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

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NOTICE

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ACKNOWLEDGMENT

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

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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 podification 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.

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- The proposed modifications can be implemented without significant impact on long-term plant operations.
- Westinghouse acceptance criteria are adequate and whether the proposed modifications meet the acceptance criteria.
- The proposed modifications can be implemented without impacting plant completion schedules.
- 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.

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

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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

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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.

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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.

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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.

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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 pre- dicted, 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 mot on a generic basis	Section 5.5/p. 5-39
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

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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)



Fig. 1.1-1. Preheat steam generator geometry

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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 stean selerator 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





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 expansi 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 Utiliries (Electronucleaire) Nublearna 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.*

^{*&}quot;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.

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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,

 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,

 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:

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

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

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- 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.
Model	D4/D5	D4/D5	D4/D5	E
No. of loops	2	3	4	A11
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

TABLE 2.3-1 INITIAL THERMAL-HYDRAULIC DESIGN OBJECTIVES

2.4. MECHANICAL DESIGN AND QUALIFICATION PROCESS/TUBE EXPANSION CRITERIA

2.4.1. Mechanical Design Criteria

The mechanical design criteria are as follows:

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

- 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.
- The parallel portion of the expanded tube length shall be greater than or equal to []a,b,c,e.
- The expanded tube shall be free to slide axially over a distance greater than or equal to []a,b,c,e.
- The diameter variation within the parallel expanded portion of the tube shall be less than or equal to []a,b,c,e.
- 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.
- 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 PEVIEWED 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
- Startup
- Step Load Increase
- Feedwater Cycling
- Oscillations About Steady State
- Loop Startup
- Temperature-Power Operating Map

Upset Conditions

- · Loss of Load
- Loss of Feedwater
- Partial Loss of Flow
- Boron Dilution
- · Reactor Trip
- Reactor Trip with Cooldown Actuating Safety Injection
- Control Rod Drop
- Excessive Load Increase

Emergency Conditions

- Small LOCA
- · Complete Loss of Flow

Faulted Conditions

- Steamline Break (Double Ended)
 Control Rod Ejection
- Feedline Break (Double Ended)
- Reactor Coolant Pump Locked Rotor

- Unit Unloading
- Shutdown
- Step Load Decrease
- Load Rejections
- Boron Concentration Equalization
- Loop Shutdown
- · Loss of Power
- Excessive Feedwater
- Rod Withdrawal
- Inadvertent RCS Depressurization
- Reactor Trip with Cooldown
- Inadvertent Startup of an Inactive Loop
- Inadvertent Safety Injection
- Bubble Collapse
- Small Steamline Break
- LOCA
- 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. ۴

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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:

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



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

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



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

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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 0.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.

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 severa! 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.



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

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Tubes selected for accelerometer installation in the Model E lead plant will be designated following tube expansion map determination by Westinghouse. 1. 26 M



TABLE 4.3-1 MODEL D4/D5 ACCELEROMETER LOCATIONS

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.

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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-tobaffle 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 baris 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



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:

- High vibration levels existed in the T-slot area but also in areas away from the T-slot.
- The wear information on the three removed tubes was consistent and is a good basis for wear prediction.
- There were no eddy current indications, which means there was no tube wear in excess of 10% of wall thickness.
- No clear correlation has been established between the tube-tobaffle-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. <u>Air Model</u>. 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. <u>Single Tube Model</u>. 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 piezeelectric 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. <u>Installation Model</u>. 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. <u>1/4-Scale Water Model</u>. 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

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velocity data, and fluctuating velocity measurements. Several configurations were tested, including a simulation of a 90-degree sector.

5.1.2.5. <u>2/3-Scale 180-Degree Water Model</u>. 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 non-linear tube vibration model described later. This model is shown in Fig. 5.1-3.

5.1.2.6 <u>Sixteen-Degree Full-Scale Water Model</u>. 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 mid-dle 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 cross-passes. 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. Flow-meters 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.

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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 Velocity (2/3-Scale Data)		Model E			
			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 lowfrequency 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 timehistories 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. <u>Work Rate Method</u>. 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.

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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. <u>G-Delta Method</u>. 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:



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. <u>G-Delta Assessment Method</u>. 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:

 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.

- Peak-to-peak accelerations. The peak-to-peak acceleration value is obtained by visual observation of the obtained time-history records.
- 3. <u>Root mean square displacement</u>. This displacement is computed from measured tube acceleration signals. An RMS displacement spectrum is obtained by double integration within the frequency range.
- 4. <u>G-Delta</u>. 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:

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

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 (Kg) FROM KRSKO REMOVED TUBE DATA $V_{wear} = K_g \Sigma$ (G-Delta)_i T_i

	R49 C56	I	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					김동물학
(100/0)/(70/30) G-Delta					일을 가지
Σ (G-Delta) _i T _i Phase I					
Phase II					1.15
Total					
Fitted Wear Coefficient					
KS					1.0
Average Kg					1.1
Worst Case Kg					
G-Delta Predictions - Scar Depth (mils)					
Nominal					1
Worst Case					



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.

 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

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 nonlinear 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 fullpower 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. <u>Work Rate Method</u>. 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. <u>Application of Wear Methods</u>. 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:

- 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.
- 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 0.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



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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 $1.2^{\circ}F$ 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. <u>Stress Criteria and Design Loads</u>. 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. <u>Analysis Method</u>. 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				
.00% load	Load/unload 15%-100%				
oss 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 power Loss of flow Inadvertent startup of inactive loop Inadvertent safety injection Loss of load				
oss 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				
arge step load decrease	Large step load decrease				
xcess feedwater	Excess feedwater				
xcess bypass feedwater	Excess bypass feedwater				
5% power	25% power				
orward flushing	Forward flushing at 32°F Forward flushing at 200°F Forward flushing at 250°F				
eatup/cooldown	Heatup/cooldown Turbine roll				
<pre>nadvertent reactor coolant system depressurization (zero thermal stress + pressure)</pre>	Reactor trip Inadvertent reactor coolant system depressurization Turbine roll				
perating basis earthquake	Operating basis earthquake				
rimary hydrotest	Primary hydrotest				
rimary leak rate test	Primary leak rate test				

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Table 5.6-1 (Continued)

Umbrella Transient

Transients Included

Secondary hydrotest Tube leakage at 840 psi

Texas and the second second

Secondary hydrotest 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.

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:



(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

^{*&}quot;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 10¹¹. 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. <u>Tube Plugging Analysis</u>. 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. <u>Structural Evaluation of Feedwater Split Modification</u>. 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. Westinghous: 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 oczzles is less than 1.0. Additionally, the following conclusions are reached:

- 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.
- 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:

- 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.
- 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 monitor-ing 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

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PRIMARY TO SECONDARY PRESSURE DIFFERENTIAL, ΔP_i (PSI)

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



AXIAL CRACK LENGTH, L (IN.)

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_1 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 <u>et al</u>. 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.

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

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

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,

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

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

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NORMALIZED AXIAL CRACK LENGTH, λ

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



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TABLE 5.6-2						
DIMENSIONS	OF	EXPANDED	STEAM	GENERATOR	TUBES	

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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, \overline{o} , 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 0.D. expansion results in an increase in the mield 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 govern-ing 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

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MEAN RADIUS TO THICKNESS RATIO, Rm/t

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

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

where R = KMIN/KMAX,

 λ = material constant, approximately 2.

For general operation of the steam generator, the value of ΔK is small compared with $K_{\rm MIN}$; 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, $K_{\rm MIN} = 11.1$ ksi $\sqrt{10}$, and $K_{\rm MAX} = 11.6$ ksi $\sqrt{10}$, 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 $\sqrt{10}$, value of 4K is about 10% of the threshold value of about 5 ksi $\sqrt{10}$.* (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

[&]quot;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. 0.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

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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.

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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 sitespecific 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 sitespecific 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, administra- tion, 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, con- struction, startup, project man- agement, design, mining, market- ing, and operation	25
J. H. Goldberg	Houston Lighting & Power Company	Member	Nuclear engineering, construc- tion, design, project manage- ment, and modifications	25
Guy A. Frederick	Electrobel (Electro- nucleaire - Belgium)	Member	Engineering, design, specifica- tion, 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
м. с	. Zaalouk	Houston Lighting & Power Company	Chairman	Reactor engineer, supervisor, professor, mechanical-nuclear, startup and construction support, and design modifications	26
A. B	. Poole	Honston Lighting & Power Company	Assistant Chairman	Mechanical-nuclear, project coordination, design, stress cal- culations, 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, frac- ture 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. R	eiss	Commonwealth Edison Company	Member	Plant operations, project engi- neering, 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 analy- sis, and program manager	11
R. Hanford	Carolina Power & Light Company	Member	Resident welding/material engi- neer, construction support, qual- ity assurance, corrosion, stress corrosion cracking, and ASME Code	15
Willy Ch. P. De Roovere	EBES (Electronucleaire - Belgium)	Member	Plant design, operation, and main- tenance, 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, con- struction, and plant startup and modification	
Vladimar Fatur	NPP, Krsko (Yugoslavia)	Member	Engineering, professor, and research	10

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TABLE 6-2 RESUMES OF STEERING COMMITTEE MEMBERS

LAWRENCE D. BUTTERFIELD, JR. SECTION ENGINEER COMMONWEALTH EDISON COMPANY

L. D. (Del) Butterfield, Jr., is a Section Engineer in the Station Nuclear Engineering Department of the Commonwealth Edison Company and has been employed by them since 1966.

At Commonwealth Edison, Mr. Butterfield is responsible for supervising the Reliability and Design Group and Special Projects. His previous assignments at Commonwealth Edison included Zion Project Engineer in the Station Nuclear Engineering Department, Supervising Reactor Analysis Engineer in the Nuclear Fuel Services Department, Nuclear Licensing Administrator for BWR's and PWR's, Administrative Assistant to the Dresden Station Superintendent, and several other technical assignments at Dresden Station and the Nuclear Fuel Services Department. Prior to joining Commonwealth Edison, Mr. Butterfield was the Reactor Supervisor for the University of Florida Training Reactor.

Mr. Butterfield received his Bachelor of Electrical Engineering and Master of Science in Nuclear Engineering Degrees from the University of Florida.

Mr. Butterfield is the Chairman of the Steering Committee of the Counterflow Steam Generator Owners Review Group. He is a registered Professional Engineer in the State of Illinois and a member of the American Nuclear Society.

A. B. CUTTER

VICE PRESIDENT

CAROLINA POWER & LIGHT COMPANY

A. B. Cutter is Vice President, Nuclear Plant Engineering, at Carolina Power & Light Company and has been employed by them since April 1980.

At CP&L, Mr. Cutter is responsible for the management of the Nuclear Plant Engineering Department. He is responsible for design, engineering, procurement, and management of A/E and CP&L resources, in support of construction of new nuclear plant facilities. In this position he is also responsible for engineering, procurement, and studies, for modifications to comply with regulatory requirements, improve performance and/or reliability of existing nuclear power plants, and contacts with A/E firms.

Mr. Cutter has held positions as Project Manager-Prairie Island and Kewaunee Nuclear Steam Supply System, Program Manager-Fast Flux Test Facility, Program Manager-Clinch River Breeder Reactor, Director-Iran Projects, and Manager-Project Operations with Westinghouse Electric Corporation. He also served 11 years with the U.S. Navy in various capacities and was Chief Engineer, SIW Prototype, Naval Reactor Facility, Idaho.

Mr. Cutter received a B.S. degree in Chemical Engineering in 1956 from the University of Rochester and an M.S. degree in Nuclear Science and Engineering in 1972 from Carnegie-Mellon University. He completed Advanced Nuclear Power Training Courses in the U.S. Navy, several graduate courses in nuclear engineering at the University of Idaho, Tuck Executive Program at Dartmouth College, Brookings Institute for Government Operation at Brookings Institute, and numerous short courses in Project Management, A/E Management, and General Management Techniques at Westinghouse.

Mr. Cutter is a registered Professional Engineer in the state of North Carolina and a member of the American Nuclear Society.

WILLIAM M. PETRO

VICE PRESIDENT

PUBLIC SERVICE COMPANY OF INDIANA

William M. Petro is Vice President, Nuclear Services, at Public Service Company of Indiana and has been employed by them since September 1980.

At Public Service Company of Indiana, Mr. Petro is responsible for engineering, procurement, construction, and startup of two 1100-MW PWR plants at Marble Hill Nuclear Generating Station.

Mr. Petro has held positions as Assistant Project Manager at Southern California Edison Company and Construction Manager at Arizona Public Service Company. He served as Site Construction Manager, Project Manager, and Regional Sales Manager with Westinghouse Electric Corporation. At Bechtel Power Corporation he served as Design Engineer, Lead Startup Engineer, and Project Mechanical Design Engineer. He also held the position of Shift Operating Foreman in Nuclear Plant Operations of Duquesne Light Company.

Mr. Petro received a B.S. degree in Marine Engineering in 1957 from the U.S. Naval Academy and took courses in Utility Executive Management at Arizona State University.

He is a member of the Project Management Institute, American Nuclear Society, and Atomic Industrial Forum. He is a past president and co-founder of the Phoenix Arizona Chapter of Project Management Institute.

J. H. GOLDBERG

VICE PRESIDENT

HOUSTON LIGHTING & POWER COMPANY

J. H. Goldberg is Vice President of Nuclear Engineering and Construction at Houston Lighting & Power Company and has been employed by them since October 1980.

At HL&P, Mr. Goldberg has direct complete responsibility for the South Texas Project.

Mr. Goldberg has held positions in Bethlehem Steel at their Quincy ship building yard where he was involved in design and construction of nuclear warships. He later worked for General Dynamics Corporation where he was involved with the construction of four nuclear submarines. Mr. Goldberg also held positions at Stone & Webster Engineering Corporation as a Project Manager of Beaver Valley I, Chief Engineer of the Engineering Mechanics Division, and Vice President and Deputy Director of Construction. At Stone & Webster he was involved in construction of a number of nuclear power projects and was in charge of the modification of two operating nuclear plants.

Mr. Goldberg received a B.S. from the U.S. Merchant Marine Academy and an M.S. in Nuclear Engineering from MIT.

Mr. Goldberg is a member of the American Nuclear Society and a registered professional engineer in seven states.

GUY A. FREDERICK MANAGER, NUCLEAR EQUIPMENT ELECTROBEL (ELECTRONUCLEAIRE - BELGIUM)

Guy A. Frederick is a Manager, Nuclear Equipment, at Electrobel, architect engineers for the three Tihange units. He has been employed by them since 1972.

At Electrobel, Mr. Frederick was responsible for the engineering management of the reactor, auxiliary and waste mechanical equipment for the Tihange units. His responsibilities included the design specification, Owners' review, and certification of contractor-supplied ASME Section III components; equipment procurement specification; ASME Section XI in-service inspection; and contractor surveillance. Mr. Frederick is also responsible for analyzing and recommending solutions to various equipment, heat exchanger, pump, piping fabrication, startup, and operating problems.

Mr. Frederick has been involved with design, engineering, testing, initial startup, and in-service surveillance of equipment associated with NSSS, auxiliary system, and waste.

Prior to joining Electrobel, Mr. Frederick worked at Metallurgy Research Center (Belgium) where he was in charge of the fracture mechanics and fatigue research work. He also worked at Stein et Roubix (Belgium) on the design of thermal power station systems.

Mr. Frederick graduated in 1966 from Liege University (Belgium), majoring in Electrical and Mechanical Engineering. He has also completed numerous additional courses in the areas of welding technology, NDT techniques, and fracture mechanics.

Mr. Frederick was a member of the Subcommittee of the International Institute of Welding, "Significance of Welding Defects." He is a member of various professional organizations in Belgium and is the Belgian representative to the Technical Advisory Committee of the Steam Generator Owners Group. TABLE 6-3 RESUMES OF TECHNICAL REVIEW COMMITTEE MEMBERS

M. G. (JIMMY) ZAALOUK PRINCIPAL ENGINEER HOUSTON LIGHTING & POWER COMPANY

M. G. Zaalouk is a Principal Engineer, Nuclear at Jouston Lighting & Power Company and has been employed by them since September 1981.

At HL&P, Dr. Zaalouk is responsible for providing the technical support for the South Texas Project. Before joining HL&P, Dr. Zaalouk held positions of Senior Engineer, Project Engineer, and Principal Engineer in the Mechanical/Nuclear Department at Carolina Power & Light Company, where he was responsible for nuclear systems design review, construction support, and review and implementation of Regulatory and Code requirements, and assured design compliance including Safety Analysis and ALARA requirements for Shearon Harris Nuclear Power Plant (four units); provided start-up/ operations support of the Brunswick plant; provided engineering management of all mechanical/nuclear plant design modifications for the H. B. Robinson and Brunswick Nuclear Power Plants; and headed the company TMI-2 Corporate Investigative Team. Prior to joining CP&L he taught at North Carolina State University as Assistant Professor, Power Systems and codirected an NSF- and NRC-funded research program to develop a temperature control system to prevent burnout of heating elements when exceeding critical heat flux values. He has also worked as a Reactor Engineer and Nuclear Engineering Unit Supervisor for the Egyptian Atomic Energy Agency and at the Norwegian Institute for Atomic Energy where he developed an advanced computer code now in use by industry in light water reactor core design and analysis.

Dr. Zaalouk received a B.S. in Electrical Engineering in 1957 from Cairo University, Egypt, and an M.S. and Ph.D., both in Nuclear Engineering, in 1962 and 1966, respectively, from North Carolina State University.

Dr. Zaalouk is a member of the American Nuclear Society where he was co-chairman of the Reactor Operations Division Technical Program Committee (1979-81) and was appointed to the ANS National Program Committee in 1982. He was an Adjunct Associate Professor (1972-81) at North Carolina State University. He served as a member of ANS Standards Committee ANS-19, "Reactor Core Design" (1973-81), and is an IEEE Technical Reviewer-Journal of Instrumentation and Control.

Dr. Zaalouk actively participated in Professional Education and Training Programme conducted by Argonne National Laboratory, the American Nuclear Society, North Carolina State University, and CP&L. He also participated as an active member in Mark I-BWR Owners Group, Steam Generator Owners Group, and TMI-2, PWR Owners Group.

Dr. Zaalouk is the author of approximately 40 technical papers and reports, including a technical film "Boiling Process with Sine Inputs: All Boiling Modes," and is a registered Professional Engineer.

A. BRUCE POOLE STAFF ENGINEER HOUSTON LIGHTING & POWER COMPANY

A. B. Poole is a Staff Engineer, Nuclear Systems at Houston Lighting & Power Company and has been employed by them since 1980.

At HL&P, Mr. Poole is responsible for the reactor coolant system and components. His specific responsibilities include the steam generator, reactor vessel, reactor coolant pump, and associated equipment.

Prior to joining HL&P, Mr. Poole held positions of Associate Engineer, Design Engineer, Senior Engineer, and Project Coordinator with Westinghouse Electric Corporation in their Bettis Atomic Power Laboratory. He worked on the Advanced Submarine project. His responsibilities included stress calculations, material testing, thermal-hydraulic analysis, detail design and structural analysis of the reactor core barrel, piping design for the emergency core cooling system, and design and calculations on the in-core instrumentation. As Senior Engineer, his responsibilities included design, analysis, fabrication follow-up, and delivery receipt inspection of the core barrel and closure head. Prior to joining Westinghouse he served as Associate Engineer at LTV Aerospace Corporation and as Project Engineer at Standard Manufacturing Company.

Mr. Poole received his B.S. in Aeronautical Engineering in 1968 from the University of Texas and is preparing to receive his MBA within the next year from the University of Houston.

Mr. Poole is the HL&P representative on the Technical Advisory Committee of the EPRI Steam Generators Owners Group. At Westinghouse Electric Corporation, he served as an editor of the ASP Core Reference Design Report provided to Naval reactors.

FRANTISEK L. EISINGER

MANAGER FOSTER WHEELER ENERGY CORPORATION CONSULTANT TO HL&P

F. L. Eisinger is Manager of the Equipment Design Department and Staff Engineering Operations at Foster Wheeler Energy Corporation and has been employed by them since 1969.

At Foster Wheeler, Dr. Eisinger is responsible for vibration and stress analysis of steam generators and associated equipment. He is also responsible for the analysis for flow-induced vibration, seismic analysis of steel structures and piping systems, noise control, dynamics of machines and rotor dynamics, stress analysis, fatigue, structural dynamics, and development of stress analysis methods.

Dr. Eisinger has held positions as Head of Applied Mechanics Department, Associate Professor of Applied Mechanics, Associate Dean for Education, Assistant Professor of Applied Mechanics, and Lecturer in Czechoslovakian universities.

Dr. Eisinger received his B.S. and M.S. in Mechanical Engineering in 1952 from the College of Engineering, Slovak Technical University, Czechoslovakia. He received his Ph.D. in Applied Mechanics in 1965 and his Associate Professor's in 1966 from the same university.

Dr. Eisinger received awards for extraordinary achievements in Engineering Education in 1962 and 1967. He is a member of the ASME Metal Properties Council, Subcommittee Three on Fatigue, and is also a member of the HTRI Vibration Task Force.

Dr. Eisinger has published 22 papers in the field of structural and vibration analysis, has 6 patents on vibration prevention and stress reduction, and is author of three college textbooks on applied mechanics and strength of materials.

ROBERT C. IOTTI CHIEF ENGINEER EBASCO SERVICES, INCORPORATED CONSULTANT TO CP&L

R. C. Iotti is Chief Engineer, Applied Physics, at Ebasco Services, Inc., and has been employed by them since 1971.

As Chief Engineer, Dr. Iotti is responsible for planning and directing all engineering efforts toward shielding and radiation monitoring of nuclear installations. He is also responsible for planning, direction, and performance of studies of transient effects in containments and enclosures resulting from high-energy pipe breaks, wind loadings, and detonations; studies of transient fluid flow phenomena; and heat transfer studies. He developed and directed work on establishing methodologies for evaluation of piping vibrations during steady-state and transient loading conditions; for simulation of transients in power plants; and for design against hydrogen detonations. He also developed a computer program to describe the boiler implosion phenomenon and managed in-house research and development programs on solar energy use. Dr. Iotti has directed and performed work required for the complete analysis of the reactor vessel supports of a PWR plant under LOCA conditions resulting in asymmetric pressure loads inside the reactor vessel and the reactor cavity. He engineered and designed novel neutron streaming shields for application in existing PWR plants.

Dr. Iotti previously was Engineer, Senior Engineer, Principal Engineer, and Supervisor in Applied Physics at Ebasco. In these capacities, he was responsible for shielding design, radiation monitoring systems, studies on turbine, detonation, and tornado driven missiles, and other developmental work.

Before joining Ebasco, Dr. Iotti taught for six years at Kansas State University where he received his B.S., M.S., and Ph.D., all in Nuclear Engineering. He is a registered Professional Engineer in New York State.

Dr. Iotti has directed work performed by consultants for a group of utilities owning CE reactors, and another group owning B&W reactors, to assess the probability of pipe ruptures. In this capacity, he acted as consultant to SAI in the required probabilistic and fracture mechanics studies.

Dr. Iotti has published over 20 papers and 5 reports. His paper on "Neutron Streaming - The Problem and Engineered Solution" won an award. He served on American Nuclear Society Subcommittee ANS 55.2 on Protection Against the Effects of Pipe Whip and is a member of the Atomic Industrial Forum Ad-Hoc Committee on Pipe Whip.

ROGER W. RILEY

NSSS SUPERVISING ENGINEER

PUBLIC SERVICE COMPANY OF INDIANA

R. W. Riley is the NSSS Supervising Engineer for Public Service Company of Indiana and was employed by them in 1982. Mr. Riley is responsible for the NSSS design at the Marble Hill Nuclear Generating Station.

Prior to joining Public Service Company of Indiana, Mr. Riley worked for the Cincinnati Gas & Electric Company for 18 years on fossil and nuclear projects.

Mr. Riley received a B.S. in Mechanical Engineering from the University of Cincinnati and has also done post-graduate work there. He has been certified as Special Inspector of Pressure Piping in the State of Ohio and also holds a secondary teaching certificate from Cincinnati Baptist College and is certified to teach in vocational schools.

Mr. Riley, while working on the W. H. Zimmer Nuclear Project, represented Cincinnati Gas & Electric Company on the Mark II BWR Owners Group, Mass Energy Subcommittee and Safety Valve Test Program. He has performed numerous heat exchanger sensitivity studies and fluid flow calculations and authored Chapter 8 of the Zimmer Design Assessment Report "Suppression Pool Temperature Response." He was also the responsible engineer assigned to respond to NUREG 0313, which addresses intergranular stress corrosion cracking in BWR's.

JOSEPH F. REISS PRINCIPAL ENGINEER COMMONWEALTH EDISON COMPANY

J. F. Reiss is Principal Engineer, Station Nuclear Engineering Department of the Commonwealth Edison Company and has been employed by them since 1976.

At Commonwealth Edison, Mr. Reiss is responsible for the independent review of modifications, as well as carrying on operational and maintenance support for the Zion nuclear power station. His previous assignments at Commonwealth Edison included field technical staff engineer at Zion station as well as several of Commonwealth Edison's fossil plants. His fossil plant experience is general in nature, and his nuclear plant experience is centered around primary and reactor safeguards systems with special emphasis on flow-induced vibration and NSSS loose parts monitoring systems.

Mr. Reiss received his B.S. in Physics from Loyola University of Chicago and is the Commonwealth Edison representative on the ASME committee studying flow-induced vibration in heat exchangers.

DAVID A. STEININGER

PROJECT MANAGER

ELECTRIC POWER RESEARCH INSTITUTE

David A. Steininger is Project Manager in the Steam Generator Project Office of EPRI and has been employed by them since 1978.

At EPRI, Mr. Steininger has initiated and managed research contracts in the areas of three-dimensional thermal/hydraulic computer code development, flow and boiling phenomena within steam generators, corrosion fatigue and "fretting and wear" characteristics of Inconel 600, and flow-induced tube vibration of steam generator tube configurations. Additionally, Mr. Steininger has been involved with individual electric power utilities such as Rochester Gas and Electric, Swedish State Power Board, and others, consulting on their steam generator problems.

Mr. Steininger has held positions as Program Manager at General Electric Company and Senior Engineer at General Atomic Company. While at General Electric, Mr. Steininger received a General Manager's Award for management and technical excellence.

Mr. Steininger received a B.S. in Aeronautical Engineering in 1970 and an M.S. in Nuclear Engineering in 1972, both from Purdue University.

Mr. Steininger is a member of the Nuclear Heat Exchanger Committee of the ASME.

RAYVINE HANFORD RESIDENT ENGINEER - METALLURGY/WELDING CAROLINA POWER & LIGHT COMPANY

Rayvine Hanford is Resident Enginer, Metallurgy/Welding in the Harris Site Management Section, Nuclear Plant Construction Department of Carolina Power & Light Company and has been employed by them since 1971.

Mr. Hanford was transferred to the Harris site at New Hill, North Carolina, in 1979 and promoted to Resident Engineer there in January 1982. His previous assignments at CP&L have included establishing a Quality Assurance Engineering Unit, including the recruiting and hiring of new personnel, the establishment of QA procedures, and an engineering surveillance program. He also worked closely with Brown & Ro t on development of the current welding procedures used at Brunswick for nuclear power plants, investigation of the boiler drum failure during hydrotesting at Roxboro in April 1973, extensive corrosion problem investigations at BSEP on a plant-wide basis, and materials engineering and failure analysis; represented utilities for CP&L as a team member on ASME surveys for Code Stamp Authorization; and was nominated as a Task Force member of the EPRI Nuclear Systems and Materials Task Force.

Mr. Hanford received his B.S. in Metallurgical Engineering from North Carolina State University in 1960 and has completed post-graduate work in thermal engineering at Old Dominion College. He is a member of the American Welding Society, American Society for Metals, and American Society for Materials & Testing (ASTM).
WILLY CH. P. DE ROOVERE ENGINEERING MANAGER EBES (ELECTRONUCLEAIRE - BELGIUM)

Willy Ch. P. De Roovere is an Engineering Manager at EBES (Electronucleaire - Belgium) and has been employed by them since 1975.

At EBES, Mr. De Roovere is responsible for operation, maintenance and refueling, maintenance and outage. At EBES, he was also responsible for electrical maintenance as Head, Electrical Maintenance Department. Prior to joining EBES, Mr. De Roovere was employed by Interescaut Thermal Power Plants, Belgium, as Head, Electrical Maintenance and Operations Department.

Mr. De Roovere has an M.S. in Mechanical Engineering, Electrical Engineering, and Nuclear Engineering, all from the University of Ghent, Belgium.

Mr. De Roovere is a member of the Board of the Mechanical Division of the Flemish Engineering Society, Steering Committee for Steam Generator Performance and Availability, and Plant Operating Review Committee for the Doel 3-4 Nuclear Power Plants, all in Belgium. He is also a registered Professional Engineer in Belgium and is in the process of becoming a recognized SRO for Doel 3 and Doel 4.

ELIE J. STUBBE PRINCIPAL ENGINEER TRACTIONEL (ELECTRONUCLEAIRE)

Elie J. Stubbe is Supervisor, Thermal Hydraulic Section of the nuclear department at Tractionel, architect/engineers for the Doel Nuclear Power Plants, and has been employed by them since 1971.

Dr. Stubbe has provided engineering and analysis support in the areas of containment and subcompartment design; equipment qualification; and system behavior under transient hydraulic loads (waterhammer, blowdown, and wave forces). His activities currently concentrate on the evaluation of primary and secondary system behavior under accidental and operational transient conditions, using second generation loop codes.

Prior to joining Tractionel, Dr. Stubbe was research assistant at the Canadian National Research Council in Ottawa working with fusion plasma devices (shock tube, pinches), where he conceived and developed a miniature eddy current probe for measuring plasma electrical conductivity. He also holds the position of Lecturer at the University of Louvain, Belgium.

Dr. Stubbe graduated in Mechanical and Electrical Engineering from the University of Louvain. He received his M.S. in Applied Science from the Institute of Aerospace Sciences, University of Toronto (Canada) and his Ph.D. in Electrical Engineering from the University of Ottawa (Canada).

Dr. Stubbe is a member of the American Nuclear Society, the IEEE, and the Board of Mechanical Division of the Flemish Engineering Society.

KRESIMIR FINK CHEMISTRY SUPERVISOR NPP, Krsko

K. Fink is Chemistry Supervisor, NPP, Krsko and has been employed by them since October 1981.

At NPP, Krsko, Mr. Fink is responsible for chemistry, radiochemistry, materials, and corrosion. Prior to joining NPP, Krsko, he taught advanced instrumental analysis at Zagreb University, Yugoslavia.

Mr. Fink received his B.S. in Chemistry in 1975 and M.S. in Analytical Chemistry in 1980, both from Zagreb University. He later specialized in modern instrumental analysis while at Southern Illinois University, Carbondale.

Mr. Fink is the author of several technical papers on analytical chemistry.

PETAR BILCAR CONSTRUCTION MANAGER NPP, Krsko

Petar Bilcar is a Construction Manager, Nuclear Power Plant at NPP, Krsko, Yugoslavia.

At NPP, Krsko, Mr. Bilcar is responsible for engineering, procurement, construction, and plant start-up of the 632 MW PWR plant. Presently, he is responsible for studies and modifications to improve performance and reliability of the nuclear power plant at NPP, Krsko.

Mr. Bilcar received his B.S. in Electrical Engineering from the University of Zagreb, Yugoslavia.

VLADIMIR FATUR HEAD OF ISEG NPP, Krsko

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Vladimir Fatur is the Head of Independent Safety Engineering at NPP, Krsko and was employed by them in 1982.

Prior to joining NPP, Krsko, Dr. Fatur was Vice President, Energy, Industry and Civil Engineering Committee of the Republic of Slovenia for five years. Dr. Fatur also held the positions of Assistant Professor at the University of Ljubljana and Research Engineer at the Institute of Materials and Construction, Ljubljana.

Dr. Fatur received his B.E., M.S., and Ph.D. from the University of Ljubljana.

Dr. Fatur is the author of 20 technical papers and reports.

7. BIBLIOGRAPHY

- "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, May 1982.
- "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, July 1982.
- "Model E, Steam Generator Preheater Tube Air Flow and Vibration Test Results," Preliminary Westinghouse Report, October 27, 1982.
- "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, November 1982.
- "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, March 1983.
- "Counterflow Preheat Steam Generator Tube Expansion," Westinghouse Report, April 1983.
- "Preheat Steam Generator Customer Working Group Meeting," Westinghouse Report, May 1983.
- "Counterflow Preheat Steam Generator Tube Vibration," Westinghouse Summary Report, June 1983.
- "Counterflow Preheater Steam Generator Tube Expansion," Westinghouse Report, June 1983.
- "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.
- "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.
- Preheat Steam Generator Status, Westinghouse Presentations, February 1982.
- Wear Assessment for the McGuire 1 Steam Generator Tubing, Westinghouse Presentation to U.S. Nuclear Regulatory Commission, April 1982.

7-1

- 15. "KR3KO", Letter of March 28, 1983, from NUS Corporation.
- 16. G. J. Theus, "Stress Corrosion Cracking Tests of Alloy 600," <u>Workshop</u> <u>Proceedings: U-Bend Tube Cracking in Steam Generators</u>, 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.
- "IGA of Oconee 1 Steam Generator," Letter from D. A. Steininger to J. L. Epstein (Westinghouse), March 23, 1983.
- J. Seivar, "Normal Operating Radiation Levels in Pressurized Water Reactor Plants," Nuclear Technology, Vol. 36, 12. 48-55, 1977.
- 20. R. D. Blevins, Flow Induced Vibrations, Van Nostrand-Reinhold, 1977.
- 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.
- Strommer <u>et al.</u>, "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.
- 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.

7-2

- D. Brock, <u>Elementary Engineering Fracture Mechanics</u>, p. 360, Noordhoff Mtf Publishing, Leyden, Netherlands, 1974.
- 29. P. G. Hodge, <u>Plastic Analysis of Structures</u>, McGraw-Hill, New York, 1959.
- L. S. Beedle, <u>Plastic Design of Steel Frames</u>, Riley Interscience, New York, 1958.
- B. G. Neale and J. S. Symonds, <u>Plastic Design of Steel Structures</u>, John Wiley, New York, 1963.
- M. Vagins, et al., "Steam Generator Tube Integrity Program Phase J Report," U.S. Nuclear Regulatory Commission, NUREG/CR 0718, September 1971.
- 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

F

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:

- To determine if the proposed modifications will eliminate major tube vibration.
- 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
- <u>Steering Committee</u> (STC) Group responsible for the management of the Owners Review Group. Group will consist of the following:
 - 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.
- <u>Technical Review Committee</u> (TRC) Group responsible for the technical review of the final proposed Westinghouse modification program. This group will consist of:
 - 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 EPXI 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.
- 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.
- 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 LEATTED 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

Bruce Poole F. L. Eisinger Joe Reiss Del Butterfield Robert C. Iotti Ray Hanford David A. Steininger Guy Frederick Willy De Roovere Elie Stubbe Charles B. Hardee Roger Riley A. B. Cutter Ed Harris J. H. Goldberg Jimmy Zaalouk Jim McGuffin* D. White* Patrick McDonough* J. Vogle* Tom Timmons* J. Epstein*

Organization

Houston Lighting & Power Company Consultant - HL&P (FWEC) Commonwealth Edison Company Commonwealth Edison Company Consultant - CP&L (Ebasco) Carolina Power & Light Company EPRI (SGPO) Electronucleaire Electronucleaire (EBES) Electronucleaire Carolina Power & Light Company Public Service, Indiana Carolina Power & Light Company Carolina Power & Light Company Houston Lighting & Power Company Houston Lighting & Power Company Westinghouse Westinghouse Westinghouse Westinghouse Westinghouse 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:

 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.

- 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:

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

Second Session

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.
- 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 ARD 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

Jimmy Zaalouk Bruce Poole Frank Eisinger Guv Frederick Elie Stubbe Roger Riley L. D. (Del) Butterfield Joe Reiss Robert Iotti David Steininger John Hopkins* Marty Wambsganss* Patric McDonough* Tom Timmons* Edward Burns* Jim McGuffin* Kresimir Fink

Organization

Houston Lighting & Power Company Houston Lighting & Power Company Consultant to HL&P (FWEC) Electronucleaire Electronucleaire Public Service, Indiana Commonwealth Edison Company Commonwealth Edison Company Consultant to CP&L (Ebasco) EPRI (SGPO) NRC Consultant to NRC (Argonne Natl. Lab) Westinghouse Westinghouse Westinghouse Westinghouse KRSKO

* Part-time attendance

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

- The members of the TRC have each identified questions they have relative to Westinghouse responses of April 8, 1983.
- This morning we will review the D2/D3 report and discuss our working relationship with the NRC.
- 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.
- This afternoon or tomorrow we will develop an outline for the final report and identify who will write the various sections.

Counter Flow Steam Generator Minutes of the Meeting Between Technical Review Committee, Westinghouse and NRC, Held on April 21, 1983 Page 2

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:

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

Counter Flow Steam Generator Minutes of the Meeting Between Technical Review Committee, Westinghouse and NRC, Held on April 21, 1983 Page 3

- If a D4/D5 unit goes to internal split flow then reverse flow will be the limiting contributing factor.
- When discussing erron bands on tubes we must evaluate if it is a safety or long term operability question.
- 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. N	1. N	lova	ak		
J. E	3. H	iopl	cins		
F. L	. E	is	inge	r	
D. A	4. 5	ite	inin	ger	
T. M	1. W	111	liam	son	
Mila	in C	op	ic		
Guy	Fre	eder	ric		
Elie	: St	ubt	be		
Roge	er F	lile	ey		
Will	y D)e F	ROOV	ere	
Robe	ert	С.	Iot	ti	
Joe	Rei	SS			
L. [). E	Butt	terf	ield	1
K. F	ink				
V. F	aty	r			
Jimn	nv Z	aa	louk		
B. F	001	e			
Ed	larr	is			
Ted	Jer	ki	ns		
Harc	ld	R.	New	th	
Albe	ort	Lat	tham		
Rob	Dar	ko	unum		
M	inc	ian.	ina		
P1. 3	niç	Japi	ara		

NRC NRC Consultant - HL&P EPRI (Steam Generator Project Office) Duke Power Company RKE - NE KRSKO (YU) Electronucleaire (Belgium) Electronucleaire (Belgium) Public Service (Indiana) Electronucleaire Doel 3-4 Consultant - CP&L (Ebasco) Commonwealth Edison Commonwealth Edison NPP - KRSKO (YU) NPP - KRSKO (YU) HL&P - STP HL&P - STP Carolina Power & Light Co. SHNPP Texas Utilities Consultant - Texas Utilities Texas Utilities Texas Utilities 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 lube 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^o 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

- We have reviewed data on 16⁰ 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.
- 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.
 - 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 ware 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 16degree 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.