Vogtle PEmails

From: Kallan, Paul
Sent: Tuesday, January 30, 2018 1:58 PM
To: Vogtle PEmails
Subject: VOG-RAI-Letter for LAR 17-023.docx
Attachments: VOG-RAI-Letter for LAR 17-023.docx

Hearing Identifier: Vogtle_COL_Docs_Public
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Mail Envelope Properties (CY1PR09MB0987D552AF89675A65E5618AF6E40)

Subject: VOG-RAI-Letter for LAR 17-023.docx
Sent Date: 1/30/2018 1:58:01 PM
Received Date: 1/30/2018 1:58:05 PM
From: Kallan, Paul

Created By: Paul.Kallan@nrc.gov

Recipients:
"Vogtle PEmails" <Vogtle.PEmails@nrc.gov>
Tracking Status: None

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Files	Size	Date & Time
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January 30, 2018

Mr. B.H. Whitley, Director
Regulatory Affairs
Southern Nuclear Operating Company, Inc.
42 Inverness Center Parkway, B237
Birmingham, AL 35242

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 2 RELATED TO IMROVEMENTS TO MAIN CONTROL ROOM (MCR) POST-ACCIDENT RADIOLOGICAL CONSEQUENCES FOR THE VOGTLE ELECTRIC GENERATING PLANT UNITS 3 AND 4 COMBINED LICENSES (TAC NO. RP9638)

Dear Mr. Whitley:

By letter dated August 31, 2017 (ADAMS Accession NO. ML17243A351), Southern Nuclear Operating Company (SNC) requested an amendment to Combined License Numbers NPF-91 and NPF-92 for Vogtle Electric Generating Plant Units 3 and 4 respectively. The requested amendment proposes to depart from Tier 2 information in the Updated Final Safety Analysis Report (UFSAR) (which includes the plant-specific DCD Tier 2 information) and involves related changes to plant-specific Tier 1 (and associated COL Appendix C) information, and COL Appendix A Technical Specifications. Specifically, the requested amendment proposes changes to the plant-specific nuclear island non-radioactive ventilation system (VBS), the main control room emergency habitability system (VES), and post-accident operator dose analyses. These changes are proposed to maintain compliance with General Design Criterion (19), which requires that main control room (MCR) personnel dose does not exceed 5 rem total effective dose equivalent (TEDE) for the duration of a design basis accident (DBA).

In the course of reviewing your request the NRC staff has identified the need for additional information. The request for additional information (RAI) is enclosed. Please respond to this RAI within 30 days of receipt of this letter.

If you have any questions or comments concerning this matter, you may contact me at 301-415-2809 or Paul.Kallan@nrc.gov.

Sincerely,

/RA/

Paul Kallan, Senior Project Manager
Licensing Branch 4
Division of New Reactor Licensing
Office of New Reactors

Docket Nos. 52-025
52-026

Enclosure:
Request for Additional Information 2

CC: see next page

If you have any questions or comments concerning this matter, you may contact me at 301-415-2809 or Paul.Kallan@nrc.gov.

Sincerely,

/RA/

Paul Kallan, Senior Project Manager
Licensing Branch 4
Division of New Reactor Licensing
Office of New Reactors

Docket Nos. 52-025
52-026

Enclosure:
Request for Additional Information 2

CC: see next page

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DATE	01/30/18	1/30/18	1/30/18

*Approval captured electronically in the electronic RAI system.

Request for Additional Information 2

Issue Date: 01/30/2018
Application Title: VEGP Units 3 and 4 - LARs
Operating Company: Southern Nuclear Operating Co.
Docket No. 52-025 and 52-026
Review Section: 06.04 - Control Room Habitability System
Application Section:

QUESTIONS

06.04-1

Regulatory Basis

10 CFR 52.79(a)(1)(vi) requires that a combined license (COL) application final safety analysis report (FSAR) provides a description and safety assessment of the site on which the facility is to be located. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.79(a)(1)(vi)(A) and 52.79(a)(1)(vi)(B). The radiological consequences of design basis accidents are evaluated against these offsite dose criteria.

10 CFR 52.79(a)(4) requires a COL application FSAR that describes the design of the facility including the principal design criteria for the facility, for which the AP1000 used the 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." For a COL application received after January 10, 1997, General Design Criterion (GDC) 19 requires that a control room be provided with adequate radiation protection to permit access and occupancy of the control room under accident conditions without the personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors,” gives guidance on review of design basis accident (DBA) radiological consequence analyses. In addition, Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” gives guidance on DBA dose analysis inputs, assumptions, and models.

Key Issue 1:

On page 23 of 60 of Enclosure 1 of the license amendment request (LAR), item O states that the licensee is clarifying that for the fuel handling accident (FHA), the depth of water above a dropped fuel assembly could be reduced below the licensing basis DBA dose analysis assumption of 23 feet currently reported in the updated final safety analysis report (UFSAR). The LAR goes on to assert that

…continued conformance with Regulatory Guide 1.183 is clarified by confirming that a minimum water height of 18.6 feet above a postulated dropped fuel bundle is needed to support an overall iodine decontamination factor of 200 as specified in Appendix B of Regulatory Guide 1.183.”

In apparent contradiction to the licensee's statement, RG 1.183, Appendix B, Position 2, "Water Depth," states:

If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method.

The LAR does not provide further discussion of this topic.

Question 1

To facilitate staff understanding of the application information, sufficient to make appropriate regulatory conclusions with respect to compliance with 10 CFR 52.47(a)(1) and GDC 19, the staff requests that the licensee:

- Provide an assessment of the iodine decontamination factor applicable to a pool water depth of 18.6 feet above fuel damaged in the design basis FHA, including the basis for the method used to determine the overall effective iodine decontamination factor. If the overall effective iodine decontamination factor is determined to be less than that used in the current license basis, revise the DBA FHA dose analysis and provide a description of the revised analysis, including dose results at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and in the MCR and technical support center (TSC). Note that RG 1.183 mentions that an overall effective iodine decontamination factor of 200 is appropriate for a water depth of 23 feet or greater and the depth cited in the LAR is only 18.6 feet.

06.04-2

Regulatory Basis

10 CFR 52.79(a)(1)(vi) requires that a combined license (COL) application final safety analysis report (FSAR) provides a description and safety assessment of the site on which the facility is to be located. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.79(a)(1)(vi)(A) and 52.79(a)(1)(vi)(B). The radiological consequences of design basis accidents are evaluated against these offsite dose criteria.

10 CFR 52.79(a)(4) requires a COL application FSAR that describes the design of the facility including the principal design criteria for the facility, for which the AP1000 used the 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." For a COL application received after January 10, 1997, General Design Criterion (GDC) 19 requires that a control room be provided with adequate radiation protection to permit access and occupancy of the

control room under accident conditions without the personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors,” gives guidance on review of design basis accident (DBA) radiological consequence analyses. In addition, Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” gives guidance on DBA dose analysis inputs, assumptions, and models.

Key Issue 2:

New UFSAR Tier 2 Table 12.2-29, “Core Melt Accident Integrated Source Strengths from MCR HVAC Filters,” includes the same information as previously reviewed and approved in the staff safety evaluation report for Levy Nuclear Plant Units 1 and 2 FSER Chapter 21 – “Design Changes Proposed in Accordance with ISG-11,” (ADAMS Accession Number - ML16068A418), as noted in LAR Enclosure 1, item A, “Changes Impacting MCR Dose for Design Basis Accidents.” Although new UFSAR Table 12.2-29 information was previously approved in the Levy COL review, the nuclear island non-radioactive ventilation system (VBS) filter loading was based on a filter efficiency of 90% for elemental and organic iodine, whereas the LAR proposes to increase the credited filter efficiency for the VBS from 90% to 99% for elemental and organic iodine (see additional change item K, “VBS Inleakage Optimization,” on pages 18 and 19 of LAR Enclosure 1).

Question 2

To facilitate staff understanding of the application information, sufficient to make appropriate regulatory conclusions with respect to compliance with GDC 19, the staff requests that the licensee:

- Clarify whether the proposed change in VBS filter efficiency would change the VBS filter loading source strengths used to evaluate the direct dose in the MCR from VBS filter shine. If so, please provide the revised source strength information and make the necessary changes to the MCR direct dose (filter shine) dose analysis. Also revise the associated information in the UFSAR, including new UFSAR Table 12.2-29, along with descriptions of the filter shine dose analyses and the total MCR dose analyses and results.

06.04-3

Regulatory Basis

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 licensees 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.

NUREG-0737, "Clarification of TMI Action Plan Requirements," and NUREG-0800 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) section 12.3-12.4 provide additional guidance on acceptable methods of meeting these requirements. These documents indicate that post accident radiation zones should consider access to, stay time in, and egress from these vital areas. NUREG-0737 specifies that any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is to be designated as a vital area. As specified, the plant should be designed so that the dose to an individual should not exceed the occupational dose criteria to perform the vital missions, including accessing and egressing from the areas.

Background

The licensee's submittal, "Request for License Amendment and Exemption Regarding Improvements to Main Control Room (MCR) Post-Accident Radiological Consequences (LAR 17-023)," dated August 31, 2017, provided a revised methodology for calculating doses to operators following an accident. LAR 17-023 Enclosure 4, "Proposed Changes to the Licensing Basis Documents, (withheld Information, in accordance with 10 CFR 2.390(d)), ND-17-1297) depicts the personnel travel path from the Annex Building into the "Elec. Penet. Room Division A", room identification Number 12412. Room 12412 is identified as a Radiation Zone VII. AP1000 DCD Tier 2 Revision 19 Figure 12.3-2 (Sheet 1 of 15) "Radiation Zones, Post-Accident Legend," identifies that the maximum design dose rate in a Radiation Zone VII area is ≤ 100 Rem/hour (hr). This travel path is not shown on AP1000 DCD Tier 2 Revision 19 Figure 12.3-2 (Sheet 7 of 15) "Radiation Zones, Post-Accident Nuclear Island, Elevation 117'6".

Key Issue 3:

LAR-17-023 appears to identify a previously unidentified post-accident mission travel path. Because of the dose rates that may be present in a radiation zone VII area, a small amount of time in the area could challenge the applicable radiation exposure limits. The amount of time is dependent on the travel time to the area, the type of task to be performed, the assumed radiological conditions (e.g., airborne activity,) and the time needed to exit the area.

Question 3

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to compliance with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f)(2)(vii), the staff requests that the licensee:

- For this new post-accident mission travel path, explain how the LAR addresses the specific requirement to 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,
- Explain/Justify the apparent change to the post-accident mission travel path, including the purpose of the mission, how many times the mission may need to be performed,
- Describe the evaluation of the mission dose and results including the methods, models and assumptions, used for determining the exposure estimate,
- As necessary, revise the Vogtle Units 3 & 4 UFSAR to include the description of the apparent mission and other relevant information,

OR

- Provide the specific alternative approaches used and the associated justification.