November 2, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
WISCONSIN ELECTRIC POWER COMPANY	)	Docket Nos. 50-266
	)	50-301
(Point Beach Nuclear Plant,	)	(OL Amendment)
Units 1 and 2)	)	

LICENSEE'S TESTIMONY OF W. D. FLETCHER

Q. Please state your name, address, and occupation.

A. My name is W. D. Fletcher. My address is Westinghouse Electric Corporation, Forest Hills Site, Avenue A & West Street, Pittsburgh, Pennsylvania 15221. I am presently Manager, Steam Generator Development and Performance Engineering in the Nuclear Technology Division of the Westinghouse Electric Corporation. A statement of my qualifications and experience is attached to this testimony.

Q. The contention in this proceeding is:

> That the license amendment should be denied or conditioned because applicant has not demonstrated that eddy current testing is adequate to detect serious stress corrosion cracking or intergranular attack, in excess of the technical specification prohibiting

> more than 40 percent degradation of the sleeve wall, in sleeves that would be inserted within steam generator tubes.

Please describe the technical specification referred to in the contention.

A. Technical Specification 15.4.2.A, "Steam Generator Tube Inspection Requirements," of the Technical Specifications which are included in the operating licenses for Units 1 and 2 of the Point Beach Nuclear Plant, specifies the requirements for inservice inspection of the steam generator tubes. Paragraph 5 of Technical Specification 15.4.2.A defines "defect" as an imperfection that exceeds 50% of the tube wall thickness, and "plugging limit" as an imperfection of 40% of the tube wall thickness. Paragraph 6 requires the licensee to plug tubes which leak or have degradation exceeding the 40% plugging limit. Paragraph 3 requires that the tubes shall be examined for degradation in accordance with Appendix IV, "Eddy Current Examination Method of Nonferromagnetic Steam Generator Heat Exchanger Tubing," of the ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Plant Components."

Technical Specification 15.4.2.A is consistent with NRC Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes."

Q. Does eddy current testing, as performed by Westinghouse, meet the requirements specified in Paragraph 3 of Technical Specification 15.4.2.A?

A. Yes. Eddy current testing (ECT) fully complies with the requirements of ASME Boiler and Pressure Vessel Code, Section XI, as required by Paragraph 3 of Technical Specification 15.4.2.A.

Q. Is the ECT sufficiently sensitive to detect intergranular attack (IGA) and stress corrosion cracking (SCC) at the 40% plugging limit in steam generator tubes?

A. Yes. While such degradation may not in all cases be detected with absolute certainty because of variations in the sizes and configurations of the degradation, and because of test signal interferences from non-uniform materials in the test vicinity, inservice ECT methodology has progressed to the stage where we can generally expect to detect IGA and SCC that has progressed to 40% of the tube wall thickness.

Q. Is the ECT sufficiently sensitive to detect IGA and SCC at the plugging limit in the sleeves to be installed in the Point Beach steam generators?

A. Yes. The ability to detect IGA and SCC in the pressure boundary portion of the sleeve between the upper and lower joints is enhanced over the ability of ECT to detect such degradation in the corresponding portion of an unsleeved tube.

ECT utilizes an electromagnetic field, emanating from the eddy current probe within the tube or sleeve to examine the tube or sleeve wall. Degradation in the wall of the tube or sleeve causes variations in the effective electrical conductivity and/or magnetic permeability of the wall material.

These variations are measured directly by changes in the coil voltage of the eddy current probe. The outer edge of the tube is only about 0.007" (7 mils) from the inner surface of the tubesheet hole. The uneven characteristics of the surface of the tubesheet hole cause signals which can interfere with the interpretation of eddy current indications. Signals can also be caused by the magnetite in the sludge surrounding the tube in the vicinity of the tubesheet, as well as by conductive impurities which may be deposited outside of the tube. These signals, referred to as "noise," can be reduced by the use of multifrequency mixing techniques such that the adequacy of the inspection is maintained, even though some residual interference remains.

Significantly less noise is present when the sleeve is tested. The outer surface of the sleeve is nominally 75 mils away from the surface of the tubesheet hole. This means that, compared to the tube, the sleeve wall being examined is now much farther away from the surface of the tubesheet hole, and much farther away from any sludge and impurities which may be present on the outer surface of the tube. The noise from these three sources is significantly reduced by the greater distance. In addition, the tube surrounding the sleeve acts as an electromagnetic shield which further reduces the interfering signals from these exterior noise sources.

The portion of the sleeve above the top of the tube sheet will also experience improved eddy current inspectability

because, as stated above, the outer surface of the sleeve is farther away from the sludge and impurities which may be present on the outer surface of the tube, and because of the electromagnetic shielding by the tube. Even if the tube were to leak in the sleeved region, exposing the outer surface of the sleeve to conductive interferences, the adequacy of the inspections is maintained through the use of multifrequency mixing techniques to reduce the resultant noise.

Q. The Licensing Board has asked that we provide additional information about ECT, the rate of progression of IGA and SCC in steam generator tubes, and the consequences of failing to detect such degradation. Can you comment on this?

A. Certainly, although I should state first that eddy current testing is not the only, or even the primary, means of providing assurance that tube degradation does not lead to an unsafe condition. A number of other factors are present, including the corrosion resistance of the tubing material, the toughness and ductility of the tubing material which results in detectable and controllable leakage before risk of rupture occurs (the "leak-before-break" characteristic), hydrostatic testing, leak rate limitations, and the conservatism of NRC's "plugging limit."

NRC's licensing requirements assume that some leakage will occur. Such leakage is monitored constantly while the plant is in operation. The leakage is detected by monitoring the secondary system condensor air ejector and steam generator

blowdown for radioactivity. The Technical Specifications require the units to be shut down for repairs if leakage exceeds a predefined rate. Eddy current testing is used to help reduce the potential occurrence of leakage, but is not expected to eliminate it totally.

The rate of corrosion in tubes or sleeves is dependent on the environment to which they are exposed. With regard to the outer diameter (OD) of the sleeve, it will not be exposed to the secondary side environment unless degradation in the original tubing propagates through-wall and opens sufficiently to permit solution to enter the annulus. However, because of the expected increased corrosion resistance of the thermally treated Inconel 600 sleeve material, compared to that of the mill annealed Inconel 600 of the tube, the sleeve would be more resistant to attack even if the same combination of environmental factors which led to attack in the original tubing were to develop in the annulus.

For IGA, a rate of about 15% of tube wall thickness per year was conservatively estimated from eddy current testing and examinations of tubes in the field. This value is consistent with laboratory data for mill annealed Inconel 600 tubing exposed at 600°F to 10% NaOH solutions. In these same laboratory tests, thermally treated Inconel 600, which represents the sleeve condition, had a lower IGA rate, by a factor of 2 or 3. A larger reduction, of the order of 10, was observed in the caustic SCC rate for thermally treated Inconel 600. An

estimate of propagation rate, should conditions exist for caustic induced IGA and SCC for the sleeve material, is:

IGA: 5-10% of wall/year (during operation)

SCC: 5-15% of wall/year (during operation)

The test data do not indicate a change in corrosion rate over time.

The characteristics of the tube and sleeve material, Inconel 600, are such that, whether or not IGA is present, SCC progressing through the tube wall will result in leakage before the time when the tube has the potential for rupture during accident or normal operating conditions. This "leak-before-break" characteristic is based on the concept that a corrosion crack, initiating from the OD, would penetrate through-wall and result in a small but detectable leak before the crack would propagate to the "critical crack length," i.e., such a length that the tube could develop a large leak rate in the event of a postulated accident. This concept is particularly applicable to materials such as Inconel 600 with high ductility and toughness, that is, with the capability to withstand high stresses by deforming rather than fracturing.

To illustrate this concept, consider the initiation of a stress corrosion crack on the OD of a sleeve. If the crack were to occur, it would most likely be axial, that is, parallel to the axis of the sleeve, due to the dominating effect of the internal pressure hoop stress. The crack would propagate in length as well as depth in such a manner that a small

penetration of the sleeve wall would eventually occur. The aspect ratio, or the ratio of the length to depth of a propagating corrosion crack prior to through-wall penetration, has been found, from examinations of degraded tubes, to be about 2 to 5. Assuming a value of 5, such a crack would propagate to penetrate the sleeve wall while its OD length would be no more than 0.250".

Primary to secondary leakage would begin to be detected at this time. This would provide an early indication of degradation, permitting an orderly shutdown for inspection and resolution should the cracking process continue with an accompanying increase in leakage rate to Technical Specification limits. The maximum through-wall crack length which could exist without exceeding the plant's Technical Specification limits for leakage, 500 gpd per steam generator (0.3 gpm per steam generator), would be about 0.4" at normal operating pressures. If, assuming a 0.4" through-wall crack length, one would also postulate the simultaneous occurrence of a steam line break accident, the leakage would not be expected to be excessive, because the 0.4" long crack could withstand the increased pressure differential without bursting.

Laboratory and operating experience confirm the validity of the leak-beforebreak concept. Degraded tubes normally do not result in large breaks, but penetrate locally resulting in minor leakage which is readily detectable and can be remedied. Virtually all leakage events in Westinghouse steam generators were of this kind.

The NRC's 40% plugging limit takes into account margins for eddy current testing uncertainty, as well as margin for continued degradation for operating intervals between inspections. When a tube has 40% indicated degradation, it still has margin to resist rupture under both normal operating and accident condition differential pressure.

The maximum primary-to-secondary pressure differential occurs following a postulated feedline break or steam line break accident which reduces the secondary side pressure to zero. Analysis of this accident condition, contained in the Point Beach Sleeving Report (WCAP-9960, Rev. 1, pgs. 6.120-6.121), indicates that for uniform thinning completely around the circumference, the sleeve can degrade to 38% of its original wall thickness and still resist rupture under both the normal operating and accident loads. This corresponds to 62% degradation or a margin of 22% beyond the 40% degradation limit. Thus, the NRC's 40% plugging limit is conservative.

The above analysis is generic, and assumes a maximum ΔP of 2560 psi. For Point Beach, this value is conservative. The effect of this conservatism on the minimum sleeve wall thickness calculation is to increase the amount of degradation that can be tolerated and still resist rupturing.

Burst tests were performed on portions of tubes removed from Point Beach which had IGA on the order of 40 to 60% penetration of the tube wall. This testing required differential pressures in excess of 5000 psi to cause bursting of the

degraded tubes. This indicates substantial additional margin over the conservatively estimated pressures resulting from postulated accidents.

The consequences of sleeve degradation would be no worse than, and in all probability less than, the consequences of degredation in the corresponding portion of an unsleeved tube. Extensive examination of removed tube samples has shown that IGA occurs in the Point Beach steam generator tubes within the tubesheet. In addition, as a result of increased knowledge and field experience with the interpretation of ECT signals from IGA over the last two years, IGA is detectable at less than 40% through-wall penetration. It is not expected that IGA would progress undetected by either eddy current testing or by primary-to-secondary leak detection to the point where rupture could occur. However, for the sake of argument, if rupture of the sleeve were assumed to occur within the tubesheet as a result of IGA (or SCC), the leak path would be obstructed by the tube-to-tubesheet narrow annulus, and the leak rate would be significantly reduced compared to the rate from a ruptured tube postulated to occur above the tubesheet.

Recent experience with Point Beach steam generators has shown that tube degradation is not now occurring above the tubesheet to any significant degree. Again, for the sake of argument, if rupture of the sleeve were assumed to occur above the tubesheet, the consequences would be no worse than the consequences of an equivalent rupture of an unsleeved tube. As

a practical matter, although not required, it is assumed that some additional leak limiting capability would be provided by the presence of the surrounding tube, even if the tube had experienced degradation.

In summary, eddy current inspectability of sleeves, the leak-before-break characteristics of the sleeves, the additional corrosion resistance of the sleeve material, and the added margin in the ability of the corroded material to resist rupture provide reasonable assurance of the protection of public health and safety against unacceptable leakage during normal and accident conditions.

Q. How does the presence of sleeves affect the safety considerations you have just discussed?

A. In the region in and above the tubesheet, sleeving, in effect, substitutes a new primary pressure boundary which adds margin to that portion of the pressure boundary provided by the original tube. As I stated earlier, inspectability of the sleeves is better than that of the unsleeved tubes in the sleeved region, the sleeves are less likely to be subjected to the caustic environment to which the tubes are exposed, and the thermally treated Inconel 600 provides additional resistance to both IGA and SCC, compared to the tube material. The leak-before-break principle described above is applicable both to the tubes and the sleeves. Moreover, even if rupture of a sleeve were to occur, the consequences would be no more, and

likely less, than the consequences of a rupture of an unsleeved tube in the sleeved region.

The NRC's General Design Criterion 14, Appendix A, 10 C.F.R. Part 50 requires that:

> The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Sleeving provides an even lower probability of occurrence of these three events in the sleeved portion of the pressure boundary.

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 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 Mechanical 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.C. 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.

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"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, et al..

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," American Nuclear Society, W.D. Fletcher, M.J. Bell, R.T. Marchese, and J.L. Gallagher.

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."

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WISCONSIN ELECTRIC POWER COMPANY	)	Docket Nos. 50-266
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Units 1 and 2)	)	

CERTIFICATE OF SERVICE

This is to certify that copies of "Licensee's Testimony of W. D. Fletcher" were served, by deposit in the U.S. Mail, first class, postage prepaid, to all those on the attached Service List, except that those marked with an asterisk were served by hand delivery or by deposit with Federal Express, this 2nd day of November, 1982.

Delissa A. Ridgway

Delissa A. Ridgway

Dated: November 2, 1982

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