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Subject: Response to Request for Additional Information Regarding License Amendment Application for Measurement Uncertainty Recapture Power Uprate (License Amendment Request No. 05-0007) (TAC No. MD5240)

Ladies and Gentlemen:

By letter dated April 12, 2007 (Serial No. 3198), the FirstEnergy Nuclear Operating Company (FENOC) submitted License Amendment Request (LAR) No. 05-0007. The proposed amendment would revise Technical Specifications for Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1 to accommodate an increase in the Rated Thermal Power from 2772 megawatts thermal (MWt) to 2817 MWt. Subsequently, by letter dated July 25, 2007 (Log No. 6528), the NRC provided a request for additional information concerning the LAR.

Attachment 1 provides responses to the NRC staff's questions. Enclosures 1 through 11 provide information in support of Attachment 1. Please note that Enclosures 1, 4, 7, and 10 are considered to be proprietary in their entirety. Pursuant to 10 CFR 2.390, and in accordance with the appropriate affidavits (included under separate Enclosures 2, 5, 8, and 11), it is requested that this information be withheld from public disclosure.

Attachment 2 identifies the commitment contained in this submittal.

If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, FENOC Manager - Fleet Licensing, at (330) 761-6071.

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The statements contained in this submittal, including its associated attachments and enclosures, are true and correct to the best of my knowledge and belief. I am authorized by the FirstEnergy Nuclear Operating Company to make this submittal. I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 18, 2007

By: _____
Mark B. Bezilla, Vice President-Nuclear

MKL

Attachments
1. Response to Request for Additional Information
2. Commitment List

Enclosures
1. Alden Research Laboratory, Inc., Report No. 310-01/C730, "Calibration of Two 18" Leading Edge Flow Meters for Caldon, Inc. Purchase Order Number 18350," October 2001 *(proprietary)*
2. Cameron Letter CAW 07-14 dated August 15, 2007, "Application for Withholding Proprietary Information from Public Disclosure" *(applicable to Enclosure 1)*
3. Cameron Letter dated March 8, 2007, "Cameron Measurement Systems Response to Transducer Replacement Sensitivity"
4. Cameron Engineering Report ER-202, Revision 3, "Bounding Uncertainty Analysis for Thermal Power Determination at Davis Besse Nuclear Power Station Using the LEFM√+ System," May 2007 *(proprietary)*
5. Cameron Letter CAW 07-12 dated August 7, 2007, "Application for Withholding Proprietary Information from Public Disclosure" *(applicable to Enclosure 4)*
6. Cameron Letter dated May 14, 2007, "Changes to LEFM CheckPlus System Uncertainty Owing to Transducer (Re)Placement"
7. Caldon Engineering Report ER-202, Revision 2, "Bounding Uncertainty Analysis for Thermal Power Determination at Davis Besse Nuclear Power Station Using the LEFM√+ System," July 2004 *(proprietary)*
8. Cameron Letter CAW 07-13 dated August 9, 2007, "Application for Withholding Proprietary Information from Public Disclosure" *(applicable to Enclosure 7)*
9. Davis-Besse Nuclear Power Station Drawings M-206D Rev. 18, M-206E Rev. 11, and M-207C Rev. 16
10. Caldon Engineering Report ER-227, Revision 1, "Profile Factor Calculation and Accuracy Assessment for the Davis Besse Unit 1 LEFM√+ Spool Pieces (Alden Report No. 310-01/C730)," September 2003 *(proprietary)*
11. Cameron Letter CAW 07-11 dated July 9, 2007, "Application for Withholding Proprietary Information from Public Disclosure" *(applicable to Enclosure 10)*

cc: Regional Administrator, NRC Region III
NRC/NRR Project Manager
Executive Director, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison) w/o Enclosures 1, 4, 7, and 10
NRC Senior Resident Inspector
Utility Radiological Safety Board w/o Enclosures 1, 4, 7, and 10

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

To complete their review, the NRC staff has requested additional information regarding the license amendment application for the measurement uncertainty recapture power uprate (License Amendment Request No. 05-0007). FENOC's response to this request is provided below.

1. **Engineering Instrumentation & Controls Branch (EICB): Enclosure 1 stated in section 5.2 that the limiting safety system setpoints (LSSS) for the reactor protection system high flux functional unit is the limiting trip setpoint specified in updated final safety analysis report (UFSAR) technical requirements manual and the allowable value relationship to the setpoint methodology and testing requirements in the technical specifications is documented in the setpoint calculation which is maintained as part of plant records. Please submit this calculation assuring that the calculation documents (including sample calculation) the methodology used for establishing the limiting set point or the nominal set point and the limiting acceptable values for the as-found and as-left set point as measured in periodic surveillance testing. This calculation should also indicate the related analytical limit and other limiting design values (and the sources of these values) for the high flux functional unit LSSS.**

DBNPS Response:

The current analytical limit for the Reactor Protection System (RPS) high flux trip setpoint is based on 112% of 2772 MWt. The analytical limit for the MUR is 110.2% of 2817 MWt. These values, 112% of 2772 MWt and 110.2% of 2817 MWt, are numerically the same.

The field trip setpoint calculation is currently under development. The calculation is based upon the analytical limit and allowable values described in the license amendment application, and employs current industry accepted methodology to determine the field setpoints for the RPS high flux (overpower) trip. Specifically, the calculation utilizes ISA-67.04.01-2000, "Setpoints for Nuclear Safety Related Instrumentation," and Method 1 from Section 7.3 of ISA RP67.04.02 – 2000, "Determination of Setpoints for Nuclear Safety Related Instrumentation," to determine Limiting and Nominal Trip setpoints.

Upon internal approval of the above-mentioned calculation, FENOC will make the calculation available for NRC staff review and discussion.

2. EICB: Section I.1 in Enclosure 2 states that the Leading Edge Flow Meter (LEFM) flow meters (one in each of the two steam generator feedwater flow headers) were calibrated at the Alden Research Laboratory using the plant's current piping configuration and variations of the plant's configuration. Please explain those variations of the plant's configuration and submit the calibration report.

DBNPS Response:

Alden Research Laboratory set up of the test piping on the inlet of the LEFM to replicate the plant piping configurations with the LEFM installed. The "variations" discussed referred to whether or not a flow straightener was installed on the inlet of the test configuration. The flow straightener variation is explained in greater detail within the response to question 21 of this RAI. A copy of the proprietary calibration report is included as Enclosure 1. The associated affidavit is included as Enclosure 2.

3. EICB: Section 2.1(2) in Enclosure 3 states that the correspondence between the plant computer IDs and the variables used in CTPA was not formally provided to AREVA NP and, therefore, the information is assumed. The heat balance uncertainty calculations in enclosure 3 was performed by AREVA NP using the assumed values. Please explain the validity of the assumed values in this calculation and how the actual values will affect the heat balance uncertainty.

DBNPS Response:

In section 2.1 of the heat balance calculation, the following statement is made regarding the core thermal power analysis (CTPA) software:

> The correspondence between the plant computer IDs and the variables used in CTPA was not formally provided to AREVA NP. Thus, the information shown is assumed.

In section 3.5 of the heat balance calculation, the following statement is made:

> A listing of Davis-Besse computer points that are input to the current (pre-Caldon instrumentation) core thermal power calculation is provided for information. This table was provided informally to AREVA NP.

This assumed information was provided for information only and not for design input.

The key parameters for the heat balance uncertainty calculations are: (1) feedwater flow rate, (2) feedwater temperature, (3) steam temperature, and (4) steam pressure. The feedwater flow and temperature uncertainties were obtained from Cameron and the steam temperature and pressure uncertainties were provided by FENOC for the Davis-Besse specific instrumentation. Therefore, there is no effect of the assumption regarding the plant computer IDs on the results of the heat

balance uncertainty calculation.

4. **EICB: Section I.1.E in Enclosure 2 states that Cameron Measurement Systems (formerly Caldon Inc.) has performed an evaluation of the uncertainty involved in replacing LEFM Check Plus transducers in the field. Please submit this calculation.**

DBNPS Response:

A March 8, 2007 letter from Ed Madera to Tim Laurer, "Cameron Measurement Systems Response to Transducer Replacement Sensitivity," (Enclosure 3) discusses the effect of transducer replacement on uncertainty. The letter cites an increase in the overall mass flow uncertainty from 0.26% to 0.29%, and proposes to revise the Davis-Besse analysis within 90 days to reflect the new terms. A copy of the revised analysis, Cameron Report ER-202 Revision 3 (proprietary) is included as Enclosure 4. The associated affidavit is included as Enclosure 5.

A May 14, 2007 letter from Ed Madera to Brian Young, "Changes to LEFM CheckPlus System Uncertainty owing to Transducer (Re)Placement," (Enclosure 6) discusses additional tests and analysis at the request of the NRC to verify the effects of transducer replacement are bounded by the LEFM CheckPlus uncertainty analysis. The effect on Davis-Besse LEFM CheckPlus uncertainty is to increase the total feedwater mass flow uncertainty by 0.02%. This letter includes Customer Information Bulletin (CIB) 125, Revision 0, which, in turn, references Cameron Engineering Report ER-551P, Revision 1, "LEFM CheckPlus Transducer Installation Sensitivity." Proprietary document ER-551P and its associated affidavit are available in ADAMS as non-publicly available documents; please refer to ADAMS Accession Numbers ML071500360 and ML071500358.

5. **Component Performance & Training Branch: Section IV.1.A.ix, Safety-Related Valves - Describe whether the design bases of safety-related valves have been evaluated for the pressure changes due to the measurement uncertainty recapture power uprate. Also explain whether pressure locking effects on safety-related power-operated valves have been evaluated for any change in differential pressure per the recommendations in Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves."**

DBNPS Response:

Table VIII.6-1 of Enclosure 2 of the April 12, 2007 license amendment application provides the calculated plant operating conditions for the power uprate. The table shows minor changes to the Reactor Coolant System (RCS) T_{hot} and T_{cold} of approximately 1°F. RCS pressure for the uprate conditions remains the same. Steam pressure, feedwater pressure and feedwater temperature also

remain essentially the same. Minor changes to steam temperature are predicted to occur with the uprate.

Pressure locking and thermal binding effects on safety related motor operated valves that were evaluated previously in accordance with the recommendations of Generic Letter 95-07 were reviewed and evaluated against the MUR operating conditions. This review determined that the MUR conditions do not produce any changes to the present assumptions, operating restrictions or design features used in demonstrating that the subject valves are not susceptible to pressure locking or thermal binding.

6. **Fire Protection-1: LAR, Enclosure 2, Attachment A, "D-B MUR [measurement uncertainty recapture] Summary Report" Section II "Accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level," mentions safe-shutdown fire analysis. However this section does not discuss the impact of measurement uncertainty recapture power uprate on the fire protection system(s). Clarify whether this request involves changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability in accordance with Title10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix R. Provide the technical justification for whether and, if so, why, existing analyses bound any impact on accidents or transients resulting from any changes.**

DBNPS Response:

DB has performed various analyses to demonstrate safe plant shutdown following a fire. Two calculations to show adequate core cooling were completed to address the two potential results of a fire. Calculation C-NSA-064.02-032, "Davis-Besse Appendix R Overheating Summary Report," analyzed the potential over-heating conditions (i.e., actuations/failures leading to a loss of feedwater, loss of Makeup/High Pressure Injection (HPI), and spurious opening of Reactor Coolant System (RCS) leak paths). Calculation C-NSA-064.02-033, "Davis-Besse Appendix R Overcooling Summary Report," analyzed the potential over-cooling conditions (i.e., full Auxiliary Feedwater (AFW) flow and spurious opening of RCS leak paths).

The analyses were performed with the objective of updating the analyses to current standards and quantifying available margins by using updated computer models. The updated analyses are used to define time critical operator actions. These calculations will also support an increase in the Davis-Besse rated core power from 2772 to 3014 MWt.

The overcooling analysis was performed using the topically-approved RELAP5/MOD2 computer code with modified boundary conditions from the main steam line break accident (MSLB). For the MSLB accidents (or overcooling type of events), a nominal core power is used to minimize heat input to the RCS. The results of the analyses using a nominal core power level of 2772 MWt is conservative for a higher core power level because the core decay heat will be

minimized. Less core decay heat results in a greater overcooling of the RCS and a greater challenge to maintaining the minimum subcritical margin. The overcooling analyses determined the minimum operator action time to manage RCS makeup flow, core reactivity, and SG overfill concerns.

The overheating analysis was also performed using the RELAP5/MOD2 computer code with modified boundary conditions from the small break loss of coolant accident (SBLOCA) evaluation model (EM). Successful core cooling was demonstrated for core thermal power levels up to 3025.32 MWt, with flow from only one AFW and ECCS train. That power level bounds the power level proposed by the MUR uprate.

Based on the above discussion, the proposed license amendment involves no changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability, as the existing analyses are bounding.

7. **Fire Protection-2: The NRC staff notes that LAR, Enclosure 2, Attachment A, "D-B MUR Summary Report" Section III, "Accidents and transients for which the existing analyses of record do not bound plant operation at the proposed uprated power level," does not include any discussion regarding changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe shutdown capability in accordance with Appendix R. Clarify whether this request involves changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability in accordance with 10 CFR Part 50, Appendix R. Provide the technical justification for whether and, if so, why, existing analyses do not bound any impact on accidents or transients resulting from any changes.**

DBNPS Response:

As indicated in the response to the previous question, the proposed license amendment involves no changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability. The existing analyses are bounding.

8. **Fire Protection-3: Section VI, "Post Fire Safe Shutdown Capability," of the NRC safety evaluation report dated May 30, 1991, on page 29 states that:**

   The NRC staff's conclusion is also based on the statements made by the licensee in its letter dated June 6, 1988, that the capability to return the pressurizer level to within prescribed instrument indication range, and to restore other process variables to within the range predicted by a loss of offsite power, will be preserved. In addition, the licensee states that the core will not be uncovered and fission product boundary integrity will not be affected during the postulated transient conditions.

The NRC criteria which is applicable to the DBNPS post-fire safe-shutdown is contained in Sections III.G and III.L of Appendix R to 10 CFR Part 50, in GL 81-12, "Fire Protection Rule (45 FR 76602, November 19, 1980)," and its subsequent clarification in GL 86-10, "Implementation of Fire Protection Requirements." The NRC staff requests the licensee to verify whether the above conclusion is valid at an increased reactor power level of 2817 megawatts thermal (MWt), 1.63% above the currently licensed power level of 2772 MWt.

DBNPS Response:

The conclusion is still valid for the revised power levels. As discussed in the response to question 6, the DB Appendix R Overheating analysis was performed at a reactor power level of 3014 MWt. One of the acceptance criteria of the overheating analysis was to maintain the core covered. The acceptance criteria (see below) references the June 6, 1988 letter (DBNPS Serial Number 1535) mentioned in the question above. Appendix B of Attachment 1 of the calculation discusses the following acceptance criteria in more detail, including the 1988 letter and the 1991 SER.

> The RCS inventory will be permitted to be depleted until an unrecoverable condition is reached. An unrecoverable condition is defined in accordance with Reference 23 [June 6, 1988 letter mentioned above] as "the loss of any shutdown function(s) for such a duration as to ultimately cause the reactor coolant collapsed liquid level to fall below the top of the active fuel height of the core and subsequent breach of the fuel cladding. Maintaining the reactor coolant level above the top of the core ensures adequate core cooling and fission product boundary integrity." This definition of minimum allowable RCS inventory ensures that core cooling will be maintained for the duration of the event.

9. **Vessels & Internals Integrity Branch (CVIB): Section IV.1.C.i, "Pressurized Thermal Shock (PTS)," of Enclosure 2 to the submittal dated April 12, 2007, indicates that the MUR power uprate projected fluence at 32 effective full power years (EFPY) for the limiting reactor vessel material, Weld WF-182-1, is 1.124 x 10^{19} n/cm^2 (E>1.0 MeV). However, Section IV.1.C.v, "Effect on Low Upper Shelf Energy," of Enclosure 2 indicates that the MUR projected inner-diameter (ID) fluence at 32 EFPY for this material is 1.02 x 10^{19} n/cm^2 (E>1.0 MeV). Please explain the discrepancy between these two ID fluence values.**

DBNPS Response:

Section IV.1.C.i states "...considering a 32 EFPY inside surface fluence of 1.124 x10^{19} n/cm^2 (E > 1.0 MeV) based on a 1992 fluence projection plus 5% to account for the MUR power uprate. The 1992 fluence projection bounds a more recent 2006 fluence projection for the

32 EFPY inside surface fluence and provides a more conservative RT_{PTS} value."

Section IV.1.C.v states "...at 32 EFPY considering an inside surface fluence of 1.02 $x10^{19}$ n/cm^2 (E>1.0 MeV) based on a 2006 fluence projection."

The PTS and USE evaluations were performed at different times using different fluence projections. The latest applicable fluence projection is from 2006 (used for the low upper-shelf energy predication), and the 1992 +5% fluence used in the PTS evaluation is bounded by it as stated in the text. The PTS evaluation was performed prior to 2006, but not updated due to the fact that the limiting fluence decreased.

10. **CVIB: Section IV.1.A.viii, "Pressurizer Structural Evaluation," of Enclosure 2 to the submittal indicates that your pressurizer structural evaluation determined that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. What temperature change (e.g., hot leg and cold leg temperature change) do you refer to? How does this temperature change affect the most critical transient that was used in the existing pressurizer integrity analysis? Identification of new pressurizer insurges in recent years has caused reevaluation of pressurizer integrity for several pressurized-water-reactor (PWR) plants. Confirm that you have considered appropriate pressurizer insurges in your design transients.**

DBNPS Response:

Pressurizer insurges and outsurges have been evaluated for Davis-Besse. The temperature change that is referred to is the hot leg temperature which will increase slightly (< 0.4F) for the MUR. This higher hot leg temperature results in a slightly lower temperature differential between the hot leg and the pressurizer and lessens the effect of the insurges. However, the critical Pressurizer insurge and outsurge transients occur during the heatup and cooldown transients which are not affected by the MUR.

11. **CVIB: Enclosure 2 to the submittal provides very little information regarding your reactor vessel (RV) internals structural evaluation. Table Matrix-1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," provides the NRC staff's basis for evaluating the potential for extended power uprates to induce aging effects on RV internals. Depending on the magnitude of the projected RV internals fluence, Table Matrix-1 may be applicable to the MUR application. In the Notes to Table Matrix-1, the NRC staff states that guidance on the neutron irradiation-related threshold for irradiation-assisted stress corrosion cracking (SCC) for PWR RV internal components are given in BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," and WCAP-14577, Revision l-A, "License Renewal Evaluation: Aging Management for Reactor Internals." The Notes to Table Matrix-1 state that for thermal and neutron embrittlement of cast austenitic**

stainless steel, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs to investigate degradation effects and determine appropriate management programs. Discuss your management of the above-mentioned aging effects on RV internals in light of the guidance in BAW-2248A and WCAP-14577, Revision 1-A. Please also confirm whether you have established an inspection plan to manage the age-related degradation in the DBNPS RV internals, or whether you have participated in the industry's initiatives on age-related degradation of PWR RV internals.

DBNPS Response:

The EPRI Materials Reliability Project (MRP) Issues Task Group (ITG), of which FENOC is an active participant, is addressing the age-related degradation effects on reactor internals components.

FENOC will continue to monitor the industry inspection findings and operating experience, as part of the general assessment of any need to increase the monitoring and inspection activities above the current ASME Section XI program. Since several other plants of similar design and configuration have been operating at extended power levels for several years and have fluences comparable to those at Davis-Besse, these plants will provide insight as to the need for any increased actions during the current license life.

The EPRI MRP committee is actively working with plants with license renewal commitments to submit inspection plans and procedures in the 2007 time frame. These plans and procedures are being developed and issued by the EPRI MRP committee for the management of the aging mechanisms during the license renewal period. The EPRI MRP committee will issue these documents through the protocols developed under NEI 03-08, "Guideline for the Management of Materials Issues."

12. **Steam Generator Tube Integrity & Chem. Engineering Branch (CSGB): Confirm that the steam generators (SG) will continue to satisfy all original design criteria under power uprate conditions. In addition, confirm that your analysis addresses the current condition of your SGs (e.g., plugs, tube repairs, loose parts, etc.) and addresses flow induced vibration.**

DBNPS Response:

The design and operational functions of the once-through steam generators (OTSGs) were reviewed as part of the MUR uprate effort. The review included items such as heat transfer, inventory, transient performance, tube loading, natural circulation heat removal, and design pressure and temperature. It was concluded in all cases that the steam generator will continue to satisfy all of the design and operational functions at the MUR power level.

A review was performed to evaluate the impact of the revised operating conditions associated with the MUR on the existing qualification reports and design calculations for the mechanical plugs, welded plugs, tube sleeves and tube stabilizers. Steam generator loads with the existing repair products remain valid. The temperature changes due to MUR are bounded by those used in the tube repair product qualifications and analyses. The effect of the flow increase was also evaluated and it was shown that all installed tube repair hardware maintained functional integrity with the increased secondary side flow rates associated with MUR.

An evaluation was performed to address flow-induced vibration (FIV) implications associated with the MUR power uprate conditions and the OTSG tube bundle and installed tube repair hardware. The evaluation was based on previous FIV analyses of virgin tubes and tubes that are plugged and stabilized, considering all of the types of stabilizers that have been installed in the DB-1 OTSGs.

The FIV analyses for plugged, stabilized tubes assumed a complete sever at the limiting location based on the uprate thermal-hydraulic conditions. Scaling factors were determined based on the ratio of dynamic pressures at the nominal conditions of 2772 MWt with 0% plugging to the dynamic pressures at the power uprate conditions of 2819 MWt with 20% plugging. The scaling factors were determined at the worst-case locations of the tube bundle entrance and exit; however, the FIV evaluation conservatively applied the maximum scaling factor over the entire length of the tube.

The maximum scaling factor was applied to the worst case tubes from previous analyses for fluid-elastic instability, random turbulence-induced vibration, and vortex shedding-induced vibration. At the uprate conditions, the margin of safety against fluid-elastic instability is at least 23% for all tubes and stabilized tubes (FSM > 1.23). The original design bases for the DB-1 OTSGs applied a minimum FSM of 1.0 for the fluid-elastic instability analysis. The minimum margin against high cycle fatigue associated with turbulence-induced vibrations considering both in-service and stabilized tubes was greater than 32.5%. Therefore, the results of the evaluation show that the tube bundle in the DB-1 OTSGs will not fail due to the high cycle fatigue effects of flow-induced vibration resulting from turbulence due to cross flow conditions at the uprate power level nor will tube-to-tube impacts occur over the remaining life of the plant.

There is no evidence of any secondary-side loose parts in the Davis-Besse once-through steam generators. Therefore, no tube wear from loose parts is expected. The flow increase is not expected to result in any loose parts.

The FIV evaluation reviewed concerns from NRC Bulletin 88-02 and NRC Information Notice 2002-02 in relation to the DB-1 power uprate. These concerns have been addressed at DB-1 or are covered by the current stabilization criteria.

13. CSGB: Provide confirmation that your SG tube plugging limit is still appropriate for power uprate conditions, given the guidance in Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes.

DBNPS Response:

A detailed review of the Davis-Besse analyses was performed to evaluate the potential effects of the power uprate operating conditions on the integrity of the steam generator tubes. The review included the tube Topical Report BAW-10146, which provides the detailed evaluation of degraded tubes performed to satisfy the requirements defined in Draft Regulatory Guide 1.121. The assessment concluded that the power uprate-associated changes in average tube temperatures, tube-to-shell temperature differences, primary-to-secondary pressures, and resulting tube axial loads are negligible and do not affect the structural integrity of either the un-degraded or degraded steam generator tubes. Therefore, the SG tube plugging limit is still appropriate for power uprate conditions.

14. CSGB: Confirm that the coating qualification temperature and pressure profile used to qualify the original maintenance Service Level I coatings continues to bound the design basis accident temperature and pressure profile under power uprate conditions.

DBNPS Response:

The Davis-Besse design basis accident (DBA) testing for Service Level 1 protective coating systems were evaluated for continued applicability for the power uprate. The changes in the DBA temperature and pressure profile under power uprate conditions remain bounded by the applicable Service Level 1 coating system DBA test temperature and pressure profiles.

15. CSGB: Please confirm the following regarding the SG blowdown system:

a. That you considered whether the additional operating time due to the power uprate will result in system components to be more susceptible to flow accelerated corrosion (FAC).

b. That your current evaluation of the SG blowdown system under power uprate conditions considered the effect of a potential increase of impurities in the SG water.

c. That any change to the inlet pressure of the SG blowdown system is still inside the range of operating parameters for the power uprate.

DBNPS Response:

For the MUR, the secondary system was evaluated for erosion and corrosion due to changes in the initial conditions. The evaluation concluded that the MUR conditions, pressure, temperature and flow were bound by the design conditions used in the Davis-Besse FAC Program. Therefore, the predicted increases in maximum component wear rates and reductions in service lives would continue to be managed by the Davis-Besse FAC program.

The SG blowdown system is not operational on a B&W plant above approximately 15% reactor power, and is removed from service above that power. The plant chemistry requirements are not being revised for the MUR. Therefore, the total dissolved solids measured with the plant secondary chemistry will be in specification prior to going above 15% reactor power. A potential increase of impurities in the SG water will be accommodated by the plant condensate demineralizer system.

The SG blowdown system is designed with a blowdown capability at power levels at or below about 15% so that solids can be removed from the SGs that do not exit in the normal steam flow. At a SG operating pressure of 925 psig, the flow rate is 150 gpm per SG. With no pressure, the system is designed to pass 32 gpm per SG. There will be a slight increase in secondary side flow (at 15% power) due to the power uprate of approximately 1.6%, but the affect on the pressure in the SG is not significant, less than 0.5 psi. Since the power uprate does not affect the pressure that the SGs are controlled, there will be no affect on the blowdown system.

16. **CSGB: You indicated that "the predicted increases in maximum component wear rates and reductions in service lives can be managed by the DBNPS FAC program." Discuss how significant the increases in wear rates and reductions in service lives are for the power uprate conditions. In addition, discuss any changes made to DBNPS FAC program (i.e., criteria used for selecting components for inspection following the power uprate, criteria for repair and replacement, increased inspection scope, etc.) due to power uprate conditions. Also, identify the systems that are expected to experience the greatest increase in wear as a result of the power uprate. Discuss whether inspections will be performed to assess wear prior to entering power uprate conditions.**

DBNPS Response:

Component wear rates and reductions in service lives were evaluated previously at a power level (~8.8% uprate) that exceeds the measurement uncertainty recapture (MUR) power uprate, using the EPRI CHECWORKS FAC monitoring program. The previous evaluation at the higher power level quantified the effect on wear rates and service lives and concluded that the impacts were small enough that the affected systems and components could continue to be managed through the Davis-Besse FAC program. The increases in pressure, temperature, and flow in the affected systems for the MUR power uprate are bounded by the previous evaluation and thus the

conclusions reached remain valid.

The previous evaluation at the ~8.8% power uprate conditions showed negligible impact for most of the systems reviewed. It was determined that the ~8.8% power uprate would significantly affect wear rates and service lives for some Condensate system components. The evaluation provided the remaining service life values for the Condensate system, broken down by Wear Rate Analysis (WRA) line segments. The worst case was 30,402 hours of service life remaining for WRA line segment "Condensate 3", based on the minimum component in the line segment. The average service life of the components in this line segment is 1,702,229 hours for the ~8.8% uprate conditions. The evaluation concluded that although this is a significant decrease in service life from existing conditions, the remaining service life is a large enough amount of time (about 3.5 years) that the decrease is manageable within the FAC program. For the MUR power uprate, the impact on the Condensate system will be less significant. For other systems, the impact was determined to be negligible even at the ~8.8% power uprate conditions.

Due to the favorable results of the evaluation, it was determined that no changes were needed to the Davis-Besse FAC program, and no additional inspections have been recommended prior to operation at power uprate conditions.

17. CSGB: Provide confirmation that your evaluation for the chemical and volume control system demonstrates that the conditions for the power uprate are bounded by the existing conditions (thermal performance, letdown and makeup requirements, etc.).

DBNPS Response:

The plant-specific system corresponding to the chemical and volume control system is the Makeup and Purification system. The principal function of this system is to provide a path for supplying makeup liquid to the Reactor Coolant System (RCS) and Reactor Coolant Pump (RCP) mechanical seals, and a means for processing primary coolant (i.e., boron concentration and sampling). There are no required changes to the Makeup System configuration for the MUR. The nominal RCS pressure, RCS flow and average system temperature are not affected by the power uprate, therefore the amount of coolant required to offset temperature changes will not be affected. The boric acid content of Borated Water Storage Tank (BWST) and Boric Acid Addition Systems are verified in the core design process so that the required ability to add adequate amounts of negative reactivity are not impaired nor affected by the MUR power uprate. The hot leg and cold leg temperatures will change by 0.4F and as a result, the letdown line will experience a slightly lower temperature as a result of the power uprate. This means that the letdown coolers are bounded by current operation and there is no adverse impact on the cooling function of the letdown coolers.

18. Reactor Systems Branch (SRXB): Please provide a copy of Caldon, Inc. Engineering Report: "Bounding Uncertainty Analysis for Thermal Power Determination at Davis Besse Nuclear Power Station Using LEFM✓+ System," July 2004.

DBNPS Response:

Cameron Report ER-202 Revision 2 (proprietary) is included as Enclosure 7. The associated affidavit is included as Enclosure 8.

19. SRXB: In the letter dated March 8, 2007, from Ed Madera, Cameron Measurement Systems Sr. Project Engineer, to Tim Laurer, Nuclear Staff Engineer, DBNPS, it is stated that Cameron proposed to provide a revised analysis that reflects the uncertainty associated with transducer replacement within 90 days. Please provide a copy of that information when it becomes available or in your response to this RAI.

DBNPS Response:

As noted in the response to question 4, Cameron Report ER-202 Revision 3 (proprietary) is included as Enclosure 4, and the associated affidavit is included as Enclosure 5.

20. SRXB: Please provide a description and drawings that illustrate the feedwater piping configuration from the outlet of the feedwater pumps to the containment pressure boundary. Identify any perturbations in the piping wall that could affect the flow profile.

DBNPS Response:

Enclosure 9 provides piping isometrics showing the requested feedwater piping configuration. As shown on the isometrics, each main feedwater pump discharge is routed through three sets of feedwater heaters. Downstream of the feedwater heaters, the lines combine into a common header, which then splits into separate 18-inch diameter lines, one line to each steam generator. The LEFMs are installed in each of the 18 inch diameter lines downstream of the common header.

The field installation of the Caldon flow meters does not contain any perturbations in the piping wall that could affect the flow profile.

21. SRXB: Please provide a description and drawings of the Alden Laboratory test configuration used for the plant's current piping configuration and variations of the plant's configuration. Identify any differences between the test and plant configurations. Reference I-10 from your submittal of April 12, 2007, may be provided to address part of this request to alleviate preparation of additional documentation.

DBNPS Response:

Reference I-10, Alden Research Laboratory Inc. Report Number ARL-310-01/C730 (proprietary), is included as Enclosure 1. This report was used to develop proprietary Caldon Engineering Report 227 (ER-227), a copy of which is included as Enclosure 10. The associated affidavit is included as Enclosure 11.

The test configuration, as described in Section 2.3 of ER-227 (pages 5 and 6), can be compared to the as-built plant configuration using drawing M-206D (included as part of Enclosure 9).

The test configuration had on the outlet of the flow straighteners, 18 feet of 18 inch diameter schedule 80 straight pipe prior to an elbow. The elbow was an 18 inch diameter schedule 80, 90 degree long radius elbow. On the outlet of the elbow was 93 inches of 18 inch diameter schedule 80 straight pipe, followed by the LEFM flow meters. Since a long radius elbow measures 27 inches from the centerline of the inlet to the edge of the outlet the total distance from the centerline of the elbow inlet to the LEFM flow meters is 10 feet.

For the plant configuration, main feedwater (MFW) train 1 has approximately 33 feet of 18 inch diameter schedule 80 straight pipe prior to the 90 degree elbow, and MFW train 2 has approximately 31 feet of 18 inch diameter schedule 80 straight pipe prior to the 90 degree elbow. On both MFW lines there is 10 feet of 18 inch diameter schedule 90 pipe between the centerline of the elbow's inlet and the LEFM. FENOC is unable to confirm from plant drawings whether the 90 degree elbow is a long radius elbow, as was used in the test configuration. However, this potential difference between test configuration and plant configuration would not be significant, based on the long length of straight pipe between the outlet of the elbow and the inlet of the LEFM.

Note that the identical configuration of MFW trains 1 and 2 downstream of the elbows allowed Alden Research Laboratory to use a single testing loop. As described in section 2.3 of ER-227, the LEFMs were interchanged during the testing. Also, test variations were run with the LEFMs rotated 90 degrees during testing for comparison. As described in section 2.4.1 of ER-227, one test variation was run with no flow straightener.

One notable difference in piping configuration between the test set up and plant installation is the length of piping on the inlet to the elbow that is prior to the LEFM. Whereas the tested configuration included flow straighteners as part of the standard test equipment used by Alden Research Laboratory, the plant has a longer length of 18 inch diameter straight pipe prior to the

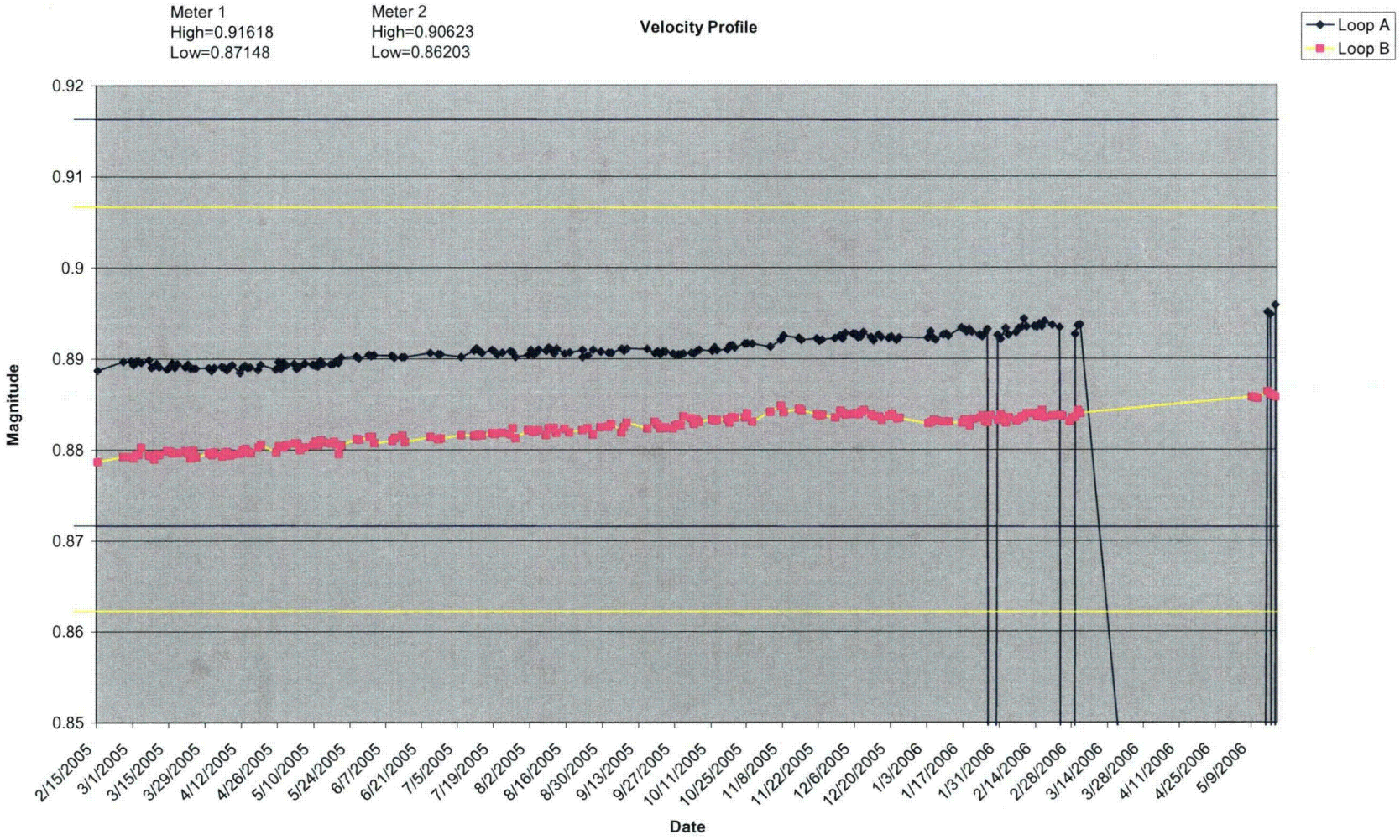
LEFM, which allows for more natural straightening of the flow prior to entering the elbow. This difference is judged to be insignificant.

The other notable difference between the test set up and the plant installation is that the test piping was laid out horizontally while the plant piping is a vertical run of piping up to the elbow, with the flow direction from bottom to top. This difference is judged to be insignificant.

22. SRXB: Please provide a summary of the Alden Laboratory test results and application/comparison of those test results to plant operation. Include a representative set of test data if not provided in Request 21, above.

DBNPS Response:

The following graph shows a trend of the velocity profiles from February of 2005 until May 2006. This plant data is consistent with velocity profile data shown in Table 3 (page 9) of ER-227 (Enclosure 10). The rise in the velocity profile of the plant data is believed to be due to the decreasing signal to noise ratio as the transducers aged over time.


Meter 1
High=0.91618
Low=0.87148
Meter 2
High=0.90623
Low=0.86203
Velocity Profile
Loop A
Loop B
Magnitude
0.92
0.91
0.9
0.89
0.88
0.87
0.86
0.85
2/15/2005
3/1/2005
3/15/2005
3/29/2005
4/12/2005
4/26/2005
5/10/2005
5/24/2005
6/7/2005
6/21/2005
7/5/2005
7/19/2005
8/2/2005
8/16/2005
8/30/2005
9/13/2005
9/27/2005
10/11/2005
10/25/2005
11/8/2005
11/22/2005
12/6/2005
12/20/2005
1/3/2006
1/17/2006
1/31/2006
2/14/2006
2/28/2006
3/14/2006
3/28/2006
4/11/2006
4/25/2006
5/9/2006
Date

23. SRXB: Please provide a summary of LEFM characteristics before and after replacement of the 32 transducers in June 2006. This summary should contain a comparison to other feedwater measurement instruments over a sufficient time span to enable a valid comparison of before and after characteristics.

DBNPS Response:

The following graphs are provided:

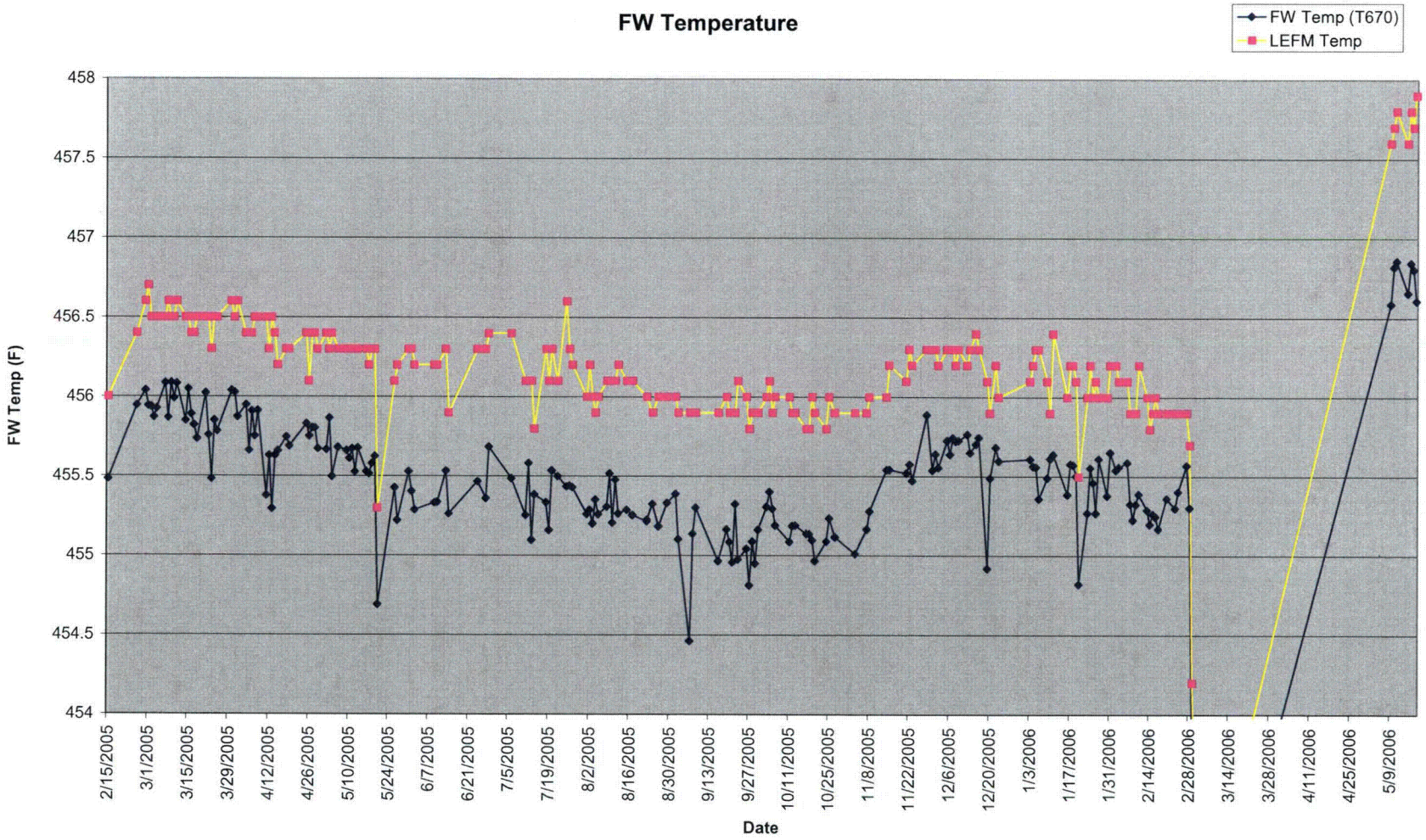
- Feedwater temperature output of the LEFM and RTD prior to and after June of 2006.
- Pathway gains prior to and after June of 2006.

A review of the data for Feedwater (FW) temperature, which is one of the inputs to the heat balance calculation, shows consistency in trending between computer point C224 (LEFM FW temperature) and computer point T670 (RTD FW temperature) before and after June of 2006. The trends in LEFM FW temperature show no changes in the performance before or after the change out of the transducers in June of 2006. The downward spike in data for the graph of FW temperature prior to June of 2006 was due to the 14th Refueling Outage (14RFO). The downward spikes in the data for the graph of FW temperature after June of 2006 were due to plant maintenance outages or maintenance on the LEFM.

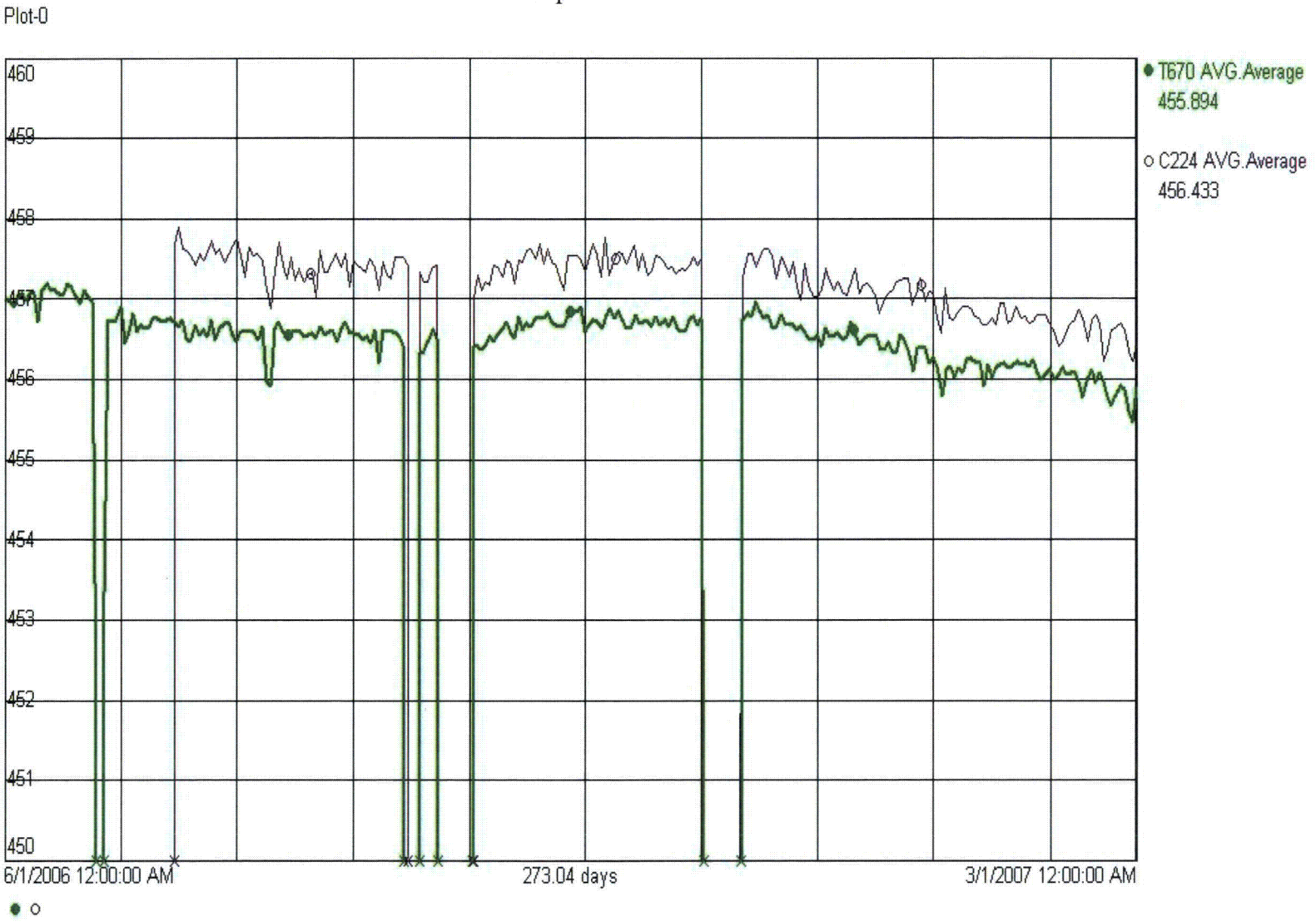
The gains of the individual pathways of the LEFM were chosen for review due to their trends indicating the overall health of the transducers. The higher the gain, the greater the amplification of the signal required to ensure an accurate flow measurement. A review of the gains prior to and after June of 2006 shows a definite improvement in gains after the transducers were changed out in June of 2006. This is expected due to the fact that the original transducers had approximately six thermal cycles on them, which is the major aging mechanism for the "F" revision of the transducers. The data shows that changing out the transducers to the "K" revision did not have an identifiable impact on the performance of the LEFM other than what was expected. On the graphs of the gains prior to June of 2006, the upward shift in gains is due a plant maintenance outage and the downward spikes in gains were due to 14RFO. The spikes on the graph after June of 2006, were due to plant maintenance outages.

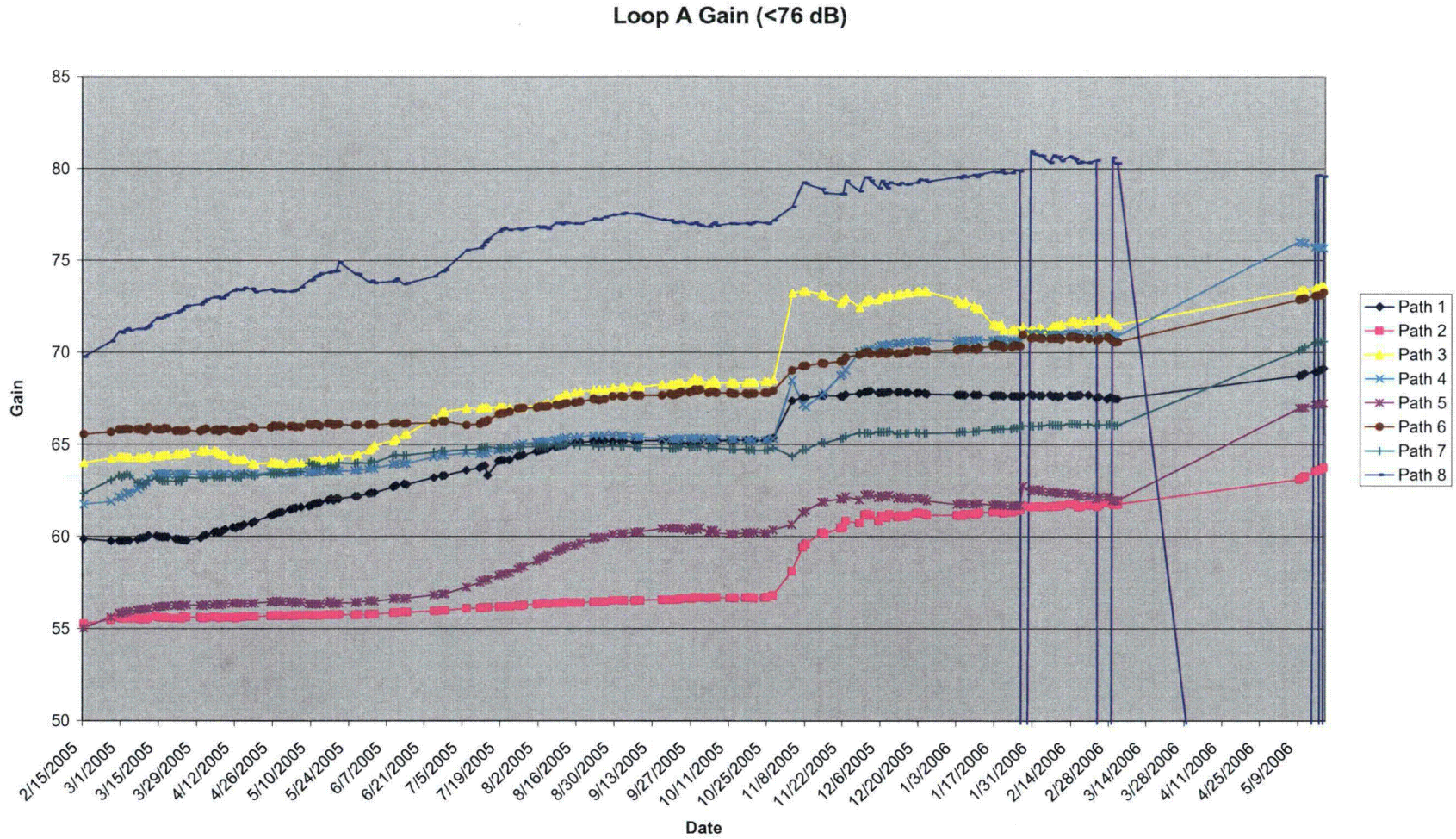
In summary, there were no identifiable effects observed in the data due to changing out of the transducers in June of 2006. Data was collected on the performance of the transducers after the change out and verified by Cameron that the LEFMs were performing within specifications.

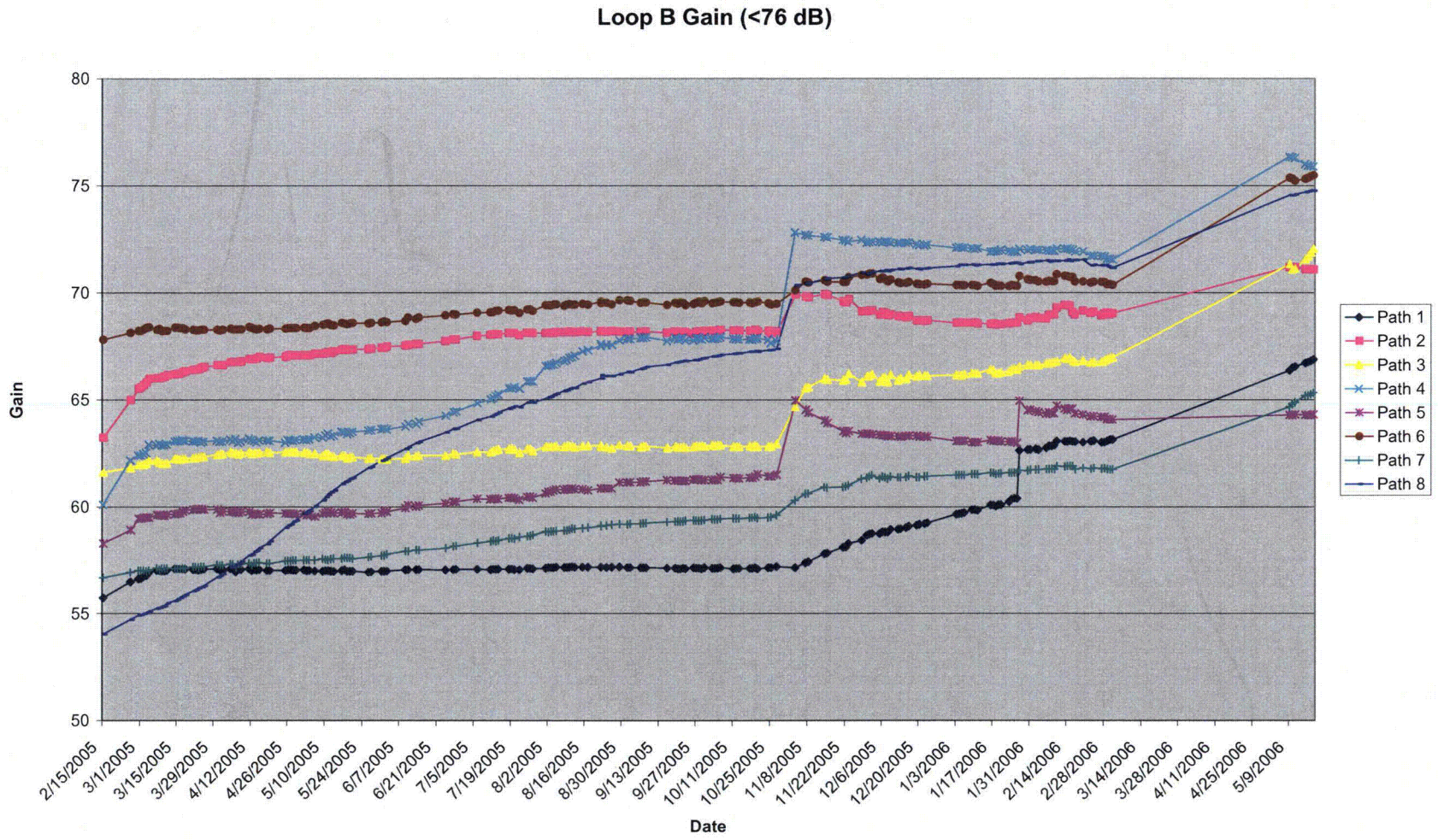
Prior to June of 2006



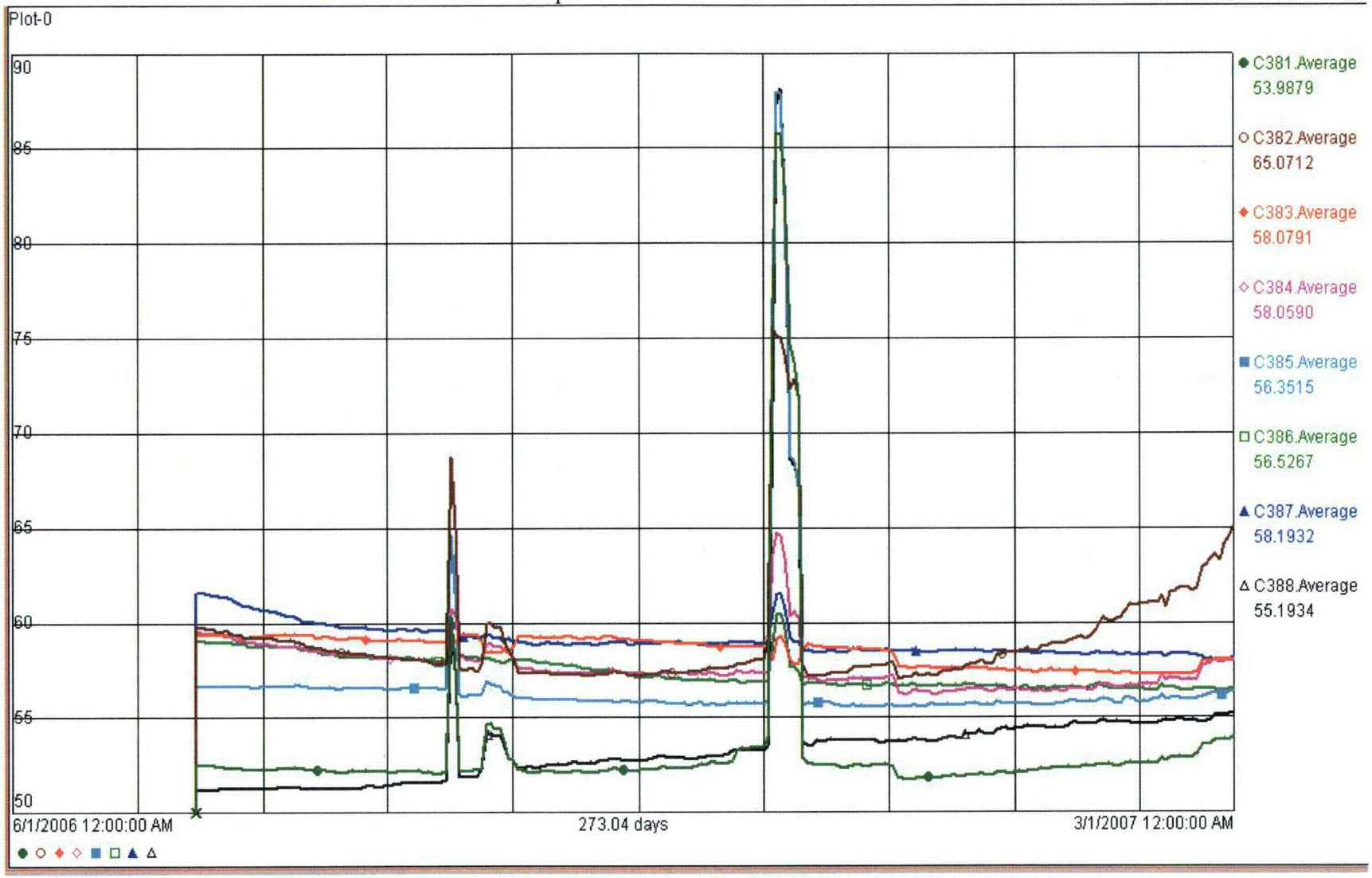
FW Temperature After June of 2006



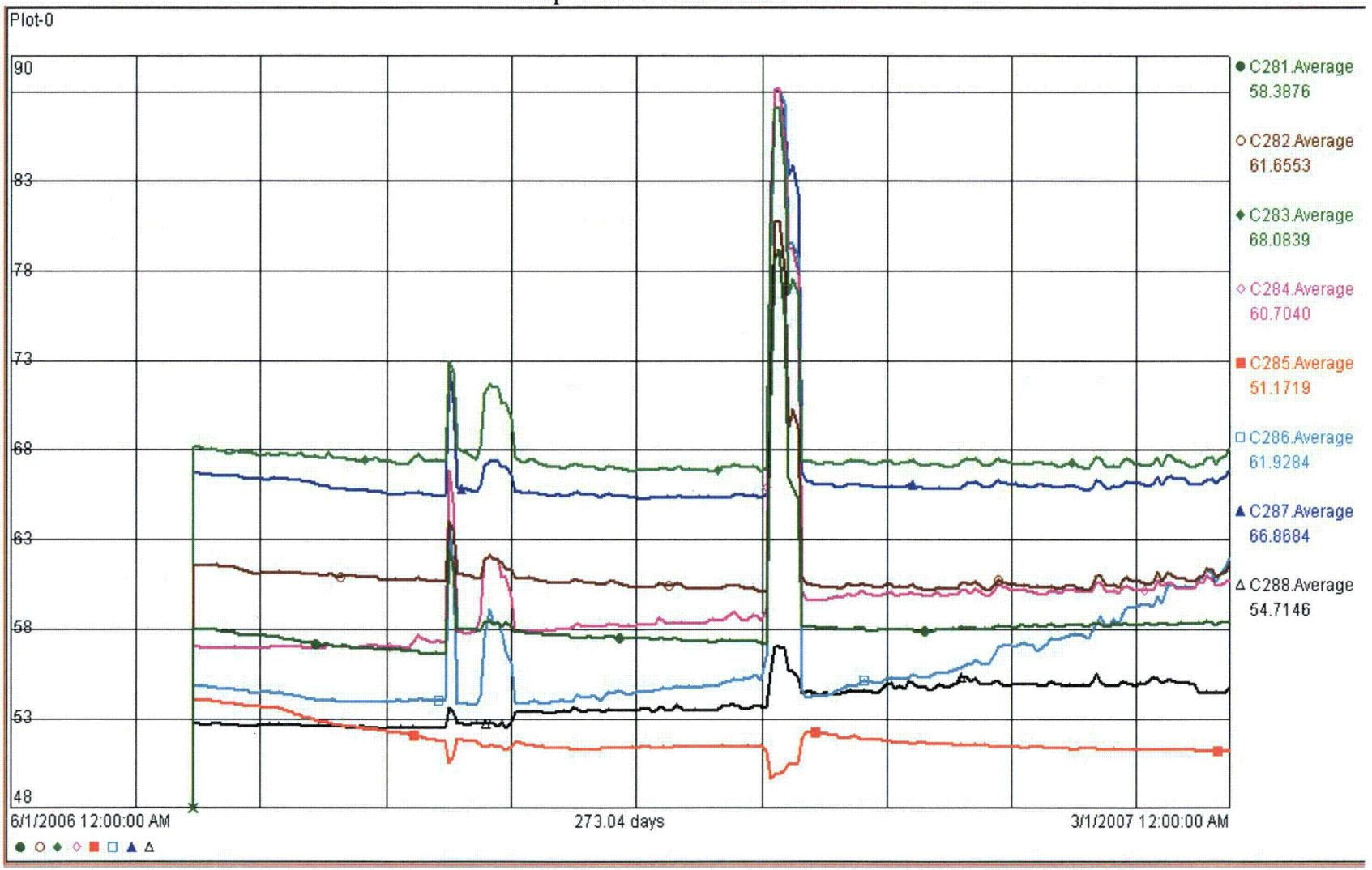

Loop A Gain (<76 dB)
Gain
85
80
75
70
65
60
55
50
2/15/2005
3/1/2005
3/15/2005
3/29/2005
4/12/2005
4/26/2005
5/10/2005
5/24/2005
6/7/2005
6/21/2005
7/5/2005
7/19/2005
8/2/2005
8/16/2005
8/30/2005
9/13/2005
9/27/2005
10/11/2005
10/25/2005
11/8/2005
11/22/2005
12/6/2005
12/20/2005
1/3/2006
1/17/2006
1/31/2006
2/14/2006
2/28/2006
3/14/2006
3/28/2006
4/11/2006
4/25/2006
5/9/2006
Date
Path 1
Path 2
Path 3
Path 4
Path 5
Path 6
Path 7
Path 8


Loop B Gain (<76 dB)
Gain
80
75
70
65
60
55
50
2/15/2005
3/1/2005
3/15/2005
3/29/2005
4/12/2005
4/26/2005
5/10/2005
5/24/2005
6/7/2005
6/21/2005
7/5/2005
7/19/2005
8/2/2005
8/16/2005
8/30/2005
9/13/2005
9/27/2005
10/11/2005
10/25/2005
11/8/2005
11/22/2005
12/6/2005
12/20/2005
1/3/2006
1/17/2006
1/31/2006
2/14/2006
2/28/2006
3/14/2006
3/28/2006
4/11/2006
4/25/2006
5/9/2006
Date
Path 1
Path 2
Path 3
Path 4
Path 5
Path 6
Path 7
Path 8

Loop A Gains After June of 2006



Loop B Gains After June of 2006



24. SRXB: If an LEFM becomes inoperative, you plan to rely upon venturis for a short time that have been calibrated with the last valid LEFM data. If a venturi defouling event should occur during this time, an overpower condition could result. Please discuss this possibility.

DBNPS Response:

Davis-Besse does not make adjustments (or calibrations) to the venturi performance based on the LEFM indication. Davis-Besse does not use the LEFM indication to provide a correction factor to the venturi flow input into the heat balance calculation. If the LEFM becomes inoperable and the venturis are selected for flow input into the heat balance calculation, no correction factor (LEFM/Venturi) is applied.

25. SRXB: How are plant personnel qualified to perform maintenance and calibration of the LEFM system?

DBNPS Response:

During the initial installation of the Caldon LEFM modification, it was determined that the Instrument & Control (I&C) Journeyman qualifications covered the general knowledge and skill requirements to support maintenance and calibration of the LEFM System. General LEFM work activities include power supply checks and sensor replacements. All I&C Technicians are considered qualified to work on the Caldon LEFM System based on having completed those Journeyman qualifications.

Caldon LEFM modification overview training was provided in 2002. In addition, vendor-supplied overview training was provided to selected individuals in 2005. Detailed training on outage related preventative maintenance activities for the LEFM equipment was provided during I&C continuing training in 2007.

Recent issues with the LEFM transducer failures have been reviewed by the Instrument and Control Curriculum Review Committee, and a subcommittee was established to review the preventive and corrective maintenance activities associated with the LEFM equipment. Currently, a formal gap analysis is being conducted to determine future training and qualification needs.

26. SRXB: Discuss the frequency of the listed preventive maintenance activities.

DBNPS Response:

Section I.1.D.1.1 of Enclosure 2 of the license amendment application identifies a preventive maintenance program has been developed for the LEFM that is to be performed every refueling outage. The listed preventive maintenance activities are described in Davis-Besse maintenance plan 83956 (Reference I-11) and are to be performed every refueling outage.

27. SRXB: The discussion on the calibration of the flow meters indicates that the meters were calibrated at the Alden Research Laboratory facility using the plant's current piping configuration and variations of the plant's configuration. Explain what in the configuration was varied and why.

DBNPS Response:

As described in the response to question 21, there are two notable differences between the tested configuration and the plant configuration: 1) the length of piping on the inlet to the elbow that is prior to the LEFM; and 2) the test piping was laid out horizontally while the plant piping is a vertical run of piping up to the elbow, with the flow direction from bottom to top. The difference in length of the piping on the inlet to the elbow was due to a lack of space within Alden Research Laboratory to accommodate any long run of straight pipe prior to the elbow. The difference in orientation (horizontal versus vertical) was also due to laboratory limitations. As discussed in the response to question 21, neither of these differences is judged to be significant.

As described in the laboratory report (Enclosure 1), a test configuration without a flow straightener in the line was included. The laboratory test data demonstrated that there was very little effect on the performance of the LEFMs without the flow straightener in the line.

28. SRXB: The submittal indicates that the failure of one transducer resulted in an alarm that would have caused the LEFM system to be removed from service. The submittal further indicates that, since this initial failure in June 2005, one additional transducer has failed.

- **Did the second failure result in an alarm also? If not, why not?**
- **Are these the only two transducer failures that have occurred over the life of the system, or have other transducer failures occurred?**

DBNPS Response:

The alarm associated with the June 2005 transducer failure, referred to in the license amendment

application, was a local (system internal) alarm, not an annunciator alarm. The LEFM System Trouble Annunciator was not in service in 2005 and LEFM data was not being used for the heat balance calculation at that time. It is not known whether the second transducer failure resulted in an internal alarm. It was identified in July 2006 that the LEFM System Trouble Annunciator circuit was not configured and wired correctly. This condition was corrected in April 2007, following which the LEFM System was then placed into service. Currently, if one or more transducer failures occur, the LEFM System Trouble Annunciator will be activated. This annunciator alarm prompts Control Room operators to follow the actions given in Alarm Procedure, DB-OP-02010, "Feedwater Alarm Panel 10 Annunciators," and remove the LEFM from service.

Since June of 2006, other transducer failures have occurred. Cameron is investigating the issue to determine the probable cause of these transducer failures. Once the results of the evaluation are available, the need for additional corrective actions will be determined in accordance with the corrective action process.

29. SRXB: Section VIII.6.3.2 of the AREVA attachment indicates non-core sources of heat addition to the RCS power, then states, "A value of 17 MWt has been used by Davis-Besse." The wording, "has been used by," is slightly unclear to the NRC staff. Provide additional comment on the basis for selection of this value.

DBNPS Response:

The original Final Safety Analysis Report stated that the nuclear steam supply system (NSSS) is designed for a power rating of 2789 MWt. Subtracting the core rated thermal power of 2772 MWt results in a value of 17 MWt for non-core heat addition sources. This is the value used in Enclosure 2 of the license amendment application, Table VIII.6-1, "Davis-Besse Operating Conditions."

The heat balance calculation determines the total NSSS power, and subtracts a value for the non-core heat addition sources to determine the core power. The core power can then be verified to be within the licensed limit. In actual practice, the value for the non-core heat addition sources is determined dynamically, and typically ranges from 14-15 MWt. Using a value of 14 MWt, the maximum allowable NSSS power, within the licensed core power, is 2786 MWt, which is conservative compared to the above-mentioned 2789 MWt value.

30. SRXB: Upon annunciation of a transducer failure, how long before the LEFM system is declared inoperable?

DBNPS Response:

Upon receipt of the annunciator alarm, the Control Room operators will follow the actions within

Alarm Procedure, DB-OP-02010, "Feedwater Alarm Panel 10 Annunciators." The procedure requires that the feedwater flow venturis, in lieu of the LEFM, be used for the heat balance calculation.

Operators are expected to evaluate annunciator alarms expeditiously. However, the length of time for the operators to take action will vary. A recent example occurred on April 24, 2007. The annunciator alarm was received at 0813 and the inoperability declaration was made at 0830.

31. SRXB: Section 15.1.2 of the UFSAR does not provide a very detailed anticipated transient without scram (ATWS) analysis. In comparing section 15.1.2 with the submittal, Section II.2, it appears that the maximum pressure criterion for an ATWS event is 3200 psig, whereas in the UFSAR, a safety limit is established as 2750 psig. The UFSAR does not discuss a maximum pressure criterion. Please explain the derivation of the maximum pressure criterion, its relationship to the American Society of Mechanical Engineers Code Section III pressure limit (2750 psig), and what pressures are predicted in an ATWS scenario.

DBNPS Response:

The NRC documented its evaluation of ATWS for Davis-Besse in a September 29, 1989 letter, Thomas V. Wambach (NRC) to Donald C. Shelton (Toledo Edison Company), "Evaluation of the Davis-Besse Nuclear Power Station Compliance with 10 CFR 50.62 Requirements for Reduction of Risk From Anticipated Transients Without Scram (ATWS) (TAC 59086)." This letter cites NRC reviewed and acceptance of Babcock and Wilcox (B&W) Document 47-1159091-00, "Design Requirements for DSS (Diverse Scram System) and AMSAC (ATWS Mitigation System Actuation Circuitry)." In this document, the peak pressures for the B&W designed plants were within the range 3621 to 4190 psia for the limiting transient (a loss of main feedwater event) assuming no reactor trip. The report further states that the design goal for the DSS was to "...prevent the RCS pressure from exceeding 3250 psig." This was to ensure that ASME Service Level C limits were not exceeded. The report also stated that analyses had been performed wherein the DSS and AMSAC were credited and that the peak RCS pressure would be less than 3250 psig. Note that the 3200 psig value referenced in the license amendment application was a typographical error and should have been 3250 psig.

A scoping calculation was performed in 2005 for Davis-Besse at a power level of 3026 MWt, crediting both the DSS and SFRCS, to determine if a power uprate would be feasible. Using the current Davis-Besse trip setpoints for the DSS and SFRCS, the RCS peak pressure was less than 2750 psig.

32. SRXB: Regarding the Control Rod Assembly Misalignment analysis, explain why a power level of 102 percent (2966MWt) was selected for analysis, and how that differs from the assumptions of the original analysis. Why was this power level selected

instead of the 102 percent generally selected for the remaining transients? In more detail, explain why this analysis is bounded by the analysis of record. Address any significant differences in peak pressure, peak temperature, and maximum reduction in DNBR margin, and any changes in the sequence of events.

DBNPS Response:

The results presented in the UFSAR are the original analysis performed for Davis-Besse and were based on 100% core power with a dropped rod worth up to 0.65% dk/k. Cases were run at beginning of life (BOL) and end of life (EOL) conditions that allowed for the most positive and most negative moderator temperature coefficients. At BOL, even the smallest worth that was modeled resulted in a reactor trip. At EOL conditions, the reactivity addition due to the temperature decrease and negative MTC prevented a reactor trip. The peak thermal increased back to the initial power level. Due to 24-month fuel cycles and current core designs, the maximum worth of a dropped control rod must be limited to less than 0.2% dk/k, including uncertainties. For each new fuel cycle, the maximum worth of a dropped rod and the core peaking are evaluated to ensure that the UFSAR analysis remains bounding.

There are no peak pressure or temperature consequences for this event as this is an over-cooling transient. Since core power does not increase above the initial value and coolant temperature is lower, the DNBR during the transient is greater than the initial value. There is really no sequence of events for this event. The rod is dropped at time = 0 seconds. The rod is fully inserted within 2.4 seconds. No safety system actuation setpoints are reached and no operator actions are modeled. The transient is over in less than 30 seconds. The worth of the dropped rod determines the initial temperature and pressure decrease. For the limiting worth, the temperature will only decrease sufficiently that RCS will remain slightly above the low pressure reactor trip setpoint. Any further pressure/temperature decrease cause the reactor to trip and would terminate the transients. The subsequent power increase is governed by the final RCS temperature and the EOC MTC.

The 102% of 2966 MWt case described, was a scoping analysis to ascertain if a larger power uprate were feasible. This case was simply used to verify that as the initial power level increases, the resulting RCS temperature decrease becomes smaller. Core power only increased up to about 95% whereas the UFSAR case returned to the initial value. In effect, this case demonstrated that use of 100% power (as presented in the UFAR) was conservative.

33. Electrical Engineering Branch (EEEB): Provide a detailed comparison of existing ratings with uprated ratings and the effect of the power uprate on the following equipment:

- **main generator rating and power factor**
- **isophase bus**
- **main power transformer**
- **unit auxiliary/startup transformer**
- **main generator breaker**

DBNPS Response:

Component	Rating		Effects
	Existing	Uprated	
Main Generator	1,069 MVA, 0.90 pf, 25kV	1,069 MVA, 0.90 pf, 25kV	See NOTE 1.
Isophase Bus	25 kV Nominal Bus Voltage; 25,000 A Continuous Current Rating	25 kV Nominal Bus Voltage; 25,000 A Continuous Current Rating	See NOTE 2.
Main Power (Generator Step-up) Transformer	980 MVA FOA rating 65°C Rise, 23.75 kV - 345 kV	980 MVA FOA rating 65°C Rise, 23.75 kV - 345 kV	See NOTE 3.
Startup Transformer (SU01 and SU02)	Max Rating 72.8 MVA at 65 °C 39/52/65 MVA OA/FOA/FOA at 55 °C rise	Max Rating 72.8 MVA at 65 °C 39/52/65 MVA OA/FOA/FOA at 55 °C rise	See NOTE 4.
Auxiliary Transformer (11)	Max Rating 58.24/77.653 MVA at 65 °C 52/69 MVA OA/FA at 55°C rise	Max Rating 58.24/77.653 MVA at 65 °C 52/69 MVA OA/FA at 55°C rise	See NOTE 5.
Main Generator Breaker	362 kV, Rated Maximum Voltage 2,000 A Rated Continuous Current	362 kV, Rated Maximum Voltage 2,000 A Rated Continuous Current	See NOTE 6.

NOTES

1. The Main Generator reactive capability curve illustrates that the Main Generator is capable of operating at a maximum real power output of 1068 MWe at a 1.0 power factor (zero megavar output). It is expected that gross generator output levels will be less than this maximum. Machine operation at a lower real output power level and a power factor (pf) of 1.0, or less, is permissible provided unit operation remains within the real and reactive power limits defined by the generator reactive capability curve. The generator reactive capability curve is contained in procedure DB-PF-06703, "Miscellaneous Operating Curves," while procedure DB-OP-06301, "Generator and Exciter Operating Procedure," contains the generator operating limit to maintain the Main Generator limits given in DB-PF-06703.

2. The voltage on the Isophase Bus supplied from the Main Generator is normally lower than the rating of 25 kV. Lowering the voltage (from 25 kV) to the Isophase Bus requires an increase in operating current to maintain the same MVA value. Under certain conditions (i.e., as reactive power is increased) the generator real power output may be limited in order to maintain current below the rating of the Isophase Bus. For example, if the unit was required to operate at a 0.9 pf lagging, the real power would be limited to approximately 935 MWe (based on a 24 kV Main Generator Output Voltage and 25000 A rated Isophase Bus current). Procedure DB-OP-02042, "Isophase Bus Cooling Alarm Panel 42," contains temperature alarms and actions associated with the alarms (including action to reduce Main Generator Load) to protect the Isophase Bus.

3. The rating of the Main Transformer is 980 MVA. When running the house loads off the Auxiliary Transformer, the MVA at the Main Transformer is less than that at the output of the Main Generator (when running off the Startups, the Main Transformer would not see the reduction due to the removal of the house loads). Similar to the Isophase Bus, under certain conditions (i.e., as reactive power is increased) the generator real power output may be limited in order to remain below the rating of the Main Transformer. For example, with a 980 MVA Main Transformer rating value, adding approximately 55 MVA for house loads equates to 1035 MVA at the Main Generator. If the unit was required to operate at a 0.9 pf lagging, the real power would be limited to approximately 931.5 MWe. Procedure DB-OP-06313, "Station Transformer Auxiliaries System Procedure," contains a load table that may be utilized for MVA loading calculations. The load table is based on the 100% MVA rating of the transformer and takes into account ambient temperature and the amount of cooling equipment.

4. The Davis-Besse load flow cases use 72.8 MVA. The expected load increases from the MUR uprate continue to be below the Maximum Rating of the Startup Transformers.

5. The Davis-Besse load flow cases use 77.65 MVA. The expected load increases from the MUR uprate continue to be below the Maximum Rating of the Auxiliary Transformer.

6. Davis-Besse does not have a Main Generator Breaker prior to the Isophase Bus, Main Transformer, or Auxiliary Transformer. However, there are two 345 kV Air-Blast Circuit Breakers (34560 and 34561) that connect the generator (through the Main Transformer) to the Switchyard. Maintaining the generator power output within the ratings of the Main Transformer (as discussed above) will ensure the ratings of the 345 kV Air-Blast Circuit Breakers will not be exceeded.

34. EEEB: Does the power uprate affect any ac distribution system loads? If so, provide a list of loads affected by the power uprate change.

DBNPS Response:

Estimated Impact of the Uprate on Plant Loads	
Affected Component	**Magnitude of Load Increase**
Condensate Pump Motor	14 horsepower (hp) per motor
Heater Drain Pump Motor	6 hp per motor
Reactor Coolant Pump Motor	4 hp per motor
Stator Water Cooling Pump Motor	2.5 hp per motor

The actual house power electrical demand was reviewed in calculation C-EE-015.03-008, "AC Power System Analysis," along with other historical data, to be a maximum of 47.5MW during normal operation (at the current licensed power level). The house power demand modeled in calculation C-EE-015.03-008 for normal operation (wintertime), at the current licensed power level, indicates a house load of slightly more than 52MW. This comparison shows that for normal operation, the analytical results are conservative compared to the actual plant historical data. There is adequate margin, approximately 4.5MW, to accommodate the slight increase in house demand due to the load increases listed in the table above.

35. EEEB: Attachment A of the LAR refers to "Davis-Besse Stability Study for FirstEnergy Corporation" (ADAMS No. ML020640288). The transient stability study assumed a 10 percent increase in gross power output, which is significantly higher than the proposed increase of 1.63 percent. The study concluded that for two of the fourteen contingencies analyzed the system response varied or was unstable. A three-phase fault at the Bayshore 345 kiloVolt (kV) bus, Contingency 4, resulted in unstable system responses for the uprated system but stable conditions for the existing ratings. A three-phase fault at DBNPS circuit breaker 34564, Contingency 8, resulted in unstable system response. The study states, "if the Davis-Besse uprate occurs, additional analysis is recommended to determine methods to improve system stability [for Contingencies 4 and 8]." Have additional analyses been performed to evaluate improving system stability for Contingencies 4 and 8? If so, what actions are being taken as a result of the additional analyses?

DBNPS Response:

The referenced stability study assumed a 10 % increase in gross power output and included analysis of two contingencies which were outside of the First Energy and North American Reliability Corporations required contingencies. Contingencies 4 and 8, which were identified as having unstable system responses for the uprate, are the basis for the license amendment application statement: "Even at 3 times the current proposed increase, this study found that all machines maintained stability for all fault cases which have a reasonable probability of occurring." No additional analyses have been performed to evaluate improving system stability for contingencies 4 and 8, as they are beyond the scope of First Energy and North American Electric Reliability Corporation requirements.

36. EEEB: Provide justification that the DBNPS Stability Study completed in May 2000 bounds the current grid conditions. Specifically, since the results of the stability analysis are based on 1999/2000 summer peak load conditions, describe the impact on grid stability when using current summer peak loads.

DBNPS Response:

The stability study completed in May 2000 was based upon a planned 10% power uprate. Since stability requirements were met at the significantly higher planned 10% power uprate, they will clearly be met for the more modest proposed 1.63% uprate.

Comprehensive stability assessments of the First Energy facilities in the Midwest Independent Transmission System Operator footprint were performed in 2005 and again in 2007. Both of these studies continue to show that Davis-Besse and the system as a whole meet all First Energy and North American Electric Reliability Corporation stability requirements with sufficient margin to accommodate a 1.63% uprate. Based on these most recent studies, the conclusion that the 1.63% uprate presents no stability issues for either Davis-Besse or the First Energy system remains valid.

37. EEEB: The DBNPS Stability Study indicates that with a 10 percent increase in gross power output, the change in power factor reduces the unit's reactive power capability by 67 mega volt ampere reactive (MVAR). For the current uprate of 1.63 percent, please address and discuss the following:

- **Identify the nature and quantity of MVAR support necessary to maintain post-trip loads and minimum voltage levels. Address how the power uprate affects MVAR support.**
- **Identify what MVAR contributions DBNPS is credited by the transmission system operator (TSO) to support the grid. Address how the power uprate changes the MVAR contributions credited by the TSO.**
- **Address the compensatory measures taken to compensate for the depletion of**

the nuclear unit MVAR capability on a grid-wise basis due to this power uprate.

- **Provide an evaluation of the impact of any MVAR shortfall listed in part C on the ability of the offsite power system to maintain post-trip voltage levels and to supply power to safety buses during peak electrical demand periods. The subject evaluation should document any information exchanges between the TSO and DBNPS on this matter.**

DBNPS Response:

The adequacy of voltage levels required at plant loads during various plant operating conditions and system configurations, for both normal and accident conditions, is evaluated in the Davis-Besse load flow and motor starting analyses, contained in calculation C-EE-015.03-008, "AC Power System Analysis." For the proposed power uprate of 1.63%, the quantity of MVAR support required to maintain post trip loads and minimum voltage levels will increase slightly. This electrical load increase creates a larger electrical system demand. The electrical demand, as determined in calculation C-EE-015.03-008, will increase from 51.651 MW and 39.355 MVAR to 51.697 MW and 39.403 MVAR. In addition, the increased electrical system demand is bound by the values of 52 MW and 45 MVAR, as described in the Midwest Independent Transmission System Operator Interconnection Agreement.

Per the Midwest Independent Transmission System Operator Interconnection Agreement, FirstEnergy Energy Delivery Planning and Protection Services and Midwest Independent Transmission System Operator (MISO) credit Davis-Besse to operate from 0.90 lagging power factor to 1.00 power factor, with the intent to improve reactive capability in the leading direction to 0.95 leading power factor following the Fifteenth Refueling Outage. The MVAR contributions credited in the Midwest Independent Transmission System Operator Interconnection Agreement are not impacted by the 1.63% uprate because Davis-Besse's Main Generator will continue to operate within its capabilities.

No measures are required to compensate for any changes in nuclear unit MVAR capability on a grid-wise basis due to this power uprate with respect to the Main Generator's capabilities. With respect to bus voltage issues, Davis-Besse will only be asked to operate up to 0.95 leading power factor (consume reactive power) if the grid voltage is higher than the plant's minimum voltage requirements. Therefore, no compensatory measures were taken as a result of the power uprate. However, Davis-Besse plans to change operations procedures following the Fifteenth Refueling Outage (15RFO) to allow energization of the house loads via the start-up transformers. This will enable Davis-Besse to operate over a wider power factor range, under normal grid system conditions, while maintaining the essential bus voltages above their minimum voltage requirements.

The Davis-Besse peak electrical demand on the system occurs post unit trip on a design basis event. The power uprate will not have an impact on the post trip voltage levels on the system. Davis-Besse procedure DB-OP-01300, "Switchyard Management," describes the interface with the Transmission System Operator. The essential bus voltages are maintained within their

required bus voltage limits as defined in Davis-Besse procedure DB-SC-03041, "On Site AC Bus Sources Lined Up, Available and Isolated (Modes 1, 2, 3, and 4)." The essential bus voltages are maintained within their required limits by adjusting the Main Generator Automatic Voltage Regulator based on the voltage schedule provided in procedure DB-OP-01300. The system dispatcher is notified of inability to meet the voltage schedule. Davis-Besse is typically only requested to consume reactive power, operate with a leading power factor, when the grid is lightly loaded and the grid voltage is high. A higher grid voltage will enable Davis-Besse to consume additional MVARs.

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station (DBNPS) in this document. Any other actions discussed in the submittal represent intended or planned actions by the DBNPS. They are described only for information and are not regulatory commitments. Please contact Mr. Thomas A. Lentz, FENOC Manager - Fleet Licensing, at (330) 761-6071 if there are any questions regarding this document or any associated regulatory commitments.

COMMITMENT	DUE DATE
Upon internal approval of the Reactor Protection System high flux field trip setpoint calculation, FENOC will make the calculation available for NRC staff review and discussion.	November 1, 2007

Docket Number 50-346
License No. NPF-3
Serial Number 3355
Enclosure 2

ENCLOSURE 2

Cameron Letter CAW 07-14

August 15, 2007

Application for Withholding Proprietary Information from Public Disclosure

(applicable to Enclosure 1)

Measurement Systems

Caldon® Ultrasonics Technology Center
1000 McClaren Woods Drive
Coraopolis, PA 15108
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CAMERON

August 15, 2007
CAW 07-14

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Calibration of Two 18" Leading Edge Flow Meters for Caldon, Inc.
Purchase Order Number 18350 October 2001 – ARL NO. 310-01/C730"

Gentlemen:

This application for withholding is submitted by Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains trade secrets and/or commercial information proprietary to Cameron and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 07-14 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 Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 07-14 and should be addressed to the undersigned.

Very truly yours,

Calvin R. Hastings
General Manager

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

August 15, 2007
CAW 07-14

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Calvin R. Hastings, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

C R Hastings

Calvin R. Hastings
General Manager

Sworn to and subscribed before me

this 15th day of

August, 2007

Joann B. Thomas

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joann B. Thomas, Notary Public
Findlay Twp., Allegheny County
My Commission Expires July 28, 2011
Member, Pennsylvania Association of Notaries

1. I am the General Manager of Caldon Ultrasonics Technology Center, 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 Cameron.

2. I am making this Affidavit in conformance with the provisions of 10CFR Section 2.390 of the Commission's regulations and in conjunction with the Cameron application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information. The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

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

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

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron 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 Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

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 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 Cameron's competitors without license from Cameron 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, and assurance of quality, or licensing a similar product.

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

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

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

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

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

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

(c) Use by our competitor would put Cameron 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 Cameron of a competitive advantage.

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

(f) The Cameron 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. 390, it is to be received in confidence by the Commission.

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

(v) The proprietary information sought to be withheld is the submittal titled "Calibration of Two 18" Leading Edge Flow Meters for Caldon, Inc. Purchase Order Number 18350 October 2001 – ARL NO. 310-01/C730" is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A, B), with the reason(s) for confidential treatment described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for LEFM CheckPlus Systems used by Davis Besse Unit 1 for an MUR UPRATE.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Cameron because it would enhance the ability of competitors to provide similar flow and temperature measurement systems 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 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 Cameron effort and the expenditure of a considerable sum of money.

In order for competitors of Cameron to duplicate this information, similar products would have to be developed, 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 analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

Docket Number 50-346
License No. NPF-3
Serial Number 3355
Enclosure 5

ENCLOSURE 5

Cameron Letter CAW 07-12

August 7, 2007

Application for Withholding Proprietary Information from Public Disclosure

(applicable to Enclosure 4)

Measurement Systems

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Coraopolis, PA 15108
Tel 724-273-9300
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CAMERON

August 7, 2007
CAW 07-12

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Caldon Ultrasonics Engineering Report: ER-202 Rev. 3, "Bounding Uncertainty Analysis for Thermal Power Determination at Davis Besse Nuclear Power Station Using the LEFM✓+ System"

Gentlemen:

This application for withholding is submitted by Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains trade secrets and/or commercial information proprietary to Cameron and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 07-12 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 Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 07-12 and should be addressed to the undersigned.

Very truly yours,

C R Hastings

Calvin R. Hastings
General Manager

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

August 7, 2007
CAW 07-12

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Calvin R. Hastings, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

C R Hastings

Calvin R. Hastings
General Manager

Sworn to and subscribed before me

this 7th day of

August, 2007

Joann B. Thomas

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joann B. Thomas, Notary Public
Findlay Twp., Allegheny County
My Commission Expires July 28, 2011
Member, Pennsylvania Association of Notaries

1. I am the General Manager of Caldon Ultrasonics Technology Center, 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 Cameron.

2. I am making this Affidavit in conformance with the provisions of 10CFR Section 2.390 of the Commission's regulations and in conjunction with the Cameron application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information. The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

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

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

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron 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 Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

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 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 Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

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

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

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

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

(f) The Cameron 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. 390, it is to be received in confidence by the Commission.

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

(v) The proprietary information sought to be withheld is the submittal titled Caldon Ultrasonics Engineering Report: ER-202 Rev. 3, "Bounding Uncertainty Analysis for Thermal Power Determination at Davis Besse Nuclear Power Station Using the LEFM✓ + System" is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A, B), with the reason(s) for confidential treatment described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for LEFM CheckPlus Systems used by Davis Besse Unit 1 for an MUR UPRATE.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Cameron because it would enhance the ability of competitors to provide similar flow and temperature measurement systems 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 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 Cameron effort and the expenditure of a considerable sum of money.

In order for competitors of Cameron to duplicate this information, similar products would have to be developed, 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 analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

Docket Number 50-346
License No. NPF-3
Serial Number 3355
Enclosure 8

ENCLOSURE 8

Cameron Letter CAW 07-13

August 9, 2007

Application for Withholding Proprietary Information from Public Disclosure

(applicable to Enclosure 7)

Measurement Systems

Caldon® Ultrasonics Technology Center
1000 McClaren Woods Drive
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Tel 724-273-9300
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www.c-a-m.com

CAMERON

August 9, 2007
CAW 07-13

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Caldon, Inc. Engineering Report: ER-202 Rev. 2, "Bounding Uncertainty Analysis for Thermal Power Determination at Davis Besse Nuclear Power Station Using the LEFM✓ + System"

Gentlemen:

This application for withholding is submitted by Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains trade secrets and/or commercial information proprietary to Cameron and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 07-13 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 Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 07-13 and should be addressed to the undersigned.

Very truly yours,

C R Hastings

Calvin R. Hastings
General Manager

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

August 9, 2007
CAW 07-13

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Calvin R. Hastings, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

CR Hastings
Calvin R. Hastings
General Manager

Sworn to and subscribed before me
this 9th day of
August, 2007
Joann B Thomas
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joann B. Thomas, Notary Public
Findlay Twp., Allegheny County
My Commission Expires July 28, 2011
Member, Pennsylvania Association of Notaries

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

(v) The proprietary information sought to be withheld is the submittal titled Caldon, Inc Engineering Report: ER-202 Rev. 2, "Bounding Uncertainty Analysis for Thermal Power Determination at Davis Besse Nuclear Power Station Using the LEFM✓+ System" is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A, B), with the reason(s) for confidential treatment described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for LEFM CheckPlus Systems used by Davis Besse Unit 1 for an MUR UPRATE.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Cameron because it would enhance the ability of competitors to provide similar flow and temperature measurement systems 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 the right to use the information.

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In order for competitors of Cameron to duplicate this information, similar products would have to be developed, 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 analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

Docket Number 50-346
License No. NPF-3
Serial Number 3355
Enclosure 3

ENCLOSURE 3

Cameron Letter

March 8, 2007

Cameron Measurement Systems Response to Transducer Replacement Sensitivity

CAMERON

Measurement Systems

Caldon® Ultrasonics Technology Center
1000 McClaren Woods Drive
Coraopolis, PA 15108
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www.c-a-m.com

March 8, 2007

Tim Laurer
Nuclear Staff Engineer
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449
Attn: Tim Laurer

Telephone Number: 419-321-7764

Reference: First Energy Nuclear Operation Corp. Order No. 7048503
Cameron Measurement Systems Contract No. CO-22776

Subject: Cameron Measurement Systems Response to Transducer Replacement Sensitivity

Dear Tim,

At the request of the NRC, Cameron conducted transducer replacement testing to create an empirical, statistical evaluation of the uncertainty involved in replacing LEFM CheckPlus transducers in the field. The results of these tests reveals a spread on the same order as the uncertainty in the testing itself. In addition, uncertainties already accounted for in the analysis could be the source of parts of the spread in the raw results.

As a conservative measure, however, Cameron has elected to create a new uncertainty term in all analyses going forward explicitly to address the transducer replacement uncertainty. The term will actually appear both in the calibration uncertainty and in the installed system uncertainty as it applies to both instances. The amount of this uncertainty term for Davis Besse's two 18 inch pipe case is 0.1%. Applying this term in both calibration and installation uncertainty cases results in a change in overall mass flow uncertainty from 0.26% to 0.29%.

It is planned that no changes will be backfit to existing analyses, but that all analyses going forward will contain these additional terms. However, as Davis Besse is in the unusual position of having an old analysis being submitted for a new approval, an exception to this plan seems to be required. Therefore, Cameron proposes to revise Davis Besse's analysis to reflect the new terms. We will deliver the revised analysis in 90 days. In the meantime, Cameron will continue with our plans to schedule a general meeting with the NRC to discuss the particulars of the issue and the proposed plan.

Please do not hesitate to give me a call if you have any questions.

CAMERON

Measurement Systems

Caldon® Ultrasonics Technology Center
1000 McClaren Woods Drive
Coraopolis, PA 15108
Tel: 724-273-9300
Fax: 724-273-9301
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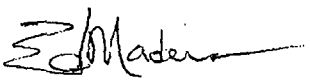
Sincerely,



Ed Madera
Cameron Measurement Systems Sr. Project Engineer

Ernie Hauser
Director of Sales
Cameron Measurement Systems
(formerly Caldon Inc.)

Docket Number 50-346
License No. NPF-3
Serial Number 3355
Enclosure 11

ENCLOSURE 11

Cameron Letter CAW 07-11

July 9, 2007

Application for Withholding Proprietary Information from Public Disclosure

(applicable to Enclosure 10)

Measurement Systems

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CAMERON

July 9, 2007
CAW 07-11

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Caldon, Inc. ER-227 Rev.1 "Profile Factor Calculation and Accuracy Assessment for the Davis Besse Unit 1 LEFM ✓+ Spool Pieces (Alden Report No. 310-01/C730)"

Gentlemen:

This application for withholding is submitted by Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains trade secrets and/or commercial information proprietary to Cameron and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 07-11 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 Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 07-11 and should be addressed to the undersigned.

Very truly yours,

C R Hastings

Calvin R. Hastings
General Manager

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

July 9, 2007
CAW 07-11

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Calvin R. Hastings, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

C.R. Hastings

Calvin R. Hastings
General Manager

Sworn to and subscribed before me

this 9th day of

July, 2007

Joann B. Thomas

Notary Public

COMMONWEALTH OF PENNSYLVANIA

Notarial Seal
Joann B. Thomas, Notary Public
City of Pittsburgh, Allegheny County
My Commission Expires July 28, 2007

Member, Pennsylvania Association of Notaries

1. I am the General Manager of Caldon Ultrasonics Technology Center, 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 Cameron.

2. I am making this Affidavit in conformance with the provisions of 10CFR Section 2.390 of the Commission's regulations and in conjunction with the Cameron application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information. The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

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

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

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron 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 Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

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 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 Cameron's competitors without license from Cameron 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, and assurance of quality, or licensing a similar product.

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

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

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

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

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

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

(c) Use by our competitor would put Cameron 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 Cameron of a competitive advantage.

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

(f) The Cameron 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. 390, it is to be received in confidence by the Commission.

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

(v) The proprietary information sought to be withheld is the submittal titled Caldon, Inc. ER-227 Rev. 1 "Profile Factor Calculation and Accuracy Assessment for the Davis Besse Unit 1 LEFM ✓+ Spool Pieces (Alden Report No. 310-01/C730)" is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A, B), with the reason(s) for confidential treatment described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for LEFM CheckPlus Systems used by Davis Besse Unit 1 for an MUR UPRATE.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Cameron because it would enhance the ability of competitors to provide similar flow and temperature measurement systems 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 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 Cameron effort and the expenditure of a considerable sum of money.

In order for competitors of Cameron to duplicate this information, similar products would have to be developed, 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 analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

Docket Number 50-346
License No. NPF-3
Serial Number 3355
Enclosure 6

ENCLOSURE 6

Cameron Letter

May 14, 2007

Changes to LEFM CheckPlus System Uncertainty Owing to Transducer (Re)Placement

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May 14, 2007

FirstEnergy
Davis-Besse
5501 N. State Route 2
Oak Harbor, OH 43449-9760

Subject: Changes to LEFM CheckPlus System Uncertainty owing to Transducer (Re)Placement

References: 1) CIB 125, dated April 2007
2) ER-202 LEFM CheckPlus Uncertainty Analysis for Davis-Besse NPP

Dear Mr. Brian Young:

Cameron has performed additional tests and analysis at the request of NRC to verify that the effects of transducer replacement are bounded by the LEFM CheckPlus Uncertainty Analyses. During the conduct of those tests and analysis, an uncertainty source was postulated related to the location of the transducers in the housings which had not been previously identified.

Test results could neither confirm nor rule out that the postulated mechanism adds uncertainty because the magnitude of the effect is approximately 0.1%. This figure is of the same order as the testing resolution, and the time measurement terms already considered in the uncertainty analysis. As a conservative measure, Cameron has elected to add the term to future analyses. Cameron has also evaluated the effects on all existing analyses. The effect on the Davis-Besse LEFM CheckPlus uncertainty is to increase the total feedwater mass flow uncertainty by 0.02%. There is no change to the feedwater temperature uncertainty.

The amount of your uprate was based on the bounding uncertainties of Revision 0 of the uncertainty analysis. However, the design basis revision of that analysis incorporates the data collected during the calibration of your flow element(s) and during the commissioning of the system in your plant. The actual uncertainty in the profile factor, determined by the calibration testing, is lower than the bounding value of Revision 0. The margin in mass flow uncertainty provided by the use of the Revision 0 profile factor uncertainty offsets the increase in mass flow uncertainty to accommodate transducer replacement. Consequently there should be no net effect on the thermal power uncertainty on which your uprate is based. Nevertheless you should confirm that your actual thermal power uncertainty is within the design basis of your uprate.

Cameron has shared the general results and its recommended actions with the NRC staff. You should keep this letter with your documentation justifying your uprate since it constitutes part of the basis for your measurement uncertainty. If you wish Cameron to provide an updated uncertainty analysis for your plant, please call Leeanne Jozwiak at 724-273-9300 to request a price and delivery quotation.

Sincerely,

Ed Madera
Cameron Senior Project Engineer

Description

LEFM Check and CheckPlus System Uncertainty Analyses have included time measurement uncertainty components associated with the critical characteristics of transducers, circuits, and coherent signal to noise ratio. These uncertainties typically aggregate to approximately 0.09%[1].

During an NRC review of the Uncertainty Analyses for LEFM Check and CheckPlus systems, the NRC asked for evidence that these uncertainties bounded results when transducers were changed or replaced. An initial demonstration test was conducted during the Seabrook calibration in January 2006, in which two transducers were replaced with no measurable change in Meter Factor. Nevertheless, in its July 5, 2006 SER, the staff asked Cameron to conduct a test with a statistically significant number of transducer replacements to confirm the preliminary results.

Summary

This CIB reports the results of those tests. The tests demonstrated that the variability in Meter Factor owing to transducer replacements has an upper bound of approximately 0.1%, depending on pipe diameter and number of pipes per system. While this variation is very close to the time measurement uncertainties already identified in the uncertainty analyses, a separate potential uncertainty source, transducer location, was identified that had not been explicitly addressed.

Since this uncertainty was of the same order as the test resolution, it could not be explicitly ruled out as a contributor. Therefore, as a conservative measure, Cameron has elected to include a separate uncertainty for the actual location of the transducer within the housings.

The average impact of this additional uncertainty is to increase the mass flow uncertainty by an amount ranging from 0.01% to 0.03% with an average of 0.017%.

Description of Test

Testing was conducted in 160 flow tests and 128 transducer replacements on an 18 inch 8 path meter. The testing methodology and results are recorded in Cameron report ER-551 Rev. 1. The scatter in the test results shows a 95% confidence interval of approximately 0.1%, roughly equivalent to the estimated test error. No definitive conclusions that errors larger than those already identified in the uncertainty analysis can be reached, but during the testing another possible error contributor was postulated that had not been explicitly bounded with a separate uncertainty term. This possible error is a function of the clearance between the transducer element and the housing wall that leads to an uncertainty in the location of the transducer centerline. Variations in the location of the transducer in the housing can cause changes in path angle and path spacing. The path angle effect is by far the most important. This potential source of error can not be excluded on the basis of the test results.

Actions

Cameron has added an uncertainty term to address the transducer location uncertainty. The value of the term varies by pipe size and number of flow elements in the system, but is typically between 0.05 and 0.1%. The term is considered both in the calibration uncertainty and again in the installation uncertainty. The term has already been added to all analyses generated after December 2006, and will be added to all future analyses.

Cameron presented its findings and ER-551 Rev 1 to the NRC staff on April 11, 2007, including its recommended actions. The report was presented for information and needs no further NRC action.

Extent of Condition

All users will be formally notified of the change in overall uncertainty specific to their system. Essentially the total mass flow uncertainty (95% confidence level) will be increased approximately by 0.01 to 0.03% (depending on pipe size and the number of flow elements. On average, the increase is 0.017%. Because of existing margins in the mass flow uncertainty as well as in other elements of the thermal power uncertainty, it is not expected that there will be any net change in the overall thermal power uncertainty. Nevertheless each licensee should confirm this conclusion with respect to his or her plant, and to retain, as part of his design basis for the MUR uprate, a record of this confirmation.

We do not consider it necessary to revise individual plant uncertainty analyses to incorporate this effect. Nevertheless, if any user wishes to have Cameron revise their uncertainty analysis, they should contact the Caldon Ultrasonics Technology Center, Project Engineering, for information on schedule and cost.

Questions pertaining to this bulletin may be addressed to: Leeanne Jozwiak at Cameron 724-273-9300.

[1] The current value applied for signal to noise ratio error is greater than that originally used in ER-157P. The current practice for computing this error more conservatively combines the errors for individual paths than the method used in ER-157P.