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L'ÉNERGIE ATOMIQUE
DU CANADA LIMITÉE

STEAM GENERATOR TUBE PERFORMANCE: EXPERIENCE WITH WATER-COOLED NUCLEAR POWER REACTORS DURING 1978

Performance des tubes de chaudière: expérience acquise en 1978 dans les réacteurs nucléaires de puissance refroidis par eau

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Résumé

On a passé en revue la performance des tubes de chaudière employés en 1978 dans les réacteurs nucléaires de puissance refroidis par eau. Des défectuosités de tube ont été relevées dans 31 réacteurs sur les 86 ayant été passés en revue. On décrit dans le présent rapport la cause de ces défectuosités et les méthodes conçues pour en venir à bout. On a constaté que le nombre des tubes bouchés en 1978 était spectaculairement inférieur à celui de l'année précédente. Cette amélioration est attribuable à l'application assidue des techniques développées au cours des récentes années à partir de l'expérience acquise *in situ* et de programmes de R&D.

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ABSTRACT

The performance of steam generator tubes in water-cooled nuclear power reactors has been reviewed for 1978. Tube failures occurred at 31 of the 86 reactors surveyed. Causes of these failures and procedures designed to deal with them are described. A dramatic decrease in the number of tubes plugged was evident in 1978 compared to the previous year. This is attributed to diligent application of techniques developed from in-plant experience and research and development programs over the past several years.

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INTRODUCTION

Steam generators are amongst the largest and the most critical components in nuclear power plants. Within them, several thousand thin-walled ($\sim$1.2 mm) tubes separate primary and secondary water circuits in an environment which is perhaps more conducive to corrosion than any other in the plant. Failure of a single steam generator tube often requires a lengthy shutdown for location and plugging. This maintenance work must be performed in considerable radiation fields and cramped quarters. In addition, the number of tubes removed from service in some plants is approaching the 20-25% reserve margin after which either progressive derating or tube bundle replacement becomes necessary. It has been estimated that 10 plants have reached the point where it is questionable whether or not the steam generators will survive to the end of the design life (1). Understanding tube failure mechanisms and controlling the rate of deterioration has potential for very high pay-off in terms of radiation exposure, power generation economics and fossil fuel conservation.

The performance of steam generator tubes during 1978 has been reviewed. Data on tube failures, their causes and methods designed to alleviate them are presented as in previous surveys conducted by Atomic Energy of Canada Limited (2-8). Water-cooled reactors of more than 50 MW(e) (except for NPD*) with at least 100 effective full-power days (EFPD) of operation at 1978 December 31 are included. Reactors in Eastern Bloc countries have been excluded because of paucity of data.

Appendix 1 lists 87 reactors and their cumulative experience with steam generator tubes. The reactors surveyed for 1978 experience are of the following types:

- 67 pressurized water reactors (PWR)
- 13 pressurized heavy water reactors (PHWR)
- 5 boiling water reactors (BWR)
- 1 water-cooled, graphite-moderated reactor

KWL Lingen is included for completeness although it is now permanently shutdown. Seven PWR's and one PHWR have been added to the survey including Korea's first power reactor and the first two of what is expected to be a long line of French PWR's.

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*NPD - Nuclear Power Demonstration Reactor, Rolphton, Ontario, 25 MW(e)

Steam generator tubes were plugged at 33 reactors, not much different than in 1977 but the total number of tubes plugged was 1247, a considerable improvement over the 4360 tubes plugged the previous year. In two reactors (Biblis A and Mihama-2) 5 tubes were removed for examination but no defects were found.

SURVEY OF 1978 FAILURES

Experience at power reactors in which steam generator tubes failed during 1978 is described below. A summary of this experience is presented in Table 1.

ARKANSAS-1, USA

Routine in-service inspection was performed during the 1978 March refuelling outage. Automated eddy-current testing of the full length of 21% of the tubes in steam generator A and 9% of those in steam generator B revealed that 5 tubes in the A steam generator had thinning in excess of 40% of the nominal tube wall thickness. Four of these tubes were plugged and the fifth removed for further examination. Three defects were located on tubes adjacent to the open lane (where one row of tubes was omitted during manufacture to facilitate inspection), near the upper tubesheet while the others were between support plates. Failure was caused by cracking and one tube may have failed by stress corrosion cracking.

Cracking, especially near the upper tubesheet, has been a problem at other reactors with Babcock and Wilcox designed once-through steam generators. It appears that a time-dependent mechanism is responsible giving rise to cracks with significant through-wall penetration between 700 and 800 effective full-power days of operation.

BEZNAU-2, SWITZERLAND

Seven tubes were plugged in steam generator B at Beznau-2 during 1978. One defect was caused by secondary-side stress corrosion cracking within the tubesheet crevice and four were caused by phosphate wastage above the tube-sheet. The cause of failure and location of the other two could not be determined.

CRYSTAL RIVER-3, USA

Failure of a burnable poison rod assembly generated debris detected by the loose-parts monitor in the upper head of steam generator B. Following cooldown, seventeen pieces of hardware on the upper tubesheet were identified to be parts of the burnable poison rod assembly.

Inspection of the steam generator A upper tubesheet showed no loose parts or damage to tube ends. Eddy-current testing, leak testing and free path checks confirmed that tubes had not been damaged. One tube, found to have a defect ($\sim$40% wall thickness), was plugged.

Repair of the steam generator B was performed in several well-rehearsed phases. After a distilled water rinse, the dose rate above the tubesheet was 3.0-3.5 R/h. To minimize radiation exposure, lead blankets were laid on the tubesheet while others were supported in the head by a dome-shaped frame. Tube ends were repaired by exposing small sections of tubesheet. This procedure halved the estimated exposure, resulting in a total of 126.2 rem for all repair work on steam generator B. A free path check of all tubes showed 19 with potential blockage. Of these, 7 were found to be clear, debris was dislodged from 5 and 7 were removed from service by plugging. Eddy-current testing was performed on the tubes from which debris had been removed and on an additional 3% of tubes. No serious defect indications were observed and the reactor was returned to power (9).

DOEL-1,2, BELGIUM

Fourteen tubes were plugged in Unit 1 because of damage caused just above the tubesheet by foreign material inadvertently left in the secondary side during manufacture. During 1977 and 1978, 24 failures have been attributed to this cause.

In Doel-2, 27 tubes were plugged because of stress corrosion cracking within the tubesheet crevice. Tubes in these steam generators are rolled once, leaving a long crevice on the secondary side. Stress corrosion cracking occurs at the high stress region of the rolled joint.

Table 1 - SUMMARY OF STEAM GENERATOR TUBES PLUGGED DURING 1978

REACTOR	TUBES PLUGGED	FAILURE CAUSE	FAILURE LOCATION	SECONDARY CHEMISTRY CONTROL	CONDENSER COOLING WATER	CONDENSER LEAKS	COMMENTS
Arkansas-1	5	1SCC, 4UD	TS, between TSP	AVT/CD	fresh	no	OTSG
Beznau-2	7	1SCC, 4W	TS, above TS	AVT	fresh	no	3 leakers
Biblis B	3	--	--	PO_4	fresh	no	removed for examination
Crystal River-3	8	1UD, 7 debris	TSP	AVT/CD	sea	NR	OTSG
Doel-1	14	debris	above TS	AVT/CD	sea	no	1 leaker
Doel-2	27	SCC	TS rolled joint	AVT/CD	sea	no	9 leakers
Dresden-1	15	UD	UD	CD	fresh	no	BWR, 15 leakers
Farley-1	1	manufacturing defect	near U-bend	AVT	fresh	yes	leaker
Fort Calhoun-1	3	EC indications	TSP	AVT	fresh	yes	
Ginna	24	W, ID defects	above TS, bundle periphery	AVT/CD	fresh	no	1 leaker, 1 removed for examination
Indian Point-2	30	D	TSP	AVT	brackish	no	22 removed for examination
Indian Point-3	4	D	TSP	AVT	fresh	yes	3 damaged by welder
KWO Obrigheim	32	2SCC, 30UD	near TS, TSP and U-bend	AVT	fresh	yes	4 leakers
Maine Yankee	15	--	--	AVT	brackish	yes	plugged for modification
Mihama-2	2	--	--	AVT	sea	yes	removed for examination
NPD	9	fretting	TSP	PO_4	fresh	NR	
N-Reactor	8	UD	UD	AVT/CD	fresh	yes	partial CD, leaks in SS tubes
Oconee-1	18	UD	various locations	AVT/CD	fresh	NR	OTSG
Oconee-2	5	fatigue	upper TS	AVT/CD	fresh	NR	1 leaker, OTSG
Point Beach-1	13	1D, 10SCC, 2UD	TSP, within TS, UD	AVT	fresh	yes	4 leakers
Point Beach-2	2	SCC, UD	near TS	AVT	fresh	yes	1 leaker
Ringhals-2	1	D	TSP	AVT	sea	yes	
Robinson-2	21	W	above TS	PO_4	sea	yes	2 leakers
San Onofre-1	18	11D, 5W, 2 fretting	TSP, above TS, U-bend	PO_4	sea	yes	1 leaker

Table 1 - cont'd

REACTOR	TUBES PLUGGED	FAILURE CAUSE	FAILURE LOCATION	SECONDARY CHEMISTRY CONTROL	CONDENSER COOLING WATER	CONDENSER LEAKS	COMMENTS
SENA (Chooz)	3	fretting	U-bend	AVT	fresh	yes	
Surry-1	167	D, W	TSP, TS	AVT/CD	brackish	NR	5 leakers
Surry-2	200	D	TSP, TS	AVT/CD	brackish	NR	
Takahama-1	35	25SCC, 10UD	within TS, U-bend	AVT	sea	no	
Three Mile Island-1	2	manufacturing defect	above 3rd TSP	AVT/CD	fresh	NR	OTSG
Trojan	1	UD	UD	AVT/CD	brackish	no	leaker
Turkey Point-3	2	D, W	TSP, above TS	AVT	sea	yes	1 leaker
Turkey Point-4	532	D, W	TSP, above TS	AVT	sea	yes	2 leakers
Yankee Rowe	30	W	above TS	AVT	fresh	yes	2 leakers

AVT - all-volatile treatment
BWR - boiling water reactor
CD - condensate demineralization
EC - eddy current
ID - inside diameter
NR - not reported
OTSG - once-through steam generator
PO_4 - phosphate treatment
SCC - stress corrosion cracking
TS - tubesheet
TSP - tube support plate
UD - undetermined
W - phosphate wastage

Steam generator tubes were inspected by automated, multi-frequency, eddy-current methods and plugging was performed by manual welding.

Chemistry experience at Doel and Tihange has been reviewed by Roofthooft (10).

DRESDEN-1, USA

Fifteen leaking tubes were plugged during 1978. Neither cause nor location were determined, but the pattern follows that of previous failures and is suspected to be caused by stress corrosion cracking of the stainless steel tubes.

FARLEY-1, USA

Primary to secondary leakage was suspected in late 1978 and confirmed early in 1979. One tube was found to be leaking at a point approximately 50 mm from the U-bend on a short radius tube. Cause of failure was attributed to a manufacturing defect. Hydro testing and eddy-current inspection were performed to ensure absence of other defects.

FORT CALHOUN-1, USA

Eddy-current inspection revealed tube-wall deterioration and a distorted support plate in steam generator A. Three tubes were plugged. This is the first occurrence of tube-related steam generator problems in the Fort Calhoun plant in over 1000 effective full-power days of operation.

GINNA, USA

During operation at 100% power a primary to secondary leak was indicated by air ejector and blowdown radiation monitors. The leaking tube, identified by hydro testing on steam generator B, was plugged and eddy-current testing was performed on several hundred others. Seven additional tubes showed degradation greater than 40% of wall thickness and were explosively plugged. The 6 defects caused by phosphate wastage were known to be present and were plugged because of continuing attack. These were located near the centre of the tubesheet.

The leaker and one tube with a 50% wall penetration had primary side defects and were located on the tube bundle periphery. The leaking tube was removed for metallographic analysis.

One tube with a primary side indication of 20% to 40% wall thickness was removed from steam generator B during the spring refuelling and maintenance outage. Multi-frequency eddy-current testing (Intercontrole) showed significant defects in 1 tube in steam generator A and 10 in steam generator B. All tubes in steam generator B had defects on the inner wall and these were plugged along with 5 which were damaged during extraction of the sample tube.

The defect in steam generator A was a small lap due to manufacturing but was plugged to prevent confusion in future in-service inspections. No tubes have been plugged in steam generator A since 1977 April and no primary side defects have been observed. Primary side cracking of tubes in steam generator B was first detected in 1977 and 20 tubes have been plugged to date. It is believed that a stress mechanism is responsible for these cracks but no corrodent could be postulated in the primary coolant which would contribute to stress corrosion cracking. However, primary side stress corrosion cracking induced by high stresses has been observed at other reactors (Obrigheim, Surry-1, -2 and Turkey Point-3,4). Denting at the tube support plates was measured and found to be less than previously thought. There is no evidence of dent growth and this is attributed, in part, to the excellent secondary chemistry control achieved since commissioning of the full-flow condensate demineralizers in 1977 December.

INDIAN POINT-2,3, USA

At Unit-2, 2 tubes in steam generator 23 and 5 in steam generator 24 were plugged because of constriction at the support plates. In-service inspection showed little change in the severity of tube denting and the condition of tube support plates since the previous year.

The first in-service inspection of steam generators at Indian Point-3 showed minor denting, in some areas too little to be quantified. Three tubes were plugged because of arc damage caused by a welder during modification. Late in 1978 a primary to secondary leak occurred on steam generator 33. One tube was found to be leaking at the second tube support plate and would not pass a 13.7 mm probe, the smallest available. This tube had been examined during the in-service inspection and a dent was observed at the 2nd support plate. Testing of tubes in the vicinity of the defect showed no progression of denting in other tubes.

KWO OBRIGHEIM, FRG

Of 32 tubes plugged at Obrigheim, 2 were caused by secondary side stress corrosion cracking above the tubesheet. The cause of the other defects was unknown. These were located at tube support plates (22 tubes), the U-bend (4 tubes) and 4 tubes for which the location could not be ascertained.

MAINE YANKEE, USA

Five tubes were removed from service in each of the three steam generators to effect modifications to partial support plates where some denting had been observed (8). These plates are fastened to the shroud by lugs and have a non-perforated rim. To relieve stresses it was decided to cut these where possible. Consequently 5 tubes had to be strengthened ("staked") and plugged to provide support.

NPD, CANADA

The steam generator in this prototype CANDU reactor consists of horizontal U-tubes in a U-shaped shell connected to an overhead steam drum by risers and downcomers. A primary-to-secondary leak occurred early in 1978 and the unit was removed from service to effect repairs. Thirty tubes in the vicinity of the downcomers were tested by eddy current and 8 tubes showed wall loss of 35% or more at the support plates. These and one tube which could not be tested were plugged. These failures are similar to one which occurred in 1969 and it is believed that downcomer flow gives rise to excessive vibration. Tubes inspected in the riser areas showed no significant wall thinning.

Late in 1978, a primary-to-secondary leak was again observed. Investigation during 1979 showed that some tubes had defects near the tubesheet which could be either cracks or wastage. These are believed to be due to corrosion and represent the first instance of corrosion failures in a CANDU steam generator.

N-REACTOR, USA

As in previous years, leaks occurred only in the two steam generators tubed with 304 stainless steel. Eight tubes were plugged in 1978. Eddy-current inspection of 406 tubes in both hot and cold legs of 2 Alloy-600 tubed steam generators showed apparent denting in 2 tubes. Further examination of the 1 accessible tube did not confirm the eddy-current indication. The 10 steam generators tubed with Alloy-600 have been very reliable at N-Reactor, having operated longer without failures than any other Alloy-600 tubed steam generators Condensate demineralization is used during startup and periods of condenser leakage.

OCONEE-1,2, USA

Unit-1 was taken off-line during 1978 May to investigate a suspected tube leak. Two tubes found to be leaking were plugged. One leak was due to a weld crack at the lower tubesheet. Three other tubes were plugged. One was cut in an attempt to gain visual access to a tube on the open lane, one had a crack near the upper tubesheet and the third was plugged because of any eddy-current signal. During the fall refuelling outage, 13 tubes were plugged.

Oconee-2 developed a primary-to-secondary leak in 1978 April. One tube was found to have a circumferential crack and another had a seal weld crack at the lower tubesheet. Two tubes were plugged as a preventive measure and the fifth was stabilized and plugged after an unsuccessful attempt to extract it (11).

POINT BEACH-1,2, USA

Thirteen tubes had to be plugged at Point Beach-1. Ten tubes, including 4 with leaks, failed by stress corrosion cracking in the tubesheet crevice while 1 would not pass the standard eddy-current probe. Nine of the failed tubes and all leakers were in steam generator A.

At Unit-2, two tubes were plugged, one having a through-wall defect caused by stress corrosion cracking just above the tubesheet. The location and cause of the second defect could not be determined.

RINGHALS-2, SWEDEN

One tube was plugged because of denting at Ringhals-2. Thorough in-service inspection showed that denting in steam generators 1 and 2 was minor with no evidence of support plate distortion. Moderate denting and support plate distortion up to 12 mm at the flow slots was observed on the hot leg of steam generator 3.

ROBINSON-2, USA

Based on eddy-current results during the refuelling and maintenance shutdown 19 tubes were plugged, 10 in steam generator A, 3 in steam generator B and 6 in steam generator C. Later in 1978, during full-power operation, 2 tubes leaked in steam generator A. The cause of all failures was attributed to phosphate wastage. Robinson-2 uses phosphate-controlled secondary water chemistry at 10-80 $mg\cdot kg^{-1}$ and a 2.3-2.4 Na/PO_4 ratio.

SAN ONOFRE-1, USA

The leading cause of steam generator tube failure at San Onofre-1 during 1976 and 1977 was fretting at the anti-vibration bars. During 1977, anti-vibration bars of different design were installed and these have had a marked effect on this type of failure as evidenced by 1978 experience. Only 2 tubes were plugged because of fretting at the U-bend. Eleven tubes were plugged because of constriction at tube support plates but, in general, inspection revealed that tube denting and support plate distortion is not progressing at a significant rate. Five tubes were plugged because of defects above the tubesheet, at least 2 of which were caused by phosphate wastage.

SENA (CHOOZ), FRANCE

Three tubes were plugged at the Chooz reactor because of fretting wear at the U-bend. This is a recurring problem shared with San Onofre (and Jose Cabrera and Haddam Neck where only isolated tubes have been affected). Chooz began commercial operation in 1967 with steam generators tubed with type 316 stainless steel. Tubes have been plugged because of wear at the anti-vibration bar in 1974, 1975, 1976 and 1978. However, the problem cannot be considered serious (23 tubes have been plugged) and there have been no failures due to other causes. This illustrates that excellent performance can be obtained from stainless steel tubes in the steam generator environment. Secondary water chemistry has been controlled by the all-volatile method.

SURRY-1,2, USA

In Unit-1, 165 steam generator tubes were plugged because of denting at support plates and 2 tubes were plugged because of phosphate wastage near the tubesheet. At Unit-2, 200 tubes were plugged because of denting at tube support plates.

An extensive program of in-service inspection was adopted for the Surry steam generators. During 1978 each tube was inspected, on average, 1.7 times in Unit-1 and all tubes were inspected in Unit-2. Full-flow condensate demineralization has been installed.

On 1979 February 4, Unit-2 was shutdown for replacement of all three steam generators. There were two options available for steam generator replacement. Westinghouse, the steam generator manufacturer, developed a tube replacement technique (12). The second option involved replacement of the lower half of the steam generator as a unit and this was chosen for Surry-2. Primary system piping and the transition cone were cut and the steam generator lower section removed through the equipment hatch. The new steam generators have broached holes in the tube support plates to preclude denting and the plates are made of type 405 stainless steel. The Alloy-600 tubes are rolled the full depth of the tubesheet to minimize crevices (13).

TAKAHAMA-1, JAPAN

Of 35 steam generator tubes plugged at Takahama-1, 25 had failed by stress corrosion cracking within the tubesheet crevice and 10 by undetermined causes at the U-bend.

THREE MILE ISLAND-1, USA

Two tubes were explosively plugged during the fuelling and maintenance shutdown. One had a manufacturing defect located about 76 mm above the third support plate. The other was plugged as a precautionary measure because of inconclusive eddy-current results.

TROJAN, USA

One leaker, believed to be within the tubesheet was plugged in steam generator B.

TURKEY POINT-3,4, USA

No in-service inspection was performed on unit-3 during 1978. Two tubes were plugged during the year. The defects were ascribed to denting and wastage, respectively.

Turkey Point-4 steam generators had 532 tubes plugged. Of these, 510 tubes, plugged as part of the preventive maintenance program, showed constriction at the tube support plates. Another 21 tubes were plugged because of continuing phosphate wastage above the tubesheet.

HISTORY OF TUBE FAILURES

The history of tube failures is summarized in Table 2. Over the period 1971-78 an average of 38% of the reactors experienced steam generator tube defects. In 1978, 36% of the reactors surveyed had tube defects, a figure close to the historical average. However, the number of tubes with defects declined sharply in 1978 as did the percentage of tubes with defects. In fact the tube defect rate in 1978 was lower than in any previous year. It appears that the experience gained from recent steam generator problems (e.g. denting) is being applied to minimize further defects. There are in excess of 1.1 million tubes in service in 86 reactors. So far 1.6% of these tubes have been plugged because of tube defects.

Table 3 shows that the number and percent of reactors and tubes with defects (cumulative) increases with increasing effective full-power days in service. A comparison with similar data from the 1977 survey shows that there has been a significant decrease in the number of defective tubes in reactors with less than 1000 EFPD. Table 4 shows 1978 tube defects as a function of effective full-power days. Again the tube defect rate increases significantly with increasing EFPD. Tables 3 and 4 may be used to compare the steam generator tube performance in a given reactor with that in other water-cooled reactors.

There were 12 reactors which exceeded 1000 EFPD without experiencing tube defects. These were Atucha, Biblis A, Borssele, Calvert Cliffs-1, Kewaunee, Stade, MZFR, Pickering-1, 3 and 4, Prairie Island-1 and Zion-1. In 1977 and 1976 there were 7 and 5 reactors respectively, in this category.

LOCATION OF 1978 TUBE DEFECTS

Table 5 shows the location of 1978 tube defects. A majority (78%) of the defects occurred at the tube support plates by denting. The next most common location in recirculating steam generators was just above the tubesheet under the accumulated sludge. Both phosphate wastage and stress corrosion cracking (SCC) were observed in this region. In once-through steam generators some defects occurred near the upper tubesheet.

TABLE 2

TUBE DEFECTS vs YEAR

YEAR	Reactors in Survey	Reactors with Defects	Reactors % with Defects	Tubes in Survey	Tubes with Defects	Tubes % with Defects
1971[a]	34	19	56	337,808	1305[b]	0.39
1972	36	13	36	364,691	1066	0.29
1973	48	11	23	553,883	3942	0.71
1974	59	25	42	742,623	1990	0.27
1975	62	22	35	783,433	1671	0.21
1976	68	25	37	879,333	3763	0.43
1977	79	34	43	1,008,893	4355	0.43
1978	86	31	36	1,119,728	1242	0.11

a Cumulative to the end of 1971
b Excludes pre-operational tube failures in N-Reactor and Tarapur-1 and -2

TABLE 3

CUMULATIVE TUBE DEFECTS vs EFPD* TO 1978 DECEMBER 31

EFPD	Reactors in Survey	Reactors with Defects	Reactors % with Defects	Tubes in Survey	Tubes with Defects	Tubes % with Defects
< 500	16	4	25	243,084	19	0.008
500-1000	21	11	52	317,941	3511	1.1
> 1000	49	36	73	558,703	14489	2.6

* Effective Full-Power Days

TABLE 4

1978 TUBE DEFECTS vs EFPD

EFPD	Reactors in Survey	Reactors with Defects	Reactors % with Defects	Tubes in Survey	Tubes with Defects	Tubes % with Defects
< 500	16	3	18	243,084	10	0.004
500-1000	21	7	33	317,941	107	0.03
> 1000	49	21	43	558,703	1125	0.2

TABLE 5

LOCATION OF 1978 TUBE DEFECTS

Location	Number of Reactors Affected	Number of Tube Defects	% of Tube Defects
Within Tubesheet	4	63	5.1
Near Tubesheet	13	131	10.5
Tube Support Plate	13	964	77.6
U-bend	5	20	1.6
Undetermined	5	64	5.2

TABLE 6

CAUSES OF 1978 TUBE DEFECTS

Cause	Number of Reactors Affected	Number of Tube Defects	% of Tube Defects
Denting	9	926	74.6
Wastage	8	90	7.2
SCC	8	81	6.5
Fretting	3	14	1.1
Fatigue	1	5	0.4
Other	5	39	3.1
Unknown	11	87	7.0

Defects within the tubesheet were observed in steam generators with a long tube-to-tubesheet crevice. SCC occurred within this crevice. In the case of Doel-2 cracking occurred in the roll-expanded zone within the tubesheet. It is not clear whether the cracks initiated from the primary or the secondary side. In most of the steam generators in operation the long tube-to-tubesheet crevice is eliminated by expansion of the tube within the tubesheet. Doel-1 and -2 and Tihange have reported minor denting at the tubesheet but no tubes were plugged because of this in 1978. A few tubes failed at the U-bends due to fretting. The causes of U-bend failures in KWO Obrigheim and Takahama-1 were not known. Previous failures at U-bends in Obrigheim were attributed to SCC.

CAUSES OF 1978 TUBE DEFECTS

DENTING

Denting continued to be the leading cause of tube defects accounting for 75% of the tubes plugged in 1978 (Table 6). However the number of dented tubes decreased from 3780 in 1977 to 926 in 1978. This was probably a consequence of the remedial actions taken by operators to minimize denting through improved control of secondary water chemistry.

PHOSPHATE WASTAGE

Phosphate wastage of tubes caused 7% of the defects and was observed in 8 reactors. It is interesting to note that only 2 (Robinson-2 and San Onofre) of these 8 reactors were using phosphate treatment. The other 6 were on all-volatile treatment and the wastage was caused by residual phosphate (from previous phosphate treatment) in the sludge above the tubesheet. However the number of tubes showing phosphate wastage is quite small in relation to the number of tubes in service and therefore the wastage problem appears to be under control.

STRESS CORROSION CRACKING

Stress corrosion cracking occurred in 8 reactors and accounted for approximately 7% of the tubes plugged in 1978. It is suspected that SCC from secondary side was caused by concentrated alkaline solutions produced from residual sodium phosphate within the sludge or through in-leakage of alkali-forming condenser cooling waters. SCC from secondary side was observed in reactors using all-volatile treatment but not in those using phosphate treatment.

In some cases SCC occurred in the tube-to-tubesheet crevice because of the stagnant conditions in this region which resulted in concentration of aggressive chemical species. Fewer tubes failed by SCC in 1978 than in 1977.

FRETTING

Fourteen tubes failed by fretting in 1978 compared with 93 in 1977. Fretting occurred at anti-vibration bars in U-bends at San Onofre and SENA. Design modifications have significantly decreased the number of fretting-induced defects in these reactors. Thus fretting was not a serious problem in 1978.

FATIGUE

Failures in Oconee-2 were ascribed to vibration-induced fatigue possibly assisted by corrosion (11). The corrosion was probably caused by concentration of impurities in water droplets carried by steam up the tube-free lane. The nature of these impurities has not been identified.

OTHER CAUSES

At Doel-1 tubes were damaged by material left in the secondary side of the steam generator during manufacture. At Crystal River-3, debris from the failure of a burnable poison rod assembly blocked some of the tubes which had to be plugged. Tubes in Farley-1 and Three Mile Island-1 were plugged because of manufacturing defects. The causes of failure of 87 tubes in 11 reactors were not known. It is often difficult and time-consuming to remove tubes for examination and this may be the reason why the cause of failure of 7% of the tubes plugged in 1978 was unknown.

In summary, 90% of the tube defects were caused by corrosion in the form of phosphate wastage, denting and SCC.

SECONDARY WATER CHEMISTRY CONTROL

Table 7 shows the secondary water chemistry control in various reactors and Table 8 shows the relationship between secondary water chemistry control and corrosion-induced defects from the secondary side. Ten out of 86 reactors used phosphate treatment and the rest were on all-volatile treatment.

TABLE 7

SECONDARY WATER CHEMISTRY CONTROL IN 1978

Reactor	Treatment	mg PO_4/kg Water	Na:PO_4 molar ratio
Atucha	Phosphate	2-6	2.2 - 2.6
Biblis A,B		2-6	2.2 - 2.6
Borssele		2-6	2.2 - 2.6
GKN Neckar		2-6	2.2 - 2.6
KKS Stade		2-6	2.2 - 2.6
Jose Cabrera		2	2.0 - 2.2
NPD		2-5	--
Robinson-2		10-80	2.3 - 2.4
San Onofre	Phosphate	5-10	2.5 - 2.8
Arkansas-1	All-volatile	With condensate demineralization	
Calvert Cliffs-1,2			
Crystal River-3			
Davis-Besse-1			
Doel-1,2			
Ginna			
Loviisa-1			
MZFR			
N-Reactor			
Oconee-1,2,3			
Rancho Seco			
Surry-1,2			
Three Mile Island-1			
Trojan			
Other reactors	All-volatile	No condensate demineralization	

Eight reactors used a low concentration of sodium phosphate (2-6 mg PO_4/kg water) with a sodium to phosphate ratio of less than 2.6. So far no corrosion-induced failures (e.g. phosphate wastage, denting, SCC) have been reported from reactors which use a low concentration of phosphate. There is strong evidence that a low concentration phosphate treatment can be used without significant phosphate wastage and is therefore an attractive alternative to all-volatile treatment particularly for reactors on brackish or sea water locations.

Some failures due to phosphate wastage were observed in Robinson-2 and San Onofre which use a higher concentration of phosphate. In San Onofre 5 tubes were plugged because of denting.

Of the 76 reactors on all-volatile treatment, 17 used full-flow condensate demineralization. Eight of the reactors in this category were equipped with once-through steam generators and the rest with recirculating steam generators. The incidence of corrosion-induced defects from the secondary side in reactors with condensate demineralizers was quite low with only Arkansas-1, Doel-2 and Ginna exhibiting such defects. Thus all-volatile treatment with condensate demineralization seems to be an effective way to minimize corrosion of steam generator tubes.

Defects caused by corrosion from the secondary side (wastage, denting and SCC) were observed in 14 out of 59 reactors on all-volatile treatment with no condensate demineralization. Phosphate wastage was attributed to corrosion from the residual sludge left over from the phosphate treatment. Denting was usually associated with leakage of brackish or sea water leading to formation of acidic chloride solution in tube-to-support-plate annuli. This solution would not be neutralized with all-volatile treatment.

Table 8 also shows the percentage of tubes with corrosion defects in 1978. It confirms the observation made in the 1977 survey that a phosphate treatment at low concentration or an all-volatile treatment with condensate demineralization seems to offer the best insurance against corrosion from the secondary side. The tube defect rate was considerably higher in reactors on phosphate

TABLE 8

SECONDARY WATER CHEMISTRY vs CORROSION DEFECTS* IN 1978

Water Chemistry	Reactors			Tubes		
	Number	With Defects	%	Number	With Defects	%
Phosphate (2-6 mg PO_4/kg Water)	8	0	0	77,390	0	0
Phosphate (5-80 mg PO_4/kg Water)	2	2	100	21,162	37	1.7×10^{-1}
AVT + CD**	17	3	18	345,458	34	9.8×10^{-3}
AVT	59	14	24	665,718	1008	1.5×10^{-1}

* Includes phosphate wastage, denting and SCC from secondary side.
** All-volatile treatment with condensate demineralization.

treatment at high concentration or those on all-volatile treatment without condensate demineralization. However there were notable exceptions to this rule. For example reactors such as Pickering-1, -3 and -4, Kewaunee, Priairie, Island-1 and Zion-1 have exceeded 1000 EFPD without tube defects although they use all-volatile treatment without condensate demineralization. It is interesting to note that they are all located on fresh water sites.

STEAM GENERATOR TUBE MATERIALS

Table 9 summarizes the experience with steam generator tube materials to the end of 1978. The number of stainless steel and Alloy-800 tubes in use has been amended in light of more recent data. The number of Alloy-600 tubes in service increased over last year. There was an increase in the number and percentage of Alloy-600 tubes with defects in 1978. However many of these defects (e.g. denting) were caused by factors other than tube material. The defect rate of Monel-400 was quite low. So far no defects have been observed in steam generators tubed with Alloy-800, one of which, KKS Stade, has exceeded 2100 EFPD.

The replacement steam generators in Surry-1 and -2 are equipped with type 405 stainless steel tube support plates with broached holes. Laboratory studies have shown that this material is more corrosion resistant than carbon steel and is unlikely to cause denting even in the presence of sea water leaks.

CONDENSER TUBE MATERIALS

The integrity of condensers against ingress of cooling water to the secondary circuit is of prime importance in prevention of corrosion failures in steam generator tubes. Condenser leakage can be due to a variety of causes such as tube impingement by debris, erosion, vibration and corrosion. Table 10 lists condenser tube materials in use in operating water-cooled reactors with steam generators. It can be seen from the table that admiralty brass is the most popular alloy at fresh water sites though the tendency in newer plants is to use type 304 stainless steel.

TABLE 9

EXPERIENCE WITH STEAM GENERATOR TUBE MATERIALS TO 1978 DECEMBER 31

Tube Material	Number of Reactors	Number of Tubes	Number of Tube Defects	% with Defects	Failure Mechanism
SS	11	69,853**	1,019	1.5	SCC,W
Alloy-600	63	809,458	16,793	2.1	SCC,W,D Fr,F
Monel-400	8	167,700	335	0.2	SCC,Fr
Alloy-800	6	72,717	0	0	

* N-Reactor has steam generators with both Alloy-600 and type 304 SS tubes
** Amended total

TABLE 10

CONDENSER TUBE MATERIAL

Tube Material	Cooling Water		
	Fresh	Brackish	Salt
Admiralty	27	3	--
90-10 Cu-Ni	1	3	1
70-30 Cu-Ni	--	3	3
Aluminum Brass	1	3	9
Titanium	1	3	3
Stainless Steel	17	--	--
Arsenical Copper	2	--	--
CuZn 28 Sn	1	--	--

On sea water sites, aluminum brass is the preferred alloy. Many condensers have tubes of a different material in impingement-air ejector sections. Plants with admiralty brass as the main condenser tubing can have 90-10 or 70-30 cupro-nickel or stainless steel in the outer tubes ($\sim$10%). The latter material is more commonly used. Plants with aluminum brass main tubing often have 70-30 cupro-nickel in peripheral zones. Titanium is being used as replacement tube material in plants on brackish or sea water sites where experience with copper alloys has been unsatisfactory and in new plants at such sites. Utilities in Japan have used aluminum brass almost exclusively and, by careful attention to quality control and in-service inspection, have had reliable service from their condensers.

INSPECTION AND REPAIR PROCEDURES

Procedures for inspecting steam generator tubes vary widely throughout the industry. In general, the amount of inspection performed is related to tube failure history. In plants where few problems have been observed, inspection is performed less frequently and on a smaller tube sample. Eddy-current has become the most widely used test method with an increase in the use of multi-frequency techniques and automated probe manipulation. Visual, photographic, fibre-optic and hydrostatic methods are widely used to complement eddy-current inspection.

Explosive plugging has become the most popular method for removing problem tubes from service, although welded plugs are still commonly used. Tube sleeving has enabled the return to service of previously plugged tubes at Palisades. This technique may be useful where the tube reserve is nearly depleted but the deterioration has been arrested. The most difficult repair operation involves replacement of the steam generators or in-situ retubing. Retubing was performed for the first time in a large commercial reactor (Shippingport steam generators have been replaced) after several years of operation, in 1979. It was found that such an operation is feasible and can be performed in reasonable time with radiation exposure, cost and waste kept to an acceptable level.

SUMMARY

Denting continued to be the major failure mechanism in 1978. However there was a significant decrease in the number of steam generator tube defects in 1978 compared to 1977. This was probably a result of more stringent control of secondary side chemistry by the operators. Reactors on phosphate treatment at low concentration or on all-volatile treatment with condensate demineralization showed very few corrosion-related defects during the year. Utilities are turning to corrosion resistant condenser tube materials to minimize the leakage of cooling-water into the steam generators.

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APPENDIX 1

STEAM GENERATOR EXPERIENCE TO
1978 DECEMBER 31

STEAM GENERATOR EXPERIENCE TO 1978 DECEMBER 31

Reactor	MW(e)	# of SG	Tubes per SG	Material	Builder	Area per SG (m^2)	Condenser Cooling	Failures per Tube Year (X10^4)	EFPD	Cumulated Defects	Comments
Arkansas-1	820	2	15531	6	BW	12304	F	0.6	978	5	OTSG
Atucha	320	2	3945	8	GHH	3454	F	0	1209	0	PHWR
Beaver Valley-1	852	3	3388	6	W	4785	F	0	287	0	
Beznau-1	350	2	2604	6	W	3097	F	263.7	2371	892	
Beznau-2	350	2	2604	6	W	3097	F	88.5	2163	273	
Biblis A	1146	4	4060	8	KWU/DBW	4510	F	0	1001	0	4 removed for examination
Biblis B	1240	4	4021	8	KWU	4335	F	0	496	0	3 removed for examination
Borssele	447	2	4234	8	Balcke	3600	S	0	1515	0	2 removed for examination
Bruce-1	750	8	4200	6	BW(Can)	2368	F	0	466	0	CANDU
Bruce-2	750	8	4200	6	BW(Can)	2368	F	2.0	482	9	CANDU
Bruce-3	750	8	4200	6	BW(Can)	2368	F	0	309	0	CANDU
Calvert Cliffs-1	850	2	8519	6	CE		B	0	1006	0	
Calvert Cliffs-2	850	2	8519	6	CE		B	0	565	0	
Cook-1	1054	4	3388	6	W		F	0	891	0	
Cook-2	1065	4	3388	6	W	4784	F	0	152	0	
Crystal River-3	825	2	15457	6	BW	12245	S	2.7	344	8	OTSG
Davis-Besse-1	906	2	15457	6	BW	12245	F	0	152	0	OTSG
Doel-1	392	2	3260	6	CKL	4130	B	11.6	1153	24	
Doel-2	392	2	3260	6	CKL	4130	B	17.9	907	29	
Douglas Point	208	8	1950	4	MLW	970	F	0.2	1962	2	CANDU
Dresden-1	200	4	1801	SS	FW	605	F	28.0	3256	180	BWR
Farley-1	829	3	3388	6	W	4784	F	1.0	359	1	
Fessenheim-1	890	3	3388	6	Fram	4780	F	0	324	0	
Fessenheim-2	890	3	3388	6	Fram	4780	F	0	278	0	
Fort Calhoun-1	457	2	5005	6	CE	4428	F	0.9	1196	3	

STEAM GENERATOR EXPERIENCE TO 1978 DECEMBER 31 - cont'd

Reactor	MW(e)	# of SG	Tubes per SG	Material	Builder	Area per SG (m^2)	Condenser Cooling	Failures per Tube Year ($X10^4$)	EFPD	Cumulated Defects	Comments
Garigliano	150	2	1785	4	KM	560	F	104.5	3249	332+	BWR
Genkai-1	529	2	3388	6	MHI	4784	S	0.5	1009	1	
Ginna	490	2	3260	6	W	4129	F	47.9	2174	186	
GKN Neckar	750	3	4021	8	GHH/Balcke	4270	F	0	616	0	
Haddam Neck (Conn. Yankee)	575	4	3794	6	W	2573	F	2.5	3117	32	
Ikata-1	538	2	3388	6	MHI	4785	S	0	428	0	
Indian Point-2	864	4	3260	6	W	4129	B	12.6	995	45	
Indian Point-3	965	4	3260	6	W	4129	B	1.8	608	4	
Jose Cabrera (Zorita)	153	1	2604	6	W	2308	F	1.6	2655	3	
KANUPP	126	6	1355	4	BW(Can)	705	S	0	875	0	CANDU
Kewaunee	540	2	3388	6	W	4785	F	0	1261	0	
KKS Stade	630	4	2993	8	DBW	2930	F	0	2135	0	2 removed for examination
Ko-Ri-1	597	2	3388	6	W	4785	S	0	170	0	
KRB Gundremmingen	237	3	1929	SS	VKW	870	F	86.2	2664	364	BWR
KWL Lingen	256	2	5000	SS	GHH	2360	F	23.4	1743	112	BWR, shutdown
KWO Obrigheim	328	2	2605	6	GHH/Balcke	2750	F	62.0	2913	258	
Loviisa-1	420	6	3200	SS	AEE	2000	B	0	525	0	horizontal SG
Maine Yankee	790	3	5703	6	CE	5405	B	2.2	1440	15	
Mihama-1	320	2	4426	6	CE	3381	S	1372.0	663	2206	
Mihama-2	470	2	3260	6	MHI	4130	S	118.7	1283	272	6 removed for examination
Mihama-3	780	3	3388	6	MHI	4785	S	0	597	0	
Millstone-2	796	2	8519	6	CE		S	238.0	686	762	
MZFR	52	2	765	SS	GHH/Balcke	920	F	0	2697	0	PHWR
North Anna-1	943	3	3388	6	W	4785	F	0	172	0	
NPD	22	1	2069	6	BW(Can)	577	F	4.6	3820	10	CANDU, horizontal SG

STEAM GENERATOR EXPERIENCE TO 1978 DECEMBER 31 - cont'd

Reactor	MW(e)	# of SG	Tubes per SG	Material	Builder	Area per SG (m^2)	Condenser Cooling	Failures per Tube Year (X10^4)	EFPD	Cumulated Defects	Comments
N-Reactor	860	10 2	1920	6 SS	CE	1486	F	0	2056	0 42	LGR
Oconee-1	871	2	15531	6	BW	12304	F	7.4	1185	75	OTSG
Oconee-2	871	2	15531	6	BW	12304	F	1.5	963	12	OTSG
Oconee-3	871	2	15531	6	BW	12304	F	1.9	1048	17	OTSG
Palisades	700	2	8519	6	CE	7368	F	741.6	1061	3673	
Pickering-1	514	12	2600	4	BW(Can)	1858	F	0	2246	0	CANDU
Pickering-2	514	12	2600	4	BW(Can)	1858	F	0.1	2127	1	CANDU
Pickering-3	514	12	2600	4	BW(Can)	1858	F	0	1773	0	CANDU
Pickering-4	514	12	2600	4	BW(Can)	1858	F	0	1540	0	CANDU
Point Beach-1	497	2	3260	6	W	4129	F	100.1	2181	390	
Point Beach-2	497	2	3260	6	W	4129	F	7.5	1787	24	
Prairie Island-1	520	2	3388	6	W	4786	F	0	1299	0	
Prairie Island-2	520	2	3388	6	W	4786	F	0.5	1111	1	
Rancho Seco	913	2	15457	6	BW	12245	F	1.3	724	8	OTSG
RAPP-1	207	8	1950	4	MLW	970	F	0	614	0	CANDU
Ringhals-2	822	3	3388	6	W	4784	S	100.3	827	231	
Robinson-2	700	3	3260	6	W	4128	F	22.4	1969	118	
Salem-1	1090	4	3388	6	W	4784	B	0	321	0	
San Onofre-1	430	3	3794	6	W	2573	S	27.5	2846	244	
SENA (Chooz)	280	4	1662	SS	CKL	1385	F	5.5	2304	23	
Shippingport	72	2 2	1692 3050	6 6	BW FW	1244 1084	F	133.8 0	3435	426 0	horizontal SG - 2 recirculating, 2 OTSG
St. Lucie-1	802	2	8485	6	CE		S	0	575	0	
Surry-1	788	3	3388	6	W	4784	B	473.3	1343	1770	
Surry-2	788	3	3388	6	W	4784	B	534.4	1293	1924	

STEAM GENERATOR EXPERIENCE TO 1978 DECEMBER 31 - cont'd

Reactor	MW(e)	# of SG	Tubes per SG	Material	Builder	Area per SG (m^2)	Condenser Cooling	Failures per Tube Year ($X10^4$)	EFPD	Cumulated Defects	Comments
Takahama-1	780	3	3388	6	W	4785	S	94.3	746	196	
Takahama-2	780	3	3388	6	MHI	4785	S	0	844	0	1 removed for examination
Tarapur-1	198	2	1589	SS	FW		S	2.6	1786	4+	BWR
Tarapur-2	198	2	1589	SS	FW		S	128.4	1869	209	BWR
Three Mile Island-1	792	2	15531	6	BW	12034	F	1.0	1200	10	OTSG
Tihange-1	880	3	3388	6	CKL	4788	F	4.9	945	13	
Trino Vercellese	242	4	1662	SS	W	1384	F	0.9	2922	5	
Trojan	1130	4		6	W		B		380	1	
Turkey Point-3	693	3	3260	6	W	4128	S	228.9	1492	915	
Turkey Point-4	693	3	3260	6	W	4128	S	494.5	1265	1676	
Yankee Rowe	175	4	1620	SS	W	1248	F	10.7	4976	95	
Zion-1	1050	4	3260	6	W	4128	F	0	1047	0	
Zion-2	1050	4	3260	6	W	4128	F	0	971	0	

ABBREVIATIONS USED IN APPENDIX I

BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium
EFPD	Effective Full-Power Days
LGR	Light Water-Graphite Reactor
OTSG	Once-Through Steam Generator
PHWR	Pressurized Heavy Water Reactor
SG	Steam Generator

CONDENSER COOLING WATER

B	Brackish
F	Fresh
S	Sea

STEAM GENERATOR TUBE MATERIALS

4	Monel-400
6	Alloy-600
8	Alloy-800
SS	Stainless Steel

STEAM GENERATOR MANUFACTURERS

AEE	Atomenergoexport
Balcke	Balcke
BW	Babcock & Wilcox
BW(Can)	Babcock & Wilcox Canada
CE	Combustion Engineering
CKL	Cockerill
DBW	Deutsche Babcock & Wilcox
Fram	Framatome
FW	Foster Wheeler
GHH	Gutenhoffnungshutte
KM	Koninklizke Machinefabrik Stork
KWU	Kraftwerk Union
MHI	Mitsubishi Heavy Industries
LMW	Montreal Locomotive Works
VKW	Vereingte Kesselwerke
W	Westinghouse

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