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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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March 21, 1979

GL-79-15

ALL PRESSURIZED WATER REACTOR LICENSEES

Gentlemen:

This letter is being sent to all licensees authorized to operate or construct a pressurized water power reactor and to all applicants for a license to operate or construct a pressurized water power reactor.

Operating problems have occurred in Pressurized Water Reactor (PWR) steam generators. The enclosed report, "Summary of Operating Experience with Recirculating Steam Generators," NUREG 0523, focuses on the problems associated with steam generators of the recirculation type, i.e., those manufactured by Combustion Engineering and Westinghouse. The report discusses the NRC staff's evaluation of these problems and the programs for resolving these problems.

The NRC has recently identified steam generator degradation as an Unresolved Safety Issue deserving the highest priority for resolution. However, for the reasons identified in the report, the NRC staff has concluded that continued operation of existing plants and licensing of new plants with recirculation type steam generators, pending completion of our review, does not constitute an undue risk to the health and safety of the public and therefore may continue.

It should be noted that a number of research efforts are currently under way which will improve our knowledge of steam generator degradation mechanisms. The information presented in the report represents our current understanding of each issue. Comments on this report and information

TTA4
CP

related to steam generator degradation mechanisms are encouraged and should be forwarded to Dr. Boen-Dar Liaw, Engineering Branch, Division of Operating Reactors, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

D. G. Eisenhut, Deputy Director
Division of Operating Reactors

Enclosure:
Summary of Operating Experience
with Recirculating Steam
Generators, January 1979,
NUREG 0523

cc w/o enclosure:
Service List

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Consolidated Edison Company of New York, Inc.

cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

Joseph D. Block, Esquire
Executive Vice President
Administrative
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Edward J. Sack, Esquire
Law Department
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N.W.
Washington, D. C. 20005

Dr. Lawrence R. Quarles
Apartment 51
Kendal at Longwood
Kennett Square, Pennsylvania 19348

Theodore A. Rebelowski
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 1051.

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NUREG-0523

SUMMARY OF OPERATING EXPERIENCE WITH RECIRCULATING STEAM GENERATORS

D. G. Eisenhut
B. D. Liaw
J. Strosnider

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

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Division of Operating Reactors
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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Figure 5 - Indian Point Unit 2, Consolidated Edison Company

Figure 9 - Westinghouse Electric Corporation

Figure 14 - Combustion Engineering

SUMMARY OF OPERATING EXPERIENCE
WITH RECIRCULATING STEAM GENERATORS

1. INTRODUCTION

Operating problems have occurred in the steam generators of each of the three manufacturers of pressurized water reactors (PWR) nuclear steam supply systems (NSSS): Babcock & Wilcox, Combustion Engineering, and Westinghouse Electric Corporation. This report focuses on the problems associated with steam generators of the recirculation type that are designed by Westinghouse and Combustion Engineering. It identifies the operational problems observed to date, including the NRC staff's evaluation of such problems, and provides a status report summarizing the NSSS, licensee, and staff programs for the resolution of each problem. (Figure 1 shows the major types of degradation for recirculation type steam generators.) Information related to the cause of these problems is discussed to the extent that such information is known and available. It should be noted that a number of research efforts related to these problems are currently under way. Therefore, some of the causal information included in this summary represents our current understanding of each issue. For those who are not completely familiar with PWRs, Appendix A briefly describes the functions of the various coolant systems of pressurized water reactors.

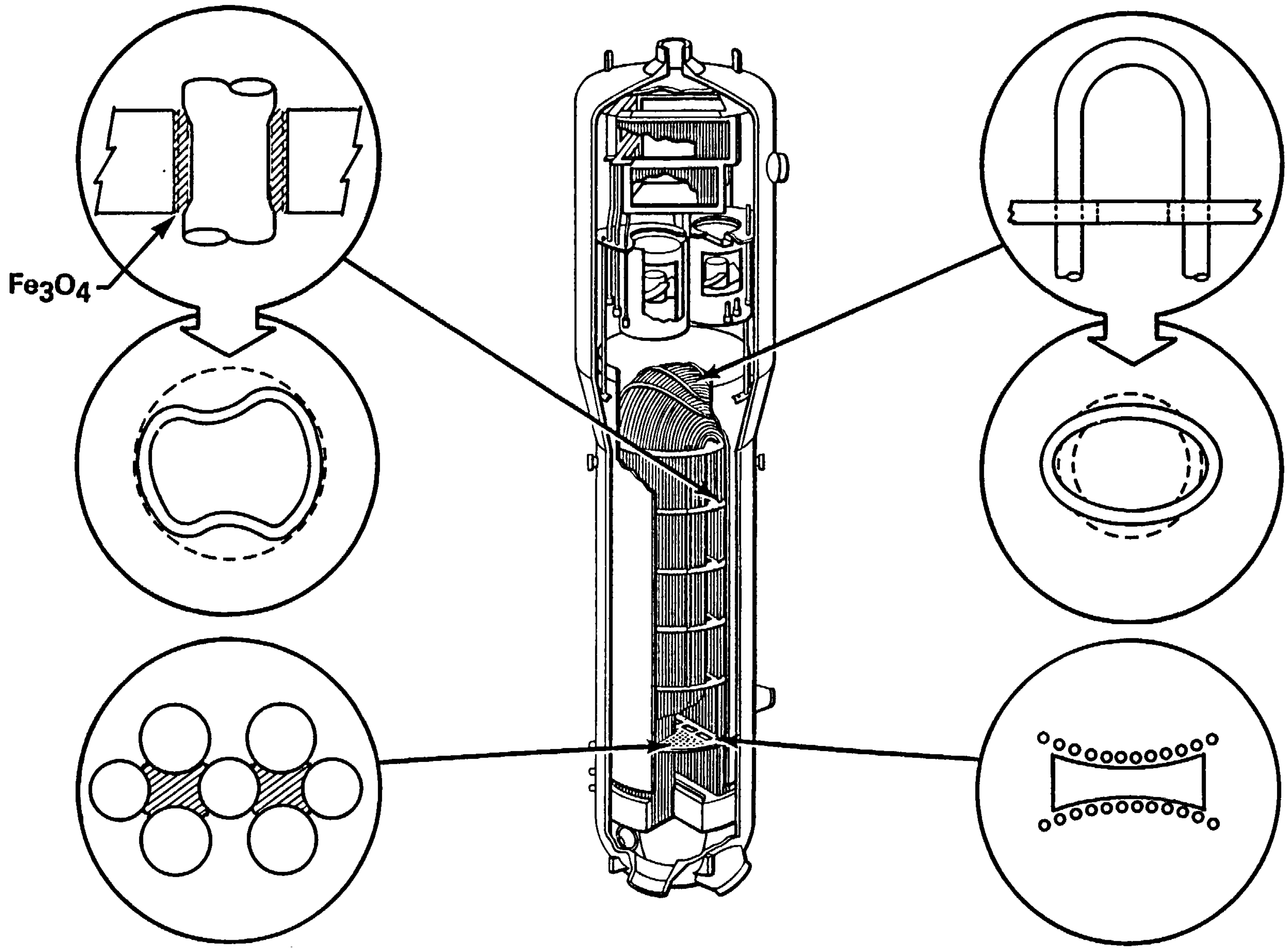


FIGURE 1. PROBLEM AREAS IN PWR STEAM GENERATOR

2. DESCRIPTION OF TYPES OF OPERATIONAL PROBLEMS

2.1 Caustic Stress Corrosion and Wastage

Inconel-600 tubing is typical of that found in most operating recirculation types (U-tube) of steam generators. Intergranular stress corrosion cracking and localized tube wall thinning (wastage) are the major types of degradation that affect the exterior surface of the tubing. "Pitting" (that is, relatively deep, small volume wastage of the exterior surface of steam generator tubing) has also been experienced.

Wastage has occurred when a coordinated phosphate treatment of the secondary coolant has been utilized and is attributed to the local concentration of residual acidic phosphates. In some cases, these acidic phosphates have not been completely removed after a changeover from a phosphate treatment to all-volatile treatment (AVT)* of the secondary coolant water. Approximately a dozen plants have experienced some degree of wastage while operating with phosphate water treatment. Since the establishment of AVT chemistry control, both the evidence and the extent of wastage have diminished and no further substantial tube degradation due to this mechanism is expected to occur. Caustic stress corrosion cracking is caused mainly by either the formation of caustic compounds in the secondary coolant (i.e., from hydrolysis of trisodium phosphate) or by caustic-forming impurities carried into the steam generator by the feedwater.

The principal cause of serious corrosion damage from either wastage or caustic stress corrosion cracking is the local concentration of aggressive chemicals within the secondary side of steam generators. The major source of these impurities is in-leakage of condenser cooling water. Because of this, the boundary between the secondary coolant system and the condenser cooling system is of significance. The concentration of these impurities is affected by thermal and mechanical design parameters of steam generators, by accumulations of chemicals and corrosion products within the steam generators as plants age, and by the nominal and transient variations in water and air environments to which steam generator internals are exposed. Both types of corrosion generally occur where regions of restricted water flow and high heat flux tube surfaces cause impurities to concentrate or phosphates to precipitate (hideout). These high concentrations may occur at crevices between the tubing and the tube support plates or the tube sheet, and in areas where sludge deposits have built up on the tube sheet or tube support plates.

*This chemistry control is called AVT because the chemicals injected into the secondary water eventually volatilize and escape with steam.

2.2 Denting and U-Bend Cracking

In December 1975, the NRC was informed by Westinghouse that several plants designed by them had experienced steam generator tube deformation in the form of a reduction in tube diameter. This reduction in tube diameter was later termed "denting."

Later laboratory reports of dented tubes indicated that the annulus between tubes and support plates was filled with hardened corrosion products (as shown in Figure 2) that continue to form by the corrosion of the support plates and, therefore, exert sufficient forces to "dent" the tube diametrically. Severe buildup of corrosion products has caused cracking of the tube support plate ligaments between the tube holes and the water circulation flow holes. The phenomenon of denting in Westinghouse plants has been attributed to acid chloride salts that concentrate in the annulus between the tubes and the tube support plates. The first incidence of denting occurred shortly after steam generator secondary water chemistry control was switched from phosphate treatment to an all-volatile treatment (AVT). Contamination of the secondary coolant by inleakage of condenser cooling water was believed to have caused a catalytic reaction with residual phosphates.

The simultaneous presence of residual phosphate in the tube/tube support plate annulus and chloride in the condenser cooling water caused accelerated corrosion of carbon steel support plates present in most plants. The corrosion product from the carbon steel support plate occupies approximately twice the volume of the material corroded. The continuing corrosion product exerts sufficient forces to dent the tube and/or crack the tube support plate ligaments between the tube holes and the water circulation flow holes. These dented tubes thus become subject to higher strains; however, they have otherwise generally retained their integrity. (That is, there have been relatively few leaks at the dent locations and no rapid failures at dent locations.) Denting has occurred more recently at plants that have used AVT exclusively.

Along the chord of the innermost rows of tubes in Westinghouse-designed steam generators, there is a row of rectangular flow slots in the tube support plate. These slots are approximately 16 inches long by 2-3/4 inches wide and are spaced about 20 inches center to center (see Figure 3). Because of the pressure built up in the tube support plate due to the denting phenomenon, the flow slots in the tube support plates have been observed to deform (the "hourglassing" effect); that is, the central portion of the parallel flow slot walls has moved closer, so that some flow slots are now narrower in the center than at the ends. Figures 4 and 5 are photographs of hourglassed flow slots from San Onofre Unit 1 and Indian Point Unit 2, respectively. Because the initial parallel slot walls have moved closer, the tube support plate material supporting the

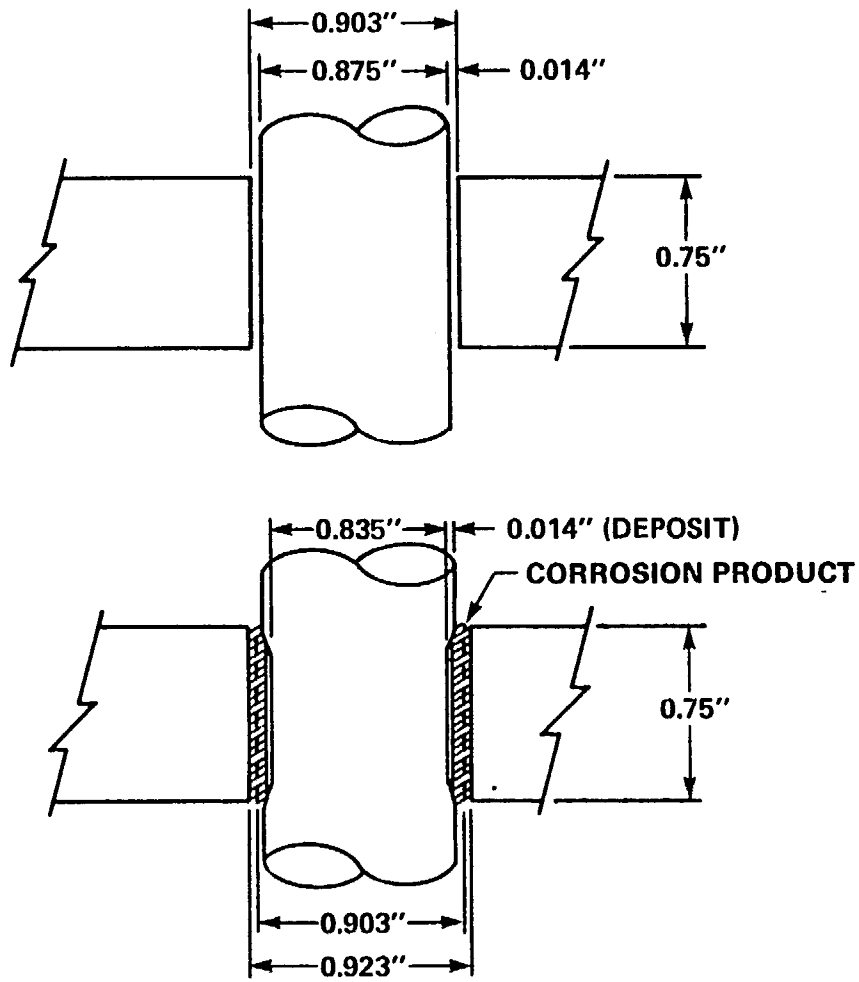


FIGURE 2. TYPICAL DENTING MECHANISM

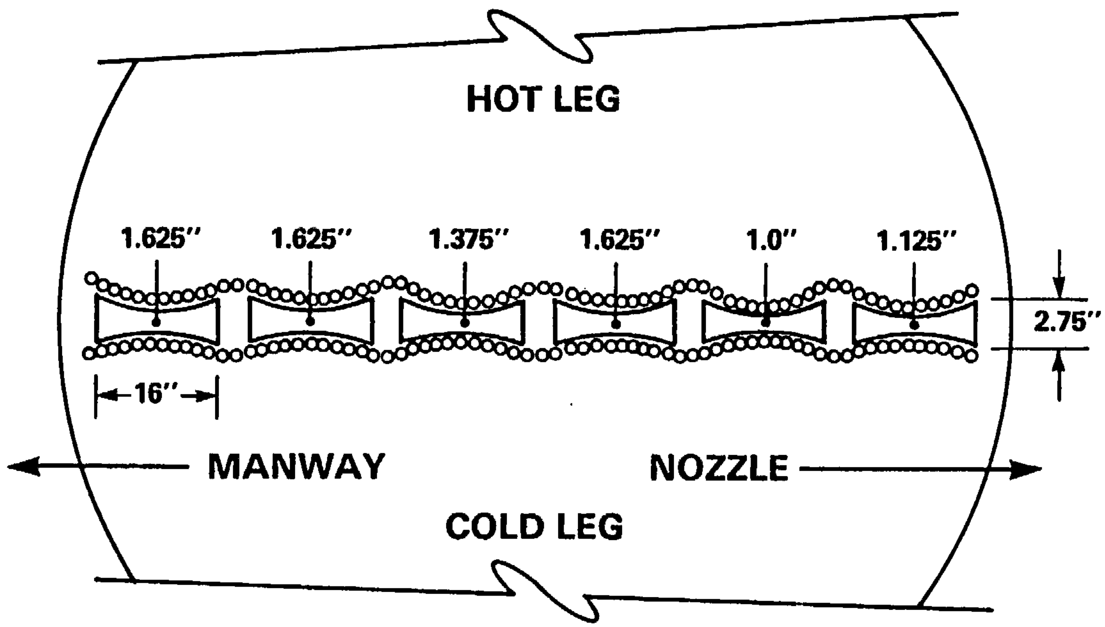


FIGURE 3. FLOW SLOT DEFORMATION

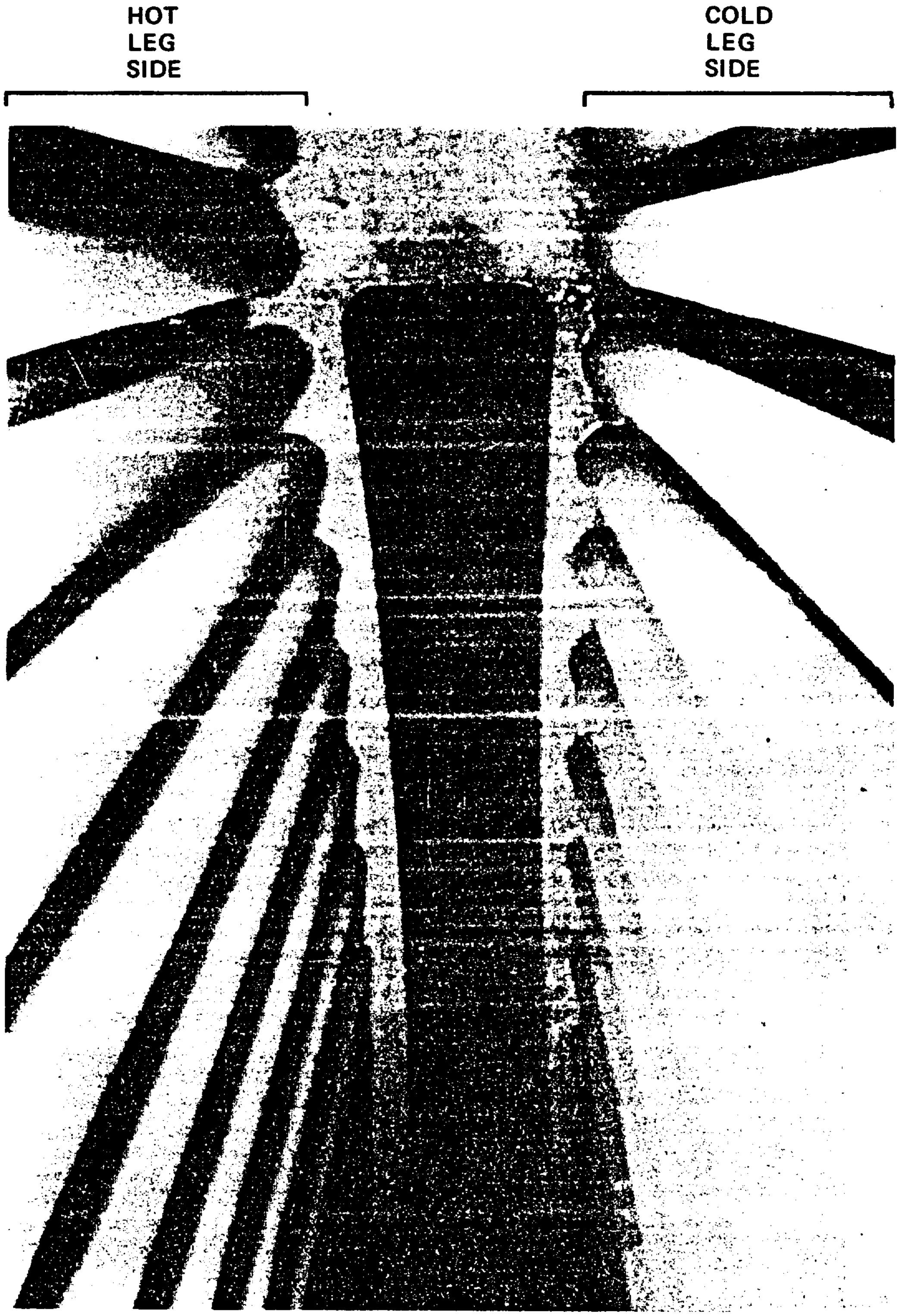


FIGURE 4. FLOW SLOT "HOURGLASSING"

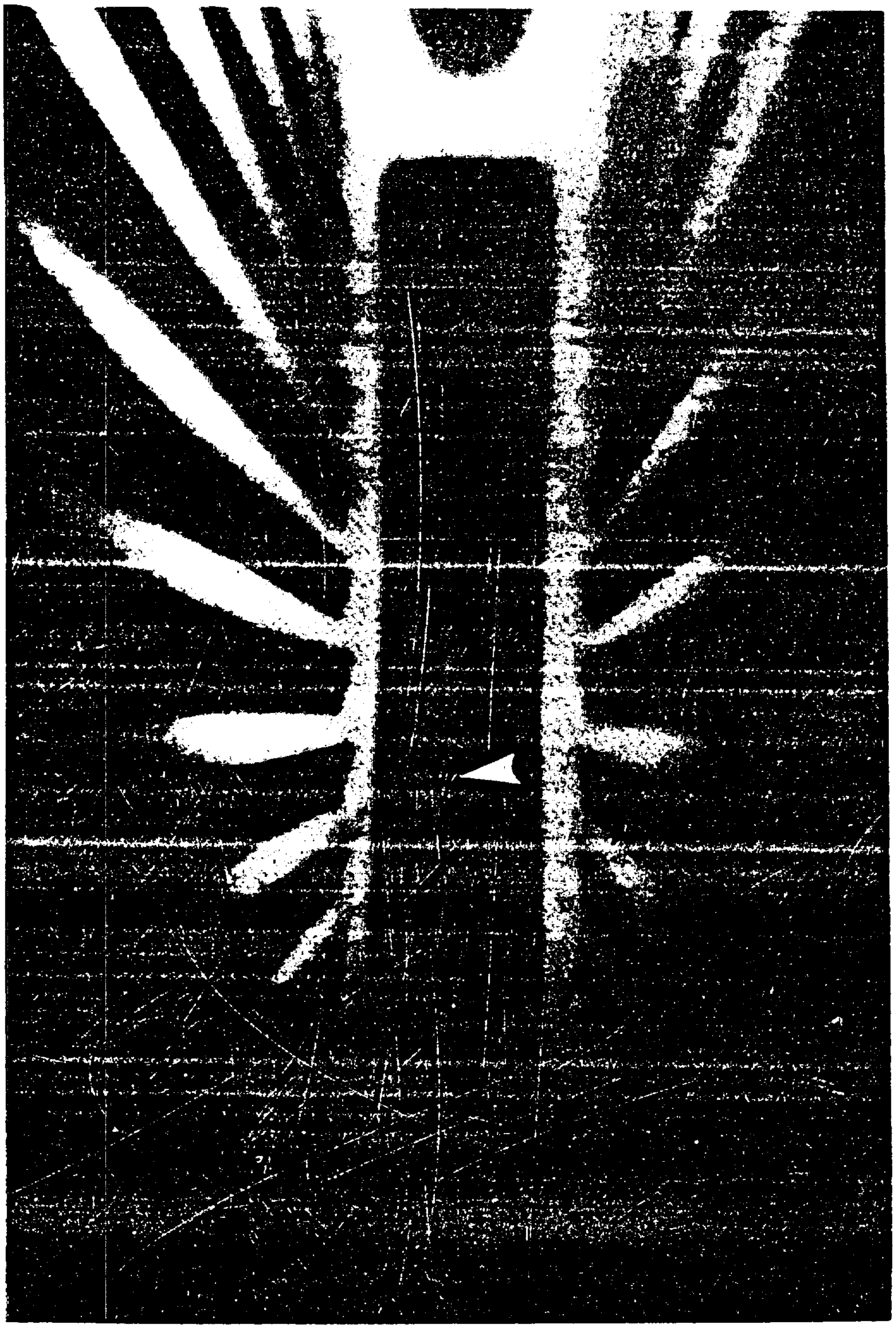


FIGURE 5. SUPPORT PLATE CRACKING AT EDGE OF FLOW SLOT

tubes nearest this central portion of these flow slots has also moved inward, which consequently forces an inward displacement of the legs of the tubes at these locations. When this inward movement of the legs of the tubes has occurred at the upper support plate, it has been shown to cause an increase in the hoop strain at the tube U-bend apex. This effect is shown in Figure 6. It is this additional increase in strain at the apex of the U-bend that is believed to be the additional factor required to initiate and increase the susceptibility of Inconel-600 alloy tubing exposed to PWR reactor coolant to stress corrosion cracking at the top of the U-bend.

Because of tube denting or ovalization (non-uniform denting, see Figure 7), tubes at tube/tube support plates have developed small stress corrosion cracks in the longitudinal direction of the tube. These small cracks are masked by the support plates. During normal operation, small leaks through these cracks have occurred in a few plants where severe tube denting has occurred.

2.3 Tube Support Plate Cracking

As a consequence of continuing magnetite growth in the tube-to-tube support plate annulus and the subsequent cracking of support plates, portions (small pieces) of the support plate material in Westinghouse-designed steam generators have moved with tubes into the flow slots to cause the so-called "islanding" phenomenon; i.e., broken support pieces moving into flow slots (see Figure 8).

This phenomenon could lead to the possible loss of lateral support of some inner-row tubes. Concern about this problem has been alleviated in many plants by the fact that many tubes in inner rows have been plugged as part of the preventive plugging programs.

2.4 Anti-Vibration Bar Wear or Fretting

Steam generator design in several plants use anti-vibration bars in the upper "curved" portion of U-bend steam generator tube bundles to provide lateral support. Inspections in two plants have shown that these bars have caused fretting between the tubes and the bars. Recent inspections of some removed bars have revealed serious degradation. The cause appears to be primarily mechanical and is affected by the material (i.e., carbon steel bars versus Inconel bars in new designs), the shape of the bars, the clearances, and the bar support design. Only two operating nuclear power plants in the United States have anti-vibration bar design that have experienced degradation and, thus, this is not considered to be a widespread problem.

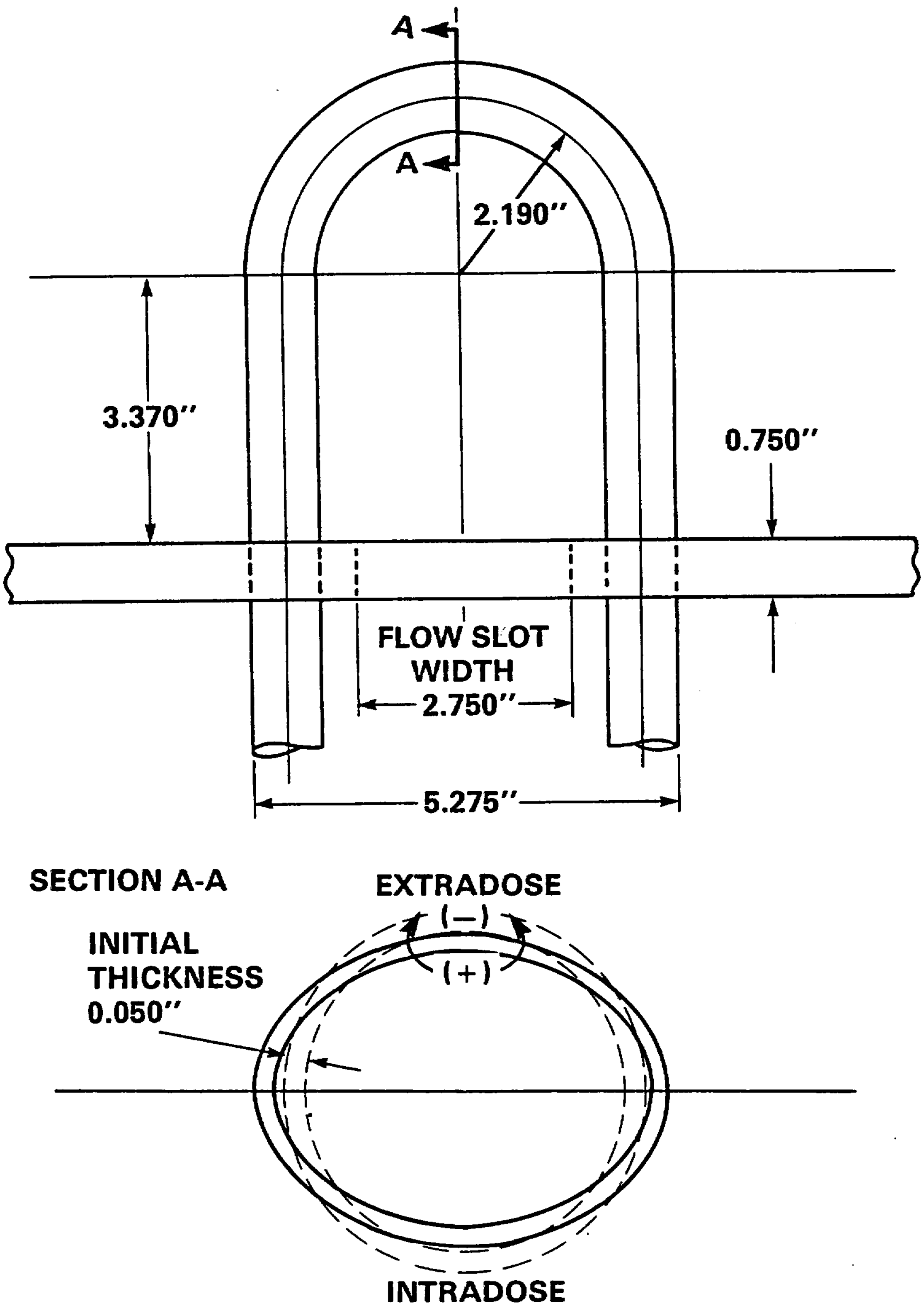
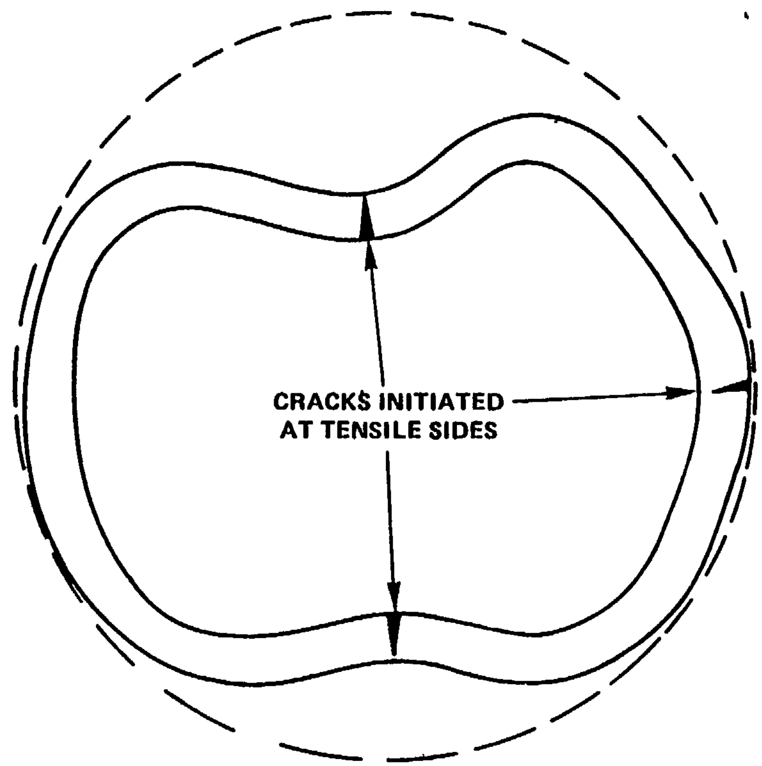


FIGURE 6. SCHEMATICS OF U-TUBE OVALIZATION



SCALE: OUTSIDE DIAMETER = 0.875 INCHES
WALL THICKNESS = 0.050 INCHES

FIGURE 7. CROSS SECTION OF DENTED TUBE SHOWING LOCATION OF LEAKAGE (TURKEY POINT UNIT 4, 4TH TUBE SUPPORT PLATE ELEVATION).

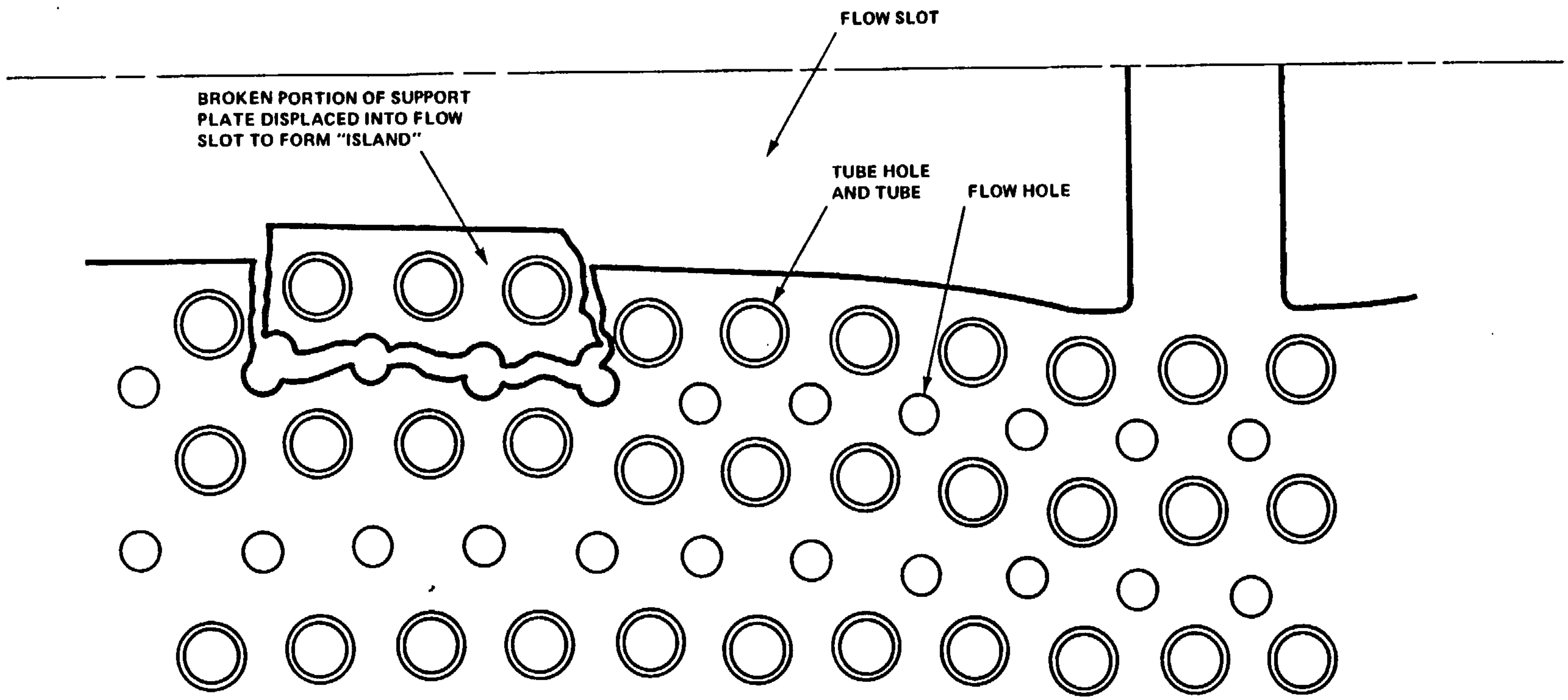


FIGURE 8. STEAM GENERATOR SUPPORT PLATE "ISLANDING"

3. OPERATING EXPERIENCE

At this time (December 1978), there are thirty-three operating PWR units with recirculation type steam generators in the United States.* Of these, seventeen have been found to have one or more forms of tube degradation. These total numbers do not include eight PWRs using the once-through steam generators designed by Babcock & Wilcox. Table 1 identifies the Westinghouse and Combustion Engineering units that have been found to have significant forms of degradation. Table 2 summarizes the extent of steam generator tube plugging that has been performed to date as a result of all modes of degradation.

3.1 Plants Designed by Westinghouse

All commercially operating Westinghouse-designed steam generators are the vertical shell recirculation type units (Figure 9). All use Inconel-600 tubing except for the Yankee Rowe unit, which uses stainless steel tubing. A major consideration in all Westinghouse designs of operating steam generators is that they use several fully extended drilled support plates (Figure 10). The drilled support plates are significant because the annular space between the steam generator tube and the support plate is related to several forms of degradation. By comparison, the typical Combustion Engineering design in operating reactors uses both drilled support plates and "egg crate" support plates (Figure 11).

The controlling parameter for the various corrosion mechanisms that lead to tube degradation appears to be related to steam generator secondary water chemistry control. The operating history and method of secondary water chemistry control for these plants is shown in Figure 12. The predominant method of chemistry control prior to 1975 was coordinated pH-phosphate control. In late 1974 through early 1975, thirteen Westinghouse plants converted from phosphate control to all-volatile treatment (AVT). Nine newer plants started up with AVT control. Two plants elected not to convert to AVT because of concern for condenser tube integrity problems and their particular satisfactory operating history with respect to steam generator tube corrosion. Both plants did make minor changes to ensure a more restrictive range of phosphate concentration. Tube corrosion (thinning) at both plants is continuing but at a much slower rate.

3.1.1 Caustic Stress Corrosion and Wastage

The purpose of the chemistry changeover for the operating plants from phosphate (PO_4) chemistry control to the AVT method was principally to arrest tube thinning (wastage) that primarily occurred near the tube sheet.

*Indian Point Unit 1 is not included.

TABLE 1
SUMMARY OF STEAM GENERATOR ADVERSE EXPERIENCE

12/29/78

NSSS	PLANT NAME	WASTAGE	U-BEND FRETTING	SECONDARY SIDE CRACKING		DENTING					CONSIDERATION FOR REPLACEMENT OR RETUBING
				HIGH CYCLE FATIGUE	SCC*	TUBE DENTING	SP HOUR-GLASSING	SP CRACKING OR ISLANDING	LEAKING DENTS	U-BEND CRACK	
CE	MAINE YANKEE					X-MINOR					
	MILLSTONE 2					X-MODERATE		X			
	PALISADES	X			X	X-MINOR					X (Tube Sleeving)
	ST. LUCIE 1					X-MINOR					
W	HADDAM NECK	X	X		X	X-MINOR					
	R.E. GINNA 1	X			X	X-MINOR					
	INDIAN POINT 2					X-MODERATE	X	X	X		
	INDIAN POINT 3					X-MINOR			X		
	POINT BEACH 1	X			X	X-MODERATE					
	POINT BEACH 2	X				X-MODERATE			X (?)		
	H.B. ROBINSON 2	X			X	X-MINOR					
	SAN ONOFRE 1	X	X			X-EXTENSIVE	X	X	X		
	SURRY 1	X				X-EXTENSIVE	X	X	X	X	X
	SURRY 2	X				X-EXTENSIVE	X	X	X	X	X
	TURKEY POINT 3	X				X-EXTENSIVE	X	X	X		X
	TURKEY POINT 4	X				X-EXTENSIVE	X	X	X	X	X
	YANKEE ROWE				X						

*SCC – CAUSTIC STRESS CORROSION CRACKING

NOTES:
1. TO DATE THERE ARE 33 OPERATING PWR UNITS (NOT INCLUDING INDIAN POINT 1) WHICH UTILIZE RECIRCULATION TYPE OF STEAM GENERATORS.
2. 17 HAVE BEEN FOUND TO HAVE ONE OR MORE FORM(S) OF DEGRADATION, AS SUMMARIZED ABOVE.
3. TROJAN AND D. C. COOK HAVE HAD INDICATIONS OF LIMITED DEGRADATION IN RECENT INSPECTIONS.

TABLE 2
STEAM GENERATOR TUBE PLUGGING SUMMARY

12/29/78

NSSS	PLANT NAME	PERCENTAGE OF TUBES PLUGGED (0 – 10 – 20 – 30)	LEAKING/LOST PLUGS	REMARKS
CE	Arkansas 2			
	Calvert Cliffs 1			0%, 1/78
	Calvert Cliffs 2			0%, 1/78
	Fort Calhoun 1			0%, 11/77
	Maine Yankee			0%, 4/77, some denting
	Millstone 2	2-78		
	Palisades 1	2-78		35 tubes are sleeved
	St. Lucie 1			0%, 2/78, some denting
W	Beaver Valley 1			
	Cook 1			0%, 2/77
	Cook 2			
	Farley 1			0%, 5/78
	Ginna 1	4-77		
	Haddam Neck	10-77		<1%
	Indian Point 2	5-78		<1%
	Indian Point 3			
	Kewaunee			0%, 4/78
	North Anna 1			
	Point Beach 1	2-78	Lost	
	Point Beach 2	3-78		<1%
	Prairie Is. 1			0%, 3/77
	Prairie Is. 2	11-77		<1%
	Robinson 2	2-78		
	Salem 1			
	San Onofre 1	4-77		
	Surry 1	12-78	Leaking	
	Surry 2	7-78	Leaking	
	Trojan			0%, 5/77, 1 tube leaking
	Turkey Point 3	12-77	Lost	
	Turkey Point 4	9-78	Lost	
	Yankee Rowe	7-77		
	Zion 1			0%, 3/78
	Zion 2			

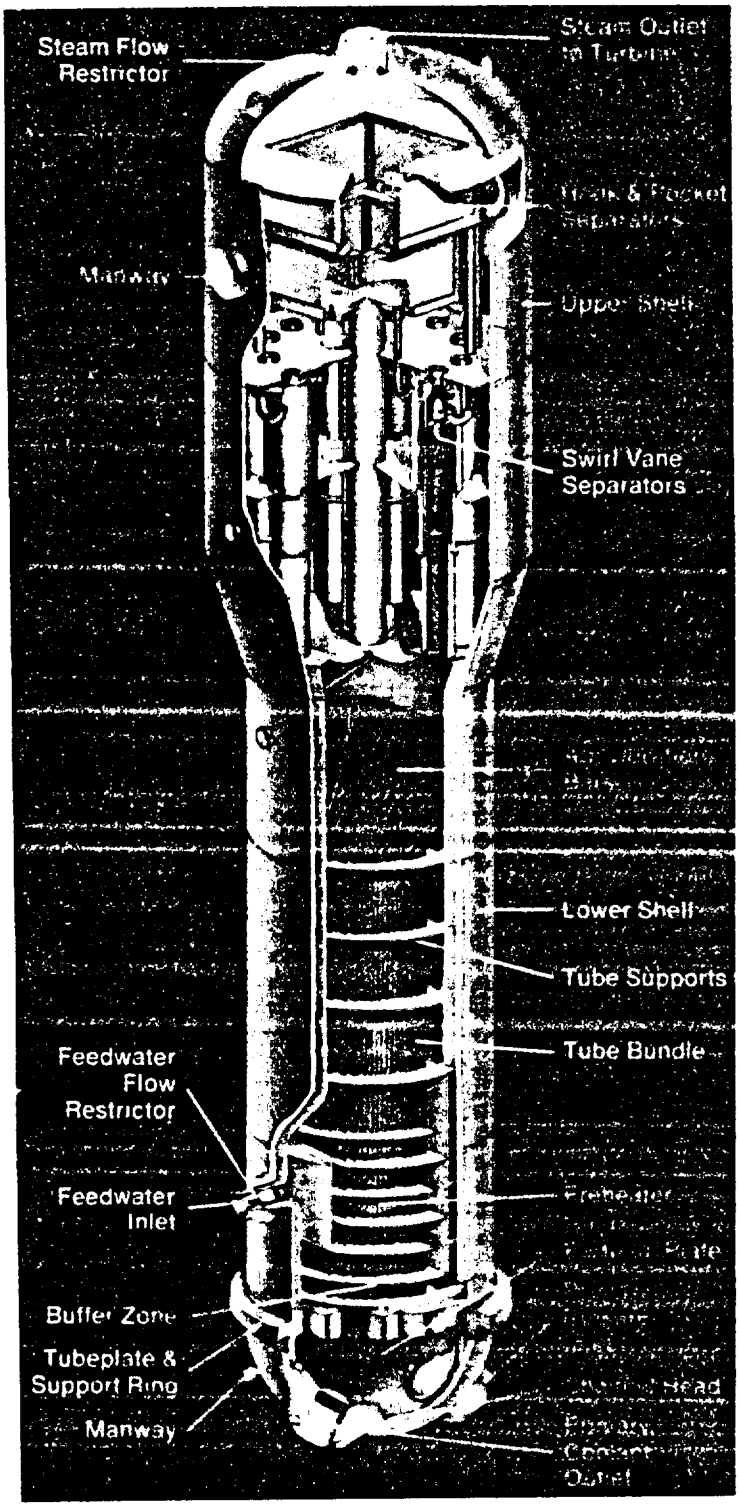


FIGURE 9. A TYPICAL WESTINGHOUSE STEAM GENERATOR

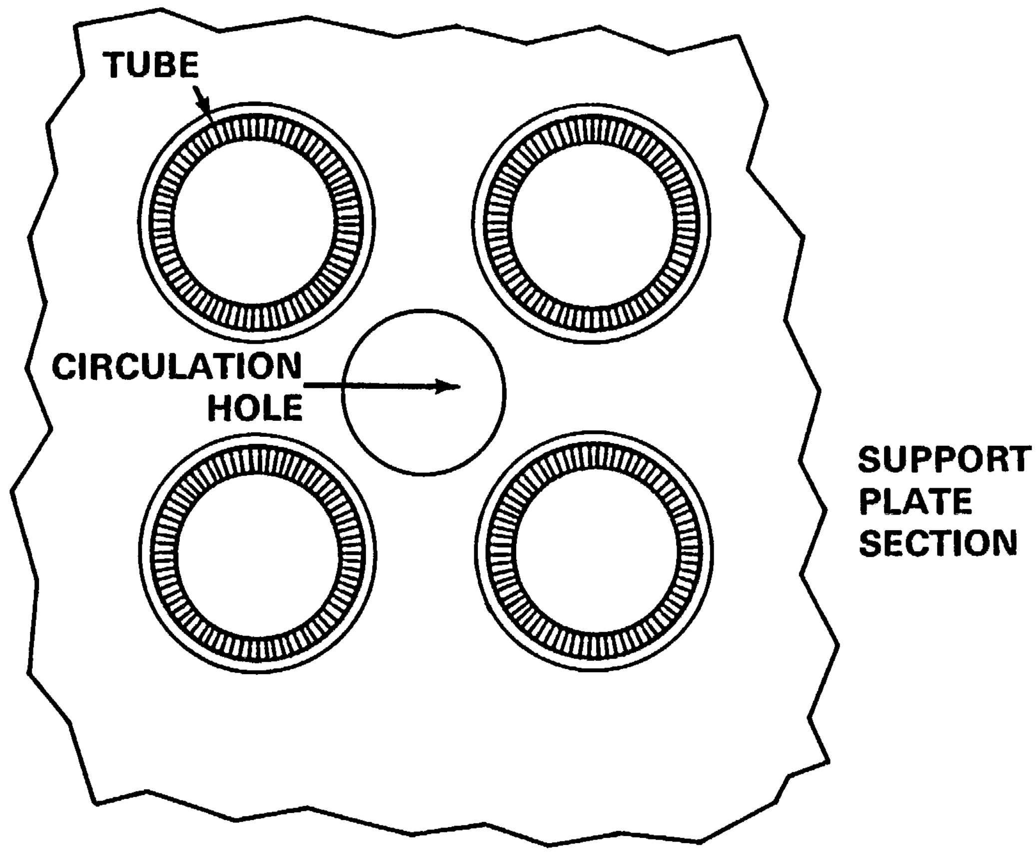


FIGURE 10. DRILLED TUBE SUPPORT DESIGN

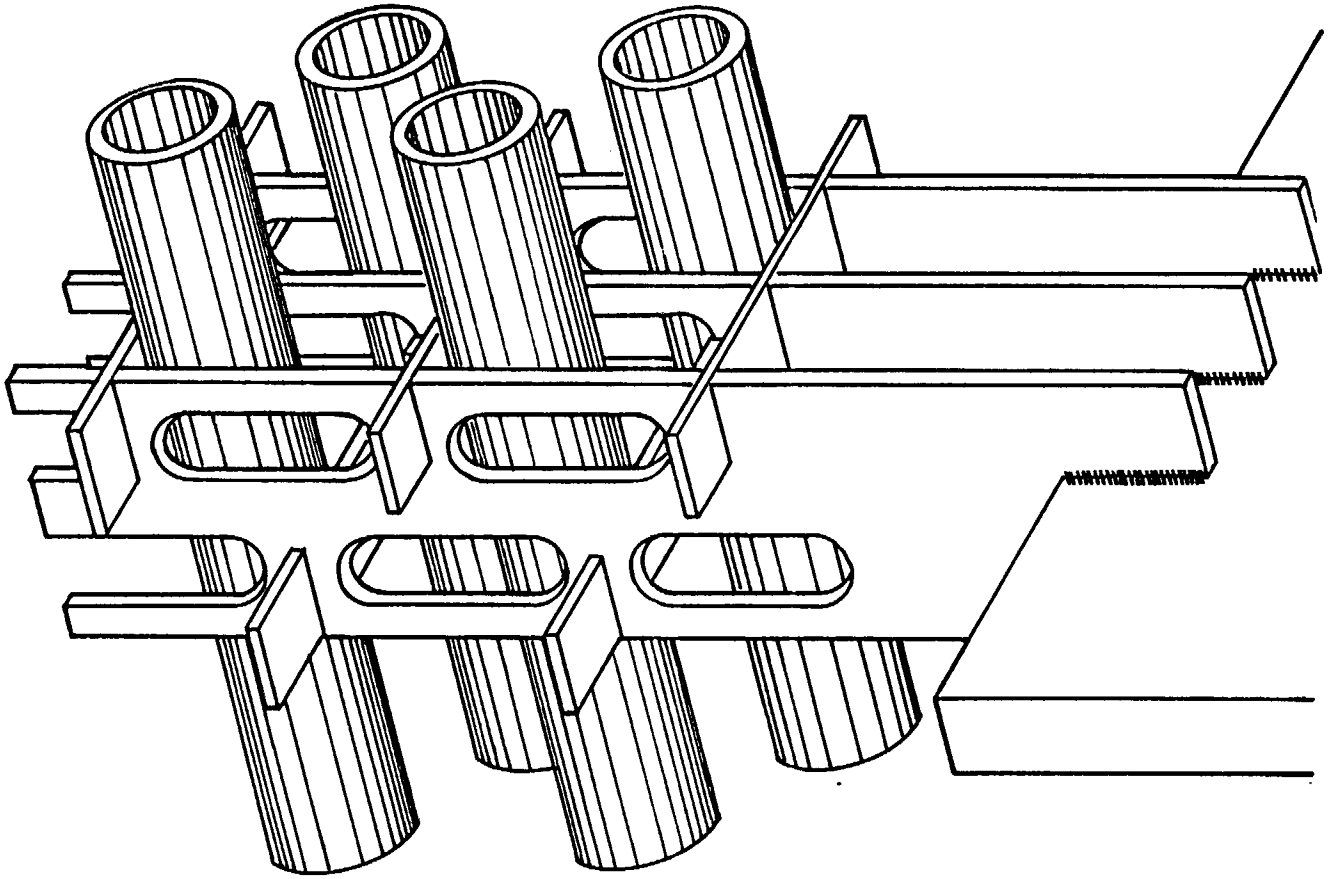


FIGURE 11. CE STEAM GENERATOR EGG CRATE TUBE SUPPORT PLATE DESIGN

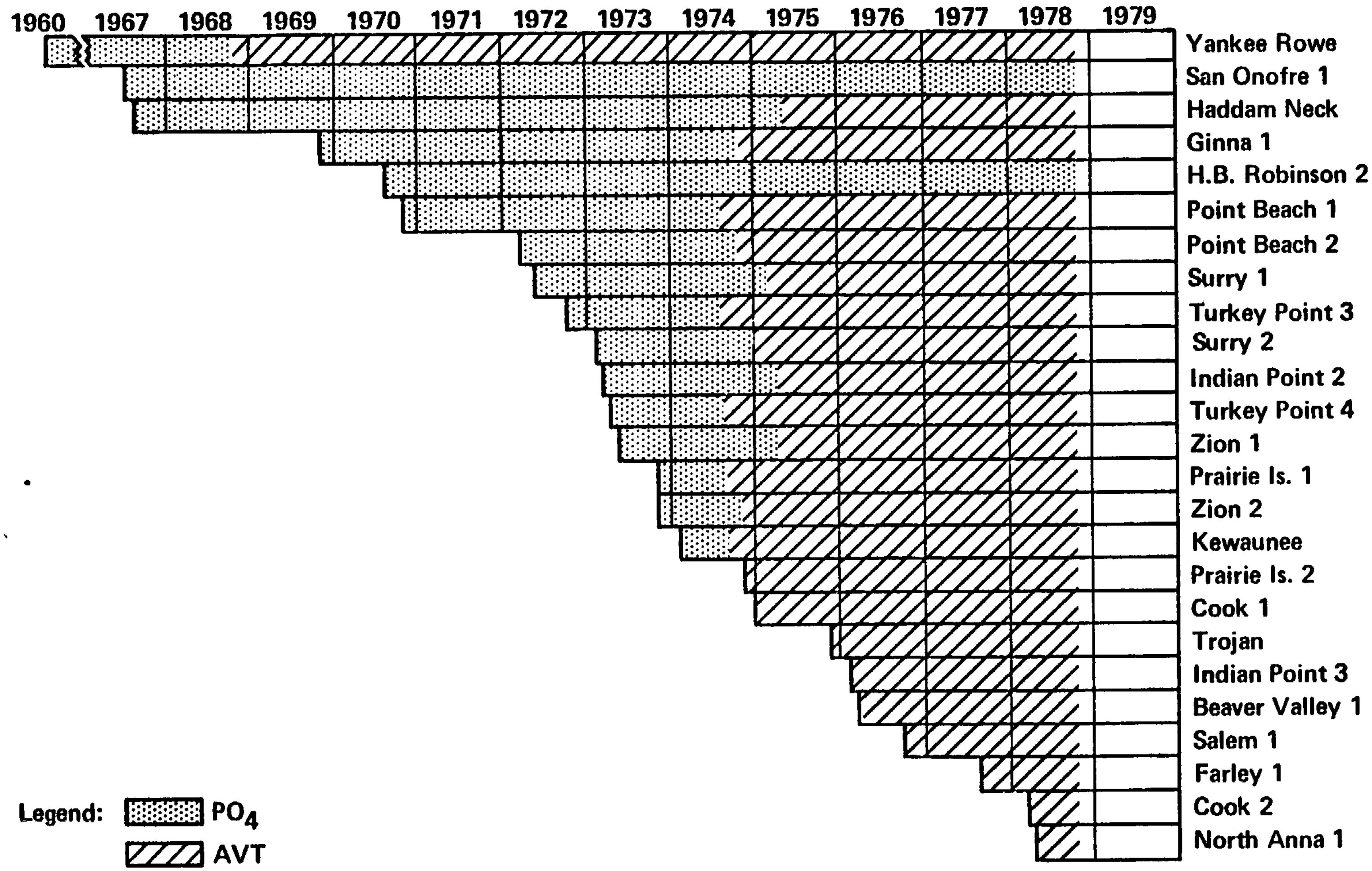


FIGURE 12. SUMMARY OF SECONDARY WATER CHEMISTRY TREATMENT IN OPERATING WESTINGHOUSE PLANTS

Turkey Point Units 3 and 4 and R. E. Ginna, whose steam generators had exhibited extensive thinning while on PO_4, experienced significantly reduced rates of wall thinnings following conversion. Improvements in technology in performing steam generator tube inspections using eddy current techniques (ECT) over the last few years have resulted in identification of an apparent increased rate of wall thinning.

The objective of using PO_4 control was to buffer inleakage of impurities from the condenser and to prevent formation of boiler scale on the steam generator tubes. Control of caustic level was also of concern. In fact, improper use of phosphate often leads to caustic stress corrosion. With the changeover to AVT control, caustic stress corrosion has remained a concern. Stress corrosion cracking in plants that converted from PO_4 to AVT control is related to previous PO_4 concentration and possibly to makeup water contamination. Plants with only short periods of PO_4 control before conversion and plants that initially started with AVT have not experienced operational problems due to tube wall thinning or caustic stress corrosion cracking.

A second significant effect of the conversion to AVT upon wastage has occurred due to a change in the character of steam generator sludge deposits. In steam generators using phosphate, the sludge is coarse, granular material that forms a cohesive mass on the tube sheet. Plants that converted after a short period of phosphate treatment have exhibited a finely divided sludge of dense particles that are more easily removed by water-lancing procedures. The sludge is similar in metal composition to the phosphated sludges because the iron impurities in the feedwater are unchanged. The improved ability to remove the "AVT sludge" should help to minimize wastage of steam generator tubes because the wastage is most severe within areas of sludge deposits.

3.1.2 Denting and U-Bend Cracking

Denting is caused by the buildup of corrosion products in the crevices between the tubes and tube support plates or between tubes and the tube sheet. The corrosion products, which are primarily derived from the carbon steel support plate and consist mainly of iron magnetite (Fe_3O_4), expand volumetrically (about 2:1) to fill the crevice and, therefore, exert forces on the tubes and on the tube support plates. Phenomena directly associated with denting include the following:

Tube diameter reduction
Tube leakage
Tube support plate hole distortion.
Tube support plate flow hole distortion (flow slot hourglassing)
Tube support plate expansion and cracking
Wrapper distortion

Denting has resulted in greater than about 0.25 inches reduction in tube diameter in the most severely affected units (3/4- and 7/8-inch tubes). The reduction in tube diameter is generally not concentric. This is dramatically illustrated in Figure 7, which shows the cross section of a dented tube removed from an operating facility. Areas of high tensile stress on either outside or inside surfaces in dented tubes are susceptible to stress corrosion cracking, and small leaks have occurred in steam generators with severe denting.

Areas of the tube support plates located near the edges of flow slots and the support plate periphery, which do not have flow holes, are stiffer than the rest of the plate. These areas, termed "hard spots," have experienced more severe denting than other regions of the steam generator. These areas are shown in Figure 13. In less-stiff areas of the support plate, it is somewhat easier for the plates to deform than it is for the tubes to be dented. Distortion of the flow holes, flow slots, and plate periphery have occurred as the volume of the corrosion products and tube support plate increases. Figure 4 shows the hourglassing (see Section 2.2) type of deformation of the support plate in the flow slot area. In some instances, extreme support plate deformation has resulted in cracking of the support plates. Figure 5 is a photograph of cracking that occurs at the flow slot edge. Cracking of the support plate behind the first row of tubes has also been observed. This form of cracking (also known as islanding, see Secton 2.3) causes a portion of the support plate and the tubes contained in it to move into the slot (see Figure 8).

The denting phenomena are believed to be directly related to the secondary water treatment history of a plant. Plants that converted to AVT after extended use of phosphate water treatment have had severe denting. However, recent experience indicates that ingress of chlorides through condenser leaks may be a significant contributor to denting. Thus far, denting has not been significant at plants with low chloride in the condenser cooling water although denting has been found to occur in plants with very low amounts of chlorides. The most severe cases are plants with brackish or seawater condenser cooling. Table 3 lists plants discussed herein and their types of condenser cooling water and condenser tube material.

Except for Indian Point Unit 3, denting has not been reported in Westinghouse plants with all AVT or limited phosphate history. The extent of denting of Indian Point Unit 3 is minor with average denting of about 3 to 4 mils. The success of AVT can be attributed to the close control of chloride ingress. Westinghouse AVT chemistry specifications establish strict chemistry guidelines regarding cation conductivity and chloride levels. When condenser problems occur, the AVT chemistry guidelines are exceeded and plants that implement timely corrective actions have avoided severe denting problems.

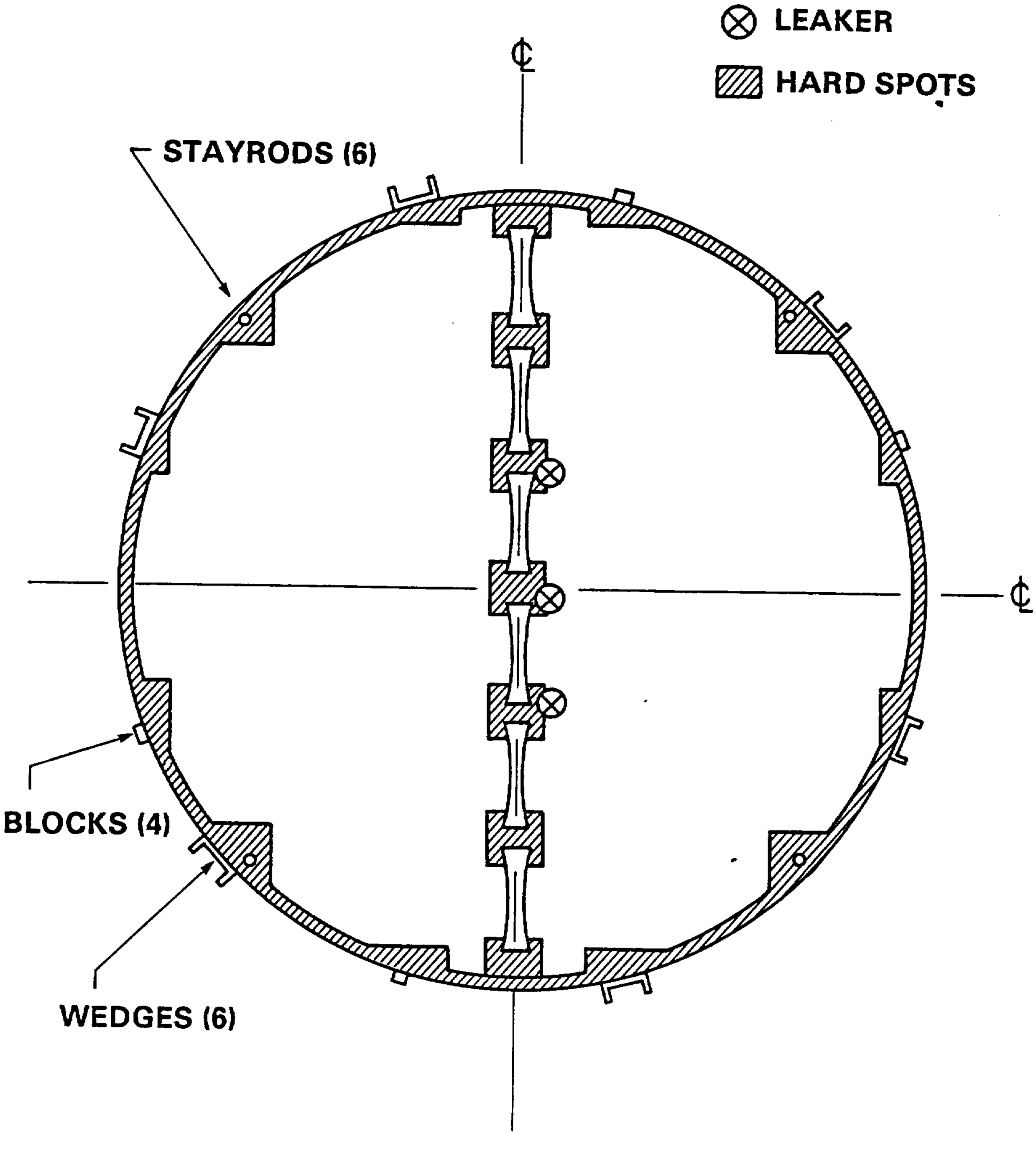


FIGURE 13. TYPICAL TUBE SUPPORT PLATE HARD SPOTS

TABLE 3
DENTING IN WESTINGHOUSE STEAM GENERATORS

PLANT	EXTENT OF DENTING*	CONDENSER TUBE MATERIAL	COOLING WATER
Surry Units 1 & 2	Extensive	90 Cu 10 Ni	Brackish
Turkey Point 3 & 4	Extensive	Al-brass	Seawater
San Onofre Unit 1	Extensive (stabilized)	2 boxes-titanium 2 boxes-90-10 CuNi	Seawater
Point Beach Units 1 & 2	Moderate	Admiralty with stainless steel impingement area	Fresh water (lake)
Indian Point Unit 2	Moderate	Admiralty	Brackish
R.E. Ginna Unit 1	Minor	Admiralty with stainless steel impingement area	Fresh water (lake)
Connecticut Yankee	Minor	Admiralty and stainless steel in impingement area	Fresh water (river)
H.B. Robinson 2	Minor	Admiralty and stainless steel in impingement area	Fresh water (lake)

*See Glossary at end of report for definitions of terms.

Turkey Point Units 3 and 4 and Surry Units 1 and 2 began commercial operation from mid-1972 to mid-1973. Like almost all units with U-tube design steam generators, these units began operation using a sodium phosphate secondary water chemistry treatment. This treatment was designed to remove precipitated or suspended solids by blowdown and was successful as a scale inhibitor. However, during early use, many PWR U-tubed steam generators with Inconel-600 tubing experienced stress corrosion cracking. The cracking was attributed to free caustic that can be formed when the Na/PO_4 ratio exceeds the recommended limit of 2.6. In addition, some of the insoluble metallic phosphates, formed by the reaction of sodium phosphates with the dissolved solids in the feedwater, were not adequately removed by blowdown. These precipitated phosphates tended to accumulate as sludge on the tube sheet and tube supports at the central portion of the tube bundle where restricted water flow and high heat flux occur. Phosphate concentration (hideout) at crevices in areas of the steam generator, noted above, caused localized wastage resulting in thinning of the tube wall. The problem of stress corrosion cracking was corrected by maintaining the Na/PO_4 ratio between 2.6 and 2.3. Although the recommended Na/PO_4 ratio was maintained in some units, it did not correct the phosphate hideout problem that caused wastage of the Inconel-600. Largely to correct the wastage and caustic stress corrosion cracking encountered with the phosphate treatment, most PWRs with a U-tube designed steam generator using a phosphate treatment for the secondary coolant have converted to an all-volatile chemistry. Surry Units 1 and 2 and Turkey Point Units 3 and 4 converted to AVT in middle to late 1974.

In 1975, deformation, or the so-called "denting," of steam generator tubes occurred in several PWR facilities, including Surry Units 1 and 2 and Turkey Point Units 3 and 4, after 4 to 14 months operation. This occurred after the conversion from a sodium phosphate treatment to an AVT chemistry for the steam generator secondary coolant. Tube denting occurs predominantly in rigid regions, or so-called "hard spots," in the tube support plates. These hard spots are located in the tube lanes between the six rectangular flow slots in the support plates near the center of the tube bundle and around the peripheral locations of the support plate where the plate is wedged to the wrapper and shell. The hard-spot areas do not contain the array of water circulation holes found elsewhere in the support plates.

The Surry Units 1 and 2 have experienced severe denting and support plate deformation throughout their steam generators. On September 15, 1976, during normal operation, one U-tube in steam generator "A" at Surry Unit 2 suddenly developed a primary-to-secondary leak of about 80 gpm. Subsequent investigations revealed that the leak resulted from an axial crack, approximately 4-1/4 inches in length, in the U-bend apex of an inner-row tube. It was also established that the crack initiated from the primary side of the tubing. Hourglassing of the flow slots in the upper tube support plate "pulled the legs" of the U-bend closer together thereby causing higher stresses in the tube material in the "U" area, which resulted in stress corrosion cracking. This ovalization phenomenon is shown in Figure 6. As a result of the event, the innermost row of tubes was removed from service by plugging. This action was taken in all Surry Unit 2 steam generators that exhibited a large degree of hourglassing in the upper support plate flow slots. The potential for dent-related cracking (at U-bends and support plates) has necessitated the preventive plugging of over 20 percent of the tubes in the Surry Units 1 and 2 steam generators (as of this report). The approach of removing the innermost row of tubes from service by plugging because of hourglassing has also been used at other facilities.

The Turkey Point Units 3 and 4 have also experienced severe denting-related phenomena throughout their steam generators and have plugged approximately 12 and 17 percent respectively of their tubes as part of their preventive plugging programs (as of this report). At San Onofre Unit 1, only two of the three steam generators have been affected to date and only in the hot leg of the lowest two (of a total of four) support plates. No significant denting has been detected in the third steam generator. In addition, it appears that the denting phenomena have slowed down in the San Onofre facility. Other Westinghouse units that have observed various levels of denting include R. E. Ginna, Indian Point Units 2 and 3, Point Beach Units 1 and 2, Haddam Neck, and H. B. Robinson Unit 2.

3.1.3 Tube Support Plate Cracking

Excessive deformation as a result of continued magnetite growth has resulted in cracking of the tube support plates at Surry Units 1 and 2, Turkey Point Unit 4, San Onofre Unit 1, and Indian Point Unit 2.

At Surry Units 1 and 2 and Turkey Point Unit 4, portions of the support plate have moved, along with tubes, into the flow slots. This phenomenon termed islanding (see Sections 2.3 and 3.1.2) could potentially lead to the possible loss of lateral support of some inner-row tubes. Concern for this lack of support was eliminated by preventive plugging of many of the tubes in the inner rows.

Support plate cracking has also been observed at San Onofre Unit 1. Deformation, hourglassing, and cracking was found in the bottom two tube support plates in steam generators "A" and "C" by inspections. During the April 1978 inspection, no evidence of deformation, hourglassing, or cracking existed in the upper two support plates in steam generators "A" and "C" or any of the four support plates in steam generator "B". Physical measurements of the three flow slots nearest the upper-hand hole entry were made in steam generator "C". These measurements indicated that the flow slots in the top support plate did not deviate from their manufactured condition.

Measurements of the flow slots in the lower support plates in steam generators "A" and "C" were not made. Results of the latest steam generator inspection, completed in October 1978, indicated that tube denting and support plate cracking may have stabilized in the San Onofre Unit 1 steam generator.

During a recent inspection at Indian Point Unit 2, cracks were found in the third flow slot from the manway side in the second tube support plate in one steam generator. The cracks are located in the ligaments between the flow slot and first-row tube holes near the center of the flow slot. Figure 5 is a photograph of such support plate cracking. In addition, it was discovered that a tube support plate is in contact with the wrapper in that steam generator. During the inspection, the licensee also removed a section of the lowest tube support plate from another steam generator. The sample was removed as part of a chemical cleaning feasibility study at Indian Point Unit 2. The support plate section contained the first two rows of tubes in columns 3 through 13 occupying an area approximately 14 by 5 inches. The section was cut out using electrical discharge machining (EDM) and was removed through a 6-inch-diameter hand hole below the lowest support plates. While the specimen was being removed from the steam generator, parts of the support plate and 6 of the 22 tubes broke loose from the specimen. Measurements of the flow hole elongation between the first and second rows and the second and third rows account for virtually all the hot leg flow slot hourglassing.

3.1.4 Anti-Vibration Bar Wear or Fretting

A few plants originally equipped with round, carbon steel anti-vibration bars (AVBs) have experienced varying degrees of tube wear at the AVB locations. The cause appears to be primarily mechanical and is dependent on the material (that is, carbon steel bars), the shape of the bars, the clearances, and the bar support design.

The most severely affected plant is San Onofre Unit 1 where, between April 1975 and October 1976, a large number of tubes exhibited a substantial increase in wear rate. An additional AVB array using square, chromium-plated Inconel bars on unworn surfaces near the original set of AVBs was installed in all three steam generators. Severely degraded tubes were removed from service by plugging.

Connecticut Yankee, and a foreign plant with similar AVB design, have also experienced tube wear. However, the extent of the wear has been small and only a few tubes have been removed from service at each of these facilities.

3.2 Plants Designed by Combustion Engineering

All commercial operating Combustion Engineering (CE) steam generators are of the vertical shell type with recirculating Inconel-600 U-bend tubing and integral steam separation equipment (see Figure 14). The CE design

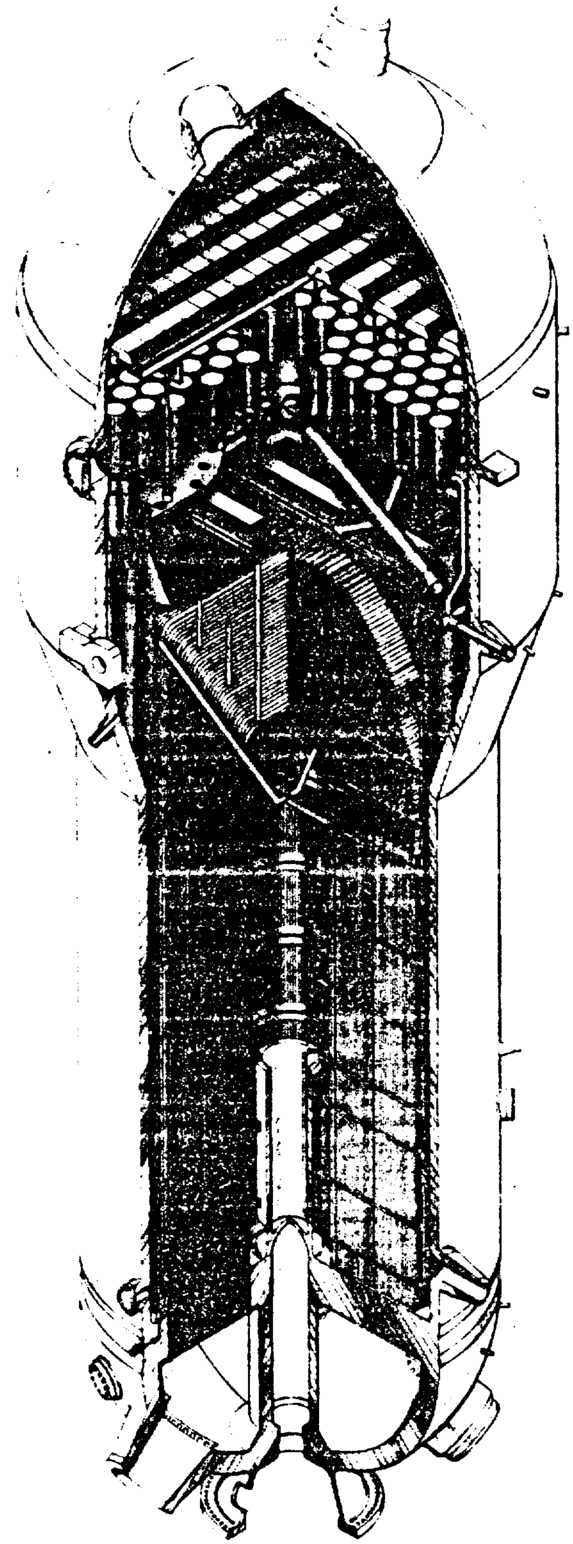


FIGURE 14. A TYPICAL COMBUSTION ENGINEERING STEAM GENERATOR

has a combination of drilled carbon steel partial support plates similar to the Westinghouse design but without flow holes (Figure 10) and carbon steel "egg crates" (Figure 11) for tube supports. With the exception of the Palisades Plant, the drilled plates are two partial plates located near the top of the tube bundle.

The methods used for secondary steam generator water treatment for CE-designed operating plants are shown in Figure 15. Palisades began operation with a phosphate treatment for the secondary water and converted to AVT to arrest the tube wastage problem encountered with phosphate. All other facilities utilized AVT chemistry for the secondary coolant from the beginning of operation. (Fort Calhoun operated a short time on phosphate control prior to commercial operation.)

3.2.1 Caustic Stress Corrosion and Wastage

Palisades conversion from phosphate to AVT secondary water chemistry control to correct the tube thinning or wastage problem seems to have eradicated the major occurrences of wastage-type tube degradation. Table 1 summarizes the present status of wastage experienced with CE steam generators.

3.2.2 Denting

Four Combustion Engineering plants are now known to have experienced some degree of "denting" (Table 4). Palisades, which uses water from Lake Michigan in a closed-cycle cooling system with cooling towers, used phosphate treatment prior to AVT. Operation at the other three plants has been totally on AVT. These plants include Maine Yankee, Millstone Unit 2, and St. Lucie. Most of the tube supports in these units are of the egg-crate design, with only two partial support plates of the drilled-hole design in each steam generator. The Maine Yankee plant has 90-10 CuNi condenser tubes, whereas Millstone Unit 2 has Al-brass. The main condenser at Millstone Unit 2 has experienced operational difficulties and has been retubed. The denting at Maine Yankee is limited to the drilled-hole partial support plates. In addition, inspection at Millstone in late 1977 showed some dent-like signals within the egg-crate region. It is unclear what phenomenon is occurring at these locations.

The Maine Yankee plant became operational in October 1972. In July 1973, steam generator 3 was inspected and only a few tubes in the partially drilled support plate were examined at that time with no indication of tube degradation. In July 1974, steam generators 1 and 2 were inspected, with the same negative result. In June 1975, steam generator 3 was inspected for the second time. No degradation of any kind was seen, although only a few tubes in the partial support plate were examined. At that time, sludge was present on the tube sheet, up to 3-1/2 inches deep. In May

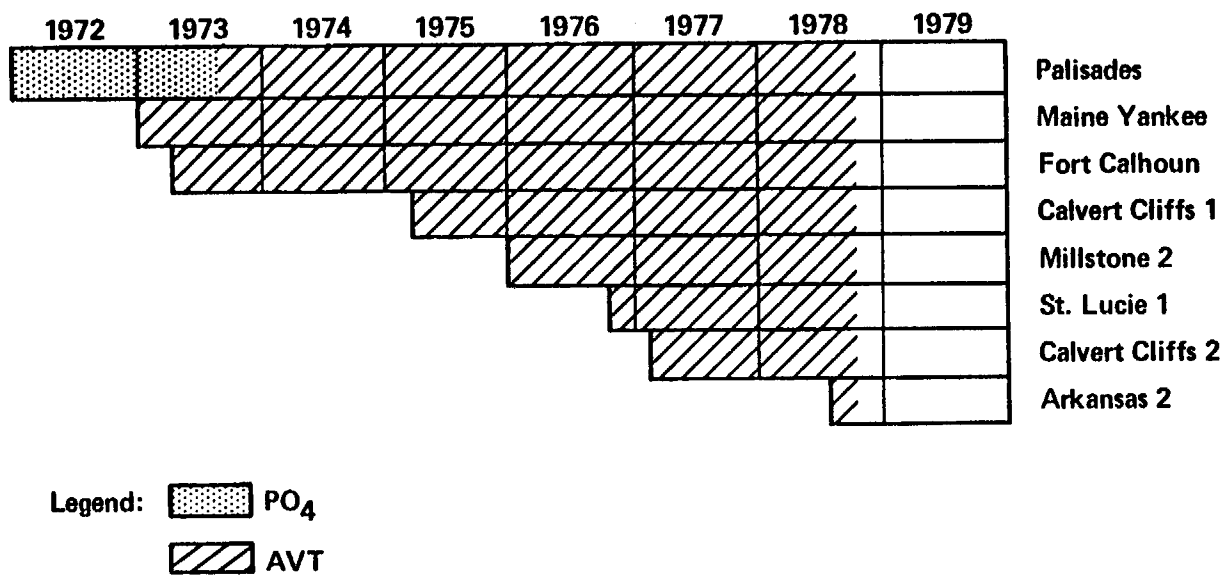


FIGURE 15. SUMMARY OF SECONDARY WATER CHEMISTRY TREATMENT IN OPERATING COMBUSTION ENGINEERING PLANTS

TABLE 4
DENTING IN CE STEAM GENERATORS

PLANT	EXTENT OF DENTING*	CONDENSER TUBE MATERIAL	COOLING WATER
Palisades	Minor	Admiralty changed to 90/10 Cu-Ni	Closed-cycle cooling (lake water)
Maine Yankee	Minor	Al-brass and some 70/30 CuNi	Brackish water
St. Lucie 1	Minor	Al-brass (70/30 Cu-Ni in Air Removal Section)	Seawater
Millstone 2	Moderate	Al-brass changed to 90/10 Cu-Ni	Seawater

*See Glossary at end of report for definitions of terms.

1977, steam generators 1 and 2 were reexamined, with relatively shallow dents found in the drilled partial support plates, averaging about 1 mil. Sludge on the tube sheet had reached a maximum depth of 4 to 4-1/2 inches over a few tubes with a little more occurring in the hot leg than in the cold leg, but no sludge was found on the drilled support plates. The partial drilled-hole support plates are at the seventh and ninth support levels with a total of about 30 percent of the tubes passing through one or the other or both of them. Radial clearance in the drilled holes is 8 mils by design.

In addition, chloride (Cl^-) and cation conductivity have been monitored at the condensate at Maine Yankee and Cl^- is checked daily in the steam generators. Minor condenser leaks have occurred about once a month. When Cl^- has exceeded 0.05 ppm in the condensate, power has been reduced, and the leak has been located and fixed in 1 to 1-1/2 days. Therefore, secondary water quality has generally been controlled and maintained, although 1 ppm Cl^- in the blowdown was exceeded about 60 times during the operating lifetime, and, on a few occasions early in operation, rose to some tens of ppm's for short periods. On the other hand, the experience at Maine Yankee is believed significant because denting occurred in spite of significant efforts to keep the secondary water relatively free of impurities. However, denting at Maine Yankee after 53 months of operation is less than at Millstone Unit 2 after 23 months of operation where condenser problems are more severe.

At Millstone Unit 2, the initial denting in the drilled plates was moderate and widespread, measuring 7 to 13 mils, but no leaks were found in the steam generator. The plant has had a relatively high level of Cl^- in the secondary water, because of almost continuous low-level inleakage through the condensers. The Millstone Unit 2 plant became operational in August 1975. The aluminum-brass (Al-brass) main condenser has since been retubed, because of condenser problems. A minimum of 0.15 ppm Cl^- has been reported, with a typical blowdown analysis in 1976 of Na = 0.2 ppm, Cl^- = 0.6 ppm, and conductivity = 6 μmho/cm. Contaminants were only a little less in the first part of 1977. Since startup, Cl^- has exceeded 1 ppm on 76 days.

Approximately 2,200 tubes in Millstone Unit 2, which pass through the two top support plates, were eddy current tested (ECT) in each steam generator with a 0.540-inch-diameter probe at 400 Hz. Essentially all the tubes inspected in both steam generators had dent indications at each of the drilled support plates. The majority of tubes not passing the 0.540-inch ECT probe are in regions of the support plate periphery adjacent to the lugs supporting the plates. Dent indications (1.0 mil) at tube/egg-crate intersections were observed in approximately 70 percent of the tubes inspected in both steam generators.

St. Lucie Unit 1 has conducted steam generator inspections during their first refueling outage. The unit, which began commercial operation in December 1976, has CE steam generators similar in design to Millstone Unit 2 and has operated continually with AVT secondary water chemistry. The inspection program to date included 380 hot leg tubes and 110 cold leg tubes in one steam generator. Preliminary results indicated 55 tubes with dent signals after one fuel cycle of operation. The average dent magnitude is in the range of 1 to 2 mils with the maximum dent magnitude observed being 4 mils.

3.2.3 Tube Support Plate Cracking

At Millstone Unit 2, tube denting has caused the two top partial support plates in both steam generators to expand against the "hard spots" at supporting lugs on the tube bundle shroud. The stresses induced by the expanding support plates has caused cracking of the ligaments between the tube holes and circulation flow holes in corner areas of the uppermost support plate along the outer band of tubes adjacent to the rim of solid metal at the outer periphery of the plates. Shear stresses have caused cracking along the inner boundary of the solid rim section and shifting at the corner areas of the plate. This produced a shearing action on the tubes and deformed the tube wall of about 20 outer peripheral tubes located in the corner areas of the plate.

The tube support plate ligament cracks observed at Millstone Unit 2 resulted from high support plate strains. These strains are the result of corrosion product growth in the annular clearance between the tubes and tube holes within the plate. The aggregate tube ligament strains are relieved by plate expansion within certain limits. At Millstone Unit 2, plate expansion was constrained by solid unyielding attachments or wedges between the plate and steam generator tube bundle shroud. As a result, the ligaments and tubes became deformed and the plate cracked to relieve the stress.

4. CORRECTIVE ACTIONS AND REPAIRS

4.1 Short-Term Program and Licensing Requirements

Increased steam generator tube inservice inspection (ISI) frequency, preventive tube plugging, and more stringent technical specifications limiting operation with steam generator tube leakage have formed the bases for continued safe operation of degraded steam generators. Other requirements, such as reduced primary coolant radioactivity limits, have also been required for severely degraded steam generator plants.

The Standard Technical Specifications for Westinghouse and CE facilities require a steam generator ISI every 12 to 24 calendar months unless two consecutive ISIs indicate good condition of the steam generators, in which case the interval may be increased to 40 months. In the event that steam generator degradation is observed, the inspection frequency must return to the original 12- to 24-month schedule. In the event of severe steam generator degradation, the NRC has required plants to perform ISI more frequently.

Tube-plugging criteria addressing wastage types of degradation are routinely included in plant technical specifications. These plugging criteria are based on the guidelines in Regulatory Guide 1.121 and are designed to ensure the integrity of degraded tubes during normal or accident conditions. Plugging criteria for tube denting is not specifically addressed in the current revision of Regulatory Guide 1.121 and is more difficult to establish. Dented tubes are susceptible to stress corrosion cracking (SCC), which is dependent on stress level, time, and environment. Tests have shown that dented tubes with small through-wall cracks near the support plate have adequate margins against tube burst or collapse under normal operation, transients, and postulated accidents. Severe SCC could, however, reduce the margins to an unacceptable level. Therefore, tube-plugging criteria for SCC of severely dented tubes must be based on the magnitude of denting (stress level), operating time, and the rate of degradation. The objective of the tube-plugging criteria is to remove from service any tubes that might develop through-wall cracks or become severely degraded before the next ISI. (The present extent of the tube plugging for Westinghouse and CE plants is summarized in Table 2.)

The continued integrity of the steam generator tubing is further monitored by a reactor primary coolant-to-secondary system leak rate limit in plant technical specifications. The technical specification leak rate limit corresponds to a leak rate from a sufficiently small defect so that it will not result in a sudden tube rupture under design basis accidents. For units with severely degraded steam generators, this limit has been reduced to as low as 0.3 gpm. Following an occurrence of leakage of 0.3 gpm or greater at such facilities, the facility is required to shut down, to plug the leaking tube, often to conduct a steam generator tube inspection, and occasionally to request NRC approval prior to restart. This approach helps to ensure that severely degraded tubes (that is, even to the point of leakage) are removed from service and that other tubes are inspected to detect any other continued degradation.

4.1.1 Turkey Point Units 3 and 4 and Surry Units 1 and 2

Operation of the Turkey Point Units 3 and 4 and Surry Units 1 and 2 is being closely regulated by the NRC. These units have been conducting inspections about every six months in order to carefully monitor the rate of steam generator tube degradation. Each inservice inspection includes eddy current inspections, tube gauging, and support plate examinations. The numbers and locations of tubes to be gauged are established using a finite element computer model of the tube support plates. As a result of each inspecton, tubes that may be susceptible to SCC and that may begin to leak prior to the next ISI are plugged. The typical plugging criteria outlined below are based on operational experience and are essentially the same for these four units.

1. All tubes that do not pass the 0.540-inch ECT probe will be plugged (nominal inside diameter of virgin tube is 0.770 inch).

2. Additionally, if attempting to justify operation for six months or longer on a severely degraded facility, two tubes beyond (that is, higher row numbers) any tube in columns 15-79 that does not pass the 0.540-inch probe will be plugged in the tube-lane region; for such tubes in column 1-14 and 80-94, five tubes beyond will be plugged on the hot leg side and four tubes beyond will be plugged on the cold leg side in the tube-lane region (see Figure 16 for column and row numbering).

3. All tubes that do not pass the 0.610-inch probe will be plugged. No surrounding tubes are plugged by this step.

4. Those tubes in any column, for which plugging under criteria 1, 2, or 3 above is implemented in the tube-lane region, will also be plugged in the lower-numbered row of tubes back to the tube lane if not already plugged.

5. As a conservative measure, all tubes immediately surrounding any known leaky tubes, including the diagonally adjacent tubes, will be plugged if they are not already covered by the foregoing criteria.

6. In any given column that is surrounded by columns containing tubes with significant tube restriction or prior plugging

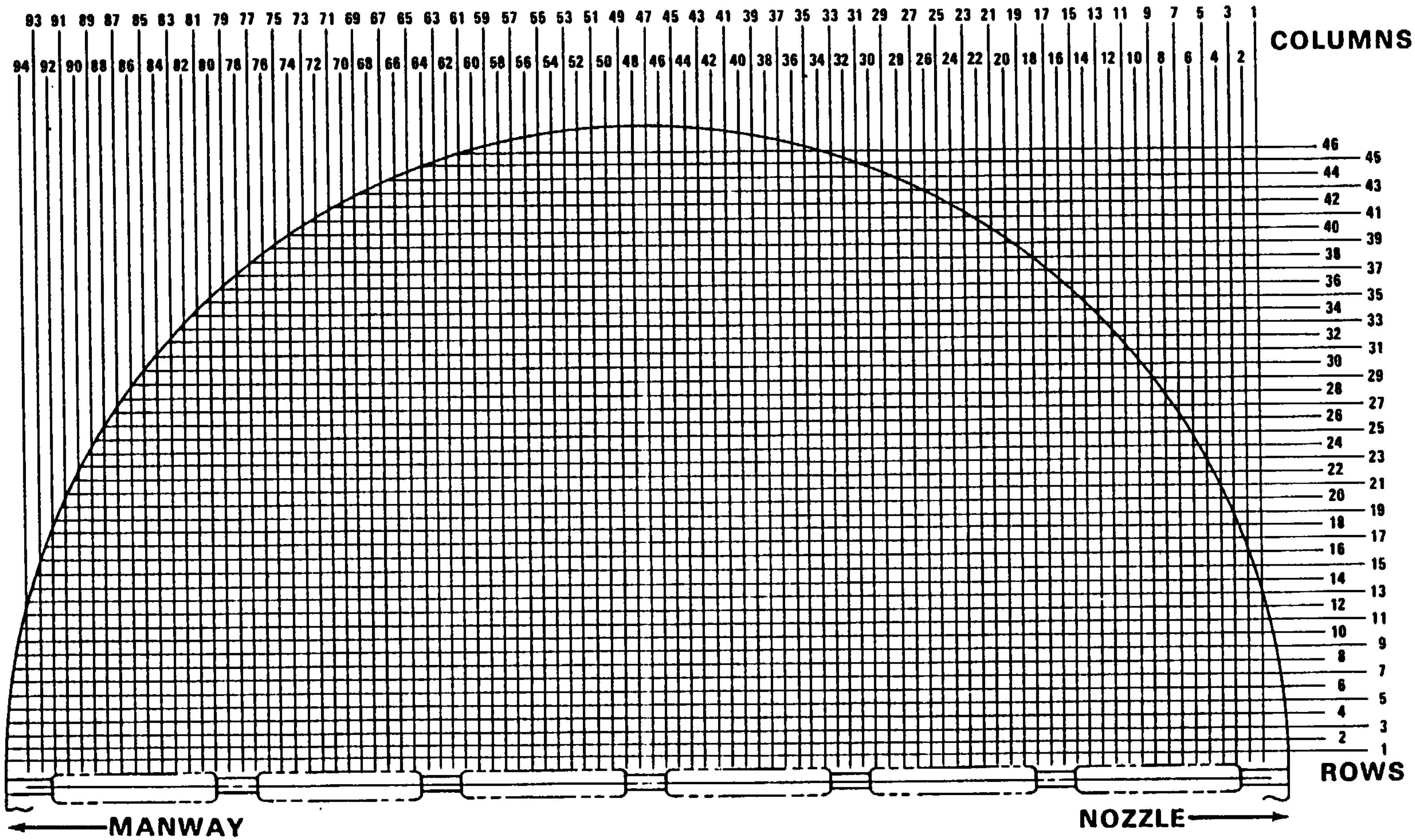


FIGURE 16. CROSS SECTION OF STEAM GENERATOR TUBE ARRAY

(thereby creating a "plugging valley" in the pattern), engineering judgment will be used to fill the bottom of the valley. In the peripheral tube-lane areas near the three and nine o'clock wedges, tubes surrounded by previously plugged tubes or tubes exhibiting high deformation activity will be plugged based on engineering judgment.

7. Additional preventive plugging will be implemented at the hot leg wedge locations. This plugging will include all tubes that:

   a. Restrict the 0.610-inch probe, or
   b. Restrict the 0.650-inch probe at the periphery, or
   c. Surround leakers and tubes that restrict the 0.540-inch probe including the diagonally adjacent tubes.

8. Application of the criteria specified in 7, above, will be made on the basis of engineering judgment for cold leg wedge locations.

9. Additional preventive plugging will be implemented in the patch-plate region. This plugging will include all tubes that:

   a. Restrict the 0.610-inch probe, or

   b. Surround leakers and tubes that restrict the 0.540-inch probe including the diagonally next tube, or

   c. Lie on either side of the patch plate boundary (plate perimeter on one side and plug welds on the other three) and restrict the 0.650-inch probe.

The above criteria indicate "hard spot" areas of the tube support plates where the tubes are more susceptible to denting. The exact column or row numbers to bound regions for tube plugging depend on the lateral support arrangement of the support plates.

In addition to the aforementioned conservative criteria for tube plugging in plants with severely degraded steam generator tubing, several other requirements are generally in place at these facilities. Typical other requirements include the following:

1. A technical specification requiring plant shutdown if a leaking dent exceeds 0.3 gpm in a steam generator. This requirement is intended to require plugging of the leaking tube and usually requires additional tube ISI.

2. A technical specification requiring plant shutdown for additional tube inspection if any two separate dented tubes are

found to leak in any 20-day period regardless of the leakage level of each tube. This requirement is intended to require an inspection to explore the rate of degradation in steam generators because, with the conservative plugging pattern, it is not expected to have two leakages in such a short period of time.

3. More restrictive limits have been incorporated in technical specifications to limit normal operation radioactivity levels in the primary coolant. This requirement is part of our defense-in-depth approach and is meant to limit the consequence of a hypothetical major loss-of-coolant accident if one postulated that major steam generator tube leakages occurred simultaneously with the accident.

4.1.2 Indian Point Unit 2

Indian Point Unit 2 is one of the six PWR facilities that were initially identified to have suffered steam generator tube denting and that have been under close monitoring since the latter part of 1976. Steam generator inspection conducted in February 1978 revealed minor but somewhat progressive denting. The 1978 inspection also revealed two cracks at a flow slot in the second support plate of one steam generator. The cracks had not been previously observed. In addition, it was observed that a tube support plate was in contact with the wrapper in the same generator. During the inspection, a section of the lowest tube support plate was removed as a part of a chemical cleaning feasibility study. While the specimen was being removed from the steam generator, parts of the support plate broke loose from the specimen. The inspection results indicated that active corrosion of the carbon steel support plate was continuing, but at a slower rate in comparison with other units.

The plugging criterion that was implemented consisted of plugging of any tube that would not pass a 610 mil or smaller ECT probe. In addition, the reactor coolant-to-secondary leakage limit was reduced to 0.3 gpm and the aforementioned "two leakages in 20 days" requirement was imposed. With these corrective actions and licensing conditions, 16 equivalent full-power months of operation were justified for Indian Point Unit 2.

4.1.3 San Onofre Unit 1

Photographs and videotapes taken in September 1977 and from inspections prior to that time showed cracking at the edge of the flow slots in the bottom two support plates of two of the three steam generators at San Onofre Unit 1. A steam generator ISI was therefore conducted during April 1978 to determine if the degradation was progressing. The results of this inspection established that no perceptible progression of denting or change in support plate condition had occurred between October 1976 and April 1978. Nine tubes that would not pass a 0.460-inch probe in April

1978 were plugged. Because the San Onofre steam generator tubes have a thicker wall and smaller diameter, this degree of denting corresponds to the same tube wall hoop strain as a Surry or Turkey Point steam generator tube dented to about a 0.500-inch inside diameter. The 0.3 gpm technical specification leak rate limit was also imposed. San Onofre Unit 1 steam generators were again inspected (ISI) during the scheduled September 1978 refueling. The results of this inspection did not alter the conclusions reached following the April 1978 inspection.

4.1.4 Millstone Unit 2

Millstone Unit 2, which has the most severe denting problem of all the CE plants, has performed extensive repairs to minimize the progression of support plate cracking and shifting and further tube damage. Approximately 80 percent of the plate constraint is attributable to the lugs supporting the plates. Analyses have shown that compressive and shear stresses associated with plate constraint would cause further cracking and shifting of the partial support plate. This condition would cause additional deformation of the peripheral tubes. The results of finite element analysis indicated that stresses in the plate adjacent to the rim would be reduced by removing the lugs and a portion of the peripheral plate rim adjacent to the tube bundle shroud. Therefore, the following modifications were made at the Millstone Unit 2 facility:

1. Removal of all lugs at each support plate and a portion of the peripheral solid rim in the uppermost plate to reduce "hard spots" and minimize the possibility of further cracking and shifting of the plates in each steam generator.

2. Preventive plugging of all peripheral tubes adjacent to the solid rim that have the greatest potential for failure, including additional tubes near the periphery in the corner regions of both support plates.

3. Plugging of all tubes not passing the 0.540-inch ECT probe and those surrounding the restricted tube.

4. Avoiding and minimizing unfavorable chemistry conditions.

5. Exclusion of seawater ingress by means of assuring condenser tube integrity (i.e., retubing the condenser with 90-10 CuNi) and a full-flow condensate polishing system, which will be available during cycle 2 operation.

In addition to the above tube-plugging pattern in items 2 and 3, the following preventive plugging was performed based on the critical tube hoop strain predicted by the finite element analysis of the tube support plate:

1. Any tubes that were damaged during the rim and support lug removal operation were plugged.

2. All tubes that lie along an apparent continuous series of ligament cracks in the plates were plugged.

3. All tubes not passing the 0.540-inch ECT probe and those surrounding the restricted tube were plugged.

Implementation of the plugging criteria resulted in plugging 290 tubes in steam generator 1 and 352 tubes in steam generator 2.

4.1.5 Other CE Facilities

Maine Yankee and Arkansas Unit 2 are other CE facilities that have removed all lugs at each drilled support plate and a portion of the peripheral solid rim in the uppermost plate to reduce "hard spots" and minimize the potential for cracking and shifting of the plates in each steam generator. At Arkansas Unit 2, these modifications were performed prior to initial startup. Other CE units, Calvert Cliffs Units 1 and 2 and Fort Calhoun Unit 1, have not experienced any form of tube degradation with an AVT chemistry. Recent inspections of St. Lucie Unit 1 steam generators revealed a buildup of corrosion products in the annulus between the tube and tube support plate, creating a potential for future denting. A proposal to chemically clean the St. Lucie Unit 1 steam generators to remove these corrosion products is being reviewed by the NRC.

4.2 Long-Term Repairs

4.2.1 Tube Sleeving

Combustion Engineering presently has under way a program to demonstrate the feasibility of installing sleeves as an alternate measure to tube plugging at the Palisades facility. The operation of the Palisades steam generators has, in the past, resulted in localized corrosive attack on the outside (secondary side) of the steam generator tubing. The reduction in steam generator tube wall thickness due to this corrosive attack may progress to the point of causing tubes to leak during operation. In addition, reduction in tube wall thickness may lessen the ability of the tube to continue to perform its function as a primary coolant pressure boundary during design accident conditions such as a loss-of-coolant accident (LOCA) or a main steam line break (MSLB).

Historically, the corrective action taken where steam generator tube wall degradation has been identified was to install welded plugs at the inlet and outlet of the steam generator tube when the reduction in wall thickness exceeded the plugging limit. This value of wall reduction requiring plugging was calculated such that adequate tube strength remained to prevent failures of the steam generator tubes during normal operation and postulated accident conditions.

Installation of tube plugs in a steam generator tube removes the heat transfer surface of the tube from service. The technique for installation of steam generator sleeves eliminates this negative aspect of steam generator tube plug installation. The sleeves are installed at the local area of tube wall reduction and impose only a minor restriction to primary coolant flow. Thus, while providing a corrective response to the weakening effect of tube wall reduction, the effects on heat transfer and primary coolant flow are minimized.

The steam generator sleeving concept consists of installing, inside the steam generator tube, a smaller diameter Inconel-600 tube to span the degraded area of the parent steam generator tube. This system is shown in Figure 17. Both ends of the inserted sleeve are hydraulically expanded into an interference fit with the parent tube. The rationale for installing the sleeves in this manner is to restore the mechanical strength of the degraded tube to a level adequate to prevent rupture during postulated accident conditions. By installing a sleeve to span the degraded area, the structural integrity of the tube is reestablished.

Although the sleeving process has been used only on a limited scale at the Palisades steam generators, the sleeving process may be applicable to all PWR generators. To qualify the sleeve for other applications, specific sizing and environmental conditions would have to be examined to ensure applicability.

4.2.2 Steam Generator Repair

Extensive preventive plugging as a result of continuing tube denting can cause excessive steam generator inspections and reduction in unit availability. For these reasons, Florida Power and Light Company (FPL) and Virginia Electric Power Company (VEPCO) are planning replacement of the lower portion of the steam generators at Turkey Point and Surry, respectively.

FPL and VEPCO are currently pursuing engineering and licensing activities to effect these steam generator repairs. Repair of the first such steam generator is tentatively scheduled to begin in early 1979. The existing steam generators are expected to be cut apart at the transition piece to the upper section of the shell. The upper section of the steam generator will then be stored inside the containment and joined to the new lower steam generator assembly, which will include the new tube bundle. The lower assemblies, including the old tube bundles, will exit the containment via the equipment hatch.

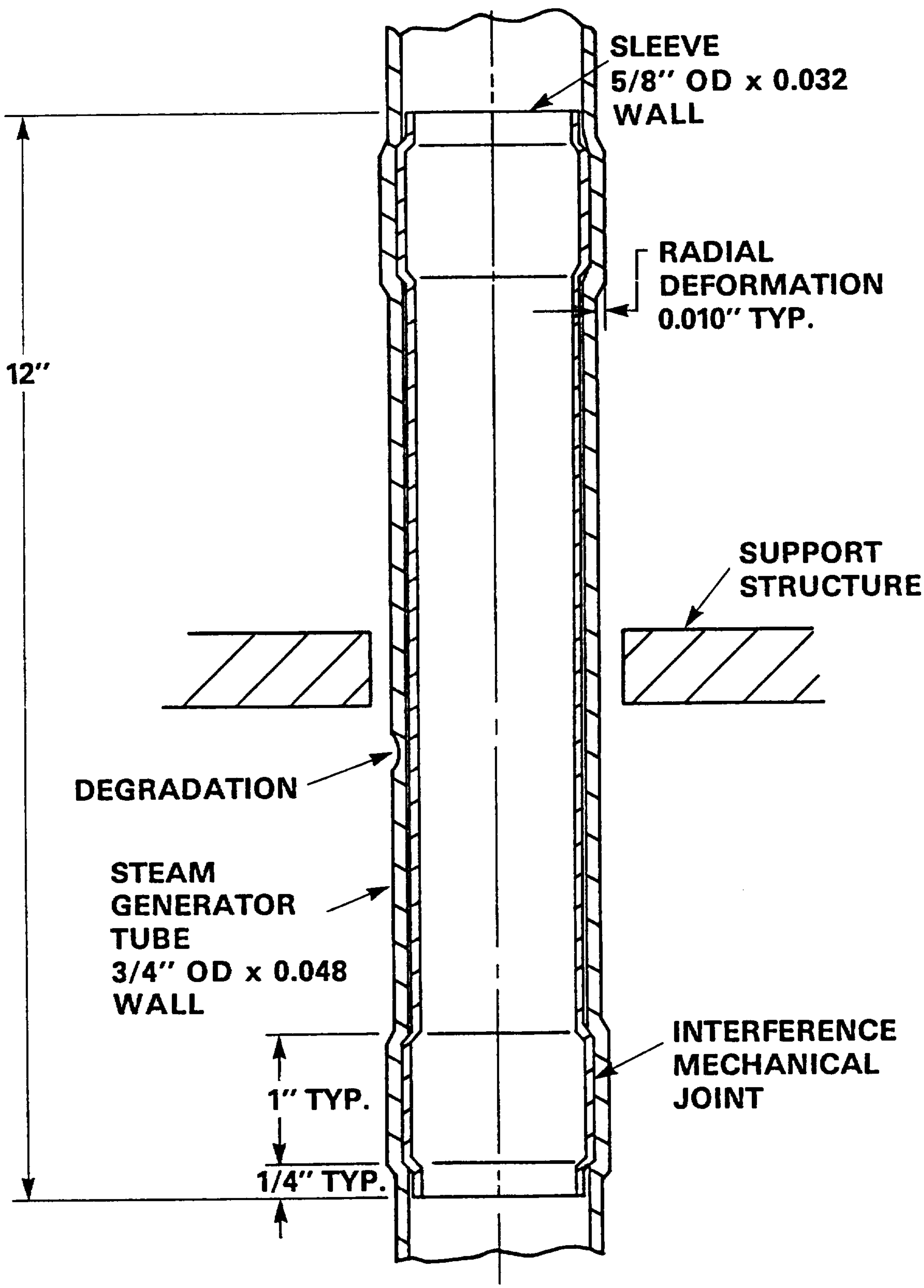


FIGURE 17. STEAM GENERATOR TUBE SLEEVE

Several secondary side design changes will likely be made in the repaired steam generators that will enhance their resistance to the previously discussed forms of degradation. A flow distribution baffle will be incorporated to improve flow velocities and circulation across the tube sheet. By directing flow to the blowdown pipe location, the effectiveness of sludge removal by blowdown would be increased. A new tube support plate design using a four-lobed broach tube hole, called a "Quatrefoil," will maximize flow along the tubes and reduce the susceptibility to corrosion product buildup and tube denting. The new plates are also fabricated from corrosion-resistant Type 405 ferritic stainless steel with Inconel-600 tubing being heat-treated to increase its resistance to stress corrosion cracking.

4.2.3 Condenser Integrity

Recent experience indicates that ingress of chlorides through condenser leaks is a principal contributor to the denting problems. Elimination of condenser leaks is, therefore, a primary concern in ensuring steam generator integrity. Improved integrity of rolled joints and selection of better materials will increase condenser reliability and reduce contaminant input to the steam generators. In particular, the use of titanium tubing in seawater cooled condensers and stainless steel tubing in fresh water cooled condensers offers improved corrosion resistance, enhances condenser integrity, and reduces the source of copper to the condensate. The use of copper-based alloys in condenser tubing is being approached with extreme caution because the presence of soluble copper and/or nickel may promote the chemical reaction that causes denting. Of the copper-based alloys, CuNi alloys offer improved corrosion resistance over brasses or bronzes commonly used in older units.

The same concern for reducing corrosion and the source of copper ions in condensers applies to the feedwater heaters and moisture separator reheaters (MSR). Westinghouse suggests the use of 300 series stainless steel for the feedwater heaters and either carbon steel or carefully chosen types of ferritic stainless steel for the MSR. The use of 90-10 CuNi alloy is acceptable in the low-oxygen environment of the MSR because significant copper pickup is not expected.

4.2.4 Condensate Polishers

Acceptable secondary coolant chemistries have been maintained both with and without the use of condensate polishing. The levels of sludge accumulation of the tube sheets are comparable in plants with or without polishers; however, there has been no evidence of caustic-induced corrosion in any plant equipped with condensate polishing. Condensate polishing can be an asset in maintaining a contaminant-free secondary coolant chemistry.

4.2.5 Steam Generator Tube Repair

Another option presently being examined is a method of removing only steam generator tubes as opposed to replacing the entire lower section (shell, tube sheet, and tube bundle) from degraded steam generators and replacing them with new tubes. This option would reuse the existing steam generator upper and lower vessel shell and the steam generator tube sheet.

5. RELATED RESEARCH PROGRAMS

5.1 Westinghouse Electric Corporation

Research into the causes of denting include operational testing, destructive analysis of tubing and support plate samples, and laboratory experimentation. The initial information on denting phenomenon was derived from examination of tube/support plate samples that revealed thick oxide buildup, tube diameter reduction, and chemical makeup of the crevice-filling materials. Only minor corrosive attack on the tube material was observed. The crevice contained a thick layer of almost pure magnetite (Fe_3O_4).

Westinghouse conducted a series of tests on the crevices with contaminants and have been able to produce denting in the laboratoy. Denting has subsequently been reproduced in model boilers equipped with plant-type geometrical configurations.

The presence of chloride has been found to be a common factor in reproducing denting. Nickel chloride solutions and ferrous or cupric chlorides have produced measureable denting. Thus far, test data indicate that certain substances, e.g., phosphates, calcium hydroxide, zinc oxide, and borates, seem to retard the denting process. Morpholine, an AVT additive, has shown a beneficial effect by reducing the corrosion rate of carbon steel.

Westinghouse has also found a correlation between net hydrogen (H_2) generation in the steam generator and the existence of denting. Testing the effects of lithium borate and boric acid additions to the steam generators has been combined with a program in which the H_2 produced by corrosion has been monitored. Some reduction in the H_2 generation rate has been seen with boric acid injection. Tests to verify and quantify the borate and boric acid effects are in progress.

5.2 Combustion Engineering

Model boiler tests have also been used by Combustion Engineering to simulate the denting mechanism as parts of the EPRI overall steam generator program. The findings are basically in agreement with those reported by Westinghouse and both experiments correlate with Potter and Mann's observations* of magnetite formation in the crevices of carbon steel in high-temperature pure water containing iron or nickel or copper chloride salts. The Combustion Engineering experiments indicate that denting could occur due to the corrosion of copper-based condenser tubes and condenser inleakage

* E. C. Potter and G. M. W. Mann, "The Fast Linear Growth of Magnetite on Mild Steel in High Temperature Aqueous Conditions," p.26, British Corrosion Journal, Vol. 1 (1965).

of cooling water containing chloride ions. The type of copper-based tubing would be important in the accumulation of CuO_2 in the secondary side of the steam generator; i.e., copper-nickel alloys being more corrosion resistant. The CuO_2 "builds up" in the crevices between steam generator tubes and tube support plates and acts as a concentrator for chloride ions. As a result of acid chloride concentration, the carbon steel support plate corrosion is accelerated with corrosion product buildup causing denting.

The Combustion Engineering tests also indicate that the phosphates to AVT transition was not necessary to initiate the denting process, even before plants with pure, uncontaminated AVT environments were found to have suffered denting. In additon to reproducing the denting phenomenon, the Combustion Engineering tests are also being used to demonstrate chemical cleaning as a means to remove corrosion products to arrest the denting process.

5.3 NRC-Funded Research Programs

The NRC is funding two research programs related to steam generator problems. The first program is investigating stress corrosion cracking of PWR steam generator tubing. The objectives of this program are (1) to develop data that will enable prediction of stress corrosion cracking of Inconel-600 in terms of factors such as temperature, stress, material history, and environment, and (2) to evaluate the aqueous environment and metallurgical structures under which stress corrosion cracking occurs. The second program is directed toward an investigation of steam generator tube integrity. The objective of this program is to develop a validated model, based upon experimental data, for prediction of margin-to-failure under burst and collapse pressure of degraded steam generator tubing. The experimental work will be performed on designated steam generator tube specimens from our operating PWR steam generator. It is expected that this pogram will provide the necessary corroboration for calculations that indicate adequate safety margins exist under theoretical accident conditions with degraded steam generator tubes.

6. CONCLUSIONS

The resolution of operational problems related to the PWR steam generators is a complex task that requires the joint efforts of the vendors and operators of these plants and the NRC staff. In this regard, the staff formulated Task Action Plans (TAP) A-3 and A-4 (copies of which are enclosed in Appendix B) for plants designed by Westinghouse and Combustion Engineering, respectively, to organize and give priority to NRC staff efforts in the final resolution of problems related to the operation of PWR steam generators.

6.1 Basis for Continued Operation

For PWRs with recirculation types of steam generators, the NRC staff concluded that, pending completion of the Task Action Plans A-3 and A-4, continued operation does not constitute an undue risk to the health and safety of the public for the following reasons:

1. Primary-to-secondary leakage rate limits, and associated surveillance requirements, have been established to assure that the occurrence of tube cracking during operation will be detected and appropriate corrective action will be taken before any individual crack becomes unstable under normal operating, transient, or accident conditions.

2. Inservice inspection requirements and preventive tube plugging criteria have been established so that the great majority of degraded tubes will be identified and removed from service before leakage develops.

3. On a case-by-case basis, additional measures have been taken (a) to minimize contamination of the secondary coolant by inleakage of condenser cooling water (for example, condenser tubes with improved corrosion resistance have been installed), and (b) to minimize buildup in the steam generators of corrosion products generated in the secondary system (for example, feedwater heaters with improved corrosion resistance have been installed). Controls or monitoring of parameters that affect steam generator water chemistry are being considered to provide additional assurance that the potential for tube degradation during operation is minimized.

4. Observed through-wall cracks at dented locations (that is, tube/support plate intersections) have been small and stable (no rapid failures) during normal operation. In addition, because such cracks are constrained by the support plates, they are not anticipated to become unstable (burst) during postulated accidents.

5. Even if a LOCA or a MSLB were to occur during operation and some tubes were in a state of incipient failure, the radiological consequences of such an event would not be severe.

6. Continuous feedback from operating experience and the TAP efforts will be utilized to update interim criteria and requirements.

6.2 Basis for Continued Operation of Plants With Severe Degradation

For plants experiencing severe degradation, the following additional interim bases were also considered:

1. The probability of a design basis accident occurring during normal operation is small, and the probability that the accident would occur during the short period of time while the plant was operating with either a slightly leaking tube or a tube at the point nearing leakage is even smaller.

2. Even if an accident occurs when there are cracked tubes, the conservatively calculated consequences are still acceptably small.

3. A small amount of leakage (e.g., less than the technical specification limit) can be tolerated during normal operation without exceeding the offsite dosage limits of 10 CFR Part 20.

6.3 Licensing of New PWR Facilities

The preceding rationale, which constitutes the basis for continued operation of licensed Westinghouse and Combustion Engineering PWR facilities, also supports continued licensing of new facilities. Furthermore, to the extent that is practicable, depending on the status of the design, fabrication, and installation of the steam generators for facilities not yet licensed for operation, "state-of-the-art" design improvements and operating procedures that eliminate, or at least reduce, the potential for steam generator tube degradation are required by the staff. The following design and operational factors are considered by the staff in the conduct of its reviews:

1. Designs to provide improved circulation to eliminate low flow areas and to facilitate sludge removal.

2. Designs to minimize flow-induced vibration and cavitation.

3. Designs to provide increased flow around the tubes at the support plates.

4. Selection of material for tube support plates that demonstrates improved corrosion resistance.

5. Material selection (composition), processing, and heat treatment to minimize the susceptibility of tubes to stress corrosion cracking.

6. Secondary water system chemistry control.

7. Designs to allow for installation of an ion exchanger (condensate demineralizer) in the secondary water system to minimize feedwater contamination.

In view of the above, the staff concluded that issuance of Construction Permits (CP) and Operating Licenses (OL), pending completion of generic studies, can continue with reasonable assurance that operation will not present an undue risk to the health and safety of the public.

APPENDIX A

PWR DESIGN CONFIGURATION

Nuclear power plants using the pressurized water reactor (PWR) design concept contain three separate cooling cycles. The three cooling cycles of a typical PWR are shown in Figure A-1. The first cooling cycle comprises the primary coolant system that pumps pressurized coolant water through the heat-generating core of the reactor where it picks up heat. The second cooling cycle consists of large heat exchangers called steam generators, a steam-driven turbine generator, a steam condenser, feedwater pumps, and associated piping systems. Heat generated in the primary coolant system is transferred to the secondary system through steam generators. The water in the secondary coolant system boils in the steam generator creating steam that is used to drive 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 steam generators by the feed pumps, thereby completing the cycle. The third cooling cycle is the condenser cooling water system that provides cold water to condense the steam back to water in the steam condenser. The basis for this closed-cycle system is to ensure that the radioactive primary coolant water, the secondary cooling water, and the condenser cooling water are separated from each other. The steam generator is the connecting link between the radioactive primary and nonradioactive secondary coolant system and is, therefore, a principal part of the reactor coolant pressure boundary. Figure A-2 shows the major components of the reactor coolant system.

Two major types of steam generators are currently in use in pressurized water reactors in the United States. These are the recirculating type, which is manufactured by Westinghouse and Combustion Engineering, and the once-through type, which is manufactured by Babcock & Wilcox. Typical recirculating types of steam generators manufactured by Westinghouse and Combustion Engineering are thoroughly described in the main report. In the recirculating type of steam generator, 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.

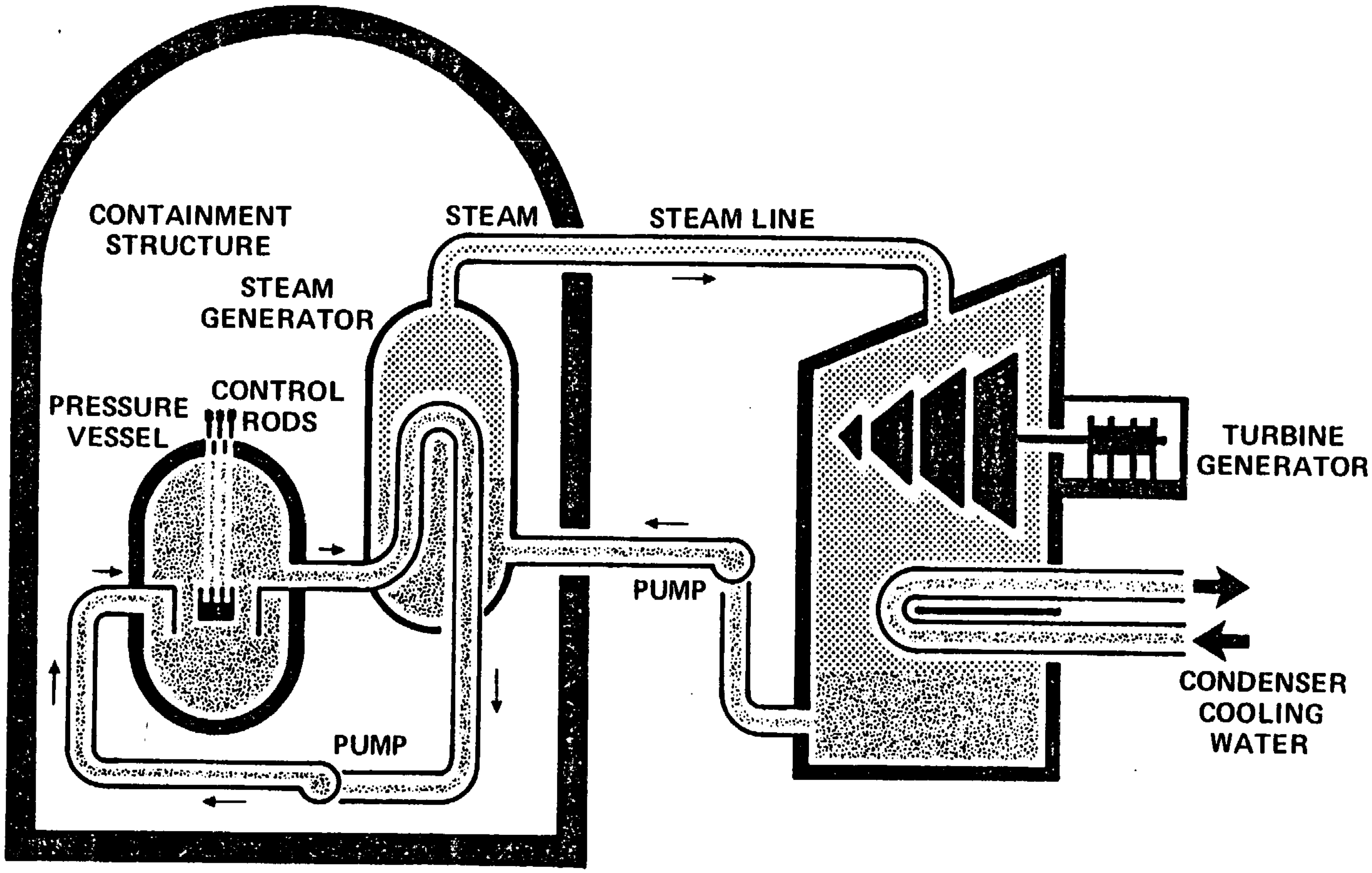


FIGURE A-1. PRESSURIZED WATER REACTOR (PWR) COOLING CYCLES

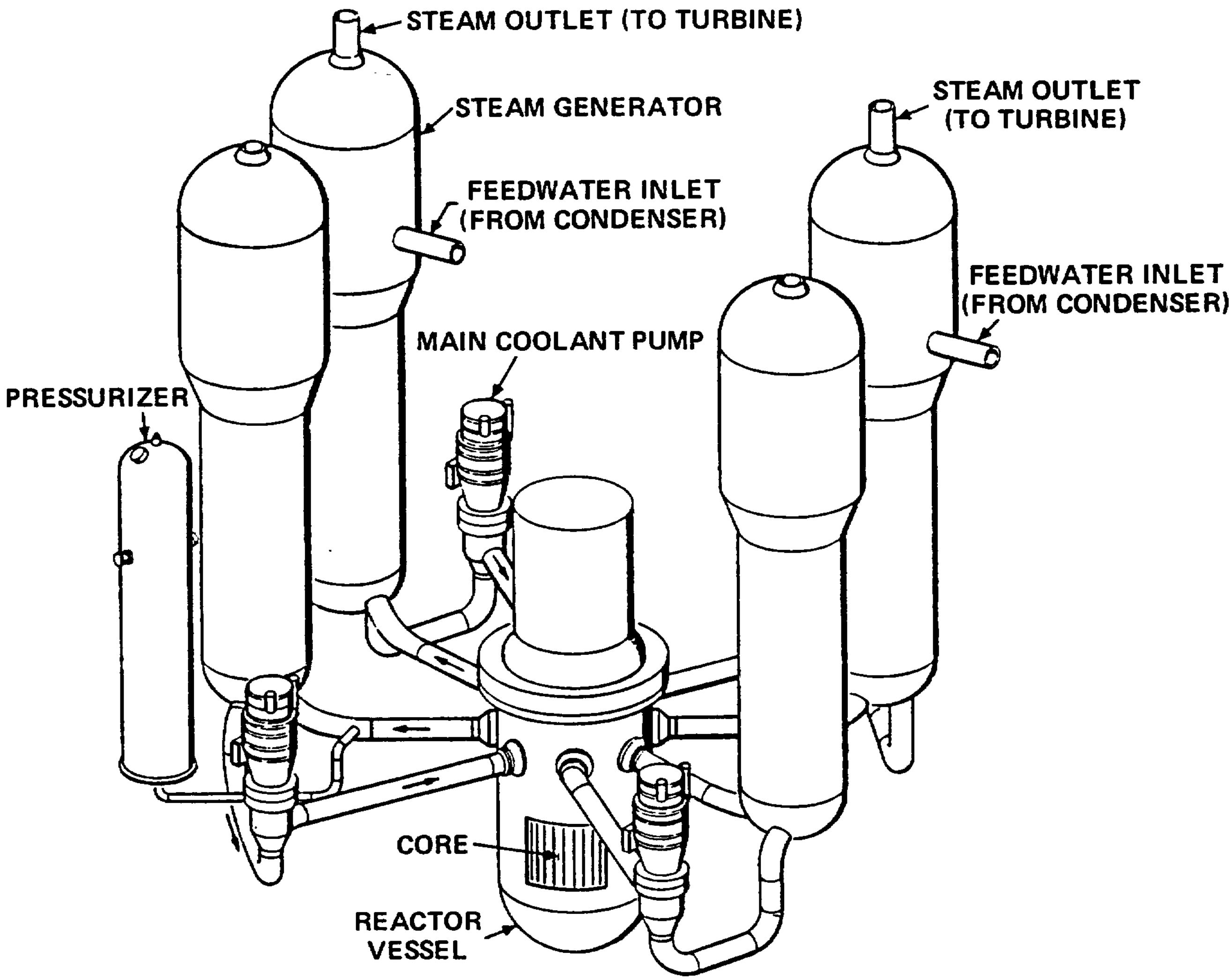


FIGURE A-2. SCHEMATIC OF REACTOR COOLANT SYSTEM FOR PWR

Several thousand tubes (the number of tubes depends on the design) are welded into and are vertically supported by the tube sheet that is 20 inches thick (or more) and is perforated with thousands of holes for the steam generator tubes. At various higher elevations, the tubes penetrate through holes in tube support plates that provide lateral support and, 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 7/8 to 3/4 inch in outside diameter with a wall thickness of about 0.050 inch. The tubes provide the heat transfer surfaces between the primary coolant water and the secondary coolant water. The steam generator tubes constitute over 50 percent 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 the steam generated that is sent to the turbine generator.

APPENDIX B

TAP A-3

Westinghouse Steam Generator Tube Integrity

TAP A-4

Combustion Engineering Steam Generator Tube Integrity

(Data in this Appendix is taken from USNRC Report NUREG-0371, "Task Actin Plans for Generic Activities, Category A," November 1978, pp. A-3/1 through A-3/11, pp. A-4/1 through A-4/10.)

Task A-3

WESTINGHOUSE STEAM GENERATOR TUBE INTEGRITY

Lead NRR Organization:	Division of Operating Reactors (DOR)
Lead Supervisor:	Darrell G. Eisenhut, A/D for Systems and Projects, DOR
Task Manager:	B.D. Liaw, EB/DOR
Applicability:	Westinghouse Pressurized Water Reactors
Projected Completion Date:	December 1979

1. DESCRIPTION OF PROBLEM

Pressurized water reactor steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in the tube diameter (denting) and vibration induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Westinghouse steam generator tubes have suffered degradation due to wastage and stress corrosion cracking. Both types of degradation have been nominally arrested; however, degradation due to denting which leads to primary side stress corrosion cracks is the major problem at present, and the principal focus of this technical activity.

2. PLAN FOR PROBLEM RESOLUTION

The major portion of the NRC staff efforts related to the resolution of the denting problem will consist of evaluation of the results of investigations by Westinghouse, EPRI, and EPRI supported contractors. In addition, NRC supported technical assistance and confirmatory research programs will be used as the basis for evaluation of applicant supplied data.

The specific activities directed at resolution of the denting problem in Westinghouse steam generators consist of the following issues and tasks:

A. Generic Evaluation of ISI Results

Review and evaluate the various eddy current inspection results; i.e., experience from operating reactors and evaluate these data as they relate to the generic determination of failure probability of degraded tubes. In addition, evaluate the test programs and analytical studies to provide staff understanding sufficient to continue to provide justification for continued safe operation of operating reactors.

B. Evaluation of Transients and Postulated Accidents

Evaluation of failure consequences under postulated accident conditions (LOCA and MSLB) to determine the acceptable levels of primary to secondary leakage rates and the effect on ECCS performance. The results will be used to define the acceptable number of tube failure that may be necessary as a licensing basis considering predicted fuel behavior and radiological dose during transients and postulated accident conditions.

C. Evaluation of Steam Generator Tube Structural Integrity

Review and evaluate te structural integrity of steam generator tubes under normal operating and postulated accident conditions (LOCA, SSE and MSLB) including licensee and Westinghouse analyses where appropriate to generic conclusions.

D. Establish Tube Plugging Criteria

Establish a generic tube plugging criteria that is consistent with the determined allowable leak rate, tube structural integrity and degradation rates. These results will allow assessment of the adequacy of the requirements defined in Regulatory Guide 1.121.

E. Secondary Coolant Chemistry Requirements

Evaluate the mechanism of tube degradation. The results will be used to define the requirements for secondary coolant chemistry control including considerations for condenser in-leakage.

F. Evaluation of ISI Methods

Review the development of improved eddy-current probes, coils and multi-frequency techniques to better quantify dents and growth of dents and increase sensitivity of detecting cracks in dented regions.

G. Establish Criteria for Revision of Regulatory Guide 1.83

Integrate experience from inservice inspection results, the results from the evaluation of various ISI improvements and the plugging and secondary water chemistry requirements into criterion for possible revision of Regulatory Guide 1.83.

H. Steam Generator Replacement (Prototype)

Review and evaluate plans for initial steam generator replacement as generic basis for subsequent replacement actions.

I. Review Design Criteria for Plants Not Yet Licensed

Review and evaluate design modifications proposed by applicants and Westinghouse to prevent denting in plants not yet licensed for operation.

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION OF TASK

The safety issue addressed by this Task Action Plan is applicable to selected Pressurized Water Reactors with Westinghouse-designed (W) steam generators.

For W PWRs currently licensed for operation, we have concluded that, pending completion of this TAP, continued operation does not constitute an undue risk to the health and safety of the public for the following reasons:

- Primary to secondary leakage rate limits, and associated surveillance requirements, have been established to provide assurance that the occurrence of tube cracking during operation will be detected and appropriate corrective action will be taken such that an individual crack will not become unstable under normal operating, transient or accident conditions.

- Augmented inservice inspection requirements and preventative tube plugging criteria have been established to provide assurance that the great majority of degraded tubes will be identified and removed from service before leakage develops.

- Steam generator water chemistry control requirements are being considered to provide additional assurance that the potential for tube degradation during operation is minimized. On a case-by-case basis, additional measures have been taken to (1) minimize contamination of the secondary coolant by in-leakage of condenser cooling water (e.g., condenser tubes with improved corrosion resistance have been installed) and (2) minimize buildup in the steam generators of corrosion products generated in the secondary system (e.g., full flow condensate demineralizers have been installed).

- Observed through-wall cracks at dented locations, i.e., tube/support plate intersections, have been small and stable (no rapid failures) during normal operation. In addition since such cracks are constrained by the support plates, they are not anticipated to become unstable (burst) during postulated accidents.

- Even if a LOCA or a MSLB were to occur during operation and some tubes were in a state of incipient failure, the radiological consequences of such an event would not be severe.

- Continuous feedback from operating experience and the TAP efforts will be utilized to update interim criteria and requirements.

For plants experiencing severe degradation, the following additional interim bases were also considered:

- The probability of the design basis accident during normal operation is small and the probability that the accident would occur during the short period of time between the leak detection and the plant shutdown is even smaller.

- Even if an accident occurs when there are cracked tubes, the conservatively calculated consequences are still acceptably small until plant shutdown.

- A small amount of leakage (e.g., less than the Technical Specification limit) can be tolerated during normal operation without exceeding the offsite dosage limits of 10 CFR Part 20.

The above-montioned rationale which constitutes the basis for continued operation of licensed W PWR facilities also supports continued licensing of new facilities. Further, to the extent that is practicable, depending on the status of the design, fabrication and installation of the steam generators for facilities not yet licensed for operation, "state-of-the-art" design improvements and operating procedures which eliminate or at least minimize the potential for steam generator tube degradation are required by the staff. The following design and operational factors are considered by the staff in the conduct of its reviews:

- Designs to provide improved circulation to eliminate low flow areas, and to facilitate sludge removal.

- Designs to minimize flow induced vibration and cavitation.

- Designs to provide increased flow around the tubes at the support plate.

- Selection of material for tube support plates with improved corrosion resistance.

- Material selection (chemistry), processing and heat treatment to minimize the susceptibility of tubes to stress corrosion cracking.

- Secondary system water chemistry control.

- Secondary side material selection (condensers, feedwater, heaters turbine discs and blades, elbows, etc.), and water cleanup system to minimize erosion and the resulting sludge and corrosion product buildup in the steam generators.

- Designs to allow for installation of an ion exchanger (condensate demineralizer) in the secondary system to minimize feedwater contamination.

- Condenser leakage detection systems.

In view of the above, we conclude that issuance of Construction Permits and Operating Licenses, pending completion of this TAP, can continue with reasonable assurance that operation will not present an undue risk to the health and safety of the public.

4. NRR TECHNICAL ORGANIZATIONS INVOLVED

A. Engineering Branch, Division of Operating Reactors, has the primary lead responsibility for the overall review and evaluation of steam generator tube integrity. This includes operational experiences, tube failure mechanisms and potential repairs, plugging criteria, ISI requirements, tube failure probability, leakage rate limits, and secondary coolant system control. This also includes the lead responsibility for determining the probability of LOCA and MSLB initiating events and the probability of tube failures during these events and responsibility for determining the number of tubes assumed to fail in LOCA and MSLB analyses. The Engineering Branch also has lead responsibility for the review of prototype steam generator tube replacement.

Manpower Estimates: 0.1 man-year FY 1977; 1.0 man-year FY 1978; 1.0 man-year FY 1979.

B. Environmental Evaluation Branch, Division of Operating Reactors, has the lead responsibility for the review and evaluation of the offsite dosage related to the consequence or probability of a Main Steam Line Break (MSLB) accident or LOCA given the physical conditions determined in item A, above. EEB will also consult with EB and provide support for the probabilistic evaluation of MSLB and LOCA initiating events, the probability of tube failures during these postulated events and evaluation of environmental aspects of steam genérator tube replacement.

Manpower Estimates: 0.1 man-year FY 1977; 0.2 man-year FY 1978; 0.2 man-year FY 1979.

C. Reactor Safety Branch, Division of Operating Reactors, has the lead responsibility for the review and evaluation of: (1) the ECCS performance related to secondary-to-primary leakage as a consequence of a LOCA, and (2) the effect of primary-to-secondary leakage during a MSLB accident on fuel failures.

Manpower Estimates: 0.1 man-year FY 1977; 0.13 man-year FY 1978; 0.13 man-year FY 1979.

D. Mechanical Engineering Branch/Materials Engineering Branch, Division of Systems Safety, has lead responsibility for the review of new design/material concepts and new system component requirements. This will apply to PWR facilities not yet licensed for operation.

The activities involved will include the review and evaluation of applicant's and Westinghouse's proposed improvements on the design and/or operation of the steam generators for items such as secondary coolant chemistry, design modifications to avoid denting, condenser design to avoid inleakage, ISI requirements, recommendation for revision of Regulatory Guides, and provisions for access opening and space in the containment to facilitate steam generator inspections.

Manpower Estimates: 0.1 man-year FY 1977; 0.5 man-year FY 1978; 0.5 man-year 1979; 0.5 man-year FY 1980.

E. Analysis Branch, Division of Systems Safety, has the lead responsibility in developing analytical capabilities (computer code, etc.) to evaluate the effects of steam generator tube rupture(s) concurrent with various reactor transients that include MSLB and LOCA accidents. The purpose is to determine the equivalent number of tube failures that can be tolerated during transient events. This information will then be factored into the overall program of determining an adequate sample plan for tube inspections.

Manpower Estimates: 0.1 man-year FY 1977; 0.2 man-year FY 1978; 0.2 man-year FY 1979.

F. Reactor Systems Branch, Division of Systems Safety. Has the responsibility of implementing new procedures on CP/OL safety analyses for plants yet to be licensed should any be required as the result of this technical activity.

Manpower Estimates: 0.1 man-year FY 1979; 0.3 man-year FY 1980.

G. Environment Project Branch No. 1, Division of Site Safety and Environmental Analysis. Responsible for the review of the nonradiological environmental aspect of steam generator replacement for the lead unit.

Manpower Estimate: 0.2 man-year FY 1978.

5. TECHNICAL ASSISTANCE

A. Contractor: Brookhaven National Laboratory (BNL) - DOR, DSS

Funds Required: $98K FY 1977; $125K FY 1978; $225K FY 1979.

This effort is funded as part of an overall program at BNL applicable to the three Category A Technical Activities (A-3, A-4, and A-5) related to PWR steam generators. Funding values under DORSAT are not included.

This program is needed to obtain technical consultation and assistance to review information in areas of water chemistry and corrosion analysis, monitored jointly by EB/DOR and MTEB/DSS. Stress and/or burst strength calculations are funded in part under DORSAT contract on an as-needed basis. This program will provide assistance in accomplishing Tasks 2C, 2E, and 2G.

B. Contractor: Idaho National Engineering Laboratory (INEL) - DSS

Funds Required: $75K FY 1977; $100K FY 1978.

This effort is generic in nature and will be applicable to the three Category A Technical Activities (A-3, A-4, and A-5) related to PWR steam generators.

The purpose of this program is to determine the effect of steam generator tube plugging on the predicted peak clad temperatures following a postulated LOCA. The primary activity is to produce a reliable computer code to aid the evaluation of the effects of tube plugging on the ECCS performance. An addition to the program will be needed to consider steam generator tube failures concurrent with MSLB or a LOCA. This program will provide assistance in accomplishing Tasks 2B and 2D.

C. Contractor: Sandia Laboratories, DOR

Funds Required: $50K FY 1977; $100K FY 1978; $150K FY 1979.

This work is of generic nature, and will be applicable to all PWR steam generators.

The purpose of this program is to perform a statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. This is a new program to augment staff effort in steam generator safety reviews and will assist in addressing Tasks 2A, 2F, and 2G.

6. ASSISTANCE REQUIREMENTS FROM OTHER NRC OFFICES

A. Office of Nuclear Regulatory Research, Division of Reactor Safety Research, Metallurgy and Materials Branch and Probabilistic Analysis Branch.

RES has funded, at the request of NRR, a major confirmatory experimental program at Pacific Northwest Laboratory. The activity of this program consists of a series of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leakage rate data. This program is managed by Metallurgy and Materials Branch, (Task 2C).

RES has funded, at the request of NRR, a program, to address the factors which determine Inconel 600 susceptibility to stress corrosion cracking in primary water. Metallurgical condition, chemistry, temperature, stress and environment will be considered, (Task 2E).

B. Office of Standards Development, Division of Engineering Standards, Structures and Components Standards Branch.

OSD has funded a confirmatory research program at Battelle Columbus Laboratory to evaluate eddy current methods for inspecting steam generator tubes as a subcontract to Brookhaven National Laboratory, (Part of Task 2F).

C. Office of the Executive Director for Operations, Applied Statistic Group.

Provide assistance to EB/DOR for statistical assessment of steam generator tube integrity, (Part of Tasks 2A, 2F, and 2G).

D. ACRS

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the committee as the task progresses.

7. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

A. Licensee(s) of Westinghouse (W) Nuclear Facilities

At present all W plants experiencing tube denting will be monitored for the progress of denting. Each licensee will submit an analysis of the consequences of tube denting on tube integrity and demonstrate that adequate safety margins exist for continued safe operation. The Turkey Point and Surry licensees will be closely monitored relative to steam generator replacement.

B. Westinghouse

The primary interaction with Westinghouse has been and continues to be on the investigation program for the resolution of the problems at Westinghouse designed plants and their generic implication such as the licensing bases or justifications for continued operation of Westinghouse plants with known tube degradations. For interim periods of operation before the cause of denting is identified and corrective measures implemented, the interaction will be needed to ensure that Westinghouse develops and improves capabilities for the evaluation of ECCS performance under postulated accidents concurrent with tube failures should such a licensing basis become necessary. Review and evaluate new designs proposed to prevent denting in facilities not yet licensed for operation.

C. EPRI, PWR Owner Group, etc.

Interactions with other organizations such as the Electric Power Research Institute (EPRI) and the "ad hoc" organization of PWR owners may also be required because of mutual interests in the safe operation of steam generators in general and, in particular, the various problems associated with the operation of steam generators.

The purpose for interactions with these organizations is to exchange information on the research works sponsored by NRC and these outside organizations in identifying potential problems or solutions to existing problems associated with the operation of steam generators. Current programs in this area include an EPRI sponsored steam generator program in conjunction with Combustion Engineering. One aspect of this program is designed to define the mechanism of tube denting, and its goal is to provide corrosion-related information for improved steam generator coolant system technology and operation. The technology will be applied to the operation of plant systems and components that affect the reliability of steam generators. Additionally, EPRI had underway an ISI round robin test program for steam generator tubes to determine the effectiveness of various ISI techniques and methods for tube inspection.

8. POTENTIAL PROBLEMS

Except for steam generator replacement, there is no apparent short term resolution of tube denting in affected Westinghouse plants. The many programs underway to resolve tube denting in presently operating plants may bring about a partial solution, by arresting denting through a cleaning program, sometime early in 1979.

However, by establishing quantitative plugging criteria for dented tubes, and requiring scheduled inspections varying with the degree of denting observed, safety concerns can be minimized to the point where continued operation can be justified.

Finally, completion of many of the indicated tasks will depend on the scheduled completion of programs sponsored by organizations outside NRR. As with most experimental investigations, periodical delays can be expected, which may delay completion of some of the tasks indicated in the Task Action Plan.

Task A-4

COMBUSTION ENGINEERING STEAM GENERATOR TUBE INTEGRITY

Lead NRR Organization:	Division of Operating Reactors (DOR)
Lead Supervisor:	Darrell G. Eisenhut, A/D for Systems and Projects, DOR
Task Manager:	Frank M. Almeter, EB/DOR
Applicability:	Combustion Engineering Pressurized Water Reactors
Projected Completion Date:	December 31, 1979

1. DESCRIPTION OF PROBLEM

Pressurized water reactor operating experience during the past five years has shown that steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in tube diameter (denting) and vibration induced fatigue cracks. Since the steam generator tubes are an integrated part of the reactor coolant pressure boundary in the PWR system, the primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Palisades has been the only Combustion Engineering designed plant to experience tube degradation due to wastage and secondary side stress corrosion cracking with the use of a phosphate treatment for the secondary coolant. Both types of degradation have been nominally arrested by conversion to AVT chemistry control. However, tube degradation due to denting (but to a lesser degree than the Westinghouse steam generators) occurred after the conversion to an AVT chemistry Recent inservice inspections at two sea coast facilities with CE designed steam generators, which used an AVT chemistry for the secondary coolant since initial startup, have shown that the prior use of phosphates is not a necessary precursor to cause denting in steam generator tubing. Denting which leads to primary side stress corrosion cracking is the major problem at present and the principal focus of this technical activity. However, as steam generator operating experience is accumulated and interpreted, it has become evident that condenser cooling water in-leakage resulting from the corrosion of condenser tubes can contaminate the secondary water of PWR steam generators and may be the principle source leaking to all types of steam generator tube degradation. It has also become evident that the maintenance of secondary coolant water quality cannot be achieved if condenser in-leakage is allowed. Because the condenser is an important component of the PWR secondary system, an approach must be developed to minimize condenser in-leakage to ensure adequate steam generator tube integrity.

2. PLAN FOR PROBLEM RESOLUTION

The problem will be resolved by reviewing the type and mechanism of tube degradation in operating reactors to evaluate the effects of tube structural integrity and failure probability under normal operation and accident conditions (LOCA, SSE and MSLB). Assessment of the effects of degraded tubes on postulated accident conditions will be factored into the development of new criteria for tube plugging, acceptable levels of primary to secondary leakage, and ISI requirements to ensure the safe operation of operating pressurized water reactors. To minimize tube degradation, priority areas where improvements in steam generator design and criteria for the secondary coolant system are needed will be identified to develop licensing positions for the CP/OL review of new plants.

The specific activities directed at resolution of the denting problem in Combustion steam generators consist of the following issues and tasks:

A. Generic Evaluation of ISI Results

Review and evaluate the various eddy current inspection results; i.e., experience from operating reactors and evaluate these data as they relate to the generic determination of failure probability of degraded tubes. In addition, evaluate the test programs and analytical studies to provide staff understanding sufficient to continue to provide justification of continued safe operation of operating reactors.

B. Evaluation of Transients and Postulated Accidents

Evaluation of failure consequences under postulated accident conditions (LOCA and MSLB) to determine the acceptable levels of primary to secondary leakage rates and the effect on ECCS performance. The results will be used to define the acceptable number of tube failures that may be necessary as a licensing basis considering predicted fuel behavior and radiological dose during transients and postulated accident conditions.

C. Evaluation of Steam Generator Tube Structural Integrity

Evaluation of licensees' and CE's analysis of structural integrity of tubes under normal operating and accident conditions (LOCA, SSE and MSLB). Information developed in this task will provide input for establishing a generic tube plugging criteria and recommendations for the revision of Regulatory Guide 1.121.

D. Establish Tube Plugging Criteria

Establish a generic tube plugging criteria that is consistent with the determined allowable leak rate, tube structural integrity and degradation rates. These results will allow assessment of the adequacy of the requirements defined in Regulatory Guide 1.21.

E. Secondary Coolant Chemistry Requirements

Evaluate the mechanism of tube degradation. The results will be used to define the requirements for secondary coolant chemistry control including considerations for condenser in-leakage.

F. Evaluation of ISI Methods

Review the development of improved eddycurrent probes, coils and multi-frequency techniques to better quantify dents and growth of dents and increase sensitivity for detecting cracks in dented regions.

G. Establish Criteria for Revision of Regulatory Guide 1.83

Integrate experience from inservice inspection results, the results from the evaluation of various ISI improvements and the plugging and secondary water chemistry requirements into criterion for possible revision of Regulatory Guide 1.83.

H. Review Design Criteria for Plants Not Yet Licensed

Review and evaluate design modifications proposed by applicants and CE to prevent denting in plants not yet licensed for operation.

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION OF TASK

The safety issue addressed by this Task Action Plan is applicable to Pressurized Water Reactors with Combustion Engineering (CE) steam generators.

For CE PWRs currently licensed for operation, we have concluded that, pending completion of this TAP, continued operation does not constitute an undue risk to the health and safety of the public for the following reasons:

Primary to secondary leakage rate limits, and associated surveillance requirements, have been established to provide assurance that the occurrence of tube cracking during operation will be detected and appropriate corrective action will be taken such that no individual crack will become unstable under normal operating, transient or accident conditions.

- Augmented inservice inspection requirements and preventative tube plugging criteria have been established to provide assurance that the great majority of degraded tubes will be identified and removed from service before leakage develops.

- Steam generator water chemistry control requirements are being considered to provide additional assurance that the potential for tube degradation during operation is minimized. On a case-by-case basis, additional measures have been taken to (1) minimize contamination of the secondary coolant by in-leakage of condenser cooling water (e.g., condenser tubes with improved corrosion resistance have be installed) and (2) minimize buildup in the steam generators of corrosion products generated in the secondary system (e.g., full flow condensate demineralizers have been installed).

- Tube denting at tube/support plate intersections in CE designed steam generators has not been severe enough to result in through-wall cracks at dented locations. However, if tube cracking were to occur in the dented region, it would be constrained by the support plates which would control crack stability and prevent tube failure (bursting) during postulated accidents.

- Even if a LOCA or a MSLB were to occur during operation and some tubes were in a state of incipient failure, the radiological consequences of such an event would not be severe.

- Continuous feedback from operating experience and the TAP efforts will be utilized to update interim criteria and requirements.

For plants experiencing severe degradation, the following additional interim bases were also considered:

- The probability of the design basis accident during normal operation is small and the probability that the accident would occur during the short period of time between the leak detection and the plant shutdown is even smaller.

- Even if an accident occurs when there are cracked tubes, the conservatively calculated consequences are still acceptably small until plant shutdown.

- A small amount of leakage (e.g., less than the Technical Specification limit) can be tolerated during normal operation without exceeding the offsite dosage limits of 10 CFR Part 20.

The above-mentioned rationale which constitutes the basis for continued operation of licensed CE PWR facilities also support continued licensing of new facilities. Further, to the extent that is practicable, depending on the status of the design, fabrication and installation of the steam generators for facilities not yet licensed for operation, "state-of-the-art" design improvements and operating procedures which eliminate or at last minimize the potential for steam generator tube degradation are required by the staff. The following design and operational factors are considered by the staff in the conduct of its reviews:

- Designs to provide improved circulation to eliminate low flow areas, and to facilitate sludge removal.
- Designs to minimize flow induced vibration and cavitation.
- Designs to provide increased flow around the tubes at the support plate.
- Selection of material for tube support plates with improved corrosion resistance.
- Material selection (chemistry), processing and heat treatment of minimize the susceptibility of tubes to stress corrosion cracking.
- Secondary system water chemistry control.
- Secondary side material selection (condensers, feedwater, heaters turbine discs and blades, elbows, etc.), and water cleanup system to minimize erosion and the resulting sludge and corrosion product buildup in the steam generators.
- Designs to allow for installation of an ion exchanger (condensate demineralizer) in the secondary system to minimize feedwater contamination.
- Condenser leakage detection systems.

In view of the above, we conclude that issuance of Construction Permits and Operating Licenses, pending completion of this TAP, can continue with reasonable assurance that operation will not present an undue risk to the health and safety of the public.

4. NRR TECHNICAL ORGANIZATIONS INVOLVED

A. Engineering Branch, Division of Operating Reactors, has the primary lead responsibility for the overall review and evaluation of steam generator tube integrity in operating plants. This includes operational experiences, tube failure mechanisms and potential repairs, plugging criteria, ISI requirements, tube failure probability studies, leakage rate limits, and secondary coolant system control. This also includes the lead responsibility for determining the probability of LOCA and MSLB initiating events and the probability of tube failures during these events and responsibility for determining the number of tubes assumed to fail in LOCA and MSLB analyses.

Manpower Estimates: 0.1 man-year FY 1977; 0.5 man-year FY 1978;
0.5 man-year FY 1979.

B. Environmental Evaluation Branch, Division of Operating Reactors, has the lead responsibility for the review and evaluation of the offsite dosage related to the consequence or probability of a Main Steam Line Break (MSLB) accident or a LOCA should such evaluation become necessary. EEB will also consult with EB and provide support for the probabilistic evaluation of MSLB and LOCA initiating events and the probability of tube failures during these postulated events.

Manpower Estimates: 0.1 man-year FY 1977; 0.2 man-year FY 1978;
0.2 man-year FY 1979.

C. Reactor Safety Branch, Division of Operating Reactors, has the lead responsibility for the review and evaluation of: (1) the ECCS performance related to secondary to primary leakage as a consequence of a LOCA, and (2) the effect of primary to secondary leakage during a MSLB accident on fuel failures should such evaluation prove necessary.

Manpower Estimates: 0.1 man-year FY 1977; 0.13 man-year FY 1978;
0.13 man-year FY 1979.

D. Mechanical Engineering Branch/Materials Engineering Branch, Division of Systems Safety, has responsibility in factoring all steam generator operating experience into the review of new design/material concepts and new system component requirements. This will apply to PWR facilities not yet licensed for operation.

The activities involved will include the review and evaluation of the applicant's and the NSSS's proposed improvements on the design and/or operation of the steam generators; for items such as secondary coolant chemistry, design modifications to avoid denting, ISI requirements, recommendations for revision of Regulatory Guides, condenser design to avoid in-leakage and provisions for access opening and space in the containment to facilitate steam generator inspections.

Manpower Estimates: 0.1 man-year FY 1977; 0.5 man-year FY 1978; 0.5 man-year FY 1979.

E. Analysis Branch, Division of Systems Safety, has the lead responsibility in developing analytical capabilities (computer codes, etc.) to evaluate the effects of steam generator tube rupture(s) concurrent with various reactor transients that include MSLB and LOCA accidents. The purpose is to determine the equivalent number of tube failures that can be tolerated during transient events. This information will then be factored into the overall program of determining an adequate sample plan for tube inspections.

Manpower Estimates: 0.2 man-year FY 1978; 0.2 man-year FY 1979.

F. Reactor Systems Branch, Division of Systems Safety, has the responsibility of evaluating the design and performance of new associated auxiliary systems for CP/OL plants yet to be licensed, should any be required as the result of this technical activity; e.g., full flow condensate demineralization, etc., for PWR secondary coolant.

Manpower Estimate: 0.15 man-year FY 1979.

5. TECHNICAL ASSISTANCE

A. Contractor: Brookhaven National Laboratory (BNL) - DOR, DSS

Funds Required: $98K FY 1977; $125K FY 1978; $225K FY 1979.

This effort is funded as part of an overall program at BNL applicable to the three Category A Technical Activities (A-3, A-4, and A-5) related to PWR steam generators. Funding values under DORSAT are not included.

This program is needed to obtain technical consultation and assistance to review information in areas of water chemistry and corrosion analysis, monitored jointly by EB/DOR and MTEB/DSS.

Stress and/or burst strength calculations are funded in part under DORSAT contract on an as-needed basis. This program will provide assistance in accomplishing Tasks 2C, 2E, and 2G.

B. Contractor: Idaho National Engineering Laboratory (INEL) - DSS

Funds Required: $75K FY 1977; $100K FY 1978.

This effort is generic in nature and will be applicable to the three Category A Technical Activities (A-3, A-4, and A-5) related to PWR steam generators.

The purpose of this program is to determine the effect of steam generator tube plugging on the predicted peak clad temperatures following a postulated LOCA. The primary activity is to produce a reliable computer code to aid the evaluation of the effects of tube plugging on the ECCS performance. An addition to the program will be needed to consider steam generator tube failures concurrent with MSLB or a LOCA. This program will provide assistance in accomplishing Tasks 2B and 2D.

C. Contractor: Sandia Laboratories - DOR

Funds Required: $50K FY 1977; $100K FY 1978; $150K FY 1979.

This work is of generic nature, and will be applicable to all PWR steam generators.

The purpose of this program is to perform a statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. This is a new program to augment staff effort in steam generator safety reviews and will assist in addressing Tasks 2A, 2F, and 2G.

6. ASSISTANCE REQUIREMENTS FROM OTHER NRC OFFICES

A. Office of Nuclear Regulatory Research, Division of Reactor Safety Research, Metallurgy and Materials Branch and Probabilistic Analysis Branch.

RES has funded, at the request of NRR, a major confirmatory experimental program at Pacific Northwest Laboratory. The activity of this program consists of a series of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leakage rate data. This program is managed by Metallurgy and Materials Branch, (Task 2C).

RES has funded, at the request of NRR, a program to address the factors which determine Inconel 600 susceptibility to stress corrosion cracking in primary water. Metallurgical condition, chemistry, temperature, stress and environment will be considered, (Task 2E).

The Probabilistic Analysis Branch funded the program to assist EEB in probabilistic analyses, (Task 2B).

B. Office of Standards Development, Division of Engineering Standards, Structures and Components Standards Branch.

OSD has funded a confirmatory research program at Battelle Columbus Laboratory to evaluate eddy current methods for inspecting steam generator tubes as a subcontract to Brookhaven National Laboratory, (Part of Task 2F).

C. Office of the Executive Director for Operations, Applied Statistics Branch.

Provide assistance to EB/DOR for statistical assessment of steam generator tube integrity, (Part of Tasks 2A, 2F, and 2G).

D. ACRS

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the committee as the task progresses.

7. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

A. Licensee(s) of Combustion Engineering Nuclear Facilities

At present all CE plants experiencing tube denting will be monitored to evaluate the progress of denting. Each licensee will submit an analysis of the consequences of tube denting on tube integrity and demonstrate that adequate safety margins exist for continued safe operation.

B. Combustion Engineering

The primary interactions with CE has been and continues to be related to their investigation program for the resolution of the tube denting problem at CE designed plants, and its generic implications, such as the licensing bases or justifications for continued operation of CE plants with known tube degradations. For interim periods of operation until the cause of tube denting is identified and corrective measures(s) implemented, this interaction will be needed to ensure that CE develops capabilities for the evaluation of ECCS performance for postulated accidents concurrent with tube failures, should such a licensing basis become necessary. In conjunction with licensees, CE will be requested to submit a test program and corrective action plan for Maine Yankee and Millstone Unit 2 and an analysis of the structural integrity of degraded tubes under normal operating and accident conditions (LOCA, SSE and MSLB).

In addition, CE will be requested to keep NRC informed of steam generator design changes and modifications in secondary water treatment systems to alleviate tube degradation in future CE plants. This information will be incorporated into all Tasks of the program.

C. EPRI, PWR Owner Group etc.

Interactions with other organizations such as the Electric Power Research Institute (EPRI) and the "ad hoc" organization of PWR owners may also be required because of the mutual interests in the safe operation of steam generators in general and, in particular, the various problems associated with the operation of steam generators. Current programs sponsored by EPRI include the CE model boiler studies and the round robin program for ISI techniques.

8. POTENTIAL PROBLEMS

It should be anticipated that required feedback from related programs funded by outside organizations may delay the timely completion of certain subtasks. However, it is hoped that effective participation of NRC representatives at "ad hoc" organizational meetings will improve mutual interests in NRC goals. Any delays in submittals required by licensees and NSSS vendors would certainly delay the review and evaluation of tasks defined in the program. Timely input is required from all technical organizations involved.

APPENDIX C

CONDENSER TUBE MATERIALS FOR OPERATING PLANTS (PWR)

NSSS Vendor	Plant Name	Condenser Tube Material
W	Beaver Valley 1	Stainless steel
	Cook 1	Cu and stainless steel
	Cook 2	Cu and stainless steel
	Farley 1	Titanium
	Ginna	Admiralty and stainless steel
	Haddam Neck	Admiralty and stainless steel
	Indian Point 2	Admiralty
	Indian Point 3	Admiralty
	Kewaunee	Admiralty brass
	North Anna 1 & 2	Stainless steel
	Point Beach 1	Admiralty
	Point Beach 2	Admiralty
	Prairie Island 1	Stainless steel
	Prairie Island 2	Stainless steel
	Robinson 2	Admiralty brass and stainless steel
	Salem 1	90-10 CuNi and titanium
	San Onofre 1	90-10 CuNi and titanium
	Surry 1[1]	90-10 CuNi
	Surry 2[2]	90-10 CuNi
	Trojan	Admiralty brass and 70-30 CuNi
	Turkey Point 3[1]	Al-brass
	Turkey Point 4[1]	Al-brass
	Yankee Rowe	Admiralty and stainless steel
	Zion 1	Stainless steel
	Zion 2	Stainless steel
CE	Arkansas 2	90-10 CuNi
	Calvert Cliffs 1	70-30 CuNi
	Calvert Cliffs 2	70-30 CuNi
	Fort Calhoun 1	Stainless steel
	Maine Yankee	Al-brass and 70-30 CuNi
	Millstone 2[2]	70-30 CuNi
	Palisades 1[3]	90-10 CuNi
	St. Lucie 1	Al-brass

APPENDIX C (continued)

NSSS Vendor	Plant Name	Condenser Tube Material
B&W	Arkansas 1	Admiralty
	Crystal River 3	70-30 CuNi
	Davis-Besse 1	Stainless steel
	Oconee 1	Stainless steel
	Oconee 2	Stainless steel
	Oconee 3	Stainless steel
	Rancho Seco 1	Stainless steel
	Three Mile Island 1	Stainless steel
	Three Mile Island 2	Stainless steel

Notes: [1] Retubing condenser with titanium.
[2] Was Al-brass up to 5/77.
[3] Previously Admiralty.

GLOSSARY

extensive denting - (a) presence of tube denting that is widespread throughout whole steam generator in which the average total reduction in tube diameter equals to or exceeds twice the tube wall thickness; (b) measurable support pate in-plane deformations, such as hourglassing of flow slots in Westinghouse plants; (c) damage has caused leaking from dents.

moderate denting - (a) presence of the tube denting that is widespread throughout whole steam generator in which the average total reduction in tube diameter exceeds 20 percent of the tube wall thickness; (b) no measurable support plate in-plane deformation; (c) damage has not caused leaking from dents.

minor denting - (a) presence of tube denting is spotty to widespread, but the average total reduction in tube diameter is less than 20 percent of the tube wall thickness; (b) no visible support plate deformation; (c) damage has not caused leaking from dents.

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16. ABSTRACT (200 words or less)

Operating problems have occurred in the steam genertors of each of the three manufacturers of pressurized water reactors (PWR) nuclear steam supply systems (NSSS): Babcock & Wilcox, Combustion Engineering, and Westinghouse Electric Corporation. This report focuses on the problems associated with steam generators of the recirculation type that are designed by Westinghouse and Combustion Engineering. It identifies the operational problems observed to date, including the NRC staff's evaluation of such problems, and provides a status report summarizing the NSSS, licensee, and staff programs for the resolution of each problem.

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