Southern Nuclear
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52-026

ND-17-2102
10 CFR 50.90
10 CFR 52.63

U.S. Nuclear Regulatory Commission
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Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3 and 4
Revised Supplement to Request for License Amendment and Exemption:
Pipe Rupture Hazard and Flooding Analysis (LAR-17-010S4R1)

Ladies and Gentlemen:

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC) requests an amendment to the combined licenses (COLs) for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 (License Numbers NPF-91 and NPF-92, respectively). The requested amendment proposes to depart from approved AP1000 Design Control Document (DCD) Tier 2 information (text, tables, and figures) [as incorporated into the Updated Final Safety Analysis Report (UFSAR) as plant-specific DCD information], and involves related changes to COL Appendix C information, with corresponding changes to the associated plant-specific Tier 1 information. Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is also requested for the plant-specific DCD Tier 1 material departures.

The requested amendment proposes changes to the COL, COL Appendix C (and to plant-specific Tier 1 information) and associated Tier 2 information to address mitigation of fire protection system flooding of the Auxiliary Building identified during completion of the pipe rupture hazards analysis (PRHA).

Enclosures 1 through 4 were provided with the original LAR. Enclosures 5 and 6 were provided on August 21, 2017, with SNC letter ND-17-1465 (LAR-17-010S1), in response to 5 of 7 NRC Staff requests for additional information (RAIs) dated July 20, 2017 (ADAMS accession number ML17201Q412). Enclosures 7 through 11 were provided on October 9, 2017, with SNC letter ND-17-1725 (LAR-17-010S2), in response to the remaining two NRC Staff RAIs dated July 20, 2017, and included updated responses to two others in response to NRC Staff follow-up questions. Enclosures 12 through 14 were provided on November 1, 2017, with SNC letter ND-17-1831 (LAR-17-010S3), in response to the NRC Staff RAIs #2 dated September 22, 2017.

Enclosure 15 provides revised responses to NRC Staff RAI #3 dated November 21, 2017 (based on clarification calls of November 15 and 16, 2017). Enclosure 16 provides revised associated additional licensing basis document revisions.

The supplemental information provided in this LAR supplement does not impact the scope, technical content, or conclusions of the Technical Evaluation, Significant Hazards Consideration Determination, or Environmental Considerations of the original LAR, LAR-17-010, provided in Enclosure 1 of SNC letter ND-17-0496.

SNC requests staff approval of the license amendment and associated exemption by January 15, 2018, to support continued construction activities and ITAAC closure activities. SNC expects to implement the proposed amendment (through incorporation into the licensing basis documents; e.g., the Updated Final Safety Analysis Report) within 30 days of approval of the requested changes.

In accordance with 10 CFR 50.91, SNC is notifying the State of Georgia of this LAR supplement by transmitting a copy of this letter and enclosures to the designated State Official.

This letter contains no regulatory commitments. This letter, including enclosures, has been reviewed and confirmed to not contain security-related information.

Should you have any questions, please contact Ms. Amy Chamberlain at (205) 992-6361.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of December 2017.

Respectfully submitted,

Amy G. Aughtman

Amy G. Aughtman
Licensing Director, Nuclear Development
Southern Nuclear Operating Company

Enclosures:

- 1 – 4) (Previously submitted with original LAR-17-010 via ND-17-0496)
- 5 – 6) (Previously submitted as supplemental information with LAR-17-010S1 via ND-17-1465)
- 7 – 11) (Previously submitted as supplemental information with LAR-17-010S2 via ND-17-1725)
- 12 – 14) (Previously submitted as supplemental information with LAR-17-010S3 via ND-17-1831)
- 15) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Revised Response to NRC Request for Additional Information #3 Regarding the LAR-17-010 Review (LAR-17-010S4R1)
- 16) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Revised Revisions to Proposed Changes to Licensing Basis Documents (LAR-17-010S4R1)

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Southern Nuclear Operating Company

ND-17-2102

Enclosure 15

Vogtle Electric Generating Plant Units 3 and 4

Revised Response to NRC Request for Additional Information #3

Regarding the LAR-17-010 Review

(LAR-17-010S4R1)

(This Enclosure consists of 7 pages, including this cover page.)

Requests for Additional Information #3

- Question 1, Radiation Protection Review (Supplemented)
- Question 2, Instrumentation and Control Review

Question 1, Radiation Protection Review:

Follow-on RAI to SNC Supplement 2 Response to RAI Questions 2 and 3, dated October 9, 2017

GDC 2, "Design bases for protection against natural phenomena," requires in part that SSCs shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions and shall reflect, in part, the importance of the safety functions to be performed.

GDC 60, "Control of releases of radioactive materials to the environment," requires that the nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

GDC 61, "Fuel storage and handling and radioactivity control," requires in part that the fuel storage and handling, radioactive waste, and other systems with may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions and s shall be designed with appropriate containment, confinement, and filtering systems, among other aspects.

10 CFR 20.1101(b) requires that the licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

10 CFR 20.1406 requires that the design minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

Please provide additional information to enable the staff to reach a reasonable assurance finding that the worst case flooding will not result in a significant release from the gaseous waste management system. The charcoal guard bed and delay beds are located in Room 12153, which are on the bottom floor of the building. Table 2 in response to Question 7.a indicates 168" of flooding in that room. The concern is that potentially the water could damage the system or the beds or that water could infiltrate the beds and cause them to loose there adsorption ability, resulting in a significant release of the content of the beds (physical integrity of the rad waste system or other system or component failure mechanisms that may lead to a significant release of gaseous activity, etc.).

If the licensee cannot provide reasonable assurance that the flooding will not result in a substantial release from the gaseous waste management system, then provide the dose consequences for a release of radioactive material. Describe the acceptance criteria for those dose consequences.

RESPONSE to Question 1:

Section I: Impacts to Waste Gas System Functionality as a Result of Buoyant Forces

The subsections of the response below address engineering consideration given to upward buoyancy effects on WGS equipment.

Delay Beds

The empty weight of the delay bed tanks (MV6H tanks) is 3,800lbs. Considering a minimum weight of activated carbon of 2247 lbs (conservative for each bed, with respect to UFSAR, Table 11.3-1), this brings the total weight during operation to at least 6,047 lbs. This is a conservative weight for each delay bed considered only for this evaluation, and, normally, each delay bed would be expected to be operating at a total weight above this value due to the presence of additional activated carbon.

Considering a tank internal volume of 89 cubic feet (conservative for this evaluation with respect to Table 11.3-2), and a shell volume of 10132.8 cubic inches, the total volume of the component is 94.86 cubic feet. During the postulated flooding event, the weight of displaced water would then equate to less than 5,910 lbs.

This conservatively illustrates that the weight of the delay beds exceeds the weight of the water displaced, and, therefore the tanks will not be subjected to upward buoyant forces even if completely submerged. This calculation is conservative in that it uses a minimum mass of carbon in the delay beds, and assumes complete submergence of the tanks, also neglecting any loads imparted by the supporting and attached piping.

Guard Beds

Similar evaluations for the guard bed (MV6G) show that the guard bed is significantly heavier, proportionally, to the weight of water it displaces. The empty guard bed tank weight is identified as 2200 lbs, and combined with a minimum carbon mass of 209 lbs, yields a total component mass of 2409 lbs. While, due to its small size (a nominal 8 cubic feet as shown in Table 11.3-1 of the UFSAR, combined with a shell volume of 4.13 cubic feet), the volume of displaced water is only 12.13 cubic feet, correlating to a weight of 755.7 lbs. This indicates there is no risk or hazard due to upward buoyant forces acting on the guard bed following the postulated flooding event.

Anchor bolts and Other Associated Margins Not Mentioned Above

Beyond these evaluations, an important consideration for the design is that the beds are bolted to a mechanical module that is then welded to the floor. This effectively anchors the beds to the structural floor (the nuclear island basemat). The anchor bolts used in this configuration for each component are also qualified to withstand significant forces, of several thousand pounds in magnitude.

The welds attaching the module to the floor are likewise designed to withstand significant forces. In addition, the components themselves are connected to other peripheral piping and supports, which help to constrain the components within the connected matrix of systems, structures, and components. This provides added margin reinforcing the conclusions above.

Section II: Potential for Moisture Ingress to WGS Components

Separately from the buoyancy evaluations described above, Subsection 11.3.1.2.2.2 of the UFSAR addresses water incursion and design features included in the plant to preclude or mitigate the potential for water ingress into the WGS. While this section is written to address the concern of moisture intrusion from other waste system components (like the degasifier separator), the features are also relevant and applicable for the postulated flooding event discussed in SNC LAR 17-010.

Additionally, note that the operating pressure of the system is ≥100 psi (UFSAR, Table 11.3-2). This exceeds the hydrostatic pressure of the postulated flood, even on the basemat elevation. Therefore, there is no driving force to send moisture from the flood into the WGS given the system operating parameters, and thus, minimal risk of moisture intrusion.

Conclusion

Based upon the discussions above – including evaluations that indicate the gaseous radwaste management system will remain intact following a postulated flood – the design conveyed in the UFSAR as modified in SNC LAR 17-010 is sufficient to conclude that the worst case postulated flood event will not result in a significant release from the gaseous waste management system.

Follow-up Information

Based on the physical arrangement of the system, the system components, and the flooding scenario described in the LAR, there would be no breach of the WGS system to allow moisture incursion into the system. There is no controlled valve that allows a bypass of the WGS Charcoal Delay beds, and the valve which isolates the normal system release (post-delay bed) is both (a) fail closed and (b) controlled to close on a high-high radiation signal (WGS-PL-V051, non-safety valve). Flooding of PLS cabinets (located in various rooms on the non-RCA side above EL. 100') is not postulated, so a spurious control signal is not postulated as a consequence of this event. There are no other valves that could cause a spurious release due to the flooding described in the LAR.

Question 2, Instrumentation and Control Review:

Follow-on RAI to SNC Supplement 3 Response to RAI Question 3, dated November 1, 2017

Title 10 of the Code of Federal Regulations (CFR) Section 50.90, "Application for amendment of license, construction permit, or early site permit," states, in part, that when a holder of a combined license (COL) desires to amend the license, the application for an amendment must fully describe, "the changes desired, and following as far as applicable, the form prescribed for original applications." 10 CFR 52.79, "Contents of applications; general information," requires, in part, that applicants for a COL provide a description and analysis of system, structures, and components of the facility sufficient to permit understanding of the system design and their relationship to safety evaluations as well as the principle design criteria for the facility. 10 CFR 50.49, "Environmental qualification of electric equipment important to safety to nuclear power plants," requires, in part, that the holder of a COL shall establish a program for qualifying the electrical equipment defined as important to safety.

License Amendment Request (LAR) 17-010 requested that, due to a revision of the Pipe Rupture and Hazard Analysis (PRHA) that resulted in higher water levels being postulated in areas of the auxiliary building, the licensee chose to add new safety-related level sensors, level transmitters and associated instrumentation and controls (I&C) equipment in the auxiliary building. Based upon the information provided in the LAR pertaining to the new safety-related level equipment (that interfaces with the protection and safety monitoring system (PMS)) and their new interface with microprocessors that perform safety function(s) within the PMS, acceptance criteria in 10 CFR 50.55a(h), "Protection and safety systems" apply to the review of this amendment request. Regarding the principle design criteria for the facility, several criteria within 10 CFR Part 50 Appendix A, "General Design Criteria for Nuclear Power Plants," apply. Specifically, General Design Criteria (GDC) 4, "Environmental and Dynamic Effects Design Basis"; GDC 13, "Instrumentation and Control"; GDC 20, "Protection System Functions"; GDC 23, "Protection System Failure Modes"; and GDC 24, "Separation of Protection and Control Systems," are also applicable.

Based upon the information provided in Chapter 7 of the licensee's updated final safety analysis report (UFSAR) and the information provided in WCAP-16675-P, "AP1000 Protection and Safety Monitoring System Architecture Technical Report," Revision 5, it identifies the PMS performing the following four functions:

• the reactor trip (RT) functions
• the engineered safety features (ESF) actuation functions
• the Qualified Data Processing System (QDPS) functions, and
• safety-related Component Control functions

The technical report acts as a secondary reference to the licensee's UFSAR. The architecture technical report describes no other functionality of the PMS, which would account for this type of new equipment and related signaling being installed within the PMS.

Based upon the licensee's responses to the questions posed that relate to the change identified in LAR 17-010, it identifies the newly installed level monitoring equipment signals interfacing with the PMS, although not as a part of the four previously identified functions of the safety-related system.

Specifically, based upon the licensee's response to a staff request for additional information discussed below, the new signals are described as being provided by their respective sensors, processed through the bistable process logic processor (BPL) and the integrated communications processor (ICP) and then sent to the safety-related displays (the staff understands that safety-related display function to be that information displayed on one or more of the primary dedicated safety panels (PDSPs).

a. In letter dated November 1, 2017 [ADAMS Accession No. ML17305B507], the licensee's response to Question 3 states, in part, "The data from the sensors input into the bistable processor logic subsystem." Based upon the information related to the addition of the level switch signals, it appears that it would increase the tasks performed by the BPL. As such, it could affect the maximum central processing unit (CPU) loading analysis completed on this given system. The staff requests the licensee to clarify the following points:

    1. Clarify how the added CPU load to process these new signals will be accounted for in a CPU loading analysis of the affected processors and identify any related reports or other documentation in which this information will be captured to determine the impact to both the CPU loading analysis and time response to the BPLs.

    2. Clarify and identify any associated documentation, and the specific location within the related documents, detailing how the requirements for these new signals being processed via the PMS have been formulated and captured and what and how testing, regression or otherwise, would be performed to account for this new PMS functionality.

b. Given this information that seems to describe a newly identified set of signals processing a safety-related alarm function within the PMS and related PDSP indications or notifications and that these new signals that are being processed through the BPL (and ICP), state whether they are considered a "new" functionality of the PMS not previously described in the UFSAR Chapter 7, or in WCAP-16675-P? If not, explain why this is not considered newly identified functionality and where the relevant information discussing this safety-related alarm, display and/or notification information, but not a safety function, is described in either the UFSAR Chapter 7 or WCAP-16675-P? If so describe the plans for documenting this new functionality.

RESPONSE to Question 2:

a. The performance of PMS tests and the documentation of the test results, including a response time test performed under maximum CPU loading on the PMS, is required per ITAAC No. 2.5.02.11. The new PMS inputs are accounted for in the CPU loading analysis according to the PMS test plan, which is part of this ITAAC. The CPU loading test documentation for the tests captured under this ITAAC include this new signal and are available for inspection.

b. It is proposed that this new PMS capability be documented in UFSAR Chapter 7, Subsection 7.1.2.12, Safety-Related Display Instrumentation, as shown in Enclosure 16.

Southern Nuclear Operating Company

ND-17-2102

Enclosure 16

Vogtle Electric Generating Plant Units 3 and 4

Revised Revisions to Proposed Changes to Licensing Basis Documents

(LAR-17-010S4R1)

(This Enclosure consists of 2 pages, including this cover page.)

In response to inquiry from the NRC Staff, SNC herein proposes the following additional Proposed Changes to Licensing Basis Documents.

UFSAR Subsection 7.1.2.12, Safety-Related Display Instrumentation - Revise the discussion to include revised information as shown below.

Safety-related display (PMS safety display subsystem) instrumentation provides the operator with information to determine the effect of automatic and manual actions taken following reactor trip due to a Condition II, III, or IV event as defined in Chapter 15. This instrumentation also provides for operator display of the information necessary to meet Regulatory Guide 1.97. A description of the equipment used to provide this function is provided in Reference 19, Section 4.1. A description of the data provided to the operator by this instrumentation is provided in Section 7.5. A description of the equipment used to provide data to the safety-related display instrumentation is provided in Subsection 7.1.2.5.

In addition to the functions of the PMS described in Section 1 of Reference 19 and the post-accident monitoring identified above, the PMS provides water level indication for the flooding events as discussed in Subsections 3.4 and 11.2. The two sensors used for water level indication provide input into the bistable processor logic (BPL) subsystem and then into the integrated communications processor (ICP) subsystem. The information is then sent to the safety displays. These sensors are categorized as Class 1E and are on the safety-related displays for flood indication, but do not perform any safety-related function.