Westinghouse Electric Company,
a division of CBS Corporation

Box 355
Pittsburgh Pennsylvania 15230-0355

June 1, 2000

CAW-00-1399

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

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Response to Selected CPSES – NRC Requests for Information (RAIs)"

Dear Mr. Collins:

The application for withholding is submitted by Westinghouse Electric Company, LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.790, Affidavit CAW-00-1399 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW-00-1399 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosure

cc: T. Carter/NRC (5E7)

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 2.790 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) contained within parentheses 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 2.790(b)(1).

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CAW-00-1399

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, 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 Corporation ("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:

Henry A. Sepp

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 5th day
of June, 2000

Lorraine M. Piplica
Notary Public

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Business Unit, of the Westinghouse Electric Company 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 rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Nuclear Services Business Unit.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 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 the Westinghouse Nuclear Services Business Unit 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 2.790 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 which 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 10CFR Section 2.790, 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 "Response to Selected CPSES – NRC Requests for Information (RAIs)". This information is being transmitted by Tennessee Valley Authority letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Samuel J. Collins. The proprietary information as submitted for use by the Tennessee Valley Authority, Watts Bar Unit 1 is expected to be applicable in other licensee submittals in response to certain NRC requirements for licensing of a 1.4% power uprate to 3459 MWt.

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

(a) Provide the applicable engineering evaluations which establish the technical basis for the 1.4% power uprate.

(b) Provide licensing information to support license amendments.

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 the methodology in the licensing process.

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 methodologies 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 for developing the methodology.

Further the deponent sayeth not.

ENCLOSURE 12

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1 - DOCKET NO. 390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-06

LIST OF COMMITMENTS

The following items will be completed prior to increasing WBN Unit 1 reactor power above 3411 MWt:

1. WBN will install an LEFM System for the purposes of uprating the Unit 1 RTP from 3411 to 3459 MWt. Included will be implementation of the necessary procedures and documents required for operation, maintenance, testing, and training at the uprated power level with the new LEFM System.

2. WBN will address the operability requirements for the LEFM System, including the appropriate actions to be taken when the LEFM is unavailable in a new Technical Requirement to be included within the WBN Technical Requirements Manual (TRM).

3. TVA will issue a change to the WBN FSAR that will stipulate that future revisions of the WBN-specific topical reports listed in TS Section 5.9.5b that currently assume 102 percent of rated power shall reflect 100.6 percent of rated power only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM).

4. TVA will perform a calculation to confirm that the existing 40% through wall plugging criteria for Steam Generator Tubes will remain adequate for the 1.4% uprate conditions.

5. With respect to the 1.4% uprate, the WBN Steam Generator Inspection Program will include consideration of the higher temperatures in growth rate analyses. Based on condition monitoring and operational assessments of inspection results, expansion of inspection plans and repairs will be made. Degradation growth rate changes will be incorporated into the operational assessment associated with potential affects of the uprate.

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1 - DOCKET NO. 390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-06

WESTINGHOUSE POWER CALORIMETRIC MEASUREMENT UNCERTAINTY CALCULATION
FOR WATTS BAR UNIT 1 POWER UPRATE TO 3459 MWT

(NON-PROPRIETARY)

Power Calorimetric Measurement Uncertainty Calculation for

Watts Bar Unit 1 Power Uprate to 3459 MWt

May 2000

WESTINGHOUSE ELECTRIC COMPANY LLC
4350 Northern Pike
Monroeville, Pennsylvania 15146-2886

1. INTRODUCTION

Westinghouse WCAP-14738, "Westinghouse Revised Thermal Design Procedure Instrumentation Uncertainty Methodology for Tennessee Valley Authority, Watts Bar Unit 1," Revision 0 - contains the current power measurement uncertainty calculations. These calculations have been supplemented by ones that calculate the uncertainty using the LEFM. These calculations are described herein and have been completed under the following assumptions:

1) Feedwater flow will be measured by a permanently installed leading edge flow meter (LEFM) located in the feedwater header.

2) Feedwater temperature will be measured using the LEFM.

3) The combined accuracy for feedwater flow and feedwater temperature (both density and enthalpy effects) is ±0.483% Rated Thermal Power.

4) The steam moisture content is less than [ ]$^{+a,c}$.

5) The net pump heat addition due to the reactor coolant pumps and reactor coolant system is [ ]$^{+a,c}$.

6) Steam generator blowdown is not secured.

7) The daily power measurement calculations are performed in the new Integrated Computer System (ICS).

8) The output of the LEFM is a digital signal transmitted to the ICS by means of afiberoptic link. Therefore there is no need for analog to digital conversion electronics.

At present a 2.0% Reactor Thermal Power uncertainty is used in selected WattsBar Unit 1 FSAR Chapter 15 analysis. The 1.4% power uprate is based on a power measurement uncertainty of ~0.6% RTP.

As shown at the bottom of Table 11a (attached), the calculated power uncertainty is ~0.6 % RTP (rounded up). Tables 9a, 10a and 11a show the individual uncertainties used to calculate the overall power measurement uncertainty. Tables 9a, 10a and 11a will be included in the next revision of WCAP-14738.

2. BACKGROUND

Watts Bar performs a secondary side heat balance (or calorimetric) measurement every 24 hours when power is above 15% of Rated Thermal Power (RTP). This heat balance is used to verify that the plant is operating within the limits of the Operating License and to adjust the Power Range Nuclear Instrumentation System when the difference between the NIS and the heat balance is greater than the limits set forth in the plant Technical Specifications. The calculation is to be performed daily, and the plant process computer is used to calculate the heat balance.

Assuming that the primary and secondary sides are in equilibrium, the core power is determined by summing the thermal outputs of the steam generators, correcting the total secondary power for steam generator blowdown (if not secured), subtracting the Reactor Coolant Pump (RCP) heat addition, adding the primary side system losses and dividing the algebraic result by the core rated Btu/hr at full power. The resulting equation for this calculation is the following:

$$RP = \frac{\{(\Sigma Q_{SG}) + Q_L - Q_P\}(100)}{H}$$

where

RP	=	Core power (% RTP)
Q_{SG}	=	Steam generator thermal output (Btu/hr)
Q_p	=	RCP heat adder (Btu/hr)
Q_L	=	Primary system net heat losses (Btu/hr)
H	=	Core rated Btu/hr at full power.

$$Q_{SG} = (h_s - h_f)W_f + (h_{sgbd} - h_s)W_{sgbd}$$

where

h_s	=	Steam enthalpy (Btu/lb)
h_f	=	Feedwater enthalpy (Btu/lb)
W_f	=	Main feedwater mass flow (lb/hr)
h_{sgbd}	=	Steam generator blowdown enthalpy (Btu/lb)
W_{sgbd}	=	Steam generator blowdown mass flow (lb/hr)

At present, main feedwater flow is measured by four Venturis (1 per loop) and the associated ΔP transmitters. Main feedwater temperature is measured by four RTDs (1 per loop). The following figures show the instrumentation layout.

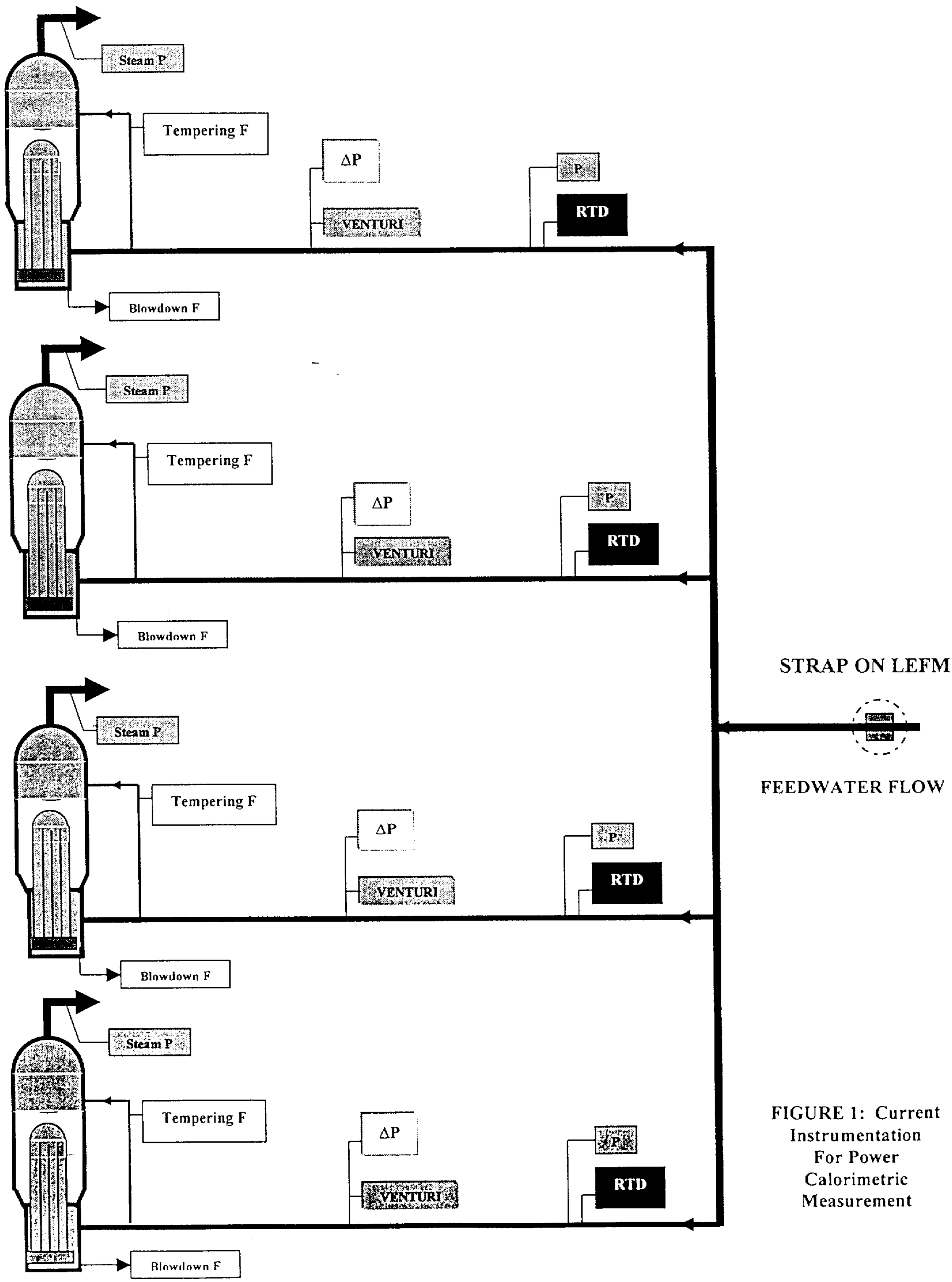
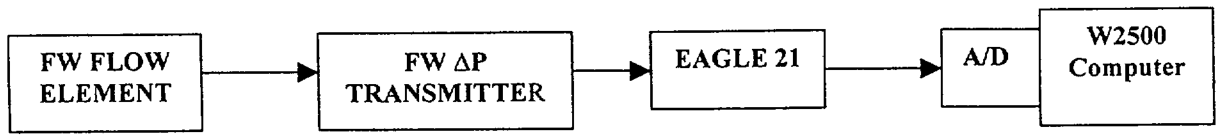
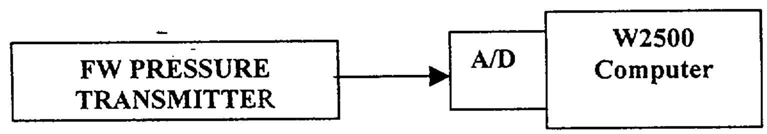


FIGURE 1: Current Instrumentation For Power Calorimetric Measurement

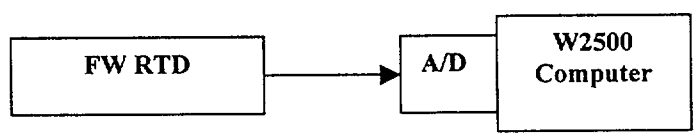


Main feedwater flow element: Venturi.

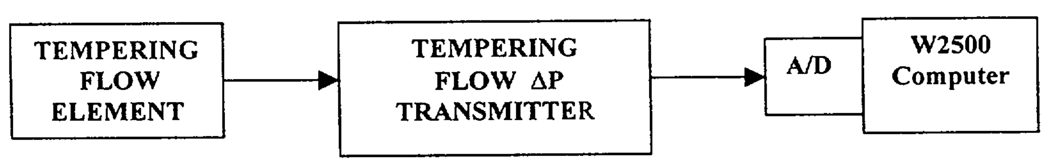
Main feedwater flow (ΔP) transmitter: 8 total; Range: 0 – $4.5X10^6$ lb/hr; Rosemount model 1152DP5 ΔP transmitter.



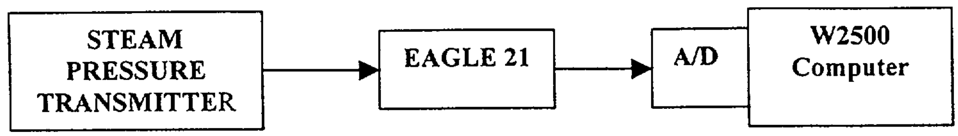
Main feedwater pressure transmitter: 4 total; Range: 0 – 1300 psig; Foxboro E11GM transmitter.



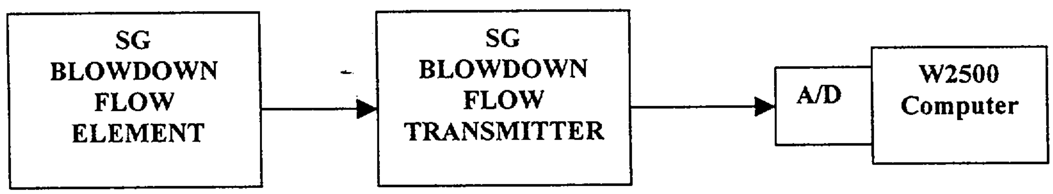
Main feedwater temperature RTD: 4 total: Range: 100 – 217.54Ω $\Rightarrow$ 32 – 600°F; Newport Labs RTD, model 600-12.



Main feedwater tempering flow transmitter: 4 total; Range 0 – 5" of H_2O column $\Rightarrow$ 0 – $84.51x10^3$ lb/hr; Rosemount 1151DP3 transmitter.



Steam pressure transmitter: 12 total; Range 0 – 1300 psig; Foxboro model N-E11GM; Foxboro model E11GM; Barton 763 Lot 7 transmitters.



Steam generator blowdown flow transmitter: 4 total; Range 0 – 15" of H_2O column
$\Rightarrow$ 0 – 87.5 gpm.

Based on the configuration above, WCAP-14738 Revision 0 establishes the uncertainty of the power measurement calculated by the current plant process computer as [ ]$^{+a,c}$.

3. POWER CALORIMETRIC MEASUREMENT UNCERTAINTY USING THE LEFM ON FEEDWATER HEADER

The power calorimetric measurement uncertainty calculation using the leading edge flow meter assumes the following conditions:

- The new LEFM is installed permanently in the feedwater header. The LEFM also provides measurement of the feedwater temperature. The combined accuracy for feedwater flow measurement and feedwater temperature measurement (density and enthalpy components), as provided by Caldon, is ±0.483% flow. The output of the LEFM electronics is a digital signal transmitted to the ICS by means of a fiberoptic link. This digital signal contains the flow measurement and the temperature measurement. The signal transmitted to the computer is digital. Therefore, there is no need for analog to digital conversion electronics at the input of the computer.

- Steam moisture content is less than [ ]$^{+a,c}$.

- Net pump heat addition is [ ]$^{+a,c}$.

- Steam generator blowdown flow is not secured.

- The existing W2500 plant process computer will be replaced by an Integrated Computer System (ICS). The analog to digital input conditioner has the following uncertainty values:

$$\begin{aligned} RCA_{A/D} &= \\ RMTE_{A/D} &= \\ RTE_{A/D} &= \\ RD_{A/D} &= \end{aligned} \left[\quad \quad \quad \quad \right]^{+a,c}$$

The following figures show the instrumentation layout.

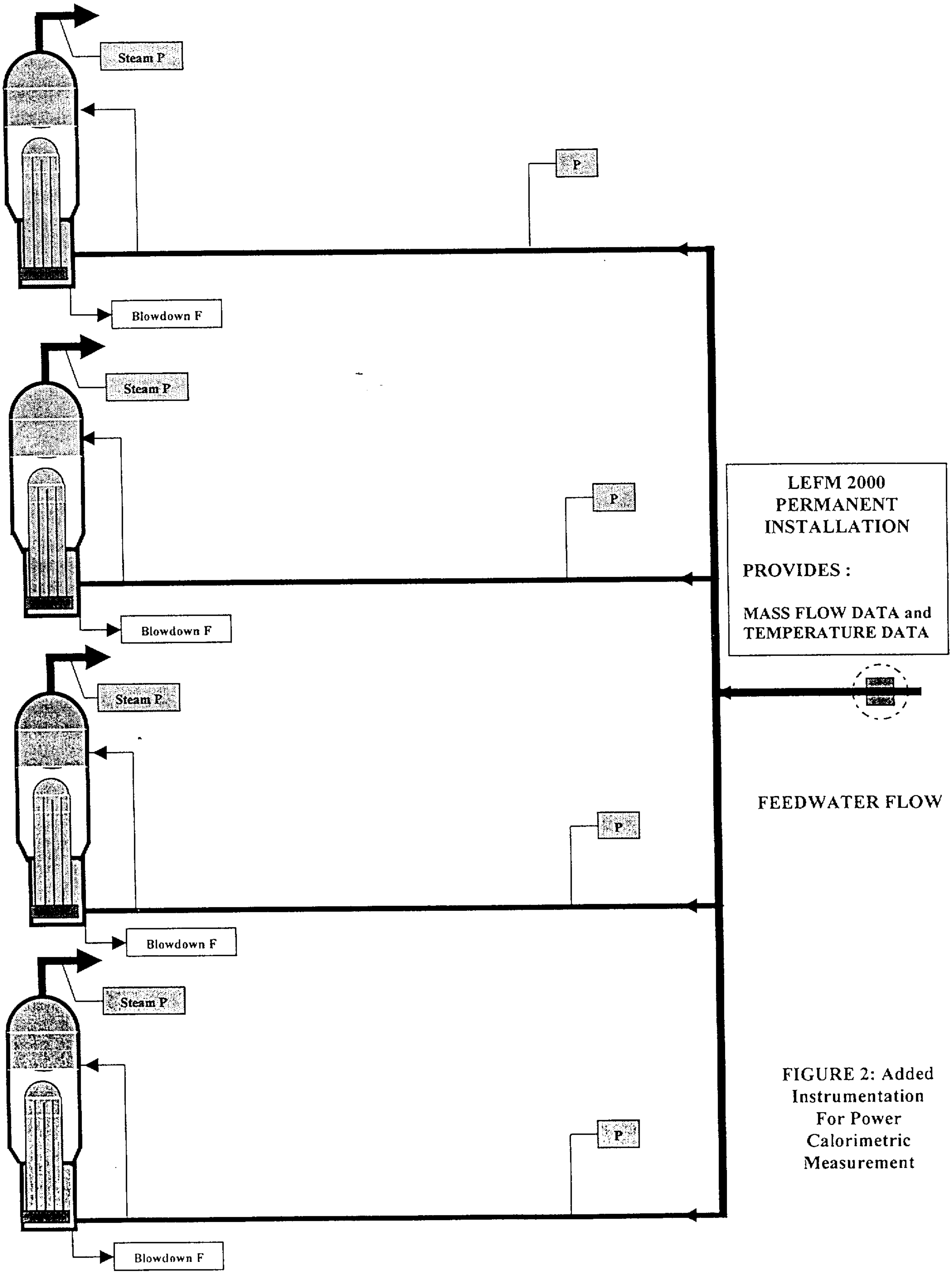
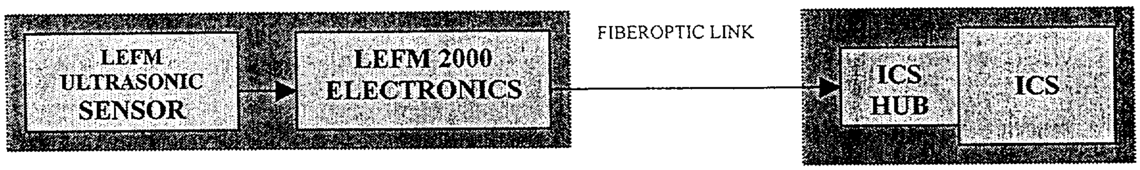


FIGURE 2: Added Instrumentation For Power Calorimetric Measurement

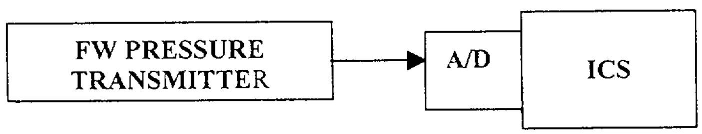
Main Feedwater Leading Edge Flow Meter (flow and temperature data) (total of 1) (Caldon, model LEFM 2000):



FMT-3-415

Feedwater Pressure Transmitter (total of 4) (Foxboro E11GM transmitter):

0 – 1300 psig;



1-PT-3-37

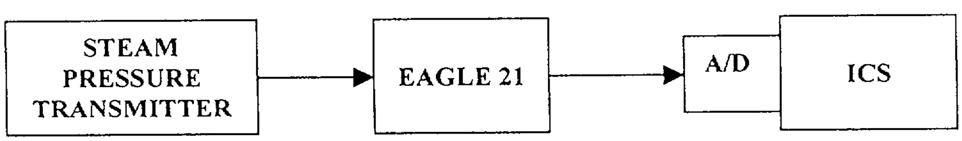
1-PT-3-50

1-PT-3-92

1-PT-3-105

Steam Pressure Transmitter (total of 12) (Foxboro model N-E11GM; Foxboro model E11GM; Barton 763 Lot 7 transmitters):

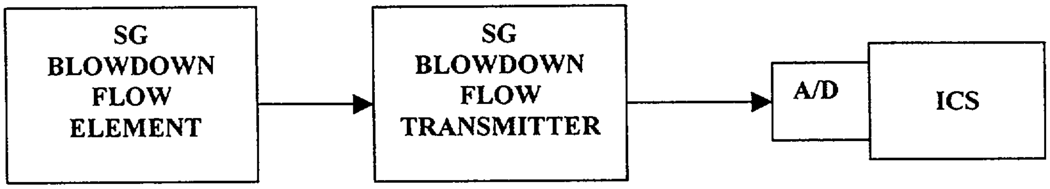
0 – 1300 psig.



1-PT-1-2A	(Loop 1) (Foxboro N-E11GM)
1-PT-1-2B	(Loop 1) (Foxboro N-E11GM)
1-PT-1-5	(Loop 1) (Foxboro E11GM)
1-PT-1-9A	(Loop 2) (Barton 763 Lot 7)
1-PT-1-9B	(Loop 2) (Barton 763 Lot 7)
1-PT-1-12	(Loop 2) (Barton 763 Lot 7)
1-PT-1-20A	(Loop 3) (Barton 763 Lot 7)
1-PT-1-20B	(Loop 3) (Barton 763 Lot 7)
1-PT-1-23	(Loop 3) (Barton 763 Lot 7)
1-PT-1-27A	(Loop 4) (Foxboro N-E11GM)
1-PT-1-27B	(Loop 4) (Foxboro N-E11GM)
1-PT-1-30	(Loop 4) (Foxboro E11GM)

Steam Generator Blowdown Flow Transmitter (total of 4) (Rosemount 1153DB3RB transmitter):

0 – 120 gpm



1-FE-1-152	1-FT-1-152
1-FE-1-156	1-FT-1-156
1-FE-1-160	1-FT-1-160
1-FE-1-164	1-FT-1-164

Tables 9a, 10a and 11a reflect the uncertainties calculated using the LEFM.

The calculated power measurement uncertainty is ~0.6% RTP (rounded up).

+a,c

[ ]

Table 9a (for WCAP-14738 Rev.1)
INTEGRATED COMPUTER SYSTEM POWER MEASUREMENT INSTRUMENTATION UNCERTAINTIES
(USING FEEDWATER LEFM ON FEEDWATER HEADER)
FOUR LOOP OPERATION

(% SPAN)	FW TEMP	FW PRES	FW (header)	FW d/p (tempering)	STM PRES	SG BLOWDOWN FLOW
						+a,c
LEFM =						
SRA =						
SMTE =						
SPE =						
STE =						
SD =						
BIAS =						
RCA_{EAI} =						
$RMTE_{EAI}$ =						
RTE_{EAI} =						
RD_{EAI} =						
RCA_{EAO} =						
$RMTE_{EAO}$ =						
RTE_{EAO} =						
RD_{EAO} =						
$RCA_{A/D}$ =						
$RMTE_{A/D}$ =						
$RTE_{A/D}$ =						
$RD_{A/D}$ =						
CSA =						
NUMBER OF INSTRUMENTS USED	1	1/LOOP	1		1/LOOP	1/LOOP
	°F	psi	%FLOW	%FLOW	psi	%FLOW
INST SPAN =		1300			1300	1.3
						+a,c
INST UNC (RANDOM) =						
INST UNC (BIAS) =						
NOMINAL =	441.8	1080 psia	100.0		980 psia	87.5 gpm/loop**

* Effects are included in the Caldon supplied feedwater uncertainty.
** The conditions analyzed for SG blowdown flow for the power measurement uncertainty are based on a maximum total flow rate of 350 gpm.

All parameters are read by the Integrated Computer System

TABLE 10a (for WCAP-14738 Rev.1)
INTEGRATED COMPUTER SYSTEM POWER MEASUREMENT SENSITIVITIES
(USING LEFM ON FEEDWATER HEADER)
FOUR LOOP OPERATION

FEEDWATER DENSITY			
TEMPERATURE	=	**	%/°F
PRESSURE	=	0.0041 *	%/psi

+a,c

FEEDWATER ENTHALPY		
TEMPERATURE	=	
PRESSURE	=	
h_s	=	
h_f	=	
$\Delta h(SG)$	=	
STEAM ENTHALPY		
PRESSURE	=	
MOISTURE	=	
S. G. BLOWDOWN FLOW		
F_a		
TEMPERATURE	=	
MATERIAL	=	
DENSITY		
PRESSURE	=	
ΔP	=	
S.G. BLOWDOWN ENTHALPY		
PRESSURE	=	

* Supplied by Caldon

** Incorporated into the feedwater flow uncertainty supplied by Caldon

TABLE 11a (for WCAP-14738 Rev.1)
INTEGRATED COMPUTER SYSTEM POWER MEASUREMENT UNCERTAINTIES
(USING FEEDWATER LEFM ON FEEDWATER HEADER)
FOUR LOOP OPERATION

COMPONENT		INSTRUMENT ERROR	POWER UNCERTAINTY +a,c
FEEDWATER FLOW (HEADER)			
FEEDWATER DENSITY			
TEMPERATURE			
PRESSURE			
FEEDWATER ENTHALPY			
TEMPERATURE			
PRESSURE			
STEAM ENTHALPY			
PRESSURE			
MOISTURE			
NET PUMP HEAT ADDITION			
STEAM GENERATOR BLOWDOWN FLOW			
ORIFICE (FLOW COEFFICIENT)			
THERMAL EXPANSION COEFFICIENT			
TEMPERATURE			
MATERIAL			
DENSITY			
PRESSURE			
DELTA P			
STEAM GENERATOR BLOWDOWN ENTHALPY			
PRESSURE			
BIAS VALUES			
STEAM PRESSURE	ENTHALPY		
SG BLOWDOWN	LIQUID ENTHALPY		
SG BLOWDOWN	LIQUID DENSITY		
POWER BIAS TOTAL VALUE			
*, ** INDICATE SETS OF DEPENDENT PARAMETERS			
4 LOOP UNCERTAINTY	(WITHOUT BIAS VALUES)		
4 LOOP UNCERTAINTY	(WITH BIAS VALUES)		0.58 % POWER

*** Effects are included in the feedwater flow uncertainty provided by Caldon

4. REFERENCES

4.1 WCAP-14738, Rev.0, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Tennessee Valley Authority Watts Bar Unit 1"

4.2 TVA letter W-7499, March 22, 1999.

4.3 TVA letter W-7517, March 14, 2000.

4.4 TVA letter W-7523, May 9, 2000.

ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1 - DOCKET NO. 390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-06

APPLICABILITY OF COMANCHE PEAK UNIT 2 RAI QUESTIONS TO WBN UPRATE

(NON-PROPRIETARY)

TVA has addressed NRC Staff written questions [Request for additional information (RAIs)] raised in the licensing process for the power uprate license amendment granted to TU for Comanche Peak Unit 2. This information is incorporated into the WBN license amendment request where practical. The RAI questions were taken from TU Electric transmittals to the NRC. The reference number of each RAI question and the associated TU Electric transmittal letter have been provided for cross reference. In addition, TVA has addressed seven questions associated with the Caldon Topical Report (TR-ER80P) which TVA considered were plant specific (TXX-98274). A list of the subject TU Electric transmittal letters and the associated responses is provided below:

TXX-99105	-	April 23, 1999	
TXX-99115	-	May 14, 1999	(Attachments 3, 6, and 7)
TXX-99195	-	August 13, 1999	
TXX-99164	-	July 9, 1999	
TXX-99203	-	August 25, 1999	
TXX-98274	-	December 17, 1998	(Response to selected questions)

Question 1 (TXX-99105):

Provide a discussion that addresses the impact of the proposed power uprate on the load, voltage, and short circuit values for all levels of the station auxiliary electrical distribution system. Include in this discussion any impact on the direct current power systems.

Response:

Refer to Enclosure 1, Section III.8.

Question 2 (TXX-99105):

For the power uprated conditions, discuss environmental qualification for the safety related electrical equipment located in harsh environmental areas. For this safety-related electrical equipment,

address the continued environmental qualification and the process for establishing qualification for any increased temperature, pressure, humidity, and radiation values.

Response:

The normal environments for the plant buildings were assessed. The 1.4% uprate has an insignificant effect on process fluid temperatures in the auxiliary and control buildings. With the exception of the main feedwater, the increase in the heat loads is caused by the increase in the decay heat load as it is transferred to the Component Cooling System and Emergency Raw Water Cooling System. The increase in these system temperatures has been evaluated and found to have an insignificant impact. The main feedwater temperature is changing by approximately 1.8°F with the Steam Generators at the maximum plugged tubes of 10%. This small change in fluid temperatures has an insignificant affect on the area temperatures. Similar conclusions were reached following the evaluations of the normal environmental conditions in the containment building.

The post-accident thermal environmental parameters were generated from computer models of the building structures that calculate the environment created by mass and energy releases during postulated pipe breaks. Evaluations concluded that through the use of the reduced 0.6% power calorimetric uncertainty to offset the 1.4% increase in reactor power, the existing mass and energy releases used in the environmental analyses for both inside and outside containment would remain valid. Because the mass and energy releases are not changed, the resulting environments are also unchanged. Therefore, the 1.4% power uprate has no impact on the WBN Unit 1 non- radiological equipment qualification program.

Generally, postulated radiation doses impacting equipment qualification depend primarily on post-accident contributions. However, normal-operating dose rate contributions are included in the design basis calculations. These normal-operating contributions are, in all cases, based on source terms which were originally generated for a power level of 104.5% RTP (i.e., 3565 MWt). Therefore, in regard to cases where normal operating equipment qualification dose rate contributions may be significant, it can safely be concluded that a power uprating of 1.4% would not cause dose rates or integrated doses to exceed design basis values.

The effects of post-accident radiological consequences on equipment qualification were also evaluated. The source term used in the original analyses was generated for operation at a thermal power of 3565 MWt. Revised core fission product inventory calculations were performed; it was concluded that the original source term remains bounding. Based on the revised core fission product inventory, the post-accident gamma source strengths for some energies were found to slightly increase as a result of the power uprate; however, when applied in specific dose rate computations, it was shown that the accumulated doses at all times remain lower than current design-basis values. The current design basis was performed in accordance with RG 1.49 which requires the normal power level to be 1.02% of the licensed power. Therefore, it was concluded that all doses used for equipment qualification remain within existing design basis values.

In summary, the 1.4% thermal power uprate has a negligible effect on normal environmental conditions and no effect on the environmental conditions currently used for equipment qualification.

Question 3 (TXX-99105):

Discuss and verify the assumptions for the station blackout analysis are valid for the power uprate conditions, particularly as they relate to issues such as the heat-up analysis, equipment operability, and battery capacity.

Response:

Equipment

To provide for an orderly and safe cooldown of the unit during and following a Station Blackout (SBO) event, the following conditions must be met:

The turbine driven auxiliary feedwater pump must operate to provide feedwater to the steam generators (SGs), a slight repositioning of the discharge valves (air operated) may be necessary, the main steam safety valves (MSSVs) are utilized to relieve steam generator secondary side pressure to maintain hot standby conditions, the SG power operated relief valves (PORVs) must cycle open to relieve steam for unit cooldown (after 4 hour SBO event), and an adequate supply of water from the condensate storage tank must be available to maintain adequate water level in the steam generators.

Control power from batteries has been provided to the turbine driven auxiliary feedwater pump for operation during an SBO. The turbine is supplied steam via the number 1 or 4 SG. A flow path is ensured from at least one of the two steam generators for pump operation.

The MSSVs have been evaluated and found acceptable - see Enclosure 1, Section III.4.2.1.

Specific air operated valves in the main steam system and the auxiliary feedwater system must be able to be operated from accumulators that have sufficient capacity to cycle the valves as needed during the controlled unit cooldown. In each case, the required number of valve cycles was established independent of and was determined to be reasonably insensitive to the actual power level. Nitrogen bottles have been included with each of these valves to meet operational requirements during the SBO event. The accumulator (i.e., nitrogen bottles) sizes are therefore sufficient to provide a safe cooldown during a SBO event.

An evaluation was performed in which it was concluded that the current minimum available condensate inventory in the condensate storage tank is sufficient for the 1.4% uprate condition (see Enclosure 1, Section III.4.2.4).

The existing calculations used to demonstrate the capability to withstand an SBO event of four hours duration without uncovering the core were reviewed for the 1.4% uprate conditions. The later stages of the existing analysis credit operator action to maintain the RCS temperature and pressure below specified limits; the SG PORVs are used to accomplish this action. The capacity of the SG PORVs was evaluated and determined to be sufficient to accommodate the 1.4% uprated condition (see Enclosure 1, Section III.4.2.1). The conclusions of the calculation remain valid, i.e., the time to uncover the core following a SBO event is greater than four hours.

Environmental

The existing loss of ventilation analyses for an SBO at Watts Bar is a 4 hour transient. The SBO room temperatures in vital areas were calculated using transient heatup computer models. The temperatures identified were the peak temperatures calculated for the 4 hour coping period. Equipment operability was assessed at those peak temperatures and no required operations were compromised by overheating.

The containment environment during a 4 hour SBO event is significantly less than the thermal profiles considered for LOCA/MSLB events. A small increase in decay heat and initial process temperatures cannot result in a change of such magnitude that the calculated LOCA/MSLB environment will be exceeded. Therefore, it was concluded that a small change in RCS temperature, decay heat, main steam and feedwater temperatures would have no effect on the equipment as evaluated for the SBO event.

The primary heat loads in the main steam and feedwater piping areas are from the main steam and feedwater piping. The power uprate results in a lower operating steam temperature at full load and no change to the no-load steam temperature. The slight increase in feedwater temperature realized from the 1.4% uprate would be insignificant during an SBO since feedwater heating would be terminated upon turbine trip.

The primary heat load in the turbine-driven auxiliary feedwater pump room is from the main steam piping feeding the turbine and the turbine casing. The power uprate results in a lower operating steam temperature and no change to the no-load steam temperature. Therefore, the current heat load resulting from the main steam lines bound the expected heat loads following the 1.4% uprate.

Based on the preceding discussions, it is concluded that the small changes in main steam and feedwater temperatures do not adversely impact the environment and equipment previously evaluated for the SBO event.

Battery Capacity

As a result of this uprate, no ac or dc auxiliary load ratings are expected to change, and the loads are not expected to experience additional demands above their ratings. Since the plant auxiliary ac/dc electrical load will not change, there is no impact on the station battery capacity due to the 1.4% uprate. See Enclosure 1, Section III.8 for additional information.

Question 4 (TXX-99105):

Provide a discussion addressing the impact of the CPSES Unit 2 power uprate on the turbine/generator, isophase bus, main transformers, and switchyards. Address in detail any non hardware changes for these items as a result of the CPSES Unit 2 power uprate.

Response:

Refer to Enclosure 1, Section III.8.

Question 5 (TXX-99105):

Discuss the impact of the CPSES Unit 2 power uprate electrical conditions on the current grid stability and reliability analysis. Describe in this discussion, how the station continues to be in conformance with General Design Criterion 17 with CPSES Unit 2 at the power uprated electrical conditions.

Response:

Refer to Enclosure 1, Section III.8.

Question 6 (TXX-99105):

Provide a pressurized thermal shock evaluation for the CPSES Unit 2 reactor vessel before implementing the power uprate and after implementing the power uprate.

Response:

Refer to Enclosure 1, Section III.5.1 for this evaluation. The evaluation concluded that the existing RT_{PTS} values remain valid and conservative.

Question 7 (TXX-99105):

What is the calculated end-of-life fluence in the current vessel design of CPSES Unit 2? What is the expected fluence for pressurized thermal shock with the revised design conditions/power uprate for CPSES Unit 2?

Response:

Refer to Enclosure 1, Section III.5.1.2 The evaluation concluded that the existing fast neutron data used in the reactor vessel design remains bounding for the 1.4% uprated conditions.

Question 8 (TXX-99105):

Does the power uprate for CPSES Unit 2 change the cold leg temperature? If so, please provide details.

Response:

Refer to Enclosure 1, Section III.2 for the change in $T_{cold,}$ and Section III.5.1.2 for an evaluation confirming that the T_{cold} is within the range assumed in the development of the equations and tables which form the bases for evaluating the neutron irradiation effects on vessel integrity.

Question 9 (TXX-99105):

Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will new operator actions be required?

Response:

The modest 1.4% power uprate is not expected to have any significant effect on the manner in which the operators control the plant, either during normal operations or transient conditions. The power uprate will lead to minor changes in several plant parameters. These parameters include, but are not limited to, the 100% value for Rated Thermal Power, Reactor Coolant System Delta Temperature, Main Turbine Impulse Pressure, Steam Generator Pressure and Main Feedwater and Steam Flows. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification, and will be included in Operator Training accordingly.

Question 10 (TXX-99105):

Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will effect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. Discuss why any reduced operator response times are needed. Discuss whether

any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to complete the required manual actions in the times allowed. Discuss results of simulator observations regarding operator response times for operator actions that are potentially sensitive to power uprate.

Response:

The modest 1.4% power uprate is not expected to have any significant effect on the manner in which the operators control the plant (including operator response times), either during normal operations or transient conditions. The power uprate will lead to minor changes in several plant parameters. These parameters include, but are not limited to, the 100% value for Rated Thermal Power, Reactor Coolant System Delta Temperature, Main Turbine Impulse Pressure, Steam Generator Pressure and Main Feedwater and Steam Flows. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification, and will be included in Operator Training accordingly.

Question 11 (TXX-99105):

Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g., normal range, marginal range, and out-or-tolerance range)? If changes will occur, discuss how they will be addressed.

Response:

No changes to control room annunciators, controls and displays are required as a direct result of the power uprate. When the power uprate is put in place, the Nuclear Instrumentation System will simply be adjusted to indicate the new 100% RTP in accordance with Technical Specification requirements and plant administrative controls. Because this power uprate is predicated on the availability of the LEFM, procedural guidance, supplemented by plant computer displays, will be developed to facilitate operation when the LEFM is unavailable. The plant computer system will provide an audible alarm for LEFM system failure or if maintenance is required. There are no new operator tasks required for safe shutdown by implementing this uprate. The operator's response has not changed. The reactor operators will be trained on the changes in a manner consistent with any other design modification.

Question 12 (TXX-99105):

Discuss all changes the power uprate will have on the Safety Parameter Display System (SPDS) and how they will be addressed.

Response:

The SPDS is unaffected by the proposed 1.4% increase in Reactor Thermal Power.

Question 13 (TXX-99105):

Describe all changes the power uprate will have on the operator training program and the plant simulator. Provide a copy of the post-modification test report (or test abstracts) to document and support the effectiveness of simulator changes as required by American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.5-1985, Section 5.4.1.

Specifically, please propose a license condition and/or commitment that stipulates the following:

(a) Provide classroom and simulator training on all changes that effect operator performance caused by the power uprate modification.

(b) Complete simulator changes that are consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be re-validated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." Simulator revalidation will include comparison of individual simulated systems and components and simulated integrated plant steady state and transient performance with reference plant responses using similar startup test procedures.

(c) Complete all control room and plant process computer system changes as a result of the power uprate.

(d) Modify operator training and the plant simulator, as required, to address all related issues and discrepancies that are identified during the startup testing program.

Response:

The modest 1.4% power uprate is not expected to have any significant effect on the manner in which the operators control the plant, either during normal operations or transient conditions. The power uprate will lead to minor changes in several plant parameters. These parameters include, but are not limited to, the 100% value for Rated Thermal Power, Reactor Coolant System Delta Temperature, Main Turbine Impulse Pressure, Steam Generator Pressure and Main Feedwater and Steam Flows. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification, and will be included in Operator Training accordingly.

In addition, the modest 1.4% power uprate is not expected to have a significant effect on any simulated systems. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification, and will be tested and documented accordingly. The WBN Simulator will be modified to match predicted plant values for 101.4% rated power. Following plant implementation, startup and operation at the uprated power, plant data will be collected and incorporated as the reference plant data for Simulator Steady State Performance Tests in accordance with the Simulator Certification annual testing program.

Question 14 (TXX-99105):

The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurement and plant operation.

Response:

New procedures for maintenance and calibration of the LEFM system will be developed per the design control process based on the vendor's recommendations.

Current Operations procedures are used to perform a calorimetric heat balance measurement for the purpose of calibrating the Power Range NIS channels. Contingencies and instructions will be added to the procedure in the event that the LEFM system becomes unavailable. This procedure will be revised per the design change control process to incorporate the requirements for the new LEFM system. In addition, more formal guidance, including routine surveillance requirement(s) for the LEFM and appropriate contingency actions, will be provided in the Technical Requirements Manual (TRM). Refer to response for Question 2 (TXX-99203) response for additional information.

Question 15 (TXX-99105):

For plants that currently have LEFMs installed, the licensee should provide an evaluation of the operational and maintenance history of the installed installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Response:

The LEFM 8300 strap-on system that is currently installed at WBN Unit 1 is only used as a basis for determining the correction factor for feedwater venturi fouling. The existing LEFM 8300 strap-on system is not as accurate as the new LEFM system and therefore will not be used as a basis for the 1.4% uprate. A complete new system will be installed at WBN that is bounded by the analysis and assumptions set forth in the Topical Report ER-80P. The new LEFM system is the same LEFM system that formed the basis of the analysis in the Topical Report. Commissioning of the system will be completed following the installation and prior to the uprate that will document that the new system is bounded by the Topical Report. This documentation will be available for inspection.

Question 16 (TXX-99105):

The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response:

The methodology used to calculate the combined feedwater mass flow and feedwater temperature uncertainty for the improved LEFM is exactly the same as the methodology presented in the Topical Report ER-80P. This value is then utilized to calculate the total power measurement uncertainty described in Enclosure 1, Section III.6.7 and Enclosure 4.

Question 17 (TXX-99105):

Licensees for plant installations where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), should provide additional justification for use. This justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response:

The LEFM system to be installed at Watts Bar Unit 1 will be calibrated to a site specific piping configuration prior to installation. The results of the calibration will provide a meter factor representative of the plant specific configuration. In addition, the accuracy with which the meter factor is determined will be incorporated into the uncertainty analysis of record for the Watts Bar LEFM system. Therefore, additional justification for use will not be required.

Question 18 (TXX-99105):

Based on the above, the staff finds that feedwater flow measurement using the LEFM can provide a thermal power measurement that will remain bounding within an uncertainty of 1% of rated thermal power.

This is premised on the assumption that no additional uncertainties beyond those included in Topical Report ER-80P are assumed to be included in the 10 CFR Part 50, Appendix K 102% thermal power margin requirement.

Response:

Refer to Enclosure 1, Section III.6.7 and Enclosure 4 for a discussion of the power measurement uncertainty calculation.

Question 19 (TXX-99105):

The amendment request proposes to reduce the margin for assumed power level for non-LOCA accident and transient analysis on the same basis as the proposed exemption to the Appendix K ECCS evaluation requirement. Staff consideration of the related Appendix K exemption request was in part based on the premise that the power level requirement is one of several conservative features that, taken together, provide substantial conservatism in ECCS analyses.

Justify the proposed margin reduction for non-LOCA analyses that currently assume 102% power. The justification should include a quantitative or qualitative discussion of conservative analysis assumptions for the non-LOCA accidents and transients and the safety margin they provide relative to the power level margin assumption.

Response:

Refer to Enclosure 1, Section III.6 For analyses that employ a 2% uncertainty, additional discussion regarding the conservative analysis assumptions has been provided.

Question 20 (TXX-99105):

Increasing licensed power level would result in an increased heat source that could affect the progression of certain accidents. Discuss the potential impact of plant operation at the higher proposed power level on ATWS progression, containment integrity analyses, and on overall IPE results.

Response:

Section 7.7.1.12 of the Watts Bar FSAR documents that TVA installed the AMSAC system to comply with the ATWS rule. Unlike CPSES, the Watts Bar FSAR does not include a section addressing an ATWS analysis. The 1.4% increase in core power will not affect the ability of the AMSAC to perform its intended functions stated in FSAR Section 7.7.1.12.

Refer to Enclosure 1, Section III.6.2 and III.6.3 for an evaluation of the mass and energy releases used as input to the containment integrity analysis. Since the current mass and energy releases remained bounding for the 1.4% uprate conditions, it was not necessary to re-perform the containment integrity analysis.

The WBN IPE model includes both level 1 systems analysis and level 2 containment analysis. The systems analysis is not impacted by the additional hardware installed for the LEFM project. Success criteria for the level 1 analysis would be minimally impacted by the minor power uprate. This is evidenced by the Best Estimate Large Break LOCA analysis which demonstrated a minor impact to peak clad temperatures (12F) from the power uprate. Many of the IPE success criteria are based on traditional analyses (e.g., 3 out of 4 accumulators for LBLOCA success is derived from the design basis analysis). The power uprate would also have minor impacts on the containment analysis. A slight decrease in time for onset of the hydrogen/zirc water reaction, earlier time to rupture for the pressurizer relief tank rupture disk, for transients, etc., would be expected although not significant in terms of damage progression. An IPE re-analysis is therefore not warranted prior to the next IPE update.

Question 21 (TXX-99105):

Discuss the impact on LOCA and non-LOCA analysis results (e.g., main steam line break) of the revised values for RCP heat addition and RCS flow rate included in the amendment request.

Response:

The license amendment does not involve a change to the design basis RCS flow rate. The revised RCP heat addition values were evaluated per the WBN 10CFR50.59 program as part of an upgrade performed prior to WBN Cycle 2. It was determined that no impact to nuclear safety existed and that no unreviewed safety questions were created by more accurately accounting for the net heat input from the RCPs. The heat input estimate was revised from a nominal 14 MWt to 16 MWt for operation following Cycle 1.

Question 22 (TXX-99105):

Provide the detailed calculational basis to substantiate the statement made in the amendment request that a 10-percent SG tube plugging level supports a peak plugging level of 15% in any one SG, provided that the average level of plugging of all four SGs is no greater than 10 percent. Explain the difference between the plugging level used in the analysis discussed in the amendment request and the plugging level assumed in the current LOCA analysis?

Response:

Refer to Enclosure 1, Section III.2. The analyzed SGTP level is 10% and does not presently address the use of asymmetric tube plugging levels above 10%.

Question 23 (TXX-99105):

Plant response to SGTR and other events depends on SG atmospheric relief valve operation. Reactor operation at higher power levels may cause these valves to operate more often in the event of certain events, thereby affecting their reliability. Discuss the effects of operation at the proposed new power level on the possible increased challenge to these valves and their expected failure frequency during a SGTR event (and other events requiring their operation).

Response:

While it is true that transients initiated from a higher power level may present more challenges to the ARVs, the frequency of such challenges is not considered to be significant. The proposed increase in reactor power is very modest. The capacity and reliability of the Steam Dump System are such that the ARVs are generally not anticipated to be operated any more frequently than they are currently cycled.

Question 24 (TXX-99105):

When considered in terms of core power, the proposed changes in power range neutron flux, and overpower N-16 nominal and allowable reactor power trip levels appear slightly non-conservative. Explain the basis for the proposed revision to the N-16 overpower and power range neutron flux trip set points given in the amendment request. Provide justification for the apparently non-conservative set point changes.

Response:

The Watts Bar design does not employ an N-16 reactor trip function. Also, no changes to the reactor trip functions have been proposed and the current values have been preserved in the applicable safety analyses.

Question 25 (TXX-99105):

The N-16 overtemperature trip setpoint was not changed in the amendment request, based on the statement that it was previously analyzed at the power level requested in the proposed amendment. Confirm that the other proposed changes to plant parameters such as RCS flow and coolant temperatures do not result in a change to the N-

16 overtemperature trip setpoint. Explain how the proposed changes in core flow rate and coolant temperatures affect the calculation of the N-16 overtemperature trip setpoint.

Response:

The Watts Bar design does not employ an N-16 reactor trip function. Refer to Enclosure 1, Sections III.6.6 - which confirms that no changes were required to the reactor trip and engineered safety feature actuation system setpoints as a result of the slight changes to the RCS temperatures for the 1.4% uprate conditions.

Question 1 (TXX-99115 - Attachment 3):

Part 1) In Attachment 2 of the submittal, the licensee states that the Balance of Plant (BOP) fluid systems were reviewed for compliance with the Westinghouse Nuclear Steam Supply System (NSSS)/BOP Interface guidelines. How does the power uprate affect the design basis of the following systems: main steam, steam dump system, feedwater and condensate system, and auxiliary feedwater system?

Part 2) In Section C of Attachment 2, the licensee states that design documentation and instrumentation and control setpoint changes are required. Which, if any, of the following systems and items would exceed the design basis: circulating water, turbine plant cooling, spent fuel pool cooling, component cooling, station service water, station blackout, spent fuel storage, HVAC systems, turbine/generator? In any, provide the new limits and explain why the new design basis is acceptable.

Part 3) In Table IV-1 of Attachment 2, "NSSS Revised Design Parameters," the licensee describes three limiting cases. Explain which case(s) was (were) used in the evaluation of the above listed BOP systems and the NSSS/BOP interfaces. If only one was used, explain why it provides conservative results.

Response to Part 1

Refer to Enclosure 1, Section III.4.2.

Response to Part 2

The existing design basis is not exceeded for any BOP components for the power uprate. The secondary side is basically a design for a turbine/generator that is guaranteed for 105% of RTP (3411MWt). One supporting systems may require rescaling - the impulse pressure for the turbine/generator will increase by approximately 15 psig. This is not a design basis concern or challenge.

Response to Part 3

Refer to Enclosure 1, Section III.2.3 and Table 2-1 for the bounding 1.4% uprate parameters used in this evaluation.

Question 2 (TXX-99115 - Attachment 3):

Solid, liquid, and gaseous radioactive waste activity are influenced by the reactor coolant activity which is a function of the reactor core power. What is the impact of these systems by the increase in power?

Response:

Offsite doses from normal effluent releases remain below referenced bounding results, which are within 10CFR50 Appendix I limits. Further, the capabilities of the plant radioactive waste processing systems were evaluated to assess the effects of the 1.4% power uprate. Thus, the capability to process and store effluents will not be significantly impacted by the 1.4% uprate.

The evaluation of the Gaseous Waste Processing System is discussed in Enclosure 1, Section III.9.

The solid waste management and liquid waste processing systems are designed to control, collect, process, store and dispose of radioactive wastes due to normal operation including anticipated operational transients. Operation of these systems are primarily influenced by the volume of waste processed, which is not expected to change as a result of the 1.4% uprate condition. Thus, the capability of the solid waste management and liquid waste processing systems are not significantly impacted by the 1.4% uprate.

In summary, the 1.4% power uprate has no significant effect on any of the waste subsystems or components of these subsystems. Because these systems are typically operated in a batch mode, the only potential effect is a slight increase in the frequency at which the batches may be processed. These systems continue to meet current design bases.

Question 3 (TXX-99115 - Attachment 3):

Discuss why the current containment analysis remains appropriate for use at power uprate conditions.

Response:

Refer to Enclosure 1, Sections III.6.2 and III.6.3.

Question 1 (TXX-99115 - Attachment 6):

In regard to Section B.4 of Attachment 2 to the reference transmittal, provide the maximum-calculated stress and cumulative fatigue usage factor (CUF) at the critical locations of the RPV and internals (such as RPV nozzles, lower and core plates, core barrel, baffle/barrel, control rod drive mechanism, and fuel assembly, etc.), the allowable code limits, the Code and Code edition used in the evaluation for the power uprate. If different from the Code of Record, provide the necessary justification. Also, provide an assessment of flow-induced vibration of the reactor internal components due to power uprate.

Response:

As noted in Enclosure 1, Section III.2, the 1.4% uprate conditions resulted in very small changes to the NSSS design conditions (e.g. - T_{cold} and T_{hot} changed by 0.4 °F). In addition, Section III.3.1 indicates that there were no changes required for the NSSS design transients. As a result, in most cases, an evaluation was performed to confirm that the existing fatigue usage factors and maximum stress intensities were either negligibly affected or bounded by margin in the existing calculations. Thus, in most cases, revised fatigue usage factors and stress intensities did not need to be calculated.

Refer to Enclosure 1, Section III.5.1.1 for a discussion of the RPV structural evaluation. The Code version used in the evaluation is the 1971 Edition of Section III of the ASME Code through the Winter 1971 Addenda, which is the same as the current Code of record for these components.

Refer to Enclosure 1, Section III.5.2.3 for a discussion of the RV Internals evaluation. The reactor internals are not licensed to a Code version and were originally designed based on sound engineering practice.

Refer to Enclosure 1, Section III.5.3 for a discussion of the CRDM evaluation. The Code version is the same as the current Code of record.

The flow-induced vibration analysis for the internals was unaffected since the power uprate did not require a change to the plant mechanical design flow.

On a cycle-specific basis, the mechanical design of the fuel assemblies is verified to meet all current design criteria. The fuel vendor performs the required analyses using methods specific to the fuel type. This evaluation is documented in the cycle-specific Reload Safety Evaluation, performed in accordance with 10 CFR 50.59.

Question 2 (TXX-99115 - Attachment 6):

On page 22 of Attachment 2 to the reference transmittal, provide the methodology and assumptions used for evaluating the reactor coolant piping systems, equipment nozzles, and supports for the increased hot leg and cold leg temperatures, increased dynamic hydraulic forcing functions, and the affected design transients due to the power uprate, as stated in the transmittal. Also, provide the calculated maximum stress, critical locations, allowable stress limits, and the Code and Code edition used in the evaluation for the power uprate.

Response:

Refer to Enclosure 1, Section III.5.4 for a discussion of the RCL piping related evaluations performed for the 1.4% uprate. Enclosure 1, Section III.3.1 indicates that none of the NSSS design transients, which include those for the reactor coolant system piping and nozzles, are affected by the uprate conditions. Section III.6.4 indicates that the current LOCA hydraulic forcing functions remained bounding for the uprate conditions.

For the Reactor Coolant System (RCS) temperature changes, an evaluation demonstrated that the current analyses for the reactor coolant loop piping, primary equipment nozzles, primary equipment supports and pressurizer surge line piping remained bounding for the uprated conditions due to the conservative nature of inputs for the current analyses. Thus, there were no new calculated maximum stresses, critical locations, and loads.

In addition to the above, an evaluation was performed to demonstrate that the existing fatigue usage factors for the reactor coolant loop piping, nozzles and auxiliary lines remained bounding. The uprated design conditions only impacted the starting and ending temperatures associated with cooldown and heat-up events. The potential slight increase in fatigue was offset by existing margin in the current analysis.

Furthermore, the evaluation performed to address the effects on the pressurizer surge line stratification analysis included a review of the fatigue analysis and the stratification loadings that were transmitted to the pressurizer nozzle from the surge line piping. The potential load increases (from the increased T_{hot}) were determined to be bounded by the current analysis since the analysis used conservative envelopes that lumped various transients under a reduced number of bounding thermal cases. Therefore, the current analysis results remain unchanged for the 1.4 % uprate conditions.

Finally, by taking credit for the conservative nature of the existing inputs, it was not necessary to re-calculate the stresses and CUF values in accordance with the applicable Code versions. Thus, it was not necessary to change or review the existing Code versions.

Question 3 (TXX-99115 - Attachment 6):

Were the analytical computer codes used in the power uprate evaluation different from those used in the original design-basis analyses? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

Response:

The analytical computer codes used in the Westinghouse NSSS and BOP analysis are either the latest revisions of the computer codes presently described in the FSAR and/or are the same computer codes used in the original design basis analyses.

Question 4 (TXX-99115 - Attachment 6):

In reference to the reactor coolant pump (RCP) structural analysis on page 23 of Attachment 2 to the reference transmittal, you stated that "an analysis was performed to determine the impact of the revised design conditions on the stresses and fatigue usage of the RCP ("CRDM" stated in your report should be "RCP") components and the results indicated that the stress and fatigue usage remain within ASME Code limits. Describe the analysis methodology and assumptions (if any), used for evaluating RCP. Also provide the maximum-calculated stress and CUF for the RCP, the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide justification.

Response:

Refer to Enclosure 1, Section III.5.5.1. The SG outlet temperature only changed by 0.5°F for the 1.4% uprate conditions, which was judged to have a minimal effect on the current stress and fatigue analyses. As a result, it was concluded that the stress intensities remained below the applicable limits and the fatigue usage was less than 1.0. The Code version used in the analysis is the same as the Code of record.

Question 5 (TXX-99115 - Attachment 6):

On page 23 of Attachment 2 to the reference transmittal, provide a comparison of the design parameters (i.e., steam pressure, temperature, primary-to-secondary pressure differential, etc.) and transients for the steam generators (SGs) Model D5 against the power uprate condition. Also, provide the maximum calculated stress and CUF for the critical locations (such as the vessel shell, secondary manway bolts, and nozzles), the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different

from the Code of record provide justifications. Also, provide an evaluation on the flow-induced vibration of the SG U-bends tubes due to power uprate regarding the analysis methodology, vibration level, computer codes used in the analysis and the calculated cross flow velocity.

Response:

Table 1 below identifies the CUF and maximum stresses for the limiting steam generator locations. The Code version used in the evaluation is the 1971 Edition of Section III of the ASME Code through the Summer 1972 Addendum, which is the same as the current Code of record for these components.

An evaluation was performed to confirm that no additional tubes required plugging to prevent high cycle fatigue caused by U-bend vibration. This evaluation used a one dimensional relative stability ratio (RSR). The RSR is the stability ratio of a tube at the 1.4 % uprated conditions relative to the conditions used for the analysis at the current design conditions.

First, the maximum allowable RSR was calculated for the most susceptible tubes in the plant. If the actual RSR for the 1.4% power uprate is below this value for any given tube, the tube will not be susceptible to high cycle fatigue. These values were calculated for the most susceptible tubes in each generator. All other tubes have higher allowable RSRs.

The one-dimensional RSR is affected by changes in steam flow, circulation ratio, steam pressure and saturation temperature resulting from the 1.4% power uprate. The maximum RSR occurs, as expected, for the 1.4% uprate with 10% plugging. This value is lower than all the allowable RSR values for the most susceptible tubes. Therefore, none of the tubes are expected to become susceptible to high cycle fatigue as a result of the 1.4% power uprating.

Table 1: Maximum Stress Intensity Range / Allowable, and Cumulative Fatigue Usage Factors for Normal / Upset Conditions

Component	Section Or location	Maximum Stress Range/ Allowable (Current)	Maximum Stress Range/ Allowable (Uprated)	Fatigue Usage Factor (Current)	Fatigue Usage Factor (Uprated)
Divider Plate	Divider Plate to T/S Jct.	[ ] +a,c	[ ] +a,c	[ ] +a,c	[ ] +a,c
Tubesheet & Shell Junction	Center of Tube-sheet	[ ] +a,c	[ ] +a,c	[ ] +a,c	[ ] +a,c
Tube to Tubesheet Weld	Weld Horiz. Section	[ ] +a,c	[ ] +a,c	[ ] +a,c	[ ] +a,c
Main Feedwater Nozzle	Nozzle Knuckle	[ ] +a,c	[ ] +a,c	[ ] +a,c	[ ] +a,c
Auxiliary Feedwater Nozzle	Liner to Nozzle Weld	[*] +a,c	[*] +a,c	[ ] +a,c	[ ] +a,c

* The maximum stress intensity range calculated elastically exceeded the $3S_m$ limit and a simplified elastic - plastic analysis was performed per NB 3228.3 of the ASME Code. The values reported correspond to the stress intensity ranges with the exclusion of thermal bending stresses.

Question 6 (TXX-99115 - Attachment 6):

On page 25 of Attachment 2 to the reference transmittal, you stated that the pressurizer structural evaluation was performed by comparing the key inputs in the current pressurizer stress report with the revised design conditions in Table IV-1 and that the results indicated that the design condition used in the current analysis remain bounding for the revised design conditions. Provide a comparison of the design parameters (i.e., RCS pressure hot let temperature, cold leg temperature, temperature differential, etc.), the stratification and cyclic design transients for the CPSES pressurizer against the power uprate condition. Also, provide the maximum calculated stress and CUF at the critical locations (such as surge nozzle, skirt support, spray

nozzle, safety and relief nozzle, upper head/upper shell and instrument nozzle) of the pressurizer, the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide justification.

Response:

The limiting locations for the pressurizer structural analysis are the surge nozzle, the spray nozzle, and the upper shell at the point of spray impingement. The pressurizer structural evaluation was performed by comparing the key inputs in the current pressurizer stress report with the uprated design conditions. The uprated design conditions affecting the pressurizer are T_{COLD} and T_{HOT}. Table 2 below compares the current and uprated T_{COLD} and T_{HOT} with the values used in the current stress report. This comparison demonstrates that the current analysis values for temperature changes in the stress report remain bounding for the uprated power conditions. Thus, the current analysis remains bounding. Enclosure 1, Section III.3.1 indicates that it was not necessary to modify any of the current primary transients for the uprated conditions.

The Code version used in the evaluation is the 1971 Edition of Section III of the ASME Code through Summer 1971 Addenda, which is the same as the current Code of record for this component.

Since the existing surge line piping loads are bounding for stratification at the uprated conditions, there were no changes required for the piping loads for the pressurizer surge nozzle. Thus, the current surge nozzle analysis remains bounding for the stratification the uprated conditions.

Table 2

Comparison of Uprated Design Conditions to Current Analysis Values for the Pressurizer

Parameter (°F)	Current Design (°F)	Revised Design (°F)	Current Analysis Value (°F)
$T_{PRESSURIZER}$	653	653	[ ] +a,c
T_{HOT}	617.3	619.1	---
T_{COLD}	559.1	557.3	---
$\Delta T_{HOT} = T_{PRESSURIZER} - T_{HOT}$	35.7	33.9	Bounded by Current Design
$\Delta T_{COLD} = T_{PRESSURIZER} - T_{COLD}$	93.9	95.7	[ ] +a,c

Question 7 (TXX-99115 - Attachment 6):

Discuss the operability of safety-related mechanical components (i.e., valves and pumps) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which operability at the uprated power level could not be confirmed.

Response:

The safety-related pumps are designed for the 10CFR50 App "K" required power level (102%). Refer to Enclosure 1, Section III.4.1 for a discussion concerning the 1.4% uprate effect on systems for residual heat removal, chemical and volume control, and safety injection. (Also see Enclosure 1, Section III.5.8) The flow requirements of the Auxiliary Feedwater Pumps (motor and turbine driven) are not effected by this modest power uprate. The steam generator MSSV setpoints will also remain the same; therefore, the steam generator pressures at which the equipment is required to pump against will be unchanged.

The air operated valves that are required to be operable are powered from the accumulators and are unchanged from present design limits. For further discussion of BOP valves, refer to Enclosure 1, Section III.4.2.1 and III.4.2.3.

No changes to the TVA procedures that address the MOV program are required as a result of this 1.4% power increase. The methodology used to document the requirements of the MOVATS and MOV program are standard for the three nuclear sites at TVA. Maximum differential temperatures and pressures (design) are used for sizing requirements for normal operation and worst case conditions are used for the accident required actions. As these conditions bound the power increase conditions, no reduction in margin of safety results from the power increase.

Question 8 (TXX-99115 - Attachment 6):

(This question has been subdivided in order to provide clearer responses.)

a) In reference to Section C on page 26 of Attachment 2 to the reference transmittal, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate.

b) Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports.

c) Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code editions used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences.

d) Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

Response:

Refer to Enclosure 1, Section III.7, "Balance of Plant" for this response.

Question 9 (TXX-99115 - Attachment 6):

Discuss the potential for flow-induced vibration in the heat exchangers following the power uprate. Provide a summary of evaluation for power uprate effects on the high energy line break analysis, jet impingement and pipe whip loads for the power uprate conditions.

Response:

Flow-induced vibration potential is a function of the shell side flow rates (i.e., flow velocities) in the various NSSS heat exchangers. Shell side flow rates in these heat exchangers are not significantly affected by the uprating. In addition, all of these heat exchangers have been designed to withstand up to 2 times the shell side design flow without encountering damaging tube vibrations. Therefore, flow-induced vibration is not a concern following the uprating.
The only area for potential problems with flow induced vibration is the Steam Generator - due to the increased steam flow and the decreased pressure. The vendor has concluded that the tube lengths and spacers are adequate for the small increase in steam velocity on the shell side of the tubes and flow induced vibration is not a concern due to the power uprate. Other heat exchangers (i.e., feedwater heaters, main condenser, etc.) on the secondary side are bounded by their design conditions.

The primary side pressures and flows are not affected by the 1.4% uprate as discussed in Enclosure 1, Section III.2.3. Therefore, the current high energy line break analyses are bounding.

The high energy line breaks on the secondary side of the plant are discussed in Enclosure 1, Section III 6.2.

Question 1 (TXX-99115 - Attachment 7):

Provide a description, references, and standards to describe CPSES configuration management/procedures including software.

Response:

The LEFM system is designed as a Quality Related system for TVA-WBN and thus configuration management of the LEFM system is maintained by TVA Standard Programs and Processes (SPP)-9.0, "Engineering." The Software and Firmware Verification and Validation Report by Caldon is described in Topical Report ER-80P, Section 6.4, "Quality Measures in Design, Fabrication and Factory Acceptance Testing of the LEFM." TVA software control for LEFM is in accordance with SPP-2.6, "Computer Software Control."

Question 2 (TXX-99115 - Attachment 7):

In response to Question 16 the methodology used to calculate calorimetric uncertainty is referenced as ASME PTC 19.1 - 1985, Measurement Uncertainty and Is the same methodology as used to determine the uncertainty using the LEFM✓ system.

A review of the CPSES FSAR and TS shows, the following Information:

* **Chapter 15 Page 15.0-16. Section 15.0.7, Instrumentation Drift and Calorimetric errors - Power range neutron Flux" Is deleted but references Section 15.0.6, "Trip Setpoints and Time Delays to Trip Assumed in Accident Analysis" references Section 7.1.2.1.9 and the CPSES Technical Specifications. This references Westinghouse setpoint methodology. PTC 19 is not referenced.**

* **The CPSES FSAR references RG 1.105 and the Westinghouse setpoint methodology not PTC 19.**

* **The CPSES Bases B 3/4 2-11 DNB parameters references the RCS total flow uncertainty as 1.8%. The uncertainty is stated to be based on Westinghouse Revised Thermal Design Procedure which includes measurements of reactor power. The methodology used to develop the associated uncertainties and includes specific treatment of feedwater flow uncertainties. PTC 19 is not referenced.**

* **FSAR Page 4.4-37 Reference 85 lists "Improved Thermal Design Procedure" as the methodology used. PTC 19 is not referenced.**

Response:

Refer to response to Question 16 (TXX-99105).

Question 3 (TXX-99115 - Attachment 7):

For Question 17 provide a calibration report from a calibration lab with accuracy traceable to NIST that indicates the accuracy of the LEFM in fully conditioned flow. Additionally provide a test report from a calibration facility that shows the LEFM accuracy is unaffected by velocity profile changes including those based on piping geometry changes (reducers, header, elbows, etc.) such that it can be confirmed the LEFM is not sensitive to plant specific piping installation effects and that the calibration facility results are directly applicable to a plant specific installation.

Response:

The Watts Bar LEFM system will be calibrated in hydraulically similar piping at Alden Research Laboratories prior to installation at Watts Bar. The results from the calibration laboratory report will be directly applicable to the plant-specific installation and will be incorporated in the site-specific uncertainty analysis for the Watts Bar LEFM system. This analysis can be made available for NRC review.

Question 1 (TXX-99195):

Provide a comparison of the relevant acceptance criterion to the appropriate design limit (e.g., DNBR, RCS pressure) for each of the following safety analyses:

15.4.2	**Uncontrolled RCCA withdrawal from power**
15.4.7	**Misloaded fuel assembly**
15.4.8	**Rod Ejection**
15.4.3	**Dropped RCCA**

Response:

Refer to Enclosure 1, Section III.6.5 for a discussion of the evaluation performed for the uncontrolled RCCA withdrawal at power, rod ejection and dropped rod events. The Watts Bar FSAR does not include the misloaded fuel assembly event.

Question 2 (TXX-99195):

The topical report detailing the analysis of an inadvertent boron dilution event (RXE-91-002-A) indicates that the analysis assumed a power level of 100 percent. Discuss the sensitivity of the analysis results to initial power level. Summarize the methods and results of any supporting sensitivity analysis and provide references.

Response:

Refer to Enclosure 1, Section III.6.5.3 for a discussion of the boron dilution event.

Question 3 (TXX-99195):

Discuss the sensitivity of the analysis results to initial power level for the SG tube rupture event. Summarize the methods and results of any supporting sensitivity analysis and provide references.

Response:

Refer to Enclosure 1, Section III.6.1 for a discussion of the SGTR event.

Question 4 (TXX-99195):

CPSES technical specifications contain a surveillance requirement (3.3.1.2) requiring that power levels measured by nuclear instruments and by the N-16 monitoring system be checked to within 2% of the daily calorimetric. Explain why this surveillance requirement is not being modified to require that the readings be within 1% of the calorimetric.

Response:

WBNP does not have the N-16 monitoring system therefore this portion of the question is not applicable.

The uncertainty associated with the accuracy of the plant calorimetric measurement is considered in the plant safety analyses. It is this uncertainty that can be reduced through the use of the improved LEFM system.

Technical Specification Surveillance Requirement (SR) 3.3.1.2 is a requirement for the re-normalization of the Nuclear Instrumentation System (NIS) power range channels if the allowed deviation (±2% RTP) between the power calculated by the plant calorimetric measurement and the NIS indicated power is exceeded. This deviation is considered in the uncertainty analyses of those reactor trip functions that are based on the NIS power range channels.

Question 5 (TXX-99195):

In response to a previous request for additional information the revised overpower N-16 allowable value of 113.5% of rated thermal power was defended as having been derived based on WCAP-12123 methods. Provide the detailed calculation showing how the allowable value for the N-16 overpower trip was determined.

Response:

The Watts Bar design does not employ an N-16 overpower trip. Thus, this RAI would not apply to the Watts Bar 1.4% uprate submittal.

Question 1 (TXX-99164):

The licensee needs to evaluate the effects of the power uprate on the tube degradation mechanisms (present and potential) including wear.

Response:

Refer to Enclosure 1, Section III.5.6.5.

Question 2 (TXX-99164):

Discuss how steam generator tube inspection plan will be assessed to monitor potential tube degradation including wear. Will additional inspections be necessary? How will TXU Electric assess their inspection plans should new degradation mechanisms be discovered?

Response:

Watts Bar's current Steam Generator (SG) program follows the EPRI SG Inspection Guidelines. The modest 1.4% uprate will not require a change to the program. We currently inspect for all active and potential degradation. Higher temperatures will be considered in growth rate analyses. The pre-outage degradation assessment includes Watts Bar specific degradation as well as industry degradation. Based on condition monitoring and operational assessments of inspection results, expansion of inspection plans and repairs will be made. Degradation growth rate changes will be incorporated into the operational assessment associated with potential affects of the uprate.

Question 3 (TXX-99164):

The licensee needs to evaluate if the Technical Specification plugging limit of 40 percent through wall degradation is still adequate.

Response:

Refer to Enclosure 1, Section III.5.6.6.

Question 1 (TXX-99203):

In section 6 of the Caldon Topical, reference is made to use of the LEFM to calibrate the NIs. How does CPSES plan to use the LEFM and explain the relation of the LEFM as M&TE with regards to Appendix B.

Response:

The requirement in Technical Specification Surveillance (SR) 3.3.1.2 is to "adjust" the NIS power range channels if the absolute difference between the calorimetric heat balance calculation and the NIS power range channels output is greater than 2%. This requirement is further clarified in the associated Bases with the words "the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP". Using these guidelines, it is more correct to state that the NIS indication of reactor power is normalized, rather than calibrated, against the reactor power calculated with the LEFM-based secondary plant power calorimetric measurement. As such, the application of M&TE is not strictly appropriate.

The improved LEFM is included in the non-Appendix B Quality Assurance program as described in FSAR Section 17.2.

Question 2 (TXX-99203):

Page 5.5 of the Caldon Topical discusses the use of the LEFM to correct the Venturi measurement. Page 8 of the TXU license amendment request also discusses the use of the LEFM for providing correction for the venturi. What are CPSES plans when the LEFM is unavailable and the venturis are used for normalizing the NIs?

Response:

The referenced page of the Caldon Topical Report discusses use of the improved LEFM for correcting the venturi-based feedwater flow indication for effects such as fouling. As detailed in response to Question 29 (TXX-98274), WBN currently uses the LEFM 8300 strap-on system measurement of feedwater flow to correct fouling effects for the venturis. This correction is used for the power calorimetric measurements only. Refer to Question 29 for additional discussion on WBN plans to continue correcting for feedwater venturi fouling based on the improved LEFM System.

Through the use of the improved LEFM, the power calorimetric uncertainty is shown to be less than 0.6% RTP. However, this uncertainty calculation is not applicable to the case where the power calorimetric is based on venturi-based feedwater flow indication, even if the improved LEFM is used to correct the venturi-based feedwater flow indications for effects such as fouling.

WBNP will be operated in accordance with the safety analyses and the applicable power calorimetric uncertainty analysis. When the improved LEFM-based calorimetric measurement is available, the plant will be operated at a nominal core power of 3459 MWt. The reactor operators will be provided procedural guidance for those occasions when the improved LEFM is not available. As summarized below, for those instances a new section of the WBN Technical Requirements Manual (TRM) will specify the appropriate actions to be taken when the LEFM is unavailable.

The WBN TRM andother appropriate plant procedures will specify that if the LEFM becomes unavailable during the interval between daily performances of the calorimetric heat balance comparison with the NIS (Technical Specification Surveillance Requirement (SR) 3.3.1.2), plant operations may remain at a thermal power of 3459 MWt while continuing to use the power indications from the NIS power range channels. However, in order to remain in compliance with the bases for operation at a Rated Thermal Power of 3459 MWt, the LEFM system must be returned to service prior to the performance of SR 3.3.1.2. If the LEFM has not been returned to service prior to the performance of SR 3.3.1.2, the procedural guidance/TRM would require that the reactor power be reduced to, or maintained at, a power level of 3411 MWt. This power level is consistent with the uncertainty previously assumed for the venturi-based indication of feedwater flow. This power reduction is intended to be performed prior to SR 3.3.1.2 being performed. The surveillance would then be performed using the venturi-based feedwater flow indications in the case where the LEFM is unavailable. Once SR 3.3.1.2 is performed using the corrected venturi-based feedwater flow indications, the assumed power uncertainty is 2% RTP even though the actual uncertainty is much better than this. In order to maintain compliance with the safety analyses , it would be necessary to operate the plant at a maximum core thermal power of 3411 MWt until the LEFM is restored.Once LEFM is restored, performance of SR 3.3.1.2 is required using the LEFM indication of feedwater flow. Upon completion of SR 3.3.1.2, the plant could again be operated at 3459 MWt.

Question 3 (TXX-98274, TR-ER80P):

Describe how the LEFM is used in calorimetric power determinations.

Response:

See general description of LEFM (Enclosure 1, Section III.1) and responses to Questions 4 (TXX-99195), 1 and 2 (TXX-99203).

Question 5 (TXX-98274, TR-ER80P):

Who is responsible and how are Calibration, Maintenance, and Training performed and achieved?

Response:

The Verification Test of the LEFM spoolpiece is contracted by Caldon and performed at Alden laboratory before the installation in the main feedwater header at WBN. The requirements for installation are within WBN Design Change Notice (DCN) 50451. The software has provisions for on-line monitoring and diagnostics and will alert the operator if the system has failed or the performance of the system indicates a maintenance/alert condition. In that event, it may become necessary for maintenance to be performed. This necessary maintenance will be procedurally controlled.

Training on the operation and maintenance of the LEFM system is contractually provided by Caldon. Maintenance will be performed by WBN plant personnel per vendor recommendations contained in vendor supplied instructions and does not require any special skills that would be beyond that encompassed in the WBN I&C technician training program.

Question 6 (TXX-98274, TR-ER80P):

How will monitoring, verification, and error reporting be handled? Provide clarification (list) of Quality Control standards used by Caldon in the design and manufacturing of the LEFM. Provide clarification (list) as to the standards followed under Caldon's verification and validation program.

Response:

TVA believes this question was intended to mainly provide clarification to Caldon's Quality Control Program which TVA will not attempt to address in this response.

WBNP will include the LEFM in the calibration and maintenance program including the preventative maintenance program. The system will be monitored by the System Engineer for reliability. As a plant instrument, all equipment problems fall under the site work control process. All adverse conditions that are identified will be documented on a Problem Evaluation Report (PER) in accordance with the WBNP Corrective Action Program. WBNP has required Caldon to maintain the LEFM software under their V & V Program with requirements that Caldon notify WBNP of any deficiencies that could affect the design basis accuracy.

Question 10 (TXX-98274, TR-ER80P):

How does the LEFM uncertainty compare to the venturi uncertainty at Comanche Peak, in measuring reactor thermal power?

Response:

See Question 16 (TXX-99105) for response.

Question 29 (TXX-98274, TR-ER80P):

How is the LEFM used currently to provide correction factors to the venturis? Is the correction determined on the basis of the absolute accuracy or the repeatability of the LEFM?

Response:

The LEFM 8300 strap-on system that is currently installed at WBN Unit 1 is only used as a basis for determining the correction factor for feedwater venturi fouling. The correction is based on the absolute accuracy of the LEFM but a high degree of repeatability is also required. WBN plans to continue correcting for feedwater venturi fouling based on the improved LEFM System. This correction factor will be based on the improved accuracy of feedwater mass flow measurement.

Question 30 (TXX-98274, TR-ER80P):

What action is taken when the LEFM fails?

Response:

See Response to Question 2 (TXX-99203).

Question 34 (TXX-98274, TR-ER80P):

Provide a figure analogous to figure 5-2 in the topical using the Comanche Peak site-specific uncertainty values for the venturi and LEFM instruments.

Response:

This question is addressed in Figure 3 and accompanying text in Caldon's report ER-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System", Enclosure 2 of this submittal.