Vogtle PEmails

From:	Hoellman, Jordan
Sent:	Wednesday, July 12, 2017 8:25 AM
To:	Vogtle PEmails
Cc:	Reyes-Maldonado, Ruth; Patel, Chandu
Subject:	DRAFT RAI - SNC LAR 17-010, Pipe Rupture Hazard and Flooding Analyses
Attachments:	DRAFT Request for Additional Information - Vogtle LAR 17-010 (RPAC ICE MEB SEB).pdf

Attachment: Draft Request for Additional Information (RAI) in support of staff's review of SNC LAR 17-010, Pipe Rupture Hazard and Flooding Analyses.

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Draft Request for Additional Information

Vogtle Electric Generating Plant Units 3 and 4

License Amendment Request, LAR 17-010

Pipe Rupture Hazard and Flooding Analyses

Questions for Radiation Protection and Accident Consequences Branch (RPAC)

Question 1

Title 10 of the *Code of Federal Regulations* (10 CFR), 52.47(a)(8) requires that the final safety analysis report provide the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

10 CFR 50.34(f)(2)(vii) requires that applicants perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.

10 CFR 50.34(f)(2)(viii) requires that applicants provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.

License amendment request (LAR) 17-010 discusses the possibility of significant flooding events in the Auxiliary Building due to the failure of non-seismically supported fire protection piping. LAR 17-010 does not contain any information regarding how the flooding events discussed within the LAR impact access to vital areas, which may require access following accidents, or any potential impacts to the mission doses for required post-accident actions discussed in the Updated Final Safety Analysis Report (UFSAR) Subsection 12.4.1.8. The vital areas that require post-accident accessibility include access to the valve area to align spent fuel pool makeup. As shown in UFSAR Figure 12.3-2 (Sheet 6 of 15), the spent fuel pool make-up valves are located in rooms 12365 and 12354. Access to these rooms also requires the operator to transverse through room 12351. As discussed in LAR 17-010, potential flooding scenarios could occur that would result in these areas being flooded. The LAR specifically mentions that room 12365 could be flooded to a maximum flood level of approximately 108 inches. The LAR specifies that this is acceptable because the spent fuel pool level transmitters located in the room are qualified for submergence and that the isolation valves only require that their pressure boundary be maintained (as discussed previously, there is no discussion of the requirement to manually operate the valves, if needed). As a result, staff has the following question:

a) How the flooding events described in the LAR will or will not impact the ability for operators to access the spent fuel pool make-up valve alignment areas and any other vital access paths or areas? Include in the discussion a description of how operators would access these areas, if required, during the maximum flooding events described in the LAR and the additional dose that would be received in accessing areas during a design basis event and the maximum flood events.

<u>Question 2</u>

General Design Criterion (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 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 shall be designed with appropriate containment, confinement, and filtering systems, among other aspects.

Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components in Light Water Cooled Nuclear Power Plants," provides guidance to licensees and applicants on methods acceptable to the staff for complying with NRC regulations for radioactive waste management systems.

RG 1.143, Revision 2, which is referenced in the UFSAR, indicates that radwaste systems and associated components will be designed for flooding. LAR 17-010 discusses the potential for major flooding in the Auxiliary Building due to potential fire protection system piping failures. The potential flooding events include the potential for flooding portions of the Auxiliary Building that contain radwaste systems and components. However, there is no discussion regarding how these potential flooding events impact the design of the radioactive waste management systems and if the radwaste systems and components in the potentially flooded areas still meet RG 1.143, Revision 2. Please provide additional information regarding how the radwaste systems and components will continue to meet RG 1.143, Revision 2, due to the potential flooding conditions discussed in the LAR.

<u>Question 3</u>

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 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.

LAR 17-010 discusses the potential for major flooding in the Auxiliary Building due to potential fire protection system piping failures. The potential flooding events include the potential for flooding portions of the Auxiliary Building that contain radwaste systems and components. This includes the potential for significant flooding of the rail car bay which, as described in UFSAR Section 11.4, contains mobile solid waste management systems, spent resin storage tanks, high-integrity containers containing resin, and other including spent filters. The potential flooding events could result in the spread and potential release of other radioactive material due to equipment damage, radioactive sumps and drains overflowing, overflowing the holdup tanks (flood water greatly exceeds the capacity of the waste holdup tanks, where sumps are routed), high-integrity containers and other stored waste being spilled or released due to the flooding, etc. There is no discussion in the LAR of the potential radiological impacts of the internal flooding events.

RG 1.29, “Seismic Design Classification for Nuclear Power Plants,” specifies that systems that contain or may contain radioactive material and the postulated failure of which would result in conservatively calculated potential offsite doses that are more than 500 mrem total effective dose equivalent be designed Seismic Category I (RG 1.29, Section C.1.g) and that those portions of SSCs of which failure could reduce the functioning of any plant feature and result in exceeding this criteria, also be designed to Seismic Category I criteria (RG 1.29, Section C.1.i). RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” contains similar guidance for classifying systems as Quality Group C (and therefore, designing to those standards). As indicated in LAR 17-010, the fire protection piping assumed to fail is not seismically qualified and in reviewing design criteria in DCD Table 3.2-3, much of the fire protection piping in the Auxiliary Building would not meet the Quality Group C criteria. The staff has the following comments/questions:

a) Update the LAR to describe the worst case radiological release from flooding scenarios due to the possible fire protection piping failures and evaluate if the release exceeds the offsite doses described above. Explain the approach used and how the results were reached. If the potential for offsite doses exceeding the regulatory criteria exists, provide additional details about how the facility will meet the regulatory requirements (this could include descriptions of relevant design changes, etc., as appropriate).

b) If no design changes are considered and the potential for the flooding events described in the LAR still exist, describe how the design is consistent with limiting occupational and public radiation exposure ALARA, consistent with 10 CFR 20.1101(b) and minimizing contamination consistent with 10 CFR 20.1406. Include in the discussion how the design will ensure that contaminated flood water will not spread to other areas of the plant beyond those areas discussed in the LAR (e.g. through piping penetrations, ventilation ducting, etc), how the spread of contamination to the environment is minimized, how the water will be collected and treated for release, and how effluent releases will be adequately controlled.

Question for Instrumentation, Controls and Electronics Engineering Branch (ICE)

Question 4

10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit." requires, in part, that, "… a holder of a… combined license… under part 52 of this chapter, desires to amend the license or permit, application for an amendment must be filed with the Commission… fully describing the changes desired, and following as far as applicable, the form prescribed for original applications."

In addition, based upon the information provided in the LAR pertaining to the new safety-related level switches that interface with the protection and safety monitoring system (PMS) and their new safety function(s), acceptance criteria in 10 CFR 50.55a(h), "Protection and safety systems" apply. Additionally, several criteria within 10 CFR Part 50 Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 4, "Environmental and Dynamic Effects Design Basis;" GDC 13, "Instrumentation and Control;" GDC 20, "Protection System Functions;" and GDC 24, "Separation of Protection and Control Systems," apply.

In Enclosure 1 of the LAR it states in the Summary Description, in part, "The proposed changes revise the COLs to modify the design of the power plant by adding two flood-up level sensors to the Auxiliary Building radiologically controlled area (RCA). These level sensors provide main control room (MCR) notification of a rise in water level that may indicate flooding in the Auxiliary Building". Further, in the Detailed Description of Enclosure 1 it states, in part, "To alert the MCR of a potential flooding condition in the Auxiliary Building RCA, two safety-related, Class 1E, seismic Category I level sensors are proposed to be installed in the Auxiliary Building at Elevation 66'-6". A safety-related display in the MCR provides indication of the flooding situation. These sensors are safety-related and Class 1E because they are connected to the protection and safety monitoring system."

The staff requests the licensee to provide the following information:

a) Provide detailed design information describing the design function of the equipment to be installed under the amendment and its impact to the instrumentation and controls (I&C) safety system, the PMS, and the components that it controls.

b) Provide any information that discusses the addition of the new safety function(s) or a modification to an existing safety function of any safety related equipment that interfaces with the protection system.

c) Provide information related to any remote equipment used by the PMS, including sensors, final actuation devices and cabling and their environmental qualification, including submersion if necessary, that may be impacted by the newly postulated flood-up levels.

d) Does the new equipment require interfaces with non-safety-related I&C equipment? If yes, provide details of this interaction.

Question from Mechanical Engineering Branch (MEB)

Question 5

10 CFR 50, Appendix A, GDC 1 requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

LAR 17-010, Enclosure 1, Section 2, "Detailed Description," in the section titled "Auxiliary Building Level 1 (Elevation 66'-6") and Level 2 (Elevation 82'-6")," (page 7 of 40) states the following regarding containment isolation valves below the maximum flood level:

> The maximum flood level on Level 2 of the Auxiliary Building RCA reaches Elevation 85'-6" and requires the limit switches located at this elevation for outside containment isolation valves WLS-PL-V057 (Sump Containment Isolation valve), WLS-PL-V068 (Reactor Coolant Drain Tank (RCDT) Gas Containment Isolation valve), the limit switch and solenoid for CVS-PL-V047 (Letdown Flow Containment Isolation valve), and valve CVS-PL-V090 (Makeup Line Containment Isolation valve) to be qualified for operation during submergence from a MELB. Qualifying the limit switches for operation during submergence allows the switches to perform their indication function and the containment isolation valves to perform their containment isolation design function in the post-MELB condition. Other containment isolation valves below the maximum flood level are either air operated and fail closed or remain closed during safe shutdown operation. The general RCA flooding discussion in Subsection 3.4.1.2.2.2 (Containment Flooding Events) indicates that these valves fail closed or remain closed during safe shutdown operation.

LAR 17-010, Enclosure 1, Section 2, "Detailed Description," in the section titled "Plant-specific Tier 2 changes" (page 11 of 40) states the following:

> Subsection 3.4.1.2.2.2, Auxiliary Building Flooding Events, General, Radiologically Controlled Areas, 1st paragraph, regarding the containment isolation valves that are located near the containment vessel and are above elevation 82'-6", is revised to indicate that the containment isolation valves below the maximum flood level are either air operated and fail closed or remain closed during a safe shutdown operation. This

change further describes components that are located below the flood level of 85'-6" on RCA Level 2.

LAR 17-010, Enclosure 1, Section 2, “Detailed Description,” in the section titled “Licensing Basis Change Descriptions for Auxiliary Building Levels 1 and 2,” (page 13 of 40) states the following regarding revisions to Table 3.11-1, “Environmentally Qualified Electrical and Mechanical Equipment,” regarding equipment qualification for submergence:

- Is revised to indicate that the Letdown Flow Containment Isolation valve outside reactor containment CVS-PL-V047 valve limit switch (CVS-PL-V047-L) and solenoid valve (CVS-PL-V047-S1) are required to be qualified for submergence resulting from a MELB because they are below the flood level on RCA Level 2;
- Is revised to indicate that valve limit switches for the Sump Containment Isolation valve outside reactor containment WLS-PL-V057, the Reactor Coolant Drain Tank Gas Containment Isolation valve outside reactor containment WLS-PL-V068, and the Makeup Line Containment Isolation valve outside reactor containment CVS-PLV090 (WLS-PL-V057-L, WLS-PL-V068-L, and CVS-PL-V090-L, respectively) are required to be qualified for operation during submergence from a MELB because they are below the flood level on RCA Level 2; and
- Is revised to indicate that Resin Flush Containment isolation valve outside reactor containment, CVS-PL-V041 (manual valve), located on RCA Level 2 is required to be qualified for submergence resulting from a MELB.

LAR 17-010, Enclosure 1, Section 2, “Detailed Description,” in the section titled “Auxiliary Building Level 3 (elevation 100’0”) and above,” (page 17 of 40) states the following regarding submerged isolation valves Level 3 of the Auxiliary Building:

- The spent fuel pool level transmitters SFS-JE-LT019A and SFSJE-LT019C and the spent fuel cooling system isolation valves are located in Room 12365. The maximum flood level in this room is approximately 108 inches. The aforementioned safe shutdown components are located below this flood level. The spent fuel pool level transmitters SFS-JE-LT019A and SFS-JE-LT019C are qualified for submergence. The isolation valves are manual valves and only require their pressure boundary to be maintained following a PRHA event. The flood elevation in Room 12354 does not affect the structural adequacy of the adjacent floor and walls. The only safety-related equipment below the flood level in Room 12354 is the valve body for PCS-PL-V026, which is unaffected by the flooding.

The staff requests the licensee to provide the following information regarding submergence of safety-related valves:

a) Identify all safety-related valves, operators, and associated subcomponents (e.g., limit switches and solenoid valves) that are submerged or partially submerged as a result of the as-designed pipe rupture hazards analysis. Identify the type of operator (i.e., motor operator or air operator). Does UFSAR Table 3.11-1, “Environmentally Qualified Electrical and Mechanical Equipment,” identify that submergence testing is required for each valve, operator, and subcomponent? If not, provide a basis for concluding that submergence testing is not required for these valves, operators, and associated subcomponents.

b) The licensee states that containment isolation valves below the maximum flood level are either air operated and fail closed or remain closed during safe shutdown operation. Are these valves required to operate when submerged? Are these valves qualified for submergence? If not, provide a basis for concluding that submergence testing is not required for these valves and operators.

c) CVS-PL-V090 is a motor operated valve that is normally open (Tier 1 Figure 2.2.1-1) and now, due to this change, it is below the water flood level (i.e., submerged). The safety function of this valve as identified in DCD Tier 2, Table 3.9-16 is maintain close/transfer close. Therefore, please explain if this motor operated valve (including the operator) is required to operate while submerged. Is this valve qualified for submergence? If not, provide a basis for concluding that submergence testing is not required for this valve and operator.

d) The proposed revision in UFSAR Section 3.11, Table 3.11-1, "Environmentally Qualified Electrical and Mechanical Equipment," identifies equipment in the as-designed pipe rupture hazards analysis that is submerged. However, note 6 of Table 3.11-1, states that these components are <u>qualified for operation with spray</u> from a moderate-energy pipe crack or <u>spray</u> from a cold high energy pipe crack. The licensee is requested to explain the basis for stating that submerged components are qualified for operation with spray.

e) The licensee is requested to address the basis for relocating the "S" designator for submergence in Table 3.11-1 and how this affects the qualification program?

<u>Question 6</u>

LAR 17-010, Enclosure 1, Section 3, "Technical Evaluation," discusses licensing basis changes to accommodate flooding of the upper levels (page 34 of 40) and states the following:

> UFSAR Subsection 3.4.1.2.2.2, Auxiliary Building Level 5 (Elevation 135'-3"), Radiologically Controlled Area, is revised to describe this evaluation and state the only safety-related equipment below the flood level is the valve body of Containment Purge Inlet Containment Isolation Valve, VFS-PL-V003. Because valve bodies are unaffected by flooding, there is no nuclear safety or operability concerns with this flooding event.

Given the description above, it appears that the containment isolation valve body (VFS-PL-V003) is below the maximum flood level. However, it is not clear to the staff whether the position indication or associated solenoid valve are impacted by submergence of the valve body. The NRC staff requests the following additional information:

a) Are VFS-PL-V003 electrical components such as position indication or the associated solenoid valve submerged or operation of the components impacted by submergence of the valve body?

Questions for Structural Engineering Branch (SEB)

Question 7

10 CFR, Part 50, Appendix A, GDC 4, requires that structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power units.

10 CFR, Part 50, Appendix A, 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.

Consistent with Standard Review Plan Section 3.8.4, the staff reviews the descriptive information, including plans and sections of each structure, to establish that there is sufficient information to define the primary structural aspects and elements relied upon for the structure to perform the intended safety function.

Staff reviewed LAR 17-010, submitted by Southern Nuclear Operating Company (SNC). As a result of the staff review, need for additional information was identified in the following areas to complete the safety evaluation. The staff requests that responses to the following structural engineering questions be incorporated in the LAR:

a) Provide a visual characterization of the area affected by the flooding including the current wall thickness and the height to which the flood water is expected to rise.

b) Explain using the load combinations that govern the wall design, how the existing wall thickness was re-evaluated to ensure that the new demand was accommodated by the existing wall capacity.

c) Provide configuration and mounting details of the flood relief louver installed in the wall.

d) Provide minimum distance between the NI structures and the new tanks in the yard to prevent external flooding and distances between objects with II/I interaction consideration.

e) Where will the flood water collected from the RCA be stored? If the collected water is stored in the auxiliary building or any other adjacent building, provide a design for the storage of this contaminated water and explain how this additional weight was considered in the building design.

The staff requests that the applicant in responding to this RAI take into consideration factors that may need to be considered from responses to other RAIs in the LAR.