FENOC
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
Route 168
P.O. Box 4
Shippingport, PA 15077-0004

L. William Pearce
Site Vice President

724-682-5234
Fax: 724-643-8069

June 28, 2004
L-04-089

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
License Amendment Request No. 328
Revised Steam Generator Inspection Scope for One Cycle of Operation

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above license for Beaver Valley Power Station (BVPS) Unit No. 1 in the form of changes to the Technical Specifications. The proposed amendment will revise the scope of the steam generator tube sheet inspections for one operating cycle using the W* methodology.

The FENOC evaluation of the proposed changes are presented in the Enclosure. The proposed Technical Specification changes are presented in Attachment A. Attachment B provides the proposed information-only changes to the Technical Specification Bases that reflect the proposed license amendment. Attachment C identifies the commitments made in this submittal. Attachment D provides one copy of WCAP-14798-NP, Revision 2, "Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions," dated March 2003 (non-proprietary). Attachment E provides one copy of WCAP-14797-P, Revision 2, "Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions," dated March 2003 (proprietary). Attachment F provides additional information supporting the application of W* to the BVPS Unit No. 1 steam generator tubes, which is proprietary to Westinghouse (Westinghouse Letter LTR-SGDA-04-211, dated June 2004). Attachment G provides a redacted (non-proprietary) version of the information in Attachment F (Westinghouse Letter LTR-SGDA-04-212, dated June 2004).

The Beaver Valley review committees have reviewed the changes. The changes were determined to be safe and do not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety evaluation and no significant hazard evaluation.

A001

FENOC requests approval of the proposed amendment by October 15, 2004, in order to support implementation of the proposed changes for the next BVPS Unit 1 refueling outage in the fall of 2004. Once approved, the amendment shall be implemented within 60 days.

Enclosed are Westinghouse authorization letters, CAW-04-1847 and CAW-04-1848, accompanying affidavits, Proprietary Information Notices, and Copyright Notices.

As Attachments E and F contain information proprietary to Westinghouse Electric Company LLC, each is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-04-1847/CAW-04-1848 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

If there are any questions concerning this license amendment request, please contact Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement at 724-682-5284.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 28, 2004.

Sincerely,

L. W. Pearce

L. William Pearce

Enclosures:

- FENOC Evaluation of the Proposed Change
- Westinghouse authorization letter, CAW-04-1847, dated June 18, 2004
- Westinghouse authorization letter, CAW-04-1848, dated June 18, 2004
- Proprietary Information Notices
- Copyright Notices

Attachments to FENOC Evaluation of the Proposed Change:

A. Proposed Technical Specification Changes (mark-ups)
B. Proposed Changes to Technical Specification Bases (Information only)
C. List of Regulatory Commitments
D. WCAP-14798-NP, Revision 2 (Non-Proprietary)
E. WCAP-14797-P, Revision 2 (Proprietary)
F. Additional Information Supporting the Application of W* to the Beaver Valley Unit Steam Generator Tubes, LTR-SGDA-04-24, dated June 18, 2004 (Proprietary)
G. Redacted Version of Westinghouse Proprietary Information Provided in Attachment F, LTR-SGDA-04-212, dated June 18, 2004 (Non-Proprietary)

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP (w/o Attachments E and F)
Mr. L. E. Ryan (BRP/DEP) (w/o Attachments E and F)

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4643
Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: CAW-04-1847

June 18, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-SGDA-04-211, "Additional Information Supporting the Application of W* to the Beaver Valley Unit 1 Steam Generator Tubes" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-04-1847 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by FirstEnergy Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-04-1847, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: W. Macon, NRC
E. Peyton, NRC

bcc: J. A. Gresham (ECE 4-7A) 1L
R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L, 1A (letter and affidavit only)
J. M. Hall, Waltz Mill
N. B. Closky, ECE-410L
G. A. Brassart, ECE-411D
G. W. Whiteman, ECE-410C
R. F. Keating, Waltz Mill
W. K. Cullen, Waltz Mill
L. A. Nelson, Waltz Mill

CAW-04-1847

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J A Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 18th day
of June, 2004

Sharon L Fiori

Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The 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) It 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) Its use 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 a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

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

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-SGDA-04-211, "Additional Information Supporting the Application of W* to the Beaver Valley Unit 1 Steam Generator Tubes" (Proprietary) dated June 2004. The information is provided in support of a submittal to the Commission, being transmitted by FirstEnergy Nuclear Operating Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Beaver Valley Unit 1 is expected to be applicable to other licensee submittals in support of implementing the W* inspection methodology addressing service induced degradation in the tube joint region of steam generators.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation of the analyses, methods, and testing for the implementation of the W* tube inspection methodology.

(b) Provide evaluation of the required W* engagement lengths for Beaver Valley Unit 1.

(c) Provide a bounding W* potential steam line break leakage evaluation for Beaver Valley Unit 1.

(d) Assist the customer to respond to NRC requests for information.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of this information to its customers in the licensing process.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar licensing support documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

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Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: CAW-04-1848

June 18, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-14797-P, Rev. 2, "Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions," March 2003 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-04-1848 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by FirstEnergy Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-04-1848, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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Regulatory Compliance and Plant Licensing

Enclosures

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AFFIDAVIT

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Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 18th day
of June, 2004

Sharon L. Fiori
Notary Public

Notarial Seal
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Monroeville Boro, Allegheny County
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This information is part of that which will enable Westinghouse to:

(a) Provide documentation of the analyses, methods, and testing for the implementation of the W* tube inspection methodology.

(b) Provide evaluation of the required W* engagement lengths for Beaver Valley Unit 1.

(c) Provide W* steamline break leakage evaluation for Beaver Valley Unit 1.

(d) Assist the customer to respond to NRC requests for information.

Further this information has substantial commercial value as follows:

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COPYRIGHT NOTICE

ENCLOSURE

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 328

FENOC Evaluation of the Proposed Change

Subject: Application for Amendment of Technical Specification 3/4.4.5 to Revise the Steam Generator Inspection Scope for One Cycle of Operation.

Table of Contents

Attachments

1.0 DESCRIPTION

FirstEnergy Nuclear Operating Company (FENOC) requests to amend Operating License DPR-66 for Beaver Valley Power Station (BVPS) Unit No. 1. The proposed amendment would revise Technical Specification 3.4.5, "Steam Generators," to change the scope of the steam generator (SG) tube sheet inspections required in the SG tube sheet region for one operating cycle using the W* methodology. (W* is defined in WCAP-14797, Revision 2 and in this document on page 3).

Specifically, the proposed change will revise the Unit No. 1 Technical Specification definition for steam generator tube inspection included in BVPS Technical Specification (TS) Surveillance Requirement (SR) 4.4.5.4.a.8 to revise the definition to exclude the portion of the tube within the tubesheet below the W* distance. The proposed change will also revise SR 4.4.5.4.a.6 on steam generator tube repair criteria, add SR 4.4.5.2.e to require a 100 percent rotating probe inspection of the hot leg tubesheet W* distance, add new W* terminology definitions in 4.4.5.4.a.11, and add new reporting criteria for W* inspection information in 4.4.5.5.d.1 and 4.4.5.5.e. FENOC's proposed change requires that any tube identified with service induced degradation in the W* distance or less than eight inches below the top-of-tube sheet (TTS), which ever is greater, must be repaired. Since FENOC proposes to repair any service induced degradation within the W* distance, FENOC's proposal is a conservative limited scope application of the complete W* methodology as described in WCAP–14797, Revision 2.

2.0 PROPOSED CHANGE

The proposed changes to the Unit 1 Technical Specification (TS) will revise the definition of steam generator tube inspection to exclude the portion of the tube within the tubesheet below the W* distance, tube to tubesheet weld and tube end extension. This will ensure that inspections may continue to be utilized to ensure the safe operation of the plant while limiting unnecessary radiation exposure to plant personnel.

FirstEnergy Nuclear Operating Company (FENOC) is proposing to modify the Beaver Valley Power Station (BVPS) Unit 1 TSs to revise SRs 4.4.5.4.a.6, 4.4.5.4.a.8, and 4.4.5.5.d.1 and add SRs 4.4.5.2.e, 4.4.5.4.a.11 and 4.4.5.5.e for one cycle of operation. This proposed revision would be applicable only to Cycle 17. SR 4.4.5.4.a.8 defines steam generator (SG) tube inspection scope. FENOC's proposed change alters the tube inspection to exclude the portion of the tube within the tubesheet below the W* distance, tube to tubesheet weld and tube end extension by crediting the methodology as described by WCAP-14797, Revision 2. The W* distance is the distance from the top of the tubesheet to the bottom of the W* length (The maximum length of tubing below the bottom of the WEXTEX transition {BWT} which must be demonstrated to be non-degraded and is defined as 7.0 inches below the bottom of the WEXTEX transition on the hot leg side) plus the distance to the BWT and uncertainties. SR 4.4.5.4.a.6 provides steam generator tube repair criteria. FENOC's proposed change requires that any service induced degradation identified in the W* distance or less than eight inches below the top-of-tube sheet, which ever is greater, be repaired.

The proposed Technical Specification change, which is submitted for Nuclear Regulatory Commission (NRC) review and approval, is provided in Attachment A. Attachment A also shows changes pending from the prior BVPS Unit 1 license amendment request 322 submitted by FENOC on January 27, 2004. The changes proposed to the Technical Specification Bases are provided in Attachment B. The proposed Technical Specification Bases changes do not require NRC approval. The Beaver Valley Power Station (BVPS) Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. The Technical Specification Bases changes are provided for information only. Attachment C provides a list of commitments associated with this License Amendment Request (LAR). The W* methodology is described in WCAP-14797-NP, Revision 2, "Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions" dated March, 2003 (nonproprietary) contained in Attachment D and WCAP-14797-P, Revision 2, "Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions" dated March, 2003 (proprietary) contained in Attachment E. These reports detail the analyses and testing performed to verify the adequacy of the W* methodology. Attachment F provides additional information which is proprietary to Westinghouse. Attachment G provides a redacted version of the information in Attachment F and is non-proprietary.

The proposed change to the Technical Specifications and Technical Specification Bases have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined or as inserted text. This presentation allows the reviewer to readily identify the information that has been deleted and added.

To meet format requirements the Index, Technical Specifications and Bases pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

3.0 BACKGROUND

3.1 Reason for Requested Changes

Industry Operating Experience from Diablo Canyon Unit 1 steam generator tubesheet inspections in April, 2004 identified unexpected axial Primary Water Stress Corrosion Cracking (PWSCC) indications about 1 inch from the tube end, just below the shop roll in the deplugged tube expansion zone (PEZ). The cause of the PEZ PWSCC is suspected to be due to high residual stresses caused by the plug expansion processes. The cracking was noted to be coincident with the location of the expanded location of the plug. It is believed that when the plug is removed, the residual stresses of the expanded roll area remains high, and in combination with high Reactor Coolant System temperature and susceptible Alloy 600 tubing material, PWSCC developed in 1 to 2 operating cycles. These indications were unexpected since this degradation was not present during prior inspections performed immediately following tube plug removal.

The steam generators (SGs) at BVPS Unit 1 are Westinghouse Model 51 with a U-tube configuration. Each tube is secured in the tubesheet above the lower plenum of the SG by an explosive expansion process (WEXTEX). The WEXTEX process expands each tube over its entire length of the tubesheet and forms an interference fit between the tube and tubesheet. This interference fit forms the interface which provides the structural and part of the leaktight boundary between the primary and secondary systems at each end of a SG tube. Located near the top of the tubesheet is a region where the tube transitions from the tubesheet hole diameter to that of the original tube. This region is referred to as the WEXTEX transition region.

Similar to Diablo Canyon Unit 1, BVPS Unit 1 has also had previously installed steam generator tube plugs removed and steam generator tubes placed back into service. Although BVPS Unit 1 does not yet have any prior inspection results which may indicate axial PWSCC indications at the SG tube end in the SG tubesheet similar to the indications reported at Diablo Canyon, it is possible that BVPS Unit 1 may be susceptible to have PWSCC indications just below the deplugged tube expansion zone following plant operation since tube plug removal as experienced by Diablo Canyon.

An alternate tube repair criteria (referred to as W*) was developed by Westinghouse Electric Company (Westinghouse) to permit tubes with predominantly axially oriented primary water stress corrosion cracking (PWSCC) in the WEXTEX tubesheet expansions to remain in service. The W* methodology defines a W* length that would permit flaws to remain in service and assure adequate strength is available to resist the axial pullout loads experienced within the tubesheet. This proposed change will revise the Technical Specification required inspection scope for SG tube inspections to exclude the length of tubing below the W* length on the hot leg side by crediting the W* methodology as described by WCAP-14797, Revision 2, for one cycle of operation. This proposed revision would be applicable only to Cycle 17.

3.2 W* Background

Existing plant Technical Specification tube repair/plugging criteria apply throughout the tube length and do not take into account the reinforcing effect of the tubesheet on the external surface of an expanded tube. The presence of the tubesheet constrains the tube and complements tube integrity in that region by essentially precluding tube deformation beyond the expanded outside diameter. The resistance to both tube rupture and tube collapse is significantly enhanced by the tubesheet. In addition, the proximity of the tubesheet in the expanded region significantly reduces the leakage of throughwall tube cracks. Based on these considerations, the establishment of W* methodology criteria to the portion of tubing expanded by Westinghouse explosive tube expansion is supported by testing and analysis results included in WCAP-14797, Revision 2.

For 51 Series steam generators with WEXTEX expansions at BVPS Unit 1, the full depth tube-to-tubesheet expansion can be defined as follows. From the lower tube end and extending upward for a length of approximately 2.75 inches is a region expanded by the roll expansion process. From the top of the rolled expansion region to the vicinity of the top of the tubesheet (TTS), the tube-to-tubesheet expansion is accomplished by the WEXTEX explosive expansion process. The resulting full depth tube-to-tubesheet expansion can be considered as four distinct areas. These are described in WCAP-14797, Revision 2 as:

1. The Roll Region – The region of tube which has been expanded by a tube rolling process. This region extends from the bottom of the tube to approximately 2.75 inches above the bottom of the tube.
2. The Roll Transition – The portion of the tube which extends from the roll expanded region of the tube to the initially unexpanded region, and which is subsequently expanded by the WEXTEX process.
3. The WEXTEX Region – The portion of the tube expanded by the explosive expansion process to be in contact with the tubesheet. This region starts at the roll transition and extends to the WEXTEX transition in the vicinity of the top of the tubesheet.
4. The WEXTEX Transition – The portion of the tube which acts as a juncture between the WEXTEX region and the unexpanded region of the tube.. The region starts at the top of the explosively expanded region and extends for approximately 0.25 inches.

A W* methodology has been developed by Westinghouse to permit tubes with predominantly axially oriented primary water stress corrosion cracking in the WEXTEX tubesheet and hardroll region to remain in service. The W* methodology defines a W* length as measured from the bottom of the tube explosive expansion transition that would permit flaws below that length to remain in service and based on the assurance that adequate strength is available to resist the axial pullout loads experienced within the tubesheet during all plant conditions.

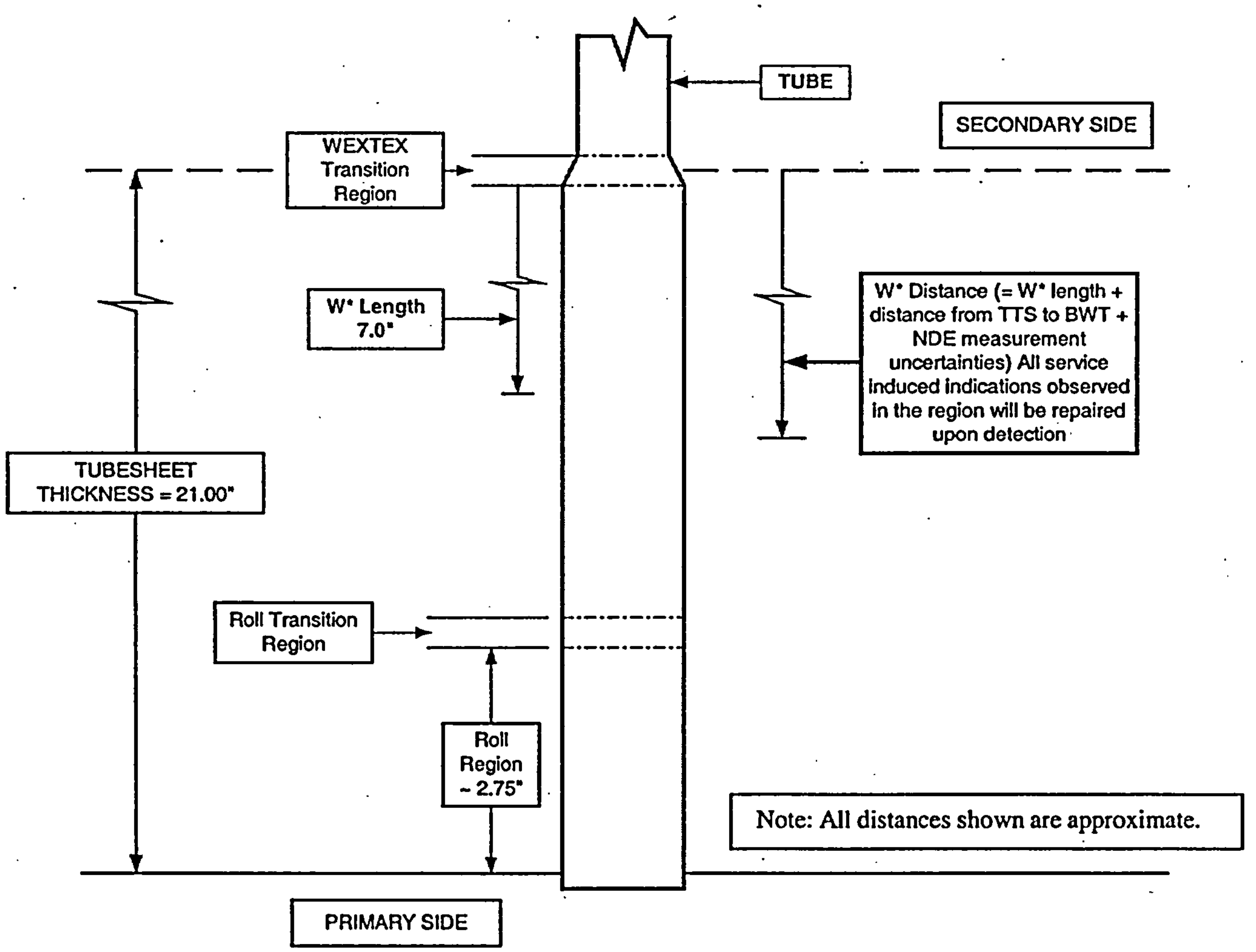
The following definitions apply:

W* length – The maximum length of tubing below the bottom of the WEXTEX transition (BWT) which must be demonstrated to be non-degraded and is defined in WCAP-14797, Rev. 2, Section 4.0 as 7.0 inches below the bottom of the WEXTEX transition on the hot leg side.

BWT – The bottom of the WEXTEX Transition as is defined in WCAP-14797, Rev. 2, as approximately 0.25 inches from the top of the tubesheet.

W* distance – The distance from the top of the tubesheet to the bottom of the W* length including the distance from the top of the tubesheet to the BWT and measurement uncertainties. Uncertainties are defined in Section 8 of WCAP-14797, Revision 2, as 0.12 inch. Therefore, the W* distance is 7.12 inches on the hot leg side plus the distance to the BWT.

Sketch of W* Distance in BVPS Unit 1 Steam Generator Tube Sheet



The W* methodology provides the basis for tubes with any form of degradation below the W* length to remain in service. The presence of the surrounding tubesheet prevents tube rupture and provides resistance against axial pullout loads during normal and accident conditions as discussed in WCAP-14797, Revision 2. In addition, any primary to secondary leakage from tube degradation below the W* length is determined to be acceptably low as discussed in Section 4.0 of this report. Both steam generator tube structural and leakage integrity will be shown to meet the required performance criteria and, thus, the necessary regulatory criteria as defined below.

General design criteria (GDC) 1, 2, 4, 14, 30, 31, and 32 of 10 CFR 50, Appendix A, define requirements for the reactor coolant pressure boundary (RCPB) with respect to structural and leakage integrity.

General design criterion (GDC) 19 of 10 CFR 50, Appendix A, defines requirements for the control room and for the radiation protection of the operators working within it. Accidents involving the leakage or burst of SG tubing comprise a challenge to the habitability of the control room.

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, construction and operation of safety related components. The pertinent requirements of this appendix applies to all activities affecting the safety related functions of these components; these include, in part, inspection, testing, operation and maintenance. Criteria IX, XI and XVI of Appendix B apply to the steam generator tube integrity program defined.

10 CFR 100, Reactor Site Criteria, established reactor-siting criteria, with respect to the risk of public exposure to the release of radioactive fission products. Accidents involving leakage or tube burst of steam generator tubing may constitute a challenge to containment and therefore involve an increased risk of radioactive release. Beaver Valley Unit 1 is licensed for the use of alternate source term in accordance with 10 CFR 50.67 for some design basis accidents.

Demonstrating compliance to the Technical Specification steam generator tube repair limits, normal operating and accident-induced primary-to-secondary leakage limits provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining the reactor coolant pressure boundary.

As discussed in more detail in Section 4.0 of this report, the generic W* methodology contained in WCAP-14797, Rev. 2 is applicable to the Beaver Valley Unit 1 steam generators and it defines the maximum hot leg W* length for pullout resistance as 7.0

inches below the bottom of the WEXTEX transition. FENOC has chosen to use the 7.0 inches length in this request since it is the most conservative of the two listed lengths (5.2 inches for Zone A and 7.0 inches for Zone B) specified in WCAP-14797.

4.0 TECHNICAL ANALYSIS

4.1 Overview

FENOC is proposing to modify the BVPS Unit 1 TSs to revise Surveillance Requirement (SR) 4.4.5.4.a.6, 4.4.5.4.a.8, and 4.4.5.5.d.1 and add SRs 4.4.5.2.e, 4.4.5.4.a.11 and 4.4.5.5.e for one cycle of operation. This proposed revision would be applicable only to Cycle 17. FENOC is currently planning to replace the BVPS Unit 1 SGs during the seventeenth refueling outage and the proposed use of W* will not apply to the replacement SGs expected to commence operation in Cycle 18.

SR 4.4.5.4.a.8 defines steam generator (SG) tube inspection scope. FENOC's proposed change alters the tube inspection scope to exclude the portion of the tube within the tubesheet below the W* distance. This exclusion does not apply to steam generator tubes with sleeves installed within the tubesheet region. SR 4.4.5.4.a.6 provides steam generator tube repair criteria. FENOC's proposed change requires repair of any service induced degradation identified in the W* distance or less than eight inches below the top-of-tube sheet, which ever is greater. The amendment is based on the Westinghouse Electric Company WCAP-14797, Revision 2 entitled, "Generic W* Tube Plugging Criteria For 51 Series Steam Generator Tubesheet Region Wextex Expansions." Since FENOC proposes to repair any service induced degradation within the W* distance, FENOC's proposal is a conservative limited scope application of the complete W* methodology as described in WCAP–14797, Revision 2. The WCAP was developed for Westinghouse fabricated steam generators that utilized the WEXTEX tube expansion process for application of W* methodology. The W* methodology accounts for the reinforcing effect that the tubesheet has on the external surface of the SG tube within the tubesheet region. The W* methodology shows that tube integrity and leakage below the W* distance remain within the existing design limits. The W* criteria were developed for the tubesheet region of 51 Series steam generators considering the most stringent loads associated with plant operation, including transients and postulated accident conditions. The W* criteria were selected to prevent tube burst and axial separation due to axial pullout forces acting on the tube, and to ensure that the steam line break (SLB) leakage limits are not exceeded.

Constraint provided by the tubesheet precludes tube burst for cracks within the tubesheet Thus, the NRC Regulatory Guide (RG) 1.121 criteria are satisfied by the tubesheet constraint. Crack lengths do not need to be limited by burst considerations, and operating leakage limits are not required since, without the potential for tube burst, there is no need for the leak-before-burst leakage limit.

Conceivably, however, a 360 degree throughwall circumferential crack or a significant number of axially oriented cracks could permit severing of the tube and tube pullout from the tubesheet when the tube is subjected to axial forces from primary-to-secondary pressure differentials. The W* criteria were developed to allow certain tubes with indications below the W* distance in the tubesheet region of the tubesheet to remain in service, while precluding tube pullout from the tubesheet under axial loading conditions. A non-degraded W* length is required such that the tube-to-tubesheet contact pressures integrated over the W* length are sufficient to compensate for the axial forces on the tube and thus prevent tube pullout.

Loading analyses were conducted per the requirements of RG 1.121 for both the 3ΔP normal operating load and the limiting faulted condition load (with the applicable safety factor applied). To prevent pullout, these loads must be reacted by the axial restraint afforded by the contact pressure between the tube and tubesheet times the friction coefficient of the tube-to-tubesheet interface acting over some interface length. Contact pressure between the tube and tubesheet is a function of the WEXTEX expansion pressure, and primary-to-secondary pressure and temperature differentials. The W* length is defined to be the length of non-degraded tube that provides assurance that tube pullout criteria are met for the most limiting loading scenario (the 3ΔP operating condition loading). The WCAP-14797, Revision 2, bounding generic methodology parameters used in the calculation of the W* length are conservative for BVPS Unit 1.

The generic W* methodology detailed in WCAP-14797, Revision 2 is applicable to the BVPS SGs and defines the maximum hot leg W* lengths for pullout resistance as 7.0 inches below the bottom of the WEXTEX transition. These distances are increased by an allowance for Non-Destructive Examination (NDE) uncertainties in measuring the W* length. The maximum NDE uncertainty on the W* length in WCAP-14797, Revision 2 is 0.12 inch. The W* methodology provides the basis for tubes with any form of degradation below the W* length to remain in service. This includes a tube with a 360 degree circumferential through wall crack located just below the W* distance, which would still not be pulled out by the worst case main steam line break (SLB) axial loads on the tube. The presence of the surrounding tubesheet prevents tube rupture and provides resistance against axial pullout loads during normal and accident conditions. In addition, any primary-to-secondary leakage from tube degradation below the W* distance contributes less than 5 percent of the total leakage assumed for a postulated SLB accident and may be considered negligible. Consequently, any tube degradation that may go undetected in this area would not affect structural or leakage margins.

To reduce the probability and consequences of SG tube rupture or tube failure, BV performs rotating pancake coil (RPC) probe examinations in critical regions for crack-like indications that would not be easily identified with the bobbin coil probe. These critical regions are based on a degradation assessment where potential and active degradation is expected in SG tubes that could challenge structural and/or leakage integrity if the tubes are not taken out of service by repair.

The critical region of the tubes in the tube-to-tubesheet expansion in Westinghouse Model 51 SGs with WEXTEX explosive expansions is defined as the W* length. The W* length is defined for BVPS Unit 1 in WCAP-14797, Revision 2, considering the most stringent loads associated with plant operation, including transients, and accident conditions. The W* distance is the total rotating pancake coil (RPC) probe inspected length as measured downward from the top-of-tubesheet including the distance to the BWT and includes the NDE uncertainty. The full W* length of tubing must be demonstrated to be non-degraded below the bottom of the WEXTEX transition (BWT). If service-induced degradation is found within the W* distance or less than eight inches below the top-of-tube sheet, which ever is greater, the tube must be repaired or removed from service. Below the W* distance, degradation is acceptable.

BVPS Unit 1 does not use WCAP-14797, Revision 2 to leave degraded tubes in service via the W* methodology. FENOC's proposed change requires that any service induced degradation identified in the W* distance be repaired. The WCAP is only used to define the length of tubing that will be inspected with an RPC probe to remove degraded tubes from service by plugging. FENOC will implement the following W* repair criteria and acceptance criteria:

1. All service-induced indications within the W* distance or less than eight inches below the top-of-tube sheet, which ever is greater, must be repaired.

2. Any type or combination of tube degradation below the W* distance is acceptable. This does not apply to steam generator tubes with tubesheet sleeves installed.

Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet. Thus, structural criterion is satisfied by the tubesheet constraint. However, a 360-degrees circumferential crack or many axially oriented cracks could permit severing of the tube and tube pullout from the tubesheet under the axial forces on the tube from primary to secondary pressure differentials. Section 4 of WCAP-14797, Revision 2 describes the testing that was performed to define the length of non-degraded tubing that is sufficient to compensate for the axial forces on the tube and thus prevent pullout. The operating conditions utilized in WCAP-14797, Revision 2 bound the operating conditions for BVPS Unit 1.

Operating experience has demonstrated negligible normal operating leakage from primary water stress corrosion cracking (PWSCC) even under free span conditions in roll transitions. PWSCC in WEXTEX expansion in the tubesheet region would be even further leakage limited by the tight tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. The steam line break (SLB) conditions provide the most stringent radiological hazards for postulated accidents involving loss of pressure or fluid in the secondary system. WCAP-14797, Revision 2, describes the methodology for calculating leakage for all cracks left in service and the justification to neglect the total SLB leak rate contributed by cracks below the W* distance. Therefore, RPC probe inspection in the area below the W* distance is not necessary to preclude normal operating or accident induced leakage.

Notwithstanding the prior discussion which states that there will be negligible leakage from any type of degradation below the W* distance, even under SLB conditions, FENOC proposes to postulate additional leakage below the W* distance to be included in the total BVPS Unit 1 SLB leak rate during SLB conditions as described in Section 4.3 of this report.

BVPS Unit 1 license amendment request (LAR) 322 submitted by FENOC on January 27, 2004 also affected BVPS Unit 1 Technical Specification 3/4.4.5. The proposed changes from LAR 322 are also shown in Attachment A. The only change shown in this LAR (328) associated with LAR 322 is that the LAR 322 footnote previously labeled with "*" is now being changed to a number to avoid potential confusion with the star listed in proposed changes for LAR 328 which use the W* terminology. Otherwise the "*" used in the footnote in LAR 322 could not be distinguished from the "*" used in the W* term. Beyond this footnote format label change, the changes requested by LAR 328 are fully compatible with the changes pending from LAR 322.

4.2 Applicability of WCAP-14797, Revision 2, to Beaver Valley Power Station Unit 1

4.2.1 Background Information

W* is the length of sound engagement of the tube within the tubesheet such that the force resisting expulsion from the tubesheet balances the force applied to the end of a presumed severed tube. The structural performance criteria are that tube burst will not occur with a margin of 3 during normal operation and 1.4 during the most severe faulted event, a postulated steam line break (SLB) for Beaver Valley Unit 1.

Applied Load

The applied force comes from the internal pressure in the tube. At the U-bend there is a component of the primary-to-secondary differential pressure acting in the axial direction. For the development of the criteria it is also assumed that the tube is severed in the tubesheet so that the differential pressure acts on the entire cross section area of the tube as calculated using the outside diameter (OD) of the tube. The applied force, F_A, is determined from the applied pressure (ΔP), area (A), and the outside diameter (D_o), as

$$F_A = \Delta PA \ where, \ \ A = \frac{\pi}{4} D_o^2 \qquad (1)$$

Here, ΔP is the difference between the primary pressure (P_P), and secondary pressure (P_S) at the top of the tubesheet, i.e., $P_P - P_S$.

Reaction Load

The reaction load in developing the W* length arises from friction between the tube and the tubesheet within the tubesheet hole. The friction force is the product of the normal force between the tube and the tubesheet and coefficient of friction between the tube and the tubesheet. The normal force arises or is affected by four sources:

1. The residual preload from the expansion process,
2. Differential thermal expansion between the tube and the tubesheet,
3. Resultant pressure in the tube within the tubesheet, and
4. Dilation of the tubesheet holes from bowing of the tubesheet.

The first three items result in a compressive normal force between the OD of the tube and the inside diameter (ID) of the tubesheet hole. The last item results in a reduction of the normal force near the top of the tubesheet and an increase in the normal force near the bottom of the tubesheet. It is noted that a lateral load applied to the center of the tube span above the TTS would tend to result in a slight lateral contraction in the axial direction. An analysis of the geometry of deflection shows that the axial contraction is a small fraction of the lateral deflection and the bounding load that would be developed at the TTS would tend to be tighter in the tubesheet hole. On this basis, the action of lateral flow loads can be neglected from further consideration.

Determination of W*

The calculation performed is to find the length, W*, that makes the following equality true between the resisting force on the left and the applied force on the right,

$$\mu\left(N_X + N_T + N_P + N_D\right) = \Delta P\ A \tag{2}$$

where N_X = The residual normal force from the expansion process,
N_T = The normal force from the differential thermal expansion,
N_P = The normal force due to the differential pressure,
N_D = The normal force resulting from dilation of the tubesheet hole,
μ = The coefficient of friction between the tube and the tubesheet.

The resisting forces are due to the interface pressure between the tube and the tubesheet. The actual force is the product of the interface pressure times the effective area of contact, the circumference of the tube times the length of contact. Conservative uncertainty adjustments are then made to the length of contact to determine W*. The diameter of the tube is constant, so an expression for the force per unit length is used and solved for the length, *L*. This means that each force term must be replaced by force per unit length term.

The solution for W* is then,

$$W = \frac{\Delta P\, A}{\mu\left(F_X + F_T + F_P + F_D\right)} \tag{3}$$

where *W* stands for the W* length and the letter *F* stands for the force per unit length of engagement. Each force per unit length term is then replaced by a corresponding pressure times circumference term, i.e.,

$$W = \frac{\Delta P\, A}{\mu\left(P_X + P_T + P_P + P_D\right)\pi D_o} \tag{4}$$

where D_o is the outside diameter of the expanded tube. Substituting for the cross section area yields,

$$W = \frac{\Delta P\, D_o}{4\mu\left(P_X + P_T + P_P + P_D\right)} \tag{5}$$

for the determination of *W*, sans adjustments for uncertainties in measurement and end effects where the assumption of a severed tube has been made. The following points are to be noted:

1. The applied load term, the numerator is affected by changes in the operation of the plant. The value used for the generic document is the same as that for the plant specific application at Beaver Valley Unit 1, hence the W* length would be expected to be the same as that for the generic application.

2. The residual expansion pressure, P_X, is not affected by changes in the operation of the plant.

3. The thermal expansion term, P_T, is affected by changes in the hot leg temperature, T_{hot}. The hot leg temperature at Beaver Valley Unit 1 is greater than the value used for the generic determination, hence the value of W* for use at Beaver Valley Unit 1 should be less than the generic value.

4. The differential pressure term, P_P, is affected by changes in the primary or secondary pressure. The differential pressure term used for the Beaver Valley Unit 1 analysis is greater than the generic analysis; therefore, the W* length would be less.

5. The dilation term, P_D, is also affected by changes in the primary or secondary pressure. The differential pressure acting across the tube sheet is the same as in the generic report. It is noted that different differential pressure conditions were used for the determination of the different contributing load terms for the analysis of the generic report. This is an acceptable approach since the structure remains elastic when the terms are superimposed and is discussed in the following section.

4.2.2 Application of WCAP-14797-P, Revision 2 to Beaver Valley Power Station Unit 1

The determination of the non-degraded tube length considers the residual preload capability of the tube expansion process, the thermal tightening effects due to thermal expansion coefficient differences between the tube and the tubesheet material, pressure tightening effects, and loss of preload due to tubesheet bow effects. The residual preload inherent in the expansion process is independent of differences between analysis and plant conditions. The generic analysis uses a hot leg temperature of 590°F, whereas the limiting BVPS Unit 1 hot leg operating temperature at 2697 MWt is approximately 610.8°F. Therefore, the generic analysis includes less thermal tightening contribution than the actual condition within the steam generators. The generic analysis uses a secondary side steam pressure of 900 psia for evaluation of pressure tightening effects whereas the minimum steam pressure required for 100% electrical power is 760 psia (at the SG steam outlet nozzle). This results in a smaller primary to secondary pressure differential for the analysis condition compared to the BVPS Unit 1 condition. Therefore, the generic analysis includes less pressure tightening contribution than the actual condition within the steam generators. The generic analyses also uses a steam pressure of 760 psi for evaluation of tubesheet bow effects, the same pressure differential that is part of the Beaver Valley Unit 1 design basis. Assumed normal operating steam pressure also influences the analysis with regard to defining the applied end cap load that acts to push the postulated separated tube out of the tubesheet hole. The generic analysis uses a steam pressure of 760 psia, whereas the minimum steam pressure will be maintained above 760 psi at the SG outlet nozzle. Moreover, the internal steam pressure losses due to moisture separation will result in a slightly higher steam pressure within the steam generator. Therefore, the generic analysis includes greater end cap loading compared to the actual conditions within the Beaver Valley Unit 1 steam generators. This end cap load must be reacted by the net residual contact load. As the end cap load is reduced, the non-degraded tube length is also reduced.

Based on the above, it is expected that the Beaver Valley Unit 1 specific W* should be less than the generic value because of the net effect of the changes. The independent considerations lead to the results similar to those presented in the WCAP-14797, Rev. 2. Table 4.2.1 summaries the comparison of Beaver Valley Unit 1 operating parameters to those considered in the generic report.

Table 4.2.1 Comparison of W* for Beaver Valley Unit 1 SG Tubes Relative to Generic Information

Item	Analysis Term & Description	Beaver Valley Unit 1	Generic WCAP-14797 Rev. 2	Application of the Result
1	ΔP, Applied Pressure	P_B = 760 psia	P_G = 760 psia	Beaver Valley Unit 1 W* ~ Generic W*
2	P_T, Thermal Tightening	T_{hot} = 610.8°F	T_{hot} = 590°F	Beaver Valley Unit 1 W* < Generic W*
3	P_P, Pressure Tightening	P_B = 760 psia	P_G = 900 psia	Beaver Valley Unit 1 W* < Generic W*

4.2.3 Beaver Valley Power Station Unit 1 Conclusions

The WCAP-14797, Revision 2 determination of the W* length of 5.2 inches Zone A and 7.0 inches for Zone B for the application to the Beaver Valley Unit 1 SGs is considered to be valid. The differences in length required for the two zones are based on a variance in the tubesheet bow between peripheral and center regions of the tubesheet. The differences between the Beaver Valley Unit 1 specific and the generic calculation values is the result of the conservative assumptions associated with performing a generic calculation, e.g., extremely low secondary side pressure (which increases the applied load and the dilation of the tubesheet holes), additional pressure considered in the crevice, and the use of a lower bound residual expansion pressure. Nevertheless, the application of W* to the Beaver Valley Unit 1 SG tubes per the WCAP-14797, Revision 2 guidance is considered to be justified.

4.3 Steam Line Break Primary to Secondary Leakage Determination

4.3.1 Evaluation

This evaluation documents the primary water stress corrosion cracking (PWSCC) history of BVPS Unit 1 to establish that the most likely point of initiation is at the top of tubesheet, and that the PWSCC initiation potential below the top of tubesheet is significantly less than the top of tubesheet region. Evaluation of the BVPS Unit 1 specific data indicates that the likelihood that a circumferentially separated tube exists below the current +Point (+Pt) coil inspection distance of 8 inches below the top of tubesheet is essentially zero. Although W* distance is not a fixed distance, the W* distance for the large majority of BVPS steam generator tubes is less than 8 inches.

This evaluation also establishes a conservative leakage allowance to be applied to the number of postulated circumferentially separated tubes in order to estimate steam line break condition leakage.

The SLB conditions provide the most stringent radiological hazards for postulated accidents involving a loss of pressure or fluid in the secondary system. To establish the leak rate criteria for faulted conditions, the SLB leakage rate is used with the feed line break (FLB) pressure differential since the FLB transient provides a more limiting primary-to-secondary pressure differential (2650 psi) than the SLB (2560 psi).

SLB leakage from the tubesheet region is limited by the contact pressure between the tube and tubesheet. The contact pressure results from the WEXTEX expansion process, thermal expansion mismatch between the tube and tubesheet, and from the differential pressure between the primary and secondary side. The contact pressure is also a function of tubesheet deflection that varies radially from the tubesheet center.

4.3.2 Estimate of Number of Indications in Service Below the Current Tubesheet Region Inspection Distance

Axial PWSCC was first reported at Beaver Valley Unit 1 at the 1R10 (1995) inspection. One axial and one circumferential indication were reported in SG B at the expansion transition region. No PWSCC was reported in Steam Generators (SGs) A or C. For the 1995, 1996, and 1997 inspections, the +Pt examination depth below the top of tubesheet was 3 inches. Twelve (12) axial indications were reported at the 1996 and 1997 outages; no circumferential indications were reported. The 1996 inspection was the first application of +Pt inspection technology for the tubesheet region. Note that due to an extended shutdown period, no SG inspections were performed in 1998 or 1999.

At the 1R13 (2000) and all subsequent outages, the +Pt examination depth below the top of tubesheet (TTS) was 8 inches, i.e., the tube length between 3 and 8 inches below TTS was not previously inspected using a +Pt coil. If the assumed indication initiation distribution was constant over the entire tube-in-tubesheet distance, then approximately thirty (30) indications would have been expected to be reported at the 1R13 inspection in the distance between 3 and 8 inches below TTS; instead, only 7 indications (6 axial, 1 circumferential) in 6 affected tubes were reported. Also, three (3) indications in three (3) tubes were reported within 3 inches of TTS.

Figure 4.3.1 presents a cumulative distribution plot of elevations of all PWSCC indications reported at BVPS Unit 1. This plot clearly shows that the PWSCC initiation potential is significantly greater for the top of tubesheet region than at deeper elevations. Recall that for the 1R11 and 1R12 outages, the +Pt inspection distance below TTS was 3 inches while for all subsequent outages, the +Pt inspection distance below TTS was 8 inches. Figure 4.3.1 also presents cumulative distribution data of PWSCC as a function of elevation for the 1R14 and 1R15 outages only. For these outages, the inspection transient effects of the increased +Pt inspection distance would not be included in the elevation distribution, therefore, they represent the most appropriate data for evaluating the future PWSCC initiation distribution as a function of elevation within the tubesheet. The elevation distributions for all data and the 1R14/1R15 data are quite similar, further supporting the argument that PWSCC initiation potential below the transition region is far less than for the transition region. Note that for these plots, indication elevations between 0.00 and 0.99" below the top of tubesheet are found in the 0" bin. The BVPS Unit 1 cumulative elevation distribution was compared against that of another plant with Model 51 SGs. The distributions for both plants are quite similar.

For the 1R14 and 1R15 outages with a +Pt inspection distance of 8 inches below TTS, 34 PWSCC indications were reported; 26 were reported within 4 inches of the top of tubesheet. For the 1R10 through 1R13 outages, 36 indications were reported; 29 were reported within 4 inches of the top of tubesheet. Thus for the BVPS Unit 1 PWSCC indications, approximately 79% were reported within 4 inches of TTS, with 21% reported between 4 and 8 inches below TTS. Thus, the number of indications between 8 inches below TTS and 12 inches below TTS would not be expected to be greater than 25% of the expected number of indications for the range between TTS and 8 inches below TTS, assuming the number of indications from 8 to 12 inches below TTS is consistent with the number of indications from 4 to 8 inches below TTS.

To date, 70 total indications have been reported, only 4 of which were circumferentially oriented. Figure 4.3.2 plots the individual indication totals by outage. A regression analysis using all data suggests that approximately 32 indications are expected for the 1R16 outage. Using only the last three outages' data, 25 indications are anticipated. [Should the total number of indications identified in 1R16 exceed the projected number of 32, the actual number of identified 1R16 indications will be used for this analysis of determining a conservative number of unidentified indications from 8 to 12 inches below TTS. However, a minimum of 32 will be used for this analysis if less than 32 indications are identified in 1R16.] Therefore, the cumulative BVPS Unit 1 total number of indications including the 1R16 projection is a maximum of 102 (70 + 32). An additional 25 indications (25% of combined historical indications plus 1R16 new indications) might be observed if the inspection distance were increased from 8 to 12 inches below TTS. A very small number of these 25 postulated unidentified indications between 8 and 12 inches below TTS would be expected to;

1. be all circumferentially oriented
2. represent a 100% TW condition, and
3. extend for 360° (circumferential cracks).

At the 1R13 inspection, which was the first inspection to 8 inches below TTS, only one circumferential indication at -2.76" was reported. For the axial indications, only one represented a 100%TW degradation potential based on flaw amplitude. For the circumferential indications reported below the expansion transition, the largest reported arc length was 38°. Therefore, for 25 postulated indications between 8 and 12 inches below TTS, 2 (4/70 x 25) would be expected to be circumferentially oriented, and the number of indications with 100%TW penetration over 360° is zero. The four circumferential indications reported to date were located at 0.33, 2.27, 2.76, and 7.66 inches below TTS. The total number of indications in the tubesheet for all SGs is 23 in SG A, 18 in SG B, and 29 in SG C.

Figure 4.3.3 presents a tubesheet map plot of all Unit 1 PWSCC indications reported to date. A slightly greater number of indications are reported for Zone A, the outboard region, which is the zone with the lesser amount of tubesheet deflection during operating or faulted conditions.

The above data was developed using the nominal inspection distance below TTS of 3 or 8 inches, based on the outage chronology. In actuality, the inspection distance applied to each tube exceeds the specified value. This is purposely done during the data collection process to ensure that the appropriate distance is examined. All SGC hot leg top of tubesheet +Pt inspection calibration groups were examined for the 1R15 outage. For each calibration group, the lowest scan line of analyzable data was measured for the first, middle, and last tube for that calibration group to define an average inspection distance for each calibration group. This evaluation shows that the calibration group minimum average distance inspected was 8.50 inches, with a mean inspection distance of 10.47 inches below TTS. This explains why the current inspection data contains flaw reports as low as 10.55 inches below TTS. Figure 4.3.4 presents a plot of the binned PWSCC elevation data for all historical indications with the best fit regression. This plot shows that the predicted number of indications in each bin is only one (1) for the 1 inch incremental bins greater than 11 inches below TTS.

Approximately half of the total reported indications occur within 1 inch of the top of tubesheet. A regression of all indication elevations and number of indications within a 1" elevation bin produces a best fit curve that predicts approximately 2 indications in the 8 to 9" below TTS bin, and approximately 1 indication within the 12" below TTS and lower bins. A regression was fit using all data with the exception of the data in the TTS to 1" below bin to determine if the initiation trend follows a similar pattern as for all data. The regression for this subset of data is essentially equal to the regression for all data. Thus, the judgment that the flaw initiation potential decreases with increasing depth below TTS remains valid for this data. Figure 4.3.4A presents the plot of all data and below expansion transition data as a function of depth below TTS. As seen from this figure, the two regressions are essentially identical.

Previous evaluation of residual stress distribution in hydraulically expanded tube-in-tubesheet joints indicates that the residual stresses below the expansion transition are likely compressive in nature. Therefore, PWSCC initiation is likely associated with a localized geometry discontinuity resultant from the tube drilling process. As all tubesheet holes were drilled from the primary face, the frequency of tube hole abnormalities would be expected to be increased as the secondary side face of the tubesheet is approached. The data of Figure 4.3.1 supports this argument. Furthermore the circumferential extent of these abnormalities would be expected to be limited. The inspection data that shows the largest circumferential arc extent is 38° for indications below TTS also supports this assumption.

Although it is likely that the 25 postulated unidentified indications between 8 and 12 inches below TTS would be spread over all three steam generators, it will be conservatively assumed for this analysis that the 25 postulated unidentified indications between 8 and 12 inches below TTS are all grouped in one steam generator.

4.3.3 Leakage Potential Evaluation of Postulated Circumferential Degradation Below W*

NRC has recently questioned the validity of the original assumption presented by WCAP-14797 Rev 2 that postulated circumferential degradation below W* would not produce leakage at SLB conditions. This section attempts to establish a basis supporting this position. The accuracy of this statement can also be qualitatively assessed by examining operating plant history. SG leakage events have been attributed to ODSCC in the freespan and at Tube Support Plate (TSP) intersections, PWSCC at Ubends, at tack roll transitions of non-expanded tubes, and due to loose parts/foreign objects. Therefore, as no operating experience has been associated with postulated circumferential degradation below W*, it is reasonable to assume that the potential for such indications is unlikely, or that the inherent leakage resistance of at least 7 inches of sound WEXTEX expansion (based on a nominal 8 inch below TTS inspection distance when the expansion transition distance is ignored) at TTS is sufficient to preclude reportable leakage (i.e., <2 gpd or <1.4 x 10^{-4} gallons per minute {gpm}).

Two sets of leakage data support the W* criteria. The first set of data was prepared by tack rolling a 7/8 inches OD x 0.050" wall thickness Alloy 600 tube into a carbon steel collar. The tube was then seal welded to the collar and then WEXTEX expanded. Ten (10) 0.125 inch diameter holes were drilled through the collar and tube at elevations referenced from the top of the test collar. The second set of leakage data examined the impact of contact pressure upon crack opening potential for axial flaws. This data may not be directly applicable to postulated circumferential degradation but it does show that at about 2500 psi contact pressure, that leak rates are essentially zero (< 1.33 x 10^{-5} gpm). These tests located the upper crack tip immediately below a channel machined in the tubesheet collar to eliminate any potential for flow restriction. The prevention of leakage at this contact pressure is expected to be independent of flaw orientation.

For the first set of leakage data, if it is assumed that the entire circumference of the drilled holes contributes to the leakage characteristic, the effective leakage flow length was 3.93 inches, which is substantially greater than the leakage flow length of a postulated circumferentially separated tube of 2.79 inches based on a tubesheet hole ID of 0.890 inches. The first set of holes was located at 3" from the top of the test collar thus resulting in a tube to collar contact length of <3 inch due to the transition geometry. The through holes in the carbon steel collar were drilled to a diameter slightly larger than 0.125 inches to permit staking of the tube holes. This staking operation inwardly deformed the tube at the tube/collar interface, in effect pulling the tube from the collar at the hole to ensure that the drilling operation did not affect the leak path. For this set of tests, the leak rate at 600°F was essentially zero at SLB pressure differential. For these samples tested at 600°F and a pressure differential of 1620 psi, leak rates were essentially identical to the 2650 psid SLB tests suggesting that the pressure expansion contribution to the total leakage resistance was not significant. A choked flow condition may have been present. After completion of the 3 inch tests, a new set of holes was drilled at 2 inches from the top of the test collar. Leak rates were elevated above the 3 inch test levels. This process was repeated again locating the holes at 1.25 inches below the top of the collar and leak rates were again elevated above the previous test levels. The tube holes were not plugged after completion of testing at the 3 inches and 2 inches nominal crevice tests. Thus, the 1.25 inches and 2 inches nominal tests included leakage effects of the holes below these elevations. In effect, the 1.25 inches nominal crevice test simulated a tube with a circumferentially separated condition at 6, 8, and 8.5

inches below TTS (based on contact pressures presented in Figure 4.3.7). All of these tests were conducted at a temperature of 600°F.

Prior to elevated temperature testing, room temperature testing was conducted at 1620 psid for each of the crevice depth conditions. For the room temperature, 3 inch tests at 1620 psid, the average leak rate was ~100 times the 600°F average leak rate. Figure 4.3.5 presents the 600°F drilled hole leak rate test data as a function of crevice depth for 2650 psid. Figure 4.3.6 presents the 600°F drilled hole leak rate test data as a function of crevice depth for the 2650 psid and 1620 psid test conditions. For the 3" nominal crevice tests at 2650 psid, the actual crevice depths were 2.37 inches, 2.29 inches, 2.37 inches, and 2.1 inches, for an average crevice depth of 2.28 inches. For the 2 inch nominal crevice tests, the average crevice length was 1.28 inches, and for the 1.25 inches nominal crevice tests, the average crevice length was 0.61 inches.

For the 3 inch tests at 600°F, 2 of the 4 samples had no leakage at 1620 or 2650 psid. The leak rates of the other two samples at 2650 psid were 1.33 x 10^{-6} gpm and 1.33 x 10^{-5} gpm. For these tests, radial contact pressure due to thermal expansion, pressure expansion, and WEXTEX expansion are inherent. There was no allowance for tubesheet hole dilation. Thus, these tests can be seen as a representation of the point where tubesheet hole dilation effects are neutral, i.e., at the neutral axis, or between 11 and 12" below TTS.

Figure 4.3.7 presents a plot of contact pressure as a function of distance below TTS for 4200 seconds into the SLB event. At this point the primary and secondary side temperatures are assumed to be constant, and tubesheet bow effects are maximized. The contact pressures in the drilled hole specimens are similar to the predicted contact pressures in the tube for depths of 6 to 8.5 inches below TTS.

WCAP-14797 Rev 2 calculates the total positive radial contact pressure as a function of depth below top of tubesheet due to thermal expansion, pressure expansion, and WEXTEX expansion. The reduction in contact pressure due to tubesheet bow is included to develop resultant radial contact pressure as a function of depth below the top of tubesheet. Figure 4.3.7 presents a plot of resultant radial contact pressure as a function of depth below top of tubesheet for the Zone A and B regions. If the Zone B data is conservatively applied to the entire tubesheet, at approximately 9.1 inches below TTS, the resultant radial contact pressure is 2500 psi. The Figure 4.3.7 data is based on plant response at 4200 seconds into the SLB event.

WCAP-14797 presents a summary of contact pressures for various conditions. Using the WCAP-14797 information for normal operating conditions, the Zone B contact pressure at 8" below TTS is approximately 1406 psi, with WEXTEX contact pressure included. However, this calculation assumes a hot leg temperature of 590°F and also assumes that the secondary side pressure (assumed 900 psi) acts on the tube OD over the entire tube length in the tubesheet. For the BVPS condition, additional contact pressure will be provided due to the actual hot leg temperature of 608°F. The drilled hole leakage test data suggests that the pressure drop through the actual crevice length of 2.25" is about 2500 psi based on the observation that 2 of the 4 samples had no leakage during the leak test with a 2560 psi pressure differential. Thus, the assumption that the secondary pressure acts on the tube OD in the crevice is quite conservative. If secondary side pressure does not act on the tube OD in the crevice at deep depths, which is a reasonable assumption, and the actual hot leg operating temperature and pressure differential for BVPS is applied, the normal operating contact pressure at the tubesheet neutral axis is approximately 2403 psi, which bounds the contact pressure of the 3" nominal samples. The SLB reference case for contact pressure evaluation used a hot leg temperature of 460°F and pressure differential of 2560 psi. The neutral axis contact pressure for the SLB case at 4200 seconds is approximately 2903 psi, which also bounds the contact pressure of the 3" nominal samples.

Contact pressures at the neutral axis were also evaluated for the Case (3) condition of Table 3.2-1 of WCAP-14797. This case considers plant conditions at 600 seconds into the SLB event. For this case the hot leg temperature is 433°F, the secondary side temperature is 212°F, and primary to secondary pressure differential is 2344 psi. The contact pressure at the neutral axis is 2683 psi, which still bounds the contact pressure of the 3" nominal leakage samples by approximately 600 psi. These calculations were performed using the same model used to develop the contact pressures contained in WCAP-14797, only the input values were changed to account for the different temperatures and pressures. These calculations were performed for a depth below TTS of 10.5". The bounding leakage methodology presented to the NRC on 6/8/2004 compares contact pressures at 12" below TTS against the 3" nominal leakage samples. The actual tube contact pressures at 12" below TTS will be several hundred psi greater than the values listed above. Therefore, the bounding leakage methodology is judged to be conservative for the SLB event. While the leakage tests were conducted at 600°F and the 4200 second SLB hot leg temperature is predicted to be 460°F, the results of the leakage test are still considered applicable as flashing of the primary water in the crevice is expected to occur at both 600°F and 460°F, thereby resulting in a choked flow condition. The contact pressure model calculates contact pressures at the neutral axis (10.5" below TTS), 6, 4, and 2" below TTS as well as at TTS. For the 6" below TTS elevation for hot leg temperature of 433°F, secondary side temperature of 212°F, and 2344 psi differential pressure, the contact pressure is 1803 psi. The bounding leakge model considers a separate leakage allowance for postulated separated tubes at 8 to 12" below TTS as well as >12" below TTS. Therefore, at 8" below TTS, the contact pressure is expected to be approximately 2194 psi, which is well above the contact pressure of the 1.25" nominal leakage samples of approximately 1352 psi. Therefore, the bounding leakage methodology is judged to be conservative for the SLB event for both the 8 to 12" below TTS elevation and the >12" below TTS elevation.

Figure 4.3.7 shows that the drilled hole test contact pressures are substantially higher for the actual crevice lengths tested compared to the resultant plot of contact pressure for a tube at the same crevice depth condition. The 1.25 inches nominal crevice tests involve a contact pressure consistent with the Zone B tube at approximately 6 inches below TTS and the 3 inch nominal crevice tests involve a contact pressure consistent with the Zone B tube at approximately 8.5 inches below TTS. The rate of change of contact pressure as function of distance below TTS is more rapid for the drilled hole specimens than for the actual tube. For the 1.25" nominal crevice test case, an actual crevice length of ~0.6 inch with positive contact pressure was provided. The corresponding tube would have a crevice of ~5.3 inches with positive resultant contact pressure for ~3.5 inches above a postulated separation. Thus, it could be argued that the drilled hole specimen data is conservative compared to the actual tube in the range of 6 to 8.5 inches below TTS due to the more rapid change in contact pressures as a function of distance below TTS and shorter crevice length with positive contact pressure. However, since the contact pressure reduction due to hole dilation was not included, a conservative estimate of potential leak rate for a postulated circumferentially separated tube below the W* inspection distance of 8 inches below TTS would be to use the predicted leak rates at some minimal crevice depth, such as 0.50 inch. This selected crevice depth bounds the 1.25 inch nominal crevice test condition. At this crevice depth, the predicted leak rate is 0.0004 gpm using a mean regression of the leak rate as a function of actual crevice depth and 0.0045 gpm at the upper 90% prediction bound. See Figure 4.3.5. As Figure 4.3.5 plots the leak rate data on a logarithmic scale, the zero leakage data points were assigned a leak rate of 1 x 10^{-6} gpm. Thus, the SLB condition conservative leak rate applied to a circumferentially separated tube between 8.0 and 12 inches below TTS is 0.0045 gpm. For a postulated circumferentially separated tube at >12 inches below TTS, substantial contact pressure and crevice lengths would exist above this location, resulting in a no leakage condition.

Thus, for Cycle 17, the steam line break evaluation will total up the number of indications which happen to be identified below the W* distance and less than 12 inch range below TTS and left in service along with the number of conservatively assumed unidentified indications within the 8.0-12 inch range below TTS (a minimum of 25 indications as described in Section 4.3.2 of this request) and multiply this total by 0.0045 gpm as identified in the above paragraph. As an example, an assumption of no identified indications left in service within the W*-12 inch below TTS range combined with the minimum of 25 unidentified indications within the 8.0-12 inch below TTS range results in an additional leakage of 0.11 gpm (25 total indications x 0.0045 gpm per indication in the 8-12" range). This applied bounding leak rate between 8 and 12 inches below TTS is extremely conservative because actual leakage would likely be zero since any degradation would likely be axial non-through-wall and not circumferential through-wall. Even if it is assumed that all 25 postulated unidentified indications between 8 and 12 inches below TTS are circumferentially oriented and separated, the leakage associated with these indications would be approximately 0.11 gpm, and would be dispersed over 3 SGs. Evaluation of the actual inspection distances indicates an average inspection distance of 10.47 inches for the 1R15 outage. At 10.47 inches the Zone B contact pressure is approximately 3000 psi. All postulated unidentified indications will be conservatively assumed to be in one steam generator. The highest number of identified indications left in service between 8 and 12 inches in any one steam generator will be included in the 8-12 inch below TTS leakage term.

Note that due to the substantial contact pressure at greater than 12 inches below the top of tubesheet, no leakage is expected for indications greater than 12 inches below the top of tubesheet. At 12 inches below the top of tubesheet, the expected Zone B contact pressure at 4200 seconds into the SLB event is 3500 psi. A conservative estimate of leakage for indications at greater than 12 inches below TTS can be accomplished by applying the upper 90% prediction leak rate at 2650 psid for the 3 inch nominal crevice data of 9×10^{-5} gpm. Note that the contact pressure for the 3 inch nominal crevice samples is approximately equal to a Zone B at 8.5 inches below TTS. If all remaining active tubes in the least plugged SG are assumed to contain a circumferential separation at 12 inches below TTS, the SLB leakage contribution would be approximately 0.261 gpm (2900 tubes times 0.00009 gpm per tube). Note that the average contact pressure for these samples was 2273 psi, or about 1200 psi less than for the 12 inch depth, that 2 of the 4 samples had no leakage at 2650 psid, and that the constrained crack leak test data suggests that at a contact pressure of greater than 2500 psi, no leakage is expected, regardless of the indication orientation. Zone A contact pressures as a function of depth below the top of the tubesheet are higher than for Zone B, but not significantly. Zone A tubes retain positive resultant contact pressure for the entire crevice length. The added length with positive contact pressure is expected to represent a substantially increased resistance to leakage during a postulated event.

Thus, for 1R16, the steam line break evaluation will conservatively assume each tube remaining in service contains a circumferential separation at 12 inches below TTS with an assumed leakage of 0.00009 gpm per tube, even though actual leakage would likely be zero since any degradation would likely be axial non-through-wall and not circumferential through-wall. This applied bounding leak rate for > 12 inches below TTS is extremely conservative. The bounding leak rate will be determined by the steam generator with the maximum amount of unplugged tubes remaining in service following 1R16. Therefore, the assumed leakage for greater than 12 inches below TTS will be equal to 0.00009 gpm times the maximum number of tubes left in service in any one BVPS steam generator.

4.3.4 Evaluation of Axial PWSCC within W* Distance

SLB Conditions Leakage Potential from Axial PWSCC within W*

At the 1R15 outage, 18 axial PWSCC indications on 18 tubes were reported. These indications ranged in elevation from 0.18" below TTS to 9.79" below TTS. The breakdown of indications per SG was; SGA, 3, SGB, 5, SGC, 10. Ten (10) of these 18 indications were noted within 1" of TTS, and are assumed to be located within the expansion transition. The largest amplitude signal of 1.89 volts by +Pt was noted on tube R20 C36, SGA. Based on a +Pt amplitude versus depth correlation developed by Westinghouse, the maximum depth of this indication is estimated at 71%TW. The phase based depth analysis indicated a maximum depth of 97%TW. This flaw was located at 0.43" below TTS, which locates it within the expansion transition. The flaw length from profiling was 0.31". This indication was in situ pressure tested to 2841 psi with no leakage and no burst reported at 4900 psi. FENOC has conservatively performed in situ pressure testing at previous outages of other axial PWSCC indications located within the W* distance. At the 1R14 (2001) outage, a 1.98 volt axial PWSCC indication located at 1.86" below TTS was in situ pressure tested with no leakage or burst reported. At Diablo Canyon, axial PWSCC indications with amplitudes up to 5.6 volts by +Pt have been in situ pressure tested with no leakage or burst reported. At 5.6 volts, this indication is judged to have contained a 100%TW penetration. No less than 5 of the in situ pressure tested indications at Diablo Canyon are judged to have contained a 100%TW penetration. None leaked during in situ pressure test.

Other industry in situ pressure test data supports the application of the voltage based sizing technique. During the Fall 2003 inspection at a plant with C-E SGs, a 1.9 volt by +Pt, 106° arc length circumferential PWSCC indication was reported at the top of tubesheet. The phase based depth assessment indicated that the maximum depth of 99%TW extended for nearly the entire flawed length. This indication was in situ pressure tested for leakage only. No leakage was reported at 2900 psi. It should be noted that the depth estimate using the amplitude correlation is approximately 73%.

A history review of each of the BVPS 1R15 indications was performed using the 1R14 +Pt inspection data. Thirteen (13) of the 18, 1R15 flaws had a precursor signal in the 1R14 data. Of the five 1R15 indications with no precursor signal in 1R14, only one had an amplitude >1 volt. This indication was found at 0.49" below TTS, placing it within the expansion transition. The only indication with a modest amplitude growth was R20 C36, which had an amplitude growth of 1.38 volts. As this indication was located within the expansion transition, the higher residual stresses of the transition would be expected to exacerbate growth. R20 C36 also had the largest length growth of 0.14", and also had the only substantial depth growth (by phase) of 59%. Cycle 15 growth data is provided in Table 4.3.4-1. The Cycle 14 growth statistics are essentially equal to the Cycle 15 growth statistics. The only appreciable difference between the growth data for Cycles 15 and 14 is maximum depth growth. The expansion transition flaws appear to have a slightly higher maximum depth growth. The average +Pt voltage growth of 0.31 volts is modest for PWSCC as PWSCC amplitudes are greater than ODSCC for equal depths. The average length growth of 0.01" suggests that the residual stress fields that initiate the PWSCC are limited in length and do not represent a potential for indications with significant lengths.

The average +Pt voltage of the 1R14 precursor signals was 0.52 volts, suggesting that the detection threshold for axial PWSCC is slightly above this value. With a maximum +Pt voltage growth for Cycle 15 of 1.38, the maximum EOC16 PWSCC amplitude expected is approximately 2 volt. Based on the correlation of +Pt voltage to maximum depth, this postulated indication would be expected to have a maximum depth of about 74%TW. Therefore, axial PWSCC within the W* distance that either is initiated during the cycle or remains in service based on probability of detection is judged to result in an indication that is well <100%TW at the end of the next operating cycle, and will not contribute to primary to secondary leakage during a postulated SLB event.

However, if 1R16 observed axial degradation within the W* distance is judged to contain 100%TW penetration, the end-of-cycle 17 evaluation of postulated SLB condition leakage will account for these indications in the calculation of SLB leakage within the zero to eight inch region below TTS. This process of addressing additional potential SLB leakage in Cycle 17 based upon the 1R16 inspections is part of the normal condition monitoring/operational assessment. This assessment will occur regardless of approval of the changes requested in this submittal.

Table 4.3.4-1

	1R15 Flaws		Cycle 15 Average Growth		
	Avg +Pt V	Avg Length	+Pt V	Length	Depth
All PWSCC	0.79	0.20	0.31	0.01	10
Exp. Transition	0.75	0.19	0.32	0.01	15
Below Transition	0.84	0.22	0.31	0.00	6

4.3.5 Conclusions Regarding Leakage Testing

The WEXTEX drilled hole leakage testing indicates the following characteristics for a WEXTEX expanded tube with a postulated circumferential separation below the W* inspection distance.

1. Comparison of the room temperature and elevated temperature tests indicates that elevated temperature leak rates were approximately 100 times less than room temperature leak rates.

2. Comparison of the elevated temperature test data for 1620 and 2650 psi pressure differential shows the leak rates are essential constant for both pressure differential conditions suggesting that pressure expansion has a limited effect on leak rates.

3. Based on the observations of items 1 and 2, contact pressure is seen as the most significant factor for restricting leak rate.

4. For the 1.25" nominal crevice test case, the contact pressure is approximately equal to the contact pressure of an expanded tube at 6" below TTS while for the 3" nominal crevice test case, the contact pressure is approximately equal to the contact pressure of an expanded tubes at 8.5 inches below TTS. The contact pressure reduction in the test samples was more rapid than in the actual tube.

5. The accumulation of the leakage effects of the holes at 2 and 3 inch below TTS can be seen as a representation of the postulated case where the tube is separated at 6, 8, and 8.5 inch below TTS. The leak rate determined for the 1.25 inches nominal crevice tests is likely a conservative estimate of leakage for a tube with a postulated circumferential separation below the W* inspection distance.

6. A 90% prediction bound for the 2650 psi data indicates that at a contact pressure of approximately 1300 psi, which represents the approximate contact pressure at 6 inches below TTS the upper bound leak rate is 0.0069 gpm, similar to the value selected of 0.0045 gpm based on a 0.50 inch crevice depth using the upper 90% prediction bound.

7. Evaluation of the actual below-the-TTS inspection distance for the BVPS 1R15 outage indicates on average the inspection depth was 10.47 inches below the top of the tubesheet. Therefore, the likelihood of any postulated circumferentially separated tube at or below the neutral axis will exhibit a leak rate exceeding 0.0045 gpm is acceptably small.

8. Evaluation of the actual below TTS inspection distance for the BVPS 1R15 outage indicates that on average, the inspection depth was 10.47 inches below TTS. Therefore, the likelihood of any postulated circumferentially separated tube left in service will exhibit a leak rate exceeding 0.0009 gpm during a postulated SLB event is acceptably small. Contact pressure for these samples was an average of 2273 psi. Contact pressure for Zone B tubes at 12 inches below the TTS is expected to exceed 3500 psi.

9. Zone B tubes were used for estimation of SLB condition leak rates. The Zone A tubes retain positive resultant contact pressure over the entire crevice length, and this increased length is expected to represent a significant increase in SLB condition leak rates compared to Zone B.

4.3.6 Summary of Steam Line Break Conclusions

The Steam Line Break leakage for Cycle 17 operation of BVPS Unit 1 is evaluated to be:

$$\text{Postulated SLB Leakage}_{\text{cycle 17}} = \text{ARC}_{\text{GL 95-05}} + \text{Assumed Leakage}_{0''\text{-}8'' < \text{TTS}} + \text{Assumed Leakage}_{8''\text{-}12'' < \text{TTS}} + \text{Assumed Leakage}_{>12'' < \text{TTS}}$$

Where: ARC $_{\text{GL 95-05}}$ is the normal SLB leakage derived from alternate repair criteria methods and the 1R16 steam generator tube inspections. This term would also include any other postulated leakage (e.g., as committed in LAR 1A-322 for Alloy 800 sleeves).

Assumed Leakage $_{0''\text{-}8'' < \text{TTS}}$ is the postulated leakage for undetected indications in steam generator tubes left in service between 0 inches and 8 inches below the top of the tubesheet (See Section 4.3.4, page 26)

Assumed Leakage $_{8''\text{-}12'' < \text{TTS}}$ is the conservatively assumed leakage from the total of identified and postulated unidentified indications in steam generator tubes left in service between 8 and 12 inches below the top of the tubesheet (See Section 4.3.2 and Section 4.3.3, page 23). This is 0.0045 gpm times number of indications (See Section 4.3.2 and Section 4.3.3, page 23). All postulated unidentified indications will be conservatively assumed to be in one steam generator. The highest number of identified indications left in service between 8 and 12 inches below TTS in any one steam generator will be included in this term.

Assumed Leakage $_{>12'' < \text{TTS}}$ is the conservatively assumed leakage for the bounding steam generator tubes left in service below 12 inches below the top of the tubesheet. This is 0.00009 gpm times number of tubes left in service in the least plugged steam generator following 1R16 (See Section 4.3.3, page 24)

Figure 4.3.1

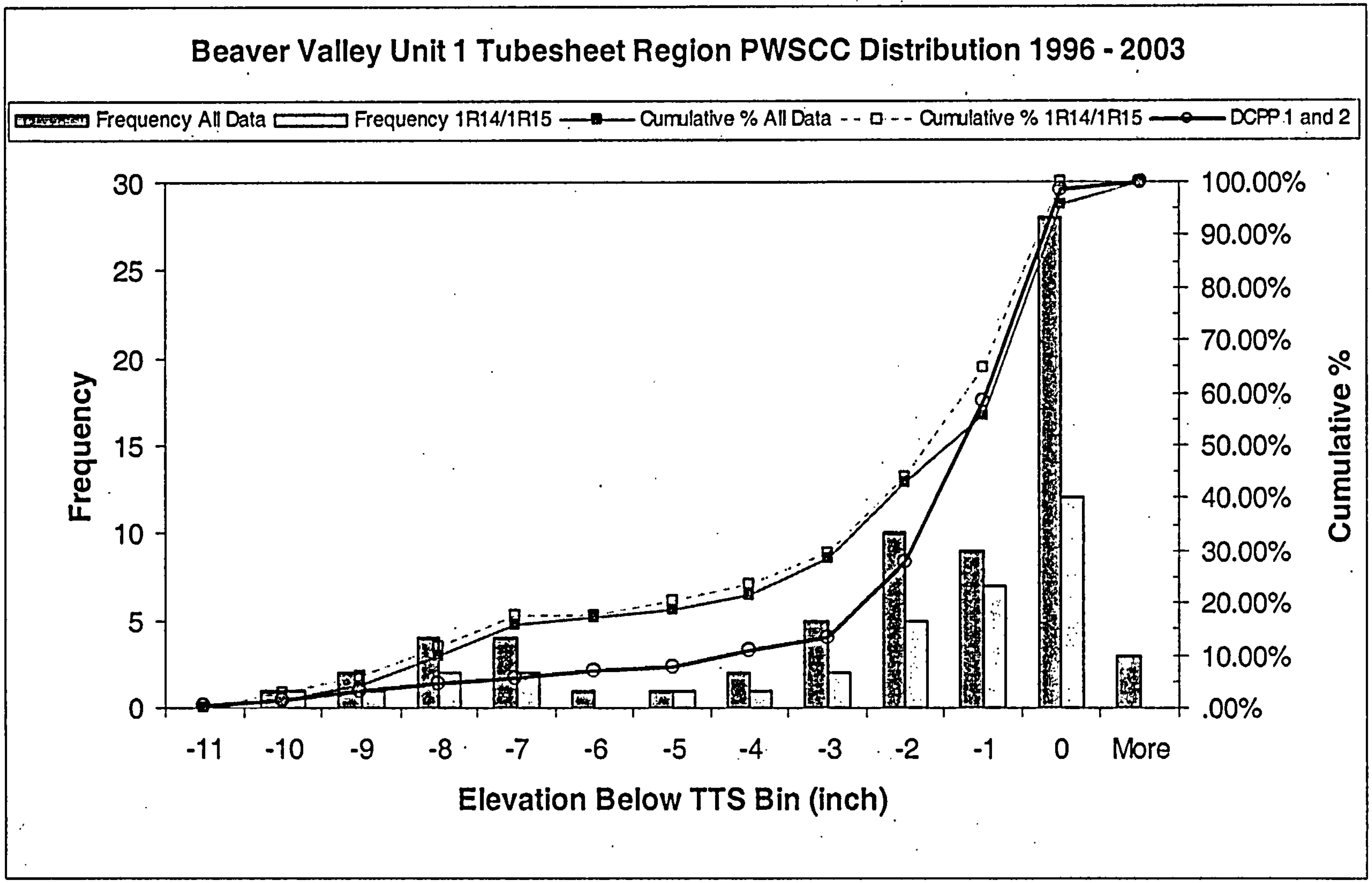

Beaver Valley Unit 1 Tubesheet Region PWSCC Distribution 1996 - 2003
Frequency All Data
Frequency 1R14/1R15
Cumulative % All Data
Cumulative % 1R14/1R15
DCPP 1 and 2
Frequency
Cumulative %
30
25
20
15
10
5
0
100.00%
90.00%
80.00%
70.00%
60.00%
50.00%
40.00%
30.00%
20.00%
10.00%
.00%
-11
-10
-9
-8
-7
-6
-5
-4
-3
-2
-1
0
More
Elevation Below TTS Bin (inch)

Figure 4.3.2

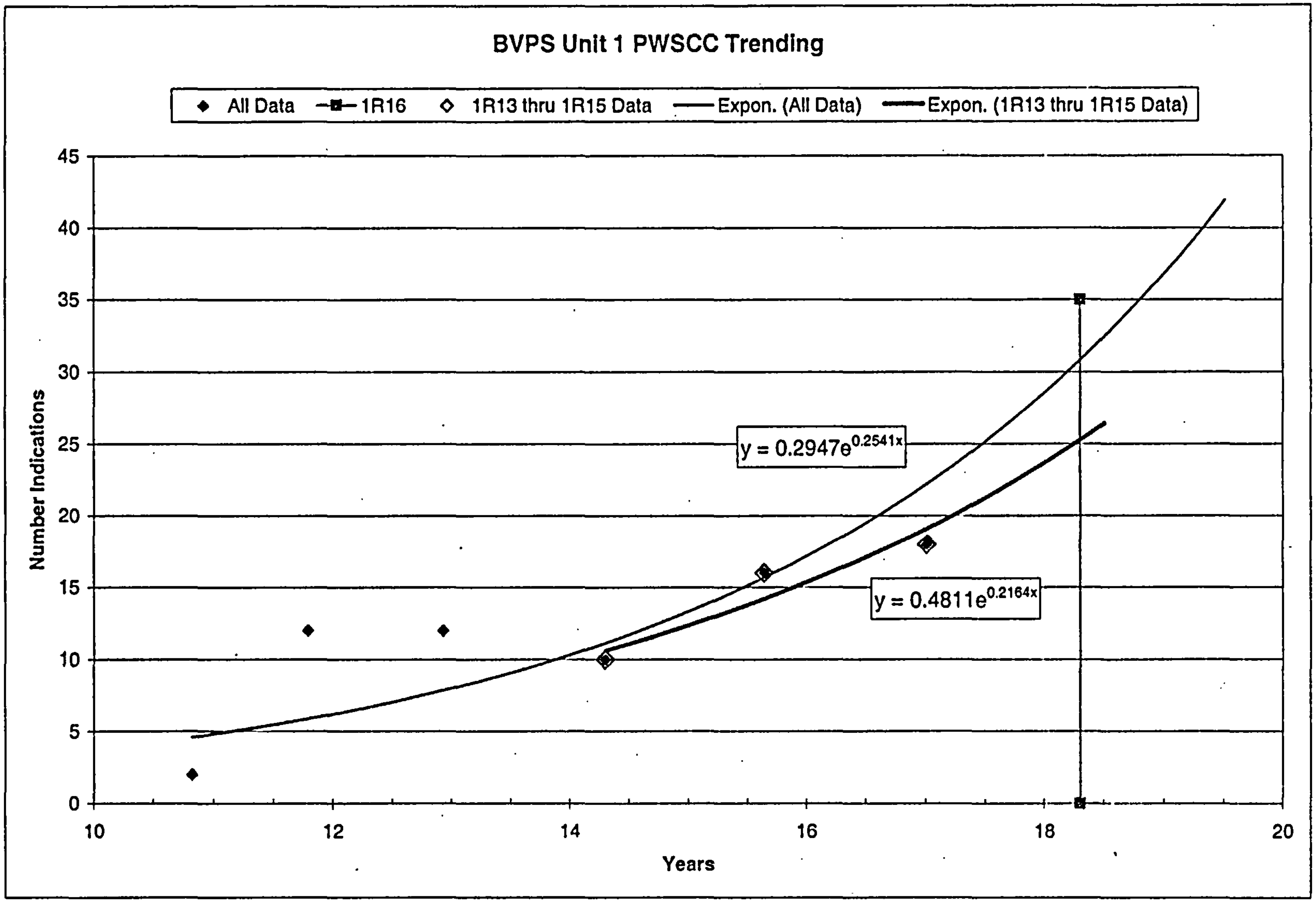

BVPS Unit 1 PWSCC Trending
All Data
1R16
1R13 thru 1R15 Data
Expon. (All Data)
Expon. (1R13 thru 1R15 Data)
y = 0.2947e^0.2541x
y = 0.4811e^0.2164x
Number Indications
Years
0
5
10
15
20
25
30
35
40
45
10
12
14
16
18
20

Figure 4.3.3

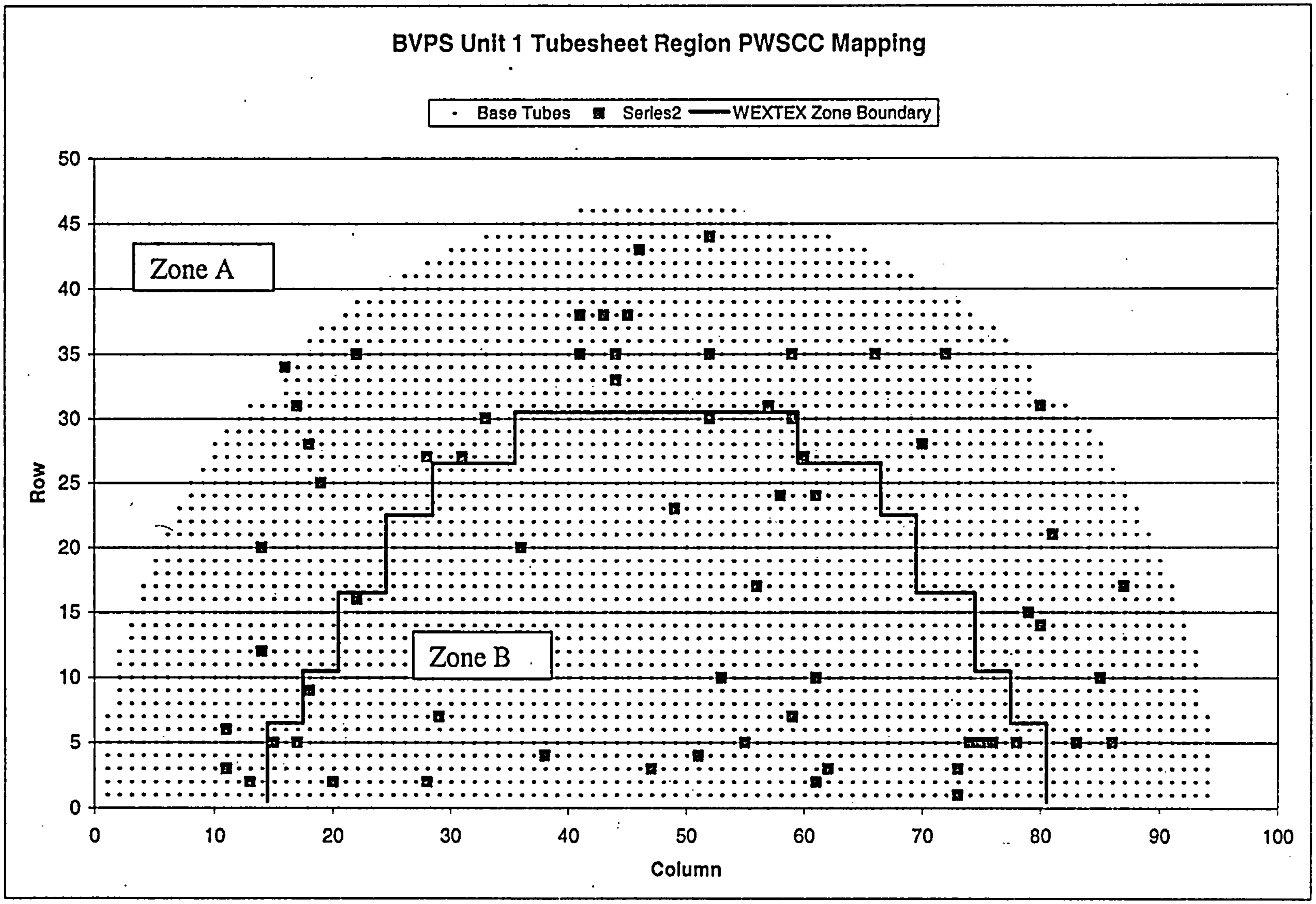

BVPS Unit 1 Tubesheet Region PWSCC Mapping
Base Tubes
Series2
WEXTEX Zone Boundary
Zone A
Zone B
Row
Column
0
5
10
15
20
25
30
35
40
45
50
0
10
20
30
40
50
60
70
80
90
100

Figure 4.3.4

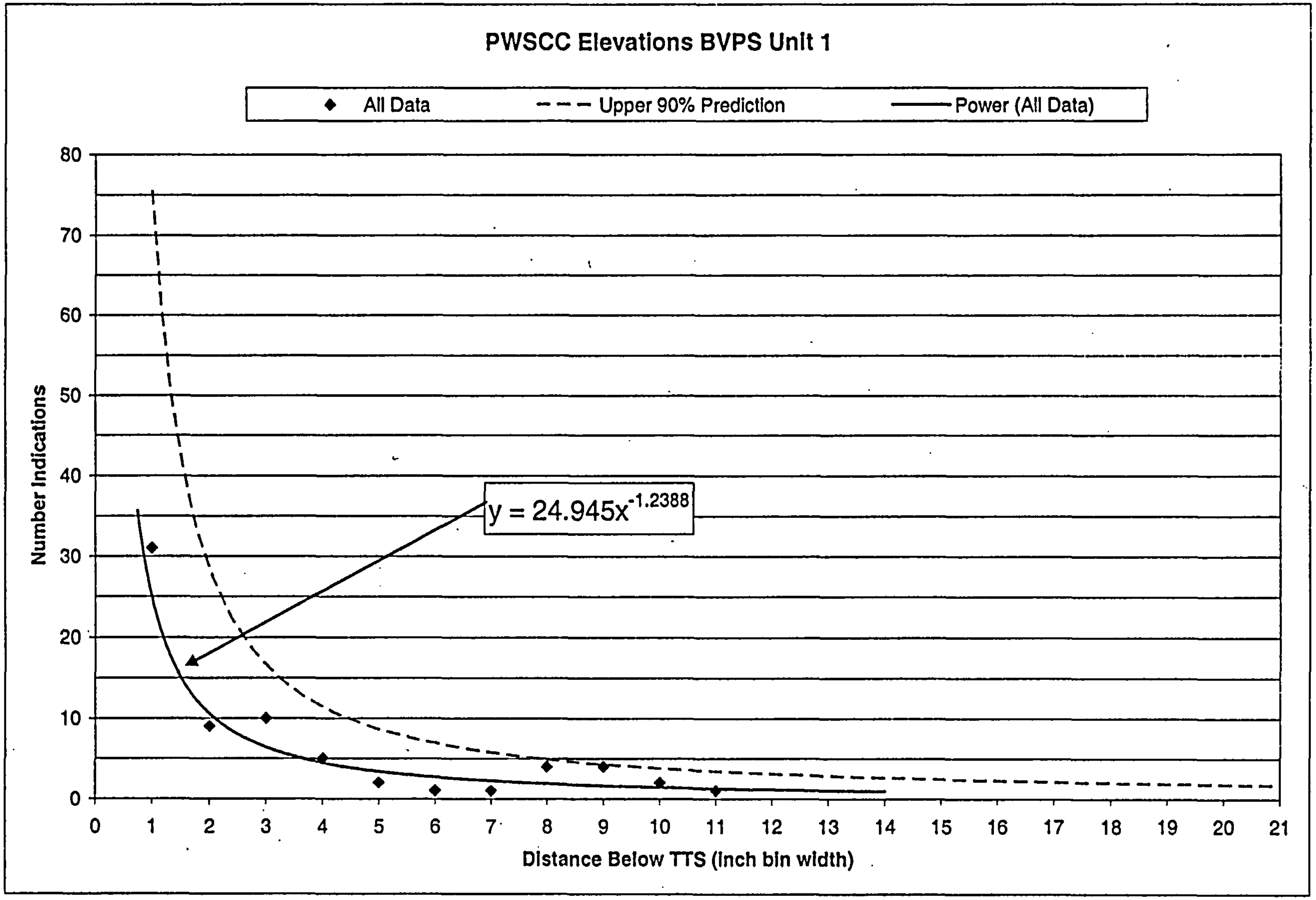
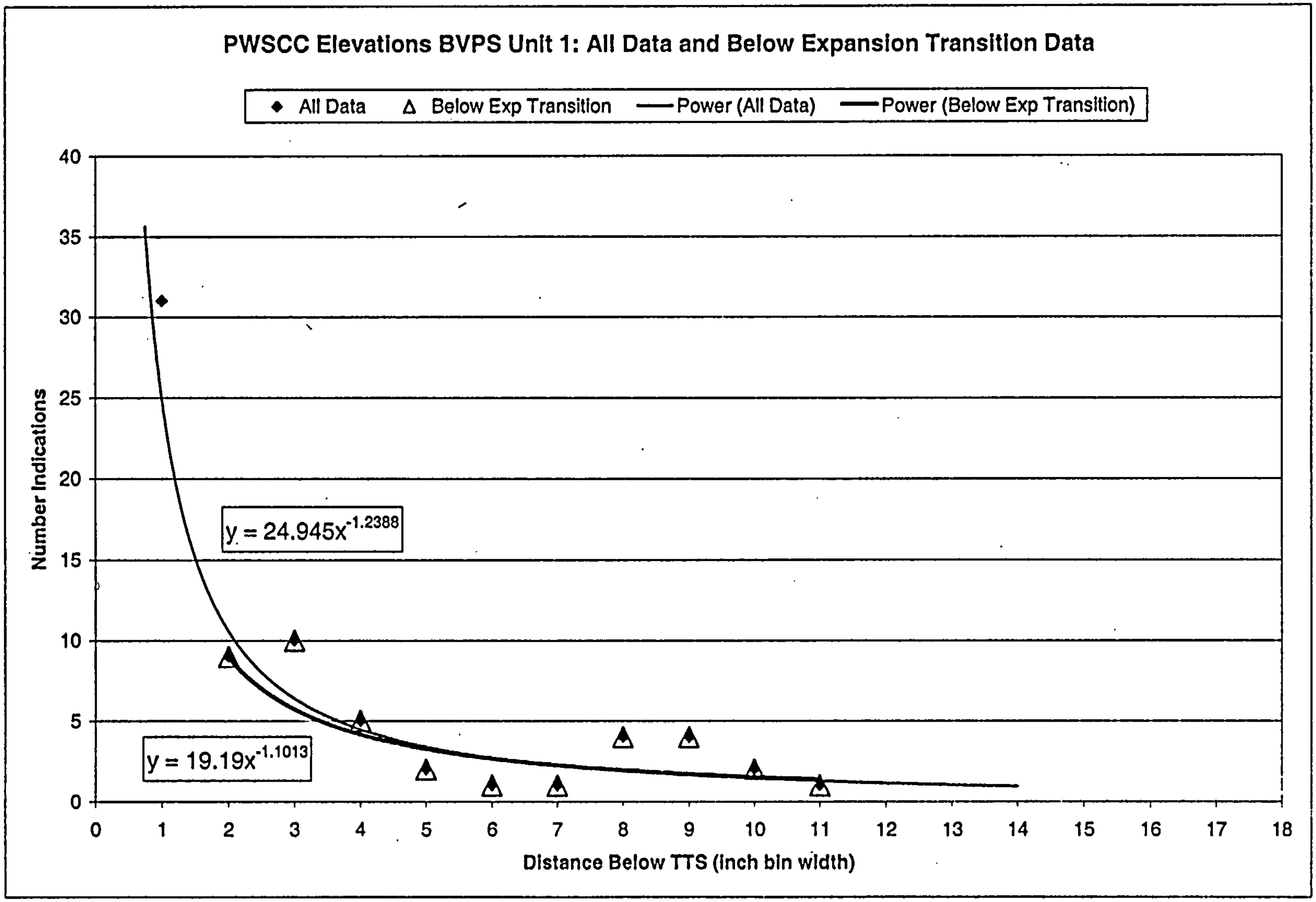

PWSCC Elevations BVPS Unit 1
All Data
Upper 90% Prediction
Power (All Data)
80
70
60
50
40
30
20
10
0
Number Indications
y = 24.945x-1.2388
0 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21
Distance Below TTS (inch bin width)

Figure 4.3.4A


PWSCC Elevations BVPS Unit 1: All Data and Below Expansion Transition Data
All Data
Below Exp Transition
Power (All Data)
Power (Below Exp Transition)
y = 24.945x^-1.2388
y = 19.19x^-1.1013
Number Indications
Distance Below TTS (inch bin width)
0
5
10
15
20
25
30
35
40
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18

Note: The Following Figure are Westinghouse Proprietary and are shown in Attachment F

Figure 4.3.5

Figure 4.3.6

Figure 4.3.7

4.4 Inspection Results Summary

4.4.1 Beaver Valley Power Station Unit 1 Inspection History

Table 4.4-1 presents a summary of the axial and circumferential PWSCC for the last six outages at Beaver Valley Unit 1.

Table 4.4-1
Axial and Circumferential PWSCC at Hot Leg TTS Tracking

	SG A		SG B		SG C	
Inspection	Axial	Circ	Axial	Circ	Axial	Circ
1R10	0	0	1	1	0	0
1R11	5	0	2	0	5	0
1R12	6	0	3	0	3	0
1R13	2	0	1	1	6	0
1R14	7	0	4	0	3	2
1R15	3	0	5	0	10	0
Total	23	0	16	2	27	2

4.4.2 Industry Operating Experience

As noted above, a distribution of indications has been conservatively defined for the WEXTEX Region of the Beaver Valley Unit 1 steam generators. The potential occurrence of indications for Model 51 steam generators will be projected uniformly over the entire length of the joint in the SLB Leakage Model. This is based on the indications being potentially caused by tubesheet hole drilling anomalies during manufacture and the propensity for tube indications at tube hole surface anomalies in a previous Westinghouse laboratory program involving roll expended (RE) joints (Alloy 600 tubes). (It is assumed that potential corrosion of the WEXTEX tube joints would be similar to the RE tube joints determined in the test program. It is also based on the lack of indications in smooth bore, anomaly-free tube holes in hydraulically expanded joints (Alloy 600 tubes) in a laboratory program performed by Westinghouse for EPRI. This is conservative because based on observations at three hydraulic expansion (HE) joint plants of the Plant A tube size, most of the irregularities occurred in the upper reaches of the tubesheet. Most of the signals recorded from one plant in the profilometry file were characterized as bulges. For the most part, the profilometry supports a declining trend of indications from the top of the tubesheet.

4.5 Structural Analysis

4.5.1 Impact of Locked Tubes on W*

Locked tube residual axial loads occur because of packing and/or denting of the tube-to-tube support plate crevice while the plant is operating, i.e., when the tube is strained in the axial direction by primary-to-secondary pressure end cap load. Hence, the residual load in the tube is equal to the normal operating load. When the plant cools, a portion of the residual strain, and the attendant residual load, remains. When the plant returns to power, the load at operating conditions is restored to the same level that was present when the locking occurred, i.e., at normal operating conditions. Since the tube support plate (TSP) is restored to the elevation it occupied at the time of locking, it imparts no additional load on the tube. In other words, as the pressure load increases strain in the tube, the residual downward load of the tube on the TSP decreases. The W* criteria were developed to provide a margin of safety of three relative to the loads corresponding to normal operating conditions.

The second effect of the locked tube loading is the Poisson contraction of the tube in the hoop direction. Again, the axial stress conditions in the tube during operation are unaffected by locking of the tube at the TSP. The tensile test results used to demonstrate that the value used for the coefficient of friction were from conservative test programs. The equations used to calculate the coefficient of friction also did not account for the axial load, hence, the calculated values for the tests are lower than the actual values. Moreover, the configuration of the tensile tests also results in a Poisson contraction of the tube over the full range of the test load. This is conservative relative to demonstrating the factors of safety relatives to the requirements of Regulatory Guide 1.121, e.g., testing to 3 times the required axial force results in a Poisson contraction three times that of the nominal condition.

In summary, implicitly ignoring the potential effects of the tube being locked at the TSPs is conservative to determining the W* value. Even if a loss of engagement is postulated to be possible, and the effect of the locked condition is evaluated, the locked condition has no effect on the required W* length. Therefore, the implicit assumption that the tubes are not locked at the TSP is either conservative or not significant to the determination of W*.

4.5.2 Impact of Tubesheet Bow on Pullout and Leak Rate Testing

Tubesheet bow is the flexing of the tubesheet in response to the primary-to-secondary side pressure difference results in a dilation of the holes above the mid-plane of the tubesheet during normal operation and postulated faulted conditions. The contact pressure decreases above the approximate[1] mid-plane of the tubesheet. The contact pressure increases below the mid-plane of the TS. The dilation does not have to be simulated since it can be treated analytically.

In situ leak rate tests are conducted at ambient conditions and there is no differential pressure across the tubesheet. Thus, there are two conditions that are atypical of normal operating and postulated accident conditions. In summary, the increase in temperature tends to make the joint tighter, and the increase in differential pressure across the tubesheet will tend to make the joint looser. In addition, for structural integrity testing the increase in pressure internal to the tube will act to tighten the joint and increase the strength of the joint. Thus, the act of testing may bias the results in a non-conservative manner at the elevation of the degradation being tested. It is noted that there is no hole dilation at the location of the neutral plane of the tubesheet, slightly below the mid-plane, and such testing which results in no leakage is indicative of operating and accident condition results to be expected at lower elevations within the tubesheet.

In situ structural testing is not likely to be meaningful in demonstrating compliance with performance criteria, i.e., demonstrating a resistance to pullout of greater than three times the normal operating pressure differential. Moreover, the difference in contact pressure during in situ testing means that the leak rate data cannot be used directly to quantify potential leak rates. This does not mean that an analytical procedure could not be developed to deal with such quantification, that is the basis for correlating the leak rate to the inverse of the loss coefficient and further correlating the loss coefficient to the contact pressure between the tubes and the tubesheet.

The effect of tubesheet bow can result in an average decrease in the contact pressure during postulated accident condition for the Model 51 tubes. For the tubes tested to date, in situ testing resulted in no measurable leakage. However, the contact pressure during the performance of the in situ test was significantly less than the contact pressure present when the laboratory tests where performed, almost all of which leaked. Thus the leak rate tests performed in situ are relevant to demonstrating whether or not an indication leaks. Although the leak rate from a leaking indication may not lend itself to a precise quantified prediction of the leak rate during operation, it can be used to estimate whether or not the leak rate would be significant during operation or postulated accident conditions. A deformation balance can be performed to determine if lower pressure test should be performed (results from the analysis of Model F SGs were that higher pressure led to higher leakage).

[1] The location of the neutral plane of the tubesheet is slightly below the mid-plane because of the membrane stress caused by the distributed pressure load.

During the Fall 2003 inspection at a plant with C-E steam generators, a 1.9 V (by +Point), 140° arc length, 99% through wall (by phase angle) circumferential crack (PWSCC) at the top of the tubesheet was in situ tested with no leakage being reported at the postulated SLB pressure difference.

4.5.3 Tube Pull Out Testing Description

This information provided in Attachment F. Portions of this information is Westinghouse Proprietary.

4.5.4 Tubesheet Finite Element Model Discussion

Loads are imposed on the tube as a result of tubesheet bowing under various pressure and temperature conditions. The finite element analysis of the tubesheet, channelhead, and lower shell were performed to determine the unit displacements throughout the tubesheet for two pressure unit loads (primary and secondary side) and three thermal unit loads (tubesheet, shell, and channel head). The analysis yielded the unit displacements throughout the tubesheet for these five unit loads. The normal operating and faulted conditions (pressure and temperature) were then applied to these unit displacements for calculating the tube-to-tubesheet contact pressure distribution from the top to the bottom of the tubesheet.

The pressure and temperature parameters for the feedline break, steam line break, and loss of coolant accident (LOCA) events were from a generic accident analysis. It is shown in Section 3.0 of this report that the generic temperature and pressure parameters used in the structural analysis bound the values used in the accident analysis of the Beaver Valley Unit 1 steam generators.

4.5.5 Ligament Tearing Discussion

One of the concerns that must be addressed in dealing with cracks in SG tubes is the potential for cracked tube radial ligament tearing to occur during a postulated accident when the differential pressure is significantly greater than during normal operation. While this is accounted for in the strength evaluations that demonstrate a resistance to pullout in excess of $3 \cdot \Delta P$ for normal operation and $1.4 \cdot \Delta P$ for postulated accident conditions, the potential for ligament tearing to significantly affect the SLB leak rate predictions needs to be accounted for.

Ligament tearing considerations for circumferential tube cracks that are located below the W* depths within the tubesheet are significantly different from those for potential cracks at other locations. The reason for this is that W* has been determined using a factor of safety of three relative to the normal operating pressure differential and 1.4 relative to the most severe accident condition pressure differential. Therefore, the internal pressure end cap loads which normally lead to an axial stress in the tube are not transmitted below about 1/3 of the W* depth. This means that the only source of stress acting to extend the crack is the primary pressure acting on the flanks of the crack. Since the tube is captured within the tubesheet, there are additional forces acting to resist opening of the crack. The contact pressure between the tube and tubesheet results in a friction induced shear stress acting opposite to the direction of crack opening, and the pressure on the flanks is compressive on the material adjacent to the plane of the crack, hence a Poisson's ratio radial expansion of the tube material in the immediate vicinity of the crack plane is induced which also acts to restrain the opening of the crack by increasing the contact pressure between the tube and the tubesheet. In addition, the differential thermal expansion of the tube is greater than that of the carbon steel tubesheet, thereby inducing a compressive stress in the tube below the W* length.

A scoping evaluation of the above effects was performed by ignoring the forces that resist the crack opening, and simply looking at the effect of the pressure acting to open the crack. If a 360° throughwall crack is considered, the stress from the pressure on the flanks is 0.206 or 20.64% of the stress that would result from an end cap pressure load for the same pressure. The primary pressure during normal operation and during a postulated accident is 2250 and 2665 psia respectively. The actual pressure difference between accident and normal operating conditions is 415 psi or 18% and the relative effect is equivalent to a change in pressure of 100 psi or 4% if the source of the stress were the end cap pressure differential.

The magnitude of the effect can also be used to conservatively estimate the ligament thickness of tube material affected. Using the ASME Code specified minimum yield stress of 35.2 ksi at 650°F, the applied forces can be calculated as if the pressures were applied to the entire cross-sectional area of the tube, thus representing the maximum force that can be applied to the tube as the result of a crack. These calculated maximum forces are 290.9 and 349.6 lbs at internal pressures of 2250 and 2665 psia respectively. So as not to exceed the yield stress, less than 9% of the cross-sectional area of the tube is required to resist the maximum force of 344.6 during a postulated accident. This equates to a circumferential crack that extends 360° and is 92% through-wall, i.e., the remaining material is less than 3.5 mils thick. The corresponding value for the normal operating condition is a little more than 3.0 mils. Thus, the difference in required wall thickness between the normal operation and accident condition pressures is on the order of 0.5 mil. If the resisting forces discussed above were to be included in this evaluation, the difference would be significantly less.

In summary, considering the worst-case scenario, the likelihood of ligament tearing from radial circumferential cracks resulting from an accident pressure increase is small since at most, only 8% of the cross-sectional area is needed to maintain tube integrity. Also, since the crack face area will be less than the total cross-sectional area used above, the difference in the force applied as a result of normal operating and accident condition pressures will be less than the 53.7 lbs calculated for the Model 51 steam generators. Therefore, the potential for ligament tearing is considered to be a secondary effect of essentially negligible probability and should not affect the results and conclusions reported for the W* evaluation. The leak rate model does not include provisions for predicting ligament tearing and subsequent leakage, and increasing the complexity of the model to attempt to account for ligament tearing has been demonstrated to be not necessary.

4.5.6 Discussion of No Contact Length for Normal/Postulated Accident Conditions

The no contact length for each of the SG zones for both normal and postulated accident conditions is the axial length over which dilation of the tubesheet causes the mechanical interference fit contact pressure between the tube OD and the tubesheet hole surface to reduce to zero. The no contact length during normal operating conditions is less than 1.0 inch in Zones A and B. The no contact length during a postulated SLB is less than 3 inches in Zones A and B.

All WEXTEX expansions are assumed to have a small gap over the upper 0.7 inches of distance below the BWT for both pullout force and leakage analysis.

4.6 Steam Line Break Leakage Considerations

The leak rate model included in WCAP-14797, Revision 2 is based on data from crevice tests and from cracked tube tests. The computer code CRACKFLO is based on the theory of two phase flow and crack opening area. Crevice flow uses a theory model with empirical data for validation at various levels of contact. An empirical effective axial crack length is used. The individual elements of which have been validated against test data. The DENTFLO model is based on a crack in series with the crevice. The model balances flow through the crack and through the crevice such that the continuity is maintained. The pressure at the exit of the crack must match the pressure at the entrance to the crevice. A deterministic corrosion calculation algorithm has been demonstrated to be conservative to a Monte Carlo simulation of uncertainties.

4.6.1 W* Leakage Model and Insitu Testing Validation Program

The following discussion is included for information only. The use of a bounding leak rate model based on test data for the Beaver Valley Unit 1 SG indications obviates the need to apply the standard W* leak rate model.

The W* leakage model was developed based upon first principles of leakage from a crack in a tight crevice. Leakage from a tight crevice is a series path through the crack with the crack opening constrained by the tubesheet, and followed by leakage through the tight crevice. In the leakage model, the total steam line break (SLB) pressure drop occurs from the tube inside diameter (ID) to near the top of the tubesheet (TTS) at the bottom of the WEXTEX transition (BWT). The crack inside the tubesheet, with very small clearances, cannot open significantly due to the constraint of the tubesheet hole ID. Leakage tests were performed to directly model this effect by measuring the leakage at the upper tip of the crack with small tube to tubesheet clearances. The leak rates for the constrained crack are correlated with contact pressure using an equivalent CRACKFLO crack length to represent leakage. Since the equivalent crack length is the length that gives the measured leak rate, the plot of Figure 6.4-1 of WCAP-14797, Revision 2, is essentially the leak rate through the crack at the crack tip versus contact pressure. The principal purpose in introducing the CRACKFLO equivalent length, rather than measured leakage, is to permit adjustments of the measured leak rates for the pressure drop across the crack in series with the crevice pressure drop for the leakage model.

The test data for leakage through the crevice is modeled using the crevice loss coefficient correlated with contact pressure. The loss coefficient is fit to each measured leak rate for the correlation with contact pressure, such that the correlation is essentially the leak rate through the crevice versus contact pressure. The use of a loss coefficient correlation, rather than leak rate, permits adjustments of the leak rate for the pressure drop across the crevice in series with the crack pressure drop.

Both the effective crack length and loss coefficient correlations are obtained as regression fits from the leak rate measurements. These values, therefore, are not analytical results. The analytical model is only used to perform the series leakage analysis, and consists primarily of adjustments to obtain equal leakage through the crack and crevice for the total tube ID to crevice exit pressure drop. This is a first principles fluid flow calculation based on prototypic, experimentally developed, effective lengths and loss coefficients. Combined crack and crevice leak tests by in situ testing were performed as described below.

To date, 14 W* indications have been in situ leak tested in the industry: 7 at Diablo Canyon Unit 2 (DCPP 2) at refueling outages 2R9 (1999) and 2R10 (2001), 1 at Sequoyah Unit 2 (SQN 2), and 6 at Beaver Valley Unit 1 (BVPS 1). The indications are listed in Table 4.6.1-1. No leakage was observed in any test. Most of the indications were located near the bottom of WEXTEX transition (BWT), such that the tubesheet provided minimal crevice restriction. The DCPP 2 indications are located in tubes that had been deplugged, were tested to normal operating pressure differential, and were returned to service. Non-deplugged W* indications at DCPP have not grown deep enough to satisfy the requirements for leak testing.

FENOC intent is to only in situ test indications that meet the requirements of the EPRI In Situ Guidelines.

Table 4.6.1-1

Industry In Situ Test Results for Axial PWSCC in WEXTEX Region

Plant	Year	SG	Tube	Deplug tube	Crack distance below BWT, (below TTS for SQN and BVPS), inch	Peak Volt	Crack Length inch	Max Depth	Approx Length > 80%, inch	Test Pressure	Leak Rate
DCPP 2	1999	1	R3C59	Yes	0.51	5.6	0.27	100%	0.23	NOP	0
DCPP 2	1999	1	R7C62	Yes	0.59	4.2	0.35	80%	None	NOP	0
DCPP 2	1999	2	R31C25	Yes	0.98	4.0	0.24	70%	None	NOP	0
DCPP 2	2001	3	R7C52	Yes	0.56	3.4	0.43	94%	0.37	NOP	0
DCPP 2	2001	4	R3C5	Yes	0.55	1.5	0.83	100%	0.63	NOP	0
DCPP 2	2001	4	R2C29	Yes	3.52	4.5	0.91	100%	0.84	NOP	0
DCPP 2	2001	4	R2C29	Yes	1.83	0.9	0.34	100%	0.1	NOP	0
SQN 2	1997	4	R7C17	No	0.15	3.6	0.32	100%	0.02	3dpNO	0
BVPS 1	1997	A	R10C51	No	0.24	0.7	0.30	77%	None	3dpNO	0
BVPS 1	1997	A	R27C28	No	3.20	1.2	0.22	35%	None	3dpNO	0
BVPS 1	1997	B	R5C83	No	0.35	1.5	0.21	30%	None	3dpNO	0
BVPS 1	1997	C	R27C31	No	0.60	0.9	0.18	44%	None	3dpNO	(1)
BVPS 1	2001	A	R7 C59	No	1.86	1.98	---	33%	N/A	SLB ΔP	0
BVPS 1	2001	B	R35C22	No	1.51	1.3	---	50%	N/A	SLB ΔP	0

(1) In situ pressure testing tooling system leakage. No leakage judged to be due to a flaw.

W* in situ leak tests are conducted at normal operating pressure differential. If the indication leaks, the test will be continued up to the SLB pressure differential, and the tube will be repaired. If leakage is not detected at normal operating pressure differential, the test would be terminated without extending the pressure differential to SLB conditions. If tubes require in situ testing, they will be removed from service regardless of the test results.

Also, it is noted that in situ leak testing of likely throughwall indications near the top of the tubesheet does provide meaningful information relative to the potential for throughwall indications located deeper in the tubesheet to leak. The in situ testing experience has been such that indications do not leak. Because the hole dilation diminishes with distance into the tubesheet, there is a location with a post accident condition radial contact load that corresponds to that achieved during the in situ testing. Thus, testing of through wall indications near the top of the tubesheet supports the evaluations that conclude that leakage from tube indications deeper in the tubesheet would be negligible.

4.6.2 Discussion of Tube Radial Contraction Effect

Equations 4.4-4 and 4.4-7 of WCAP-14797-P, Rev. 2 do not include a term to account for a contraction of the tube by the Poisson effect due to tube end loading. Had this been accounted for, the term

$$\frac{P_i}{E_t}\left[\frac{2a^2b}{b^2-a^2}\right] \qquad \text{(Current 4.4-4 Term)}$$

in both equations 4.4-4 and 4.4-7 would have been written as

Corrected Equation

$$\frac{P_i}{E_t}\left[\frac{(2-\nu)a^2b}{b^2-a^2}\right]. \qquad \text{(Modified 4.4-4 Term)}$$

The net result of including the appropriate Poisson effect in the expression for the outward movement of the tube OD due to internal pressurization, i.e., accounting for the end cap load on the tube, would be a 15% reduction in the calculated value for the outward movement of the tube OD due to internal pressure. This is based on a value of Poisson's ratio, ν, of 0.3 so 2-ν or 1.7 should have been used instead of 2 as the constant in the numerator. Calculations have been performed to determine the net result of including the aforementioned Poisson effect in the expression for the tube-to-tubesheet contact pressure, with equation 4.4-7 written as:

$$P=\frac{\frac{P_i}{E_t}\left[\frac{(2-\nu)a^2b}{b^2-a^2}\right]+b(\alpha_{tube}-\alpha_{TS})(T-70)}{\frac{b}{E_{TS}}\left[\frac{c^2+b^2}{c^2-b^2}+\nu\right]+\frac{b}{E_t}\left[\frac{b^2-a^2}{b^2-a^2}-\nu\right]}. \qquad \text{(Modified 4.4-7)}$$

For the range of contact pressures associated with the data in Figure 6.4-2 of WCAP-14797, Rev. 2, including the Poisson effect due to the tube end cap load produces net reductions in the contact pressure ranging from 50-350 psi, depending on the magnitude of the contact pressure plotted (which is a function of the internal pressure in the leak test specimen). A slight reduction of 50-100 psi occurs in the contact pressure for data points on the lower end of Figure 6.4-2, when accounting for the Poisson effect, thus shifting the data points slightly to the left. For the highest data points of Figure 6.4-2, a contact pressure reduction on the order of 250-350 psi would be calculated to occur, shifting these data points farther to the left. Hence, overestimation of the contact pressure through not accounting for the contraction of the tube due to the end cap load causes the scale on the crevice resistance correlation of Figure 6.4-2 and the effective crack length correlation of Figure 6.3-8 of WCAP-14797, Rev. 2 to shift outward. Modifications to account for the Poisson effect will cause both curves to become steeper, resulting in higher crevice resistance and a smaller effective crack length for a given calculated contact pressure. Hence, neglecting the Poisson effect in calculating the contact pressure for the test specimens produced conservative values of effective crack length and loss coefficient for the given contact pressures. This effect, however, is neutralized in the application of these correlations because lower contact pressures will be applicable for applying the correlations to calculate leak rates as described in Section 6.4 of WCAP-14797, Rev. 2. That is, the same contact pressure formula used to develop the leak rate correlations is also used to apply the correlations for calculating leak rates in Section 6.4. Hence, calculated leak rates using contact pressures which account for the Poisson effect would be expected to be negligibly different (estimated as less than a few percent) from those calculated neglecting the contraction of the tube due to the end cap load. The conclusions from the leak rate analysis would not be affected by the absence or presence of the Poisson's term in the calculation of the contact pressure.

In retrospect, it would have been more appropriate to include the Poisson contraction effect in the analyses. However, the expected negligible impact does not justify performing a complete reanalysis of the data with the Poisson effect included. Moreover, it is noted that because the load transmitted along the tube diminishes with depth into the tubesheet, the original equation is correct below what would be the W* distance without the application of a safety factor to the end cap load. This would further diminish any effect below that distance and further support the conclusion that a correction, really a modification, to the analyses is not necessary.

4.6.3 Discussion of Secondary to Primary Leakage Following Postulated LOCA

During normal operation, the tube-to-tubesheet contact pressure in the region above the tubesheet neutral bending axis is reduced by upward bending of the tubesheet due to the primary-to-secondary ΔP. This reduction in contact pressure is a function of elevation and radial location from the center of the tubesheet, and is accounted for in the determination of the W* lengths. During a postulated LOCA event, the primary side pressure drops to atmospheric conditions while the secondary side remains at 1005 psia. The component of the tube-to-tubesheet contact pressure resulting from primary pressure inside the tube is lost, and the external pressure on the tube acts to further reduce the tube-to-tubesheet contact pressure. However, the reversal of the ΔP across the tubesheet causes the tubesheet to bow downward, providing an increase in that component of the tube-to-tubesheet contact pressure above the neutral axis of the tubesheet. In the top four inches of the W* region near the top of the tubesheet, the increase in contact pressure due to downward tubesheet bending more than offsets the reduction in contact pressure due to the reversed ΔP across the tube wall. For instance, at a distance of 2" below the TTS, the tube-to-tubesheet contact pressure resulting from the primary-to-secondary ΔP is 790 psi during normal operation, while the maximum loss of contact pressure due to tubesheet bending is -1549.5 psi; the net minimum contact pressure (including an additional 509.8 psi contact pressure due to thermal expansion and 693 psi due to the residual WEXTEX contact pressure) is 443.3 psi. For postulated LOCA conditions, the loss of primary side pressure in conjunction with a secondary side pressure of 1005 psi results in a negative contact pressure due to the secondary-to-primary ΔP of -1225.6 psi. However, tubesheet bow in the opposite direction adds 970.6 psi in addition to the 509.8 psi contact pressure due to thermal expansion and 693 psi from the WEXTEX expansion residual preload; the net minimum contact pressure is 947.8 psi, which is 504.5 psi greater than the contact pressure during normal operating conditions. The net effect is a tighter joint at the top of the tubesheet during a postulated LOCA event than exists during normal operating conditions. Past analyses performed by Westinghouse have shown that secondary-to-primary in-leakage through free span cracks occurs at a slower rate than primary-to-secondary leakage at the same ΔP. Based on this experience, along with the lower magnitude of the LOCA ΔP relative to that during normal operation, and with the increase of tube-to-tubesheet contact pressure at the TTS caused by the reverse tubesheet bow, in-leakage to the primary side during a LOCA event would be expected to occur at a slower rate than primary-to-secondary leakage during normal operation, which is limited to 150 gpd (0.1 gpm). In-leakage to the primary side through W* tubes during a LOCA event is therefore assured of occurring at a slower rate than 0.1 gpm and would therefore not affect the plant LOCA analyses.

4.6.4 Discussion of Consistency of Leak Rate Results – C* Test Data Applicability

4.6.4.1 Introduction:

WCAP-14797, Revision 2 describes the development of criteria, designated W* (W-star), developed for utilities to use to disposition steam generator (SG) tube degradation when such degradation is found in the tube length that is within the tubesheet for tubes explosively expanded into the tubesheet, i.e., Westinghouse SGs with WEXTEX tube-to-tubesheet joints. A presentation was given to the NRC staff in 1996, Reference 4, and again in September 2003, Reference 5, further explaining the basis for the establishment of the criteria and its implementation.

Following the development of the W* criteria, two additional criteria were developed, designated H* for dealing with indications in tubes hydraulically expanded into the tubesheet in Westinghouse SGs and C* for dealing with indications in tubes explosively expanded, *explanded,* into the tubesheet in Combustion Engineering (CE) fabricated SGs. The information developed from these programs has been used to technically justify limiting the length of eddy current test (ECT) rotating pancake coil (RPC) inspection of the tubes within the tubesheet, notwithstanding the full length inspection using bobbin coil technology. The rationale is that the regulatory criteria relative to margin against burst during normal operation and postulated steam line break (SLB) events is precluded by the presence of the tubesheet and that the total leak rate from all indications within the tubesheet would be less than the UFSAR accident analysis assumption for primary-to-secondary leakage in the faulted loop during a postulated SLB, typically less than 1.0 gpm. The only possible mode of burst for tubes with degradation within the tubesheet is expulsion from the tubesheet of a severed tube end. It has been found that the length of engagement required to preclude burst relative to the length required to meet leakage requirements is a function of the joint fabrication process. The length required to meet leak rate requirements is usually longer than the length required to prevent burst for explosively and hydraulically expanded joints. The number of circumferential indications in WEXTEX tube-to-tubesheet joints is very small, and the testing program demonstrated that axial indications would be expected to exhibit no or miniscule leak below the W* distance, hence the conclusion was expressed that degradation of any extent could be tolerated below W*.

Recent leak rate results obtained from a second series of tests of explanded joints were preliminarily found to be potentially significantly greater than those obtained from the first series of tests. This prompted one utility to conservatively increase the length of the RPC inspection within the tubesheet because the final analysis of the test data was, and still is, incomplete, hence the final results of the analysis for Combustion Engineering explansion joints were and are unknown. However, a thorough review has since confirmed the validity of the original testing. Several of the steps taken by the test operators effectively deoxygenated the deionized water that was used in the W* leak tests. Nitrogen sparging, dissolved oxygen measurements or frequent venting during heatup was adequate to remove oxygen from the test water. Cooldown procedures and between test machining steps were conducted in a manner such that the sample was not exposed to air while at an elevated temperature. Therefore, it is highly unlikely that oxides formed within the tight annulus between the tube and the tubesheet stimulant that would have effected the leak rate results.

4.6.4.2 Testing Information:

There have been several series of tests performed on specimens expanded into tubesheet simulating collars using installation techniques typical of those used to fabricate the original SGs. The results of these tests have led to the establishment of plugging or inspection criteria designated as W* for WEXTEX, H* for hydraulic, and C* for Combustion Engineering tube-to-tubesheet joints. The predecessor criteria were F* for force and L* for leak rate from hard rolled tube-to-tubesheet joints. The F* criterion involved no leakage from the tubesheet crevice and is of no interest in this discussion except to note that F* was the length of engagement necessary to resist pullout of the tube from the tubesheet under all operating condition loads. The L* criteria, the length of engagement to prevent leakage, was similar to the W* criteria and the philosophy of development was the same. Because the length of engagement to prevent leakage, L*, in hard rolled joints was so short, shorter than the F* length, there was no consideration of allowing cracks to remain in service within the L* length.

W* Testing:

The leakage from cracks inside the tubesheet is comprised of two leak paths. One path is the tube to tubesheet crevice above the top of the crack. The second leak path is related to the ability of a crack to open inside the tubesheet where the tubesheet constrains opening of the crack flanks in the radial and hoop directions. The W* leak tests included two series of tests to separately measure leakage from the two leakage paths. The two tasks supported a W* leak rate calculation model incorporating the two leakage paths in series and satisfying conservation of mass and momentum relations.

The leakage path for the tube-to-tubesheet crevice leakage was achieved by drilling holes through the collar simulating the tubesheet and tube. The area of the drilled holes in the tube was sufficient for the tests to approximate leakage from a tube severed at the area of the holes. In particular, some of the tubes were staked, i.e., deformed inward, at the holes to assure separation of the tube from the collar, effectively opening the leak path at the interface. The measured leakage from these tests represents crevice restriction as a function of tube to tubesheet contact pressure as the only limitation on leakage.

The second series of tests were performed as constrained axial crack leak tests. These tests were performed to reflect the limited ability of the crack to open inside the tubesheet as a reduction in leakage compared to a freespan crack or a severed tube.

Tube sections were prepared with fatigue cracks, which conservatively model primary water stress corrosion cracks (PWSCC) since the fatigue cracks are uniformly throughwall with no ligaments and have a smoother crack face surface (less tortuosity) than corrosion cracks. The tubesheet collars were drilled with tight tolerances and the test tubes were ground on the OD in order to obtain small, well defined initial gaps to improve the accuracy of the contact pressure analyses. The collars were machined with a 360° groove with axial grooves to the top of the collar providing a large leakage path from the circumferential groove to the top of the collar. For the leak tests, the tip of the fatigue crack was located at the bottom of the groove in the collar to provide a leak path from the crack tip to the top of the collar with a negligibly small resistance. These tests measure only the constraining effect of the tubesheet on crack opening and leakage to the tip of the crack. Tests were performed in two collars with initial ambient condition diametral tube-to-collar gaps of 0.4 and 0.9 mils to obtain variability in the contact pressures. The elevated temperature tests were performed with borated and lithiated water to simulate primary water conditions. These tests thus measured leakage from a constrained crack to the tip of the crack as a function of contact pressure. These test results in combination with the crevice leakage test results provide test data for the entire leakage path in support of the leakage model for tubesheet crack leakage.

The results from the crevice tests were used to develop a flow loss coefficient or resistance as a function of contact pressure between the tube and the tubesheet. This was necessary because the tubesheet bows upward during operation resulting in a varying contact pressure that is a minimum at the top of the tubesheet and a maximum at the bottom. The leak rate equation is,

$$Q = \frac{1}{\mu K}\frac{dP}{dx} \tag{1}$$

where Q is the leak rate, μ is the effective viscosity of the fluid (water or steam or a combination), K is the loss coefficient, and dP/dx is the axial primary-to-secondary pressure drop driving the fluid through the crevice. The test programs are conducted to solve the rearranged equation,

$$K = \frac{1}{\mu Q}\frac{dP}{dx}, \tag{2}$$

and results in the determination of the logarithm of K as a linear function of the contact pressure, Pc, between the tube and the tubesheet, see Figure 4.6.4-1. [Note: Figure 4.6.4-1 is shown in Attachment F.] Three-dimensional finite element analysis was performed to map the tube-to-tubesheet contact pressure as a function of depth into the tubesheet at various radii from the center of the SG. This later step was taken to account for the dilation of the TS holes because the TS bows upward due to the primary-to-secondary pressure difference. In summary, a prediction of the potential leak rate from any and all indications within the tubesheet could be made based on a knowledge of the flaw distance below the top of the tubesheet, the temperature within the tubesheet, the radial location of the tube hole in the tubesheet (by row and column) and the primary-to-secondary operating conditions. The second series of tests was performed to measure the effect of the tubesheet on the leak rate from axial cracks in the SG tubes. The results of the analysis of the test data demonstrated that the loss coefficient would be extremely large for indications located below about 6 inches into the tubesheet.

The crevice testing was conducted at ambient and elevated temperature conditions using deionized (DI) water, and also deoxygenated water. A review of the test data the test equipment and test conduct indicates no apparent anomalies and no reason to question the efficacy of the obtained data. A total of four specimens were tested at a variety of internal pressure and temperature conditions. The results of the tests exhibited similar trends with no single specimen dominating the results. Moreover, the room temperature test results were consistent with the elevated temperature results according to the model, i.e., the major contributor to the leak rate behavior should be the viscosity for the same contact pressure conditions. Finally, a review of the testing program information from the time, including a meeting with the test engineer (now retired), indicates that the level of oxygen in the test water was minimized, a heated reservoir was used and the reservoir was vented periodically to control the pressure and remove oxygen.

H* Testing Programs:

The initial H* tests were conducted in the late 1990's using Westinghouse Model F tube specimens hydraulically expanded into carbon steel collars to simulate the tube-to-tubesheet joint. The justification of the H* length also made use of test results documented in a 1988 Westinghouse report relative to demonstrating the efficacy of the hydraulically expanded joints in Model F SGs. Tests were conducted for two different lengths of engagement and a variety of induced contact pressures between the tube and the tubesheet. The contact pressure can be controlled by the internal pressure in the tube and the temperature at which the test is performed. The Model F SG tests were conducted using DI water at ambient and elevated temperatures and at various primary-to-secondary differential pressures simulating normal operation and postulated accident conditions. The analysis approach was the same as that for W* in that a loss coefficient as a function of contact pressure was calculated and used to predict the leak rate of potentially throughwall tube indications within the tubesheet. A review conducted in 2001 of the earlier test data indicated that the leak rate resistance could be a function of the time-at-temperature of the specimen, i.e., the leak rate may decrease with operating time.

It was postulated at the time by Westinghouse that the interaction of the hot deionized, but not deoxygenated, water could have resulted in oxidation of the tubesheet in the crevice leading to the decrease in the leak rate as a function of time. It was also postulated that a similar corrosive reaction would take place in primary water as the carbon steel would interact with boron carried through the crack. Replenishment of the boron would not take place once the flow was stopped and further corrosion would not take place. Thus, the test results would be expected to be in concert with field data wherein an operating plant with Model F SGs with hydraulically expanded tubes was concluded to have a tube that was severed or nearly severed several inches into the tubesheet without attendant leakage during operation or during in situ testing. A second laboratory testing program was undertaken using tube specimens representative of Westinghouse Model D5 SGs to extend the H* results to another class (different tube size) of Westinghouse SGs with hydraulically expanded tubes. Deoxygenated primary water was used as the testing medium for the second series of tests. The results from the tests indicated a slight, but not statistically significant, decrease in the leak rate as a function of joint length and/or contact pressure. In other words, the resistance to leaking primary water increased relative to the use of DI water. This information provided direct confirmation relative to the adequacy of the use of DI water for the H* crevice leak rate testing and probably the W* test program, both of which were conducted at the Westinghouse laboratory in Churchill, Pennsylvania. Subsequent information indicates that the deionized test water would have been depleted of oxygen also.

C* Testing Programs:

The approach taken to demonstrating compliance was to determine the depth within the tubesheet that degradation of any extent could be postulated for all of the tubes in the SG and the leak rate requirement of the UFSAR accident analyses leak rate assumption in the faulted loop during a postulated SLB event would not be exceeded. The required length of engagement was determined from a series of tests performed for the Combustion Engineering Owners Group (CEOG). A length of engagement on the order of 7 inches, including measurement error, was calculated to be sufficient to meet a typical leak rate requirement of 0.1 gpm, apportioned to the tube circumferential indications within the tubesheet, during a postulated SLB event if all of the tubes were considered to have 360° by 100% deep circumferential cracks at that elevation. The tests were conducted using typical tube sizes for CE designed plants, test blocks and collars representative of the tubesheet fabrication process, and also included a tubed section of tubesheet from a plant that was cancelled during fabrication of the components. The explansion installation process for the tubes into the tubesheet holes was either duplicated or the tube-to-tubesheet joints already existed. Tubesheet holes in CE designed units are classified as being either rough bore or smooth bored depending on the plant. The rough bore drilling process results in holes more like, but not necessarily identical to, those in Westinghouse designed SGs, also a drilling process, than the smooth bore trepanning process used for a few plants' SGs.

Most of the initial test program was conducted using specimens at ambient conditions, i.e., 70°F, with a few specimens tested at elevated temperature. All of the first series of tests used deionized water as the testing medium. A significant fraction of the specimens exhibited no leakage at pressures approximating normal operation and SLB conditions. The results of the tests were analyzed using the W* methodology for the Reference 6 presentation in response to an NRC query regarding whether the results obtained by Westinghouse were similar to' those obtained by CE (now a part of Westinghouse). It was found that the loss coefficients from the C* tests were not inconsistent with those from the W* tests. This is an expected result since the explosive expansion process would be expected to be somewhat consistent, but not identical, from one SG fabricator to another. As noted above, it is possible that the finish results from the drilling process would also be different and the differences could affect the leak rate test results.

A second series of tests was initiated at the same time as the Model D5 testing program to increase the amount of data available at elevated temperature and to obtain results using primary water as the testing medium. A total of nine (9) specimens remaining from the initial test program were tested to obtain additional leak rate data points. The preliminary results from the analysis of the test data indicated an average decrease in the crevice resistance, i.e., loss coefficient, of about a factor of eight relative to the prior C* test data. Since the final results from the analysis of the data were not expected to be available for several weeks, one utility, with a plant in a SG inspection outage, decided to significantly increase the length within the tubesheet that would be inspected by RPC.

Discussion:

There have been several series of tests run to measure the resistance of the tube-to-tubesheet crevice for a variety of joints fabricated by both Westinghouse and Combustion Engineering. Most of the tests employed deionized water as the test medium but some of the recent tests employed a primary water simulant, borated and lithiated. Two sets of test results from the recent tests provided a direct comparison of results using DI water with those using borated and lithiated water. One set of those tests led to results indicating the later medium leads to reduced leakage relative to DI water while the other led to somewhat contradictory results. The first set was performed using test specimens simulating the hydraulic expansion of tubes into the tubesheet while the second simulated the CE explosive expansion process, a.k.a. explansion. Hydraulically expanded joints are looser than explosively expanded joints and offer less resistance to flow. In addition, models of the joint behavior would indicate that the W* joint loss coefficient should exhibit a higher dependency, i.e., correlation slope, that that of H* joints. The W* results are totally consistent with these technically based expectations. In addition, the conduct of the W* tests was such as to minimize any of dissolved oxygen in the testing medium.

There were some inconsistencies noted from the results of the second series of C* tests. The leak rates were of similar magnitude to those obtained from tests of hydraulically expanded, i.e., H*, joints. This is contradictory to physical expectation since explosively expanded joints have been verified to lead to a higher pullout resistance. This higher residual contact pressure results in a smaller leak path through the crevice and should result in a smaller leak rate for the same thermal and differential pressure conditions. In addition, the leak rates from the second series of C* tests did not result in significant differences between smooth and rough bore tubesheet specimens. The first series of tests demonstrated a higher crevice leakage resistance for rough bore tubesheet samples. This is consistent with some models of the tube-to-tubesheet interface and how it is changed during the expansion process.

Conclusions:

There is no direct evidence to suggest that any changes should be made to the W* inspection length. Indeed, final analyzed test data, the H* results, indicate that there should be no difference between results obtained with deionized or primary water. There are enough consistencies between the results obtained from the original W* tests relative to the later H* tests to conclude that the data were valid and formed an appropriate basis for establishing the W* criteria values. There are inconsistencies noted with the C* test results to conclude that they should not be used as a basis for modifying the W* criteria. Information gathered regarding the conduct of the tests supports this conclusion. Thus, the direct evidence that is available confirms the validity of the W* test results.

Finally, there are so few circumferential indications in the Beaver Valley Unit 1 SG tubes within the tubesheet that the margin relative to the allowable leak rate is very significant. The conclusion from these considerations is that the W* criteria should not be modified for the Beaver Valley Unit 1 tube inspections.

4.6.5 Address How Leak Model Addresses 360° Circumferential Crack

The bounding leak rate model is directly based on the test data from effectively severed tubes as described in Section 4.3.3 and thus obviates the concern that led to the other prior licensee questions on this topic.

The prediction of leak rates is discussed in Section 4.3 of this report. The leak rate from a 360° crack between at the W* distance and 12 inches below the top of the tubesheet at the worst case radial location within the tubesheet is calculated to be 0.0045 gpm at 90% prediction interval. At greater than 12 inches below the top of the tubesheet, the bounding per tube leakage allowance is 9 x 10^{-5} gpm. Modeling of future leak rates is a function of the number of circumferentially oriented cracks that are in the SGs at elevations below the W* distances.

4.6.6 Leak Rate Loss Coefficient RAI Discussion

The bounding leak rate model is directly based on the test data from effectively severed tubes as described in Section 4.3.3 and thus obviates the concern that led to the other prior licensee questions on this topic. The response that was provided to support the application of W* for another licensee is provided for information.

Background:

Mr. Phillip Rush, then of the NRC staff, transmitted a facsimile to Mr. Thomas Pitterle of Westinghouse on September 29, 1998, with the following discussion:

"Attached are copies of some Excel graphs I made to investigate the possibility of specimen dependence on the unusual scatter evident in Figure 6.4-2 of the W* topical report. I first separated the data into four groups by specimen ID, i.e., W4-001, W4-004, W4-008, and W4-018. Then I broke down the data in some of the groups into three subgroups (short, medium and long). Short medium and long specimens are those with crevice lengths on the order of 0.5 inches, 1 inch, and 2 inch respectively. As you can see from the attached figures, tests of W4-004 specimens and W4-018 clearly demonstrate that something unusual has cropped up in the test results. Please look at this and get back to me with your conclusions. Feel free to call me to discuss this observation."

The facsimile from Mr. Rush looks at the data base used to establish crevice loss coefficients for WEXTEX tube-to-tubesheet crevices. This data base was developed from leak tests run on simulated WEXTEX crevices and appears as Figure 6.4.2 of WCAP-14797, Revision 2. The data set is listed in Table 6.2-3 of WCAP-14797, Revision 2.

The information plots transmitted by Mr. Rush depicted data from the response (dated 1/1/1999) to NRC RAI on Diablo Canyon W* Loss Coefficient, which was prepared in response to earlier RAIs and updated Table 6.2-3 of the WCAP to include calculated loss coefficients (flow resistance) for the test specimens. The essence of Mr. Rush's concern is summarized as "... tests of W4-004 specimens and W4-018 clearly demonstrate that something unusual has cropped up in the test results."

The "something unusual" to which Mr. Rush refers is apparently that the loss coefficient appears to be dependent on crevice length. Physically, the loss coefficient should be independent of the length of the crevice.

Purpose:

A review of the WEXTEX leak rate data set and resulting calculated loss coefficients was conducted to determine if an unwarranted crevice length effect on loss coefficient is evident in the test data (the response, dated 1/1/1999, to NRC RAI on Diablo Canyon W* Loss Coefficient).

Loss Coefficient vs. Contact Pressure:

Mr. Rush re-plots the data set of loss coefficient vs. contact pressure (Figure 6.4-2 of Reference WCAP-14797-P, Rev 2). The loss coefficient test data used by Mr. Rush were provided via the response, dated 1/1/1999, to NRC RAI on Diablo Canyon W* Loss Coefficient. A review of the data set in this letter shows that most of the loss coefficients are an order of magnitude higher than those plotted on Figure 6.4-2. The corrected data set is attached as Table 4.6.6-1 to this letter and supercedes the data set previously provided via the response, dated 1/1/1999, to NRC RAI on Diablo Canyon W* Loss Coefficient. It is noted that Table 4.6.6-1 represents the data analyzed in support of the conclusions of WCAP-14797, Revision 2. [Note: Table 4.6.6-1 is shown in Attachment F.]

Mr. Rush's plots of the data, separated by sample number and crevice length, were re-plotted using the correct data set. These plots appear as Figures 4.6.6-1 through 4.6.6-4. [Note: Figures 4.6.6-1 through are shown in Attachment F.] Except for a few of the points, the plots have the same appearance as Mr. Rush's with the loss coefficient scale an order of magnitude lower. The data scatter on Figures 4.6.6-1 through 4.6.6-4 is consistent with the scatter on Figure 6.4-2 of WCAP-14797, Revision 2. Recall that the data appear to have constant variance about the regression line of WCAP-14797, Revision 2 and do not contradict the assumption of normality for the distribution of the residuals from the regression analysis.

On the surface, Figure 4.6.6-2 (for Sample W4-004) and to a lesser degree, Figure 4.6.6-4 (for Sample W4-018) seem to indicate that loss coefficient is increasing with decreasing crevice length. Samples W4-001 and 008 do not show this pattern. Figures

4.6.6-5 through 4.6.6-8 re-plot the same loss coefficient data sets versus crevice length with operating conditions as a parameter. When viewed in this manner, the data shows no consistent trend of loss coefficient variation with length. Sample WP4-001 shows a general increase of loss coefficient with length although one set of operating conditions shows the reverse trend. The medium lengths for Sample WP4-004 have lower loss coefficients than the short lengths, but the trend is reversed for the single set operating conditions tested at the long length. Sample WP4-008 shows a general increase of loss coefficient with length. Sample WP4-018 shows a decrease of loss coefficient with length for the medium and long lengths, but the reverse trend for the single case of short and medium length tested.

Conclusions:

The above review of the loss coefficient data for WEXTEX crevices shows no consistent length effect. The absence of a length effect is expected based on physical grounds.

Overall, it is noted that the leak rate data include considerable scatter, but do not show unacceptable bias toward the variables influencing crevice leakage. While scatter is common for leak rate data, the W* tests may include more than typical data scatter. The leak rates are small and minor variations in the crevice can influence the observed leak rates. Regardless, the spread in the loss coefficient data and associated uncertainties on the leak rates are included in the W* analyses, for both the loss coefficient and effective crack length.

Note: Table 4.6.6-1 and Figures 4.6.6-1 through 4.6.6.8 are provided in Attachment F.

<u>Precedent</u>

WCAP-14797 Revision 1 has been previously approved by the Nuclear Regulatory Commission for Diablo Canyon Units 1 and 2 in license amendments 129 and 127, respectively. Revision 2 of WCAP-14797 only address minor changes from Revision 1. The pages changed in Revision 2 include 1) page 6-20, 6-21 where loss coefficients for all specimens were added, and crevice length (1), average contact pressure (3) and leak rate (2) were corrected, 2) page 7-9 corrected a typo on growth rate, 3) page A-12 Table A1 added shading, and 4) Figure 6.2-3 titles were revised and included change bars. A similar, but not identical, W* methodology was also previously approved by the Nuclear Regulatory Commission for Sequoyah Unit 2 in license amendment 266 as documented in WCAP-13532, Revision 1, for one cycle of operation.

5.0 REGULATORY SAFETY ANALYSIS

FirstEnergy Nuclear Operating Company (FENOC) is proposing to modify the BVPS Unit 1 Technical Specifications (TS) to revise Surveillance Requirement (SR) 4.4.5.4.a.6, SR 4.4.5.4.a.8, and 4.4.5.5.d.1 and add SRs 4.4.5.2.e, 4.4.5.4.a.11 and 4.4.5.5.e for one cycle of operation. This proposed revision would be applicable only to Cycle 17. Specifically, the proposed change will revise the Unit No. 1 Technical Specification definition for steam generator tube inspection included in BVPS Technical Specification (TS) Surveillance Requirement (SR) 4.4.5.4.a.8 to revise the definition to exclude the portion of the tube within the tubesheet below the W* distance. The proposed change will also revise SR 4.4.5.4.a.6 on steam generator tube repair criteria, add SR 4.4.5.2.e to require a 100 percent rotating probe inspection of the hot leg tubesheet W* distance, add new W* terminology definitions in 4.4.5.4.a.11, and add new reporting criteria for W* inspection information in 4.4.5.5. FENOC's proposed change requires that any tube identified with service induced degradation in the W* distance or less than eight inches below the top-of-tube sheet (TTS), which ever is greater, be repaired. Since FENOC proposes to repair any service induced degradation within the W* distance, FENOC's proposal is a conservative limited scope application of the complete W* methodology as described in WCAP–14797, Revision 2.

FENOC's proposed change alters the tube inspection to exclude the portion of the tube within the tubesheet below the W* distance. The W* distance is defined in an analysis (WCAP-14797, Rev. 2) that accounts for the reinforcing effect that the tubesheet has on the external surface of the steam generator (SG) tube within the tubesheet region. This analysis shows that the tube integrity and leakage below the W* distance remain within the existing design limits. FENOC's proposed change requires that any service induced degradation identified in the W* distance or less than eight inches below the top-of-tube sheet, which ever is greater, be repaired. These changes will ensure that inspections may continue to ensure the health and safety of the public and safe operation of the plant while limiting unnecessary radiation exposure to plant personnel.

5.1 No Significant Hazards Consideration

FirstEnergy Nuclear Operating Company (FENOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10CFR50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed change modifies the BVPS Unit 1 TSs to incorporate steam generator (SG) tube inspection scope based on WCAP-14797, Revision 2. Of the various accidents previously evaluated in the BVPS Unit 1 Updated Final Safety Analysis Report (UFSAR), the proposed changes only affect the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB) accident evaluation. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Series 51 steam generators has shown that axial loading of the tubes is negligible during an SSE.

For the SGTR event, the required structural margins of the steam generator tubes will be maintained by the presence of the tubesheet. Tube rupture is precluded for cracks in the Westinghouse explosive tube expansion (WEXTEX) region due to the constraint provided by the tubesheet. Therefore, Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," margins against burst are maintained for both normal and postulated accident conditions.

The W* length supplies the necessary resistive force to preclude pullout loads under both normal operating and accident conditions. The contact pressure results from the WEXTEX expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. The proposed changes do not affect other systems, structures, components or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of an SGTR or SLB accident.

The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the WEXTEX expansion by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and collapse is strengthened by the tubesheet in that region. At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the W* length is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

SLB leakage is limited by leakage flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of crack face opening compared to free span indications. The total leakage, that is, the combined leakage for all such tubes meet the industry performance criterion, plus the combined leakage developed by any other alternate repair criteria, will be maintained below the maximum allowable SLB leak rate limit, such that off-site doses are maintained less than 10 CFR 100 guideline values and the limits evaluated in the BVPS Unit 1 UFSAR.

Therefore, based on the above evaluation, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed changes do not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the W* methodology.

   The proposed changes do not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   No. The proposed changes maintain the required structural margins of the steam generator tubes for both normal and accident conditions. NRC Regulatory Guide (RG) 1.121 is used as the basis in the development of the W* methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For primarily axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. WCAP-14797, Revision 2, defines a length, W*, of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). Application of the W* criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the W* criteria.

Plugging of the steam generator tubes reduces the reactor coolant flow margin for core cooling. Implementation of W* methodology at Beaver Valley Unit 1 will result in maintaining the margin of flow that may have otherwise been reduced by tube plugging.

Based on the above, it is concluded that the proposed changes do not result in a significant reduction of margin with respect to plant safety as defined in the Final Safety Analysis Report Update or bases of the plant Technical Specifications.

Based on the above, FENOC concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

A review of 10CFR50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants" (Reference 1), was conducted to assess the potential impact associated with the proposed changes. The following table lists the criterion potentially impacted, and an assessment of the need for a modification to the UFSAR description of BVPS design conformance to the criterion.

General Design Criteria		Impact
14	Reactor coolant pressure boundary	None
15	Reactor coolant system design	None
30	Quality of reactor coolant pressure boundary	None
31	Fracture prevention of reactor coolant pressure boundary	None
32	Inspection of reactor coolant pressure boundary	None

Steam generator (SG) tube inspection and repair limits are specified in Section 3/4.4.5 of the BVPS Unit 1 TS. The current BVPS TS require that flawed tubes be repaired if the depths of the flaws are greater than or equal to 40 percent through-wall, unless the degradation is subject to voltage-based repair criteria. The TS repair limits ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with GDC 14, 15, 30, 31, and 32 of 10 CFR 50, Appendix A. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Leakage integrity refers to limiting primary to secondary leakage to within acceptable limits.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) in the W* length is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. Primary-to-secondary leakage flow due to a postulated Steam Generator Tube Rupture event is not affected since the tubesheet enhances the tube integrity in the region of the WEXTEX expansion by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and collapse is strengthened by the tubesheet in that region. Steam Line Break (SLB) leakage is limited by leakage flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of crack face opening compared to free span indications. The combined leakage for all such tubes, plus the combined leakage developed by any other steam generator Alternate Repair Criteria, is maintained below the allowable SLB leak rate limit such that off-site doses are maintained less than the 10 CFR 100 guideline values. The W* criteria maintain the Regulatory Guide 1.121 margins against leakage for both normal and postulated accident conditions.

For design basis events, the required structural margins of the SG tubes will be maintained by the presence of the tubesheet. Tube rupture is precluded for cracks in the WEXTEX region due to the constraint provided by the tubesheet. The W* distance, which includes consideration for Non-Destructive Examination uncertainties, provides the necessary resistive force to preclude tube pullout under normal operating and accident conditions. Therefore, Regulatory Guide 1.121 margins against burst are maintained for both normal and postulated accident conditions.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. Beaver Valley Power Station Unit No. 1 Updated Final Safety Analysis Report

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

4. Westinghouse Electric Company WCAP-14795-P (Proprietary), "NRC/Utility Meeting on Model 51 Steam Generator Tube Integrity ARC Methodology", dated December, 1996.

5. Presentation Material, "The Application of W* Criterion to TVA SGs at Sequoyah Nuclear Station," Tennessee Valley Authority Meeting with the NRC Staff, Rockville, MD, September 10, 2003.

ATTACHMENT A

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 328

Proposed Technical Specification Changes

The following are the only affected pages:

3/4 4-8	#	
3/4 4-9		##
3/4 4-10	#	
3/4 4-10a	#	
3/4 4-10b		##
3/4 4-10c	#	##
3/4 4-10d	#	
3/4 4-10e		
3/4 4-10f		

\# No proposed changes for LAR 328. Included for readability only.

\## Pages also affected by BVPS Unit No. 1 pending License Amendment Request 322.

No Proposed LAR 328 Changes. Included for Readability Only.

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing Tavg above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with Article 8 of Section V ("Eddy current Examination of Tubular Products") and Appendix IV to Section XI ("Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing") of the applicable year and addenda of the ASME Boiler and Pressure Vessel Code required by 10CFR50, Section 50.55a(g). When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3 percent of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50 percent of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations greater than 20 percent, and

Includes One LAR 322 Change.
Included for Readability / Compatiblity.

2. Tubes in those areas where experience has indicated potential problems, and

3. At least 3 percent of the total number of sleeved tubes in all three steam generators. A sample size less than 3 percent is acceptable provided all the sleeved tubes in the steam generator(s) examined during the refueling outage are inspected. These inspections will include both the tube and the sleeve, and

4. A tube inspect n
4.4.5.4.a.8. If a e
passage of the edd e
inspection, this t
tube shall be selected and subjected to a tube inspection.

For Information Only.
4.4.5.2.b.3 proposed changes due to LAR 322 shown on next page.

5. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (4.4.5.4.a.10) shall be inspected by bobbin coil probe during all future refueling outages.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2. The inspections include those portions of the tubes where imperfections were previously found.

d. Implementation of the steam generator tube-to-tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

Insert 1

The results of each sample inspection shall be classified into one of the following three categories:

(3) Applicable only to Cycle 17.

<u>Insert 1</u>

4.4.5.2

e.[3] Implementation of the steam generator WEXTEX expanded region inspection methodology (W*), requires a 100 percent rotating probe inspection of the hot leg tubesheet W* distance.

<u>For Information Only</u>

<u>LAR 322 Insert</u>

3. <u>Except for Alloy 800 leak limiting sleeves</u>, at least 3 percent of the total number of sleeved tubes in all three steam generators. A sample size less than 3 percent is acceptable provided all the sleeved tubes in the steam generator(s) examined during the refueling outage are inspected. <u>All inservice Alloy 800 sleeves shall be inspected over the full length using a plus point coil or equivalent qualified technique during each refueling outage.</u> These inspections will include both the tube and the sleeve, and

No Proposed LAR 328 Changes. Included for Readability Only.

Category	Inspection Results
C-1	Less than 5 percent of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1 percent of the total tubes inspected are defective, or between 5 percent and 10 percent of the total tubes inspected are degraded tubes.
C-3	More than 10 percent of the total tubes inspected are degraded tubes or more than 1 percent of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10 percent) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under All Volatile Treatment (AVT) conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

   2. A seismic occurrence greater than the Operating Basis Earthquake,

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   4. A main steamline or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

   1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20 percent of the nominal tube wall thickness, if detectable, may be considered as imperfections.

   2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve.

   3. Degraded Tube means a tube or sleeve containing imperfections greater than or equal to 20 percent of the nominal wall thickness caused by degradation.

   4. Percent Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

   5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

Includes One LAR 322 Change.
Included for Readability / Compatiblity.

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

1.0)

a) Original tube wall 40%

This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.10 for the repair limit applicable to these intersections.

Insert 2

b) ABB Combustion Engineering TIG welded sleeve wall 32%

c) Westinghouse laser welded sleeve wall 25%

LAR 322 Insert

7. Unserviceable descr[...]n of a tube if it leaks or contains a defe[...] affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steamline or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot-leg side) completely around the U-bend to the top support ~~to the cold leg~~.

of the cold-leg, excluding the portion of the tube within the tubesheet below the W* distance, the tube to tubesheet weld and the tube end extension. This exclusion is applicable only to Cycle 17. This exclusion does not apply to steam generator tubes with sleeves installed within the tubesheet region.

9. Tube Repair refers to sleeving which is used to maintain a tube in-service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure. The following sleeve designs have been found acceptable:

a) ABB Combustion Engineering TIG Welded Sleeves, CEN-629-P, Revision 02 and CEN-629-P Addendum 1.

b) Westinghouse laser welded sleeves, WCAP-13483, Revision 1.

(3) Applicable only to Cycle 17.

Change applies to both LAR 322 and 328

Insert 2

2.0)[3] This definition does not apply to service induced degradation identified in the W* distance. Service induced degradation identified in the W* distance or less than eight inches below the top-of-tube sheet (TTS), which ever is greater, shall be repaired on detection.

LAR 322 Insert

6. d) Westinghouse Alloy 800 leak limiting sleeve[3]: Plug on detection of any service induced imperfection, degradation or defect identified in the (a) sleeve and/or (b) pressure boundary portion of the original tube wall in the sleeve/tube assembly (i.e., the sleeve-to-tube joint).

Includes One LAR 322 Change.
Included for Readability / Compatiblity.

(e) Westinghouse Alloy 800 sleeves: WCAP-15919-P, Revision 00

LAR 322 Inserts

10. Tube Support Plate Plugging Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented o[illegible] stress corrosion cracking confined [illegible]ess of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

    a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

    b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged, except as noted in 4.4.5.4.a.10.c below.

    c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit(1) may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit(1) will be plugged or repaired.

    d) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

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(1) The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

(3) Applicable only to Cycle 17.

No Proposed LAR 328 Changes. Included for Readability Only.

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left(\frac{CL - \Delta t}{CL}\right)$$

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on VMURL and time into cycle
Δt	=	length of time since last scheduled inspection during which VURL and VLRL were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC) [(2)]

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

---

(2) The NDE is the value provided by the NRC in GL 95-05 as supplemented.

Insert 3

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be submitted in a Special Report in accordance with 10 CFR 50.4.

b. The complete results of the steam generator tube and sleeve inservice inspection shall be submitted in a Special Report in accordance with 10 CFR 50.4 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes and sleeves inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the Commission prior to returning the steam generators to service (MODE 4) should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

Insert 4

2. If circumferential crack-like indications are detected at the tube support plate intersections.

(3) Applicable only to Cycle 17.

Insert 3

4.4.5.4

a.

11.[3] a) Bottom of WEXTEX Transition (BWT) is the highest point of contact between the tube and tubesheet at, or below the top-of-tubesheet, as determined by eddy current testing.

b) W* Distance is the non-degraded distance from the top of the tubesheet to the bottom of the W* length including the distance from the top of the tubesheet to the bottom of the WEXTEX transition (BWT) and Non-Destructive Examination (NDE) measurement uncertainties (i.e., W* distance = W* length + distance to BWT + NDE uncertainties).

c) W* Length is the length of tubing below the bottom of the WEXTEX transition (BWT) which must be demonstrated to be non-degraded in order for the tube to maintain structural and leakage integrity. For the hot leg, the W* length is 7.0 inches which represents the most conservative hot leg length defined in WCAP-14797, Revision 2.

Insert 4

For Cycle 17, the postulated leakage resulting from the implementation of the voltage-based repair criteria to tube support plate intersections shall be combined with the postulated leakage resulting from the implementation of the W* criteria to tubesheet inspection depth.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1 X 10-2, notify the Commission and provide an assessment of the safety significance of the occurrence.

e.[(3)] The aggregate calculated steam line break leakage from the application of tube support plate alternate repair criteria and W* inspection methodology shall be submitted in a Special Report in accordance with 10 CFR 50.4 within 90 days following return of the steam generators to service (MODE 4). In addition, the total number of indications that are identified from 1R16 rotating probe inspections that are performed as part of the W* inspections will be included in this report.

[(3)] Applicable only to Cycle 17.

Attachment B

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 328

Proposed Technical Specification Bases Changes

Technical Specification Bases changes are provided for information only.

The following is the only affected page:

B 3/4 4-2c

Provided for Information Only.

3/4.4.5 STEAM GENERATORS (Continued)

for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b (c) criteria.

Bases Insert

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

Bases Insert

The W* criteria incorporate the guidance provided in WCAP-14797, Revision 2, "Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions." W* length is the undegraded length of tubing into the tubesheet below the bottom of the WEXTEX transition (BWT) that precludes tube pullout in the event of a complete circumferential separation of the tube below the W* length. W* distance is the undegraded distance from the top of the tubesheet to the bottom of the W* length including the distance from the top of the tubesheet to the BWT and measurement uncertainties. Indications detected within the W* distance or less than eight inches below the top-of-tube sheet (TTS), which ever is greater, will be repaired upon detection.

Tubes to which WCAP-14797 is applied can experience through-wall degradation up to the limits defined in Revision 2 without increasing the probability of a tube rupture or large leakage event. Tube degradation of any type or extent below W* distance, including a complete circumferential separation of the tube, is acceptable. As applied at Beaver Valley Unit 1, the W* methodology is used to define the required tube inspection depth into the hot leg tubesheet, and is not used to permit degradation in the W* distance to remain in service. Thus while primary to secondary leakage in the W* distance need not be postulated, primary to secondary leakage from potential degradation below the W* distance will be assumed for every inservice tube in the bounding steam generator. The postulated leakage during a steam line break for Cycle 17 shall be equal to the following equation (as described in LAR 1A-328, Section 4.3.6):

$$\text{Postulated SLB Leakage}_{\text{cycle 17}} = \text{ARC}_{\text{GL 95-05}} + \text{Assumed Leakage}_{0''-8''<\text{TTS}}$$
$$+ \text{Assumed Leakage}_{8''-12''<\text{TTS}} + \text{Assumed Leakage}_{>12''<\text{TTS}}$$

Where: ARC $_{\text{GL 95-05}}$ is the normal SLB leakage derived from alternate repair criteria methods and the 1R16 steam generator tube inspections. This term would also include any other postulated leakage (e.g., as committed in LAR 1A-322 for Alloy 800 sleeves).

Assumed Leakage $_{0''-8''<\text{TTS}}$ is the postulated leakage for undetected indications in steam generator tubes left in service between 0 inches and 8 inches below the top of the tubesheet.

Assumed Leakage $_{8''-12''<\text{TTS}}$ is the conservatively assumed leakage from the total of identified and postulated unidentified indications in steam generator tubes left in service between 8 and 12 inches below the top of the tubesheet. This is 0.0045 gpm times number of indications. All postulated unidentified indications will be conservatively assumed to be in one steam generator. The highest number of identified indications left in service between 8 and 12 inches below TTS in any one steam generator will be included in this term.

Assumed Leakage $_{>12" <TTS}$ is the conservatively assumed leakage for the bounding steam generator tubes left in service below 12 inches below the top of the tubesheet. This is 0.00009 gpm times number of tubes left in service in the least plugged steam generator following 1R16.

The aggregate calculated SLB leakage from the application of all alternate repair criteria and the above assumed leakage shall be reported to the NRC in accordance with applicable Technical Specifications.

The combined calculated leak rate from all alternate repair criteria must be less than the maximum allowable steam line break leak rate limit in any one steam generator in order to maintain doses within 10 CFR 100 guideline values and within GDC-19 values during a postulated steam line break event.

Attachment C

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 328

Commitment Summary

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for Beaver Valley Power Station (BVPS), Unit No. 1 in this document. These commitments are only applicable to Cycle 17. Any other actions discussed in the submittal represent intended or planned actions by Beaver Valley. These other actions are described only as information and are not regulatory commitments. Please notify Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement, at Beaver Valley on (724) 682-5284 of any questions regarding this document or associated regulatory commitments.

Commitment	Due Date
The Steam Line Break leakage for Cycle 17 operation of BVPS Unit 1 is evaluated to be: Postulated SLB Leakage $_{\text{cycle 17}}$ = ARC $_{\text{GL 95-05}}$ + Assumed Leakage $_{0''\text{-}8'' <\text{TTS}}$ + Assumed Leakage $_{8''\text{-}12'' <\text{TTS}}$ + Assumed Leakage $_{>12'' <\text{TTS}}$ Where: ARC GL 95-05 is the normal SLB leakage derived from alternate repair criteria methods and the 1R16 steam generator tube inspections This term would also include any other postulated leakage (e.g., as committed in LAR 1A-322 for Alloy 800 sleeves). Assumed Leakage $_{0''\text{-}8'' <\text{TTS}}$ is the postulated leakage for undetected indications in steam generator tubes left in service between 0 inches and 8 inches below the top of the tubesheet (as described in Section 4.3.4, page 26 of this request) Assumed Leakage $_{8''\text{-}12'' <\text{TTS}}$ is the conservatively assumed leakage from the total of identified and postulated unidentified indications in steam generator	Process to address this commitment will be in place upon implementation of the amendment which approves the proposed Technical Specification changes supporting application of the W* methodology.

tubes left in service between 8 and 12 inches below the top of the tubesheet (See Section 4.3.2 and Section 4.3.3, page 23). This is 0.0045 gpm times number of indications (as described in Section 4.3.2 and Section 4.3.3, page 23 of this request). All postulated unidentified indications will be conservatively assumed to be in one steam generator. The highest number of identified indications left in service between 8 and 12 inches below TTS in any one steam generator will be included in this term.

Assumed Leakage $_{>12'' <TTS}$ is the conservatively assumed leakage for the bounding steam generator tubes left in service below 12 inches below the top of the tubesheet. This is 0.00009 gpm times number of tubes left in service in the least plugged steam generator following 1R16 (as described in Section 4.3.3, page 24 of this request)

The above commitments are only applicable to Cycle 17.

Attachment D

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 328

Westinghouse WCAP-14798-NP, Revision 2,
"Generic W* Tube Plugging Criteria for 51 Series
Steam Generator Tubesheet Region WEXTEX Expansions"

Note: The attached WCAP is Westinghouse non-proprietary.

(See enclosed WCAP-14798-NP, Revision 2)

Attachment G

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 328

Redacted Version of Westinghouse Proprietary Information
Provided in Attachment F

Westinghouse Letter LTR-SGDA-04-212, dated June 18, 2004
"Additional Information Supporting the Application of W* to the
Beaver Valley Unit 1 Steam Generator Tubes"
[Non-Proprietary]

Note: The attached information is not proprietary.

Westinghouse Non-Proprietary Class 3

Westinghouse

To: G. Brassart
cc: L.A. Nelson
J.M. Hall
W.K. Cullen

Date: 6/18/04

From: Steam Generator Design and Analysis
Ext: 724-722-5584
Fax: 724-722-5889

Your ref: LTR-SGDA-04-192
Our ref: LTR-SGDA-04-212

Subject: **Additional Information Supporting the Application of W* to the Beaver Valley Unit 1 Steam Generator Tubes (Non-Proprietary)**

Please transmit the attached engineering information to FirstEnergy Nuclear Operating Company (FENOC) in support of the implementation of a subset of the steam generator tube W* inspection methodology for Beaver Valley Unit 1.

The proposed change to the Unit 1 Technical Specifications (via LAR 1A-328) revises the definition of steam generator tube inspection to exclude the portion of the tube within the tubesheet below the W* distance, tube-to-tubesheet weld and tube end extension by crediting the methodology as described by WCAP-14797, Rev 2. FENOC's proposed technical specification change requires that any service induced degradation identified in the W* distance must be removed from service. This information is included in ATTACHMENT F and G of LAR 1A-328.

If you have any questions regarding the content of this letter, please contact me at the above telephone number.

Verified by:

Gary W. Whiteman
Steam Generator Design and Analysis

R.F. Keating
Major Component Replacements and Engineering

5182.doc:49 06/18/2004

Official Record Electronically Approved in EDMS 2000

Attachment F Figure 4.3.5

a,c,e

a,c,e

Attachment F Figure 4.3.6

[ ]

Attachment F Figure 4.3.7

a,c,e

Figure 4.6.4-1 WEXTEX Loss Coefficient as a Function of Added Contact Pressure

a,c,e

[ ]

Table 4.6.6-1 WEXTEX Loss Coefficient Reanalysis
Arranged by Sample #/Crevice Length

a,c,e

Figure 4.6.6-1 Loss Coefficient vs. Crevice Length – Sample W4-001

a,c,e

Figure 4.6.6-2 Loss Coefficient vs Contact Pressure – Sample W4-004

a,c,e

Figure 4.6.6-3 Loss Coefficient vs. Crevice Length Sample W4-008

a,c,e

Figure 4.6.6-4 Loss Coefficient vs. Contact Pressure –Sample W4-018

a,c,e

Figure 4.6.6-5 Loss Coefficient vs. Crevice Length

Sample WP4-001

a,c,e

Figure 4.6.6-6 Loss Coefficient vs. Crevice Length Sample WP4-004

a,c,e

Figure 4.6.6-7 Loss Coefficient vs. Crevice Length

a,c,e

[ ]

Figure 4.6.6-8 Loss Coefficient vs. Crevice Length

a,c,e

[ ]

4.5.3 Tube Pull Out Testing Description

The pullout testing program and results are described in the "Pullout Test Specimen Descriptions" section of WCAP-14797, Rev. 2, Section 4.2.2 and is summarized below.

The W* pullout test samples were selected from a number of specimens prepared in the W* program to provide a lower bounding case condition with regard to pullout resistance. The W* pullout samples consisted of carbon steel collars approximately [

]^{b,c,e}. The nominal unexpanded outside diameter of the tube was 0.875 inches.

Since the WEXTEX expansion is a high energy process which causes the tube OD to impact and to be deformed into even small variations in the tubesheet hole bore surface, the feature which is of most significance is the surface finish of the tubesheet hole. The Series 51 SG tubesheet hole requirement was [ ]^{a,c} microinch rms maximum. The samples in the test program were procured to a [ ]^{a,c} rms requirement. For the W* pullout tests, two tubesheet collar specimens were selected, one specimen which appeared to have a bore surface roughness at the upper end of the bare finish requirements, and one with a smooth. Both of tubesheet collar specimens had uniform diameter profiles.

[

]^{a,c}

The WEXTEX samples fabricated for pull force testing were of a double ended configuration. After WEXTEX expansion, the samples were qualitatively checked for joint leak tightness. About 70% of the specimens exhibited a slight degree of looseness in the hoop direction in that it was possible to rotate the tube by a estimated 1 or 2 degrees (1° is approximately equal to 7 mils).

The samples were also noted to have [

]^{b,c,e}