UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 14, 2010

Mr. Preston Gillespie
Site Vice President
Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (ONS) - REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING LICENSE AMENDMENT REQUEST, TRANSITION TO TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS* (10 CFR), PART 50, SECTION 50.48(c), *NATIONAL FIRE PROTECTION ASSOCIATION STANDARD* (NFPA) 805 (TAC NOS. ME3844, ME3845, AND ME3846)

Dear Mr. Gillespie:

By letter dated May 30, 2008, as supplemented by letters dated October 31, 2008, January 30, February 9, February 23, May 31, August 3, September 29, and November 30, 2009, and April 14, 2010, Duke Energy Carolinas, LLC (the licensee), submitted a license amendment request (LAR) to transition the fire protection licensing basis at ONS from 10 CFR 50.48(b) to 10 CFR 50.48(c), NFPA 805. By letter dated April 14, 2010, the licensee resubmitted the LAR and superseded the content of the LAR submitted by letters dated May 30, 2008, and October 31, 2008. This resubmitted LAR, however, does not supersede previous responses to RAI questions submitted by letters dated October 31, 2008, January 30, February 9, February 23, May 31, August 3, September 29, and November 30, 2009.

By letters dated July 30, 2010, and August 25, 2010, the U.S. Nuclear Regulatory Commission (NRC) issued RAIs. By letters dated September 13, 2010, and September 27, 2010, the licensee responded to the RAIs. Upon review the NRC staff found that some of the RAI responses needed clarification. Therefore, to complete our review of the LAR, the NRC staff needs additional clarifying information. The NRC staff's RAIs are enclosed. Unless otherwise agreed to, please submit all responses to these RAI questions within 30 days.

If you have any questions, please call me at 301-415-1345.

Sincerely,

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure:
RAI

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REQUEST FOR ADDITIONAL INFORMATION (RAI)

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (ONS)

TRANSITION TO

TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS* (10 CFR), PART 50, SECTION 50.48(c)

NATIONAL FIRE PROTECTION ASOCIATION STANDARD (NFPA) 805

By letter dated May 30, 2008 (Agencywide Documents Access and Management System (ADAMS), Accession No. ML081650475), as supplemented by letters dated October 31, 2008 (ADAMS Accession No. ML083120362), January 30, 2009 (ADAMS Accession No. ML091040205), February 9, 2009 (ADAMS Accession No. ML090480143), February 23, 2009 (ADAMS Accession No. ML090700134), May 31, 2009 (ADAMS Accession No. ML091590045), August 3, 2009 (ADAMS Accession No. ML092190212), September 29, 2009 (ADAMS Accession No. ML092740624), November 30, 2009 (ADAMS Accession No. ML093410007), and April 14, 2010 (ADAMS Accession No. ML101121032), Duke Energy Carolinas, LLC (the licensee), submitted a license amendment request (LAR) for the U.S. Nuclear Regulatory Commission (NRC) staff's review and approval. The proposed LAR would approve the transition of the fire protection licensing basis at ONS from 10 CFR 50.48(b) to 10 CFR 50.48(c), NFPA 805.

By letters dated July 30, 2010, and August 25, 2010, (ADAMS Accession Nos. ML102110394 and ML102310174, respectively) the NRC issued RAIs. By letters dated September 13, 2010, and September 27, 2010, (ADAMS Accession Nos. ML102640110 and ML102720409, respectively) the licensee responded to the RAIs. Upon review, the NRC staff found that some of the RAI responses needed clarification. Therefore, to complete our review of the LAR, the NRC staff has determined that additional clarifying information is required.

RAI 2-5c

The NRC staff requires follow-up information on the response to RAI 2-5. During the review of the B-1 Table in the ONS LAR, the NRC staff identified the following issues that are linked to specific B-I Table elements. The licensee should review the LAR submittal and ensure that these and any similar conditions are resolved appropriately.

B-1 Table: NFPA 805, Chapter 3 3.2.2 – Management Policy Direction and Responsibility

NFPA 805, Section 3.2.2 states: "A policy document shall be prepared that defines management authority and responsibilities and establishes the general policy for the site fire protection program." Please provide a compliance statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3 – Prevention

NFPA 805, Section 3.3 states, in part: A fire prevention program with the goal of preventing a fire from starting shall be established, documented, and implemented as part of the fire protection program. The two basic components of the fire prevention program shall consist of both of the following:

(1) Prevention of fires and fire spread by controls on operational activities

(2) Design controls that restrict the use of combustible materials

Please provide a compliance statement that addresses how these two requirements are met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.1 – Fire Prevention for Operational Activities

NFPA 805, Section 3.3.1 states: "The fire prevention program activities shall consist of the necessary elements to address the control of ignition sources and the use of transient combustible materials during all aspects of plant operations. The fire prevention program shall focus on the human and programmatic elements necessary to prevent fires from starting or, should a fire start, to keep the fire as small as possible." Please provide a compliance statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.1.1 - General Fire Prevention Activities

NFPA 805, Section 3.3.1.1, states: "The fire prevention activities shall include but not be limited to the following program elements..." Please provide a compliance statement that addresses how the "but not be limited to" aspect of the NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.1.2 - Control of Combustible Materials

NFPA 805, Section 3.3.1.2, states: "Procedures for the control of general housekeeping practices and the control of transient combustibles shall be developed and implemented. These procedures shall include but not be limited to the following program elements..." Please provide a compliance statement that addresses how both the requirements to develop and implement procedures for the control of general housekeeping and transient combustibles and the "but not be limited to" aspect of this NFPA 805 requirement are met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.7 – Bulk Flammable Gas Storage

NFPA 805, Section 3.3.7, states: "Bulk compressed or cryogenic flammable gas storage shall not be permitted inside structures housing systems, equipment, or components important to nuclear safety." Please provide a compliance statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.12 – Reactor Coolant Pumps

The NRC staff has determined that this parent element contains requirements related to seismic and other accident/off-normal conditions that are not addressed in the detailed sub-parts to this element. Please provide a compliance statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.4.2 – Pre-Fire Plans

NFPA 805, Element 3.4.2, states: "Current and detailed pre-fire plans shall be available to the industrial fire brigade for all areas in which a fire could jeopardize the ability to meet the performance criteria described in Section 1.5." Please provide a compliance statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

RAI 2-11c

The NRC staff requires follow-up information on the response to RAI 2-11. The following documents are used in Table B-1 of the LAR to document previous NRC approval of alternatives to certain NFPA 805, Chapter 3 requirements. [Bracketed statement indicates the Chapter 3 element where the document is used].

- NRC Safety Evaluation (SE) dated August 11, 1978 (ADAMS Accession No. ML7911280619) [3.5.3; 3.5.4; 3.5.15; 3.5.16; 3.6.1]
- NRC SE dated June 7, 1988 (ADAMS Accession No. ML8806170310) [3.6.1]
- NRC Letter dated April 14, 1981 (ADAMS Accession No. ML810422024) [3.11.2]
- NRC Exemption dated August 21, 1989 (ADAMS Accession No. ML8908290074) [3.11.2; 3.11.4]

Please justify the continued validity of each of these approvals, with regard to their use in the B-1 Table.

Used in this context, the term "valid" means that the technical basis for approval still applies. For example, plant modifications or other changes have not invalidated the assumptions or analysis

that formed the basis for the approval, or new information has not surfaced that would invalidate the original finding.

RAI 2-19c

The NRC staff requires follow-up information on the response to RAI 2-19, Approval Request # 2, Fuel-Fired Heaters.

NFPA 805 is a consensus code developed by a diverse group of individuals that represented subject matter experts in the field of fire protection, nuclear insurers, manufactures, enforcement, and lastly experts from the utilities. Their collective expertise concluded, in Section 3.3.1.3.4 of NFPA 805 that fuel-fired heaters shall not be permitted in plant areas containing equipment important to nuclear safety or where there is a potential for radiological releases resulting from fire. The NRC staff finds that the licensee's response to RAI 2-19 does not justify the use of fuel-fired heaters in areas containing equipment important to nuclear safety or where there is a potential for radiological releases resulting from fire. Therefore, the NRC staff is not prepared to approve this request.

Should the licensee choose to pursue this request, the NRC expects to receive all of the following information, for each area these heaters may be used:

1. The increase in core damage frequency due to the use of these heaters;
2. The increase in large early release frequency due to the use of these heaters;
3. A demonstration, in detail, that defense-in-depth is maintained despite the addition of combustible liquids and ignition sources due to these heaters;
4. A demonstration, in detail, that safety margins are maintained despite the use of these heaters;
5. A demonstration, in detail, that the radioactive release goal, objectives, and performance criteria are met, despite the use of these heaters
6. A demonstration, in detail, that the nuclear safety goal, objectives, and performance criteria are met, despite the use of these heaters

RAI 2-22c

The NRC staff requires follow-up information on the response to RAI 2-22, Approval Request #8. Demonstrate the capability to extinguish expected fires inside containment during all operational modes. Specifically, in light of the limited ability of the current low-pressure service water (LPSW) system that supplies the existing standpipe system, explain how additional equipment and capabilities, are sufficient to handle the expected hazards in this area during all modes of operation.

Additionally, explain in summary detail, how the above capabilities are integrated into the Fire Protection Program (for example in procedures, fire pre-plans, fire brigade training, etc.).

RAI 3-38c

The NRC staff requires follow-up information on the standby shutdown facility (SSF) design assumption.

DISCUSSION

Response to RAI 3-1 stated the following:

The licensee requested that the NRC formally document as a "prior approval" recognition that within the first 10 minutes following the identification of a confirmed_active fire requiring activation of the SSF growth will not reach a point where fire damage will:

- Result in spurious equipment operation
- Result in a loss of offsite power condition
- Preclude operation of plant equipment from the control room

Response to RAI 3-38 stated the following:

Alignment basis will be revised (per corrective action program) to reflect that the 10-minute assumption is only used for SSF risk areas. It is not used for non-SSF risk areas. Other spurious operations beyond this assumption were postulated to occur until mitigating actions are taken. This is consistent with the scope of Prior-Approval Clarification Request 1 of Attachment T: "As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC formally document as a "prior approval" recognition that during the 10 minutes required to activate the SSF, fire growth will not have reached a point where fire damage will preclude operator actions from the control room nor will any spurious operations or loss of offsite power conditions occur within the first 10 minutes following the identification of a confirmed active fire."

The licensee's letter dated September 20, 1982: The licensee responsed to an NRC staff RAI regarding spurious actuations of components which may affect the safe shutdown capability, which states:

> The originally accepted design concept for SSF operation was based on a 10 minute capability to transfer control to the SSF. Hot shorts or spurious actuation due to fire within the first 10 minutes of the event are not part of the design basis.
>
> This justification was based on the extreme unlikelihood of multiple spurious operations resulting in unacceptable coolant loss essentially coincident with loss of multiple mitigating systems within a 10-minute period...
>
> Upon confirmation of a fire in the plant, operating personnel will be dispatched to the SSF where they will establish communication with the control room.

Since time zero (t_0) for the event is not defined, the phrase "within the first 10 minutes of the event" could be interpreted to mean that no spurious operations are assumed to occur for 10 minutes following fire initiation. If a detector response time of 5 minutes is assumed, upon receipt of an

alarm in the control room, operators would have 5 minutes to man the SSF prior to spurious actuations. This time would be reduced further if the time for operators to confirm the type and size of the fire is considered.

The stated justification (likelihood of fire damage causing unacceptable coolant loss coincident with loss of multiple mitigating systems within a 10-minute period) is limited to high/low pressure interface valves identified in the NRC staff's original RAI and does not bound the current assumption for all potential spurious operations.

The statement "Upon confirmation of a fire in the plant" infers that operators will man the SSF in response to all fires not just those that are deemed to be of a certain size, thus minimizing the time to enter the procedure. RAI 3-1 stated that "A Confirmed Active Fire is defined as a locally observed fire with smoke and either radiant heat or visible flame".

Regulatory Guide (RG) 1.205 and Industry Guidance Nuclear Energy Institute (NEI) 00-01

RG 1.205 states when the requirements in Chapter 4 of NFPA 805 are not met for the protection of required circuits, circuit analysis assumptions regarding the number of spurious actuations, the manner in which they occur (e.g., sequentially or simultaneously), and the time between spurious actuations should be supported by engineering analysis, test results, or both, that are accepted by the NRC.

Within NEI 00-01, Section 3, Deterministic Methodology, the requirements of Appendix R Sections III.G.1, III.G.2 and III.G.3 apply to equipment and cables required for achieving and maintaining safe shutdown in any fire area. Also assume that the fire may affect all unprotected cables and equipment within the fire area. This assumes that neither the fire size nor the fire intensity is known. This is conservative and bounds the exposure fire that is postulated in the regulation.

Loss-of-Offsite Power

NEI 00-01 assumes the loss-of-offsite power has the potential to affect safe shutdown capability. In addition, the regulatory requirements for offsite power differ between the redundant and alternative/dedicated shutdown capability. Therefore, consideration must be given for the loss-of-offsite power when evaluating its effect on safe shutdown. The Appendix R requirement to consider a loss-of-offsite power is specified in Section III.L.3.

ONS Licensee Event Reports (LERs)

The NRC staff reviewed the following LERs associated with the SSF concerning the 10-minute assumption.

- LER 269/2006-004, Design Oversight Results in Appendix R Deficiencies (ADAMS Accession No. ML063470037)

- LER 269/2003-01 (8-4-2003), Design Oversight Results In Appendix R Control Cable Separation Issue (ADAMS Accession No. ML032250125)

- LER 269/2002-02 (5-28-2002), Potential for Fire to Indirectly Damage Mitigation Component (ADAMS Accession No. ML021580287)

The LERs documented concerns with the current licensing basis and the 10-minute assumption associated with the SSF.

Armored Control Cable Testing

The NRC staff witnessed ONS proprietary armored cable fire testing and observed that the armored cable material was combustible, had horizontal flame spread, and did not self-extinguish when the test burner was removed.

The NRC requests the following:

1. Provide the definition of a confirmed active fire requiring activation of the SSF.

2. Provide additional documentation that confirms the basis that spurious operations will not occur within the first 10 minutes following the identification of a confirmed active fire and that the assumption remains valid in light of:
    a. "prior approval"
    b. acceptable engineering analysis or test results
    c. LERs
    d. armored cable fire tests

3. Provide documentation that confirms the basis that no loss-of-offsite power conditions occur within the first 10 minutes following the identification of a confirmed active fire and that the assumption remains valid in light of:
    a. "prior approval"
    b. LERs
    c. armored cable fire tests
    d. the method used to meet NEI 00-01

4. Provide documentation that confirms the basis that fire growth will not have reached a point where fire damage will preclude operator actions from the control room within the first 10 minutes following the identification of a confirmed active fire and that the assumption remains valid in light of:
    a. "prior approval"
    b. LERs
    c. armored cable fire tests

RAI 5-71c

Question 5-71 requested the licensee to describe the process to confirm that the final estimates developed for the as-built, as-operated, proposed protected service water (PSW) system are consistent with, or is bounded by, the initial estimates. The description should include the quantitative criteria that will be developed and the relation of these criteria to the functional

reliability estimates in the response to RAI 5-70. The process should include the actions to be taken if the final estimates cause the acceptable change-in-risk guidelines to be exceeded.

Licensee's Response

A table was provided in related RAI 5-70 that gives the change in core damage frequency (CDF) associated with each of the three PSW functions.

The installed PSW system will be treated similarly in maintenance rule space as the current ONS auxiliary service water system. Given that the proposed PSW system represents a significant risk offset for the risk of variance from deterministic requirements (VFDRs) associated with the transition to NFPA-805, the post-PSW system installation fire probabilistic risk assessment (FPRA) results will be compared to the current pre-PSW system installation FPRA results. The FPRA update will address additional changes not associated with installation of PSW system. If there is an increase of more than 10 percent in the current FPRA results, the individual scenarios that are driving that increase would be investigated. If the increase is attributed to a difference in PSW system credit, then the impacted fire risk evaluations will be revisited to ensure that the delta risk results remain within the range of the results used to support the previous conclusion.

NRC Staff's Comment

The proposed "monitoring" does not appear to be capable of monitoring the availability, reliability, and performance of the PSW system to ensure that the assumptions in the performance-based methods remain valid. The maintenance rule may be an acceptable monitoring process, but the statement that the PSW system will be treated similarly is inconclusive for the quantitative criteria. Conversely, a quantitative criterion of 10 percent in the current FPRA results of individual scenarios bears no clearly definable relation to the availability, reliability, and performance of the PSW system credited in the transition risk analysis. Changes to the scenario frequencies are the cumulative effect of changes to the facility, changes in the probabilistic risk assessment (PRA) models, changes in the PRA modeling assumption. Changes to the modeling assumptions, for example, could easily lead to reductions in scenario frequencies despite potentially large reductions in the PSW system availability, reliability, and performance.

Please provide the estimate for the availability of the PSW system functions credited in the current PRA.

RAI 5-73c

RAI 5-73 requested the licensee to provide the delta risk for all submersible pump deployment and activation actions.

Licensee's Response

Thus, if modeled, the delta risk associated with the failure to deploy the submersible pump is expected to be epsilon for fire events.

NRC Staff's Comment

The expectation of an impact is unusual. Without a quantitative estimate it is difficult to conclude that the licensee has identified the particular and detailed impact in the PRA and evaluated that impact.

Please provide a quantitative estimate of the delta risk associated with failure to deploy the submersible pump.

RAI 5-75c

RAI 5-75 requested the licensee to explain how fire detection and suppression are modeled in the risk assessment, how fire detectors are credited in this modeling, and why fire detectors are required to meet the risk criteria.

Licensee's Response

Other than a few cases where automatic suppression is credited, no credit for detection is provided in the quantification of fire risk for Oconee. However, the Oconee Fire PRA assumed no damage beyond the initial zone of influence provided the formation of a hot gas layer did not occur until after an assumed fire brigade response time of 20 minutes.

NRC Staff's Comment

The two sentences appear to be inconsistent. Assuming no damage beyond the initial zone of influence does credit detection in the quantification of fire risk. The licensee assumes that fire suppression is always successful based on rapid detection. Successful manual fire suppression is required or damage beyond the initial zone of influence can be expected. Therefore, detection and suppression are credited in the quantification of fire risk because non-suppressed fire scenarios leading to core damage are not included in the risk analysis. The impact of properly including these missing fire scenarios will always increase risk and could increase the risk substantially. Similarly, the change in risk estimates could be unaffected or could increase substantially depending solely upon the particular plant equipment configuration nearby each ignition source. Therefore, these are non-conservative assumptions.

Please identify the potential impact on the change in risk estimates of removing these non-conservative assumptions. If no impact is expected, please provide a discussion on how this conclusion is support based on fire development and other site specific analysis parameters.

RAI 6-2c

The NRC staff requires follow-up information on your response to RAI 6-2, Monitoring Program. The response did not provide the criteria being used to determine the structure, system, and components (SSCs) included in the monitoring program. Specifically, the response states that performance monitoring goals (PMGs) in high safety significant fire zones will be monitored. Please provide the following:

1. Define "high-safety significant" fire zones and the criteria (e.g., core damage frequency (CDF), large early release frequency (LERF), etc.) used to select SSCs to be monitored.

2. Clarify how SSCs credited to meet the nuclear safety performance criteria but not located in a "high-safety significant" fire zone are evaluated for inclusion in the monitoring program.

3. Clarify how SSCs credited for meeting the radioactive release performance criteria are evaluated for inclusion in the monitoring program.

4. Describe the process for how system failures are evaluated to determine if the target reliability and/or action levels have been exceeded.

5. The response states that Electric Power Research Institute (EPRI) 1006756 will be used to develop availability and reliability criterion for PMGs. With regard to the EPRI document, provide the following:

   a. The EPRI document provides the results of an analysis of FPP system availability for several plants, but does not provide any specific recommendations for target availability. Describe how target availability values are developed for ONS.

   b. The EPRI document provides the recommended target reliability values: 1) valve position checks – 99 percent, and 2) all other activities – 95 percent. Clarify that these are the initial target reliability values being used at ONS. If not, describe how target reliability values are being developed (e.g., is plant-specific data being used).

   c. The EPRI document provides the recommended test/inspection frequencies for various FPP systems. Clarify that these are the initial test/inspection frequencies being used at ONS. If not, describe how test/inspection frequencies are being developed (e.g., is plant-specific data being used). Justify the use of these test/inspection frequencies in lieu of the frequencies specified in applicable NFPA codes (e.g., NFPA 72 for fire detection systems).

   d. The EPRI document provides the recommended action level values: 1) valve position checks – 97 percent, and 2) all other activities – 92 percent. Clarify that these are the initial action level values being used at ONS. If not, describe how action level values are being developed (e.g., is plant-specific data being used).

   e. The EPRI document provides failure criteria for the various FPP systems. Clarify if these failure criteria are being used at ONS or provide the alternative failure criteria being used.

If you have any questions, please call me at 301-415-1345.

Sincerely,

/RA/

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

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