



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

October 18, 2017

Mr. James J. Hutto  
Regulatory Affairs Director  
Southern Nuclear Operating Co., Inc.  
P.O. Box 1295, Bin 038  
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 – AUDIT PLAN  
RE: SYSTEMATIC RISK-INFORMED ASSESSMENT OF DEBRIS TECHNICAL  
REPORT (CAC NOS. MF9685 AND MF9686; EPID L-2017-TOP-0038)

Dear Mr. Hutto:

By letter dated April 21, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17116A096) as supplemented by letter dated July 11, 2017 (ADAMS Accession No. ML17192A245), Southern Nuclear Operating Company submitted a plant-specific technical report for Vogtle Electric Generating Plant, Units 1 and 2, and requested U.S. Nuclear Regulatory Commission (NRC) approval of the methods and inputs described in the technical report. The plant-specific technical report describes a risk-informed methodology to evaluate debris effects with the exception of in-vessel fiber limits.

The NRC staff will conduct a regulatory audit to support its review of the technical report. The audit will be conducted at ENERCON Services, Inc.'s offices in Albuquerque, NM on October 24-26, 2017. The audit plan is enclosed. The logistics and scope of the audit was discussed with your staff on October 5, 2017.

If you have any questions, please contact me by telephone at 301-415-2871 or by e-mail at [Michael.Marshall@nrc.gov](mailto:Michael.Marshall@nrc.gov).

Sincerely,

A handwritten signature in black ink that reads "Michael L. Marshall, Jr." with a stylized flourish at the end.

Michael L. Marshall, Jr., Senior Project Manager  
Plant Licensing Branch I  
Division of Operating Reactor Licensing  
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosure:

1. Audit Plan
2. Audit Information Needs

## AUDIT PLAN

### GENERIC SAFETY ISSUE 191

#### SOUTHERN NUCLEAR OPERATING COMPANY

#### VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

#### DOCKET NOS. 50-424 AND 50-425

### **I. BACKGROUND**

By letter dated April 21, 2017 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML17116A096), as supplemented by letter dated July 11, 2017 (ADAMS Accession No. ML17192A245), the Southern Nuclear Operating Company (SNC) submitted a plant-specific technical report for Vogtle Electric Generating Plant, Units 1 and 2 (VEGP) and requested U.S. Nuclear Regulatory Commission (NRC) approval of the methods and inputs described in the technical report. The plant-specific technical report provides plant-specific conditions and models related to Generic Safety Issue (GSI) - 191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance" and Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," (ADAMS Accession No. ML042360586). The technical report also provides risk quantification, a description of a plant-specific probabilistic risk assessment model, and defense-in-depth and safety margin evaluations.

### **II. REGULATORY AUDIT BASES**

The purpose of the audit is to gain a more detailed understanding of the analyses performed by SNC to resolve GSI-191 through a risk-informed approach. The main focus of the audit will be the risk-informed approach and related computations that are documented in Enclosure 3 to SNC's letter dated April 21, 2017. Additional audit discussions will cover the topics contained in the submittals dated April 21, 2017 and July 11, 2017. The objectives of the audit are to (1) gain a better understanding of the technical approaches implemented in support of the risk-informed methodology, (2) gain a better understanding of the methods used in the computer models, (3) identify related verification and validation activities, and (4) gain a better understanding of differences between weld break locations where debris generation and transport may result in exceeding a limit that would result in a core damage sequence.

At the end of the audit, the audit team expects to have a more complete understanding of the features that determine the set of welds that cause failure. The audit team expects to confirm "near miss" locations (i.e., those breaks with low margin to failure) were classified correctly.

### **III. REGULATORY AUDIT SCOPE AND METHODOLOGY**

The scope of the audit includes key components of the risk-informed methodology, specifically:

- Approaches to compute the amount of debris generated and transported to the strainer(s) given a break of a specified size, at a specified weld location, a range of

orientations, and the operating state of the plant, starting from the computer-aided design model of the nuclear power plant.

- Approaches to compute the conditional failure probabilities.
- Approaches to compute the core damage frequency using break frequencies in NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," April 2008, Volumes 1 and 2 (References 1 and 2, respectively), and the NRC initiating event database for small LOCAs.
- Approaches to validate the application of the Break Accident Debris Generation Evaluator (BADGER) and Nuclear Accident Risk Weighted Analysis (NARWHAL) computer codes as applied to VEGP.
- Methods used to ensure the computer assisted design (CAD) model and BADGER as described in the submittal accurately represents the as-built, as-operated plant.
- Methods used by the licensee to model the generation, transport, and effects of debris are consistent with approved methodologies (e.g., NRC staff's safety evaluation for Nuclear Energy Institute (NEI) 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology" (References 3 and 4)).
- Modelling of plant conditions to provide a realistic or conservative estimation of the effects of debris on recirculation.
- Use of guidance reports (see Section XI, "References" of this audit plan) in the computation of debris amounts and transported amounts.
- Methods used to verify and validate the risk-informed computations, including verification and validation documentation.

To accomplish these objectives, the audit team will use BADGER and NARWHAL computer models through the assistance of licensee staff, review documentation and assumptions, discuss questions with the licensee technical experts, and review computational results. The audit team will require assistance from licensee technical experts who can identify and walk through a series of computations (i.e., examples), starting from visual representations (i.e., CAD model) of the nuclear power plant system that to core damage frequency (CDF) and large early release frequency. The examples should identify weld locations where LOCAs are postulated, amounts of debris generated, and amounts of debris transported to the strainer and the reactor. Each example should continue through the calculation of CDF and explain the application of the methodology. The examples should also demonstrate how to operate the user interface of the software.

#### **IV. INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT**

The information needed for the regulatory audit is listed in Enclosure 2. The audit team will not remove non-docketed information from the audit site. NRC contractors will maintain control of proprietary materials in accordance with NRC procedures and non-disclosure agreements.

## **V. AUDIT TEAM ASSIGNMENTS**

The members of the audit team will be:

- Victor Cusumano, team lead, NRC
- Michael Marshall, project manager, NRC
- Candace De Messieres, technical reviewer, probabilistic risk assessment, NRC
- Steve Smith, technical reviewer, debris generation/transport, NRC
- Paul Klein, technical reviewer, chemical effects, NRC
- Matt Yoder, technical reviewer, coatings and chemical effects, NRC
- Osvaldo Pensado, NRC contractor
- Stuart Stothoff, NRC contractor

## **VI. LOGISTICS**

The NRC staff and NRC's contractor will conduct the audit on October 24 - 26, 2017, in the offices of ENERCON Services, Inc. in Albuquerque, NM, or other locations agreed upon by the licensee and NRC staff that facilitates access to the licensee's computer models, documentation, and technical experts performing the work on GSI-191 resolution. The NRC Project Manager will coordinate any changes to the audit schedule and location with the licensee.

## **VII. SPECIAL REQUESTS**

The NRC staff would like access to the following equipment and services:

- Telephone with a speaker or speaker phone.
- Enclosed conference room (or comparable space) with a table, chairs, and white board.
- A projector and screen.
- Wireless internet access (if available in the work space).
- A computer running the version of the NARWHAL and BADGER software used to support the VEGP systematic risk-informed assessment of debris technical report.

## **VIII. DELIVERABLES**

An audit summary will be prepared within 90 days of the completion of the audit. If information evaluated during the audit is needed to support a regulatory decision, the NRC staff will identify it in a request for additional information. The NRC staff, if needed, will provide the request for additional information to the licensee in separate docketed correspondence.

## **IX. REFERENCES**

- (1) U.S. Nuclear Regulatory Commission, NUREG-1829, Volume 1, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process, Main Report," April 2008 (ADAMS Accession No. ML082250436).
- (2) U.S. Nuclear Regulatory Commission, NUREG-1829, Volume 2, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process, Appendices A through M," April 2008 (ADAMS Accession No. ML081060300).

- (3) Nuclear Energy Institute, NEI 04-07, Volume 1, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December 2004 (ADAMS Accession No. ML050550138).
- (4) Nuclear Energy Institute, NEI 04-07, Volume 2, "Pressurized Water Reactor Sump Performance Evaluation Methodology: Safety Evaluation by the Office of Nuclear Reactor Regulation re to NRC Generic Letter 2004-02, Revision 0," December 6, 2004 (ADAMS Accession No. ML050550156).
- (5) U.S. Nuclear Regulatory Commission, "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses," November 2007 (ADAMS Accession No. ML073110278).

## AUDIT INFORMATION NEEDS

### Documentation

- 1) Calculations of the amount of debris generation and amount of debris transport for select sample weld locations (see the table in Other Information Needs No. 7).
- 2) Documentation of any verification(s) performed on the NARWHAL code for debris mass balance computations.
- 3) NARWHAL User Manual.

### Other Information Needs

- 1) Demonstrate how BADGER and NARWHAL computes the amount of debris, including pipe and equipment insulation, fire barrier material, and concrete and steel coatings surfaces for select sample weld locations (see the table in Other Information Needs No. 7).
  - a) Include examples of the presence of robust barriers to limit the zone of influence.
  - b) Provide comparisons of the CAD model to other methods used to validate its completeness and accuracy.
  - c) Describe how the information is imported and used in NARWHAL.
  - d) Present documentation of the verification efforts to demonstrate that the NARWHAL computations are reasonably complete and accurate, such as benchmarking using hand or other methods based on a double-ended guillotine break (DEGB) at the weld.
- 2) Discuss benchmarking using a DEGB at the weld location 11201-053-1-RB or another illustrative weld to demonstrate that BADGER and NARWHAL computations to determine debris amounts are reasonably accurate.
- 3) Discuss how the transport fractions, including erosion into fines, are used to compute the potential amount of debris that could be transported to the screens and reactor core. In particular, explain the assumptions and methodology used for the determination of erosion of large and small fibrous debris retained in the pool and above the pool, and on the strainer, if applicable.
  - a) Describe how this is implemented in NARWHAL.
  - b) Explain why some cases result in more debris on one residual heat removal or containment spray strainer while some cases result in balanced loading between strainers in the A and B trains.
  - c) Explain why some cases result in different amount of unqualified coatings transport as compared to other cases.
- 4) Discuss the approach to establish strainer failure given a break, the different failure criteria (flashing, deaeration, excess differential pressure (structural or net positive suction head, etc.)), and how this determination is made on a per strainer basis.
  - a) Clarify if credit is taken for one train of emergency core cooling system remaining operational after failure of another train.

- b) Discuss the incorporation of key uncertainties affecting the amount of debris, including chemical effects.
  - c) Provide plots of debris type and size mass at the strainer versus break size and orientation for a number of weld locations, and provide comparisons of these plots to the strainer failure criteria.
- 5) Provide examples that demonstrate the calculation of the CDF using the NARWHAL conditional failure probabilities (based on counting the number of simulated breaks causing failure), pump state probabilities, and integration with the probabilistic risk assessment model.
- 6) Discuss the methodology used to calculate the mass balance on the strainer and in the core for both cold-leg and hot-leg breaks.
- a) Discuss how pre-defined and different transport fractions for strainers are treated in the model (considering that strainers are assumed to experience identical flows).
  - b) Discuss key model and parameter uncertainties (e.g., fiber penetration experiments and their application to the plant configuration, chemical effects timing, etc.).
  - c) Discuss competing assumptions regarding debris buildup on strainers and in the reactor core (e.g., assumed high pump flows enhance buildup on the strainer at the expense of the core and vice versa).
  - d) Demonstrate that the adopted treatment and assumptions do not underestimate strainer and in-vessel debris amounts, associated conditional failure probabilities, and associated core failure frequencies.
- 7) For items 1 through 6 above, consider welds (i.e., break locations) that both do and do not result in failure. A list of welds of interest is provided in the table below.

List of Welds of Interest		
Inner diameter (inches)	Breaks with Failure	Breaks without Failure
12.814	11201-053-1-RB (SG)	11201-004-2-RB (SG)
27.5	11201-009-8-RB (Cavity)	11201-012-8-RB (Cavity)
29	11201-001-3-RB (SG)	All 29 inch DEGBs result in failure. All are hot-leg breaks with containment spray.
31	11201-008-2-RB (SG)	All 31 inch DEGBs result in failure. All are located in the SG compartments.
VEGP may select different welds that result in failures in different criteria or those that are considered to be more illustrative of various aspects of the analysis.		

- 8) Discuss the differences in the sensitivity cases examined in Table 3-17 of Enclosure 3 of the letter dated April 21, 2017 (e.g., the continuum break model vs. the DEGB only model, various hybrid LOCA frequency allocation methodologies, etc.).

- 9) Discuss the chemical effects methodology, including:
  - a) Head loss stabilization criteria between additions of chemical debris batches during strainer head loss testing.
  - b) Credit taken for phosphate inhibition of aluminum corrosion and sensitivity studies showing the impact of credit for phosphate inhibition.
  - c) Comparison of the Vogtle aluminum release model to the autoclave results from WCAP-17788-NP, Volume 1, Revision 0, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," July 2015 (ADAMS Accession No. ML15210A669).
- 10) Provide a walkthrough of the large early release frequency calculations, including an explanation of how random failures and unavailability of the containment fan coolers were modeled.
- 11) Discuss the test report theoretical bed thickness calculations for the thin-bed test. On page 103 of Calculation No. ALION-CAL-SNC-7410-005, Revision 1 "Head Loss Testing of a Prototypical Vogtle 1 and 2 Strainer Assembly," August 13, 2015 (ADAMS Accession No. ML15293A187), the bed thickness is listed as 0.625 inches. The NRC staff calculated that the thickness was 0.57 inches.
- 12) Discuss the extrapolation methodology that is described beginning on page 107 of Calculation No. ALION-CAL-SNC-7410-005. Specifically, please clarify what is meant by:

[...] the measured head loss values plotted in Figure 4.3-1 will be reduced, based on the ratio of the maximum full calcium phosphate head loss as compared to the flow sweep chemical head loss, in order to perform the same temperature corrections for the full calcium phosphate head loss values.
- 13) Enclosure 2, Table 3.g.16-1, shows that the residual heat removal pumps have significant margins for net positive suction head. Provide the dominant failure modes that result in increases in delta CDF.
- 14) Table 3-16 and Table 3-17 in Enclosure 3 of the submittal list how alternative models were evaluated to address uncertainty in the aluminum metal release equation. Provide additional details related to the sensitivity study that evaluated aluminum release without phosphate inhibition credited. For example, state if a multiplication factor was applied to the aluminum release equation from WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191" (ADAMS Accession No. ML081150379).

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\* via email

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