MATERIALS TRANSMITTED HEREWITH
CONTAINS 2.790 INFORMATION

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

August 24, 2000

TVA-WBN-TS-00-06 10CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of) Docket No. 50-390
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 00-06 - INCREASE UNIT 1 REACTOR POWER TO 3459 MWt - RESPONSE TO REQUESTS FOR ADDITIONAL INFORMATION (TAC NO. MA9152)

TVA's letter of June 7, 2000, provided the NRC with the subject license amendment request which would increase the WBN full core thermal power rating by 1.4% from 3411 MWt to 3459 MWt. The NRC Staff requested additional information from TVA in order to complete the review of TVA's application. The questions and TVA's proposed responses were discussed in a meeting between TVA and NRC in Washington on August 3, 2000. The purpose of this letter is to provide a formal response to the requested information. Enclosure 1 (Non-Proprietary) and Enclosure 2 (Proprietary) provide TVA's responses to these questions.

As Enclosure 2 contains information proprietary to Westinghouse, Enclosure 3 includes a Westinghouse Electric Company Application for Withholding Proprietary Information from Public Disclosure, and an accompanying Affidavit CAW-00-1414 signed by Westinghouse, the owner of the information. Also included are a Proprietary Information Notice and a Copyright Notice.

The above affidavit sets forth the basis on which the requested information may be withheld from public disclosure by the Commission, and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.790 of the Commission's regulations. Accordingly, TVA requests that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.790.

D030

August 24, 2000

Correspondence regarding the proprietary aspects of the Westinghouse information listed above, the Copyright Notice, or the supporting affidavit, should reference Westinghouse letter CAW-00-1414 and be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

There are no new regulatory commitments in this submittal and TVA's previous determination that there are no significant hazards considerations associated with the proposed change remains valid.

If you have any questions about these responses, please contact me at (423) 365-1824.

Sincerely,

P. L. Pace
Manager, Site Licensing
and Industry Affairs

Enclosures
cc (Enclosures):

NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Mr. Robert E. Martin, Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

TECHNICAL SPECIFICATION CHANGE TS-00-06

(NON-PROPRIETARY)

I. MECHANICAL ENGINEERING BRANCH

Question 1

In Section III.5.1.1 of Enclosure 1 and Page E6-16 of Enclosure 6, you stated that in most cases (but not all), revised fatigue usage and stress intensities of the reactor vessel components did not need to be calculated for the power uprate. Please identify components that are impacted by the power uprate and require further calculation. For these components evaluated for the uprated conditions, provide the maximum calculated stress and cumulative fatigue usage factor (CUF) at the critical locations of these components. Also, provide 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 the necessary justification.

Response to Question 1

The evaluation assesses the effects that the 1.4% uprating conditions have on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions as identified in the reactor vessel stress reports and addenda. The evaluation considers a worst case set of operating parameters from the current design basis parameters and the 1.4% uprate parameters.

The normal vessel outlet temperature increases from 618.7°F to 619.1°F, thereby increasing the T_{hot} variation in the outlet nozzles during normal plant loading and plant unloading. Therefore, the normal plant loading and plant unloading are considered to be more severe transients at the outlet nozzles. The vessel inlet temperature for the 1.4% uprate (557.3°F) resulted in a reduced temperature variation during normal plant loading (557°F to 557.3°F versus 557°F to 557.7°F) and plant unloading for those regions of the reactor vessel assumed to be in contact with vessel inlet water during normal reactor operation. Thus, only the outlet nozzles required evaluation for the worst case operating parameter effects. All of the remaining regions (including the upper head, main closure, inlet nozzles, vessel shell and bottom head) are analyzed to be in contact with the vessel inlet water. Therefore, the current reactor vessel stress reports remain applicable for the limiting locations in these regions.

The reactor vessel outlet nozzles were evaluated for the effects of the increased T_{hot} variations during the normal plant loading and plant unloading transients. The evaluation results show that the maximum ranges of primary plus secondary stress intensity and maximum cumulative fatigue usage factors reported for the outlet nozzles are negligibly affected by the 0.4°F increase in the vessel outlet temperature. The outlet nozzle stress intensity is [ ]$^{+a,c}$ ksi which is well below the the allowable 3 S_m limit of [ ]$^{+a,c}$ ksi; also, the fatigue usage is [ ]$^{+a,c}$ which is well below the 1.0 acceptance criteria. Therefore, all of the maximum ranges of stress intensity and maximum cumulative fatigue usage factors remain valid for the 1.4% uprate conditions.

As noted on Page E6-16 of the submittal, 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.

Question 2

In regard to Section III.5.2.3 of Enclosure 1, provide the maximum calculated stress and CUF at the critical locations of the reactor internal components (such as lower and upper core plates, core barrel, baffle/barrel, and fuel assembly) for the power uprate condition. If codes are used in the evaluation for the power uprate, provide the allowable Code limits, and the Code and Code edition. Confirm that methodology, assumptions and allowable limits used for the power uprate evaluation are the same as those in the current licensing basis of record.

Response to Question 2

The reactor internals are not licensed to a Code version and were designed based on sound engineering practice.

Baffle-Barrel Region Components:

No new CUF calculations were performed. The existing design transients remained valid for the 1.4% power uprate. The heat generation rates seen by the baffle-barrel region for the 1.4% power uprate were bounded by the existing analysis. The existing analysis remains applicable for the 1.4% power uprate conditions.

Upper Core Plate:

No new CUF calculations were performed. The existing design transients remained valid for the 1.4% power uprate. The effect of heat generation on the upper core plate is negligible (for the current analysis as well as the 1.4% uprate evaluation) due to the distance between the active fuel and the upper core plate. The analysis of record remains applicable for the 1.4% power uprate conditions.

Lower Core Plate:

New CUF calculations were performed for the 1.4% power uprate conditions. Although not required as a part of the licensing basis, an analysis was done to meet the intent of the ASME B&PV Code

(Section III, Division I, Appendices 1989 Edition). This analysis indicates that the maximum stress intensity is [ ]$^{+a,c}$ ksi, which is well below the 3 S_m limit of [ ]$^{+a,c}$ ksi. The fatigue usage factor is less than [ ]$^{+a,c}$, which is well below 1.0.

Question 3

In regard to Section III.5.2.2 of Enclosure 1, provide an assessment of flow-induced vibration of the reactor internal components due to the changes of T_{hot} and T_{cold} for the power uprate.

Response to Question 3

The reactor internals are not licensed to a Code version and were designed based on sound engineering practice. The T_{COLD} and T_{HOT} fluid densities increase by less than 0.3% (557.7 to 557.3 °F) and 0.4 % (618.7 to 619.1 °F) respectively for the 1.4% uprate conditions. These changes are insignificant and the flow induced vibration stresses would still remain well below the structural limits. The most limiting location from flow induced vibration, the upper support plate, would be expected to have a stress less than [ ]$^{+a,c}$ psi which is well below a representative allowable limit of [ ]$^{+a,c}$ psi.

Question 4

In reference to Section III.5.3 of Enclosure 1, provide an evaluation of the control rod drive mechanism with regard to the stress and fatigue usage as a result of the power uprate. Also, provide the allowable Code limits for the critical components evaluated, and the Code and Code edition used for the evaluation. If different from the Code of record, justify and reconcile the differences.

Response to Question 4

The evaluation included reviewing the various generic and plant specific analyses for the CRDMs. The CRDMs are affected by the cold leg temperature. The 1.4% uprate results in a T_{cold} of 557.3°F. This results in a maximum 1.5% change in ΔT for the heat-up/cooldown transient analysis. Assuming that the primary plus secondary stress range (both thermal and mechanical) is increased by 1.5% for all analyzed components, there would still be adequate margin in the 3 S_m ASME Code allowables. Also, the maximum fatigue usage at the limiting location in the CRDM resulting from all of the normal and upset transients is [ ]$^{+a,c}$. The usage contribution from the 200 heat-up/cooldown cycles at this location is a negligible [ ]$^{+a,c}$. Thus, the generic CRDM reports remain valid for the 1.4% uprate conditions.

As noted on Page E6-16 of the submittal, the Code version is the same as the Code of record. The Code of record is the ASME B&PV Code Section III 1971 Edition through Winter 1972 Addenda for the full length CRDMs.

Question 5

In reference to Section III.6.4 of Enclosure 1, you stated that the 2-percent increase in forces (loop forces increase due to a reduction of T_{cold}) was offset by a more representative characterization of the loop at the break location. Explain more about the approach using "the more representative characterization of the loop," which was claimed to result in 17-percent reduction in loop force at the break location. Is this approach currently used by WBN for a licensing basis documented in the UFSAR?

Response to Question 5

The licensing basis methodology is contained in WCAP-8708-P-A and the modeling of the branch connections is discussed in WCAP-8082-P-A. Consistent with this methodology, branch line ruptures are modeled at the girth butt weld between the branch pipe and safe end of the branch connection in the largest auxiliary lines of the primary coolant loop.

In the existing licensing basis calculation, the postulated branch line break forces were calculated in a conservative manner that did not require additional detailed information about the branch line nozzle. A conservatively short branch line nozzle leg was modeled with a 1 foot length and placement of the break was closer to the main loop piping than is required by WCAP-8082-P-A. The flow area of this branch line nozzle leg was conservatively assigned the flow area of the main coolant loop piping. This modeling was considered conservative relative to the requirements of WCAP-8708 and WCAP-8082, and modeling the details of the branch line nozzles was not necessary.

For the 1.4% uprate, the calculation was revised to more accurately reflect the branch line flow area which is consistent with the licensing basis methodology. The results demonstrate that the LOCA hydraulic forcing functions for the 1.4% uprate conditions are bounded by those used in the current design. Thus, the existing hydraulic forcing functions remain valid for the 1.4% uprate conditions.

Question 6

Provide evaluation of the potential of flow induced vibration for the steam generator U-Bend tubes quantitatively based on the increase in feedwater flow and the increase in pressure difference between the primary system pressure (unchanged at 2250 psi) and the decreased steam pressure for the proposed power uprate.

Response to Question 6

The assessment of the 1.4% power uprate on the small radius U-bends is accomplished using the one dimensional relative stability ratio (RSR). The RSR is the stability ratio of a tube at the uprated conditions relative to the conditions used for the analysis at the current conditions.

First, the maximum allowable RSR is 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. The minimum value for all tubes is [ ]$^{+a,c}$ for R9C8 in SG4 and R9C22 in SG1.

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. The maximum RSR value was calculated as [ ]^{a,c}. This value is lower than all the maximum allowable RSR values for the most susceptible tubes. As noted above, the minimum value for all tubes was [ ]$^{+a,c}$. Therefore, no additional tubes will become susceptible to high cycle fatigue as a result of the 1.4% power uprating.

Question 7

In Section III.7, "Balance of Plant," you stated that as part of design change process for the power uprate, additional heat balance studies will be performed at higher ambient conditions to assess potential impact on individual BOP components. Please provide such an evaluation and identify systems and components that will be affected by the higher ambient conditions for the power uprate.

Response to Question 7

The secondary side plant systems were originally designed to support the operation of the Westinghouse supplied turbine/generator. At the valves wide open or stretch condition the turbine/generator is rated at 1,269,837 kW with a steam flow of 15,900,800 lb/hr. This equates to operation at approximately 104% reactor thermal power. Based on this, no major impacts to the balance of plant were expected.

Using the revised NSSS parameters (see Table 2-1 of TVA's June 7, 2000 amendment request), TVA has performed a heat balance at 101.4% reactor thermal power for the proposed uprate. As reported in TVA's amendment request, a comparison of the uprate heat balance with the current 100% heat balance revealed no significant differences in pressures, temperatures, or flows for the secondary side plant systems.

As part of the design change process for the power uprate, additional heat balance studies have been performed at higher ambient conditions to assess potential impacts on individual BOP components. Heat balances were performed at the current 3427 MWt and the proposed uprate thermal input of 3475 MWt with the following boundary conditions:

Boundary Condition	Units	3427 MWt	3475 MWt
Steam Generator Outlet Pressure	psia	1000.0	980.0
Main Steam Throttle Pressure	psia	975.0	955.0
Condenser Circulating Water (CCW) Temperature	°F	90.5	90.5
Condenser Cleanliness	%	80	80
Steam Generator Blowdown (SGBD) Flow	gpm	350	350

The boundary conditions are representative of the following:

- Peak CCW temperature experienced during summer operation
- Condenser cleanliness representative of the condenser following a long continuous run with macro-fouling on the tubesheets
- Maximum allowable SGBD flow

The results of the two heat balances are shown in Table 1 (attached). As in the previous heat balance comparison, no notable differences existed which would warrant further investigation. However, areas of consideration that were explored further included the main condenser backpressure, condensate polishing inlet temperature, main feed pump turbine and associated condenser, high pressure turbine impulse pressure, flow instrumentation range limitations, and the high pressure reheater operating vent line. The following paragraphs discuss each of these items:

Main Condenser Backpressure

Watts Bar's Low Pressure Turbine "C" currently has a backpressure limitation of 6.2 in-HgAbs. Current operation at the listed boundary conditions would result in a backpressure of 5.79 in-HgAbs. Likewise, for the uprate conditions, the backpressure is expected to increase to 5.9 in-HgAbs. Since the predicted backpressure remains below the limit, the main condenser can support the proposed uprate.

Condensate Polishing Inlet Temperature

The condensate polishers have an inlet temperature limit of 140°F. SGBD is fed into the condensate system directly upstream of the polishers. The blowdown flow is cooled by condensate cooled heat exchangers prior to entering the condensate system. The blowdown temperature is expected to increase by approximately 0.6°F following the uprate (141.8°F). Since maximum SGBD flow has been assumed for the heat balance evaluation, it can also be assumed that maximum condensate flow is directed to the polishers to aid in cycle cleanup. With this combination, the SGBD flow (350 gpm) is mixed with approximately 18,000 gpm of condensate and the condensate temperature entering the polishers will stay well below the limit. Therefore, the condensate polishers will not be impacted by the uprate.

Main Feedwater Pump Turbine (MFPT) and Associated Condenser

Watts Bar has one motor-driven and two turbine-driven feedwater pumps available to supply feedwater to the steam generators. The MFPTs driving the feedwater pumps exhaust to MFPT Condensers which are cooled by condensate. When the CCW inlet temperatures to the main condenser approach the higher values, the condensate temperatures supplied to the MFPT condenser also rise. Each MFPT has a condenser backpressure limitation of 10 inches vacuum (~20 in-HgAbs). Additional heat balances have been performed to demonstrate that the unit can operate below these limits at the uprated condition; however, the plant typically operates the motor-driven feed pump in parallel to achieve a overall higher efficiency. Thus, it is determined that the uprate has no significant impact on MFPT condenser operation.

High Pressure (HP) Turbine Impulse

As a result of the proposed uprate, the steam flow through the HP turbine increases, such that the Impulse Pressure also must increase. An increase of approximately 13 psi is anticipated following the uprate. This increase will result in rescaling the Impulse Pressure transmitters and the AMSAC bistable arming setpoints. In addition, runback setpoints associated with the Impulse Pressure have been evaluated and no setpoint modification to these instruments are required. Applicable instrumentation changes are being included in the engineering design package which is currently being finalized to support the proposed 1.4% uprate.

Instrumentation Range Limitations

The change in measured parameters (i.e., affected by the uprate) did not impact the instrumentation supporting BOP operation (except as noted above - Impulse Pressure). However, the total power calorimetric uncertainty using LEFM was evaluated by Westinghouse and resulted in the uncertainties for several BOP instrument channels having to be re-calculated using current Westinghouse methodology. This required several BOP instrument loop accuracies to be revised to comply with the Westinghouse calculation.

HP Reheater (Second Stage) Operating Vent Line

The HP reheater operating vent lines pass two-phase flow from the exit of the each second stage reheater 4th tube pass to the number 1 extraction piping. The design limit of these lines indicate that they would not be able to handle any significant increase in flow. Steam to the second stage reheaters is supplied off main steam. Due to the reduction in steam generator pressure (thereby directly reducing the saturation temperature of the steam) the achievable reheat temperature is reduced along with the steam demand to the reheater. Therefore, since the steam supply to the HP reheater is reduced with the uprate, the operating vent line is not impacted.

Based on TVA's evaluations, the Balance of Plant systems are deemed adequate for the increase in thermal loads produced by the power uprate with exceptions as noted above.

TABLE 1 - HEAT BALANCE RESULTS				
FIELD DESCRIPTION	**UNITS**	**WATTS BAR 3427 MWt**	**WATTS BAR 3475 MWt**	**% Diff**
1ST STAGE REHEATER TUBE SIDE INLET FLOW	#/HR	568,292	583,164	2.62%
2ND STAGE REHEATER TUBE SIDE INLET FLOW	#/HR	793,096	768,230	-3.14%
CCW INLET FLOW	GPM	412,800	412,800	0.00%
CONDENSER HOTWELL DRAIN FLOW	#/HR	8,533,612	8,677,104	1.68%
HP TURBINE EXHAUST FLOW	#/HR	12,378,778	12,582,604	1.65%
LP TURBINE A EXHAUST STAGE FLOW	#/HR	2,714,600	2,759,291	1.65%
LP TURBINE B EXHAUST STAGE FLOW	#/HR	2,714,600	2,759,291	1.65%
LP TURBINE C EXHAUST STAGE FLOW	#/HR	2,714,600	2,759,291	1.65%
MAIN STEAM AT LP TURBINE INLET FLOW	#/HR	9,862,364	10,038,637	1.79%
MAIN STEAM THROTTLE FLOW	#/HR	14,312,852	14,571,540	1.81%
MFPT CONDENSER INLET FLOW	#/HR	8,708,612	8,852,104	1.65%
MOISTURE SEPERATOR DRAIN OUTLET FLOW	#/HR	1,422,136	1,426,005	0.27%
NO 1 EXTRACTION STEAM OUTLET FLOW	#/HR	644,437	671,323	4.17%
NO 2 EXTRACTION STEAM OUTLET FLOW	#/HR	644,657	658,673	2.17%
NO 3 EXTRACTION STEAM OUTLET FLOW	#/HR	914,407	933,576	2.10%
NO 4 EXTRACTION STEAM OUTLET FLOW	#/HR	362,046	372,708	2.94%
NO 5 EXTRACTION STEAM OUTLET FLOW	#/HR	514,572	525,645	2.15%
NO 6 EXTRACTION STEAM OUTLET FLOW	#/HR	345,774	353,084	2.11%
NO 7 EXTRACTION STEAM OUTLET FLOW	#/HR	143,170	145,119	1.36%
NO. 1 FWH DRAIN FLOW	#/HR	1,468,135	1,469,715	0.11%
NO. 1 FWH TUBE OUTLET FLOW	#/HR	15,295,600	15,529,420	1.53%
NO. 2 FWH DRAIN FLOW	#/HR	2,721,128	2,751,032	1.10%
NO. 3 HDT DRAIN FLOW	#/HR	5,057,671	5,110,613	1.05%
NO. 4 FWH DRAIN FLOW	#/HR	362,046	372,708	2.94%
NO. 4 FWH TUBE OUTLET FLOW	#/HR	10,237,929	10,418,807	1.77%

TABLE 1 - HEAT BALANCE RESULTS				
FIELD DESCRIPTION	**UNITS**	**WATTS BAR 3427 MWt**	**WATTS BAR 3475 MWt**	**% Diff**
NO. 5 FWH DRAIN FLOW	#/HR	876,618	898,353	2.48%
NO. 6 FWH DRAIN FLOW	#/HR	1,282,514	1,314,557	2.50%
NO. 6 FWH TUBE INLET FLOW	#/HR	9,602,415	9,782,169	1.87%
NO. 7 FWH TUBE INLET FLOW	#/HR	8,073,098	8,215,466	1.76%
NO. 7 HDT DRAIN FLOW	#/HR	1,529,317	1,566,703	2.44%
REHEAT STEAM TO MFPT LP INLET FLOW	#/HR	179,871	184,386	2.51%
SGBD HX TUBE SIDE STAGE 1 INLET FLOW	#/HR	635,514	636,638	0.18%
SGBD HX TUBE SIDE STAGE 2 INLET FLOW	#/HR	250,000	250,000	0.00%
STEAM GENERATOR BLOWDOWN OUTLET FLOW	#/HR	175,000	175,000	0.00%
STEAM GENERATOR STEAM OUTLET FLOW	#/HR	15,120,600	15,354,420	1.55%
MAIN TURBINE CONTROL VALVE INLET MOISTURE	%	0.3864	0.3823	-1.04%
2ND STAGE REHEATER SHELL SIDE OUTLET TEMPERATURE	°F	520.3	517.8	-0.48%
CCW INLET TEMPERATURE	°F	90.5	90.5	0.00%
CCW OUTLET TEMPERATURE	°F	128.0	128.6	0.47%
HP TURBINE 1ST EXTRACTION OUTLET TEMPERATURE	°F	449.1	450.9	0.40%
HP TURBINE 2ND EXTRACTION OUTLET TEMPERATURE	°F	410.2	411.7	0.37%
HP TURBINE EXHAUST STEAM OUTLET TEMPERATURE	°F	369.8	371.1	0.35%
LP TURBINE 4TH EXTRACTION OUTLET TEMPERATURE	°F	361.5	359.2	-0.64%
LP TURBINE 5TH EXTRACTION OUTLET TEMPERATURE	°F	284.7	282.5	-0.77%
LP TURBINE 6TH EXTRACTION OUTLET TEMPERATURE	°F	215.9	216.7	0.37%
LP TURBINE 7TH EXTRACTION OUTLET TEMPERATURE	°F	176.8	177.6	0.45%
LP TURBINE A EXHAUST STEAM OUTLET TEMPERATURE	°F	115.3	115.7	0.35%
LP TURBINE B EXHAUST STEAM OUTLET TEMPERATURE	°F	127.4	127.9	0.39%
LP TURBINE C EXHAUST STEAM OUTLET TEMPERATURE	°F	139.4	140.1	0.50%
LP TURBINE REHEAT STEAM INLET TEMPERATURE	°F	519.7	517.2	-0.48%
MAIN CONDENSER DRAIN OUTLET TEMPERATURE	°F	131.5	132.2	0.53%
MFPTC TUBE SIDE INLET TEMPERATURE	°F	133.3	133.9	0.45%

TABLE 1 - HEAT BALANCE RESULTS				
FIELD DESCRIPTION	**UNITS**	**WATTS BAR 3427 MWt**	**WATTS BAR 3475 MWt**	**% Diff**
NO 1 FWH DRAIN COOLER APPROACH (DCA)	°F	7.8	7.8	0.00%
NO 1 FWH OUTLET TEMPERATURE	°F	439.8	441.5	0.39%
NO 1 FWH TERMINAL TEMPERATURE DIFFERENCE (TTD)	°F	4.2	4.4	4.76%
NO 2 FWH DRAIN COOLER APPROACH (DCA)	°F	9.1	9.3	2.20%
NO 2 FWH TERMINAL TEMPERATURE DIFFERENCE (TTD)	°F	4.1	4.3	4.88%
NO 3 FWH TERMINAL TEMPERATURE DIFFERENCE (TTD)	°F	1.9	2.0	5.26%
NO 4 FWH DRAIN COOLER APPROACH (DCA)	°F	0.4	0.5	25.00%
NO 4 FWH TERMINAL TEMPERATURE DIFFERENCE (TTD)	°F	0.9	0.9	0.00%
NO 5 FWH DRAIN COOLER APPROACH (DCA)	°F	11.3	11.6	2.65%
NO 5 FWH TERMINAL TEMPERATURE DIFFERENCE (TTD)	°F	5.3	5.5	3.77%
NO 6 FWH DRAIN COOLER APPROACH (DCA)	°F	10.8	11.0	1.85%
NO 6 FWH TERMINAL TEMPERATURE DIFFERENCE (TTD)	°F	3.5	3.6	2.86%
NO 7 FWH TERMINAL TEMPERATURE DIFFERENCE (TTD)	°F	3.5	3.6	2.86%
SGBD HX SHELL SIDE STAGE 1 OUTLET TEMPERATURE	°F	164.0	164.5	0.30%
SGBD HX SHELL SIDE STAGE 2 OUTLET TEMPERATURE	°F	141.2	141.8	0.42%
SGBD HX TUBE SIDE STAGE 1 OUTLET TEMPERATURE	°F	265.3	264.8	-0.19%
SGBD HX TUBE SIDE STAGE 2 INLET TEMPERATURE	°F	132.1	132.7	0.45%
SGBD HX TUBE SIDE STAGE 2 OUTLET TEMPERATURE	°F	148.0	148.6	0.41%
STEAM GENERATOR BLOWDOWN OUTLET TEMPERATURE	°F	544.6	542.1	-0.46%
NET TURBINE HEATRATE IN BTU/KWH	BTU/KWH	10090.6	10114.5	0.24%
LP TURBINE A EXHAUST STEAM OUTLET PRESSURE	IN-HG	3.02	3.06	1.32%
LP TURBINE B EXHAUST STEAM OUTLET PRESSURE	IN-HG	4.22	4.28	1.42%
LP TURBINE C EXHAUST STEAM OUTLET PRESSURE	IN-HG	5.79	5.90	1.90%
MFPT EXHAUST STEAM OUTLET PRESSURE	IN-HG	10.62	10.86	2.26%
MAIN FEED PUMP TURBINE OUTPUT	KW	9929.9	10132.9	2.04%
TOTAL GENERATOR OUTPUT	MWe	1,158,841	1,172,243	1.16%
2ND STAGE REHEATER SHELL SIDE OUTLET PRESSURE	PSIA	161.5	164.1	1.61%

TABLE 1 - HEAT BALANCE RESULTS				
FIELD DESCRIPTION	**UNITS**	**WATTS BAR 3427 MWt**	**WATTS BAR 3475 MWt**	**% Diff**
2ND STAGE REHEATER TUBE SIDE INLET PRESSURE	PSIA	965.3	945.5	-2.05%
HP TURBINE 1ST EXTRACTION OUTLET PRESSURE	PSIA	418.8	426.3	1.79%
HP TURBINE 2ND EXTRACTION OUTLET PRESSURE	PSIA	277.2	282.1	1.77%
HP TURBINE EXHAUST STEAM OUTLET PRESSURE	PSIA	172.9	175.7	1.62%
HP TURBINE IMPULSE PRESSURE	PSIA	743.0	757.4	1.94%
LP TURBINE 4TH EXTRACTION OUTLET PRESSURE	PSIA	68.6	69.7	1.56%
LP TURBINE 5TH EXTRACTION OUTLET PRESSURE	PSIA	43.2	43.9	1.55%
LP TURBINE 6TH EXTRACTION OUTLET PRESSURE	PSIA	15.9	16.1	1.64%
LP TURBINE 7TH EXTRACTION OUTLET PRESSURE	PSIA	7.0	7.1	1.71%
LP TURBINE REHEAT STEAM INLET PRESSURE	PSIA	158.3	160.8	1.58%
MAIN TURBINE CONTROL VALVE INLET PRESSURE	PSIA	975.0	955.0	-2.05%
NO 1 FWH SATURATION PRESSURE	PSIA	397.9	405.0	1.78%
NO 2 FWH SATURATION PRESSURE	PSIA	263.3	268.0	1.79%
NO 3 FWH SATURATION PRESSURE	PSIA	164.3	166.9	1.58%
NO 4 FWH SATURATION PRESSURE	PSIA	65.2	66.2	1.53%
NO 5 FWH SATURATION PRESSURE	PSIA	41.0	41.7	1.71%
NO 6 FWH SATURATION PRESSURE	PSIA	15.1	15.3	1.32%
NO 7 FWH SATURATION PRESSURE	PSIA	6.6	6.8	3.03%
STEAM GENERATOR STEAM OUTLET PRESSURE	PSIA	1000.0	980.0	-2.00%
MAIN FEED PUMP TURBINE SPEED	RPM	4,622	4,649	0.59%

Question 8

On Page E6-22 of the reference, you indicated that the licensing basis conditions for the motor-operated valves (MOVs) program by TVA bound the uprated conditions and therefore, the safety-related MOVs at WBN will be capable of performing their intended function(s) following the power uprate. Please discuss effects of the proposed power uprate on the pressure locking and thermal binding of the safety-related power-operated gate valves for Generic Letter (GL) 95-07 and on the evaluation of overpressurization of isolated sections of piping segment for GL 96-06. Identify mechanical components for which functionality at the uprated conditions could not be confirmed.

Response to Question 8

GL 96-06 Response:

Generic Letter GL 96-06 addresses the overpressurization of isolated piping segments. Two other issues are addressed in the GL but were not pertinent to the RAI.

The isolated segments of pipe have been previously evaluated and have been upgraded, where required, to meet the criteria of the GL. Isolated segments of piping that are susceptible to overpressurization due to thermal stresses imposed by the environment or due to internal heat sources have been provided with thermal safety-relief valves or have been determined to be structurally adequate to withstand the stresses imposed by the thermal loading. LOCA analyses that affect the containment side of the isolated segment of piping have been performed at 102% of 3411 MWt and remain bounding for this power uprate.

The environmental conditions, imposed by LOCAs or secondary side line breaks, have not been affected in an adverse manner that would compromise the piping that is capable of being isolated by segments. The main steam temperature has decreased and this evaluation bounds the small increase in the feedwater temperature in the environmentally qualified rooms.

There is no increase in the possibility of overpressurization of isolated segments of piping.

Response to GL 95-07:

A review of the documentation and evaluations of GL 95-07 was performed to determine if the proposed 1.4% power increase would adversely affect any conclusions or qualifications that were approved by the NRC upon closure of the subject Generic Letter. Of particular interest was the TVA Pressure Locking and Thermal Binding (PLTB) Evaluation Matrix Notes and the NRCs Safety Evaluation (TAC No. M93537) and the comments and conclusion therein. The conditions that were in the evaluation remain bounding for the 1.4% power uprate conditions and the conclusions, and the NRCs understanding of the basis for those conclusions, remain valid.

The support systems that are in close proximity to the NSSS loop were reviewed for impact. The second isolation valves for the support systems were assumed to be unaffected by the small NSSS hot leg temperature increase. The valves that were listed in the GL 95-07 as being modified to eliminate the potential for PLTB remain unaffected. (Refer to Table 2, Item A, attached). Valves that were listed in the GL 95-07 response as being evaluated to ensure their ability to open under pressure locking conditions were reevaluated for the hot leg temperature increase of 0.4 degrees F. (Refer to Table 2, Item B, attached). In accordance with the guidance of Information Notice 95-14 (Susceptibility of Containment Sump Recirculation Gate Valves to Pressure Locking), a 1 degree F temperature rise may result in a 33 psi pressure increase. The 0.4 degrees F temperature rise results in a pressure increase of approximately 13 psi. The increase in thrust per the Commonwealth Edison Company thrust-prediction methodology due to the 0.4 degrees F is insignificant and is bounded by the current calculations. Additionally, the evaluation assessed the impacts of the 1.4% uprate on the GL 89-10 and Limitorque Technical Bulletin 98-01 update programs and found these to be acceptable. No impacts were identified on the secondary side due to the lower Main Steam system operating pressure. The feedwater temperature increase is bounded by the existing evaluations.

This proposed power uprate does not introduce any increased challenge for thermal binding and/or pressure locking and the responses and conclusions of GL 95-07 and GL 96-06, as well as GL 89-10, remain valid.

TABLE 2

LISTING OF VALVES EVALUATED UNDER GL 95-07

A. Valves that have been previously modified to eliminate PLTB:

1-FCV-63-008	Residual Heat Removal (RHR) to Safety Injection (SI) and High Pressure SI Pump Suction
1-FCV-63-011	RHR to SI and High Pressure SI Pump Suction
1-FCV-63-072	Containment Sump to RHR Pump Suction
1-FCV-63-073	Containment Sump to RHR Pump Suction
1-FCV-63-156	SI Hot Leg Injection
1-FCV-63-157	SI Hot Leg Injection
1-FCV-63-172	RHR Hot Leg Injection
1-FCV-72-040	RHR Spray Header Isolation
1-FCV-72-041	RHR Spray Header Isolation
1-FCV-72-044	Containment Sump to Containment Spray Pump Suction
1-FCV-72-045	Containment Sump to Containment Spray Pump Suction
1-FCV-74-001	RHR Pump Suction from Hot Legs
1-FCV-74-002	RHR Pump Suction from Hot Legs
1-FCV-74-008	1-FCV-74-002 Bypass
1-FCV-74-009	1-FCV-74-001 Bypass
1-FCV-74-033	RHR Crosstie
1-FCV-74-034	RHR Crosstie

B. Valves re-evaluated to ensure operability with the power uprate:

1-FCV-01-018	Steam Supply to Turbine Driven Auxiliary Feedwater Pump
1-LCV-62-135	High Pressure SI Pump Suction from Refueling Water Storage Tank (RWST)
1-LCV-62-136	High Pressure SI from The RWST
1-FCV-63-006	SI Pump Suction from RHR
1-FCV-63-007	SI Pump Suction from RHR
1-FCV-68-332	Pressurizer Power Operated Relief Valves (PORV) Block
1-FCV-68-333	Pressurizer PORV Block
1-FCV-72-002	Containment Spray Pump Discharge
1-FCV-72-039	Containment Spray Pump Discharge

<u>Question 9</u>

Describe superscripts "a" and "c" which are not defined in Tables 1 and 2 on Pages E6-20 and E6-21

<u>Response to Question 9</u>

These scripts refer to the classification of the proprietary information and are not intended to be footnotes. A description of these classifications was provided in Enclosure 11 of TVA's June 7, 2000, letter.

<u>Question 10</u>

Do you project modifications to piping or equipment supports for the proposed power uprate? If any, provide examples of pipe supports requiring modification and discuss the nature of these modifications.

<u>Response to Question 10</u>

We have not identified any modifications to piping or fluid system component supports as a result of the proposed 1.4% uprate.

II. REACTOR SYSTEMS ENGINEERING BRANCH

Question 1:

The SG Atmospheric Relief Valves (ARVs) are discussed on Page E1-16 of TVA's application. Provide additional information to justify the adequacy of the ARVs' design relief capacity for the 1.4% uprate.

Response to Question 1:

The ARV sizing criteria in the report is based on core power of 3564 MWt and an additional 2% uncertainty is applied to account for the power measurement uncertainty. This power level bounds the 1.4% uprate power level of 3459 MWt. Thus, the ARV is adequately sized for the 1.4% uprate conditions.

Question 2:

With respect to the discussion of BELBLOCA, Page E1-35 of TVA's application, discuss the relationship between the MONTEC computer Code and WCOBRA/TRAC and whether it may be used separately from WCOBRA/TRAC.

Response to Question 2:

The BELBLOCA methodology is contained in WCAP-12945 which was approved by the NRC via an NRC SER dated June 28, 1996. Page 6 of WCAP-12945 indicates that a Monte Carlo process is used to determine the total uncertainty. The MONTEC computer code is used to complete this process and is a part of the WCOBRA/TRAC evaluation model. The MONTEC Code is also mentioned on Page 27-14 of WCAP-12945.

Question 3:

Section 6.5.1, beginning on page E1-37 of TVA's application provides a discussion of the affects on the Non-LOCA/Transient Analyses for the 1.4% power uprate. Please provide additional information to justify the conclusion that DNBR margins remain acceptable.

Response to Question 3:

Please see Table 3 for a summary of the DNBR assessments for the applicable events. The current design limit is [ ]$^{+a,c}$ thimble cell ([ ]$^{+a,c}$ typical cell), and the safety analysis limit is [ ]$^{+a,c}$ thimble cell ([ ]$^{+a,c}$ typical cell).

Table 3: Margin Assessment for non-LOCA DNB	
Event	**Assessment**
Core Limits Changes	These limits were slightly changed to adjust the vessel exit boiling limit for the 1.4 % increased core power. DNBR margin between the design and safety analysis limit was allocated to offset the power increase, instead of changing DNB core limits. DNBR is still above the design limit.
Rod Withdrawal from Subcritical	The calculated DNBR value is [ ] $^{+a,c}$ which is greater than the safety analysis limit.
Rod Withdrawal from Power	The calculated DNBR value is [ ] $^{+a,c}$ which is greater than the safety analysis limit.
Dropped Rod Event	DNB margin between the design and safety analysis limit was assigned to accommodate the 1.4 % increased core power. DNBR is still above the design limit.
Loss of Flow	The calculated DNBR value is about [ ] $^{+a,c}$ which is greater than the safety analysis limit.
Loss of Load / Turbine Trip	The calculated DNBR value is about [ ] $^{+a,c}$ which is greater than the safety analysis limit.
Feedwater System Malfunction – Excessive Heat Removal	The calculated DNBR value is about [ ] $^{+a,c}$ which is greater than the safety analysis limit.
Accidental RCS Depressurization	The calculated DNBR value is about [ ] $^{+a,c}$ which is greater than the safety analysis limit.
Inadvertent ECCS Actuation at Power	Limiting DNBR occurs at event initiation and gets higher during event. The calculated DNB value is greater than 2.0 which is greater than the safety analysis limit.
Locked Rotor (Single RCP)	The number of rods in DNB is still less than 13%. This value is confirmed every cycle as part of the reload process.
Excessive Load Increase	Bounding statepoints of temperature, pressure, and power have been compared to core limits. The DNBR limit continues to be met, with negligible (~0.3°F) change in proximity to limit lines. Event is not DNB limiting.
Steamline Break at Power Coincident with RWAP	DNB margin between the design and safety analyses limit was assigned to accommodate the 1.4% increased core power. DNBR is still above the design limit.
Steamline Break at Hot Zero Power	Event initiates at zero power. The calculated DNBR value is greater than 2.0 which is greater than the safety analysis limit.

Question 4:

TVA's application discusses the Rod Ejection Event, on Page E1-44. Please discuss the acceptance criteria for the fuel pellets with respect to 10 CFR 50, Appendix-A, General Design Criteria 28.

Response to Question 4:

Acceptance Criteria for Fuel Pellet:

The fuel pellet enthalpy criterion presented in the submittal is the same as that found in Chapter 15.4.6.1.2 of the Watts Bar UFSAR: 200 cal/gm for irradiated fuel and 225 cal/gm for unirradiated fuel. In the Westinghouse methodology, a limit of 200 cal/gm is used since it bounds both irradiated and unirradiated fuel, as well as the Standard Review Plan value of 280 cal/gm.

The issues associated with the current Rod Ejection criterion and high burnup fuel were recently discussed in a meeting of the Westinghouse Fuel Working Group held on May 4, 2000, in Columbia S.C. NRC representatives also attended this meeting. In response to a question, an NRC representative indicated that there were no current plans to backfit any reduced fuel limits for the RCCA Ejection accident to plants that stay within the current licensed burnup limit (62,000 MWd/mtU). The intent is to apply these revised analysis limits, when available, only to plants requesting an increase in the licensed burnup limit. This position is consistent with the NRC's Memorandum "Agency Program Plan for High Burnup Fuel", from L. J. Callan to the ACRS, dated July 6, 1998. Current fuel license limits prohibit lead fuel rod burnups greater than 62,000 MWd/MTU. Therefore, the 200 cal/gm peak fuel enthalpy requirement applied by Westinghouse, which bounds the current 280 cal/gm criterion, is still applicable to the Watts Bar. Based upon the use of a 200 cal/gm limit, the Watts Bar Rod Ejection analysis is consistent with section 15.4.8 of the Standard Review Plan and meets the requirements of General Design Criterion 28.

With respect to the reactor pressure criterion in Section 15.4.8 of the Standard Review Plan, Westinghouse has generically shown that this criterion is met as documented in WCAP-7588 Revision 1-A. Since this generic evaluation is applicable to Watts Bar, the reactor pressure criterion is met for the Rod Ejection analysis.

Clad Temperature Value:

The clad temperature criterion presented in the submittal is the same as that found in Chapter 15.4.6.1.2 of the Watts Bar UFSAR: average clad temperature at the hot spot below 3000°F. This is actually an internal criterion that was used by Westinghouse to provide an indication of core coolability and is not a licensing limit. As discussed in letter NS-NRC-89-3466 (Reference 1), Westinghouse recognizes that the fuel pellet enthalpy limit and not clad temperature is the accepted criterion for confirming core coolability following the event. The criterion and discussion regarding the fuel cladding temperature limit was provided for informational purposes. The fuel enthalpy criterion, 200 cal/gm, continues to be used to demonstrate core coolability. It was determined that the enthalpy was less than 200 cal/gm for the 1.4% uprate conditions. Therefore, the core coolability would be expected to be maintained.

Reference 1: Westinghouse Letter NS-NRC-89-3466, "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," W. J. Johnson of Westinghouse to R. C. Jones of USNRC, October 23, 1989.

Question 5:

Please provide additional information to justify TVA's conclusion on Page E1-45, that Reactor Trip and ESFAS Setpoints remain acceptable for the 1.4% Power Uprate.

Response to Question 5:

The change in operating steam generator pressure and steam flow has no effect on the steam pressure effect for the narrow range steam generator water level low-low and high-high trip setpoints. It also has no effect on the fluid velocity effect for the narrow range steam generator water level high-high turbine trip setpoint.

The maximum fluid velocity effect occurs at approximately 80% of Rated Thermal Power; and the maximum fluid velocity effect of 0.6 psi used in the current steam generator water level high-high turbine trip setpoint is unaffected by the 1.4% power uprate.

The steam pressure effect on the steam generator water level high-high turbine trip setpoint is unaffected by the 1.4% power uprate since the limiting case is determined by the 0% power case.

The steam pressure effect on the steam generator water level low-low reactor trip setpoint is unaffected by the 1.4% power uprate since the limiting case is determined by the 0% power case.

III. MATERIALS AND CHEMICAL ENGINEERING BRANCH

STEAM GENERATOR RELATED QUESTIONS

The following questions relate to steam generator tube degradation as discussed in Sections 5.6.5, 5.6.6, and 5.6.7 in TVA's submittal dated June 7, 2000.

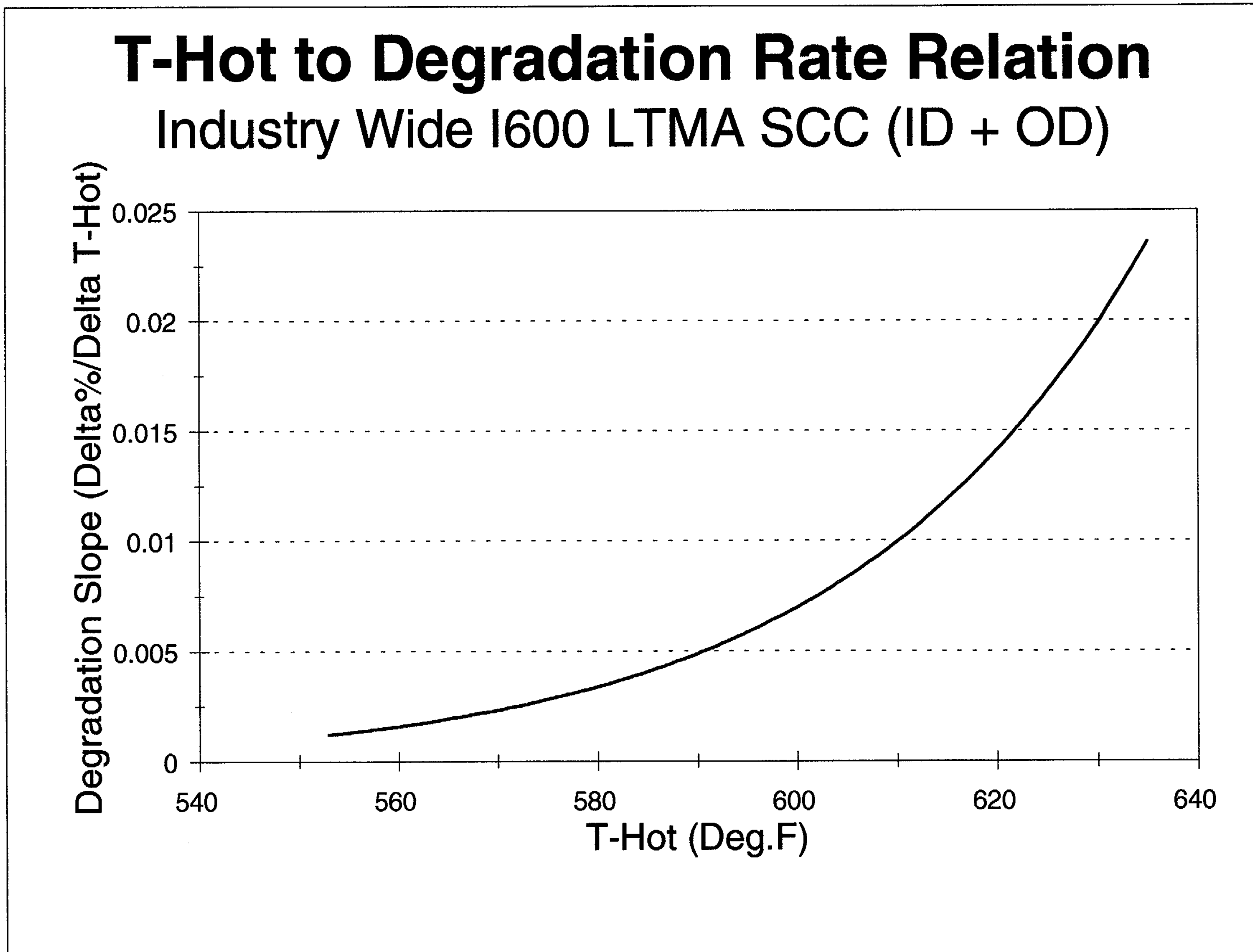
Question 1:

Section 5.6.5 - TVA stated that ". . . T_{hot} is expected to increase by 0.4 degree F for the 1.4% uprate and is considered to be the most sensitive operating parameter with respect to corrosion . . . " TVA also stated that ". . . these changes are expected to have an insignificant effect on the tube corrosion mechanisms since they are relatively minor and are comparable to the range of uncertainties used in assessing corrosion . . . " (1) TVA should expand on why the increase in T_{hot} is the most sensitive operating parameter with respect to corrosion. (2) If the increase in T_{hot} is within the range of uncertainties used in assessing corrosion and is relatively minor, TVA needs to describe the uncertainties in terms of quantitative or qualitative analysis to support the above statement.

Response to Question 1:

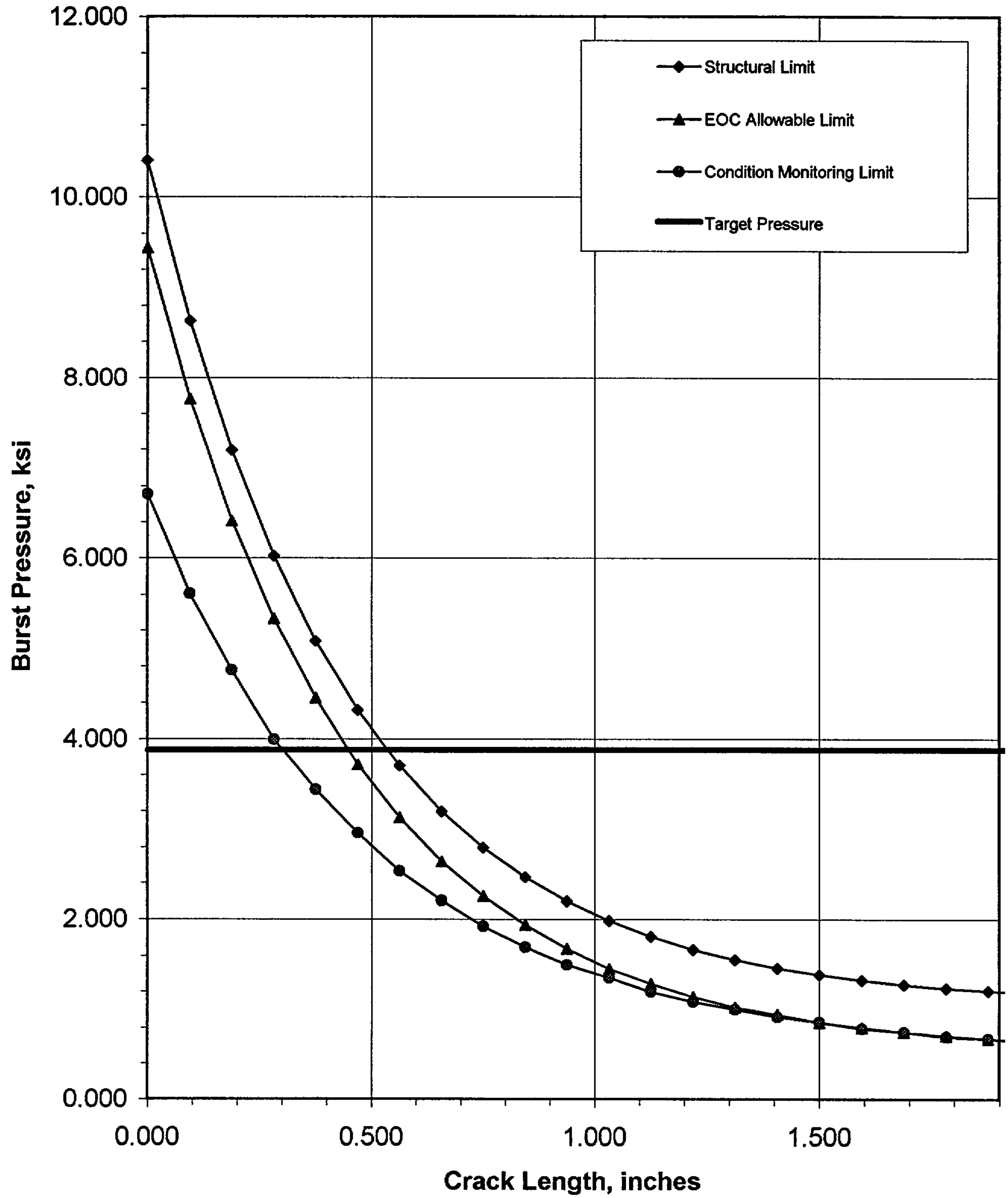
(1) Of the changes proposed in the 1.4% uprate, the minor change in temperature and the minor change in secondary pressure are the only changes that would affect corrosion rates. The T_{hot} change is considered to be the most sensitive to corrosion rates. The attached graph (T-Hot Degradation Rate Relation) represents industry data for Inconel 600 tubing. The graph illustrates the impact of temperature to corrosion rates, however, a 0.4 degree F increase affect can not be quantified. When assessing structural integrity of indications identified during an inspection, the change in the secondary pressure would be a sensitive parameter; however, as depicted in the two additional attached graphs, the change in $3\Delta P$ from 3876 psi to 3909 psi is also too small to be a quantifiable impact. The graphs illustrate structural limits for axial through wall cracking. One graph has the structural limit, the condition monitoring limit, and the end of cycle allowable limit with a $3\Delta P$ of 3876 and the second graph has the same limits with a $3\Delta P$ of 3909. The graphs illustrate the length of a through wall axial crack and its associated burst pressure. There is no difference in the acceptable length of a through wall crack at these two pressures.

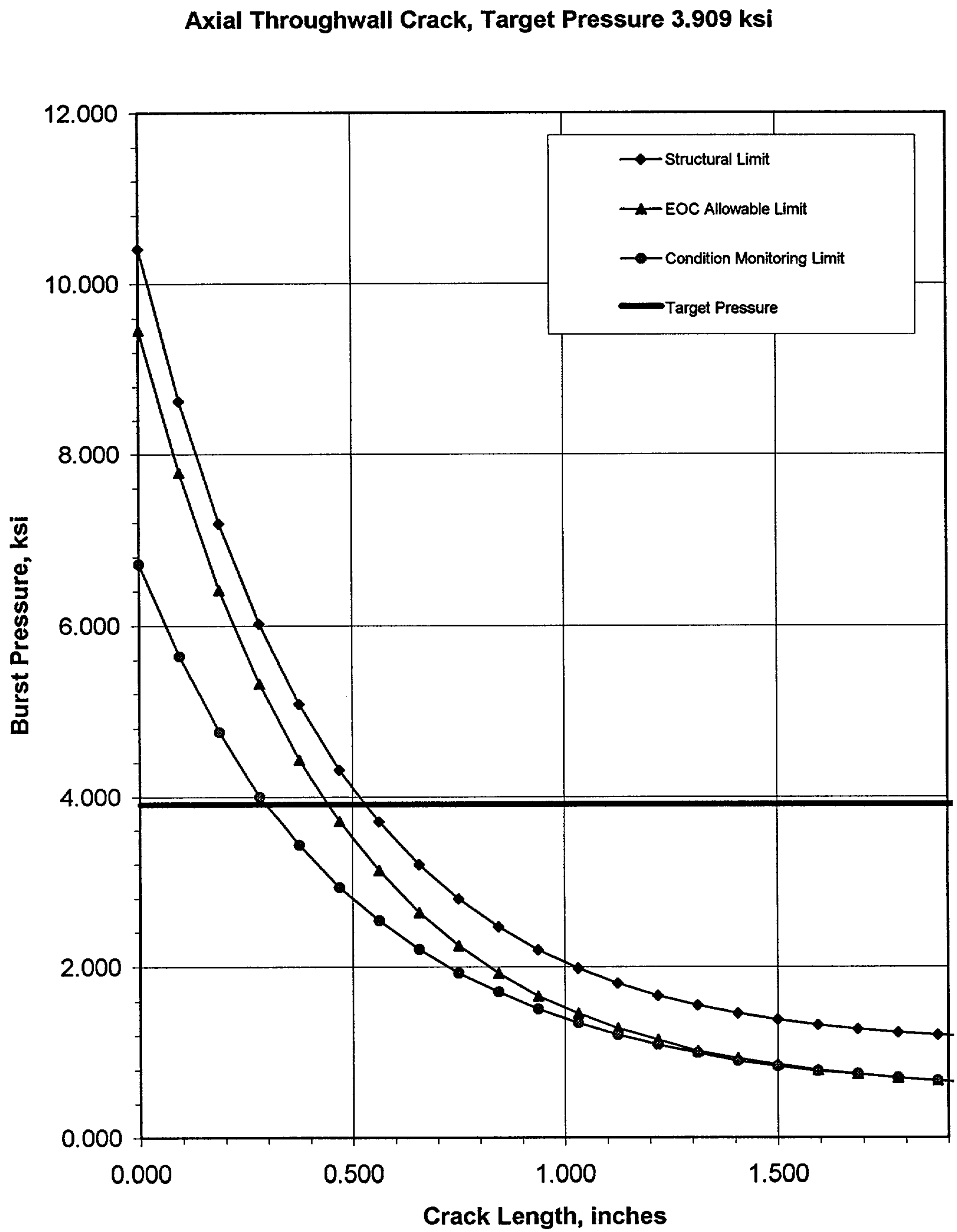
(2) The uncertainties mentioned in section 5.6.5 are burst equation uncertainties and material property uncertainties.


T-Hot to Degradation Rate Relation
Industry Wide I600 LTMA SCC (ID + OD)
Degradation Slope (Delta%/Delta T-Hot)
0.025
0.02
0.015
0.01
0.005
0
540
560
580
600
620
640
T-Hot (Deg.F)

E1-21

Axial Throughwall Crack, Target Pressure 3.876 ksi





Axial Throughwall Crack, Target Pressure 3.909 ksi
Structural Limit
EOC Allowable Limit
Condition Monitoring Limit
Target Pressure
Burst Pressure, ksi
12.000
10.000
8.000
6.000
4.000
2.000
0.000
0.000
0.500
1.000
1.500
Crack Length, inches

Question 2:

Section 5.6.5 - TVA stated that "...With regard to pre-heater wear, the 1.4% uprate conditions result in a slight increase in flow through the main feedwater nozzle which can impact the rate of wear. This slight increase in flow is not expected to result in a significant increase in the wear rate, and the resultant flow is within the pre-heater design flow..." (1) What is the flow rate through the main feedwater nozzle after the uprate? (2) What is the design flow rate for the pre-heater? (3) Does increase in T_{hot} affect the pre-heater wear?

Response to Question 2:

1) The flow rate through the main feedwater nozzle after the 1.4% uprate is 3.84 million pounds per hour.

2) The design flow rate for the preheater is [ ]$^{+a,c}$ million pounds per hour.

3) With increased temperature, an increase in the wear coefficient, along with tube wear rate could potentially occur. However, the magnitude of the temperature increase that would be necessary to produce a noticeable change would have to be significant (e.g., much greater than tens of degrees). Since the increase in T_{hot} is very small (i.e. 0.4 degree F), there would not be any noticeable change in the wear coefficient. Therefore, the small increase in T_{hot} will not produce a noticeable increase in the preheater tube wear.

Question 3:

Section 5.6.5 - TVA stated that "...For anti-vibration bar (AVB) wear, the slightly increased steam flow and reduced steam pressure can impact the flow induced vibration and wear. The revised design conditions will have a negligible impact on the projected AVB wear rate..." These two statements seem to be incongruent. The first statement indicates that the increase in steam flow and pressure reduction will affect the AVB wear. The second statement indicates that these changes will have negligible impact on the AVB wear rate. TVA needs to clarify the ambiguity.

Response to Question 3:

For anti-vibration bar (AVB) wear, evaluations have shown that a significant increase in steam flow (> 5 %) and a significant decrease in steam pressure (> 100 psi) can impact the flow induced tube vibration and wear. However, a 1.4% uprating, will slightly increase the steam flow rate and slightly decrease the steam pressure. These changes will have negligible impact on the projected AVB wear rate. Thus, the 1.4% uprate will not significantly impact future tube wear at the AVB sites.

Question 4:

Section 5.6.5 - TVA needs to address (1) whether the steam generator tubes would satisfy Regulatory Guide 1.121 under the power uprate condition. (2) the impact of the power uprate on the tube inspection during future outages.

Response to Question 4:

(1) The attached graphs referenced in the response to Question 1 illustrate the structural limit for axial through wall cracks. These graphs demonstrate that there is no affect in the 11 psi decrease in secondary side pressure. Therefore, the requirements of RG-1.121 are still satisfied.

(2) Tube inspections are driven by the degradation assessment. This uprate has not affected the degradation assessment; therefore, no changes will be made to the inspection plan for the upcoming outage. Future inspections will be determined by active degradation, potential degradation, industry experience, and plant-specific operating experience. If the temperature change affects degradation growth rates, the repair limit will be assessed during the operational assessment.

Question 5:

Section 5.6.6 - TVA performed a preliminary assessment to confirm that the existing 40% through wall plugging criteria will remain adequate for the power uprate condition. Provide the final assessments for staff review.

Response to Question 5:

The only degradation at WBN that is left in service based upon the 40% repair criteria is thinning and wear. Both of these degradation types can be bound by uniform wall thinning calculations.

High temperature burst pressure data for steam generator tubing in NUREG/CR-0718 and NUREG/CR-2336 was used to develop burst pressure equations for steam generator tubing with various types of wall thinning. For a tube exhibiting uniform thinning over the full tube circumference, the average normalized burst pressure is represented by:

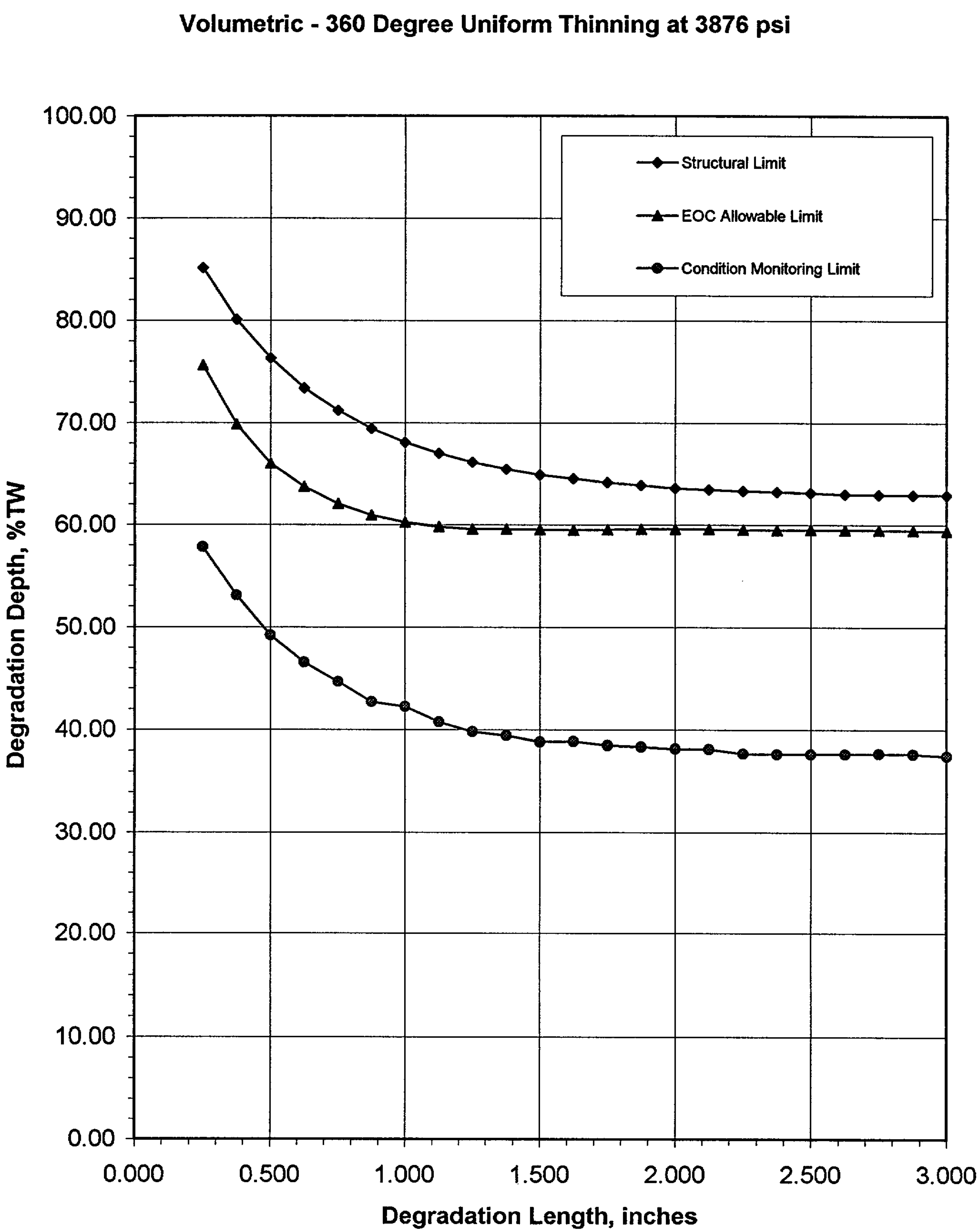
$$P_N = (1-d/t)^{\beta}\ (0.597)$$

$$\beta = 1 - EXP\left(\frac{-0.139\lambda}{\sqrt{1-d/t}}\right)$$

$$\lambda = \frac{L}{\sqrt{R_m t}}$$

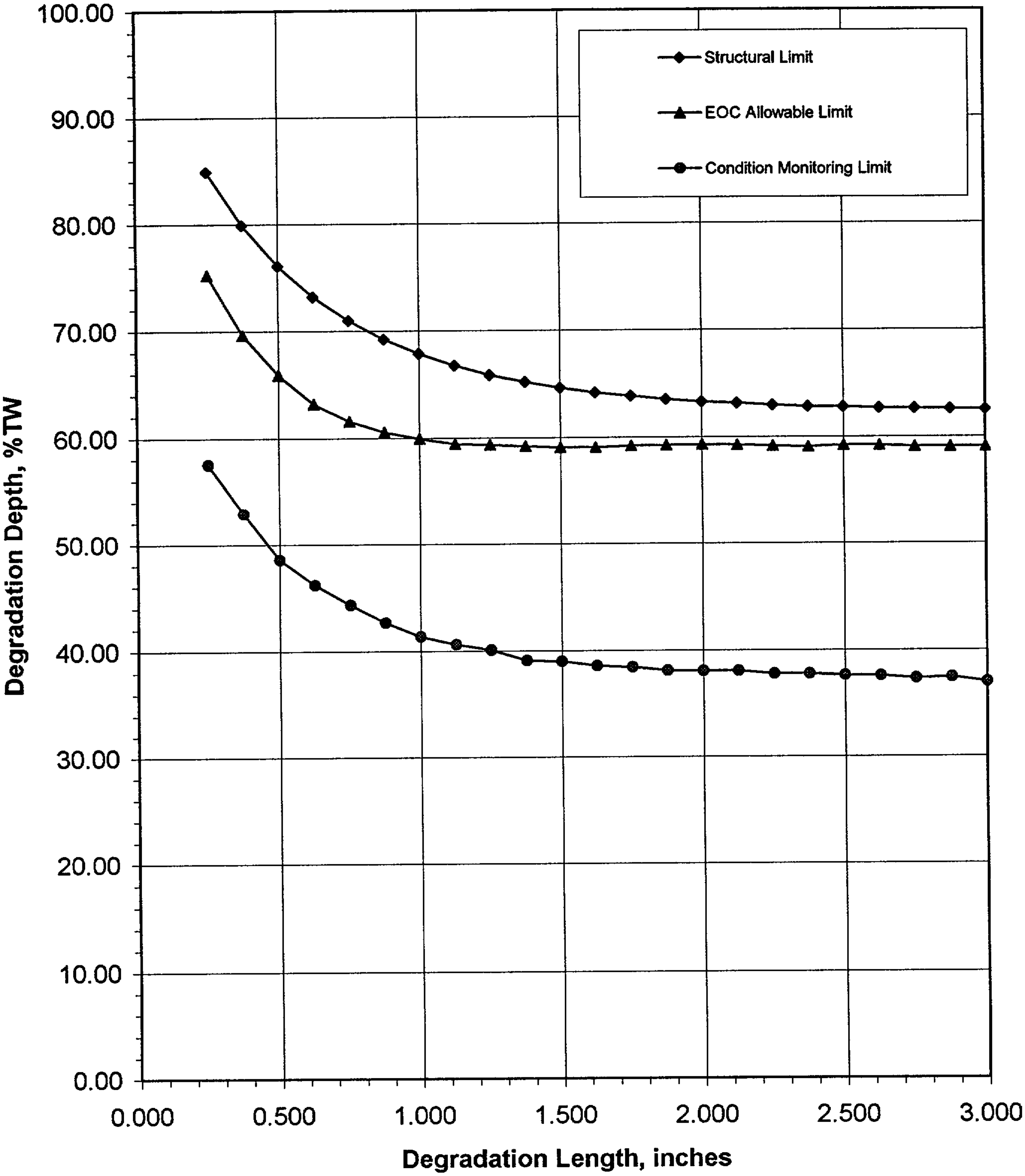
Where: P_N = Normalized burst pressure
d = degradation depth
t = tube wall thickness
L = degradation length
R_m = Mean tube radius

A Monte Carlo simulation was performed which considers the scatter about the equation and the scatter in the material properties. The end of cycle allowable limits have a 95% probability of meeting 3ΔP at a 95% confidence. The analysis results are illustrated on the attached two graphs for Volumetric - 360 Degree Uniform Thinning at both 3876 psi and 3909 psi. The end of cycle limit illustrates the operational assessment limit. Also given is the structural limit and the condition monitoring limit which considers NDE uncertainty. The length of degradation at WBN for thinning or wear would be bound by 0.5 inches. Therefore, 40% was conservative for 3876 psi and remains conservative for 3909 psi performance criteria.



Volumetric - 360 Degree Uniform Thinning at 3876 psi
Structural Limit
EOC Allowable Limit
Condition Monitoring Limit
100.00
90.00
80.00
70.00
60.00
50.00
40.00
30.00
20.00
10.00
0.00
Degradation Depth, %TW
0.000
0.500
1.000
1.500
2.000
2.500
3.000
Degradation Length, inches

Volumetric - 360 Degree Uniform Thinning at 3909 psi



Question 6:

Section 5.6.7 - Discuss whether the increase in T_{hot} would affect the proposed outside diameter stress corrosion cracking (ODSCC) voltage-based alternate repair criteria (ARC).

Response to Question 6:

As discussed in question 1, the affect of the 0.4 degree F T_{hot} increase is non quantifiable. However, if the T_{hot} increase affects growth rates of the ODSCC ARC tubes, this will be evaluated as part of the operational assessment.

Question 7:

Section 5.6.7 - TVA stated that ". . . The ODSCC ARC was developed to replace the application of the generic 40% depth plugging criterion for tube cracking at elevations corresponding to tube support plate intersections. . . " It should be noted that the ODSCC ARC are applicable only to predominate axial tube cracking at tube support plates. The ARC are not applicable to circumferential cracking. Clarify if that is the intent of the above statement.

Response to Question 7:

ODSCC ARC are only applicable to predominate axial tube cracking at tube support plates.

Question 8:

Section 5.6.7 - TVA stated that "…The loading conditions compared to applicable criteria are only operative during faulted conditions, since the tube degradation is confined to the tube/tube support plate intersection crevice during normal operation…." (1) Clarify the above statement. Specifically, what is meant by ". . .the loading conditions compared to applicable criteria are only operative during faulted conditions . . .?" (2) Do the temperature and primary-to-secondary pressure differential change for the faulted condition under power uprate?

Response to Question 8:

During normal operation, the presence of tube support plates (TSPs) prevents the burst of ODSCC indications at the TSP intersections even if the tubes are overpressurized. During faulted conditions, however, TSPs may be displaced exposing ODSCC indications to free span conditions. Therefore, the limiting condition to be considered for demonstrating acceptability of the voltage-based repair criteria stipulated in Generic Letter 95-05 is a postulated main steam line break (MSLB) event. The peak pressure differential across the steam generator tube wall during the design-basis MSLB event is determined by the pressurizer safety valve setpoint (as the secondary side is assumed depressurized to atmospheric pressure), which is not affected by the proposed uprate. Therefore, the uprating has no impact on the applicability of voltage-based repair criteria. As noted in the uprating program submittals, the F* length is also unaffected by the 1.4% uprate.

As noted above, the limiting condition for evaluating the applicability of voltage-based repair criteria is the design-basis MSLB event, and the peak primary-to-secondary pressure differential for this event is not affected by the proposed uprate. The primary temperature during the event is expected to change to the same extent as the T_{hot} increase (0.4°F), and its impact is insignificant.

Question 9:

Section 5.6.7 - TVA stated that "...the structural and leakage criteria do apply during the application of faulted loading conditions; however, these are unaffected by the 1.4% uprate....
(1) Discuss how the conclusion was reached. (2) Was there any calculations or assessments performed?"

Response to Question 9:

The structural and leakage criteria in question are those applicable to ODSCC indications in TSP crevices. As noted in the response to Question 8, these criteria are based on the peak primary-to-secondary pressure differential occurring during the design-basis MSLB event, which is not affected by the proposed 1.4% uprate. Therefore, the applicability of the Generic Letter 95-05 voltage-based repair criteria is unaffected by the 1.4% uprate.

No analysis was needed to evaluate the impact of 1.4% uprate on the applicability of the Generic Letter 95-05 voltage-based repair criteria.

Question 10:

Section 5.6.7 - TVA needs to address (1) the impact of the power uprate on tube degradation itself, i.e., would the power uprate affect the ODSCC degradation mechanism? (2) The impact of power uprate on the methodology (the assumptions and parameters used) for condition monitoring and operational assessments.

Response to Question 10:

(1) Refer to the response to question 1. The affect of the 0.4 degree F temperature change is non quantifiable. (2) The uprate would change nothing in the methodology used to perform condition monitoring and operational assessments. The primary and secondary pressure are inputs to calculations performed. These pressures are verified each inspection to ensure the limiting steady state delta pressure for the past cycle is used in calculations.

Question 11:

TVA needs to make an overall conclusion as to the structural and leakage integrity of steam generator tubes under power uprate conditions.

Response to Question 11:

The responses to the above questions illustrate that the minor change in temperature and secondary side pressure will have non-quantifiable affects on degradation rates, structural integrity, and/or leakage integrity. Steam generator inspections are driven by degradation assessments and repair decisions are driven by operational assessments. The current methodology of performing condition monitoring and operational assessments will not change due to this minor uprate.

Question on Section 4.2.5 - Steam Generator Blowdown System:

In the submittal you have indicated that the required flow rates in the steam generator blowdown system are not expected to be significantly affected by the 1.4% power uprate. The reason you gave was that the power uprate will not significantly impact addition of dissolved solids and particulates into the steam generators. Please, provide technical basis justifying that the power uprate will not significantly change dissolved solids and particulates introduced into the steam generators and there will be no need, therefore, for changing the flow rates in the blowdown system.

Response: 4.2.5 Steam Generator Blowdown System

The rate of addition of dissolved solids to the secondary systems is a function of condenser leakage and the quality of secondary makeup water. The rate of generation of particulates is a function of erosion-corrsion (E/C) within the secondary systems. Since neither condenser leakage nor the quality of secondary makeup water are impacted by power uprate, the rate of blowdown required to address dissolved solids should not be impacted by power uprate. Theoretically, the potential for E/C increases with any increase in secondary system flowrates that may result from the slightly increased flows from the uprate. However, the overall effect of the minor increases in secondary system velocities is not expected to alter the E/C rates appreciably. Therefore, the evaluation of the SGBD System concluded that the required blowdown to control secondary chemistry and particulates will not be significantly impacted by power uprate.

WBN currently maintains SG secondary chemistry within EPRI guidelines. Site goals are based upon achieving and maintaining INPO Top Decile Chemistry Performance Indicator (1.01). These limits are maintained using ETA, hydrazine, boric acid and ammonium chloride secondary chemical injection to maintain Steam Generator chemistry within the following limits:

SG Sodium	less than or equal to 0.80 ppb
SG Chloride	less than or equal to 10 ppb
SG Sulfate	less than or equal to 1.70 ppb
SG Molar Ratio	0.10 to 0.50
SG Boron	1.00 to 10.00 ppm
SG Cation Conductivity	less than or equal to 0.80 microsiemens/cm
FW ETA	2.70 to 3.30 ppm
FW Hydrazine	20 to 500 ppb
FW pH	8.80 to 10.00
FW Dissolved Oxygen	less than or equal to 5 ppb
FW Sodium	less than or equal to 5 ppb
Condensate DO	less than or equal 3.30 ppb
Hotwell Sodium	less than or equal to 0.10 ppb
Air inleakage	less than or equal to 15 cfm

WBN operates with SGBD flow of about 100 gpm to Cooling Tower blowdown for secondary chemistry control. Condensate Polishers are maintained in standby to support transient response to a Main Condenser tube leak, a Steam Generator tube leak, or other corrosion product or contaminant transients induced by large load changes or removing balance of plant equipment from service then returning it to service. ETA chemistry has pacified the secondary system components to minimize corrosion products and corrosion product fouling of stainless steel surfaces such as feedwater venturis. An initial temporary increase in corrosion products associated with increased secondary flow rates due to power uprate of 1.4% is anticipated. Secondary chemistry should remain within site goals and return to normal equilibrium within a week to ten days of this load increase. Steam Generator Blowdown flow may be increased to control this transient, but flow would be returned to normal values when chemistry values returned to normal equilibrium values. Placing Condensate Polihers in service is not anticipated to maintain site chemistry goals.

REACTOR VESSEL FLUENCE

Question:

In Section 5.1.2, TVA indicates that existing neutron fluence projections bound the corresponding projections for the 1.4% uprated conditions. What are the existing values and the uprated values?

Response to Question:

Calculated and best estimate neutron exposures [i.e., $\Phi(E > 1.0\ \text{MeV})$, $\Phi(E > 0.1\ \text{MeV})$, and dpa] at several key azimuths along the reactor vessel inner radius were determined for the 1.4% uprate fluence evaluation based on the use of historical plant operating information. In addition, corresponding future projections were determined using average historical exposure rates in conjunction with the 1.4% uprated power level.

A subset of the existing (Reference 1) versus uprated vessel fluence exposure / projection results is presented in the table below. (Although this particular subset of data is for the fast (E > 1.0

MeV) neutron fluence at the 45° azimuth only, which is where the maximum values occur, the complete set of data that forms the 1.4% uprate fluence evaluation exhibits the same trend discussed below.) The existing exposures / projections were compared to the uprated values at three discrete points in time, i.e., at 3.96 EFPY, 16 EFPY, and 32 EFPY. The trend observed from the subset of data presented below shows that the existing results always bound the uprate values.

Watts Bar Unit 1 Fast Neutron Fluence (E > 1.0 MeV)
At the Reactor Vessel Inner Radius -- 45° Azimuth

Irradiation	Calculated		Best Estimate	
Time (EFPY)	Existing Values	Uprate Values	Existing Values	Uprate Values
3.96	3.66E+18	2.38E+18	3.96E+18	2.57E+18
16.0	1.55E+19	9.79E+18	1.68E+19	1.06E+19
32.0	3.13E+19	1.96E+19	3.38E+19	2.12E+19

The existing neutron exposure parameters at the reactor vessel inner radius bounded the corresponding 1.4% uprate values for all of the neutron exposure parameters at every azimuthal position and irradiation time that was investigated. It was concluded that the existing neutron fluence projections bound the corresponding projections for the 1.4% uprate conditions. Therefore, the reactor vessel will continue to meet the requirements of Appendices G and H to 10 CFR 50 and of 10 CFR 50.61, and no additional testing beyond normal surveillance capsule analysis is currently planned.

Reference

1) WCAP-15046, "Analysis of Capsule U from the Tennessee Valley Authority Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program," June 1998 submitted to NRC on October 13, 1998.

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATION CHANGE TS-00-06
APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION
AFFIDAVIT CAW-00-1414
PROPRIETARY INFORMATION NOTICE
COPYRIGHT NOTICE

Westinghouse Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

August 21, 2000
CAW-00-1414

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 "Responses to NRC RAIs for the Watts Bar 1.4% Uprate Program"

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-00-1414 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 the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-00-1414 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: T. Carter/NRC (5E7)

CAW-00-1414

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

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 Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 21st day
of August, 2000

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 Division, of the Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 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 Westinghouse Electric Company LLC 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 "Responses to NRC RAIs for the Watts Bar 1.4% Uprate Program". 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: Mr. 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 obtaining power uprates.

(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 services 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 testing and analytical methods and performing tests.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

In order to conform to the requirements of 10 CFR 2.790 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).

COPYRIGHT NOTICE