UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

SOUTH CAROLINA ELECTRIC & GAS
COMPANY Docket No. 50-395

(Virgil C. Summer Nuclear
Station, Unit 1)

COUNTY OF ALLEGHENY
STATE OF PENNSYLVANIA ss

AFFIDAVIT OF W. D. FLETCHER

My name is W. D. Fletcher. I am Manager, Steam Generator Development and Performance Engineering in the Nuclear Technology Division of the Westinghouse Electric Corporation. My business address is P.O. Box 855, Pittsburgh, Pennsylvania 15230. A statement of my educational, professional qualifications and experience is attached and forms a part of my affidavit.

I have reviewed the "Petition to Intervene and Request for Hearing" filed by Fairfield United Action ("FUA") dated April 9, 1982 in this proceeding. The purpose of this affidavit is to address the safety issues raised in the contention identified as "B1" by FUA. I have reviewed the results of the data gathered from operating plants,

utilizing Model D steam generators. In my opinion, the interim operation program developed by Westinghouse and South Carolina Electric & Gas Company ("SCE&G") for the V. C. Summer plant will preclude any significant steam generator tube wear until modification to the steam generators can be implemented. Contention B1 does not present, in my opinion, significant safety issues.

I am familiar with data from the operating plants utilizing Model D3 steam generators. Wear has been observed at plants utilizing Model D3 steam generators which have operated for extended periods of time at high main feed flow. Significant tube motion has been determined from instrumented steam generators and model tests to begin at about 60-65% main feed flow. It is believed that tube wear is related to significant tube motion. Significant wear is not expected below about 60-65% power level. As explained in greater detail below, I have concluded that the interim operating program for the Summer plant within the specified parameters will minimize the potential for significant tube wear.

An interim operating program for the Summer Plant has been developed by Westinghouse and South Carolina Electric and Gas Company with the objective of precluding any significant steam generator tube wear until the permanent modification to the steam generators can be implemented. This interim program includes:

- Instrumentation installed inside steam generator
- Low power testing
- Power escalation testing to main feed flow up to 50% power
- Power operation with main feed flow at 50% power for specified time intervals (including operation above 50% power based on information available at that time)
- Tubing eddy current inspections

For present purposes, power level and main feed flow can be taken to be equal. The 50% main feed flow power conditions identified for the interim period have been based on instrumentation data obtained from three operating plants (having Model D steam generators of similar design as installed at Summer) and hydraulic, vibration test model results, analysis of operating plant eddy current inspection data and tube inspections made on operating plants before and after 50 percent power operation.

During the interim operating period, it is understood that data from installed instrumentation will be examined and inspection of the steam generator tubing will be performed at appropriate intervals as a means of verifying that significant tubing degradation has not occurred.

In D3 operating plants where tube wear was observed, instrumentation was installed to measure tube motion, which has provided data relating motion to power level. This data supports the power level which can be used during the interim operating period. Characterization of tube motion in the Model D-3 steam generators at operating conditions showed that significant motion begins to occur at about 60% to 65% feed flow through the main nozzle. It is concluded from these data that operation at 50% feed flow through the main nozzle is an acceptable power level for the interim period to minimize the potential for significant tube wear.

A pertinent comparison of operating data which follows shows the operating history of three plants with similar steam generators, in terms of hours and power levels. The data are based upon the time of operation prior to tube inspection in which wear indications were measured.

COMPARATIVE POWER HISTORIES

	Number of Hours of Operation		
% Power	Plant 1 (Non-domestic) as of Oct. 1981	Plant 2 (Non-domestic) as of Nov. 1981	Plant 3 (Domestic) as of Feb. 1982
>90 to 100	1640	1456	72
>75 to 90	540	95	252
>50 to 75	1580	537	1176[1]
>20 to 50	1390	738	(negligible)
	Wear Observed	Wear Observed	No significant eddy current indications

[1]Most of this time period was at 50% power

At Plant 1 and Plant 2, where wear indications were observed in the tubes, significant periods of operation were at high power levels, i.e., greater than 90%. Plant 3 has operated for about 55% of the time since initial startup in December, 1981. At Plant 3, where the power level has been primarily at 50%, no significant wear was indicated. The judgment derived from analysis of this data is that no significant wear is to be expected from interim time periods of operation at the lower power levels.

Additionally, the results of other plant operating experience at 50% power, equivalent to the program for the Summer D-3 steam generators is relevant. At Plant 2 (D-3 steam generators) a program of 50% power operation for 1500 hours (with short intervals of higher power operation to obtain tube instrumentation data) was completed in March of 1982. Comparing the tubing inspection data in the preheater region prior to (i.e., in Nov. 1981) and after (i.e., in March 1982) this operating interval showed no significant change, including tubes which previously exhibited indications of wear.

Experimental testing has provided a data base which also supports the results observed from the tube instrumentation and plant inspection data. The tube instrumentation data indicate that the onset of tube motion is a result of turbulent feed flow at the preheater inlet at the higher equivalent power levels. The testing results have been obtained from several hydraulic models which demonstrate levels of turbulence and associated flow fluctuation at various feed flow rates. Data from a scale model (∿4/10 scale) of the D-3 preheater demonstrates tube motion which increases with the feed flow rate in a manner similar to the data from the operating plant instrumentation. These test results support the conclusions derived from the field data.

Westinghouse currently has an engineering design program to modify Model D steam generators to alleviate tube wear. The anticipated design modification can be implemented in the field. An extensive testing program is in progress to verify that the anticipated modification to the steam generator preheater section addresses and corrects tube wear. The testing program incorporates a variety of testing facilities such as 0.417, 2/3 and full scale model testing. These test results, when combined with actual plant performance data, are expected to provide verification that the design modification will perform as predicted by engineering design analysis.

The other issues raised in FUA Contention B1 are not concerns during the V. C. Summer interim program as described above. Operation during this period is designed to minimize tube wear so that tube rupture, multiple tube rupture, actions of the PORV and LOCA events are not relevant.

Notwithstanding this, a brief status of these issues is as follows:

Row 1 U-bends

The staff analysis of tangent point cracking of Row 1 tubes concludes that while there remains a potential that this issue may occur in Model D Row 1, it is not a safety concern. Westinghouse supports the technical basis for this conclusion as detailed in Section 5.4.2 of the V. C. Summer SER, Supplement 3, January, 1982.

Tube Rupture Events

Westinghouse has performed analyses for the postulated double-ended steam generator tube rupture event for all Westinghouse-designed NSSS plants. Systems installed in these plants are designed to accommodate these events. The results of these analyses and of other analyses have been used to formulate Emergency Response Guidelines (ERG) for use by utilities in writing plant-specific Emergency Operating Procedures. Westinghouse has also performed evaluations of those significant tube leakage events that have occurred and has used the results of these evaluations to further improve the guidance provided in earlier versions of the Westinghouse ERG's.

Additional evaluations of these significant tube leakage events and the manner in which those leaks developed, leads Westinghouse to believe that multiple tube rupture is very unlikely. This belief is further reinforced by the results of steam generator tube inspections that are routinely performed in operating steam generators and also as described above for the Summer interim operation program. These inspections are specifically aimed at early detection of any condition that would have the potential for tube rupture. In addition, tube conditions are continuously monitored for tube leakage during normal operation.

As noted above, Westinghouse believes that concurrent multiple tube ruptures or concurrent tube ruptures in multiple steam

generators are very unlikely. Notwithstanding this belief, Westinghouse has performed, for the purposes of developing additional ERG's, evaluations of multiple tube ruptures and of tube ruptures in multiple steam generators. These ERG's take into consideration past operating experience, post-TMI lessons learned and multiple equipment failure contingencies. These procedures have been written to minimize radioactive release from the plant. This effort was sponsored by the Westinghouse Owners Group (WOG) of which South Carolina Electric and Gas is a member.

The latest revision of these guidelines were submitted by the WOG to the NRC in November 1981 and are currently under review. In a meeting with the ACRS on 3/24/82, NRC staff expressed basic agreement with the Westinghouse guidelines.

PORV

With respect to a tube rupture, the PORV provides a means to depressurize the primary side of the steam generators. If the PORV were to stay in the open position during use, the operator could isolate the PORV with the PORV isolation (block) valves, such as used at both Ginna and Three Mile Island, to effectively isolate the PORVs. The presence of three PORVs and associated block valves in the Summer plant allows the operator sufficient flexibility to enable him to decrease and control RCS pressure. The NRC has requested certain operational data for pressurizer PORV's through NUREG-0737, Item II.K.3.2. In responding to this request, Westinghouse conducted a survey of domestic Westinghouse NSSS operating plants, and the results of such survey were provided to the NRC. Notwithstanding the experience in the past that some PORVs have in a few instances remained in the open position and did not close on demand, the results of the Westinghouse survey and operating experience subsequent to the survey, still indicates that the PORV reliability is acceptable.

Moreover, the industry has implemented certain testing. For example, the Electric Power Research Institute (EPRI) has data from such tests that are directly applicable to the PORVs at Virgil Summer, since EPRI tested the same exact model PORV as installed in the Virgil Summer nuclear facility [Copes-Vulcan PORV Model D-100-160 (316 w/Stellite plug and 17-4Ph cage)]. As reported in the "EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report," (April 7, 1982), twenty tests were conducted on this valve during which the valve fully opened and fully closed on demand.

Accident Sequences

Postulated accident sequences, such as the loss-of-coolant accident (LOCA) and the feedline break (FLB) accident, impose increased loads on the steam generator tubes. Such postulated increased loads have been evaluated by Westinghouse in an analytical program. The results of these analyses show that tubes which exhibit degradation less than that specified in the tube plugging criteria will maintain their integrity for all postulated design basis accident sequences.

These studies have further shown that for a LOCA, the maximum steam generator tube stresses, caused by rarefaction waves, blowdown and vibrational forces, occur in the U-bend region of the steam generator. The tube stresses near the tubesheet and in the preheater region are lower. In addition, the pressure force mechanism during the postulated LOCA is in the direction of the potential for tube collapse rather than tube rupture since the primary side has depressurized. A postulated tube collapse has small potential for creating a leak path between the secondary side and primary side of the steam generator.

Westinghouse studies have shown that the postulated feedline break accident produces the highest steam generator tube stresses of any of the design basis accident sequences. During the postulated feedline break accident, the primary steam generator tube stresses result from the pressure differential across the tube. The calculated stresses resulting from the postulated feedline break accident are within acceptance limits for a tube with wear even more than that limit specified in the tube plugging criteria.

Postulating a steam generator tube rupture in conjunction with another design basis accident sequence constitutes a double failure. As such, the ECCS Acceptance Criteria in 10CFR50.46 and the requirements of 10CFR50 Appendix K do not require evaluation of such an event. Accordingly the evaluation models do not apply to such postulated double failures and evaluation of such a postulated double failure has been performed using modified codes and assumptions.

Westinghouse has performed calculations of a postulated loss-of-coolant accident ("LOCA"), with and without secondary to primary steam generator tube leakage, for a typical Westinghouse plant. The calculation was performed for a double-ended cold leg break of the reactor coolant pipe using codes and assumptions modified to more realistically represent the system than is performed for design basis calculations. Contrary to FUA contention (page 4), these calculations showed that adequate core cooling was maintained. The results of these calculations were that less than a 10°F increase in peak fuel cladding temperature was calculated when a 250 gpm secondary to primary tube leak was modelled. This leak rate is consistent with a postulated double-ended break of a steam generator tube. Since the pressure force mechanism during

a LOCA is postulated to occur in the direction of the potential for tube collapse and since wear patterns on tubes removed from two operating plants exist on primarily one side of the tube, a postulated double-ended tube break resulting from a LOCA is considered highly unlikely. Moreover, the data from the two operating plants indicates that one tube tended to lead the rest by at least 10 percent of the tube wall thickness reduction, thus the participation by leakage of more than one tube is considered unlikely.

Further, affiant sayeth not.

AFFIDAVIT

Before me, the undersigned authority, personally appeared W. D. Fletcher, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this affidavit are true and correct to the best of his knowledge, information, and belief:

W. D. Fletcher, Manager
Steam Generator Development
and Performance Engineering

Sworn to and subscribed
before me this 22 day
of April 1982.

Paulette Slonska

Notary Public
PAULETTE SLONSKA, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES MARCH 10, 1986
Member, Pennsylvania Association of Notaries

STATEMENT OF QUALIFICATIONS AND EXPERIENCE

W. D. FLETCHER

EXPERIENCE

My name is W.D. Fletcher; I am presently Manager, Steam Generator Development and Performance Engineering in the Nuclear Technology Division of the Westinghouse Electric Corporation.

I graduated from Hardin-Simmons University in 1950 with a Bachelor's degree in Chemistry and from Fordham University in 1960 with a Masters degree in Chemistry.

I was employed with the Vitro Laboratories from 1951 to 1955, where I performed research on organo-phosphorus compound synthesis, reaction kinetics and mechanisms of organo-phosphorus compounds, phase studies, bench scale and pilot plant production of organo-phosphites, high and low temperature kinetic studies of boron hydride synthesis, and electro-kinetic studies of electrophoretic deposition of inorganic oxides in the manufacture of reactor fuel elements.

In 1957 I began my employment with Westinghouse and have been engaged in development work on the heterogeneous catalysis of reactions between hydrogen and oxygen produced through radiolysis of reactor coolants, reaction kinetics and mechanisms, catalyst development and evaluation in high temperature and pressure aqueous solutions; evaluation and study of reactor coolant contaminants

and means of coolant purification; study of behavior of fission and corrosion products in reactor coolants; in-pile studies of reactor coolants as pertains to chemical shim technology; reactor plant chemistry control, analyses, and data collection and interpretation of all operating reactor systems designed by Westinghouse.

Since 1970, I have been directly involved in development and design activities related to Westinghouse steam generators. Under my direction, steam generator programs related to operations have been executed involving chemistry and materials as well as specific design configurations.

As Manager, Steam Generator Development and Performance Engineering, I am responsible for three design-development groups that involve steam generator thermal/hydraulics, advanced concepts design and analysis and design of field modification to steam generators.

I am a member of the American Chemical Society, the National Association of Corrosion Engineers, the American Nuclear Society, and the American Society of Mecnanical Engineers.

PUBLICATIONS

"Update of Operations with Westinghouse Steam Generators" American Nuclear Society 1977, D.C. Malinowski and W.D. Fletcher.

"Operating Experience with Westinghouse Steam Generators", Nuclear Technology, 1975 W.D. Fletcher and D.D. Malinowski.

"Water Technology for Nuclear Power/PWR's", Industrial Water Engineering, 1971, W.D. Fletcher.

"Primary Coolant Chemistry of PWR's", W.D. Fletcher, the International Water Conference of the Engineers Society of Western Pennsylvania, Pittsburgh, October, 1970.

"Post Accident Iodine Cleanup by Containment Filters and Sprays." Presentation at Tampa, Florida, May 21, 1968, J.D. McAdoo and W.D. Fletcher.

"Effects of Coolant Chemistry on Corrosion and Corrosion Products", W.D. Fletcher, Am. Nuc. Soc., Seattle, June 1969.

EURAEC-1972 (WCAP-3690-4) - "Description and Evaluation of the Boron Concentration Meter Utilized at the SENA (Franco-Belge) Reactor Plant", January 1968, W.D. Fletcher.

WCAP-3269-57 - "The Post-Irradiation Examination of Saxton Fuel Cladding Corrosion Products", March 1966, L.F. Picone and W.D. Fletcher.

WCAP-3269-63 - "Fission Products from Fuel Defect Test at Saxton", April 1966, W.D. Fletcher and L.F. Picone.

WCAP-2964 - "Stability of Alkali in Reactor Coolant", 1964, W.D. Fletcher.

WCAP-2656 - "Analysis of Fission Products in Saxton Primary Coolant", August 1964, W.D. Fletcher.

"Water Technology of the Saxton Nuclear Experiment", Division of Water and Waste Chemistry, 4, 46 (1964), W.D. Fletcher and R.F. Swift.

"Flame Photometric Determination of Lithium Produced by B-10 (n,a) Li-7 to Measure Boron-10 Burnup in Reactors Utilizing Chemical Shim Control,: Presentation at Gatlinburg, Tenn., Oct. 6-8 1964, B.D. LaMont and W.D. Fletcher.

WCAP-3716 - "Ion Exchange in Boric Acid Solutions with Radioactive Decay", November 1962, W.D. Fletcher.

WCAP-1689 Rev. - "The Behavior of Stainless Steel Corrosion Products in High Temperature Boric Acid Solutions", May 1961, W.D. Fletcher, A. Krieg and P. Cohen.

WCAP-4097 - "Inorganic Ion-Exchanger Materials for Water Purification in CVTR", August 1961 (CVNA-135), N. Michael, W.D. Fletcher, <u>et al</u>..

WCAP-3730 - "Interactions Between Stainless Steel Corrosion Products and Boric Acid Solutions", March 1960, W.D. Fletcher.

"Some Performance Characteristics of Zirconium Phosphate and Zirconium Oxide Ion Exchange Materials", Trans. Am. Nuc. Soc., 3, 46 (1960), N. Michael and W.D. Fletcher.

WCAP-1206 - "Internal Recombination Catalyst Studies", May 4, 1959, W. D. Fletcher and D.E. Byrnes.

WCAP-1110 - "A Semi-Flow System for the Study of Catalytic Combination of Hydrogen and Oxygen in Aqueous or Slurry System", February 1959, W.D. Fletcher and W.E. Foster.

"Electrophoretic Deposition of Metallic and Composite Coatings", Plating 42, 1255 (1955).

"Post LOCA Hydrogen Generation in PWR Containments", W.D. Fletcher, M.J. Bell, R.T. Marchese, and J.L. Gallagher, American Nuclear Society.

PATENTS

U.S. Patent, "Information Storage Systems and Methods for Producing Same".

U.S. Patent, "Boron Concentration Meter".

U.S. Patent, "Electrophoretic Coating Dispersion Formulations".