June 29, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
	)	(Steam Generator Repair)
(Three Mile Island Nuclear Station, Unit No. 1)	)	

TESTIMONY OF CONRAD E. McCRACKEN
AND PAUL C. WU ON TMIA CONTENTION 1.b

Q.1 Please state your names and positions with the NRC.

A.1 My name is Conrad E. McCracken. I am the Section Chief of the Chemical and Corrosion Technology Section, Chemical Engineering Branch, NRC Division of Engineering. A copy of my professional qualifications is attached.

My name is Paul C. Wu. I am a Chemical Engineer in the Chemical and Corrosion Technology Section, Chemical Engineering Branch, NRC Division of Engineering. A copy of my professional qualifications is attached.

Q.2 What is the purpose of this testimony?

A.2 The purpose of this testimony is to address that portion of TMIA Contention 1.b. dealing with the potential for the kinetic expansion repair process to increase the probability of simultaneous tube ruptures involving both TMI-1 steam generators.

Q.3 Why is the alloy Inconel-600 chosen as the steam generator tube material?

A.3 Inconel-600 is a nickel-base alloy. Like the austenitic stainless steels, Inconel-600 is a standard engineering material for applications which require resistance to corrosion and heat. The alloy has excellent mechanical strength, its room temperature yield strength is about 40,000 psi and, at 1,000°F, its yield strength still remains above 32,000 psi. The alloy does not embrittle after long exposure to high temperatures. Test results* indicate that the alloy maintains its room temperature ductility after 2159 hours of creep test at 1000°F under 10,000 psi. Inconel-600 is highly corrosion-resistant in water and particularly it is more resistant than stainless steels to chloride-induced stress corrosion cracking. Consequently, Inconel-600 has been chosen as the steam generator tube material for the commerical nuclear industry.

Q.4 Will the Inconel-600 tubing lose its strength or material properties after a certain period of service in an operating steam generator?

A.4 As pointed out in response to Q.3, mechanical strength and ductility of Inconel-600 does not change significantly even after prolonged exposure at 1000°F. At the normal nuclear power plant operating temperature of about 600°F, there will be no significant change in mechanical properties, microstructure or carbide precipitation of Inconel-600. Consequently, the alloy is expected to maintain its original strength and ductility even after prolonged service in operating steam generators under normal operating conditions.

* "Inconel-600", Huntington Alloys, Huntington, West Virginia, 1973, at 9.

Q.5 The TMI-1 steam generator tubes are sensitized. What does sensitized mean?

A.5 Inconel-600, like the austenitic stainless steels, containing more than approximately 0.025% carbon by weight will, when heat-treated in the temperature range between 800°F and 1600°F, form grain boundary networks of chromium (Cr) carbides. The precipitation of Cr carbides along grain boundaries results in a network of Cr-depleted zones adjacent to the grain boundaries in which the Cr concentrations fall below those in the bulk of the grains. This enables the formation of electrochemical cells between the grain boundaries and the grains. This phenomenon is commonly referred to as sensitization. Austenitic alloys, containing networks of Cr carbides along grain boundaries with adjacent Cr-depleted zones, are said to be sensitized.

Q.6 What caused the Inconel-600 tubes at TMI-1 to become sensitized?

A.6 When the manufacturing process is completed, the entire OTSG is placed in a furnace and heated to approximately 1100°F to relieve stresses in the structural welds. This heat treatment leads to precipitation of Cr carbides along gran boundaries with adjacent Cr depleted zones, thus causing the Inconel-600 to become sensitized.

Q.7 What effect does sensitization have on the structural integrity of the kinetically expanded repair joint?

A.7 Sensitization does not significantly alter the mechanical strength or ductility of Inconel-600. The microstructural changes which occur when Inconel-600 is sensitized affect primarily its resistance to various types of localized corrosion. Because sensitization does not significantly alter the mechanical strength or ductility of Inconel-600, it does not adversely influence the structural integrity of the repaired joint.

Q.8 What is the safety margin for degraded steam generator tubes?

A.8 The TMI-1 steam generators, as fabricated and installed, have tube wall thickness which exceed the Section III requirements of the ASME Boiler and Pressure Vessel Code. At tube wall degradations of approximately 70% (30% tube wall remaining) it has been demonstrated both analytically and experimentally that the boiler and pressure vessel code requirements are met and that steam generator tube integrity will be maintained for design basis accident conditions. Because the code itself contains a built-in conservatism of approximately a factor of 2, additional margin to failure exists for design basis accidents even if a tube is degraded in excess of 70% through wall.

Q.9 In NUREG-1019, its supplement and affidavit accompanying Staff's Motion for summary disposition it is concluded, in part, that the kinetic expansion repair process is acceptable because the OTSG's are returned to their original licensing basis. Considering that the Inconel-600 tubes have been exposed to various service and shutdown conditions for a period of ten years, how is it possible to conclude that they have been returned to the original licensing basis?

A.9 As discussed in response to Q.3 through Q.7, Inconel-600 tubing maintains its mechanical strength and ductility even after extended service in the steam generators. Inconel-600 was specifically selected for application in nuclear steam generators because its strength and ductility will not significantly change during the design life of the plant. In NUREG-1019 at Pg. 2 and 4 and Supplement No. 1 to NUREG-1019, at Pg. 1 and 2, the repair process is described. The repair process relies on a six-inch defect-free kinetic expansion joint plus a 2-inch or 7-inch defect-free unexpanded section. As discussed in Response to Q.8, the repaired joint plus the unexpanded sections could have met the ASME boiler and pressure vessel code requirements and therefore the original licensing basis even if degradation had been present. Because the licensee elected to remove from service all tubes which had defects within the kinetic expansion repair area and the defect-free unexpanded sections, the kinetic expansion repair process has returned the steam generators to the original licensing basis.

Q.10 Has the kinetic expansion repair increased the probability of simultaneous tube ruptures involving both TMI-1 steam generators?

A.10 No. The kinetic expansion repair has returned the TMI-1 steam generators to their original licensing basis. Because Inconel retains its strength and ductility despite previous operation and because the repair itself did not affect that strength and ductility, the tubes are as resistant to rupture now, after the repair, as they were when they were new and had not experienced operation. Therefore, the probability of simultaneous tube ruptures involving both steam generators is no greater now than it was at the time of the original licensing.

Q.11 Does the NRC consider the potential for tube ruptures in multiple steam generators to be any greater for TMI-1 than for any other plant?

A.11 The staff does not consider the potential for tube ruptures in multiple steam generators to be any greater for TMI-1 than for any other plant. A number of factors influence this conclusion.

1. The licensee has thoroughly quantified the corrosion condition of the steam generators by conducting 100% eddy current testing (ECT)* of both steam generators.

2. When considering steam generator tube ruptures, tubes in the free span (the 52 feet open area between upper and lower tubesheets) are the primary concern, because this is the only location where the classic guillotine break is possible. Tubes which have been repaired within the tubesheets are restrained from separating within the tubesheet crevice. Therefore, although leakage in the tubesheet is possible, "tube rupture" in the classic sense is not. Greater than 95% of all corrosion at TMI-1 took place within the upper tubesheet crevice, where separation is restrained. All tubes in the free span of both TMI-1 steam generators that were identified as defective have been removed from service. Therefore, both TMI-1 steam generators will be returned to service under the same criteria

* ECT is a means whereby the electrical conductivity of a tube is checked by passing a coil with an induced voltage along the tube. If some form of tube degradation has occurred (such as corrosion) which has separated the metal, an electrical discontinuity exists. The electrical discontinuity will be proportional to the amount of metal which is missing. If 40% or more of the tube wall is missing, the tube is classified as defective and has to be repaired or removed from service.

as other units which have experienced corrosion. These criteria have been demonstrated to be successful in reducing the potential for tube rupture.

3. The most limiting initiating event for a steam generator tube rupture is the main steam line break (MSLB) accident. Under MSLB, maximum differential pressure will exist on the tubes. For a tube to rupture during a MSLB, it would have to be uniformly degraded through by greater than 70% of its wall thickness. The tube plugging criteria of less than 40% includes a corrosion allowance for the next operating period and an uncertainty allowance. Because most corrosion mechanisms do not result in uniform degradation that would cause structural failure before an unacceptable leakage occurs, 40% plugging criterion is very conservative. This is evidenced by the fact that no steam generator tube ruptures due to corrosion have occurred since 1976, and only two occurred prior to that time.

4. In addition to the conservatism of the tube plugging criteria, a number of other factors contribute to making tube rupture in multiple steam generators at TMI-1 no more likely than before the corrosion problem.

   a. The vast majority of defects are within the upper tubesheet. The tubesheet structural restraint would act to prevent tube rupture, even if continued degradation occurs.

   b. All tubes with detected defects in the tube free span have been plugged.

c. The extent of ECT for the TMI-1 steam generators is greater than that performed at any other operating plant. The techniques used and extent of ECT provide reasonable assurance that defects which may be present have been detected. (NUREG-1019, Section 3.3).

d. A significant difference exists in the extent of corrosion between the two steam generators. Although both steam generators have been repaired to the same criteria, a statistical difference exists as to the potential for continued corrosion. This factor reduces the probability that ruptures would occur in multiple steam generators, even in the event of a MSLB.

e. The corrosion which has been found is circumferential, and in most cases involves less than one-third the tube circumference. This results in sufficient tube wall remaining to maintain structural strength, even for a MSLB, in the event a defective tube has not been identified.

f. Extensive pre-critical hot functional testing has been performed to verify reliability of the steam generators.

g. Subsequent to criticality, power escalation will be slow (approximately 8 weeks to reach 75% power). Once 50% power is reached, the plant will be shutdown within 90 to 120 days and the steam generators examined by ECT to monitor for continued corrosion.

h. Extensive efforts have been conducted to identify and remove the contaminents to mitigate the possibility of recontamination of the reactor coolant system (NUREG-1019 and Supplement No. 1 to NUREG-1019).

In summary, the probability of steam generator tube ruptures in multiple steam generators is no greater for TMI-1 than for any other plant.

Conrad E. McCracken

Professional Qualifications

I am Section Chief of the Chemical Technology Section in the Chemical Engineering Branch of the Division of Engineering, Office of Nuclear Reactor Regulation. My responsibilities in this position include supervision of the evaluation of all PWR's for compliance with chemistry and corrosion requirements of the Commission. Specifically, this includes evaluating the chemistry and corrosion control measures that are instituted to minimize corrosion of steam generator materials. I have served in this capacity since April 1982. Between February 1981 and April 1982 I served as a senior chemical engineer with the same branch, where my duties included the evaluation of steam generator chemistry and corrosion programs at both operating plants and plants in the licensing process.

From 1966 to 1981 I was employed by Combustion Engineering Corporation in a variety of management and engineering positions, the last of which was Manager of Chemistry Development from 1977 to 1981. During this 15-year period, my prime technical responsibility was support to operating nuclear power plants and nuclear plants in construction in the area of chemical and radiochemical sampling, analysis, data interpretation, establishing chemistry specifications and conducting laboratory experiments to verify or support nuclear plant requirements. In this capacity I made frequent visits to nuclear power plants where I physically conducted sample and analysis programs or audited the utilities' capabilities in the chemistry and radiochemistry area. During the last twelve years at Combustion Engineering, approximately fifty percent of my time was expended in areas associated with understanding and resolving steam generator corrosion problems.

From 1958 to 1966 I served in the United States Navy where I was Qualified in submarines for all nuclear duties. For three years of this period I was an instructor, responsible for teaching officer and enlisted personnel in the area of chemistry, corrosion and mechanical systems operations and control. My final duty station in the Navy was on the USS Nautilus, where I was responsible for all chemistry and corrosion control and personnel radiation exposure.

Education

I attended the University of Hartford School of Engineering and completed course work in 1970. I am a Registered Professional Corrosion Engineer.

Paul C. S. Wu
Chemical Engineering Branch
Division of Engineering

Professional Qualifications

FIELD: Corrosion and Water Chemistry

TOTAL EXPERIENCE: 19 years

NUCLEAR: 16 years

KEY RELEVANT EXPERIENCE

- Principal Engineer, Materials and Corrosion Programs at Westinghouse ARD
- Supervisor, Mechanical Properties Laboratory
- Lead Engineer, Advance Nuclear Control Material Development
- Lead Engineer, Materials and Corrosion Evaluation for Power Generating Equipments
- Lead Engineer, High Temperature Design Criteria and Method
- Lead Engineer, Low Friction and High Wear Resistant Materials Development
- Lead Engineer, Liquid Metal Corrosion and Sodium Technology

RELATED PROFESSIONAL BACKGROUND

Before joining the Metallurgy and Materials Research Branch at NRC, I was employed as a Principal Materials Engineer at the Westinghouse Advanced Reactors Division. I was responsible for many materials and corrosion programs at ARD. From 1976 to 1979, I was in charge of all materials and corrosion programs concerning aqueous corrosion and pertinent to nuclear fuel reprocessing and waste management. I was responsible for proposal preparation, research execution, and program coordination among various Westinghouse divisions and national laboratories. Prior to 1976, I was in charge of the Mechanical Properties Laboratory at ARD, and was responsible for characterizing the creep, fatigue, and stress-rupture of stainless steels and nickel-base alloys for the national program on high temperature design criteria and methods. Before joining Westinghouse, I was a research scientist at the Ames Laboratory of USAEC engaging in sodium technology and nuclear materials research.

EDUCATIONAL BACKGROUND

B.S. (Metallurgical Engineering, 1964), National Cheng-Kung University, Taiwan

M.S. (Metallurgical Engineering, 1967), University of Missouri at Rolla, Rolla, MO

Ph.D. (Materials Science and Inorganic Chemistry, 1972), Iowa State University, Ames, Iowa

MANAGEMENT TRAINING

- Management Techniques, Westinghouse Learning Corporation
- Decision Making, Westinghouse Learning Corporation
- Communication Skills, Westinghouse Learning Corporation
- Fracture Mechanics, Westinghouse Headquarters Engineering

HONORS AND ACHIEVEMENTS

- Who's Who in Technology
- Member of the American Honorary Chemical Society
- Member of the Review Board of TMS and ASM Publications
- Member of the NACE Committee on Stress Corrosion Cracking
- Member of the NACE International Relations Committee
- Westinghouse Advanced Reactors Division Cost Saving Award (1975)
- Technical Program Chairman, Pittsburgh Diffraction Conference (1976)

PUBLICATIONS AND PRESENTATIONS

I have published more than 30 technical papers and reports on subjects covering corrosion, materials evaluation and selection, mechanical properties of engineering alloys, sodium technology, friction and wear of materials, nuclear control material development, fuel reprocessing technology, and waste management. In addition, 8 invited presentations at Stanford Research Institute, ANL, ORNL, Sandia Laboratory and other research institutions have also been accredited to me.

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CERTIFICATE OF SERVICE

I hereby certify that copies of "TESTIMONY OF CONRAD E. McCRACKEN AND PAUL C. WU ON TMIA CONTENTION 1.A" and "TESTIMONY OF CONRAD E. McCRACKEN AND PAUL C. WU ON TMIA CONTENTION 1.B" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, this 29th day of June, 1984:

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