UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
GEORGIA POWER COMPANY, et al.	)	Docket Nos. 50-424 (OL)
	)	50-425 (OL)
(Vogtle Electric Generating Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF CARL W. HIRST

County of Allegheny)
) ss.
Commonwealth of Pennsylvania)

I, Carl W. Hirst, being duly sworn according to law, depose and say as follows:

1. I am Manager of Reactor Coolant System Components Licensing for the Nuclear Technology Division of Westinghouse Electric Corporation. My business address is Westinghouse Electric Corporation, Monroeville Nuclear Center, P.O. Box 355, Pittsburgh, PA 15230. A summary of my professional qualifications and experience is attached hereto as Exhibit A, which is incorporated herein by reference.

2. This Affidavit is offered in support of "Applicants' Motion for Summary Disposition of Joint Intervenors' Contention 11." The affidavit describes the concepts of vibration-induced fatigue cracking and bubble-collapse water-hammer, and explains

why these phenomena are not a concern for the Westinghouse Model F steam generators used at Plant Vogtle. I have personal knowledge of the matters stated herein and believe the following to be true and correct.

I. INTRODUCTION

3. The Georgia Power Company's Vogtle Electric Generating Plant (VEGP) utilizes two Westinghouse-designed nuclear steam supply systems (NSSS's) consisting of four recirculating reactor coolant loops. Each loop contains a Westinghouse Model F steam generator. See Figures 1 and 2.

4. The Model F steam generator is of the feedring type. The Model F is a vertical, inverted U-tube heat exchanger, which uses high temperature pressurized water on the primary side as a heat source, and produces essentially dry, saturated steam on the secondary side.

5. The primary water, which is heated in the reactor vessel, enters the bottom of the steam generator at the channel head inlet nozzle. The primary water flows upward through the inside of the inverted U-tubes where the heat transfer takes place and exits from the opposite side of the channel head at the bottom of the steam generator. The primary and secondary sides are separated by a thick forged plate called the tubesheet. The inverted U-tubes are held firmly in the tubesheet, and are supported laterally along their length by tube support plates. In the U-bend area, additional support is provided by antivibration bars.

6. On the secondary side, the main supply of feedwater enters the steam generator through a feedwater nozzle at an elevation above the top of the U-tubes. The water entering through the main feedwater nozzle is distributed circumferentially around the steam generator by means of a feedring. The main feedwater enters the feedring via a welded sleeve that connects the feedwater nozzle and feedring, and leaves the feedring through inverted-J tubes located at flow holes along the top of the feedring. The feedwater then flows down the annulus (downcomer) between the tube bundle wrapper and the outer shell, entering the bottom of the tube bundle at the tubesheet elevation. As the flow rises through the tube bundle, heat is transferred from the primary to the secondary fluid and boiling occurs. The steam-water mixture that leaves the top of the tube bundle is passed through two moisture separation stages, to produce essentially dry steam at the steam nozzle. The water that is separated from the steam is mixed with the entering feedwater and recirculated through the steam generator.

7. The VEGP Model F steam generator incorporates an auxiliary feedwater (bypass) nozzle in addition to the main feedwater nozzle. Flow through the auxiliary feedwater nozzle is discharged into the upper plenum of the steam generator through an upward sloping discharge pipe. Feedwater discharged into the upper plenum region mixes with the bulk steam generator water which is recirculated through the secondary side of the steam generator. The auxiliary feedwater nozzle is used to

introduce feedwater into the steam generator during low flow operation such as startup, hot standby and power escalation. The primary purpose of the auxiliary feedwater nozzle in the VEGP steam generators is to minimize thermal gradients in the main feedwater piping and thereby minimize the potential for thermal stress induced pipe cracking that might otherwise occur at low main feedwater flow rates of low temperature water. For the low flow, low temperature conditions at VEGP, feedwater will be supplied through the auxiliary nozzle.

II. VIBRATION-INDUCED FATIGUE CRACKING

8. Fatigue refers to the degradation of material due to cyclic or repeated loading. If the stresses due to the loads exceed the endurance limit (i.e., the stress below which an infinite number of cycles can be accommodated) and the number of repetitions of the loading is sufficiently large, material degradation such as cracking could occur. With respect to the specific phenomenon raised in contention 11, vibration-induced fatigue cracking, the cause of the repeated tube loading is vibration.

9. Vibration-induced fatigue cracking has not been observed in any Westinghouse-designed steam generator, as confirmed by periodic inspection of the steam generator tubes at operating plants. As of May 31, 1985, this history includes experience from 110 plants having between one and 24 years of operation and covers various model steam generators, including

nineteen model F steam generators and sixteen other feedring type units with flow and support configurations similar to those of the Vogtle steam generators.

10. This operating experience is reflected by the NRC in NUREG-0886, "Steam Generator Tube Experience" (February 1982) and NUREG-0606, "Unresolved Safety Issues Summary" (August 17, 1984). In both of these reports, the NRC associates fatigue cracking only with non-Westinghouse steam generators using a once-through steam generator design (Figure 3). See NUREG-0886, Tables 1 and 3; NUREG-0606 at 10 (Problem Description). The Model F, feedring-type steam generator has a structure and flow substantially different from the once-through steam generator.

11. Confidence that vibration-induced fatigue cracking will not occur in a Westinghouse-designed steam generator, however, is not based on historical experience alone. The possibility of tube degradation due to mechanical or flow-induced vibration has been thoroughly evaluated. For the Model F steam generator, this evaluation included detailed analysis of the tube support systems as well as a comprehensive research program with tube vibration model tests and a lead plant test program.

12. The primary source of tube vibrations is attributed to hydrodynamic excitation by the secondary fluid on the exterior of the tubes (the effects of primary fluid flow and mechanically-induced vibration being negligible in comparison).

The evaluation of vibration induced by secondary fluid flow considered the effects of both parallel flow along the straight sections of the tube and cross flow experienced at the entrance of downcomer feed to the tube bundle and in the curved tubed section of the U-bend.

13. For both types of flow, thermal-hydraulic analysis was used to calculate flow velocities for various modes of plant operation. For the case of parallel flow, the maximum vibratory deflections were then calculated. This analysis confirmed that the parallel flow velocities result in negligible vibratory amplitudes.

14. In the evaluation of cross-flow excitation, three vibration mechanisms were identified and studied: vortex shedding, fluidelastic excitation, and turbulence. To evaluate cross-flow at the exit of the downcomer feed to the tube bundle and at the top of the tube bundle in the U-bend area, Westinghouse performed an experimental research program of cross-flow in tube arrays with the specified parameters of the steam generator. Air and water model tests were employed.

15. The results of the research indicate that under design and flow conditions typical of Westinghouse steam generators, vortex shedding does not provide detectable tube bundle vibration. Flow turbulence in the downcomer and tube bundle inlet region inhibit the formation of Von Karman's vortex train; the spatial variations in cross-flow velocities along the tube preclude vortex shedding at a single frequency; and parallel flow velocities disrupt the Von Karman vortices.

16. Cross-flow induced vibrations due to flow turbulence were observed. The stresses caused by these vibrations, however, were two orders of magnitude below the endurance limit of the tube material. Fluidelastic excitation was also observed, but the amplitudes of the vibrations were two orders of magnitude smaller than those produced by turbulence. In sum, flow-induced vibration was too small to cause fatigue.

17. To confirm the Model F design, Westinghouse conducted the Westinghouse Partial Full Scale Test Model Program. The Partial Full Scale Test Model Program was a full scale test of a 15 degree sector of the Model F steam generator tube bundle region. The test nominal flow rate duplicated that of the Model F, and the overflow condition was 140 percent of the nominal flow rate. Two tube arrays were tested in the model to represent the bundle inlet flow at different peripheral locations around the tube bundle and were found to have tube response characteristics consistent with the design basis for the Model F. Various Model F support configurations were tested, using the adjustable support positioning capability of the model. In no case did the tubes exhibit any unstable vibrational characteristics. The test model results were consistent with design calculations, and the model results demonstrated that the design method was conservative.

18. To provide further data concerning the capability of the Model F steam generators to withstand vibration-induced degradation, Westinghouse conducted a Lead Model F Vibration

Instrumentation Program. The first Model F steam generators to begin operation in a plant (in September, 1983) were instrumented to monitor tube vibration amplitudes under various modes of operation. Peripheral tubes of the tube bundle were instrumented at the bundle inlet in the tubing straight leg and also in the U-bend region using both externally-mounted strain gages and internally-mounted accelerometers. The pressure field in the annulus between the tube bundle wrapper and the lower shell was also monitored using pressure probes. The data from the Lead Model F Instrumentation Program indicated no significant tube motions and confirmed that the vibration amplitudes measured inservice are consistent with the design assessment.

19. Thus, on the basis of historical, design, and operational assessments, vibration-induced fatigue cracking of VEGP steam generator tubes is considered to be an extremely unlikely event. However, in the event that vibration-induced fatigue cracking should occur, the extent of cracking during plant operation would be restricted by the plant's technical specification limiting steam generator tube leakage. Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. The technical specifications' maximum permissible leak rate of 0.35 gpm per steam generator for VEGP assures that a tube wall crack will be detected and repaired before the crack reaches the critical crack length (the crack length at which

tube failure could occur under postulated design basis accident conditions). Operating plants have demonstrated that primary-to-secondary leakage of the magnitude of the technical specification limit can be readily detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and inspection, during which the leaking tubes could be located and repaired.

20. In addition, as indicated in the FSAR at § 1.9.83, Vogtle has committed to conduct an inservice inspection program that conforms to Regulatory Guide 1.83 and plant technical specifications. Such a program will monitor the integrity of the steam generators and would detect tube degradation should it occur.

21. For all of these reasons, there is reasonable assurance that vibration-induced fatigue cracking will not occur in the VEGP steam generators. Furthermore, even if vibration-induced fatigue cracking were to occur, such cracking would be detected and repaired before it reached an extent at which tube failure could occur.

III. FRETTING

22. In response to one of Applicants' interrogatories asking for the basis for Joint Intervenors' contention that Westinghouse PWR steam generators have shown signs of vibration-induced fatigue cracking, Joint Intervenors referred to "fretting" and cited September 12, 1984 testimony of David

Schlissel before the Maine PUC. Letter from T. Johnson to J. Joiner (Feb. 7, 1985) (third unnumbered page of supplemental information from Howard Deustch). Fretting, however, is not a fatigue-related phenomenon. Rather, fretting is a type of wear -- a loss of tube material caused when parts come in contact with each other and have relative motion. Fretting in steam generators can be the result of vibration, but can also result from loose parts in the secondary system. In sum, fretting is a phenomenon wholly separate and distinct from vibration-induced fatigue cracking.

23. Furthermore, while the testimony of David Schlissel did refer to fretting that had been attributed to vibration, such fretting involved only preheater-type steam generators. In the preheater type of steam generator, main feedwater enters the bottom of the tube bundle directly in the preheater section. See Figure 4. This feature of the preheat-type steam generator, prior to corrective modification, resulted in the outer rows of tubes in the preheater being directly exposed to a non-uniform, highly turbulent flow entering through the main feedwater nozzle. See Figure 5.

24. By comparison, in a feedring type steam generator, feedwater enters through the elevated feedring, flows down through the annulus between the tube bundle wrapper and the outer shell, and enters the bundle at the tubesheet. Because of the differences in the internal flow configuration, distribution, and velocities between the preheat and feedring type

steam generators, the VEGP feedring type units are not susceptible to the feedwater flow-induced fretting mechanism experienced in preheat units.

IV. BUBBLE-COLLAPSE WATER-HAMMER

25. The only phenomenon involving "bubble-collapse" associated with steam generators is that which is commonly referred to as bubble-collapse water-hammer. Bubble-collapse water-hammer refers to a potential condition where initially a volume of steam is trapped within an enclosed region, for example, a horizontal section of pipe with water slugs on both sides. See Figure 6. If the temperature of the water in the slugs is the same as that of the steam, the water and steam will be in equilibrium. However, if the slugs contain cold water which comes into contact with the steam, the steam will condense rapidly resulting in a sudden local decrease in pressure. A higher pressure behind the water slugs will cause them to accelerate towards each other. When they collide, an increase in pressure will result. This change in pressure will propagate as a wave back and forth until it dissipates due to friction. The magnitude of the pressure change depends on the volume of entrapped steam, the rate at which the steam is condensed, and the pressure behind the water slugs.

26. Bubble-collapse water-hammer is a secondary side phenomenon that does not take place in the primary system. Early operating experience with feedring type steam generators

indicated a potential for draining of the feedring and the horizontal piping connected to the feedwater nozzles, permitting the formation of steam in these areas which could potentially permit a bubble-collapse water-hammer event to occur. This phenomenon, however, has never resulted in damage to steam generator tubes. Moreover, as discussed below, the likelihood of a bubble-collapse water-hammer has been significantly reduced by changes in steam generator design and operation, with the result that this type of water-hammer is no longer classified as an unresolved safety issue by the NRC.

27. Westinghouse has designed the feedwater nozzle and feedring of the Model F steam generator to inhibit the formation of steam voids in the feedring during all normal and transient plant conditions. This is accomplished principally by employing inverted J-tubes along the top of the feedring and a welded thermal sleeve. In conjunction with the steam generator design, the feedwater system is also designed to minimize the potential for steam entering the feedring and feedwater line due to backleakage from the steam generator.

28. In the model F steam generator, the inverted, top-discharging J-tubes along the top of the feedring (Figure 7) replace exit openings along the bottom of a feedring in the early feedring models. When the steam generator water level drops below the feedring, the J-tube configuration prevents rapid draining and steam filling of the feedring. Similarly, the thermal sleeve welded at the nozzle to feedring entrance

has been designed to replace the slip-fit joint employed in earlier designs. The slip-fit joint had permitted drainage of the feedring at the joint. The welded thermal sleeve precludes this possibility.

29. The separate auxiliary feedwater nozzle on the VEGP steam generators also serves to provide additional margin in minimizing the potential for bubble collapse water-hammer in the steam generators. Following plant operating or transient conditions which result in steam generator water level dropping below the level of the feedring, system design is such that feedwater for recovering the steam generator water level will enter the steam generator through the auxiliary nozzle. Thus, this potentially cold water will not be introduced into the feedring but will enter the steam generator upper plenum.

30. In addition to the design features which address the potential for bubble-collapse water-hammer in the steam generator feedring, additional design and operational features address the potential for bubble-collapse water-hammer in the adjacent main feedwater or bypass piping. (The bypass piping is the external piping that runs to the auxiliary feedwater nozzle). The main and auxiliary feedwater connections on each of the steam generators are the highest point of each feedwater line downstream of the respective isolation valves. The feedwater lines contain no high-point pockets that could trap steam and lead to water-hammer. An elbow, with a short transition piece, is connected directly to the steam generator main and

auxiliary feedwater nozzles, which tends to minimize the portion of feedwater piping that could drain into the steam generator and become filled with steam. The horizontal pipe length from the main and auxiliary feedwater nozzles of each steam generator is minimized, reducing the potential steam volume and thus the magnitude of slug formation and impact.

31. One particular postulated phenomenon considered in the design of the VEGP feedwater systems is that of steam backleakage from the steam generator into the feedwater piping. In the main feedwater piping, the potential for steam backleakage is minimized by closing the Main Feedwater Isolation Valve (MFIV) to isolate the main feedwater nozzle when the main feedwater nozzle is not in use. Additionally, the main feedwater system piping is provided with temperature sensors close to the nozzle which will alert the operator if backleakage should occur so that the operator can take corrective action.

32. With respect to the auxiliary feedwater system, a combination of operational procedures and design features prevent backleakage. The auxiliary nozzle in the VEGP steam generators connects inside the steam generator to an upwardly inclined pipe extension, the discharge end of which is below the normal operating water level in the steam generator. The feedwater control system is designed to maintain the steam generator water level above the top of the auxiliary feedwater discharge pipe inside the steam generator. If the water is kept at the normal operating level, steam cannot enter the internal

extension and thus, cannot enter the bypass piping. In addition, four check valves are provided in series between the auxiliary nozzle and the auxiliary feedwater system pump recirculation lines to minimize the potential for backleakage. For steam to push back into the bypass piping, it would be necessary for the check valves, which are provided to restrict reverse flow, to be leaking and for the steam generator water level to be below the auxiliary nozzle internal extension.

33. Moreover, steam backleakage during normal power operation is very unlikely since system design is such that normally continuous flow is provided through the steam generator auxiliary nozzle which effectively prevents the backflow of steam from the steam generator. Although during heatup, cooldown and hot standby operations, relatively small amounts of feedwater are supplied to the steam generator by the Auxiliary Feedwater System, this system is still designed to provide continuous feed rather than intermittent feed as much as possible.

34. An additional design feature of the feedwater bypass system to minimize the potential for a water hammer of this type is the installation of two temperature sensors on the bypass piping inside containment close to the auxiliary feedwater nozzle of each steam generator. If the measured temperature values exceed a predetermined setpoint, an alarm is activated in the control room. In the event that the presence of steam is suspected in the bypass line, based on temperature data and water level status and history, the system can be

recovered by slowly purging the bypass line using the Auxiliary Feedwater System at a rate of approximately 15 gpm.

35. Based on the design features of the auxiliary nozzle and its internal extension, the normal operating conditions, and the means provided for alarming and recovery from backleakage of steam if it should occur, the probability of bubble-collapse water-hammer in the feedwater bypass line is minimized. This conclusion is consistent with that reached in NUREG/CR-3090, "Evaluation of Water Hammer Potential in Preheat Steam Generators," (Dec. 1982) which evaluated the potential for water-hammer occurrence during Auxiliary Feedwater operation. (Although steam generators evaluated were preheat types, their auxiliary feedwater systems were substantially the same as that in the Model F.) This report concluded that the likelihood of water-hammer occurrence is extremely low. NUREG/CR-3090 also concluded that even if a water-hammer event were to occur, the event should have no adverse effects on Auxiliary Feedwater system operation or plant safety.

36. The conclusions set forth above are consistent with those reported by the NRC Staff in NUREG-0927, "Evaluation of Waterhammer Occurrence in Nuclear Power Plants-Technical Findings Relevant to Unresolved Safety Issue A-1" (Rev. 1 March 1984). In that report, the Staff concluded that the overall incidence of water-hammer in nuclear power plants has declined considerably in recent years. Although the Staff found that total elimination of water-hammer is not feasible, they

concluded that the frequency and severity of water-hammers is significantly reduced through proper design. Moreover, the NRC Staff reported that none of the water-hammer events which have occurred placed the plant in a faulted or emergency condition, resulted in damage to the integrity of the Reactor Coolant Pressure Boundary (including steam generator tubes), or resulted in a radioactive release. Id., §§ 1.2(b), 2.2.1. On the basis of these and other key findings, the Commission resolved USI A-1 without imposing any new regulatory requirements. NUREG-0993, "Regulatory Analysis for USI A-1 Waterhammer" (Rev. 1 1984).

37. Specifically for feedring-type steam generators, a key finding of the NRC's evaluation of water-hammer is stated in paragraph 1.3(e) of NUREG-0927.

> Following the implementation of design features and testing contained in BTP ASB 10-2 [Branch Technical Position (Auxiliary Systems Branch) 10-2: Design Guidelines for Avoiding Water-hammers in Steam Generators, NUREG-0800 (SRP) at 10.4.7-8)], the frequency of steam generator water-hammer in top feedring design steam generators has been essentially eliminated.

38. The guidelines in BTP ASB 10-2 were reiterated in NUREG-0927 as measures to prevent or mitigate waterhammers. For top feeding steam generators, these measures are:

- prevent draining of the feedring by means such as J-tubes
- minimize horizontal pipe run adjacent to steam generator nozzle (preferably less than seven feet long)
- perform preoperational testing for recovery of steam generator water level following loss of normal feedwater

- provide for automatic initiation of auxiliary feed-water

NUREG-0927, §§ 2.5.2.1(g)(1), 3.13(a). Each of these measures has been adopted in the design and operation of the VEGP steam generators.

39. NUREG-0927 also enumerated design and operating measures related to the separate auxiliary feedwater nozzle (which was evaluated as a design feature of preheat steam generators). These additional measures are as follows:

- minimize horizontal pipe run adjacent to the nozzle
- provide a check valve upstream of the auxiliary feed-water connection to the top feedwater [bypass] line
- maintain the line to the auxiliary nozzle full at all times

NUREG-0927, §§ 2.5.2.1(g)(2), 3.13(b). Each of these measures has been included in the Vogtle design.

40. For all these reasons, the occurrence of a bubble-collapse water-hammer in the VEGP steam generators is unlikely. Furthermore, even if a bubble-collapse water-hammer were to occur in the VEGP steam generators, it would not adversely affect the steam generator tubes.

Carl W Hirst

Carl W. Hirst

Subscribed and sworn before me
on this 28th day of June, 1985.

My commission expires

Lorraine M. Piplica

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

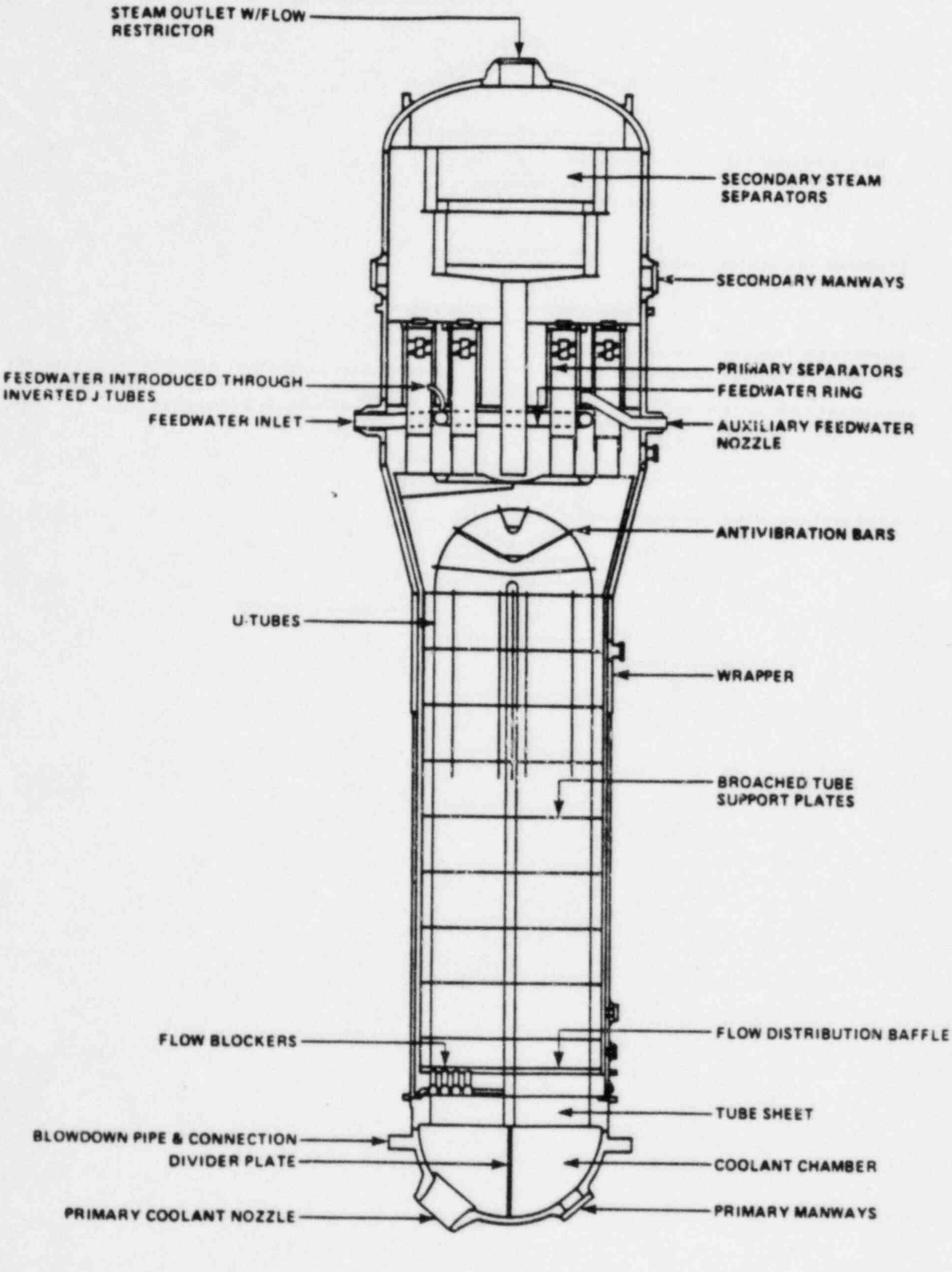
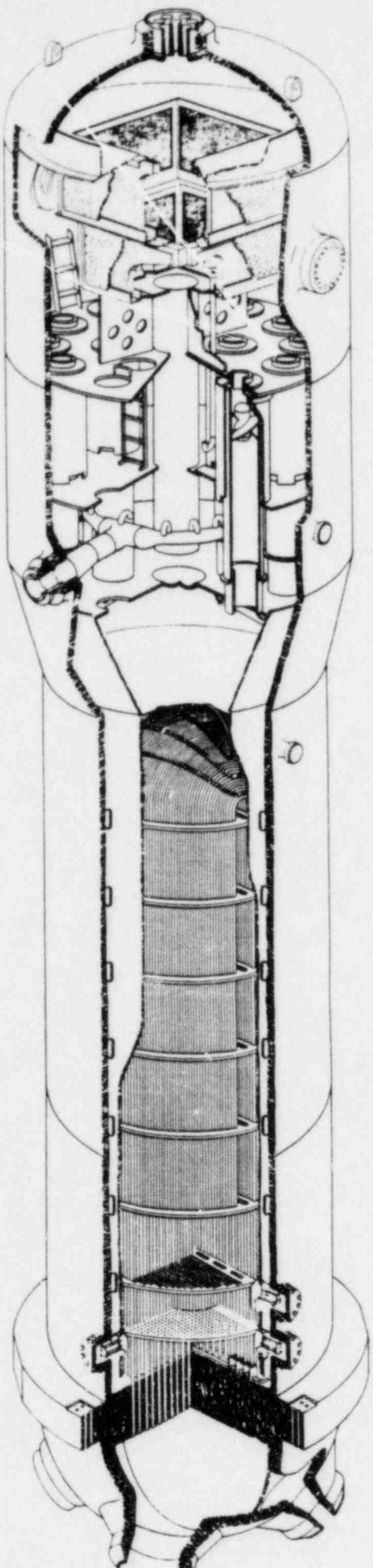


Figure 1 MODEL F STEAM GENERATOR

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2



MODEL F
STEAM GENERATOR

Figure 2

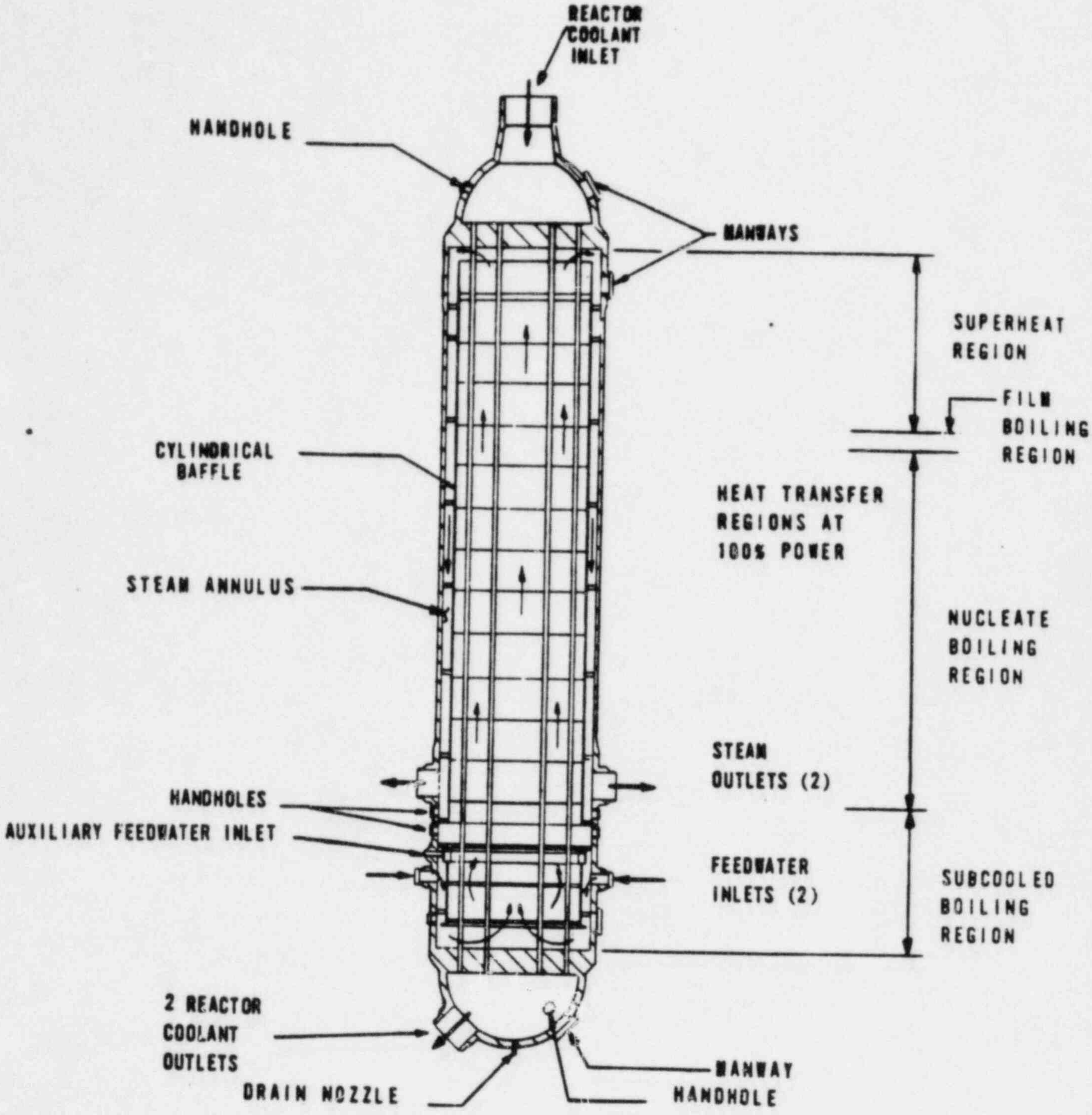


Figure 3 ONCE-THROUGH STEAM GENERATOR

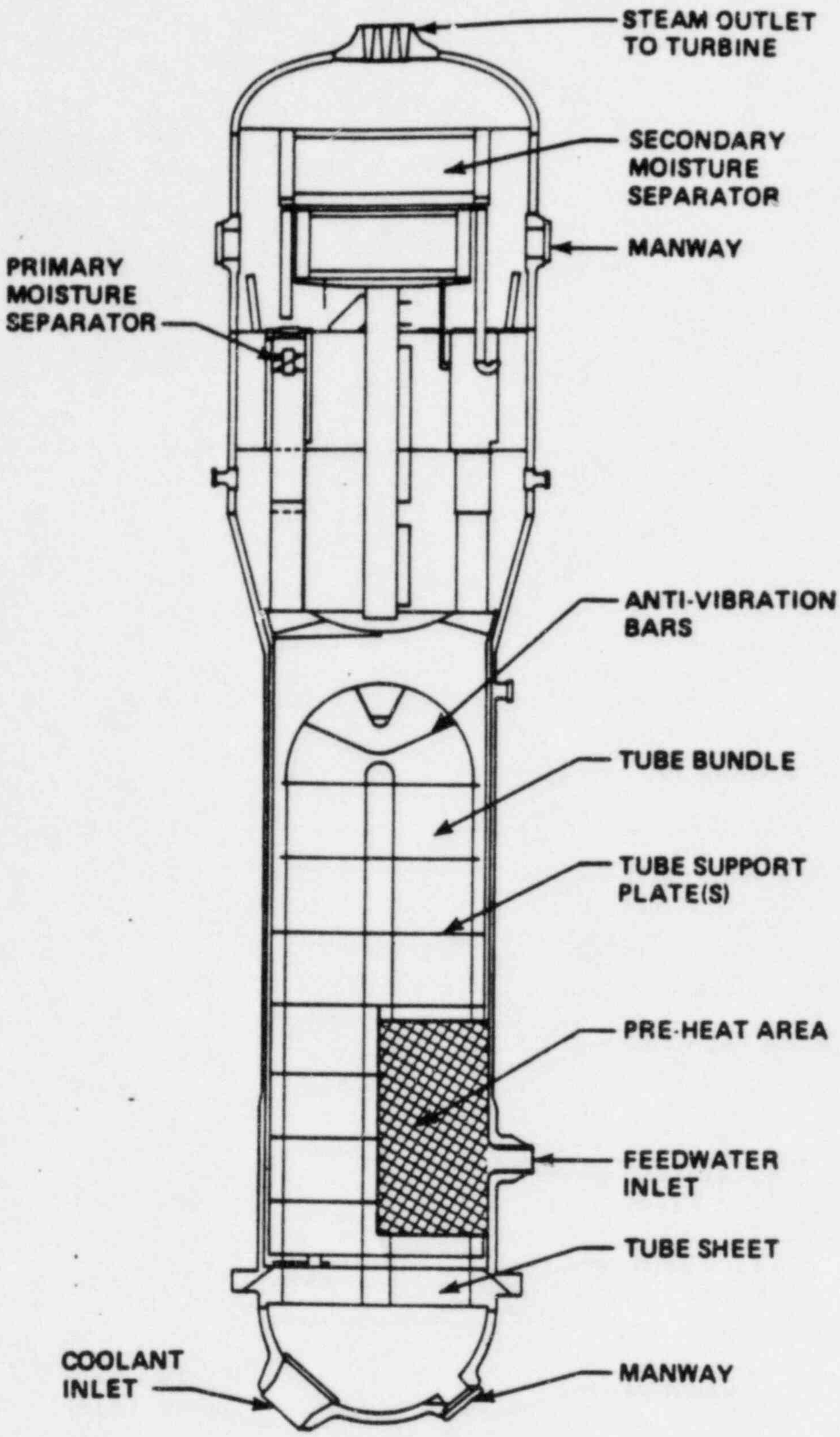


Figure 4 Preheat Steam Generator

D3 PREHEATER INLET MODIFICATION

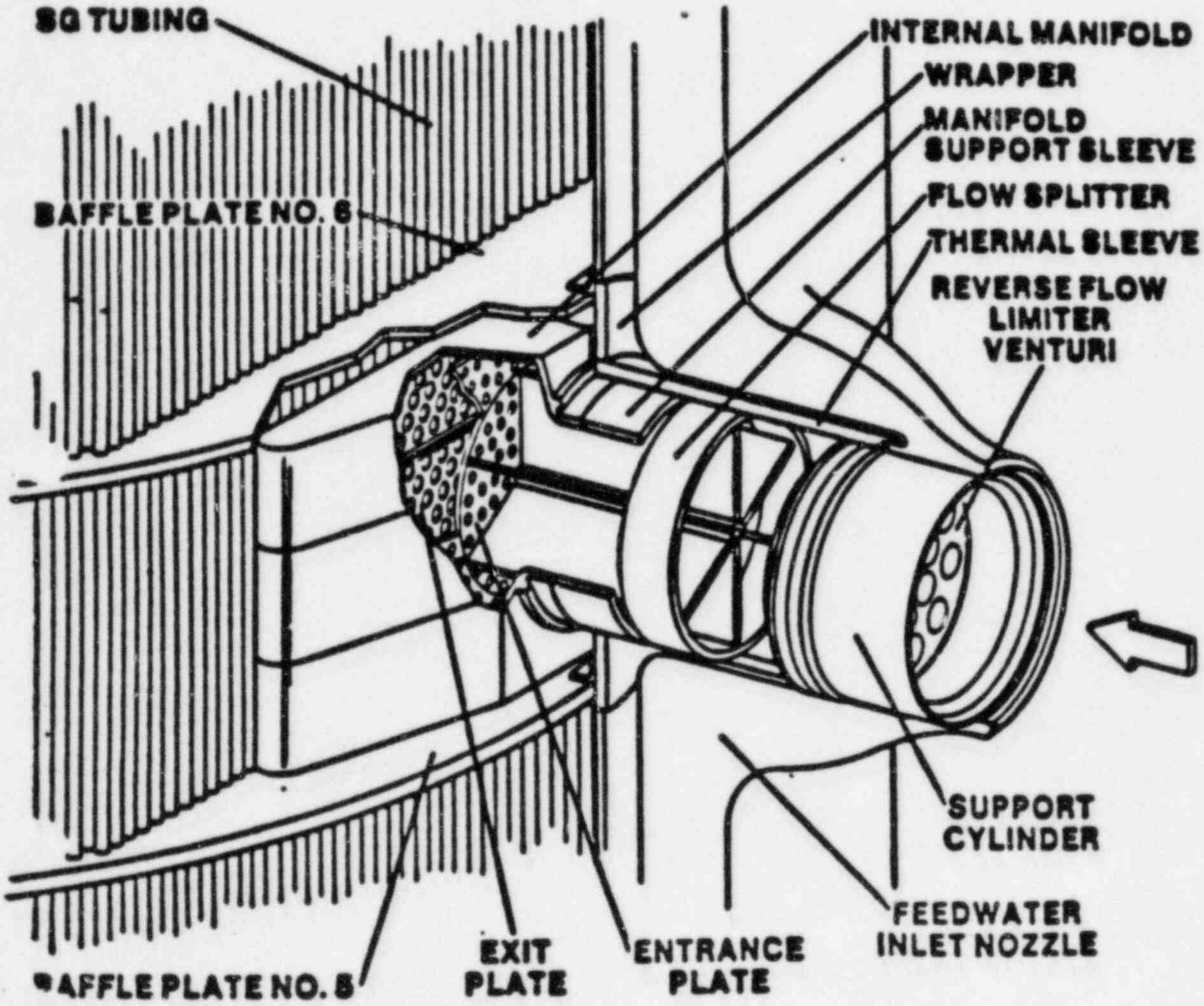
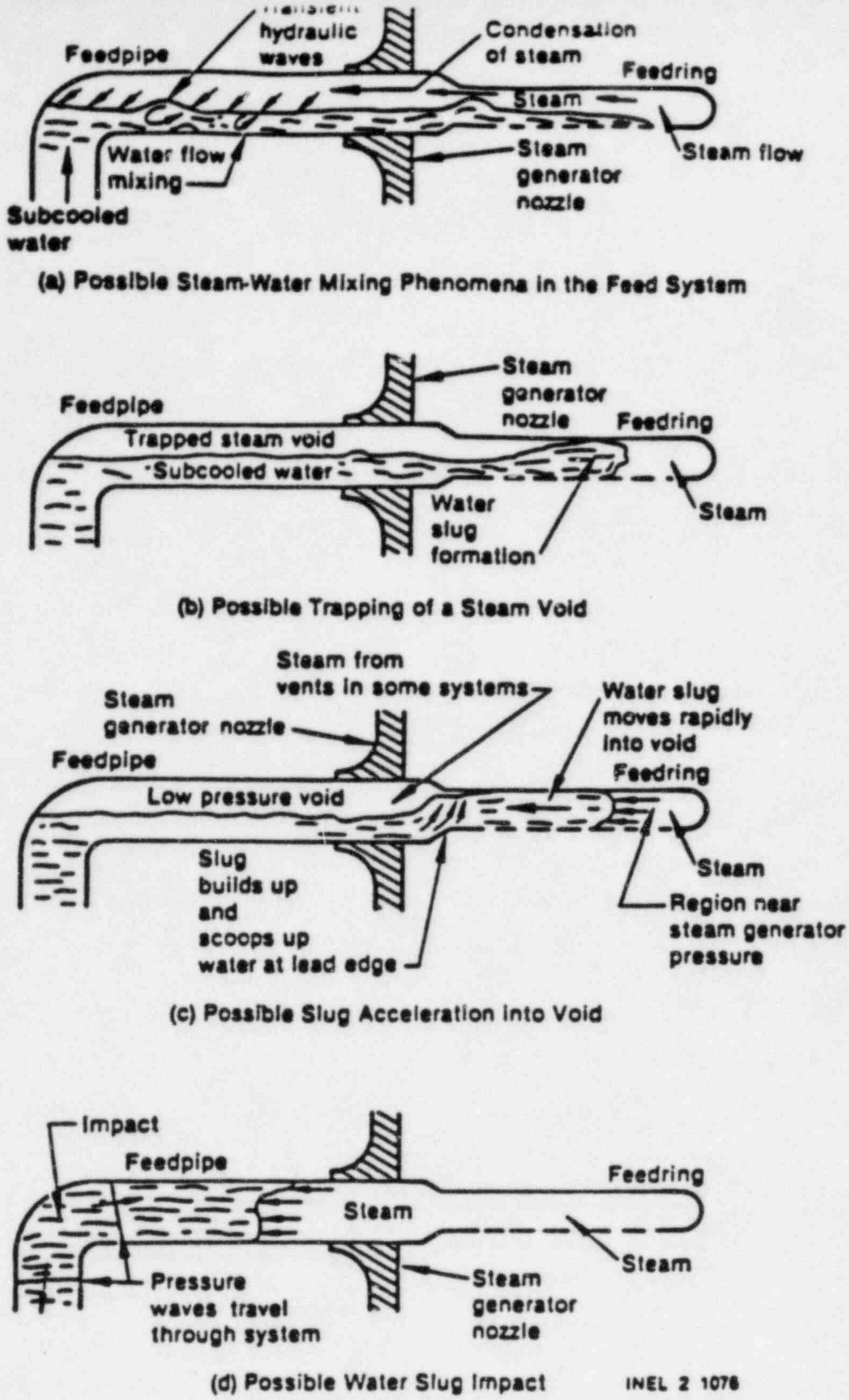


Figure 5



Possible sequential events leading to steam generator water hammer

Figure 6

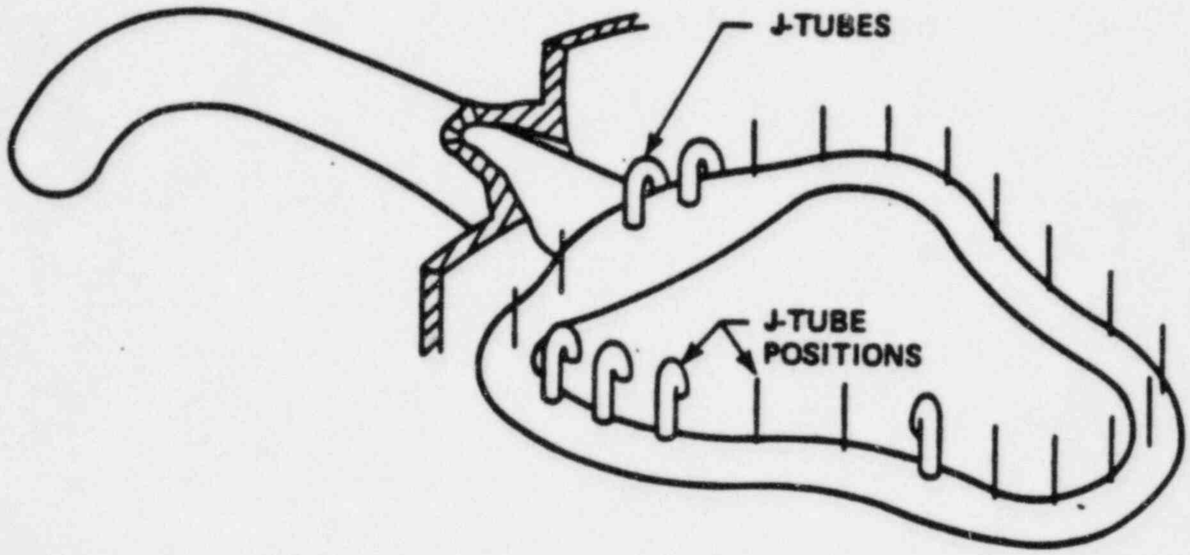


Figure 7 **J-Tube Configuration**

EXHIBIT A

SUMMARY RESUME

CARL W. HIRST

Education

B.S. Degree Aerospace Engineering, 1964, Pennsylvania State University

Work Experience

1984 - Present

Manager of Reactor Coolant System Components Licensing in the Product Licensing Group of the Nuclear Safety Department. This group is responsible for the licensing aspects of the individual components in the reactor coolant system. These include the steam generator, reactor vessel, pressurizer, reactor coolant pump, reactor vessel internals and the reactor coolant pipe.

1983 - 1984

Principle Licensing Engineer in the Nuclear Safety Department of the Westinghouse Nuclear Technology Division. Task team leader for the Regulatory Control Program to supply consulting services to operating plant utilities; for identifying present, past and future regulatory requirements, providing regulatory base definitions and assessments and benefits of regulatory requirements.

1978 - 1982

Principle Licensing Engineer in the Nuclear Safety Department of the Westinghouse Nuclear Technology Division. Responsible for all licensing issues related to steam generators. These activities cover a wide range of topics which include direct interface with the Nuclear Regulatory Commission on behalf of Westinghouse directly and Westinghouse in support of various utility customers. The scope of these activities include:

Coordinate meetings and the submittal of reports to customers and the NRC on steam generator sleeving programs.

Safety Analysis Report preparation and response to NRC questions for new plants.

Coordinate meetings and the submittal of a topical report to the NRC on the Westinghouse steam generator retubing operation.

Coordinate meetings with the NRC to support utilities with operating plants in their presentations on the results of steam generator inspections and tube examinations.

Coordinate meetings with the NRC on behalf of the Feedwater Cracking Owners Group and the investigation of feedwater line cracking.

1970 - 1978

Senior Engineer in the Hydraulic Equipment group of Westinghouse PWR Systems Division. Responsible for the preparation of equipment specifications, quotation evaluation and implementation of requirements associated with the design and manufacture of valves for nuclear power plant service.

1964 - 1970

Engineer with Mesta Machine Company. Responsible for the design, fabrication and start-up of hydraulic systems for steel mill applications.

Professional

Registered Professional Engineer (019008-E) with the State of Pennsylvania since 1972.

July 5, 1985

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NUCLEAR REGULATORY COMMISSION

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	)	50-425 (OL)
(Vogtle Electric Generating Plant,	)	
Units 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of (1) "Applicants' Motion for Summary Disposition of Joint Intervenors' Contention 11 (Steam Generators)", dated July 5, 1985, and (2) "Applicants' Statement of Material Facts as to Which There is No Genuine Issue to Be Heard Regarding Joint Intervenors' Contention 11 (Steam Generators)", dated July 5, 1985, and (3) "Affidavit of Carl W. Hirst," were served upon those persons on the attached Service List by deposit in the United States mail, postage prepaid, except where indicated by an asterisk (*) by hand delivery, this 5th day of July, 1985.

David R. Lewis

David R. Lewis

Dated: July 5, 1985

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