UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket Nos. 50-400 OL
and NORTH CAROLINA EASTERN	)	50-401 OL
MUNICIPAL POWER AGENCY	)	
	)	
(Shearon Harris Nuclear Power	)	
Plant, Units 1 and 2)	)	

AFFIDAVIT OF THOMAS F. TIMMONS

County of Allegheny	)	
	)	ss.
Commonwealth of Pennsylvania	)	

THOMAS F. TIMMONS, being duly sworn according to law, deposes and says as follows:

1 My name is Thomas F. Timmons. My business address is P.O. Box 355, Pittsburgh, Pennsylvania 15230. I am employed by Westinghouse Electric Corporation as the Manager of Reactor Coolant System Components Licensing in the Nuclear Safety Department of the Nuclear Technology Division.

2 I was graduated from Marquette University in 1968 with a Bachelors degree in Mechanical Engineering. Upon graduation, I received a commission as an Ensign in the U.S. Navy and was assigned to the U.S. Navy Nuclear Power Program. From August 1968 to August 1969, I successfully completed the courses of study at the U.S. Navy Nuclear Power School at Mare Island, California and at the U.S. Navy Nuclear Power Prototype School at Idaho Falls, Idaho.

3 From September 1969 to September 1972, I was assigned to the Engineering Department of the USS Bainbridge (DLGN-25), a nuclear-powered, guided-missile frigate. I served as the Reactor Laboratories Officer in charge of technicians who controlled reactor coolant system and steam generator chemistry and all radiological monitoring and control for the twin nuclear reactor plants of the USS Bainbridge. My subsequent duties included serving as the Electrical Officer in charge of the ship's electrical generation and distribution systems and components, including the electrical systems and components of the nuclear power plants. I was also qualified as Engineering Officer of the Watch (EOOW) and as Engineering Duty Officer (EDO). As EOOW, I supervised the operation and maintenance of the nuclear reactor plants while they were in operation and as EDO, I supervised the Engineering Department, including operation and maintenance of the nuclear reactor plants while the ship was in port. My separation from the U.S. Navy was as a Lieutenant, senior grade.

4 From January 1973 to August 1975, I was employed by the WEDCO Corporation, a subsidiary of Westinghouse Electric Corporation, at the Indian Point Nuclear Station as an Electrical Startup Engineer in the Operations Department. My duties included reconciliation of as-built conditions with drawings, post-installation electrical checkout of control circuits, motors, circuit breakers, etc. and supervision of startup testing.

5 From September 1975 to January 1980, I was employed in the Safety Standards Group in the Nuclear Safety Department of the PWR Systems Division of the Westinghouse Electric Corporation located at Monroeville, Pennsylvania. There I held positions as an engineer and as a senior engineer. My duties included development, evaluation and application of safety criteria in safety evaluations of nuclear power plant components and systems. During 1978 and 1979, I also coordinated a research program on nuclear power plant operator response to accident situations.

6 In February 1980, I was appointed Manager of Mechanical and Fluid Systems Evaluation in the Nuclear Safety Department. In this capacity, I was responsible for licensing activities and safety evaluations in the areas of fluid systems and mechanical components, including steam generators, for operating and non-operating plants with Westinghouse Nuclear Steam Supply Systems. In March 1982, I was appointed Manager of Reactor Coolant Systems Components Licensing, a position that I currently hold. In this position, I am responsible for all licensing activities and safety evaluations for the reactor coolant system and its components, including steam generators, for operating and nonoperating plants. Between March 1982 and September 1983, I was also assigned a collateral position as Manager, Licensing for the Model D Steam Generator Task Force where I was responsible for the licensing support activities necessary for resolution of the flow induced vibration issue.

7 I make this Affidavit in support of Applicants' Motion for Partial Summary Disposition of Joint Contention VII (Steam Generators). I have personal knowledge of the matters stated herein and believe them to be true and correct. Part I of this affidavit addresses Joint Intervenors' Contention VII insofar as this contention concerns flow induced tube vibration in the preheater section of Westinghouse Model D steam generators. Part II of this affidavit addresses Joint Intervenors' Contention VII insofar as this contention concerns the effect of secondary water chemistry on tube integrity and explains the history and development of the water chemistry guidelines recommended by Westinghouse to utilities for control of the water chemistry of the secondary side of operating Pressurized Water Reactors.

<u>Part I</u>

8 The Carolina Power & Light Company's (CP & L) Shearon Harris Nuclear Power Plant Unit 1 utilizes a Westinghouse designed nuclear steam supply system (NSSS) consisting of three recirculating reactor coolant loops. Each loop contains a Westinghouse Model D4 steam generator. Within each steam generator there are 4578 inverted U-shaped steam generator tubes, collectively referred to as the tube bundle. The tubes act as the

pressure boundary between the primary (reactor coolant) water and the secondary (steam producing) water. The tubes are secured at the end of each leg of the "U" to a thick steel plate known as the tubesheet. The tube sheet also separates the primary water from the secondary water. The hot reactor coolant water flows through the inside of the tubes. The tube bundle is immersed in relatively cool secondary water which is raised to steam producing temperatures by the transfer of heat through the walls of the steam generator tubes from the primary water to the secondary water. High quality steam exits the top of the steam generator and is used to drive turbines which in turn drive a generator to produce electricity.

9 An outline drawing of a Model D4 preheat steam generator is shown at Attachment 1. As illustrated in Attachment 1, the preheat region is located on the "cold leg" (coolant exit) side of the tube bundle and faces the feedwater inlet nozzle. Within the Model D preheat steam generator series, there are two general types of steam generators. These are the split-flow type and the counter-flow type. These two types of steam generators differ in the configuration of the inlet nozzle water box area and in the flow paths for the feedwater in the preheater itself.

10 In the split-flow type (see Attachment 2), the incoming feedwater enters at the midsection of the preheater, is directed outward over the front of the outer row of tubes, and then enters the tube bundle. At the rear of the inlet pass (at the centerline of the steam generator) the flow "splits" with portions being directed upward and downward around the tubes and baffles.

11 In the counter-flow type (see Attachment 3), the incoming feedwater enters the inlet water box and impinges on a wall that directs the water outward to fill the water box volume, and downward to the preheater inlet pass located near the bottom of the steam generator, before entering the tube bundle. The water then enters the tube bundle at the inlet pass, flows around the tubes and then upward around the tubes and baffles. This upward flow is "counter" to the direction of the flow of

the primary water inside the steam generator tubes. The counter-flow models are also equipped with a "T"-shaped blowdown pipe to minimize the accumulation of sludge on the tubesheet. This blowdown pipe is accommodated by a "T"-shaped lane within the tube bundle (see Attachment 4). Model D4 steam generators such as those installed in the Harris Plant are of the counter-flow preheat design.

12 Because of experience involving tube vibrations in operating split-flow type preheat steam generators, Westinghouse undertook an extensive program to investigate, understand and define vibration and tube wear in Model D4 steam generators and to conceive, develop, test and evaluate any modifications necessary to allow safe operation of Model D4 steam generators at full power. This program included gathering, reviewing and analyzing data from operating plants and from laboratory and model tests.

13 Eddy current testing (ECT) of the the only Model D4 steam generators in operation at that time began in November 1981. Prior operation of these steam generators had been at power levels up to 50 percent. The results of ECT revealed no indications of tube wear. Based on these data, Westinghouse recommended an interim maximum operating condition of 50 percent main feed flow rate for the next operating interval.

14 After a period of operation at 50 percent main feed flow rate, another ECT inspection was performed and vibration instrumentation was installed in January 1982. Again, the ECT data did not reveal indications of tube wear. Upon startup, vibration data were obtained at various power levels and feedwater configurations. This additional ECT data and the tube vibration data were reviewed and analyzed by Westinghouse.

15 This analysis of the operating D4 steam generator data was used to determine the extent of tube vibration and to determine the main feed flow rates below which wear-producing tube vibration would not be expected to occur. This flow rate was determined to be 70 percent of the full main feed flow rate. Based on this plant data and laboratory

test data available at that time, Westinghouse increased the recommended maximum main feed flow to 70 percent and recommended 1500 hours as a prudent interim operating period.

16 After the November 1981 and January 1982 ECT inspections, a new calibration method for eddy current testing was qualified. The accuracy of the new method was verified by metallurgical examination of tubes removed from operating preheat steam generators. This new calibration method for ECT provided more accurate data on the depths of tube wear marks and was used in all subsequent ECT inspections of preheat steam generators.

17 During the period from November 1981 until May 1982, Westinghouse performed analyses, evaluation and correlation of available data from removed tubes, ECT, vibration instrumentation and laboratory testing. As a result, Westinghouse developed a reliable, empirical data base with respect to operation of Model D4 steam generators. This data base permitted Westinghouse to make conservative estimates of potential tube wear for operating conditions extending to 100 percent main feed flow.

18 In May 1982, the next ECT inspection was performed on the operating D4 steam generators. Additional vibration measurement instrumentation was installed, and one tube was removed from a steam generator for visual and metallurgical examination and analysis. No indications of tube wear were detected by ECT. The removed tube had some wear with a depth below the limit of ECT detectability. In addition to the inspections and installation of instrumentation, modifications were made to the feedwater bypass system to allow operation of the plant at 100 percent power with up to 30 percent of the feedwater flow bypassing the preheater of the steam generator.

19 After resumption of operation, vibration data was collected at various power levels and various combinations of main feed and bypass feed flows (e.g. 70/0, 70/30, 90/10, 100/0, etc.). From this data it was observed that the tube vibrations at the 70 percent main feed/30 percent bypass

feed combination were slightly greater than those at the 70/0 combination and that the vibrations observed at 70/30 had limited wear potential that would not preclude extended plant operation.

20 In November 1982, another ECT inspection was performed on the operating D4 steam generators. Additional tube vibration measurement instrumentation was installed, two tubes were removed and one tube was expanded at baffle plate intersections. No indications of tube wear were observed from the ECT inspection. The two removed tubes had wear marks of .001 to .002 inches in depth, values which are below the limit of ECT detectability. After resumption of operation, tube vibration data were again obtained. From this data, it was observed that the tube vibrations in the two operating D4 steam generators were similar and had not changed with time. The expanded tube, which had been previously instrumented for vibration measurement during the operating interval prior to expansion was re-instrumented following expansion for this test. Pre-expansion vibration data for this tube were compared with post-expansion data. From this comparison, it was concluded that expansion reduced tube vibrations by at least a factor of 5. Based on these ECT and vibration data, Westinghouse recommended that the operating interval at 70/30 main feed/bypass feed combination be increased to approximately 4500 hours before the next ECT inspection.

21 In addition to use for determining appropriate operating levels and intervals for the operating D4 steam generators, the ECT data, tube vibration data and inspection data from the removed tubes provided baseline information that was used to calibrate and qualify various size laboratory test scale models and a flow-induced-vibration dynamic analysis model. These models were used to obtain data not available from the operating plant. These data were used for the development and evaluation of further modifications to reduce flow induced vibration.

22 For the counter-flow steam generator program a 0.95 scale air model, a 2/3 scale water model, a single tube vibration model, and a 16 degree full scale water model were constructed.

23 The 0.95 scale air model was used to determine flow velocity distributions within the preheater. The flow distribution patterns obtained from this model were then verified in the 2/3 scale water model. In addition to determining the shell side local water velocities and flow distribution patterns, the 2/3 scale water model permitted measurement of the drag forces on the tubes and the pressure drops at various locations within the preheater.

24 A single tube vibration model was used to characterize tube response under various excitation and support conditions. Here, a device was used to vibrate a full length tube. Support plates were located at the same elevations as in the actual steam generator.

25 A full scale model of a 16 degree steam generator tube bundle sector was used to replicate in the laboratory the tube vibration response observed in operating steam generators. This model consists of one half of the preheater inlet region (the other half is symmetrical). Within this region, all of the tubes were installed, with those tubes contained within a 16 degree sector being full length (up to the U-bend elevation). This model was used to test various tube/support plate interactions with various inlet flow velocities and distributions.

26 The use of these various test models provided the additional capability of testing various concepts designed to reduce tube vibrations. In this manner, several design concepts were rejected while others were optimized.

27 Concurrent with collecting and analyzing operating plant and laboratory test data, a flow-induced-vibration dynamic analysis computer model was developed to predict tube behavior. This model is a multi-span dynamic analysis model which uses annular gap elements at the support plate locations and is thus able to simulate tube response within the support plate clearance. The gap elements can be offset to simulate various support conditions. Results from this model were correlated with data obtained from operating plants and scale models.

28 On the basis of the results of the analysis and test program, Westinghouse has recommended that Carolina Power and Light Company make modifications to the Harris plant to reduce the potential for tube vibrations in the Harris steam generators. These modifications are: 1) the expansion at baffle locations of 124 tubes per steam generator and 2) the bypassing of approximately 18 percent of the flow from the main feedwater nozzle to the auxiliary feedwater nozzle. The expansion of tubes at baffle plate locations will limit the tube movement at the baffle plate intersections to a few thousandths of an inch. The bypassing of 18 percent of the main feed flow to the auxiliary nozzle of the steam generator will reduce the main feed flow at the inlet to the preheater to approximately 82 percent and will further reduce the potential for vibration of the tubes in the preheater.

29 Westinghouse developed a proprietary process that was used to expand steam generator tubes at baffle locations. The process involves the insertion of tools into the tubes from the primary side of the steam generator tubesheet. The tools are then used to locate the baffle plate intersection and to expand the tube at the appropriate location. The expansion zone is entirely within the thickness of the baffle plate.

30 As part of the expansion procedure, tubes are eddy current inspected before and after the expansion process. These inspections would provide an indication of cracks or pits which may have existed prior to or following tube expansion. Results of these inspections for the Shearon Harris steam generator tube expansion program revealed no such indications. Also, in an extensive program of expansion process qualification, laboratory examination of tubes following expansion indicated that tube integrity is maintained.

31 Expansion of the tubes which occurs only at the ends of tubes within the tubesheet and at the B and D baffles (see Attachment 3), in no way precludes the ability to sleeve such tubes. Further, because all tubes are expanded at their ends, and tube plugs have always been designed for such expanded tubes, the integrity of plugs in the expanded tubes at the B and D baffles would be the same as that in non-expanded tubes.

32 Expansion of tubes in steam generators has long been utilized in the manufacture of steam generators. Favorable field experience indicates the acceptability of the tube expansion process. To supplement the field experience data base, Westinghouse established a program to evaluate the effect of tube expansion at the baffle plate. This program included an evaluation of the levels of residual stresses in expanded tubes, corrosion testing, structural analysis and an evaluation of the effect of expansion on the required tube minimum wall thickness.

33 Westinghouse conducted an extensive test program to assess the levels of residual stress and the potential for stress corrosion cracking in expanded tubes. Test results indicate that any change in the level of residual stress due to the expansion process is not detectable using established testing techniques. In addition, testing has shown that the baffle plate expansion results in lower residual stress than does the tube sheet expansion which has been shown to be acceptable from favorable operating field experience. These test results, combined with the retarding effect on stress corrosion cracking rates of the relatively low operating temperatures in the preheater, are the basis for the conclusion that tube expansion does not significantly affect the potential for stress corrosion cracking.

34 Westinghouse has also conducted accelerated corrosion testing to assess the effects of the reduced tube-to-tube-hole clearance on the potential for denting of the expanded tubes. The results of this testing indicate that the potential for denting is not increased for tubes expanded at the baffle intersections.

35 Westinghouse has also performed structural analyses of the expanded tube for design basis transients and accidents. The amount of tube wall reduction resulting from the expansion process was determined from the expansion process qualification program. This expanded tube configuration was evaluated for faulted plant conditions per ASME Code, Section III, Subsection NB requirements. Finite element analysis was applied using conservatively established boundary conditions. The results of such structural analyses indicate that the ASME Code allowable values for stresses and fatigue usage factors are not exceeded for expanded tubes.

36 An evaluation of the effect of tube expansion on the minimum required tube wall thickness was also completed. The minimum tube wall thickness is that required to meet applicable stress and burst strength requirements with specified margins of safety. Westinghouse determined that an increase of less than 0.5 mils in the minimum tube wall thickness satisfies stress and burst strength requirements in the expanded region. The draft Shearon Harris Technical Specifications currently define tube plugging criteria which would require that tubes be removed from service or repaired when tube wall thickness is less than the tube plugging limit wall thickness. The margin between the tube plugging limit wall thickness and the increased minimum tube wall thickness for the expanded region is conservative and meets applicable regulatory requirements.

37 The above evaluation considered the geometric effects of tube expansion (i.e., wall thickness and diameter). The effect of tube expansion on tube material properties was also considered. The plastic deformation induced by the expansion process results in a strain hardening of the tube, which elevates the burst pressure of the tube. While this strain hardening effect represents a small decrease in ductility, the resultant tube yield and ultimate strengths for the range of steam generator operating temperatures provide for a tube material well within the ductile range. Thus, while the tube response to postulated loading conditions remains ductile, the increased burst strength of the material provides additional margin in the determination of minimum required wall thickness in the expanded region.

38 Another aspect of the overall evaluation of the expanded tube is the effect of expansion on standard tubing repair procedures such as sleeving and plugging. Both sleeving and plugging processes have been qualified considering a change in tube inside diameter along the length of the tube. Steam generators have always had some form of tube expansion within the tubesheet. The variation of tube inside diameter along the length of the tube has thus been a factor in all previous field sleeving and plugging operations. The large data base of favorable field experience provides confidence that expansion at baffle plates will not impair the ability to sleeve or plug the tube.

39 In addition to tube expansion, Westinghouse has recommended a feed system modification that would allow bypassing a percentage of feed flow to the auxiliary feedwater nozzle during power operation. For the Harris plant, the feedwater bypass modification will result in approximately 82 percent of the feedwater flow entering the main feedwater nozzle and the remainder of the feedwater flow entering the steam generator through the auxiliary feedwater nozzle for full power operation.

40 Westinghouse has tested the tube expansion and feed system modifications in the 16 degree Model and in the operating Model D4 steam generators. In the 16 degree Model, a number of tubes were expanded and testing was conducted to determine the effect of the modifications on tube vibration. At a flow rate equivalent to 82 percent of the Harris full power main feed flow rate, the expanded tubes exhibited vibration levels that were less than those observed at flow rates equivalent to 70 percent of the operating D4 steam generators full power main feed flow rate without tube expansion. As I indicated earlier, a 70 percent main feed flow rate limits tube wear potential so as not to preclude extended plant operation.

41 The Westinghouse modification evaluation program and results were reviewed in detail by a third-party review group consisting of representatives of utilities with counter-flow preheat steam generators. The objective of this review group was to determine the acceptability of the counter-flow preheat steam generator modifications recommended by Westinghouse and to submit a report to the NRC with the review group's conclusions. The third-party review included the following areas: thermal hydraulics, model testing, radiological considerations, structural mechanics, stress analysis, inservice inspection, tooling, chemistry and installation considerations. On the basis of this review, the group concluded that the modification to the counter-flow steam generators can be made, that it does not introduce any unresolved safety issues and that the modified steam generators can be operated safely at rated capacity.

42 The NRC has reviewed the report and conclusions of the third-party review group. Both the third-party review and the NRC review are documented in NUREG-1014, "Safety Evaluation Report related to the D4/D5/E Steam Generator Design Modification." The NRC found that, pending plant-specific verification and documentation of safety analyses, the implementation of Regulatory Guide 1.121 concerning minimum tube wall thickness and plugging criteria for expanded regions, and in-service inspection (ISI) performed according to NUREG-0452 with inspection of all expanded tubes after 6 effective full power months (EFPM) and before 12 EFPM, that the modifications of counter-flow preheat steam generators are acceptable and that the modified steam generators can be operated at 100 percent of their design capacity. It is my understanding that CP L has committed to meet these requirements at the Harris plant.

43 Based on the extensive program that has been conducted by Westinghouse and has been reviewed by the Counter-Flow Steam Generator Owners Group and the NRC, it is my judgement that the Harris plant steam generators can be safely operated at all power levels.

Part II

44 A simple schematic of the secondary side of a typical Westinghouse Pressurized Water Reactor (PWR) is shown in Attachment 5. This identifies the interrelationship of the secondary side system (commonly called the "Balance of Plant") to the turbine and steam generator. Secondary side water chemistry controls are aimed at minimizing the input of impurities into the secondary system. These impurities include dissolved and suspended solids which, if not controlled, could lead to corrosion of the secondary side materials.

45 Water chemistry control of the secondary side of Pressurized Water Reactor (PWR) Nuclear Power Plants has been developed from the extensive experience accumulated by operation of industrial fossil-fueled boilers. The two traditional methods for chemistry control of the steam side of power plants are:

1. Sodium phosphate treatment.
2. All Volatile Treatment (AVT)

46 The sodium phosphate treatment uses dissolved solids which act as a buffer to the concentrated contaminant species and has traditionally been used in recirculating boilers; AVT, as the name implies, utilizes only all volatile chemicals, has been used in once-through boilers, and is presently used in the majority of PWR steam generators of several different manufacturers throughout the world.

47 The operation of the secondary side of a PWR and its associated water chemistry control program is the responsibility of the utility. Westinghouse has traditionally consulted with utilities on chemistry control programs and has provided guidelines for secondary side chemistry control to each utility. The objectives of the Westinghouse PWR secondary water chemistry guidelines are:

1. To minimize metal corrosion.

2. To control the extent of sludge accumulation in the steam generator.

3. To minimize scale formation on heat transfer surfaces.

4. To minimize the potential for the formation and accumulation of corrosive species (such as free caustic or acid).

5. To minimize the level of dissolved oxygen and carbon dioxide.

48 Westinghouse steam generators are of the recirculating type. As a result, the initial recommendation in the early 1960's was to use phosphate as the steam generator chemistry control agent. The phosphate was maintained at a desired concentration to provide the pH buffering action against excursions in alkalinity or acidity due to ingress of contaminants.

49 The initial guidelines stipulated a phosphate concentration of up to 10 ppm phosphate with a sodium-to-phosphate molar ratio of less than 2.6. The ratio was chosen because earlier work had shown that higher ratios led to formation of sodium hydroxide, a caustic, which is detrimental to the Inconel 600 steam generator tubing.

50 The initial guidelines had allowed the utility, should it so desire, to operate with no added phosphate. Using the initial water chemistry guidelines, some instances of stress corrosion cracking were observed in plants that operated with excessive condenser inleakage and inadequate phosphate control.

51 The initial guidelines were revised to address this issue. Ranges of phosphate concentration with minimum levels of phosphate were recommended. The recommended sodium-to-phosphate molar ratio remained at less than 2.6.

52 The recommended treatment (i.e., sodium-to-phosphate molar ratio less than 2.6 with higher phosphate concentrations) was successful in mitigating the incidence of caustic stress corrosion cracking of the steam generator tubing which had resulted from condenser inleakage and inadequate phosphate control. However, after approximately two years, another corrosion phenomenon called "thinning" was observed in some plants.

53 Laboratory experimentation was performed using high temperature autoclaves and model boilers to understand the thinning phenomenon. The results of these studies together with field experience showed that the thinning was caused by low sodium-to-phosphate molar ratio solutions. This led to a further adjustment in the guideline to a sodium-to-phosphate molar ratio between 2.3 and 2.6. However, results from the lead plants using this guideline showed that the thinning was not abated. Therefore, in the latter part of 1974, all volatile treatment (AVT) was recommended in lieu of phosphate control.

54 As the name implies, the basis for the All Volatile Treatment is that only volatile chemicals are intentionally added as control agents. These agents do not concentrate in the steam generator but are removed via the steam and circulate through the secondary system.

55 In general, two chemicals are added, a volatile amine (usually ammonia as ammonium hydroxide) for pH control of the feedwater and an oxygen scavenger (hydrazine). Hydrazine scavenges oxygen according to the following simple equation:

$$O_2 + N_2H_4 > N_2 + 2H_2O$$

producing innocuous byproducts, nitrogen and water. As the hydrazine moves through the feedwater system and is subjected to higher temperatures, any unreacted hydrazine can decompose to form such compounds as ammonia, nitrogen and hydrogen. These by-products are volatile compounds. Since ammonia is produced by thermal decomposition of hydrazine, the addition of ammonium hydroxide for pH control is adjusted to compensate for that produced from the thermal decomposition of the excess hydrazine.

56 The change to AVT mitigated the thinning of the steam generator tubing caused by acidic phosphate species. However, in the immediate years following the conversion to AVT, another form of corrosion called "denting" was observed.

57 Denting is a localized radial reduction in the diameter of steam generator tubes, resulting from corrosion of the carbon steel tube support plates in the tube-to-tube-support-plate annulus. Initially, it was thought to be associated only with plants with prior extended phosphate treatment. However, laboratory and field data indicated that corrosion, which can produce denting, occurred in chemistry environments where no phosphate was present. The data showed that copper and chloride are active in the denting process. Plants with copper alloys in their balance of plant and situated on the sea coast or utilizing brackish water as a cooling water were observed to be more prone to this form of corrosion because of the high concentrations of acid-chloride forming constituents in this cooling water.

58 Ongoing laboratory programs and specific field tests have established that the denting corrosion process is multivariate with an interrelationship between acid chlorides, and reducible metal compounds like copper oxide and oxygen. It has also been shown from the field tests that the steam generator corrosion can be initiated by contaminant ingress from many locations in the secondary system. Contaminants introduced into the steam generator bulk water can also be carried to the turbine and to the rest of the equipment in the secondary plant cycle by either carryover or by the volatility of the particular chemical compound.

59 Two changes were recommended to address the denting phenomenon. One was to reduce the input of oxidizing agents such as copper oxide and/or oxygen and the other was to further restrict the input of chloride ions into the system.

60 In 1976, Westinghouse recommended that the balance of plant materials be considered relative to reduction of sludge generation and corrosion product transport into the steam generators. It was stressed that (1) the presence of copper alloys in the system may preclude the optimum chemistry control for carbon steel protection and (2) copper corrosion products entering the steam generators appear to remain in the sludge rather than be removed by blowdown. As indicated above, these copper compounds together with chloride ions from condenser inleakage play a major role in the initiation of the corrosion leading to denting.

61 In 1977, interim AVT guidelines were established with the following operating practices for restricting chloride ion and oxygen input:

1. Operate to the normal power operation AVT specifications only.

2. Implement rigorous control of the condensate and feedwater chemistries during both shutdown and power operation to minimize secondary system corrosion and transport of the corrosion products into the steam generators.

3. Identify sources of contaminant ingress and eliminate them immediately upon detection.

62 These changes were recommended mainly because data collected from operating plants, field programs, and laboratory experiments indicated that a coordinated systems design approach is desirable. The steam generator is only one component of a complex secondary system that integrates system design and operational considerations. A joint effort by the nuclear equipment supplier, the utility, and the architect-engineer is essential to produce a coordinated systems approach.

63 Based on this understanding of the denting phenomenon and its relationship to chemistry control, plants that have only operated with an AVT chemistry program in accordance with Westinghouse and industry guidelines have successfully operated without signs of denting corrosion. In addition, plants which had previously experienced denting have had denting arrested after adopting a chemistry control program in conformance with established guidelines. As indicated earlier, plants with fresh water condenser cooling such as Harris are much less prone to this form of corrosion due to the low concentrations of acid-chloride forming constituents in the condenser cooling water.

64 Although there may be some corrosion even using AVT, corrosion per se is not a safety issue because corrosion generally develops slowly and corroded tubes should be detected in the required, periodic in-service inspections (ISI) and can be plugged or sleeved before exceeding tube structural integrity limits. Even if corrosion were such that a leak would occur, it has been shown that the leak should be small and detectable. Continuous monitoring would allow the operator to follow the leak and shut the plant down before the leak approaches such limits. Such a leak should be small enough that any leakage through the tube would be less than Technical Specification limits. Because of the non-uniformity of the degradation that has been observed in operating plants and because of the inherent toughness of the Inconel 600 material, the tube would be expected to leak rather than be susceptible to large ruptures.

65 Other corrosion mechanisms have been observed in plants where AVT has been the exclusive water chemistry control; however, these have been limited occurrences. For example, one plant experienced pitting of the Inconel tubing, which is believed to be due to an acidic chloride condition involving copper and chloride ions. Also, plants using AVT since initial startup have experienced a limited incidence of stress corrosion cracking or thinning incidents.

66 Although the principles of the interim guidelines established by Westinghouse in 1977 have been maintained, certain modifications to the guidelines have evolved since 1979. These include expected parameter levels considerably lower than the previous normal operation limits and increased attention to the control of air ingress to the subatmospheric sections of the condensate system.

67 The revised guidelines make recommendations concerning achieving, monitoring and maintaining desirable secondary side chemistry operating conditions; minimizing the generation and transport of corrosion products to the steam generators; and selecting appropriate materials for use in the main feed and condensate systems.

68 The original conversion from a phosphate treatment to the All Volatile Treatment recommended by Westinghouse guidelines has eliminated the steam generator tube corrosion ascribed to the presence of phosphate. In addition, full depth expansion of the tubes in the tubesheet minimizes the crevice between the tube and the tubesheet and, together with AVT chemistry control, minimizes the potential for crevice corrosion previously experienced by operating plants without full depth expansion. Over the last nine years, the utility industry has come to understand more fully the importance of good secondary side water chemistry control. Experience has shown that with diligent operation of the plant by the utility, the Westinghouse guidelines can be achieved and, in my opinion, will minimize the potential for corrosion.

69 In addition, I should emphasize that these measures are under continual review by the nuclear industry. Westinghouse, with other nuclear power

system vendors, is involved in a joint endeavor with utilities to emphasize the importance of maintaining appropriate water chemistry conditions in the secondary systems of PWR nuclear power plants. In order to reinforce the need for rigorous chemistry control, the Steam Generator Owners Group (SGOG) has issued AVT guidelines as a model for the industry.

70 I have been informed that Harris has adopted chemistry guidelines that are consistent with the SGOG and Westinghouse chemistry guidelines and has established procedures and controls to address any potential off-normal secondary-side chemistry conditions. Careful attention to the maintenance of proper chemistry conditions on the secondary side of the plant and proper maintenance of the feed and condensate trains should provide the necessary conditions to minimize tube corrosion and cracking and to permit long-term integrity of the steam generators.

71 Because of the recent emphasis on secondary side water chemistry, the realizations by the utilities that strict chemistry controls are necessary to minimize corrosion and the fact that Harris has adopted the above guidelines, I am confident that Harris can be operated at all power levels in a safe manner in such a way as to minimize corrosion.

Thomas F. Timmons

Sworn to and subscribed before me
this 15th day of May, 1984.

Lorraine M. Piplica
Notary Public

My Commission Expires:

LORRAINE M. PIPLICA, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES DEC 14, 1987
Member, Pennsylvania Association of Notaries

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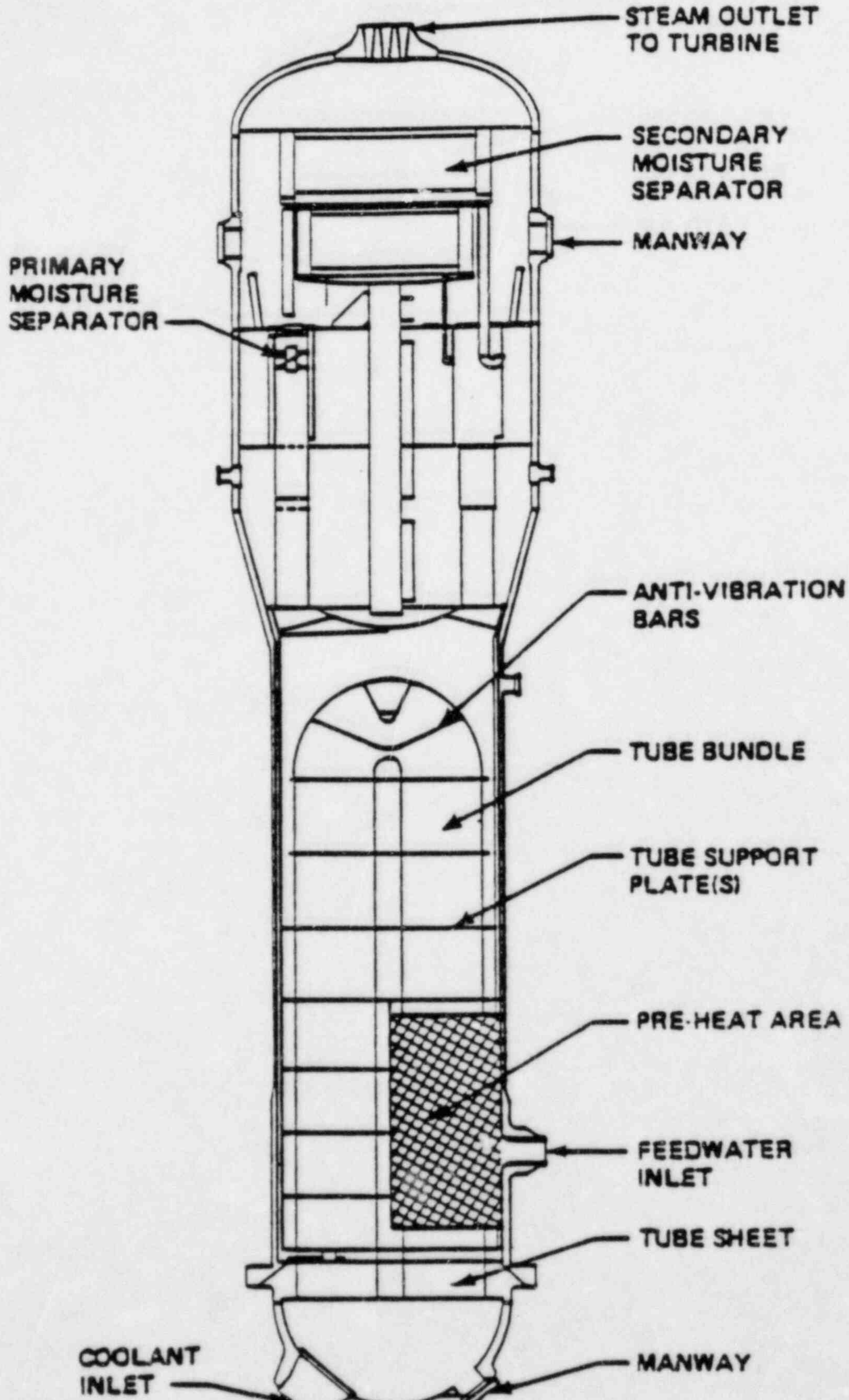
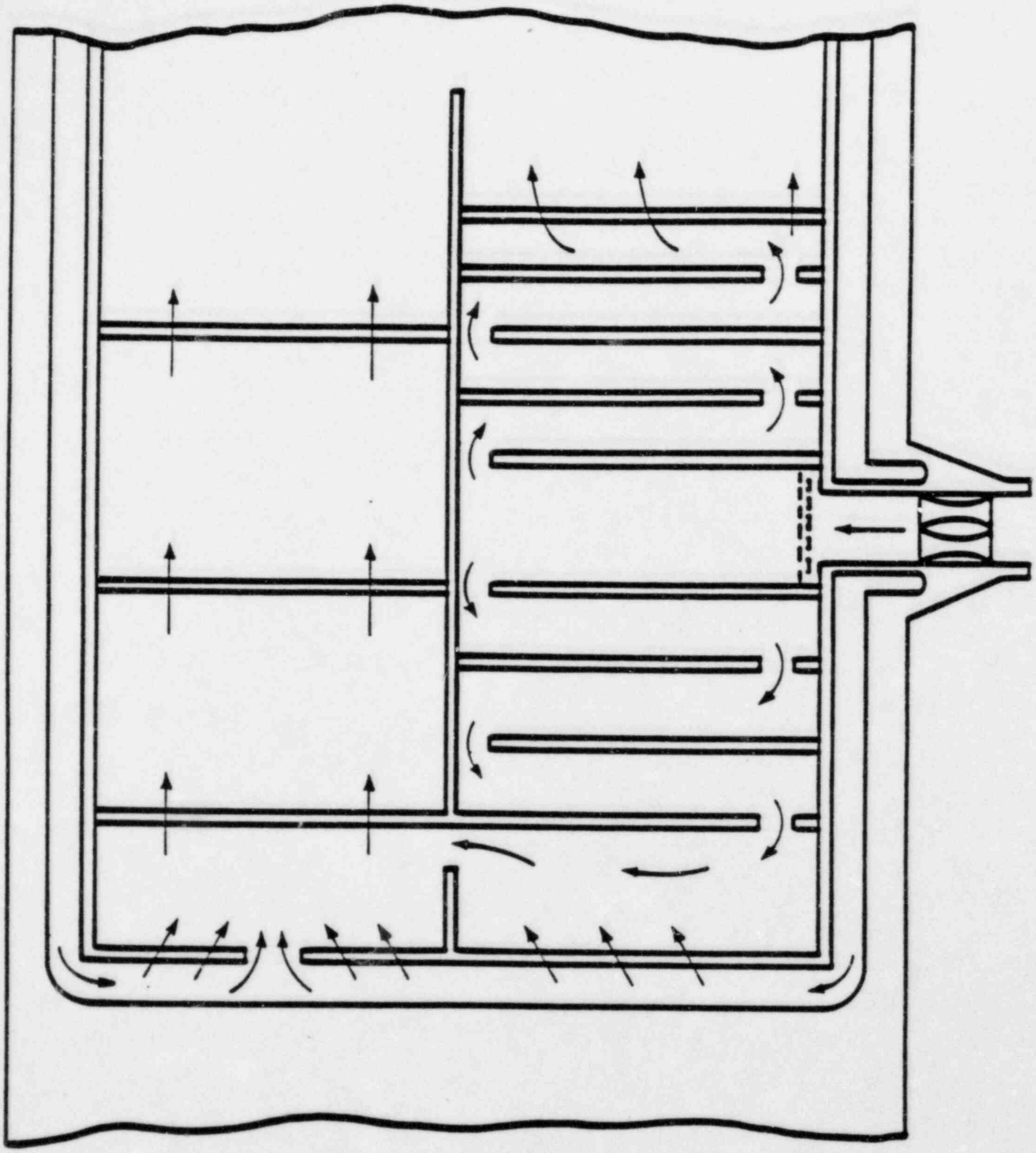
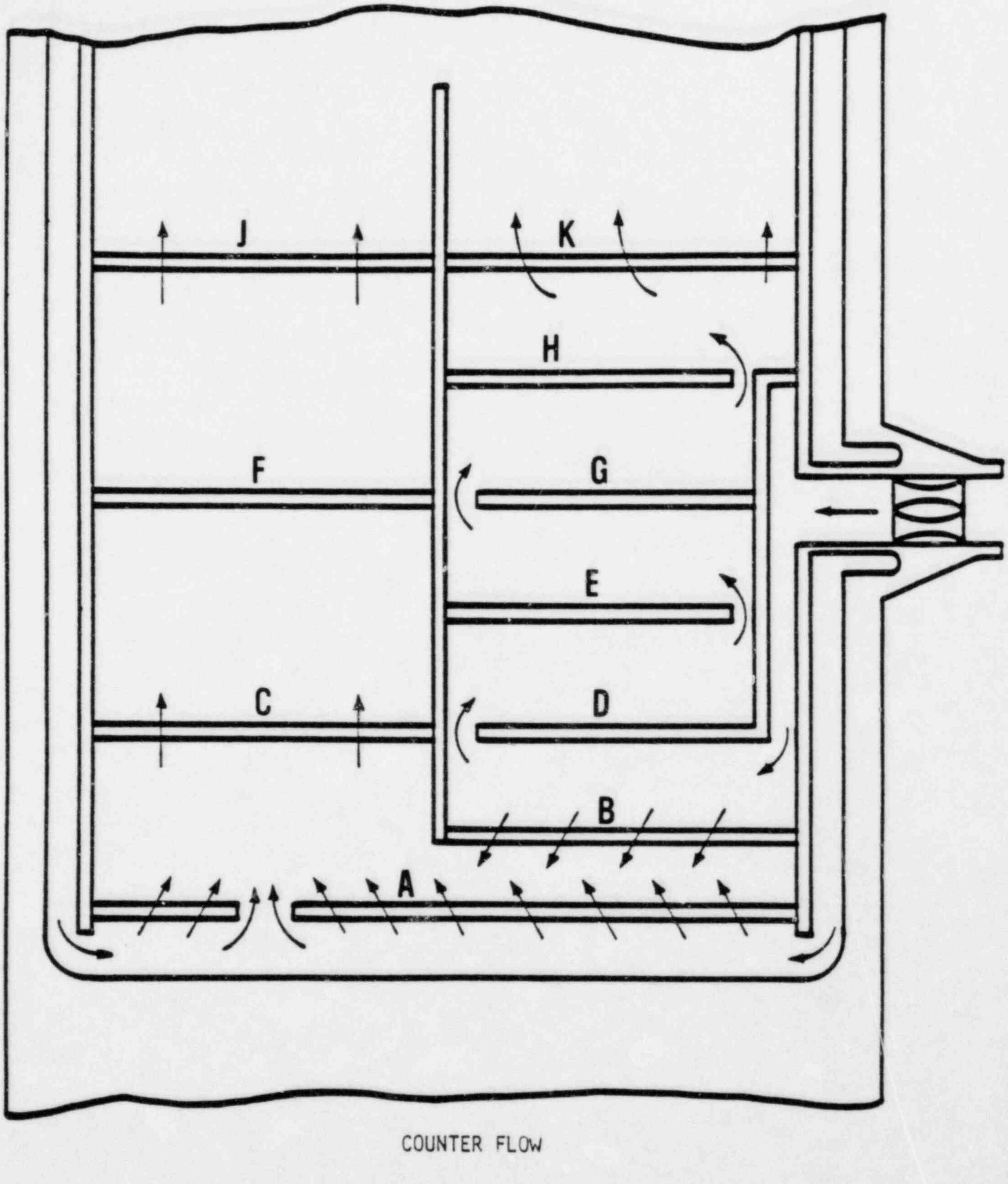


Preheat Steam Generator



SPLIT FLOW

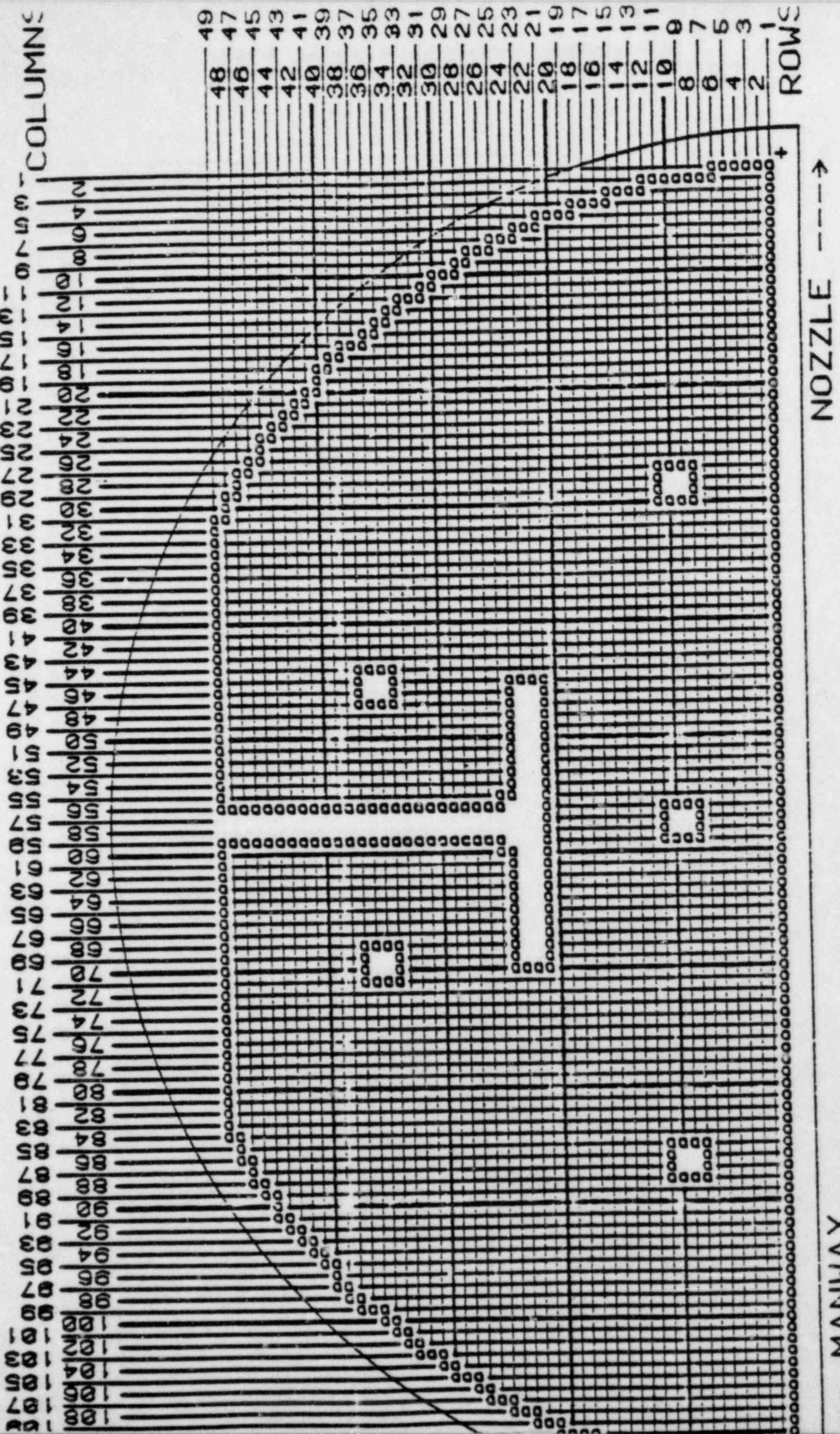
Preheat Steam Generator



COUNTER FLOW

Preheat Steam Generator

Attachment 4



Attachment 5