UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket No. 50-454-OLA
	)	50-455-OLA
(Byron Station, Units 1 and 2)	)	

TESTIMONY OF MAHENDRA R. PATEL CONCERNING STEAM GENERATOR TUBE INTEGRITY (TUBE PLUGGING CRITERIA)

Submitted on behalf of
the Applicant, Commonwealth Edison
Company in Response to DAARE/SAFE
Contention 9c and League Contention 22

February 25, 1983

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SUMMARY

The testimony of Dr. Mahendra R. Patel sets forth his qualifications as an expert and addresses the bases for establishing the tube plugging criterion for use at Byron Station. Specifically, Dr. Patel establishes that a 40% tube plugging criterion is conservative for application at the Byron Station based on:

1. The tubes in the Byron Station steam generators meeting the minimum tube wall thickness needed to sustain the loads from the normal operating and design basis accident conditions as established and required by paragraph IWB-3521.1 of Section XI of the ASME Code.

2. The leak-before-break principle, as implemented by a technical specifications' maximum permissible leak rate limit at the Byron Station of 0.35 gpm per steam generator, assuring that a tube wall crack no greater than 0.43 inches will be detected and repaired before the crack reaches the critical crack length (0.51 inches) at which point tube rupture could occur under postulated loss-of-coolant accident and main steam line break accident conditions.

3. The 40% tube plugging criterion including an allowance of 10% tube wall thickness, as established by the ASME Working Group, to accommodate continued tube degradation between inspections and eddy current measurement uncertainties.

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CONCERNING STEAM
GENERATOR TUBE INTEGRITY
(TUBE PLUGGING CRITERIA)

Q.1. State your name, address and present occupation.

A.1. My name is Mahendra R. Patel. My business address is P.O. Box 355, Pittsburgh, Pennsylvania 15230. I am employed by the Westinghouse Electric Corporation as a manager of a structural mechanics group in the Nuclear Technology Division of the Water Reactor Division headquartered at the Monroeville Nuclear Center near Pittsburgh.

Q.2. Please state your educational background.

A.2. I graduated from high school at the top of my class (of about 300) in 1963 in India and was awarded the Government of India National Scholarship which I retained throughout my undergraduate school by passing each grade First Class with Distinction (top

1% of university enrollment). After two years of science, I transferred to engineering and graduated with a Bachelor's degree in Mechanical Engineering in May 1968 from Gujarat University, India.

In September of 1968, I enrolled in the University of Missouri at Rolla for post-graduate studies in Mechanical Engineering/Engineering Mechanics and graduated in December 1973 with a Ph.D. in Mechanical Engineering. My post-graduate study was supported through various research and teaching assistantships in the area of heat-transfer, structural and solid mechanics, vibrations and mechanical testing.

Q.3. Please state your work experience following completion of your post-graduate studies.

A.3. In March of 1974, I joined the Pressurized Water Reactor Systems Division of the Westinghouse Water Reactor Division. My duties consisted of development and interpretation of structural criteria and design loads used for the nuclear steam supply system and mechanical components and performing of detailed stress analyses required as a basis for design and criteria verification and resolution of technical issues. As part of my duties, I acquired

an in-depth knowledge of mechanical design and safety requirements on nuclear power plant systems and components in accordance with the ASME Boiler and Pressure Vessel Code and applicable USNRC Regulatory Guides.

Q.4. Have your duties changed since that time?

A.4. Yes. In July 1977, I transferred to the Nuclear Steam Generator Division of the Water Reactor Division. My responsibilities included: (1) mechanical design, analysis and test support in the development of future generation steam generators with advanced features, (2) in-depth testing and analysis to verify all aspects of structural integrity of steam generator tubing, and, (3) development and implementation of plugging criteria for structurally degraded tubing to ensure continued safe operation of steam generators. During the course of this work, I amassed an extensive background and experience relative to the material and mechanical strength characteristics of PWR steam generator tubing under the typical full operating range of thermal, mechanical and chemical environment in a steam generator. The tube mechanical response evaluation includes in-depth analyses and testing for the determination of burst strength,

leak rate, external pressure collapse, thermal-mechanical fatigue and flow-induced vibration/wear characteristics of virgin as well as service exposed tubing representative of modes of degradation such as thinning, wastage, denting, pitting, stress corrosion cracking and intergranular attack. Results of these studies are used in conjunction with the stress analyses of steam generator tubing subject to the postulated accident condition (loss of coolant, main feedline/steam line break and maximum design basis safe shutdown earthquake) loads to establish the tube plugging criterion for continued safe operation.

In November 1980, I was promoted to Principal Engineer in Nuclear Technology Division and continued working on additional aspects of steam generators including tube sleeving and waterhammer evaluation.

Q.5. What are your present management responsibilities at Westinghouse?

A.5. In August 1981, I became manager of the Auxiliary Equipment Analysis group. In addition to the design/stress analysis of auxiliary equipment (auxiliary heat exchangers, tanks, demineralizers,

filters, etc.), I retained the scope on steam generator tube integrity evaluation and plugging criteria. Most recently, I had the lead technical responsibility to determine and evaluate the failure mechanism that led to a tube rupture on January 25, 1982 at the Ginna Station and to assure structural integrity following the repairs.

Q.6. What is the purpose of your testimony?

A.6. My testimony addresses one of the guidelines used to assure structural integrity of steam generator tubes. Specifically, the testimony explains the bases for establishing steam generator tube plugging criteria i.e., determining the wall thickness limit below which tubes are to be removed from service by plugging them to avoid the possibility of tube failure due to degradation. In addition, I discuss the tube plugging criterion applicable to the Byron Station.

Q.7. Why are tube plugging criteria necessary?

A.7. The steam generator tubing which is part of the primary system pressure boundary represents an integral part of a major barrier against the release of radioactivity to the environment. Accordingly, conservative design criteria for tube wall sizing

have been established to assure structural integrity of the tubing under normal operating and the postulated design-basis accident condition loadings. In reality, however, steam generator tubes are manufactured with a wall thickness much greater than the minimum thickness indicated by the design rules (of Section III of the ASME Boiler and Pressure Vessel Code). Heavier wall thicknesses than required by design rules are used in procurement primarily to accommodate fabrication, installation, and handling procedures.

However, over a period of time under the influence of the operating loads and environment in the steam generator, some tubes may become degraded due to localized wall degradation or cracking. Partially degraded tubes are acceptable for continued service as long as it is assured, through in-service inspections (using eddy current techniques) and leakage monitoring, that degraded tubes meet the applicable tube wall and associated strength requirements to safely withstand all operating and design basis accident condition loads. Standards and guidelines are provided for establishing the "limiting safe conditions" of tube degradation beyond which defective tubes must be repaired or

removed from service. The amount of degradation, expressed as a percentage of the design nominal tube wall thickness below which the tube shall not continue in service, is customarily defined as the "tube plugging criterion." The acceptable tube degradation is referred to as the plugging margin.

Q.8. How is the "limiting safe condition" of tubing established?

A.8. To establish the limiting safe condition, the following three factors must be considered:

(1) Minimum tube wall thickness needed in order to sustain the imposed normal operating and postulated design basis accident condition tube loads.

(2) Maximum (technical specification) permissible leak rate during normal operation to preclude a tube rupture during a postulated main steam line break accident, and

(3) Allowance for continued degradation between inspections and eddy current measurement uncertainties.

The minimum required tube wall thickness is based on the results of detailed analyses and strength testing. Both the thermal-hydraulic and stress

analyses are performed to determine tube loads and the resulting stresses during the normal as well as various postulated accident conditions.

From the viewpoint of mechanical loads on tubing, the governing design basis accidents are the primary and the secondary side full blowdown accidents and the safe shutdown earthquake (SSE). During a loss-of-coolant accident (LOCA), the primary pressure could theoretically drop to almost zero resulting in an external pressure differential approximately equal to the secondary side operating pressure. This is the worst credible collapse mode accident for the tubing. Additionally, tubing is subjected to high bending stresses due to primary loop shaking and the pressure differential across the U-bend resulting from the rarefaction wave. In the case of a secondary side blowdown accident, main feedline or steamline break (FLB/SLB), the secondary pressure could drop to zero, resulting in an internal pressure differential equal to the full primary loop pressure. This is the worst credible burst mode accident for the tubing.

On the basis of extensive burst and collapse pressure testing, it has been demonstrated that the

minimum tube wall thickness required to safeguard against tube rupture during a postulated FLB/SLB accident is always more limiting than the minimum tube wall thickness required to safeguard against collapse during a postulated LOCA. Also, collapse of tubing made of highly ductile material such as Inconel 600 does not result in cracking. Thus, the only concern with tube collapse during a LOCA is the loss of primary flow area in the tube bundle and its impact on core cooldown rate following the LOCA. Based on detailed analyses of steam generators similar to the design at the Byron Station, the loss in primary flow area in the faulted steam generator is approximately 2 to 3 percent of nominal full flow area, and has no consequential impact on the rate of core cooldown or the peak clad temperature.

Plant Technical Specifications provide for an orderly plant shutdown and corrective action in case of primary-to-secondary tube leakage. To preclude the possibility of rupture of a leaking tube during the governing design basis accident condition(s), leak rate and burst pressure tests are performed. The maximum permissible leak rate during normal operation is then established to assure leak-before-break.

The final determination of tube plugging margin includes allowances for uncertainties in inspection measurements and additional probable degradation that may occur during the next operating interval. Analytical procedures and the operating history of the same or similar type of steam generator are used to determine these allowances.

Plugging margins established in accordance with the above requirements ensure that, at the end of an operating period, a degraded tube with loss of wall or a leak (1) will not undergo progressive yield, that is permanent deformation during operation, (2) will not burst or rupture during either the normal operation or the governing design basis accident(s), and (3) will meet, under the postulated accident condition loadings, the applicable stress limits specified in Appendix F of Section III of the Code.

Q.9. What standards or guidelines are used for establishing tube plugging criteria?

A.9. There is only one standard, Section XI of the ASME Boiler and Pressure Vessel Code (Code). The Code provides for two alternative methods. Paragraph IWB-3521.1 allows a depth of outside diameter degradation not to exceed 40% of the wall thickness

for tubing from SB 163 material when the mean tube radius to wall thickness ratio is less than 8.70.

Alternatively, Paragraph IWB-3630 provides that a plant-specific evaluation may be performed by analysis and testing consistent with the guidelines of U.S. NRC Reg. Guide 1.121 to determine the allowable tube wall degradation.

Regardless of which of the above two approaches is used, the maximum permissible leak rate must be established consistent with the leak-before-break principle and set forth in the Technical Specifications.

Q.10. What is the leak-before-break principle?

A.10. The mechanical and metallurgical properties of ductile materials such as the Inconel 600 used in steam generator tubing are such that degradation in the form of cracking penetrates through wall and a small leak occurs long before the crack reaches a linear length called "critical length," at which tube rupture can occur. When a crack first penetrates through wall, the result is a primary-to-secondary leak. The degradation thus becomes detectable through leakage monitoring. This is

known as the leak-before-break principle and it provides a mechanism for positive detection of tube degradation before tube rupture can occur. By establishing a conservative limit on the maximum permissible leakage during operation, continued safe operation can be assured without the potential of tube rupture under the postulated design basis accident condition loading.

Q.11. How is the maximum permissible leak rate established for the Byron Station?

A.11. From the viewpoint of tube rupture, the most limiting crack orientation is along the tube axis. An extensive test program consisting of leak rate and burst pressure tests was undertaken to establish correlations among the length of the crack, the associated leak rate during normal operation and burst strength. The maximum permissible leak rate during normal operation for the Byron Station has been established at 0.35 gpm per steam generator. This corresponds to the maximum allowable crack length of 0.43 inch based on results of the leak rate tests. On the other hand, the critical crack length corresponding to the maximum accident condition pressure during a postulated FLB/SLB was conservatively determined to be 0.51 inch using the

results of the burst pressure tests. Since the critical crack length is greater than the maximum permissible length of 0.43 inch for continued operation, the unit is safeguarded against tube rupture during a postulated FLB/SLB accident.

Q.12. What tube plugging criterion will be used at the Byron Station?

A.12. Currently, a plugging criterion of 40% indication of wall thickness has been established for the Byron Station per paragraph IWB-3521.1 in Section XI of the ASME Code.

Q.13. What is the basis for the 40% tube plugging criterion suggested by the ASME Code?

A.13 The plugging criterion of 40% indication in paragraph IWB-3521.1 in Section XI of the ASME Code is based on the work performed by its Working Group (WG) on Steam Generator In-Service Inspection. The theoretical bases used by the WG in determining the plugging criterion were the same as those discussed previously in A.8.

In its evaluation, the WG considered four different steam generator designs: the Combustion Engineering Non-Economizer and System 80 designs and the

Westinghouse Series 51 and Model D designs. The tubing material for all four designs is Inconel 600 (SB-163), and the ratio of mean radius to thickness ranges from 7.31 to 8.7. Each design was evaluated generically (as opposed to plant specific loads) using its load envelopes during normal operation and postulated design basis accidents. A conservative analytical correlation was derived between the burst pressure and percent tube wall degradation enveloping the tubing materials and geometries in all four designs. The conservatism in this correlation was verified using the actual burst pressure test results on representative tubing with various types of simulated defects including through-wall and part through-wall cracks, machined slots, and elliptical and uniform circular wastage.

Based on this evaluation, the WG concluded that for the four designs considered, the tubing can sustain degradation in excess of 50% of wall thickness and still meet all applicable stress and strength requirements. Finally, a conservative factor of 10% wall thickness was assumed to account for both eddy current measurement uncertainty and corrosion allowance for the continued plant operation until next inspection. This factor was assumed on the

basis of some developmental work and historical data on eddy current testing and corrosion rates on operating plants. Thus, a plugging criterion of 40% (50%-10%) tube wall degradation was established and adopted by the WG on standards for ASME Section XI.

Q.14. Are the models and designs reviewed by the ASME Working Group representative of the Byron Steam Generators?

A.14. Yes. The steam generators at the Byron Station are Westinghouse Model D design which is one of the designs considered by the WG. The tubing material is Inconel 600 (SB-163) and the mean radius to thickness ratio is 8.22 which is less than 8.70, the upper limit specified in Section XI. Thus, the 40% plugging margin established per Section XI standard is conservatively applicable to the Byron Station.