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Worldwide Assessment of Steam-Generator Problems in Pressurized-Water-Reactor Nuclear Power Plants

H. H. Woo

S. C. Lu

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September 15, 1981

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CONTENTS

Abstract	v
Acknowledgments	vii
1. Introduction	1
1.1 History and Background	1
1.2 Objective and Contents of this Report	2
1.3 Approach	2
2. PWR Steam Generator Designs and Materials	3
2.1 Operation of a Typical PWR Plant	3
2.2 Role of the Steam Generator in a PWR	4
2.3 Types of PWR Steam Generators	4
2.4 Materials	8
2.5 Design of Tube to Tube-Sheet Joints	8
2.6 Design of Support Plates	10
2.7 World Distribution of PWR Plants	10
3. PWR Steam Generator Problems	13
3.1 Overview of Experience in Steam Generator Performance	13
3.2 Corrosion Damage Problems	17
3.3 Mechanical Damage Problems	23
4. Avoidance or Mitigation of Steam Generator Damages	27
4.1 Inspection and Detection of Damages	27
4.2 Cost and Benefit for Nondestructive Examination	29
4.3 Chemistry Control	30
4.4 Mechanical Modification	31
4.5 Operational Procedure	31
4.6 Repair and Replacement Options	32
5. Foreign Experience	33
5.1 German PWR Steam Generator Experience	33
5.2 French PWR Steam Generator Experience	38
5.3 Japanese PWR Steam Generator Experience	43
6. Improved Steam Generator Designs	44
6.1 Material Alternatives	44
6.2 Mechanical Design Alternatives	47

7. Summary and Conclusions	49
8. References	51
Appendix A	55
Appendix B	71
Glossary	73

ABSTRACT

The objective of this investigation is to assess the reliability of steam generators of pressurized water reactor (PWR) power plants in the United States and abroad. The assessment is based on operation experience of both domestic and foreign PWR plants. Our approach is to collect and review papers and reports available from the literature as well as information obtained by contacting research institutes both here and abroad.

This report presents the results of our assessment. It contains a general background of PWR plant operations, plant types, and materials used in PWR plants. A review of the worldwide distribution of PWR plants is also given. The report describes in detail the degradation problems discovered in PWR steam generators: their causes, their impacts on the performance of steam generators, and the actions to mitigate and avoid them. One chapter is devoted to operating experience of PWR steam generators in foreign countries. Another discusses the improvements in future steam generator design. Finally the summary and conclusion of the study is presented.

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1. INTRODUCTION

1.1 HISTORY AND BACKGROUND

Steam generators are among the largest and most critical components in pressurized water reactor (PWR) nuclear power plants. Since the first commercial PWR plant manufactured by Westinghouse (Yankee, Rowe, Massachusetts) began operations in 1961, the nuclear industry in the United States and abroad has had a series of problems with degradation of the steam generators from such mechanisms as vibration, "fretting" (rubbing of adjacent surfaces), high-cycle fatigue, water hammer, cracking, "wastage" (wall thinning), "denting" (distortion that occurs when a tube support plate made of carbon steel corrodes to the point where it squeezes down the steam generator tubing), and erosion corrosion. Of 78 PWRs currently in operation, 24 units have experienced tube denting, 22 units have had wastage of tubes, at least 5 have had tube cracking in the tube sheet crevice, 11 have had tube cracking in the U-bend, and at least 9 have experienced water hammer (Mundis, 1981).

Failure of a single steam generator tube often requires a lengthy shutdown while the tube is located and plugged. This maintenance work must be performed where the radiation level is high and quarters are cramped. In addition, the number of tubes removed from service in some plants is approaching the 20% to 25% reserve margin after which either progressive derating or replacement of tube bundles becomes necessary. It has been estimated that the steam generators at 10 plants have reached the point where it is questionable that they will survive to the end of their design life, which is 40 years (Tatone, 1980). At least 2 utilities have already decided to replace steam generators in 4 units before these units have operated for 10 years. These replacements are expensive (\$200-300 million/unit, including replacement power costs), involve significant radiation exposure, and, more importantly, result in significant reduction of power generation because of extensive outages (Green, 1980). Therefore, understanding the mechanisms of tube failure and controlling the rate of degradation have potentially high payoff in terms of worker safety, power-generation economics, and fossil-fuel conservation. Although a concerted industry effort by steam-generator owners and their contractors has progressed significantly toward solving these problems, this effort is far from being completed. Sound technical bases for satisfactorily improved steam-generator performance and reliability can still be established. In view of the current problems associated with PWR steam generators, whether or not continued steam-generator degradation may eventually lead to an unrecoverable reduction of worldwide power generation is a legitimate concern.

1.2 OBJECTIVE AND CONTENTS OF THIS REPORT

The objective of this report is to assess the reliability of steam generators in PWR plants worldwide. Chapter 2 contains a general background for PWR plant operations, steam generator types, and materials; a review of the worldwide distribution of PWR plants is also given. Chapter 3 is concerned with details of degradation problems in PWRs, their causes, and their respective impacts on the performance of steam generators. Chapter 4 deals with how to mitigate and avoid steam generator problems. Chapter 5 describes operating experience for PWR steam generators in foreign countries. Chapter 6 discusses the improved PWR steam-generation designs for the future. Chapter 7 contains a summary and the conclusions of our study; an outlook on current PWR steam-generator reliability is included.

1.3 APPROACH

Our assessment of PWR steam-generator degradation problems is based on operating experiences of both domestic and foreign PWR plants. We have collected and reviewed papers and reports from the open literature. We have also contacted personnel at the Electric Power Research Institute (EPRI) (Green, 1981) and the Nuclear Regulatory Commission (NRC) (Strosnider, 1981). Foreign data from sources other than the open literature were augmented in the course of a foreign trip. It should be understood that our time and budget for this investigation were limited. Therefore, the results reported herein should be considered the authors' assessment made on the basis of the best available information.

1. INTRODUCTION

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2. PWR STEAM GENERATOR DESIGNS AND MATERIALS

2.1 OPERATION OF A TYPICAL PWR PLANT

The basic constituents of a PWR are illustrated in Fig. 1. Uranium dioxide pellets are inserted into Zircaloy tubes to form a fuel rod, which is about 1 cm in diameter and 4 m long. The core of the reactor is an assembly each with approximately 200 fuel rods interspersed with control channels in which the control rods move. The core is contained in a stainless steel pressure vessel whose walls are more than 20 cm thick. Water under high pressure is forced through the spaces between fuel rods to cool them and to slow the fission neutrons. This primary water picks up heat from the reactor core and transfers heat to secondary coolant water through a steam generator. Steam from the same generator passes through a steam turbine to generate electricity. The low-pressure steam leaving the turbine enters a steam condenser where it is recondensed to water so that it

FIG. 1. Schematic diagram of the PWR reactor and electricity generating system.

can be pumped back to the feedwater line to complete the cycle. The condenser requires a large volume of ambient-temperature cooling water, which is usually obtained from artificial cooling panels or cooling towers. A containment shell, which houses both the pressure vessel and the steam generator, acts simultaneously as a biological shield against radiation and as a containment vessel in the event of leakage from the radioactive part of the system. The water in the secondary circuit is essentially free of radioactivity.

2.2 ROLE OF THE STEAM GENERATOR IN A PWR

All large water-moderated nuclear power plants now operate on a steam cycle: the heat generated by nuclear fission is used to convert water into high-pressure, high-temperature steam. PWR plants contain three separate cooling cycles, as shown in Fig. 1. The first cooling cycle consists of the primary coolant system that pumps pressurized coolant water through the heat-generating core of the reactor. The second cooling cycle consists of large heat exchangers (steam generators), a steam-driven turbine generator, a steam condenser, feedwater pumps, and associated piping systems. The water in the secondary coolant system boils in the steam generator and creates the steam that drives the turbine generator. After it passes through the turbine generator, the steam is condensed back into water in the steam condenser. The secondary cooling water is returned to the steam generators by the feed pumps, thereby completing the cycle. The third cooling cycle, the condenser cooling water system, provides cold water to condense the steam back to water in the steam condenser. The cooling system is closed to ensure that the radioactive primary cooling water, the secondary cooling water, and the condenser cooling water are kept separate. The steam generator is the connecting link between the radioactive primary and the nonradioactive secondary coolant system. Therefore, the steam generator in a PWR plant not only exchanges heat but also acts as a principal part of the reactor coolant pressure boundary.

2.3 TYPES OF PWR STEAM GENERATORS

The two basic PWR designs now in commercial operation are a recirculating type and a once-through type.

2.3.1 Recirculating Type

The recirculating-type steam generator produces dry steam at the saturation temperature (D&S). A vertical U-tube D&S unit is shown in Fig. 2. Westinghouse (W),

FIG. 2. Schematic of a recirculating D&S steam generator (Westinghouse Model F is the example).

Combustion Engineering (CE), and Kraftwerk Union (KWU, West Germany) produce variants of the inverted U-tube design; there are important differences in detail between vendors and between older and current designs. In the following discussion, we will illustrate the functions of a recirculating-type steam generator manufactured by Westinghouse (Fig. 2).

The hot coolant water from the reactor enters the steam generator through the primary coolant inlet nozzle and flows into the inlet side of the steam generator lower plenum. The coolant water then flows through a large number of U-tubes to the outlet side of the lower plenum where it exits the steam generator through the primary coolant outlet. A vertical divider plate separates the inlet and outlet plenums. The secondary coolant water enters the steam generator through the feedwater inlet and flows through the feedwater ring into an annulus area between the wrapper and the shell. It then flows to the bottom of the steam generator and up through the tube bundle region.

Several thousand tubes (the number of tubes depends on the design) are welded into and are vertically supported by the tube sheet, which is at least 20 in. thick and is perforated by thousands of holes for the steam generator tubes. At higher elevations, the tubes penetrate through holes in tube support plates that provide lateral support: in the U-bend area, anti-vibration bars are sometimes laced through the tubes to minimize flow-induced vibrations. The steam generator tubes are approximately 0.75 to 0.88 in. in outside diameter (o.d.); the wall is about 0.050 in. thick. The tubes are the surfaces where heat transfers from the primary to the secondary coolant water. The steam generator tubes constitute over 50% of the area of the total primary coolant system pressure-retaining boundary. Above the U-tubes, in the upper portion of the steam generator, there is a moisture-separating system for improving the quality of generated steam that is sent to the turbine generator.

2.3.2 Once-Through Type

Figure 3 shows a typical once-through steam generator (OTSG) manufactured by Babcock and Wilcox (B&W). There are nine operating PWR nuclear units with B&W steam generators in the United States. In this steam generator design, approximately 16,000 straight tubes terminate in a tube sheet at each end of the steam generator. These tubes are approximately 0.625 in. in o.d. and their walls are about 0.035 in. thick. Primary coolant water enters the steam generator inlet plenum, flows from top to bottom through the steam generator tubes, and exits through the steam generator outlet plenum. The secondary coolant water enters through the steam generator feedwater nozzles, flows down through an annulus around the downcomer section, and flows up around the tubes.

FIG. 3. Schematic of a once-through steam generator (Babcock and Wilcox model).

The auxiliary feedwater is introduced through an internal header located in the upper region of the steam generator (Fig. 3). Boiling occurs about two-thirds of the way up the tube bundle. Tubes in the steam region (area above the boiling region) are exposed to steam (in contrast to the U-tubes in the recirculating types of steam generator, which remain covered with water). The steam is superheated in this region. After passing through the superheated region near the upper tube sheet, the steam flow shifts from vertically upward, to radially outward, and then abruptly downward through the annulus between the shell and the baffle. The steam then flows through the steam generator outlet nozzles. The tubes are supported vertically by the tube sheets and are restrained from horizontal movement along their length by tube support plates (the exact number depends on design, but it is generally 15 or 16). The support plate-tube intersection design shown in Fig. 4 is used for all support plates, except that a portion of the 15th support plate has drilled tube holes near the periphery to enhance steam flow through the center of the bundle and into the upper annulus of the steam generator.

2.4 MATERIALS

Table 1 summarizes the material currently used in the manufacture of steam generators and condensers for PWR plants. A variety of tube materials has been used, including 1 Cr-1/2 Mo steel, 300 stainless steel, Alloy 800, and Inconel 600; experience with them has been reviewed by Berge and Donati (Berge, 1978).

The tube-sheet and tube-support plate materials include mild steel (which has been shown to be unsatisfactory and will therefore be replaced by 12 Cr material in all new designs from Westinghouse and CE and by 13 Cr materials in some French units) and 300-series stainless steel in KWU units.

2.5 DESIGN OF TUBE TO TUBE-SHEET JOINTS

In the old designs the method of joining the tube to the tube sheets produces a crevice between tube and plate on the secondary side in which aggressive solutions can be generated. In more recent D&S designs the tubes are expanded into the tube sheet, thereby virtually eliminating the crevice. The method of expanding the tube into the tube sheet can also influence corrosion vulnerability; however, because the extent of residual stress and cold work produced in the tube will affect stress-corrosion cracking (SCC) susceptibility. Mechanical rolling can produce high residual stresses, whereas expansion by explosion or static hydraulic pressure has been shown to be far superior and is now the standard practice for new design (Garnsey, 1979).

FIG. 4. B&W tube support plate design.

Table 1. Materials used in PWR steam generators and condensers.

Reactor component	Material
Steam generator	
Shell	Low-alloy steel (SA533 Gr. B) ^a
Support plates	Carbon steel (SA515 Gr. 60) 300 series SS
Tubes	Inconel 600: Incoloy 800 (SB407.513)
Condenser	
Tubes	Al-Bronze: Al-brass; Cupronickel; titanium; Type 304 SS ^b

^aSA and SB numbers are specifications from American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

^bSS = stainless steel.

2.6 DESIGN OF SUPPORT PLATES

Figure 5 shows various designs for support plates: drill hole, B&W trifoil, KWU egg crate, and Westinghouse quatrafoil support plates. The drill-hole type of support plates has caused problems which will be discussed later. Other designs--either in operation or scheduled--are much more open and flexible than drilled plate.

2.7 WORLD DISTRIBUTION OF PWR PLANTS

A recent survey (Nuclear News, June 1981) of nuclear power plants worldwide shows that 114 PWR plants are in commercial operation or are ready for operation and 131 PWR plants are under construction. Of these, approximately 34% of those in operation and 42% of those under construction are in the United States. France is second in both categories; the Soviet Union and Japan share third place. Table 2 presents the worldwide distribution of PWR plants. Detailed information regarding size, reactor supplier, generator supplier, architect engineer, construction type, and construction stage can be found in Appendix A.

FIG. 5. Tube support plate designs: (a) drilled hole; (b) B&W trifoil plate for OTSG; (c) KWU egg crate; (d) Westinghouse quatrafoil for Model F units (Garnsey, 1979).

TABLE 2. World distribution of FWR plants as of June 30, 1981.
(Operable, under construction, or on order with 30 MWe or over.)

Region and country	Plants in commercial operation or ready	Plants under construction	On order
U.S.	39	58	11
Europe			
Belgium	3	4	0
Bulgaria	3	1	0
Czechoslovakia	2	6	2
Finland	1	1	0
France	22	24	3
Germany (East)	5	2	0
Germany (West)	6	4	6
Hungary	0	4	0
Italy	1	1	1
Luxembourg	0	0	1
Netherlands	1	0	0
Poland	0	0	2
Rumania	0	0	1
Spain	3	6	2
Sweden	3	0	0
Switzerland	3	0	0
Yugoslavia	1	0	0
Subtotal	54	53	18
U.S.S.R.	10	6	0
Asia			
Japan	10	5	0
Korea	1	7	0
Philippines	0	1	0
Taiwan	0	2	0
Subtotal	11	15	0
Others			
Brazil	0	3	0
Egypt	0	0	1
Iraq	0	0	1
Libya	0	0	1
South Africa	0	2	0
Turkey	0	0	1
Subtotal	0	5	4
Non-U.S. total	75	79	22
World total	114	137	33

3. PWR STEAM GENERATOR PROBLEMS

3.1 OVERVIEW OF EXPERIENCE IN STEAM GENERATOR PERFORMANCE

Steam generators have been consistently troublesome for PWR plants, and a large variety of problems have contributed to the plant-capacity losses due to this component (Cottrell, 1981). A series of analyses (Stevens-Grulle, 1975a; Stevens-Grulle, 1975b; Hare, 1976; Hare, 1977; Tatone, 1978; Pathania, 1979; Tatone, 1980) of steam generator problems indicated that the tube-related problems are the principal cause of component downtime. Recently, Mundis and Noble (Mundis, 1981) surveyed 78 PWR steam generators worldwide and found that 24 plants have experienced tube derating, 22 plants have had wastage (or thinning) of tubes, at least 5 have had tube cracking in the tube crevice, 11 have had pieces of tube in the U-bend, and at least 9 have experienced water hammer. The results reported by Mundis are summarized in Table 3.

Most steam generator problems result in plugging of the affected tubes. Figure 6 is a histogram showing industry's experience in tube plugging from all causes. The tube-plugging rate has varied from 0.16 to 1.6% of tubes in service each year. Since a steam generator can operate at full power with no more than 20% (0.5% per year for 40 years) of the tubes plugged, the overall plugging rate, particularly from 1972 to 1977, was an industry-wide concern. In 1978 and 1979 tube-plugging rates were reduced significantly, however. This was because wastage was virtually eliminated as a cause for tube plugging and denting-related plugging was reduced, as also shown in Figure 6. (Denting is the distortion that occurs when a tube support plate made of carbon steel corrodes to the point where it squeezes down the steam generator tubing.) These improvements, though reassuring, are not enough and much more work is needed. The increase in tube plugging due to "other" causes in 1979 is just one indicator that new problems continue to be discovered.

The major experience with steam generator performance has been gained with the U-tube D&S design (Fig. 2). Some Westinghouse and CE designs have experienced severe corrosion damage, which falls into three categories (Eisenhut, 1979): (1) caustic cracking of the table within the tube sheet or just above the tube sheet in regions submerged in crude deposits; (2) tube wastage (phosphate thinning) in the region just above the tube sheet; and (3) denting at tube/table-support intersections on some plants that operate with all volatile (alkali) treatment (AVT). The distortion caused by denting has in some cases lead to stress-corrosion cracking of the tubing from the inside. Figure 7 schematically illustrates the type and location of corrosion damage, which is primarily restricted to the secondary side of the system.

Table 3. Summary of steam-generator problems (number of units in each group known to be affected by each problem). (Mundis, 1981).

								Fresh water/ cooling	Brackish seawater
	OTSG	Recirculating steam generators					Total PWR		
Problems	All	US ^b	US ^c	Japan	France	Other	units	tower	
	(8) ^a	(8)	(32)	(8)	(10)	(13)	(78)	(47)	(31)
Denting									
Tube support corrosion	0	5	14	0	0	5	24	11	13
Tube sheet corrosion	0	0	9	0	0	4	13	5	8
Tubing corrosion									
Wastage	0	3	12	3	0	4	22	12	10
Pitting	1	0	0	0	0	0	1	1	0
i.d. cracking ^d	0	0	8	1	0	4	13	6	7
o.d. SCC/IGA	0	0	7	2	0	4	13	6	7
Mechanical damage									
Fretting	3	0	2	0	0	2	7	-	-
Fatigue cracking	4	0	0	0	0	0	4	-	-
Impingement	1	0	1	0	0	0	2	-	-
Waterhammer	1	1	6	0	0	1	9	-	-
Carryover	1	0	5	0	2	0	8	-	-
Clad separation	0	0	10	0	0	1	11	-	-

^a Numbers in parentheses indicate total number of operating units in each group.

^b Units with egg-crate tube supports.

^c Units with drilled-hole support plate.

^d i.d. = inner diameter, o.d. = outer diameter.

FIG. 6. The effect of steam generator problems on PWR availability is reflected in this plot of percent of all operating steam generator tubes plugged each year from 1972 through 1979 (Mundis, 1981).

With the exception of the Obrigheim reactor (see Appendix A) in West Germany, KWU plants have not experienced caustic cracking, but very thick crude deposits have accumulated on the tube sheets, and progressive tube thinning has occurred in this region on some plants. Surprisingly, none of these plants has experienced denting.

The B&W OTSG plants (Fig. 3) have not had any of the above problems but have experienced mechanical problems such as high-cycle fatigue failure, fretting (that is, movement between the rubbing surfaces that is oscillatory and small in amplitude), pitting, and erosion/corrosion of tubes—all of which originate on the secondary side of the system (Liau, 1980).

In the following, corrosion and mechanical damage will be discussed in detail. However, it is important to note that not all the problems listed are encountered on plants of a single design. For any one design, the severity of a particular problem can range from extreme to undetectable. Secondary-system corrosion damage, as any

FIG. 7. Corrosion problems in PWR steam generators (Layman, 1979).

steam-generator problem, is a complex function of boiler design, past operating conditions, boiler water chemistry, and the extent of ingress of air and cooling water contaminants.

3.2 CORROSION DAMAGE PROBLEMS

3.2.1 Process of Corrosion Damage

The process of corrosion damage has three steps in a steam generator: (1) the ingress of impurities into the secondary system, (2) the concentration of impurities in the steam generator, and (3) corrosion attack on steam-generator materials by the concentrated impurities. A schematic of these steps and the factors that are pertinent to them is shown in Fig. 8.

FIG. 8. Local corrosion process in steam generators (Stahlkopf, 1978).

3.2.2 Causes of Corrosion Damage

The sources of impurities that enter steam generators are: condenser cooling water leakage; makeup water; demineralizer regenerating chemicals; various soluble and insoluble corrosion and erosion products; some water treatment chemicals, such as phosphates; and miscellaneous contaminants, such as air, oil, preservatives, and packing, which enter when systems are under construction or are opened for repair and inspection.

Factors that act to concentrate chemical impurities in steam generators are shown in Fig. 8. The dominant driving force for concentration is the boiling or evaporation process, which produces relatively pure steam and a chemical residue in the liquid phase. The tendency toward residue formation is counteracted by good hydraulic conditions, which redilute the local build-up area and restore the bulk chemistry conditions that are controlled by blowdown. However, the large "open" volume on the secondary side, in combination with thermal-hydraulic asymmetries, makes it very difficult to assure fluid-flow conditions to adequately wet heat-transfer surfaces on a local scale approximating the size of a single tube. Furthermore, sludge and occluding conditions, such as scale and crevices, are important and complex factors that affect impurity concentration.

On the basis of the field data, it appears that local corrosion damages PWR steam generators. Local accelerated corrosion that has occurred to date has been of various forms--pitting, wastage, intergranular stress-corrosion cracking (IGSCC) and denting. For the most part, local corrosion tends to be restricted to certain zones of the steam generator: tube sheet crevices; areas just above the tube sheet and just underneath the top of the tube-sheet sludge piles; at or near support-plate crevices; and at the upper U-bend regions, particularly at the tube support devices (refer to Fig. 8.). Chemical concentration factors can be associated with all these zones. This observed correlation is the basis for the belief that chemical concentration is the probable cause of tubing corrosion damage.

3.2.3 Corrosion of Tube Support Plate and Tube Sheet

3.2.3.1 Denting and Its Consequences. The most widespread and serious problem observed thus far in recirculating steam generators is denting, the distortion that occurs when a tube support plate made of carbon steel corrodes to the point where it squeezes down the steam generator tubing. As this process continues, the support plate deforms and "hourglassing" of the flow slots results. The cause of denting is best explained by Potter-Mann-type (Potter, 1965) linear accelerated corrosion in which a nonprotective

oxide is formed as the corrosion progresses. The linear growth of the oxide layer, whose volume is greater than that of the parent metal, fills the original tube-tube support crevice and subsequently exerts a compressive force on the tube. Figure 9 illustrates a dent, so-named for the radial reduction in the outside surface of the tube. As the denting proceeds, both the tube and the tube support plate can crack.

Chloride concentrations of over 400 ppm have been measured in sample intersections of tubes and support plates removed from dented steam generators. Laboratory results indicate that denting is an acid chloride reaction. To date, no denting has been obtained with sodium chloride, which is neutral. Results from laboratory capsule tests also indicate that denting increases with increasing chloride concentration.

Local thermal/hydraulic conditions in the crevice where the tube and the support plate meet may be responsible for the high local chloride concentration. Most dents occur in drilled support plates that contain flow holes. In these support plates, it is estimated that only 3 to 8% of the total flow passes through the open (unplugged) tube/tube support crevice. (These estimates were made on the basis of single-phase flow calculations.) The remainder of the flow passes through the flow holes, as shown in Fig. 10. Because a fixed pressure drop exists across the tube support plate, flow hole, and crevice, boiling in the crevice or any buildup of solids in the crevice will increase the resistance to flow through

FIG. 9. Denting at tube/tube support plate intersection.

FIG. 10. Flow between crevice and flow hole.

the crevice. This in turn leads to decreased flow through the crevice and accelerates flow starvation in the crevice. Such starvation in turn accelerates the concentration and deposition of chemicals. This starvation is aggravated because, regardless of the type of tube support, many tubes will touch some part of the support. This contact is illustrated in Fig. 11 for a drilled support plate. Metal-to-metal contact exists over a small but finite angle around the circumference. Near the contact area, a region of local flow starvation exists. This starvation leads to heat transfer dryout, or vapor blanketing, or alternate wetting and drying, all of which will cause the concentration or deposition of chemicals. Boiling with high heat-transfer rates exists in the remaining circumference.

The major consequences when denting progresses uncontrolled are tube cracking and leaking at the U-bends, tube cracking and leaking at the constricted crevice (dents), tube-support-ligament failure (cracks), and excessive deformation of the tube support plate. Damage to the tube sheet due to denting has not been observed. Although the causes of denting are not fully understood, plant experience has shown that denting can be controlled if the chemistry is rigidly controlled (i.e., eliminate as far as possible the ingress of impurities into the steam generator). The successful experience at one unit is shown in Fig. 12. The implications of field experience with tube-support and tube-sheet corrosion problems are clear. It is very difficult to avoid denting, particularly in units with drilled-hole carbon-steel support plates. However, experience at a number of units where great attention has been paid to reducing impurities shows that new denting can be greatly retarded, if not prevented, and denting which has started can be stopped.

FIG. 11. Tube contact within support plate.

FIG. 12. Reduction of impurity ingress (particularly, chlorides, oxygen, and copper oxides) into the steam generator does reduce the rate of corrosion damage progression (Mudis, 1981).

3.2.4 Corrosion of Tube Materials

Corrosion of Inconel 600 tube material can be categorized as wastage (thinning) and stress-corrosion cracking. The following is a discussion of these corrosion problems.

3.2.4.1 Wastage (Thinning). Tube-wall thinning has been observed in the region above the tube sheet (refer to Fig. 7) when phosphate chemistry has been used--hence the term "phosphate thinning." A significant amount of field experience and laboratory work has confirmed that wastage results from the deposit and buildup of sludge on the tube sheet and subsequent concentration of aggressive species (e.g., phosphate) as a consequence of steam blanketing. Figure 13 illustrates the thinning process, which starts when sludge builds up to a critical height on the lower tube sheet so that liquid can no longer penetrate to the lower sludge region to replace the liquid that is being vaporized by boiling. Furthermore, the liquid flows down the space between the tubes and a vapor/liquid mixture flows up the tubes, absorbing heat transferred through the tube walls. At large sludge heights, a steam-blanketing condition likely exists near the tube sheet: there is no introduction of liquid and hence no buildup of solids. The temperatures in this region approach the primary coolant temperature. Near the upper surface of the sludge, adequate circulation of liquid exists and chemicals do not concentrate. The most corrosive region is the intermediate region where alternate wetting and drying occurs.

Since the industry shifted to AVT chemistry in 1974 the progression of tube-wall thinning appears to have diminished to the point where it is not a serious problem (Green, 1980). However, thinning continues to be observed in the plants that have remained on phosphate chemistry.

FIG. 13. Sludge on tube sheet.

3.2.4.2 Stress-Corrosion Cracking. In recirculating steam generators, intergranular cracking of the Inconel 600 tubing has occurred at various locations (e.g., within the tube sheets, just above the tube sheets, in dents at support plates, and in tight-radius U-bends). It is an interesting contrast that cracking of Alloy 800 tubes in KWU-designed steam generators has never been reported (Garnsey, 1979).

Although we do not fully understand the mechanisms that cause the Inconel 600 cracks at these various locations, crack characteristics and the prevailing environment support the view that stress corrosion is dominant (Layman, 1979): First, concentrated caustic environments occur when boiler water (which contains water that has leaked in from condensers cooled by certain fresh-water sources) dries out. Second, although normal operating stresses for the tubes are far below the yield stress, analyses have shown that once support-plate corrosion products close the clearance annulus, additional corrosion can readily cause the tube to yield.

Stress-corrosion cracking was also found in a tube pulled from an OTSG (Layman, 1979). Sulfur was present and may have been involved since it has long been recognized that sulfur can induce cracking of Inconel 600. Sulfur can come from resin beads, resin fines, demineralizer regeneration with sulfuric acid, and treatment of cooling tower water with sulfuric acid.

3.3 MECHANICAL DAMAGE PROBLEMS

Mechanical damage has to date been restricted to OTSGs, but conditions are present in recirculating steam generators that could also result in fretting, wear, and fatigue over the long term once the short-term corrosion problems are eliminated. Again, this type of damage process originates on the secondary side of the system, although effects of the primary side may contribute to crack initiation.

3.3.1 Fretting and Wear

Tube fretting and wear is caused by fluid-induced vibration. In PWR steam generators, a clearance exists between the tube and the tube support plate through which the tube passes. This clearance is necessary for manufacturing and design considerations. Vibration can cause a tube to impact and slide tangentially against the tube support plate, which results in local wear (i.e., loss of tube and tube support plate metal). Because the movement between the rubbing surfaces, (i.e., the tube and the tube support plate) is oscillatory and usually small in amplitude, the rubbing process is termed fretting.

Tube-metal loss is detrimental, and fretted regions are highly sensitive to fatigue crack. Under fretting conditions, fatigue cracks can begin close to the material surface at a stress level well below the fatigue limit of nonfretted tube test material (Green, 1980).

3.3.2 Fatigue Cracking

High-cycle fatigue conditions exist in PWR steam generators. In recirculating plants, the Inconel tubing is most likely to vibrate and fail from fatigue just above the tube sheet and in the U-bend region where the flow velocities of the primary side are highest and cross-flow from the secondary side is significant. Although tube cracking has not yet been observed in these regions, it is judged that the steam generators have not been operated long enough for such cracking to be seen.

Peripheral tubing observed in OTSGs has been found to be liable to fatigue (Layman, 1979). Cracks observed at the upper tube sheet and highest tube-support plate began on the outside and were oriented circumferentially. Various cracks, including through-wall types, were found. All cracks were located at defects (i.e., at regions of localized wall thinning) on the outer surfaces of the tubes. Examination of the crack striations indicated that the propagation phase of the crack experienced many cycles. It has been estimated that a stress level of only about 89 MPa was required for crack propagation--an indication that this tubing is much weaker than would be assumed from the usually reported fatigue strength of Inconel tubing materials (Green, 1980). Therefore, initiating mechanisms such as fretting fatigue, localized corrosion, or corrosion fatigue are also considered to be possible contributing factors.

3.3.3 Impingement and Erosion-Corrosion Effects

Impingement of solid particles may damage tubes in some OTSGs. Metallurgical examinations of three tubes removed from the affected plants could not pinpoint the exact cause of the observed damage, which was greater than 40% through-wall. However, impact of solid particles has been identified as a highly probable cause. Impingement of a liquid or solid particle is known to mechanically damage materials under certain environments. Potential sources of solid particles include any particles entering the steam generator from the feedtrain and particles entrained from within the steam generator. Internally entrained particles can be found in sludge previously deposited or products generated within the steam generator, e.g., magnetite at tube-support plate

crevices. In lieu of a precise understanding of impingement and erosion corrosion, limiting the ingress of particulates is the only way likely to stop this problem in operating plants.

3.3.4 Water Hammer

Water-hammer incidents have occurred in feedwater piping of the steam generator for five feed-ring-type steam generators and can occur in the preheat section of economizer-type steam generator designs, which are now in construction (Layman, 1979).

Four of these plants sustained extensive and permanent pipe deformation and pipe-support damage or component fracture. Most of these water-hammer events occurred when cold auxiliary feedwater was introduced into the main feedwater piping after the feedwater ring sprayers uncovered, drained, and filled with steam during an abnormal operating transient (such as main feedwater pump trip). The process of water hammer can be described step by step as follows:

1. The flow is interrupted and susceptible regions fill with steam.
2. When the feedwater flow resumes, a volume of steam is trapped and rapidly collapses because of heat transfer to the colder water.
3. The rapid collapse of the steam volume creates a pressure differential that accelerates a slug of water through the susceptible region.
4. The accelerating slug of water impacts (hammers) a barrier.

Because steam-pocket formation is a complex process, data on direct-contact condensation rates is lacking, and the liquid slug's length and interface geometry is incompletely understood, the forces resulting from a water-hammer event cannot be accurately predicted. Therefore, current industry practice includes combining design modifications and administrative controls (to prevent formation of a critical steam volume) with testing (to verify that either the event will not occur or, if it does occur, the event will not result in damage). For example, adding design features to prevent the feed ring and inlet piping from draining after a loss of feedwater flow has been effective in preventing water hammer in feed-ring steam generators.

3.3.5 Carryover

Several steam generators have been altered in the field to increase the operating recirculation ratio of the units. This increase in recirculation ratio is intended to decrease tube-sheet sludge deposits and to reduce the tendency for dryout, thereby

decreasing chemical concentration in the crevice. However, an increased recirculation ratio increases the loading on the primary separators--it is possible to exceed the allowable carryover value, i.e., the limit of moisture or bad quality of steam carried over from the steam generator to the turbine, in some applications.

Work is underway to evaluate the potential gains from improving the performance of primary steam separators. Possible improvements include reduction of carryover for increased loading and reduction of overall pressure drop, which would lead to an increase in recirculation flow. Further analytical and experimental projects are planned (Layman, 1979).

4. AVOIDANCE OR MITIGATION OF STEAM GENERATOR DAMAGES

4.1 INSPECTION AND DETECTION OF DAMAGES

Before any actions can be taken to prevent or mitigate damage to the steam generator resulting from corrosion or mechanical problems, one first has to locate the damaged regions and to determine the degree of damage.

4.1.1 Hydrogen Evolution Monitoring

A system to measure hydrogen evolution across a steam generator has been successful at several operating plants in determining relative corrosion rates (or change in corrosion rate) within the steam generator. The system operates on the principle that hydrogen is produced as a result of the corrosion processes that occur in the steam generator. Such a system will help operating personnel to identify actions that cause a change in corrosion rate, to take corrective action where appropriate, and to evaluate the effectiveness of the corrective actions taken.

4.1.2 Nondestructive Examination (NDE)

"Rules for Inservice Inspection of Nuclear Power Plant Components," stated in the Code of Federal Regulations (NRC, 1979), is a mandatory requirement for operating nuclear power plants in the United States. The following (Layman, 1979) is a summary of the state of the art of NDE methods presently available for field use and a summary of present development efforts.

4.1.2.1 Eddy Current Testing. The NRC Regulatory Guide 1.83 specifies the use of eddy current testing (ECT) for periodic inspection of steam generator tubing. This test is normally performed with a single frequency--usually 400 kHz for routine inspections and other frequencies for special inspections such as sludge height. While many advances have been made in the 15 years since ECT was developed, numerous limitations exist and have become more apparent from recent problems, such as tubing denting and vibration-induced cracks. The limitations include inability to discriminate tube flow signals near tube support plates and the tube-sheet area, influence of work hardening on the signal, need for increased precision in measuring tube thickness (particularly in the range of 40% of wall thickness and below), and the need to increase the speed and reliability of data interpretation, which is presently slow and depends on the operator's

skill and experience. Single-frequency eddy-current systems are currently being developed to overcome these limitations, with emphasis on the development of special probes and the use of automatic signal processing.

To compensate for the drawback of ECT mentioned above, multifrequency eddy current testing (MECT) procedure has been developed and used extensively in Europe. Recently, MECT was successfully used in the United States for inservice inspection of steam generators in an operating plant. This method involves simultaneously recording eddy current signals from several frequencies, followed by electronic manipulation and subtraction to suppress unwanted signals from support plates, etc., so that defect signals will be more easily detected and interpreted. This technique is a significant improvement over conventional single-frequency methods. Major benefits include increased ability to detect and size flaws near support plates and dented regions and the capability to record data at several different frequencies with one pass through a tube. It has been found that the system's usefulness would be increased by recording additional channels of data and by operating in parallel with conventional single-frequency instruments to facilitate comparison with previous inspection results and interpretation methods.

Because MECT is likely to improve inservice inspections, work is presently underway to further characterize its capabilities and provide a system for use by public utilities. Various commercially available MECT components have been combined into a system that has been used by utilities for inservice inspection with excellent results.

4.1.2.2 Visual Inspection Techniques. Two optical methods of inspecting tubing are available, but each has limitations. Use of a fiber optics probe is presently limited by the rapid decrease in sensitivity and image intensity as the fiber bundle lengthens. Thus, fiber optics can only be used to inspect a few feet of a tube. The miniature TV cameras available are relatively fragile, have limited resolution, and do not fit all tube sizes. Therefore, a probe that uses a solid-state linear diode array is now under development; this probe will also provide data in a format convenient for additional analysis by digital electronic techniques.

4.1.2.3 Ultrasonic Techniques. Conventional ultrasonic techniques are possibly useful for tubing inspection, but the requirement for a fluid couplant between transducer and tube wall makes this method impractical. However, generating ultrasonic waves in a tube with electromagnetic excitation has been shown to be feasible. These electromagnetic acoustic transducers, which are currently under development, require no fluid couplant.

4.1.2.4 Others. Except for direct visual examination where possible, no common NDE technique exists for determining support plate integrity. Low-frequency eddy-current testing looks promising, but this method has not been fully investigated.

Radiographic methods of inspecting the support plate ligaments between adjacent tubes have been shown to be feasible and further development continues. Also under investigation is a technique to measure the effect of support-plate cracks on the magnetic flux pattern from a direct-current coil probe. It is important to accurately determine dent size and shape if one is to determine the stresses present in the tube and support plate. Three methods under development may address this problem: One device uses eight fingers that ride on the inner surface of the tube. The fingers are attached to strain gages, which measure finger deflection at dents and measure eight radii of the dent. A second method will deliver a Silastic compound to a dent and make a cast of its shape. The third method was previously mentioned--use of a solid-state diode-array scanner for inspecting the tube interior. This device measures the dent profile quite accurately.

4.2 COST AND BENEFIT OF NONDESTRUCTIVE EXAMINATION

Satisfactory resolution of the problem of degradation of steam generator tubing requires periodic inspection to assess tube damage. NDE or in-service inspection (ISI) provides the principal measure of the tubing response to the working environment and, therefore, the principal guide to improving design, operation, and maintenance practices that can optimize this response. ISI, in conjunction with degradation limits (e.g., plugging limit) on serviceability, permits censoring of tubes that reach the "unacceptable" region. Naturally, the question is raised about the cost of various ISI plans for steam generators and the benefits of these plans. It is known that unscheduled outages due to steam generator problems can result in power plant shutdown. The cost of downtime* for power plants in various regions of the United States has been estimated (Zima, 1980). For California and the northeastern United States, the average cost of downtime is 30 mills/kWh. The average value for downtime in the northwest is 17.5 mills/kWh. For the rest of the nation, the average cost of downtime is 23.75 mills/kWh. Downtime costs do not include the maintenance cost for increased inspection and repair of steam generator tubes. Because downtime is expensive, a planned ISI of steam generator performance is a justifiable precaution to lower the risk of plant outage. On the basis of the results reported in Zima (1980), it was concluded that the cost of a thorough plan of ISI over a nine-year evaluation period is only one-third the cost of steam generator downtime for all power levels. Therefore, to adopt a thorough plan of ISI is very promising from the cost/benefit standpoint for utility owners.

*From the view of the utility, the fuel costs are "saved" and the downtime cost is simply electricity replacement cost minus the fuel cost saved.

4.3 CHEMISTRY CONTROL

Corrosion damage in PWR steam generators can be avoided by eliminating the ingress of chemicals or by lowering their concentration, i.e., control of the local chemistry. The following subsections (Layman, 1979) describe various methods to control the chemistry for steam generators.

4.3.1 Deoxygenation of Secondary Water System

The fact that oxygen is present in the secondary plant water and systems both during plant shutdown periods and during operation is significant in studying the corrosion process. This oxygen contributes to sludge formation in the steam generators by corroding secondary plant piping and components.

Oxygen also directly corrodes steam generator internal parts and may also directly cause denting. Because current plant designs and operating procedures make it difficult to eliminate oxygen from the secondary plant, development work is needed in this area.

4.3.2 Chemical Cleaning

Utilities that have either incipient denting or moderately dented units have considered cleaning the affected units with chemicals. Experience with the cleaning process itself (solvents, inhibitors, conditions) and its side effects is still limited. Chemical cleaning processes have been somewhat successful in removing steam-generator sludge that is high (up to 50%) in copper content.

Recently, a utility cleaned three steam generators before operation began. The effort concentrated on removing hematite with an ammonia citrate solution coupled with a proprietary inhibitor.

4.3.3 All-Volatile Treatment (AVT) Chemistry

Denting continues to occur in recirculating steam generators even though chloride levels are low, which indicates that drilled carbon-steel support plates are susceptible to accelerated corrosion when AVT chemistry is used. One plant operating with chloride levels in the 20- to 50-ppb range has experienced denting. Extraordinary measures in both operation and chemistry control may be required in such units to avoid denting.

4.3.4 Neutralization of Crevice Acids

Crevice acids and, in particular, acid-chlorides have been identified as media that cause denting. Therefore, it is thought that denting can be stopped or reduced if the crevice acid is neutralized by adding a base. If the pores are plugged, acid chlorides cannot penetrate the crevice magnetite and the reaction will be stopped because the corrodent is limited. Boric acid is the pore-plugging agent of choice. Sodium phosphates and boric acid, chemicals used for the near-term remedy of denting, have shown the most promise in the laboratory capsule, heated crevice, and pot/model tests. An accelerated laboratory test program is underway to quantify both benefits and detrimental side effects of using phosphates off-line.

4.4 MECHANICAL MODIFICATION

Many plants have installed or plan to install either deep-bed or precoat condensate polishers. Full-flow condensate polishing systems can reduce the impurities that enter the steam generator from the condensate system and thus avoid or reduce denting. Such systems can provide a degree of protection from limited leakage of cooling water into the condensers. Recent information from operating plants indicates that very low levels of impurities in boiler water (<50 ppb chlorides) can cause denting in sea-water plants that use AVT. The use of condensate polishers may be necessary in such plants if denting is to be prevented. Additionally, they act as filters, both before startup and during operation, by removing insoluble corrosion products which would otherwise be deposited as sludge in the steam generators. Laboratory tests to quantify the benefits and potential side effects that result from operating with condensate polishers are underway. Preliminary results indicate that resin specification and quality control, improved resin regeneration procedures, and proper monitoring are required (Harhay, 1979).

4.5 OPERATIONAL PROCEDURE

Impurity (i.e., chloride, sulfate, silica, calcium, magnesium, etc.) concentration in crevices can be reduced by operating at reduced power and by performing water soaks. At some level of reduced power, the cycle of wetting and drying in the crevices may be so reduced that the crevice stays wet. If this is the case, impurities diffuse out of the crevices into the bulk fluid and are removed from the steam generator by blowdown. The power level at which this occurs varies from plant to plant and probably from crevice to

crevice within the same steam generator. A test program now underway will quantify (and optimize) the conditions under which water soaks effectively reduce concentrations of impurities in the crevices and thereby reduce denting in steam generators. Preliminary results indicate that repeated water flushes reduce even stifle denting.

4.6 REPAIR AND REPLACEMENT OPTIONS

To sustain operability over plant lifetime, the ability to maintain and upgrade nuclear steam generators is desirable. The available methods include plugging tubes, replacing the lower shell and tube-bundle assembly, retubing, and replacing entire steam generators.

Plugging is the first step in maintaining the service availability of a steam generator when the tube integrity is questioned. The second step is retubing, a systematic process developed for the in situ replacement of the tubing in PWR recirculating steam generators. The alternative to retubing is replacing the lower shell and tube bundle assembly. The very last step to keep the power plant operating is to replace the whole steam generator when the number of plugged tubes in the steam generator exceeds a certain limit (usually 30% of the entire generator tubing) or when retubing is no longer economical. Replacing a steam generator is costly (about \$100 to \$300 million; Green, 1980) and needs to be thoroughly planned. Recently, three steam generators at Surry Unit 2 nuclear power station were successfully replaced (Parrish, 1980).

5. FOREIGN EXPERIENCE

5.1 GERMAN PWR STEAM GENERATOR EXPERIENCE

5.1.1 Description of German PWR Steam Generators

As of June 30, 1981, 11 nuclear power plants were in operation in the Federal Republic of Germany (FRG) with a total generating capacity of more than 8500 MWe (six of the plants are PWRs; see Appendix A). This represents about 14% of the total electricity generated in FRG. As can be seen from Appendix A, the German nuclear reactor market is dominated by one major firm, Kraftwerk Union (KWU). In addition to supplying power plant reactors in FRG, KWU also exports to Argentina, Austria, Brazil, the Netherlands, Spain, and Switzerland. Presently, 10 power plants with KWU PWRs are in operation worldwide. The accumulated service life of their 30 steam generators amounts to about 180 years.

The steam generators in power plants with PWRs built by KWU have the following features:

a. KWU steam generators use a steam separation system that consists of 50 cyclones that separate the generated steam from the circulating water. The separated steam is then dried in chevron-type driers.

b. The egg-crate tube support system is used. Each grid consists of two rows of austenitic stainless steel strips. This design reduces the pressure loss on the secondary side and minimizes the danger of particle deposition.

c. Curved stainless steel strips support the U-bends. They prevent the tubes from vibrating during normal operation and are designed to support the tubes if a steam line breaks. The hazard of steam blanketing or particle deposition is thereby also minimized.

d. The feedwater spargers are equipped with syphon-type elements to prevent the ingress of steam if the water level should drop below them. Furthermore, a welded connection between the thermal sleeve and the nozzle does not allow the steam to enter the feedwater line. Because of this design no water-hammer incidents have ever been observed in KWU steam generators.

e. Incoloy 800 has been used as the tube material in all KWU steam generators, except for Obrigheim steam generators, where the two steam generators are tubed with Inconel 600.

f. In addition to the welded joints between tubes and the tube sheet, tubes are expanded by a rolling procedure at the upper and lower ends of the tube sheet.

Consequently, the crevice between each tube and tube sheet is closed, thus preventing crevice corrosion.

g. The secondary side of the steam generator is operated with phosphate chemistry. An equilibrium concentration of 2 to 6 ppm PO_4 is maintained in the steam generator water by continually adding disodium phosphate to the feedwater. Only the Obrigheim steam generators are operated with volatile alkalization of the water.

5.1.2 Operating Experience

Schücktanzen, Riess, and Stieding (1981) delivered a paper at the American Power Conference on April 28, 1981, to summarize German nuclear steam generator experience. Highlights of this paper are described in this subsection.

Operating Experience of Obrigheim Power Station. All KWU steam generators have operated to date without tube leakage, except for Obrigheim steam generators. Steam generator tubes were inspected by ECT with the aid of automatically operating manipulators developed by KWU. During a steam generator inspection in 1972, after 2.5 years of operation of a Obrigheim power station, 25 slightly leaking tubes were found in steam generator No. 1. The leaks were in the center of the tube bundle on the hot side of the steam generator, just above the tube sheet. Three tubes were removed for further investigations. The metallographic findings indicated

- a 15-mm axial intergranular crack that ran from the outer to the inner surface of the tube, and
- several longitudinal and circumferential intergranular cracks, starting from the inner surface at the transitions of the three expanded portions of every tube.

The cause of the cracks was determined to be intergranular stress corrosion. The sensitivity of Inconel 600 to this kind of corrosion was one of the major reasons why KWU adopted Incoloy 800 as a tube material.

Operating Experience of Stade Power Station. The first steam generators equipped with Incoloy 800 tubing were those at Stade nuclear power station. Stade is a 630-MW four-loop plant which began commercial operation in May 1972. The design of the Stade steam generators is essentially identical to that of the Obrigheim ones except for tube material and water chemistry.

After a few weeks of operation with volatile alkalization, followed by a chemical cleaning, the water chemistry on the secondary side was changed over to phosphate treatment with 2 to 6 ppm PO_4 .

The Stade steam generators have been ECT-inspected five times to date. During the first seven years of operation, five tubes were removed for metallographic examination.

Sludge deposits were revealed by ECT on the tube sheet in the center of the tube bundle on the hot and cold legs up to a height of 80 and 180 mm, respectively. The height of these deposits has not increased during the last five years of operation. The same behavior has been observed in the steam generators of other plants. In 1979, ECT indications of wall degradation were found in several tubes just above the tube sheet on the hot-leg side in the region of sludge deposition. No such indications were found in any other portions of the U-tubes, namely the regions around the tube supports, the bends, and the areas of sludge deposition on the cold side of the tube sheet. Two tubes were removed and examined metallographically. The metallography revealed wastage just above the tube sheet, which extended axially about 15 mm and covered about one-third of the tube circumference. No other type of corrosion attack could be detected, on either the outer or inner surfaces of the tubes.

A comparison of the ECT and metallographic results showed that the depth of the defects was over-estimated by ECT by about 10%. In spite of the conservative ECT results, it was recommended to the operating utility that all the tubes yielding ECT indications of more than 50% be plugged.

From a comparison of the ECT indications with the results of preceding inspections it was deduced that the annual corrosion rate of the affected tubes amounted to 6% on average with a maximum of 10%. A tube-wall degradation of no more than 70% of the wall thickness was demonstrated, however, to meet all the requirements of the applicable German codes. An inspection interval of two years would, therefore, be sufficient to ensure that no tube would start leaking during the operating periods between inspections.

Typical Operating Experience of KWU Steam Generators. Results similar to those of Stade were obtained from the ECT inspections in other plants, namely:

- No stress corrosion. Incoloy 800 is not susceptible to either intergranular or transgranular stress corrosion under the operating conditions of a steam generator. Nevertheless, in order to avoid any risk of stress corrosion, the latest steam generators are equipped with shot-peened tubes. A glass-bead peening process after tube bending introduces compressive stresses into the outer surface layers of the tubes, thus eliminating the possibility of stress corrosion by definition.

- No fretting. The egg-crate tube support is designed to avoid any unacceptable tube vibration. The flexibility of this system of stainless-steel strips also reduces the danger of fretting.

- No denting. A prerequisite for denting is the existence of a carbon-steel tube-support structure with the potential for the concentration of aggressive contaminants, such as chlorides, in the steam generator water. Tube supports, including the Obrigheim ones, are made of stabilized austenitic stainless steel (1.4550 or ASTM 347) in all steam generators. A series of tests has verified that this material does not suffer from fast linear growth of magnetite, even in the highest chloride concentrations.

- Tube thinning was detected on 1 to 10% of the steam generator tubes in the center of the hot-leg tube bundles. A total of 142 tubes--approximately 0.1% of all the tubes in operation--in 26 steam generators had to be plugged.

Phosphates are added to the steam generator water to maintain the high pH level required to prevent any selective corrosion from ingress of salts originating from condenser leaks. However, phosphate treatment causes wastage corrosion when the phosphates are concentrated to a critical extent in flow-stagnation zones. This may occur in sludge accumulations on the tube sheet. Investigations of the accumulated material, as well as of the removed tubes, indicate that a critical concentration does occur in the zone between 10 and 20 mm beneath the upper surface of the sludge pile. Corrosion in that zone can then be caused by alternate wetting and drying of the tubes. As evidenced by ECT procedures, the sludge pile on the hot leg of the tube-sheet reaches an almost stable configuration within the first one or two years of operation. The critical concentration of phosphates, therefore, always occurs on the same portion of the steam generator tubes, resulting in wastage corrosion of the tube material.

Remedial Actions. Since wastage corrosion is caused by the concentration of salts in flow-stagnation zones of the steam generator, three possible remedial actions must be considered.

- Prevention of any low-flow zones in the steam generators. Because of the egg-crate tube support system in the steam generators, only the area above the tube sheet need be considered for improvement in flow distribution. A flow distribution baffle was designed to avoid accumulation on the tube sheet. A drilled stainless-steel plate placed 350 mm above the tube sheet forces the circulating water to flow towards the center of the bundle, thus minimizing the flow-stagnation zone and preventing particles from dropping out. Most of these particles remain in suspension and are removed from the steam generator via the three blow-down lines located at the lower end of the downcomer.

The flow distribution plates were first installed in the steam generators of the Goesgen plant in Switzerland. An ECT inspection in the summer of 1980, after about 18 months of operation, revealed that the amount of deposition on the tube sheet was

reduced to about 20% of that in comparable steam generators. Since this means of flow control did not completely prevent deposition, additional measures should be adopted.

- Prevention of condenser leakages. Corrosion attack on Incoloy tubes can only occur if salts are concentrated at the tube surfaces to a critical extent. The salts arise from condenser leakages (chlorides) or from the chemical treatment of the steam generator water (phosphates). Operation without any salts would, therefore, effectively preclude corrosion. The elimination of the phosphates could, however, result in the danger of selective corrosion, e.g., pitting, or stress corrosion caused by salts originating from condenser leakages. This type of corrosion need not be limited to the areas of sludge deposits, but may affect the whole tube length. Eliminating phosphates while the possibility still exists of salt ingress into the steam generators is, therefore, no solution to the corrosion problem. Such a measure could under certain circumstances intensify or expand the problem.

Consequently, KWU decided to eliminate salts from the feedwater steam circuit by redesigning its condensers. To ensure the maximum possible tightness against leakage, only corrosion/erosion-resistant and copper-free materials, such as stainless or titanium, will be used for the condenser tubes. Furthermore, the condenser tubes will be welded to the tube sheets and subsequently expanded. The resulting leak-tightness will be comparable to that of the tube-to-tube-sheet connections in the steam generators.

Additional advantages expected from using copper-free tube material include (1) chemical cleaning of the steam generators will be easier, and (2) the pH value of the feedwater can be increased, which will result in a lower corrosion rate of the material in the feedwater train and thus in fewer corrosion products flowing into the steam generators.

The installation of full-flow demineralizers is presently not recommended by KWU because of the risks that might be involved in operating such equipment, e.g., steam-generator-tube corrosion by resin fines. If a demineralizer is installed, it should only be operated in the event of major condenser leakages as a result of tube breaks in order to obtain more flexibility in scheduling repairs.

- Reduction of the corrosion products flowing into the steam generators. Obviously it is technically impossible to avoid any stagnant zones in the steam generator. On the other hand, there are no means of completely preventing salt ingress into the steam generators. KWU recommends, therefore, that the quantity of corrosion products being carried into the steam generators be minimized. This will be achieved by installing recirculation lines and filters in the secondary system.

The recirculation line enables the steam/feedwater system to be flushed before start-up after every plant shutdown period. The provision of mechanical filters in the

main feedwater line ahead of the feedwater storage tank, as well as the installation of electromagnetic filters in the condensate lines from the high-pressure turbine, reduces the corrosion products being transported during the operation of the plant. The entry of corrosion products into the steam generators can be reduced by a factor of 5 by all these measures.

In addition to these equipment improvements, procedures for mechanical and chemical cleaning of the steam generators have been developed and can be used if necessary. The above-mentioned measures will be adopted in all future nuclear stations and, as far as necessary, in those plants under construction. The remedial actions to be taken in operating plants depend on the service experience with each individual installation.

5.2 FRENCH PWR STEAM GENERATOR EXPERIENCE

5.2.1 Introduction

The French have developed a consistent nuclear energy policy despite political changes. In the early 1950s a modest start was made; the result was that 2.6 GWe of mainly gas-cooled graphite reactors was in operation by 1972. In 1970 Electricite de France (EdF) decided to make 50% of its future investment in nuclear plants. In 1973 the commitment was increased to two-thirds and in 1974 it was decided that nearly all new power plants would be nuclear.

Similarly the choice of reactor type has evolved over the years. The early gas-graphite systems were followed by a first PWR at Chooz, which began to operate in 1967. The current French program involves building 35 GWe of PWR stations by 1985. The first PWR units to become operational under this plan were at Fessenheim, followed by further units at Bugey, Dampiere, Tricastin and Gravelines. The PWRs are grouped on 20 sites, mostly on major rivers rather than at the coast.

The consistent program has enabled the industrial infrastructure to develop in step with the requirements. Framatome was set up in 1958 by several firms in the Empain-Schneider group. Its plants at Chalon and Le Creusot now have an annual capability to build 8 reactor vessels, 24 steam generators, and 8 pressurizers. Two Empain-Schneider subsidiaries, Creusot-Loire and Jeumont Schneider, have the capacity to produce 8 sets of reactor internals, 24 primary rotor pumps, and 8 sets of control rod mechanisms per year. With the French market contracting from 5.5 GWe/year to 5.0 GWe/year in the late 1980s, these companies have been looking for overseas sales to maintain their plants at full capacity. That 25% of world PWR orders were secured by Framatome during 1973-77 augurs well for their export program.

Since 1969, Framatome has received a total of 59 orders for 900-MWe and 1300-MWe nuclear steam supply systems (NSSS). Among them, 9 orders are export orders--all for 900-MWe units. The French nuclear program includes 34 900-MWe units and 16 1300-MWe units.

5.2.2 Operating Experience

Sixteen nuclear plants are already in operation. In 1980, 10 units were fuel-loaded and full-power operation was achieved on 7 plants. In 1981, 8 units will be fuel-loaded and started. Up to 1988, 5 to 6 units will be put in operation each year.

The mean availability for the first five EdF plants was 72% in early 1980 and 87.2% in December 1980. According to Berge and Vignes (1981), severe corrosion problems have been encountered on some French PWR steam generators. The remainder of this subsection describes the actions taken by Framatome-EdF and improvements achieved to date to solve the corrosion problems.

Steam Generator Design and Manufacturing Improvements. The need for a standard design to minimize manufacturing and plant costs has led Framatome to manufacture only two types of steam generators, after the Fessenheim and Bugey units: The 900-MWe units are 51 B Model and the 1300-MWe units are 68/19 Model. The original 51 Model steam generator was a W-design similar to the Beaver Valley Unit.

The Fessenheim and Bugey steam generators incorporate some of the modifications made to the original design, which led to the 51 B Model. Most of these modifications were aimed at improving the thermal-hydraulic conditions at the bottom of the tube bundle, in order to decrease the sludge accumulation (or sludge-pile buildup) on the tube sheet. The sludge accumulation between the tube and the tube sheet accelerated the risk of tube corrosion and stress corrosion during operation. The 68/19 Model is the original Framatome design with all the features of the 51 B incorporated. Both 51 B and 68/19 models have no economizer sections.

The main steam generator design and manufacturing improvements incorporated on the 51 B and on the 68/19 were aimed at (1) decreasing the risks of tube corrosion in order to avoid stress corrosion cracking or thinning of the tube, and (2) decreasing the risks of tube-support-plate corrosion in order to avoid denting.

The risks of tube corrosion have been decreased by increasing the circulation ratio (3.8 to 4.2) through improvement (on the 51 B Model) or redesign (on the 68/19 Model) of the primary moisture separators, by removing the flow restrictor, and by lengthening the tube wrapper in order to increase the water speed on the tube sheet. Other measures include adding a flow baffle plate to insure flow penetration at the center of the bundle

and increasing the blowdown capacity and efficiency in removing solid particles. The latter was achieved by a redesign of a central blowdown line and the use of lane blockage to extract fluid from low flow area (the most probable area for solid concentration). This redesign of the blowdown line included a material change to Inconel, which allowed an increase in the fluid velocity resulting in increased capacity. Tube corrosion can also be minimized with the use of thermally treated Inconel 600 tubes, which reduce the material susceptibility to caustic stress-corrosion cracking. Another tactic is to "eliminate" the crevice between the tube and the tube-sheet through mechanical rolling of the tube along the full depth of the tube sheet. Also, greatly reducing the residual stresses on the o.d. tube surface in the transition zone of the mechanically expanded tube at the secondary side of the tube sheet will lower the risk. A special roll procedure with a specific tool has been developed by EdF and Framatome, in order to get a transition zone with a very low level of residual stresses.

Tube-support-plate corrosion, which can lead to denting, can be decreased in a number of ways. Tube-support plates with quatrofoil design direct the bulk fluid along the tube surface. This prevents the gap between the tube and the tube plate from being filled with solids (magnetite and sea water salts) and avoids the hide-out of acid chloride. Other designs use broached tube-support plates in 405 stainless steel or a French version of the 13% Cr steel. Some steam generators are equipped with broached tube support plates of carbon steel. Model boiler studies have shown that the quatrofoil design prevents the accumulation of solids between the tube and the tube support plate and so, corrosion of the carbon steel. The use of 13% Cr steel as the tube-support-plate material is a supplementary conservative means. The use of a broached support plate also results in lower tube-wall superheat and offers less flow resistance, which leads to higher recirculation. Other design and manufacturing improvements include a sealed, thermal-sleeve, J-nozzle-type feed ring to minimize water hammer potential; perforated plates for the steam dryers to improve the steam quality; and a special procedure for the tubing of the steam generator to avoid any contact between the tubes in the U-bends and to suppress all risk of wear between two tubes by friction.

Secondary Circuit Material and Design Improvements. Although Framatome is not responsible for the design of the balance of the secondary circuit, the necessity of considering the integrity and compatibility of the steam-water circuit as a whole is fully appreciated by Framatome and EdF. The choices of materials and of some design features have been guided by several aims:

- To avoid the ingress of river-water or sea-water components into the steam generator, by ensuring the maximum leak-tightness of the condenser tubes. In river-water plants, condenser tubes are generally made of brass. In all sea-water plants,

titanium is and will be used for condenser tubes. The 0.5-mm-thick titanium tubes are rolled-welded. Furthermore, to avoid any leak due to tube expansion in the tube plates, the latter are made of twin plates with an intermediate space filled with demineralized water under slight positive pressure, acting as a hydraulic barrier. The tube plate material, on the sea-water side, is a copper-aluminium alloy with 5% of nickel. A model of another type of tube plate is now under test. This is a single plate in which each bore is recessed at mid-thickness. All recesses are inter-communicating and filled with demineralized water under slight positive pressure as a sealing barrier. On the other hand, on Unit 2 of the Paluel plant, one copper-aluminium plate will be replaced by a steel plate that is plated with a sheet (5-mm thick) of titanium. After expansion, the titanium tubes will be welded to the titanium sheet.

- To reduce the amount of corrosion products from feedwater-train materials transported into the steam generator. In all sea-water plants and in some river-water plants, carbon steel and stainless steel will be used for the feedwater heater tubes and for the moisture-separator reheater tubes. In the other river-water plants, only the first low-pressure feedwater heaters will be in brass. Because non-copper tube material is used, the pH value of the feed water can be increased and the corrosion rate of the feedwater heater tubes will be lower; thus, the height of the sludge pile on the tube sheet will be reduced. When copper-bearing material is used, large quantities of copper produce a coherent mass of deposit that is difficult to remove by blowdown or by sludge lancing. Furthermore, copper dissolved from the feed-train materials enters as Cu^{2+} ions and can act as an oxidising agent, increasing the risks of denting on sea-water plants because of the ingress of chloride in the steam generator. The all-volatile treatment, compared with the phosphate treatment, strongly affects sludge accumulation and sludge hardness. The removal of corrosion products from sludge pile by blowdown and sludge lancing is easiest with the AVT treatment.

- To reduce air ingress in the steam generator. Air ingress affects denting either directly or indirectly through corrosion of the copper bearing material. Air ingress cannot be prevented completely, but the amount can vary widely, depending on feed-train design. Deaerators will be installed on all 900-MWe units of the CP2 program (10 units) and on all 1300-MWe units. No full-flow condensate polishing plant is planned on any EdF plant although space has been provided for future installation on some sea-water plants. Each Koeberg plant will be equipped with a full-flow condensate polishing plant and a titanium condenser.

Chemistry specifications. All plants manufactured by Framatome have been operated with AVT specifications.

The feedwater pH is controlled by a volatile amine, (ammonia or morpholine) and oxygen is removed by adding hydrazine. The use of morpholine to control the pH of the feed water has a specific advantage in reducing the amount of corrosion-erosion occurring in the carbon steel piping between the turbine and the main steam recirculation (MSR). Tests performed on Fessenheim 2 (3.8 ppm of morpholine in the condensate water) have shown that the morpholine treatment reduces the corrosion rate in the MSR by a ratio of 3 to 4 compared with the corrosion rate with the ammonia treatment. The morpholine treatment also reduces the amount of corrosion product that is produced in the feedwater reheaters and enters the steam generator, particularly when copper alloys are used in the low power feedwater heaters. Nevertheless, the use of morpholine produces an increase of the cationic conductivity at the steam generator blowdown. This increase is a major drawback with respect to the detection of low-level ingress of pollutants.

The last orders for 900-MWe units and all orders for 1300-MWe units specify the use of stainless steel piping between the turbine and the MSR and for MSR tubes. Stainless steel piping will reduce the erosion-corrosion problem and will decrease the major disadvantage of the morpholine treatment. The oxygen content of the condensate water can vary widely from plant to plant, depending on feed-train design. For example, at Tihange 1, the oxygen content of the condensate water is so low (<3 ppb) that no hydrazine is added. At Fessenheim 2, the oxygen content of the condensate water is also very low (<4 ppb). But at Fessenheim 1, the oxygen content has been for some periods of the order of 25 ppb. To decrease the oxygen content, EdF plans to add hydrazine directly in the low-pressure side of the turbine. With this procedure, already tested in thermal plants, the efficiency of consumption of oxygen is much higher.

The level of contaminants in the steam generator water is controlled at the steam generator blowdown by two reliable, automatic, continuous measurements: cation conductivity and sodium content. These two parameters allow for excellent control of the steam generator water quality and offer a simple method to determine the nature (acid-forming or alkaline-forming) of the pollutants. The alternative is the free hydroxide method, which is slow, manual, and unreliable.

The steam-generator-blowdown chemistry specifications are insufficient to guarantee the integrity of the steam generator because of the phenomenon of hide-out of the contaminants. This hide-out in the deposits (in the sludge pile and in the gaps between the tube and the tube-support plate) leads to corrosion of the tubes under the sludge pile or to denting. So, all the impurities entering the steam generator are not eliminated by the blowdown. It is therefore very important to limit the amount of contaminants entering the steam generators. This has led to the introduction of another parameter in the chemistry specifications: The Framatome specification requires that

the annual integrated pollution amount (expressed in day per microsiemen per centimetre or $d/\mu S\text{ cm}^{-1}$) be maintained below $500\text{ d}/\mu S\text{ cm}^{-1}$ per year. EdF has specified more stringent values on this parameter: For sea-water plants, I.P. $< 300\text{ d}/\mu S\text{ cm}^{-1}$; for river plants, I.P. $> 400\text{ d}/\mu S\text{ cm}^{-1}$. To satisfy these requirements, the daily cation conductivity should be maintained well below the allowable limit of $2\text{ }\mu S\text{ cm}^{-1}$. With this requirement, the contaminant ingress in the steam generator is maintained at a low level.

5.3 JAPANESE PWR STEAM GENERATOR EXPERIENCE

The Central Electric Power Council of Japan announced its fiscal 1980 edition of a long-term electric power plan under a severe energy situation in which the petroleum supply is restricted and its price increases (The Institute of Energy Economics, 1980). Under this plan, nuclear power generation will be expanded in the next 10 years to 3 times that existing now to reduce the use of petroleum for electric power generation. The goal is to reduce the rate of dependence on petroleum from 43% of the current figure to 26% in ten years. Nuclear power generation amounting to about $2.57 \times 10^7\text{ kW}$ will be realized during the next 10 years. As a result, nuclear power generation installations in fiscal 1989 will be about $4.07 \times 10^7\text{ kW}$ and the percentage of all power generation installations will be increased to 19% from 13% at the present time.

There are 23 nuclear power plants operational in Japan. Among them, 10 are PWR plants (see Appendix A). It is indicated by Thomas and Surrey (1981) that all Japanese nuclear power plants have suffered prolonged outages, many exceeding 6 months. Japan's PWRs had persistent steam-generating problems. As a result of this, the performance of Japan's PWRs is worse than that of either the United States or the German PWRs.

6. IMPROVED STEAM GENERATOR DESIGNS

6.1 MATERIAL ALTERNATIVES

From the discussions in Chapters 3 and 4, it is evident that a highly corrosive and variable environment is possible in a PWR steam generator, and, consequently, radical changes in both design and operation will be essential to assure that a steam generator lasts for 40 years. Material optimization certainly plays an important role in the long-range plans. Ideally, tubing used in the future will resist either stress-corrosion cracking or corrosion-fatigue cracking, while retaining high overall resistance to local attack or wastage. For tube support plates and tube sheets, the primary need is for a material that is protected against corrosion. Several materials and material processes are being evaluated for replacement tubes (nickel-base alloys such as Inconel 600, Inconel 691, and Alloy 800) and support plates (ferritic stainless steels such as types 405 and 409); these evaluations are conducted with standard laboratory stress-corrosion and general-corrosion tests and long-term model boiler tests under aggressive fault-chemistry conditions (Green, 1980). The following subsections (Roberts, 1981) summarize the progress being made on the most resistant steam generator materials.

6.1.1 Inconel 600

When used as mill-annealed tubing, Inconel 600 is susceptible to intergranular SCC. Therefore, research metallurgists have attempted to improve the SCC resistance by modifying the microstructure of the grain boundary. Laboratory-scale experiments have been successful and long-term pot boiler tests are underway (Green, 1980). Two heat treatment processes were used (viz. a high-temperature "purification" annealing and a thermal treatment); the thermally treated tubing was superior to the mill-annealed product in all cases in resisting caustic SCC.

Thermal treatment both increases the critical stress level above which rapid crack propagation occurs and decreases the crack propagation rate. Airey (1979) therefore attributes the improved SCC resistance to a reduction in residual stress (retarding crack initiation) and modification of grain boundary structure (inhibiting crack growth). The results of this program also indicated that one benefit of thermal treatment is to reduce the variability in the caustic SCC resistance. As implied earlier, there is a large scatter in the data on the caustic SCC resistance of mill-annealed Inconel 600--presumably because materials with different thermomechanical processing histories were tested at different heats. (Was, 1978). If the role of thermal treatment is to reduce this scatter, it

could also explain some of the conflicts in the data; in some cases thermal treatment improves the caustic SCC resistance and in others it does not.

6.1.2 Alloy 800

Although Alloy 800 is still being evaluated in the United States, it has been used for several years for steam generator tubes in German-designed PWRs. Alloy 800 has also been selected as steam-generator-tubing material for those reactors such as the high-temperature gas-cooled reactor (HTGR) and the liquid-metal fast breeder reactor (LMFBR).

Alloy 800 is not completely immune from cracking in water that contains chloride ions or oxygen, or in acid or alkaline environments. Laboratory investigations of cracking of Alloy 800 showed that Alloy 800 has an apparently higher stress threshold for crack initiation in caustic environments at low concentrations than does Inconel 600, but that it fails rapidly at low stresses in highly caustic conditions (Blanchet, 1978).

Although susceptible to cracking in certain chloride environments containing oxygen, Alloy 800 proves less susceptible than either the austenitic Types 304 and 316 or Inconel 600. Results of a number of studies have shown: (a) that a chloride concentration threshold exists at 473°K and between 0.5 and 2 ppm: at lower temperatures, stress cracking in Alloy 800 either does not occur or is strongly delayed; and (b) that cold work, sensitization, and low Ti/C + N ratios reduce the cracking resistance of Alloy 800 (Blanchet, 1978).

PWRs with Alloy 800 steam generator tubes have been operating in Germany for several years now. Careful examination of the tubes failed to reveal any signs of cracking or pitting, but some general corrosion was observed, which resulted in metal loss to a thickness of 600 μm (Pugh, 1975). On the basis of this good experience, KWU has decided to continue to use Alloy 800, and the alloy is being evaluated in model boiler tests in the United States.

6.1.3 Inconel 690

Inconel 690, a relatively new high-chromium (30 wt%) nickel-base alloy, is designed to provide a stable austenitic structure with thermal and mechanical properties close to those of Inconel 600 combined with a resistance to corrosion and oxidation in aggressive environments that makes it comparable to Alloy 800.

The corrosion resistance of Inconel 690 has been studied in a wide variety of media of interest to PWR operators (Pement, 1979). Results of caustic-stress-corrosion tests

indicated that Inconel 690 is the most resistant of the alloys tested (Type 304 stainless steel, Alloy 800, Inconel 690, and Inconel 600) in oxygenated 50% sodium hydroxide and deoxygenated 10% sodium hydroxide. The cracking resistance of Inconel 690 was also highest in tests in hot water containing oxygen, and this property was not impaired by various heat treatments, cold work, or the presence of welds. Besides stress-corrosion-cracking resistance, this alloy exhibits a high general corrosion resistance (i.e., low weight loss) in oxidizing acids and in flowing hot-water environments.

6.1.4 SCR-3 Alloy

A new Ni-Cr-Fe alloy, SCR-3, has recently been developed in Japan for use as PWR steam generator tubing. This alloy contains almost equal amounts of nickel (25 to 26 wt%) and chromium (24 wt%) and small amounts of vanadium and titanium; the carbon level is approximately 0.02 wt% and silicon levels range from 1.7 to 2 wt%. Thus, SCR-3 is lower in chromium than Inconel 690, but is higher than either Inconel 600 (15.6 wt%) or Alloy 800 (21 wt%).

The SCC susceptibility of this new alloy has been compared with Type 304 stainless steel, Inconel 600, and Alloy 800 in the "standard" media (i.e., chlorinated water, caustic solutions, phosphate solutions, etc.), and in all instances it proved to be the most resistant alloy (Kowaka, 1979).

6.1.5 Ferritic Stainless Steels

Although improved SCC-resistant tube materials are available, the situation is less promising for support-plate materials. Ferritic materials are expected to be resistant to SCC but prone to severe general attack and pitting in steam or water environments. Their resistance to general corrosion should, however, improve as the chromium content is increased. Nevertheless, model boiler testing of the 400 series ferritic steels with seawater chemistry has resulted in tube locking within the drilled holes (Green, 1980). This could be a result of material deposition in the crevice or corrosion buildup from the support plate itself. However, these preliminary results cast doubt on the adequacy of these steels as a replacement for carbon steel.

6.2 MECHANICAL DESIGN ALTERNATIVES

6.2.1 Design for Tube Support Plates and Tube Sheets

Because of the denting problems occurring in recirculating steam generator PWR plants, the design for tube-support plates and tube sheets becomes an important issue. The industry trend is toward a more "open" support structure, as shown in Fig. 5. Although open structure will minimize the likelihood of tube touching, it will increase the risk of vibration fatigue and fretting problems. A redesign of anti-vibration bars at the U-bends should help solve this, however. Because vibration-related problems have been observed in OTSGs, the new designs eliminate the vacant tube lanes and, therefore, stiffen the support structure to minimize damage from fatigue and wear.

6.2.2 Fabrication Process

For fabrication, the major concern is residual stresses. Peak tensile stresses of more than 150 MPa have been measured in the U-bends; stress may exceed 200 MPa on both inner and outer surfaces of the bending radii (Berge, 1976). These stresses are very high and are likely to produce stress-corrosion cracking under undesirable chemical conditions. Therefore, a design that reduces residual stresses that arise during the fabrication process is a necessity.

A radical approach to this problem for OTSGs in the United States is to heat-treat the entire steam generator after assembly. This approach could create the following problems: (1) the tube bundle in the recirculating steam generator design may be distorted, (2) the temperature limitation (870°K) may prevent full reduction of tube residual stresses, and (3) tube materials may be sensitized if certain temperature regimes persist long enough.

A more conventional approach to the reduction of residual stresses is to reduce them during tube fabrication. These (Berge, 1978) include (1) avoiding tube distortion during fabrication, (2) mechanically relieving stress of the straight tubes by applying tension and internal pressure, (3) introducing residual compressive stresses at the outer surface, by shot peening for example, and (4) heat treating at 970°K or above (the final stress level after 16 hour treatment at this temperature is about 35 MPa for Inconel 600 tubing).

6.2.3 Thermal and Hydraulic Simulation Analysis

Computer codes currently being developed by EPRI (URSULA 2 and THEDA 2) can be used to simulate steam generator performance. Tests are planned in model and full-size steam generators to experimentally verify the computer codes. Bouecke (1980) described how a computer code was used to optimize certain features of the steam generator and also described analytical and experimental studies of flow distribution within a steam generator. As known, flow distributions in the steam generator can greatly influence the amount of sludge deposited.

6.2.4 Access for Inspection, Repair, and Replacement

Regardless of how advanced the design techniques and operating procedures are, a certain amount of damage to the steam generator cannot be avoided. To better define the problems, one of the design concerns should be to consider the access from which steam-generator damage can be inspected and repaired. This design concern can relieve the dependence on nondestructive examination and also can accelerate the repair process. Research also should be focused on the design option that views the steam generator as replaceable after 15 or 20 years if severe degradation of the steam generator is inevitable before the end of 40-year plant life.

7. SUMMARY AND CONCLUSIONS

The nuclear industry in the United States and abroad has had persistent problems with degradation of the steam generators used in PWR nuclear power plants. These problems are caused by either erosion/corrosion effects or mechanical damages (vibration, fretting, impingement, and water hammer). The results are tube denting, cracking of support plates, fatigue- and stress- corrosion cracking of tube wall, tube wastage (wall thinning), tube wear, pipe deformation, and pipe support failure. Available information indicated that the extent of these problems is significant. It has become a major concern of the nuclear industry whether or not the steam generators used in operational PWR nuclear power plants can last to the end of a 40-year design life. As a matter of fact, there are utilities that have already decided to replace the steam generators used in a few units of their plants.

Problems in the recirculating steam generators used by most Westinghouse and Combustion Engineering PWR units are primarily caused by corrosion of the carbon steel support plates, tube sheets, and Inconel 600 tubes, although there have been some fretting and water-hammer problems. The principal damages are denting of the tubes in the tube support plates and tube sheets, as well as thinning and cracking of the tubes on top of and within the tube sheets.

The German-built KWU units, with the exception of the Obrigheim reactor, have not experienced stress-corrosion cracking, denting, or fretting, but progressive tube thinning was detected in the center of the hot-leg tube bundles near the tube sheet; this thinning is a result of wastage corrosion of the tube material.

Problems in once-through steam generators used by PWR plants are primarily caused by mechanical factors. The resulting damages are cracking and fretting wear of the Inconel 600 tubes.

In recent years the nuclear industry in the United States and abroad has initiated significant efforts to solve the problems related to PWR steam generators. One of the major efforts was undertaken by EPRI for a steam generator owners group (see Appendix B). This group consists of 20 U. S. utilities joined by organizations representing Great Britain, Japan, France, and Sweden. Significant progress has been made toward solving the problems.

Field experience and laboratory work over the past few years have provided a greatly improved understanding of the known problems with steam generator tubing. The increased insight has led to at least partial solutions to these problems. Several utilities have proceeded with corrective actions based on the growing understanding of the causes

of the steam generator problems. Initial observations by some plants that have taken these steps are quite promising. One of the important results is that the progression of denting has been significantly slowed if not completely eliminated. This is indicated by a worldwide survey (Mundis and Noble, 1981) that indicated that the rate of tubes plugged in PWR steam generators was reduced drastically in 1978 and 1979 because wastage was virtually eliminated and tube denting was reduced significantly.

These improvements, though reassuring, are not sufficient and much more work needs to be done. According to the same worldwide survey, the increase in tube plugging due to "other" causes in 1979 is just one indicator that new problems continue to be discovered. Clearly, there is a need to continue to develop a technical basis for each utility's choices of alternatives for prolonging steam generator life. Specifically, there is a need to obtain quantitative information, from both the laboratory and operating plants, regarding the required purity of the feedwater current and improved chemistry control. There is also a need to determine the effectiveness of balance-of-plant actions in improving the feedwater purity. Finally, for the newer steam generator problems (such as tube cracking and intergranular attack in the open crevices of the tube sheet and tube cracking in the first row U-bends and other regions of high stress), there is a significant need for evaluation.

As a final remark, we feel that, while the extent of reliability problems associated with current PWR steam generators should be properly recognized, the seriousness of the problems need not to be over-emphasized. Commercial nuclear energy development is one of the most advanced and complex technologies that man has ever undertaken. Most of the problems related to nuclear power plant steam generators are not totally surprising to us. Human beings are accustomed to learning and progressing from experience.

The nuclear industry has launched a great effort toward solving the steam generator problems. The preliminary results are promising. The next few years represent a crucial period in which it has to find satisfactory means to improve the reliability of the nuclear power plant steam generators so they can last until the end of their design lives.

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APPENDIX A

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial Operation orig. sched- ule+	actual or ex- pected
BELGIUM								
Indivision Doel								
• Doel 1 (Antwerp)	390	ACECOWEN	COP/TOSI/ ACEC	TE		100	6/73	2/75
• Doel 2 (Antwerp)	390	ACECOWEN	COP/TOSI/ ACEC	TE		100	3/75	11/75
Societe Belgo-Francaise d'Energie Nucleaire Mosane (SEMO)								
• Tihange 1 (Huy, Liege)	870	ACLF	Alsthom/Rateau/EdF/ La Meuse/ ACEC-JS	Electrobel	CFE-Blaton/ADF SPIE-SNTP/C-8	100	12/74	9/75
Societe Intercommunale Belge de Gaz et d'Electricite (INTERCOM)								
Tihange 2 (Huy, Liege)	900	FRAMACECO	Alsthom/Rateau La Meuse/ACEC	Electrobel	CTAFMC	90	4/80	9/82
Tihange 3 (Huy, Liege)	1000	ACECOWEN	Brown Boveri/ Alsthom/ACEC	Electrobel	CTAFMC	30	4/82	9/84
Societes Reunies d'Energie du Bassin de l'Escaut (EBES)								
Doel 3 (Antwerp)	900	FRAMACECO	Alsthom/ Atlantique/ACEC	TE	TE	90	2/80	mid 82
Doel 4 (Antwerp)	1000	ACECOWEN	Alsthom/ Atlantique/ACEC	TE	TE	40	12/82	mid 84
BRAZIL								
Furnas								
Angra 1 (Itaorna)	626	W	W	G&H/PE	W	97	3/77	12/81
Angra 2 (Itaorna)	1245	KWU	KWU	Nuclen	KWU	10	12/82	6/87
Angra 3 (Itaorna)	1245	KWU	KWU	Nuclen	KWU	1	6/84	12/88
BULGARIA								
• Kozloduy 1 (Kozloduy)	440	AEE				100		12/74
• Kozloduy 2 (Kozloduy)	440	AEE				100		12/75
• Kozloduy 3 (Kozloduy)	440	AEE				100	/78	12/80
• Kozloduy 4 (Kozloduy)	440	AEE				80	/79	/82

* Information taken from Nuclear News, American Nuclear Society, August, 1981, Vol. 10, pp. 85-104.

+ Estimated date of startup, announced at time reactor was ordered.

• Units in commercial operation.

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
 Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial orig. sched- ule+	actual or ex- pected
CZECHOSLOVAKIA								
• Bohunice 2A (Jaslovske Bohunice)	440	AEE				100		12/78
• Bohunice 2B (Jaslovske Bohunice)	440	AEE				100	/79	3/80
• Bohunice 3 (Jaslovske Bohunice)	440	AEE					/82	
• Bohunice 4 (Jaslovske Bohunice)	440	AEE					/83	
• Dukovany 1 (Dukovany)	440	AEE					/82	/83
• Dukovany 2 (Dukovany)	440	AEE					/83	/84
• Dukovany 3 (Dukovany)	440	AEE					/83	/84
• Dukovany 4 (Dukovany)	440	AEE					/84	/85
• Leviece 1	440	AEE						/87
• Leviece 2	440	AEE						/88
EGYPT								
Egyptian Electricity Authority								
• Sidi-Krier-1 (Sidi Krier)	622	W	W	Gilbert	Jones	0	/83	/88
FINLAND								
Imatran Voima Osakeyhtio (IVO)								
• Loviisa 1 (Loviisa)	420	AEE	AEE	IVO	IVO	100	6/76	5/77
• Loviisa 2 (Loviisa)	420	AEE	AEE	IVO	IVO	98	4/78	12/80
FRANCE								
Electricite de France (EdF)								
• Fessenheim 1 (Haut-Rhin)	890	Fra/CL	Alsthom	EdF	C-B	100	10/75	12/77
• Fessenheim 2 (Haut-Rhin)	890	Fra/CL	Alsthom	EdF	C-B	100	7/76	3/78
• Bugey 2 (Ain)	920	Fra/CL	Alsthom	EdF	Bouygues	100	12/76	2/79
• Bugey 3 (Ain)	920	Fra/CL	Alsthom	EdF	Bouygues	100	8/77	2/79
• Bugey 4 (Ain)	900	Fra/CL	Alsthom	EdF	Bouygues	100	5/78	7/79
• Bugey 5 (Ain)	900	Fra/CL	Alsthom	EdF	Bouygues	100	11/78	12/79
• Dampierre 1 (Loiret)	900	Fra/CL	Alsthom	EdF	CM/SeB/ Ballot	100	7/79	10/80
• Dampierre 2 (Loiret)	900	Fra/CL	Alsthom	EdF	CM/SeB/ Ballot	100	1/80	2/81
• Dampierre 3 (Loiret)	900	Fra/CL	Alsthom	EdF	CM/SeB/ Ballot	100	7/80	6/81
• Dampierre 4 (Loiret)	900	Fra/CL	Alsthom	EdF	CM/SeB/ Ballot	100	4/81	11/81
• Gravelines B1 (Dunkerque)	920	Fra/CL	Alsthom	EdF	SGE	100	4/79	10/80
• Gravelines B2 (Dunkerque)	920	Fra/CL	Alsthom	EdF	SGE	100	10/79	12/80
• Gravelines B3 (Dunkerque)	920	Fra/CL	Alsthom	EdF	SGE	100	5/80	5/81
• Gravelines B4 (Dunkerque)	920	Fra/CL	Alsthom	EdF	SGE	100	2/81	10/81
• Gravelines C5 (Dunkerque)	920	Fra/CL	Alsthom	EdF	SGE	30	12/84	12/84
• Gravelines C6 (Dunkerque)	920	Fra/CL	Alsthom	EdF	SGE	17	8/85	8/85
• Tricastin 1 (Drome)	920	Fra/CL	Alsthom	EdF	C-B-C	100	2/79	12/80
• Tricastin 2 (Drome)	920	Fra/CL	Alsthom	EdF	C-B-C	100	8/79	12/80
• Tricastin 3 (Drome)	920	Fra/CL	Alsthom	EdF	C-B-C	100	3/80	5/81
• Tricastin 4 (Drome)	920	Fra/CL	Alsthom	EdF	C-B-C	100	9/80	10/81

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
 Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial orig. sched- ule+	actual or ex- pected
● Saint-Laurent-des-Eaux B1 (Loir-et-Cher)	880	Fra/CL	CEM	EdF	GTM	100	1/81	6/81
Saint-Laurent-des-Eaux B2 (Loir-et-Cher)	880	Fra/CL	CEM	EdF	GTM	100	6/81	11/81
Blayais 1 (Gironde)	920	Fra/CL	Alsthom	EdF	SB/Dumez	100	2/81	10/81
Blayais 2 (Gironde)	920	Fra/CL	Alsthom	EdF	SB/Dumez	83	9/81	8/82
Blayais 3 (Gironde)	920	Fra/CL	Alsthom	EdF	SB/Dumez	70	9/82	2/83
Blayais 4 (Gironde)	920	Fra/CL	Alsthom	EdF	SB/Dumez	60	2/83	7/83
Chinon B1 (Indre-et-Loire)	875	Fra/CL	CEM	EdF	GTM	85	2/82	6/82
Chinon B2 (Indre-et-Loire)	875	Fra/CL	CEM	EdF	GTM	70	6/82	12/82
Chinon B3 (Indre-et-Loire)	875	Fra/CL	CEM	EdF	GTM	5	5/86	5/86
Chinon B4 (Indre-et-Loire)	875	Fra/CL	CEM	EdF	GTM	5	1/87	1/87
Paluel 1 (Seine-Maritime)	1285	Fra/CL	Alsthom	EdF	CM/Baillet/ Chag	50	2/83	8/83
Paluel 2 (Seine-Maritime)	1285	Fra/CL	Alsthom	EdF	CM/Baillet/ Chag	40	5/83	2/84
Paluel 3 (Seine-Maritime)	1285	Fra/CL	Alsthom	EdF	CM/Baillet/ Chag	30	6/84	10/84
Paluel 4 (Seine-Maritime)	1285	Fra/CL	Alsthom	EdF	CM/Baillet/ Chag	15	9/85	9/85
Cruas 1 (Ardeche)	880	Fra/CL	CEM	EdF	C-B-C	60	7/83	7/83
Cruas 2 (Ardeche)	880	Fra/CL	CEM	EdF	C-B-C	50	12/83	11/83
Cruas 3 (Ardeche)	880	Fra/CL	CEM	EdF	C-B-C	40	6/84	5/84
Cruas 4 (Ardeche)	880	Fra/CL	CEM	EdF	C-B-C	30	12/84	11/84
Saint-Maurice Saint-Alban 1 (Isere)	1285	Fra/CL	Alsthom	EdF	Bouygues	30	11/84	11/84
Saint-Maurice Saint-Alban 2 (Isere)	1285	Fra/CL	Alsthom	EdF	Bouygues	15	7/85	10/85
Flamanville 1 (Manche)	1285	Fra/CL	Alsthom	EdF	SGE/SCREG	20	2/85	1/85
Flamanville 2 (Manche)	1285	Fra/CL	Alsthom	EdF	SGE/SCREG	10	12/85	12/85
Cattenom 1 (Moselle)	1270	Fra/CL	Alsthom	EdF	SB/Dumez	17	7/85	8/85
Cattenom 2 (Moselle)	1270	Fra/CL	Alsthom	EdF	SB/Dumez	5	3/86	5/86
Belleville 1 (Cher)	1270	Fra/CL	Alsthom	EdF	GTM	5	10/80	10/86
Belleville 2 (Cher)	1270	Fra/CL	Alsthom	EdF	GTM	0	8/87	8/87
Nogent s/Seine 1 (Aube)	1270	Fra/CL	Alsthom	EdF	Dumez	0	5/87	5/87
Nogent s/Seine 2 (Aube)	1270	Fra/CL	Alsthom	EdF	Dumez	0	3/88	3/88
Societe d'Energie Nucleaire Franco-Belge des Ardennes (SENA)								
● SENA (Chooz)	310	ACECOMEN/ Fra	Rateau/C-L	G&H/SPIE	SGE/CITRA	100		4/67
GERMANY (DEMOCRATIC REPUBLIC)								
● Rheinsberg 1 (Rheinsberg, Granesse region)	70	AEE				100	/60	5/66
● Nord 1 - 1 (Lubmin, Greifswald region)	440	AEE				100	12/74	12/73
● Nord 1 - 2 (Lubmin, Greifswald region)	440	AEE				100	/75	2/75

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial orig. sched- ule+	actual or ex- pected
• Nord 2 - 1	440	AEE				100	/77	6/78
• Nord 2 - 2	440	AEE				100	/78	/80
Magdeburg 1	440	AEE				1	/80	
Magdeburg 2	440	AEE				1	/80	
GERMANY (FEDERAL REPUBLIC)								
Bayernwerk AB								
Grafenrheinfeld KKG (Grafenrheinfeld)	1225	KWU	KWU	KWU	KWU	98	11/78	12/81
Gemeinschaftskernkraftwerk Neckar (GKN)								
• GKN 1 (Neckarwestheim)	805	KWU	KWU	KWU	KWU	100	2/76	10/76
GKN 2 (Neckarwestheim)	805	KWU	KWU	KWU	KWU	0	/81	/86
Kernkraftwerk Lippe-Emsland GmbH Emsland (Ligen)	1222		KWU	KWU	KWU	KWU	0	late '80s
Kernkraftwerk Brokdorf GmbH Brokdorf (Brokdorf)	1290	KWU	KWU	KWU	KWU	0	9/77	/86
Kernkraftwerk Hamm BmbH (KKH) Hamm (Hamm-Uentrop)	1300	KWU	KWU	KWU	KWU	0	/81	late 80s
Kernkraftwerk Isar (KKI) Isar II	1227	KWU	KWU	KWU	KWU	0		1/88
Kernkraftwerk Obrigheim GmbH (KWO) • Obrigheim KWO (Obrigheim)	328	Siemens	Siemens	Siemens	Siemens	100	3/69	3/69
Kernkraftwerk Philippsburg (KKP) KKP 2 (Philippsburg)	1281	KWU	KWU	KWU	KWU	45	/77	3/85
Kernkraftwerk Stade GmbH (KKS) • Stade KKS (Stade)	630	Siemens	Siemens	Siemens	Siemens	100	4/72	5/72
Kernkraftwerk Sud GmbH (KWS) KWS (Wyhl/Rhein)	1284	KWU	KWU	KWU	KWU	0	/79	indef.
Kernkraftwerk Unterwesser GmbH (KKU) • KKU (Esensham)	1230	KWU	KWU	KWU	Arge/KKU	100	1/76	10/79
Gemeinschaftskernkraftwerk Grohnde GmbH (KWG) KWG (Grohnde)	1294	KWU	KWU	KWU	Arge/KWG	45	8/79	/84
Rheinisch-Westfalisches Elektrizitätswerk AG RWE								
• Biblis A (Worms/Rhein)	1146	Siemens	KWU	KWU	Hochtief	100	6/74	3/75
• Biblis B (Worms/Rhein)	1240	Siemens	KWU	KWU	Hochtief	100	8/76	1/77
Biblis C (Worms/Rhein)	1232	KWU	KWU	KWU	Hochtief	0	/81	indef.
Kaerlich	1227	BBR	BBC	BBC	Hochtief	45	5/78	/85
Neupotz 1 (Neupotz)	1246	BBR	BBC	BBC	Hochtief	0	/83	indef.
HUNGARY								
Hungarian Electrical Works								
Paks 1 (Paks)	440	AEE		ERBE		84	/80	/82
Paks 2 (Paks)	440	AEE		ERBE		50	/80	/83
Paks 3 (Paks)	440	AEE		ERBE		5		/84
Paks 4 (Paks)	440	AEE		ERBE		5		/85

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial orig. sched- ule+	actual or ex- pected
IRAQ								
Iraq 1	900	Fra				0		
ITALY								
Ente Nazionale per l'Energia Elettrica (ENEL)								
• Trino Vercellese (Vercelli)	247	W	Franco Tosi Marelli	G&H	Recchi	100		1/65
ENEL 5 (site not yet approved)	952	El/W	Franco Tosi/ Marelli	Bechtel		0	/80	indef.
JAPAN								
Kansai Electric Power Co., Inc.								
• Mihama 1 (Mihama-cho)	320	W	MHI/MEL	KEPCO/ Gilbert	Maeda/ Kum/Obay	100		11/70
• Mihama 2 (Mihama-cho)	470	MHI	MHI/MEL	KEPCO/MAPI	Maeda/ Kum/Obay	100		7/72
• Takahama 1 (Takahama-cho)	780	W	MHI/MEL	KEPCO/ Gilbert	Maeda/ Haz/Taisei	100	8/74	11/74
• Takahama 2 (Takahama-cho)	780	MHI	MHI/MEL	KEPCO/MAPI	Maeda/ Haz/Taisei	100	7/75	11/75
Takahama 3 (Takahama-cho)	830	MHI	MHI/MEL	KEPCO/MAPI	Maeda, Haz, Kum/ Takenaka, Obay, Taisei, Haz	8	8/80	2/85
Takahama 4 (Takahama-cho)	830	MHI	MHI/MEL	KEPCO/MAPI	Maeda, Haz, Kum/ Takenaka, Obay, Taisei, Haz	2	8/80	8/85
• Mihama 3 (Mihama-cho)	780	MHI	MHI/MEL	KEPCO/MAPI	Haz/Taisei	100	7/76	12/76
• Ohi 1 (Ohi-cho)	1120	W	MHI/MEL	KEPCO/ Gilbert	Kum/Obay	100	7/76	12/76
• Ohi 2 (Ohi-cho)	1120	W	MHI/MEL	KEPCO/ Gilbert	Kum/Obay	100	1/77	12/79
Kyushu Electric Power Co., Inc.								
• Genkai 1 (Genkai, Saga)	559	MHI	MHI/MEL	MAPI	Obay	100	7/75	10/75
• Genkai 2 (Genkai, Saga)	559	MHI	MHI/MEL	MAPI	Obay	100	3/81	3/81
Sendai 1 (Sendai, Kagoshima)	890	MHI	MHI/MEL	MAPI	Obay	50.4	7/84	7/84
Sendai 2 (Sendai, Kagoshima)	890	MHI	MHI/MEL	MAPI	Obay	0.5	3/86	3/86
Shikoku Electric Power Co.								
• Ikata 1 (Ikata-cho, Ehime Pref.)	538	MHI	MHI/MEL	MAPI	MHI/Taisei	100	4/77	9/77
Ikata 2 (Ikata-cho, Ehime Pref.)	538	MHI	MHI/MEL	MAPI	MHI/Taisei	96	10/81	3/82

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial Operation orig. sched- ule+	actual or ex- pected
KOREA								
Korea Electric Co.								
• Ko-Ri 1 (Ko-Ri, near Pusan City)	564	W	GEC	Gilbert	W	100	12/75	4/78
Ko-Ri 2 (Ko-Ri, near Pusan City)	605	W	GEC	Gilbert	W	64.5	11/79	2/83
Korea Nuclear 5 (Ko-Ri, near Pusan City)	900	W	GEC	Bechtel	Utility	48.4	9/84	9/84
Korea Nuclear 6 (Ko-Ri, near Pusan City)	900	W	GEC	Bechtel	Utility	48.4	9/85	9/85
Korea Nuclear 7 (Ko-Ri, Young Kwang-Kun)	950	W	W	Bechtel	Utility	11.3	3/86	3/86
Korea Nuclear 8 (Ko-Ri, near Kwang-Kun)	950	W	W	Bechtel	Utility	11.3	3/87	3/87
Korea Nuclear 9 (Ul Jin-Kun, near Yeongju City)	950	Fra	undecided	Fra	Utility	3.4	12/87	3/88
Korea Nuclear 10 (Ul Jin-Kun, near Yeongju City)	950	Fra	undecided	Fra	Utility	3.4	12/88	3/89
LIBYA								
Libya 1	300	AEE				0		Indef.
LUXEMBOURG								
Luxembourg Nuclear Power Company (SENU) Remerschen (Remerschen)	1250	BBR	BBR	BBC	BBC	0	1/82	Indef.
NETHERLANDS								
NV Provinciale Zeeuwse Energie Maatschappij (PZEM) • Borssele (Borssele)	445	KWU/RDM	KWU/Stork	KWU	KWU/Bredero	100	7/73	10/73
PHILIPPINES								
Philippine National Power Corp. PNPP 1 (Morong, Bataan Luzon)	620	W	W	B&R	W	21	12/82	1/85
POLAND								
Zarnowiec 1 (Zarnowiec)	440	AEE				0	/85	/85
Zarnowiec 2 (Zarnowiec)	440	AEE				0	/86	/86
RUMANIA								
Rumania-1 (Olt)	440	AEE					/80	/83
SOUTH AFRICA								
Electricity Supply Commission (ESCOM) Koeberg 1 (Koeberg)	922	Fra	Alsthom		SB	70	12/82	1/83
Koeberg 2 (Koeberg)	922	Fra	Alsthom		SB	50	12/83	1/84

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial orig. sched- ule+	actual or ex- pected
SPAIN								
Fuerzas Electricas de Cataluna, S.A. Asco 1 (Asco, Tarragona)	880 W		W/ENB	Bechtel/ Initec/lyP	NUCEA	96	7/77	5/82
Fuerzas Electricas de Cataluna, S.A., Empresa Nacional Hidroelectrica Ribagorzana, Hidroelectrica de Cataluna, S.A., and Fuerzas Hidroelectrica del Segre, S.A. Asco 2 (Asco, Tarragona)	880 W		W/ENB	Bechtel/ Initec/lyP	NUCEA	71	12/77	3/84
Compania Sevillana de Electricidad SA, Hidroelectrica Espanola SA, and Union Electrica SA								
Almaraz 1 (Almaraz, Caceres)	930 W		W	Ea/G&HE	EyT	100	11/76	12/81
Almaraz 2 (Almaraz, Caceres)	930 W		W	EA/G&HE	EyT	85	12/77	/83
Electricas Reunidas de Zaragoza SA, Energia e Industrias Aragonesas SA, Union Electrica SA								
Trillo 1 (Trillo, Guad.)	997 KWU		KWU/ENSA/ E.N.Bazan	EA	ETOCEA	2	6/82	1/86
Trillo 2 (Trillo, Guad.)	1200 KWU					0		Indef.
Empresa Nacional Hidroelectrica del Ribagorzana SA, and three other utilities								
Vardellos 2 (Tarragona)	982 W		W	Initec/ Bechtel	VANEA	5	12/81	/86
FENOSA (Fuerzas Electricas del Noroeste, S.A.)								
Regodola (Lugo)	1000 KWU		KWU					
Iberduero SA								
Lemoniz 1 (Lemoniz, Vizcaya)	900 W		W	Iber/Bech/ Sen/Initec	Iberduero	100	12/76	5/82
Lemoniz 2 (Lemoniz, Vizcaya)	900 W		W	Iber/Bech/ Sen/Initec	Iberduero	55	7/78	12/84
Sayago (Sayago, Zamora)	1075 W		W	Iber/Bech/ Sen/Initec	Iberduero	0	6/81	7/89
Union Electrica (UE)								
• Jose Cabrera (near Madrid)	153 W		W	G&H	EyT	100	7/68	8/69
SWEDEN								
Statens Vattenfallsverk (SSPB)								
• Ringhals 2 (Varberg)	800 W		Stal-Laval	SSPB/G&H/S-L	SSPB	100	7/74	5/75
• Ringhals 3 (Varberg)	900 W		Stal-Laval	VBB-TS/S-L	SSPB	100	12/77	4/81
Ringhals 4 (Varberg)	900 W		Stal-Laval	VBB-TE/S-L	SSPB	100	7/79	11/82

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
 Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial Operation orig. sched- ule+	actual or ex- pected
SWITZERLAND								
Nordostschweizerische Kraftwerke AG (NOK)								
• Beznau 1 (Doettingen)	350	W	BBC	G&H/BBC	Zschokke	100	12/69	12/69
• Beznau 2 (Doettingen)	350	W	BBC	G&H/BCC	Zschokke	100	3/72	3/72
Kernkraftwerk Gosgen-Daniken AG								
• Gosgen (Daniken, SO)	920	KWU	KWU	KWU	KWU	100	11/77	11/79
TAIWAN								
Taiwan Power Co.								
Maanshan 1 (Maanshan)	907	W	GE	Bechtel	TPC	36	4/81	4/84
Maanshan 2 (Maanshan)	907	W	GE	Bechtel	TPC	19	4/82	4/85
TURKEY								
Akkuyu (Akkuyu)	440	AEE				0		/87
UNITED STATES								
NORTHEAST								
Baltimore Gas & Electric Co.								
• Calvert Cliffs 1 (Lusby, Md.)	850	C-E	GE	Bechtel	Bechtel	100	1/73	5/75
• Calvert Cliffs 2 (Lusby, Md.)	850	C-E	W	Bechtel	Bechtel	100	1/74	4/77
Boston Edison Co.								
• Pilgrim 2 (Plymouth, Mass.)	1150	C-E	GE	Bechtel	Bechtel	0	8/80	indef.
Connecticut Yankee Atomic Power Co.								
• Haddam Neck (Haddam Neck, Conn.)	575	W	W	S&W	S&W	100	11/67	1/68
Consolidated Edison Co.								
• Indian Point 2 (Indian Point, N.Y.)	873	W	W	UE&C	Wedco	100	6/69	7/74
Duquesne Light Co.								
• Beaver Valley 1 (Shippingport, Pa.)	833	W	W	S&W	S&W	100	6/73	4/77
• Beaver Valley 2 (Shippingport, Pa.)	833	W	W	S&W	DLC/S&W	41.6	10/78	5/86
Maine Yankee Atomic Power Co.								
• Maine Yankee (Wiscasset, Me.)	825	C-E	W	S&W	S&W	100		12/72
Metropolitan Edison Co.								
• Three Mile Island 1 (Londonderry Twp., Pa.)	792	B&W	GE	Gilbert	UE&C	100	9/71	9/74
• Three Mile Island 2 (Londonderry Twp., Pa.)	880	B&W	W	B&R	UE&C	100	5/73	12/78
Northeast Utilities								
• Millstone 2 (Waterford, Conn.)	870	C-E	GE	Bechtel	Bechtel	100	4/74	12/75
• Millstone 3 (Waterford, Conn.)	1150	W	GE	S&W	S&W	36	3/78	5/86
Power Authority of the State of New York								
• Indian Point 3 (Indian Point, N.Y.)	965	W	W	UE&C	Wedco	100	7/71	8/76
Public Service Co. of New Hampshire								
Seabrook 1 (Seabrook, N.H.)	1150	W	GE	UE&C	UE&C	48	11/79	2/84
Seabrook 2 (Seabrook, N.H.)	1150	W	GE	UE&C	UE&C	8	8/81	5/86
Public Service Electric & Gas Co.								
• Salem 1 (Salem, N.J.)	1090	W	W	Utility	UE&C	100	/71	6/77
• Salem 2 (Salem, N.J.)	1115	W	W	Utility	UE&C	100	/73	8/81

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial orig. sched- ule+	actual or ex- pected
Rochester Gas & Electric Corp. ● Robert F. Ginna (Ontario, N.Y.)	490	W	W	Gilbert	Bechtel	100	11/69	3/70
Yankee Atomic Electric Co. ● Yankee (Rowe, Mass.)	175	W	W	S&W	S&W	100	1/61	6/61
MIDWEST								
Commonwealth Edison Company ● Zion 1 (Zion, Ill.)	1040	W	W	S&L	Utility	100	4/72	10/73
● Zion 2 (Zion, Ill.)	1040	W	W	S&L	Utility	100	5/73	9/74
Byron 1 (Byron, Ill.)	1120	W	W	S&L	Utility	76	10/78	10/83
Byron 2 (Byron, Ill.)	1120	W	W	S&L	Utility	60	10/79	10/88
Braidwood 1 (Braidwood, Ill.)	1120	W	W	S&L	Utility	62	10/79	10/85
Braidwood 2 (Braidwood, Ill.)	1120	W	W	S&L	Utility	48	10/80	10/86
Commonwealth Edison Company, Interstate Power Company, and Iowa-Illinois Gas and Electric Company								
Carroll County 1 (Savanna, Ill.)	1120	W		S&L		0	10/87	Indef.
Carroll County 2 (Savanna, Ill.)	1120	W		S&L		0	10/88	Indef.
Consumer Power Co. ● Palisades (South Haven, Mich.)	740	C-E	W	Bechtel	Bechtel	100	7/70	12/71
Midland 1 (Midland, Mich.)	530	B&W	GE	Bechtel	Bechtel	66	5/78	7/84
Midland 2 (Midland, Mich.)	805	B&W	GE	Bechtel	Bechtel	66	5/79	12/83
Indiana & Michigan Electric Co. ● Donald C. Cook 1 (Bridgman, Mich.)	1054	W	GE	AEPSC	AEPSC	100	4/72	8/75
● Donald C. Cook 2 (Bridgman, Mich.)	1094	W	BB	AEPSC	AEPSC	100	4/73	7/78
Iowa Power & Light Co. Central Iowa (Vandalia, Iowa)	1270	B&W		Bechtel		0		Indef.
Kansas Gas & Electric Co. and Kansas City Power & Light Co. Wolf Creek (Burlington, Kans.)	1150	W	GE	Bech S&L	Daniel	74	4/81	4/84
Northern States Power Co. ● Prairie Island 1 (Red Wing, Minn.)	520	W	W	FPS	Utility	100	5/72	12/73
● Prairie Island 2 (Red Wing, Minn.)	520	W	W	FPS	Utility	100	5/75	12/74
Omaha Public Power District ● Fort Calhoun 1 (Fort Calhoun, Neb.)	478	C-E	GE	G&H	G&H	100	6/71	9/73
Public Service Indiana Marble Hill 1 (Jefferson County, Ind.)	1130	W	W	S&L	Utility	25	/82	late 86
Marble Hill 2 (Jefferson County, Ind.)	1130	W	W	S&L	Utility	11	/84	late 87
Toledo Edison Co. ● Davis-Besse 1 (Oak Harbor, Ohio)	906	B&W	GE	Bechtel	Bechtel	100	12/74	11/77
Union Electric Co. Callaway 1 (Fulton, Mo.)	1150	W	GE	Bechtel	Daniel	80	10/81	4/83
Callaway 2 (Fulton, Mo.)	1150	W	GE	Bechtel	Daniel	0.5	4/83	/90

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
 Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage	Commercial Operation	
						%	orig. sched- ule+	actual or ex- pected
Wisconsin Electric Power Co.								
● Point Beach 1 (Two Creeks, Wis.)	497	W	W	Bechtel	Bechtel	100	4/70	12/70
● Point Beach 2 (Two Creeks, Wis.)	497	W	W	Bechtel	Bechtel	100	4/71	10/72
Wisconsin Public Services Corporation								
● Kewaunee (Carlton, Wis.)	535	W	W	FPS	FPS	100	6/72	6/74
SOUTH								
Alabama Power Company								
● Joseph M. Farley 1 (Dothan, Ala.)	829	W	W	SCSI/ Bechtel	Daniel	100	4/75	12/77
● Joseph M. Farley 2 (Dothan, Ala.)	829	W	W	SCSI/ Bechtel	Daniel	99	4/76	7/81
Arkansas Power & Light Co.								
● Nuclear One 1 (Russellville, Ark.)	836	B&W	W	Bechtel	Bechtel	100	7/72	12/74
● Nuclear One 2 (Russellville, Ark.)	858	C-E	GE	Bechtel	Bechtel	100	12/75	3/80
Carolina Power & Light Co.								
● Robinson 2 (Hartsville, S.C.)	665	W	W	Ebasco	Ebasco	100	5/70	3/71
● Shearon Harris 1 (Newhill, N.C.)	900	W	W	Ebasco	Daniel	43	3/77	9/85
● Shearon Harris 2 (Newhill, N.C.)	900	W	W	Ebasco	Daniel	3	3/78	3/88
● Shearon Harris 3 (Newhill, N.C.)	900	W	W	Ebasco	Daniel	1	3/79	3/94
● Shearon Harris 4 (Newhill, N.C.)	900	W	W	Ebasco	Daniel	1	3/80	3/92
Duke Power Co.								
● Oconee 1 (Seneca, S.C.)	860	B&W	GE	Utility/ Bech	Utility	100	5/71	7/73
● Oconee 2 (Seneca, S.C.)	860	B&W	GE	Utility/ Bech	Utility	100	5/72	9/74
● Oconee 3 (Seneca, S.C.)	860	B&W	GE	Utility/ Bech	Utility	100	6/73	12/74
● McGuire 1 (Cornelius, N.C.)	1180	W	W	Utility	Utility	99	3/76	7/81
● McGuire 2 (Cornelius, N.C.)	1180	W	W	Utility	Utility	92	3/77	6/83
● Catawba 1 (Clover, S.C.)	1145	W	GE	Utility	Utility	83.4	3/79	3/84
● Catawba 2 (Clover, S.C.)	1145	W	GE	Utility	Utility	32	3/80	9/85
● Perkins 1 (Mocksville, N.C.)	1280	C-E	GE	Utility	Utility	0	1/81	indef.
● Perkins 2 (Mocksville, N.C.)	1280	C-E	GE	Utility	Utility	0	1/82	indef.
● Perkins 3 (Mocksville, N.C.)	1280	C-E	GE	Utility	Utility	0	1/83	indef.
● Cherokee 1 (Gaffney, S.C.)	1280	C-E	GE	Utility	Utility	18	9/82	indef.
● Cherokee 2 (Gaffney, S.C.)	1280	C-E	GE	Utility	Utility	0	9/83	indef.
● Cherokee 3 (Gaffney, S.C.)	1280	C-E	GE	Utility	Utility	0	9/84	indef.
Florida Power & Light Co.								
● Turkey Point 3 (Florida City, Fla.)	666	W	W	Bechtel	Bechtel	100	8/70	12/72
● Turkey Point 4 (Florida City, Fla.)	666	W	W	Bechtel	Bechtel	100	8/71	9/73
● St. Lucie 1 (Hutchinson Island, Fla.)	777	C-E	W	Ebasco	Ebasco	100	1/73	12/76
● St. Lucie 2 (Hutchinson Island, Fla.)	777	C-E	W	Ebasco	Ebasco	75	9/79	5/83
Florida Power Corporation								
● Crystal River 3 (Red Level, Fla.)	825	B&W	W	Gilbert	Jones	100	9/72	3/77

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*

Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial Operation orig. sched- ule+	actual or ex- pected
Georgia Power Co.								
Vogtle 1 (Waynesboro, Ga.)	1100	W	GE	SS Bechtel	Utility	18	2/78	5/85
Vogtle 2 (Waynesboro, Ga.)	1100	W	GE	SS Bechtel	Utility	10	2/79	11/86
Louisiana Power & Light Co.								
Waterford 3 (Taft, La.)	1104	C-E	W	Ebasco	Ebasco	85	1/77	3/83
South Carolina Electric & Gas Co.								
Virgil C. Summer 1 (Parr, S.C.)	900	W	GE	Gilbert	Daniel	99	10/77	6/82
Tennessee Valley Authority								
Sequoyah 1 (Daisy, Tenn.)	1148	W	W	Utility	Utility	100	10/73	7/83
Sequoyah 3 (Daisy, Tenn.)	1148	W	W	Utility	Utility	99	4/74	7/82
Watts Bar 1 (Spring City, Tenn.)	1177	W	W	Utility	Utility	74	10/76	1/84
Watts Bar 2 (Spring City, Tenn.)	1177	W	W	Utility	Utility	58	4/77	10/84
Bellefonte 1 (Scottsboro, Ala.)	1213	B&W	BB	Utility	Utility	75	7/77	12/85
Bellefonte 2 (Scottsboro, Ala.)	1213	B&W	BB	Utility	Utility	59	4/78	9/86
Yellow Creek 1 (Luka, Miss)	1285	C-E	GE	Utility	Utility	26	4/83	4/88
Yellow Creek 2 (Luka, Miss)	1285	C-E	GE	Utility	Utility	3	4/84	Indef.
Virginia Electric & Power Co.								
• Surry 1 (Gravel Neck, Va.)	775	W	W	S&W	S&W	100	3/71	12/72
• Surry 2 (Gravel Neck, Va.)	775	W	W	S&W	S&W	100	3/72	5/73
• North Anna 1 (Mineral, Va.)	865	W	W	S&W	S&W	100	3/74	6/78
• North Anna 2 (Mineral, Va.)	890	W	W	S&W	S&W	100	7/75	12/80
• North Anna 3 (Mineral, Va.)	907	B&W	W	S&W	S&W	8.8	3/75	/89
SOUTHWEST								
Arizona Public Service Co.								
Palo Verde 1 (Wintersburg, Ariz.)	1270	C-E	GE	Bechtel	Bechtel	88	5/81	5/83
Palo Verde 2 (Wintersburg, Ariz.)	1270	C-E	GE	Bechtel	Bechtel	61	11/82	5/84
Palo Verde 3 (Wintersburg, Ariz.)	1270	C-E	GE	Bechtel	Bechtel	21	5/84	5/86
Houston Lighting & Power Company								
South Texas Project 1 (Palacios, Tex)	1250	W	W	Brown	Brown	60	10/80	/84
South Texas Project 2 (Palacios, Tex)	1250	W	W	Brown	Brown	23	3/82	/86
Texas Utilities Generating Company								
Comanche Peak 1 (Glen Rose, Tex.)	1150	W	Allis	G&H	B&R	88	1/80	/82
Comanche Peak 2 (Glen Rose, Tex.)	1150	W	Allis	G&H	B&R	52	1/82	/84
WEST AND NORTHWEST								
Pacific Gas & Electric Company								
Diablo Canyon 1 (Avila Beach, Ca.)	1084	W	W	Utility	Utility	100	5/72	1/82
Diablo Canyon 2 (Avila Beach, Ca.)	1106	W	W	Utility	Utility	93	7/74	8/82
Portland General Electric Co.								
• Trojan (Prescott, Ore.)	1130	W	GE	Bechtel	Indep	100	9/74	5/76
Pebble Springs 1 (Arlington, Ore.)	1260	B&W	GE	Bechtel	Indep	0	7/80	9/88
Pebble Springs 2 (Arlington, Ore.)	1260	B&W	GE	Bechtel	Indep	0	6/83	9/90
Sacramento Municipal Utility District								
• Rancho Seco (Clay Station, Ca.)	913	B&W	W	Bechtel	Bechtel	100	5/73	4/75

APPENDIX A (cont'd)

WORLD LIST OF PWR NUCLEAR POWER PLANTS*
Operable, Under Construction, or on Order (30 MWe and Over) as of June 30, 1981

	Net MWe	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Con- struc- tion stage %	Commercial Operation orig. sched- ule+	actual or ex- pected
Southern California Edison and San Diego Gas & Electric Co.								
• San Onofre 1 (San Clemente, Ca.)	436	W	W	Bechtel	Bechtel	100		1/68
San Onofre 2 (San Clemente, Ca.)	436	W	W	Bechtel	Bechtel	98	6/75	6/82
San Onofre 3 (San Clemente, Ca.)	436	W	W	Bechtel	Bechtel	78	6/75	7/83
Washington Public Power Supply System								
WNP-1 (Richland, Wash.)	1250	B&W	W	UE&C	UE&C	49.6	9/80	6/86
WNP-3 (Satsop, Wash.)	1240	C-E	W	Ebasco	Ebasco	30.6	3/82	12/86
WNP-4 (Richland, Wash.)	1250	B&W	W	UE&C	UE&C	24.2	3/82	6/87
WNP-6 (Satsop, Wash.)	1240	C-E	W	Ebasco	Ebasco	13.7	3/83	12/87
U.S. PWR Total (108 Units)	107846							
USSR								
Ministry of Electric Power								
• Novo-Voronezh 1 (Voronezh)	210					100		10/64
• Novo-Voronezh 2 (Voronezh)	365					100	12/69	
• Novo-Voronezh 3 (Voronezh)	440					100		6/72
• Kola 1 (near Murmansk)	440					100		10/73
• Kola 2 (near Murmansk)	440					100	74	12/74
Kola 3 (near Murmansk)	440						/80	
Kola 4 (near Murmansk)	440						/80	
• Novo-Voronezh 4 (Voronezh)	440					100		4/73
• Novo-Voronezh 5 (Voronezh)	1000					100		/80
• Armenia 1 (Ararat Valley)	400					100	/75	/76
• Armenia 2 (Ararat Valley)	400					100	/75	/81
• West-Ukrainian 1 (Rovno)	440					100		/76
West-Ukrainian 2 (Rovno)	440						/80	
South-Ukrainian 1	1000						/80	
Chernobyl 3	1000							/82
Chernobyl 4	1000							/82
YUGOSLAVIA								
Savske Electrane (Slovenia) and Elektroprivreda (Croatia)								
Krsko (Krsko)	615	W	W	Gilbert	W/local	100	12/78	7/81
Non-U.S. PWR Total (174 units)	143,724							
WORLD PWR TOTAL (282 units)	251,570							

ABBREVIATIONS USED IN THIS TABLE

ACEC: Ateliers de Constructions Electriques de Charleroi S.A. (Belgium)

ACECOWEN: ACECO with Westinghouse (Belgium)

ACLF group: ACECO Creusot-Loire/Framatom/Westinghouse Electric Energy Systems Europe (France)

ADF: Auxeltra-Delens-Francois

AECL: Atomic Energy of Canada Ltd.

AEE: Atomenergoexport (USSR) (formerly TPE. Technopromexport)

AEG: Allgemeine Elektricitäts-Gesellschaft. Age-Telefunken (W. Germany)

AEI: Associated Electric Industries Ltd. (U.K.)

AEPSC: American Electric Power Service Corp. (U.S.)

Allis: Allis-Chalmers (U.S.)

Alsthom: Ste Generale de Constructions Electriques et Mechaniques (France)

Arge/KKU: Dyckerhoff & Widmann AG; Wayss & Freitag AG; Hegdkamp (FGR)

BCC: Brown Boven et Cie (Switzerland)

BBR: Babcock-Brown Boveri Reaktor GmbH (W. Germany)

Bech: Bechtel Corporation (U.S.)

B&R: Burns & Roe, Inc. (U.S.)

B&W: Babcock & Wilcox Co (U.S.)

BRAUN: C. F. Braun & Co. (U.S.)

Brown: Brown & Root, Inc. (U.S.)

C-B: Campenon-Bernard (France)

C-E: Combustion Engineering, Inc. (U.S.)

CEM: Compagnie Electro Mechanique (France)

CFE: Cie d'Enterprises CFE S.A. (Belgium)

Chag: Chagnaud (France)

CITRA: Compagnie Industrielle de Travaux (France)

CL: Creusot-Lorie (France)

CM: Chantiers Modernes (France)

COP/TOSI/ACEC: Cokesrill Ougree-Providence Franco Tosi SpA Atehers de
Constructions JElectriques de Charleroi SA (Belgium)

CTAFMC: CFE Travaux, Astobel General Contractors, JFrancois et Fils Maurice
Delens, Campenon Bernard (Belgium)

Daniel: Daniel Construction Co. (U.S.)

EA: Empresanos Agrupados (Spain)

Ebasco: Ebasco Services, Inc. (U.S.)

ERBE: Hungarian Company for Power Plant Investment

ETOCEA: Entrecanales y Ocisa (Spain)

EW: ElectroWatt Ltd. (Switzerland)

EyT: Entrecañales y Tavora SA (Spain)

FPS: Fluor Power Services (U.S.)

Fra: Framatome Societe Franco-Americanine de Constructions Atomiques SA
(France)

GE: General Electric Co. (U.S.)

G&H: Gibbs & Hill, Inc. (U.S.)

G&HE: Gibbs & Hill Espanola SA (Spain)

Gilbert: Gilbert Associates, Inc. (U.S.)

GKW: Germeinschaftkraftwerk Weser GmbH (FRG)

GTM: Grancs Travaux de Marseille (France)

Haz: Hazama Gumi Co. (Japan)

Hoch: Hochtief AG (W. Germany)

Iber: Iberduero SA (Spain)

INITEC: Empresa Nacional de Ingenena y Tecnologia SA (Spain)

Jones: J.A. Jones Construction Co. (U.S.)

Kaiser: Kaiser Engineers (U.S.)

Kum: Kumagaya Gumi Co. (Japan)
KWU: Kraftwerk Union AG (W. Germany)
MAPI: Mitsubishi Atomic Power Industries, Inc. (Japan)
MEL: Mitsubishi Electric Corporation (Japan)
MHI: Mitsubishi Heavy Industries, Ltd. (Japan)
Obay: Obayashi Gumi Co. (Japan)
PE: Promon Engenharia SA (Brazil)
PWR: pressurized water reactor
Rateau: Rateau. Ste. (France)
RDM: Rotterdamse Droogdok Maatschappij (The Netherlands)
SB: Spie Batignolles SA (France)
SeB: Sainrapt et Brice (France)
SGE: Societe Generale d'Enterprises (France)
S&L: Sargent & Lundy Engineers (U.S.)
S-L: Stal-Laval Turbin AB (Sweden)
SO: Siemens Osterreich (Austria)
SS: Souther Services, Inc. (U.S.)
SSPB: Swedish State Power Board
Stork: Koninklijke Machinefabriek Stork (The Netherlands)
S&W: Stone & Webster Engineering Corp. (U.S.)
TE: Traction-Electricite
TPC: Taiwan Power Company
UE&C: United Engineers & Constructors (U.S.)
VANE: Vandellos-Empresarios Agrupados (Spain)
VBB: AB Vattenbyggnadsbyran
W: Westinghouse Electric Corporation (U.S.)
Wedco: a subsidiary of Westinghouse Electric Corporation (U.S.)

APPENDIX B

STEAM GENERATOR OWNERS GROUP MEMBER UTILITIES

Arizona Nuclear Power Project	Florida Power & Light Company
Arkansas Power and Light Co.	Georiga Power Company
Baltimore Gas and Electric Co.	Houston Lighting and Power Company
Boston Edison Co.	Northeast Utilities
Carolina Power and Light Co.	Portland General Electric Co.
Central Electricity Generating Board (CEGB, United Kingdom)	Power Authority of the State of New York
CRIEPI (Japan)	Public Service Electric Gas Co.
Commonwealth Edison Company	Rochester Gas and Electric Corp.
Consolidated Edison Company of New York	Southern California Edison Company
Consumers Power Company	Swedish State Power Board (Sweden)
Duke Power Company	Tennessee Valley Authority
Electricite de France (France)	Wisconsin Electric Power Company

GLOSSARY

AVT	all volatile (alkali) treatment
B&W	Babcock and Wilcox
CE	Combustion Engineering
cyclone	a cone-shaped air-cleaning apparatus operated by centrifugal separation that is used to collect particles
deaerator	a device in which oxygen and carbon dioxide are removed from boiler feedwater
denting	distortion that occurs when a tube support plate made of carbon steel corrodes to the point where it squeezes down the steam generator tubing
D&S	dry steam at the saturation temperature
ECT	eddy current testing
EdF	Electricite de France
EPRI	Electric Power Research Institute
fretting	rubbing of adjacent surfaces
HTGR	high-temperature gas-cooled reactor
IGSCC	intergranular stress-corrosion cracking
ISI	in-service inspection
KWU	Kraftwerk Union
LMFBR	liquid-metal fast breeder reactor
MECT	multifrequency eddy current testing
MSR	Molten salt reactor
NDE	Nondestructive examination
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply systems
OTSG	once-through steam generator
PWR	Pressurized water reactor
SCC	stress-corrosion cracking
sparger	a liquid-distribution device consisting of a length of tubing with holes at spaced intervals along the length; used in spray columns
U-bend	reversed-U-shaped bent tubing used in recirculating steam generators.
W	Westinghouse
wastage	wall thinning

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