of 0.5 μS/cm max. for feedwater cation conductivity. The guidelines now list a value of 0.2 μS/cm max. since operating experience has shown that plants normally have measured values of 0.15 μS/cm.

At the other end of the spectrum there is the tendency to specify values that are difficult to achieve or maintain. The reason is that the specified value is often well below the value detectable by analytical procedures that are available or practical for use at power plants. This condition often leads to situations where the specifications are ignored.

The parameter specified should be in a chemical form which is meaningful or is technically justifiable. An example is the recent concern with sulfur species and the corrosion of high nickel alloys (e.g., Inconel 600 and austenitic stainless steels). In most cases the specified values have been listed in the form of sulfate (SO₄²⁻) which is the only oxidized specie of sulfur. The reduced forms, of which there are many, (e.g., sulfite or SO₃²⁻, sulfide of S⁻, thiosulfate of S₂O₃⁻) have been long known to be the species involved in the corrosion mechanism. The reduced species are extremely difficult to analyze individually. However, the reduced species can be determined collectively by oxidizing them to the sulfate form and measuring the total sulfur as sulfate and then subtracting the amount of sulfate in the original sample. In this way the sulfur forms of real concern can be determined or evaluated.

CORRECTING OUT-OF-SPECIFICATION CONDITIONS

Corrective actions for out-of-specification chemistry conditions are often not well defined. Sometimes they are overly restrictive and do not provide any alternatives to the instructions. At other times they tend to be too lax and do not provide any meaningful instructions.
Proper corrective-actions should be related according to:

- Potential for the out-of-specification condition to cause damage or otherwise impact plant operations
- Conditions of the plant or equipment at the time of out-of-specification condition (e.g., temperatures and pressures)
- Values of other variables that may have an effect on the potential damage (e.g., if there are high sulfur levels in the steam generator water, is the pH basic enough to help arrest the potential damage effects?)

The EPRI/SGOG secondary water chemistry guidelines\(^1\) presents a graded system of action levels or corrective actions for out-of-specification conditions for recirculating steam generators. These are considered a good approach to the subject and are reviewed below:

"Three action levels have been defined for taking remedial action when monitored parameters are observed and confirmed to be outside the normal operating value. Normal operating value as it is used here refers to the value of a parameter which is consistent with long-term system reliability. Action Level 1 is implemented whenever an out-of-normal value is detected. The normal values given for Action Level 1 are practical and achievable in the field. Maintaining parameter values within the normal range will provide a high degree of assurance that corrosive conditions will be avoided. Action Level 2 is instituted when conditions exist which have been shown to result in some degree of steam generator corrosion during extended full (100%) power operation. Action Level 3 is implemented when conditions exist which will result in rapid steam generator corrosion and continued operation is not advisable.

Action Level 1. Objective: To promptly identify and correct the cause of an out-of-normal value without power reduction.

Actions:

a) Return parameter to within normal value range within one week following confirmation of excursion.

b) If parameter is not within normal value range within one week following confirmation of excursion go to Action Level 2 for those parameters having Action Level 2 values. The lack of progressive action criteria for many parameters is not intended to imply that remaining outside the normal range is satisfactory. In these cases, other chemical parameters, specifically associated with known corrosion conditions, are utilized for control."
Action Level 2. Objective: To minimize corrosion by operating at reduced power while corrective actions are taken. Power reduction should be to a level which will reduce available steam generator superheat and heat flux in the crevices where concentration of aggressive chemicals can occur, while providing sufficient system flow to maintain automatic operation while the source of the impurity is corrected. This reduced power level is typically 30% of full power or less.

Actions:

a) Reduce power to appropriate level (typically 30% or less) within four hours of initiation of Action Level 2.

b) Return parameter to within normal value range within 100 hours or go to Action Level 3 for those parameters having Action Level 3 values.

Action Level 3. Objective: To correct a condition which may result in rapid steam generator corrosion during continued operation. Plant shutdown will avoid ingress and eliminate further concentration of harmful impurities.

Actions:

a) Shut down within four hours and clean up by feed and bleed or drain and refill as appropriate until normal values are reached. The judgment on maintaining the steam generator in a hot condition or progressing to cold shutdown should be based on the corrosion concern imposed by the specific impurity and the most rapid means to effect cleanup.

Written instructions for corrective actions should be prepared for out-of-specification conditions for each parameter so that plant personnel can respond in a timely manner. A program should also be established for cognizant personnel to formally review the written instructions on a regularly scheduled basis. The following may be considered typical corrective actions:

- Compare results of various analyses and readings from continuous monitors for consistancy.
- Identify and isolate sources of impurity ingress.
- Increase steam generator blowdown (if applicable) to maximum levels for removal of specific impurities.
- Increase sample and analysis frequencies for short-term trending and confirmatory analysis of critical parameters.
CONTROLLING CHEMICAL PROCESSES AND EQUIPMENT

The ability to maintain the proper chemistry environments in the steam generators is only as good as the manner in which the chemistry processes and equipment are controlled.

- The deaeration equipment (in the condenser hotwell or in the feedwater cycle) should be capable of handling the normal ingress of air (5-6 scfm).

- The fresh makeup added to the steam plant cycle should be deaerated.

- The chemical addition systems (e.g., for the pH control additive and the hydrazine) should be capable of compensating for variations in feedwater flow and other variables.

- The condenser should have capability of isolating compartments to minimize the effects of condenser tube leaks.

- Moisture separator reheater (MSR) drains should be routed to the condenser or routed through in-line purification equipment to reduce the contaminants recycled with the drains. The use of in-line purification equipment reduces the heat losses that result from routing the drains to the condenser.

- In preparing for startup from a cold shutdown condition, the pre-boiler or feedwater cycle should be flushed to reduce the contaminants to the desired levels before the feedwater is added to the steam generators.

- During cold shutdown conditions, a recirculation and chemical addition system should be available to control and maintain the specified layup chemistry.

KNOWING THE SOURCES OF AND PREVENTING CONTAMINATION AND RECONTAMINATION

Another aspect of proper chemical and environmental controls is to understand the potential sources of contamination and recontamination and to take the steps necessary to prevent them from occurring.
Some of the common sources of contamination and recontamination are as follows:

- Air in-leakage at valve packings, pump seals, condenser shells and other low pressure and vacuum areas that allows oxygen and carbon dioxide to enter the steam cycle. The problems with oxygen are well known. Carbon dioxide forms carbonic acid which depresses (reduces) the pH.

- Impurities in chemical additives. With the low parts per billion control ranges or specifications for some parameters, this has the potential of being a significant source.

- Condensate polishing system resin leakage. Cation resins contain sulfur radicals in their chemical formulations and thus are sources of sulfur contamination.

- Improper regeneration of condensate polishing systems resins. Sodium hydroxide (NaOH) and sulfuric acid (H₂SO₄) are used as regenerant solutions and thus are potential sources of sodium and sulfur contamination.

- Condenser tube leaks. This is a source of various contaminants depending on the source of the cooling water (cooling tower, lake water, sea water, brackish water, river water, etc.).

- Improper control of organics, lubricants, and oils. These are sources of sulfur in particular and of organics that can foul demineralizer ion exchange resins and possibly create heat transfer problems by depositing on tube surfaces. This type of contamination is most apt to occur during shutdown periods for maintenance and repair work.

IMPLEMENTING CONTROLS DURING MAINTENANCE, REPAIR AND MODIFICATION WORK

During maintenance, modification, and repair activities, the secondary side of steam generators are highly susceptible to periods when the chemical environments are not properly controlled. There seems to be a general feeling that the control requirements can be relaxed; in addition, the inherent nature of such activities makes it more difficult to provide the desired controls as the steam generator usually has to be open and thus can
be exposed to the external environment. In any event, efforts should be made to control the chemistry to the greatest extent possible as undesirable contaminants can enter the steam generators during these periods, and the conditions in the steam generator can promote some forms of corrosion (e.g., underdeposit and crevice corrosion).

In normal cold shutdown conditions, the steam generators are usually maintained in a wet layup condition with water conditioned with ammonia and hydrazine and with an inert blanket of nitrogen. In nearly all cases the generators must be drained to perform any maintenance, repairs, or modifications. Crevices and areas under deposits remain wet and create corrosive environments.

If the work on the steam generator involves a long term outage and exposure of the interior to the exterior environment, the unit should be dried out to the extent practical by using a recirculation system containing a fan, filter (to remove particulate contaminants), and a drying arrangement using a dessicant or a dehumidifier. The drying can be assisted by pulling a vacuum with the main condenser before the recirculation system is used.

If ventilation is needed while the work is being performed, a closed system would be preferred as it would tend to limit the ingress of undesirable material. However, if a "once-through" system is employed, it should employ a filter on the supply line for control of particulate contaminants.

Openings which are not required to be open to perform the work should be kept closed. During short periods when no work is being performed (e.g., weekends) all openings should be closed with temporary covers. For longer periods, the air should be purged out with nitrogen.

As soon as the work is completed, the steam generator should be restored to its normally maintained cold shutdown condition.

**INDOCTRINATING AND TRAINING PERSONNEL**

Plant chemistry personnel should be trained to have an in depth understanding of the parameters involved in the proper control of the chemical environment in the steam generators. This includes chemistry personnel who are associated with the operation of equipment as well as personnel who
perform laboratory analysis. Within the level of their positions and responsibilities, they should be taught the reasons for the parameters, the impact that they have on plant and equipment operations, and the corrective actions that should be instituted for out-of-specification situations. Other plant personnel that deal with activities involving environmental chemistry controls (e.g., steam production personnel) should also be trained and indoctrinated so that they have a similar understanding of the parameters. This will aid in providing a harmonious relationship between the various plant personnel in matters dealing with chemistry control.

The training should be a continuous review process to keep the personnel abreast of new developments in experience and technology and to reacquaint them with items which are usually not a part of their normal or routine activities.

The Institute of Nuclear Power Operations (INPO)\(^2\) has issued several publications defining the requirements for indoctrinating and training chemistry personnel. This training includes some formal and basic types of training.

**USING QUALITY CONTROL AND QUALITY ASSURANCE PROGRAMS**

In a way chemical and environmental controls are forms of preventive maintenance, quality control and quality assurance. In the vernacular of a well known American television advertisement slogan, "you can do it now, or suffer the consequences later". In this context, quality controls and quality assurance must be an integral part of any chemistry and environmental control program. All procedures and practices should utilize the principles of quality control and quality assurance. Typical examples are:

- Personnel performing chemical analyses and chemical process functions should be trained and qualified for their work.
- On-line chemical monitoring equipment should be checked by the frequent chemical analyses of grab samples.
- Cross-checks of chemical analysis results and procedures should be performed on a regular scheduled basis.
- Chemicals used in the plant and in the chemical laboratory should be verified to meet the chemical specifications and other requirements for the intended use.
When an unusual result of a chemical analysis is obtained in the laboratory the first order of business should be to reanalyse the sample or obtain another sample and analyze it.

**SUMMARY**

Control of the chemistry environments that affect the secondary side is vital to maintaining the operability and increasing the operating life of steam generators. The controls must be maintained during all operating modes (power, shutdown, maintenance, repairs, and modifications). The institution and maintenance of the controls must be a cooperative effort between management, production, maintenance, and chemistry personnel. Chemistry environmental controls are forms of preventive maintenance, quality control, and quality assurance that relate to all of the activities associated maintenance of the proper environments.

**REFERENCES**


SESSION 5.5: SECONDARY SIDE CONCEPT TO
PREVENT STEAM GENERATOR TUBE CORROSION

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1. INTRODUCTION

Pressurized water reactor (PWR) nuclear power plants have experienced widespread corrosion problems on the secondary side of the steam generators. The problems include stress corrosion and fatigue cracking, tube denting at support intersections, tube thinning (wastage) and pitting within sludge piles above the tube sheet, and intergranular cracking within the tube sheet (Fig. 1). However, Kraftwerk Union's experience gained in Germany, the Netherlands, Switzerland and Argentina, yield more favorable operating results, especially with the present design concept which was introduced in 1972 when the 662 MWe Stade PWR power plant was put into service.
2. SG DESIGN FEATURES TO MINIMIZE CORROSION

KWU presently has more than 30 U-tube steam generators in operation, which are essentially designed and manufactured in accordance with present KWU standards employing INCOLOY 800 tube materials. INCOLOY 800 tubing has been tested and shown to be highly corrosion resistant (see Fig. 2). Adoption of this tube material together with design improvements, as applied to these steam generators has yielded an excellent reliability record:
U-bend stress corrosion cracking has not occurred on the INCOLOY 800 tubing. As an additional measure against stress corrosion, glass-bead shot peening after bending introduces compressive tube surface stresses which eliminate stress corrosion per definition.

Tube denting has been avoided by utilizing austenitic strip spacers which form an open grid with only linear contact to the tubes, as shown in Fig. 3. Tube denting has not occurred with this design, which has also been confirmed by tests in high chloride concentration.
Tube wastage caused by phosphate dosing normally takes place at the tubesheet after corrosion products have formed sludge build-up. On top of a sludge pile, as indicated in Fig. 4, a dry-wet zone appears with concentrated impurities or phosphates. Avoiding sludge build-up would eliminate the problem. The design feature which has been adopted is a flow distribution baffle to minimize local tubesheet regions of low velocity flow where sludge typically would deposit (see Fig. 5).
In addition, cleaning of the sludge build-up from the tubesheets has become an important maintenance procedure for steam generators. The triangular tube pitch of KWU steam generators allows easy lancing and cleaning with high-pressure water jets as shown in Fig. 6.

Fig. 6 — Sludge Removal from Tubesheet.
- Intergranular cracking in the tube-sheet region has been eliminated by the material characteristics of INCOLOY 800 tubing.

The welding and expansion procedures for this tubing incorporate two expansion sections, the upper one extending within 0.12 inch of the tubesheet surface (see Fig. 7). This design minimizes the formation of crevices and subsequent stress corrosion cracking in this steam generator region.

3. KWU SECONDARY SIDE CONCEPT TO MINIMIZE CORROSION

In addition to SG design features there are important parameters influencing the corrosion behaviour of the tubing material depending on the steam water cycle concept. These are:

- corrosion products (amount and composition transported to the SGs)
- impurities (salts and organic)
- oxygen
The basic KWU plant arrangement provides all features to control above mentioned parameters.

These features are:

- Avoid the ingress of contaminants into the secondary side by using high integrity condensers. They are characterized by a copper-free corrosion/erosion resistant material and a tube to tubesheet welding. In addition, a close chemistry control of other sources of contaminants has to be performed.

- Full flow deaeration of the feedwater and make up water

- "High AVT" Chemistry (pH > 9.8) with excessive $N_2H_4$

- Full flow mechanical/electromagnetic filtration of the feedwater and of the heater drains, respectively

4. COMPARISON OF NON KWU AND KWU PLANTS SECONDARY SIDE CONCEPT

A basic difference of Non-KWU and KWU plants is the use of a feedwater tank in the KWU units. The employment of such a tank necessitates a completely different arrangement of secondary side components.

4.1 Oxygen control

Main objective for oxygen control with regard to hardware provisions is:

- prevent $O_2$ ingress into the system
- immediate detection of unacceptable $O_2$ levels
- provisions for fast counteractions.

It is very important to fulfill these objectives, because oxygen is involved in many corrosion mechanisms, especially in the steam generators.
A comparison of foreign and KWU secondary side concepts with respect to oxygen control (i.e. \( O_2 \) sources, \( O_2 \) detection and corrective actions) is summarized in figure 8. Based on this summary and considering the oxygen control objectives the following conclusions may be drawn:

**POWER OPERATION**

- As shown in figure 8 the number of \( O_2 \) sources in foreign plants is higher than in KWU units. Consequently the risk of \( O_2 \) ingress is higher. In detail the different sources can be assessed as follows:

  . Air inleakages into the turbine condenser area may be similar in both concepts. Usually these air inleakages can be counteracted by condenser vacuum and air removal system and will be partially supported by hydrazine treatment. However, to KWU's understanding some condenser designs may cause either subcooling or insufficient deaerating of the condensate which are both associated with higher \( O_2 \) levels in condenser effluent.

  . \( O_2 \) ingress via demineralized water storage tank (DWST) in the KWU concept gives no concern because of full flow deaeration in the feedwater tank and high hydrazine treatment (no hydrazine limitation). In most of the foreign plants condensate storage tank is designed to supply make up water for the secondary side. It must be emphasized that the condensate storage tank is a key component for \( O_2 \) control in this plants if the possibilities for full flow deaeration in the feedwater system do not exist. Proper sealing of this component against atmosphere is mandatory.
Fig 8: O$_2$ Control

**Sources:**

1. Turbine Condenser (Air inleakage+Subcooling)
2. Condensate Storage Tank

1. Turbine Condenser (Air inleakage)
2. Demineralized Water Storage Tank

**Detection:**

6. Main Condensate (Air-inleakage: CO$_2$ detection via Ka)

6. Main Condensate and Feedwater (Air-inleakage: CO$_2$ detection via Ka)

**Corrective Action:**

- ▲ Hydrazine injection (excess limited)
- ▲ Operation of steam spargers?
- ▲ Fullflow deaeration in Feedwater Tank
4.2 Corrosion product control

The main objective for corrosion product control with regard to hardware provision is:

- minimize the formation of corrosion products within the steam-water cycle
- provide a proper instrumentation to detect the corrosion product transportation into SGs
- prevent the ingress of corrosion products into SGs.

It is very important to fulfill these objectives, because corrosion products may form crud accumulation in low flow areas within the SGs. Such crud accumulation is a prerequisite for higher salt concentration and thereby a prerequisite for SG corrosion problems. Moreover, the corrosion products may lead to a fouling problem of the SG tubes, i.e. the heat transfer from primary to secondary side will be reduced.

From a comparison of Non-KWU and KWU secondary side concepts with respect to corrosion product control (i.e. sources, detection and preventive actions) the following conclusions can be drawn:

POWER OPERATION

- The main source for corrosion products may be turbine condenser and the forward pumped heater drains. At new KWU units major sources cannot be identified. Reasons are:
  - the selection of erosion corrosion resistant material in extraction steam systems and
  - the operation at high pH-values (> 9.8)

- The evolution of corrosion product transport rates can be taken from figure 9.
The sampling and analyzing techniques at foreign plants are often based on grab samples. This is neither sufficient nor representative.

The only possibility for counteraction with a hardware system at the selected foreign plant is the condensate polishing system, if available. This covers only one of the main sources.

STARTUP, SHUTDOWN AND POWER TRANSIENTS

- Many new units (i.e. Plant X and KWR) have the hardware to recirculate all heater drains and the feed-water upstream of a mechanical filter device. It must be emphasized that these possibilities should be used extensively.

4.3 Impurity control (salts)

Main objective for impurity control with regard to hardware provision is:

- prevention of impurity ingress into the system
- immediate detection of unacceptable impurity levels
- provisions for fast cleanup

It is very important to fulfill these objectives because salts as impurities are involved in many corrosion mechanisms specially in SGs (see introduction of this paper).
SESSION 5.6 : PREVENTIVE AND CORRECTIVE ACTIONS FOR DOEL 2 STEAM GENERATORS

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1. INTRODUCTION

Units 1 and 2 of the Doel Nuclear Power Plant were commissioned (start of power operation) respectively in September 74 and June 75; they are twin units (2 x 400 MW) of the 2 loops PWR type.

Both units are provided with identical steam generators, of Westinghouse design (model 44) and Cockerill construction; the 7/8" tubing in mill annealed Inconel 600 was supplied by Mannesman.

Both units were similarly operated by the same team, with secondary full flow demineralization and all volatile treatment (AVT), with the only exception of Doel 1 hot run test where phosphate treatment was used.

However, while unit 1 is still nowadays practically defect free, unit 2 has shown since 1978 leaks from axial cracks in the tubes roll transition. Eddy current inspection results show that up to 90% of tubes in SG-A, and close to 60% in SG-B, are presently affected by the problem which is expected to further progress.

In June 1979, a tube rupture (from a longitudinal crack in bend) was experienced by SG-B and attributed to excessive ovality in the tube bend. This
led to the preventive plugging of 26 (first row) tubes in SG-B and 50 tubes in SG-A. For the latter SG, the roll transition problem further extended the plugging to a total amount of about 140. The extent of plugging is thus about 4% for SG-A and 1% for SG-B.

Several repair techniques were attempted on about 9% of the tubes in SG-A while no significant work has yet been performed on SG-B, which is believed to follow the SG-A trend with a 3 year time lag.

The reason for the drastically different behaviour of Units 1 and 2 (and, to a lesser extent, of SG-A and B for unit 2) has been extensive-ly investigated. While no difference could be traced in the tube mecha-nical properties, a recent review of all micrographs from the 4 SG'S allowed a statistically meaningful comparison with a clear correlation between grain size and actual service behaviour (Fig. 1). The larger grain size (lower ASTM number), probably resulting from a higher tem-perature of the final annealing treatment, is associated with the good performance of both SG's of unit 1; the smaller grain size is associated with the generic cracking problem of unit 2 SG-A, while an intermediate behaviour is observed for SG-B.

2. DEFECT CHARACTERIZATION

Defects presently known belong to the 4 following categories.

2.1 Primary side stress corrosion cracking (PS SCC)

This is by far the most significant problem presently experienced. The so called "pure water" corrosion process affects locations of high (service or residual) stress levels i.e. :

- Ovalized tube bends (believed to be eliminated by preventive plugging of ovalized first row tubes).
- Tube roll transitions (60 to 90% of tubes affected).
Tubes are only partially rolled, for about 70 mm in the lower part of the tube sheet, leaving a nominal .15 mm crevice (radial clearance) higher up (Fig. 2).

The combined information from eddy current testing (ECT) and tube pull expertise characterizes the PS SCC as a pattern of multiple (about 10 in the same roll transition cross section) axial cracks of length up to an estimated maximum of 9 mm. The cracks are very tight and through-thickness cracks are difficult to locate even with the very sensitive fluoresceine leak detection method used with secondary pressurization during plant shutdown.

As it has been the plant policy to repair (or plug) any detected leaker the integrated leak (from PS SCC) observed during service conditions has also been rather low (from 2 to 4 l/h for SG-A and less than 1 l/h for SG-B). It is estimated that about 300 leakers would be required to exceed the Technical Specification limit of 28 l/h.

2.2 Secondary side stress corrosion cracking (SS SCC)

A total number of 3 tubes with SS SCC has been detected in SG-A. The cracks are longitudinal and located between 100 and 300 mm from the bottom. The main cracks were isolated, up to 20 mm long and very tight (making them difficult to detect by ECT even when close to through-thickness. The first crack was detected in October 1982 by a leak rate of about 60 l/h (significantly in excess of the Tec. Spec. limit); this tube was pulled and expertized; similarity with Ringhals II and Takahama problems (where SS SCC was known to have increased at an exponential rate) led to a preventive "crevice cleaning" in order to remove the potential causative (supposed to be caustic) elements.

Ongoing expertise of the three affected tubes (all pulled from SG-A) has not detected any significant amount of caustic element on the tube OD and the cracks surfaces; the causative species remains thus presently unknown.
2.3 Intergranular attack (IGA)

IGA initiated IGA has been identified in the rolled region of two pulled out tubes: This may be locally quite extensive and affect almost the full wall thickness for a significant part of the circumference.

The IGA is undetectable by ECT and it is not known to what extent it is present in the tube bundle (nor whether a similar effect exists in unit 1).

2.4 Tube denting

There is but a limited extent (about 15% of tubes) of moderate denting (mean value: 0.1 mm/maximum: 1 mm) which affects similarly both SG's of units 1 and 2.

The denting is mainly located at the secondary face of the tube sheet (in the sludge pile area) and, to a lesser extent, at the periphery of the lower support plates.

The phenomenon is no more significantly evolutive.

3. REPAIR PHILOSOPHY

As can be seen from the hereabove description, all significant problems of the Doel 2 unit are located rather deep within the tube sheet.

Because of this particular location it was possible to show that
- Longitudinal cracks cannot grow up to a critical length, so that no unstable propagation (out of the tube sheet) is to be feared.
- Circumferential cracks, leading to tube severance, do not allow tube disengagement from tube sheet (because of restraining from adjacent tubes and/or antivibration bars), except for the 6 outermost columns (3 columns, both sides of the tube bundle).

* While all of the original cracks are in the axial direction, a further circumferential deviation cannot be ruled out and some repairs have also initiated additional circumferential cracks (see § 5).
Thus a SGTR occuring in those locations will not result in a safety problem and, because of that, all repair techniques should preferably not affect the tube outside of the tubesheet, in order no to create a potential for a more significant problem (safety versus operational).

Additional criteria retained for the selection of repair techniques are:
- Allow the further implementation of another repair (not necessarily satisfying the same criteria/ "long sleeves" for instance).
- Allow plugging by conventional means.
- Minimize cost, time and radioactive exposure.

Two such local repair techniques have already been applied (§ 4 and 5) and more are contemplated (§ 7).

4. MECHANICAL REROLLING

A rerolling repair technique, at the lower third of the tubesheet, has been jointly developed by Westinghouse, KCD and Laborelec. This superposes a "soft" hydraulic expansion (about 5 " long) and a 2 steps "hard" mechanical rolling (about 2" long, centrally located).

Some variations were also applied to a few tubes, either deliberately or not (wrong identification of tube, incomplete repair sequence, ...). A total of 111 leaking tubes was thus repaired in SG-A (1980 and 1981).

The method was then abandoned because of a number of disadvantages:
- Not applicable to a significant % of tubes because of the tight tolerances required by hydraulic expansion.
- Not fully effective for a significant % of tubes (leak tightness not obtained)
- Sporadic loss of leaktightness during further service operation
- Significant cost, time and radioactive exposure.
Moreover the repair technique was not expected to provide a long term solution as
- The residual stress level associated with the new transitions might initiate further PS SCC.
- The closed gap left between the original and the repair rolled regions might initiate a process of progressive tube deformation (inward bulging), as was experienced in Obrigheim with a similar geometry.

However none of those two harmful effects has been detected until now.

5. EXPLOSIVE MINISLEEVE

The explosively welded "mini-sleeve" repair technique was specifically developed by B&W for KCD, in an attempt to remedy the deficiencies of the previous technique.

A short (40 mm long), thin (0.4 mm) inconel 600 (thermally treated) sleeve is explosively expanded and welded over the cracked area (roll transition) of the tube; the sleeve acts as a leak-tight membrane without any structural function (Fig. 3).

The installation requires prior cleaning of the tube (by brushing) and provides an effective metallurgical bond all over the sleeved surface, except for a few mm from both ends.

Based upon preliminary good results from a still incomplete qualification program, some hundred tubes of SG-A were repaired that way when leakers were detected during the successive shutdowns of late 1982.

However B&W final autoclave tests of mock-ups in NaOH (under imposed potential) revealed that, while no axial cracks were produced in the new expansion transitions, systematic circumferential cracking (undetectable by conventional NDT methods) was initiated at the weld limits, just below both ends of the minisleeve. (Fig. 4).
This was also confirmed by another set of mock-ups of sensitized inconel 600 tested by Laborelec in tetraethylammonium; the extensive circumferential cracking at both ends of minisleeve was indicative of a residual stress level significantly in excess of 150 MPa.

There is presently additional evidence from pulled tubes (from SG-A)
- 2 tubes pulled immediately after sleeving, show no crack.
- Among 2 tubes pulled after a few weeks service conditions, 1 shows circumferential cracking at both sleeve ends (however this sleeve was practically unbonded because of the entire lack of prior cleaning).
- Among 7 tubes pulled after a few months service conditions, all showed circumferential cracking of various extent (up to 3/4 of wall thickness) at least at one of the sleeve ends. One of the sleeve was again unwelded, with evidence of no prior tube cleaning.

While ECT is unable to detect these circumferential cracks (because of the interference with the adjacent geometrical discontinuity) a UT inspection was performed by KWU on 25 sleeved tubes (7 of which were pulled).

The UT results, when calibrated on artificial defects (circumferential machined grooves) indicated a large % of tubes with cracks up to 25% (in one case 60%) of tube wall thickness. The method was further (and more reliably) calibrated against the actual crack depth measurements (7 x 2 sleeve ends \(\rightarrow\) 14 data points). The agreement was quite satisfactory, apart from a systematic underevaluation (by a factor of 3).

Based upon this updated calibration, it can be concluded that after a few (6 to 9) months of service conditions about 60% of the sleeved tubes are significantly cracked (more than 15% of wall thickness) at least of one sleeve end.

A total number of 187 tubes has been sleeved (184 in SG-A and 3 in SG-B) of which 11 have been pulled (from SG-A).
6. CREVICE CLEANING

After the first appearance of SS SCC it was deemed advisable to perform, as early as possible, a "crevice cleaning" to remove the caustic species from the tube to tubesheet crevice. A comparative study between the American and Japanese procedures led to selection of the latter one, on basis of a better demonstrated efficiency (progressive stop of SS SCC on the Takahama 1 and Mihama 2 units, after a few successive cleaning operations).

The selected method calls for an electrical heating (from the dry channel head) of the tubesheet, with an imposed axial thermal gradient (130°C on primary side and 100°C on secondary side). The associated hydraulic system is illustrated by Fig. 5.

The "cleaning" effect is obtained by successive "pressurization-depressurization" cycles, initiating boiling in the crevice and flushing of undesirable sequestrated species.

When the procedure was applied to the 2 SG's in March 1983, it allowed to extract but slightly more than 1 g of Na per SG after 6 cycles (i.e. about 10 times less than has been quoted for similar operations performed in Ringhals 2, Japan or the USA).

This apparently deceiving result should nevertheless not be interpreted as a lack of efficiency. Unlike the other units subjected to crevice cleaning, Doel 2 has always been operated under AVT conditions (phosphate was never used). It is thus likely that very little caustics were present in the crevice, as further confirmed by the laboratory expertise of tubes pulled before the cleaning operation. Owing to this result, the "crevice cleaning" was no more applied during the following outage of Doel 2.

While the causative species of SS SCC has not yet been identified, it is possible that the present lack of any further leak or ECT signal is indicative of an effective beneficial result from the performed "crevice cleaning".
7. FUTURE PLANS

SS SCC is not presently considered a major concern and no conclusion is yet available about IGA. So no generic remedies are contemplated in this respect.

The PS SCC is expected to further progress (new cracks, propagation of existing cracks, both in depth and in length) and requires remedies for which the following strategy has been defined; the "in depth defence" relies on the successive consideration of following approaches.

7.1 Local repairs (complying with the criteria of § 3)

7.1.1 Induction stress relief heat treatment (SRHT) at both ends of minisleeves
Field application of such SRHT has been proven feasible by B&W during the October 1983 outage of Doel 2 (15 tubes of which 8 were pulled and 7 remained in SG-A, including 5 "old" sleeves and 2 "new" sleeves). However mock-up tests with sensitized inconel tested in tetrathionate have shown the residual stresses to remain high. Work is still going on to optimize SRHT parameters.

7.1.2 Minisleeve redesign
A feasibility study is aimed at avoiding cracks by the use of softer material (Ni) and explosive charge shaping. This would remove the need for SRHT to be performed after sleeving.

7.1.3 Explosive reexpansion
A preliminary study is aimed at providing a new leaktight barrier by explosive expansion of the tube for some height above the initial roll transition (a similar procedure was applied to TMI-1)
7.1.4 Roll transition stress relieving
    The same induction SRHT as for § 7.1.1 could be directly applied to the "unsleeved" roll transitions. However the potential of this method for preventing extension of existing cracks needs to be established.

7.2 Conventional sleeving

A number of "long sleeves" repairs are presently proposed by various NSSS suppliers and would be considered in case of failure of the local attempts.

7.3 Steam generator rotation

The temperature dependance of "pure water" SCC is well established (a useful "rule of thumb" calls for a lifetime increase by a factor of 2 per 10°C decrease) and is best illustrated by the excellent service performance of the cold side of all SG's at the world wide level.

A feasibility study has been undertaken by Tractionel to evaluate the potential and economics of a "S.G. rotation" (the tube bundle being rotated by 180°C in order to bring the defective tubes to the "cold side" and the intact tubes to the "hot side").

The main interest of this solution lies in the relatively short notice required for starting implementation.

This would allow further investigation of the various repair possibilities (§ 7.1 and 7.2) without requiring the long lead decisions associated with a S.G. replacement.

7.4 Steam generator replacement

As this cannot be ruled out as the "ultimate solution", preliminary studies (and enquiries for bids) have also been performed.
Fig. 2 Cutaway View of Tubesheet showing Roll Transition Tube Cracking

Fig. 3 Cutaway View of Tubesheet showing Mini-sleeve installed

Fig. 4 Schematic showing weld Transition Tube Cracking
**Fig. 1**

**Fig. 5**
SESSION 5.7: GENERIC PREVENTIVE ACTIONS
FOR MITIGATING M.A. INCONEL 600 SUSCEPTIBILITY
TO PURE WATER STRESS CORROSION CRACKING

G. Frédéric
Electrobel+
P. Hernalsteen
Tractionel++

1. INTRODUCTION

Belgian utilities are presently operating 5 PWR nuclear units (Doel 1, 2 and 3/ Tihange 1 and 2) while 2 more units (Doel 4 and Tihange 3) are due to be commissioned in 1985.

As many other utilities (1, 2), they suffered of primary stress corrosion cracking on their SG.

The worst experience up to now was the roll transition cracking of Doel 2 (3).

More recently Doel 3 experienced leaks after only 10,000 hours of operation. The cracks were demonstrated to be inside the tubesheet, in the full depth roll.

It was decided at that time to define generic preventive actions applicable not only to hot plants like Doel 3 but also to plants under construction with a similar design.

2. DOEL 3 AND TIHANGE 2 EXPERIENCE

The steam generators of the last four units in Belgium have full depth roll (Fig. 1).

++ 1, Place du Trône, 1000 Brussels
++ 31, Rue de la Science, 1040 Brussels
The depth of the tubesheet is covered by 21 passes, with a 5mm overlap between each pair of adjacent passes. At the top of the tubesheet a "Kiss roll" expansion is performed to reduce the OD residual stresses in the roll transition.

The tool used for roller expansion is semiautomatic. It expands a tube at a given pass until a preset torque is achieved, and then the tool is moved to the next pass. The torque value used was calibrated to provide about a 3% tube wall reduction after tube contact with the tubesheet was achieved.

Since the onset of Doel 2 cracking in the roll transition, special attention was given in Belgium to the profile of the tube inside the tubesheet.

An Eddy Current technique was developed and used by Laborelec (4) to systematically record the rolled region profilometry of all the tubes of the last four units (100,000 tube ends). The profilometry uses the lift-off signal (Fig. 2) to evaluate the distance between a surface coil and the tube surface. A diameter is then obtained by adding the lift-off signal of two surface coils at 180° from each other to the distance between these two coils. The measurement of a profile is made on four diameters within the same cross section (Accuracy and reproducibility for on-site conditions better than \( \pm 15 \mu m \)).
A lot of anomalies were found and subsequently corrected, for example skip roll steps, non overlapping between two steps, inadequate kiss roll...

Waves (see record on Fig. 2) were not considered harmful and were left in the steam generators. For the units under construction they were also corrected when greater than 0,15mm (diametral). The origin of these waves is not clear; they appear to be experienced by most SG manufacturers.

In octobre 83, three leaks and two additional cracked tubes were detected at Doel 3. In four cases, the cracks were associated with rolling irregularities at locations with oversized tube holes, where proper contact with the tubesheet was not obtained (see Fig. 3 for an example case; the corresponding gap allowed detectable leakage.

The fifth crack occurred at a location with no significant anomaly.
Two tubes were pulled and rupture was experienced during that operation at a half of the normal load.

The examination of the two tubes allowed to establish the ID origin of the cracks, their intergranular aspect and their position, dimension and number. A very good correlation was obtained between laboratory radiography and metallographic examination performed by EDF and on site Eddy Current performed by L/E with a specially designed rotating probe. Outside the ruptured area where some circumferential components are present the cracks are axial and their length is 4mm maximum.

Subsequent to the identification of this problem at Doel 3, the EC rotating probe was improved (speed and sensitivity) and used for the detailed examination of about 500 tubes in each of two Tihange 2 steam generators.

One of the steam generators was free of indications. However, significant indications were noted in the other steam generator:

- 16 tubes had short deep longitudinal cracks,
- 10 additional tubes had "probable" cracks and
- 100 tubes had small indications, which may or may not prove to be cracks.
The cracks and indications were generally not associated with oversized holes or with rolling irregularities exceeding procedure or drawing limits and one of the cracks was in the roll transition area.

The difference between the two SG was attributed to the difference in final thermal annealing conditions. Table 1 presents these conditions for the belgian last four units.

It appears that the tubing of the SG without any indication was processed during final annealing treatment at a higher temperature which is known to be beneficial for improved resistance to IGSCC (5). The improvement is likely to delay the onset of cracking but is not considered sufficient for a 40 years lifetime.

Three heats from spare tubes of Tihange 3 and Doel 4 (processed at the same higher temperature) exposed to primary chemistry at 365°C (Reverse U bends) were cracked in about two months which can be correlated to about 2 or 3 years in the SG conditions (6).

The existence of important residual stresses was confirmed by tests performed on roll expanded tube mockups using stainless steel tubes in boiling magnesium chloride and sensitized alloy 600 in tetraethionate. These tests have shown that cracks can occur at the overlap areas of the roll expanded region even if tolerance limits on tube ID waviness and roller torque are met.

The earlier appearance (10,000 h) of larger through thickness cracks at Doel 3 is attributed to the higher residual stress levels induced by rolling expansion without tubesheet contact.

3. SIGNIFICANCE OF THE CRACKS FOUND IN FULL DEPTH ROLL

The cracks associated with out of tolerance holes in the tubesheet are not significant, since few tubes are concerned and may be plugged.

The significance of cracks occurring at the overlap regions of normally rolled tubes is not clear. Compressive stresses should exist in the correctly rolled regions, which would be expected to prevent crack extension
in the axial direction and should also prevent leakage.
There is some concern that expansion of water trapped in crevices behind cracks could cause progressive plastic deformation (inward bulging) of tube during plant transients. This type of deformation has occurred at Obrigheim (7); however, in that case the tube had been expanded only at selected elevations in the tubesheet such that there was a larger crevice than would be expected behind a fully expanded tube.

Cracks in overlaps near the roll transition and cracks at this transition are sources of increased concern: with only a short (or nonexistent) properly rolled area above the cracks, primary to secondary leaks are probable.

4. PREVENTIVE ACTIONS

The main approaches which are being pursued in regard to the cracking problems in steam generators at the 4 last units are the stress relief heat treatment of the entire tubesheet area and the roto- or shot-peening of the tube ID for the full depth of the tubesheet, up to above the roll transition.

In the analysis of preventive measures, it must be recognized that different approaches may be required for operational units (Doel 3 and Tihange 2) and for units under construction (Doel 4 and Tihange 3).

Stress relieving appears attractive for operational plants because of the relatively small amount of radioactive exposure and contamination release. On the other hand, the non operational plants are not radioactive which makes roto-peening or shot-peening easier. Also, these units would be more difficult to stress relieve because they have preheater sections (nonsymmetrical conditions) wherein some tubes have been rolled against the support plates.

On the other hand, global heat treatment is probably effective on precracked material (which is the case of operational units) whereas the peening techniques are more questionable. It is generally considered that the axial propagation of existing cracks would be arrested while the penetration would not.
5. GLOBAL HEAT TREATMENT OF THE TUBESHEET

The objective of the global heat treatment is to heat the full depth of the tubesheet and also include the tubing in the kiss roll region immediately above the tubesheet. Decreased susceptibility to primary side IGSCC would be expected from a combination of microstructural improvement and stress relief.

Two stress relief temperatures are being considered, 610° and 550°C. They would be applied for about 10 hours. These temperatures and times are still acceptable for the tubesheet material:

- Based on relevant literature ("temper embrittlement"), the nil ductility transition temperature increase should be limited to about 20°C;
- Total treatment time qualified during manufacturing of the SG is not exceeded.

At the considered temperatures, the stress relief effect is effective but moderately. A residual stress reduction of about 50% was evaluated from available data (8). The main effect is expected from the microstructural improvement.

A research program was launched in two directions:

- Feasibility study of this treatment
- Verification of the microstructural improvement.

5.1 Feasibility

Two approaches are being evaluated. The first involves using electric strip heaters installed around the lower part of the steam generator shell and inside the channel head, while the second involves circulation of a heated gas in the tube bundle as well as use of strip heaters outside the SG.

A SG model was fabricated and stress relieved to provide data for qualification of analytical models and optimization of heating technology.
The model was not to scale but was directed at obtaining information for verification of analytical methods, heating control and temperature monitoring. It included effects of heat losses due to radiation, convection and conduction.

Tests and analysis realized on the mockup indicate that both the electric heater and circulating hot gas/electric heater approaches would be satisfactory. The electric heater approach, without circulating hot gases, currently appears somewhat more favorable because of its simplicity.

This approach is similar to the treatment performed, with another objective, during the SG replacement of Turkey Point and Robinson. The tubesheet would be heated by heaters inside the channel head and the whole channel head itself would be externally heated. Separate controls would be provided for divider plate heaters so as to prevent harmful differential expansion of the divider plate relative to the rest of the head. Heaters would also be installed along the shell of the steam generator for about one steam generator diameter above the tubesheet in order to minimize temperature gradients in the tubesheet and match stayrod and steam generator wall temperatures so as to reduce tube support plate distortion and avoid resulting tube deformations. The stress relief would be performed with a vacuum on the secondary side (e.g., 0.1 bar) in order to reduce convective heat losses.

Tests have been performed on actual steam generator tubing removed from Doel 3 to determine emissivity over the temperature range 100 to 600°C. Based on these tests, an average emissivity of about 0.6 to 0.7 was measured, with various areas ranging from 0.4 to 0.85.

Stress analysis is going on to verify criteria put on temperatures, stresses, residual stresses and strains.

5.2 Microstructural improvement

It is well known that carbide precipitation at grain boundaries associated with inconel thermal treatment (500 to 800°C) provides improved resistance to primary side IGSCC and secondary side caustic attack (9, 10).
It is recognized that the sensitization would increase susceptibility to attack by sulfur species under oxidizing conditions (11). However, the successful operating experience of many plants with sensitized tubing (Tihange 1 for example) makes concerns regarding sensitization less significant, as compared to concerns related to primary side IGSCC of the tubing.

Sensitization tests have been performed for numerous heats (from spare tubes of Doel 3, Doel 4 and Tihange 3). These tests show that about 9 out of 10 heats are sensitized by heat treatment in the 550-610°C range for 10 hours.

Qualification tests to verify the benefit obtained by stress relief are being performed by tests of mockups and Reverse U bends. These involve controlled potential tests at 288°C as well as tests at 360°C in NaOH solutions and in pure water. The pure water tests are not expected to be complete until early next year. The tests include some highly susceptible material, rolled into carbon collar in a manner to provide a "wavy" ID. Some samples will be first exposed to the test environment until cracks appear, then will be stress relieved, and finally will be retested in the accelerated environment. This will provide information regarding the effects of stress relief on propagation of existing cracks.

5.3 Summary

In summary, investigations performed to date indicate that stress relief at 550-610°C is practical to perform, and probably effective in improving resistance to primary side IGSCC. However, both aspects require further confirmation. The Doel 3 and Tihange 2 units are candidate for such a treatment; sufficient information to support a final decision will not be available before early next year.

6. PEENING TECHNIQUES

The objective of the peening techniques is to induce a thin layer (0.2mm) of residual compressive stresses on the inside surface of the tube in order to inhibit IGSCC (12, 13).
The main concern regarding these technique is that excessive peening could increase OD residual stresses in the tube above the top of tubesheet and thus could increase the risk of OD initiated IGSCC.

The adequacy of two techniques was examined:

- Roto-peening which uses tungsten carbide beads, bonded to a fabric in a flapper wheel arrangement, that impinge the tube inner surface
- Shot-peening which uses beads (glass, ceramic or steel) blasted against the tube inner surface.

For the latter technique, glass beads and stainless steel shots were investigated as well as the ceramic beads which are currently favored. Glass beads were discarded because of excessive fragmentation, while stainless steel shots were discarded because the associated higher intensities developed excessive tensile residual stresses at the OD.

Accelerated corrosion tests of rolled specimens were used to evaluate the merits of the techniques (stainless steel in boiling magnesium chloride and sensitized inconel in tetrathionate). Table 2 summarizes the tests performed.

Tests of shot peened specimens indicate that Almen intensities of 4N to 11N are satisfactory, as long as coverage (number of times the surface is covered) is kept close to 100% (tests showed that peening at 9N for a coverage of over 200% caused excessive OD stresses). Intensities higher than 11N induce significant stresses on OD.

The results for roto-peening are similar with a more important scatter for the ID results, probably due to an unsufficient control of the process. The acceptable intensity is about 10A which is similar to the intensity found in shot-peening when adequate corrections are made (magnetic fixing in the Almen device, tube size).

A first evaluation of tools suitable for field use was made. The shot-peening technique can be easily coupled with the tools used for Eddy Current inspection (pusher puller and polar manipulator). If the good
performance of the polar manipulator in dust environment can be assumed the
shot-peening of one tube and the displacement of the peening nozzle to the
following tube takes about 1 minute (Peening nozzles changes and tube
inspection are to be considered separately).

Roto-peening needs a more sophisticated tool with various movements (rotative,
alial, orbital). The peening time of one tube is in excess of 20 minutes;
this leads to the necessity of a multihead tool.

For shot-peening, there is some concern regarding ceramic fragments embedded
in the tube material by the beads. These fragments tend to range in size
around 10 microns and can be numerous. However it is not clear if these
embedments have any significance on the good performance of the tubes.

On the other hand, there is a concern for the residue from shot-peening that
could inadvertently remain in the system.(about 5 tons of beads would be used
for the 3 SG) whereas this problem is reduced for roto-peening by having the
beads attached to fabric (about 10.000 flappers to be used for 3 SG).

A chemical evaluation for primary water compatibility and a study of the
significance of these beads for the bearings and seals in the primary pumps
is required.

For shot-peening the cleaning feasibility has to be demonstrated whereas it
seems not necessary for roto-peening.

In summary, the two techniques have been shown to provide adequate protection
against primary side IGSCC without increased susceptibility to OD IGSCC if
intensity and coverage are limited.

The limited time available before commissioning of Doel 4 and Tihange 3 was
not considered sufficient to clear the uncertainties relative to cleaning
procedures effectiveness and the significance of the ingress of ceramic beads
in the primary reactor coolant. Thus the decision was taken to select the
roto-peening technique.
Further qualification testing of tools suitable for field use and confirmation
of earlier corrosion test results are underway.
If roto-peening is found adequate, it will be applied this year on Doel 4 and
Tihange 3.
Table 1

FINAL ANNEALING CONDITIONS FOR ALLOY 600 TUBING
USED IN THE DOEL AND THIANGE STEAM GENERATORS

<table>
<thead>
<tr>
<th>PLANT</th>
<th>SG</th>
<th>APPROX. PROCESSING DATE</th>
<th>FURNACE TEMP. (°F)</th>
<th>BELT SPEED (FT/MIN)</th>
</tr>
</thead>
<tbody>
<tr>
<td>DOEL 3</td>
<td>R</td>
<td>May 1976</td>
<td>1875</td>
<td>3.0</td>
</tr>
<tr>
<td></td>
<td>G</td>
<td>May 1976</td>
<td>1875</td>
<td>3.0</td>
</tr>
<tr>
<td></td>
<td>B</td>
<td>Oct. 1976</td>
<td>1875</td>
<td>3.0</td>
</tr>
<tr>
<td>THIANGE 2</td>
<td>3</td>
<td>Oct. 1976</td>
<td>1875</td>
<td>3.0</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>Mar. 1977</td>
<td>1925</td>
<td>3.5</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>April 1977</td>
<td>1925</td>
<td>3.5</td>
</tr>
<tr>
<td>DOEL 4</td>
<td></td>
<td>DATA NOT RETRIEVED; MOST LIKELY IDENTICAL TO THIANGE 3.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>THIANGE 3</td>
<td>1</td>
<td>April 1979</td>
<td>1925</td>
<td>3.5</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>April 1979</td>
<td>1925</td>
<td>3.5</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>May 1979</td>
<td>1925</td>
<td>3.5</td>
</tr>
</tbody>
</table>

NOTES:
- NUFFLE FURNACE HOT ZONE IS APPROXIMATELY 25 FEET.
- FROM A BATCH PROCESSED IN OCTOBER 79, THE PEAK MEASURED METAL TEMPERATURE IS 1800 °F DURING 0.7 MINUTES FOR 1950 °F FURNACE TEMPERATURE.

Table 2

ACCELERATED CORROSION TEST MATRIX

<table>
<thead>
<tr>
<th>SHOT-PEENING (LABORELEC)</th>
<th>120 SPECIMENS</th>
</tr>
</thead>
<tbody>
<tr>
<td>TESTING METHOD</td>
<td>Tubing Materials</td>
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<tr>
<td>Boiling MgCl₂</td>
<td>Stainless steel</td>
</tr>
<tr>
<td>Polythionate</td>
<td>Sensitized inconel</td>
</tr>
<tr>
<td>ROTO-PEENING (WESTINGHOUSE)</td>
<td>84 SPECIMENS</td>
</tr>
<tr>
<td>Boiling MgCl₂</td>
<td>Stainless steel</td>
</tr>
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REFERENCES

1. J. Engström, K. Norring, Primary and secondary cracking at Ringhals 2, EPRI Workshop, Nov. 1983, Clearwater Beach.


3. P. Hernalsteen, R. Houben, Preventive and corrective actions for Doel 2 steam generators, NEA/CSNI - UNIPEDE Specialist meeting on steam generator problems, paper 5.6, Oct. 1984, Stockholm.


5. C.M. Owens, A historical view of the importance of the final anneal on primary side SCC resistance of alloy 600 steam generator tubing, EPRI Workshop, Nov. 1983, Clearwater Beach.


7. G. Schucktanz, Inspection findings at U tube steam generators of German PWR, Kerntechnik, n° 5, 1978.


BALANCE OF PLANT PROBLEMS CONNECTED TO STEAM GENERATOR PERFORMANCE by Ivar Multer SSPB

Introduction

Steam generators (SG) in pressurized water reactor (PWR) service were designed for a 30 to 40 years operating life. However, in the 20 years since PWR's began operating commercially, there have been steam generator tube failures which relate to corrosion and have resulted in numerous long plant outages. Steam generator replacement has been required in US PWR's with cost exceeding $100 million per PWR unit.

In order to decrease the corrosion in the SG's, more strict water chemistry guide lines have been implemented. The primary goal is to reduce or eliminate corrosion of the SG internals by reducing the ingress of corrodents and corrosion products. This has been possible due to improved balance of plant systems.

Chemistry considerations

Any steam plant requires that effective water chemistry be maintained to minimize corrosion of steam system components and to prevent the formation of deposits on SG heat transfer surfaces toward this end, several water chemistry parameters must be controlled.

a) To prevent general corrosion of ferrous materials throughout the steam cycle, an elevated pH and minimum oxygen concentration must be maintained in the working fluid.

b) To prevent formation of scale on steam generator tubes, dissolved and suspended impurities must be minimized in the feed water.

c) Since concentrated alkalinity can induce corrosion in steam generator tube materials, free caustic must be eliminated from the steam generator. Chloride is causing denting and pitting.

At the beginning of the seventies a congruent sodium phosphate treatment program was provided for proper pH control. This concept of water treatment was in use to the beginning of 1975. KWU is still using it for their older plants for SG water blow down.
<table>
<thead>
<tr>
<th>Parameter</th>
<th>Westingh.</th>
<th>KWU</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH value at 25°C</td>
<td>8,5-10,6</td>
<td>8,8-9,5</td>
</tr>
<tr>
<td>conductivity uS/cm</td>
<td>10-80</td>
<td>&lt;50</td>
</tr>
<tr>
<td>Phosphate ppm</td>
<td>-</td>
<td>2-6</td>
</tr>
<tr>
<td>Molar ratio Na:Po₄</td>
<td>&lt;2,6</td>
<td>&lt;2,6</td>
</tr>
<tr>
<td>Chloride ppm</td>
<td>&lt;75</td>
<td>&lt;1</td>
</tr>
<tr>
<td>Silica ppm SiO₂</td>
<td>&lt;5</td>
<td>&lt;4</td>
</tr>
</tbody>
</table>

As operational time increased, several types of failures occurred at these operating conditions. They included transgranular stress corrosion, intergranular stress corrosion and mainly tube wall thinning (wastage). The clearly chemistry related problems of tube wastage were related to localized chemical concentration areas and under porous deposits. Many variations of congruent pH phosphate control were attempted to prevent low-pH (acid-phosphate) tube wastage or high-pH (caustic) stress corrosion problems. While bulk water conditions were carefully monitored, there was no way to make an absolute control of local conditions.

Especially US units which were cooled by brackish water (e.g. Turkey Point) and had leaking Al-brass condensers were badly deteriorated by wastage. Also KWU has experienced some tube wastage, especially in Borsele and Stade but to a much lesser extent.

Finally a decision was made by Westinghouse to completely change the chemistry to all-volatile treatment (AVT) control in 1975, succeeded in arresting any further significant wastage by phosphates, but caustic stress corrosion cracking (SCC) has continued to be a concern, particularly in plants with significant periods of phosphate operation before conversion to AVT. Only 3 plants manufactured by Westinghouse are still on phosphate treatment: Robinson 2, San Onofre 1 and Zorita.

The Westinghouse AVT guide lines at that time were

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH 25°C</td>
<td>8,5-9</td>
</tr>
<tr>
<td>Cation cond. uS/cm</td>
<td>&lt;2,0</td>
</tr>
<tr>
<td>Free hydroxide: ppm as CaCO₃</td>
<td>&lt;0,15</td>
</tr>
<tr>
<td>NH₃ ppm</td>
<td>0,25</td>
</tr>
<tr>
<td>Na ppm</td>
<td>&lt;0,10</td>
</tr>
<tr>
<td>Cl ppm</td>
<td>&lt;0,15</td>
</tr>
<tr>
<td>SiO₂ ppm</td>
<td>&lt;1,0</td>
</tr>
</tbody>
</table>

The change of water chemistry to AVT introduced a new problem: denting, which is squeezing of tubes at support plate or tubesheet intersections caused by the corrosion of the carbon steel support plates.
and tubesheet. Approximately 24 Westinghouse and Combustion Engineering plants have reported denting, including eight plants where denting is considered extensive. With the exception of a few plants, all currently operating plants are potentially susceptible to denting if sufficient condenser leakage occurs. Because copper oxide has been demonstrated to be a catalyst in these reactions, those plants with copper in their systems are even more susceptible. The denting problems started activities in US at EPRI and the following waterchemistry specifications have been recommended.

<table>
<thead>
<tr>
<th></th>
<th>Feed water</th>
<th>S.G Blow down</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH (ferrous system)</td>
<td>9.3-9.6</td>
<td>&gt;9.0</td>
</tr>
<tr>
<td>pH (ferrous+Cu syst)</td>
<td>8.8-9.2</td>
<td>8.5-9.2</td>
</tr>
<tr>
<td>Cation. cond. us/cm</td>
<td>&lt;0.2</td>
<td>&lt;0.8</td>
</tr>
<tr>
<td>Na ppb</td>
<td>&lt;3</td>
<td>&lt;20</td>
</tr>
<tr>
<td>Chloride ppb</td>
<td></td>
<td>&lt;20</td>
</tr>
<tr>
<td>Silica ppb</td>
<td></td>
<td>&lt;300</td>
</tr>
<tr>
<td>Total Fe ppb</td>
<td>&lt;20</td>
<td></td>
</tr>
<tr>
<td>Total Cu ppb</td>
<td>&lt;2</td>
<td></td>
</tr>
<tr>
<td>Dissolved O₂ ppb</td>
<td>&lt;3</td>
<td></td>
</tr>
</tbody>
</table>

This specification has been difficult to achieve for the US utilities without condensate polisher plants for the secondary systems.

The Babcock & Wilcox OTSG units have not had any of the above mentioned corrosion problems but have experienced more mechanical and erosion type of failures.

The water chemistry is not important only for the S.G but also the turbine is damaged by poor water chemistry.

Westinghouse turbines associated with PWR S.G have suffered numerous outages because of blade failures, particularly in the low pressure turbine L-1 blade row and adjacent blade rows. The L-1 blade falls within the area of the Wilson line when the steam alternates from wet to dry. In addition to blade failures, certain Westinghouse low pressure turbine discs have also suffered stress corrosion cracking at keyways near the Wilson line. Both chlorine and caustic are causing these failures contributed by oxygen. The corrodents deposit preferentially at the Wilson line.
Feed train design

American S.G vendors are not usually responsible for the design of the rest of the secondary system, as KWU and ASEA-ATOM. It has been unfortunate because it is clear that the corrosion problems occurred because the necessity of considering the integrity and compatibility of the steam-water circuit as a whole was not sufficiently appreciated.

It is quite clear that the condenser is the major component, which can control the environment of the S.G, although there are other sources of contamination. A typical 1000 MW(e) condenser might have 600 km of tubing and over 100,000 expanded or rolled tube to tubesheet joints exposed to an aggressive environment so that occasional leaks are inevitable. Condenser integrity problems vary widely from one plant to another. On some of the best plants chloride levels in the steam generator are a few tens of ppb, but levels over 100 ppb have been common during normal operation. The tendency in Europe is to construct zero leaking (seal welded) condensers.

Both fresh and sea water cooled stations in US have condenser polishing plants (CPP) with resin in the ammonia form. Deep mixed bed plant with resin in the hydrogen form has a greater capacity for salt and is more appropriate for seawater cooled stations. Especially in Japan many plants have consistently maintained low chloride levels in the steam generator blow down: 2-6 ppb, fully comparable with a system with sealwelded condensers. In Japan during every refueling a complete condenser maintenance is performed and tubes with deeper defects than 20% are replaced. In the USA the results have not been so successful and the CPP has created corrosion problems on the S.G tubing. If the condensers are tight, the CPP become the major source of contamination, especially if the regeneration is not properly performed. An excess of sulphate ion over sodium has occured with mixed bed system in the hydrogen form which are regenerated with sulphusic acid.

Condensate powdered resin type filter demineralizers such as Powdex are not recommended for operation in PWR secondary systems for sea water cooling. They rapidly exhaust to the ammonium ion form. In this form, the powdered resin has little capacity for sodium ion removal and most of the sodium ions present will slip through.

Chloride leakage, which causes denting, will also be higher. The Powdex filter is superior to deep bed concerning the ability to remove iron oxides and should be used only in this capacity, e.g for start
up cleaning system. A seal welded condenser costs almost half the price of a full flow deep bed CPP, the operational and maintenance costs not included. Air ingress to the steam generator is also impossible to prevent completely but the amount can vary widely depending on feed train design and method of operation. Full flow steam deaerators are installed only in Japanese and KWU units. The dissolved oxygen content should be kept <3 ppb.

Another very important problem is the material selection of the feed train. Most of the PWR secondary systems now operating contain condenser and heat exchanger tubing constructed of copper alloys. Corrosion products from oxidation of the copper alloys are transported to the S.G when they act both as oxidizer and catalyst in the S.G denting phenomenon. Moreover the presence of copper and copper bearing alloys prevents operation with a secondary system pH above 9.2 which would reduce corrosion of carbon steel. Therefore, not only do the copper alloys corrode, but their presence indirectly leads to corrosion of carbon steel. As a result, even well operated plants containing copper alloys in the secondary system generate sufficient iron oxide corrosion products to prevent them from staying within the 10 ppb iron feed water concentration limit recommended by Westinghouse and Combustion Engineering. This leads to an accumulation of substantial quantities of sludge in the S.G, containing not only copper and copper oxides, but large amounts of iron oxides. Both Millsstone 2 and Indian Point 3 have severe pitting due to leaking condensers, high oxygen level and copper in the sludge. Sleevings has been the temporary solution for both of them.

Improvements of balance of plants

Ringhals 2 (850 MWe), which is a three loop PWR of Westinghouse design, started with PO4 chemistry in 1974. After 3 months of operation, the water chemistry was changed to AVT after cleaning the steam generators. The condensers of Al-brass were cooled by sea water with a Cl- content of 1.2 ppm and the condensate system is not equipped with any full-flow, mixed bed polishers. Compared with other sea water cooled PWR's the Cl- content in the S.G blow down (15 to 30 ppm) was reasonably low, but despite this denting was discovered in the S.G's in 1977. In 1978 operational restrictions concerning condenser leaks were introduced to decrease the rate of denting. The Cl- content in the S.G blow down was limited to 20 ppm. During 1978 many discussions took place on whether to improve the system with full-flow, mixed bed polishers or Ti-seal welded
condenser. Owing to the fact the condensers had to be retubed anyway and that continuous leakage from the polishers in the ammonia cycle can be a problem, a decision was taken in favour of a Ti-seal welded condenser. Other benefits, compared with full-flow polishers, were a shorter delivery time, a lower price and simpler operation. The first condenser was replaced in 1979 and the second in 1980. (1)

Other improvements which have been implemented: recovery of S.G blow down (0.5%), with mixed bed filters and replacing of LP-heater Al-brass tubes with welded 18/8 SS tubes.

Ringhals 2 has been operating without any condenser failure since 1980 and denting has been stopped. As a comparison the water chemistry values from 1977 and 1983 are given in the following table:

<table>
<thead>
<tr>
<th></th>
<th>1977</th>
<th>1983</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH in FW</td>
<td>8.5-9</td>
<td>9.2-9.6</td>
</tr>
<tr>
<td>S.G blow down:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cation cond. US/cm</td>
<td>2</td>
<td>0.2-0.3</td>
</tr>
<tr>
<td>Na ppb</td>
<td>1-3</td>
<td>1</td>
</tr>
<tr>
<td>Cl ppb</td>
<td>15-30</td>
<td>&lt;5</td>
</tr>
<tr>
<td>Fe in FW ppb</td>
<td>20-50</td>
<td>4-5</td>
</tr>
<tr>
<td>Cu in FW ppb</td>
<td>1</td>
<td>0</td>
</tr>
</tbody>
</table>

Ringhals 4 started with Ti-condensers from the beginning. For R3 the hot functional test was performed with Al-brass condenser, but before nuclear operation the condenser was completely rebuilt to a seal welded Ti-condenser. All Al-brass tubing in LP-heaters was changed to 18/8 SS welded tubes in order to avoid Cu in the system before start up in both stations. In R2 it was changed one year ago.

Due to the organic colloids in the make-up water, the S.G blow down cation conductivity was 2 µS/cm in R2. In 1980 a system for clean up and re-use of blow down water was introduced and the cation conductivity decreased to 0.3 µS/cm.

Both R3 and R4 have the same clean-up system and the water chemistry values are as good as in R2 after modification. In order to decrease the iron content during start up, R3 and R4 have been fitted out with 50% capacity by pass Powdex filters.

The water chemistry values have been excellent and no condenser leaks have occurred after introducing the Ti-seal welded condensers.
As it was mentioned in the chapter "chemistry considerations", an elevated pH and minimum oxygen concentration is necessary to prevent general corrosion of ferrous materials throughout the steam cycle.

KWU's S.G had some wastage problems in the sludge layer above the tubesheet with PO₄ dosage. This problem forced KWU to change their water-chemistry to AVT.

KWU went the whole way with tight condensers and increased pH to 10 in order to minimize the corrosion products. (2) Many of the corrosion products passing into the feed water stem from erosion-corrosion processes in the plant sections and pipes which carry wet steam. Erosion-corrosion only occurs when passivity is not present (i.e when chrome components are lacking in steel) and a protective coating must be formed for the metal to be resistant.

Tests performed by KWU (2) show (fig 1,2) very clearly the influence of pH, temperature and material composition on erosion-corrosion. Alkalisation to a pH level of >9.5 or an oxygen content of >150 ppb can extensively prevent erosion-corrosion even in carbon steels. In order to operate at such a high pH-value all Al-brass condenser tubes have to be replaced by SS for fresh water cooling as in Biblis and Grafenrheinfeld.

Grafenrheinfeld was fitted out with Al-brass condenser but after that the water chemistry was changed to AVT, the condenser was retubed with SS.

The water chemistry values for Grafenrheinfeld are the following:

<table>
<thead>
<tr>
<th>S.G blow down</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH</td>
</tr>
<tr>
<td>cat uS/cm</td>
</tr>
<tr>
<td>Cl ppb</td>
</tr>
<tr>
<td>Silica ppb</td>
</tr>
</tbody>
</table>

Iron content in feed water is only 1-2 ppb.

These excellent values, specially iron content, can be compared with other PWR's in Germany on fig 3.

KWU's philosophy is to improve the system so much that the iron content in the feed water will be less than 1 ppb. For this reason cartridge and electromagnetic filters will be used in Isar-2 and Philipsburg.
The selection of materials for cross over piping, moisture separator (MS) and HP-drains is very important from the erosion point of view. At several PWR stations in US and also in Europe, the HP heater drains which are pumped forward, contributed with 20 to 30% of the total iron concentration of the feed water, 20 to 40% of the nickel, and 30% of the copper. Measurements at Prairie Island Unit 2, which has an all-ferrous secondary system, indicate that 70% of the total iron content in the HP heater drains was contributed by the MS drains. Even with a high rate of corrosion product removal in the condensate demineralizer, it is difficult to meet the Westinghouse and Combustion Engineering limit of <10 ppb total iron in the feed water for recirculating type steam generators. Even Genkai Unit 1 and also Doel, which are considered to be well operated plants incorporating a condensate demineralizer but pumping all drains forward have a feed water iron concentration 10-15 ppb during normal operation. Due to the fact that condenser tubes are of copper alloy, the pH-value will be limited to 9-9.2.

In France where most of the plants are river-water cooled, the condensers are of Brass 70/30 and a lot of LP-preheaters are also of the same material. Due to the limited pH-value, EDF has some erosion problems in the steam phase. In order to decrease the erosion problems morpholine (pH 9.1-9.3) is added.

As mentioned before MS is a source for corrosion product and MSR-tubes of 439 (19% Cr) have been introduced in France. They are superior to 18/8 tubes and not sensitive to stress corrosion cracking as 18/8.

Besides Doel only in Lovisa (Finland) CPP is used. Lovisa is operating with neutral water pH 7. In order to decrease the corrosion of carbon steel in MSR's and steam piping, 25-30 ppb O₂ is injected. Lovisa 2 operating normally only with precoat filter has less than 5 ppb Cl⁻ and Na. Iron content in the FW to the S.G is 10-15 ppb.

Conclusions

To obtain the minimum content of impurities in the system the following measures should be taken:

Tight Ti or SS condensers, if possible they should be seal welded.

High pH 10 and ferritic system, with erosion resistant material.
No condensate polishing plant (mixed bed). Only cartridge or electromagnetic filters for removal of iron. Start up cleaning system is very useful.

Feed water tank or other means for good deaeration.

Improved make-up water systems and recovery of S.G blow down.

References


SESSION 5.9: PREVENTIVE AND CORRECTIVE ACTIONS AGAINST DEGRADATION

Nobuaki Mori
Japanese Ministry of International Trade and Industry (MITI)

1. INTRODUCTION

In order for us to maintain the integrity of steam generators, it is important to implement the preventive and corrective actions against the recurrence of those experience cases of tube degradation as well as to deter the propagation of tube degradation. Accordingly, in Japan whenever any tube degradation is first detected, all the operating units will be reviewed and inspected, and the preventive and corrective actions will, if deemed necessary, be implemented. Presently, preventive and corrective actions for the operating units are performed as follows:

(1) Discharge of impurity by strict water chemistry control
(2) Repairing of degraded tube
(3) Structural improvement of regions subjected to degradation
(4) Tube inspection
2. WATER CHEMISTRY CONTROL

The water chemistry of secondary side has been changed to all volatile treatment since 1974 after tube wall wastage due to localized concentration of sodium phosphates was detected. Because of the fact that all volatile treatment tends to degrade the tube when such impurities as oxygen, sodium and chloride exists in the system, the secondary side has been equipped with the leak tight condenser, condensate polisher and deaerator.

(Table 1)

As the maintenance work, the defect of the condenser tube will be detected by the eddy current test during plant outage and defect tubes are plugged.

And also the leak tightness of condenser is confirmed by the air leak test.

The water chemistry control guideline is specified to meet with ALAP concept.

In fact, monitors are installed at each point of the secondary system so that the adequate action can be taken as soon as water chemistry is deviated off the spec. (Table 1)

The sludge lancing is executed to remove the sludge piled on the tubesheet and the hot water flushing is executed to remove the slight soluble impurities such as sodium during plant outage period.

3. REPAIRING AND STRUCTURAL IMPROVEMENT

The plugging and the sleeving are adopted as the repair work for degraded tubes.

Explosive plug or welding plug was used in early years.
Mechanical plug has been adopted since 1981 to reduce the amount of radioactive radiation and to shorten the repair time. The sleeving technique was developed in 1980 and has been applied to the repair of secondary side IGA/SCC at the tubesheet crevice. The technique has also been applied to the repair of primary side SCC at the expanded area in tubesheet. (Fig. 2) The mini-sleeving process (braze welding) has also been developed for the repair of secondary side IGA/SCC at the tube support crevice, and the verification test is now proceeding.

For the preventive maintenance work against degraded tube, zone plugging and the crevice blockage for partially expanded tubes were executed.

It is too difficult to improve the degraded tube caused by the high residual stress during tubing such as primary side SCC at tight U-bend, therefore, plugging was executed to all row-1 tubes of Takahama Unit 1 and all row-1 and -2 tubes of Ohi Unit 1.

As the blockage of tubesheet crevice, hydraulic and mechanical rerolling were performed for the rest of partially expanded area of Mihama Unit 2, Takahama Unit 1 and Unit 2, and, therefore, all crevices at hot leg sides have just eliminated from 1981 to 1984. (Fig. 3)

4. PRACTICAL PROCEDURE

When the preventive and corrective work will be executed, the electric power company must apply to the regulatory authority for the construction permit.

The authority technically judges the documents in accordance with codes and rules, and gives the construction permit. After the works done, they are authorized by severe inspection by the authority.
In principle, sampling inspection is applied to in-service tube inspection, but, for the unit in which tube defect was detected during operation, 100% tubes are inspected every year. When defected tubes are found through ECT inspection, we evaluate whether the defect may cause leaking or not during plant operation and then decide countermeasures for the defected tubes. But, in fact, it is so difficult to make quantitative judgements on the defect indications, therefore, all the tubes that have defect indications are usually plugged or sleeved.
### TABLE - 1

**INSTALLATION STATUS OF CONDESATE POLISHER AND DEAERATOR**

<table>
<thead>
<tr>
<th>Location</th>
<th>Condensate Polisher</th>
<th>Deaerator</th>
</tr>
</thead>
<tbody>
<tr>
<td>MIHAMA - 1</td>
<td></td>
<td>○</td>
</tr>
<tr>
<td>MIHAMA - 2</td>
<td></td>
<td>○</td>
</tr>
<tr>
<td>MIHAMA - 3</td>
<td>○</td>
<td>○</td>
</tr>
<tr>
<td>TAKAHAMA - 1</td>
<td></td>
<td>○</td>
</tr>
<tr>
<td>TAKAHAMA - 2</td>
<td></td>
<td>○</td>
</tr>
<tr>
<td>OHI - 1</td>
<td>○</td>
<td>○</td>
</tr>
<tr>
<td>OHI - 2</td>
<td>○</td>
<td>○</td>
</tr>
<tr>
<td>IKATA - 1</td>
<td>○</td>
<td>○</td>
</tr>
<tr>
<td>IKATA - 2</td>
<td>○</td>
<td>○</td>
</tr>
<tr>
<td>GENKAI - 1</td>
<td>○</td>
<td>○</td>
</tr>
<tr>
<td>GENKAI - 2</td>
<td>○</td>
<td>○</td>
</tr>
<tr>
<td>SENDAI - 1</td>
<td>○</td>
<td>○</td>
</tr>
</tbody>
</table>
| TABLE - 2

STANDARD AND ACTUAL VALUE OF
SECONDARY WATER CHEMISTRY

<table>
<thead>
<tr>
<th>Condensate</th>
<th>Standard</th>
<th>Actual (Sample)</th>
</tr>
</thead>
<tbody>
<tr>
<td>CATION CONDUCTIVITY</td>
<td>(&lt;μS)</td>
<td>&lt; 0.2</td>
</tr>
<tr>
<td>DISSOLVED OXYGEN</td>
<td>(ppb)</td>
<td>&lt; 50</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Feed Water</th>
<th>Standard</th>
<th>Actual (Sample)</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH</td>
<td>8.8 ~ 9.3</td>
<td>9.1</td>
</tr>
<tr>
<td>CONDUCTIVITY</td>
<td>(&lt;μS)</td>
<td>&lt; 5</td>
</tr>
<tr>
<td>DISSOLVED OXYGEN</td>
<td>(ppb)</td>
<td>&lt; 5</td>
</tr>
<tr>
<td>TOTAL IRON</td>
<td>(ppb)</td>
<td>&lt; 20</td>
</tr>
<tr>
<td>TOTAL COPPER</td>
<td>(ppb)</td>
<td>&lt; 5</td>
</tr>
<tr>
<td>TOTAL NICKEL</td>
<td>(ppb)</td>
<td>&lt; 5</td>
</tr>
<tr>
<td>HYDRAZINE</td>
<td>(ppb)</td>
<td>&gt; 2</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Steam Generator Blowdown</th>
<th>Standard</th>
<th>Actual (Sample)</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH</td>
<td>8.5 ~ 9.1</td>
<td>9.0</td>
</tr>
<tr>
<td>CONDUCTIVITY</td>
<td>(&lt;μS)</td>
<td>&lt; 5</td>
</tr>
<tr>
<td>CATION CONDUCTIVITY</td>
<td>(&lt;μS)</td>
<td>&lt; 2.0</td>
</tr>
<tr>
<td>SILICA</td>
<td>(ppb)</td>
<td>&lt; 500</td>
</tr>
<tr>
<td>CHLORIDE</td>
<td>(ppb)</td>
<td>&lt; 100</td>
</tr>
<tr>
<td>SODIUM</td>
<td>(ppb)</td>
<td>&lt; 100</td>
</tr>
<tr>
<td>FREE ALKALI</td>
<td>(ppb)</td>
<td>&lt; 150</td>
</tr>
<tr>
<td>TURBIDITY</td>
<td>(ppm)</td>
<td>&lt; 1</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Steam</th>
<th>Standard</th>
<th>Actual (Sample)</th>
</tr>
</thead>
<tbody>
<tr>
<td>SILICA</td>
<td>(ppb)</td>
<td>&lt; 20</td>
</tr>
</tbody>
</table>
FIG. 2 SLEEVING IN TUBE SHEET

Steam Generator Water Chamber

Type-1 Sleevings

Type-2 Sleevings
FIG. 3 CREVICE ELIMINATION BY RE-ROLLING

Steam Generator
Water Chamber

Hydraulic Expansion
Re-Rolling
SESSION 6

STEAM GENERATOR

REPLACEMENT
SUMMARY OF SESSION 6 - STEAM GENERATOR REPLACEMENT

Session Chairman: Mr. B. Agrenius

This session dealt with two different topics; SG-replacement and aspects on SG-design.

The presentations concerning SG-replacement were made by vendors who had the over-all responsibility for the whole replacement job.

The papers show that it is possible to replace SG:s within a time of 4-6 months if careful planning and preparation is performed.

Different methods of replacement were used in Obrigheim and Point Beach, reflecting the strong influence of containment lay-out and design on the replacement method. Two important advances in replacement technology took place at those plants.

In Obrigheim, a computer controlled welding machine with a TV monitor was used successfully for the reactor coolant pipe welds. In Point Beach, special templates were developed and used which made it possible to reduce the number of R.C. pipe welds two per steam generator.

The dose figure of about 300 manrem per steam generator confirms the trend of decreasing collective dose figures.

The two papers show that important know-how and experience exist in this field, which should be considered by those being close to replacing SG:s.

The other two papers discussed various aspects of new SG-design. The first paper points out important aspects of SG design improvements, particularly in the areas of thermo-hydraulic and mechanical design.
The second paper considered the concept with SG:s equipped with economizers. It is evident that the introduction of economizers improves the thermal efficiency but it also shows that thorough design evaluation, analyses and testing is required.
SESSION 6.1

DESIGNING AN IMPROVED U-TUBE STEAM GENERATOR FOR REPLACEMENT PURPOSES

M. Sala ANSALDO COMPONENTI DGV
R. De Santis ANSALDO COMPONENTI DGV

ABSTRACT

The replacement of many operating PWR Steam Generators is becoming an operational necessity and Utilities facing this problem are steadily increasing. Providing a replacement Steam Generator requires combined efforts of an experienced Manufacturer and of a System Engineering Company, both cognizant of the operating plant principles. Basic requirements for a replacement Steam Generator are:

- to provide the same or a greater thermal power output than the replaced one
- to provide an increased reliability
- to provide system and site compatibility.

Modern design cognizances and capabilities allow to reach the above mentioned scope by means of:

- more corrosion resistant materials
- improved mechanical design of the component
- improved fabrication technologies
- improved secondary side flow-distribution
- minimization of secondary side corrosion
- control of vibration & wear phenomena
- improvement of moisture separation
- improved inspectability and maintainability.

Ansaldo DGV in cooperation with ENEA has set-up an extensive Steam Generator Research and Development program, with relevant test rigs, including all aspects from Thermal-Hydraulic behaviour of U tube Steam Generators to improved design and manufacturing methods and technologies.

As a result of this program a complete component design capability has been reached so as to allow ANSALDO DGV to meet any Utility need; a new Steam Generator
Design, which could be adapted also to SG replacement, has been developed including specific features like:
- a new tube bundle support system (bi-structural grid)
- improved primary moisture separators.

INTRODUCTION

Costs associated with SG replacement are so high (113 $ million were spent from VEPCO to replace generators on Surry 1 and 2 - each 775 MWe W PWR's on line 1972 and 1973) that replacement, when necessary, can be considered only once in the plant life.
It is however imperative that the replacement SG incorporates the most advanced solutions which guarantee a satisfactory behaviour of the component for the residual plant life.
Any SG replacement has a major or minor impact on the NSSS system, depending upon importance of changes introduced in respect to the first model; the impact on the system, mainly in terms of Safety Evaluation, must be carefully investigated prior to a final decision.
Design features of A - DGV SG have been developed during extensive analytical and large scale experimental activities under progress since 1977 under a cooperation agreement between ANSALDO and ENEA (1).
The SG characteristics of A - DGV are compared with the W model 51 which is installed in most plants facing replacement problems.

DESIGN BASIS

The A - DGV design incorporates the main improvements currently provided by all manufacturers and some specific features, that is:
a) current improvements:
   - Additional Heat Transfer Area
   - Higher margins for tube plugging
   - Flow Distribution Baffle
   - Tubelane blockage device
- J-nozzle feedwater distribution ring and improved feedwater distribution concept
- Increase of the minimal U bend radius
- Increased number of maintenance and inspection accesses
- Full depth expansion of tubes in the tube - sheet

b) **Specific features**
- New tube bundle support system (bi-structural grid)
- Improved primary moisture separators
- Channel - head in an integral piece with hot extruded nozzles.

The combination of the above design improvements with the high standards of fabrication of A-DGV are considered capable to provide a reliable SG free from the currently known troubles.

**THERMAL HYDRAULIC DESIGN**

The TH design objective has been that of providing:
- minimal impact on the system (by preserving, as much as possible, both primary and secondary side volumes as well as primary side pressure drops)
- increase of performance margins (by assuming reasonable values for plugging factor and fouling resistance)
- increase of thermal power with the same secondary steam pressure.

The consequent increase of heat transfer surface has been achieved by designing a more compact bundle with increased tube number, reduced tube diameter, and reduced pitch to diameter ratio. The square pitch has however been maintained for not departing too markedly from the existing design.

The use of a square pitch has been allowed by the availability of a fully engineered new tube support system (bi-structural grid), characterized by very low pressure drops and almost zero dry-out. This tube support system obviates to the drawbacks on the circulation ratio of the increased thermal power and of the higher resistance of the more efficient moisture separators.

Tables 1 and 2 provide, respectively, the geometric and TH characteristics of the A-DGV SG compared to the reference W model 51.
TABLE 1
Comparison of the Geometrical Characteristics of the Replacement S.G.
and the reference Model 51

<table>
<thead>
<tr>
<th>ITEM</th>
<th>UNIT</th>
<th>REPLACEMENT S.G.</th>
<th>REPLACEMENT MODEL 51</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tube Material</td>
<td></td>
<td>Alloy 690</td>
<td>Inconel 600 MA</td>
</tr>
<tr>
<td>Tube O.D. ((d_o))</td>
<td>mm</td>
<td>19.05</td>
<td>22.225</td>
</tr>
<tr>
<td>Nominal Wall Thickness</td>
<td>mm</td>
<td>1.092</td>
<td>1.27</td>
</tr>
<tr>
<td>Tube Pitch ((P))</td>
<td>mm</td>
<td>24.9</td>
<td>32.54</td>
</tr>
<tr>
<td>Pitch/diameter Ratio</td>
<td>--</td>
<td>1.31</td>
<td>1.46</td>
</tr>
<tr>
<td>Ligament ((P-d_o))</td>
<td>mm</td>
<td>5.85</td>
<td>10.315</td>
</tr>
<tr>
<td>Tubesheet Thickness</td>
<td>mm</td>
<td>710.</td>
<td>550.</td>
</tr>
<tr>
<td>Number of U-Tubes</td>
<td></td>
<td>4996.</td>
<td>3388.</td>
</tr>
<tr>
<td>Installed Transfer Area</td>
<td>m²</td>
<td>4623.</td>
<td>4023.</td>
</tr>
<tr>
<td>Outer Tube Limit ((OTL))</td>
<td>mm</td>
<td>2973.</td>
<td>3117.</td>
</tr>
<tr>
<td>Minimum Bend Radius</td>
<td>mm</td>
<td>87.15</td>
<td>67.</td>
</tr>
<tr>
<td>Top of Tube Bundle</td>
<td>mm</td>
<td>8140.</td>
<td>8960.</td>
</tr>
<tr>
<td>Straightened Active Length of an &quot;average&quot; U-tube</td>
<td>mm</td>
<td>15460.</td>
<td>17006.</td>
</tr>
<tr>
<td>Tube Support System</td>
<td></td>
<td>7 Grids + 1 FDB</td>
<td>7 Drille Hole Plates</td>
</tr>
<tr>
<td>Primary Volume at BOL/EOL(*)</td>
<td>m³</td>
<td>28.13/26.25</td>
<td>27.99/27.39</td>
</tr>
<tr>
<td>Secondary Volume</td>
<td>m³</td>
<td>157.2</td>
<td>156.8</td>
</tr>
<tr>
<td>Moisture Separator I.D.</td>
<td>mm</td>
<td>495.</td>
<td>1422.</td>
</tr>
<tr>
<td>Moisture Separator Number</td>
<td>--</td>
<td>16</td>
<td>3</td>
</tr>
<tr>
<td>Tube Bundle Weight</td>
<td>kg</td>
<td>42.3x10³</td>
<td>43.3x10³</td>
</tr>
<tr>
<td>Lower Shell ID</td>
<td>mm</td>
<td>3286</td>
<td>3286</td>
</tr>
</tbody>
</table>

(*) BOL = Beginning of Life (≠ No Tubes Plugged, No Fouling Resistance)
EOL = End of Life (≠ Design Tube Plugging, Design Fouling Resistance)
<table>
<thead>
<tr>
<th>ITEM</th>
<th>REPLACEMENT S.G.</th>
<th>REFERENCE MODEL 51</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>107%</td>
<td>100%</td>
</tr>
<tr>
<td>Thermal Power (MW)</td>
<td>870.</td>
<td>813.</td>
</tr>
<tr>
<td>Hot Leg Temperature (°C)</td>
<td>324.611</td>
<td>324.611</td>
</tr>
<tr>
<td>(from the specification)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Primary Flowrate (kg/s)</td>
<td>4212.11</td>
<td>4212.11</td>
</tr>
<tr>
<td>Steam Pressure at Nozzle Outlet (bar)</td>
<td>59.8</td>
<td>62.9</td>
</tr>
<tr>
<td>Steam Pressure in Bundle (bar)</td>
<td>60.6</td>
<td>63.7</td>
</tr>
<tr>
<td>Circulation Ratio (-)</td>
<td>3.5</td>
<td>3.9</td>
</tr>
<tr>
<td>Design Tube Plugging (%)</td>
<td>10.</td>
<td>10.</td>
</tr>
<tr>
<td>Fouling Resistance (m²K/W)</td>
<td>1.76x10⁻⁵</td>
<td>1.76x10⁻⁵</td>
</tr>
<tr>
<td>Primary Side Pressure Drops at:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>BOL (No Plugging) (bar)</td>
<td>1.53</td>
<td>1.53</td>
</tr>
<tr>
<td>EOL (Design Plugging) (bar)</td>
<td>1.82</td>
<td>1.82</td>
</tr>
</tbody>
</table>

(*) Results of DGV thermal-hydraulic analysis
MECHANICAL DESIGN

The mechanical design of A-DGV SG is characterized by:
- channel head forging with extruded primary nozzle
- full depth expansion of tubes in the tube-sheet
- increased hand-holes number on the secondary side to allow access to both the tube-sheet and flow distribution baffle for purpose of inspection and sludge lancing
- use of seven bistructural grids at different elevations
- use of Inconel 690 tubing with stress relieving of all bends with radius lower than 300 mm.

BISTRUCTURAL GRID

The tube support system specific of A-DGV SG is the product of a Joint Development Agreement between W ENEA and A-DGV.

The grid principle is outlined in figure 1. The basic concept is to withstand the seismic loads by using bars of equal strength containing a structural part and a support part; the tube is in contact with low resistance tabs which yield before the tube deforms in case of denting occurrence. Hooks located every ten tube pitches avoid vertical relative movements of the two sets of bars while guides prevent horizontal sliding. The bars are made by a punching process followed by stress relief. Materials were selected on the basis of corrosion and wear concerns: the bars are made by Incoloy 800 for coupling with Inconel tubes. Inner and outer rings are also made by Incoloy 800 to match the thermal expansion coefficient with the bars. Vertical restraints are provided by stayrods, while jacking blocks connect the grid to the shell. Figure 2 provides an idea of the bars assembling in the manufactured grid prototype (fig. 3).
Pressure loss coefficients have been measured in single phase conditions; the results reported in table 3, confirm the favourable effect on the circulation ratio expected by using these supports. Table 4 summarizes the effects of the grid in preventing tube damages.

**Table 3 - PRESSURE LOSS COEFFICIENT FOR SUPPORTS**

<table>
<thead>
<tr>
<th>Type of support</th>
<th>Pressure loss coefficient</th>
</tr>
</thead>
<tbody>
<tr>
<td>Drilled plate</td>
<td>9.0</td>
</tr>
<tr>
<td>Broached plate (quaterfoil)</td>
<td>3.5</td>
</tr>
<tr>
<td>Grid for triangular pitch (KWU type)</td>
<td>2.3</td>
</tr>
<tr>
<td>Grid for square pitch (bi structural)</td>
<td>1.5</td>
</tr>
</tbody>
</table>

**Table 4 - EFFECTS OF GRID FOR PREVENTING THE TUBE DAMAGE**

<table>
<thead>
<tr>
<th>Typical feature</th>
<th>Effects</th>
</tr>
</thead>
<tbody>
<tr>
<td>- Large flow area</td>
<td>- avoids retention of concentrates at supports</td>
</tr>
<tr>
<td>- Low pressure losses</td>
<td>- enhances the circulation ratio</td>
</tr>
<tr>
<td>- flat land contact geometry between tube and support</td>
<td>- reduces load on the tube support</td>
</tr>
<tr>
<td>- elasticity</td>
<td>- avoids dryout and denting; reduces wear</td>
</tr>
<tr>
<td></td>
<td>- allows local deformation.</td>
</tr>
</tbody>
</table>
PRIMARY MOISTURE SEPARATORS

An extensive experimental development work has been carried out on three test rigs (1) namely:

LARA                  air water loop, ambient pressure, low scale
ARAMIS                air water loop, low pressure, full scale
(fig. 4)
GEST-SEP              water steam loop reproducing in full scale
the environment of the moisture separator
(fig. 5)

Based on experimental results A - DGV has developed some separator prototypes characterized by high separation efficiency which can be utilized in existing internal arrangement providing beneficial effects (patent pending). Use of secondary improved dryers will further enhance overall separator efficiency.
In addition measures are foreseen to avoid jet impingement which can lead to local dryer overloading.

REFERENCES

FIG. 3 - PROTOTYPE GRID UNDER MANUFACTURING
FIG. 4: General view of ARAMIS loop. The facility consists of a water line and of an air line which merge in a mixing device and in a tank containing the test section. It operates in scaled working conditions ensuring similar separator performance as in a real SG.
FIG. 5: View of GEST-SEP loop. The facility consists of an adiabatic closed loop with a steam line joining a water line in a mixer to produce the two phase mixture. The full scale separator is located into a vessel (length 12.5 m outside diameter 2.2 m) where the real SG environment is reproduced.
SESSION 6.3: EXPERIENCE WITH KWU STEAM GENERATORS
PART 1
STEAM GENERATOR REPLACEMENT
AT THE NUCLEAR POWER PLANT OBRIGHEIM

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Kraftwerk Union AG, Erlangen
Federal Republic of Germany

1. Introduction
The two steam generators of the 345 MW_e Obrigheim power plant which, after 4 years of construction, went into operation in 1968, were built in accordance with now obsolete design standards which specified INCONEL 600 tubing. As early as in 1971, corrosion cracking had occurred in the steam generator tube bundles at the tubesheet region and the narrow U-bends. Extensive investigations concluded that replacement of these obsolete steam generators was advisable. Two new steam generators designed in accordance with KWU standards were manufactured and supplied to the site in 1976.

After about fifteen years of operation, the stress corrosion problems increased to a point where replacement of the two steam generators became necessary.

2. Planning of the Steam Generator Replacement
The decision to replace the steam generators was made after the outage in 1982. At that time, it was decided to replace the steam generators during an extended Refueling outage in the time from June 4, 1983 - to September 9, 1983.
In September 1982 the Kraftwerk Union AG (KWU) was awarded the contract for the replacement and the overall responsibility for the planning.

The power plant design concept made it possible to remove and reinstall the steam generators as a complete unit without major alterations to the containment inside or the building structure.
(Fig.1 and 2)
The equipment hatch had to be removed and replaced by a larger temporary erection hatch. The lifting capacity of the containment building crane was sufficient to carry the steam generators (Weight of the old steam generators 168 T. - Weight of the new steam generators 158 T. Lifting rig and lifting support equipment 23 T).

In addition to the crane a special lifting device had to be employed to lift and permit tilting of the steam generators in a horizontal position. To protect the fuel storage pool from possible outside contamination and crashing parts, a cover of adjoining steel girders and a layer of steel plates was assembled.

Additionally, the pool had to be protected against a hypothetical steam generator crash or a crane failure during the steam generator transport across the fuel storage pool. The time for preparation, a period between the decision to replace the steam generators and the actual replacement was relatively short. During this time the following activities had to be performed:

- Activities for commissioning and licensing.
- Planning and construction of storage facilities for the old steam generators with a space of approximately 250 m².
- Construction of a building to temporary storage of parts removed from the containment. (Storage space approximately, 600 m²).
- Construction of a building for preparatory activities to be performed on the new steam generators.
- Preparatory work activities on the new steam generators such as:
  - Sand blasting to remove the protective coating, grinding of the welds, inservice inspections at selected areas, installation of sniphon, at the feedwater nozzles, etc.
- Manufacturing of eight new main coolant pipe subsections.
- Manufacturing of several main steam- and feedwater pipe sections.
- Operational planning incorporating the new steam generators.
- Development of a suitable decontamination method (electro polishing) to decontaminate the main coolant pipes, including the fabrication and functional testing of the manipulators.
- Qualifying the automated and remote controlled welding operation.
- Training of personnel employing a full scale steam generator channel head mockup to test the handling of the required equipment.
- Fabrication of the lifting and support straps for the old and the new steam generators.
- Fabrication and functional testing of the lifting rig.
- The construction of a new cafeteria for this outage to serve approximately 900 people.
- Social facilities for the additional personnel had to be created.

66 house trailers were installed on site.

- Preparatory measures for the health physics coverage.
- Enlargement of the entrance to the controlled area.

The power plant was shut down on June 4, 1983. After opening and defueling of the reactor, the equipment hatch was removed and a temporary erection hatch installed permitting a directed air flow to the containment.

The disassembly activities began inside the containment, starting with the dismantling of the steam generator and main coolant pipe insulations, the removal of the steam generator missile shield compartment slaps, platforms and ventilation ducts.

The shielding was brought in to the containment, auxiliary scaffolding was erected, a survey of the old steam generators performed. The anti-tilting protection was removed. Parallel to these activities, the cut off of the connecting pipes to the main steam-, the feedwater pipes and the auxiliary systems began.
The steam generators were securely fastened to the containment building crane and the support straps before the main coolant pipes were cut. After cutting-off the main coolant pipes and the removal of the steam generator supports, the steam generator was lifted out of the containment and placed on a heavy load transport vehicle positioned at the entrance of the equipment hatch for transport to the storage facility. The steam generator outside surfaces were cleaned and nozzle covers welded in to place inside containment.

Sections of the remaining main coolant pipe ends and segments of the pump loops were subsequently decontaminated utilizing an electropolishing method. The remote controlled manipulators and the associated electrolyt recirculation system used for the electropolishing decontamination have been successfully employed. The decontamination factor was above 200. The positive decontamination factor significantly improved the subsequent erection activities. The radiation level inside the main coolant pipes was reduced from approximately 10,000 mR/h to approximately 50 mR/h. Outside the main coolant pipes a radiation level in the range of 500 mR/h to 1000 mR/h was measured prior to decontamination. After decontamination and the installation of appropriate shieldings, the level was reduced to 50 mR/h. In comparison to the steam generator removal, the replacement was performed in the reversed order of events. Extensive optical surveys and mechanical measurements were necessary to incorporate the new components into the already existing system. A successful fitup was performed. The main coolant pipes were placed within a clearance of less than .5 mm/.0197 inch. Steam generator removal and steam generator replacement inside containment took approximately five weeks. It took another five weeks, up to the middle of August, until all the welds were completed. To weld the main coolant pipe circumferential welds an automatic remote controlled pulsed are GTA orbital welding method was employed. Welding was performed from the outside without a backup weld.

* Fig. 3 time schedule
Fig. 4 personnel participation
On the primary loops, a total of 14 circumferential welds were made. The average welding time for one complete weld amounted to 100 hours. 130 passes were necessary to complete a weld with a wall thickness of approximately 50 mm at a weld rod diameter of 0.8 mm. At the secondary side, 10 circumferential welds at the main steam pipe area and 18 circumferential welds at the feedwater area were completed. (Fig. 5). Including the signal lines and auxiliary systems, (167 welds on the primary side and 275 welds on the secondary side) a total amount of 484 circumferential welds have been prepared, welded and examined. Reinstallation and reassembly of all other building sections and plant parts, such as the equipment hatch, platforms, air vent ducts, anti-tilting safeguards etc. dismantling of erection aids, decontamination and removal of all the equipment from the containment were performed simultaneously with the welding of the remaining pipe works.

After completion of all examinations and heat treatments, the plant was prepared for the pressure test. The pressure test on the secondary side, (86.5 bar) took place in the time from August 13- to 15-1983, the pressure test at the primary side (224 bar) was successfully completed on August 14- to 17-1983. During the pressure test KWU performed acoustic emission measurements on the primary and the secondary side.

3. Radiation Exposure

The radiation exposure for the personnel envolved during the steam generator replacement amounted to 690 rem, this was considerably lower than previously calculated. Fig. 6, depicts the accumulated radiation exposures at the various work activities. In addition to that, 100 rem have to be added for monitoring and supervisory activities not directly related to the actual work activities.
4. Conclusion

Experience gained by replacing the two steam generators in the Obrigheim power plant has proven that it is possible to exchange completely assembled steam generators within 10 weeks. Only if transportation conditions within the reactor building absolutely do not allow such an exchange of completely assembled steam generators, should the method of removing old and installing new steam generator in sections be considered. A main factor for the success in Obrigheim was the very close working relationship between the plant personnel and KWU as overall supplier and contractor. This working relationship began in the initial planning phase, about one year before the unit was shut down.
<table>
<thead>
<tr>
<th>Month</th>
<th>June '83</th>
<th>July '83</th>
<th>August '83</th>
<th>September '83</th>
</tr>
</thead>
<tbody>
<tr>
<td>Weeks</td>
<td>22</td>
<td>23</td>
<td>24</td>
<td>25</td>
</tr>
<tr>
<td></td>
<td>26</td>
<td>27</td>
<td>28</td>
<td>29</td>
</tr>
<tr>
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<td>30</td>
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<td></td>
<td>34</td>
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<td>36</td>
<td>37</td>
</tr>
<tr>
<td></td>
<td>38</td>
<td>39</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1 Plant shut down</td>
<td></td>
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</tr>
<tr>
<td>Fuel removal</td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>2 Equipment hatch</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>removal</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3 Auxiliary hatch</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>installation</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>4 Removal of steam generator No 2</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>5 Removal of steam generator No 1</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>6 Decontamination of main coolant pipes</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>7 Replacement of stg. No 1 including pipe installation</td>
<td>4.6.83</td>
<td>6.6.83</td>
<td></td>
<td>17.8.83</td>
</tr>
<tr>
<td>8 Replacement of stg. No 2 including pipe installation</td>
<td></td>
<td></td>
<td></td>
<td>18.9.83</td>
</tr>
<tr>
<td>9 Removal of the auxiliary hatch</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>10 Re - installation of the equipment hatch</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>11 Pressure test. (primary and secondary system)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>12 RPV inservice inspection Loading &amp; closing of the RPV</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>13 Start – up to full load</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Steam generator replacement from 6.6.83 to 17.8.83
Revision from 4.6.83 to 18.9.83

Nuclear Power Plant Obrigheim
Steam Generator Replacement
SESSION 6.3: EXPERIENCE WITH KWU STEAM GENERATORS
PART 2
KWU STEAM GENERATOR CONCEPT WITH ECONOMIZER

R. Bouecke, G. Schücktanz
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Federal Republic of Germany

1. Introduction

The integrity of the tubing of steam generators is a fundamental prerequisite for a high availability of nuclear power plants with pressurized water reactors. During the past few years steam generator tubes have suffered worldwide from various corrosion mechanisms such as stress corrosion, denting, pitting and wastage. In addition to that the phenomenon of fretting had to be paid special attention recently in steam generators equipped with preheaters. Several steam generators of this design experienced tube fretting already after a few thousand hours of operation. The fretting defects resulted in primary to secondary side leakage in some cases and caused severe restrictions in the operation of all plants affected.

KWU has completed the fabrication of 12 steam generators with preheater so far. 10 steam generators with preheaters are still being manufactured, 4 of them for the NPP Brockdorf in the FRG and 6 for the NPPs Trillo 1 and Trillo 2 in Spain.

2. Design

The design principle of the KWU steam generator with economizer (Fig. 1) is that of the natural circulation U-tube steam generator using a heat transfer area of about 5400 m². The heating tubes are made out of INCOLOY 800 (22 mm outer diameter, 1.2 mm wall thickness).
The primary coolant enters the primary channel head through the inlet nozzle, flows through the 4100 tubes and transfers heat to the secondary water via the tube walls. In this process the primary coolant is cooled down by about 35 K to 291 °C and leaves the steam generator by way of the outlet nozzle.

The feedwater passes through two nozzles into the economizer, positioned at the cold side end of the tube bundle.

Approximately 50% of the feedwater flows upwards through the preheater in a direction opposite to the primary coolant, and approximately 40% flows downwards. This so-called split flow design protects the tube sheet against cold feedwater shocks and on the other hand grants a high thermal efficiency of the steam generator. The remaining 10% of the feedwater is fed into the downcomer for condensation of possible carry under from the separators. The 10% auxiliary feedwater nozzle is also used for feeding of the steam generators during low load conditions.

The feedwater is heated in the economizer to just below boiling temperature. After leaving the economizer it mixes with the circulating water. Steam separators are located on a horizontal separator supporting plate above the tube bundle and these separate the generated steam from the circulating water. The separated steam then flows through steam driers and leaves the steam generator with a residual moisture of less than 0.25% and at a steam pressure of approximately 68.7 bar.

The water leaving the separators flows downwards into the downcomer between the shroud and the vessel wall to the hot leg to enter the tube bundle again above the tube sheet.

A horizontal flow distribution plate is located above the tube sheet, which is to effect equal distribution of water in the tube bundle. The circulation ratio, i.e. the ratio of the mass flow in the riser above the economizer to the generated steam flow, is approximately 3.5 at full load.
The preheaters of KWU-steam generators are characterized by the following design features (Fig. 2):

- Flow limiters (Fig. 3) in each of the feedwater nozzles. These are to reduce the loads on the preheater structure in case of a postulated break of a feedwater line.

- The baffles are not welded to the shroud of the preheater but connected to the tubesheet by tie rods. Thus the thermal expansion of the baffles is not restricted and thermal stresses are minimized. Well defined clearances between the baffles and the shroud minimize the risk of stagnant flow areas in the preheater.

- The material applied for the baffles is stabilized austenitic stainless steel (1.4550 equivalent to type 347), which excludes any risk of denting.

- In each preheater two water distribution boxes (Fig. 4) are installed in order to distribute the feedwater uniformly over the inlet portion of the tube bundle. All parts exposed to higher velocities are either made out of stabilized austenitic stainless steel or are protected by an overlay cladding out of this material.

3. Thermohydraulic design

In order to verify the design of the preheater extensive experimental work was performed. Some of these tests are mentioned in the following:

- Vibration tests of tube bundles

In order to determine the critical velocities for tube excitation vibration tests were performed with tube bundles of the same geometry as existing in the steam generators. The tests confirmed that vortex shedding need not be considered in the preheaters, but when water velocity exceeded a
certain limit value the phenomenon of fluid elastic instability was observed (Fig. 5).

- **Flow distribution testing and vibration testing of the preheater.**

A 1:1 scale model of a complete preheater was tested at full feedwater flow rates. Several subcomponents of the preheater such as the secondary side partition plate as well as some tubes were equipped with a special instrumentation for monitoring any vibration. The test results did not indicate any vibration worth mentioning.

- **Fretting tests**

A tube bundle consisting of 144 tubes was tested for 205 days under operating conditions of a preheater (68 bar, 250 °C, simulation of thermal expansion of the baffles etc.). Several intermediate inspections resulted in the conclusion that also after an operating period of 40 years an unacceptable wall degradation need not be expected (Fig. 6).

- **Water hammer tests (Fig. 12)**

These tests confirmed that also after a not intended interruption of the feedwater flow, and a subsequent feeding of the preheater, water hammer does not occur (Fig. 7).

4. **Commissioning Tests**

The NPP Grafenrheinfeld has been the first station equipped with preheating steam generators. This plant started commercial operation in early 1982. During commissioning of the plant the extensive instrumentation of one of the 4 Grafenrheinfeld steam generators permitted the monitoring of the thermohydraulic behaviour of steam generator equipped with preheaters.
Various operating and upset conditions were simulated. In all cases the behaviour of the steam generators and of the feed-water control corresponded completely to the predicted one. An interruption of the feedwater flow at 30 % load resulted in the formation of steam in the preheater. The subsequent continuation of the feedwater flow did not cause any water hammer, thus confirming the results of the respective model tests.

The loose particle monitoring system permitted already during commissioning of the plant the statement that unacceptable tube vibration should not exist. This statement was possible by comparing the signals of the loose particle monitoring system gained at various load steps.

5. Inservice Inspections

The Grafenrheinfeld plant was shut down for first refuelling and inspection in spring 1983 after some 10 000 hours of operation. The tubes near feedwater inlet of all 4 steam generators were inspected applying the multifrequency eddy current technique. There were no indications of fretting damage. The eddy current inspection of the other portions of the tube bundle showed that there is also no indication of any other kind of corrosion.

In May 1984 after about 18,000 hours of operation two of the Grafenrheinfeld steam generators were once again inspected applying a more sensitised eddy current inspection technique which is able to detect wall degradations down to 5% of the wall thickness. There were no indications of any kind of tube wall attack.

These result agree completely with the predicted one. Grafenrheinfeld is like all other pressurized water reactors of KWU characterized by a steam generator concept, which minimizes the hazard of tube corrosion.

The feedwater steam cycle is free of any copper alloyed material thus permitting an operation of the feedwater train at a pH of more than 9.8. The high pH reduces the transport of corrosion
products into the steam generators and thereby minimizes the risk of concentration of salts in corrosion deposits. The high integrity condenser is equipped with stainless steel tubing and was manufactured at the highest quality standards. Condenser leaks are therefore minimized. This permits the elimination of a phosphate addition which could cause wastage corrosion. The steam generator tubes made out of INCOLOY 800 are glass bead peened at the outer surface and grant the maximum safety against any kind of stress corrosion as has been demonstrated by the operating experience with other NPPs of KWU. The stabilized austenitic stainless steel used as material for the tube spacers eliminates any risk of denting.

6. Summary

The steam generators of the NP? Grafenrheinfeld are the first KWU steam generators equipped with preheaters. Inspections performed after one respectively two years of operation did neither indicate any fretting damage nor any other corrosion of the steam generator tubes. Extensive experimental investigations performed for confirmation of the preheater design, such as vibration tests, fretting tests, water hammer tests etc. had indicated that neither fretting damages nor water hammer needed to be expected. The absence of any kind of corrosion can be traced back to the consequent implementation of the KWU steam generator concept, which is characterized by the minimization of the ingress of corrosion products and other impurities into the steam generators as well as by a careful design of the steam generators and an appropriate selection of the materials.
Amplitude of Tube Vibration

- Authorized Operation
- Steam Line/Feedwater Line Leak
- Random Turbulent Vibration
- No Dominant Vortex Shedding

\[ \phi = \frac{w^2}{2} \]

- Flow Pressure
- Critical Velocity
- Fluid Elastic Instability

Single - Phase: Water
Two - Phase: Air - Water

Steam Generator Tube Bundle
Pitch Diameter = 1.36
Straight Tube Region: Water and Water - Steam - Mixture
Tube Bend Region: Water - Steam - Mixture

Cross Flow Induced Vibration Tests:
Excitation of the S.G. Tubes

\[ w = \text{Gap Velocity between Tubes} \]

Fretting Tests: Measured Fretting Wear Versus Duration of Tests

Pressure Peak

\[ \sigma = 150 \text{ K} \]

Basic Run: \( \sigma = 68 \text{ K} \)

\[ \sigma : \text{Degree of Subcooling} \quad T_{\text{sat}} - T_{\text{sat}} \]

Influence of Various \( \sigma \) on Pressure Peak

- Steam Quality

Water Hammer Tests

Fig. 5

Fig. 6

Inspection Stops

Fretting Wear

[\mu m]

Specified Maximum Roughness of SG - Tubes

Baffle - No. 7
Baffle - No. 9
Baffle - No. 5
Baffle - No. 8

Duration of Test

Fig. 7
SESSION 6.4: POINT BEACH UNIT 1
STEAM GENERATOR REPLACEMENT PROJECT

P. T. Conroy and R. C. Jain
Westinghouse Electric Corporation

1. INTRODUCTION

In replacing the steam generators at the Point Beach Unit 1 nuclear power plant, Westinghouse Electric Corporation utilized innovative concepts that, together with extensive planning and thorough training, resulted in completion of the project ahead of schedule and with manrem exposure significantly below that of previous steam generator replacement projects.

This turnkey project undertaken for Wisconsin Electric Power Company began in June of 1982. The outage started in early October 1983 and, following defueling activities, the containment was made accessible to Westinghouse on October 15, 1983. Westinghouse critical path activities were completed on January 25, 1984 when the new steam generators were presented to WEPCO for hydrostatic testing, 31 days ahead of schedule. This schedule is significantly shorter than previous domestic steam generator replacement projects, and compares favorably with the shortest international replacement project when adjustments are made for scope differences.

2. DESCRIPTION OF THE PROJECT

Westinghouse established a project organization within its Nuclear Services Integration Division (NSID) to implement this activity. The elements of
engineering, construction, contract administration and quality assurance were contained within the project team. The project office also directed the resources of other Westinghouse divisions and subcontractors as required to augment the project team's capabilities.

The NSID Quality Assurance Program Plan was supplemented to accommodate the special requirements of this construction project. The supplement describes the responsibilities of the site organization, recognizes the customized procedures required for project activities, and authorizes other Westinghouse divisions to provide design and procurement controls. The supplement embodies the philosophy of a single quality system covering all project work. This single system was accomplished by coupling the construction subcontractor's quality control system with the design and procurement control systems of other Westinghouse divisions.

Initial plant investigations were conducted during a plant outage in November 1982. Physical and radiation measurements were taken, exiting plant facilities were evaluated, interferences were identified, and numerous photographs and video tapes were prepared for subsequent review. The information gathered during this outage permitted detailed planning to proceed.

Temporary facilities were used to support the replacement outage. Site offices, a warehouse, a machine shop, training facilities, parking areas and the steam generator storage facility were erected, and a Containment Access Building (CAB) was constructed adjacent to the containment. The CAB, a unique concept developed for this project, was designed to minimize the loss of worker productivity associated with entering and exiting the containment work area. The CAB contains lavatory and change facilities, a lunch room, a communications center, and health physics facilities. Both the CAB and the containment were declared devitalized zones. Integral with the CAB was a weather barrier directly outside the containment equipment hatch to provide protection from the winter weather for movement of personnel and equipment through the hatch.
Prior to the replacement outage, a complete training program was developed. All project personnel received a basic project introduction as well as security and health physics indoctrination. Craft labor received detailed activity training, including full-scale mock training. Remote TV and videotapes of actual work were used to fine-tune crew performance on special activities. This training was highly successful in maximizing the productivity of the workers and reducing manrem exposures.

The initial activities during the replacement outage included performing local decontamination; installing control barriers, temporary shielding, scaffolding, and work platforms; removing or protecting plant components; removing interferences; installing temporary power, ventilation, and lighting; and upgrading the polar crane lift capacity from 100 tons to 250 tons. Also, a steam generator transport system consisting of an upender, low profile saddle, and transfer skid was installed.

Cutting methods were specially selected for each operation. Before the steam generators were removed, the various major secondary system piping connections (including the main steam, feedwater, blowdown, and instrumentation lines) were flame cut. The upper shell was separated from the lower assembly using an oxygen-acetylene cutting device traveling on a track. The reactor coolant piping was cut at the steam generator nozzles using a track-mounted plasma arc torch.

The upper shell was placed in a storage stand where the moisture separator package and feed ring were replaced. Unlike the method used in previous replacement projects, this operation was performed with the shell in the upright position. This innovation saved several days in the schedule by eliminating the upper shell upending process.

Steel cover plates with provisions for securing shielding were welded in place to seal the lower shell. The polar crane was then attached to the steam generator lower assemblies using a lift beam arrangement. The lower
assemblies were lifted from the steam generator cubicles, placed on the transport system, winched through the equipment hatch and moved to the storage facility.

The new lower shell was installed in basically the reverse order of the removal operation. A template was made of the as-built dimensions of the new lower assemblies to identify the weld prep dimensions required on the reactor coolant pipe ends. This technique identified that fitup was achievable using only one cut on each reactor coolant pipe, a first for the industry, resulting in significant cost, schedule and radiation exposure reductions. The reactor coolant pipe was rewelded using automatic welding equipment after decontamination of the pipe ends. The refurbished upper shell was welded to the new lower shell using manual welding techniques, as was the secondary side piping. Radiography of the steam generator girth weld was accomplished with a specially designed film holder which permitted filming without reduction of the pre-heat temperature, resulting in additional time savings. Final post-weld heat treatment, NDE, insulation, hydro testing, and clean-up were then completed.

An extensive radiological and health physics program was implemented to control manrem exposures during the project. The health physics program was prepared by Westinghouse and issued by WEPCO as a supplement to their existing site program. This supplementary program was geared to the construction effort of the steam generator replacement, as opposed to the daily operation of the Point Beach Nuclear Plant. As a result, the replacement project was able to proceed independently of WEPCO's operations, but within the site guidelines and overall site health physics program. The Westinghouse computerized Radiation Exposure Monitoring System was utilized to maintain exposure records on-line, allowing daily tracking of allowable exposures for each person and each activity. A budget of 1390 manrem was estimated in the repair report sent to the NRC in 1982. Extensive mock-up training, decontamination, shielding, and the one-cut per pipe replacement method resulted in an actual exposure of about 590 manrem for the total project. This is well below previous replacement projects that have ranged from over 2000 manrem to 690 manrem for the best international project.

11s/2s/081584
3. CONCLUSION

The Point Beach Unit 1 steam generator replacement project was a highly successful nuclear repair activity. Performing the project on a turnkey basis provided the management control necessary to maintain staffing, material, cost, schedule, and ALARA controls. Significant advances in the state of the art of steam generator replacement technology were developed in the areas of:

- Advanced outage planning
- Training
- Containment access control
- Radiation exposure control
- Pipe fitup templating
- Upright moisture separator modifications
SESSION 7

CURRENT LICENSING

POSITIONS AND ISSUES
SUMMARY OF SESSION 7 - CURRENT LICENSING POSITIONS AND ISSUES

Session Chairman: Mr. G. Holahan

The papers in this session addressing current licensing positions and issues were prepared by the regulatory bodies of France, Italy, Spain, Sweden and the U.S. and by the IAEA.

Three of the papers address the broad regulatory approaches used to address steam generator design, analysis, operation and inspection in the licensing process. The other three papers address the specific regulatory responses to the flow induced vibration problem in Westinghouse model D2/D3 steam generators.

In general, the papers reflect a remarkable, international consensus on the nature and magnitude of the safety concerns relating to steam generators. The licensing approaches to both the specific and the general problems vary somewhat in the details but all address the issues in a manner consistent with its safety importance. Steam generator tubes are viewed as a special safety barrier since they function both as a reactor coolant boundary and as a containment boundary. Simply stated, the safety issues are viewed as:

- most steam generator tube ruptures (∼10⁻²/RY) would be expected to have negligible radiological consequences;

- a few steam generator tube ruptures (∼1/100 or ∼10⁻⁴/RY) would be expected to have moderate to large offsite radiological consequences

- a very few steam generator tubes (∼1/1000 to 1/10,000 or ∼10⁻⁷ to 10⁻⁹/RY) would be expected to result in core damage or core melt

- steam generator problems are an important contributor to occupational radiation exposure.

The papers on the flow induced vibration problem clearly identify that significant and timely licensing action was needed and taken in terms limiting operation, requiring modification, and augmenting inspection and monitoring.
SESSION 7.1: A REAPPRAISAL OF STEAM GENERATOR TUBE RUPTURE IN THE FRENCH LICENSING PROCESS

M.M. CONTE, A. GOUFFON, P. MORIETTE
I.P.S.N. FRANCE

1. INTRODUCTION

In 1970, when the French PWR program was launched, the safety authorities decided that as much as possible should be gained from American experience, for design, regulation and general safety principles.

The list and classification of internal events into 4 categories which are taken into account during design complied with the American Standard: ANSI 18-2 (August 1973).

Whilst this standard list, which is drafted according to the principle of a single initiator, e.g. steam line break, has to be finalized at one stage or another so as to build the project, it should be open to change regarding any new factor which may affect the assumptions made during the list's definition.

Significant technological and theoretical improvements were based on the experience acquired from implementing and operating French nuclear reactors: 28 standardized 900 MWe reactors and then 18 standardized 1300 MWe reactors, and has been accompanied by a progressively French approach to construction, principles and evaluation of reactor safety.
This experience has been further increased by the operation of reactors worldwide and from the incidents and accidents which have occurred. In the light of this experience, the French safety authorities have reexamined this events list and have specified acceptable risks for each category of events.

As early as 1977, upon examination of the safety options submitted by Electricité de France (E.D.F.) for the 1300 MWc plants and the safety evaluation of the preliminary report on the first 1300 MWe reactor (Paluel 1), safety authorities:

1) specified the risk requirements for the design:

   - "As a general objective, PWR design should aim at a probability of unacceptable consequences of less than $10^{-6}$ per reactor per year".

   - "if the utility decides not to take into account in the design a "family of events", it must demonstrate that its probability to induce unacceptable consequences is less than $10^{-7}$ per reactor/year".

2) accepted the values proposed by the designer in respect of radiological effects associated to category III and IV events.

Table 1 summarizes the classification which associates frequency of and consequences from the situations used for the design.
Table 1

<table>
<thead>
<tr>
<th>Class</th>
<th>Annual Occurrence</th>
<th>Radiological Consequences</th>
</tr>
</thead>
<tbody>
<tr>
<td>I and II</td>
<td>$&lt;10^{-2}$</td>
<td>Authorized releases</td>
</tr>
<tr>
<td>III</td>
<td>$10^{-2} &lt; F &lt; 10^{-4}$</td>
<td>$5.10^{-5}$ SV whole body dose (500 mrem)</td>
</tr>
<tr>
<td>IV</td>
<td>$10^{-4} &lt; F &lt; 10^{-6}$</td>
<td>$15.10^{-2}$ SV wbd in 2 hours at the limits of the site (15 rem)</td>
</tr>
</tbody>
</table>

3) estimated that a total break of a steam generator tube, would correspond to a primary break with a diameter of around 17 mm, did not appear to be less likely than small breaks in the primary circuit having diameters between 9.5 and 25 mm. Hence this event should be classed under category III.

During this period, around 40 steam generator tube incidents occurred, for which the corresponding primary/secondary leak rates were measured or estimated.

Since then, a broad study has been made by the safety authorities to specify the steam generator tube rupture events which shall be included in the design of the plant and to define the criteria and any changes to be made for the installation to satisfy the risk requirements set for the design.

This study also enabled identification of certain cumulated failures which had not been taken into account under conventional design and which were deemed to be sufficiently significant to warrant in-depth analysis.
2. INTRODUCTION OF GENERATOR TUBE RUPTURES IN REACTOR DESIGN

2.1 Classification of steam generator tube rupture events

The safety evaluation of nuclear power plant design should enable identification of all potential "weak spots" in the installation whose failure would be the source of radioactive releases into the environment and to determine the action necessary to mitigate the consequences.

Taking under consideration the two following points specific to PWR design, this type of "weak spot" was rapidly identified:

- steam generator tubes which take the place of the 2nd and 3rd confinement barriers are a particular design point and are an exception to the concept of 3 independent barriers usually specified between the fission products source (fuel) and the environment.

Consequently, any tube failure could induce a primary coolant leakage outside of the 3rd barrier.

- non filtered releases into the environment from steam generator safety valve discharge which can occur during an accident situation, and in particular, following the transient which occurs during tube failure.

The quantity and the activity of the releases depend on primary coolant activity and on the valve closing time.

Operating experience of pressurized water reactors bore out the fact that safety considerations on steam generator bundle strength were well founded. The incident which occurred on January 25, 1982 on the Ginna Reactor, showed the necessity of evaluating in detail all solutions which would prevent or at least mitigate any radioactive releases into the environment.
Consequently, upon the examination of the safety options submitted by EDF for a new pressurized water reactor design (N4: 1400 MWe), the French safety authorities decided that the conventional list of events to take under consideration should be amended as follows:

- failure of 1 steam generator tube: category 3 event,

- failure of 2 steam generator tubes: category 4 event.

2.2 Impact on Design and Operating Criteria

To meet these objectives, design improvements were decided and new operating criteria were required by the technical specifications.

Some of these improvements have already been implemented on the 900 and 1300 MWe reactors presently in operation and the new requirements have been enforced.

These measures contribute to an improvement in the overall safety of the plants and satisfy two fundamental in-depth defense principles: prevention of accidents and mitigation of consequences.

2.2.1 Prevention

Various preventive measures have been adopted by E.D.F. to reduce tube degradation risks.

2.2.1.1 At the design stage

- Above the tube sheet, the water flowrate has been improved by means of flow distribution baffles.

  This change reduces deposit regions.

- Special tube heat treatment limits generalized corrosion phenomena.
- Fabrication residual stresses have been reduced: improvement of tube rolling and stress relieving heat treatment (of straight and small bended tubes) to reduce corrosion under stress.

- Quadrifoil tube support plates made of Z10 C13 stainless steel, prevent denting and improve secondary feedwater thermohydraulics near the tube support plates.

2.2.1.2 Improvement of secondary feedwater quality

The first incidents observed on steam generator tubes - leakage induced by stress corrosion, tube thinning due to acid corrosion - lead to the replacement of phosphate treatment of the secondary feedwater by a volatile treatment as of 1974.

The quantity of denting observed in 1976 in American reactors which had changed over to volatile treatment instigated a major research program to stop and if not, to prevent denting.

Although the conclusions drawn from this program have not been definitively established, secondary feedwater chemical specification has been decided for French reactors which is compatible with operating requirements and, a priori, sufficiently restrictive to prevent denting from occurring. This specification is now applied.

The volatile treatment adopted consists in the injection of volatile products, only, into the secondary feedwater.

- an amine (morpholine) to obtain the pH level that limits corrosion of the materials used for the vessels and piping,
- hydrazine to reduce oxygen content,

To guarantee treatment efficiency, stringent control is performed to maintain the specified chemical properties of the secondary feedwater:

- Testing of feedwater treatment: pH and acceptable morpholine level, hydrazine and oxygen concentrations.

- Testing of steam generator blowdown pollution: measurement of non-volatile impurities.

2.2.1.3 Enhanced quality control

In compliance with the requests of the French Government, a major research and development program has been undertaken to improve steam generator tube defect detection and to evaluate their hazard.

We have only just cited these controls, since the progress in these areas and results have been covered elsewhere (cf session 2: "Regulatory Requirements for Steam Generator Tube Surveillance and In-service Inspection", P. LEDERMANN, J.M. GIRES: "Contribution to the Improvement of Flaw Detection in and Sizing of Steam Generator Tubes using Eddy Current Techniques", R. LEVY, R. SAGLIO; Intercontrôle).

2.2.2 Mitigation of consequences

The radiological consequences of generator tube integrity failure can be mitigated if any or all of these conditions are fulfilled:

- primary coolant activity is low,
- tube flaw detection is rapid,
- release time is short,
- operating procedure is suitable and easily implemented.
2.2.2.1 Primary coolant activity

Operating experience shows that all reactors operate with failed fuel elements.

To mitigate the releases into the environment, primary coolant activity, which depends on the number and seriousness of the fuel failures and on the reactor operation, should be kept as low as possible and still be compatible with the reactor operation.

Previously, technical specifications for continuous operating conditions required that:

- the safety authorities have to be informed and that surveillance activity be increased when the primary coolant mass activity reaches \(4.44 \times 10^9\) Bq (0.12 Ci/t) in I 131 equivalent and \(3.7 \times 10^{11}\) Bq (10 Ci/t) in rare gas activity,

- the value, \(3.7 \times 10^{10}\) Bq (1 Ci/t) in I 131 equivalent and \(1.11 \times 10^{13}\) Bq (300 Ci/t) rare gas activity must not be exceeded.

When these values are exceeded, various operating constraints are applied: reduction of operating time, no load follow operation, new corrective actions and surveillance.

Hot shutdown status was required for the reactor within 6 hours after primary coolant activity exceeded \(1.11 \times 10^{11}\) Bq (3 Ci/t) in I 131 equivalent.
Since July 1980, additional limitations have been imposed. They are based on the mass activity of the primary coolant measured under continuous operation conditions and require reactor shutdown:

- during reactor start-up, activity shall not exceed:

  $1.11 \times 10^9$ Bq (0.03 Ci/t) in I 131 equivalent
  $9.25 \times 10^9$ Bq (2.5 Ci/t) total of rare gases.

- When the mean value of mass activity in I 131 equivalent reaches $2.22 \times 10^{10}$ Bq (0.6 Ci/t), as computed from the start of the cycle, the reactor is shut down within 2 months.

- When this same mean value reaches $2.96 \times 10^9$ Bq (0.08 Ci/t), the reactor is shut down within 15 days.

Furthermore, to foresee any occurrence of a serious tube leakage or degradation of minor failures, it is specified that:

- When instantaneous mass activity in I131 equivalent exceeds $1.85 \times 10^{10}$ Bq (0.5 Ci/t) and $1.48 \times 10^{12}$ Bq (40 Ci/t) for rare gases, the reactor shall be shutdown within 48 hours.

- When these same values reach $1.11 \times 10^{11}$ Bq (3 Ci/t) and $9.25 \times 10^{12}$ Bq (250 Ci/t), respectively, the reactor shall be shutdown within 6 hours.

The utility must inform safety authorities whenever the values specified in the technical specifications are reached.
To satisfy these operating rules, which limit the maintenance of failed fuels in the reactor, specifications on the quality control and elimination of failed assemblies during reactor loading has been established by the EDF. Studies made by EDF have lead to the establishment of a relationship between primary coolant activity, the number and location of failures. This relationship enables determination of rod leak rate, $\gamma g$, which is representative of the fission products transfer time into the primary coolant.

This initial cladding material estimate is checked out during the sipping tests and enables discard of assemblies with serious failures ($\gamma g$ near $10^{-2} s^{-1}$).

Application of these rules on the 25 plants in operation have confirmed the soundness of this approach and has enabled significant improvements in the accuracy of failure's detection and location of failed fuel rods.

2.2.2.2 Detection of steam generator tube flaws during operation

During operation, several tests enable detection of flaws which occur on the steam generator bundle: primary/secondary total leakage, activity measurements made on the steam generator secondary-side, from the condenser and blowdowns.

Measurements from the condenser are accurate and involve no time delay. However, they are insufficient when taken alone, for operation under accident conditions. Moreover, measurement is inoperative after steam isolation or if the condenser is unavailable.

Continuous measuring of activity has been installed on each steam generator blowdown to improve location and rapid isolation of a faulty steam generator. This additional information also facilitates implementation of accident procedures.
2.2.2.3 Mitigation of releases

The radiological consequences of steam generator tube failures are related to the actuation of the steam generator isolation devices.

Since efficient filtration cannot be installed between the active devices and the environment, the release value is dependent on the closing capacity of these devices, i.e., their qualification. To fulfill this qualification, a special loop is now under construction; next year operation tests on the safety valves in steam phase will be carried out under conditions simulating accident conditions.

Regarding water discharge problems, studies have been undertaken so as to be able to determine the safety margins with respect to the risk of filling steam generators with water and will determine if qualification of active devices under this type of operation is necessary.

The preliminary results have shown that in the event of failure of one tube, the risk of filling the generator with water is low, even negligible.

2.2.2.4 Operating procedure

Plant design and evaluation of radioactive releases are based on accident studies whose effect on the plant and the environment are considered as envelopes of the consequences of all accidents of the same category. There was no cumulative failure assumed in these studies.

Regarding steam generator tube failures, safety authorities requested that release evaluations should be based on a two-fold working hypotheses, one set of assumptions which maximize effects on the core and the other which maximizes releases.
These studies should be carried out in applying the following rules:

- the operation of regulation systems should be discounted to the extent that their effect will be unfavorable with respect to the consequences of the accident,

- in applying the single failure criterion, the failure of each steam generator isolating device should be considered in so far as each of these devices is actuated and any single failure which delays a drop in pressure in the cooling circuit should also be considered.

- with respect to operator actions, especially under manual operation from the control room, for actuators which are normally operated by the regulation system, no action should be required during the first 10 minutes following identification of the accident. Afterwards, operator actions will be assumed to be effective except in scenarios where they constitute the single failure to be considered.

- the total lost of external electrical power supplies during an emergency shutdown should be foreseen.

The progress made in this direction is given in the lecture by Mrs. N. TELLIER and Mr. M.C. ZILLIOK: "Steam Generator Tube Rupture: Studies to improve Plant Procedure" (Session 3).

The design improvements that should result from these studies will be introduced in the design of the new project N4 and the feasibility of retrofitting them to existing plants will be examined.
3. CONSIDERATION OF CUMULATIVE FAILURES

The probability of the occurrence of the event "single generator tube rupture" has been estimated to be several $10^{-3}$ per reactor per year. To take a census of the set of scenarios whose frequency is estimated at being greater than $10^{-7}$ per reactor per year, the multiple failures likely to occur in a plant have to be taken into consideration. Although these cumulative failures are never requested to justify the PWR design, the French safety authorities have deemed that such studies should be made for the new project N4. Given the complexity and the number of possible combinations, it has been decided to analyze two accidents whose radioactive consequences were estimated to amply cover those caused by potential multiple failures:

- non-isolatable steam pipe break, cumulated over 1,2 or 10 steam generator tubes,

- single steam generator tube failure cumulated with a steam generator safety valve open.

The results of these studies will be taken into consideration for the design of 1400 MWe reactors.

4. CONCLUSION

The principles adopted in France for operating pressurized water reactors are: priority is given to safeguarding the core in association with low core activity with respect to fission products (clean core concept) changes in design and qualification testing of present or scheduled discharge devices tend towards improved safety by contributing to mitigating the radioactive consequences of steam generator tube rupture accidents.

Further, studies now underway for evaluating, under penalizing assumptions, releases related to steam generator tube failure accidents cumulated with or without a steam pipe break will provide the data needed to decide any new changes in design.

Finally, analysis of the operating experience should enable the validation of the solutions adopted and if necessary their improvement.
SESSION 7.2: ITALIAN REGULATORY TRENDS ON
STEAM GENERATOR TUBE RUPTURE

S. Benassai, G. Bisceglie
ENEA - DISP

1. INTRODUCTION

The Italian National Energy Plan, approved by the Parliament in 1981, establishes the construction of eight nuclear power reactors according to a standardized PWR type design.

In this framework ENEA/DISP, the Italian regulatory authority, has given first priority to the setting up of General Design Criteria (GDC), as to give timely indication on the fundamental safety and protection objectives to be achieved in the design. Quantitative design objectives have then been set up for the radiological protection of workers and general public, design objectives for general public being defined with reference to different plant conditions. For severe accidents leading to core degradation, safety objectives, in terms of maximum probability of their occurrence, have been established.

In accordance to the "defense in-depth" principle and consistently with the radiological and safety design objectives, specific design requirements have been set up for all relevant safety systems, including the Steam Generators Tubes, considered a critical component, since their failure:

- allows primary coolant to be released to the environment;
may constitute an initiating event of accidental sequences leading to core melt.

2. RADIOLOGICAL AND SAFETY DESIGN OBJECTIVES

Radiological design objectives for the individuals of the population reference groups are established as follows:

- for operational conditions, including both normal operation and transients, the effective dose equivalent and the organ dose equivalent shall not be respectively higher than 0.1 mSv/y (10 mrem/year) and 0.3 mSv/y (30 mrem/year);
- for accidents whose annual probability is not higher than $3 \times 10^{-2}$ the effective dose equivalent and the organ dose equivalent shall not be respectively higher than 5 mSv/event (500 mrem/event) and 15 mSv/event (1.5 rem/event).
- for accidents whose annual probability is not higher than $10^{-3}$, including design basis accidents, the effective dose equivalent and the organ dose equivalent shall not be respectively higher than 100 mSv/event (10 rem/event) and 150 mSv/event (15 rem/event); however the effective dose equivalent shall be, as far as possible, limited to 5 mSv/event (500 mrem/event).

Moreover the occurrence probability of those accident sequences which entail radiological consequences higher than those above established, shall be kept as low as practicable. With regard to workers exposure, it is required that the annual collective dose, for normal and planned conditions, is aligned with the best international standards, but not higher than 4 manSv (400 manrem) for each 1000 MWe unit.

Safety design objectives, in terms of reference probability values of sequences leading to core degradation are also established, as follows:
- the single sequence annual probability shall not be higher than $10^{-6}$;
- the overall annual probability shall not be higher than $10^{-5}$.

Moreover, proven design alternatives must be taken into consideration in developing the design so as to approach $10^{-7}$, and $10^{-6}$ respectively for any single sequence and for the overall core degradation probability.

3. CONSISTENCY WITH RADIOLOGICAL AND SAFETY DESIGN OBJECTIVES

3.1 Selection of important events

In order to verify the compliance with the radiological and safety design objectives, events for the analysis will be selected according to their occurrence probabilities. In particular it is requested the evaluation of:

- the probabilities of single and multiple SGTR in one or different steam generators, also considering failure propagation from one tube to others through impact or jet forces;
- the probabilities of single or simultaneous multiple SGTR, as a consequence of accidents (such as Feed Water Line Break, ATWS, LOCA, etc.) entailing significant increase in steam generator tubes stress conditions;
- conditional core melt probability following a SGTR.

The probability of a single SGTR has been assumed not less than $10^{-2}$ /reactor year.

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(1) Core degradation has been conservatively assumed to occur when core coolability conditions, established in accordance to US-NRC 10 CFR 50.46, are exceeded.
3.2 Evaluations

Preliminary results of the system designer analysis on SGTR for the Italian reference design are presently available only for a single tube rupture, considered alone or combined with loss of off-site power (LOOSP) and Stuck Open Valve (SOV). This analysis are presently being evaluated by the licensing body.

Probability values and radiological consequences evaluated by conservative models used for licensing purposes, in terms of thyroid dose for the most exposed infant, are respectively:
- for SGTR: $\approx 10^{-2}$/reactor year and negligible radiological consequences
- for SGTR + LOOSP: $\approx 10^{-3}$/reactor year and doses up to $3 \times 10^{-2}$ Sv
- for SGTR + LOOSP + SOV: $\approx 10^{-5}$/reactor year and doses up to some Sv.

Moreover, very preliminary results of a level 1 risk assessment performed for the Italian reference design give a core degradation probability from sequences initiated by a SGTR in the order of $10^{-6}$/reactor year.

Multiple SGTR are still under evaluation. They may entail much more severe conditions, in particular with reference to the possibility of heat removal from the reactor, if more than one steam generator is affected.

Preliminary results, limited to ruptures consequent to accidents entailing a significant stress increase in the steam generator tubes, give probabilities in the order of $10^{-6}$ for the rupture of two tubes, and negligible for more tubes.

3.3 Preliminary assessment

As it can be seen, radiological consequences associated with SGTR + LOOSP and the annual frequency of $10^{-6}$ for sequences leading to core degradation are slightly above the design objectives mentioned in
paragraph 2. However, also if not conclusive opinion can be expressed on the bases of these very preliminary and partial evaluations, there is a wide consensus that a more accurate accident analysis and the benefits from the adoption of preventive and mitigating provisions which are under study (and will be discussed later), will be effective in reducing consequences and probabilities within the acceptable limits.

4. PREVENTIVE AND MITIGATING PROVISIONS

Over and above the previous considerations, SGTR may entail, together with radiological consequences, other important impacts such as:
- plant outage;
- decontamination and repair cost;
- workers exposure for decontamination and repair;
- operator behaviour in respect of conflicting demands of avoiding either the radioactivity releases to the environment or damage to the fuel;
- general public aptitude in relation to any radioactivity release to the environment.

It is then required that provisions, aimed to prevent SGTR and to mitigate consequences, are taken into consideration, where they reflect advanced design developments, whose implementation can be achieved in the present situation of technological maturity. Some of these provisions are discussed in the paper, in connection with the following items:
- improvements in design and construction
- improvements for ISI
- pressure peak prevention
- technical specifications
4.1 Design and construction of steam generator

The steam generator should meet the following requirements:

- design, construction and inspectionability shall be in compliance with ASME Sec. III and Sec. XI for class I components; an analysis is also required of the fracture mechanics capable of demonstrating the steam generator integrity, particularly in design accident conditions;
- all workshop welds shall undergo UT control at the workshop following the same procedures as laid down for the ISI;
- resilience tests using Charpy V Standard test pieces with transverse bearing must be carried out on each batch of ferritic material used in the construction of the steam generator (including gates, water tank and the upper hemispherical casing); such tests shall be carried out on surface material and at half thickness; the minimum Charpy V Energy upper shelf value shall not be lower than 100J;
- the partition plates shall not be of ferritic steel;
- improved toughness, at operating temperatures, of the base material, weld material and heat affected zone (HTZ);
- control of inclusions and segregations and of factors, mainly the welding parameters, which can lead to the formation of cracks;
- design and construction capable of reducing the number of areas to be periodically inspected;
- the pipes-pipe plate combination shall be such as to prevent interstitial areas which can trigger off corrosive phenomena or inadmissible residual tensile stress conditions;
the deposition process for the protective coating on the pipe plate and on the hemispherical bottom shall be of the low heat transfer type.

Moreover demonstration shall be given that:

- the relative movement between tubes and separating plates does not lead to excessive thinning of the pipe by rubbing;
- post-weld local thermal treatments carried out using induction techniques do not leave high residual stresses in the weld zone and HTZ;
- the coolant flow does not cause, throughout the entire flow velocity range considered, any self-sustaining vibrations of the pipes bundles which could lead to impacts or friction between adjacent pipes;
- the coating deposition processes, especially on forget casing, do not introduce cracks on the undercoating; destructive tests used to provide an adequate demonstration shall be carried out after baking.

4.2 Improvements for ISI

SGTR can be partially prevented, according to the principle of leak before break, if an extensive program of ISI is performed. At the moment USNRC Regulatory Guide 1.83 constitutes the main reference; but there is a tendency to improve steam generator ISI program, with reference to intervals between two ISI, choice of tubes to be inspected, etc.

However, improvement in ISI shall be considered together with the associated workers exposure, in compliance with the requirement that the risk reduction for the general public shall be balanced with the possible related increase in worker exposure. In this regard,
operating experience from Trino PWR has shown that each ISI on the steam generator primary and secondary sides leads to collective dose from $10^{-1}$ to 1 manSv (10 - 100 manrem). In particular, during the last ISI, the collective workers dose was in the order of 0.4 manSv (40 manrem), the eddy current test being the major responsible for the dose (0.23 manSv).

A final decision on improvements for ISI will then be taken with regard to the use of advanced ISI techniques able to reduce workers exposure, and on the basis of a sensitivity study concerning the effects of these improvements on the reduction of SGTR probability.

4.3 Prevention of pressure peaks

Provisions to prevent or reduce stress conditions in the primary circuit and in the steam generator tubes in case of accidents such as Anticipated Transient Without Scram (ATWS), Main Steam Line Break (MSLB), Feed Water Line Break (FWLB) are positively considered.

One of the alternative considered is increasing the pressurizer safety valve relief capacity, which would reduce the peak pressure in the primary coolant and yield an higher probability of the pressure boundary to survive with no damage a pressure transient.

The peak pressure could also be reduced by adding burnable poisons in the core; this will modify the core behaviour, reducing the fraction of time the moderator temperature coefficient is unfavorable. The adoption of a check valve inside the primary containment would avoid rapid pressure decrease in the secondary side following a FWLB, reducing stress conditions on SGT's.
Analogously, the adoption of a single forged element (including branches or in such a way as requires only head welds) for the steam line section between the primary containment and the isolation valve, will reduce the probability of MSLB without isolation.

4.4 Technical specifications

Radiological consequences associated to SGTR, are strongly effected by:

- the number (percentage) of failed fuel rod and, consequently, the radioactivity concentration in the primary coolant;
- the leakage flow from the primary to the secondary loop through the steam generators and, consequently, the radioactivity concentration in the secondary coolant.

Operating experience for PWR, in Italy and in the world, has shown in this regard that radioactivity concentration in the primary coolant and SGT leakage rate are generally well below the values of 1 uCi/g (I 131 equivalent) and 1 gpm which are normally considered in the technical specifications and accident analysis. Therefore it is under consideration the possibility of fixing more limiting conditions for the plant operation as regard the radioactivity concentration limits in the primary coolant and the SGT leakage rate limit.

4.5 Radioactivity release containment

A study has been required with reference to possible design options aimed to reduce the radioactivity release to the environment in case of SGTR.
Design options under consideration spread from a system able to collect the secondary coolant released via relief and safety valves and to condensate the steam, up to the more simple provision of a circuit able to discharge the secondary fluid, in case of steam generator overfilling, to a tank located inside the primary containment.
SESSION 7.3

FLOW-INDUCED VIBRATION ISSUES
IN PREHEATER STEAM GENERATORS - THE IAEA INITIATIVES

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FLOW-INDUCED VIBRATION ISSUES
IN PREHEATER STEAM GENERATORS - THE IAEA INITIATIVES

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1. INTRODUCTION

In response to the requests and the perceived needs of some of its Member States, the International Atomic Energy Agency (IAEA) undertook initiatives following the October 1981 incident at the Ringhals-3 nuclear power plant in Sweden which involved the leakage of primary coolant in one of its preheater steam generators. The cause was subsequently determined to be high wear rate due to flow-induced vibration of the steam generator tubes. The wear was most pronounced in tubes close to the feedwater inlet.

The Ringhals-3 incident has international implications. There are plants in other countries, namely, Brazil, Spain and the United States, which have similar steam generators. In addition there are plants in Belgium and Yugoslavia which are equipped with steam generators of slightly different design, but utilising the same concept of preheating the feedwater, which may experience the same problem. Overall, more than 100 steam generators in 30 PWR plant units worldwide have been affected (see Table 1). The safety concern with regard to this problem stems from the fact that if the rate of tube wear is too high, or if the wear remains undetected, tube failure may occur even during normal operating conditions. Such events may have potential radiological consequences, both to the plant personnel and the members of the public.

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Paper to be presented at the NEA/SNI-UNIPEDE Specialist Meeting on Steam Generator Problems, Stockholm, Sweden, 1-5 October 1984
Various investigations and studies were initiated by organizations affected by the problem (i.e., the manufacturer of the steam generators, utilities and regulatory bodies in affected countries). Both short- and long-term actions for resolution of the issue were developed.

Table 1. Plants affected by the steam generator tube vibration problem

<table>
<thead>
<tr>
<th>Country</th>
<th>No. of units</th>
<th>No. of steam generators per unit</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Split-flow models</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Brazil</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>Spain</td>
<td>6</td>
<td>3</td>
</tr>
<tr>
<td>Sweden</td>
<td></td>
<td>3</td>
</tr>
<tr>
<td>USA</td>
<td>6</td>
<td>4</td>
</tr>
<tr>
<td><strong>Counter-flow models</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Belgium</td>
<td>2</td>
<td>3</td>
</tr>
<tr>
<td>Spain</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>USA</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td>10</td>
<td>4</td>
</tr>
<tr>
<td>Yugoslavia</td>
<td>1</td>
<td>2</td>
</tr>
</tbody>
</table>

The IAEA initiatives in assisting its Member States affected by this problem have been in the form of: (1) sending safety advisory missions to the Angra-1 nuclear power plant in Brazil and the Krsko plant in Yugoslavia in 1982, at the request of these countries, and (2) the convening of an Advisory Group on Safety of Preheater Steam Generators in October 1983.

These initiatives are part, respectively, of the advisory services and exchange of information components of the nuclear safety programme of the Agency. Among the other components are the development of internationally agreed safety codes and guides for nuclear power plants (referred to by the acronym NUSS); establishment of an international Incident Reporting System (IAEA-IRS) that will, it is hoped, eventually include all countries with nuclear power programmes; development of international mechanisms to respond to requests for assistance in the event of a serious nuclear
accident; and, the identification of areas and means by which international co-operation in nuclear safety research can be possible.

2. SAFETY ADVISORY MISSIONS ON STEAM GENERATORS

The safety advisory missions to Brazil and Yugoslavia on the steam generator vibration problem were each of one week's duration and were each composed of a team of five experts from Spain, Sweden, USA and the IAEA. These missions conducted discussions with the regulatory bodies and utilities in the requesting countries.

In Brazil, the Angra-1 nuclear power plant was nearing critical stage at the time of the visit of the mission in January 1982. This mission was requested to provide information and advice in the following areas (Ref. 1):

- Status of the steam generator problem in Spain, Sweden and the U.S., and the corresponding regulatory and industry positions on the problem in these countries;

- Comments on the safety review and licensing decisions of the Brazilian regulatory body for the Angra-1 nuclear power plant with respect to the tube vibration problem;

- Effectivity of certain tests and instrumentation for detection of steam generator tube vibration and wear.

In Yugoslavia, on the other hand, at the time of the mission in June 1982, some tests and measurements had been completed concerning the vibration in the steam generators of the Krsko plant (Ref. 2). As a result of these tests, a proposal was made to make modifications in the feedwater flow in order to alleviate the problem. The mission was requested to give comments and advice on the following:

- Proposed feedwater system modification, including the changes that need to be done to other plant systems;
Plans and preparations for the operating programme following the modification.

The sharing of information and experience during the safety advisory missions have been useful, not only for the recipient countries, but also to the mission members in that they contributed towards providing bases for decisions relating to the vibration issue.

3. ADVISORY GROUP MEETING ON SAFETY STEAM GENERATORS

By 1983 results were already available on the actions taken to resolve the steam generator problem. In particular, modifications designed to eliminate or reduce vibration to acceptable levels have been introduced in several affected steam generators.

These modifications were in the form of either: (1) the installation (for the split-flow models) of a device to improve the flow distribution at the entrance of the preheater; or, (2) improvement of the tube support conditions (for counter-flow models) by reducing the clearance between selected tubes and support plates, i.e. by tube expansion. In addition, a reduction in the velocity of the flow in the preheater for some counter-flow steam generators was also provided by splitting the feedwater flow between the main and auxiliary nozzles.

In view of the above developments, the Agency deemed it useful to convene in Vienna a one-week Advisory Group meeting in October 1983 in order to review the available experiences, test results and other relevant safety information on preheater type steam generators installed in plants which are either operating or under construction. The countries represented at the Advisory Group were Belgium, Brazil, Canada, Germany (F.R.), Spain, Sweden, U.S.A. and Yugoslavia. The Advisory Group developed several recommendations on the subject which should be of value to Member States with nuclear plants equipped with the type of steam generators which are affected by the problem. They will also be of value to the designers of nuclear power plant systems in terms of the important considerations that should be taken into
account in future designs of nuclear steam generators. The general lessons learned from the actions taken in order to solve the problem should be of benefit to utilities and regulatory bodies of Member States with nuclear power programmes.

The following recommendations were made by the Advisory Group relative to general plant design and preheater steam generator design (Ref. 3):

3.1 General Recommendations

1. New design and design changes, prior to their implementation, should be thoroughly investigated through appropriate analytical studies and experimental modelling of the actual design. The extent, degree of replication and scale selected for the experimental model should be appropriate for the conditions to be modelled and the phenomena to be studied.

2. Changes in the design of systems important to safety, including their implications for the other plant systems and components, should be highlighted by those who performed the change and should be analyzed in the safety documentation.

3. Review of these design changes should be performed by personnel who have knowledge of the previous design and who will have subsequent responsibility for operating the modified system.

4. Organizations charged with licensing and regulatory functions should ensure that adequate attention has been paid to these design changes by all organizations responsible for their review, implementation and operation.

5. Resolution of generic problems, particularly those that are important to safety, can be effectively implemented through co-operation and pooling of resources by the affected organizations within a country and on an international basis.
3.2 Specific Recommendations for Preheater Type Steam Generators

1. An adequate flow distribution scheme appropriate for the tube support conditions should be provided to minimize the risks of adverse tube vibration.

2. Preheater type generators are designed for various reasons, among which is the desire to increase the heat transfer and thus reduce the size of the equipment needed for a given capacity. The entire preheater section, including passes other than the inlet pass, should be considered in the evaluation of all aspects of preheater performance.

3. Loadings considered in the design evaluations of the preheater components and actual plant loading conditions should be reviewed to ensure that the plant loadings are within the envelope of the design loadings. This review should be completed for the initial plant and component designs and following any subsequent design changes in the plant or component. For example, plant operating thermal transients, feedwater line pressure pulsations and piping vibrations should be verified as being consistent with preheater component design assumptions.

4. If modification is contemplated, a review of the existing surveillance programme should be made to establish the need for changes in scope and frequency of the inspection programme. Where necessary to ensure continued safe plant operation, consideration should be given to other measures of inspection. The need to trend results from the surveillance programme should be recognized for the purpose of deciding whether any additional actions may be necessary.

5. As regards the implementation of modifications to installed components, the following recommendations can be given:

   a. The modifications should preferably be performed prior to plant start-up in order to avoid radiological exposure of plant personnel and to allow unlimited amount of work time in the affected area;
b. Strict control should be established regarding the tools and materials used for the modifications in order to ensure that no foreign objects or loose parts are left in the system. Following the modification, thorough inspection should be performed to verify that this has been achieved.

c. In carrying out the modification, care should be taken that the steam generator internals are not damaged and to verify, after the modification, that no damage has occurred;

d. Personnel assigned to the implementation of the modification should be given appropriate training, preferably using mock-ups. A detailed planning of the operations should be done in advance to reduce the exposure of personnel to a level as low as reasonably achievable;

e. To avoid unexpected failures during the implementation of the modification, special tooling should be qualified and tested in full scale mock-ups. That the tooling is functional should be confirmed after receipt on site;

f. Operational procedures should be reviewed and, if necessary, revised to reflect system changes or equipment modifications implemented. Appropriate operator training should be carried out in such circumstances.

6. After the modification and following the review of several in-service inspections, the potential exists that some additional actions may be necessary. Tests and analyses should be conducted prior to the implementation of such actions.

The bases for the above recommendations are discussed in detail in Ref. 3, along with specific lessons learned in the modifications of the split-flow and counter-flow types of preheater steam generators.
4. CONCLUDING REMARKS

International co-operation has proven to make an indispensable contribution towards assuring nuclear safety. The broad membership of the IAEA makes it ideally suited for making such co-operation possible. Through advisory safety missions the IAEA can make directly available to requesting Member States the knowledge gained from the experience of others. Through the hosting of meetings, seminars and symposia, it provides a forum for exchange of technical information to pool the collective knowledge and experience of the international community.

REFERENCES


SESSION 7.4: TUBE WEAR DUE TO FLOW–INDUCED VIBRATION IN WESTINGHOUSE STEAM GENERATORS.
LICENSING REQUIREMENTS IN SPAIN

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1. INTRODUCCION

On October 21, 1981 there was a leakage of primary coolant in a steam generator of the Ringhals–3 nuclear power plant in Sweden. The cause of the incident was the high wear rate due to flow–induced vibration of the steam generator tubes which led to the leakage of the primary coolant through the tube.

There are 6 units in Spain with the same steam generator model as Ringhals 3. Since each unit has 3 steam generators, there are 21 steam generators directly affected by the flow–induced vibration problem.

The aim of this paper is to present the licensing requirements and actions of the Spanish Regulatory Body, the Consejo de Seguridad Nuclear (CSN) regarding this matter. Although a few words are said about all the affected units, the enfasis is put on Almaraz Nuclear Power Plant Unit 1 since most of the licensing requirements were imposed initially in this unit and extended later on to the other units.
2. LICENSING REQUIREMENTS AND ACTIONS.

When the incident at Ringhals-3 occurred, only Almaraz-1 was in operation (100% power). Once the cause of the incident was determined, Westinghouse advised the owner to shutdown the plant to check whether the problem in Ringhals was generic (affecting other plants) or specific to that plant. Almaraz-1 was shutdown on November 1981 to conduct ECT inspection of the steam generator tubes. The results of the inspection showed that Almaraz-1 tubes were undergoing the same type of degradation previously found at Ringhals-3. However the level of degradation was lower since Almaraz had been in operation for a shorter period of time. The ECT inspection performed at that time used the multifrequency differential method.

Shortly after the problem of tube wear was discovered it became apparent that a quick solution to eliminate it would not be available. To reduce the energy production losses to a minimum and to gain experience about the threshold of the vibrations, the owner of Almaraz-1 applied for a permit to operate the plant 1500 hours at 50% power including a power escalation at the beginning of the 1500 hours run. Previously the utility asked for a change in the plugging limit from 40 to 50% claiming that the ECT method used was overconservative based on a comparison of the defects measured by ECT and the laboratory examination of two pulled tubes; that change was approved by the CSN. With the plugging limit of 50% a total of 38 tubes had to be plugged. Regarding the damage distribution it is worth to mention that, although a pattern could be recognized, there was a considerable dispersion in damage levels. For instance, tubes symmetrically situated in the same steam generator or in the same position in different steam generators have suffered substantially different damage. This fact revealed that one factor, i.e., the tube support conditions, was playing an important role in the phenomena. To allow the determination of a threshold of tube vibration the owner of the plant installed four internal accelerometers inside two tubes (2 per tube) of the Steam Generator number 1 and installed three external accelerometers on the outside surface of each steam generator.

Although some analytical studies and scale model test results were available at that time, the license to proceed was given mainly to allow the power escalation and therefore to obtain information that could be used to determine a safe level of operation. Only 40 hours were allowed for the power
escalation and periodic reports were required along the 1500 hours run. At the end of this period the unit was shutdown to conduct ECT examinations of the tubes. This time ECT was performed using the previous method and an improved method (i.e., the absolute method). While the results of the absolute method gave a clear indication that the differential method results were normally overconservative, it was not easy to infer a reliable rate of wear at 50% power since previous measurements were done only with the differential method. In order to obtain an idea of the real damage of the plugged tubes, one tube per steam generator (the tube with the highest indications) were unplugged and ECT performed again. All the results of the absolute method were below 11% while some of those tubes had ECT indications, using the differential method, as high as 83%. At this time two additional tubes were pulled out for laboratory examination.

The utility applied again for a 1500 hours run at 50% power. The application was supported by: (1) the accelerometer measurements showing that the vibration level at 50% power was very small compared with that at 100%; (2) the comparison of ECT results with previous inspections; and (3) the information arising out of scale models. The CSN accepted the application, and later the permit was extended to 2000 hours.

The unit was shutdown and a new ECT examination of the tubes was performed. This time both ECT methods were used again. The comparison of the absolute method readings obtained this time with those obtained in previous shutdowns showed a very small, if any, increase in wear during the 2000 hours at 50% power level. At one time the utility considered the possibility of increasing the power by feeding the steam generators through the auxiliary feedwater nozzle (13% of the flow) while maintaining the previous feeding (50%) through the main feedwater nozzle. Although some modifications and tests were performed the idea was later neglected. After the 2000 hours run, and in as much as the modification to be developed by Westinghouse was far from being ready, the utility applied this time for a 4000 hours run at 50% power. Based mainly in the previous ECT results the CSN granted the permit. Near the end of this period the utility requested an extension of 840 hours to maintain the plant in operation until the modification was ready to be installed in the plant. The permit was granted again by the CSN.
Up to this point the Spanish problems were mainly related with the definition of safe levels of power operation while waiting for a modification. We will discuss briefly what the utility and the regulatory body did to deal with this problem.

First of all, it has been the normal practice of the Spanish Regulatory Body to require that none of the nuclear power plants constructed in Spain would have "first-of-a-kind" components. Every plant must have a "reference plant" that must start operation before the similar plant in Spain. Clearly, this concept did not bring adequate protection since a steam generator model, which has not been sufficiently proven in operation was procurred. The cause might have been the lack of standardization in designs at that time which made difficult to detect the change in the design of the new steam generator.

From the beginning it was noticed that while scale or mathematical models would play an important role in the identification of a safe level of operation, the most reliable element to define such a level would be ECT inspection of the tubes. Therefore, the utility sponsored a programme, in collaboration with another affected utility, to improve the precision of the ECT methods. The CSN has paid special attention to the ECT results in assessing the applications for operation at 50% power. A very small wear increase has been allowed considering that the total avoidance of this type of degradation would be not cost-effective, when other sources of tube degradation would remain, and that a very small wear rate may be an economical problem rather than a safety problem. However, the CSN has carefully reviewed the potential for an active tube to fail either due to excessive wear of the tube itself or due to the previous failure of a plugged tube. Apart from ECT data the mathematical models developed by Westinghouse to predict wear progression were reviewed. It was found that the "best estimate" model could underpredict the wear in the most unfavorable tubes. The last ECT examination of Almaraz1 after operation at 50% power, showed a wear increase exceeding the predictions of Westinghouse, but in any case far from being a safety problem since there were considerable margins in the wall thickness of the tubes before that operational period.

The criteria used to decide about the acceptability of a period of operation at a specific power level were the following:

- Regarding the active tubes, the most worn tube must not suffer
additional wear which will be enough to cause the tube failure under accident conditions;
- The plugged tubes must not reach the point where a complete severance of the tube can happen (since that tube may damage neighbouring tubes);

Two aspects were considered concerning the plugged tubes:
- To determine their actual wear, since they had been plugged based on ECT data taken with a method affected by large uncertainties. Several tubes were unplugged and a correlation between the differential and expected readings was employed.
- To determine if the wear models were applicable to the plugged tubes. By testing and analysis it was shown that both active and plugged tubes would behave in a similar manner.

In summary, the experience of operation at 50% power has shown that, although small wear rates are still present, they are low enough to permit plant operation during short periods of time without endangering the integrity of the tubes.

Some time before the scheduled installation of the modification the utility applied for a license to implement the modification and to reinitiate the operation of the plant. The CSN faced the assessment of the proposed modification at the same time that many other safety reviews had to be done. To reduce the expenditures of resources to a minimum the following approach was used. The topics for review were divided into: (1) Generic, i.e., those applicable to every plant with model D2/D3 steam generators; and (2) Specific, i.e., those applicable to a plant given its physical design or schedule. Since the generic aspects had been reviewed by the Design Review Panel (22 experts) and by the NRC (7 experts) it was considered not cost-effective to proceed to an in-depth review of such aspects. However, to make possible a correct interpretation and application of the DRP and US NRC conclusions, an overview was performed and a continuous contact and collaboration with the U.S. NRC was maintained. Since Sweden had plants with the same problems, a collaboration with the Swedish Regulatory Body, the SKI, was also maintained.

Regarding the specific aspects, the work of the CSN was divided in two main areas: (1) The review required to grant a license for the installation of
the modifications; and (2) The review required to grant a license for the operation of the modification. This split up of the work permitted the completion of the review required to authorize the installation when there were still some points under discussion regarding the operation. However, at the time the installation license was issued there was a reasonable assurance that the operational requirements would be feasible.

Concerning the installation of the modification several aspects received attention: (1) The review of the technical and quality assurance procedures for the installation; (2) The inspection of some of the installation operations; (3) The review of the installation incidents; and (4) Alara requirements.

Regarding the operational requirements of the modification the following aspects were reviewed (using as a reference the U.S. NRC Safety Evaluation Report for Mc Guire-1):

1. Monitoring of pressure oscillations.
   The intent is to monitor all pressure variations which could affect the fatigue usage factors of the bolts and welds. This was accomplished by using the pressure transducers installed in the feedline. Measurements were performed from 0 to 100% power. Power escalation was carried out in 3% increments. Measurements were taken during the period when power was increasing as well as at each 3% increment. The oscillations measured are bounded by those used in the design of the modification and are, therefore, acceptable.

2. Feedwater System changes.
   When main feedwater is initiated during plant warmup, the stagnant colder water downstream of the main isolation valve is injected into the steam generator causing conditions giving rise to usage factors over 1.0 of several manifold bolts. To avoid this problem a new line has been mounted to permit the backflow flushing of the main feedwater line. Since the implementation of the purge may cause other problems a safety analysis covering pipe break, missiles, flooding, containment isolation and design basis events was required. Additionally, operational procedures to use the purge had to be developed.

3. Loose parts monitoring.
   Since the Loose Parts Monitoring System may detect that one of the
manifolds is loose, requirements regarding the operability of the system have been introduced into the Technical Specifications. The requirements are those in the Standard Technical Specifications and in the Regulatory Guide 1.133.

4. Tube vibration monitoring.

Data from the accelerometers were recorded at the following power levels during steady state conditions: 40, 50, 60, 70, 75, 80, 85, 90, 95 and 100%. This data was recorded during the power escalation following the modification. Additional data were recorded at 100% power after three months of equivalent full power operation. Due to the uncertainty in the relationship between tube vibration and wear and the different behaviour of different tubes, no short term acceptance criteria was established. However, the results of these measurements were expected to be useful in assessing the long term potential for the manifold to reduce the wear rate to an acceptably small value.

After the assessment of the data obtained during August of 1983, Westinghouse reached the following conclusions:

- Post-modification plant vibration data are consistent with data obtained from full scale models.
- The data indicate that the instrumented tubes are supported at more plates in the modified configuration than in the unmodified configuration.
- The post-modification values of displacement, peak-to-peak acceleration, and G.Δ at 100% power are comparable to or less than the corresponding pre-modification values at 50% power.

These conclusions are still under consideration by the CSN staff because in tube R49C51 light impacting is observed in the time histories at 86% power, (although as power is increased the tube gains increased support at one or more plates resulting in no impacting at 100% power). In a similar way for tube R49C71 the onset of tube to plate impacting occurs at 86% power, no impacting is observed at 90% power, but light impacting is apparent for 100 percent power.
5. Eddy Current Testing (ECT) Program.

The following surveillance program has been required: In the first inspection after approximately six months of operation the first five rows (45 to 49) and all tubes with indications in previous inspections will be examined. In the second inspection at least two steam generators will be examined including the first two rows (48 and 49) and all tubes with indications in previous examinations. In subsequent inspections only the first row (49) and the tubes with previous indications will be inspected in the steam generators selected according to the technical specifications.

Almaraz-1 was shutdown for refueling in January 27, 1984. An ECT inspection of the first 6 rows of the preheater, 2 rows in the tube bundle periphery and the tubes in a matrix 5 x 5 was performed. As part of it all the previously plugged tubes in the three steam generators were unplugged and ECT inspected. During the unplugging operation some plugs broke-down and had to be machined. After the inspection of the 38 previously plugged tubes it was found that 13 had indications close to or over 40%, therefore they have been plugged again with the use of stabilizers.

The stabilizer is comprised of a length of cable with sleeves swaged onto the cable and also a swaged-on plug. On to one end of the cable is swaged an steel tip. This tip has a pointed end and is the first part of the stabilizer which is to be inserted into the steam generator tube. The positions and spacings between sleeves is such that the stabilizers sleeves rest opposite to the preheater baffle plates when the stabilizer is installed. The sleeves are spaced so that the stabilizer can be bent and manipulate easily inside the channelhead for simple installation into the tube.

According to Westinghouse stabilizers have undergone an extensive testing and qualification program which have demonstrated satisfactory performance. In addition to material testing, the stabilizer performance was tested by random vibration, flow induced vibration and autoclave wear testing. Although it is expected that reduction in tube vibration will reduce tube wear rate, it was not the requirement for the design of the stabilizer qualification. Continued wear of a stabilized tube is not of concern since the tube is inactive due to the plug used with the stabilizer and the tube motion is arrested and prevented from forming loose parts by the sleeved cable.
Although plugged tube stabilization may not be strictly required the utility believes that with this solution plugged tube surveillance will not be needed. It appears that similar stabilizers had been installed at other plants like: Catawba-1, Watts Bar-1 and Byron-1 in the United States and Krsko in Yugoslavia.

The ECT results showed no wear change for most of the inspected tubes, the rest of the tubes had wear increments ranging from -3% to +4% of the nominal wall thickness. These increments are within the range of ECT measurement uncertainties.


The visual inspections are intended to provide an early indication of any unexpected loss of structural integrity of the manifolds. The inspections are performed in accordance with ASME XI, IWA-2211. Visual Examination VT-1, using a fiber optics boroscope. A visual inspection of the manifold is required every time the steam generators are undergoing ECT inspection.

After shutdown in January 1984 a visual inspection of the modification was performed. Although some deposits, erosion and slightly increased gaps were found they had been considered as acceptable.

Once the refueling shutdown was completed Almaraz-1 went back to operation. The presently approved surveillance program requires ECT of the tubes and visual examination of the manifold in subsequent refueling outages.

Regarding Almaraz 2, and Ascó 1,2 all these units have installed the modification after a review process similar to the one described for Almaraz-1 with several differences:

- Since these units were not in operation at the time the Ringhals-3 leak occurred they had no worn tubes, neither was necessary to approve partial power operation.
- Of these units, Almaraz 2 and Ascó 1 were close to be lead plants in Spain after the manifold installation, therefore they have installed accelerometers and have been required to perform tube vibration measurements. Ascó 2 does not have accelerometers.
- Since there is a reasonable assurance that the modification
substantially reduces the vibration level these units will probably be exempted from a shutdown for ECT after six equivalent full power months.

Pressure oscillation monitoring, feedwater system changes, loose parts monitoring, eddy current testing and visual inspection requirements are basically the same for these units.
3. CONCLUSIONS.

This paper has presented the licensing requirements and actions of the CSN regarding the tube vibration problem in steam generators in Spain. There were three distinct stages in the licensing process of this matter:

- Definition of safe levels of power operation while a modification was not ready (1981 & 1982)
- Licensing of the modification installation and operational requirements (1983)
- Surveillance program of the modification performance. (1983, 1984 & beyond)

Throughout the whole license process the main idea was to establish the appropriate licensing requirements to return the plant conditions to those existent before the vibration problem appeared, conditions that had been considered acceptable based on the plant safety analysis.

It has been possible to carry out a reasonable review with limited resources due to the excellent international cooperation between the countries affected by this problem.
SESSION 7.5: THE SWEDISH REGULATORY APPROACH TO VIBRATION-INDUCED TUBE WEAR PROBLEMS

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1. EVALUATION OF SAFETY CONCERNS

The vibration induced wear of tubes in Westinghouse type D3 steam generators was first detected in the Swedish Ringhals 3 plant in October, 1981. The inspectorate immediately notified authorities in other countries with similar types of steam generators. Subsequently, the generic nature of the problem was confirmed by measurements in the Almaraz plant in Spain. It was realized that the rapid wear at tube support plate positions was a new type of steam generator tube failure mechanism to which old safety concepts, plugging criteria, etc could not be applied uncritically.

The inspectorate's initial analysis of the safety aspects of the vibration induced wear phenomena led to the following conclusions, endorsed by the inspectorate's advisory committee on reactor safety:

The wear mechanism was such that many tubes could be worn thin simultaneously, substantially increasing the probability of single or multiple tube failure. There was also an increased risk that plugged tubes were worn through, generating loose parts which in turn might damage earlier unaffected tubes. Probabilistic risk assessments and other safety studies pointed unambiguously to the safety significance of steam generator tube breaks as initiating events in transient sequences
involving high operator stress and possibilities to bypass the containment by blowing mixed primary/secondary steam into open air through the main steam safety valves. Hence, the design deficiencies in Ringhals 3 and the number of already worn tubes raised obvious safety concerns that must be addressed before continued operation of Ringhals 3 could be granted by the inspectorate. This conclusion was reached in accordance with the inspectorate's basic safety policy which states that before reactor operation is resumed after a planned or unplanned shutdown, the following criteria must be fulfilled:

- the extent of any faults discovered, which are considered relevant to safety, must be determined with acceptable precision
- the mechanism causing the faults must be understood, to the extent that a prognosis of fault growth rates can be made
- necessary remedial and control measures - at least of a provisional nature - must be implemented on the basis of the estimated extent and growth rates. Some uncertainties in these estimates can be accepted if it can be shown that the probability of serious consequences with regard to safety is low.

The implementation of this basic safety policy of the inspectorate in the case of Ringhals 3 is described in the following sections.

Some confirmation of the safety concerns expressed in the inspectorate's first evaluation, mentioned above, was provided by the January 25, 1982 incident\textsuperscript{2} at the Ginna plant in the United States.

2. DETERMINATION OF THE EXTENT OF THE WEAR

The problem of determining the extent of the wear with acceptable precision was addressed by applying a new eddy current evaluation technique presented by Zetec Inc. and the Swedish State Power Board (SSPB). The new evaluation technique was based on amplitude measurements instead of phase angle measurements. Calibration standards tailored to
the specific problem, i.e. determining circumferential wear in the tube support position, were developed. The inspectorate stressed the importance of intercalibration with micrometer measurements on pulled tubes which had been eddy current tested before pulling. A number of such intercalibrations were subsequently performed, substantially improving confidence in the eddy current measurements. Also, computerization of eddy current test read-outs and comparison with calibration standards have reduced measurement uncertainties. Such improved measurement techniques are of course also beneficial to the utility in their assessment of deterioration rates and remedial actions with respect to steam generator operational performance.

3. ANALYSIS AND MODELLING OF VIBRATION PHENOMENA

Obviously, the modelling of flow patterns and so called fluid-elastic instabilities used by the vendor in the design phase could not be accepted. Using its main office engineering staff and its Aelvkarleby hydrological laboratory the SSPB launched a substantial programme, carried out together with westinghouse and aimed at better understanding and modelling of the vibration phenomena to be able to resume operation at reduced power and to evaluate more long term remedial measures. The full scale experimental model built at Aelvkarleby soon revealed the complex and turbulent flow patterns at the feedwater inlet in the preheater section. To facilitate the safety evaluation of the inspectorate, basic assumptions, models and experimental results were discussed at frequent meetings between SSPB and inspectorate technical staff. The position taken by the inspectorate was that operation at reduced power, pending design modifications, could be resumed only if it was shown there was a low probability of further damage to the tubes so as not to affect long term solutions adversely. Inter alia, it was pointed out by the inspectorate that the increased clearance between tube and tube support plate due to wear could impair the tube support situation allowing for longer free vibrating lengths and hence higher future wear rates.

Based on an evaluation of experimental results from Aelvkarleby and a comparison with measured wear from the first period of operation in
Ringhals 3 and Almaraz, SSPB proposed a model for predicting wear rates as a function of main feedwater flow. Allowing for the uncertainties in the knowledge available at that time, it was concluded that operation at 40% of full main feedwater flow for 1500h would result in a very low risk of further wear, except perhaps to a few tubes which might be especially exposed in terms of local flow pattern and large clearance between tube and tube support plate. The first wear prediction models were then gradually improved, using more detailed and sophisticated measurements from Aelvkarleby, vibration data from instrumented tubes in Ringhals 3 and other reactors and, finally, eddy current measurements on large numbers of tubes in different places after extended periods of operation at reduced power.

4. TUBE PLUGGING AND TUBE PULLING CRITERIA

As already pointed out, plugging criteria developed for other types of steam generator tube problems could not be uncritically applied to the vibration induced wear situation. The following approach was adopted by the inspectorate:

- For each operating period between eddy current inspections there should be adequate safety margins against tube breaks for unplugged tubes. Based on NRC Regulatory Guide 1.121 it was concluded that a remaining wall thickness of 35% would provide sufficient structural integrity margins for all operation and transient conditions. To these 35% were initially added a 15% margin for measurement and calibration uncertainties and another 10% to provide margins for additional wear. Hence, tubes with a remaining wall thickness less than 60% should be plugged.

- For each operating period there should furthermore be adequate margins against tube breaks for already plugged tubes so as to prevent risk of generating loose parts. Calculations showed that the wall thickness of 15% is a reasonable structural integrity limit for such tubes. Adding margins for measurement uncertainties and additional wear it was concluded that tubes with remaining wall thickness smaller than 35% should be pulled.
On the first plugging occasion (March 1982) the inspectorate prescribed use of both the amplitude and phase angle method, whichever gave the most stringent requirement for plugging. Later on, as experience with the amplitude measurement method grew, the inspectorate in effect dropped the phase angle method for evaluation of indications in tube support plate positions. The original margin of about 15% for measurement uncertainties was reduced as more intercalibration data from pulled tubes became available. However, the main part of the original margin was proved prudent as the new data showed that earlier calibration curves had been non-conservative in the high-wear region.

5. THE "TELL-TALE" PLUG CONCEPT.

Obviously, the inspectorate was concerned about possibilities to monitor if the most worn tubes - after plugging - were subject to continuous wear as they could be expected to be the most exposed tubes in terms of adverse flow patterns and support situation. Continuous monitoring during operation of any developing leak in those tubes would provide early warning of:

- Gross errors in the prediction of maximum wear rates at the chosen power.

- Rapidly increasing risk of generating loose parts.

In response to the inspectorate's concerns about such continuous monitoring, so called "tell-tale" plugs were proposed by SSPB. These plugs have a small hole providing a controlled primary to secondary leak path if a plugged tube is worn through. The leakage rate is limited by the size of the hole to values above detection limits but well below the rate normally allowed in the technical specifications. Such plugs were used on all tubes requiring plugging. The "tell-tale" plugs can be described as an independent safety net, catching abnormal tube wear. Obviously, the use of "tell-tale" plugs eliminated any immediate need for stabilization of plugged and worn tubes, thus providing time to develop and evaluate long term solutions to the worn tube problem.
In this context it might also be mentioned that loose part monitoring systems have been installed on all Ringhals PWRs.

6. OPERATIONAL PERMITS FOR RINGHALS 3 BEFORE MODIFICATION

In April 1982, the inspectorate granted Ringhals 3 an operating permit for 1500h (e.g. about 2 months) at 40% power. In its decision the inspectorate cited the improved modelling of the wear mechanism presented by SSPB, based on Aelvkarleby data and on the first vibration measurements in Almaraz, as well as the application of the plugging criteria described above using "tell-tale" plugs. After the 1500h of operation Ringhals 3 was shut down for eddy current testing in June, 1982. Although some additional wear could not be fully excluded, both the eddy current tests and the vibration measurements in instrumented tubes during operation indicated that previous estimates of maximum wear rates had been conservative. Hence, an operational permit for an additional 1500h period of operation was granted in August, 1982. In September, an unplanned shutdown and eddy current inspection took place due to the detection of loose parts in the steam generators - without connection with vibration induced wear. The operational permit was then extended to Christmas, 1982, when the reactor was shut down for renewed eddy current testing. The eddy current testing also included previously plugged tubes after removal of the plugs. Furthermore, one worn tube was pulled, providing improved calibration data for the eddy current measurements. As the eddy current tests indicated very small - if any - wear, an operation permit for another 2000h, i.e. about 3 months, at 40% power was granted in January 1983. The operation permit was later extended to June 1983 without requiring any inspection before that time. The extension was based on increasing operational experience from other reactors of the same type.
7. REVIEW OF STEAM GENERATOR MODIFICATION

The inspectorate's review of the modified feed water inlet design proposed by Westinghouse consisted of two parts. The first part was a review of the structural integrity of the presented manifold design. This review is outside the scope of this presentation. The other part was a review of the functional efficiency of the manifold with respect to reduced vibration-induced tube wear.

The SSPB and Westinghouse evaluation of the functional efficiency of the manifold design was largely based on experimental data from the Aelvkarleby laboratory, comparing flow and vibration patterns at various main feedwater flows with corresponding data without manifold. Based on the results of such comparisons, SSPB proposed operation after the modification at 100% power with a maximum of 82% of main feedwater flow through the pre-heater section of the steam generators; the remainder of the feedwater being fed through the top nozzle normally used only during start-up.

On February 11, 1983, the inspectorate granted a construction permit for the modification of Ringhals 4 - the first plant modified in Sweden. The modification was completed by the end of March. Ringhals 3 was modified in the same way during the summer of 1983.

8. OPERATIONAL PERMITS FOR RINGHALS 3 AND 4 AFTER MODIFICATION

Before modification, the inspectorate had only permitted Ringhals 4 to run low power testing, not using the preheater inlet, before modification. During the period April - August, 1983, Ringhals 4 went through a normal start-up programme with stepwise power increases. The reactor was shut down in August for inspection including a base line eddy current test. In September the inspectorate granted an operation permit for 3000h at 100% power with a maximum feedwater flow of 82% through the preheater section. At the end of 1983 Ringhals 4 was shut down for eddy current testing. No tube wear could be detected. The inspectorate then granted an extension of the operation permit to the normal refueling and maintenance shut-down planned for August, 1984.
After completion of the modification of Ringhals 3 in September 1983, the inspectorate granted an operational permit for 3000h at 100% with a maximum feedwater flow of 82% through the pre-heater section. On the basis of the eddy current test results received during this period from Ringhals 4, McGuire and, especially, Almaraz (the only other operating plant with extensive tube wear), the inspectorate granted a continued operation permit without requiring shut-down for testing until the normal maintenance and refueling outage, starting May, 1984. Eddy current tests performed during the outage showed that some tube wear could not be excluded, but that any additional wear from one full fuel cycle of operation could be expected to fall within the margins already provided in the plugging criteria. No additional plugging was required and the operational permit was extended to the next refueling and maintenance shut-down in 1985.

9. CONCLUDING REMARKS

Throughout the operational history of Ringhals 3 and Ringhals 4, the inspectorate has thus adjusted operation periods and inspection intervals as growing operational experience demonstrated that early predictions of wear rates were on the conservative side. It should be pointed out in this context that it is the maximum wear rates of the most exposed tubes that are of interest from a safety point of view. Analysis of the vibration-induced wear phenomena shows that maximum wear rates are to a considerable extent determined by factors of a probabilistic nature, such as variations in local flow patterns and the individual tube support situation within the manufacturing tolerances. Hence, the statistical spread in wear rates is more interesting than simple estimates of mean values. The inspectorate’s requirement to use "tell-tale" plugs for all worn tubes is an example of the conclusions drawn by this analysis.
When faced with a new problem with obvious safety aspects, such as the vibration-induced wear, it is technically sensible from a safety point of view to resume operation only under stringent conditions with respect to power limits, inspection intervals, etc, keeping margins on the safe side with respect to uncertainties in the detailed understanding of the problem and in the efficiency of applied remedial measures. As operational experience resolves such uncertainties, the operational limitations should then be rapidly restored to normal.

Rapid international sharing of operational experience between regulatory authorities as well as between utilities is a key factor in this process. In this respect, the granting of the continued operation of Ringhals 3 for a period of 8 months after modification through step by step inspectorate decisions provides a good example. Considering that Ringhals 3 has the most worn tubes of all the reactors affected, this long operation period without eddy current testing could hardly have been granted if rapidly disseminated operational experience from other reactors had not been available, showing that actual wear rates (if any) and their statistical spread were well within the safety margins provided in the plugging criteria. Thus the experience with the vibration-induced wear problem has shown the great value to both authorities and utilities of a well established international information exchange network - both of a formal and informal nature.

10. REFERENCES

1. SKI Review Memorandum on damaged tubes in Ringhals 3, (Dec 16, 1981)


3. SKI Review Memorandum on resumed operation of Ringhals 3 at limited power, (March 31, 1982)

4. SKI Review Memorandum on extended operation of Ringhals 3, (March 15, 1984)
SESSION 7.6: UNITED STATES NUCLEAR REGULATORY COMMISSION
INTEGRATED PROGRAM FOR THE RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3, A-4, AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY

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1 INTRODUCTION AND SUMMARY

This paper presents a summary of the NRC staff activities relative to resolving Unresolved Safety Issues A-3, A-4, and A-5. The staff's recommended resolution of these issues is currently under review by the Commission and is therefore not complete at this time.

1.1 Background

Degradation of steam generators (SGs) manufactured by each of the U.S. three pressurized water reactor (PWR) vendors has resulted from a combination of problems related to steam generator mechanical design, materials selection, fabrication techniques, and secondary system design and operation. To date, many different forms of steam generator tube degradation have been identified including: stress corrosion cracking, wastage, intergranular attack, denting, erosion-corrosion, fatigue cracking, pitting, fretting, support plate degradation, and mechanical damage resulting from impingement of foreign objects of loose parts on steam generator internal components. One or more of these forms of degradation have affected at least 40 operating PWRs and have resulted in extensive SG inspections, tube plugging, repair, or replacement.
The majority of the SG tube failures that have occurred under normal operating conditions were small stable leaks; some required plant shutdown, inspection, and corrective actions; but others were small enough (e.g., below the leak rate limit of the technical specifications) so that plant operation continued until a scheduled shutdown. However, four significant SG tube ruptures have occurred in U.S. PWRs since 1975. These events occurred on February 26, 1975, at Point Beach Unit 1; September 15, 1976, at Surry Unit 2; October 2, 1979, at Prairie Island Unit 1; and on January 25, 1982, at R. E. Ginna.

The first three of these events were evaluated, including evaluations of systems response, operator actions, and radiological consequences. The event at the Ginna plant and subsequent plant restart were also evaluated\(^2\),\(^3\). These evaluations include descriptions of the event, U.S. Nuclear Regulatory Commission (NRC) findings; an evaluation of system response, operator response, steam generator inspection and repair programs, emergency preparedness, and radiological consequences.

An overall summary of operating experience of steam generator tubes was prepared by the U.S. Nuclear Regulatory Commission by identifying the types of problems which have occurred in steam generators, that report particular emphasis on recent operating experience\(^4\). In addition, that report briefly addressed the status of the NRC Unresolved Safety Issue Task Action Plant (TAPs) A-3, A-4, and A-5 regarding steam generator tube integrity; the short- and long-term corrective actions being pursued by the industry; and the inspection and repair requirements being recommended to ensure continued safe plant operation and minimization of the associated radiation exposures. An update of that report was the subject of discussion at the first Session of this meeting.

Concerns relative to steam generator tube degradation stem from the fact that the steam generator tubes are a part of the reactor coolant system (RCS) boundary and that tube failures result in a loss of primary coolant. In addition, the steam generator tubes constitute a particularly important part of the RCS boundary since their failure allows primary coolant into the steam generators where its isolation from the environment is not fully ensured. The
release of primary coolant into the environment has two major safety implications. The first is the direct release of radioactive fission products; and the second is the loss of cooling water which is needed to prevent core damage. An extended loss of coolant outside of containment would result in the depletion of the initial RCS inventory and emergency core cooling system (ECCS) water without the capability to recirculate the water as would be the case for any LOCA inside containment.

The U.S. NRC regulations provide a framework with which to assess the importance of steam generator tube integrity. Specifically, the General Design Criteria (GDC) state that the reactor coolant system boundary shall "have an extremely low probability of abnormal leakage" (GDC 14), shall "be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing ... to assess ... structural and leak tight integrity" (GDC32). 10 CFR Part 50, Appendix B, is also pertinent to the maintenance of steam generator tube integrity.

1.2 Development of Integrated Program

Steam generator tube integrity was designated an unresolved safety issue (USI) in 1978 and Task Action Plans (TAP) A-3, A-4, and A-5 were established to evaluate the safety significance of degradation in Westinghouse, Combustion Engineering and Babcock & Wilcox steam generators, respectively. These studies were later combined into one effort because many problems being experienced by PWR vendors were similar. The staff prepared a draft report regarding this issue in early 1982. This draft report primarily considered corrosion-related failure mechanisms, including the "denting" mechanism, since those failures were the main concern during the period when most of the technical studies were performed.

In May 1982, subsequent to the Ginna steam generator tube rupture (SGTR) event, the staff initiated an integrated program to consider the lessons learned from the Ginna SGTR event \(^2,3\) from the three previous domestic SGTR events \(^1\), and to consider the recommendations identified in the draft USI report prepared in early
1982, the objective of the integrated program was to complete resolution of USIs A-3, A-4, and A-5, including identification of staff positions that should be addressed by operating license (OL) applicants and licensees and identification of further efforts that should be undertaken by the NRC. This paper described the results of this integrated program.

To develop an integrated set of requirements that addresses the safety concerns discussed above, the appropriate NRC offices and divisions were asked, subsequent to the steam generator tube rupture at the Ginna plant, to review the steam generator issue and identify recommendations for any generic actions they believed appropriate. After assessing the list of proposed recommendations, issues were grouped into one of the five major subject areas: (1) steam generator integrity, (2) plant system response, (3) human factors consideration, (4) radiological consequences, and (5) organizational response.

The issues within the above five major areas were then categorized with respect to whether they were appropriate for consideration by the staff as potential positions to be addressed by all pressurized water reactors licensed applicants. Issues outside this first category fell into a second general category, which consisted of issues warranting further staff study, issues already receiving attention as part of an existing regulatory program, or issues for which corrective actions had already been taken. Issues within this second category are discussed later in this paper.

The issues within the first category, were grouped into the technical areas listed below and were subjected to a value-impact evaluation by the staff.

1. Prevention and Detection of Loose Parts or Foreign Objects
   (a) Secondary Side Visual Inspection and Improved QA/QC Procedures
   (b) Loose Parts Monitoring System
2. Inservice Inspection of Steam Generator Tubes
   (a) Supplemental Tube Inspections
   (b) Full-Length Tube Inspections
   (c) Denting Inspections
   (d) Steam Generator Inservice Inspection Interval
   (e) Inspections Following Shutdown for Repair of Leakage

3. Improved Eddy Current Techniques

4. Upper Inspection Ports

5. Secondary Water Chemistry Program

6. Condenser Inservice Inspection Program

7. Stabilization and Monitoring of Degraded Tubes

8. Primary to Secondary Leakage Limits

9. Coolant Iodine Activity Limit

10. Reactor Coolant System Pressure Control

11. Safety Injection Signal Reset

12. Containment Isolation and Reset

1.3 Scope of Value-Impact Evaluation

The value-impact analysis of the issues in the first category addressed the potential for reductions in (1) the probability of core melt, (2) the probability of significant radiological releases comparable to NUREG-75-014* PWR

*Formerly WASH-1400
release categories 8 and 9 during SGTR events not leading to core melt, (3) public risk, and (4) occupational radiation exposure (ORE). The potential requirements were also evaluated for their conformance with good operational and engineering practices, enforcability with respect to NRC regulations relating to steam generator tube integrity and SGTR mitigation capability. Economic costs were also a consideration of this analysis. Net economic costs were considered in relation to the benefits to public health and safety. Net economic benefits (deriving from the cost-effectiveness of some of the potential requirements) were considered neither as a barrier nor a basis for the disposition of the potential requirements under consideration.

The NRC's evaluation considered input from a report by the staff's contractor, Science Application, Inc. (SAI), "Value-Impact Analysis of Recommendations Concerning Steam Generator Tube Degradation and Rupture Events." The SAI report provided much of the basis for the staff's evaluation of the effectiveness of the potential requirements in reducing the incidence of steam generator tube degradation, tube ruptures, and occupational radiological exposures (ORE) and for the staff's evaluation of the potential cost benefits and cost impacts.

1.4 Risk from SGTR-Related Causes

The NRC's analysis of risk from SGTR-related causes indicates that the core melt probability from these events is small, no more than $5 \times 10^{-6}$/reactor year (RY) as an industry average. This probability is a relatively small fraction (10% or less) of the overall probability of core melt events from all causes based on probabilistic risk assessments that have been performed for a number of PWRS. The corresponding risk to the public is estimated to be limited to $2.5 \times 10^{-3}$ latent fatalities and $1.1 \times 10^{-5}$ early fatalities per reactor year from steam generator tube rupture accidents associated with core melt.

The relatively low estimates are attributed in part to the effectiveness of existing steam generator related requirements in minimizing the frequency of SGTR occurrences. This includes requirements which have been imposed by the staff for specific plants which have experienced severe degradation of the
steam generator tubes. However, the above estimates are also attributable to conservative operating practices in excess of minimum requirements being implemented by many utilities to varying degrees to minimize the occurrence of excessively degraded tubes.

The staff also evaluated the potential for significant radiological releases (comparable to NUREG-75/014* PWR release categories 8 and 9) during SGTR events not leading to core melt. SGTRs create the potential for steam generator overfill, as evidenced by the Ginna experience, which can lead to steam generator safety valve challenges in which water relief occurs, and the safety values may not properly close. This creates the potential for a prolonged release of radioactive fluid from these valves.

The probability of significant non-core melt releases during SGTRs is estimated to be about $2 \times 10^{-4}$/RY.

1.5 Disposition of Potential Positions

The above generic risk estimates notwithstanding, degradation of steam generator tubing has become widespread throughout the industry where the average age of the steam generator is still less than a quarter of their intended lifespan. Necessary repairs have caused high radiological exposure to workers and now contribute 20% of the average annual plant occupational dose for PWRs. Four SGTR events have already occurred as a direct result of steam generator tube degradation. SGTR events represent a significant challenge to the plant operators since a variety of diagnoses and manual actions must be taken in a relatively short time. In addition, SGTR events have a relatively high potential for causing undesirable radiological releases to the public. It is also likely that some plants have experienced limited periods during which they were vulnerable to rupture in the event of a sudden, loading transient such as a main steam line break.

*Formerly WASH-1400
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<td>1. Prevention and Detection of Loose Parts</td>
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<td>(a) Visual Inspection of Secondary Side</td>
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<td>and Quality Assurance Work Procedures</td>
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<td>(b) Loose Parts Monitoring System</td>
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<td>2. Steam Generator Tube Inservice Inspection</td>
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<td>(b) Full Length Inspections</td>
<td>Industry Action*</td>
</tr>
<tr>
<td>(c) Denting Inspections</td>
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</tr>
<tr>
<td>(d) Steam Generator ISI Interval</td>
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</tr>
<tr>
<td>(e) Inspections following Shutdown for Repair of Leakage</td>
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<tr>
<td>3. Improved Eddy Current Techniques</td>
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<tr>
<td>4. Upper Inspection Ports</td>
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<tr>
<td>5. Secondary Water Chemistry Program</td>
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<td>6. Condenser Inservice Inspection</td>
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<tr>
<td>7. Stabilization and Monitoring of Degraded Tubes</td>
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<tr>
<td>8. Primary to Secondary Leakage Limit</td>
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</tr>
<tr>
<td>9. Coolant Iodine Activity Limit</td>
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<tr>
<td>10. Reactor Coolant System Pressure Control</td>
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</tr>
<tr>
<td>11. Safety Injection Signal Reset</td>
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</tr>
<tr>
<td>12. Containment Isolation and Reset</td>
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</table>

* These items have been incorporated as staff positions.
The NRC has concluded that improvements from past performance are needed to further ensure continued compliance with its regulations relating to steam generator tube integrity, SGTR mitigation, and to reduce occupational exposures.

As a result, the integrated program herein, the potential positions identified previously were categorized as (1) items warranting industry action (2), items warranting further study or development as staff actions, and (3) items which should not be proposed as industry or staff actions. A summary of the staff's dispositions of the potential positions is provided in Table 1.

The staff has concluded on the basis of the value-impact evaluation that the indicated actions as a group will be effective in significantly reducing (1) the incidence of tube degradation, (2) the frequency of tube ruptures and the corresponding potential for significant non-core melt releases, and (3) occupational exposures, and are consistent with good operating and engineering practices. Adoption of these measures would further reduce public risk and provide added assurance that risk is and will continue to be small. In addition, these measures would be expected to reduce the need for plant-specific licensing actions by the staff to ensure that plants with unusually severe or active degradation can continue to be operated safely.

Of those actions listed above, items 1a, 5, 6, and 11 were found to be the most effective actions for reducing tube degradation, ruptures, offsite releases, core melt, and occupational radiological exposures.

1.6 Staff Actions

The integrated program has identified a number of staff actions and studies related to steam generator tube integrity, plant systems response, human factors considerations, radiological consequences, and organizational response to events. These staff actions are summarized in Table 2. A number of these actions have been completed, as noted in Table 2. Other staff actions identified in Table 2 involve broad generic issues extending beyond strictly steam generator related issues and are being addressed by other existing regulatory programs.
### TABLE 2
**RELATED STAFF ACTIONS AND STUDIES** *(1)*

<table>
<thead>
<tr>
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<tr>
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<td>4.6</td>
<td>6 of 7 Complete</td>
</tr>
<tr>
<td></td>
<td>Note 5</td>
</tr>
</tbody>
</table>

**NOTES:**

1. These tasks are discussed in detail in Section 4 of Draft NUREG-0844.
2. Not high priority. Will be scheduled pending availability of staff resources.
3. A proposed rule relating to this issue is currently under review by the Commission.
4. Not scheduled pending completion of tasks described in Sections 4.3.1 and 4.5.1.
5. 6 of 7 Completed. Remaining task by 06/84.
The remaining staff actions identified in Table 2 involve other issues related to steam generators which the staff finds warrant further study or action by the staff. Completion of these tasks may lead to additional staff proposals if warranted on the basis of additional reductions in risk or to ensure compliance with the NRC regulations. However, those tasks do not appear to involve issues relating to either (1) major reductions in the degree of protection to public health and safety, or (2) significant risk to public health and safety. Thus, resolution of USIs A-3, A-4, and A-5 are not contingent on completion of these actions.

1.7 Conclusions Stemming From the Integrated Program

1. The integrated program has found public risk from SGTR related causes to be small.

2. The integrated program has re-affirmed the need for the fundamental regulatory requirements pertaining to RCS boundary integrity in assuring public health and safety, including the need for in-service inspections and preventing maintenance on steam generator tubes and for SGTR mitigation.

3. The NRC has developed additional potential actions identified in Table 1 as warranting industry action were found to be effective reducing (1) the incidence of tube degradation, (2) the frequency of tube ruptures and the corresponding potential for significant non-core melt releases, and (3) occupational exposures, and are consistent with good operating and engineering practices.

4. The NRC has identified a number of broad generic issues extending beyond strictly steam generator related issues which are being addressed as part of other regulatory programs. Resolution of these issues is considered to be outside the scope of USI A-3, A-4 and A-5 resolution. In addition, the NRC has identified other steam generator related issues warranting further study or action by the staff.
Although these tasks may lead to additional proposals by the NRC if found to be warranted, resolution of USIs A-3, A-4, and A-5 is not contingent upon completion of these tasks. Table 2 provides a summary of the followup staff actions and studies.
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STEAM GENERATORS

STOCKHOLM, SWEDEN 1–5 OCTOBER 1984

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