

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)	
)	
COMMONWEALTH EDISON COMPANY)	Docket Nos. 50-454 OL
)	50-455 OL
(Byron Nuclear Power Station,)	
Units 1 & 2))	

COMMONWEALTH EDISON COMPANY'S PROPOSED
FINDINGS OF FACT AND CONCLUSIONS OF LAW
REGARDING STEAM GENERATOR TUBE INTEGRITY

June 7, 1983

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OPINION

* * * * *

II. CONTENTIONS

* * * * *

- D. Rockford League of Women Voters' (League) Contention 22 and DAARE/SAFE Contention 9(c) -- Steam Generator Tube Integrity

League Contention 22 and DAARE/SAFE Contention 9(c) both allege generally that the steam generator tubes at the Byron Station are likely to become significantly degraded, presenting the hazard that radioactive primary-system water

* These proposed findings are presented in the form of a partial initial decision which addresses one of the eight litigated issues, specifically, steam generator tube integrity. The proposed findings on the other seven issues have been or will be submitted in accordance with the schedule stated in the "Procedural History" section of "Commonwealth Edison Company's Proposed Findings of Fact and Conclusions of Law Regarding Seismology, Waterhammer, and ALARA" filed on May 31, 1982, into which this document is incorporated.

may leak out of containment either during normal operation or under accident conditions, thereby imperiling the public health and safety. The League contention asserts that "corrosion-induced wastage, cracking, reduction in tube diameter, and vibration-induced fatigue cracks" may cause loss of integrity of the Byron steam generator tubes. The DAARE/SAFE contention similarly points to "corrosion, cracking, denting and fatigue cracks" and suggests that a possible solution to this problem would be redesign of the steam generator.

Applicable Law

Prior to issuance of an operating license, the NRC must find reasonable assurance exists "that the activities authorized by the operating license can be conducted without endangering the health and safety of the public" and that such activities be conducted in compliance with the NRC's regulations. 10 C.F.R. § 50.57(a)(3). Section 50.57(a)(3) is implemented with respect to steam generator tubes by satisfying 10 C.F.R. Part 50, Appendix A, General Design Criteria 14, 31 and 32, which provide in pertinent part:

Criterion 14 - Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 31 - Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under op-

erating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

Criterion 32 - Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 14, which is primarily applicable, is germane because the steam generator tubes fall under the ambit of the "reactor coolant pressure boundary." In order to establish steam generator tube integrity under this criterion and under Criterion 31, Applicant must demonstrate that design measures have been developed and a detection system is in place so that under either normal operating conditions or under accident conditions there is an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Criterion 32, which requires that reactor coolant pressure boundary components be designed to permit periodic inspection and testing of critical areas, is, in essence, a means of satisfying Criteria 14 and 31. Implementing guidelines to the above criteria have been issued as NRC Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." The evidence presented by Applicant and the NRC Staff is directed toward demonstrating that the design and detection programs necessary for fulfillment of General Design Criteria 14, 31 and 32 have been established.

Additionally, it should be noted that steam generator tube integrity has been designated as an Unresolved Safety Issue (USI) by the NRC Staff. The role to be assigned to USI's in individual licensing proceedings was the subject of a decision by the Atomic Safety and Licensing Appeal Board of the NRC, Gulf States Utilities Co. (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977), in connection with the Board's consideration of the Gulf States Utility Company application for the River Bend Station. The Appeal Board succinctly set forth the policy to be followed for unresolved safety issues. It held that the NRC Staff should make clear in the SER its perception of the nature and extent of the relationship between each significant USI and the extended operation of the reactor under scrutiny. The furnished information should shed light on whether (1) the problem has already been resolved for the reactor under study; (2) a reasonable basis exists for concluding that a satisfactory solution will be obtained before operation; or (3) the problem would have no safety implications until several years of reactor operation at which time either a solution or alternate means will be available to prevent an undue risk to the public.

It is against the foregoing guidelines that the Board will weigh the evidence on the issues raised by Contentions 22 and 9(c).

Types of Tube Degradation

The steam generator tubing, which is part of the reactor coolant pressure boundary, represents an integral part of a major barrier against the release of radioactivity to the environment. Accordingly, conservative design criteria for tube wall sizing have been established to assure structural integrity of the tubing under normal operating and the postulated design-basis accident condition loadings. (Finding 152.) However, early in the operating experience of the first PWR's, it was recognized that over a period of time under the influence of the operating loads and an aqueous environment in the steam generator, some tubes may become degraded and leak. (Finding 154.) Implicit in the language of General Design Criteria 14 and 31, which require a low probability of abnormal leakage and of rapidly propagating failure, is the recognition that some degradation and leakage may inevitably occur.

Dr. Michael J. Wootten, Manager of Chemistry Field Development of Westinghouse and Mr. Daniel D. Malinowski, Manager of Field Data Analysis in Steam Generator Programs of the Water Reactor Division of Westinghouse, testifying for Applicant, described the following forms of steam generator tube degradation: tube wall thinning, pitting, stress-corrosion cracking, intergranular attack and tube wear. (Finding 155.) The first four forms of degradation result from corrosive mechanisms. (Findings 156-58.) The

fifth form, tube wear, results from a mechanical abrasion of the tube surface caused by the impact of adjacent structures or loose objects on the tubing. (Finding 160.) Deformation, as opposed to degradation, is manifested by tube denting and results from corrosion. (Finding 159.)

Applicant has undertaken a comprehensive program to minimize the potential for corrosion and tube wear via numerous design implementations and a comprehensive AVT secondary water chemistry program and to detect, monitor, and, if necessary, remedy any tube degradation that may occur. (Findings 161, 162, 166, 182-184, 189, 192.)

Design Features

Steam generator design has been a continually evolving process, with each generation incorporating design improvements over the preceding one. Dr. Lawrence Conway, Advisory Engineer with the Steam Generator Turbine Generator Division of Westinghouse, and Mr. Wilson D. Fletcher, Manager of Steam Generator Materials and Chemistry of Westinghouse, testified with respect to numerous design features that have been implemented in the Model D-4 and D-5 steam generators at the Byron Station to address the above forms of tube degradation and deformation.

Inconel 600 was chosen for the steam generator tube material as being the most suitable for the temperatures and chemical environment present within a steam generator and to withstand the stresses and pressures calculated to occur with respect to certain postulated design basis

accidents. To minimize chemical concentration areas (such as at the tube sheet and between the tube and tube support plate), recirculation rates were optimized, the ports in the blowdown pipe were modified, and the tubes within the tubesheet hole were expanded to eliminate the crevices at the tubesheet. In addition to low stresses being inherent in a U-shape tube design, the widest spacing between tube support plates that is functionally acceptable was selected and the holes in the flow distribution baffle plates and in the top tube support plate were modified to minimize tube stresses further. (Finding 161.)

In addition, the design of the Model D-5 steam generator in Unit 2 has been enhanced by (1) utilizing stainless steel, a more corrosion-resistant material, as the material for the tube support plates and baffles, (2) changing the shape of the holes in the tube support plates from circular to a quatrefoil shape to improve flow, (3) expanding the tubes within the tube sheet by means of a hydraulic device in lieu of mechanical rollers to reduce stresses, (4) thermally treating the Inconel 600 tubes to enhance their resistance to corrosion, and (5) changing the holes in the flow distribution baffles from slotted to a circular shape to improve flow. (Finding 162.)

One design change that is being implemented on the Model D-4 and D-5 steam generators at Byron is to address the "flow-induced vibration" phenomenon. This phenomenon

refers to certain tubes in the preheater region of a Model D steam generator vibrating against the baffle plates as a result of turbulence created by the feedwater flow entering from the main feedwater nozzle, thereby causing tube wear. (Finding 163.) Tube wear resulting from flow-induced vibration was initially identified in October, 1981 in two non-domestic plants with Westinghouse Model D-3 steam generators, Ringhals Unit 3 and Almarez Unit 1. (Finding 164.)

As a result, Westinghouse undertook an extensive investigation of the flow-induced tube vibration phenomenon. The investigation included eddy current testing at the Ringhals and Almarez plants, as well as at two plants with Model D-2 and D-4 steam generators, McGuire 1 and Krsko. The latter two plants, which had not operated above 50% power, showed no indications of possible tube wear. (Finding 164.) Based upon its investigation, Westinghouse has recommended that Applicant make the following modifications to the Byron plant to reduce the potential for significant steam generator tube vibration: (1) the expansion at baffle-plate locations in the preheater region of approximately 100 tubes per steam generator and (2) bypassing approximately 10% of the flow from the main feedwater nozzle to the auxiliary feedwater nozzle. The expansion of tubes at baffle-plate locations will limit the tube movement at the baffle-plate intersections to a few thousandths of an inch. Bypassing 10% of the main feed flow to the auxiliary nozzle will reduce the main feed

flow at the inlet to the preheater to approximately 90% and will further reduce the potential for vibration of the tubes in the preheater region. (Finding 166.)

Westinghouse has developed a proprietary process that will hydraulically expand the steam generator tubes. A long-established testing program reveals that residual stress from the expansion combined with the relatively low temperature in the preheater region does not significantly increase the potential for stress-corrosion cracking in the expanded location. (Finding 168.)

The feedwater bypass modification will ensure that approximately 90% of the feedwater flow will enter via the main feedwater nozzle and the remainder of the feedwater flow will enter the steam generator through the auxiliary feedwater nozzle. (Finding 172.) The efficacy of the Westinghouse proposed modifications has been the subject of an extensive review and verification process involving numerous meetings among Westinghouse, Applicant and the NRC Staff. (Finding 167.)

Applicant's witness, Mr. Thomas F. Timmons, Manager of Reactor Coolant Systems Components Licensing in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse, underwent extensive cross examination with respect to the 90% main feedwater-10% auxiliary feedwater

split. Intervenors suggested that a 70%-30% split would be more appropriate because the investigatory data revealed that main feed flow rates exceeding 70% resulted in significant vibration and greater tube wear than main feed flow rates at 70% or less. (See Tr. 6010-19; Finding 173.) (It should be noted that Krsko, where a 70% flow rate did not result in significant tube wear, has a flow rate 7% greater than Byron. (Finding 174.))

Mr. Timmons acknowledged that if there were no tube expansion, a split equivalent to a 70-30 split at Krsko (approximately 75-25 for Byron) would be needed to significantly reduce flow-induced tube vibration. (Timmons, Tr. 6016.) However, Westinghouse has performed model testing on the effect of tube expansion with different flow velocities as well as testing at the Krsko plant, which involved taking vibration readings, expanding a tube and comparing the post-expansion readings to the pre-expansion readings. The model and Krsko tests yield the same result--expansion reduces tube vibration by a factor of 5 resulting in a negligible level of vibration. (Finding 174.) Accordingly, the approximately 100 tubes at Byron most susceptible to flow-induced vibration will have significantly lower vibration levels with a 90-10 split and tube expansion than they would with a 70-30 split and no tube expansion. (Timmons, Tr. 6230.) Concomitantly, Mr. Timmons concluded that the tube vibration data demonstrate that at a 90-10 flow split, none

of the approximately 100 expanded tubes in the Byron steam generators will degrade to the 40% tube wall degradation level requiring plugging as a result of flow-induced vibration over the 40-year life of the plant. (Finding 176.)

The NRC Staff's preliminary review concurred in the conclusion that the proposed solution to the flow-induced vibration problem should succeed. (Finding 177.) The tube expansion process is scheduled to begin in mid-July. Applicant has committed to completing the recommended modifications prior to unit operation. (Finding 178.)

The Board finds that the proposed modifications will effectively reduce tube wear resulting from flow-induced vibration such that the Byron steam generator tubes are not expected to reach the 40% plugging level as a result of flow-induced vibration over the 40-year life of the plant. Thus, the Board concludes that based upon the testimony of Dr. Conway and Messrs. Fletcher and Timmons, the design of the Byron D-4 and D-5 steam generators effectively reduces the potential for steam generator tube degradation. Although the Model D-5 steam generators in Unit 2 have added design features to combat tube degradation, Applicant's witnesses concluded, and the Board finds, the Model D-4 design to be equally adequate. The Model D-5 design enhancements merely optimize an already safe design concept represented by the Model D-4. (Tr. 4435-36.)

Secondary System Water Chemistry

The Byron Station will implement an AVT secondary system water chemistry program that includes the addition of volatile chemicals as control agents in the secondary system and includes rigorous control of the condensate and feedwater chemistries to minimize the formation of corrosion products that are delivered to the steam generators. (Findings 180-85, 189.) Mr. Malinowski, testifying for Applicant, Mr. Marsh, Mr. Frank and Mr. McClacken, testifying for the NRC Staff, and Mr. Bridenbaugh, testifying for the Intervenor, agreed that some degree of corrosion will always take place in any metal that maintains contact with an aqueous environment. Accordingly, AVT Secondary System Water Chemistry implementations are designed to minimize, not eliminate, steam generator tube corrosion. (Finding 190.)

Westinghouse AVT guidelines, introduced in 1977 and modified subsequently from time to time, recommend a rigorous program of water chemistry control to reduce the introduction of chloride ions, oxygen, and copper bearing alloys. (Finding 183.) In addition, the Steam Generator Owners' Group (SGOG), of which applicant is a member, has issued in concert with the Electric Power Research Institute (EPRI) even more restrictive water chemistry controls. (Board Findings 184-85.) Applicant will be following both the Westinghouse and SGOG or EPRI guidelines for strict water chemistry control at the Byron Station. (Finding 182.)

Plants that have operated only on rigorously applied AVT have experienced little or no denting. Although there have been a limited number of incidents of other corrosive mechanisms since the institution of AVT water chemistry control, they have been minimal when considered in the context of the approximately 60 plants in operation. (Finding 187.) In any event, as Intervenor's witness acknowledged, there have been no tube ruptures due to corrosion during the past seven years. (Finding 188.) Adherence to the strict AVT water chemistry guidelines issued by both Westinghouse and EPRI minimizes the potential for tube corrosion, thereby enhancing the long-term integrity of the reactor coolant pressure boundary. (Finding 189.)

The Board finds that the design features combined with a strict AVT water chemistry program will minimize tube wall degradation at the Byron plant.

Monitoring, Detection and Remedial Measures

1. Pre-Service Inspections and Eddy Current Testing.

In addition to the foregoing measures employed to mitigate tube degradation in the Byron steam generators, Applicant has instituted numerous measures for the detection, monitoring, and remediation of degradation or deformation of steam generator tubes. A 100% pre-service inspection has been performed on the steam generator tubes in Unit 1 and will be performed on the tubes in Unit 2 pursuant to NRC Regulatory Guide 1.83. (Findings 192-194.) NRC Regulatory

Guide 1.83, with which Applicant will comply, requires that in-service inspections of a percentage of the tubes be performed every 12 to 24 months, depending upon the condition of the steam generators. Additional tube samples are required to be inspected when degraded or defective tubes are found. In cases where the degradation processes have been highly active, the Staff has required that the inspections be performed at more frequent intervals, consistent with the rate at which degradation is occurring. (Findings 195, 197-199.)

The method to be used for monitoring steam generator tube integrity at Byron is an inspection program using eddy current testing as the principal inspection tool. (Finding 200.) From eddy current testing, it is possible to determine the depth, length and volume of tube-wall material affected. The sensitivity of eddy current testing to tube wall degradation varies depending upon the size, shape and nature of the degradation. Eddy current testing will reliably detect the various types of tube degradation at the following sensitivity levels: tube wall thinning at 20% depth of the tube wall; pitting at 20% tube wall depth; tube wall cracking at 40% tube wall depth; intergranular attack at 40% tube wall depth and tube wear at 20% tube wall depth. Denting, a form of tube deformation, is also readily detected by eddy current testing. (Finding 201.) Eddy current inspections at Byron

will be performed according to the provisions of the Byron Technical Specifications and according to NRC Regulatory Guide 1.83. (Finding 195.)

The testimony of Intervenor's witness, Mr. Bridenbaugh, questioned the ability of eddy current testing to detect tube degradation, particularly where the tube passes through the support plate or other structural members. (Bridenbaugh, Tr. 6459.) Mr. Malinowski's testimony addressed the discrimination question. He testified that through use of multifrequency eddy current testing, discrimination can be made among tube degradation, support plates, and external deposits. Inasmuch as the responses for external discontinuities vary according to the test frequency, it is possible by linear combinations of the responses at different frequencies to reduce unwanted signals from a composite response. On cross examination, Mr. Bridenbaugh, though having reservations about the degree of experience with eddy current testing, acknowledged that the multifrequency testing currently in use is a significant improvement over earlier methods and "may very well prove to be adequate." (Finding 200.)

A review of the sensitivity of the eddy current method for detecting the degradation phenomena encountered in steam generator operating experience, coupled with the conservatism incorporated in the Byron tube plugging criterion to compensate for eddy current measurement uncertainty (discussed infra), demonstrates that tube wall penetration

at or below the plugging limit of 40% is detectable. Thus, significant tube degradation is expected to be detected by eddy current testing. (Finding 202.)

Applicant has also implemented detection measures for tube wear resulting from the impact of adjacent structures or loose objects on the tubing. Mr. Blomgren, testifying for Applicant, and Mr. Frank, testifying for the NRC Staff, described the Loose Parts Monitoring System (LPMS) that will be installed on the steam generators at the Byron Station. The system includes two sensors on the secondary side of each steam generator which monitor the steam generator for noise produced by loose parts or foreign objects. In addition, the secondary side will be visually inspected from time to time. (Finding 207.) Applicant has also instituted a program of strict inventory control of all tools and materials entering and leaving the secondary side of the steam generators during maintenance operations. (Finding 206.)

2. "Leak Before Break"

Commonwealth Edison also will monitor the Byron reactor systems for primary to secondary leakage. To ensure tubes are not degraded to the point where rupture is possible, licensees are required to shut down and repair steam generator tubes should the primary to secondary leakage exceed a maximum allowed by technical specification. (Finding 208.)

The maximum permissible leak rate will be set forth in the Byron Technical Specifications consistent with the "leak-before-break" principle. A crack in the steam generator tube material, Inconel 600, will leak long before the crack reaches a linear length called the "critical length" where tube rupture could occur under postulated accident conditions. Thus, the degradation becomes detectable through leakage monitoring, thereby providing a mechanism for positive detection of tube degradation and appropriate operator action before tube rupture can occur. (Finding 209.)

The maximum permissible leak rate for Byron was established pursuant to an extensive testing program consisting of leak rate and burst pressure tests to determine correlations among the length of the crack, the associated leak rate during normal operation and burst strength. The maximum permissible leak rate during normal operation at Byron has been established at the Standard Technical Specification limit of .35 gpm per steam generator. This corresponds to the maximum allowable crack length of .43 inch. (Finding 210.) The critical-crack length corresponding to the maximum accident condition pressure during a postulated Main Feedwater Line Break (MFLB) or Main Steam Line Break (MSLB) was conservatively determined to be .51 inch using the results of the burst pressure tests. Inasmuch as the critical crack length is greater than the maximum permissible length of .43 inch for continued operation within leakage limits,

the unit is safeguarded against tube rupture during a postulated MFLB/MSLB accident. (Findings 211-12.)

3. Tube Plugging Criteria

Tube plugging criteria are established in accordance with Section XI of the ASME Boiler and Pressure Vessel Code guidelines to provide the wall-thickness limit below which tubes are to be removed from service by plugging. (Finding 215.) The tube plugging criterion established by Section XI of the Code for steam generators of the design that are in use at Byron is an outside wall degradation not to exceed 40%. The steam generator wall tubing can sustain degradation in excess of 50% of wall thickness and still meet all applicable stress and strength requirements. The 40% figure is reached by conservatively allowing for a 10% uncertainty factor in both eddy current measurement and corrosion allowance for continued plant operation until the next inspection. (Finding 218.)

The Board finds that the numerous measures instituted by Applicant for monitoring and detection of all forms of degradation and for detection of potential sources of degradation, combined with the establishment of a conservative 40% tube plugging criterion, will provide a safe margin for implementing any necessary remedial action before the potential for a tube rupture occurs.

SAI Report

Applicant's and Staff's witnesses were cross-examined extensively about potential steam generator recommendations contained in a report entitled "Value-Impact Analysis of Recommendations Concerning Steam Generator Tube Degradations and Rupture Events" (Joint Intervenors' Exhibit 9) prepared by Science Applications, Inc. (SAI) for the NRC. Although Applicant did not rely upon the SAI Report, Commonwealth Edison's testimony reveals that the design, water chemistry and many detection practices to be implemented at Byron reflect the potential recommendations proposed in the SAI Report that are germane to Byron. (Tr. 4253-4312.) For example, the first proposed recommendation is for prevention and detection of loose parts and foreign objects. As discussed above, the Byron plant will have a thorough maintenance inventory control program, a loose parts monitoring system containing sensors on the secondary side of the steam generators and will conduct periodic inspections of the steam generators for the prevention and detection of loose parts and foreign objects. (Blomgren, Tr. 4256-57.)

Notwithstanding the similarity of the practices to be instituted at Byron to the contents of the pertinent potential recommendations, the record clearly reveals that the SAI Report is a document in preliminary draft form that will undergo extensive internal review before it is ultimately submitted to the Commission. (Finding 221.) For example, an NRC Staff witness, Mr. McCracken, testified that the

loose parts control proposal is being considered in three areas: loose parts monitoring, inventory control and visual examination. He stated that the NRC Staff may recommend implementation of one of the three, a combination of the three or none of the three, depending upon the Staff's final evaluation. (McCracken, Tr. 4503.) The SAI Report simply represents a cost-benefit analysis of recommendations initiated by the NRC Staff and is one part of the overall staff assessment of potential recommendations with respect to Unresolved Safety Issue A-3. (Marsh, Tr. 4476-77.) Upon completion of the extensive internal review process, it is not anticipated that the final recommendations will become a rule in any sense. (Finding 221.) Accordingly, the Board finds that Applicant's adherence to the standards mandated by 10 C.F.R. Part 50, Appendix A should not be measured by the potential recommendations contained in the SAI cost-benefit analysis.

Overall Assessment

Mr. Fletcher assimilated the conclusions provided by the expert witnesses testifying with respect to their specific disciplines and reached an overall assessment as to steam generator tube integrity at the Byron Station. Based upon the design, water chemistry, monitoring, detection, and remedial measures undertaken by Applicant, Mr. Fletcher concluded that steam generator tube degradation at the Byron Station should not be a safety concern and that tube rupture

should not occur, even under conditions of Main Steam Line Break (MSLB) or Loss of Coolant Accidents (LOCA's). (Finding 223.)

Mr. Fletcher's qualitative judgment that tube rupture should not occur under the foregoing accident conditions is confirmed quantitatively by an independent evaluation performed by Mr. Michael Hitchler, Manager of Probabilistic Risk Assessment with the Nuclear Safety Department of Westinghouse. Mr. Hitchler's unrefuted quantitative assessment demonstrates that the frequency of multiple tube ruptures combined or as a consequence of (i) large LOCA events is 5×10^{-7} ; (ii) small LOCA and transient events with normal pressure differentials is 2×10^{-5} ; and (iii) main steam/feedwater line break events is 3×10^{-5} . (Finding 224.) The frequencies of occurrence for these accidents are beyond the range of probabilities established generally for design basis accidents. (Finding 224.) Thus, as Applicant and the Staff aver, the frequency of postulated tube ruptures combined or as a consequence of transient conditions and accident conditions such as MSLB's and LOCA's is calculated to be extremely low over the 40-year life of the plant. (Findings 223-27.)

It should be noted that the Board conducted an examination of Intervenor's witness with respect to recommendations made in his prepared testimony that he considered to be of prime importance to the Byron Station. Although the witness believed more stringent regulatory requirements would be desirable in certain instances, he could point to

no area in which Applicant failed to meet the applicable regulations. (Bridenbaugh, Tr. 6499-6509.)

In evaluating the evidence, the Board also observes that Applicant's experts have considerable first-hand expertise in the area of Westinghouse steam generator design, operation, and the phenomena associated therewith. Intervenor's witness has gleaned his information on Westinghouse steam generators principally from the publicly available documents. (Bridenbaugh, Tr. 6408-09.) Intervenor's witness has never designed a Westinghouse steam generator (id. at 6406-07, 6416); has never been involved in the fabrication or maintenance activities with respect to Westinghouse steam generators (id. at 6407, 6417), has never devised a water chemistry program for the secondary side of a Westinghouse steam generator (id. at 6428), has never conducted eddy current testing (id.), has never evaluated a particular steam generator to determine if tubes should be plugged (id. at 6429) and has never testified prior to this case on the specific issue of steam generator tube degradation as a health and safety concern (id.).

Unresolved Safety Issue

In direct response to the River Bend decision, the NRC Staff included Appendix C to the Byron SER (Staff Exhibit 1) to give its assessment of the Unresolved Safety Issues. With respect to steam generator tube integrity, the NRC Staff concluded:

Pending completion of Task A-3 [the task applicable to Byron identified with the steam generator unresolved safety issue], the measures taken at this facility should minimize the steam generator tube problems encountered. Further, the inservice inspection and Technical Specification requirements will ensure that the applicant and the NRC Staff are alerted to tube degradation should it occur. Appropriate actions such as tube plugging, increased and more frequent inspections and power derating could be taken if necessary. Because the improvements that will result from Task A3 will be procedural (such as an improved inservice inspection program), they can be implemented by the applicant after this facility begins to operate, if necessary.

(Staff Exhibit 1, Appendix C at C-10.) (The Staff will report the results of its review of the mechanical vibration phenomenon in a forthcoming SER Supplement.) It is observed that many of the actions that result in removing questions posed by the Unresolved Safety Issue are the same actions that contribute to a satisfaction of the earlier-cited General Design Criteria. In any event, in view of the Staff's evaluation, it is clear that the Unresolved Safety Issue status of steam generator tube integrity poses no obstacle to granting operating licenses to the Byron facility.

Conclusion

Based upon (1) the design of the Byron steam generators, including the 90-10 bypass and tube expansion modification program to address the flow-induced vibration phenomenon, (2) the comprehensive AVT secondary system water chemistry program, (3) the various monitoring, detection, and remedial measures instituted by Applicant, including

preoperational inspections, inservice eddy current testing, the loose parts monitoring system, the leak-before-break and critical-crack length principles, and the tube plugging criterion, and (4) the resistance of the steam generator tubes to rupture even in the face of low-probability events such as MSLB's, MFLB's, or LOCA's, the Board finds the reactor coolant pressure boundary is designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The Board also finds that the Byron steam generators are designed to permit periodic inspection and testing of critical areas to assess their structural and leaktight integrity. Accordingly, the Board concludes that 10 C.F.R. Part 50, Appendix A, General Design Criteria 14, 31 and 32, is satisfied. Based upon the above conclusion and the NRC Staff's evaluation of the Unresolved Safety Issue, Contentions 22 and 9(c) are dismissed.

FINDINGS OF FACT

III. CONTENTIONS

* * * * *

D. Rockford League of Women Voters' (League) Contention 22 and DAARE/SAFE Contention 9(c) -- Steam Generator Tube Integrity

150. Rockford League of Women Voter's (League)

Contention 22, as litigated, provides:

An extremely serious problem occurring at other plants such as Consumers' Palisades plant and C.E.'s Zion plant, and likely to occur at C.E.'s Byron plant, is presented by degradation of steam generator tube integrity due to corrosion induced wastage, cracking, reduction in tube diameter, and vibration induced fatigue cracks. This affects, and may destroy, the capability of the degraded tubes to maintain their integrity, both during normal operation and under accident conditions, such as a LOCA or a main steam line break. The Commission Staff has correctly regarded this problem as a safety problem of a serious nature, as evidenced both by NUREG-0410 and the Black Fox testimony cited above. As a result of this serious and unresolved problem the findings required by 10 C.F.R. §§ 50.57(a)(3)(i) and 50.57(a)(6) cannot be made.

DAARE/SAFE Contention 9(c) as litigated, provides:

Steam generator tube integrity. In PWRs steam generator tube integrity is subject to diminution by corrosion, cracking, denting and fatigue cracks. This constitutes a hazard both during normal operation and under accident conditions. Primary loop stress corrosion cracks will, of course, lead to radioactivity leaks into the secondary loop and thereby out of the containment. A possible solution to this problem could involve redesign of the steam generator, but at FSAR, Section 10.3.5.3 the Applicant notes its intent to deal with this as a maintenance problem which may not be an adequate response given the instances noted in Contention 1, above.

151. To address the contentions, Applicant presented the testimony of eleven witnesses. Mr. John C. Blomgren, of Commonwealth Edison Company, addressed various measures that will be employed at the Byron Station to minimize steam generator tube degradation. Dr. Mahendra R. Patel, of Westinghouse Electric Corp., addressed the "leak-before-break" principle and steam generator tube plugging criteria. Other witnesses from Westinghouse included Mr. Daniel Malinowski, who addressed the inspection measures used to detect steam generator tube degradation; Dr. Michael J. Wootten, who addressed the water chemistry measures used to minimize tube degradation on the secondary side of the steam generators at Byron; Dr. Lawrence Conway, who addressed the design changes in D4 and D5 steam generators at Byron that enhance resistance to tube degradation; and Mr. Thomas F. Timmons, who addressed the flow-induced vibration phenomenon. Mr. Lawrence D. Butterfield, of Commonwealth Edison, addressed Applicant's modifications with respect to the flow-induced vibration phenomenon. Applicant also presented the testimony of Mr. Kenneth J. Green, Mechanical Project Engineer for Sargent and Lundy Engineers, who addressed the issue of whether the proposed modifications to the Byron steam generators described by Messrs. Timmons and Butterfield might increase the likelihood of a waterhammer event in the Feedwater Bypass Systems of the Byron steam generators, and Mr. Rodolfo Paillaman, a Senior Quality Assurance Non-

Destructive Examination Specialist with Ebasco Services, Inc., who addressed the pre-service inspection of the steam generator tubes at Byron.

Mr. Wilson D. Fletcher, Manager of Steam Generator Materials and Chemistry of Westinghouse, provided an overview of the steam generator tube integrity issue and Mr. Michael Hitchler, Manager of Probabilistic Risk Assessment with the Nuclear Safety Department of Westinghouse, quantitatively assessed the probability of steam generator tube ruptures under various conditions.

The NRC Staff presented the testimony of Dr. Jai Raj N. Rajan, a mechanical engineer, who addressed the flow-induced vibration phenomenon and the probability of tube rupture under various conditions; Mr. Ledyard B. Marsh, who addressed various steam generator tube degradation issues; Mr. Louis Frank, a senior materials engineer, who addressed secondary side water chemistry measures and in-service inspections; and Mr. Conrad McCracken, who addressed steam generator design and secondary side water chemistry measures to reduce corrosion.

The League presented the testimony of Mr. Dale G. Bridenbaugh, a nuclear engineer and President of MHB Technical Associates in California, who addressed various aspects of the steam generator tube integrity issues concerning the Byron Station.

152. The steam generator tubing, which is part of the reactor coolant pressure boundary, represents an integral

part of a major barrier against the release of radioactivity to the environment. Accordingly, conservative design criteria for tube wall sizing have been established to assure structural integrity of the tubing under normal operating and the postulated design-basis accident condition loadings. (Patel, Applicant Prepared Testimony at 5, ff. Tr. 4126.)

153. Steam generator tubes are manufactured with a wall thickness approximately twice as thick as the minimum indicated by the design rules of Section III of the ASME Boiler and Pressure Code to accommodate fabrication, installation and handling procedures. (Id. at 6; Patel, Tr. 4370.)

154. However, early in the operating experience of the first PWR's, it was recognized that over a period of time under the influence of the operating loads and environment in the steam generator, some tubes may become degraded and leak. (Patel, Applicant Prepared Testimony at 6, ff. Tr. 4126; Malinowski, Applicant Prepared Testimony at 5, ff. Tr. 4126.)

155. Degradation of steam generator tubes is manifested in the following forms: tube wall thinning, pitting, cracking, intergranular attack and tube wear. (Malinowski, Applicant Prepared Testimony at 13, 15, ff. Tr. 4126.)

156. Tube wall thinning is a localized reduction in the tube thickness resulting from corrosion by phosphates

in high concentrations. It is caused by low sodium to phosphate ratio solutions. (Wootten, Applicant Prepared Testimony at 8, ff. Tr. 4126.) Tube wall thinning has been observed within sludge piles at the top of the tubesheet and at tube support plate elevations, where lower flow velocities allowed concentration of phosphates to saturation levels. (Malinowski, Applicant Prepared Testimony at 15-16, ff. Tr. 4126.)

157. Pitting is a form of tube degradation that involves small discrete roughly circular regions of tube penetration typically less than 100 mils in diameter. Pitting can occur separately or in bands wherein each pit acts independently of others within the band. (Malinowski, Applicant Prepared Testimony at 16-17, ff. Tr. 4126.) Pitting is believed to be due to an acidic chloride condition involving copper and chloride ions. (Wootten, Applicant Prepared Testimony at 14, ff. Tr. 4126.)

158. Tube wall cracking generally occurs in a local region and the crack may extend through the entire wall thickness. Depending upon the orientation of tube wall stresses, cracks may initiate from the outside diameter or the inside diameter of the tube. All tube wall stress-corrosion cracks detected in Westinghouse-designed steam generators have been intergranular in nature. Intergranular attack is a form of tube degradation usually characterized by general grain boundary dissolution. It occurs in conjunction with stress-corrosion cracking on the outside diameter of the tube; it usually occurs within crevices

between the tube and the tubesheet. (Malinowski, Applicant Prepared Testimony at 17-18, 20, ff. Tr. 4126.)

159. Tube wall deformation can result from denting. Denting is a localized radial reduction or deformation in the diameter of steam generator tubes, resulting from corrosion of the carbon steel tube support plates in the steam generator. (Wootten, Applicant Prepared Testimony at 10-11, ff. Tr. 4126.)

160. Tube wear is a form of tube degradation that results from a mechanical abrasion of the tube surface. Such wear progressively reduces the thickness of the tube area affected. Wear results from the impact of adjacent structures or loose objects on the tubing; it has been observed at antivibration bar intersections, the baffle plates in preheat sections and locations in contact with foreign objects. (Malinowski, Applicant Prepared Testimony at 21, ff. Tr. 4126.)

161. Steam generator design has been a continually evolving process, with each generation incorporating design improvements over the preceding one. Numerous measures are incorporated into the design of the Byron steam generators, the Model D-4 in Byron 1 and Model D-5 in Byron 2, to protect against potential tube degradation. Inconel 600 was chosen for the steam generator tube material as being the most suitable for the temperatures, chemical environment, and design basis accident conditions present within a steam generator. To minimize chemical concentration areas (such as at the tube sheet and between the tube and tube support

plate), recirculation rates were optimized, the ports in the blowdown pipe were modified, and the tubes within the tube-sheet hole were expanded to eliminate the crevices at the tubesheet. In addition to low stresses being inherent in a U-shape tube design, the widest spacing between tube support plates which is functionally acceptable was selected and the holes in the flow distribution baffle plates and in the top tube support plate were modified to minimize tube stresses further. (Conway, Applicant Prepared Testimony at 14-15, ff. Tr. 4126.)

162. In addition, the design of the Model D-5 in Unit 2 has been enhanced by (1) utilizing stainless steel, a more corrosion-resistant material, as the material for the tube support plates and baffles, (2) changing the shape of the holes in the tube support plates from circular to a quatrefoil shape to improve flow, (3) expanding the tubes within the tube sheet by means of a hydraulic device in lieu of mechanical rollers to reduce stresses, (4) thermally treating the Inconel 600 tubes to enhance resistance to corrosion, and (5) changing the holes in the flow distribution baffles from slotted to a circular shape to improve flow. (Conway, Applicant Prepared Testimony at 14-15, ff. Tr. 4126; Fletcher, Applicant Prepared Testimony at 6-7, ff. Tr. 5908.)

163. A design problem identified for the Model D steam generators at Byron results from certain tubes in the

preheater region of the steam generator vibrating against the baffle plates as a result of the turbulence created by the feedwater flow entering from the main feedwater nozzle. This phenomenon is called "flow-induced vibration." (Rajan, Tr. 4765-67; Rajan, NRC Staff Prepared Testimony at 1, ff. Tr. 4473; Timmons, Applicant Prepared Testimony at 8-9, ff. Tr. 5908.)

164. Tube wear in Model D steam generators resulting from flow-induced vibration was initially identified in Sweden at Ringhals Unit 3, a plant with Model D-3 steam generators. The degradation was in the form of a through-wall hole in a single tube at a baffle plate. Eddy current testing also indicated tube wall wear in other tubes at Ringhals. Possible tube wear was also detected at another D-3 plant, Almaraz 1. Eddy current testing was also performed at McGuire 1, a domestic plant with Model D-2 steam generators and at Krsko, a non-domestic plant with D-4 steam generators. The latter two plants had not operated above 50% power and showed no indications of possible tube wear. (Timmons, Applicant Prepared Testimony at 9, ff. Tr. 5908.)

165. Based upon the early eddy current testing indications, Westinghouse established an extensive program to investigate, understand, and define vibration and tube wear in Model D steam generators and to develop, test, and evaluate any modifications necessary to allow operation of Model D steam generators without significant tube wear.

The program involved gathering, reviewing, and analyzing data from operating plants and from laboratory and model tests. (Id. at 9-10; Timmons, Tr. 6056.)

166. Premised upon its extensive investigation, Westinghouse has recommended that Applicant make the following modifications to the Byron plant to reduce the potential for significant tube vibrations in the Byron steam generators: (1) the expansion at baffle-plate locations in the preheater region of approximately 100 tubes per steam generator and (2) the bypassing of approximately 10% of the flow from the main feedwater nozzle to the auxiliary feedwater nozzle. The expansion of tubes at baffle-plate locations will limit the tube movement at the baffle-plate intersections to a resultant few thousandths of an inch clearance. The bypassing of 10% of the main feed flow to the auxiliary nozzle of the steam generator will reduce the main feed flow at the inlet to the preheater to approximately 90% and will further reduce the potential for vibration of the tubes in the preheater. (Timmons, Applicant Prepared Testimony at 22-23, ff. Tr. 5908; Butterfield, Applicant Prepared Testimony at 4, ff. Tr. 5908.)

167. The efficacy of the Westinghouse proposed modifications has been the subject of an extensive review and verification process involving numerous meetings among Westinghouse, Commonwealth Edison and the NRC Staff. (Timmons, Tr. 6276-81, 6044-45; Butterfield, Tr. 6044-45.)

168. Westinghouse has developed a proprietary process that will be used to hydraulically expand the steam generator tubes. (Timmons, Applicant Prepared Testimony at 23, ff. Tr. 5908; Butterfield, Tr. 6147.) The process involves the insertion of tools into the tubes from the primary side of the steam generator tubesheet. The tools are then used to locate the baffle plate intersection and to expand the tube at the appropriate location. The expansion zone will be entirely within the thickness of the baffle plate. (Baffle plates are provided within the preheater section of the steam generator to direct the flow past the tubes.) After the expansion has been effected, it is verified by the use of eddy current testing. (Timmons, Applicant Prepared Testimony at 23-24, ff. Tr. 5908.)

169. Westinghouse has a long-established program to evaluate the effect of tube expansion, including evaluation of the levels of residual stresses in expanded tubes. Westinghouse has concluded that the levels of residual stresses in the expanded tubes combined with the relatively low temperature in the preheater region does not significantly increase the potential for stress-corrosion cracking in the expanded location. (Id. at 24; Frank, Tr. 4701-02.)

170. Westinghouse has also conducted accelerated corrosion testing to assess the effects of the reduced tube-to-tube hole clearance on the potential for denting of the expanded tubes. The results of this testing indicate that

the potential for denting is not increased for tubes expanded at the baffle intersections. (Timmons, Applicant Prepared Testimony at 24-25, ff. Tr. 5908.)

171. Westinghouse has also performed structural analyses of the expanded tube for design basis transients and accidents. The results of the structural analyses indicate that the ASME Code allowable values for stresses and fatigue usage factors are not exceeded for expanded tubes. (Id. at 25.)

172. The feedwater bypass modification will ensure that approximately 90% of the feedwater flow will enter via the main feedwater nozzle and the remainder of the feedwater flow will enter the steam generator through the auxiliary feedwater nozzle. (Id. at 25; Butterfield, Applicant Prepared Testimony at 4, ff. Tr. 5908; Green, Tr. 6213.)

173. The modification takes into consideration the collected data indicating that high main feed flow rates with unexpanded tubes could produce significant vibration of some tubes, resulting in greater wear than at lower feed flow rates. (Timmons, Tr. 5958.) To date, operation of Model D-4 steam generators with main feed flow rates up to 70% and unexpanded tubes has not produced any tube wear that can be detected by eddy current testing, although visual examination of three removed tubes did disclose some small amount of wear, approximately 0.001 to 0.0025 inches in depth. (Timmons, Applicant Prepared Testimony at 21-22, ff. Tr. 5908.)

174. Westinghouse has tested the proposed modifications in a 16° model and in the Krsko plant, a non-domestic plant with Model D-4 steam generators. The vibration data from the 16° model correlates extremely well with the data from the Krsko plant, both for expanded tubes and for nonexpanded tubes. (Timmons, Tr. 5992-93, 6063-64.) In the 16° model, a number of tubes were expanded and testing was conducted to determine the effect of tube expansion on tube vibration. At a flow rate equivalent to 90% of the Byron main feed flow rate, the expanded tubes exhibited vibration levels that were less than those observed at flow rates equivalent to 70% of the Byron main feed flow rate without tube expansion. As noted above, a 70% main feed flow rate at Krsko, which has a 7% greater feed flow than Byron, did not result in significant tube wear. In addition to the testing in the 16° model, one tube at the Krsko plant that had been previously instrumented was expanded at baffle plate locations. Previous tube vibration data was compared with the data obtained after the tube had been expanded and it was concluded that tube vibrations were reduced by at least a factor of 5 from the non-expanded case. This reduction resulted in a negligible level of vibration for that tube. (Timmons, Applicant Prepared Testimony at 26, ff. Tr. 5908.)

175. The approximately 100 tubes in each of the four steam generators at Byron that are candidates for

expansion are the tubes most susceptible to flow-induced vibration. (Timmons, Tr. 6164-66.)

176. Tube vibration data collected by Westinghouse established a vibration level at or below which tube degradation is not expected to progress through 40% of the tube wall. (Timmons, Tr. 6198.) Westinghouse's proprietary tube vibration data demonstrate that with a 90-10 flow split none of the approximately 100 expanded tubes in each steam generator will reach the 40% tube wall degradation level requiring plugging as a result of flow-induced vibration over the 40-year life of the plant. (Id. at 6198-99, 6202, 6265.) Similarly, none of the tubes not requiring expansion are expected to experience 40% wear through the tube wall over the 40-year life of the plant. (Id. at 6198.)

177. The model and test data demonstrate that flow-induced vibration in the Byron steam generators will be minimized to the point where tube wear will not significantly affect the structural integrity of the Byron steam generator tubes. (Timmons, Applicant Prepared Testimony at 26, ff. Tr. 5908; Butterfield, Applicant Prepared Testimony at 5-6, ff. Tr. 5908.) The NRC Staff has reached a similar conclusion during its preliminary review of the matter. (Rajan, NRC Staff Prepared Testimony at 5, ff. Tr. 4473; Rajan, Tr. 4639; 6326.) In addition, Intervenor's witness testified that if the proprietary data demonstrating the benefit of expansion is applicable to the Byron Station, the technical solution for the flow-induced tube vibration problem should succeed. (Bridenbaugh, Tr. 6507.)

178. The tube expansion process is scheduled to begin in mid-July prior to start up of the Byron Station. Changes to the control circuitry of the feedwater preheater bypass valve and the installation of a feedwater bypass line flowmeter will occur at the same time as the tube expansion program. (Butterfield, Applicant Prepared Testimony at 5, ff. Tr. 5908.) Applicant is committed to complete these modifications prior to unit operation. (Blomgren, Applicant Prepared Testimony at 17, ff. Tr. 4126, Blomgren, Tr. 4120, 4251.)

179. The Board finds that the proposed modifications to which Applicant has committed will effectively reduce tube wear resulting from flow-induced vibration such that the Byron steam generator tubes are not expected to reach the 40% plugging level as a result of flow-induced vibration over the 40-year life of the plant.

180. Operating plant experience has shown the need for rigorous control of the secondary side water chemistry environment, including the condensate and feedwater systems. (Wootten, Applicant Prepared Testimony at 7-9, ff. Tr. 4126; Fletcher, Applicant Prepared Testimony at 7-8, ff. Tr. 5908.) Accordingly, the Applicant is implementing a strict AVT water chemistry program on the secondary side of the reactor systems at the Byron Station. (Blomgren, Applicant Prepared Testimony at 3-11, ff. Tr. 4126; Fletcher Applicant Prepared Testimony at 11, ff. Tr. 5908.)

181. AVT involves the addition of volatile chemicals as control agents. These agents do not concentrate in the

steam generator but are removed via the steam to the remainder of the secondary system. Generally two chemicals are added, a volatile amine (usually ammonium hydroxide) for pH control of the feedwater and an oxygen scavenger (hydrazine).

Hydrazine scavenges oxygen producing innocuous byproducts such as nitrogen and water. As the hydrazine moves through the feedwater system and is subjected to higher temperatures, any unreacted hydrazine can decompose to form volatile compounds such as ammonia, nitrogen and hydrogen. The addition of ammonium hydroxide for pH control is adjusted to compensate for that produced from the excess hydrazine thermal decomposition. (Wootten, Applicant Prepared Testimony at 9-10, ff. Tr. 4126.)

182. Applicant's AVT water chemistry program is based on Westinghouse and EPRI guidelines (Applicant Exhibit 17). (Blomgren, Applicant Prepared Testimony at 3-11, ff. Tr. 4126.)

183. The Westinghouse guidelines, introduced in 1977 and modified subsequently from time to time, recommend that: (1) the guideline chemistry conditions should be achieved prior to unit loading and maintained during power changes; (2) any source of contamination should be identified, the source corrected and no operation allowed with locatable contaminant ingress; (3) dissolved oxygen at the condensate pump discharge should be less than 10 ppb to minimize the inventory of corrosion product transported to the steam

generator; (4) continuous monitoring of the chemistry of the steam generator blowdown should be performed; measured values should be compared to theoretical values in order to identify whether or not excess alkalinity or acidity is present; (5) copper bearing alloys should be eliminated from the secondary system to permit greater flexibility and optimization in chemistry control; (6) main condenser integrity should be upgraded to minimize the ingress of impurities in the condensate in order to improve the reliability of the steam generators and turbine; and (7) if a full-flow condensate polishing system is installed, it must be carefully controlled and properly operated in order to optimize the quality of the treated condensate. (Wootten, Applicant Prepared Testimony 14-16, ff. Tr. 4126.)

184. The so-called EPRI Guidelines were developed under the aegis of the Steam Generator Owner's Group (SGOG), of which Applicant is a founding member. These guidelines incorporate even more restrictive water chemistry controls than the Westinghouse guidelines and include a staged corrective action plan. In addition to the more restrictive water chemistry controls, the EPRI Guidelines include recommendations for data management, surveillance, and analytical methods. The EPRI Guidelines include a recommendation that specific management responsibilities regarding secondary water chemistry control be assigned from the plant chemist to senior corporate management. (Blomgren, Applicant Prepared Testimony at 9, ff. Tr. 4126; Applicant Ex. 17.)

185. In addition to the Westinghouse Guidelines noted above, the Byron Station Chemistry Monitoring Program incorporates the following elements from the EPRI Guidelines: (1) more restrictive EPRI water chemical controls coupled with a corrective action plan to require prompt station response to a chemistry excursion before unit shutdown is required; (2) a staged corrective action plan based upon the level and duration of contaminant ingress, requiring specific corrective actions, including staged reductions in power; (3) a data management and surveillance program providing for prompt identification of negative trends or inconsistencies in chemical control data; (4) an analytical program to supplement and verify the continuous on-line chemistry monitoring system data. Although not specifically included in the Byron Chemistry Monitoring Program, the statement of management responsibilities recommended in the EPRI Guidelines is being addressed in a Commonwealth Edison corporate PWR Secondary Water Chemistry Control Program. (Id. at 10-11.)

186. Plants that have only operated on AVT have experienced some denting. (Wootten, Applicant Prepared Testimony at 13-14, ff. Tr. 4126.) However, the denting problem can be mitigated by rigorous implementation of a system-wide AVT program that includes (1) reduction of the ingress of oxidizing agents such as copper and/or oxygen and (2) further restriction on the introduction of chloride ions into

the secondary system. (Id. at 12-14.) Applicant has undertaken these mitigation measures. (Wootten, Tr. 4180-82, 4188, 4194-95; Blomgren, Tr. 4183, 4195-96, Blomgren, Applicant Prepared Testimony at 6-7, ff. Tr. 4126.)

187. There have been limited occurrences of other corrosion mechanisms in plants where AVT has been the exclusive water chemistry control. One plant experienced pitting of the Inconel tubing which is believed to be due to an acidic chloride condition involving copper and chloride ions. There have been an insignificant number of stress-corrosion cracking incidents. A form of tube thinning has been observed at lower tube support plate elevations around the periphery of the bundle at two all-AVT plants. (Wootten, Applicant Prepared Testimony at 14, ff. Tr. 4126.) These few occurrences must be kept in perspective inasmuch as approximately 60 plants are in operation using AVT as the exclusive water chemistry control process. (Malinowski, Tr. 4173.)

188. As Intervenor's witness acknowledged, there have been no tube ruptures due to corrosion during the past 7 years. (Bridenbaugh, Tr. 6454; NRC Staff Prepared Testimony at 5, ff. Tr. 4473; see Fletcher, Applicant Prepared Testimony at 11-13, ff. Tr. 5908.)

189. Over the last eight years, the utility industry has come to understand more fully the importance of rigorous secondary side water chemistry control. Applicant's com-

mitment to strict adherence to AVT guidelines, issued by both Westinghouse and EPRI, enhances the long-term integrity of the reactor coolant pressure boundary by minimizing the corrosion of condenser and feedwater materials, which minimizes the formation of corrosion products that are delivered to the steam generator. The result is reduced potential for tube corrosion. (Wootten, Applicant Prepared Testimony at 16, ff. Tr. 4126; Fletcher, Applicant Prepared Testimony at 7-9, ff. Tr. 4126.)

190. Although some degree of corrosion will always take place in any metal that maintains contact with an aqueous environment, the design and secondary water chemistry implementations will minimize the degree of steam generator tube corrosion. (McCracken, NRC Staff Prepared Testimony at 4, ff. Tr. 4473; Malinowski, Tr. 4231; Marsh, Frank, Tr. 4713-15.)

191. The Board finds that the design features combined with the rigorous AVT water chemistry control program will minimize tube wall degradation at the Byron Plant.

192. In addition to the design and secondary water chemistry implementations for minimizing steam generator tube wall degradation, Applicant has taken numerous measures for the monitoring, detection, and remediation of any such degradation. A 100% pre-service inspection has been performed on the steam generator tubes in Unit 1 and will be performed on the steam generator tubes in Unit 2 prior to

its operation, pursuant to NRC Regulatory Guide 1.83. The purpose of performing the pre-service inspections is to establish a baseline against which subsequent in-service inspections can be compared. (Paillaman, Applicant Prepared Testimony at 4, ff. Tr. 4818; Blomgren, Applicant Prepared Testimony at 11, ff. Tr. 4126.)

193. Although NRC Regulatory Guide 1.83 allows use of fabrication shop examinations as an adequate baseline examination, Commonwealth Edison performed a baseline examination in the field under conditions and with equipment and techniques equivalent to those that are expected to be employed for subsequent in-service examinations. (Paillaman, Applicant Prepared Testimony at 4, ff. Tr. 4818.) The data was reported in accordance with Article IV-6000 of ASME Section XI, which requires the reporting of tube wall penetrations in excess of 20% of the tube wall thickness and tube wall dents. (Id. at 6.)

194. Based upon the inspection, a baseline was established for Unit 1. The inspection revealed that two tubes were partially blocked. Based on a Westinghouse recommendation, the two tubes were plugged. (Id. at 7; Paillaman, Tr. 4821.) Although some processing-induced denting was discovered, the dents or "dings" were not considered sufficiently significant to warrant tube plugging. (Id. at 8; Paillaman, Malinowski, Tr. 4825-26.) Finally, no tube locations were found with 20% or greater wall loss.

(Paillaman, Applicant Prepared Testimony at 10, ff. Tr. 4818.)

195. The condition of the Byron steam generators will be periodically monitored through a comprehensive inspection program. Inspections will be performed according to the provisions of the Byron Technical Specifications and according to NRC Regulatory Guide 1.83. (Blomgren, Tr. 4282-4282 A; Blomgren, Applicant Prepared Testimony at 11, ff. Tr. 4126; Frank, NRC Staff Prepared Testimony at 4, ff. Tr. 4473.)

196. The results of these periodic inspections will be compared to the 100% pre-service baseline examination. This comparison provides an ongoing evaluation of the steam generator tubing condition and allows time for the initiation of appropriate measures for steam generator maintenance prior to any occurrence of primary to secondary leakage. (Blomgren, Applicant Prepared Testimony at 11, ff. Tr. 4126.)

197. NRC Regulatory Guide 1.83 requires that in-service inspections be performed every 12 to 24 months. (NRC Regulatory Guide 1.83, Section C6(b).) In cases where the degradation processes have been highly active, the Staff has required that the inspections be performed at more frequent intervals, consistent with the rate at which degradation is occurring. (Frank, NRC Staff Prepared Testimony at 4-5, ff. Tr. 4473.)

198. The Technical Specifications also call for sampling 3% of all the tubes in the plant in the first in-service inspection. The scope of the inspection may be expanded to cover 100% of the tubes in circumstances where either (i) greater than 10% of the tubes inspected have eddy current indications greater than 20%, or (ii) greater than 10% of the tubes inspected exceed the plugging criterion. (Malinowski, Applicant Prepared Testimony at 7, ff. Tr. 4126.)

199. Subsequent inspections will relate to all unplugged tubes previously identified to have eddy current indications greater than 20% as well as the 3% total sample provided by the Technical Specifications (Id. at 7-8.)

200. Eddy current testing is the primary inspection technique and it is usually performed using an instrument that impresses four different test frequencies on the coils simultaneously. The frequencies are selected on the basis of providing definitive information on tube degradation, support plates and external deposits. Because the responses for external discontinuities vary according to the test frequency, it is possible by linear combinations of the responses at different frequencies to reduce unwanted signals from a composite response. (Id. at 12; Frank, Tr. 4726.) Although Intervenor's witness questioned the amount of experience with eddy current testing, he acknowledged that multifrequency testing is a significant improvement over earlier methods and may be adequate. (Bridenbaugh, Tr. 6461.)

201. Eddy current testing enables one to determine the depth, length and volume of material affected. (Malinowski, Applicant Prepared Testimony, at 12, ff. Tr. 4126.) The sensitivity of eddy current testing to tube wall degradation varies depending upon the size, shape and nature of the degradation. Eddy current testing will reliably detect the various types of tube degradation at the following sensitivity levels: tube wall thinning at 20% depth of the tube wall (id. at 16); pitting at 20% tube wall depth (id. at 17); tube wall cracking at 40% tube wall depth (id. at 18-20); intergranular attack at 40% of tube wall depth (id. at 21); tube wear at 20% of tube wall depth (id. at 21-22). Denting, a form of tube deformation, is readily detected by eddy current testing. (Id. at 13-14.) Eddy current development now in progress in the industry is expected to further improve detection limits and characterization of possible tube degradation. (Id. at 22; Blomgren, Applicant Prepared Testimony at 12, ff. Tr. 4126.)

202. A review of the sensitivity of the eddy current method for detecting the degradation phenomena encountered in steam generator operating experience demonstrates that, for each case, tube wall penetration at or below the plugging limit of 40% is detectable. Thus, significant tube degradation is expected to be detected by eddy current testing. (Id. at 22.)

203. If a significant rate of tube degradation is determined as a result of an in-service inspection, measures

are available to reduce the probability that tube leakage will occur before the next scheduled inspection. These measures include alterations of the steam generator environment to reduce the degradation rate such as temperature changes, water lancing, flushing and chemical cleaning; mechanical or design modifications; fore-shortening the interval between inspections; and increasing the allowance for corrosion by lowering the tube plugging limit. (Id. at 8.)

204. Tube wear due to foreign objects can cause a tube rupture. Two such incidents have occurred at Prairie Island Unit 1 in 1979 and at Ginna in 1982. (Fletcher, Applicant Prepared Testimony at 14, ff. Tr. 5908.)

205. To guard against the introduction of foreign objects in the secondary side of the Byron steam generators, Applicant has implemented a loose parts control program. (Blomgren, Applicant Prepared Testimony at 12, ff. Tr. 4126.)

206. First, materials and tools used in the steam generators during maintenance and inspection will be controlled through written tool and material inventory control procedures. The control procedures will require an inventory and accounting of all tools and materials entering the secondary side of the steam generators prior to return to operation. In addition, hold points will be required in maintenance procedures for cleanliness operations. (Id. at 12-13.)

207. Second, a Loose Parts Monitoring System (LPMS) will be employed. (Id. at 14; Frank, NRC Staff

Prepared Testimony at 3, ff. Tr. 4473.) The LPMS for Byron is a monitoring, alarm and diagnostics system that provides real-time information to the operator on a variety of mechanical vibration phenomena that may occur in a reactor coolant system. The system includes two sensors on the secondary side of each steam generator which monitor the generator for noise generated by loose parts. Moreover, the secondary side will be visually inspected from time to time. (Blomgren, Applicant Prepared Testimony at 14, ff. Tr. 4126; Blomgren, Tr. 4256-57.)

208. Commonwealth Edison also will monitor the Byron reactor systems for primary to secondary leakage. (Patel, Applicant Prepared Testimony at 11-13, ff. Tr. 4126.) To ensure tubes are not degraded to the point where rupture is possible, licensees are required to shut down and repair steam generator tubes should the primary to secondary leakage exceed a maximum allowed by technical specification. (Rajan, Frank, NRC Staff Prepared Testimony at 3, 7, ff. Tr. 4473.)

209. The maximum permissible leak rate is established consistent with the "leak-before-break" principle set forth in the Technical Specifications. This principle is effected by utilizing Inconel 600 in steam generator tubing, a material in which degradation in the form of cracking penetrates through the wall causing a small primary to secondary leak long before the crack reaches a linear

length called the "critical length", where tube rupture can occur. (Patel, Applicant Prepared Testimony at 11-12, ff. Tr. 4126.)

210. The maximum permissible leak rate during normal operation at Byron has been established through leak rate and burst pressure tests at the Standard Technical Specification limit of .35 gpm per steam generator. (Id. at 12, Marsh, Tr. 4735.) This corresponds to a maximum allowable crack length of .43 inch. (Patel, Applicant Prepared Testimony at 12, ff. Tr. 4126.)

211. The critical crack length corresponding to the maximum accident condition pressure during a postulated Main Feedwater Line Break (MFLB) or Main Steam Line Break (MSLB) was conservatively determined to be .51 inch using the results of the burst pressure tests. (Id. at 12-13.)

212. Thus, shutdown for remedial action will take place long before the critical crack or calculated burst point of .51 inch is reached. (Id. at 13.)

213. In addition to eddy current testing and leak-before-break characteristics, periodic hydrostatic testing of the steam generators is performed to provide an overall measure of the tube bundle integrity. (Fletcher, Applicant Prepared Testimony at 17, ff. Tr. 5908.)

214. Partially degraded tubes are acceptable for continued service as long as it is assured, through in-service inspections (utilizing eddy current techniques) and leakage monitoring, that degraded tubes meet the applicable

tube wall and associated strength requirements to safely withstand all operating and design basis accident condition loads. (Patel, Applicant Prepared Testimony, at 6, ff. Tr. 4126.)

215. Section XI of the ASME Boiler and Pressure Vessel Code (Code) provides guidelines for establishing the "limiting safe conditions" of tube degradation beyond which defective tubes must be repaired or removed from service. (Patel, Applicant Prepared Testimony at 6, 10, ff. Tr. 4126.) The limiting safe condition takes into account (1) the minimum tube wall thickness needed in order to sustain the imposed normal operating and postulated design basis accident condition tube loads; (2) maximum (Technical Specification) permissible leak rate during normal operation to preclude a tube rupture during a postulated main steam line break accident and (3) allowance for continued degradation between inspections and eddy current measurement uncertainties. (Id. at 7.)

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

F of Section III of the Code. (Id. at 10.)

217. Applicant follows Section XI of the Code in lieu of the plant-specific program described in NRC Regulatory Guide 1.121. The amount of tube wall degradation above which the tube shall not continue in service, the tube plugging criterion, established by Section XI of the Code, paragraph IWB-3521.1, is a depth of outside wall penetration not to exceed 40% of the wall thickness for tubing from SB 163 material when the mean tube radius to wall thickness ratio is less than 8.70. (Id. at 10-11.)

218. The 40% plugging criterion is based upon ASME calculations concluding that steam generator wall tubing can sustain degradation in excess of 50% of wall thickness and still meet all applicable stress and strength requirements. (Id. at 14; Patel, Tr. 4133, 4372, 4433.) The 40% figure is reached by conservatively allowing for a 10% uncertainty factor in both eddy current measurement and corrosion allowance for continued plant operation until the next inspection. (Patel, Applicant Prepared Testimony at 14-15, ff. Tr. 4126; Malinowski, Tr. 4433.) Moreover, based on Regulatory Guide 1.121, it could be concluded that the thickness necessary for a tube to bear accidents or normal operation generally approaches 40%, i.e., 60% degradation, thereby building in an extra 10% conservatism. (Malinowski, Tr. 4433-34.) Thus, significant tube degradation will be detected well within the sensitivity limits of eddy current

testing, 20% to 40% tube wall loss depending upon the type of tube degradation. (See Finding 201.)

219. The steam generators at Byron are of the same design as those that were evaluated in arriving at the 40% plugging criteria in Section XI of the Code. The tubing material is Inconel 600 (SB-163) and the mean radius to thickness ratio is 8.22 (less than 8.70, the upper limit specified in Section XI). Accordingly, the 40% plugging criterion established by Section XI is conservatively applicable to the Byron Station. (Id. at 15; Patel, Tr. 4143; Blomgren, Applicant Prepared Testimony at 18, ff. Tr. 4126.)

220. The Board finds that the numerous measures instituted by Applicant for monitoring and detection of all forms of degradation and for detection of potential sources of degradation, combined with the establishment of a conservative 40% tube plugging criterion, will provide a safe margin for implementing any remedial action before the potential for a tube rupture occurs.

221. The Science Applications, Inc. (SAI) Report, entitled "Value-Impact Analysis of Recommendations Concerning Steam Generator Tube Degradations and Rupture Events", Joint Intervenors' Exhibit 9, is simply a cost-benefit analysis and is only one part of the overall NRC Staff assessment of potential recommendations with respect to Unresolved Safety Issue A-3. (Marsh, Tr. 4476-77.) At present, the SAI

Report is a document that will undergo extensive internal review and eventually be sent to the Commission. In any event, it is not expected by the NRC Staff to become a rule in any sense. (Marsh, Tr. 4753-54.) Accordingly, the Board finds that Applicant's adherence to the standards mandated by 10 C.F.R. Part 50, Appendix A should not be measured by the potential recommendations contained in the SAI cost-benefit analysis.

222. The NRC's response to the River Bend decision, which instructs the Staff to evaluate the degree to which an unresolved safety issue (USI) affects the operating license proceeding, reveals that the design and inspection measures implemented by Applicant should minimize any steam generator problems that are encountered. (NRC Staff Exhibit 1, Appendix C at C-10.) Accordingly, the Board finds that the USI poses no obstacle to granting an operating license to the Byron facility.

223. Mr. Fletcher assimilated the conclusions provided by the Applicant's expert witnesses testifying with respect to their specific disciplines and reached an overall assessment as to steam generator tube integrity at the Byron Station. Based upon the design, water chemistry, detection and remedial measures undertaken by Applicant, Mr. Fletcher concluded that steam generator tube degradation at the Byron Station should not be a safety concern and that tube rupture should not occur, even under conditions of Main Steam Line

Break (MSLB) or Loss of Coolant Accidents (LOCA's). (Fletcher, (Fletcher, Applicant Prepared Testimony, at 18, ff. Tr. 5908.)

224. Mr. Fletcher's qualitative judgment that tube rupture should not occur under accident conditions was confirmed quantitatively by an evaluation performed by Mr. Hitchler. Mr. Hitchler's undisputed quantitative assessment demonstrates that single and multiple tube ruptures as initiating events are predicted to occur at frequencies of 3×10^{-2} and 3×10^{-5} per year respectively. The frequency of multiple tube ruptures combined or as a consequence of (i) large LOCA events is 5×10^{-7} and (ii) small LOCA and transient events with normal pressure differentials is 2×10^{-5} . The frequency of multiple tube ruptures combined or as a consequence of main steam/feedwater line break events is 3×10^{-5} . (Hitchler, Applicant Prepared Testimony at 5, ff. Tr. 5908.) The frequencies of occurrence for the foregoing accidents are beyond the range of probabilities established generally for design basis accidents. (Id. at 6-8.)

225. Although Intervenor's witness indicated that there is an increased probability of accidents initiated by tube failures occurring under normal conditions or as a result of the degraded condition of steam generator tubes, he performed no calculations to support that statement. (Bridenbaugh, Tr. 6474-75.) Mr. Hitchler's quantitative results disclose that the frequency of postulated tube ruptures

combined or as a consequence of transient conditions and accident conditions such as MSLB's and LOCA's is postulated to be extremely low over the 40-year life of the plant.

(Hitchler, Applicant Prepared Testimony at 6, ff. Tr. 5908; Rajan, Frank, NRC Staff Prepared Testimony at 7, ff. Tr. 4473.)

226. Even if either of these extremely low probability accidents occurred, tube rupture should not occur. (Frank, NRC Prepared Testimony at 7-9, ff. Tr. 4473; Fletcher, Applicant Prepared Testimony at 18, ff. Tr. 5908.) The steam generator tubes are designed to withstand the differential pressures likely to be experienced during the most severe LOCA, MSLB, or MFLB accidents. The tube wall thickness and internal support arrangements are such that, even with some tube wall degradation, a MSLB, MFLB, or LOCA will not result in breakage of a steam generator tube. The likely effect, if any, of a large MSLB or MFLB would be that degraded steam generator tubes would develop a small leak that would be detected through monitoring, followed by reactor shutdown, and then repair. (Patel, Applicant Prepared Testimony at 8-13, ff. Tr. 4126; Fletcher, Applicant Prepared Testimony at 16-17, ff. Tr. 5908; see Finding 208.)

227. The PWR operating experience (presently 350 reactor years) of no steam generator tube ruptures occurring as a result of MSLB's, LOCA's or other events should be expected to continue. (Hitchler, Applicant Prepared Testimony at 6, ff. Tr. 5908.) Applicant's and NRC Staff's witnesses

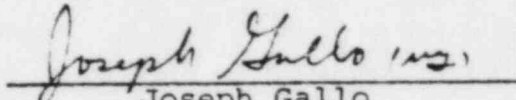
concur that rupture occurring as a result of steam generator tube degradation under normal operating conditions as well as under accident conditions is not a safety concern. (Id. at 9; Fletcher, Tr. 6258; Marsh, NRC Staff Prepared Testimony at 8-9, ff. Tr. 4473.)

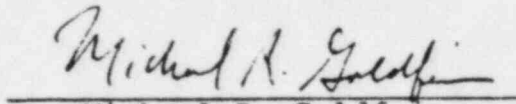
228. The Board finds that rupture of steam generator tubes either in normal operating or in accident conditions is not a safety concern.

229. Based upon (1) the design of the Byron steam generators, including the 90-10 bypass and tube expansion modification program to address the flow-induced vibration phenomenon, (2) the comprehensive AVT secondary water chemistry program, (3) the various monitoring, detection, and remedial measures instituted by Applicant, including preoperational inspections, in-service eddy current testing, the loose-parts monitoring system, the leak-before-break and critical-crack principles, and the tube plugging criterion, and (4) the resistance of the steam generator tubes to rupture even in the face of low probability events such as MSLB's, MFLB's or LOCA's, the Board finds the reactor coolant pressure boundary is designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The Board also finds that the Byron steam generators are designed to permit periodic inspection and testing of critical areas to assess their structural and leaktight

integrity. Accordingly, the Board concludes that 10 CFR Part 50, Appendix A, General Design Criteria 14, 31, and 32 are satisfied.

The foregoing document, "Commonwealth Edison Company's Proposed Findings of Fact and Conclusions of Law Regarding Steam Generator Tube Integrity" is respectfully submitted by the undersigned attorneys for Commonwealth Edison Company.


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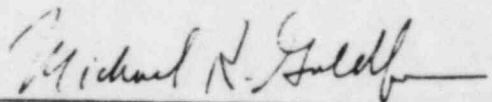
UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)	
)	
COMMONWEALTH EDISON COMPANY)	Docket Nos. 50-454 OL
)	50-455 OL
(Byron Nuclear Power Station,)	
Units 1 & 2))	

CERTIFICATE OF SERVICE

The undersigned, one of the attorneys for Commonwealth Edison Company, certifies that he filed the original and two copies of the attached "COMMONWEALTH EDISON COMPANY'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW REGARDING STEAM GENERATOR TUBE INTEGRITY" with the Secretary of the Nuclear Regulatory Commission and served a copy of the same on each of the persons at the addresses shown on the attached service list. Service on the Secretary and all parties, unless otherwise indicated, was made by deposit in the U.S. Mail, first-class postage prepaid, this 7th day of June, 1983.



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