UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER AND LIGHT COMPANY AND NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY	)	Docket Nos. 50-400 OL 50-401 OL
	)	
(Shearon Harris Nuclear Power Plant, Units 1 and 2)	)	

AFFIDAVIT OF JAI RAJ N. RAJAN, HERBERT F. CONRAD AND PAUL C.S. WU, IN SUPPORT OF NRC STAFF'S RESPONSE TO APPLICANTS' MOTION FOR PARTIAL SUMMARY DISPOSITION OF JOINT CONTENTION VII

I, Jai Raj N. Rajan, being duly sworn do depose and state:

1. I am employed by the Nuclear Regulatory Commission as a Mechanical Engineer in the Mechanical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached to this affidavit as Exhibit A. The statements made are true and correct to the best of my knowledge, information and belief.

2. I am responsible for the review of flow-induced vibration problems and structural integrity concerning the steam generators to be used at the Shearon Harris Nuclear Power Plant.

I, Herbert F. Conrad, being duly sworn do depose and state:

3. I am employed by the Nuclear Regulatory Commission as a Senior Materials Engineer, in the Materials Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation. A copy of my

professional qualifications is attached to this affidavit as Exhibit B. The statements made are true and correct to the best of my knowledge, information and belief.

4. I am responsible for the review of inservice inspection and surveillance concerning the steam generators to be used at the Shearon Harris Nuclear Power Plant.

I, Paul C. S. Wu, being duly sworn do depose and state:

5. I am employed by the Nuclear Regulatory Commission as a Chemical Engineer, in the Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached to this affidavit as Exhibit C. The statements made are true and correct to the best of my knowledge, information, and belief.

6. I am responsible for the review of corrosion aspects and secondary water chemistry control concerning the steam generators to be used at the Shearon Harris Nuclear Power Plant.

7. This affidavit is provided as part of the Staff's response to Applicants' Motion for Summary Disposition of Joint Contention VII subparts 1, 2 and 3.

JOINT CONTENTION VII - (1)

Joint Contention VII states in pertinent part:

> Applicants have failed to demonstrate that the steam generators to be used in the Harris Plant are adequately designed and can be operated in a manner consistent with the public health and safety and ALARA exposure to maintenance personnel in light of

(1) vibration problems which have developed in Westinghouse Model D4 steam generators; (2) tube corrosion and cracking in other Westinghouse steam generators with Inconel-600 tubes and/or carbon steel support plates and AVT water chemistry; (3) present detection capability for loose metal or other foreign objects; and (4) existing tube failure analyses.

Flow Induced Vibration

8. The Shearon Harris nuclear plant has Westinghouse Model D-4 steam generators. A flow-induced vibration problem resulting in increased wear of some tubes in the preheater section of these steam generators was first discovered at the KRSKO nuclear plant in Yugoslavia, which also has model D-4 steam generators. As Applicants have indicated, Westinghouse performed extensive laboratory testing and analyses to study this vibration problem and find a solution which would reduce the tube vibration to acceptable levels. Timmons Affidavit at ¶ 11. As a result of these investigations, Westinghouse proposed the use of pressure controlled expansion of selected tubes in the preheater region in conjunction with bypass of the main feedwater flow via the auxiliary nozzle. These modifications were evaluated by Westinghouse through a comprehensive test and qualification program. An extensive evaluation of these modifications by the NRC Staff and its consultants indicates that the tube vibration and consequent wear will be reduced to an acceptable level as a result of these modifications. These evaluations are contained in Sections 1, 2 and 3 of NUREG-1014 (Safety Evaluation Report Related to the D4/D5/E Steam Generator Design Modification), attached hereto as Exhibit D. To ensure that the modifications made to the Model D-4 steam generators and/or main/auxiliary feedwater systems would result in an acceptable design, Westinghouse

established certain design objectives which have been met. The NUREG-1014 Appendix B at 16-17 pertinent design objectives are discussed in the paragraph 9 through 12.

9. G-Delta is a measure of the magnitude of the vibration and is related to the wear producing capability of the vibration. A value of G-Delta that is considered appropriate for long-term operation of the plant was established and all vibration levels were held below this value.

10. The vibrational levels were limited such that the maximum wear scar depths based on conservative predictions were projected to be less than 65% of the wall thickness (structural integrity limit) for an interval of 18 equivalent full power months. The safety requirements governing the hydraulic performance criteria for the modified Model D-4 steam generator is that tube wear due to flow-induced vibration shall not result in tube wall reduction in excess of the safety limit for tubes in service. The safety limit for tube wall reduction is the amount of wall loss the tube can sustain and maintain integrity under the most severe accident conditions. For preheat steam generator tubing, this limit has been determined by analysis and test to be a 65% wall reduction.

11. Effects of tube expansion process were evaluated using the ASME Boiler and Pressure Vessel Code, Section III which forms the Staff basis for the structural design.

12. The effects of the modifications were not allowed to violate required reactor thermal margins or any other plant operational safety parameters. The tube expansion provides an area of close contact between the tube and baffle plate. This close support condition significantly

changes the response frequency of the tube and also the G-Delta value. The split feedwater flow reduces the mass flow and velocity of the fluid in the preheater section. Both modifications combine to provide a substantial improvement by reducing the potential for tube wear. The design modifications and their consequences for steam generator and plant performance were reviewed extensively by the Staff concurrent with other independent reviews. These reviews are documented in Sections 1 through 5 of NUREG-1014. These reviews address the following major areas: (1) effectiveness of the modifications in reducing tube vibration and we ~, (2) assurance that the tube expansion would not induce any large residual stresses or defects sufficient to cause early tube leaks, (3) assurance that the modifications do not produce conditions within the preheater that accelerate tube corrosion, (4) assurance that non-destructive examination (NDE) is capable of detecting wear and/or cracks at the tube expansion regions, and (5) assurance that leak before break criteria are met. The Staff and independent reviewers concluded that the proposed modifications assure substantial improvements by reducing the potential for tube wear. This conclusion was reached after a thorough review of test models and testing results, as well as evaluation of analytical models and analysis results.

13. The Staff, its consultants and independent reviewers performed a detailed review of the stress analysis, which included examination and evaluation of analytical methods and models. It was concluded that these modifications (tube expansion and split feedwater flow) meet the established structural design criteria. The review of the safety analysis led to the conclusion that the expanded tubes will meet the

"leak before break" criteria and that minimum wall thickness requirements are changed by only an acceptably small amount. Thus, normal periodic tube inspection and/or prompt remedial actions, whenever required, provides adequate assurances that an extremely low probability of tube rupture exists, consistent with the requirements of GDC-14.

14. The Staff considers that the corrosion potential of the expanded tubes is not significantly changed from that of the nonexpanded tubes. The Staff recognizes that in certain tubes the expansion conditions (i.e., baffle plate hole size and tube fit) may lead to local strains in the tubes. The larger strain levels are associated with tube expansions greater than 30 mils diameter. The strain levels of the tube expansion in the baffle plate are no larger than strain levels of similar expansions at the tube sheet which are currently operating successfully in all Westinghouse steam generators. Stress corrosion cracking in the cold leg tube sheet expansions has not been a problem in operating plants. The Staff considers the number of tubes subject to large expansions to be relatively small compared with the total number of tubes being expanded. Some tube wear may occur over the projected 40-year life of the steam generators. Also, some tubes in the expansion area may require plugging during the steam generator lifetime. This is judged to be acceptable since proper equipment for tube inspection and remedial action, as required, is available. In addition, the Shearon Harris steam generator tubes will be subject to periodic inservice inspections in accordance with Regulatory Guide 1.83, "Inservice Inspection Requirements of Pressurized Water Reactor Steam Generator Tubes", Rev. 1, and NUREG-0452, Rev.2, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors:

(STS). Operational limits of allowable primary to secondary leakage will provide added assurance of adequate tube integrity.

15. Operations with split feedwater flow, i.e., diverting part of the feedwater through the auxiliary feedwater nozzle, will be required for Model D4 steam generators. In considering the functional feasibility of split feedwater flow, the Staff and consultants have reviewed the steam generator stress analysis and has also examined and evaluated the analytical methods and models. It was concluded that steam generator operation with split feedwater flow is feasible and acceptable. The Staff is cognizant of the fact that virtually every facility having Model D4 steam generators has unique feedwater system designs. For this reason, the Staff in its evaluation in NUREG-1014 did not include all feedwater piping designs, but only the generic effect of split feedwater flow on the steam generators. The feedwater system modifications and operating procedures for the Shearon Harris plant were reviewed and found feasible and acceptable.

16. Each individual plant has differences in its design and its Safety Analysis Report (SAR). The following areas of the SAR were examined for potential impact due to the implementation of the steam generator modification at Shearon Harris.

Chapter 3. Design of Structures, Components, Equipment, and Systems

3.6. Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.9. Mechanical Systems and Components

Chapter 4. Reactor

4.4 Thermal and Hydraulic Design

Chapter 5. Reactor Coolant System and Connected Syster

5.1. Summary Description

5.2. Integrity of the Reactor Coolant Pressure Boundary

5.3. Reactor Vessel

5.4 Components

Chapter 6. Engineered Safety Features

6.2. Containment Systems

Chapter 7. Instrumentation and Controls

Chapter 9. Auxiliary Systems

9.2. Water Systems

9.3. Process Auxiliaries

Chapter 10. Steam and Power Conversion System

10.1. Summary Description

10.4. Other Features of Steam and Power Conversion System

Chapter 15. Accident Analyses

Chapter 16. Technical Specifications

It was determined that the impact of the modifications on the safety analyses is within the tolerance of the available plant margins at Shearon Harris.

17. In light of the modifications made to the Shearon Harris Steam Generators and the surveillance requirements and operational limits to be imposed on these steam generators, the Staff concludes that the vibration problems which developed in Westinghouse Model D-4 steam generators have been satisfactorily resolved.

AVT Water Chemistry

18. Tube degradation problems at Westinghouse steam generators have included the following: (a) wastage and thinning corrosion, (b) pitting, (c) denting, (d) intergranular attack, (e) stress corrosion cracking, (f) wear caused by flow-induced vibration, and (g) wear and/or impact damage as a result of foreign objects or loose parts.

19. Measures which have been taken by Westinghouse and the Applicants to minimize the potential for degradation due to corrosion including improved design features, use of all volatile treatment (AVT) secondary water chemistry, and an improved program to monitor and control secondary water chemistry.

20. Numerous design changes and operational procedures have been specifically incorporated at Shearon Harris to minimize steam generator tube corrosion. These include: (1) elimination of tubesheet crevices; (2) counterflow (axial flow) preheater which minimizes the propensity for steam blanketing; and (3) increased blowdown capability, plus a blowdown tee in the middle of the hot leg bundle to remove corrosion products on the tubesheet.

21. Improved condenser design, including integrally grooved tube sheets and continuously monitored sampling points to enable detection of tube leaks, and installation of a full flow condensate polishing system for feed water purification during startup and to aid in rapid cleanup when condenser leakage occurs further enhance the steam generator tube integrity.

22. The Applicants also incorporated a condition in the plant technical specifications which requires a secondary cycle water

chemistry program. The Shearon Harris' secondary water chemistry control program was reviewed against the criteria of the Standard Review Plan § 5.4.2.1, Rev. 1 and Positions II.2 and II.3 of Branch Technical Position MTEB 5-3, Rev. 2 and was found acceptable. Contrary to the Joint Intervenor's belief, Robinson 2 was operated from the start and continued until it was shut down in January 1984 under phosphate water chemistry control. It has never been operated under AVT chemistry. The Westinghouse All-AVT plants have thus far experienced only minor denting effects. None of the more advanced cases of denting (moderate or extensive) has occurred at Westinghouse plants in the All-AVT category. All instances of moderate or extensive denting have been observed in the group of plants that were in operation prior to August of 1974, and most of those plants are located on seaside or brackish water sites.

23. Joint Intervenors contend that it is not certain that any available water chemistry controls by themselves will be sufficient to minimize or prevent tube corrosion or cracking and satisfy this contention. All metals which maintain contact with an aqueous environment will corrode at a finite rate, depending on the corrosion potential of the aqueous environment. Water chemistry controls are implemented to minimize the potential for corrosion so that adequate assurances are provided against tube rupture between inservice inspections, consistent with GDC-14. With the implementation of an inservice inspection program as required by the technical specifications, the Applicants will periodically and systematically inspect and monitor the steam generator tube integrity to uncover any defect or degradation before they deteriorate

into serious problems. If the degradation exceeds the limit defined in the technical specifications, repairs or replacement will be made according to accepted techniques.

24. The modifications that have been incorporated at Shearon Harris in conjunction with the technical specification requirements for Inservice Inspection make it highly unlikely that a steam generator tube rupture will occur due to corrosion. To date there have only been two steam generator tube ruptures at domestic operating PWR's as a consequence of corrosion. A 125 gpm rupture occurred at Point Beach in 1975 and 50 to 80 gpm rupture occurred at Surry in 1976. It has been over eight years since a steam generator tube rupture due to corrosion has occurred.

25. The rupture at Point Beach was caused by secondary side intergranular stress corrosion cracking which occurred as a consequence of reactions between condenser inleakage impurities and residual phosphates. Shearon Harris will use all volatile chemistry treatment (AVT); consequently, the chemical reactions which caused the Point Beach rupture cannot occur at Shearon Harris.

26. The rupture at Surry was initiated from the primary side of the tube caused by excessive tube stress. The excessive tube stress resulted from extensive tube denting which first froze the tube in place and then physically moved the tube support plates, resulting in a significant deformation of the tube and resultant high stress. The water chemistry control requirements at Shearon Harris in conjunction with ISI requirements will combine to make it highly unlikely that extensive denting will occur.

27. Various corrosion phenomena aspects attributed to the steam generator modification such as tube expansion and gap size, were discussed in § 5.4.2 in the SER (NUREG-1014) and the TRC report (Appendix B of NUREG-1014). The Staff concluded that the susceptibility to denting, stress corrosion cracking, and the propensity for wastage have been adequately addressed and tested and are acceptable.

28. The Staff concluded that D4 steam generator constitutes an improved design over the D2/3 steam generator design. Therefore, we determined that GDC 1, 14, 15 and 31 have been met as they apply to minimize the possibility of a rapidly propagating failure of the RCS pressure boundary and that assurances exist that the public health and safety is protected and a significant hazard does not exist.

29. Accordingly, a technical basis has not been provided by Joint Intervenors in support of Joint Contention VII-(2).

Loose Parts Monitoring

30. The Harris Plant loose parts monitoring program is described in Applicants' letter of October 28, 1983 to H.R. Denton, NRC and SER § 4.4.4 reports on the Staff review and acceptance of that program. See also, Attachment 1, to Lang Affidavit. In the October 28, 1983 letter, the Applicants provided detailed information regarding the loose parts monitoring system (LPMS) including system description and operational procedures. In attachment 1 of the Applicants' letter, installation instruction and procedures for locating the metal impact monitoring system sensors are also discussed. By providing descriptions on sensor location and system sensitivity, channel separation and data acquisition

system, and alert level and channel operability test, the LPMS meets all regulatory position items noted in Section C of Regulatory Guide 1.133 "Loose Parts Detection Program for the Primary System of Light-Water-Cooled Reactors".

31. Shearon Harris used the Westinghouse metal impact monitoring system (MIMS), which is the same system previously reviewed according to Regulatory Guide 1.133 and approved for the Virgil Summer plant. Installation of the MIMS at the Harris plant will reduce the potential for foreign objects or loose parts remaining in the steam generator for long periods of time and potentially causing damage to tubes as in the Ginna event.

32. Intervenors have indicated that they believe that both the sensors and the systems other components should be safety grade. Installation of a primary side loose parts monitoring system (LPMS) is recommended in Regulatory Guide 1.133 for the reactor vessel and primary coolant boundary. However, there is no requirement that it should be a safety grade system. Safety grade systems are predominantly required in instances where a systems function is needed in the event of an accident condition. During any significant accident condition the plant is shutdown. Therefore, flow in the steam generator is reduced to such low velocities that vibrational wear even in the presence of loose parts would be nonexistent. Consequently, all MIMS hardware has been procured under engineering design specifications that require performance under normal, non-accident, environmental conditions. No Staff requirements or recommendation exists or is contemplated that a secondary side LPMS be installed. Since there is no regulatory requirement, the secondary side

LPMS which the Applicant has elected to install does not have to be safety grade.

33. Accordingly, a technical basis have not been provided by Joint Intervenors in support of Joint Contention VII. For the reasons discussed above the Staff concludes that the proposed loose parts monitoring system for Harris is adequate, and there is no technical basis to support the contention. .

CONCLUSION

For the reasons discussed above the Staff concludes, with respect to the issues raised in Contention VII, subparts 1, 2, and 3, that the steam generators to be used at Harris are adequately designed and can be operated in a manner consistent with the public health and safety and minimization of exposure to maintenance personnel.

Jai Raj N. Rajan
Jai Raj N. Rajan

Herbert F. Conrad
Herbert F. Conrad

Paul C.S. Wu
Paul C.S. Wu

Subscribed and sworn to before me
this 31st day of May, 1984

Edythe L. Becker
Notary Public

My Commission expires: July 1, 1986

EXHIBIT A

PROFESSIONAL QUALIFICATIONS

JAI RAJ N. RAJAN

U. S. NUCLEAR REGULATORY COMMISSION

MECHANICAL ENGINEERING BRANCH

DIVISION OF ENGINEERING

I am a mechanical engineer responsible for reviewing and evaluating safety analysis reports with regard to mechanical engineering aspects of components, the dynamic analyses and testing of safety related systems and components and the criteria for protection against the dynamic effects associated with postulated failures of fluid systems for nuclear facilities. I am the Mechanical Engineering Branch's principal reviewer on the issue of the structural integrity and plugging criteria of degraded steam generator tubes. I am also responsible for the review and evaluation of vibration problems of a generic nature in the piping systems and components of nuclear facilities.

I received a B.S. degree in 1953 from Lucknow University India majoring in Physics, Mathematics and Chemistry. In 1956 I received a B.S. in Civil Engineering from Roorkee University, India majoring in Structural and Hydraulic Engineering. In 1962 I received a M.S. degree from Duke University majoring in Applied Mechanics and Ph.D. degree in 1966 from the same university with majors in Fluid Mechanics. From 1960 to 1962 I was an instructor in structural engineering at Duke University. From 1962 to 1966 I was employed by the U.S. Army Research Office in Durham, N.C. as a research engineer conducting theoretical and experimental research in high pressure pneumatic and hydraulic shock tubes and investigating wave propagation phenomenon in pipes. From 1966 to 1973 I worked as a project mechanical engineer and subsequently as a senior project mechanical engineer at the Naval Research and Development Center at Annapolis, Md. Major projects involved design analysis, test and evaluations of fluid piping systems and power fl d systems of advanced nuclear submarines. Investigations were multidisciplinary in scope utilizing advanced techniques. Mathematical models of power plant machinery and piping systems of nuclear submarines were developed and analyzed to determine system response to flow induced vibrations and hydraulic shock. Thermodynamic and hydrodynamic analyses of naval boilers and steam plants were conducted including full scale tests.

In April of 1974 I joined the U. S. Atomic Energy Commission prior to the formation of the U. S. Nuclear Regulatory Commission and have remained with the Mechanical Engineering Branch of the Division of Engineering as a mechanical engineer performing the type of work as previously described.

I have taught at the University of Maryland on a part-time basis since 1967 both at the graduate and undergraduate levels in courses of mechanics of materials, fluid mechanics and applied mechanics.

Publications include Journals of AIAA and ASME and I am an associate member of Sigma Xi honor society.

U.S. NUCLEAR REGULATORY COMMISSION
HERBERT F. CONRAD
PROFESSIONAL QUALIFICATIONS

My present position is Senior Materials Engineer, Material Engineering Branch, Office of Nuclear Reactor Regulation. In this capacity I am responsible for technical safety review and evaluation of materials used in the construction of nuclear power plant components. Specifically, the responsibilities include evaluation of materials application, heat treatment, fabrication, inspection and corrosion control. I am a former member of the American Society of Mechanical Engineers Nuclear Code Committee Subgroup on Fabrication and Examination (Section III).

I hold a MS in Metallurgy (1959) and a BS in Mechanical Engineering (1957) from the Massachusetts Institute of Technology. I am registered by the State of California as a Professional Engineer in Mechanical Engineering and in Metallurgical Engineering with more than 24 years of professional experience. I am a member of the American Society for Metals (ASM). I have several publications in metallurgy, the most recent is a contribution to the ASM Metals Handbook, Volume 10, Failure Analysis (ASM, 1975).

I have been with the Nuclear Regulatory Commission since February 1973, two years of which were as a loan employee on detail from the University of California. Prior to my assignment to Washington, I was employed by the Lawrence Livermore Laboratory of the University of California as a Metallurgist.

EXHIBIT C

U.S. NUCLEAR REGULATORY COMMISSION
PAUL WU
PROFESSIONAL QUALIFICATIONS

My present position is Chemical Engineer, Chemical Engineering Branch, Office of Nuclear Reactor Regulation. In this capacity I am responsible for technical safety review and evaluation of materials and coolant chemistry control in LWRs. Specifically, the responsibilities include evaluation of materials application, corrosion prevention, and secondary water chemistry control in steam generators.

I hold a BS (1964) and a MS (1967) in Metallurgical Engineering, and a Ph.D. in Metallurgy (1972) from Iowa State University. I have more than 17 years of nuclear experience. I have more than 30 publications in materials engineering and corrosion, the most recent is a contribution to the Encyclopedia of Materials Science and Engineering to be published by the Pergemon Press in 1984.

I have been with the Nuclear Regulatory Commission since March 1980. Prior to joining the NRC, I was employed as a principal engineer of the Westinghouse Electric Corporation in Pittsburgh. Before that, I was a research scientist of the Ames Laboratory in Ames, Iowa.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

'84 JUN -6 A11:29

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

In the Matter of

CAROLINA POWER AND LIGHT COMPANY AND NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY

(Shearon Harris Nuclear Power Plant, Units 1 and 2)

Docket Nos. 50-400 OL
50-401 OL

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE TO APPLICANTS' MOTION FOR PARTIAL SUMMARY DISPOSITION OF JOINT CONTENTION VII" 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, through deposit in the Nuclear Regulatory Commission's internal mail system (1), or by express mail or overnight delivery(**) this 5th day of June, 1984.

James L. Kelley, Chairman*
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. Glenn O. Bright*
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. James H. Carpenter*
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Daniel F. Read
CHANGE/ELP
5707 Waycross Street
Raleigh, NC 27605

Atomic Safety and Licensing Appeal Board Panel*
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Richard D. Wilson, M.D.
729 Hunter Street
Apex, NC 27502

Travis Payne, Esq.**
723 W. Johnson Street
P.O. Box 12643
Raleigh, NC 27605

Dr. Linda Little
Governor's Waste Management Building
513 Albermarle Building
325 North Salisbury Street
Raleigh, NC 27611

John Runkle, Executive Coordinator
Conservation Counsel of North Carolina
307 Granville Rd.
Chapel Hill, NC 27514

Ruthanne G. Miller, Esq.*
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Bradley W. Jones, Esq.*
Regional Counsel, USNRC, Region II
101 Marietta St., N.W. Suite 2900
Atlanta, GA 27701

Robert P. Gruber
Executive Director
Public Staff - NCUC
P.O. Box 991
Raleigh, NC 27602

George Trowbridge, Esq.
Thomas A. Baxter, Esq.
John H. O'Neill, Jr., Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N.W.
Washington, DC 20036

Wells Eddleman**
718-A Iredell Street
Durham, NC 27701

Atomic Safety and Licensing Board Panel*
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Richard E. Jones, Esq.
Associate General Counsel
Carolina Power & Light Company
P.O. Box 1551
Raleigh, NC 27602

Janice E. Moore
Counsel for NRC Staff

NUREG-1014

EXHIBIT D

Safety Evaluation Report

related to the
D4/D5/E Steam Generator Design Modification

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

October 1983

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations,* and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $6.00

NUREG-1014

Safety Evaluation Report

related to the
D4/D5/E Steam Generator Design Modification

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

October 1983

ABSTRACT

This Safety Evaluation Report (SER) related to the D4/D5/E steam generator design modification has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The purpose of this SER is to issue the staff's evaluation of the acceptability of the design modification for both installation and full-power operation in D4/D5/E steam generators based on the Counterflow Steam Generator Owners Review Group's Technical Review Committee Report of July 1983. Those contributing to this report are listed in Appendix A.

CONTENTS

ABBREVIATIONS

CECO	Commonwealth Edison Company
CP&L	Carolina Power and Light Company
CSGORG	Counterflow Steam Generator Owners Review Group
DRP	Design Review Panel
ECT	eddy-current testing
FSAR	final safety analysis report
HL&P	Houston Lighting and Power Company
NRC	U.S. Nuclear Regulatory Commission
PSI	Public Service Company of Indiana
TRC	Technical Review Committee

SAFETY EVALUATION REPORT RELATED TO THE
D4/D5/E STEAM GENERATOR DESIGN MODIFICATION

1 INTRODUCTION AND SUMMARY

1.1 History

On October 21, 1981, a steam generator tube leak occurred at Ringhals Unit 3 (Varberg, Sweden), a three-loop Westinghouse plant with Model D3 steam generators, causing plant shutdown. From the resulting investigation, it was concluded that some type of accelerated wear mechanism involving interaction between the steam generator tubes and the tube support plates was occurring. Eddy-current testing (ECT) was performed on all three steam generators. The ECT results indicated that preferential wear was occurring in the outer three rows of tubes in the preheater section (Rows 47, 48, and 49).

Westinghouse established a task force to identify and correct the cause of the problem. To accomplish this objective, the task force gathered information relative to the problem, such as ECT data from operating plants, pulled tube data, tube vibration data, information from analytical models, and information from a series of air and water scale-model test facilities. Westinghouse determined that this type of accelerated tube wear is characteristic of only the preheat section of its Model D and E steam generators.

Several conceptual modifications to reduce the tube vibration and the resultant wear were developed by Westinghouse during the above investigations. From these, Westinghouse proposed to install an internal manifold in the Models D2 and D3 (D2/D3) steam generators. The staff reviewed the D2/D3 steam generator modification and concluded that it was acceptable (NUREG-0966).

The modification that Westinghouse proposed for the Model D4 and D5 steam generators consists of expanding the steam generator tubes and splitting some main feedwater flow through the auxiliary feedwater nozzle. For the Model E steam generators, only tube expansion is proposed. This report discusses the modification proposed for the D4, D5, and E (D4/D5/E) steam generators.

1.2 Technical Review Committee

During the first several months after Westinghouse identified the Model D steam generator problem, Nuclear Regulatory Commission (NRC) staff (the staff) worked with Westinghouse and utilities on an individual basis. In an effort to conserve NRC staff resources, the concept of a third-party design review of the proposed D2/D3 steam generator modification was initiated. This review lessened the need for a detailed technical review by the NRC, and the third-party report served as the basis for the NRC Safety Evaluation Report (NUREG-0966).

On February 4, 1983, a Counterflow Steam Generator Owners Review Group (CSGORG) was formed consisting of the following utilities:

Carolina Power & Light Company (CP&L)
Commonwealth Edison Company (CECO)
Houston Lighting & Power Company (HL&P)
Public Service Company of Indiana (PSI)
Belgian Utilities (Electronucleaire)
Nuklearna Elektrarna Krsko (Yugoslavia)

The intent of the CSGORG was to perform a review of the Westinghouse proposed D4/D5/E modification similar to the review done by the Design Review Panel (Tennessee Valley Authority, South Carolina Electric and Gas Company, and Duke Power Company) for the D2/D3 steam generator modification. The CSGORG established a Technical Review Committee (TRC) to perform this review. Table 1.1 lists all of the U.S. plants with D4/D5/E steam generators.

Table 1.1 U.S. plants with D4/D5/E steam generators

Utility	Plant	S/G model	No. of loops
Texas Utilities Generating Co.	Comanche Peak 1	D4	4
Texas Utilities Generating Co.	Comanche Peak 2	D5	4
Commonwealth Edison Co.	Byron 1	D4	4
Commonwealth Edison Co.	Byron 2	D5	4
Commonwealth Edison Co.	Braidwood 1	D4	4
Commonwealth Edison Co.	Braidwood 2	D5	4
Public Service Co. of Indiana	Marble Hill 1	D5	4
Public Service Co. of Indiana	Marble Hill 2	D4	4
Houston Lighting & Power Co.	South Texas 1	E2	4
Houston Lighting & Power Co.	South Texas 2	E3	4
Carolina Power & Light Co.	Shearon Harris 1	D4	3
Carolina Power & Light Co.	Shearon Harris 2	D4	3
Duke Power Co.	Catawba 2	D5	4

The TRC consisted of 13 members, with one member serving as chairman. The objective of the TRC was to determine the acceptability of the D4/D5/E modification selected by Westinghouse for installation in the D4/D5/E steam generators and then submit a report to the NRC with TRC's conclusions. The review performed by the TRC covered many different areas. These include thermal-hydraulics, model testing, radiological exposures, structural mechanics, stress analysis, inservice inspection, tooling, chemistry, and installation. By meeting with the TRC and by reviewing the qualifications of the TRC, the staff finds that the TRC was made up of a very competent and experienced group of engineers. From a review of the TRC report it is clearly recognizable that the TRC performed a very thorough review of the Westinghouse investigation into the problem and the proposed modification.

The TRC report of July 1983 (Butterfield letter, July 18, 1983) relies extensively on technical data summarized in the Westinghouse reports (Westinghouse, 1983a and 1983b); however, these data were made available to the TRC throughout the review process. Attached as Appendix B is a nonproprietary version of the

TRC report (July 18, 1983). The TRC has concluded that the modification to the D4/D5/E steam generators can be made, does not introduce any unresolved safety issues, and the modified steam generators can be operated safely at rated capacity.

1.3 Summary

The staff has reviewed the TRC report and finds the TRC report acceptable with some exceptions and comments in the areas of thermal hydraulics and inservice inspection and testing. This report discusses these exceptions and comments that, along with the TRC report (Appendix B), form the NRC staff's safety evaluation of the D4/D5/E steam generator modification. As a result of this review, the staff has established the following additional requirements:

(1) On a plant-specific basis, the minimum wall thickness and plugging criteria for the expanded regions shall be established in accordance with Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

(2) The staff requires that the basic document for the selection of tubes to be tested and for the frequency of testing shall be NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors." For the lead plant, the first inservice inspection of the steam generator tubes shall be conducted after 6 full-power months and before 12 full-power months of operation, including a special inspection of the expansion regions in all expanded tubes. For the remaining plants, the first inservice inspection requirements will be based on available data. Future inservice inspections on all plants shall always include the expanded regions in a sample of expanded tubes as a special subset of inspections.

The staff finds, pending plant-specific verification and documentation of safety analyses (Section 3.5 of this report and Section 2.5 of Appendix B) and implementation of the additional requirements noted above that the modification of the D4/D5/E steam generators is acceptable and that the modified steam generators can be operated at 100% of their design capacity.

2 DESCRIPTION OF MODIFICATION

An elevation view of a counterflow preheat steam generator (Model D4/D5) is shown in Figure 2.1). The preheater region is located on the cold-leg side of the tube bundle and faces the feedwater inlet. The lower shell internals including the preheater region are shown in Figure 2.2. Incoming feedwater enters the inlet waterbox and encounters the impingement plate which directs the water outward to fill the waterbox volume and downward to the preheater inlet pass located between B and D support plates. The water enters the tube bundle, then flows upwards around the tubes and baffles. It is in the outer rows of the tube bundle facing the incoming feedwater that the high tube vibration levels have been observed.

The Westinghouse proposed modification for Model D4/D5 plants consists of expanding approximately 124 tubes per steam generator at the B and D baffle plate locations and splitting feedwater flow by diverting a fraction of the main feedwater flow through the auxiliary feedwater nozzle. For Model D4/D5 two- and three-loop plants, approximately 18% flow diversion is required, and for Model D4/D5 four-loop plants, approximately 10% flow diversion is required. Typical feedwater configuration and flow distribution for a modified three-loop plant are shown in Appendix B, Figures 3.1-1 and 3.1-2, respectively.

The Westinghouse proposed modification for Model E plants consists of tube expansion at the B and D baffle plates with no feedwater flow split.

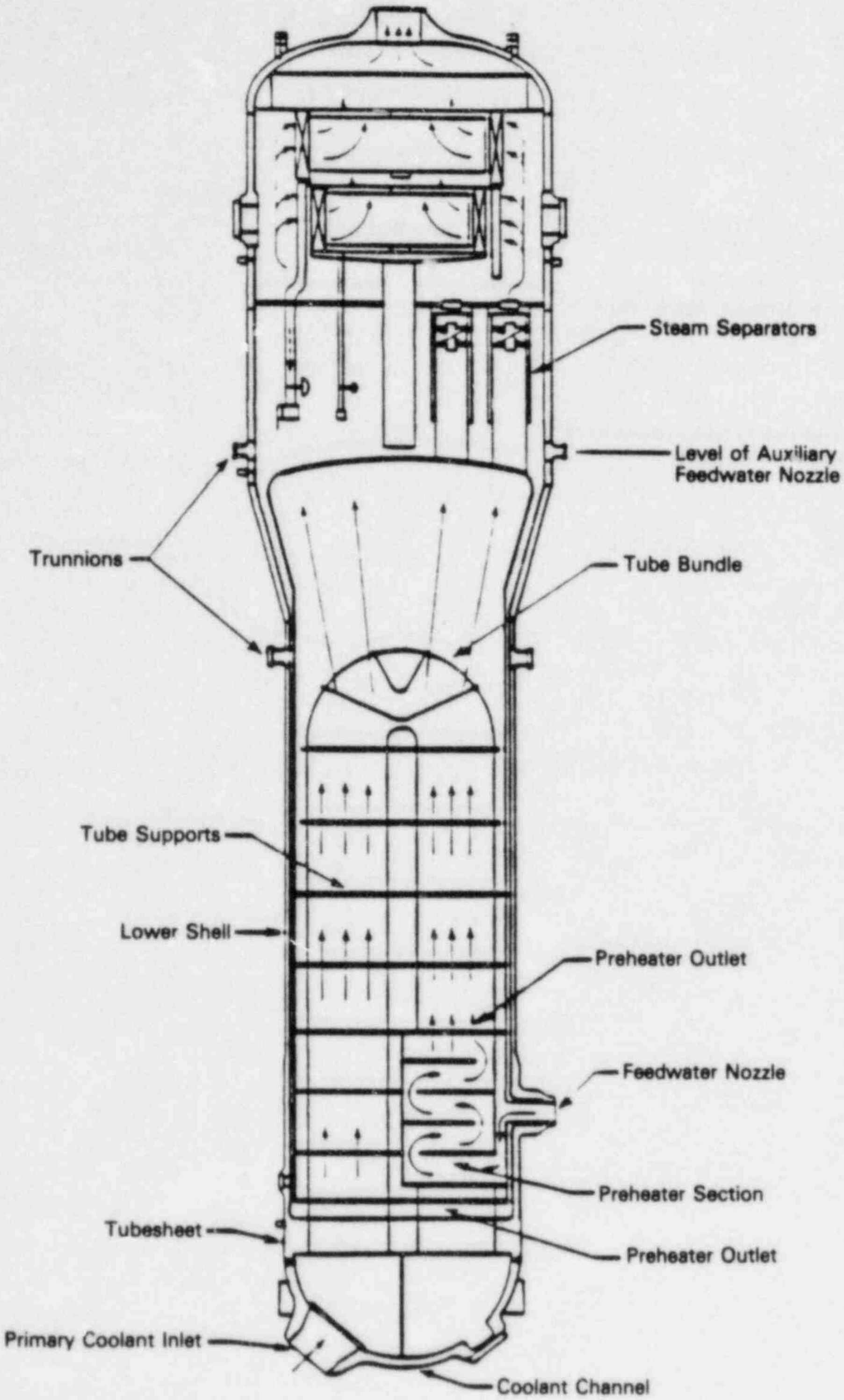


Figure 2.1 Counterflow preheat steam generator

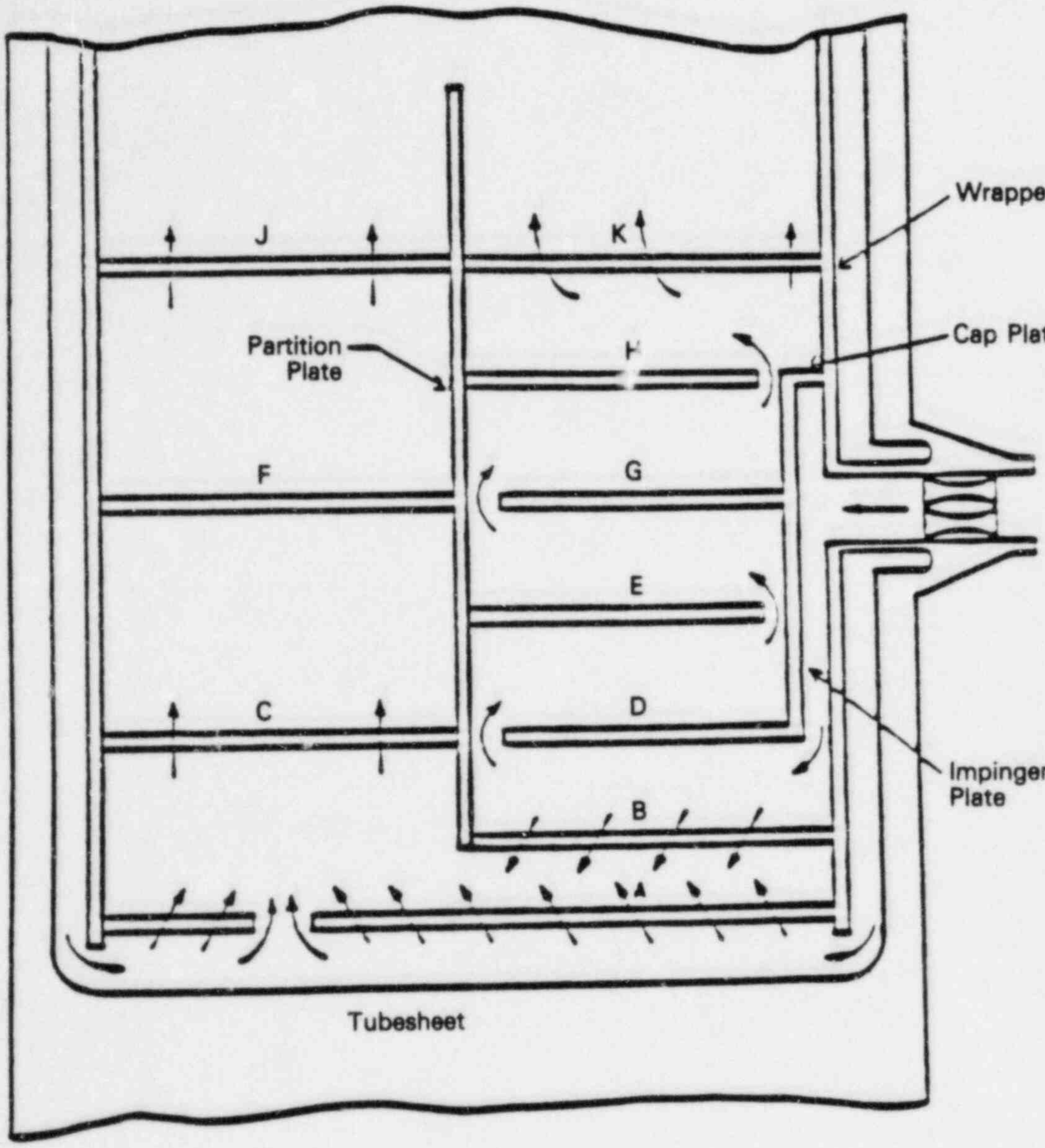


Figure 2.2 Counterflow preheat steam generator lower shell internals

3 EVALUATION

3.1 Thermal Hydraulics and Model Testing

3.1.1 Background

Wear, attributed to tube vibration, has been experienced in the preheat section of the Westinghouse Model D3 preheat steam generator. The wear was measured on tubes in steam generators from the Ringhals 3 nuclear plant in Sweden and the Almaraz 1 plant in Spain. Westinghouse assembled a task force to evaluate the problem and to develop and implement a design modification that eliminates detrimental tube wear. The Westinghouse Model D4, D5, and E steam generators also incorporate a preheat section. However, the baffling and the geometry of the feedwater inlet to the tube bundle is substantially different from the Model D3 unit. Also, there are differences in the tube bundle layout from Model D3. The major difference is the T-slot which is used in blowdown for sludge removal. There are also differences in the Model D4, D5, and E designs. Important differences are the larger size of the Model E and the fact that the first 5 rows of D4/D5 steam generator tubes are "window" tubes with no support at baffles E and H; in Model E, the tubes pass through all baffle plates.

The Krsko nuclear plant in Yugoslavia is the first Westinghouse plant with Model D4 steam generators to operate. As a result of the wear detected at Ringhals and Almaraz, Westinghouse carried out a test program at Krsko which involved the measurement of tube vibration on a select number of tubes using internally mounted tube accelerometers. In addition, three tubes were pulled and examined for wear scars at baffle plate locations.

Among other things, the Krsko data indicate impacting between tube and baffle plates at a dominant frequency starting at power levels between 65% and 85%, and root-mean-square displacements at a power level of 70% comparable to displacements measured at Almaraz at 50% power. Examination of the pulled tube showed wear at baffle plates B, D, and G; wall thinning of 6% of tube wall thickness was measured at baffle plate B.

The Westinghouse task force assembled to study tube vibration in the D4/D5/E steam generators consisted of many of the same members assigned to the D2/D3 study. Consequently, the team benefited greatly from what was learned in the evaluation of the D2/D3 problem and associated design modification development. In addition to field data from Krsko, Westinghouse performed extensive laboratory testing involving the following models:

- 0.95-scale air model
- single tube model
- 1/4-scale water model
- 2/3-scale water model
- 16° full-scale water model

The air model provided information on inlet and third pass flow velocities. The single tube model gave insights relating to tube dynamics. The 2/3-scale and 16° full-scale water models provided information on flow velocities, fluid forces, and tube vibration. Over the inlet pass, the flow distribution was found to be very nonuniform; some flow reversal occurred. Upper pass velocities were much lower than inlet pass velocities. Therefore, it was concluded that the dominant excitation source is flow over the inlet span.

As with the D2/D3 units, the low frequency response detected in the field data can only be explained by a tube vibrating within a baffle hole so that the particular baffle plate is ineffective insofar as providing support for that tube. In particular, the Krsko data can only be explained by an "inactive" baffle plate at D.

With regard to the potential to vibrate within a baffle plate hole, Westinghouse measured diametral gaps on 92 tubes at various baffle plates at Krsko. Similar measurements were taken at Comanche Peak and as-built measurements were taken on a unit fabricated at the Westinghouse plant in Tampa, Florida. Results of the measurement program showed that relatively large gaps exist. This increases the probability that under adverse conditions of flow and mechanical fit-up, determined in part by operating conditions, a tube can "float" within a baffle plate. However, no quantitative correlation between gap size and vibration level has been established.

The 16° full-scale model was designed to allow for lateral shifting of the baffle plates to account for misalignment associated with as-built tolerances, as well as differential thermal expansion or other effects related to operation. The occurrence of the dominant frequency is very sensitive to plate alignment. With appropriate plate shifting, Westinghouse was able to reasonably simulate the Krsko field data as determined from examination of frequency spectra. In addition, the simulation was shown to be repeatable. Subsequently, Westinghouse performed plate shifts to obtain the maximum vibration response at the dominant frequency for each tube in the window region.

On the basis of the results from the model tests, field data, and nonlinear analysis, Westinghouse concluded that the excitation mechanism responsible for vibration is primarily turbulence in the flow over the inlet span. Westinghouse further concluded that some fluidelastic instability contribution is likely. The staff evaluation and conclusions are discussed in the following section of this report.

3.1.2 Design Modification

In comparing Models D2/D3 with D4/D5/E, Westinghouse concluded, and the staff and its consultants have concurred, that the primary cause of the wear problem is the same in both cases, that is, relatively large tube-to-baffle-plate clearances (which make it possible for the tubes to vibrate within the clearance), coupled with relatively high inlet pass flow velocities that are nonuniform and turbulent. The TRC did not state the primary cause in their report. The design modification for the D2/D3 steam generators focused on the excitation source. The approach was to install a flow distributor which serves to both reduce turbulence levels and encourage uniform flow over the inlet area to the tube bundle. On the other hand, the design modification proposed for the D4/D5/E steam generators focuses on the vibrational characteristics of the tube, with flow splitting further employed in the D4/D5 to reduce velocities. The approach is to expand selected tubes at baffle plates B and D. As a consequence, the baffle plates will be more effective as supports, forcing vibration nodes at those points, and thereby increasing the vibration frequency. The split flow involves diverting a specified percentage (18% for 2- and 3-loop plants and 10% for 4-loop plants) of feedwater through the auxiliary feedwater nozzle.

A pressure-controlled expansion process is proposed to implement the tube expansion phase of the modification. Among other things, Westinghouse's process qualification tests have shown that:

(1) Expansion length and diametral gaps were consistently achieved over the design range of the process parameters.

(2) Expansions are consistently located within the baffle plate centerline, resulting in no tubal bulges occurring outside of the baffle plate surfaces.

3.1.3 Evaluation

3.1.3.1 Excitation Mechanism in the Preheat Region

As with the D2/D3 steam generators, Westinghouse considers the dominant excitation mechanism to be turbulent buffeting in the inlet pass. The TRC believes that the fluidelastic mechanism is probably present; however, it cannot be clearly distinguished from turbulence-induced vibration, because of amplitude limitations and the indeterminate nature of the support condition (see Appendix B, Section 5.2.3). The staff and its consultants at Argonne National Laboratory believe that the fluidelastic instability is a contributing factor. This is based on the following considerations: (1) there appears to be a threshold flowrate (corresponding to approximately 70% power level) above which relatively rapid wear occurs and below which wear is acceptable (the fluidelastic mechanism is a "threshold" type of mechanism, whereas turbulent buffeting is present at all flow rates and increases with approximately the square of the flow); (2) some of the responses show a relatively sharp well-defined peak indicative of fluidelastic instability; and (3) a reduced flow velocity is well within the range where instability can be expected, based on a stability diagram developed from idealized laboratory tests (Chen, 1982). It should be recognized that the instability is not of the classical type resulting from the changing support conditions and the nonlinearities inherent in such a situation.

3.1.3.2 Modification Concept

Regardless of the excitation mechanism, i.e., whether or not turbulent buffeting is the dominant mechanism, the TRC, the staff, and its consultants agree that the proposed modification consisting of tube expansion and flow splitting will prove effective in reducing the vibration. Tube expansion ensures support for the tubes at baffle plates B and D, thereby presenting a "stiffer system" to the incoming flow. The feedwater flow split results in lower flow velocities and hence less energy available to excite vibration. For example, the nondimensional flow velocity ($\bar{U} = U/fd$) used in assessing the potential for fluidelastic instability will be reduced by a factor greater than 2. With regard to turbulent buffeting, the system will be more difficult to excite (as a result of being stiffer), and there will at the same time be less energy to excite it. The staff believes that the modification directly addresses the problem.

3.1.3.3 Modification Performance

The effectiveness of the modification, particularly tube expansion, as a means to reduce tube vibration to acceptable levels was evaluated both in field tests at Krsko and in full-scale water tests in the 16° model. Westinghouse relies very heavily on the "G - Δ method" as a measure of effectiveness in reducing vibration and also, as will be discussed later, as a criterion for acceptability relative to wear.

The G - Δ method was developed by Westinghouse, under the Model D2/D3 investigative program, to relate tube midspan motion to wear experienced at the tube/baffle plate interface. The method (Westinghouse, 1983a) employs an "engineering type" equation of the same general form as Archard's equation for adhesive wear. In particular, the G - Δ factor is computed as the product of the maximum peak-to-peak acceleration and the root-mean-square displacement as obtained from accelerometers located midspan between baffles B and D or at the E baffle location for window tubes in the D4/D5 units. A series of tests at Krsko showed a reduction in G - Δ values by a factor of 5 to 15 when the tubes were expanded. Similar reductions in G - Δ were obtained on expanded tubes in the 16° model test.

3.1.3.4 Acceptability of the G - Δ Method

Westinghouse, the TRC, and the staff and its consultants all agree that the tube expansion at the baffle plates is effective in reducing vibration levels; however, a criterion for a vibration level that will result in acceptable wear had to be developed. The G - Δ method discussed above is the method used by Westinghouse. In equation form it is written as

$$V_T = K_g \, (G - \Delta) \, T$$

where V_T is total wear volume, K_g is the wear coefficient, and T is the time period over which the wear is occurring. The wear coefficient must be determined from experimental data. Westinghouse determined a wear coefficient from examination of three pulled tubes from Krsko. Based on this, values of G - Δ were established that result in an acceptable level of vibration for long-term operation.

The wear process is extremely complex and it is difficult to relate tube vibration (especially motion measured midspan) to wear occurring at a baffle plate. In view of these inherent difficulties, the G - Δ method represents a reasonable engineering approach. There are, however, some uncertainties associated with the use of this method; Westinghouse recognizes them, as does the TRC which calls attention to some of the limitations (Appendix B, Section 5.3). Some additional concerns identified by the staff and its consultants are listed below.

- Application of the method is somewhat subjective as it relies on visual inspection of an acceleration time history to determine the maximum peak-to-peak acceleration.

- The acceleration signals are low-pass-filtered to give response in a frequency range. The upper frequency limit of the range may not be high enough to include impact-generated frequencies.

- The wear coefficient employed in the method is based on very limited data - only three pulled tubes from Krsko.

- When a particular G - Δ value is established as an acceptance criterion, it must be related to a particular measurement station.

- The value of G - Δ for vibration of a particular tube will be dependent on the measurement location. The test data bear this out. For example, at 70% power tube R45C56 from Krsko Phase I data has a lower G - Δ value measured at the B-D midspan than that measured at baffle E elevation. It should further be noted that G - Δ values do not change proportionally with power level. For tube R45C56, referred to above, the 100% power G - Δ values, compared to the 70% G - Δ values, are increased approximately 12 and 3 times at measurement stations B-D midspan and elevation E, respectively.

- The acceptance criterion for Model D4 is based on the Model D2/D3 G - Δ values at 50% power before modification. However, one has to exercise care when comparing G - Δ values from different designs; for example, comparing data from Model D3 with those from Model D4. Westinghouse does increase the D2/D3 measured G - Δ values to account for a difference in the wear coefficients between D2/D3 and D4. Nevertheless, there may be other factors involved.

- Impacting of adjacent tubes may be "picked up" by an accelerometer located in the tube of interest. As a result, the G - Δ value could be different from what it would be without the interaction. In fact, Westinghouse uses this to explain why the first set of data from expanded tubes in Krsko (Test TE-1) shows a modest reduction in vibration levels, whereas the second set (Test TE-2) shows the substantial reduction expected. In the first case it is speculated that impacting of neighboring tubes was picked up. This illustrates an inherent difficulty in application of the method and further stresses the need for care in interpreting results.

The TRC concluded that the G - Δ method is acceptable for predicting tube wear and for assessing the effects of the steam generator modification. The staff and its consultants have determined that the cumulative uncertainties, including those discussed above in wear prediction based on the G - Δ method, do not result in a safety issue because, after modification, the wear rate is expected to be reduced to a relatively low value. The G - Δ method is considered reliable enough to predict order of magnitude wear. Any remaining uncertainty associated with the G - Δ method can be determined from the postmodification monitoring.

The vibration and wear are extremely complex phenomena. Other analytical methods could have been used to predict wear; however, the staff and its consultants are of the opinion that had other analytical techniques been used, the conclusion that the modification is acceptable would be the same.

3.1.3.5 Selection of Tubes To Be Expanded

Field data from Krsko together with results from the 16° model test have demonstrated that tube expansion at baffle plates B and D leads to a significant reduction in vibration level. Further reduction, as necessary, is accomplished by reducing the main feedwater flow by a flow split with the auxiliary feedwater. The acceptance criterion regarding G - Δ values was established and is used in selecting tubes to be expanded. The data base for selection of tubes to be expanded is shown in Figure 2.7-1 of the Westinghouse report (1983b).

Here the concern is the need to rely on the 2/3-scale model data and nonlinear model tube vibration analyses to predict the G - Δ values for tubes outside the bounds of the 16° model. The TRC also recognizes the uncertainty and discusses it in some detail (Section 5.3.2.1, Appendix B). However, Westinghouse's approach is reasonable and the TRC concludes and the staff concurs that it provides an acceptable solution.

3.1.3.6 Structural Evaluation of the Expanded Tubes

Tubes which are expanded at the support plate were evaluated for design, and for normal, upset, emergency, and faulted conditions in accordance with the requirements of the ASME Code, Section III, Subsection NB.

Primary stress levels were evaluated against design allowables and fatigue usage factors were generated. The tube loads considered effects of tube wall temperature and pressure differential, tube/baffle-plate lateral mismatch, axial interaction loads due to tube/baffle-plate interference, and tube axial temperature gradient. The Westinghouse analysis was performed using finite element techniques (see Appendix B, Section 5.6.2.2). Also a fatigue analysis was performed on the expanded portion of the tube and tubesheet locations (see Appendix B, Section 5.6.2.2). Analyses were also performed to simulate a possible locked tube condition at support locations B and D. Westinghouse concludes, and the TRC and staff concur, that the results of these analyses show acceptable fatigue usage factors of less than 1.0.

3.1.3.7 Evaluation of Feedwater Split Modification

The major effect of the feedwater bypass is lowering the heat transfer film coefficient in the main feedwater and increasing it in the auxiliary feedwater nozzle (see Appendix B, Section 5.6.2.4). The conclusions reached from the Westinghouse analysis of the feedwater split are that ASME design Code allowables are met for operational transients and that the fatigue usage factor for both nozzles is less than 1.0. Westinghouse concludes, and the TRC and staff concur, that the effect of split flow on primary and secondary stress and fatigue usage for the central drain, intermediate plate, and auxiliary nozzle discharge pipe is insignificant. Also, the fatigue usage contribution from thermal striping on the upper internals components is negligible for conditions modified by split flow.

3.1.3.8 Evaluation of Flow Excitation Mechanisms in the Tube Bundle

The areas of the tube bundle other than the preheater were analyzed by Westinghouse to determine flow-induced vibrations caused by parallel and cross flows.

These analyses indicate that the cross-flow velocities are sufficiently low so that they result in negligible fatigue and vibratory amplitudes. The support system is therefore deemed adequate [illegible] with regard to parallel flow excitation. To evaluate cross flow at the exit of the downcomer flow to the tube bundle and at the top of the bundle in the U-bend area, Westinghouse established an experimental research program investigating cross flow in tube arrays, given the specific parameters of the steam generator. Air and water model tests were employed. The results of this investigation indicate that these regions of the bundle are not subject to the vortex shedding mechanism of tube excitation. Vortex shedding was found not to be a significant mechanism in these two regions because flow turbulence in the downcomer and because the tube bundle inlet region inhibits the formation of vortices. In addition, the axial flow component disrupts the vortices. This research program also formed the basis for evaluating the fluid elastic mechanism due to cross flow at the tubesheet. The Westinghouse evaluation showed the adequacy of the tube support arrangement. Flow turbulence can result in some excitation in these regions. The staff finds this excitation of little concern, however, because the maximum stresses in the tubes are at least an order of magnitude below the fatigue endurance limit of the tube material. The TRC did not address these analyses in their report.

3.1.3.9 Evaluation of the Tube Plugging Criteria

Westinghouse has provided an assessment of the tube wall thinning that can be tolerated under accident conditions. The results of a study made on "D series" tubes under accident loading show that a minimum wall thickness of 0.026 inches would have a specified maximum faulted condition stress (i.e., due to combined LOCA and safe shutdown earthquake loads) that is less than the ASME Code-allowable Level D limit. Tubing of 0.043-inch nominal wall thickness and 0.039-inch minimum wall would have an available tube wall thickness of 0.013 inch to provide margin for uncertainties, general erosion, and corrosion loss. The TRC concluded that the calculated minimum wall thickness requirement satisfied external collapse pressure, burst strength, and leak before break criteria (see Appendix B, Sections 5.6.2.3, 5.6.2.4, and 5.6.2.5).

The staff finds that the requirements of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," which specify a factor of safety of three under normal operating conditions, have not been addressed by Westinghouse or the TRC. Therefore, on a plant-specific basis, the staff will require that the minimum wall thickness and plugging criteria for the expanded regions be established in accordance with Regulatory Guide 1.121.

3.1.3.10 Postmodification Vibration Monitoring

The TRC states (Appendix B, Section 4.3.1), and the staff concurs, that the purpose of the postmodification monitoring instrumentation is to obtain vibration levels for comparison with the objectives of the modification. Values exceeding the modification objectives will be reviewed. A vibration-monitoring plan proposed by Westinghouse calls for the first modified Model D4/D5 plant and the first modified Model E plant to be instrumented. The first Model D4/D5 plant has Model D4 steam generators; it is Comanche Peak Unit 1. The first Model E plant is Doel 4 in Belgium.

The vibration monitoring plan calls for accelerometers to be installed in four tubes of one steam generator for vibration measurements prior to extended operation. The accelerometers will be selectively located to provide an adequate representation of the expanded and unexpanded tube population in the region most susceptible to vibration and wear.

Westinghouse has recommended that accelerometers be installed at the following locations at Comanche Peak Unit 1.

Tube no.	Number and location of accelerometer
R49C59*	One between plates B and D; one at plate E
R49C41*	One between plates B and D; one at plate E
R48C41**	One between plates B and D; one at plate E
R48C33**	One at plate E

*Expanded tubes.
**Unexpanded tubes.

The TRC made no conclusion regarding the postmodification vibration monitoring program. The staff and its consultants consider these representative choices and concur with the Comanche Peak Unit 1 vibration monitoring program. For the remaining U.S. Plants with D4/D5/E steam generators, the staff will decide if vibration monitoring is needed, based on available data.

3.1.4 Summary and Conclusions

The use of pressure-controlled expansion of selected tubes in the preheater region in conjunction with bypass of the main feedwater flow via the auxiliary nozzle has been evaluated by Westinghouse through a comprehensive test and qualification program. An extensive evaluation of these two modifications by the TRC, the staff, and its consultants indicates that the tube vibration and consequent wear will be reduced to an acceptable level as a result of these modifications.

The expanded tube configuration was evaluated for design, normal, upset, emergency, and faulted conditions in accordance with ASME Code, Section III, Subsection NB requirements. As a result of its review, the TRC concludes and the staff concurs that the maximum calculated usage factors for the locally expanded region of the tube are all less than the ASME Code-allowable factor of 1.0. The primary stress levels in the expanded regions were also found to be within the ASME Code allowables. The structural evaluation of the feedwater split modification indicates that ASME design Code allowables are met for operational transients and that the fatigue usage factors for both the nozzles is less than 1.0.

Although the staff, its consultants, and TRC have previously discussed reservations about the G - Δ method for tube wear prediction and, in particular, its extrapolation to tubes where experimental data at full scale are not available, the TRC and the staff consider the approach to be reasonable, especially considering the conservatisms that are incorporated into the method.

The staff and its consultants concur with the TRC conclusion that any tube wear resulting from tube vibration is expected to be limited to a small number of tubes and its progress is expected to be within acceptable limits. This allows use of a periodic tube inservice inspection program for detection and followup of tube wear. In addition to the inservice inspection program, a vibration-monitoring program has been recommended by Westinghouse for selected tubes in the preheat region. The staff concurs with this proposed vibration-monitoring program for the lead plant.

The testing focused on Models D4 and D5. As identified by the TRC, some additional work remains relative to the Model E units. However, with regard to the Model E steam generator, Westinghouse correctly points out that the design parameters of the Model E units are more favorable relative to vibration than the equivalent parameters of Models D4 and D5.

3.2 Tooling

A pressure-controlled expansion process will be used to expand hydraulically the tubes at two baffle locations. A combination of eddy-current and expansion mandrel probe was developed for both locating the baffle plate centerline and producing the pressure-controlled hydraulic expansion.

Two circumferentially wound eddy-current coils are located along the lower part of this probe. The coils will be utilized in a single-channel, multi-frequency, eddy-current system in the differential mode for location of the support plate.

The staff concludes that the pressure-controlled expansion process and associated tooling have been satisfactorily evaluated through testing and qualification and have resulted in tube expansion within specified tolerances. Tubes not meeting tube expansion criteria within the specified tolerances shall be dispositioned on a tube-by-tube basis.

The staff did not review the radiological considerations involved with performing this modification after a plant has operated (Appendix B, Section 4.4), because it is expected that all U.S. plants will perform this modification before fuel load. A plant-specific evaluation of radiological considerations will be performed for any U.S. plant that does not perform the modification before fuel load.

3.3 Inservice Inspection and Testing

Post-expansion quality control inspection of the expanded zone of the tubes is performed using the expansion mandrel and the eddy-current probe. The two circumferentially wound coils are used in an absolute mode to verify the location of the expansion zone within the baffle plate.

A post-expansion, eddy-current examination will also be conducted to obtain a baseline signature for comparison analysis with future inservice inspection eddy-current readings. The standard bobbin eddy-current coil will be used for these examinations.

Several new eddy-current probe systems are being developed to have improved sensitivity for detecting defects in the transition regions of the expanded tubes. One such probe uses eight surface-riding pancake coils located at 45-degree increments around the inner surface. Each of the coils has an eddy-current field that overlaps the area covered by the adjacent coils, so that the entire circumference of the tube is inspected. This or another new eddy-current probe system will be used to obtain supplemental information during future in-service inspections if necessary.

The staff concludes that the proposed eddy-current techniques to be used for post-expansion, quality control inspection and post-expansion, baseline inspection are acceptable.

The TRC states that the eddy-current testing should comply with the procedures established by the ASME Code, Section XI, "In-Service Inspection," and Regulatory Guide 1.83, "In-Service Inspection of Pressurized Water Reactor Steam Generator Tubes," for the selection of tubes to be tested and for the frequency of testing. The staff does not agree and will require that the basic document for the selection of tubes to be tested and for the frequency of testing shall be NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors." Furthermore, the staff will require for the lead plant that the first inservice examination, in accordance with NUREG-0452, shall be conducted after 6 full-power months and before 12 full-power months of operation, including a special inspection of the expansion regions in all expanded tubes. For the remaining plants, the first inservice inspection requirements will be based on available data. Future inservice inspections shall always include the expanded regions in a sample of expanded tubes as a special subset of inspections.

3.4 Chemistry

Various corrosion phenomena aspects attributed to the steam generator modifications are discussed in Section 5.4 of the TRC report (Appendix B). As a result of its review of both the Westinghouse modifications reports (Westinghouse, 1983a and 1983b) and the TRC evaluation, the staff has concluded that the susceptibility to denting, susceptibility to stress corrosion cracking, and the propensity for wastage have been adequately addressed and tested and are acceptable.

The TRC report also discusses the possibility of pitting and fretting of the components because of the modification. The staff agrees with TRC's conclusion that these mechanisms will not be significantly changed from what existed before the modification.

3.5 Safety Analysis Requirements

The TRC recognized that each facility is unique in its design and safety analyses (Appendix B, Section 2.5). The TRC therefore examined, on a generic basis, the implementation of the steam generator modification (primarily split feedwater flow) for impact on areas of the final safety analysis report (FSAR).

The TRC concluded that only minor revisions to FSARs will be required, that the impact of the modification on safety analyses is believed to be within the tolerances of the available plant margins, and that this will be verified and documented on a plant-specific basis, or plant-specific parameters will be adjusted.

The staff has reviewed and concurs with the TRC conclusions including the course of action to verify and document on a plant-specific FSAR basis.

4 BIBLIOGRAPHY

American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section XI, "In-Service Inspection."

Butterfield, Jr. L. D. (CSGORG), letter dated July 18, 1983, to D. G. Eisenhut (NRC), subject: "Independent Evaluation of Proposed Modifications to Westinghouse D4, D5, and E Steam Generators" (proprietary).

Chen, S. S., "Tube Instability Flow Velocity of Tube Arrays in Crossflow," Paper F2, Int'l Conf. on Flow Induced Vibrations in Fluid Engineering, Reading, England, September 14-16, 1982.

Technical Review Committee, Counterflow Steam Generators Owners Review Group, "Independent Evaluation of Proposed Modifications to Westinghouse D4, D5, and E Steam Generators," CSGORG-001 (see Butterfield letter dated July 18, 1983 (above) and Appendix B (this report)).

U.S. Nuclear Regulatory Commission, NUREG-0452, "Standard Technical Specifications (STS) for Westinghouse Pressurized Water Reactors," September 1980.

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981 (formerly NUREG-75/087).

---, NUREG-0966, "Safety Evaluation Report related to the D2/D3 Steam Generator Design Modification," March 1983.

---, Regulatory Guide 1.83, "In-Service Inspection of Pressurized Water Reactor Steam Generator Tubes," July 1975.

---, Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976.

Westinghouse Electric Corporation, "Counterflow Preheat Steam Generator Tube Expansion," Westinghouse Report (proprietary), SGPR-8301, June 1983.

---, "Counterflow Preheat Steam Generator Tube Vibration," Westinghouse Summary Report (proprietary), SGPR-8301, June 1983b.

APPENDIX A

PRINCIPAL CONTRIBUTORS

This Safety Evaluation Report is a product of the NRC staff and a consultant The following were the principal contributors of this report.

Name	Title
T. L. Chan	Mechanical Engineer
L. Frank	Senior Materials Engineer
J. B. Hopkins	Project Manager
B. Mann	Nuclear Engineer
J. R. Rajan	Mechanical Engineer
B. Turovlin	Chemical Engineer
M. W. Wambsganss	Consultant (Argonne National Laboratory)

CSGORG-002

INDEPENDENT EVALUATION OF PROPOSED MODIFICATIONS TO WESTINGHOUSE D4, D5, AND E STEAM GENERATORS

Prepared by the
Counterflow Steam Generator Owners
Review Group - Models D4, D5, and E

MEMBERS

Carolina Power & Light Company
Commonwealth Edison Company
Houston Lighting & Power Company
Public Service Company of Indiana
Belgian Utilities (Electronucleaire)
Nuklearna Elektrarna Krsko

JULY 29, 1983

NOTICE

This report was prepared as an account of work performed by the Technical Review Committee of the Counterflow Steam Generator Owners Review Group. The Counterflow Steam Generator Owners Review Group assumes no legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

ACKNOWLEDGMENT

This report was edited and printed by Torrey Pines Technology under contract to Commonwealth Edison Company.

PREFACE

This is the report of the Technical Review Committee (TRC) of the Counterflow Steam Generator Owners Review Group on Westinghouse Electric Corporation's proposed modification of their Model D4, D5, and E steam generators. The Counterflow Steam Generator Owners Review Group is an independent technical committee that includes representatives from utilities and consulting engineering firms having interests both in the United States and in Europe. The purpose of this report is to issue the group's evaluation of the problem definition and the acceptability of proposed solutions to problems related to the implementation of design modifications and full-power operation of the Model D4, D5, and E steam generators.

The TRC has reviewed the June 1983 Counterflow Preheat Steam Generator Tube Vibration Summary Report and the June 1983 Counterflow Preheat Steam Generator Tube Expansion Report provided by Westinghouse. The TRC supplemented its review with the documents listed in the bibliography of the present report. In addition, the TRC actively participated in meetings with Westinghouse, exchanged data with Westinghouse, and made independent studies. All the contents reflect direct technical information exchange between the TRC and Westinghouse.

The purpose of this Owners Review Group was to complete a design review of the final proposed modification program for counterflow steam generators as proposed by Westinghouse. This modification program is relative to the limitation of major vibration and wear within the preheater section of the subject steam generators. The review guidelines address the following areas to determine if:

1. The proposed modifications will limit tube vibration.

2. The proposed modifications could impact licensing of the plants.

3. The proposed modifications can be implemented without significant impact on long-term plant operations.

4. Westinghouse acceptance criteria are adequate and whether the proposed modifications meet the acceptance criteria.

5. The proposed modifications can be implemented without impacting plant completion schedules.

6. Any other areas should be addressed during the course of the review.

The Owners Review Group addressed technical areas related to the following disciplines:

1. Thermal-hydraulics.
2. Vibration analysis.
3. Structural design and analysis.
4. Feedwater system analysis.
5. Metallurgy/welding.
6. Stress corrosion/chemistry.
7. Post-modification monitoring/installation.
8. Field modification.

EXPLANATORY NOTE

Westinghouse Electric Corporation proprietary information contained in the original report has been deleted. In most instances where deletions caused discontinuity of the text, sections were slightly reworded to maintain clarity and coherence of this report. In other cases, where excessive rewording would have been required, deletions were bracketed and lower-case alphabetical code letters, outside the brackets, were used to indicate the criteria or basis upon which information was determined to be proprietary. The letters used for coding the brackets are as follows:

a. Information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

b. Information consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

c. Information which if used by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

d. Not used.

e. Information reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

f. Information contains patentable ideas, for which patent protection may be desirable.

SUMMARY AND CONCLUSIONS

The design modifications developed by Westinghouse for the preheater section of Model D4, D5, and E steam generators provide a substantial reduction in tube vibration. As a result, the potential for tube wear has been greatly decreased.

The extent of the modifications depends on the steam generator design. In the Model D4 and D5 steam generators, the modifications consist of expanding the tubes into the baffle plates, located above and below the first pass feedwater inlet (B and D plates), and splitting the feedwater flow by diverting a specified percentage of the main feedwater flow through the auxiliary feedwater nozzle. For Model E steam generators, only tube expansion is proposed.

The tube expansion provides an area of close contact between the tube and the baffle plate. This close support condition significantly changes the response frequency of the tube and also the G-Delta value. The split feedwater flow reduces the mass flow and velocity of the fluid in the preheater section. Both modifications combine to provide a substantial improvement by reducing the potential for tube wear. The design modifications and their consequences for steam generator and plant performance were reviewed extensively by the TRC concurrent with the normal Westinghouse design review process. The TRC review is documented in Sections 1 through 5 of this report.

Westinghouse performed a safety evaluation of the preheater and feedwater flow (where applicable) modifications. The conclusion from this evaluation is that modification of the Model D4, D5, and E steam generators does not represent an unreviewed safety question. The TRC concurs with this conclusion. The TRC's evaluation of Westinghouse's compliance of Model D4, D5, and E steam generator modifications with their own design objectives and the TRC's design criteria is summarized in Table 1.

The five major areas of interest to the TRC were (1) effectiveness of the modifications in reducing tube vibration and wear, (2) assurance that the tube expansion would not induce any large residual stresses or defects sufficient to cause early tube leaks, (3) assurance that the modifications do not produce conditions within the preheater that accelerate tube corrosion, (4) assurance that nondestructive examination (NDE) is capable of detecting wear and/or cracks at the tube expansion regions, and (5) assurance that leak before break criteria are met. The TRC concluded that the proposed modifications assure substantial improvement by reducing the potential for tube wear. This conclusion was reached after a thorough review of test models and testing results, as well as evaluation of analytical models and analysis results.

The TRC recognizes that in certain tubes the tube expansion conditions (i.e., baffle plate hole size and tube fit) may lead to local strains in the tubes. The larger strain levels are associated with tube expansions greater than 30 mils diametral. However, the TRC judges the primary water temperatures to be low enough in the cold leg to reduce the potential for stress corrosion cracking of the expanded tube portion relative to that which exists for the tubes in the hot leg of the steam generator. The strain levels of the tube expansion in the baffle plate are no larger than strain levels of similar expansions at the tube sheet which are currently operating successfully in all Westinghouse steam generators. Stress corrosion cracking in the cold leg tube sheet expansions has not been a problem in operating plants. The TRC considers the number of tubes subject to large expansions to be relatively small compared with the total number of tubes being expanded. Some tube wear may occur over the projected 40-year life of the steam generators. Also, some tubes in the expansion area may require plugging during the steam generator lifetime. This is judged to be acceptable since proper equipment for tube inspection and remedial action, as required, is available.

The TRC performed a detailed review of the stress analysis, which included examination and evaluation of analytical methods and models. The TRC concluded that these modifications (tube expansion and split feedwater flow) meet the established structural design criteria. The TRC's review of the safety analysis led to the conclusion that the expanded tubes will meet the "leak before break" criteria and that minimum wall thickness requirements are changed by only an acceptably small amount. Thus, normal periodic tube inspection and/or prompt remedial actions, whenever required, will prevent the potential for developing any multiple tube fractures.

The TRC considers that the corrosion potential of the expanded tubes is not significantly changed from that of the nonexpanded tubes.

Operations with split feedwater flow, i.e., diverting part of the feedwater through the auxiliary feedwater nozzle, will be required for Model D4 and D5 steam generators. In considering the functional feasibility of split feedwater flow, the TRC has reviewed the steam generator stress analysis and has also examined and evaluated the analytical methods and models. The TRC concludes that steam generator operation with split feedwater flow is feasible and acceptable. The TRC is cognizant of the fact that virtually every facility having Model D4 and D5 steam generators has unique feedwater system designs. For this reason, the TRC has not evaluated all feedwater piping designs but only the generic effect of split feedwater flow on the steam generators. It is expected that the individual utilities considering split feedwater flow will have to prepare feedwater system modifications, operating procedures, etc., on a site-specific basis.

The TRC has reviewed the tooling, procedures, and NDE requirements for the tube expansion process. The TRC concludes that Westinghouse has addressed each of these areas and that adequate consideration has been given for as low as reasonably achievable (ALARA) principles in the proposed modification methods, although there is only one operating unit with counterflow steam generators. Chemistry and cleanliness control during the modification were reviewed, and it was concluded that both areas have been sufficiently considered.

Quality assurance measures relative to design, testing, analysis, manufacturing, and installation activities were reviewed. It was concluded that these measures are sufficient to assure the adequacy of the modifications.

The monitoring and testing program recommended by Westinghouse for the lead plants was reviewed. Eddy current examination of the tubing using the industry standard bobbin probe will be employed to obtain baseline signatures after expansion is completed. Westinghouse will continue development of new eddy current inspection techniques for quantifying changes observed in the baseline signatures should they occur.

Based upon its review, the TRC concluded that (1) the proposed modifications to the Model D4, D5, and E steam generators can be made, (2) they do not introduce any unresolved safety concerns, and (3) the corresponding units can be operated safely at rated capacity.

Because geometry differences between Model D and E steam generators are not fully reflected in certain of the test models, it is the TRC's position that the TRC will continue to evaluate Westinghouse's ongoing test and analysis program related to the Model E to confirm the number of tubes to be expanded, the required expansion location, analytical long-term wear prediction, and the post-modification monitoring program. Any reliability issues resulting from this effort will be resolved between Westinghouse and the Owners.

TABLE 1
SUMMARY TRC EVALUATION

TRC Criteria and Westinghouse Objectives	TRC Evaluation	Report Reference
TRC Criteria		
1. Limit vibrational levels so that maximum wear scar depths, conservatively predicted, will be less than 65% of the wall thickness (structural integrity limit) for a time interval of 18 equivalent full-power months.	Criterion met	Section 5.3.2.3/p. 5-32
2. Effects of tube expansion process shall be evaluated using the ASME Boiler and Pressure Vessel Code, Section III.	Criterion met	Section 5.6/p. 5-40
3. The modification shall not cause unacceptable effects for required reactor thermal margins or any other plant operational safety parameters.	Criterion met on a generic basis	Section 5.5/p. 5-39
Westinghouse Objectives		
1. G-Delta is a measure of the magnitude of the vibration and is related to the wear-producing capability of the vibration. The value of G-Delta that Westinghouse considers appropriate to indicate acceptable levels of vibration is [ ]a,b,c,e or less for long-term operation.	Objective met	Section 5.3.2.3/p. 5-32
2. Minimization of potential for fluidelastic instabilities.	No effect on safety	Section 5.3.2.1/p. 5-27
3. Predicted values of 40% wall reduction for the design basis case and 65% wall reduction for the safety case are used as guidelines.	Objective met	Section 5.3.2.3/p. 5-32
4. Effects of tube expansion process shall be evaluated using the ASME Boiler and Pressure Vessel Code, Section III.	Objective met	Section 5.6/p. 5-40
5. The modification shall not cause unacceptable effects to required reactor thermal margins or any other plant operational safety parameters.	Objective met on a generic basis	Section 5.5/p. 5-39

CONTENTS

FIGURES

FIGURES (Continued)

TABLES

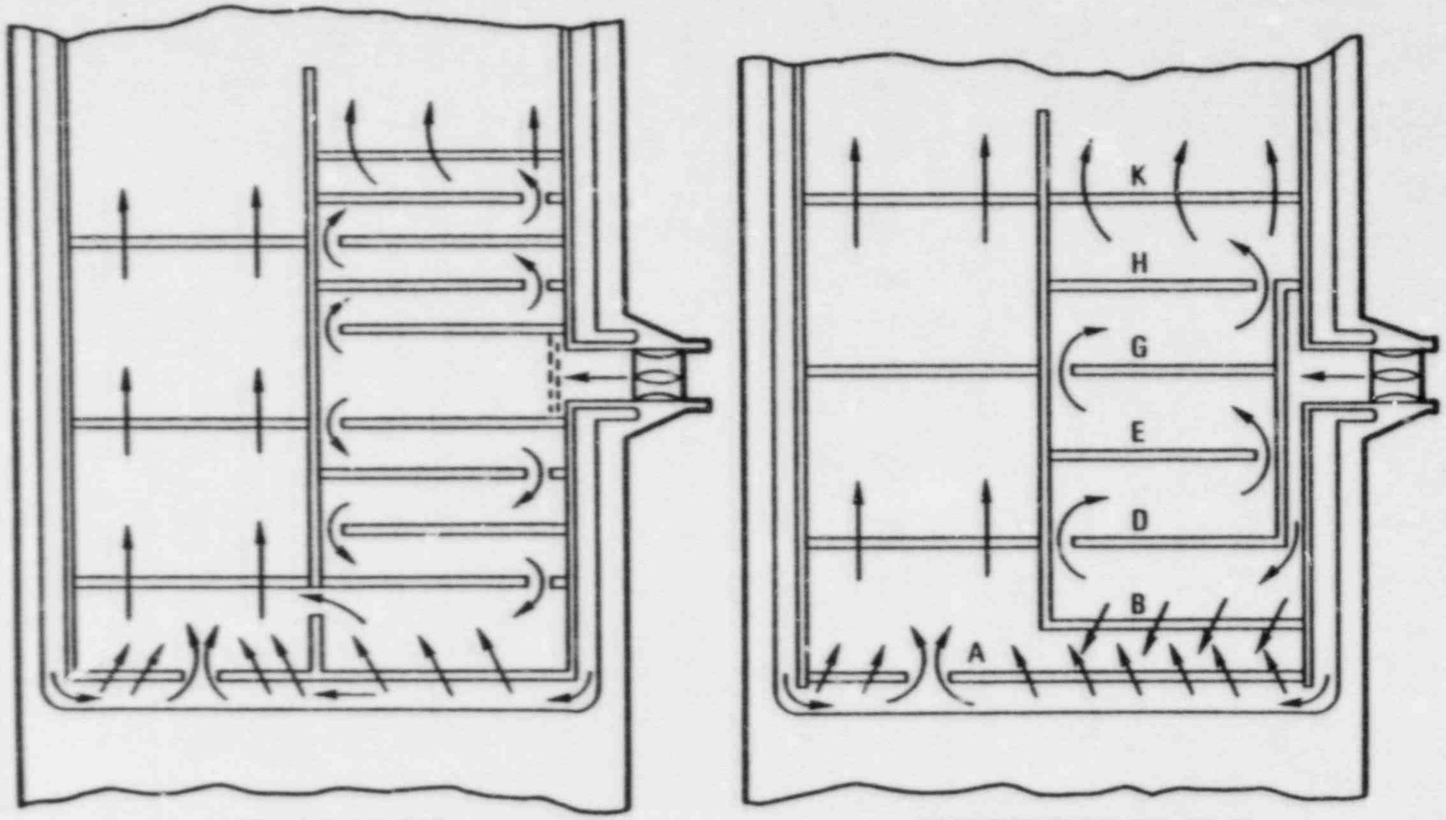
1. INTRODUCTION

1.1. HISTORY AND NEED FOR PREHEATER MODIFICATION

On October 21, 1981, the Ringhals Unit 3, a three-loop Westinghouse nuclear plant with preheater type Model D3 split flow steam generators, was shut down due to a steam generator tube leak of 2.5 gpm. Investigation revealed that tube R49 C55 (cold leg) had a small through-wall hole at baffle plate 3. The unit had operated for about 2200 hours at power levels above 75% at the time of the leak. It was apparent that some type of accelerated wear mechanism involving interaction between the baffle plates and tubes was occurring. Eddy current testing performed on all three steam generators at Ringhals 3 in Sweden and at Almaraz 1 in Spain indicated that preferential wear was occurring in the outer three rows of tubes in the preheater section (rows 47, 48, and 49). Several tubes were removed from the affected steam generator to better characterize the wear phenomena.

The Krsko nuclear plant is a two-loop Westinghouse plant containing two preheater type Model D4 counterflow steam generators. Both the D3 and D4 steam generators contain a preheat section where the feedwater enters the steam generator. However, the inlet flow geometries of the D3 and D4 designs are different, as shown in Fig. 1.1-1. After wear on tubes in the D3 steam generators was detected at Ringhals and Almaraz, Westinghouse proposed a test program including measurements of tube vibrations at Krsko. This program was executed in three phases from February through July 1982.

During the first phase at Krsko, two accelerometers were installed in each of four tubes of one steam generator with the axial location of the accelerometers at different elevations. The tubes instrumented were R49 C56, R48 C55, R46 C56, and R45 C56. The vibration magnitudes of these tubes were measured in February 1982 with full feed flow through the main nozzle at 100% power. The measured data were reduced and root mean square (RMS)



SPLIT FLOW (D3)

McGUIRE 1
ALMARAZ 1
ALMARAZ 2
ANGRA 1
ASCO 1
ASCO 2
CATAWBA 1
LEMONIZ 1
LEMONIZ 2

McGUIRE 2
RINGHALS 3
RINGHALS 4
VC SUMMER
WATTS BAR 1
WATTS BAR 2

COUNTER FLOW (D4, D5, E)

D4

BRAIDWOOD 1
BYRON 1
COMANCHE PEAK 1
KRSKO
MARBLE HILL 2
SHEARON HARRIS 1
SHEARON HARRIS 2

D5

BRAIDWOOD 2
BYRON 2
CATAWBA 2
COMANCHE PEAK 2
MARBLE HILL 1

E

DOEL 4
IBERDUERO 3
SOUTH TEXAS 1
SOUTH TEXAS 2
TIHANGE 3

Fig. 1.1-1. Preheat steam generator geometry acceleration and displacement values were obtained. These data showed that the impact between the tube and baffle plates, which lead to tube wear, started at power levels between 60% and 85% at Krsko. In comparison with the results obtained at Almaraz, it was shown that, because of the different tube support conditions, the critical vibration frequency at Krsko is lower than at Almaraz. RMS displacement at Krsko for a power level of 70% was found comparable to the Almaraz displacement for a power level of 50%.

On the basis of these results, it was concluded that the feedwater flow into the Krsko steam generators through the main nozzle should be limited to 70% of full flow to reduce the possibility of early wear on the tubes.

Since the feedwater flow through the main nozzle was limited to 70% at Krsko, 100% operation was considered possible by feeding the remaining 30% flow through the auxiliary nozzle (located in the upper portion of the steam generator). In order to achieve the 30% flow through the auxiliary nozzle, the installation of new control valves and new bypass piping was required to reduce the hydraulic resistance in the lines to the auxiliary nozzle.

Until the beginning of the piping modification (March and April 1982), Krsko was operated for 1500 hours at 75% power (70% flow to the main nozzle and 5% through the auxiliary nozzle). Parallel to the piping and valve modifications, new accelerometers were installed and tube R49 C56 was removed for examination. Examination of this tube showed wear at baffle plates B, D, and G. The major wear was at plate B with local wall thinning of 2.5 mils (6% of the tube thickness).

The results of instrumentation data taken at 70%/0% and 70%/30% conditions have shown that RMS displacement of certain tubes was higher with the 70%/30% split flow than with 70% flow through the main nozzle. This result is attributed to different feedwater temperatures at 70% and 100% power, which cause different differential thermal expansion conditions between the baffle plates and the tubes. However, the increase in vibration is small and the 70%/30% split flow is sufficient to reduce wear on the tubes to small rates. These reduced wear rates are roughly equivalent to the Model D2/D3 steam generator design operating at 50% power.

From the data taken at Krsko and from model test data, it was evident that modifications via split feedwater flow and/or other methods would be required to operate the D4 steam generators at 100% power over the life of the plant. Due to differences in the design details between the split flow and the counterflow steam generators, the flow distribution modification used for the split flow steam generators was determined not to be applicable to the counterflow steam generators.

1.2. DESCRIPTION OF MODEL D4/D5/E STEAM GENERATORS

1.2.1. General Description

Heat generated in the Westinghouse pressurized water reactor (PWR) is removed from the core by the primary coolant water that is transported to the steam generators by reactor coolant pumps. Each primary coolant loop in the Westinghouse PWR design has one reactor coolant pump and one vertically mounted U-tube steam generator. The steam generators are designed with the following integral sections: a preheater section, an evaporator section, and a steam drum section (see Fig. 1.2-1). The steam drum section is the upper part of the steam generator containing the moisture separators. The evaporator section is an inverted U-tube heat exchanger containing either 4578 (D4), 4570 (D5), or 4864 (E) Inconel tubes (3/4-in. diameter). Primary coolant water is circulated through these tubes to transfer heat from the primary coolant to water in the secondary side of the steam generator, which causes the secondary side water to boil. The primary coolant water enters the hot leg of the U-tubes at approximately 618°F (D4/D5) and 626°F (E), then flows through the U-tubes to the cold leg, and exits the steam generator at approximately 558°F (D4/D5) and 560°F (E). The primary coolant then returns via the reactor coolant pump to the reactor core where it is reheated and the cycle repeated.

Feedwater is pumped into the secondary, or shell, side of the steam generators, where it boils and generates steam to drive the turbine generator. In order to enhance heat transfer to the incoming feedwater, the Model D and E steam generators incorporate a series of baffle plates around a portion of the cold leg, which forms the preheater section. Feedwater flowing into the steam generator first passes through a venturi insert in the main feed nozzle that serves as a backflow restrictor to limit the rate of blowdown from the steam generator in the event of a main feedwater line break. Upon entering the preheat section, feedwater is diverted by a flat plate that forms the side of the water box. The feedwater turns through 90 degrees and is diverted to the lower section of the preheater. At the lower section of the preheater, or "first pass," the feedwater enters the tube

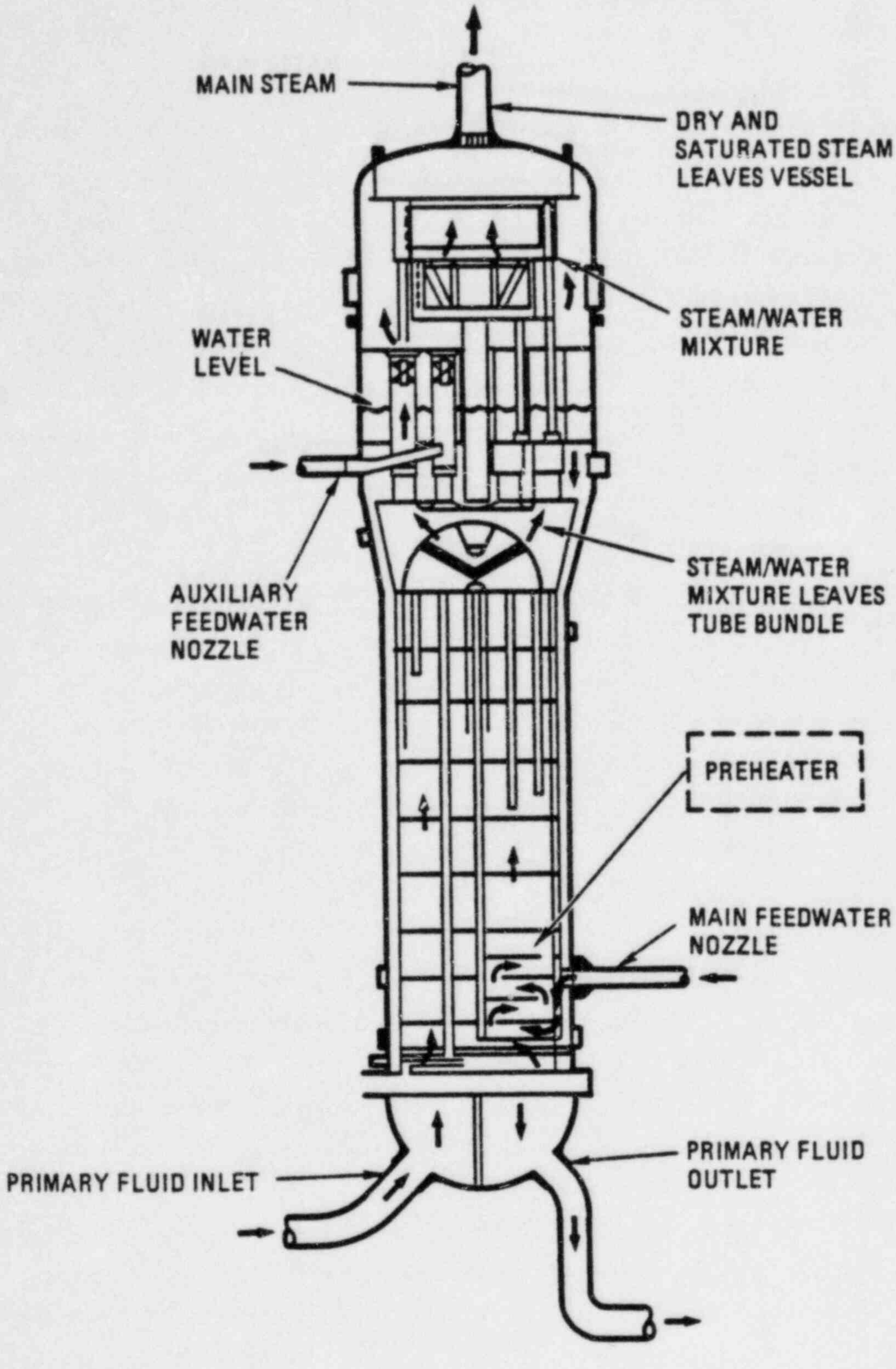


Fig. 1.2-1. Preheat steam generator design

bundle. This feedwater inlet geometry is different from that of the D2/D3 steam generators. A comparison of all Westinghouse preheat steam generators is shown in Fig. 1.2-2.

1.2.2. Preheater

The preheater section of the steam generator is made up of a series of seven semicircular baffle plates through which the cold leg tubes of the steam generator are routed (Fig. 1.1-1). The plates are spaced in the first [ ]a,b above the tube sheet. The lowest plate in the preheater is plate B, which is approximately [ ]a,b above the tube sheet. The other five plates are located above plate B with an even spacing of [ ]a,b between each plate. The flow geometry for Model D4/D5 is shown in Fig. 1.2-3 and that for Model E is shown in Fig. 1.2-4. Feedwater enters the preheat section between plates B and D. The incoming feedwater flows horizontally across the cold leg tubes. At full flow, approximately [ ]a,b leaks down through the gaps in plate B and [ ]a,b leaks out of the preheater through slots at the end of plate B adjacent to the center partition plate at the end of the preheater. This center partition plate is located between the legs of the row 1 U-tubes.

The feedwater flow is then turned 180 degrees at the center partition plate and flows into the second pass between plates D and E. At the end of plate E near the inlet feedwater box, the flow again turns 180 degrees into the third pass between plates E and G. This turn between the second and third pass has a different geometry for the Model D4/D5 and E steam generators. For the Model D4/D5 steam generators, plate E stops at tube row 44 and the last five rows of tubes are unsupported between plates D and G. The turning flow between the second pass and the third pass is upward through the tube bundle. For the Model E steam generators, plate E extends to the outer row of tubes, row 48, and the returning flow is completely through an open area and around the end of plate E.

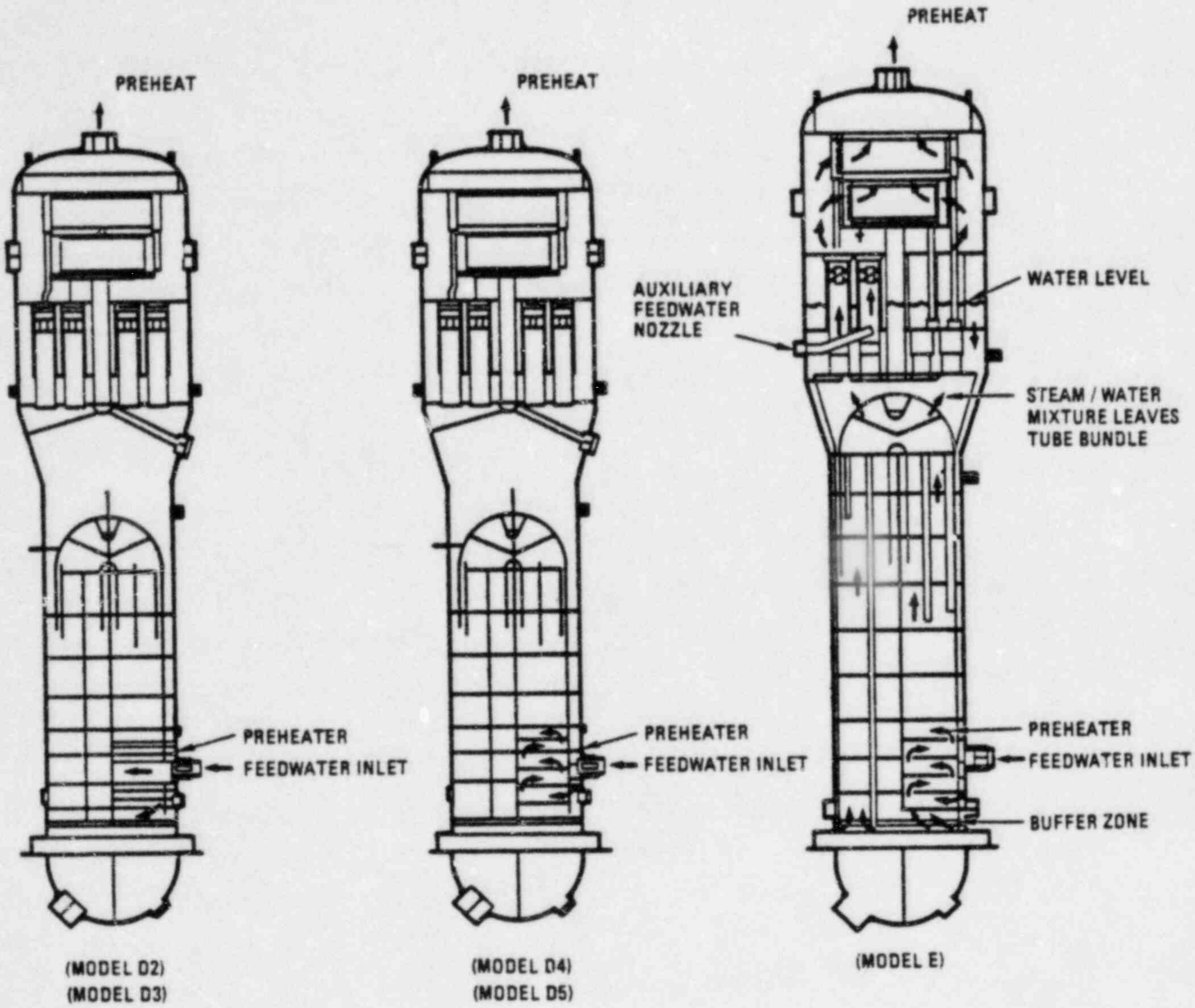


Fig. 1.2-2. Westinghouse steam generator comparison

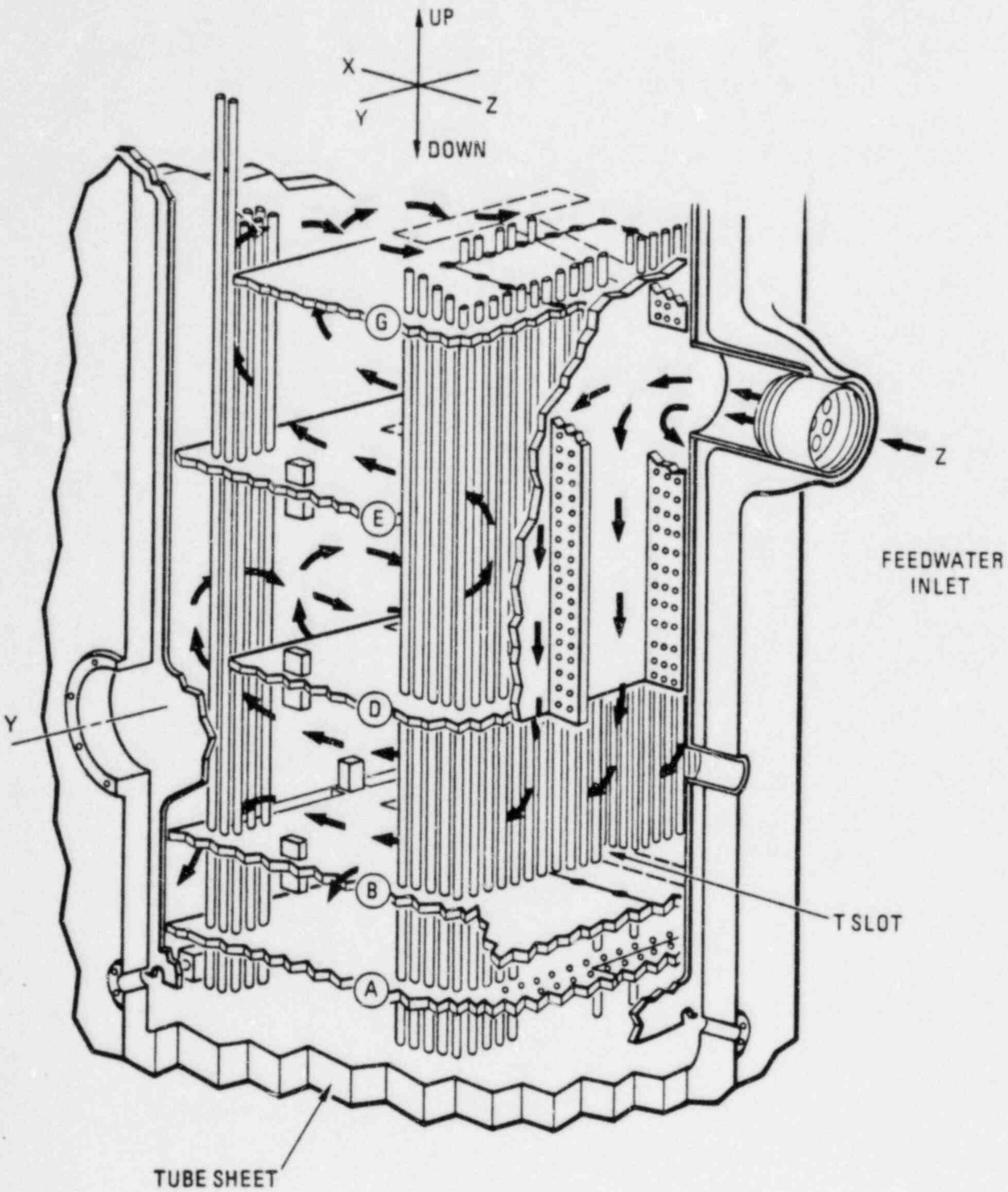


Fig. 1.2-3. Flow geometry for Model D4/D5

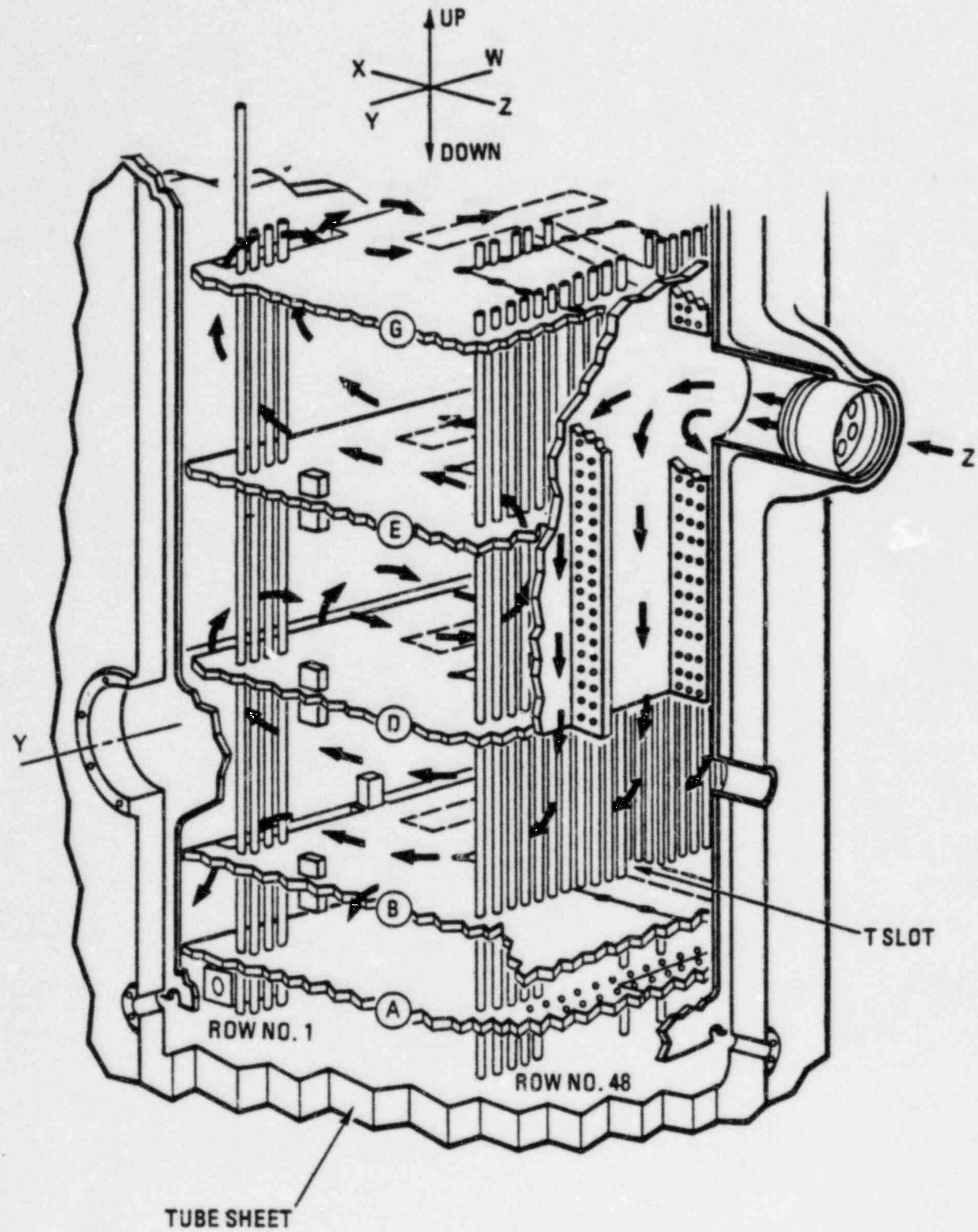


Fig. 1.2-4. Flow geometry for Model E

1.3. PROPOSED MODIFICATIONS

Modifications to limit preheater tube vibrations in the Model D4/D5 and E steam generators were proposed by Westinghouse and have been reviewed by the TRC.

In the D4/D5 units, a combination of velocity reduction and tube expansion has been considered and tested as a means of limiting tube vibration and subsequent tube wear. Peak velocity reduction is achieved by splitting the flow between the main feedwater nozzle and the auxiliary feedwater nozzle. Tube expansion consists of local expansion (increasing the diameter) of the tubes where they pass through the lowest two baffles in the preheater (B and D baffles). This process results in lower levels of tube vibration.

In the Model E steam generator, the only modification proposed by Westinghouse is tube expansion.

1.4. DESCRIPTION OF REVIEW PROCESS

On February 4, 1983, representatives for several owners of Model D4, D5, and E steam generators met to discuss the Westinghouse program. The consensus of the group was that the tube vibration problems associated with the preheater section of the equipment were of sufficient complexity to warrant a thorough technical review by the owners of both problem definitions and suggested Westinghouse remedies. To that end, the owners formed a Counterflow Steam Generator Owners Review Group consisting of the following utilities:

Carolina Power and Light Company (CP&L)
Commonwealth Edison Company (CECO)
Houston Lighting and Power Company (HL&P)
Public Service Company of Indiana (PSI)
Belgian Utilities (Electronucleaire)
Nuklearna Elektrarna Krsko (Yugoslavia)

The intent of this group, whose charter is presented in Appendix A, was to complete the same type of review that was done by the Design Review Panel (Tennessee Valley Authority, South Carolina Electric and Gas Company, and Duke Power Company) for the Model D2/ D3 steam generators. The Owners Review Group established a Technical Review Committee (TRC) to complete the detailed design review of the final modification program, proposed by Westinghouse, for limiting major tube vibration and wear in the preheat section of D4, D5, and E steam generators. This group brought to bear a broad spectrum of technical expertise from the participating owners to ensure a thorough technical review and the resolution of the steam generator tube vibration problem.

The Counterflow Steam Generator Owners Review Group is composed of two subgroups. The Technical Review Committee (TRC), concerned with the technical review of the final proposed Westinghouse modification program, is composed of utility representatives, utility consultants, and an Electric Power Research Institute (EPRI) representative. The second subgroup is the Steering Committee, which consists of one executive (Vice President level) or his appointed representative from each owner. A listing of the members of both subgroups and a summary of their backgrounds and qualifications are presented in Section 6.

The first meeting of the Counterflow Steam Generator Owners Review Group with Westinghouse was held on March 18, 1983, and the first questions from the TRC were provided to Westinghouse by letter on March 22, 1983.

The second meeting of the TRC and Westinghouse was held on April 22, 1983, to review the Westinghouse responses to initial questions. A representative from the Krsko plant in Yugoslavia was present at this meeting. After this meeting, Nuklearna Elektrarna Krsko in Yugoslavia stated that they wanted to join the Counterflow Steam Generator Owners Review Group.

The third meeting of the TRC was held on May 16, 1983, to provide the committee members with questions and/or clarifications required on the Westinghouse responses of May 5, 1983, prior to the meeting with the NRC and Westinghouse on May 18, 1983. A representative from Duke Power Company was present at the meeting. A fourth "minigroup" meeting of the TRC was held on the morning of May 18, 1983, to resolve questions prior to the afternoon meeting with Westinghouse.

The TRC met again on June 8-10, 1983, along with Torrey Pines Technology, to finalize the draft report. On June 9 and 10, 1983, Westinghouse provided additional information.*

*"Westinghouse - Counterflow Preheat Steam Generator Tube Vibration Summary Report," June 1983; "Westinghouse - Counterflow Preheat Steam Generator Tube Expansion Report," June 1983; and additional answers to TRC questions were incorporated into this report.

The TRC invited the NRC to participate in the review process. A representative attended several of the meetings described above to respond to questions concerning report details and NRC concerns.

2. DESIGN CRITERIA AND OBJECTIVES

2.1. GENERAL CRITERIA AND OBJECTIVES

2.1.1. General Criteria

The following criteria were established by the TRC for accepting the modification program:

For safety assessment purposes,

1. Limit vibrational levels so that maximum wear scar depths, conservatively predicted by methods outlined hereinafter, will be less than 65% of the wall thickness (structural integrity limit) for a time interval of 18 equivalent full-power months.

The safety requirement governing the hydraulic performance criteria for the modified counterflow preheat steam generator is that tube wear due to flow-induced vibration shall not result in tube wall reduction in excess of the safety limit for tubes in service. The safety limit for tube wall reduction is the amount of wall loss the tube can sustain and maintain integrity under the most severe accident conditions. For preheat steam generator tubing, this limit has been determined by analysis and test to be a 65% wall reduction.

For long-term reliability assessment purposes,

2. Limit tube vibration to levels comparable to those of the D2/D3 models equipped with the manifold and the Krsko 70%/30% split flow condition.

These limits are based upon projected minimum tube wear for extended periods of operation per NUREG-0966.

The second set of TRC acceptance criteria ensures that the design modifications remain within acceptable industry design standards. These criteria are:

1. Effects of tube expansion process shall be evaluated using the ASME Boiler and Pressure Vessel Code, Section III.

2. The effects of the modification shall not violate required reactor thermal margins or any other plant operational safety parameters.

2.1.2. General Objectives

To ensure that modifications made to Model D4, D5, and E steam generators and/or main/auxiliary feedwater systems would result in an acceptable design, Westinghouse established the following general design objectives:

1. G-Delta is a measure of the magnitude of the vibration and is related to the wear-producing capability of the vibration. The value of G-Delta that Westinghouse considers appropriate to indicate acceptable levels of vibration is [ ]a,b,c,e or less for long-term operation.

2. Minimization of potential for fluidelastic instabilities.

3. Predicted values of 40% wall reduction for the design basis case and 65% wall reduction for the safety case are used as guidelines.

4. Effects of tube expansion process shall be evaluated using the ASME Boiler and Pressure Vessel Code, Section III.

5. The effects of the modification shall not violate required reactor thermal margins or any other plant operational safety parameters.

The above objectives are consistent with TRC acceptance criteria.

2.2. MATERIALS AND CORROSION OBJECTIVES

The principal objective regarding materials and corrosion processes is that the steam generator modification will not degrade the preheater tubing when the plant is operated according to Westinghouse interface requirements.

The various forms of corrosion considered in the evaluation include:

1. Stress corrosion cracking.
2. Processes leading to tube denting.
3. Wastage.
4. Pitting.
5. Fretting corrosion.

2.3. THERMAL-HYDRAULIC DESIGN OBJECTIVES

The thermal-hydraulic design objectives are established so as not to affect plant performance criteria, i.e., rated power at rated steam pressure and temperature. An initial set of performance values to achieve this is defined in Table 2.3-1.

TABLE 2.3-1
INITIAL THERMAL-HYDRAULIC DESIGN OBJECTIVES

Model	D4/D5	D4/D5	D4/D5	E
No. of loops	2	3	4	All
Maximum main feedwater nozzle flow (% of design flow)	84	84	92	No flow split required
Primary coolant average temperature increase (°F)	1.2	1.2	0.6	No flow split required
Nominal main feedwater nozzle flow (%)	82	82	90	No flow split required

2.4. MECHANICAL DESIGN AND QUALIFICATION PROCESS/TUBE EXPANSION CRITERIA

2.4.1. Mechanical Design Criteria

The mechanical design criteria are as follows:

1. Tubes for which the expansion process is applied shall be evaluated for design, normal, upset, emergency, and faulted conditions in accordance with the requirements of the ASME Code, Section III, Subsection NB.

2. Steam generator pressure boundary components affected by changed thermal-hydraulic conditions caused by the diversion of feedwater flow to the auxiliary nozzle shall be evaluated against ASME Code requirements.

3. The modification shall not compromise the tube plugging margin requirements specified in NRC Regulatory Guide 1.121.

4. For purposes of the ASME Code calculations, a 40-year design life requirement shall be considered.

5. The effect of the tube expansion process shall be evaluated for its potential effect on the reduction of the tube fatigue life. Special attention will be addressed to the effect of the wear marks of the worn tubes of the Krsko plant.

6. The residual stress as inferred from the tube wall strain gradient at the expansion shall not be larger than that in a mechanically rolled tube sheet transition expansion.

2.4.2. Qualification Process/Tube Expansion Criteria

The criteria placed on the qualification of the tube expansion process are as follows:

1. The tube-to-baffle plate gap clearance after application of the process shall be less than or equal to [ ]a,b,c,e at the peak expansion point of the tube.

2. The parallel portion of the expanded tube length shall be greater than or equal to [ ]a,b,c,e.

3. The expanded tube shall be free to slide axially over a distance greater than or equal to [ ]a,b,c,e.

4. The diameter variation within the parallel expanded portion of the tube shall be less than or equal to [ ]a,b,c,e.

5. Tube expansion outside the limits of the baffle plate, if found in the field, will be evaluated for acceptability on a case-by-case basis.

6. The expanded tube region shall be centered within [ ]a,b,c,e of the baffle plate centerline.

2.5. ADDITIONAL SAFETY ANALYSIS REPORT REQUIREMENTS

Each individual plant has differences in its design and its Safety Analysis Report (SAR). This section describes, on a generic basis, the areas of the SAR that were examined for potential impact due to the implementation of the steam generator modification.

Table 2.5-1 lists the sections of a typical Final Safety Analysis Report (FSAR) that would be reviewed for impact due to this modification. Table 2.5-2 lists the events considered when evaluating the impact on the FSAR sections.

After several meetings with Westinghouse that included generic plant data as well as specific information for the Comanche Peak plant, it was concluded that only minor revisions to the FSAR will be required. The impact of the modification on the safety analyses is believed to be within the tolerance of the available plant margins. This will be verified and documented on a plant-specific basis or plant-specific parameters will be adjusted.

TABLE 2.5-1
FSAR SECTIONS REVIEWED FOR STEAM GENERATOR MODIFICATION IMPACT

Chapter 3. Design of Structures, Components, Equipment, and Systems

- 3.6. Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping
- 3.9. Mechanical Systems and Components

Chapter 4. Reactor

- 4.4. Thermal and Hydraulic Design

Chapter 5. Reactor Coolant System and Connected Systems

- 5.1. Summary Description
- 5.2. Integrity of the Reactor Coolant Pressure Boundary
- 5.3. Reactor Vessel
- 5.4. Components

Chapter 6. Engineered Safety Features

- 6.2. Containment Systems

Chapter 7. Instrumentation and Controls

Chapter 9. Auxiliary Systems

- 9.2. Water Systems
- 9.3. Process Auxiliaries

Chapter 10. Steam and Power Conversion System

- 10.1. Summary Description
- 10.4. Other Features of Steam and Power Conversion System

Chapter 15. Accident Analyses

Chapter 16. Technical Specifications

TABLE 2.5-2
REACTOR PLANT DESIGN TRANSIENTS INCLUDED IN EVALUATION OF STEAM GENERATOR MODIFICATION IMPACT ON FSAR

Normal Conditions

- Unit Loading
- Unit Unloading
- Startup
- Shutdown
- Step Load Increase
- Step Load Decrease
- Feedwater Cycling
- Load Rejections
- Oscillations About Steady State
- Boron Concentration Equalization
- Loop Startup
- Loop Shutdown
- Temperature-Power Operating Map

Upset Conditions

- Loss of Load
- Loss of Power
- Loss of Feedwater
- Excessive Feedwater
- Partial Loss of Flow
- Rod Withdrawal
- Boron Dilution
- Inadvertent RCS Depressurization
- Reactor Trip
- Reactor Trip with Cooldown
- Reactor Trip with Cooldown Actuating Safety Injection
- Inadvertent Startup of an Inactive Loop
- Control Rod Drop
- Inadvertent Safety Injection
- Excessive Load Increase
- Bubble Collapse

Emergency Conditions

- Small LOCA
- Small Steamline Break
- Complete Loss of Flow

Faulted Conditions

- Steamline Break (Double Ended)
- Control Rod Ejection
- Feedline Break (Double Ended)
- LOCA
- Reactor Coolant Pump Locked Rotor
- Steam Generator Tube Rupture

2.6. TRAINING AND OPERATIONS REQUIREMENTS

Training and operations requirements will be addressed by each utility on a plant-specific basis.

3. DESCRIPTION OF MODIFICATIONS

3.1. SPLIT FLOW ASPECTS OF DESIGN

3.1.1. Introduction

The split flow modification involves diverting a specified percentage of the feedwater flow through the auxiliary nozzle. The percentage of flow diversion is dependent on the steam generator model and the number of loops at a specific site, the ultimate objective being to reduce the flow velocity to the point where a minimum number of tubes would require expansion. Coupled with tube expansion, approximately 18% flow diversion is required for Model D4/D5 two- and three-loop plants, approximately 10% flow diversion is required for Model D4/D5 four-loop plants, and no flow diversion is proposed for Model E plants.

3.1.2. Model D4/D5 Two- and Three-Loop Plants

Approximately 18% flow diversion is required for two- and three-loop plants. In order to keep the steam pressure constant, this modification will require approximately a 1.2°F increase in the average temperature of the primary coolant.

Modifications to the feedwater system that are necessary to achieve this flow diversion include:

1. Reduce the bypass flow resistance in the feedwater line to the auxiliary nozzle.

2. Increase the flow resistance in the feedwater line to the main nozzle.

3. Flow-measuring devices.

It is expected that Item 1 can be achieved in most plants by increasing the C_v of the bypass valve. However, in some plants pipe sizes and geometry may require alterations to reduce flow resistance to acceptable levels. It is expected that Item 2 can be obtained by valve C_v changes or through the addition of restrictor or orifice plates in the feedwater line to the main nozzle. Appropriate logic and alarms will be added to inform the operator of high flow through the main nozzle.

A typical feedwater configuration and a typical feedwater flow distribution for a three-loop plant are shown in Figs. 3.1-1 and 3.1-2, respectively.

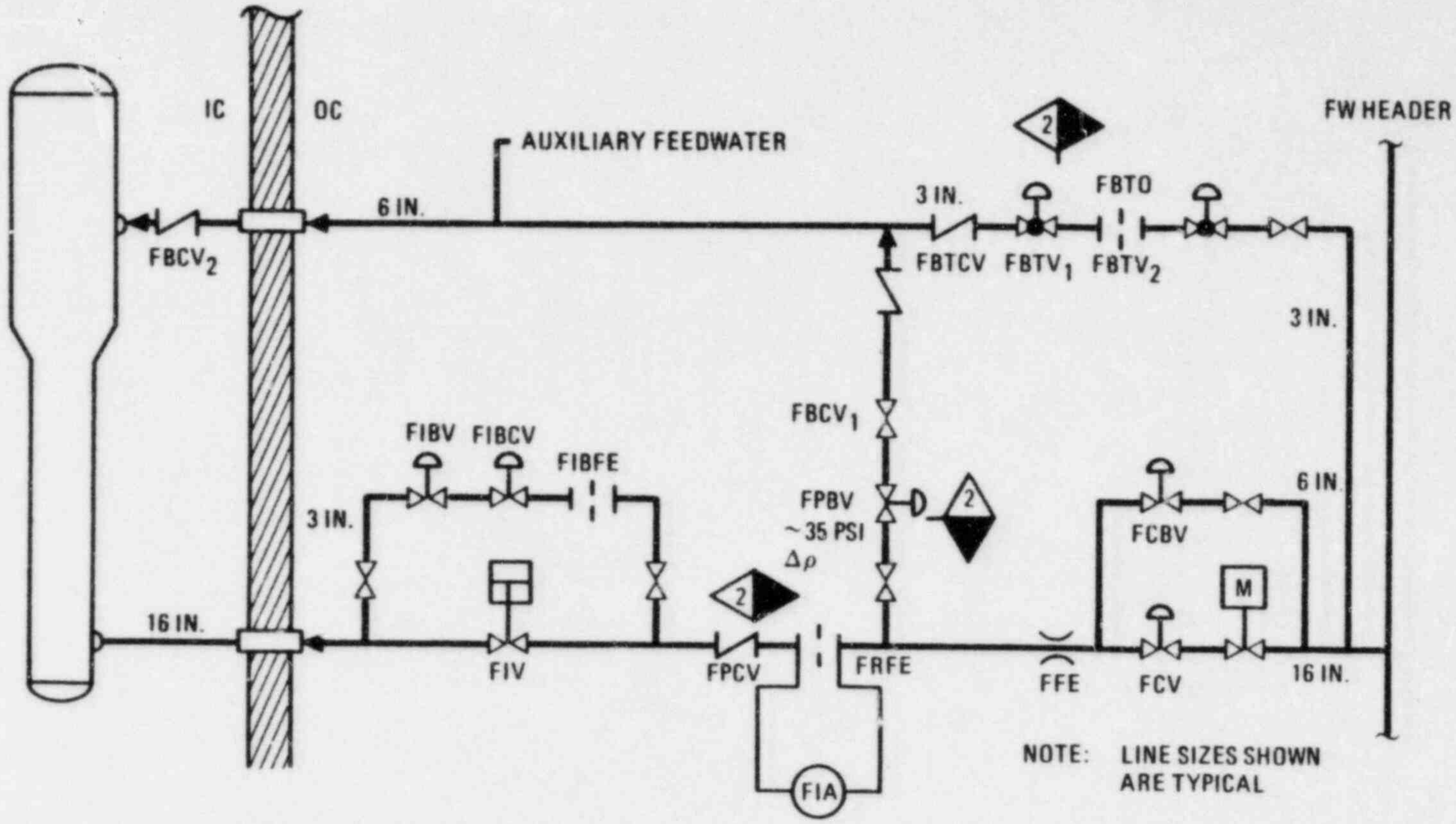
3.1.3. Model D4/D5 Four-Loop Plants

For a four-loop plant, approximately 10% flow diversion to the auxiliary nozzle will be required. The average temperature of the primary coolant will be increased by approximately 0.6°F to maintain the original steam pressure. A typical four-loop plant feedwater configuration is shown in Fig. 3.1-3.

To achieve this flow diversion, it may be necessary to reduce the bypass system flow resistance, to add a flow-measuring device in the bypass line, and to add appropriate logic and alarms to inform the operator of high flow through the main nozzle.

3.1.4. Model E Plants

Westinghouse proposes that for the Model E plants no flow diversion through the auxiliary nozzle is needed, primarily because the peak inlet pass velocities are lower in these units than in the Model D4/D5 steam generators. These conditions in the Model E steam generator are a result of a larger preheater crossflow area.



NOTES:

FBCV	FW BYPASS CHECK VALVE	FIBV	FW ISOL. BYPASS VALVE
FBTCV	FW BYPASS TEMPERING CHECK VALVE	FIBFE	FW ISOL. BYPASS FLOW ELEMENT
FBTO	FW BYPASS TEMPERING ORIFICE	FIV	FW ISOL. VALVE
FBTV	FW BYPASS TEMPERING VALVE	FPBV	FW PREHEATER BYPASS VALVE
FCBV	FW CONTROL BYPASS VALVE	FPCV	FW PREHEATER CHECK VALVE
FCV	FW CONTROL VALVE	FRFE	FW RESTRICTOR/FLOW ELEMENT
FFE	FW FLOW ELEMENT		
FIBCV	FW ISOLATION BYPASS CONTROL VALVE		

Fig. 3.1-1. Typical main feedwater bypass arrangement for three-loop plants

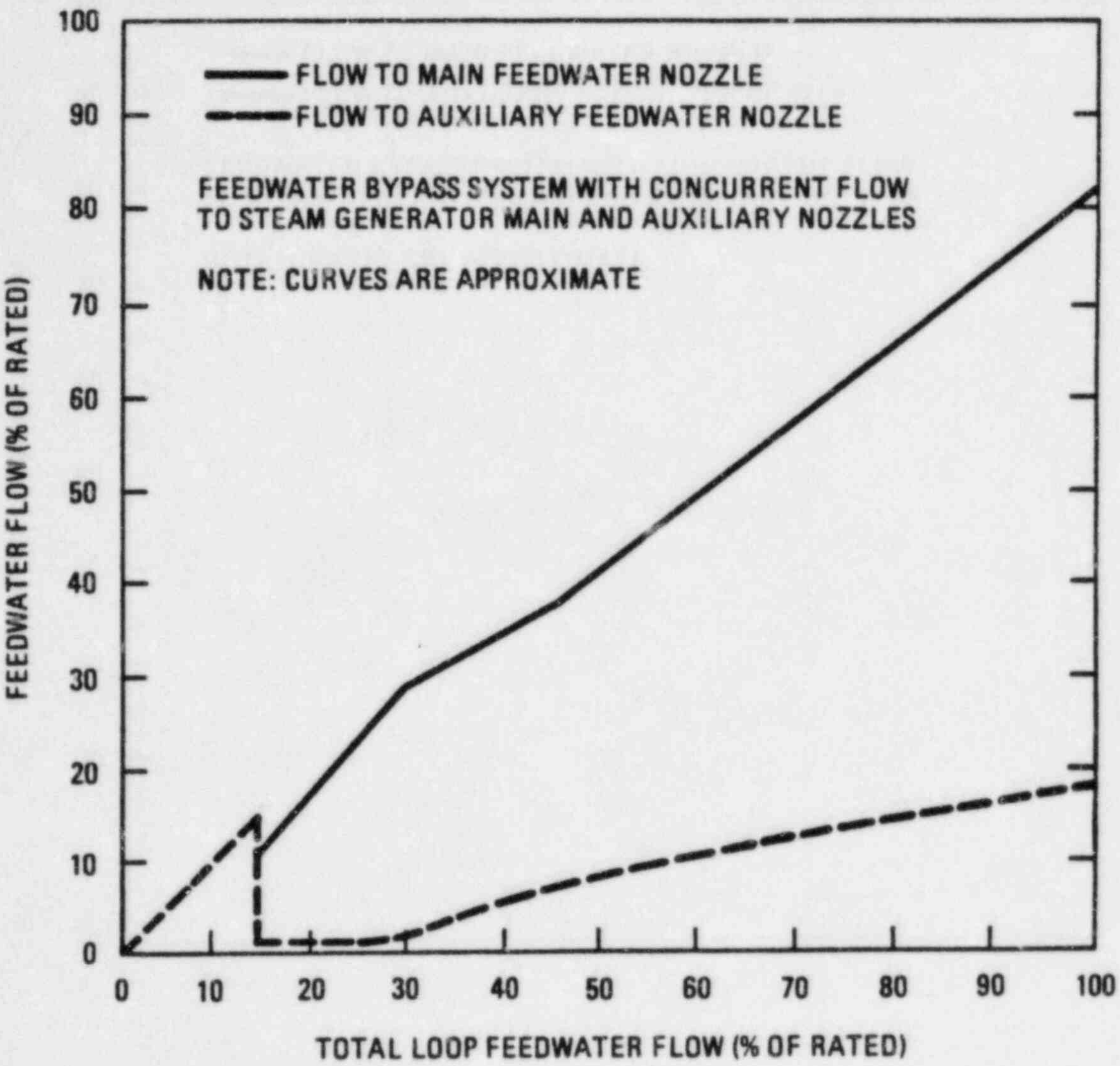
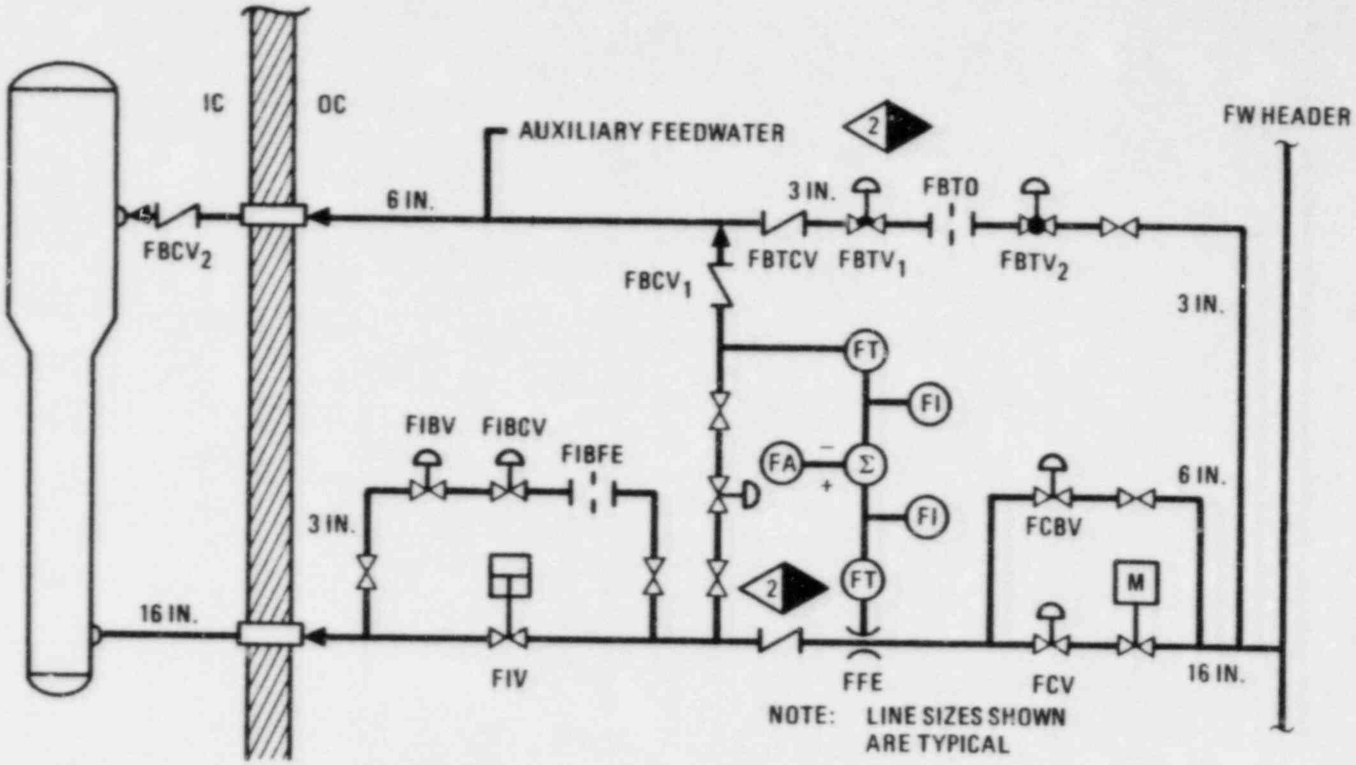


Fig. 3.1-2. Typical feedwater flow distribution for three-loop plants



NOTES:

FBCV	FW BYPASS CHECK VALVE
FBTCV	FW BYPASS TEMPERING CHECK VALVE
FBTO	FW BYPASS TEMPERING ORIFICE
FBTV	FW BYPASS TEMPERING VALVE
FCBV	FW CONTROL BYPASS VALVE
FCV	FW CONTROL VALVE
FFE	FW FLOW ELEMENT
FIBCV	FW ISOLATION BYPASS CONTROL VALVE
FIBV	FW ISOL. BYPASS VALVE
FIBFE	FW ISOL. BYPASS FLOW ELEMENT
FIV	FW ISOL. VALVE
FPBV	FW PREHEATER BYPASS VALVE
FPCV	FW PREHEATER CHECK VALVE

Fig. 3.1-3. Typical main feedwater bypass arrangement for four-loop plants

Flow in a Model E steam generator is higher than in a Model D4/D5 steam generator. However, the inlet area to the tube bundle is larger, yielding average velocities entering the region in front of the first pass that are lower. The Model E inlet conditions are approximately equivalent to those of a two- or three-loop Model D4 with approximately 20% flow diversion. However, flow velocities in the upper passes are higher than for the D4/D5 models.

The distance between the tubes and the preheater side of the impingement plate is larger in the Model E than in the Model D4. This additional turning distance may further reduce the turbulence of the water entering the tube bundle.

In addition, unlike the Model D4/D5 units, the base Model E steam generator design provides for no window or unsupported tubes at the E and H baffles. This greatly reduces the susceptibility for tube vibration in the outer five rows of the Model E units. However, certain steam generators may not totally meet the base design. These steam generators will be evaluated on a plant-specific basis.

3.2. DESCRIPTION OF TUBE EXPANSION PROCESS

Westinghouse has proposed expansion of tubes in the preheater region (see Fig. 3.2-1). This method is used to limit levels of steam generator tube vibration and involves locally expanding selected tubes at baffles B and D within the preheater. This increase in the diameter of the tube within the plates decreases the diametral gap existing between the tube and the baffle plate. A baffle plate in which a tube has been expanded acts as a nodal point in defining tube mode shape and frequency. Increasing the number of nodal points along a given length of tube increases its natural frequencies and results in a reduction of tube vibration amplitude.

The physical criteria placed on the qualification of the tube expansion process are given in Section 2.4.2. The tools used to perform the modifications to the tubes are described in Section 4.1.

Each step of the expansion process has been tested on a full-scale mockup and actual plants, and this process has been found to consistently result in a tight predictable gap between the tube O.D. and the baffle plate.

The remaining portion of this page, the write-up on page 3-9, and Figs. 3.2-1 (p. 3-8) and 3.2-2 (p. 3-10) found in the proprietary version of this report are omitted from this non-proprietary report since they describe the tube expansion process, which Westinghouse has classified as proprietary.

Four pages of the original proprietary report have been omitted...

4. IMPLEMENTATION OF MODIFICATIONS

The modifications consist of tube expansion and splitting of feedwater flow as discussed in Section 3. The flow split will be implemented on a site-specific basis and discussion of the flow split implementation is not included in this report. The following section describes implementation of the tube expansion modification.

4.1. TOOLING

4.1.1. Introduction

Westinghouse has developed tooling and processes for all phases of the steam generator modification. All tooling has been extensively tested on full-scale mock-ups for verification of task performance and tooling reliability and suitability. A detailed description of the tooling and processes is found in Westinghouse Report SGPR-8301, "Counterflow Preheat Steam Generator Tube Expansion," June 1983.

The initial development phase for tooling has concentrated on equipment to be used on a steam generator that has not been in service. Additional remote operating features will be employed at the Krsko plant owing to the expected radiation levels within the steam generators. Westinghouse has proven satisfactory operability of its coordinated transport (CT) system on several domestic plants.

4.1.2. Description

A hydraulic power source will be used to develop the high ramp pressures required for the tube expansion process. This package unit will supply pressure at a controlled rate up to its maximum shutoff rating.

The write-up on this page and Figs. 4.1-1 (p. 4-3), 4.1-2 (p. 4-4), 4.1-3 (p. 4-5), 4.1-4 (p. 4-6), 4.1-5 (p. 4-7), and 4.1-6 (p. 4-8) found in the proprietary version of this report are omitted from this non-proprietary report since they describe different tools used for tube expansion, which Westinghouse has classified as proprietary.

4.2. NDE REQUIREMENTS FOR EXPANDED TUBES

4.2.1. Introduction

Nondestructive examination of the expanded regions of the tubes is required for the quality control of the process and to establish a baseline for use in comparisons with future ISI. Westinghouse has proposed eddy current techniques for both purposes. These techniques are discussed below.

4.2.2. Post-Expansion Eddy Current Inspection

The two circumferentially wound eddy current coils will be used to locate baffle plates and will provide an indication of the profile of the expanded region of the tubes. The coils use a single channel, multifrequency eddy current system in the differential mode to locate the baffle plates and in an absolute mode to inspect the expanded zone of the tubes.

Signatures (tube profiles) will be recorded on a strip chart and compared with signatures obtained from reference expansions of known depth and profile. Coupled with this recording on the strip chart will be a baffle plate recording. A comparison of these two recordings verifies the location of the expansion zone within the baffle plate. This process was used successfully in late 1982 for measuring the expanded and hardrolled regions on approximately 3000 tubes in an operating plant.

The proposed technique appears acceptable for quality control purposes, although the profile examination obtained from encircling coils is not able to distinguish ovalized expansion or abrupt transition. These off-nominal expansion conditions did not occur during the qualification process (more than 400 expansions).

4.2.3. Post-Expansion Baseline Eddy Current Inspection

The baseline data for the expanded zones of the tubes will be used for a comparison analysis with future eddy current readings. The industry standard bobbin eddy current coil will be used to obtain these baseline signatures, which will be used to identify significant wear indications in the expansion zones.

In addition, Westinghouse is conducting a program with Zetec to determine the sensitivity of a new eddy current testing probe. Total qualification results are not yet available for the TRC review. This new probe utilizes pancake coils located around the circumference of the probe. Each of the coils has an eddy current field that overlaps the area covered by the adjacent coils so that the entire circumference of the tube is inspected. This new eddy current probe will be used to obtain supplemental information during future in-service inspections if necessary.

Several factors combine to provide the rationale for the acceptability of the existing capability for in-service inspection of expanded tubes. These include expansion process qualification, leak before break assurance, chemistry and corrosion studies, structural analyses, and single tube failure analysis.

The extensive program of expansion process qualification provides assurance that cracks will not be introduced into the transition region as a result of expansion. Westinghouse has examined, by dye penetrant, over 80 expansions of Inconel tubes and found no cracks. In addition, six Inconel samples have been metallographically examined and no cracks have been found. The leak before break analysis provided by Westinghouse provides assurance that in the event that a crack did occur, its presence could be detected and corrective action taken before a break occurred.

Chemistry and corrosion studies and structural analyses of the expanded tube configuration have been performed. These have demonstrated that no preferential tendency for cracking relative to other areas of the steam generator exists. Analyses of the results of a double-ended guillotine break of a single tube are evaluated for each plant and the results are provided in FSAR Section 15.6.3. The conservative single tube analysis provides sufficient confidence in the acceptability of plant operation.

In conclusion, Westinghouse believes that the current state-of-the-art level of in-service detection capability is sufficient in combination with other existent design and analysis features to assure safe plant operation.

4.3. POST-MODIFICATION MONITORING

4.3.1. Introduction

The purpose of the post-modification monitoring instrumentation is to obtain vibration levels for comparison with the objectives of the modifications. Values exceeding the modification objectives would be reviewed. A vibration monitoring plan proposed by Westinghouse calls for the first modified Model D4/D5 and the first modified Model E steam generators to be instrumented. The lead plant having Model D4 steam generators is Comanche Peak 1, which began tube expansion on June 17, 1983. The lead plant having a Model E steam generator is expected to be Doel 4 in Belgium, scheduled for modification in October 1983.

The vibration monitoring plan calls for accelerometers to be installed in tubes of one steam generator for vibration measurements prior to extended operation in each of the lead plants. The accelerometers will be selectively located to provide an adequate representation of the unexpanded and expanded tube populations in the region most susceptible to vibration and wear.

The criterion used for accelerometer location is to instrument unexpanded tubes and expanded tubes, selecting tubes in each category that are typical of higher vibration level tubes.

4.3.2. Model D4/D5 Accelerometer Locations

The tubes for accelerometer installation in the lead plant having D4/D5 steam generators are located at [ ]b,c,e. Figure 4.3-1 shows the locations of the instrumented tubes. The limiting expanded tube having the highest predicted wear will be instrumented.

a,c,e

Fig. 4.3-1. Model D4/D5 steam generator estimate of tube expansion

On the D4/D5 steam generators, the first five rows of tubes, rows 49 through 45, have no support at plate E and are termed "window" tubes. Table 4.3-1 summarizes the accelerometer locations.

4.3.3. Model E Accelerometer Locations

Tubes selected for accelerometer installation in the Model E lead plant will be designated following tube expansion map determination by Westinghouse.

TABLE 4.3-1
MODEL D4/D5 ACCELEROMETER LOCATIONS

Tube	Expanded	Plate E Support	Number of Accelerometer Assemblies	Location of Accelerometers

a,c,e

4.4. RADIOLOGICAL CONSIDERATIONS

With the exception of Krsko, which will have been operating for approximately 9000 hours (1-1/2 year) prior to full implementation of the tube expansion process, all other plants will have the modification completed prior to initial operation. Therefore, radiological considerations apply only to Krsko or in the unlikely case that post-expansion monitoring were to indicate the necessity or desirability of expanding one or more additional tubes.

In Krsko's case the estimated collective personnel doses, with and without the use of a CT (coordinated transport) system, have been determined by Westinghouse to be 200 and 350 person-rem per steam generator, respectively. They have shown the clear benefits, in person-rem, of employing a CT system when expanding many tubes. The TRC has reviewed the assumptions and the radiation fields used in their time and motion study. The radiation fields are consistent with data available in the literature for locations in and near steam generators having an operating history similar to Krsko.

In case of the expansion of one or a few tubes, it is not likely that semiautomated means will be employed since they would not be person-rem effective. In such cases the installation of the CT system would result in more person-rem expenditure than manual operation. Moreover, certain tubes cannot be reached by the CT system and can only be expanded manually. For expansion of a few tubes, the expected person-rem exposures would not be much larger than those accrued by the normal eddy current testing, even if the expansion were to take place later during operation of the plant when radiation fields have stabilized at near-equilibrium levels. As indicated in the literature, the equilibrium levels may be roughly double those expected at Krsko, normally occurring near the fourth year of operation. The TRC estimates that for a single tube expansion, radiation exposures would be in the range of 20 to 40 person-rem per steam generator. Approximately half of the exposure would be due to performing the eddy current test before and after expansion. Additional numbers of expanded tubes, if few and in close proximity, would not increase the exposure significantly.

Studies conducted by Westinghouse on the expansion of tubes within plate holes have indicated that the expansion process can be successfully carried out under conditions that simulate those that might be expected following a significant period of operation of the plant. These studies included expansions intentionally offset or tilted with respect to the baffle plate and in plates with the gap partially packed to simulate corroded baffle plates. Westinghouse has reported the results in their report on the expansion process for scarred tubes, and the TRC finds no evidence that the expansion process could not be implemented successfully after operation. There is no insurmountable radiological impediment that would prevent expansion of tubes after several years of operation, if that need arises.

5. EVALUATION

Westinghouse has completed a comprehensive development program in the area of thermal-hydraulics and vibration. The program included a series of model tests, analytical investigations, and full-scale data from the Krsko plant. A more detailed description of Westinghouse's work in this area follows.

5.1. DESCRIPTION OF TESTING

A large number of full-scale and model tests were used by Westinghouse for the evaluation of the flow and vibratory conditions of the original designs and of the proposed modifications. These tests included:

1. Krsko D4 steam generator, full scale.
2. D4 and E air models (0.95 scale).
3. D4 and E single tube models with exciter.
4. D4 installation model.
5. D4 1/4-scale water model.
6. D4 16-degree full-scale water model.
7. D4 2/3-scale water model.

In addition to full-scale wear data, Krsko also provided vibration and thermal-hydraulic data. The 2/3-scale and the 16-degree full-scale models gave the most valuable information regarding flow velocities, fluid forces, and tube vibration. The air model gave inlet and third pass flow velocities, and the single tube model was used to study tube dynamics.

5.1.1. Tests at Krsko

The testing at the Krsko two-loop D4 steam generator plant was accomplished in three phases. During Phase I (February-May 1982), vibration data were obtained prior to feedwater system modifications by installing two accelerometers in each of four tubes at different elevations. Figure 5.1-1 shows the locations of these accelerometers and of additional accelerometers employed in the subsequent phases.

At the end of Phase I, a tube (R49 C56) was removed from the steam generator for direct wear measurement. Examination of the tube showed wear at plates B, D, and G, the major wear being at plate B with a local wall thinning of 2.5 mils (6%).

During Phase II (July 1982), vibration data were obtained following the modification to the feedwater system. In this phase additional accelerometers were installed, as shown in Fig. 5.1-1. Moreover, thermal-hydraulic data were obtained by placing thermocouples in the fourth pass and downcomer and pressure transducers in the inlet pass, downcomer, and inlet box.

Phase III (November-December 1982) provided vibration data at additional tube locations, wear data on two additional removed tubes (R46 C56, R49 C35), eddy current testing of 900 tubes in each generator, and tube-to-baffle plate gap measurements in 64 tubes in steam generator 1 and four tubes in steam generator 2 at plates B and D. The wear data obtained from the three tubes at Krsko were used to establish the coefficients for prediction of wear in the D4/D5 and E model steam generators.

During Phase III testing, one tube (R48 C55) in steam generator 2 was hydraulically expanded at both the B and D baffle plate locations. Vibration data obtained on this tube have indicated that the expansion process further lowers the G-Delta levels by a factor which varies depending on main nozzle flow.

On the basis of these results, Westinghouse concluded that the feedwater flow into the Krsko steam generators through the main nozzle should be limited to 70% of full flow to reduce the possibility of early wear on the

a,c,e

Fig. 5.1-1. Krsko field test accelerometer locations tubes. Prior to the modification necessary to reduce main feed flow to 70% and still operate at full power, Krsko operated for about 1500 hours at 75% power (5% flow through the auxiliary nozzle).

The 70% limitation on main feedwater flow at Krsko was obtained by feeding the remaining 30% through the auxiliary nozzle. This necessitated the installation of new control valves and new bypass piping to reduce the hydraulic resistance to the feedwater in the bypass system. Subsequent successful operations at 70%/30% split flow indicate that somewhat higher vibration levels occurred in certain tubes than had been observed with just 70% flow through the main nozzle and no auxiliary nozzle flow. The increase in vibration is attributed by Westinghouse primarily to the difference in feedwater temperature at 70%/30% flow and at 70%/0% flow, causing different thermal expansion between the tubes and the baffle plates.

With full feedwater flow through the main nozzle, the Krsko G-Delta values were comparable to and in excess of those for the Almaraz and the Swedish State Power Board (SSPB) full-scale test facility in the D2/D3 program (NUREG-0966, March 1983).

Comparison of G-Delta values obtained for full main nozzle flow and 70%/30% nozzle flow has shown that the vibration levels have been reduced, with some tubes exhibiting larger and some smaller reductions. The removed tubes have shown good consistency in wear data as related to the G-Delta values obtained from accelerometer data. On the basis of conservative values of wear coefficients developed from the Krsko data and the G-Delta levels that have been measured and are predicted following feedwater split modification, there is considerable confidence that wear resulting from the lower vibration levels expected after modification will not exceed the limit established by the acceptance criteria given in Section 2.1. Any significant wear will be identified during periodic in-service inspection.

Westinghouse has reached the following major conclusions from the available data:

1. High vibration levels existed in the T-slot area but also in areas away from the T-slot.

2. The wear information on the three removed tubes was consistent and is a good basis for wear prediction.

3. There were no eddy current indications, which means there was no tube wear in excess of 10% of wall thickness.

4. No clear correlation has been established between the tube-to-baffle-plate gap sizes and tube vibration responses.

5. Tube expansion significantly reduced vibration levels.

The Krsko plant has been operating at full power with the 70%/30% split feedwater flow without significant operational problems that can be ascribed to the feedwater flow modification.

5.1.2. Various Model Tests

5.1.2.1. Air Model. There are two air models. Both are 0.95-scale, 90-degree segment models simulating three and five cross-passes of the preheater of the D4/D5 and E models, respectively. The air models accurately simulate the inlet conditions with the venturi nozzle, the inlet box with the ribs and impingement plate, and the cap plate. They include the flow breakers (flow restrictors) situated on the tube bundle periphery. Figure 5.1-2 shows the air model configuration.

The air models were used for measuring tube surface velocities in gaps between tubes using velocity probes on a large number of tubes in the first

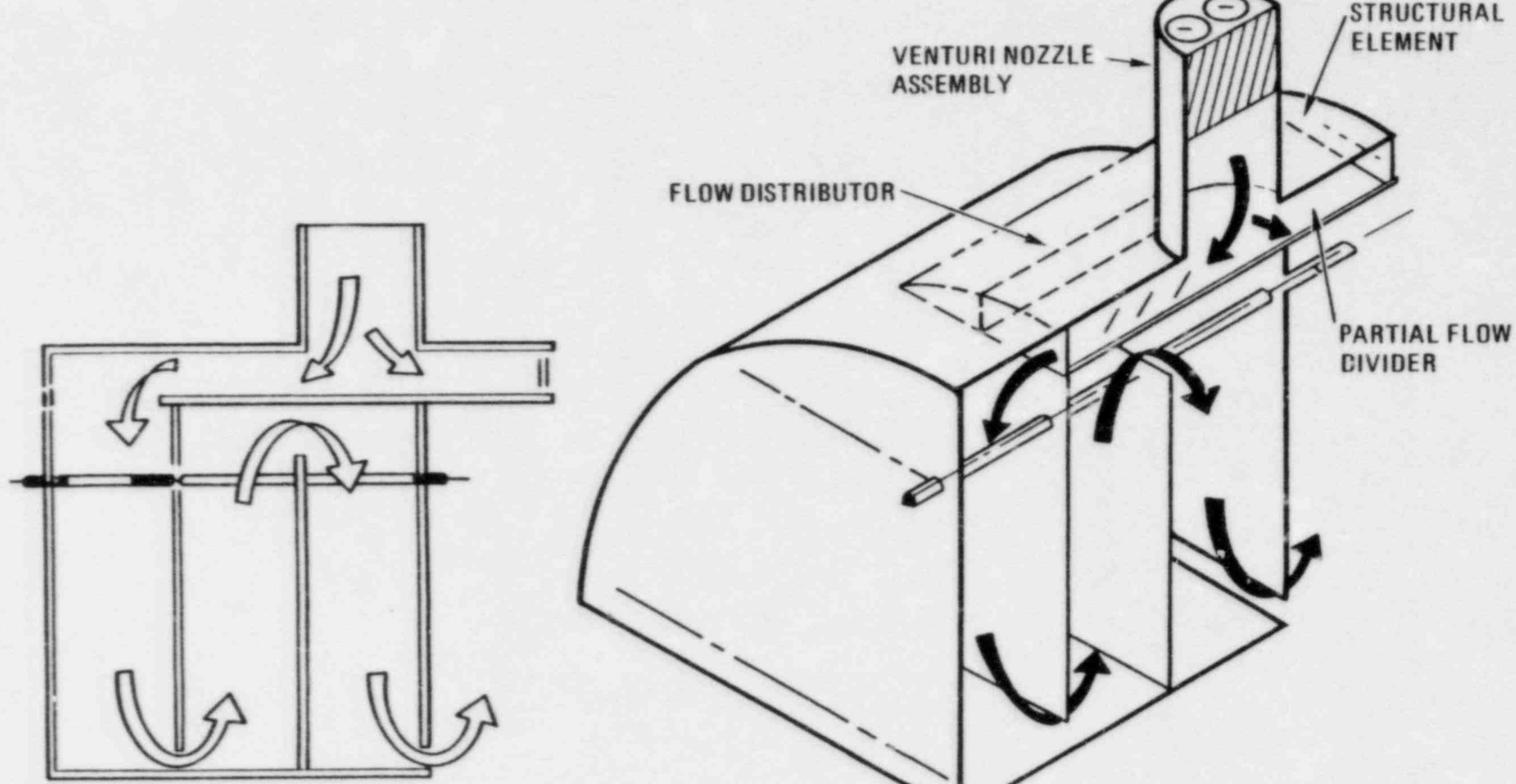


Fig. 5.1-2. 0.95-scale D4 model

and third cross-passes. A total of 42 Model D4, six Model D5, and three Model E configurations were tested. The models provided gap velocity information for evaluating modifications.

5.1.2.2. Single Tube Model. The single tube model is a structural model simulating a "loosely" supported tube at the baffle plates for the Model D4/D5 and E steam generators. The model is used to study the general behavior of a "loosely" supported tube. It is also used specifically for comparative studies of Model D4/D5 and E configurations and for evaluation of special or trimmed baffle plate configurations.

The tube is excited by applying a random forcing function normally at midspan between plates B and D (in the inlet pass). A piezoelectric force transducer is placed in series between the shaker and the tube to measure the force acting upon the tube.

Two accelerometers are used to measure tube vibration response. The accelerometers [one positioned in-line with the shaker motion (X-direction) and another perpendicular to it (Y-direction)] were placed midway between plates D and E in the flow windows. For Model E, this elevation corresponds to the midspan location; for the D4/D5 configuration, it is at the quarter-span location. The output signals were processed to obtain RMS force, RMS acceleration, and RMS displacement over a specified frequency range.

5.1.2.3. Installation Model. A simulation of the general geometric configuration of the preheater inlet region and primary tubeside inlet region is provided for developing skills and testing of alternative modifications. The qualification process/tube expansion criteria given in Section 2.4.2 were met during the installation model tests.

5.1.2.4. 1/4-Scale Water Model. This model represents the full preheater region (A to K baffle). It is a 180-degree sector model. The model was used for flow visualization studies (by air injection), obtaining gap velocity data, and fluctuating velocity measurements. Several configurations were tested, including a simulation of a 90-degree sector.

5.1.2.5. 2/3-Scale 180-Degree Water Model. This model simulates the feedwater inlet region between plates B and D, including the leakage through plate B and upflow at the back of plate D, by a specially designed outlet manifold with valving. The inlet piping configuration with the reverse flow limiter and the water box are an integral part of this model. A full complement of "rigid" cylinders simulates the tube bundle in the inlet pass.

This model is used for studies of flow velocities and fluid forces acting upon the tubes. Special instrumentation consisting of velocity probes and load-sensing cells was used for flow velocity measurements and measurements of static and dynamic forces acting upon the tubes.

The 2/3-scale model is one of the more important models providing information for detailed velocity maps and tube loading spectra that are used in the vibration and wear evaluations in conjunction with the nonlinear tube vibration model described later. This model is shown in Fig. 5.1-3.

5.1.2.6 Sixteen-Degree Full-Scale Water Model. The 16-degree model is a full-scale representation of the Model D4 preheater section up to plate L. The inlet section consists of one-half of the water box and a 90-degree section of the inlet region. The additional passes are represented by the middle portion of the tube bundle with a section 16 tubes wide (tubes within a 16-degree cut-out) from the centerline of the bundle. Valving is provided proportioning the outflow below plate B and the flow to the upper crosspasses. The entire structure is supported on a separate foundation block, dynamically isolated from the surrounding structures. Dampers in the flow path are provided to minimize pressure surges in the flow stream. Flowmeters and pressure transducers for flow parameter measurements are inserted at the inlet to the tube bundle. The 16-degree model is shown in Fig. 5.1-4.

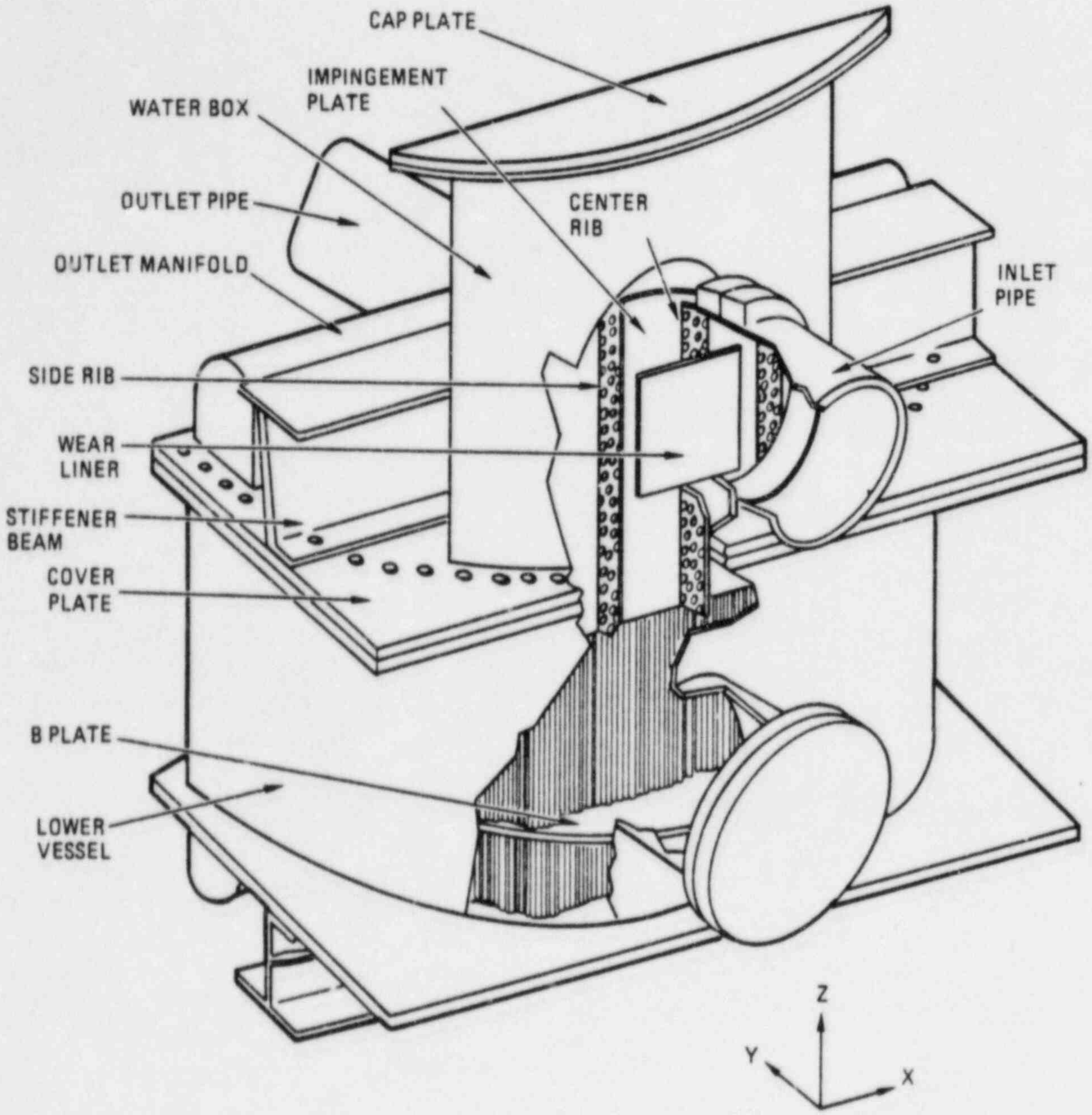


Fig. 5.1-3. Isometric view of 2/3-scale model

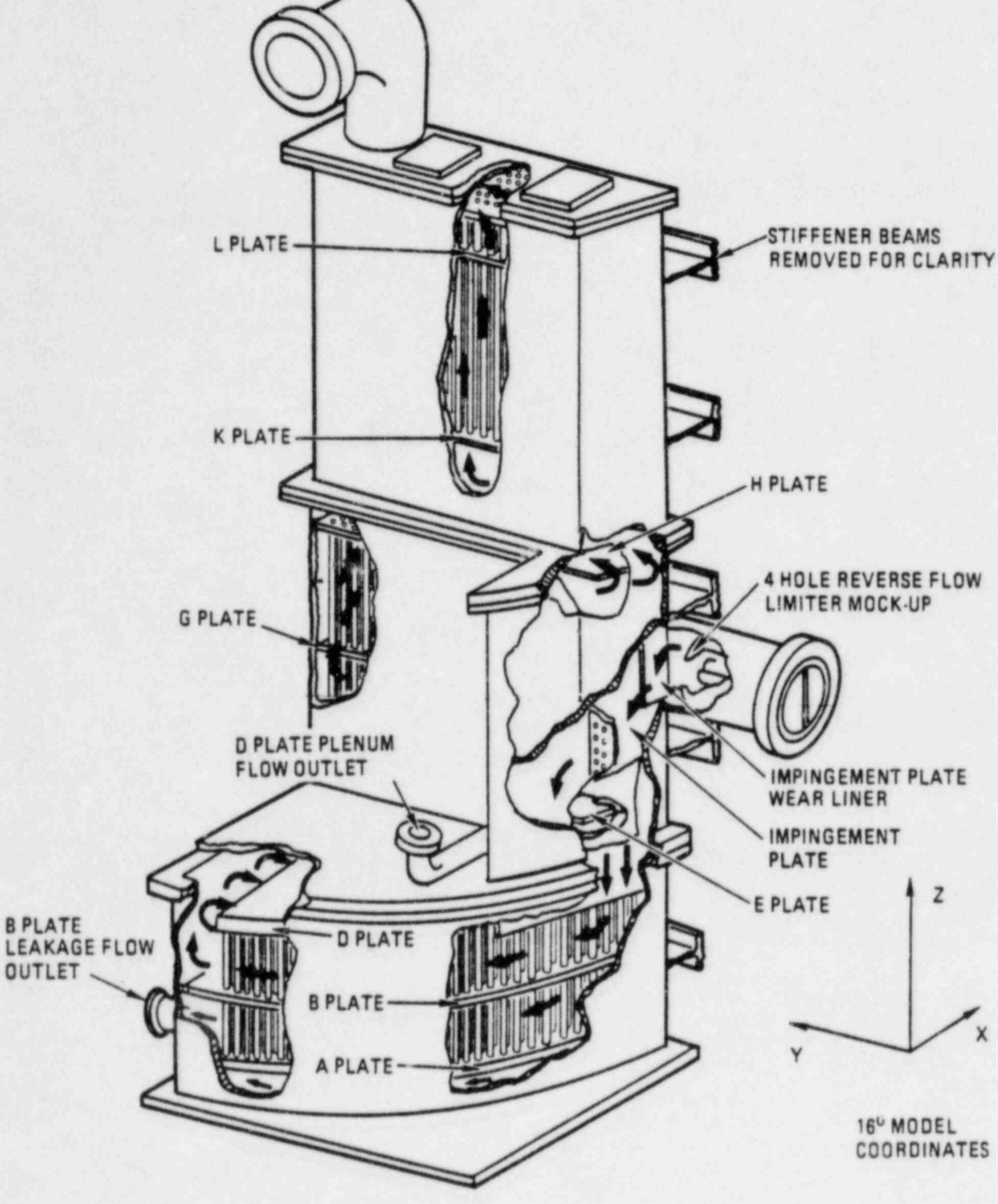


Fig. 5.1-4. Sixteen-degree full-scale model

The prime purpose of this model is tube vibration testing under the conditions of full exposure of tubes to flow at simulated tube support conditions.

All baffle plates can be repositioned biaxially to simulate operating and as-built conditions with prototypic baffle plate hole diameters. Principal data produced are tube vibration responses in two directions at several elevations (with the majority of data taken at midspan between plates B and D and at plate E).

Provisions have been made for installation of gap velocity probes and tube-to-baffle plate impact gages. If long-term reliability studies are needed, the TRC believes that this instrumentation could be utilized.

The two major test modifications, i.e., the split flow effect (with bypass flow through the auxiliary nozzle) and the effect of tube expansion against the baffle plates, were tested. The goal was to establish the extent of the higher-vibration region at full flow and at split flow conditions in order to isolate those tubes that will have to be expanded. A large number of plate shifts were utilized in this evaluation. The vibratory behavior of the expanded tubes was also tested extensively.

5.2. RESULTS OF TESTING AND VIBRATION ANALYSIS

The results of the thermal-hydraulic and model tests and analysis can be grouped into the following categories:

1. Flow velocity and fluid forces.
2. Tube vibration.
3. Vibration forcing mechanisms.
4. Computer single tube vibration model.
5. Tube wear prediction.
6. Proposed tube expansion.

Each of these categories is discussed in this section.

5.2.1. Flow Velocities and Fluid Forces

Flow velocity information for the D4/D5 configuration was obtained primarily from the 2/3-scale water model, complemented by test results on the 0.95-scale air model, the 16-degree model, and the 1/4-scale water model. The 2/3-scale model also provided information on fluid forces exerted upon tubes. All these results addressed only the conditions in the first (inlet) pass. Flow velocities for the Model E steam generator were generated in the 0.95-scale air model. This model also provided flow velocity profiles in the first and third passes.

The highest flow velocities in the gaps between tubes occur in the center of the tube bundle in the area of the T-slot, reaching the following values in the first pass:

	Model D4/D5		Model E	
	Velocity (2/3-Scale Data)		Velocity (0.95-Scale Data)	
Flow (%)	Maximum V_{avg} (ft/sec)	Peak (ft/sec)	Maximum V_{avg} (ft/sec)	Peak (ft/sec)
100	[			]b,c

The term "Maximum V_{avg}" refers to the maximum average gap velocity obtained by averaging the velocity distribution in the tube gaps between baffle plates B and D and selecting the largest average gap velocity over all gaps. The term "peak velocity" is then the largest point velocity measured. For the Model D4/D5, based on the air tests, the upper pass velocities are significantly lower. The velocities in the inlet pass, away from the T-slot, are lower, leveling off on the sides of the front tube row. For the Model E, based on the air tests, the upper pass velocities are higher than for the Model D4/D5. The side-to-side velocity distribution in the inlet pass of the Model E is rather uniform.

For both Model D4/D5 and Model E steam generators, there is a large variation of flow velocities between plates B and D in the vertical direction. The 2/3-scale and 16-degree model test results show that the flow field in front of the outer tube row is complex with significant vertical and sweeping components as well as a region with some reverse flow.

The turbulent (time varying) forces in the inlet pass are relatively small. For 18% bypass flow for the two- and three-loop plants or 10% bypass flow for four-loop plants of the D4/D5 types or the E type, the highest values range over the tube span between plates B and D.

For Model E, additional information on the turbulent (alternating) forces acting upon the tubes in the upper passes (passes 2 through 5) is being prepared by Westinghouse for TRC review of the long-term reliability performance of expanded tubes.

5.2.2. Tube Vibration Results and Their Interpretations

Tube vibration was measured on a total of 16 instrumented tubes in both steam generators in the Krsko plant. Tube vibration was also measured in the 16-degree test model at a total of 53 instrumented tube locations. The instrumented tubes supported two accelerometers, which sensed vibratory accelerations in two perpendicular directions. The typical locations of the accelerometers were at midspan between plates B and D, with fewer accelerometers located at the elevation of plate E.

The processed data included RMS accelerations, G_{RMS}, and corresponding RMS displacements, $Delta_{RMS}$. In addition to the RMS data, peak-to-peak accelerations, G_{p-p}, were obtained from real time data.

The acceleration response as a function of power for one unmodified tube in the Krsko steam generators, as presented by Westinghouse, exhibited a response similar to responses observed in some of the Model D3 tubes.

The 16-degree model closely simulated the results of the Krsko vibration measurements on a number of tubes. By shifting baffle plates (several hundred plate shifts were made), a close simulation of the first set of measurements at Krsko (Phase I measurements) was achieved in a reference plate position that Westinghouse identified as the 5R3 baseline position. (Tube R49 C56 was used as the basis for this simulation.) The 5R3 plate configuration was used for evaluation of several design concepts. It was found that the vibration results were very sensitive to baffle plate position.

Westinghouse stated that moving the lower plates also affected the tube to plate contact points at the upper plates through repositioning of the entire tube, which made moving the upper plates unnecessary. To simulate the behavior of the entire tube bundle (cold and hot leg) in operation, relative to the plate and tube movements, Westinghouse has used a computer structural model as a guide for the predictions of plate positions.

An empirical relationship has been established between the G-Delta values and flow velocity from the Krsko data. A similar expression was produced using the 16-degree model test data. For long-term reliability concerns, the TRC believes that a similar relationship should be established for the higher-frequency tubes. Westinghouse is in the process of evaluating the need for such a relationship.

The effect of bypass flow through the auxiliary nozzle has also been tested. A 20% and 30% bypass in the Model D4 steam generator leads to average reductions of G-Delta values. These results are a reflection of the very strong dependence of the G-Delta levels on flow velocity.

Using the single tube vibration model in conjunction with the turbulence force data in the first pass derived from the 2/3-scale test model, a relationship between the RMS force and the G-Delta values has been established, again by a curve fitting procedure. The derived relationships between G-Delta and force are limited to correlations established for low-frequency responses to inlet pass velocity flow fields only. The appropriateness of describing the vibratory forcing function solely by turbulence depends on the degree to which fluidelasticity is present.

It is to be noted that the type of tube vibratory motion, i.e., whether the tube is impacting upon the baffle plate while vibrating or only smoothly sliding, was established by visual inspection of the time-history traces, as was the peak-to-peak acceleration. The TRC believes that the processing of the results within the low-frequency range is reasonable, since the higher frequencies do not affect the RMS displacements significantly, while the peak-to-peak values of acceleration reflect all the frequency components.

The effect of the reductions of the tube-to-baffle plate clearances has also been tested in the 16-degree test setup. The reduction of clearances was achieved by tube expansion in the area of the baffle plates. Tube expansion to a minimum diametral clearance was found most effective. Expansion at plates B and D was determined in two tests. In the first test (TE-1), three tubes were expanded (R49 C56, R48 C55, and R48 C53). The second test (TE-2) included 24 expanded tubes (rows 46 to 49 and columns 51 to 56).

The frequency response of the expanded tubes differs significantly from that of the unexpanded tubes. In general, the first set of data (test TE-1) shows a relatively small reduction of vibration parameters, while the second test of 24 tubes (test TE-2) shows a substantial reduction. The small reductions in the first test were explained by Westinghouse as reflecting the vibration of neighboring tubes.

In the second test the displacements for the expanded tubes dropped, as did the average peak-to-peak accelerations. The computed G-Delta values dropped from their original values in the 16-degree test model. At power levels in excess of 70%, the one instrumented and expanded tube at Krsko showed a reduction in the RMS displacements, in peak-to-peak accelerations (or impact accelerations), and in G-Delta values.

Inspection of the individual data reveals that although significant reductions of vibration levels do occur on the average, in some individual cases the reductions are less pronounced. Also, the low frequency is still present in some tubes. Westinghouse maintains that isolated instances of expanded tubes vibrating in a low-frequency mode can be expected because of the small clearance between the tube and baffle plate. The TRC believes that an additional possible cause of these vibrations is the result of vibration of the unexpanded portions of the tube. Westinghouse also notes that while the vibration in the unexpanded tube regions may not have been reduced, it certainly has not been increased owing to expansion at plates B and D.

Westinghouse also evaluated the effect of the upper passes on the vibration. Based on extensive plate shift searches with the 16-degree model, Westinghouse concluded that the upper passes have negligible effect on the vibration of Model D4/D5. Based on TRC independent analytical evaluation, the results have indicated higher velocities occur in the upper passes in Model E than in Model D4/D5. This is confirmed by tests in the 0.95-scale air models. The TRC will continue to follow the Westinghouse evaluation of the impact of the upper pass velocity profiles on the long-term reliability of the Model E.

5.2.3. Tube Vibration Forcing Mechanisms

The dominant vibration excitation mechanism in the counterflow steam generator designs, similar to the split flow D2/D3 designs, is considered to be turbulence or turbulent buffeting in the inlet pass. This conclusion was reached based on analysis of the response spectra of the measurements taken at the operating plants of the Model D2/D3 steam generators and also of the Krsko data. It was further confirmed by matching the wear pattern of tubes pulled from operating steam generators with the predicted vibratory motion of a non-linear tube model excited by turbulence excitations. This was done as follows: from the 2/3-scale model tests, an equivalent forcing function acting upon the tube along its length in the inlet pass was established. Using a non-linear tube model, tube motions were predicted at several points along the tubes. Since these motions were consistent with the wear pattern of removed tubes, it was possible to determine the type of excitation the tubes receive.

According to Westinghouse, a review of the experimental data obtained from the Krsko steam generators, as well as the data from the 16-degree model testing (including the large number of cases generated during the plate shifts and overloads), does not indicate a continuous existence of the fluidelastic mechanism. Bursts of sinusoidal motions, or orbiting type motions, were observed pointing toward a possible intermittent fluidelastic mechanism affecting some tubes at certain flow and support conditions. A sustained fluidelastic mechanism was not visible, at least not within the range of the available test data. Overload simulations in the 16-degree model of up to 110% to 120% of the equivalent full flow were generated to test for instabilities. No signs of disproportionately large increases in vibration were detected.

Based on all the test results available from the D2/D3 and D4/D5/E programs, the TRC believes that the fluidelastic mechanism is probably present although not at sufficient amplitudes for it to be clearly distinguished from turbulence-induced vibration. The indeterminate nature of the support condition further increases the difficulty of distinguishing between the turbulence and fluidelastic mechanisms.

5.2.4. Computer Single Tube Vibration Model

A three-dimensional, non-linear model of a single tube from the tube sheet up to the U-bend has been developed with the Westinghouse WECAN computer program. This model is multispan and utilizes annular gap elements at the baffle plate locations simulating the tube-to-baffle plate clearances and permitting orbital tube motion. Baffle plate offsets can also be modeled.

The output is in the form of displacement time-histories from which response spectra for analysis of frequency compositions, RMS accelerations, RMS displacements, and G-Delta values can be generated. The generated time-histories also serve as a basis for computing the integrated tube-to-baffle plate contact forces times travel distance per unit time, which are then used to determine wear rates by the work rate method.

5.2.5. Tube Wear Prediction Methods

5.2.5.1. Work Rate Method. Wear prediction is based on the Archard relationship between work (product of normal force and sliding distance) and wear volume. The wear coefficient, an empirical constant reflecting the type of wear and the materials of the contacting surfaces, relates the expanded work to the wear volume.

In order to incorporate the time-history and the total wear time aspects, individual work rates performed within time intervals are integrated and total work is obtained as the product of the integrated work rates and total time. The force and displacement time-histories required for this integration were obtained from the output of the three-dimensional non-linear computer model discussed in Section 5.2.4.

Application of the work rate method depends on a relationship between work and wear. Wear rates, wear scar volume, and wear depth were established experimentally from Krsko and other full-scale wear data and from AECL model tests. It was found that the type of relative motion between the two interacting surfaces defines two different wear coefficients. The test data were also used to establish a relationship between wear scar depth and wear volume.

Because the work rate method requires information on the details of magnitude and time-history of tube-baffle impact and interaction forces, it is difficult to apply this method to a large number of tubes. To overcome this problem Westinghouse has chosen to utilize the G-Delta method as the primary wear assessment method in the D4/D5 and E Model Modification Program. The work rate method was used for confirmation.

5.2.5.2. G-Delta Method. The G-Delta method calculates the total wear volume as the product of the wear coefficients, total wear time, and G-Delta. The wear coefficient is numerically different from that used in the work rate method. The numerical value of the wear coefficient based on the G-Delta method was extracted from the three removed tubes at Krsko. These tubes exhibited wear scars at plates B, D, and G, with maximum penetrations into the tube wall 2.5 mils at plate B, 1 mil at plate D, and 1.5 mils at plate G. The nominal and worst case values of the wear coefficient were extracted from these worn tubes.

Maximum single scar volume, V_S, versus total wear volume, V_T, relationships were developed using the non-linear single tube vibration program for the Model D4/D5 steam generators as follows:

b,c,e

Design basis: [ ]

Upper bound:

where V_S and V_T have units of 10^{-4} cubic inches.

Based on the single scar volume to scar depth relationship, the following single scar wear volumes could be tolerated:

	Unexpanded	Expanded	
Design basis* (40%):	[	]	b,c,e
Safety limit (65%):			

*Tube plugging limit.

Based on backup calculations, using the above-described relationships, Westinghouse has shown predicted design basis wear depths on the pulled tubes to be greater by a minimum of 20% than those actually measured, with the worst case giving a minimum of 2.5 times deeper wear scars than those actually measured. Thus, there is a built-in conservatism in the prediction method.

5.2.6. Model E Design Differences

The Model E steam generators have a larger shell diameter and operate at higher feedwater flow rates. Although the flow path in the preheater is similar, it is not identical. The major difference between the Model E and D4/D5 steam generators is that there are no window tubes, plates E and H provide additional supports, and thus all tubes pass through the same number of baffle plates.

Westinghouse considered all the design parameters of E models as more favorable relative to vibration than the equivalent parameters of the D4/D5 models (Section 3.1.4). Evidence to support this contention was based on the 0.95-scale air model tests, the single tube structural vibration model, and analytic predictions using the non-linear vibration model. Test data from the non-window tubes in the 16-degree model were also used for this comparison.

The flow regime in the front tube rows will be different from those of the D4/D5 configuration, and air test data have shown that velocities in the upper passes will be higher than those experienced in Model D4/D5. There are also specific structural differences in the various E models.

Because the major flow testing and vibration work completed by Westinghouse have been evaluated using the D4/D5 configuration in the 16-degree model, the TRC undertook an independent flow and vibration analysis for Model E. This analysis has been completed and conveyed to Westinghouse. The TRC will continue the evaluation of Westinghouse's responses regarding the impact of these results on the Model E long-term reliability.

5.2.7. Map of Proposed Tube Expansions

Maps of proposed tube expansions are model specific. An example of a typical Model D4 tube expansion map is shown in Fig. 3.2-1.

5.3. EVALUATION OF WEAR ANALYSIS AND TESTING

5.3.1. General Considerations

The G-Delta method is the primary wear assessment method, while the work rate method is employed to provide independent confirmation. Both methods employ a nominal procedure used for the design modification evaluation and a conservative one used for safety analysis. Confidence in using these methods is enhanced by their validation in the D2/D3 evaluation program.

The G-Delta method employs values either derived from direct experimental data or calculated from velocity correlations. The appropriateness of the G-Delta prediction procedure is unquestionable when it is based on direct experimental data. In the counterflow steam generators, however, data on worn tubes are very limited, both from the extent of wear and the distribution of wear aspects. This limited data base, although reinforced by the availability of vibration data obtained directly on the worn tubes, must be treated with care. Conservatism has been employed to compensate for the small amount of experimental data.

While the procedure used for wear prediction in cases where direct experimental data are unavailable is also plausible, there are some areas of uncertainty, which are discussed more fully in the following sections. The TRC agrees that these uncertainties have no safety implications. Provisions for in-plant periodic inspection during normal ISI and possible analysis of tube wear are viable means of quantifying such uncertainties.

5.3.2. Evaluation of Westinghouse Acceptance Criteria

5.3.2.1. G-Delta Assessment Method. The data upon which the G-Delta method is formulated are experimental data acquired from steam generators 1 and 2 at the Krsko plant and the full-scale 16-degree model. The output of the accelerometers was used to determine four parameters of importance in evaluating tube vibration:

1. Tube response frequencies. Free tube span lengths are related to particular response frequencies. Vibration data indicate that the highest vibration levels are present for a particular dominant response frequency that varies with tube configuration (i.e., expanded/non-expanded, window/non-window).

   Expansion of window tubes at baffle plates B and D would have the effect of reducing the unsupported span length to that between plates D and G and increasing the response frequency. Expansion of non-window tubes would introduce definite supports at B and D and minimize the likelihood of tube response at low frequencies.

2. Peak-to-peak accelerations. The peak-to-peak acceleration value is obtained by visual observation of the obtained time-history records.

3. Root mean square displacement. This displacement is computed from measured tube acceleration signals. An RMS displacement spectrum is obtained by double integration within the frequency range.

4. G-Delta. The G-Delta parameter is a value that has been correlated with wear. A distinct feature of this method is that since the G-Delta values are obtained directly from experiments (either at Krsko or the 16-degree model), they reflect response due to turbulence and/or fluidelastic excitation. Thus, as long as G-Delta values obtained from experiment are used, correct identification of the nature of the exciting force is strictly not essential. This is a limitation in the work rate method discussed in Section 5.3.2.2, which utilizes only turbulent buffeting as the excitation force and hence would be correct only if fluidelastic excitation were absent or minimal.

To use G-Delta values for wear prediction, Westinghouse employs the following equation:

$$V_T = K_g \text{ (G-Delta) } T,$$

which is based on Archard's wear volume-work relationship. T is the time period over which wear occurs. This equation correlates G-Delta values with the total wear volume, V_T, via a coefficient K_g that was determined from examination of the three tubes removed from Krsko. The wear coefficients, K_g, determined from the Krsko test data are given in Table 5.3-1. There is good agreement between the three values of K_g.

Since acceptable wear is ultimately determined by the maximum single scar depth, it is necessary to relate the total wear volume predicted by the G-Delta method to a maximum single scar volume, which in turn can be related to maximum single scar depth by geometric considerations.

The relationship between single scar volume and wear depth is based on geometric analysis and correlation with model field data. Westinghouse developed a geometric model that computes maximum wear depth versus single scar volume with inclination of the tube relative to the baffle plate as a parameter. As shown in Fig. 5.3-1, analyses conducted for small angles of inclination closely approximate the best fit curve derived from Model D2/D3 steam generator data. The Krsko data on wear are also shown in Fig. 5.3-1. The maximum scar volume observed at Krsko was about [ ]a,b,c,e for a scar depth of 2.5 mils (~6% of wall thickness).

Because of the sensitivity of tube response to baffle plate position, the baffle plates in the 16-degree model were positioned through a series of plate searches until the maximum response at each tube location was found. By the plate search procedure, the maximum vibration levels realizable at each tube accelerometer location are obtained. This leads to a "worst case" baffle plate configuration for tubes of interest. The soundness of the plate search approach was verified by attempting to reproduce in the

TABLE 5.3-1

G-DELTA WEAR COEFFICIENT (K_g) FROM KRSKO REMOVED TUBE DATA

$$V_{wear} = K_g \Sigma (\text{G-Delta})_i T_i$$

	R49 C56	R46 C56	R49 C35	
Removed Tube Data				
Total Volume (in.^{3})				b,c,e
Max. Single Scar/Total Vol.				
Scar Depth (mils) at:				
B				
D				
G				
Acceleration and History				
(100/0)/(70/30) G-Delta				
$\Sigma (\text{G-Delta})_i T_i$				
Phase I				
Phase II				
Total				
Fitted Wear Coefficient				
K_g				
Average K_g				
Worst Case K_g				
G-Delta Predictions - Scar Depth (mils)				
Nominal				
Worst Case				

a,b,c,e

Fig. 5.3-1. Relationship between single scar volume and wear depth

16-degree model the vibration levels experienced at Krsko by moving plates B and D. Table 5.3-2 shows the good agreement obtained.

Westinghouse established a correlation between measured G-Delta values and corresponding tube gap velocity data and fluctuating force data using data from the 16-degree and 2/3-scale models (see Section 5.2.2).

The G-Delta values predicted for those tubes outside the bounds of the 16-degree model have an element of uncertainty that is not present in those measured directly in the 16-degree model through the plate searches. This is due to the fact that the fluctuating forces and gap velocities at the inlet pass are measured in the 2/3-scale model but not in the 16-degree model, and it is tacitly assumed that the fluid forces would be the same in the two models. In turn, this implies that:

1. Fluidelastic excitation is absent or negligible.

2. The excitation forces in the upper passes are negligible in comparison with the inlet pass.

Westinghouse conducted a number of plate searches that involved movement of plates B and D after tube expansion at B and D. Movement of the B and D plates also affects the support conditions at the upper plates through movement of the entire tube. During this process, Westinghouse did not find evidence of an increased vibration response in areas away from the expanded region.

It is an objective of Westinghouse to minimize the potential for fluidelastic vibration within the scope of the proposed modification. Westinghouse maintains that their application of the G-Delta method to the proposed modification implicitly satisfies their initial objective to minimize fluidelastic vibration. The TRC concurs with this approach for cases where G-Delta values are based on direct experimental data.

TABLE 5.3-2
COMPARISON OF 16-DEGREE MODEL AND KRSKO BASE TEST RESULTS

b,c,e

In such instances the TRC believes that a wear correlation based on G-Delta provides adequate conservatism for fluidelastic instability since this correlation was established using full-scale 16-degree testing and Krsko data. Even if not clearly distinguished, the fluidelastic mechanism is bounded by the test data since, if present, it is inherently included in the correlation.

Where direct experimental data are not available, Westinghouse has conservatively calculated (and the TRC has independently verified) stability ratios for the different support conditions that could occur, using the non-linear model. In general, stability ratios are less than unity except for a few tubes. For such instances, the TRC has determined that stability ratios above unity do not present a safety problem since G-Delta values for tubes having comparable or larger stability ratios will not result in wear exceeding the safety limit of 65% of the wall thickness in 18 equivalent full-power months of operation.

With the reservations identified above, the TRC has accepted Westinghouse's approach that extrapolates G-Delta values to tubes where those G-Delta values cannot be directly measured. Further, the TRC concurs with the validity of using the G-Delta method as a means to predict wear in tubes and to assess the effects of the steam generator modifications. Further validation of the soundness of the G-Delta method as a wear predictor was obtained by comparison of wear data from the Model D2/D3 steam generators at Ringhals and Almaraz. The wear predicted using the G-Delta method is generally greater than the observed wear volumes.

5.3.2.2. Work Rate Method. The work rate method is the second method employed by Westinghouse to assess the wear caused by vibration and thus judge the acceptability of the modification. This method was previously used by Westinghouse in predicting wear rates in the Model D2/D3 steam generators.

The work rate method is predicated on an analytical non-linear model of a single tube from the tube sheet up to the U-bend previously described in Section 5.2.4. The wear evaluation is performed using Archard's wear equation, just as in the G-Delta wear evaluation. The total work performed is given by the product of the total work rate and the total time T. The total wear is given by the product of the total work and the appropriate wear coefficient.

Once the total wear volume has been obtained, predictions of the lifetime to plugging of tubes require the same correlations between maximum single scar volume and total wear volume and between maximum scar depth and single scar volume that are required in the application of the G-Delta method.

Verification of the work rate method had been done as part of the effort on the Model D2/D3 steam generators. The independent review team that reviewed this method found it acceptable for modeling and bounding tube wear.

Unlike the Model D2/D3 case, in which this method provided the main evaluation tool, in this study its wear prediction capability is a secondary objective. It was used to provide confirmatory information regarding the extent of the tube expansion zone in the bundle determined by the G-Delta method and to provide supporting information regarding the effect of bypassing flow to the auxiliary nozzle in the Model D4/D5 steam generators. The primary functions of the work rate method in this program were to demonstrate the effectiveness of tube expansion as a means of controlling tube vibration prior to tube expansion in the 16-degree model and to develop the relationship between the total wear volume in a tube and the single maximum wear scar volume. The work rate method is used to estimate the wear times to 40% depth for expanding tubes due to the long-term changes in expanded tubes as the tube wears.

Finally, the method is used to provide information relative to the effect of the Model E baffle plate support and hence the wear characteristics of the Model E steam generator tubes.

When applied to the Krsko wear data, the model predicted reasonable agreement with scar location and wear volume values. This is consistent with observations made for the Model D2/D3 steam generator evaluation. Moreover, predicted lifetimes of tubes with this model are shorter or equivalent to those predicted by the G-Delta method, which is used as the design basis method for unexpanded tubes.

5.3.2.3. Application of Wear Methods. Two wear prediction procedures are used: (1) the nominal (design basis) procedure and (2) the safety analysis procedure. The safety analysis approach includes the following conservative steps:

1. Use worst case wear coefficient.
2. Use upper bound of wear distribution factor.
3. The work rate model assumes no contact between the tube and any of the baffle plates as an initial condition for vibration predictions (only imminent contact is contemplated). The work rate model is further discussed in Section 5.3.2.2.

It is of interest to note that the design basis calculations also include conservatisms. Among the most obvious is that they are based on the worst case G-Delta results obtained from plate searches. However, at the same time the calculations have some limitations:

1. For extrapolation of G-Delta values outside of the 16-degree model (columns < 42), a correlation of G-Delta to turbulent force obtained from inlet pass excitation is used.

2. Because of a lack of experimental wear distribution data for the Model D4/D5 steam generators, the wear distribution factor has not

been validated directly. However, theoretically derived upper bound wear relationships are being used.

3. The effect of fluidelastic type excitation may not be fully bounded by the present method where wear predictions are based on theoretical calculations (non-linear model).

The safety criterion is that in a period of 18 equivalent full-power months, the wear scar depths must be less than 65% of wall thickness. A minimum G-Delta value required for a tube to wear to 65% depth of its wall thickness in 18 full-power months of operation was determined. For the nomimal case, G-Delta values of [ ]b,c,e are used by Westinghouse as an objective for tube expansion. Since such a value leads to an estimated lifetime to tube plugging (40% wear) in excess of 20 years, the TRC considers this value acceptable. Since tubes with G-Delta values in excess of [ ]b,c,e will be expanded and expanded tubes have G-Delta values less than [ ]b,c,e, there will be no tubes in excess of this G-Delta value after the modification. Therefore, the proposed modification meets both the TRC acceptance criterion and the Westinghouse design objective.

5.3.3. Effectiveness of Proposed Modification in Limiting Tube Vibration

The TRC has evaluated the modifications proposed by Westinghouse to determine their effectiveness in limiting tube vibration. It was concluded that the expansion process in the tubes at plates B and D will significantly limit the vibration level that could be expected from unexpanded window tubes, which are present in the Model D4/D5 steam generators. For Model E steam generators, characterized in general by the absence of window tubes (specific plants may have a partial support of the front row tubes at plate E), G-Delta values measured before and after expansion are not available. The TRC must therefore rely on predictions of the Westinghouse work rate model and its own independent modeling to estimate the effectiveness of the expansion process in reducing vibration. From the information available, it is concluded that it is plausible that a similar reduction will occur following expansion.

However, while the TRC was able to conclusively establish that upper pass excitation forces can be considered negligible in the Model D4/D5 steam generators, it could not reach as definitive a conclusion with regard to the Model E steam generator. The TRC therefore considers that the possible contribution of the upper pass fluid excitation forces is an uncertainty that could somewhat reduce the benefits of tube expansion in the Model E steam generator. Moreover, even though there is considerable indirect evidence that fluidelasticity is not a dominant contributing factor to vibration, such evidence is not conclusive. The TRC will continue to follow the Westinghouse evaluation of the impact of the upper pass velocity profiles on the long-term reliability of the Model E steam generator.

The combined effect of split flow and tube expansion will greatly improve the vibration-associated problems by substantially reducing the vibration levels, the tube-to-baffle plate impact forces, and the corresponding tube wear. The vibration-associated wear will not be entirely eliminated; tubes will still vibrate at lower levels and some will experience wear. Nevertheless, the predicted wear rates for the modified design, even under the most conservative assumptions, are sufficiently low to permit ISI to be conducted prior to the time when very severe wear would occur, thus preventing wear-induced steam generator tube ruptures.

5.4. MODIFICATION EFFECTS ON CORROSION PHENOMENA

5.4.1. Introduction

The TRC believes that judging the effect of the modification relative to long-term plant reliability criteria is a complex and ongoing process. The TRC has reviewed the safety aspects of the modification relative to corrosion phenomena and will continue to evaluate the long-term reliability data as it is produced by Westinghouse.

The effect of the modification on the various corrosion phenomena listed in Section 2.2 is discussed below.

5.4.2. Susceptibility to Tube Denting

The effect of gap size on the extent of denting has been investigated by Westinghouse in a series of Single Tube Model Boiler tests. Tests with prepacked and non-prepacked crevice conditions have been conducted. This testing has shown that average dent size increases with increasing diametral gap and that denting rates may, in fact, be reduced for smaller gaps. The TRC concurs that tube expansion and flow split modification do not increase the concern for denting within the preheater.

5.4.3. Stress Corrosion Cracking

Westinghouse has performed a combination of polythionic acid and controlled potential electrochemical tests on Inconel 600 and magnesium chloride tests on 304 stainless steel in order to evaluate the effect of tube expansion on residual stresses. Both nominal and off-nominal tube expansions were tested for the tube-to-tube sheet and baffle plate configurations. Westinghouse concludes that the polythionic acid tests show that there is no definitive increase in tube O.D. or I.D. residual stresses for expansions in excess of the maximum expected field expansion.

The initial results from the off-nominal tube expansion tests were unexpected in that stainless steel laboratory specimens exhibited cracks within [ ]b,c,e hours in a magnesium chloride solution. These results suggested that the off-nominal expansions may induce unexplained high residual stresses. Because of this unexpected result, Westinghouse performed limited tests comparing two heats of mill-annealed Inconel 600 tubing expanded into tube sheets and baffle plate collars. Westinghouse concluded that these tests showed that the baffle plate expansion resulted in lower residual stresses than did the tube sheet expansion.

To further validate this conclusion, the TRC requested Westinghouse to provide plots of tube expansion profiles of tubes in tube sheets and those that result from baffle plate expansion. Figure 5.4-1 presents the radial expansion tube profiles as a function of axial distance along the transition for both tube sheet and baffle plate expansions.

Figure 5.4-1 shows that the nominal baffle plate expansion (Curve A) is more gradual than the hydraulic tube sheet expansion (Curve C). Therefore, it may be concluded that the residual stresses for the nominal baffle plate expansion are less than the residual stresses for the equivalent tube sheet plate expansion. The off-nominal baffle plate expansion (Curve B) is very similar to the hydraulic tube sheet expansion (Curve C) and is clearly better than the mechanically rolled tube sheet expansion (Curve D). In addition, the mechanical expansion inherently produces more residual stress than the equivalent hydraulic expansion.

Based on the above, the TRC believes that if this strain gradient relationship is maintained for the tube expansion process, the modification will not produce a stress condition on the tubes which is greater than what already exists for these tubes in the tube sheet region. Also, the existence of this condition lends considerable support to the conclusions on relative stress levels arrived at by the limited testing discussed previously. Therefore, having the strain gradient of the tube expansion equivalent to or less than that in the tube sheet provides, in conjunction with

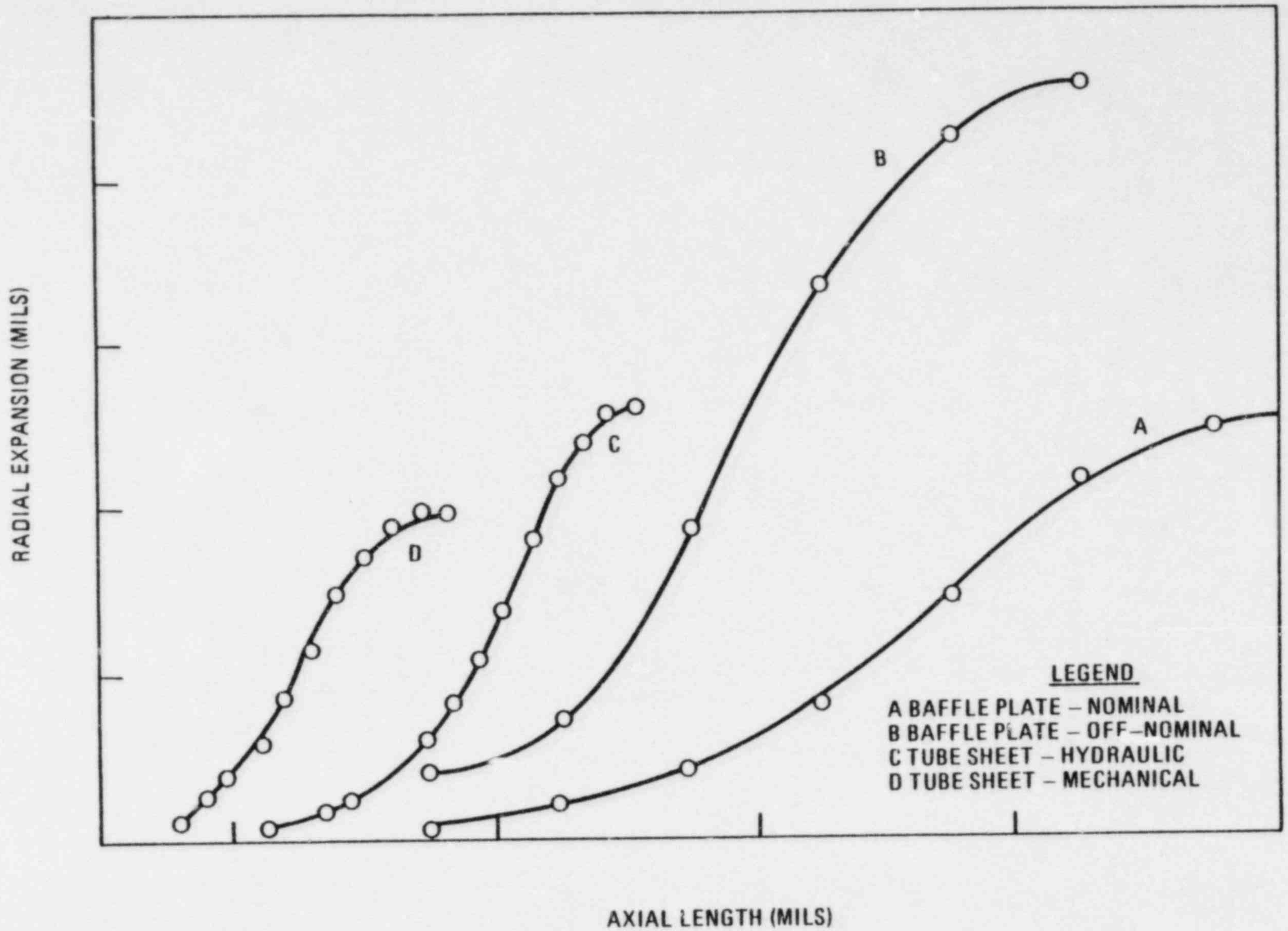


Fig. 5.4-1. Radial profiles of various tube expansions

supporting test data, an acceptance criterion for residual stresses within the tube.

Westinghouse has also presented experimental data which tend to show a retarding effect of lower environmental temperature on the stress corrosion cracking rates of Inconel 600 material. This effect is verified by the lack of stress corrosion cracking on the cold side of the steam generator in the vicinity of the tube sheet for any operating Westinghouse steam generator. The TRC agrees that lower environmental temperatures will extend the time to crack initiation for Inconel material. Therefore, since the steam generator preheater operates with the lowest average tube wall temperature within the unit, tube portions within this section of the steam generator should have the lowest stress corrosion cracking susceptibility. In addition, should a crack occur in the tube wall, Westinghouse analysis has shown that the tube will achieve a through-wall crack and leak before a critical crack length can develop as discussed in Section 5.6.2.4.

5.4.4. Wastage

The transport processes and chemical environment resulting in tube wastage are not completely understood. However, expanding the tubes into the baffle plate makes the geometry more similar to tube sheet joints, and there are no known instances of wastage in tube sheet crevices of operating Westinghouse units. Consequently, Westinghouse concludes and the TRC agrees that although wastage concerns cannot be precluded, there is no information to suggest that expanding the tubes will aggravate the potential for tube wastage at the expanded tube portions, although reduced gaps may enhance chemical concentration processes.

5.4.5. Pitting

The conclusions reached in Section 5.4.4 relative to tube wastage can also be applied to the phenomenon of pitting. On the cold leg of some steam generators, pitting has occurred in the sludge pile. The relationship of the chemical-hydraulic characteristics in a sludge pile on the cold side of the generator may be similar to those that can exist in tight crevices of the expanded tubes in baffle plates. This relationship has not been experimentally established to date. The TRC believes that since pitting has not been observed in baffle plate crevices in operating plants to date, pitting should not be considered a potential problem.

5.4.6. Fretting Corrosion

On alloys that have a passive film, such as Inconel 600, fretting (the rubbing of tube against baffle plate) in the presence of an otherwise noncorrosive aqueous medium can remove the protective film, resulting in a slight amount of metal dissolution, or corrosion, a process that is continually opposed by the relatively rapid repassivation kinetics.

This phenomenon does not appear to have caused problems for the unexpanded tubes in all Westinghouse operating plants to date, and the TRC does not expect the tube expansion process to present a safety concern related to the fretting corrosion phenomenon.

5.5. EFFECT OF PROPOSED MODIFICATION ON PLANT LICENSING

As described in Section 2.5, the proposed modification will result in minor revisions to the FSAR. These revisions will be processed on an individual plant basis. Three distinct areas must be specifically addressed:

1. Transient analysis margins.
2. Tube plugging margin.
3. First cycle inspection interval.

It is not believed that increasing the reactor coolant average temperature as much as 1.2°F will adversely affect the plant ability to load follow. The major effect will be in reducing the operating margin to the overtemperature ΔT reactor trip, and the transient most affected by this reduction in operating margin is the large load rejection transient. The effect of the modification has been evaluated as being within available plant margins on a generic basis. Each utility expects to address as necessary the quantitative aspects of the impact upon the transient analysis margins of its own plant. The Westinghouse recommended first cycle inspection interval is at first refueling.

Based upon the information supplied by Westinghouse, as well as the experience of the Model D2/D3 modified plants, the proposed modification for the Model D4, D5, and E steam generators will not have any large, adverse impact upon the licensing process.

5.6. EVALUATION OF STRUCTURAL ANALYSIS

5.6.1. Introduction

As noted in Section 2.4, the preheat modification does not involve the addition of components within the steam generator or supporting systems. Therefore, extensive new stress analysis is not required.

The modification does include locally expanding the steam generator tubing. The tubing is considered to be part of the primary pressure boundary. This modification must not compromise the integrity of this pressure boundary. Therefore, the modified tubing must meet the original structural safety criteria for the tubing and the effect of the expansion process on the tube must be evaluated.

The preheat modification also includes the redirection of a specified amount of feedwater from the steam generator main nozzle to the steam generator auxiliary nozzle. This modification changes the magnitudes of structural loading on the affected nozzles and parts of the steam generator internals for both steady-state and transient operation. These changes must not cause detrimental effects on pressure boundary and internal steam generator structural components.

The TRC has reviewed the necessary calculations and experimental data dealing with the complete modification and has concluded that structural integrity of all steam generator components has been maintained and applicable safety criteria have not been compromised.

5.6.2. Tube Expansion Structural Considerations

5.6.2.1. Stress Criteria and Design Loads. Two tube locations where additional structural analysis are required to evaluate the effects of tube expansion have been identified by Westinghouse. These locations are the expanded region and tube areas away from the expansion region.

The expanded tube configuration is evaluated for design, normal, upset, emergency, and faulted plant conditions per ASME Code, Subsection NB requirements.

The transient conditions considered in the analysis were developed by subdividing all design transients into groups. The grouping was based on the following criteria:

1. Secondary and primary pressure differences.
2. Secondary and primary fluid temperature differences.
3. Feedwater flow rates.
4. Number of occurrences.

Within each grouping the most severe transient was chosen as an "umbrella" transient on which all stress calculations are based. Table 5.6-1 presents the grouping arrangements and associated design basis umbrella transients.

Primary stress levels were evaluated against design allowables and fatigue usage factors were generated. The tube loads considered involved effects of:

- Tube wall pressure differential.
- Tube wall temperature differential.
- Tube axial temperature gradient.
- Local tube/baffle plate lateral mismatch.
- Axial interaction loads due to tube/baffle plate interference.

5.6.2.2. Analysis Method. The analysis was performed using finite element techniques. Most of the analysis involved elastic calculations. When the calculated loads were sufficiently large to cause yielding of the material, an elastic-plastic analysis was performed.

The overall Westinghouse approach consisted of using a global finite element tube model of an outermost affected expanded tube. This model

TABLE 5.6-1
TRANSIENT GROUPINGS

Umbrella Transient	Transients Included
100% load	Load/unload 15%-100%
Loss of load at 40 sec	Small step increase Small step decrease Reactor trip Control rod drop Loop out of service Load/unload 0%-15% power Feedwater cycling Loss of power Loss of flow Inadvertent startup of inactive loop Inadvertent safety injection Loss of load
Loss of load at 120 sec	Control rod drop Loss of power Inadvertent startup of inactive loop Inadvertent safety injection Loss of load Small step load increase Small step load decrease Loop out of service Load/unload 0%-15% power Feedwater cycling
Large step load decrease	Large step load decrease
Excess feedwater	Excess feedwater
Excess bypass feedwater	Excess bypass feedwater
25% power	25% power
Forward flushing	Forward flushing at 32°F Forward flushing at 200°F Forward flushing at 250°F
Heatup/cooldown	Heatup/cooldown Turbine roll
Inadvertent reactor coolant system depressurization (zero thermal stress + pressure)	Reactor trip Inadvertent reactor coolant system depressurization Turbine roll
Operating basis earthquake	Operating basis earthquake
Primary hydrotest	Primary hydrotest
Primary leak rate test	Primary leak rate test

Table 5.6-1 (Continued)

Umbrella Transient	Transients Included
Secondary hydrotest	Secondary hydrotest
Tube leakage at 840 psi	Tube leakage at 840 psi Tube leakage at 600 psi Tube leakage at 400 psi Tube leakage at 200 psi
Secondary leak rate test	Secondary leak rate test

generated tube loads for use as input to local stress analysis for the expanded portion of the tube and at the secondary side face of the tube sheet.

The local finite element model analysis for the expanded portion of the tube consists of two parts. Analysis was first performed using all transients to identify those which would cause tube/baffle plate interaction. By model definition these local tube deflection calculations do not involve stresses produced by any tube/baffle plate interaction. Those identified transients that resulted in local tube/baffle plate interaction, as determined by tube deflection values, were then rerun for final stress determination. These calculations placed radial displacement boundary conditions at those nodes which were identified to be involved in tube/baffle plate interference. Tube axial interaction loads calculated from the global model are also included. Therefore, the local moment-induced stresses are calculated in one of two ways, depending on whether or not tube/baffle plate interference exists for the transient condition being considered.

Boundary conditions used for all of the normal, upset, emergency, and faulted analyses that have been performed for the expanded tube in D4, D5, and E steam generators include the loads produced due to the tube being locked in place. The analysis was performed using two WECAN finite element models. The first model, a three-dimensional pipe model of the entire tube with appropriate boundary conditions at the baffle and support plates, was used to generate boundary condition input for the second model, a fine mesh axisymmetric model of an expanded region. The pipe model contained axial fixity at the B and D plate elevations in order to simulate a possible locked tube condition. This fixity was used in all of the analysis that has been performed. Therefore, all of the previously reported normal, upset, emergency, and faulted condition analysis results include the effects of locked tubes. The results of the analyses have shown fatigue usage factors of less than 1.0.

A fatigue analysis was performed at the expanded portion of the tube and the tube sheet locations. The fatigue analysis was performed for three post-expansion gap sizes. For each initial gap condition, analysis was performed for expansions at both baffle plates B and D of the preheater. Fatigue usage factors were calculated at the tubesheet and at both plate B and D locations. The analysis was performed on an elastic basis using ASME Code prescribed methods and includes total stresses for each transient and steady-state condition evaluated.

The maximum usage factors occurred at baffle plate B when considering tube expansion at both plates B and D. These factors are as follows:

Initial Diametral Gap (in.)	Total Fatigue Usage Factor[a]
	b,c,e

[a] These usage factors take into account the effect of the expansion process by the use of high cycle fatigue curves that incorporate the effect of the maximum residual strain.

ASME fatigue curves, which were incorporated into the Code in the Winter 1982 Addenda, were utilized in the above evaluation. Recent information from EPRI* on the fatigue properties of Inconel 600 confirm that the ASME fatigue curves are applicable for environments of AVT steam generator boiler water. Application to primary side water conditions is considered suitable, although this condition has not been tested.

Prior fatigue damage to worn tubes, such as exists at the Krsko plant, has been addressed by Westinghouse. The amplitude of alternating stresses based on Krsko vibration measurements is less than 1000 psi. Based on the examination of tubes with a total of 20 scars that were taken from

*"Fatigue Performance of Ni-Cr-Fe Alloy 600 Under Typical PWR Steam Generator Conditions," Electric Power Research Institute Report EPRI NP-2957, March 1983.

Ringhals 3 and Almaraz 1, which exhibited deeper wear scar than Krsko, the maximum theoretical stress concentration factor expected owing to wear was determined. This value was based on measurements of the deepest notch identified in the wear scar. Based on the appropriate ASME Code fatigue curve, the allowable number of cycles corresponding to an alternating stress of 2350 psi is greater than 10^{11}. Therefore, it is concluded that prior fatigue damage designated as accumulated usage due to vibration of the tubes in forming the wear scar is negligible.

5.6.2.3. Tube Plugging Analysis. In order to evaluate the tube expansion modification against the plugging criteria established by NRC Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes," Westinghouse has conducted analysis and testing on expanded tubes. These evaluations encompassed the following items:

1. Minimum wall thickness calculations.
2. External collapse pressure calculation.
3. Burst strength calculation.
4. Leak before break verification.

Westinghouse has determined that an increase in minimum allowable wall thickness at the expanded region of the tube is necessary to meet the minimum wall thickness requirements. This thickness satisfies external collapse pressure and burst strength criteria. Further discussion is given in Section 5.6.2.4.

5.6.2.4. Structural Evaluation of Feedwater Split Modification. The major effect of the feedwater bypass is lowering the heat transfer film coefficient in the main feedwater and increasing it in the auxiliary feedwater nozzle. The upper internals structure of the steam generator where the auxiliary nozzle feeds the unit is also affected. Other steam generator components, such as the downcomer, are not significantly affected. These changes occur because of the flow rate change within the nozzles. This results in different generated temperature gradients for these components during transient operation, which may impact fatigue usage. Westinghouse has indicated that the effect is significant only for normal operational transients between 15% and 100% power. Specifically, plant unloading and loading conditions are the most affected. Additionally, the upper steam generator internals are evaluated for thermal striping effects due to the addition of higher subcooled feedwater flows into this region from the auxiliary nozzle.

The analytical approach taken by Westinghouse for the nozzle consisted of scaling previous nozzle stress analysis developed for the unmodified flow split configuration. The conclusions reached from this analysis are that ASME design code allowables are met for operational transients and that the fatigue usage factor for both nozzles is less than 1.0. Additionally, the following conclusions are reached:

1. The effect of split flow on primary and secondary stress and fatigue usage for the central drain, intermediate plate, and auxiliary nozzle discharge pipe is insignificant.

2. The fatigue usage contribution from thermal striping on the upper internals components is negligible for conditions modified by split flow.

Westinghouse has performed a thermal striping test for the upper internals structure in the vicinity of the auxiliary feedwater piping discharge. This test consists of a scale model of the affected region using appropriate scaling factors. Surface temperatures of scaled structural components and their time variation are measured and used as input to a simplified heat transfer model to calculate thermal loadings on the upper internals structure. These results provide the basis for fatigue usage calculations.

Westinghouse believes and the TRC concurs that crack propagation analysis and past field experience will show limited crack growth for these components, and, therefore, crack growth is not considered a safety issue.

The feedwater flow split was also evaluated in terms of its effect on water hammer initiation and valve-generated water hammers. It does not appear that the flow split increases the potential for bubble collapse within the preheater, causing water hammer. The plant administrative controls implemented to prevent preheater water hammer are therefore unchanged.

The valve-generated water hammer was evaluated in terms of its effect on the main nozzle and auxiliary nozzle. Because of the reduced feedwater flow through the main nozzle due to the modification, water hammer loads are reduced. Also, the modification does not change the design transients used to determine the maximum water hammer loading effects on the auxiliary nozzle. The TRC believes that water hammer is not an issue relative to the split feedwater flow modification.

5.6.2.5. Leak Before Break for Expanded Tubes, Models D4, D5, and E. Westinghouse provided the following evaluation to substantiate that leak before break will occur for expanded tubes in Models D4, D5, and E. The TRC reviewed this evaluation and concurs with its basis, methodology, and conclusions.

The demonstration of leak before break for steam generator tubes that have been expanded differs only slightly from the equivalent demonstration for tubes in the unexpanded condition. In fact, the effects of the expansion process may act to enhance leak before break arguments. There are four primary reasons for this:

1. The expansion process is over a limited length and results in only a small amount of thinning in the tube wall. This amount of thinning is not sufficient to change the burst pressure significantly.

2. The residual stresses induced by the tube expansion process do not affect the bursting behavior of the tubes because the bursting process is governed by plastic instability. Residual stress does

not affect this mode of material behavior, a fact that is both theoretically and experimentally established.

3. The plastic deformation induced by the expansion process results in a strain hardening of the tube, which elevates the burst pressure of the tubes.

4. The leak rate behavior of the tubes is primarily elastic; that is, significant measurable leakage can occur without gross plastic deformation. Therefore, the hardening effect will not affect the leak rate behavior.

The following paragraphs describe the leak before break verification for these models, including the above arguments.

The rationale behind the leak before break requirement is to limit the maximum allowable (primary-to-secondary) leak rate during normal operation so that the associated crack length (through which the leakage occurs) is less than the critical crack length corresponding to the maximum postulated accident condition pressure loading. Thus, on the basis of leakage monitoring during normal operation, it is assured that an unstable crack growth leading to tube rupture would not occur in the unlikely event of the limiting accident.

For the Model D4, D5, and E steam generators, the maximum technical specification allowable leak rate is 0.35 gpm per steam generator. Results of leak rate tests were used to determine the maximum allowable crack length during normal operation corresponding to this specification limit. Typical results from one leak test series are shown in Fig. 5.6-1. These results indicate that a tube with a crack length of [ ]b,c,e will leak at a rate of [ ]b,c,e under the influence of normal operating pressure differentials. Similar data reductions were performed on the other tube test results to obtain the correlation shown in Fig. 5.6-2. From this correlation, the largest permissible crack length (associated with the limit of 0.35-gpm leak rate) during normal operation is [ ]b,c.e. Beyond this

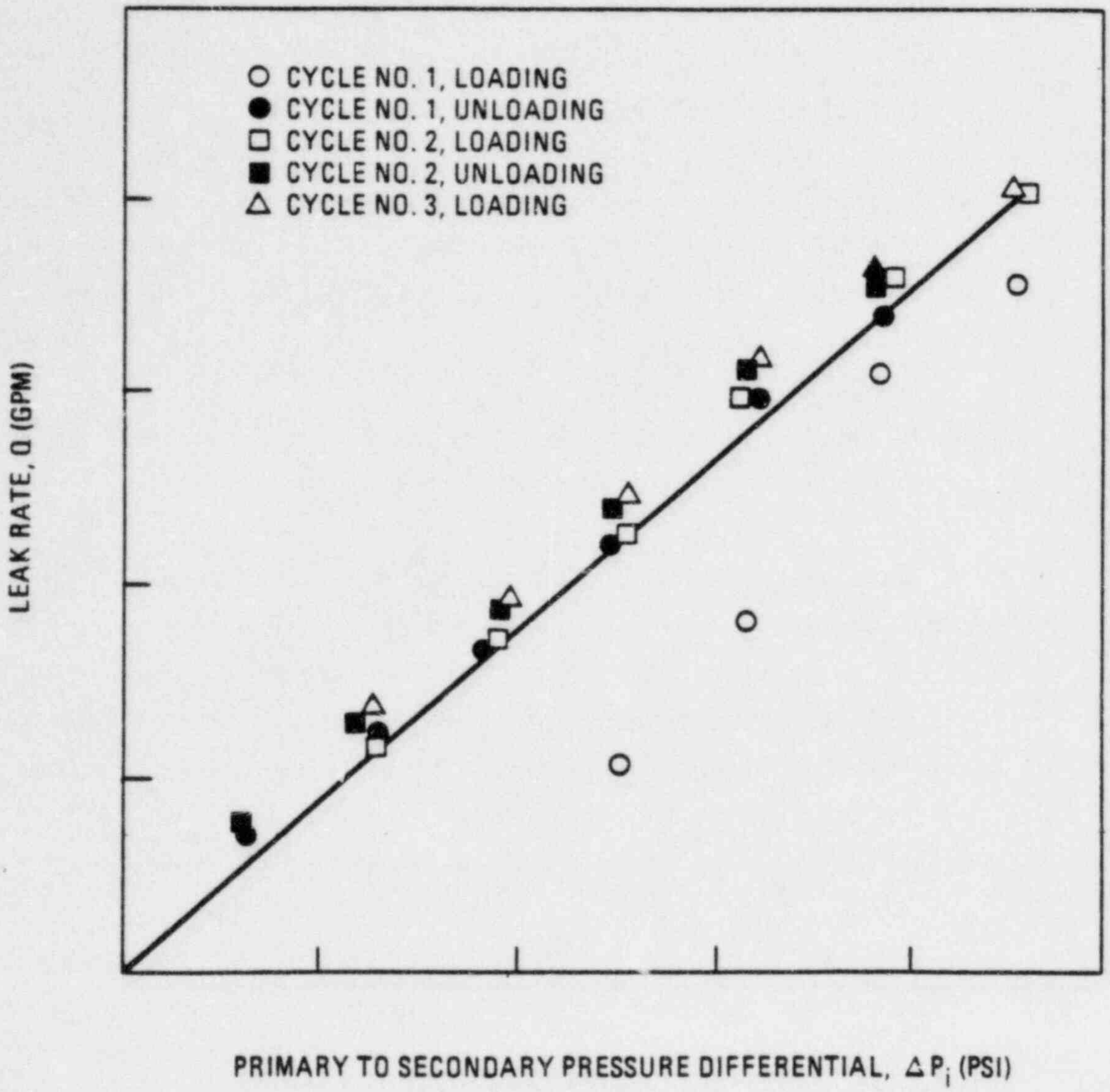


Fig. 5.6-1. Results from a typical leak rate test (test No. SGTLR-40)

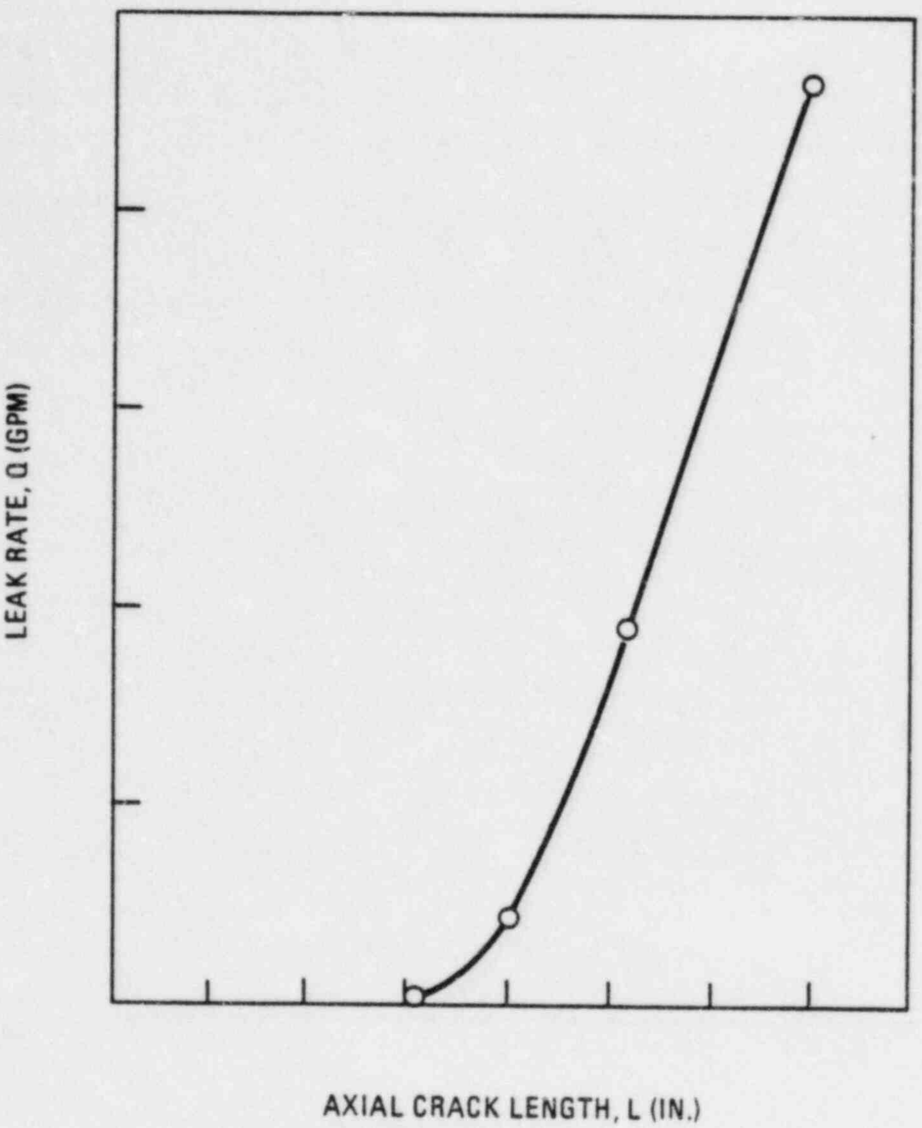


Fig. 5.6-2. Crack length versus leak rate during normal operation

length, the leakage would exceed the specification limit, requiring a plant shutdown for a corrective action.

The second part of this evaluation involves determination of the critical crack length under an accident condition maximum ΔP_i of 2650 psi. A data base was created by compiling the results of a large number of burst pressure tests performed on various Westinghouse steam generator tubing, within Westinghouse and elsewhere. Because of the variations in tube sizes and mechanical properties, the data were non-dimensionalized and are shown in Fig. 5.6-3. The figure shows excellent agreement between all the test results.

As shown in Fig. 5.6-3, the burst pressure of the tubing is a function of both the tube dimensions and the size of the flaw present. An empirical correlation for burst pressure for pipes and tubes has been developed by Hahn et al. and will serve to illustrate the functional relationship between the burst pressure, the tube dimensions, and mechanical properties. For a through-wall axial crack of length 2a,

$$\sigma = \bar{\sigma}(1 + 1.61\ a^2/Rt)^{-0.5}$$

In this correlation the failure stress, σ, is set equal to the flow stress, $\bar{\sigma}$, divided by a shell curvature correction factor $(1 + 1.61\ a^2/Rt)^{0.5}$.

The flow stress was set at the average of the yield and ultimate strength of the tube. The failure stress is related directly to internal pressure through the relationship $\sigma = PR/t$, where R = mean tube radius and t = tube wall thickness. Thus,

$$P = \bar{\sigma}\frac{t/R}{(1 + 1.61\ a^2/Rt)^{0.5}}$$

The above equation can be used to visualize the effect of the expansion process on the burst pressure. Table 5.6-2 shows a series of measurements

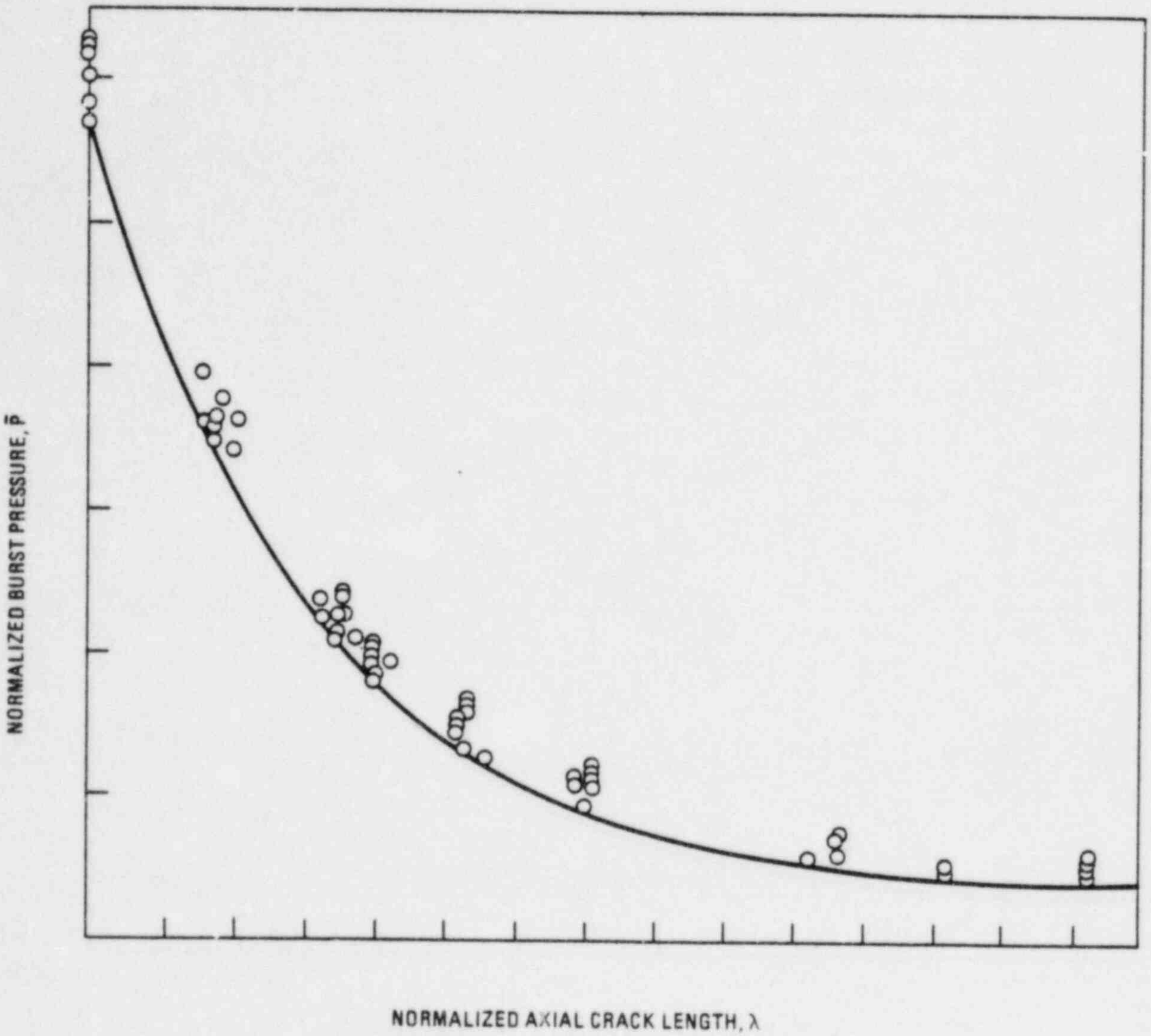
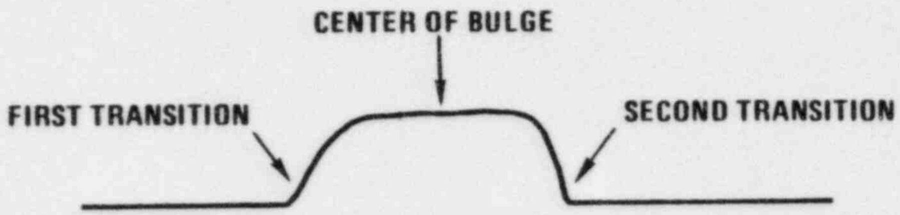


Fig. 5.6-3. Relationship between normalized burst pressure and axial crack length of steam generator tubing

TABLE 5.6-2
DIMENSIONS OF EXPANDED STEAM GENERATOR TUBES

Expansion	Original Thickness (in.)	First Transition Thickness (in.)	Center of Bulge Thickness (in.)	Second Transition Thickness (in.)	Original R_m/t	Expanded R_m/t

b,c,e



that were made on expanded tubing. The table shows that the wall thickness is decreased by a maximum of only 2%. The equation shows that the key parameter of the tube dimensions is the fraction t/R, which decreases by less than 8%. This will act to decrease the burst pressure by an equal amount. However, a more significant increase in the burst pressure is expected from the elevation of the flow stress, $\bar{\sigma}$, through the hardening of the material from the expansion process. For example, a representative material at room temperature with a yield of 39 ksi and an ultimate strength of 92.5 ksi has a flow stress of 66 ksi. A 37-mil O.D. expansion results in an increase in the yield strength to 60 ksi and the flow stress to 75 ksi, or a 14% increase. The net result would be a 5% increase in burst pressure.

Both the leak rate and burst behavior of steam generator tubing are a function of the flaw size that exists in the tube and the tube dimensions. The margin can be easily visualized in Fig. 5.6-4, where both a leak rate curve and a burst curve are plotted for a heat of tubing for which leak rate and burst tests were carried out. The burst curve shown is for the governing faulted condition for the tubes, the feedline break, while the leak curve is a plot of the flaw length at which the specification limit of 0.35 gpm is obtained at operating pressure.

The effect of the expansion process will be to elevate the burst curve, as explained above, while no significant change would be expected in the leak curve. The mean radius to thickness ratio of the tubes increases slightly due to the expansion, from 8.8 to 9.9, but Fig. 5.6-4 shows there is no change in the leak-break margin as a result of this change.

Therefore, it is seen that the effect of the tube expansion process is to increase the margin between leak and break for the steam generator tubes.

The effect of residual stresses from the tube expansion is not as straightforward to determine. Examining the mechanics involved without regard for stress corrosion cracking, a through crack can be treated as occurring in an infinite plate. The effect of the residual stress field will be to increase the mean stress intensity factor, K_I, in fatigue. In

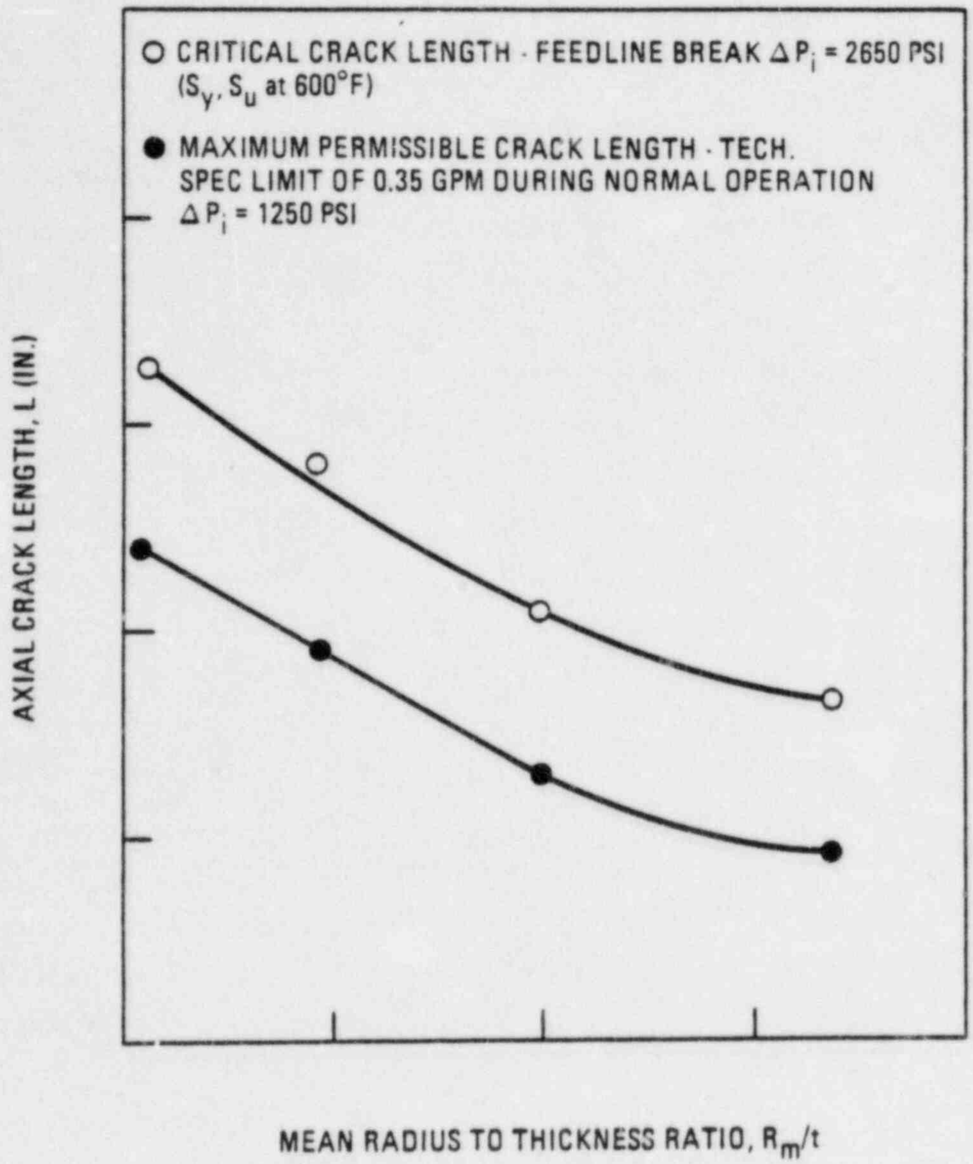


Fig. 5.6-4. Variation in margin to failure by burst as a function of R_m/t ratio

general, fatigue crack growth rates for varying mean values and ranges of stress intensity factor can be correlated by calculating an effective range as

$$\Delta K_{EFF} = \frac{\Delta K}{(1 - R/\lambda)} \quad ,$$

where $R = K_{MIN}/K_{MAX}$,

λ = material constant, approximately 2.

For general operation of the steam generator, the value of ΔK is small compared with K_{MIN}; thus R is relatively large, i.e., approaching unity. The addition of residual stress will then only increase the value of R by a small amount.

For example, secondary side pressure is on the order of 940 psi at 0% power and 880 psi at 100% power. For a primary side pressure of 2250 psi, K_{MIN} = 11.1 ksi $\sqrt{in.}$ and K_{MAX} = 11.6 ksi $\sqrt{in.}$ are calculated for a 0.44-in.-long crack. Thus, R is 0.96. If a postulated 45-ksi residual stress is included, R becomes 0.99. This would raise the SIF range by about 3%, which is not significant. In fact, the 0.5 ksi $\sqrt{in.}$ value of ΔK is about 10% of the threshold value of about 5 ksi $\sqrt{in.}$* (corresponding to 10^{-7} in./cycle growth) and without consideration of environmental influences could be significantly increased without appreciably affecting the margin to burst. It should also be noted that the use of the term critical crack length is relative to tube bursting, not catastrophic crack growth.

Other factors should also be considered since environmental effects can influence the above argument, the most powerful being that the magnitude of the residual stress field is displacement controlled. Thus, crack growth serves to reduce the residual stresses acting on the crack. Based on these

*For plane strain testing. The tube conditions relative to crack growth will be more like plane stress due to thickness effects at the crack tip, resulting in a higher threshold value.

considerations, it is probable that the residual stress field will not appreciably influence the fatigue crack growth rate. Since the controlled leak rate occurs under elastic conditions, and the burst strength is increased due to work hardening of the material, accompanied by a corresponding increase in critical crack length, the total amount of growth required between leak and burst is also increased. It can therefore be concluded that considerable margin between leak and burst will exist for the expanded tubes.

The leak before break concept has been confirmed by laboratory and plant operating experience in the case of axially oriented cracks. Typically, stress corrosion cracks in this orientation have an aspect ratio, i.e., L/D, of 4 to 6. Thus, for a tube of 0.043-in. wall, a crack would propagate through the wall (and leak) before exceeding a length of about 1/4 in., i.e., leak before break. When stress corrosion cracking has occurred, it almost always has been axially oriented because of the predominating effect of the hoop pressure stresses. In only a few instances have circumferential cracks been observed in operating plants: part-wall penetrations in a domestic plant and through-wall cracks at tube sheet transitions in a non-domestic plant. In the latter case, short, through-wall cracks, initiating from the I.D. at the tube/tube sheet transitions, were observed, each about 0.2 in. in length. This confirms that the aspect ratio previously observed for axial cracks also applies to circumferentially oriented cracks.

Given this aspect ratio of 4 to 6, leak before break would be the anticipated mode of behavior in the circumferential situation, as in the axial, and, in fact, there is a greater margin for the former case. Experiments show that the critical crack length for through-wall circumferential cracks in 3/4-in. O.D. tubing subjected to (1) residual stresses due to mechanical expansion, (2) internal pressure, and (3) imposed bending stresses was well in excess of the length that could result in leakage greater than the specification limit.

In summary, leak before break has been demonstrated for circumferential cracks in the residual stress field remaining after expanding the tube in the tube sheet. For hydraulic expansion of the tube in a baffle plate, leak before break is expected to characterize the behavior of circumferential cracks in the hydraulic expansion transition.

5.7. QUALITY ASSURANCE

Each utility will apply its 10CFR50, Appendix B, quality assurance program to the steam generator modification. All field work will be performed in accordance with appropriate Westinghouse WCAP's and any additional site-specific quality assurance requirements. The utility members will follow actions at other sites to ensure that quality assurance programs are consistent.

5.8. TOOLING AND IMPLEMENTATION PROCEDURES

Tooling and implementation procedures will be generated on a site-specific basis and will be reviewed in accordance with quality assurance requirements as set forth in Section 5.7.

5.9. IN-SERVICE INSPECTION

The eddy current testing should comply with the procedures established by the ASME Code, Section XI, "In-Service Inspection," and Regulatory Guide 1.83, "In-Service Inspection of Pressurized Water Reactor Steam Generator Tubes," for the selection of tubes to be tested and for the frequency of testing.

5.10. CLEANLINESS EVALUATION

The proposed modification requires entry into the primary system, which may be preconditioned for service. Hence, there is a need for appropriate cleaning procedures. The procedures should address the methods to account for and remove tools and equipment, clean the interior of the expanded tubes, remove solid and liquid residues from the plenum, and final wipe before closure. This will be established on a site-specific basis.

6. MEMBERSHIP OF THE COUNTERFLOW STEAM GENERATOR OWNERS REVIEW GROUP

The Counterflow Steam Generator Owners Review Group is composed of two subgroups, the Steering Committee and the Technical Review Committee. The names of the individual members, together with their profile of expertise and experience, are summarized in Table 6-1. Resumes of the members are presented in Table 6-2 (Steering Committee) and Table 6-3 (Technical Review Committee).

TABLE 6-1
COUNTERFLOW STEAM GENERATOR OWNERS REVIEW GROUP
PROFILE OF EXPERTISE AND EXPERIENCE

Members of Steering Committee and Technical Review Committee

STEERING COMMITTEE

Name	Organization	TRC Responsibility	Area of Experience	Years of Experience
L. D. Butterfield	Commonwealth Edison Company	Chairman	Project management, design, project engineering, reactor analysis, licensing, administration, and operations	17
A. B. Cutter	Carolina Power & Light Company	Member	U.S. Navy, startup, maintenance, project management, nuclear steam supply, engineering, procurement, and construction	27
W. M. Petro	Public Service Company of Indiana	Member	Engineering, procurement, construction, startup, project management, design, mining, marketing, and operation	25
J. H. Goldberg	Houston Lighting & Power Company	Member	Nuclear engineering, construction, design, project management, and modifications	25
Guy A. Frederick	Electrobel (Electronucleaire - Belgium)	Member	Engineering, design, specification, in-service inspection and contractor surveillance, analysis and recommendation of solutions to various equipment problems, and fracture mechanics and fatigue research	17

TABLE 6-1 (Continued)

TECHNICAL REVIEW COMMITTEE

Name	Organization	TRC Responsibility	Area of Experience	Years of Experience
M. G. Zaalouk	Houston Lighting & Power Company	Chairman	Reactor engineer, supervisor, professor, mechanical-nuclear, startup and construction support, and design modifications	26
A. B. Poole	Houston Lighting & Power Company	Assistant Chairman	Mechanical-nuclear, project coordination, design, stress calculations, project engineering, and thermal calculations	15
F. L. Eisinger	Consultant to HL&P (FWEC)	Member	Equipment design, vibration and stress analysis, and professor	31
R. C. Iotti	Consultant to CP&L (Ebasco)	Member	Radiation analysis, thermal hydraulics, heat transfer, fracture mechanics, continuum mechanics, and vibration analysis and testing	18
R. W. Riley	Public Service Company of Indiana	Member	NSSS design, mechanical-nuclear, heat exchanger and fluid flow calculations, and certified special inspector of pressure piping	20
J. Reiss	Commonwealth Edison Company	Member	Plant operations, project engineering, flow-induced vibration, and NSSS loose parts monitoring systems	7

TABLE 6-1 (Continued)

TECHNICAL REVIEW COMMITTEE

Name	Organization	TRC Responsibility	Area of Experience	Years of Experience
D. A. Steininger	EPRI (Steam Generator Project Office)	Member	Initiation and management of research contracts, flow and boiling phenomena, corrosion fatigue and "fretting and wear" characteristics, vibration analysis, and program manager	11
R. Hanford	Carolina Power & Light Company	Member	Resident welding/material engineer, construction support, quality assurance, corrosion, stress corrosion cracking, and ASME Code	15
Willy Ch. P. De Roovere	EBES (Electronucleaire - Belgium)	Member	Plant design, operation, and maintenance, fracture mechanics, in-service inspection, refueling procedures, and solving steam generator problems	13
Elie J. Stubbe	Tractional (Electro-nucleaire - Belgium)	Member	Engineering, analysis, evaluation of primary and secondary system behavior, research, and teaching	19
K. Fink	NPP, Krsko (Yugoslavia)	Member	Water chemistry control, radio-active chemistry, materials and corrosion, and teaching	8
P. Bilcar	NPP, Krsko (Yugoslavia)	Member	Engineering, procurement, construction, and plant startup and modification	
Vladimar Fatur	NPP, Krsko (Yugoslavia)	Member	Engineering, professor, and research	10

Twenty pages of the report, consisting of résumés of CSGORG members, have been omitted.

7. BIBLIOGRAPHY

1. "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, May 1982.
2. "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, July 1982.
3. "Model E, Steam Generator Preheater Tube Air Flow and Vibration Test Results," Preliminary Westinghouse Report, October 27, 1982.
4. "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, November 1982.
5. "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, March 1983.
6. "Counterflow Preheat Steam Generator Tube Expansion," Westinghouse Report, April 1983.
7. "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, May 1983.
8. "Counterflow Preheat Steam Generator Tube Vibration," Westinghouse Summary Report, June 1983.
9. "Counterflow Preheater Steam Generator Tube Expansion," Westinghouse Report, June 1983.
10. "D2/D3 Steam Generator Design Modification," Utility Design Review Panel Evaluation Report (Duke Power Co., Tennessee Valley Authority, and South Carolina Electric and Gas Co.), January 1983.
11. "Safety Evaluation Report Related to the D2/D3 Steam Generator Design Modification," NUREG-0966, U.S. Nuclear Regulatory Commission, March 1983.
12. ASME Boiler and Pressure Vessel Code, Section III, Appendix II, and Section IX.
13. Preheat Steam Generator Status, Westinghouse Presentations, February 1982.
14. Wear Assessment for the McGuire 1 Steam Generator Tubing, Westinghouse Presentation to U.S. Nuclear Regulatory Commission, April 1982.

15. "KRSKO", Letter of March 28, 1983, from NUS Corporation.
16. G. J. Theus, "Stress Corrosion Cracking Tests of Alloy 600," Workshop Proceedings: U-Bend Tube Cracking in Steam Generators, EPRI WS-80-136, June 1981.
17. T. S. Bulischek and C. Van Rooyen, "Stress Corrosion Cracking of Alloy 600 Using the Constant Strain Rate Test," The International Corrosion Forum Devoted Exclusively to the Protection and Performance of Materials, Paper No. 183, March 3-7, 1980.
18. "IGA of Oconee 1 Steam Generator," Letter from D. A. Steininger to J. L. Epstein (Westinghouse), March 23, 1983.
19. J. Seivar, "Normal Operating Radiation Levels in Pressurized Water Reactor Plants," Nuclear Technology, Vol. 36, pp. 48-55, 1977.
20. R. D. Blevins, Flow Induced Vibrations, Van Nostrand-Reinhold, 1977.
21. J. M. Biggs, Introduction to Structural Dynamics, McGraw-Hill, New York. 1964.
22. S. T. Rolf and J. M. Barson, Fracture and Fatigue in Structure: Applications of Fracture Mechanics, Prentice-Hall, Inc., Englewood Cliff, New Jersey, 1977.
23. Strommer et al., "Growth Characteristics of Post-Through Cracks in Thick-Walled Plates and Tubes," ASME Paper No. 76-PVP-39, October 25, 1976.
24. G. T. Hahn and A. R. Rosenfield, "Sources of Fracture Toughness: The Relation Between KIC and the Ordinary Tensile Properties of Metals," ASTM STP 432, pp. 5-32, 1968.
25. G. T. Hahn and A. R. Rosenfield, "Plastic Flow in the Locale of Notches and Cracks in Fe-35 Steel Under Conditions Approaching Plane Strain," Ship Structure Committee Report SSC-191, U.S. Coast Guard, Washington, November 1968.
26. Brown and Frawley, "Standard Method of Testing for Plane Strain Fracture Toughness of Metals," ASTM Standard Part 10, E-399, 1974.
27. G. R. Irwin, "Relation of Crack-Toughness Measurement to Practical Application," ASME Paper No. 62-MET-15, Presented at Metals Engineering Conference, Cleveland, April 9-13, 1962.

28. D. Brock, Elementary Engineering Fracture Mechanics, p. 360, Noordhoff Mtf Publishing, Leyden, Netherlands, 1974.

29. P. G. Hodge, Plastic Analysis of Structures, McGraw-Hill, New York, 1959.

30. L. S. Beedle, Plastic Design of Steel Frames, Riley Interscience, New York, 1958.

31. B. G. Neale and J. S. Symonds, Plastic Design of Steel Structures, John Wiley, New York, 1963.

32. M. Vagins, et al., "Steam Generator Tube Integrity Program - Phase I Report," U.S. Nuclear Regulatory Commission, NUREG/CR 0718, September 1971.

33. D. Tillack to R. Keating, "Stress-Strain Curves for Inconel 600," Huntington Alloys Letter TS-52974-C4, May 29, 1974.

APPENDIX A

COUNTERFLOW STEAM GENERATOR MODIFICATION PLAN
FOR OWNERS REVIEW GROUP STUDY

Plan for Counterflow Steam Generator

Modification Owners Review Group Study

I. Purpose

The purpose of this Owners Review Group is to complete a design review of the final proposed modification program for counterflow steam generators as proposed by Westinghouse. This modification program is relative to the elimination of major vibration and wear within the pre-heater section of the subject steam generators. The guidelines for the review is to address the following areas:

1. To determine if the proposed modifications will eliminate major tube vibration.

2. To determine if the proposed modifications could impact licensing of the plants.

3. To determine if the proposed modifications can be implemented without significant impact on long-term plant operations.

4. To determine if Westinghouse acceptance criteria is adequate and whether the proposed modification meets the acceptance criteria.

5. To determine if the proposed modification can be implemented without impacting plant completion schedules.

6. To determine if any other areas should be addressed during the course of the review.

The Owners Review Group will also address technical areas related to the following disciplines:

1. Thermal Hydraulics

2. Vibration Analysis

3. Structural Design and Analysis

4. Feedwater System Analysis

5. Metallurgy/Welding

6. Stress Corrosion Chemistry

7. Post Operating Monitor/Installation

8. Field Modification

II. Representatives

The Owners Review Group is composed of the following Utilities:

1. Carolina Power & Light Company

2. Commonwealth Edison Company

3. Houston Lighting & Power Company

4. Public Services Company of Indiana

5. Belgium Utilities (Electronucleaire) as represented by Electrobel

6. Nuklearna Elektrana Krsko

III. Organization

1. The Counterflow Steam Generator Owners Review Group will be composed of the following two sub-groups:

   a. Steering Committee

   b. Technical Review Committee

2. Steering Committee (STC) - Group responsible for the management of the Owners Review Group. Group will consist of the following:

   a. Members - One executive (Vice President level) or his appointed representative from each owner.

   b. Chairman - Elected by STC members from among its members. Mr. L. D. (Del) Butterfield of Commonwealth Edison Company, is the elected Chairman.

3. Technical Review Committee (TRC) - Group responsible for the technical review of the final proposed Westinghouse modification program. This group will consist of:

   a. Chairman - Dr. M. Z. (Jimmy) Zaalouk of Houston Lighting & Power Company (selected by the Steering Committee)

   b. Assistant Chairman - Mr. A. B. Poole of Houston Lighting & Power Company

The TRC will consist of no more than twelve (12) people who will be supplied by the Owners. At least one (1) representative from each Owner will be included on the TRC. No Owner shall provide more than two (2) representatives, except for the Owner who furnishes the TRC Chairman. That Owner is allowed to have two (2) representatives in addition to the Chairman. Representatives may include consultants and/or other utility representatives with experience in various technical areas as mentioned in paragraph I. The NRC and EPRI Steam Generators Project Office were invited to provide one (1) Ad HOC (non-voting) TRC representative each. Technical recommendations will be established based upon simple majority vote with each Owner having one (1) vote to be cast by its designated representative.

IV. Functions

The TRC Chairman will be responsible for formal communication between the Review Committee and Westinghouse. TRC comments on the Westinghouse program and additional recommendations by the committee shall be identified to Westinghouse. Other technical functions of the TRC Chairman are as follows:

1. To establish the location, date and schedule for all Technical Review Committee meetings.

2. To ensure that all specific requests and/or questions from the TRC are formally transmitted to Westinghouse.

3. To ensure that formal meeting minutes are completed and issued.

4. To ascertain that Westinghouse has answered all questions and that all needed information has been transmitted between the TRC and Westinghouse.

5. To provide the Steering Committee with the results of the technical review and to issue copies of this final review report to Westinghouse and the NRC after its acceptance by the Steering Committee.

V. Meeting Minutes

After each meeting, the TRC Chairman will issue minutes to attendees. Where practical, minutes will incorporate Westinghouse written and/or verbal answers to TRC's outstanding questions and/or clarifications.

VI. Report

A final report consisting of the TRC findings, minutes of the meeting, relavent correspondences and summary of the resumes of the representatives will be provided, after review by the Steering Committee, to NRC for use in preparation of appropriate Safety Evaluation Report. Copies of Final Report will be provided to each of the participating utilities.

- Final report will have both proprietary and non-proprietary information. Non-proprietary information will be so identified.

APPENDIX B

MINUTES OF SELECTED TECHNICAL MEETINGS

B.1. Counterflow Steam Generator - Minutes of the Meeting Held between TRC and Westinghouse, March 18, 1983.

B.2. Counterflow Steam Generator - Minutes of the Meeting between TRC, Westinghouse, and NRC, Held April 21 1983.

B.3. Counterflow Steam Generator - Minutes of the Meeting Held May 18, 1983, at Pittsburgh between TRC and NRC.

B.1. COUNTERFLOW STEAM GENERATOR - MINUTES OF THE MEETING HELD BETWEEN TRC AND WESTINGHOUSE, MARCH 18, 1983

OWNERS REVIEW GROUP

(March 18, 1983)

AGENDA

I. GROUNDRULES

1. THE REVIEW WILL BE LIMITED TO THE ADEQUACY OF DESIGN CHANGE NOT THE CHOICE OF MODIFICATION.

2. QUESTIONS WILL BE TRANSMITTED FROM THE ORG CHAIRMAN TO THE COUNTERFLOW PROJECT OFFICE.

3. WESTINGHOUSE WILL ATTEMPT TO RESOLVE QUESTIONS VERBALLY AT MEETINGS OR BY PHONE.

4. WRITTEN RESPONSES, WHERE REQUIRED, WILL BE TRANSMITTED FROM THE COUNTERFLOW PROJECT OFFICE TO THE ORG CHAIRMAN.

5. WESTINGHOUSE DOES NOT PLAN, IN GENERAL, TO PROVIDE RAW DATA OR CALC NOTES.

6. WESTINGHOUSE RETAINS DESIGN RESPONSIBILITIES.

7. CONSULTANTS MUST SIGN PROPRIETARY AGREEMENTS WITH WESTINGHOUSE.

8. WESTINGHOUSE RESERVES ITS RIGHT TO PROPRIETARY DATA AND MUST REVIEW ANY REGULATORY SUBMITTALS FOR PROPRIETARY MATERIAL.

II. SCHEDULE

1. WESTINGHOUSE TO ISSUE LICENSING REPORT ON TUBE EXPANSION MID APRIL.

2. PROPOSE MEETING WITH ORG LATE APRIL ON TUBE EXPANSION.

3. WESTINGHOUSE FINAL DESIGN REVIEW MID MAY.

4. PROPOSE MEETING WITH ORG LATE MAY ON TUBE EXPANSION VIBRATION, SPLIT FEEDWATER.

5. WESTINGHOUSE ISSUE COMBINED REPORT IN EARLY JUNE.

COUNTERFLOW STEAM GENERATOR

MINUTES OF THE MEETING HELD BETWEEN TECHNICAL REVIEW COMMITTEE

AND WESTINGHOUSE ON MARCH 18, 1983

Participants

Name	Organization
Bruce Poole	Houston Lighting & Power Company
F. L. Eisinger	Consultant - HL&P (FWEC)
Joe Reiss	Commonwealth Edison Company
Del Butterfield	Commonwealth Edison Company
Robert C. Iotti	Consultant - CP&L (Ebasco)
Ray Hanford	Carolina Power & Light Company
David A. Steininger	EPRI (SGPO)
Guy Frederick	Electronucleaire
Willy De Roovere	Electronucleaire (EBES)
Elie Stubbe	Electronucleaire
Charles B. Hardee	Carolina Power & Light Company
Roger Riley	Public Service, Indiana
A. B. Cutter	Carolina Power & Light Company
Ed Harris	Carolina Power & Light Company
J. H. Goldberg	Houston Lighting & Power Company
Jimmy Zaalouk	Houston Lighting & Power Company
Jim McGuffin*	Westinghouse
D. White*	Westinghouse
Patrick McDonough*	Westinghouse
J. Vogle*	Westinghouse
Tom Timmons*	Westinghouse
J. Epstein*	Westinghouse

* Attended the second session

First Session

The first session was a meeting between the members of Steering Committee and members of Technical Review Committee. The session started with each member present, introducing himself.

Then, the TRC Chairman Dr. Jimmy Zaalouk read and explained the charter of the TRC. After that the following topics were discussed:

1. NRC Participation - Technical Review Committee has no objection for the participation of NRC in our work. The need and method of getting NRC involved in the work was discussed.

   Steering Committee will identify the method of getting NRC's involvement.

2. The whole study was divided into the following three major groups:

   1. Thermal Hydraulics and Vibration

   2. Stress Corrosion Chemistry

   3. Post Operating Monitor/Installation

   Three subgroups of the members were formed representing each of the above major groups. Membership for each subgroup was left to the choice and interest of each member. A member can participate in more than one subgroup.

   Following members were nominated as subgroup leaders:

   1. A. B. Poole - Thermal Hydraulics and Vibration

   2. D. A. Steininger - Stress Corrosion Chemistry

   3. Joe Reiss - Post Operating Monitor/Installation

   The three subgroups then had different sessions to discuss and formulate first series of questions on the general presentation given by Westinghouse to the Owners Group on March 17, 1983. These questions are contained in the attached letter of March 22, 1983 from Dr. M. Z. Zaalouk, Chairman TRC to Mr. Patrick J. McDonough of Westinghouse.

   While subgroups were meeting, the members of the Steering Committee along with Dr. Jimmy Zaalouk met and made the following decisions:

   1. Dr. Zaalouk and Mr. Del Butterfield should attend the meeting to be held after a week at Bethesda where Westinghouse will present a summary of their report to NRC.

   2. At Bethesda, Dr. Zaalouk should find out our contact with NRC.

<u>Second Session</u>

Second Session was held between members of Steering Committee, Technical Review Committee and Westinghouse and the following transactions took place:

1. Westinghouse distributed the ground rules and schedule which is attached (Attachment 2) and the same was discussed.

2. Westinghouse said that the contact for TRC with Westinghouse would be Mr. Patrick J. McDonough.

B.2. COUNTERFLOW STEAM GENERATOR - MINUTES OF THE MEETING BETWEEN TRC, WESTINGHOUSE AND NRC, HELD APRIL 21, 1983

AGENDA FOR
APRIL 21 AND 22
COUNTERFLOW STEAM GENERATOR
OWNERS REVIEW GROUP MEETING

Location: Howard Johnsons, Monroeville, Neptune Room

Day: April 21, 1983

8:00 am Meeting of TRC

- Discuss comments on D2/D3 documents
- Identify reference and information needed for TRC Report
- Develop Outline for TRC Report
- Develop plan of work on the TRC Report

9:30 am Break

10:00 am Meeting with Westinghouse

- Westinghouse to provide presentation of Tube Expansion Licensing Submittal

12:00 pm Lunch

1:00 pm Closed Meeting of TRC and Trip to APD to View Models

- Review Westinghouse responses to the TRC questions.
- Determine comments and/or new questions to be provided to Westinghouse
- Summary

3:00 pm Summary Meeting with Westinghouse

- Present TRC comments to Westinghouse

Counter Flow Steam Generator

Minutes of the Meeting Between Technical Review

Committee, Westinghouse and NRC, Held on April 21, 1983

Participants	Organization
Jimmy Zaalouk	Houston Lighting & Power Company
Bruce Poole	Houston Lighting & Power Company
Frank Eisinger	Consultant to HL&P (FWEC)
Guy Frederick	Electronucleaire
Elie Stubbe	Electronucleaire
Roger Riley	Public Service, Indiana
L. D. (Del) Butterfield	Commonwealth Edison Company
Joe Reiss	Commonwealth Edison Company
Robert Iotti	Consultant to CP&L (Ebasco)
David Steininger	EPRI (SGPO)
John Hopkins*	NRC
Marty Wambsganss*	Consultant to NRC (Argonne Natl. Lab)
Patric McDonough*	Westinghouse
Tom Timmons*	Westinghouse
Edward Burns*	Westinghouse
Jim McGuffin*	Westinghouse
Kresimir Fink	KRSKO

* Part-time attendance

The TRC Chairman, Dr. Zaalouk, opened the meeting stating the following objectives for the meeting:

1. The members of the TRC have each identified questions they have relative to Westinghouse responses of April 8, 1983.
2. This morning we will review the D2/D3 report and discuss our working relationship with the NRC.
3. This afternoon Westinghouse has arranged for a tour of the 16 degree model. People who have not seen the model will probably want to go on the tour.
4. This afternoon or tomorrow we will develop an outline for the final report and identify who will write the various sections.

Discussion on D2/D3

The following questions were raised relative to the D2/D3 report:

1. Who wrote the final report?
2. How long did it take to complete the report?
3. In what manner did the group operate?

Since Frank Eisinger had been a member of the D2/D3 group, he was requested to discuss the above items.

Frank Eisinger said that the leaders of the various sections wrote their corresponding sections. The integration of the sections was coordinated by the Steering Committee. The report was based upon discussions with Westinghouse. The decision of the Design Review panel (DRP) was to model the report after the safety evaluation report and have it deal with safety aspects of the work. The writing of the report covered approximately three months of October, November, and December, 1982.

John Hopkins of the NRC was then requested to discuss the working relationship between NRC and DRP. Mr. Hopkins replied that NRC and their consultants were not involved in the day to day questions between DRP and Westinghouse. However, NRC had fairly close contact with the DRP. It would be consistent for the same type of working arrangement to be used with the TRC.

In addition to the above, the following discussions and/or decisions were made:

1. This is the first full blown meeting of the TRC, NRC and Westinghouse. In this meeting we want to establish the procedure for good working arrangements. There will be internal discussions between Westinghouse and the TRC group. Westinghouse may not want the whole group interacting. It may be preferable to have smaller groups formed on the basis of different major topics of discussions.

2. NRC does not feel the need or want to be at the internal meetings of TRC.

3. Westinghouse must show that the results of data, calculations, etc., of D4/D5 models envelopes model E also.

4. It was said that the group on the study of D2/D3 models had access to calculations. Will the TRC on the study of D4/D5/E models have access to calculations and data? Westinghouse will give calculations on specific items.

5. If a D4/D5 unit goes to internal split flow then reverse flow will be the limiting contributing factor.

6. When discussing erron bands on tubes we must evaluate if it is a safety or long term operability question.

7. We should have someone start looking at the preparation and editing of the report.

8. How did NRC handle the D2/D3 report?
   NRC had the advantage of looking at Westinghouse report. NRC had specific questions on certain areas and they looked at how the review panel reviewed the same areas. They tried to use as much as possible, acceptable findings of the panel on these questions in their review of their report.

9. TRC Report Outline/Plan

   We should visualize in what format the report should be and what materials should go into the format. The sources for writing of the report shall be Westinghouse report along with our questions and their responses to the same, consultants input, KRSKO data and EPRI reports and data.

Mostly Westinghouse is the heaviest source of input to TRC report. If we need verifications on the Westinghouse report and/or responses, the subgroups shall do the main work of getting verifications.

Mr. Ed Burns will forward Westinghouse final report in about six (6) weeks. The final version of the supplement report on tube wear will be forwarded by Westinghouse by June 1, 1983. Mr. Ed Burns will also give a report on Licensing Evaluation.

Mr. Joe Reiss of Commonwealth Edison will coordinate efforts of writing the report.

B.3. COUNTERFLOW STEAM GENERATOR - MINUTES OF THE MEETING HELD MAY 18, 1983 AT PITTSBURGH BETWEEN TRC AND NRC

ATTACHMENT III

AGENDA

WESTINGHOUSE SUMMARY PRESENTATION TO THE NRC AND REVIEW GROUP

Wednesday, May 18, 1983 - Howard Johnson's

8:30 AM	Introduction
8:45 AM	Technical Overview Wear Assessment Tube Expansion Split Feedwater Safety Considerations
12:15 PM	Lunch
1:00 PM	General Discussions as Required NRC, Owners Review Group
2:30 PM	Adjourn

COUNTERFLOW STEAM GENERATOR

MINUTES OF THE MEETING HELD ON MAY 18, 1983

AT PITTSBURGH BETWEEN TECHNICAL REVIEW COMMITTEE AND NRC

Participants

Name	Organization
T. M. Novak	NRC
J. B. Hopkins	NRC
F. L. Eisinger	Consultant - HL&P
D. A. Steininger	EPRI (Steam Generator Project Office)
T. M. Williamson	Duke Power Company
Milan Copic	RKE - NE KRSKO (YU)
Guy Frederic	Electronucleaire (Belgium)
Elie Stubbe	Electronucleaire (Belgium)
Roger Riley	Public Service (Indiana)
Willy De Roovere	Electronucleaire Doel 3-4
Robert C. Iotti	Consultant - CP&L (Ebasco)
Joe Reiss	Commonwealth Edison
L. D. Butterfield	Commonwealth Edison
K. Fink	NPP - KRSKO (YU)
V. Fatvr	NPP - KRSKO (YU)
Jimmy Zaalouk	HL&P - STP
B. Poole	HL&P - STP
Ed Harris	Carolina Power & Light Co. SHNPP
Ted Jenkins	Texas Utilities
Harold R. Newth	Consultant - Texas Utilities
Albert Latham	Texas Utilities
Bob Dacko	Texas Utilities
M. Singapura	HL&P

The TRC Chairman Mr. Jimmy Zaalouk opened up the meeting with the introduction of NRC representatives and briefly described the purpose of this meeting. The purpose of this meeting he said, is to introduce the members of the TRC to NRC representatives, the progress TRC has made thus far, what is left to be done, outstanding resolutions with Westinghouse, the schedule and to answer any questions which NRC might have.

Members then introduced themselves after which the chairman gave a brief description on the other aspects of this meeting as follows:

1. On the progress TRC has made thus far, we have reviewed Westinghouse reports and have several questions and/or clarifications. We have had several correspondences and meetings with Westinghouse. He said Westinghouse has done a good job and are very cooperative. We have

resolved quite a few questions from yesterday and we hope to resolve the rest in subsequent meetings to be held this afternoon and tomorrow. He also said that there are a few gray areas which concern us and will have to be resolved before the committee can come to any definite conclusions. After resolving all outstanding items, the committee can prepare the preliminary draft Safety Report. We are primarily concentrating on Safety Report which we want to issue first followed by Long Term Operability Report. With the progress we have made thus far, we will be able to keep up to the schedule which is to submit the preliminary final report by the end of June 1983.

The chairman gave corrosion stress cracking as an example of one of the gray areas with which the committee is concerned. He said in their test the last specimen cracked in [ ]b,c,e hours. The reason for this is not known but Westinghouse is investigating. This has to be satisfactorily resolved before we can come to some definite conclusion.

After the introductory remarks by the Chairman, the following questions and answers were exchanged:

NRC: In the report on D2, D3 Steam Generators, that panel embraced topics like fabrication and other practical aspects well. Is it the same case in this report?

Answer: Yes. Long term reliability will be maintained. In view of this the committee will concentrate on the following areas:

1 - Stress corrosion chemistry area

2 - Thermal Hydraulics Vibration area

3 - Tube expansion process and post operating area

Based on the above areas of concentration, consultants and committee members, who are very well qualified in these areas were selected. We will send resumes of these personnel to NRC. We have expertise to tackle these areas very well.

We have had several highly technical meetings and discussions with Westinghouse on the above areas. Westinghouse is very cooperative and giving responses to our questions and required clarifications. Before Westinghouse presentation of May 17, 1983, we had 54 questions which were unsatisfactorily answered in their response of May 5, 1983. After the presentation and floor discussions, the outstanding questions were reduced to 30. Subsequent to the presentation, TRC members had one to one working session with Westinghouse. TRC left this session with 3 outstanding questions left to be answered.

NRC: In your report identify the items to be reviewed by NRC. We would like to review both your report and Westinghouse report. However by and large we will rely on your report.

Answer: To start writing our report, the three outstanding questions will have to be satisfactorily resolved which we hope to do in the afternoon working session. The leaders of the subgroup covering the areas will present the unresolved problems to Westinghouse. The three areas of concerns are:

1 - In Magnesium Chloride test for stress corrosion cracking conducted by Westinghouse, the last specimen cracked in [ ]^{b,c,e} hours. Response to this cracking by Westinghouse was that it is an anomaly and will trace the reason. They could not repeat this anomaly. They said they will investigate. These tests have been done on stainless steel tubes. Our concern is how good these results applicable to Inconel 600 tubes.

2 - Resolve the differences of experimental velocity data vs. our calculated higher velocity on the Westinghouse 16° Model. This deserves discussion to come to conclusion. This is not a very significant problem and will try to resolve in this afternoon session.

3 - This refers to tube expansion and long term operation. The lead plants where expansion has been done, will have monitoring and instrumentation

for vibration analysis. Since this is not a safety requirement according to Westinghouse, Utilities will have the option to implement. Westinghouse does not have specific recommendation for monitoring and instrumentation.

Based on resolutions with Westinghouse on the above gray areas, we will be able to issue the Safety Report in June.

The Chairman then requested the Leaders of the three areas to present their respective problems and the following were the presentation:

Stress Corrosion Chemistry Area

1 - On structural aspect and impact on the steam generators, Westinghouse has done an adequate job. They are conducting tests in their Forest Hills facility, particular to stress corrosion fatigue on the main and auxiliary feedwater nozzles and upper internal structures of the steam generator portion. This was a concern to us because Westinghouse had not covered this either in their reports of responses.

NRC: How will the water chemistry be controlled on the secondary side with the proposed modifications and was this reviewed?

Answer: Water chemistry on the secondary side remains unchanged and the proposed modifications do not warrant any change. Westinghouse is putting up a package on this considering EPRI's report.

2 - Westinghouse is actively pursuing the anomaly, where one tube cracked in [ ]^{b,c,e} hours in the Stress Corrosion Cracking test and they intend to resolve it. As already explained in the introductory remarks by the Chairman, this is one of the items of our concern.

Thermal Hydraulic Vibration Area

1. We have reviewed data on 16^{0} model and model data on 2/3 model. These were in depth reviews. Also reviewed air Model for D4, D5 and E Models. We have done detailed calculations on the E Model. This has produced one question in the upper pass of the Steam Generator. We will discuss this question with Westinghouse in the afternoon working discussion.

2. We have reviewed data of KRSKO. Data on D4 is generally satisfactory and will not have vibration. However, we are not satisfied with E Model. This will also be discussed in the afternoon.

3. Methods used for vibrational analysis is the same as that used for D2 and D3. The g method was used for D4, D5 and E. From KRYSKO data it has been seen that wear coefficients are not compatible with that obtained for D2, D3. Predicting less wear for the D4, D5 on the basis of D2, D3 results, for the same acceleration, force and vibration is less accurate on this non-linear tube analysis. We have to build conservatism into this. This will be discussed with Westinghouse.

4. Feedwater is flowing all over the full length of the tube in the preheater. Wear prediction by inlet flow pressure only may not be completely accurate. There are other energies which will impact on wear prediction. We do have some reservations on the Westinghouse assumption to wear prediction. This will be resolved with Westinghouse.

Tube Expansion Process and Post Operating Area

We are satisfied with most of the responses given by Westinghouse to write Safety Reports. However some aspects like fluid elastic have to be considered and will be discussed. Westinghouse has covered the expansion tools very well. The responses on tube support plates and standard eddy current results are not clear. Westinghouse is still working with eddy current vendors to obtain clearer results. We are satisfied with modifications for one cycle operation. As far as long term operability is concerned we have some questions which we hope will be resolved with Westinghouse.

General Discussions

Chairman: Requested NRC to give guidelines to help us to give report by the middle of June.

NRC: Will report contain proprietary items and will non-proprietary items be included?

NRC suggested to put together both proprietary and non-proprietary reports as a continuous document for issue and to identify the non-proprietary items.

Answer: Both proprietary and non-proprietary reports will be issued as a continuous document identifying non-proprietary items. Proprietary report will be issued by the middle of June and non-proprietary report will be issued a week later.

NRC: Does panel provide any additional information other than what Westinghouse has provided on KRSKO data and/or information?

Answer: Till now the panel has information provided by Westinghouse on KRSKO. Now that we have KRSKO representatives on the panel, they will provide any additional information if required. For the present, we do not require more information.

NRC: In the split flow design, what areas will the panel concentrate on the flow through the auxiliary nozzle?

Answer: Westinghouse response on this is adequately satisfactory.

NRC: Will you review the process of tube expansion regarding tolerances etc.?

Answer: We will review, but how much will be incorporated in our report we do not know at the present time. When we get all responses from Westinghouse, we will review the process. At present we do not have data on eddy current impact. Westinghouse is investigating on this.

NRC: What is downside of expanding more than 96 tubes?

Answer: No excessive effect. It does not warrant more than 96 tubes. In our opinion it is preferable not to expand more than this.

When preliminary report is ready it is our idea to give a presentation of the report to NRC by group leaders.

NRC: It is a good idea and we will have the presentation.

With this the meeting was adjourned for the Westinghouse presentation to NRC and other members of the owners group.

APPENDIX C

LISTING OF FIRST ROUND TRC QUESTIONS TO WESTINGHOUSE

LISTING OF FIRST ROUND TRC QUESTIONS TO WESTINGHOUSE

I. Thermal Hydraulics/Tube Vibration/Model Testing/Safety Analysis

A. The Committee would like Westinghouse to provide a review of thermal hydraulic analysis that has been completed to support operation and design stress analysis of steam generators with expanded tubes and/or split feedwater flow. The following areas are of interest:

1. Design basis for operation of preheater during transients.

2. Increased flow in upper passes of preheater over life as a result of decreased outleakage due to baffle plate hole corrosion.

B. Provide the detailed design basis criteria for wear prediction and wear limits on useful tube life.

C. Provide basis for criteria used in selecting tubes which are recommended for expansion in order to limit vibration potential.

D. Provide basis for flow simulation in the 16-degree model to the actual operational steam generator. Review and compare the tube forcing functions obtained from the 16-degree model and the 2/3-scale model.

E. Discuss the criteria used to ensure that the 16-degree model testing bounds the vibrational aspects which may exist in all of the "as-built" steam generators (i.e., tolerances on baffle plate holes and distortion effects on operation). There are structural differences which exist between D4, D5, and E generators and several steam generators have special individual structural modifications which may not be generic. How were these structural differences evaluated in the development of the test bounding criteria?

F. Provide details of criteria changes that might require changes to the safety analysis. Operation of split feedwater conditions could result in an increase in reactor core average temperature. Identify acceptance criteria for these average temperature changes and any impact on safety analysis or core performance.

G. Provide the basis for extending D4 model test data to make predictions of flow velocities and tube vibration in the Model E steam generator. Discuss specific modifications needed to the 16-degree model to provide for testing of the exact Model E design. Provide an estimate of time required to complete such model modifications.

H. Discuss the possibility of reducing the tube vibration in Model E via splitflow (i.e., internal and external). What splitflow percentage would be required to be equivalent to tube expansion?

I. Discuss the dominant sources of vibration excitation and steps needed to ensure sufficient safety against fluid elastic vibration. Evaluate the effect of possible tube "binding" at the points of expansion upon the vibrational characteristics of the tubes.

II. Structural Integrity/NDE/Vibration Monitoring

A. Discuss the need for post-fix vibration monitoring. Provide criteria to ensure that the vibration zone has been bounded during tube expansion.

B. Provide criteria to be used on eddy current examination of the expanded tube region. Discuss what special eddy current equipment and testing procedures will be required to monitor the tube expansion zones.

C. Discuss the possibility of monitoring tubes during expansion with acoustical emission. Provide basis which will be used to ensure that flaws have not been induced into the tube wall during expansion.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG-1014

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Safety Evaluation Report related to the D4/D5/E Steam Generator Design Modification

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)

5. DATE REPORT COMPLETED
MONTH October YEAR 1983

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

DATE REPORT ISSUED
MONTH October YEAR 1983

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as 9 above

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

13. TYPE OF REPORT
Safety Evaluation Report

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

This Safety Evaluation Report (SER) related to the D4/D5/E steam generator design modification has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The purpose of this SER is to issue the staff's evaluation of the acceptability of the design modification for both installation and full-power operation in D4/D5/E steam generators based on the Counterflow Steam Generator Owners Review Group's Technical Review Committee Report of July 1983. Those contributing to this report are listed in Appendix A.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
Unclassified

21. NO. OF PAGES

20. SECURITY CLASS (This page)
Unclassified

22. PRICE
S

NRC FORM 335 (7-77)