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June 21, 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 May 22, 2012 NRC request for additional information (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 questions 5 thru 19 and Duke Energy's responses are provided in Enclosure 1 and Enclosure 2. Enclosure 2, which contains Westinghouse document DPC-12-58, is non-proprietary. Note that responses to MNS MUR LAR RAI questions 1 thru 4 were provided to the NRC via correspondence dated May 29, 2012.

The conclusions reached in the original determination that the 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.

This submittal contains no regulatory commitments.

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

A001
NRR

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: June 21, 2012
Date

Notary Public

My commission expires: July 1, 2012
Date

Enclosure 1

McGuire Nuclear Station's
Response to Request for Additional Information

NRC Question 5

In the LAR, Enclosure 2, "Summary of [Regulatory Information Summary] RIS 2002-03 Requested Information," Section VI1.6.A, "Fire Protection Program," states that " ... Additional building heat-up will be minimal such that currently credited fire protection manual actions will not be prevented from being accomplished by their required time ..." The NRC staff requests the licensee to verify that implementation of the LAR will not require any new operator actions. If implementation of the LAR requires any new operator actions, then please describe them. (AFPB 1)

McGuire Response to Question 5

Implementation of the McGuire MUR LAR will not require any new fire protection operator manual actions.

NRC Question 6

In the LAR, Enclosure 2, Section 47, "Safe Shutdown Fire," states that, " ... For specific site fire area, the Standby Shutdown Facility is the assured method to achieve and maintain the unit in a stable hot shutdown condition. While the plant is in the hot standby mode, damage control measures can be taken, as necessary, to restore the capability to achieve cold shutdown .... " The NRC staff requests that the licensee describe how McGuire 1 and 2 meet the 72-hour requirements contained in 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L, with increased decay heat at MUR power uprate conditions. (AFPB 2).

McGuire Response to Question 6

The McGuire Fire Protection Program is discussed in the Updated Final Safety Analysis Report (UFSAR) Section 9.5.1. As described in UFSAR Section 9.5.1.3, the Standby Shutdown System (SSS) is the assured method to secure and maintain the unit in a stable hot standby condition for certain Fire Areas and, while the plant is in the hot standby mode of operation, damage control measures can be taken as necessary to restore the capability to achieve cold

shutdown. As stated in Supplement 6 of the McGuire Safety Evaluation Report (NUREG-0422), the NRC concluded that McGuire met the requirements of 10 CFR 50, Appendix R, Paragraphs III.G and III.L.

A review of the impact of the MUR power uprate on the design of the SSS identified the following three potential impacts due to the small (< 2%) increase in decay heat:

- As part of the SSS function, the Main Steam safety valves are credited to release steam from the Steam Generators to maintain hot shutdown conditions. The small increase in decay heat at MUR power uprate conditions would result in a slightly higher frequency of Main Steam safety valve cycling.

- Upon commencement of the unit cooldown, the operator throttles the credited Main Steam Power Operated Relief Valves. The small increase in decay heat at MUR power uprate conditions would result in an incremental increase in valve opening position.

- Auxiliary feedwater inventory is initially from a condensate grade source of water with approximately 16 to18 hours capacity. However, the regulatory required auxiliary feedwater source is from the plant main condenser intake and discharge embedded piping for McGuire Unit 2 and Unit 1, respectively. Each unit's respective embedded piping has a nominal 3 to 3-1/2 days supply of cooling water which is readily replenished by gravity flow from Lake Norman. This is a sufficient supply of cooling water to accommodate the small increase in decay heat at MUR power uprate conditions and ensure that McGuire continues to meet the 72-hour requirements contained in 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L.

The above impacts due to the small increase in decay heat at MUR power uprate conditions would not have a material effect on the ability of McGuire 1 and 2 to implement damage control measures in hot standby as needed to achieve cold shutdown. These small impacts would not invalidate the NUREG-0422, Supplement 6 conclusions that McGuire met the requirements of 10 CFR 50, Appendix R, Paragraphs III.G and III.L, including the 72-hour requirements.

<u>NRC Question 7</u>

Some plants credit aspects of their fire protection system for other than fire protection activities (e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems). If McGuire 1 and 2 credit its fire protection system in this way, the LAR should identify the specific situations and discuss to what extent, if any, the LAR affects

these "non-fire-protection" aspects of the plant fire protection system. If McGuire 1 and 2 do not take such credit, then the NRC staff requests that the licensee verify this as well. In your response discuss how any non-fire suppression use of fire protection water will impact the ability to meet the fire protection system design demands. (AFPB 3)

McGuire Response to Question 7

The McGuire Fire Suppression System is described in UFSAR Section 9.5.1.2. The system provides fire protection throughout the plant and exterior yard areas through a system of pumps, hydrants, sprinklers, water spray, and hose stations. The suction source for the three full capacity motor driven fire pumps is Lake Norman which provides an essentially unlimited source of water.

Current procedures use the McGuire Fire Suppression System for two non-fire-protection beyond design basis events. For a loss of feedwater/auxiliary feedwater event, the system may supply up to 300 gallons per minute to Steam Generators as a secondary side heat sink. The loss of feedwater/auxiliary feedwater event assumes an initial power level of 102% rated thermal power (RTP) so the MUR power uprate will not impact this non-fire-protection aspect of the McGuire Fire Suppression System. The Fire Suppression System may also be used to supply backup make-up water to the Spent Fuel Pool for a loss of Spent Fuel Pool level event. As discussed in UFSAR Section 9.1.3.1.1, the spent fuel pool heat load is limited by procedure prior to core offload and therefore the MUR power uprate will not impact this non-fire-protection aspect of the McGuire Fire Suppression System.

As discussed above, the MUR power uprate will not impact the non-fire-protection aspects of the McGuire Fire Suppression System. Therefore, the non-fire-protection use of fire suppression water under MUR power uprate conditions will not impact the ability to meet the McGuire Fire Suppression System design demands.

NRC Question 8

The LAR takes credit in several instances for administrative controls of the McGuire 1 and 2 "design change process," or "engineering change process." The NRC staff requests that the specific numbers and titles of controlling procedures be provided, along with a reference to the sections that are relevant to identifying impacted procedures, controls, displays, alarms, the Safety Parameter Display System, and other operator interfaces, the simulator, and training, so that the NRC staff can conclude its review." (AHFB 1)

McGuire Response to Question 8

Engineering Directives Manual (EDM) 601: Engineering Change Manual, Section 601.6.3, "Reviews of Training, Procedures, and Other Programs for Design, Equivalent and Editorial Document-Only Changes", along with Appendix K, "Engineering Review Screen for Design Changes", provides the administrative controls relevant to identifying impacted procedures, controls, displays, alarms, the Operator Aid Computer (which includes the Safety Parameter Display System), and other operator interfaces, the simulator, and training.

NRC Question 9

Section 3.6.1.2 of the McGuire 1 and 2 updated final safety analysis report (UFSAR) stipulates the criteria used to define moderate energy lines at McGuire 1 and 2. Additionally, Section 3.6.2.2.1 outlines the criteria used to postulate through-wall cracks in moderate energy piping systems. However, the information contained in LAR, Enclosure 2, Section IV, does not include an assessment of the impact of implementation of the LAR on moderate energy piping or the postulation of moderate energy line cracks. Based on the McGuire 1 and 2 UFSAR criteria related to moderate energy piping, state the effects of the proposed MUR power uprate on moderate energy piping, including whether the LAR results in the required postulation of additional moderate energy line cracks. (EMCB 1)

McGuire Response to Question 9

UFSAR Section 3.6.1.2 identifies the criteria for moderate energy piping at McGuire. UFSAR Table 3-19 identifies the McGuire moderate energy systems or portions of systems that meet the criteria in UFSAR Section 3.6.1.2. Operation of these systems does not change as a result of the power uprate. Since the temperature and pressure conditions for the moderate energy systems identified in UFSAR Table 3-19 do not change under MUR power uprate conditions, there is no effect on McGuire's moderate energy piping and the LAR will not result in the postulation of additional moderate energy line cracks. Furthermore, local environmental effects including flooding, humidity, compartment pressurizations, temperature, etc. due to system pressure and/or temperature changes at the existing postulated crack locations are unchanged from those previously evaluated.

NRC Question 10

Section IV.1.D of RIS 2002-03 stipulates that the content of MUR power uprate LARs should include the codes of record used in the evaluations of structures, systems and components (SSCs) to determine their structural adequacy at MUR conditions. However, in the LAR, Enclosure 2, Table IV.1.D-1, it does not contain the code of record used in assessing the reactor vessel internals (RVIs) or balance-of-plant (BOP) and interfacing piping systems for structural adequacy at the uprated conditions. Please state the design basis code of record for the RVIs, including core support and non-core support structures, BOP piping and interfacing piping systems and confirm that the evaluations performed in support of the LAR were performed consistent with the provisions stipulated by the design basis codes of record for those SSCs identified above. (EMCB 2)

McGuire Response to Question 10

As stated in UFSAR, Section 4.2.2.4, the January 1971 draft of the ASME Code for Core Support Structures, Subsection NG, is the design basis code of record and the allowable stress limits used in the design of the Reactor Internals are based on that code. Evaluations performed in support of the LAR were performed consistent with the applicable provisions stipulated by the design basis codes of record for the SSCs identified above.

Per the McGuire Specification for the Procurement of Power Piping Systems Materials and Components, the design basis code of record for the BOP piping and interfacing piping systems evaluated for the MUR is ASME 1971 Edition including Summer 1971 and Winter 1971 Addenda. Exceptions to the design basis code are as follows: 1974 Edition including Summer 1976 Addenda is the base code for ASME Section III Paragraphs N(B)-4250, N(C)-4250, and N(D)-4250, 1974 Edition including Winter 1976 Addenda is the base code for ASME Section III Paragraph ND-4130, 1983 Edition including Winter 1983 Addenda is the base code for ASME Section III Paragraph NC-4122, and the Nuclear Service Water System conforms to ANSI B31.7 – Nuclear Power Piping, Class III, 1969 Edition. The operating parameters experienced by the BOP and interfacing piping were evaluated consistent with the applicable provisions stipulated by the design basis codes of record for the BOP piping and interfacing piping systems.

NRC Question 11

In the LAR, Enclosure 2, Section IV.1.A.ii, describes the impact of the proposed LAR on the reactor core support structures and vessel internals (i.e., the RVIs). The discussion in this

section of the LAR indicates that the revised core parameters at the proposed MUR power level are bounded by the values used in the analyses of record (AOR) for the RVIs when the analytical uncertainties in the AOR are considered. Confirm that the AOR related to the structural evaluation of the RVIs, including all core support and non-core support structures, did not require a revision to support implementation of the LAR. Additionally, quantify the uncertainty relied upon in the AOR to demonstrate that the calculation tolerance available in the AOR sufficiently bounds the core parameters proposed in the LAR. (EMCB 3)

McGuire Response to Question 11

The AOR related to the structural evaluation of the RVIs, including all core support and non-core support structures, did not require a revision to support implementation of the LAR. The uncertainty relied upon in the AOR to demonstrate that the calculation tolerance available in the AOR sufficiently bounds the core parameters proposed in the LAR is based on the core power level. The Reactor Internals were originally designed to support a core power level of 3479 MWt with a licensed core power level of 3411 MWt, which allows for approximately 2% uncertainty in the core power level. The evaluations performed in support of the MUR uprate consider a power level of 3469 MWt with the design core power level remaining at 3479 MWt, providing for approximately 0.3% uncertainty in core power level.

NRC Question 12

In the LAR, Enclosure 2, Section IV.1.A.v, details the evaluation of the BOP and interfacing piping systems to determine whether these systems will maintain adequate structural integrity at the MUR power level. Please state whether the AOR for any of these systems required revision to support the LAR (i.e., identify systems which were not bounded by the current AOR). For those systems not bounded by the current AOR, identify whether the revision to the AOR was necessitated based on an increase in temperature, pressure, flow rate, or a combination of any of the three operating parameters. (EMCB 4)

McGuire Response to Question 12

The structural integrity of BOP and interfacing piping systems listed in Enclosure 2, Section IV.1.A.iv of the LAR remain bounded by the current analysis of record at the MUR conditions. The MUR does not result in an increase in operating parameters for these systems above current parameters within the analysis of record. Therefore, no revision of the analysis of record for these BOP and interfacing piping systems is required to support the LAR.

NRC Question 13

In the LAR, Enclosure 2, Section IV.1.A.ix, discusses the review performed to evaluate the effects of the LAR on safety-related valves. The discussion in this section of the LAR focuses on the effects of MUR implementation on the pressurizer overpressure protection valves and other safety-related valves. Please address the following issues related to the safety-related valve evaluation:

a) With respect to the pressurizer overpressure protection valves, the LAR states that the current design basis event analysis is bounding and, as such, there is no adverse impact on the subject valves. Please state whether the review performed for the subject valves considered all MUR operating conditions, including normal operating conditions and abnormal operating conditions. Additionally, confirm whether the AOR for the subject valves remains bounding at MUR conditions. If the analyses are not bounded, confirm that the design basis criteria used to structurally qualify the valves remain satisfied at the proposed MUR power level.

b) For the other safety-related valves reviewed in support of MUR implementation, the information in the LAR did not identify which valves were included in this review or whether these valves were found to be structurally adequate at the proposed MUR power level, as evidenced by satisfying their respective design basis criteria. State whether any of the other safety-related valves reviewed to support MUR implementation were found to be unbounded by their current AOR. If the AOR for these valves was found to be not bounding, confirm that the respective design basis acceptance criteria remain satisfied.

(EMCB 5)

McGuire Response to Question 13

a) The review of normal and abnormal transients and the impact of these transients on the pressurizer overpressure protection valves is discussed in Sections II and IV.1.A.ix, respectively. This review, which considered all MUR operating conditions including abnormal and normal operating conditions, determined that the MUR power uprate had no adverse impact on the pressurizer overpressure protection valves. Based on this review, it was determined that the analysis of record for the pressurizer overpressure protection valves remains bounding at all MUR conditions.

b) The review of normal and abnormal transients for each system is discussed in Section II of the LAR. The impact of these transients on the safety related valves is considered as part of the review of each system, as discussed in Section IV.1.A.ix. Based on this review, it was determined that the analysis of record for the other safety related valves remains bounding at MUR conditions.

NRC Question 14

In the LAR, Enclosure 2, Section IV.1.B.viii, "Jet impingement and thrust forces," only discusses the impact of the LAR on the McGuire 1 and 2 leak-before-break evaluations and does not mention jet impingement nor thrust forces. In Section IV.1.B.vii it states that " ... there is no impact on the [high energy line break] HELB analysis that was originally performed for McGuire Units 1 and 2 ...... and describes the current AOR regarding HELBs as bounding. Please confirm that the statement in Section IV.1.B.vii applies to jet impingement and thrust forces (or other dynamic effects loadings) resulting from postulated HELBs, such that a conclusion can be made regarding whether the dynamic effects resulting from postulated HELBs remain valid at the proposed MUR power level. (EMCB 6)

McGuire Response to Question 14

The statement “there is no impact on the HELB analysis that was originally performed for McGuire Units 1 and 2” and “The MUR is bounded by the existing analysis of record for the plant” within Section IV.1.B.vii applies to jet impingement and blowdown forces resulting from postulated HELBs. Since system parameter (i.e. temperature and pressure) changes as a result of the MUR are insignificant and have no effects on previously evaluated pipe rupture loads, jet impingement loads, compartment pressurizations, and environmental conditions, blowdown forces and resultant jet impingement/pipe whip evaluations will not be impacted by the MUR.

NRC Question 15

With regards to the structural evaluations and analyses performed to support the LAR, please confirm that all analyses and evaluations for SSCs which were within the scope of the McGuire 1 and 2 license renewal efforts were done in accordance, and consistent, with the methodologies approved and referenced in NUREG-1772, "Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2." Otherwise, please state the changes for all structural evaluations and analyses

performed to support the proposed MUR conditions that were not performed in accordance with NUREG-1772. Justification should also be provided with regards to the acceptability of these changes. (EMCB 7)

McGuire Response to Question 15

All analyses and evaluations for SSCs performed to support the LAR which were within the scope of the McGuire 1 and 2 license renewal efforts were done in accordance, and consistent, with the methodologies approved and referenced in NUREG-1772.

NRC Question 16

For McGuire 1, the limiting reference temperature for pressurized thermal shock (RTPTS) reported in the LAR, Enclosure 2, Section IV.1.C.i, is 203 °F for the lower shell longitudinal welds 3-442A, B, and C. The NRC staff has the following questions:

- The NRC staff found that the limiting RTPTS value for McGuire 1 in this LAR differs from that reported the licensee's license renewal LAR (LRLAR). NUREG-1772 states "Using a limiting fluence of 2.73 x 10^{19} n/cm2 at EOLE [end of life extended], the applicant's revised PTS assessment projected the RTPTS values for these welds [the McGuire 1 limiting weld] to be 253 °F using all relevant surveillance capsule data for the heat No. 21935/12008, as obtained from docketed information from the Diablo Canyon [Power Plant], [Unit] 2 [Diablo 2] RV [reactor vessel] material surveillance program (inclusive of fracture toughness tests performed on test specimens from Diablo 2 capsules U, X, Y, and V)." The NRC staff used the LAR fluence of 2.13 x 10^{19} n/cm2 to reduce the licensee's reported LRLAR RTPTS value from 253 °F to 238 °F, but this value is still far greater than the LAR limiting RTPTS value of 203 °F. Please provide your calculations of the chemistry factors and RTPTS values from the LRLAR (253 °F) and the LAR (203 °F) and resolve this apparent discrepancy. If the resolution results in a revision of the LAR limiting RTPTS value, then Tables IV.1.C-5, IV.1.C-6, and IV.1.C-9, related to heatup and cooldown curves, need to be revised also because the LAR limiting adjusted reference temperatures (ARTs) will be changed accordingly.
- Unlike the LRLAR, where the fluence values were different for intermediate shell plate longitudinal welds 2-442A, 2-442B, and 2-442C, the LAR fluence values reported for these welds are identical. Please confirm that the peak fluence value for the intermediate shell plate longitudinal welds 2-442A, 2-442B, and 2-442C has been used for all three

welds to simplify the classification of these welds in the LAR. Please also confirm that the similar simplification has been applied to the lower shell plate longitudinal welds 3-442A, 3-442B, and 3-442C in the LAR.

(EVIB 1)

McGuire Response to Question 16

This response is provided in Westinghouse document DPC-12-58, Revision 1, which is in Enclosure 2.

NRC Question 17

For the upper-shelf energy (USE) evaluation, the LAR, Enclosure 2, Section IV.1.C.v, states that the projected EOLE Charpy USE values due to the MUR power uprate fluence at the one quarter RV wall thickness (¼ T) location were 60.5 ft-lbs for the intermediate shell longitudinal welds 2-442A, 2-442B, and 2-442C, using surveillance data, and 61.8 ft-lbs for the bottom head ring 03 for McGuire 2. The limiting USE values reported in the LRLAR are 53 ft-lbs for the nozzle shell plate B5011-2 for McGuire 1 and 55 ft-lbs for nozzle shell to intermediate shell weld for McGuire 2. These apparent discrepancies involve a change of limiting materials for both McGuire units. Please explain the following:

- The nozzle shell plate B5011-2 is no longer the limiting USE material for McGuire 1 in the LAR because its fluence at the ¼ T location has been revised from 1.83×10^{19} n/ cm^2 (LRLAR) to 0.033×10^{19} n/ cm^2 (LAR).

- The fluence at the ¼ T location for the intermediate shell longitudinal welds 2-442A, 2-442B, and 2-442C of McGuire 1 decreased from 1.63×10^{19} n/cm^2 (LRLAR) to 1.269×10^{19} n/cm^2 (LAR) indicating that higher USE value should be expected for the LAR. Instead, the USE value decreased from 72 ft-lbs (LRLA) to 60.5 ft-lbs. This appears to be caused by considering additional surveillance data. Please plot all surveillance data in Figure 2 of RG 1.99, Revision 2, to support the USE decrease that was used in estimating the LAR USE value for this material.

- The nozzle shell to intermediate shell weld is no longer the limiting USE material for McGuire 2 in the LAR because its fluence at the ¼ location has been revised from 1.73×10^{19} n/cm^2 (LRLAR) to 0.043×10^{19} n/cm^2 (LAR). Please justify this significant change. Further, the initial USE of ">71" ft-lbs in Table IV.1.C-12 indicates that the value

was not determined based on data from the Certified Materials Test Report (CMTR). Please explain how the initial USE of 71 ft-lb was determined for the nozzle shell to intermediate shell weld of McGuire 2 and justify use of this initial USE for this weld. If this 71 ft-lbs value was determined statistically, you must revise this value based on the "mean minus two-sigma" approach.

- The limiting USE material for McGuire 2 in the LAR is the bottom head ring 03 with an initial USE of ">71" ft-lbs in Table IV.1.C-12. Please explain how the initial USE of > 71 ft-lb was determined for the bottom head ring 03 of McGuire 2 and justify use of this initial USE for this ring. If this 71 ft-lbs value was determined statistically, you must revise this value based on the "mean minus two-sigma" approach.

(EVIB 2)

McGuire Response to Question 17

This response is provided in Westinghouse document DPC-12-58, Revision 1, which is in Enclosure 2.

NRC Question 18

For reactor core support structures and vessel internals, the Materials Reliability Program (MRP) Report 1022863 (MRP-227-A), "Pressurized Water Reactor [PWR] Internals Inspection and Evaluation Guidelines [I&E]," dated December 2011 (ADAMS Accession No. ML 12017A194) was issued recently, which contains the NRC staff safety evaluation (SE) for this report. MRP-227-A provides recommended I&E guidelines, as modified by the NRC staff, for PWR RV internals as a result of the industry effort on this issue for the past few years. The LAR, Enclosure 2, Section IV.1.A.ii, concluded that, "there is no impact, adverse or otherwise, from the McGuire Units 1 and 2 MUR uprate on the plant-specific implementation of the MRP-227 requirements." However, Section IV.1.A.ii did not mention submitting of the plant-specific inspection plan in accordance with MRP-227-A to demonstrate that the degradation of the RV internals will be managed appropriately after the LAR. Please confirm that you plan to submit the plant-specific inspection program consistent with the MRP-227-A report guidelines, and indicate the approximate date of submittal to support the LAR effectively. (EVIB 3)

McGuire Response to Question 18

A letter of intent to adopt MRP-227, as approved by the NRC for Oconee, McGuire, and Catawba Nuclear Stations was sent to the NRC on June 6, 2010. McGuire does not have a

license renewal commitment to submit an inspection plan/AMP, nor is there a requirement in MRP-227-A to do so. However, Duke Energy plans to submit an applicant's/licensee's application to implement MRP-227, as defined in Applicant/Licensee Action Item 8 contained in Revision 1 of the Safety Evaluation of MRP-227, Revision 0, approximately two years before the period of extended operation. The expiration of the initial license for McGuire Unit 1 and Unit 2 are June 12, 2021 and March 3, 2023 respectively. The approximate dates for submittal of McGuire Unit 1 and Unit 2 applicant's/licensee's application to implement MRP-227 containing an inspection plan are June 12, 2019 and March 3, 2021 respectively. The submittal of these applications has been entered into the McGuire corrective action program.

NRC Question 19

What is the impact of LAR on the air operated valve program for McGuire 1 and 2? (EPTB 1)

McGuire Response to Question 19

The Air Operated Valve (AOV) Program for McGuire Units 1 and 2 is not impacted by the MUR. The AOV Program consists of all active AOVs for McGuire Units 1 and 2. These valves are divided into two categories depending on determination of a valve's safety-significance. The systems that contain these AOVs were evaluated and determined to continue to be within design after implementation of the MUR. The required operating thrust/torque and actuator output capability for AOVs are determined based on worst case operating conditions within the licensing basis of the plant. These worst case conditions, which the AOVs are required to operate, remain unchanged due to the MUR. The MUR does not alter the basis, scope, or content of the AOV Program. No AOVs will be added or deleted from the program due to the MUR.

Enclosure 2

Westinghouse Document DPC-12-58, Revision 1

Response to RAI Questions 16 and 17

Westinghouse

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Our ref: DPC-12-58 Rev. 1

June 11, 2012

DUKE ENERGY
Transmittal of McGuire Units 1 and 2 Measurement Uncertainty Recapture Power Uprate Requests for Additional Information: Reactor Vessel Integrity Recommended Responses

Dear Mr. Bradley:

Attached for your information and use is Westinghouse letter MCOE-LTR-12-7, Rev. 0, "McGuire Units 1 and 2 Measurement Uncertainty Recapture Power Uprate Requests for Additional Information: Reactor Vessel Integrity Recommended Responses." This letter transmits our Westinghouse recommended responses per Reference 1 identified below.

If you have any questions or require additional information, please contact John Victor at 412-374-2532 or Michael Ryan at 412-374-4943. Thank you.

Very truly yours,

(Michael Ryan for)
Daniel C. Beddingfield
Customer Project Manager

Reference:
1. MUR LAR RAIs 05222012.pdf, "NRC RAI Letter for McGuire 1 and 2", May 22, 2012.

cc: Rick Nance, Duke Energy
David Whitaker, Duke Energy
P. R. Harden, Westinghouse
M. W. Ryan, Westinghouse
J. F. Victor, Westinghouse
A. E. Freed, Westinghouse
E. J. Long, Westinghouse
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Westinghouse

To: John F. Victor
cc: Amy E. Freed
Frank C. Gift

Date: June 7, 2012

From: Elliot J. Long
Ext.: 412-374-2728
Fax: 724-940-8565

Your ref.: N/A
Our ref.: MCOE-LTR-12-7, Revision 0

Subject: **McGuire Units 1 and 2 Measurement Uncertainty Recapture Power Uprate Requests for Additional Information: Reactor Vessel Integrity Recommended Responses**

Reference:

1. U.S. Nuclear Regulatory Commission Letter, "McGuire Nuclear Station, Units 1 and 2, Request for Additional Information Regarding License Amendment Related to Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME8213 and ME8214)," dated May 22, 2012.

As part of the McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) uprate project, Requests for Additional Information (RAIs) (Reference 1) were provided by the Nuclear Regulatory Commission (NRC). Attachment A to this letter contains recommended responses to the following RAIs pertaining to Reactor Vessel Integrity (RVI):

RAI #s 16 and 17

Please transmit this letter to Terry Bradley and David Whitaker (Duke Energy).

Do not hesitate to contact the undersigned with any questions regarding the contents of this letter.

ELECTRONICALLY APPROVED[1]
Elliot J. Long
Materials Center of Excellence – I

Verified by: **ELECTRONICALLY APPROVED**[1]
Amy E. Freed
Materials Center of Excellence – I

Approved: **ELECTRONICALLY APPROVED**[1]
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[1]***Electronically approved records are authenticated in the electronic document management system.***

ATTACHMENT A

McGuire Units 1 and 2 Measurement Uncertainty Recapture Power Uprate Requests for Additional Information: Reactor Vessel Integrity Recommended Responses

RAI # 16:

For McGuire 1, the limiting reference temperature for pressurized thermal shock (RTPTS) reported in the LAR, Enclosure 2, Section IV.1.C.i, is 203°F for the lower shell longitudinal welds 3-442A, B, and C. The NRC staff has the following questions:

- *The NRC staff found that the limiting RTPTS value for McGuire 1 in this LAR differs from that reported the licensee's license renewal LAR (LRLAR). NUREG-1772 states "Using a limiting fluence of 2.73 x 10^{19} n/cm^2 at EOLE [end of life extended], the applicant's revised PTS assessment projected the RTPTS values for these welds [the McGuire 1 limiting weld] to be 253°F using all relevant surveillance capsule data for the heat No. 21935/12008, as obtained from docketed information from the Diablo Canyon [Power Plant], [Unit] 2 [Diablo 2] RV [reactor vessel] material surveillance program (inclusive of fracture toughness tests performed on test specimens from Diablo 2 capsules U, X, Y, and V)." The NRC staff used the LAR fluence of 2.13 x 10^{19} n/cm^2 to reduce the licensee's reported LRLAR RTPTS value from 253°F to 238°F, but this value is still far greater than the LAR limiting RTPTS value of 203°F. Please provide your calculations of the chemistry factors and RTPTS values from the LRLAR (253°F) and the LAR (203°F) and resolve this apparent discrepancy. If the resolution results in a revision of the LAR limiting RTPTS value, then Tables IV.1.C-5, IV.1.C-6, and IV.1.C-9, related to heatup and cooldown curves, need to be revised also because the LAR limiting adjusted reference temperatures (ARTs) will be changed accordingly.*

Response:

The Diablo Canyon 2 surveillance capsule data was reevaluated in 2011. This work was documented in WCAP-17315-NP (Reference 1). This document was referenced in the "Annual Update to the DCPP License Renewal Application and License Renewal Application Amendment Number 45" (Reference 2) as a new License Renewal Application (LRA) reference for Diablo Canyon, Reference 39. WCAP-17315-NP updated the surveillance capsule fluence values at the clad/base metal interface in accordance with WCAP-14040-A, Revision 4 (Reference 3). This assessment was performed based on the guidance specified in Regulatory Guide 1.190 (Reference 4). The previous analysis of record for Diablo Canyon 2 (WCAP-15423, Reference 5) used an older methodology for the fluence assessment.

In addition, the Diablo Canyon 2 credibility assessment was updated in WCAP-17315-NP. Tandem weld heat No. 21935/12008 was deemed credible in Appendix A of the report. McGuire 1 took credit for the credible surveillance data and used a reduced margin term (28°F) for the RT_{PTS} calculations of this weld material. The previous RT_{PTS} assessment for McGuire 1 (Reference 6) utilized a full margin term for this weld material.

Lastly, the chemistry factor calculation in the LRLAR evaluations did not take into consideration the temperature difference between the McGuire 1 reactor vessel and the Diablo Canyon 2 surveillance capsules. This temperature adjustment was included in the chemistry factor calculations for the MUR LAR. Incorporation of the temperature adjustment reduced the chemistry factor value of the McGuire 1 vessel weld

because the surveillance capsules in Diablo Canyon 2 were irradiated at a lower temperature than the McGuire 1 reactor vessel.

Therefore, in summary, all Diablo Canyon 2 surveillance data for tandem weld heat No. 21935/12008 was utilized in the MUR LAR RT_{PTS} calculations for McGuire 1. The resulting reduction in the calculated RT_{PTS} value for this weld material is due to the updated fluence values, which impacted the chemistry factor for this weld, the inclusion of the temperature adjustment, which also impacted the chemistry factor for this weld, and the credibility conclusion, which impacted the margin term used in the RT_{PTS} calculation. Specifically, the updated fluence values and temperature adjustment inclusion reduced the chemistry factor from 194.4°F to 186.4°F and the updated credibility conclusion reduced the margin term from 56°F to 28°F. See Tables 1 through 4 below for more details.

Table 1 Calculation of McGuire Unit 1 Weld Heat # 21935/12008 Chemistry Factor Using Surveillance Capsule Data from Diablo Canyon Unit 2 for the MUR LAR

Material	Capsule	Capsule f[a] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Diablo Canyon Unit 2 Surveillance Weld (Heat # 21935/12008)	U	0.330	0.695	161.37 (173.0)	112.16	0.483
	X	0.906	0.972	191.27 (203.2)	185.97	0.945
	Y	1.53	1.118	199.39 (211.4)	222.84	1.249
	V	2.38	1.234	212.36 (224.5)	262.01	1.522
				SUM:	782.99	4.200
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (782.99) \div (4.200) =$ **186.4°F**					

Notes:

(a) f = fluence.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. All measured (pre-adjusted) values are listed in parentheses and are taken from Reference 1. The Diablo Canyon 2 ΔRT_{NDT} values are adjusted first by the difference in operating temperature then using the ratio procedure to account for differences in the surveillance weld chemistry and the beltline weld chemistry (pre-adjusted values are listed in parentheses). The temperature adjustment is -10°F (Temperature adjustment = 1.0*($T_{capsule}$ - T_{plant}), where T_{plant} = 555°F for McGuire 1 and $T_{capsule}$ = 545°F for Diablo Canyon 2). Ratio = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 208.2°F / 211.2°F = 0.99.

Table 2 Calculation of McGuire Unit 1 Weld Heat # 21935/12008 Chemistry Factor Using Surveillance Capsule Data from Diablo Canyon Unit 2 for the LRLAR

Material	Capsule	Capsule f[(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Diablo Canyon Unit 2 Surveillance Weld (Heat # 21935/12008)	U	0.338	0.701	170.57	119.57	0.491
	X	0.919	0.976	200.38	195.57	0.953
	Y	1.55	1.121	208.43	233.65	1.257
	V	2.41	1.237	221.33	273.79	1.530
				SUM:	822.58	4.231
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (822.58) \div (4.231) =$ **194.4°F**					

Notes:

(a) f = fluence.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. The Diablo Canyon 2 ΔRT_{NDT} values were only adjusted using the ratio procedure to account for differences in the surveillance weld chemistry and the beltline weld chemistry. Ratio = 0.986. No temperature adjustment used for conservatism.

Table 3 RT_{PTS} Calculations for the McGuire Unit 1 Reactor Vessel Materials at 54 EFPY for the MUR LAR[(a)]

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF (°F)	Fluence ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	IRT_{NDT} (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[(b)] (°F)	Margin (°F)	RT_{PTS} (°F)
Lower Shell Longitudinal Welds 3-442A,B,C (Heat # 21935/12008)	1.1	208.2	2.13	1.2055	-50	251.0	0	28	56	257
Using credible Diablo Canyon Unit 2 surveillance data	2.1	186.4	2.13	1.2055	-50	224.7	0	14	28	**203**

Notes:

(a) The 10 CFR 50.61 methodology was utilized in the calculation of the RT_{PTS} values.

(b) Per Appendix A.2 of WCAP-17315-NP (Reference 1), the Diablo Canyon Unit 2 weld surveillance data was deemed credible. Per the guidance of 10 CFR 50.61, the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Table 4 RT_{PTS} Calculations for the McGuire Unit 1 Reactor Vessel Materials at 54 EFPY for the LRLAR[(a)]

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF (°F)	Fluence ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	IRT_{NDT} (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[(b)] (°F)	Margin (°F)	RT_{PTS} (°F)
Lower Shell Longitudinal Welds 3-442A,C (Heat # 21935/12008)[(c)]	1.1	208.2	2.73	1.27	-50	264.4	0	28	56	270
Using non-credible Diablo Canyon Unit 2 surveillance data	2.1	194.4	2.73	1.27	-50	246.9	0	28	56	**253**

Notes:

(a) The 10 CFR 50.61 methodology was utilized in the calculation of the RT_{PTS} values.

(b) Based on the credibility evaluation contained in Appendix D of WCAP-15423 (Reference 5), the Diablo Canyon Unit 2 weld surveillance data was deemed non-credible for use in the McGuire 1 PTS calculations. Per the guidance of 10 CFR 50.61, the weld metal σ_Δ = 28°F for Position 1.1 and for Position 2.1 using non-credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

(c) The fluence value on lower shell longitudinal welds 3-442A and C bounds the fluence value on 3-442B.

- *Unlike the LRLAR, where the fluence values were different for intermediate shell plate longitudinal welds 2-442A, 2-442B, and 2-442C, the LAR fluence values reported for these welds are identical. Please confirm that the peak fluence value for the intermediate shell plate longitudinal welds 2-442A, 2-442B, and 2-442C has been used for all three welds to simplify the classification of these welds in the LAR. Please also confirm that the similar simplification has been applied to the lower shell plate longitudinal welds 3-442A, 3-442B, and 3-442C in the LAR.*

Response:

The fluence value reported for the intermediate shell plate longitudinal welds 2-442A, 2-442B, and 2-442C in this MUR LAR corresponded to the 30° azimuthal location. Welds 2-442B and 2-442C are located at the 30° azimuthal location; however, weld 2-442A is located at the 0° azimuthal location. The fluence value at this location is lower than the 30° location; therefore, to simplify the calculations, the conservative fluence value from the 30° location was utilized for all reactor vessel integrity calculations performed on this weld material. This simplification was also applied to the lower shell plate longitudinal welds 3-442A, 3-442B, and 3-442C as well as the nozzle shell longitudinal welds 1-442A, 1-442B, and 1-442C.

RAI # 17:

For the upper-shelf energy (USE) evaluation, the LAR, Enclosure 2, Section IV.1.C.v, states that the projected EOLE Charpy USE values due to the MUR power uprate fluence at the one quarter RV wall thickness (¼ T) location were 60.5 ft-lbs for the intermediate shell longitudinal welds 2-442A, 2-442B, and 2-442C, using surveillance data, and 61.8 ft-lbs for the bottom head ring 03 for McGuire 2. The limiting USE values reported in the LRLAR are 53 ft-lbs for the nozzle shell plate B5011-2 for McGuire 1 and 55 ft-lbs for nozzle shell to intermediate shell weld for McGuire 2. These apparent discrepancies involve a change of limiting materials for both McGuire units. Please explain the following:

- *The nozzle shell plate B5011-2 is no longer the limiting USE material for McGuire 1 in the LAR because its fluence at the ¼ T location has been revised from 1.83 x 10^{19} n/cm^2 (LRLAR) to 0.033 x 10^{19} n/cm^2 (LAR).*

Response:

The USE evaluations in the LRLAR utilized the peak vessel fluence for all of the three shell course plates (nozzle, intermediate and lower). This was a conservative approach for the nozzle shell plates, since they are located above the active core. In the MUR LAR, the material specific fluence value was used for the nozzle shell plates. The projected surface fluence on the nozzle shell plates is 0.0547 x 10^{19} n/cm^2, which resulted in the 1/4T fluence value of 0.033 x 10^{19} n/cm^2 at 54 EFPY.

- *The fluence at the ¼ T location for the intermediate shell longitudinal welds 2-442A, 2-442B, and 2-442C of McGuire 1 decreased from 1.63 x 10^{19} n/cm^2 (LRLAR) to 1.269 x 10^{19} n/cm^2 (LAR) indicating that higher USE values should be expected for the LAR. Instead, the USE value decreased from 72 ft-lbs (LRLAR) to 60.5 ft-lbs. This appears to be caused by considering additional surveillance data. Please plot all surveillance data in Figure 2 of RG 1.99, Revision 2, to support the USE decrease that was used in estimating the LAR USE value for this material.*

Response:

The USE calculation for the intermediate shell longitudinal welds 2-442A, 2-442B, and 2-442C of McGuire 1 considered all reactor vessel surveillance data. The USE data was plotted on Figure 2 of Regulatory Guide 1.99, Revision 2 (Reference 7) and an upper bound line was drawn parallel to the existing lines on the Figure. See Figure 1 below. This new line was used to determine the percent USE decrease values at the 1/4T fluence value of 1.269 x 10^{19} n/cm^2 for this weld material. The resulting USE percent decrease was higher (46%) than the values documented in the McGuire 1 LRLAR (36%), which resulted in the end of life extended (EOLE) USE dropping from 72 ft-lbs to 60.5 ft-lbs.

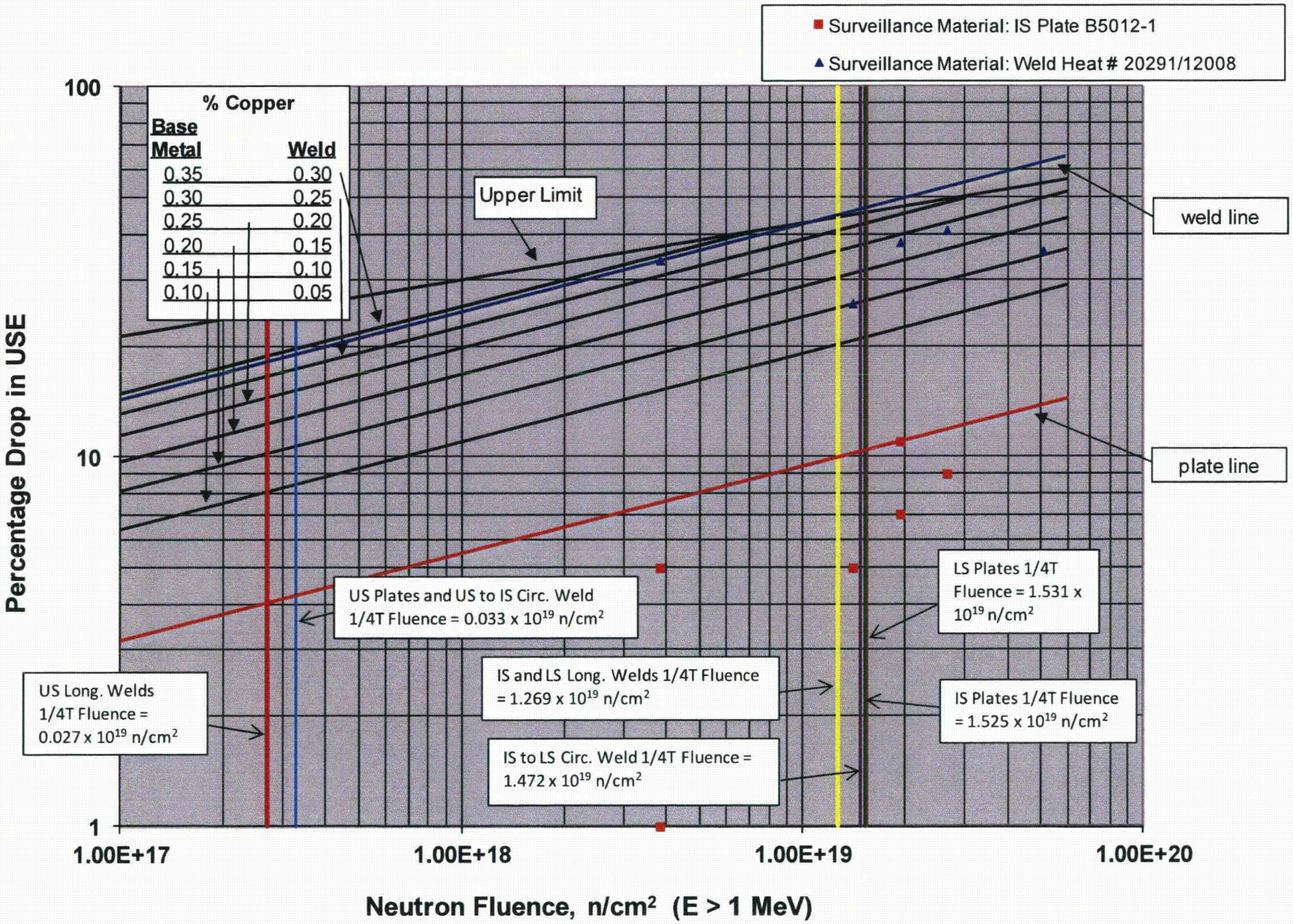

Figure 1

Regulatory Guide 1.99, Revision 2 Predicted Decrease in USE as a Function of Copper and Fluence for McGuire 1

- *The nozzle shell to intermediate shell weld is no longer the limiting USE material for McGuire 2 in the LAR because its fluence at the 1/4T location has been revised from 1.73 x 10^{19} n/cm^2 (LRLAR) to 0.043 x 10^{19} n/cm^2 (LAR). Please justify this significant change. Further, the initial USE of ">71" ft-lbs in Table IV.1.C-12 indicates that the value was not determined based on data from the Certified Materials Test Report (CMTR). Please explain how the initial USE of 71 ft-lb was determined for the nozzle shell to intermediate shell weld of McGuire 2 and justify use of this initial USE for this weld. If this 71 ft-lbs value was determined statistically, you must revise this value based on the "mean minus two-sigma" approach.*

Response:

The USE evaluations in the LRLAR utilized the peak vessel fluence for all of the reactor vessel materials. This was a conservative approach for the nozzle shell forging and nozzle to intermediate shell circumferential weld W06, since they are located above the active core. In the MUR LAR, the material specific fluence value was used for the nozzle to intermediate shell circumferential weld W06. The projected surface fluence on the nozzle to intermediate shell circumferential weld W06 is 0.0711 x 10^{19} n/cm^2, which resulted in the 1/4T fluence value of 0.043 x 10^{19} n/cm^2 at 54 EFPY.

Based on the Certified Material Test Report (CMTR), only three Charpy impact tests were performed for the nozzle shell to intermediate shell weld of McGuire 2. All three tests were performed at 10°F. The initial USE value of >71 ft-lb is based on the highest energy achieved of the three impact tests at 10°F. No other Charpy data is available for this weld material. Therefore, since the only test temperature was 10°F, the actual initial USE energy should be higher and assigning a minimum value of 71 ft-lb to this material is a conservative approach.

- *The limiting USE material for McGuire 2 in the LAR is the bottom head ring 03 with an initial USE of ">71" ft-lb in Table IV.1.C-12. Please explain how the initial USE of > 71 ft-lb was determined for the bottom head ring 03 of McGuire 2 and justify use of this initial USE for this ring. If this 71 ft-lbs value was determined statistically, you must revise this value based on the "mean minus two-sigma" approach.*

Response:

According to the CMTR, the initial USE value for the bottom head ring 03 is 109 ft-lbs. However, the CMTR indicates that the test direction was tangential (strong). Therefore, per Section B.1.1(3)(a) of NUREG-0800 Branch Technical Position 5-3 (Reference 8), the value should be reduced to 65% of the strong direction value to approximate the weak direction. This results in the reported initial USE value of 71 ft-lbs for this material.

References

1. WCAP-17315-NP, Revision 0, "Diablo Canyon Units 1 and 2 Pressurized Thermal Shock and Upper-Shelf Energy Evaluations," July 2011.
2. Pacific Gas & Electric Company Letter DCL-11-136, "10 CFR 54.21(b) Annual Update to the DCPP License Renewal Application and License Renewal Application Amendment Number 45," dated December 21, 2011.
3. WCAP-14040-A, Revision 4, "Methodology Used to Developed Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.
4. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.
5. WCAP-15423, Revision 0, "Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program," September 2000.
6. Duke Power Letter, "Comments on the U.S.N.R.C. Safety Evaluation with Open Items Related to the License Renewal of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2, Docket Nos. 50-369, 50-370, 50-413 and 50-414," dated October 2002.
7. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.
8. Branch Technical Position 5-3, Revision 2, "Fracture Toughness Requirements," Chapter 5 of Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, NUREG-0800, March 2007.