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ONS-2014-096 10 CFR 50.90

June 30, 2014

Attn: Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2746

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2 and 3
Renewed Facility Operating License Numbers DPR-38, 47 and 55
Docket Numbers 50-269, 50-270 and 50-287
Response to Request for Additional Information Associated with License Amendment Request Regarding Standby Shutdown Facility Quality Requirements
License Amendment Request No. 2012-11, Supplement 1

On October 24, 2013, Duke Energy Carolinas, LLC (Duke Energy) submitted a License Amendment Request (LAR) requesting Nuclear Regulatory Commission (NRC) approval to amend Section 3.1.1.1 of the Updated Final Safety Analysis Report (UFSAR) of Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station (ONS) Units 1, 2, and 3. This change addresses the quality requirements of the Standby Shutdown Facility (SSF) and interconnected systems. On June 4, 2014, Duke Energy received an email request for additional information (RAI). The NRC requested Duke Energy to provide the additional information within 30 days of the request, July 4, 2014. The enclosure to this letter responds to the request for additional information.

There are no Regulatory Commitments within this LAR supplement. Inquiries on this submittal should be directed to Sandra Severance, Oconee Regulatory Affairs Group, at (864) 873-3466.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 30, 2014.

Sincerely,

Scott L. Batson
Vice President
Oconee Nuclear Station

Enclosure: Duke Energy Response to Request for Additional Information

A001
NRR

cc w/enclosures:

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ENCLOSURE

Duke Energy Response to Request for Additional Information

Enclosure
Duke Energy Response to Request for Additional Information

NRC Background

Oconee Nuclear Station Updated Final Safety Analysis Report (UFSAR) Section 3.1.1 identifies the major components of the Standby Shutdown Facility (SSF) as Quality Assurance classification 1 (QA-1). The quality assurance program provides assurance that plant equipment will perform satisfactorily in service. The License Amendment Request (LAR) states that plant equipment used for the SSF function that was not QA-1 prior to the construction of the SSF was not intended or required to be upgraded to QA-1. If the equipment is not QA-1, assurance that the equipment is capable of performing satisfactorily and supporting the credited SSF function must be provided by another quality assurance program.

Equipment that is relied upon to support the SSF function should remain functional in those scenarios where the SSF is credited and the quality assurance requirements applied to this equipment should establish assurance that the equipment will function properly.

RAI-1

Describe how plant equipment used for the SSF function but pre-dating the SSF installation has been verified to withstand, without loss of function, the seismic and environmental conditions associated with scenarios that credit the SSF. Describe how the quality assurance classification(s) (other than QA-1) applied to this equipment provides adequate assurance that this equipment will perform satisfactorily in service.

Duke Energy Response

- How plant equipment used for the SSF function but pre-dating the SSF installation has been verified to withstand, without loss of function, the seismic and environmental conditions associated with scenarios that credit the SSF.

The Standby Shutdown Facility (SSF) is designed as a standby system for use under certain emergency conditions. The system provides additional "defense in-depth" protection by serving as a backup to existing safety systems. The SSF has had a dynamic design and licensing basis history. Originally developed to augment existing plant capabilities relative to mitigating postulated occurrences such as fires, turbine building flooding, and security incidents, the SSF was later credited in response to other scenarios. The scenarios for which the SSF is credited are briefly discussed below:

- Fire, Turbine Building Flood, and Sabotage - original design considerations [SSF Design Safety Evaluation (SE) dated April 28, 1983]
- Station Blackout (SBO) - SSF diesel generator credited as an alternate ac (AAC) power source. [Supplemental SE for SBO dated December 3, 1992]
- Emergency Feedwater (EFW) Single Failure Backup - SSF Auxiliary Service Water (ASW) credited to compensate for the lack of complete EFW single failure design. [UFSAR 10.4.7 Revision, Amendments 325/325/326 dated June 6, 2011]

- Generic Letter (GL) 81-14 review of adequacy of EFW Seismic Design - Seismically qualified SSF ASW System credited as an alternate method of providing secondary heat removal to substitute for the lack of EFW seismic qualification. [EFW System Seismic Qualification SE dated January 14, 1987]
- Tornado missiles - The SSF ASW pump and its associated emergency diesel generator power supply credited as a secondary source of feedwater since they are located in the fully tornado missile protected SSF structure. [Protection of EFW Against Tornado Missiles SE dated July 28, 1989]

In the 1980's, when the SSF design was submitted (and subsequently approved), Duke Energy committed to design the SSF structure and the associated sub-systems (i.e., new equipment required for event mitigation added when SSF design was installed) to seismic Category 1 requirements. As part of the SSF design process, Duke Energy established a position of minimizing interconnections between the systems being added as part of the SSF activities and the then-existing plant systems, structures, and components (SSCs). Where connections to existing systems and components were made, the established position was that the existing SSCs would not be upgraded to the same requirements as the SSF-related systems that were being added. As re-stated in the License Amendment Request submitted October 24, 2013, Duke Energy did not commit to upgrade any pre-existing plant system, structures, or components to QA-1 nor did the NRC impose any additional requirements on pre-existing plant equipment during the original SSF licensing efforts. As such, the codes and standards applicable to the SSF structure and associated sub-systems are different than those applied to pre-existing plant equipment.

Plant equipment used to support the SSF for event mitigation is required to meet its design function. The equipment design function includes all functions required to safely and reliably operate the plant in a broad spectrum of accident and normal operating conditions for which it is credited. A calculation was developed to address the non-QA-1 equipment credited for the Turbine Building Flood event by examining the non-QA-1 SSF-related equipment, determining the environmental conditions that may be present when the equipment is used to support the SSF for event mitigation, and evaluating that equipment's suitability for that environment. Bounding environmental conditions for all SSF-mitigated events were conservatively used as inputs for this calculation. Additionally, the calculation verifies the adequacy and ability of the same equipment to function, post event. This evaluation demonstrated that all the identified non-QA-1 components are fully capable of performing their required functions with the seismic and environmental conditions to which they may be exposed.

The design functions required for accident and normal plant operations are protected as part of the plant design basis, as documented in the Design Basis Documents. Functions may only be added or changed through the design change process. As pointed out in the March 6, 2012, Confirmatory Action Letter (CAL 2-12-001) from NRC to Duke Energy, NRC findings associated with the SSF have called into question the clarity and implementation of the SSF design and licensing basis. In response to that CAL, Duke Energy committed to, and performed, a comprehensive design, licensing, and operational review of the SSF. The goal of the review was to ensure that SSCs associated with the SSF functions are capable of performing their design functions. This extensive review did not identify any new instances of pre-existing equipment that was unable to withstand the seismic and environmental conditions associated with SSF-mitigated scenarios. During an inspection the week of December 2, 2013, the NRC reviewed the overall approach of the SSF review, the issues identified, and the actions planned

or taken. The NRC found that the SSF review was adequate to satisfy the commitment documented in CAL 2-12-001.

In summary, the comprehensive design, licensing, and operational review of the SSCs associated with SSF functions, combined with calculation and design control, provides assurance that original plant equipment used to support SSF event mitigation is expected to function in scenarios that credit the SSF.

- How the quality assurance classification(s) (other than QA-1) applied to this equipment provides adequate assurance that this equipment will perform satisfactorily in service.

A backup to existing safety systems, the SSF is designed to maintain the reactor in a safe shutdown condition for a period of 72 hours following a fire, turbine building flood, sabotage, or tornado missile events and for a period of four hours following a station blackout. These are not design basis events. Accordingly, the SSF is designed in accordance with criteria associated with these events. It is clear in the Oconee licensing basis that there are non-QA-1 SSCs at Oconee for which credit is taken to mitigate accidents. As described in the October 24, 2013, License Amendment Request, Duke Energy formalized an augmented quality assurance program, QA-5, to ensure that components that were required to perform a safety-related function, but which had not originally been purchased to QA-1 standards, would be maintained and tested in a manner consistent with QA-1 components. This process is described in detail in an April 12, 1995, letter to the NRC, "Oconee QA-1 Licensing Basis and Generic Letter 83-28, Section 2.2.1, Subpart 1 Supplemental Response" (Reference 1). By letter dated August 3, 1995 (Reference 2), the NRC accepted this response. As part of this effort, a methodical process was utilized to determine the appropriate classification for the various SSCs. As a result of this effort, SSCs were categorized as QA-1, QA-5, or non-QA.

If the review conducted per Reference 1 determined that the SSC routinely operates during normal plant operation in the same mode that it would function during an accident and that the limiting operational and design parameters under normal operating conditions bound the limiting operational and design parameters under accident conditions, the SSC retained its non-QA classification. Although non-QA, the SSCs are recognized as being important to the safe and reliable operation of Oconee Nuclear Station. As such, these SSCs are considered controlled plant equipment and are subject to the programmatic requirements associated with 10 CFR 50.65 (Maintenance Rule), Configuration Management, and Equipment Reliability. If the review performed per Reference 1 determined that the SSC does not operate during plant operation in the same mode that it would function during an accident, then the SSC is classified as QA-5. These SSCs warrant coverage under an augmented quality assurance program. In addition to the practices described above for non-QA equipment, the QA-5 program incorporates additional requirements to enhance the reliability of important, non-QA-1 equipment by focusing on the maintenance and testing of this equipment under selected Appendix B criteria (e.g., tested using the procedures equivalent to and on a frequency commensurate with those for QA-1 equipment procedures, inspected by qualified QA personnel with a similar scope to that applied to QA-1 activities).

In summary, QA-5 program does not impose special design control processes. QA-5 components are not procured per Appendix B, rather when replacement equipment or parts are needed, they are procured as good or better than original based on engineering judgment. This approach has been consistently presented in discussions during the development of the

QA-5 program. The NRC accepted this approach as documented in the August 3, 1995, Safety Evaluation which reiterated that the augmented QA program should provide enhancement to assure that equipment important to the mitigation of accidents and transients will perform their intended function.

Application of sound engineering practices and the use, monitoring, trending, maintenance and testing of these important SSCs, provides adequate assurance that this equipment performs satisfactorily when placed into service.

References:

1. Letter, J. W. Hampton (Duke Power) to USNRC, Oconee Nuclear Station, Docket Nos. 50-269, -270, -287, Oconee QA-1 Licensing Basis and Generic Letter 83-28, Section 2.2.1, Subpart 1 Supplemental Response, April 12, 1995.
2. Letter, Leonard A. Wiens (USNRC) to J. W. Hampton (Duke Power), Generic Letter 83-28 Supplemental Response - Oconee Units 1, 2, and 3, August 3, 1995.