Duke Energy®

REGIS T. REPKO
Vice President
McGuire Nuclear Station

Duke Energy
MG01VP / 12700 Hagers Ferry Rd.
Huntersville, NC 28078

980-875-4111
980-875-4809 fax
regis.repko@duke-energy.com

March 23, 2011

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 Numbers 50-369 and 50-370

Response to Request for Additional Information Related to License Amendment Request for Emergency Core Cooling System (ECCS) Water Management Initiative (TAC Nos. ME4051 and ME4052)

Reference: Letter from Duke Energy to NRC, dated May 28, 2010
Electronic Mail from NRC to Duke Energy, dated January 31, 2011
Electronic Mail from NRC to Duke Energy, dated February 28, 2011

On May 28, 2010, Duke Energy submitted a license amendment request (LAR) for the Renewed Facility Operating Licenses (FOL) and Technical Specifications (TS) for McGuire Nuclear Station Units 1 and 2 to allow the manual operation of the Containment Spray System (CSS) in lieu of automatic actuation, and revise the minimum volume and low level setpoint on the Refueling Water Storage Tank (ML101600256).

On November 15, 2010, Duke Energy provided a response to the Request for Additional Information (RAI) transmitted by the NRC's letter of October 12, 2010.

On January 31, 2011, the NRC electronically transmitted an RAI. On February 14, 2011, a telephone conference was held to clarify questions 2 and 3. On February 28, 2011, the NRC electronically transmitted clarified questions 2 and 3, superseding these same questions transmitted on January 31, 2011.

Duke Energy's responses to these RAI questions are contained in Attachment 1 to this letter.

Regulatory Commitments contained in Attachments 3 to the May 28 and November 15, 2010 submittals are combined with one (1) additional commitment as shown in Attachment 2 to this letter.

The responses provided by this letter do not result in any impact to the original No Significant Hazards Consideration or the Environmental Consideration contained in the May 28, 2010 submittal.

Pursuant to 10 CFR 50.91, a copy of this letter is being sent to the designated official of the State of North Carolina.

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NRR

If you have any questions or require additional information, please contact K. L. Ashe at (980) 875-4535.

Very truly yours,

R. T. Repko

Attachments

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

Regis T. Repko, Vice President, McGuire Nuclear Station

Subscribed and sworn to me: March 23, 2011
Date

Lori C. Tibby, Notary Public

My commission expires: July 1, 2012
Date

SEAL

xc (with attachments):

V. M. McCree
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Ave., NE Suite 1200
Atlanta, Georgia 30303-1257

J. B. Brady
NRC Senior Resident Inspector
McGuire Nuclear Station

J. H. Thompson (addressee only)
NRC Senior Project Manager (McGuire)
U.S. Nuclear Regulatory Commission
Mail Stop O-8 G9A
Washington, DC 20555-0001

W. L. Cox III, Section Chief
North Carolina Department of Environment and Natural Resources
Division of Environmental Health
Radiation Protection Section
1645 Mail Service Center
Raleigh, NC 27699-1645

Attachment 1
Response to NRC Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION REGARDING LICENSE AMENDMENT REQUEST RELATED TO EMERGENCY CORE COOLING SYSTEM (ECCS) WATER MANAGEMENT INITIATIVE

McGUIRE NUCLEAR STATION, UNITS 1 AND 2

1. In Section 3.2.3 of Attachment 1 of the LAR, the licensee describes that the sump temperature at the time of switchover is approximately 197°F, which is above the previous design temperature of 190°F, and indicated that the piping and components affected by this increase in the sump temperature have been evaluated at a temperature of 200°F. The licensee is requested to provide a comparative summary of the results of the maximum piping stresses for the containment spray system (CSS), along with the allowable stress limits and the applicable code version for the reanalysis and design basis analysis.

Duke Energy Response:

Auxiliary Building piping stress calculations impacted by ECCS water management were revised. The maximum thermal related stress is 27,793 psi with an allowable of 27,950 psi for a ratio of 0.994. The comparative pre-ECCS water management pressure, weight and thermal expansion stress at this location is 24,324 psi with an allowable of 27,975 psi. These stresses and allowable values are in accordance with ASME III, Subsection NC-3650, equation 10.

Reactor Building piping stress calculations impacted by ECCS water management were also revised. The maximum thermal related stress is 45,416 psi with an allowable of 45,750 psi for a ratio of 0.993. This stress and allowable value is in accordance with ASME Section III, Subsection NC-3650, equation 11. The comparative 1981 pre-ECCS water management thermal expansion maximum stress is 15,310 psi with an allowable of 28,150 psi. This stress and allowable value is in accordance with ASME III, Subsection NC-3650, equation 10.

Pipe stresses were calculated using equations 8, 9, 10 and 11 per the ASME Boiler and Pressure Vessel Code, Section III, 1974 edition. This code edition is used because the majority of stress analysis performed at Duke Energy is done using the SUPERPIPE computer analysis program which does not perform stress checking for code editions prior to 1974. By adopting the 1974 equations, consistency is provided between existing and future analyses. The equations from the 1974 ASME Code are being used as a convenience only and do not constitute a fundamental change from the 1971 ASME Code. The stress intensity factors and other requirements of the 1974 ASME Code may be used, but the provisions of the 1971 ASME Code, which is the McGuire Code of Record, remain valid.

2. In Section 3.2.3 of Attachment 1 of the LAR, the licensee describes that as a result of the revised loadings from the evaluation of the CSS pipe stress analysis, 13 pipe support restraints in the auxiliary building were re-qualified and 224 pipe supports in the reactor building will be re-qualified.

    A. The licensee is requested to clarify if the increased loadings on such a significant number of pipe supports are caused by the small increase in the sump temperature, only, or if any other effects contributed to the re-qualification of the supports.

Duke Energy Response:

The revised piping stress analysis for the CSS piping in the Reactor Building chose a temperature of 200°F for the thermal load cases. The original piping analysis of this piping chose a temperature of 120°F for the thermal load cases. The original analysis justified the reduction in temperature from the 190°F design temperature because temperatures above 120°F up to the 190°F design temperature were considered a faulted thermal load.

The revised piping stress analysis for the CSS piping in the Reactor Building analyzed load cases where water hammer and seismic loading are concurrent. These load case combinations are per the direction of McGuire Nuclear Station specification for performing piping analysis. The original piping analysis of this piping did not analyze water hammer loading and seismic loading concurrently.

B. The licensee is also requested to complete the re-qualification of the affected pipe support restraints and provide a summary of the results of this analysis by a letter dated no later than April 15, 2011. The letter should also inform the NRC staff whether the requalification of these support restrains is complete. This information is needed for the NRC staff to complete its review.

Duke Energy Response:

The entire Unit 2 affected pipe support re-qualification is complete and the entire Unit 1 affected pipe support re-qualification is expected to be completed by the end of April, 2011. A letter confirming completion of the Unit 1 pipe support re-qualification will be provided to the NRC by May 2, 2011.

Many of the old support frame qualifications were performed by using hand calculations during plant construction. Due to increase in loads per new pipe stress analysis, these pipe supports are being re-qualified for new loads using computer programs like GTStrudl for frames and SmartBASE (in house finite element analysis for base plate and concrete anchors) for base plates, as found appropriate, to ensure conformity with design specifications for pipe supports and concrete anchors.

As a summary, eight (8) re-qualified pipe supports on Unit 2 CSS piping system at azimuth (AZ) 232° from Reactor Shield Building to the ring header inside the Steel Containment Vessel (SCV) are randomly selected for brief description as below:

(1) Supports inside the SCV: 2MCR-NS-4071, 4074, 4075, 4077, 4079 & 4081.

All supports are similar in design configuration. Each support has a pair of rigid sway struts in pipe tangential direction and upwards from the horizontal plane that provides a vertical and tangential restraint.

One of the strut rear end brackets is welded to an attachment plate which is welded to the pipe. The attachment plate is welded to the pipe using full penetration double-bevel groove weld plus fillet welds. The other end of the strut rear end bracket is welded to a plate which in turn is welded to a rectangular base plate attached to the SCV. The attachment base plate is welded to the SCV using all around fillet weld.

The new enveloped loads from all above supports are 987 lbs (Normal), 1505 lbs (Upset), and 5894 lbs (Faulted). The supports are conservatively re-qualified using 4250 (Normal/Upset) and 8500 lbs (Faulted) load. Calculation considered faulted load case as the governing load case. Below is brief summary of results.

(a) The smallest strut used is #1, Grinnell Fig. 211 rigid sway strut assembly:

Design Normal/Upset loads = 4250 lbs < 8000 lbs allowable (Design, Level A & B)

Design Faulted loads = 8500 lbs < 9600 lbs allowable (Design, Level C).

Calculated required fillet weld size at rear end bracket = 0.158 inch < 5/16 inch provided.

(b) Pipe attachment plate, 3/4 inch x 1 1/2 inches x 5 inches long (SA 240, Gr. 304):

Calculated axial and bending stress (faulted) = 19543 psi < Fy = 25000 psi allowable at 200°F.

(c) Plate 3/4 inch x 4 inches x 9 inches long (Cut to suit)(SA 516, Gr 70): (Considered 9 inches long for analysis)

Calculated axial and bending stress interaction ratio = 0.8 < 1.0 allowable.

Calculated required fillet weld size at 3/4 inch x 4 inch plate = 0.162 inch < 1/4 inch provided.

(d) Base plate 3/4 inch x 9 inches x14 inches long (SA 516, Gr. 70):

Calculated plate bending stress = 33304 psi < Fy = 34600 psi allowable at 200°F.

Calculated required fillet weld size at 3/4 inch base plate = 0.06 inch < 1/4 inch provided.

(2) Support in the Reactor Building annulus: 2MCR-NS-4665.

This is upper most support in the annulus. This is a two way rigid restraint located on the pipe riser. One of the restraints is in the radial direction and is a trapeze support using two rigid sway struts and base plates attached to the Reactor Building shell wall. The restraint in other direction is a rigid sway strut assembly and a base plate attached to the shell wall.

The new maximum loads for this support are:

F_{XL} (radial direction) = + 5515 / - 12390 lbs and F_{ZL} = +1711/ - 2098 lbs. The support is conservatively re-qualified using F_{XL} = ± 14000 lbs and F_{ZL} = ± 3000 for all load cases. Below is brief summary of results.

(a) #3, Grinnell Fig. 211 rigid sway struts for trapeze support:

Total design load supported by two struts = 14000 lbs < 15700 lbs allowable in a single strut (Design, Level A & B)

Calculated required fillet weld size at the rear end bracket = 0.080 inch < 1/4 inch provided.

(b) Calculated bending stress in trapeze beam W8x31 = 6387 psi < 21600 psi allowable.

Calculated deflection = 0.011 inch < 1/16 inch allowable.

(c) Load in each Grinnell Fig. 137S, U-bolt (7/8 inch rod size) = 7000 lbs < 7540 lbs allowable (Design, Level A & B)

(d) 1 inch x 13 1/2 inch x 13 1/2 inch base plate with four 1 inch dia. Phillips Concrete fasteners (WS-10090):

Calculated bolt interaction ratio for tension and shear loads = 0.72 < 1.0 allowable.

Calculated plate bending stress = 8964 psi < 27000 psi allowable.

(e) #5, Grinnell Fig. 211 rigid sway struts assembly:

Design load = 3000 lbs < 27200 lbs allowable (Normal, Level A & B)

Calculated required fillet weld size at the rear end bracket = 0.077 inch < 3/8 inch provided.

(f) 1 1/2 inch x 13 inch x 13 inch base plate with four 1 1/4 inch dia. Phillips Concrete fasteners (WS-12590):

Calculated bolt interaction ratio for tension and shear loads = 0.17 < 1.0 allowable.

Calculated plate bending stress = 1669 psi < 27000 psi allowable.

(3) Support in the Reactor Building annulus: 2MCR-NS-4659.

This support is located in the annulus. This is a vertical and lateral rigid restraint on the horizontal pipe run. A structural steel frame supporting the pipe is attached to an embedded strip plate and a base plate with concrete expansion anchors in the Reactor Building shell wall.

The new maximum loads for this support are:

F_{XL} (radial direction) = + 482 / - 461 lbs and F_{ZL} (tangential direction) = +821/ - 1217 lbs. The supports are conservatively re-qualified using $F_{ZL} = F_{XL}$ = ± 1500 lbs for all load cases. The frame is qualified using GTSTRUDL.

Below is brief summary of results:

(a) The calculated maximum deflection in the restrained direction = 0.001 inch < 1/16 inch allowable deflection.

(b) All members passed '69AISC' code check per GTSTRUDL analysis.

(c) Maximum calculated axial and bending stress interaction ratio = 0.085 < 1.0 allowable.

(d) Calculated tension and bending moment interaction ratio for the Type 1 embedded strip plate = 0.60 < 1.0 allowable.

(e) Calculated required weld size for welds at W4 x 13 = 0.014 inch < 3/16 inch provided.

(f) Calculated required effective throat for flare bevel weld at TS 4 x 4 x 0.250 to base plates = 0.042 inch < 0.156 inch provided.

(g) 3/4 inch x 9 inch x 9 inch base plate with four 1/2 inch dia. Phillips Concrete fasteners (HN-1230):

Calculated bolt interaction ratio for tension and shear loads = 0.83 < 1.0 allowable.

Calculated bending plate stress = 1869 psi < 27000 psi allowable.

All of the above supports are qualified using current pipe support and concrete expansion anchor specifications.

3. In Section 3.2.3 of Attachment 1 of the LAR, the licensee describes that the re-qualification of the Reactor Building pipe supports (approximately three pipe support restraints in the Unit 2 Reactor Building annulus, and one pipe support restraint in the Unit 1 Reactor Building annulus) potentially require modification and will be performed in accordance with the requirements of 10 CFR 50.59. It appears that the licensee has not definitively established the actual modifications for the pipe support restraints.

A. While the installation of the modified supports can be done under 10 CFR 50.59, the licensee is requested to provide a summary of the pipe support restraints requiring modification.

Duke Energy Response:

The Initial screening that was performed to identify pipe support restraints requiring modification remains valid. This screening identified three pipe support restraints in the Unit 2 Reactor Building annulus, and one pipe support restraint in the Unit 1 Reactor Building annulus requiring modification. The modification designs for all of the identified supports are complete. The re-qualifications of these supports are documented in the pipe support restraint calculations shown below:

Unit 2: Engineering Change 101538 scheduled for implementation during outage 2EOC20 in spring 2011.

- Pipe support 2MCR-NS-4767: Modification per Rev. 3A is re-qualified in calculation MCC-1206.12-16-2106.
- Pipe support 2MCR-NS-4516: Modification per Rev. 3B is re-qualified in calculation MCC-1206.12-16-2101.
- Pipe support 2MCR-NS-4608: Modification per Rev. 2A is re-qualified in calculation MCC-1206.12-16-2103.

Unit 1: Engineering Change 101539 scheduled for implementation during outage 1EOC21 in fall 2011.

- Pipe support 1MCR-NS-767: Modification per Rev. 1A is re-qualified in calculation MCC-1206.16.59-1006.

100% of the Unit 2 supports have been re-qualified with no new modifications identified.

The re-qualification of the remaining affected Unit 1 supports is in progress. Based upon the re-qualification completed thus far; high confidence exists that there will not be any additional supports requiring modification.

B. The licensee is requested to complete the pipe support restraint re-qualification for those support restraints used in support of this LAR and provide a summary of the results of this analysis by a letter dated no later than April 15, 2011. The letter should also inform the NRC staff whether the requalification of these support restraints is complete. This information is needed for the NRC staff to complete its review.

Duke Energy Response:

See Response 2B, above.

C. The licensee is also requested to provide a summary table of the support restraints requiring modification, including a brief description of the support restraint direction and type, and the required modification

Duke Energy Response:

The support restraints requiring modifications are listed below:

Pipe Support Restraint No	Description
2MCR-NS-4767 (Unit 2)	Direction: Lateral horizontal restraint on horizontal pipe. Type: One way rigid restraint. The rigid sway strut supported by a steel frame is attached to the concrete reactor shell wall and dome. The existing flexible connections consist of horizontal wide flange members connected to the flange of a vertical wide flange member. The flanges of the horizontal members are coped and the webs are welded to the flange of the vertical member. Modification: The connections are made rigid by adding welded plates at the coped flanges and connecting them to the flange of the vertical member.
2MCR-NS-4516 (Unit 2)	Direction: Lateral horizontal restraint on horizontal pipe. Type: One way rigid restraint. The rigid sway strut supported by a steel frame is attached to the concrete reactor shell wall and dome. One of the existing flexible connections consists of horizontal wide flange member connected to the flange of a vertical wide flange member. The flanges of the horizontal member are coped and the web is welded to the flange of the vertical member. The other flexible connection consists of a vertical wide flange member connected to the base plate at the dome. The flanges of the vertical member are coped and the web is welded to the base plate. Modification: The connection between the horizontal and the vertical members is made rigid by adding welded plates at the coped flanges and connecting them to the flange of the vertical member, and adding stiffener plates in the vertical member. Also, the connection between the vertical member and the base plate at the dome is redesigned using clip angles for moment release.

Pipe Support Restraint No	Description
2MCR-NS-4608 (Unit 2)	Direction: Containment radial and tangential direction on vertical pipe. Type: Two way rigid restraint (vertical guide on the pipe riser). The support consists of a rigid steel frame attached to the concrete reactor shell wall. Modification: Additional welds added at the corners between the structural tubes of the existing frame.
1MCR-NS-767 (Unit 1)	Direction: Lateral horizontal restraint on horizontal pipe. Type: One way rigid restraint. The support consists of a rigid steel frame attached to the concrete reactor shell wall. The frame is attached to an embedded strip plate on the shell wall at one end and the other end is connected to the shell wall using concrete expansion anchors. Modification: Due to the spacing constraints on the wall, the entire support is redesigned using a rigid sway strut supported by a steel frame attached to the concrete shell wall and the dome using base plates and concrete expansion anchors.

4. Section 3.2.2 "NPSH Analysis", second paragraph, last sentence states: "The reanalysis results demonstrated that greater than two (2) feet of water head margin was afforded for the limiting available NPSH condition (i.e., one train of CS) above that required."

   A. What is the basis of the <u>required NPSH</u> that was used to compare with the available NPSH?

   Duke Energy Response:

   The subject limiting NPSH margin was associated with the Containment Spray (CS) pump, aligned in the ECCS sump recirculation mode of operation. The CS pump "required NPSH" was determined to be 17.5 feet of water, and was based on a flow rate of 3800 gpm for the limiting CS pump NPSH curve. The specified flow rate bounds the maximum predicted CS pump flow-rate for ECCS sump recirculation operation (>3% margin).

B. What are the uncertainties included in evaluation of the required NPSH?

Duke Energy Response:

Based on the aggregate conservatisms outlined in response to items 4(A) and (C) here-in, no explicit additional uncertainties were accounted for within the NPSH calculation.

C. Outline all the conservatisms in the calculation of the limiting value of available NPSH.

Duke Energy Response:

- No credit for elevated accident containment pressure.
- Utilized a bounding containment sump fluid temperature of 200°F to determine the vapor pressure penalty. This value bounded the calculated peak temperature of approximately 196°F. This peak sump temperature only exists for a brief period of time (<10 minutes) subsequent to initiation of ECCS sump recirculation. After this timeframe, the sump temperature remains below 190°F (pre-ECCS Water Management containment analyses peak).
- The maximum ECCS sump strainer head-loss was assumed concurrently with peak sump fluid temperature. The strainer debris head-loss was based on the limiting large break loss of coolant accident (LBLOCA) event.
- Three (3) feet containment sump water level was credited, which is the minimum required level for strainer submergence. This is conservative, in that the predicted sump water level for a LBLOCA would be significantly higher.

Attachment 2
NRC Commitments

NRC Commitments

The following identifies those actions committed to by Duke Energy in this letter and previous letters of May 28 and November 15, 2010. Any other statements made in this licensing submittal are provided for informational purposes only and are not considered to be regulatory commitments. Please direct any questions you may have in this matter to K. L. Ashe at (980) 875-4535:

May 28, 2010 Commitments:

1. Prior to actually utilizing the provisions afforded by the approved amendments, McGuire will have in place all required design, document, process changes and personnel training necessary to support these provisions on the affected Unit.

2. The requalification and potential modification of Containment Spray System pipe supports will be completed prior to utilizing the provisions of the approved amendment on the affected Unit.

3. Within 180 days of the installation of the associated modifications for the final unit, McGuire will submit a follow-up administrative license amendment request to delete the superseded TS requirements.

November 15, 2010 Commitment:

4. The EDG load sequencer logic and load acceptance testing will be completed prior to the first entry into Mode 4 operations following the implementing refueling outage.

March 23, 2011 Commitment:

5. A letter confirming completion of the Unit 1 pipe support re-qualification will be provided to the NRC by May 2, 2011.