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Enclosure 4 contains proprietary information.
Withhold from public disclosure per 10 CFR 2.390.
Upon removal of Enclosure 4, this letter is uncontrolled.

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July 6, 2012

10 CFR 50.90

U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Document Control Desk

Subject: Duke Energy Carolinas, LLC (Duke Energy)
McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370

Response to Request for Additional Information Regarding License Amendment Related to Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME8213 and ME8214)

This letter provides the responses to a April 27, 2012 Nuclear Regulatory Commission (NRC) request for additional information (RAI) and a June 6, 2012 RAI related to a March 5, 2012 McGuire Nuclear Station (MNS) Units 1 and 2 License Amendment Request (LAR) submitted pursuant to 10 CFR 50.90 in support of a measurement uncertainty recapture (MUR) power uprate.

NRC MUR LAR RAI question 4 and Duke Energy's response are provided in Enclosure 1. NRC MUR LAR RAI questions 20 thru 31 and Duke Energy's responses are provided in Enclosure 2, Enclosure 3, and Enclosure 4. A Duke Energy regulatory commitment related to NRC RAI question 29 is provided in Enclosure 6. Responses to MNS MUR LAR RAI questions 1 thru 3 and questions 5 thru 19 were provided to the NRC via correspondence dated May 29, 2012 and June 21, 2012 respectively.

Enclosure 4 contains proprietary information. Duke Energy requests that Enclosure 4 be withheld from public disclosure per 10 CFR 2.390. An affidavit supporting this request for withholding is contained in Enclosure 5. Included in Enclosure 4 are Cameron Caldon documents ER-819 Revision 2 and ER-822 Revision 2 which replace the Revision 1 versions provided in Attachment 4 of the March 5, 2012 MNS MUR LAR submittal.

The conclusions reached in the original determination that this LAR contains No Significant Hazards Considerations and the basis for the categorical exclusion from performing an Environmental/Impact Statement have not changed as a result of the RAI responses provided in this submittal.

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Upon removal of Enclosure 4, this letter is uncontrolled.

A001
NRR

Please contact Kenneth L. Ashe at 980-875-4535 if additional questions arise regarding this LAR.

Sincerely,

R. T. Repko

Enclosures

cc: w/enclosures

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OATH AND AFFIRMATION

Regis T. Repko affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

Regis T. Repko, Vice-President, McGuire Nuclear Station

Subscribed and sworn to me: July 6, 2012
Date

Notary Public

My commission expires: July 1, 2017
Date

bxc w/attachments:

McGuire Master File (MG02DM)
NRIA/ELL (EC05O)
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Enclosure 1

McGuire Nuclear Station's Response MUR LAR RAI Question 4 In The April 27, 2012 NRC Request for Additional Information

NRC Question 4

In Enclosure 2 of the LAR, Section V.1.C, "Environment Qualification [EQ] of electrical equipment," on page E2-73, the licensee stated that:

> In accordance with the McGuire [1 and 2] design change process, any specific component modifications that may be required to support the MUR uprate will be evaluated against the EQ Program requirements.

The NRC staff expects that any specific component modification that may be required to support the MUR power uprate will be evaluated against the EQ Program requirements prior to approval of the LAR by the NRC staff. Therefore, please complete the evaluation of the MUR power uprate against the EQ Program requirements and provide the results of this evaluation.

McGuire Response to Question 4

The review of McGuire Nuclear Station (MNS) EQ Program documentation included review of both Duke Energy EQ program-level documents and discrete EQ files/calculations for specific components installed at MNS. This review was conducted to focus on the EQ parameters of temperature, pressure, and radiation, with respect to any potential parameter changes due to the MUR power uprate.

Temperature and Pressure:

Temperature and pressure were evaluated as part of the engineering evaluations for the MUR power uprate. The potential changes in ambient temperatures, system temperatures, system pressures, and potential accident external pressures (high energy line break) and accident temperatures were considered during the review. Radiation was evaluated in a separate review.

The potential impact of the MUR uprate on ambient plant temperatures was addressed via the HVAC evaluations for the MNS Reactor Building(s), the Auxiliary Building(s), Fuel Handling Building, Diesel Building and Control Area and for the Doghouse Building which has no ventilation system. The evaluation for the upper and lower Containment HVAC Systems showed that the MUR uprate would not increase the overall heat load for the Containment, and also showed that the Containment ambient temperature would be unaffected since the normal operating temperature in upper and lower Containment is controlled by the Technical Specifications for both current and post-MUR conditions. Therefore, the temperatures used for

EQ analysis of Containment components at MNS are unchanged. The 2°F increase in Main Feedwater system temperature due to the MUR uprate will not affect the heat loading in the Doghouse since this area is vented to the environment and has no ventilation system. The evaluation also showed that the MUR uprate will not impact the HVAC in other areas of the Auxiliary Building at all. Therefore, the temperatures used for EQ analysis of Auxiliary Building components at MNS are unchanged. The potential impact of the MUR uprate on system temperature changes was evaluated as part of the MUR uprate engineering system reviews. The BOP systems review considered the Main Steam and Feedwater Systems. The results of these evaluations showed that Main Feedwater process temperatures changed approximately 2°F and that Main Steam process temperatures changed approximately 0.04°F (a decrease at the Steam Generator outlet). These slight parameter changes do not affect the qualification of any EQ components at MNS because the temperatures have already been evaluated as not impacting the ambient temperatures of the Containment and Auxiliary Building. As noted in Section VI.1.F of the License Amendment Request (LAR) the Control Area, Auxiliary Building (including Fuel Building) and Diesel Building Ventilation Systems remain bounded for the MUR power uprate conditions.

Temperature and pressure conditions in Containment following a Large Break LOCA or Main Steam Line Break were discussed in LAR Enclosure 2, Section II.1.D.43. As noted, the Updated Final Safety Analysis Report (UFSAR) analyses for these events were performed at 102 percent RTP (3479 MWt) and bound the MUR conditions. LAR Enclosure 2, Section II.1.D.44 addressed a postulated secondary system pipe break in the Doghouse. This analysis was also performed at 102 percent RTP (3479 MWt) and likewise bounds MUR conditions.

To summarize, the evaluation of the temperature and pressure review (due to the MUR uprate), the BOP systems were determined to show some slight parameter changes, but these minor changes were shown to have no impact on the EQ components at MNS. The evaluation of the systems inside Containment and in the Doghouse for accident temperature and pressure conditions showed that the current design basis analyses were performed at 102 percent RTP, which bounds the MUR uprate. There is no EQ impact with respect to temperature or pressure due to the MUR uprate. No areas went from mild to harsh as a result of the MUR uprate based on temperature.

Radiation:

The potential impact of the MUR uprate on radiation dose was evaluated for MNS EQ equipment in the equipment data base. No items were identified that were impacted by the MUR power uprate dose change (i.e. were qualified for the pre-MUR Total Integrated Dose but not qualified for the post-MUR Total Integrated Dose).

This review evaluated one existing operable but degraded/non-conforming condition (OBDN) and one operability issue. Certain reactor vessel level indication RTDs were previously determined to be OBDN dependent on their installed location. In addition, one of three MNS Unit 1 pressurizer level transmitters was determined to be inoperable for post accident monitoring but operable for other required normal operation functions. Resolution of these

existing conditions, which are being tracked in the corrective action program, are applicable for current operating conditions and conditions after MUR implementation.

In order to evaluate the EQ components for the MUR, the pre-MUR Total Integrated Doses (TIDs) in the MNS Plant Environmental Parameters (PEP) Manual were increased by 2 percent. This 2 percent dose increase is assumed to envelope the MUR power level increase of 1.7 percent. This 2 percent dose increase is applied to the normal operation dose for the eleven years of operation after the planned MUR uprate, but not to the 29 years prior to the MUR uprate. The 2 percent dose increase is also applied to the accident dose rate. The TID then equals the sum of the 29 year non-MUR operating dose, the eleven year MUR operating dose and the accident dose. No new areas became harsh as a result of the MUR uprate based on radiation.

As a result of this review (temperature, pressure, and radiation), there are no impacts to EQ equipment as a result of the MUR and no specific component modifications are required to support the MUR.

Enclosure 2

McGuire Nuclear Station's Response to MUR LAR RAI Questions 20 thru 31 In The June 6, 2012 NRC Request for Additional Information

NRC Question 20

The LAR, Attachment 4, provided the following Caldon® Ultrasonics Engineering Reports (ER):

(1) ER-819, "Bounding Uncertainty Analysis for Thermal Power Determination at McGuire Unit 2 Using the LEFM CheckPlus System," Rev. 1,

(2) ER-822, "Bounding Uncertainty Analysis for Thermal Power Determination at McGuire Unit 1 Using the LEFM CheckPlus System," Rev. 1,

(3) ER-822, "Bounding Uncertainty Analysis for Thermal Power Determination at McGuire Unit 1 Using the LEFM CheckPlus System," Rev. 1,

(4) ER-823, "Meter Factor Calculation and Accuracy Assessment for McGuire Unit 2," Rev. 0,

(5) ER-874, "Meter Factor Calculation and Accuracy Assessment for McGuire Unit 1," Rev. 1.

These reports provide the analysis of the uncertainty contribution of the LEFM CheckPlus System to the thermal power uncertainty of McGuire, Units 1 and 2. These Reports contain detailed calculations, the results of which are summarized in Appendix C, Table I, of ER-819 and ER-822. These calculations are based on the following references:
(a) Cameron Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," and
(b) Cameron Engineering Report ER-157P, "Supplement to Cameron Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus," dated May, 2008, Rev. 8 and Rev. 8 Errata.

To assist the NRC staff review in completing its review, please provide:

a. A detailed and explicit cross reference between the plant-specific Cameron Engineering Reports ER-819, ER-822, ER-823, and ER874 and the Cameron Topical reports ER-80P and ER-157P.

b. Confirmation that the assumptions listed in Cameron Caldon® Ultrasonics Engineering Report No. ER-157P, Rev. 8 and Rev. 8 Errata, Appendix A, are valid for the McGuire 1 and 2 application.

c. The LAR in Enclosure 2, Item 1.1.D, Criterion 1, from ER-157 (Rev 8, Page E2-6), requires the licensee to justify continued operation at the pre-failure level for a pre-determined time and the decrease in power that must occur following that time. The response provided in the LAR states that an engineering evaluation was performed to justify an allowed outage time upon loss of the LEFM signal. The LAR further states that the analysis established a bounding uncertainty of 0.045% RTP over a 7-day period. Provide an engineering evaluation on how the bounding uncertainty of 0.045% RTP was established and justifications for the selection of a 7 -day time period.

d. The LAR states in Table 1.1.E-1, Total Thermal Power Uncertainty Determination, that the McGuire Specific Gains/Losses are +0.088% and -0.087%. Provide detailed information on how these parameters have been established.

(EICB 1)

McGuire Response to Question 20

20 a. This response is provided in Enclosure 3 and Enclosure 4.

20 b. This response is provided in Enclosure 3 and Enclosure 4. See the below text for additional information related to this response.

Confirmation of an assumption in ER-157P, Revision 8, related to feedwater line pressure accuracy (reference response 20.b.5 in Enclosure 3) indicates that Duke is to confirm pressure input provided to LEFM is +/- 15 psi. Duke Energy confirms that the pressure in the feedwater line at the point of flow measurement is measured with an accuracy of +/- 15 psi or better.

Confirmation of an assumption in ER-157P, Revision 8, related to steam supply pressure accuracy (reference response 20.b.6 in Enclosure 3) indicates that Duke is to confirm the response. A description of the Duke Energy McGuire Nuclear Station (MNS) site specific uncertainty analysis is provided in the response to Question 20.d below. Included in the site specific uncertainty is a steam pressure uncertainty of approximately 50 psi. The other elements of the thermal power uncertainty not included as part of the LEFM uncertainty are also discussed in the response to Question 20.d below.

Confirmation of an assumption in ER-157P, Revision 8, related to reactor thermal power (reference response 20.b.9 in Enclosure 3) indicates that Duke is to confirm the response. The equation listed on page A-1 of ER-157 (P-A) contains the same terms as identified in the response to Question 20.d below. The LEFM portion of the equation is calculated and provided by Cameron in ER-819 and ER-822, but it only accounts for the feedwater mass flow rate uncertainty and feedwater enthalpy uncertainty. The remaining uncertainties associated with reactor coolant system losses, reactor coolant

pump heat addition, and main steam uncertainties are accounted for as discussed in the response to Question 20.d below.

20 c. Duke Energy contracted Cameron to support the justification of continued operation with a failed LEFM at the uprated Rated Thermal Power (RTP) for a period of seven days. If the LEFM is not returned to service within the 7 day period, then power will be reduced to the pre-MUR licensed RTP.

The justification is based on the recognition that power can be controlled using the secondary power calorimetric which is calculated continuously by the Operator Aid Computer (OAC) without the LEFM. The OAC uses flow signals from the Main Feed Water system flow nozzles. When the LEFM is operation, the signals from the flow nozzles are calibrated to it. If the LEFM fails, then the calibration is locked to the last good value. If the plant continues to run at 100 percent power, the indicated power calculated by the OAC can drift over time, and eventually degrade the confidence in the OAC calculated power.

To determine the OAC drift rate, a statistical analysis was performed studying one year of full power operation OAC calorimetric data compared to Turbine First Stage pressure. Turbine pressure was used as a surrogate for reference power since the LEFM is not yet operating. First Stage pressure is not constant and also fluctuates, so the variances in this data are higher than would be if a true reference power existed. Ten minute average power data was collected at fifteen minute intervals, a total of more than 34,000 raw data points. The data was tested for normality in distribution, and inspected for any outliers. Because the analysis spanned an entire year, any drift associated with seasonal variations could be detected and adjusted if desired, although this proved to be unnecessary. The analysis was performed separately for each MNS nuclear unit.

The data was then subjected to a statistical analysis with Microsoft Excel software, using the STDEVP function, which calculates standard deviation based on the entire population. The standard deviation is a measure of how widely values are dispersed from the average value (the mean). STDEVP uses the following formula:

$$\sqrt{\frac{\sum(x-\bar{x})^2}{n}}$$

where x is the sample mean AVERAGE(number1,number2,...) and n is the sample size.

The spreadsheet calculations were inspected by Duke Energy, and run independently on a Duke Energy computer, with identical results. Using the calculated standard deviation, an estimate of the maximum expected drift over seven days was calculated, at a 95 percent probability and 95 percent confidence interval. The calculations were also performed for three day and 10 day periods. The results reported are the 10 day drift results, as follows:

	Secondary Calorimetric 95/95 Uncertainty	Number of consecutive 10 day points tested
Unit 1	0.023 percent	25,536
Unit 2	0.045 percent	27,386

Although the analysis supports using a 10 day out-of-service (OOS) restoration time, Duke Energy elected to restrict the interval to seven days for conservatism. Also, the bounding Unit 2 value was applied to both Units for the sake of simplicity for operators. Because the statistical analysis used only full power data, it is not directly applicable to lower power levels. Therefore, these results will only be applied when at full power. If it is necessary to down-power the unit for any reason while the LEFM is out of service, power will not be returned to RTP until after the LEFM is repaired.

The additional LEFM OOS allowance of 0.045 percent is added algebraically to the normal MUR uncertainty, i.e. as a bias to the 0.3 percent uncertainty allowance when the LEFM is available, resulting in a temporary allowance of 0.345 percent.

20 d. MNS is a 4 loop Westinghouse Pressurized Water Reactor. The equation for reactor thermal power from a specific loop is given by:

$$Q_{loop} = w_{steam} \cdot h_{steam} - w_{feed} \cdot h_{feed} + w_{bd} \cdot h_{bd} + Q_{loss} - Q_{RCP} - w_{chrg} \cdot h_{chrg} - w_{seal} \cdot h_{seal} + w_{letdown} \cdot h_{letdown} + w_{excess\ letdown} \cdot h_{excess\ letdown}$$

Where: Q_{Loop} = Thermal power of the loop (btu/hr)
w_{steam} = loop steam flow (lbm/hr)
w_{feed} = loop feed flow (lbm/hr)
w_{bd} = loop steam generator blowdown flow (lbm/hr)
w_{seal} = reactor coolant pump seal injection flow - seal leakoff flow (lbm/hr)
w_{chrg} = charging flow (lbm/hr)
$w_{letdown}$ = letdown flow (lbm/hr)
$w_{excess\_letdown}$ = excess letdown flow (lbm/hr)
h_{steam} = main steam enthalpy (btu/lbm)
h_{feed} = main feed enthalpy (btu/lbm)
h_{bd} = steam generator blowdown enthalpy (btu/lbm)
h_{chrg} = charging enthalpy (btu/lbm)
$h_{letdown}$ = letdown enthalpy (btu/hr)

$h_{excess\_letdown}$ = excess letdown enthalpy (btu/lbm)
Q_{loss} = radiative heat losses to containment atmosphere (btu/hr)
Q_{RCP} = heat input to reactor coolant system from the reactor coolant pumps (btu/hr)

Not all loops contain all terms (e.g., Loops A & D contain the charging term while Loops B & C do not and Loop C contains the letdown terms while Loops A, B, & D do not).

Plant data is obtained from a snapshot in time that represents nominal full power conditions. The data is then manipulated to obtain the units necessary for the calculation or, in the case of enthalpy, the values are calculated from the appropriate pressure/temperature indication. The power to each loop is calculated and the sum of all four loops then represents the total power calculated.

To calculate the total uncertainty of the non-LEFM portion of the thermal power calculation, the change in power for a change in one of the inputs is calculated for each input to the calculation. This difference is squared. The square root of the sum of the squares is the uncertainty of the final calculation. Expressed mathematically,

$$\text{Uncertainty}_{\text{Power}} = \frac{\sqrt{\sum_{n} \left(\text{Power}_{\text{Ref}} - \text{Power}_{n}\right)^2}}{\text{Power}_{\text{Ref}}} \cdot 100$$

Where $\text{Uncertainty}_{\text{Power}}$ = Total uncertainty of the calculated thermal power (%)
$\text{Power}_{\text{Ref}}$ = Calculated thermal power based on the stated inputs (MWt)
Power_{n} = Power calculated with parameter n adjusted by the uncertainty of the measurement (MWt)

The total uncertainty for the non-LEFM portion of the calorimetric is calculated using the programming capability of Mathcad. For this calculation, the feedwater pressure, pressurizer pressure, and reactor coolant system cold leg temperature uncertainties are known from previous Duke Energy uncertainty calculations. All other parameters have an assumed uncertainty of 5 percent of reading. The uncertainties are multiplied by the value of x which is either 1 or -1 to account for the direction the uncertainties are to be applied (positive or negative.)

$$\text{U1\_Uncertainty}_{\text{Power}}(\text{Data},x) := \left| \begin{array}{l} \text{Power}_{\text{Ref}} \leftarrow \text{Power}(\text{Data}) \\ \Delta\text{Power} \leftarrow 0 \\ \text{for } i \in 1..64 \\ \quad \left| \begin{array}{l} A \leftarrow \text{Data} \\ A_{i,2} \leftarrow A_{i,2} + x \cdot A_{i,3} \\ \Delta\text{Power} \leftarrow \Delta\text{Power} + \left(\text{Power}(A) - \text{Power}_{\text{Ref}}\right)^2 \end{array} \right. \\ \dfrac{\sqrt{\Delta\text{Power}}}{\text{Power}_{\text{Ref}}} \end{array} \right.$$

Mathematically, the above Mathcad coding is equivalent to a do-loop in FORTRAN. Essentially, there is an Excel spreadsheet that contains all the parameters (64 total) that go into the calculation of thermal power at one snapshot in time. The coding reads in the spreadsheet as an array, A. The array is arranged as $A_{row,column}$ such that $A_{i,2}$ is row i, column 2. Delta power is calculated assuming a nominal value for each of the 64 parameters except for row i, which assumes the parameter is equal to the nominal value (column 2) plus (or minus) the uncertainty (column 3). The power uncertainty is then calculated as the square root sum of the squares for all the delta powers divided by the reference power value (calculated power without any uncertainties applied).

The values for the Unit 1 uncertainties are as provided in Table I.1.E-1 of the LAR, or:

$\text{U1\_Uncertainty}_{\text{Power}}(\text{Data},1)^2 = 0.088$ %RTP

$\text{U1\_Uncertainty}_{\text{Power}}(\text{Data},-1)^2 = -0.087$ %RTP

The same calculation is then performed for Unit 2 and the same uncertainties are calculated.

NRC Question 21

In Section IV.1.A.V, "Balance-of-Plant (BOP) piping," of the LAR, the licensee stated that at MUR-PU [measurement uncertainty recapture -power uprate] conditions the steam generator blowdown system (SGBS) will continue to remain within its design basis. Please confirm that the SGBS will continue to perform its intended function given the higher flow and potentially higher impurity content under the proposed MUR-PU conditions. (ESGB 1)

McGuire Response to Question 21

The Steam Generator Blowdown System will continue to remain within its design basis at MUR power uprate conditions. The purpose of the Blowdown System is to maintain acceptable steam generator (SG) shell side water chemistry by removing impurities from the secondary side. Impurities within the secondary side result from corrosion, demineralized water makeup, steam generator tube leaks, and main condenser tube leaks. The Blowdown System operates continuously with the system flow rate set based on plant chemistry requirements. Any increase in Blowdown System flow rate caused by potentially higher impurity content under MUR conditions would be bounded by the increase in overall secondary side flow of 2.4 percent resulting from the MUR. Therefore, the Blowdown System was evaluated conservatively with a bounding increase in flow of 2.4 percent. The evaluated 2.4 percent increase in Blowdown flow at the uprate conditions remain below the current design flow of the system. The Steam Generator Blowdown System will continue to perform its intended function given the potentially higher flow and impurity content under the proposed MUR conditions.

NRC Question 22

Please discuss whether components susceptible to flow-accelerated corrosion (FAC) in the SGBS will continue to be managed in accordance with the FAC program. (ESGB 2)

McGuire Response to Question 22

The components of the Stream Generator Blowdown System susceptible to flow-accelerated corrosion will continue to be managed in accordance with the FAC Program. The MUR Power Uprate will not result in the removal of components currently managed in the FAC Program.

NRC Question 23

On page E2-69 of Enclosure 2 of the LAR, the licensee stated that FAC-related piping wear rates will be impacted by the MUR-PU; however, the changes will be small. The FAC monitoring program includes the use of a predictive method to calculate the wall thinning of components susceptible to FAC. In order for the NRC staff to evaluate the accuracy of these predictions, the NRC staff requests a sample list of components most affected by FAC for which wall thinning is predicted and measured by (UT) or other methods. Include the initial wall thickness (nominal), current (measured) wall thickness, and a comparison of the as-measured wall thickness to the thickness predicted by the CHECWORKS™ modeling software. (ESGB 3)

McGuire Response to Question 23

As a result of plant and industry experience with pipe degradation in process systems, a Flow Accelerated Corrosion (FAC) program was developed at MNS. The purpose of the program is to monitor piping systems that are subject to FAC degradation, and to mitigate pipe wall loss. The FAC program is based on the guidance of EPRI NSAC-202L-R3.

MNS uses the EPRI CHECWORKS Steam/Feedwater Application (SFA) monitoring software to model operating conditions, material data, and UT inspection data to provide a calculated estimate of component wear. The thermodynamic changes associated with the MUR will impact corrosion rates for components located on FAC susceptible systems. All changes required to reflect the MUR conditions are being incorporated in the Checworks models and databases are undergoing formal validation.

A preliminary wear rate analysis has been performed to assess the impact of the MUR on susceptible FAC components and results are shown in the Tables at the end of this Enclosure. Only run definitions that yielded an increase in wear are tabulated. They provide a comparison of the pre-MUR and post-MUR wear rates. Per this analysis, the increase in wear rates due to the MUR power uprate is considered minor and the existing FAC Program is adequate to incorporate the updated predictions. Final data is expected to confirm the preliminary results and will be available around August 31, 2012.

NRC Question 24

Please provide the differential pressure across the steam generator (SG) tubes, the temperature of the secondary water, and the feedwater and steam flow rates through the SG for power uprate conditions. Confirm that the SGs will still satisfy all original design criteria under these conditions. If not, please provide a reassessment to address the current condition of your SGs (e.g., the plugs, any repairs to the as-built configuration, loose parts, etc.) at the proposed power uprate conditions. (ESGB 4)

McGuire Response to Question 24

As shown in LAR Table IV-1, the SG outlet pressure decreases from 1021 psia at current full power conditions to 1020.7 psia at 102 percent uprate conditions and the reactor coolant system pressure remains unchanged at 2250 psia. Therefore, the normal operating differential pressure across a SG tube increases from 1229 psid at current conditions to 1229.3 psid at 102 percent uprate conditions.

As shown in LAR Table IV-1, the feedwater temperature increases from 440 °F at current conditions to 442 °F at 102 percent uprate conditions. As shown in this same table, the steam flow rate increases from 15.1 E6 lb/hr at current conditions to 15.5 E6 lb/hr at 102 percent

uprate conditions. The feedwater flow rate increases from 15.2 E6 lb/hr at current conditions to 15.6 E6 lb/hr at 102 percent uprate conditions

As discussed in LAR Section IV.1.A.vi, the MUR power uprate conditions are bounded by the replacement SG design conditions. The SGs will still satisfy all original design criteria under MUR conditions.

NRC Question 25

Confirm that your SG plugging limit is still appropriate for the proposed MUR-PU conditions, given the guidance in Regulatory Guide (RG) 1.121. (ESGB 5)

McGuire Response to Question 25

Duke Energy confirms that the MNS Unit 1 and Unit 2 SG plugging limit of 40 percent is still appropriate for the proposed MUR-PU conditions given the guidance in RG 1.121.

NRC Question 26

On Table II.1-1: McGuire Analyses of Enclosure 2 of the LAR, the licensee stated that the feedwater system malfunction causing an increase in feedwater flow and the excessive increase in secondary steam flow analyses were performed at 3469 MWt and are bounding for MUR-PU conditions. The NRC staff notes that the power used in the analyses is 101.7 % of 3411 MWt (authorized core power level), which is equal to the MUR-PU conditions. Please discuss how these analyses are bounding for MUR-PU conditions. (ESGB 6)

McGuire Response to Question 26

As stated in the Feedwater System Malfunction Causing an Increase in Feedwater Flow discussion preceding Table II.1-1 in the March 5, 2012 LAR, the analysis is performed to ensure departure from nucleate boiling (DNB) does not occur. Since this is a DNB analysis, the uncertainty in power is included in the statistical core design (SCD) DNBR limit as documented in the NRC approved SCD methodology report DPC-NE-2005-PA, "Duke Energy Carolinas Thermal-Hydraulic Statistical Core Design Methodology". The SCD methodology allows for the uncertainty in core power, reactor coolant system flow, reactor coolant system temperature, and reactor coolant system pressure to be statistically combined with the uncertainties in nuclear peaking to produce a statistical DNBR limit. This allows the deterministic system analyses presented in MNS Updated Final Safety Analysis Report (UFSAR) Chapter 15 for the DNBR acceptance criterion to be initiated from nominal values for those parameters. The SCD DNBR limit was generated assuming a thermal power uncertainty of ± 2 percent Reactor Thermal Power, which is greater than the proposed thermal power uncertainty of ± 0.3 percent Reactor Thermal Power. Since the analysis was performed at the proposed nominal power level of 3469

MWt and the statistical DNBR limit includes an allowance for thermal power uncertainty that bounds the proposed thermal power uncertainty, the analysis presented in UFSAR Chapter 15.1.2 is bounding for MUR-PU conditions.

NRC Question 27

Confirm that the coating qualification temperature and pressure profile used to qualify the original maintenance Service Level 1 coatings continue to bound the Design Basis Accident temperature and pressure profile under the proposed power uprate conditions. (ESGB 7)

McGuire Response to Question 27

Containment design pressure and temperature profiles were used to qualify the Service Level 1 coatings. The proposed power uprate does not change the current Design Basis Accident temperature and pressure profiles. Therefore, the coating qualification temperature and pressure profiles used to qualify the original maintenance Service Level 1 coatings continue to bound the Design Basis Accident temperature and pressure profiles under the proposed power uprate conditions.

NRC Question 28

Test fidelity, such as test versus planned plant configuration, test variations to address configuration differences, and potential effects of operation on flow profile and calibration, should be addressed on a plant-specific basis. Applicant requests must provide a comparison of the test and plant piping configurations with an evaluation of the effect of any differences that could affect the ultrasonic flow meter (UFM) calibration. Further, sufficient variations in test configurations must be tested to establish that test-to-plant differences have been bracketed in the determination of UFM calibration and uncertainty. The turbulent flow regimes that exist when the plant is near full power result in limited upstream flow profile perturbation from downstream piping. Consequently, the effects of downstream equipment need not be considered for normal CheckPlus operation, provided changes in downstream piping, such as the entrance to an elbow, are located greater than two pipe diameters downstream of the chordal paths. However, if the CheckPlus is operated with one or more transducers out of service, the acceptable separation distance is likely a function of transducer to elbow orientation. In such cases, if separation distance is less than five pipe diameters, it should be addressed. Therefore the NRC staff requests that the licensee provide downstream distances from the UFM to the next non-straight pipe component at the Alden labs test setup to verify their applicability to the in plant setup. (SRXB 1)

McGuire Response to Question 28

This response is provided in Enclosure 3 and Enclosure 4.

NRC Question 29

Each applicant for a power uprate must conduct an in-depth evaluation of the UFM following installation at its plant that includes consideration of any differences between the test and in-plant results and must prepare a report that describes the results of the evaluation. This should address such items as calibration traceability, potential loss of calibration, cross-checks with other plant parameters during operation to ensure consistency between thermal power calculation based upon the LEFM and other plant parameters, and final commissioning testing. The process should be described in written documentation and a final commissioning test report should be available for NRC inspection. Therefore the staff requests the licensee to provide a summary of the data comparing the LEFM Checkplus operating data to the Venturi data for the past six months to verify consistency between thermal power calculation based on LEFM and other plant parameters. (SRXB 2)

McGuire Response to Question 29

As indicated in the response to Criterion 2 From ER-80P on Page E2-4 of the MNS MUR License Amendment Request dated March 5, 2012, the LEFM Checkplus system was not installed on either MNS Unit at the time the MUR LAR was submitted on March 5, 2012. The LEFM Checkplus system is scheduled to be installed in the Fall 2012 MNS Unit 2 refueling outage and the Spring 2013 MNS Unit 1 refueling Outage.

Given the above, the requested data cannot be provided at this time. After the LEFM Checkplus system is installed and operational on the respective Unit, six months of data will be collected comparing the LEFM Checkplus operating data to the Venturi data to verify consistency between thermal power calculation based on LEFM and other plant parameters. This data will be available for NRC inspection seven months after the LEFM Checkplus system is installed and operational on the respective Unit. This is identified as a commitment in Enclosure 6.

NRC Question 30

RIS 2002-03, Section II, provides for accidents and transients for which the existing analyses of record bound plant operation at the proposed power level. Please explain how the 0.3% feedwater flow measurement uncertainty is treated in departure from nucleate boiling analysis for which the stated analysis is bounded by the current analysis of 101.7%. (SRXB 3)

McGuire Response to Question 30

The 0.3 percent is not an uncertainty on feedwater flow measurement. It is an uncertainty in core thermal power as a result of the uncertainty on the measured feedwater flow, measured feedwater pressure/temperature, and measured steam pressure/temperature. The explanation for how the 0.3 percent uncertainty is treated in departure from nucleate boiling analyses for which the stated analysis is bounded by the current analysis of 101.7 percent is provided in the response to Question 26 above.

NRC Question 31

Table II.1-1, "McGuire Analyses," of the LAR provides information regarding, the McGuire 1 and 2 confirmation that the bounding event remains valid (i.e. Enclosure 2, Section II.1.D, of the LAR) and the references for the NRC staff approval of these events. Many of the Section II.1.D evaluations of the impact of the MUR on the dose analysis address the impact of the MUR on the offsite doses, but not control room doses. Many of the references for the NRC approval reference only the NRC's Standard Review Plan (SRP) or RGs rather than the documents that provide the NRC's approval for LARs such as an NRC Safety Evaluation or sections in the Updated Final Safety Analyses Report (UFSAR).

The regulation at 10 CFR, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19 (GDC 19) states that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. It also states that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, or 5 rem Total Effective Dose Equivalent as applicable.

SRP 15.0, "Introduction – Transient and Accident Analysis," Revision 3, dated March 2007 (ADAMS Accession No. ML070710376) states:

> The reviewer considers the possible case variations of AOOs [anticipated operational occurrences] and postulated accidents presented to verify that the licensee has identified the limiting cases.

Using the information provided in the LAR and in the UFSAR, the NRC staff is unable to verify that the radiological limits in GDC 19 are met with the proposed change. Please provide the references to the NRC-approved evaluation or provide an evaluation of the impact of the proposed change on control room doses for all accidents and AOO's in the design bases so that the NRC staff can verify that the limiting cases have been identified and confirm that GDC 19 is met or state where this information can be found. If there are many references for the information, it would be helpful to summarize the accident doses for the control room in a table. (AADB 1)

McGuire Response to Question 31

Previously NRC-Reviewed Accidents:

The NRC issued a Safety Evaluation Report (SER) detailing approval for full scope implementation of the method of Alternative Source Term (AST). This approval was based on the staff review of an AST analysis of the MNS Loss of Coolant Accident (LOCA - cf. ADAMS Accession No. ML090890627). More recently, the dose consequences were reviewed and approved pertaining to the adoption of Water Management (LOCA - cf. ADAMS Accession No. ML11131A133). The NRC earlier issued an SER approving partial scope implementation of the method of AST at McGuire based on its review of AST analyses of the Fuel Handling Accident (FHA) and Weir Gate Drop (WGD - cf. ADAMS Accession No. ML060250098). These calculations contained both offsite and control room doses, per Regulatory Guide 1.183. The source term for these accidents was developed assuming a reactor power of 102 percent current rated power.

Also, the SG Replacement SER (SGTR - cf. ADAMS Accession No. ML013230307) contains information regarding acceptable control room dose consequences (in accordance with GDC-19) for the Main Steam Line Break (MSLB), Locked Rotor Accident (LRA), Rod Ejection Accident (REA), and Steam Generator Tube Rupture (SGTR).

Accidents With Fuel Failure:

The following accidents have a source term that contains a failed fuel aspect:

- Loss of Coolant Accident (LOCA)
- Locked Rotor (LRA)
- Fuel Handling Accident (FHA)
- Weir Gate Drop (WGD)
- Rod Ejection Accident (REA)
- Single Rod Withdrawal (SRW)
- Tornado Missile Accident (TMA)
- Cask Drop

For each of these accidents, the calculated values of control room radiation doses have been shown to be within the limits of GDC-19. The radioactive source term for each of these accidents was developed based on reactor operation at 102 percent current rated power, which envelopes the impact of the proposed MNS uprate and associated uncertainties.

As allowed by the NRC approval of full scope implementation of AST at MNS, Duke Energy will update the UFSAR for the LRA, REA, SRW, TMA, and Cask Drop accidents as they are reanalyzed using the AST method. Duke Energy will also update the UFSAR to report the calculated control room doses.

Accidents Without Fuel Failure:

The following accidents do not include any fuel failure and use activity limits provided in the Technical Specifications, except for the Waste Gas Decay Tank Failure, where the radioactive source term is taken from MNS Selected Licensee Commitment (SLC) 16.11-20:

- Steam Generator Tube Rupture (SGTR)
- Main Steam Line Break (MSLB)
- Instrument Line Break (ILB)
- Loss of External Load (LEL)
- Waste Gas Decay Tank Failure
- Liquid Tank Failure

Control room doses have been calculated for these accidents except for the Loss of External Load and shown to be within the limits of GDC 19. The radiation doses for the Loss of External Load, both at offsite locations and in the control room, are bounded by the corresponding radiation doses of the Main Steam Line Break. As allowed by the NRC approval of full scope implementation of AST at MNS, Duke Energy will update the UFSAR for the SGTR, MSLB, ILB, and liquid tank failure accidents as they are reanalyzed using the AST method. Duke Energy will also update the UFSAR to report the calculated control room doses.

Since these accidents are not followed by fuel failure, and the activity is associated with either the Technical Specifications or SLC, the proposed power uprate has no impact on radiation doses following these accidents, either at offsite locations or in the control room.

Anticipated Operational Occurrences (AOOs):

The NRC defines "anticipated operational occurrences" so that the term means "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." As defined, the term "anticipated operational occurrence" includes ANS Condition I occurrences such as shutdown, heatup, refueling, operation with the reactor and secondary coolant contaminated with fission products, step and ramp load changes, and load rejection up to a complete load rejection. The term also includes ANS Condition II and III events.

The conduct of operations at MNS ensures that the control room is habitable in conformance to GDC 19 during all phases of normal plant operations, including any ANS Condition I event.

The acceptability of control room radiation doses for many of the ANS Condition II and III events in the MNS design and current license bases has already been discussed above. It has been shown that the proposed MUR uprate will not have an effect on the control room radiation doses

calculated for these events. For many other ANS Condition II and III events in the MNS design and current license basis, there are no explicit analyses of radiological consequences. Instead, the UFSAR sections on the environmental consequences for many of these events reports a qualitative evaluation that concludes that "the radiological consequences of ... [any of these events] will be less severe than the steamline break event analyzed in Section 15.1.5.3 since no fuel damage is expected to occur." These events include the following:

- Inadvertent Opening of a SG Safety or Relief Valve (UFSAR 15.1.4)
- Turbine Trip (UFSAR 15.2.3)
- Loss of Offsite Power (UFSAR 15.2.6)
- Loss of Main Feedwater (UFSAR 15.2.7)
- Feedwater Line Break (UFSAR 15.2.8)
- Partial Loss of Forced Reactor Coolant Flow (UFSAR 15.3.1)
- Complete Loss of Forced Reactor Coolant Flow (UFSAR 15.3.2)
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR 15.4.2)
- Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (UFSAR 15.4.4)
- Chemical and Volume Control System (CVCS) Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (UFSAR 15.4.6)
- Inadvertent Operation of the Emergency Core Coolant System (ECCS) During Power Operation (UFSAR 15.5.1)

As noted above, it was determined that the radiation doses following a steam line break are within the limits of GDC 19. It follows that the radiation dose in the control room for each of the events listed immediately above would be within the limits of GDC 19.

The UFSAR report of environmental consequences of the ANS Condition II Inadvertent Opening of a Pressurizer Safety or Relief Valve (UFSAR 15.6.1) states that the environmental consequences of this event are bounded by those of the LOCA. It is also noted that the DNBR remains above the limit value "throughout this transient." It follows that no fuel damage would occur for this accident. It follows that the control room radiation dose for this event will be under the limits of GDC 19.

For each the following events, the UFSAR report of the evaluation of environmental consequences states that "There will be no radiological consequences ... [for the event] since radioactivity is contained within the fuel rods and the Reactor Coolant System within design limits."

- Feedwater System Malfunction Causing an Increase in Feedwater Flow (UFSAR 15.1.2)
- Excessive Increase in Secondary Steam Flow (UFSAR 15.1.3)
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Condition (UFSAR 15.4.1)

- Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (UFSAR 15.4.7)

It follows that following any of these events, the control room will remain habitable in conformance to GDC 19.

- UFSAR Chapter 15 reports the evaluation of the plant response to additional anticipated operational occurrences, stating that their consequences are bounded by the consequences of one of the events listed above. These events included the following:
- Feedwater System Malfunction Resulting in a Reduction in Feedwater Temperature (UFSAR 15.1.1 - consequences bounded by those of a Feedwater System Malfunction Causing an Increase in Feedwater Flow - UFSAR 15.1.2)
- Inadvertent Closure of a Main Steam Isolation Valve (UFSAR 15.2.4 - consequences bounded by those a Turbine Trip - UFSAR 15.2.3)
- Loss of Condenser Vacuum or Other Events Causing a Turbine Trip (UFSAR 15.2.5 consequences bounded by those of a Turbine Trip - UFSAR 15.2.3)

The UFSAR states that "There will be no radiological consequences" for an increase in main feedwater flow. It follows that the control room remains habitable in conformance to GDC 19 following a reduction in feedwater temperature. Since the consequences of a inadvertent closure of a main steam isolation valve, and loss of condenser vacuum and any other event causing a turbine trip would be bounded by those a turbine trip, the control room radiation dose for these events would be bounded by those of a main steam line break and so within the limits of GDC 19.

No fuel damage occurs following each of these additional "anticipated operational occurrences." The radiological source term associated with these events then would be activity in the reactor and secondary coolant up to the limits specified in the plant Technical Specifications. As noted earlier, the proposed MUR uprate has no impact at all on the source terms for any of these events. As such, radiation doses for each of these events, at offsite locations and in the control room, will remain within the corresponding limits with implementation of the MUR uprate.

Summary:

All of the MNS design basis accidents and Anticipated Operational Occurrences (AOOs) described in Chapter 15 of the UFSAR have been reviewed to assess any potential impacts resulting from the uprate. In the analysis of radiological consequences for those accidents with fuel damage, the source term already incorporated a 2 percent increase in power, which envelopes the impact of the proposed uprate. The analysis of radiological consequences of the remaining accidents in the MNS current design basis take radioactive source terms that are developed independent of reactor power. Radiation doses for each of the AOOs, at offsite locations and in the control room, will remain within the corresponding limits with implementation

of the MUR uprate. Therefore, this proposed uprate has no impact on either offsite or control room dose consequences and the radiological limits in GDC19 are met under post-MUR uprate conditions.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION BY THE OFFICE OF NUCLEAR REGULATION REGARDING A MCGUIRE LICENSE AMENDMENT TO SUPPORT A MEASUREMENT UNCERTAINTY RECAPTURE (MUR) POWER UPRATE

Enclosure 3

Cameron Caldon Ultrasonics

Response to McGuire MUR LAR RAI Questions 20a, 20b, and 28

CAMERON

Caldon® Ultrasonics

RAI Responses for McGuire Nuclear Station

Prepared by: John Spadaro
Reviewed by: David Markowski DTM
Checked by: Ryan Hannas RH

July 2012

CAMERON

RAI Responses for McGuire Nuclear Station

Table of Contents

CAMERON

Response to 20.a. Cross reference between plant specific Engineering reports and Topical Reports:

1. Response: ER-972 revision 2 provides a detailed cross reference between the plant-specific engineering reports ER-822 and ER-819 (Unit 1 and 2 respectively) and the Cameron Topical reports. ER-157 (P-A), Appendix A contains the information necessary to perform the cross reference between the plant-specific uncertainty and the topical reports. The methodology of ER-157 (P-A) Appendix A is the same as Appendix E of ER-80P with differences in numerical uncertainty estimate as indicated on page A-1 of ER-157 (P-A).

   With regards to the Meter Factor Calculation and Accuracy Assessment Reports, ER-823 and ER-874, the uncertainties calculated in these documents are traceable to ER-157 (P-A) Appendix A, pages A-17 to A-19. Specifically, a discussion identifying sources of error in the meter factor as well as a methodology for combining these terms is presented.

Response to 20.b. Validation of ER-157 (P-A) Assumptions as listed on page A-1:

1. ER-157 (P-A) - The LEFM system measures total feedwater flow. The Plant is either a PWR or a BWR.

   Response: McGuire Units 1 and 2 are pressurized water reactors (PWRs).

2. ER-157 (P-A) - The feed line has an internal diameter of 24 inches, and a feedwater velocity of 20 ft/sec.

   Response: The McGuire Units 1 and 2 applications utilize nominal 18" feedwater lines. The spool pieces internal diameters and feedwater velocities are reflected in the McGuire engineering reports (ER-819, and ER-822 for Units 2 and 1 respectively). The resulting uncertainties in the reports were calculated based on the actual plant specific spool piece dimensions and velocities.

3. ER-157 (P-A) - The feedwater flow measurement is downstream of the final stage of feed heating. The LEFM thus measures final water temperature.

   Response: The LEFM spool pieces are located downstream of the final feedwater heating for the McGuire Units 1 and 2 applications.

4. ER-157 (P-A) - The steam pressure for the PWR and the BWR is 1000 psia.

   Response: This assumption is replaced with the appropriate plant specific value in ER-819 and ER-822.

CAMERON

5. ER-157 (P-A) - The pressure in the feedwater line at the point of flow measurement is measured with an accuracy of ± 15 psi or better.

   Response: Duke to confirm pressure input provided to LEFM is ± 15 psi.

6. ER-157 (P-A) - The steam supply pressure is measured by one or more utility supplied transmitters having an accuracy of ± 15 psi. Departures from the assumed accuracy of this instrument, as well as other elements of the thermal power determination that are not part of the LEFM system, will be treated in the site specific uncertainty analysis submitted by a utility as part of their Appendix K uprate package.

   Response: Duke to confirm.

7. ER-157 (P-A) - The final feed temperature is 430 °F.

   Response: The final feed temperature is 445 °F. This value is used in place of the assumed value in the plant specific uncertainty analyses (ER-819 and ER-822 for McGuire Units 2 and 1 respectively).

8. ER-157 (P-A) - Fluid properties at and adjoining the operating point are noted in Attachment 1, excerpted from the 1967 ASME steam tables.

   Response: The 1967 ASME Steam Tables were used in the plant specific uncertainty analyses (ER-819 and ER-822 for McGuire Units 2 and 1 respectively).

9. ER-157 (P-A) - Reactor Thermal Power is determined from the following equation (see equation (1) ER-157 (P-A) page A-1.

   Response: Duke to confirm.

10. ER-157 (P-A) - All errors and biases will be calculated and combined according to the procedures defined in ASME-PTC-19.1 (1985). This document defines the contributions of individual error elements through the use of sensitivity coefficients defined as follows: (refer to topical report ER-157 (P-A) Page A-3).

    Response: This assumption is satisfied in the McGuire Units 1 and 2 engineering reports (ER-822 and ER-819 respectively).

11. ER-157 (P-A) - The LEFM Algorithm is as follows: (Refer to topical report ER-157 (P-A), Rev. 8, Pages A-3 and A-4 for Equation).

    Response: This assumption is satisfied in the McGuire Units 1 and 2 engineering reports (ER-822 and ER-819 respectively).

CAMERON

12. ER-157 (P-A) - Spool piece and other data are in accordance with Attachment 2 to this Appendix.

    Response: McGuire Units 1 and 2 plant specific values have been used and are reflected in site specific engineering reports (ER-822 and ER-819 respectively).

Response to 28. Test Fidelity:

Response: All spool piece calibrations included a model of the site installation piping as well as parametric tests which vary the model inlet conditions and/or piping components to vary the hydraulics. The parametric tests are performed to determine the sensitivity of the meter factor (also referenced as the calibration constant or profile factor) to changes in the upstream hydraulics. Engineering reports ER-874 and ER-823 (units 1 and 2 respectively) provide sketches of the site piping models (figures 1, 2, 3, and 4) as well as a discussion of the individual parametric tests (table 3). Table 4 of the Meter Factor Calculation and Accuracy Assessment reports provide a listing of the meter factors, flatness ratios, and swirls (%) for all model and parametric tests. All of the tests are used to determine an average meter factor such that installation uncertainties or variations are accounted for.

In all Alden calibration tests the distance from the outlet of the LEFM to the first downstream hydraulic disturbance was 6 feet (approximately 5 pipe diameters with respect to the 15 inch diameter acoustic measurement section of the spool piece).

In the plant configuration the shortest distance from the outlet of the LEFM to the first downstream hydraulic disturbance is 4.75 feet (approximately 4 pipe diameters with respect to the 15 inch diameter acoustic measurement section of the spool piece).

The difference between the location of the downstream disturbance used in the calibration and that which exists in the plant has no impact on the UFM uncertainty. As discussed in ER-80P, page F-14, testing of a downstream bend located 0.4 diameters from the UFM produced a bias of 0.003 – 0.004 consistent with the findings of Weske (referenced in ER-80P) showing significant transverse velocity components beginning ½ diameter upstream of a bend. Cameron's spool piece design guarantees at least 1 diameter between the acoustic paths and the downstream disturbance. McGuire's spool is 1.6 diameters from the centerline to the spool discharge.

With regards to transducer failure, the LEFM system will revert to a maintenance mode uncertainty with the failure of a single path. If there is a failure of two transducers in different planes the LEFM system will be in fail mode and will not report mass flow.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION BY THE OFFICE OF NUCLEAR REGULATION REGARDING A MCGUIRE LICENSE AMENDMENT TO SUPPORT A MEASUREMENT UNCERTAINTY RECAPTURE (MUR) POWER UPRATE

Enclosure 5

Cameron Caldon Ultrasonics

Affidavit for Withholding Enclosure 4 Information From Public Disclosure

Measurement Systems

Caldon® Ultrasonics Technology Center
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Coraopolis, PA 15108
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CAMERON

July 3, 2012
CAW 12-08

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-819 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at McGuire Unit 2 Using the LEFM CheckPlus System"
- Caldon® Ultrasonics Engineering Report ER-822 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at McGuire Unit 1 Using the LEFM CheckPlus System"
- Caldon® Ultrasonics Engineering Report ER-972 Rev. 2 "Traceability Between Topical Report (ER-157P-A Rev. 8 & Rev. 8 Errata) and the System Uncertainty Report"

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 12-08 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 12-08 and should be addressed to the undersigned.

Very truly yours,

Ernest M. Hauser
Director of Sales

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

July 3, 2012
CAW 12-08

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Ernest M. Hauser, 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.

Ernest M. Hauser
Director of Sales

Sworn to and subscribed before me

this 3rd day of

July, 2012

Joann B. Thomas
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joann B. Thomas, Notary Public
Findlay Twp., Allegheny County
My Commission Expires July 28, 2015
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

1. I am the Director of Sales 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 10 CFR 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 10 CFR §§ 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 are the submittals titled:

- Caldon® Ultrasonics Engineering Report ER-819 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at McGuire Unit 2 Using the LEFM CheckPlus System"
- Caldon® Ultrasonics Engineering Report ER-822 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at McGuire Unit 1 Using the LEFM CheckPlus System"
- Caldon® Ultrasonics Engineering Report ER-972 Rev. 2 "Traceability Between Topical Report (ER-157P-A Rev. 8 & Rev. 8 Errata) and the System Uncertainty Report"

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further 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 the LEFM CheckPlus Systems used by McGuire Unit 1 and McGuire Unit 2 for MUR UPRATES.

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION BY THE OFFICE OF NUCLEAR REGULATION REGARDING A MCGUIRE LICENSE AMENDMENT TO SUPPORT A MEASUREMENT UNCERTAINTY RECAPTURE (MUR) POWER UPRATE

Enclosure 6

List of Commitments

Commitment	Commitment Date
After the LEFM Checkplus system is installed and operational on the respective Unit, six months of data will be collected comparing the LEFM Checkplus operating data to the Venturi data to verify consistency between thermal power calculation based on LEFM and other plant parameters.	The data will be available for NRC inspection seven months after the LEFM Checkplus system is installed and operational on the respective Unit.