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Vogtle Electric Generating Plant Units1&2
Systematic Risk-Informed Assessment of Debris Technical Report SNC Response to NRC Request for Additional Information (RAIs #37-39)

Ladies and Gentlemen:

By letter dated April 21, 2017 (Agencywide Documents Access and Management System Accession No. ML17116A098) as supplemented by letters dated July 11, 2017; November 9, 2017; January 2, 2018; January 9, 2018; February 6, 2018, February 12, 2018 and February 21, 2018; Southern Nuclear Operating Company, Inc. (SNC) submitted a plant-specific technical report for Vogtle Electric Generating Plant (VEGP), 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. By letter dated May 1, 2018, the NRC staff notified SNC that additional information is needed for the staff to complete their review. The Enclosure provides the SNC response to the NRC requests for additional information (RAIs) for RAIs 37-39.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 23 day of May 2018.

Respectfully submitted,

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Director, Regulatory Affairs
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CAG/PDB/SCM

Enclosure: SNC Response to NRC Request for Additional Information

cc: Regional Administrator, Region II
NRR Project Manager – Vogtle 1 & 2
Senior Resident Inspector – Vogtle 1 & 2
State of Georgia Environmental Protection Division
RType: CVC700

Vogtle Electric Generating Plant - Units 1 & 2
Systematic Risk-Informed Assessment of Debris Technical Report SNC Response to NRC Request for Additional Information (RAIs #37-39)

Enclosure

SNC Response to NRC Request for Additional Information (RAIs)

NRC RAI 37

Paragraph (b) of Title 10 of the Code of Federal Regulations, Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires, in part, that an emergency core cooling system (ECCS) be provided for long-term cooling after successful initial operation of the ECCS. Water for long-term cooling is recirculated through the plant's sump strainer. GL 2004-02 contained a request that licensees provide verification that the sump screens (i.e., sump strainers) are capable of withstanding the loads imposed by the accumulation of debris and pressure differentials caused by blockage under flow conditions. Item 3.k of the NRC staff's revised content guide for GL 2004-02 supplement responses (ADAMS Accession No. ML073110389) requests licensees summarize the structural qualification results and design margins for various components of the sump strainer structural assembly.

Table 3.k.1-3 of Enclosure 2, states that the crush pressure on the strainer due to suction strainer operation is equivalent to 10.1 ft. of head loss. This pressure is used in the load combinations for the structural analysis of the strainer. However, several locations in Enclosure 2 (e.g., Tables 3.f.14-1 and 3.g.16-1) identify strainer head loss values greater than 10.1 ft. and the supplemental response to GL 2004-02 item 3.f.7 notes that the strainer structural margin is 24 ft. It is not clear what the head loss limit is for the strainer.

a. Please identify the head loss limit for structural qualification of the strainer and explain how this value was determined.

b. If the value does not bound all postulated head loss values, please provide a justification for any exceedances.

SNC Response to RAI 37

a. The results and methodology presented in Sections 3.k.1 and 3.k.2 of Enclosure 5 (Reference 1) are based on the structural qualification performed by the hardware vendor, GEH, prior to the strainer installation. The head loss limit used for the structural qualification (referred to as crush pressure hereafter) presented in Table 3.k.1-3 (Reference 1 pp. E5-116) was an assumed value based on what was believed to be adequate at that time. As part of the change to a risk-informed methodology, GEH revisited the stress model and results to determine if a higher crush pressure could be justified. As shown in Table 3.k.2-2 of Enclosure 5 (Reference 1 pp. E5-120), the most limiting stress is in the welds between the perforated plate to finger and the perforated plate to frame for the Service Level D Load combination. Linear scaling was used conservatively assuming that total stress is scalable with crush pressure. The resulting max allowable crush pressure is 10.7 psi for the plate to finger weld condition, and 5.03 psi for the plate to frame weld. However, this approach is too conservative and not appropriate for the plate to frame weld. A single disk FEA model was run using ANSYS with the same mesh as the original analysis and determined that the impact of the crush pressure has

very limited impacts to the stress of the plate to frame weld. Using this new model, the allowable crush pressure for the plate to frame weld was calculated to be 30.47 psi. This concluded that the limiting factor for the crush pressure was the plate to finger weld which has a max allowable crush pressure of 10.7 psi.

Using ASME Code Section III Level D allowables, and the methodology above, the crush pressure was determined to be 10.39 psi for a 15-disk strainer, compared to 10.7 psi for the 18-disk strainer. The lower crush pressure for the 15-disk strainer is due to the larger debris weight for each disk (same total debris weight with fewer disks). Therefore, a crush pressure of 10.39 psi (24.0 ft) was conservatively used for the final configuration of 16 disks, as shown in Section 3.f.7 of the submittal (Reference 1 pp. E5-73).

b. The crush pressure of 10.39 psi (24.0 ft) bounded all postulated head loss values, as shown in Table 3.g.16-1 (Reference 1 pp. E5-103).

NRC RAI 38

Title 10 of the Code of Federal Regulations, Section 50.46, Subsection (a)(1) requires in part that cooling performance be calculated with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents (LOCA) of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Additionally, the subsection requires that the evaluation includes sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. In its risk-informed methodology, the licensee used guidance and acceptance guidelines described in Regulatory Guide (RG) 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (ADAMS Accession No. ML 100910006).

Section 2.3.3 of RG 1.174, Rev. 2, states that:

> ...technical adequacy will be understood as being determined by the adequacy of the actual modeling and the reasonableness of the assumptions and approximations.

Enclosure 1, Section 2.0, of the submittal states that a screening process can be used to eliminate scenarios that are not relevant, not affected by debris, or have an insignificant contribution. The results of this screening process are provided, but not a description (i.e., justification) of the systematic approach implemented to identify relevant initiating events and how scenarios are eliminated.

a. Please describe in detail the process leading to the identification of relevant internal initiating events (e.g., LOCA, open safety relief valve, water hammer-induced LOCAs, non-piping LOCAs). Please include any criteria (quantitative or qualitative) used in the process for screening (i.e., eliminating) any initiating events or scenarios.

b. Please describe in more detail how the high-likelihood scenarios were determined and how the change in risk associated with low likelihood scenarios were determined for this application. Please include a summary of the process used to make these determinations.

c. Please describe in detail the systematic process applied to evaluate the impact of secondary side breaks. Please include a summary of how and why these breaks were screened in or out.

d. Please explain which secondary side breaks were screened from detailed analyses, and the basis for their screening.

SNC Response to RAI 38

a. A systematic process was used to determine the hazards, initiating events, and operating modes to be addressed in the Vogtle GSI-191 analysis. The process was based on the identification of hazards and initiating events with the potential to (1) generate debris inside containment, (2) require sump recirculation for mitigation of the event, and (3) result in debris transport to the containment sump. Hazards or initiating events that do not meet these three criteria were excluded from the analysis.

   Among internal plant hazards, the following initiating events do not have the potential to generate debris inside containment and were screened from the analysis:

   - transients,
   - steam generator tube rupture,
   - inadvertent safety injection,
   - inadvertent or stuck-open power operated relief valves (PORVs) that discharge to the pressurizer relief tank (PRT),
   - secondary side breaks outside containment, and
   - interfacing systems loss of coolant accidents (LOCAs) that discharge outside containment

   The internal initiating events that do have the potential to generate debris inside containment are LOCAs (small, medium, and large) due to breaks inside containment and secondary side breaks inside containment.

   Internal flood hazards do not have the potential to generate debris inside containment. Pipe breaks that flood inside containment are evaluated as LOCA or secondary side break internal events.

   The internal hazards and initiating events identified above that have the potential to generate debris inside containment may also require sump recirculation for mitigation of the event and result in debris transport to the containment sump. Therefore, the following

events were included in the scope of the Vogtle GSI-191 analysis and considered with a detailed or conservative quantitative assessment or a qualitative evaluation:

1. Large, medium, and small LOCAs due to:
    i. Pipe breaks
    ii. Failure of non-piping components
    iii. Water hammer
2. Secondary side breaks inside containment that result in a consequential LOCA upon failure to terminate safety injection or a stuck open PORV, requiring sump recirculation.

b. The Vogtle PRA model was used to identify the high-likelihood equipment configurations that can occur in response to a LOCA, to focus the analysis of debris generation, transport, and resulting GSI-191 phenomena. The identification of high-likelihood equipment configurations followed a systematic process:

1. All possible combinations of system/train failures that affect the likelihood of debris-induced failure of ECCS following a LOCA were identified. The configurations considered were specific to each LOCA size (small, medium, or large). Examples of such configurations include successful operation of all ECCS equipment (e.g., RHR, containment spray, and containment coolers following a large LOCA), failure of one train of RHR, failure of both trains of containment spray, etc.

2. For each possible configuration, the functional failure probability (FFP) was quantified for each system/train failed in that configuration, using the associated logic in the PRA model. Those FFP values were then used to calculate total probability for each equipment failure combination.

3. For each possible configuration identified for each LOCA size, the annual scenario frequency was calculated based on the associated initiating event frequency and total probability of the equipment failure configuration. The scenario frequencies for each LOCA size were summed and the significant scenarios were identified as those that comprised 95%, or individually contributed 1%, of the total frequency. The equipment failure combinations from those significant scenarios represent the high-likelihood configurations.

The high-likelihood equipment configurations were included in the detailed NARWHAL analysis for GSI-191 effects and were explicitly modeled in the PRA for quantification of the risk impacts due to GSI-191 phenomena.

The remaining low-likelihood configurations also have a risk impact from GSI-191 phenomena, but were not evaluated in detail. Instead, a bounding risk impact was

calculated for the low-likelihood configurations. Calculation of the bounding risk impact for the low-likelihood configurations also followed a systematic process, as described below.

1. Because NARWHAL analysis was not performed to determine sump strainer or core cooling conditional failure probability (CFP) values for all low-likelihood configurations, a representative or bounding CFP was selected for each low-likelihood configuration using the following logic:

    i. Each low-likelihood configuration was binned based upon the impact on sump strainer debris accumulation, relative to the high-likelihood configurations evaluated in NARWHAL. For example, the failure of a containment cooler was captured by bounding temperature profiles used in the NARWHAL analysis, implicitly assuming the containment cooler failed. Therefore, low-likelihood configurations with containment cooler failure (e.g., RHR train A and containment cooler A failed) were considered equivalent to the high-likelihood configuration with only RHR train A failed.

    ii. Based on the binning performed, the appropriate CFP was selected to use as a representative or bounding value for the low-likelihood configuration. For example, the low-likelihood configuration with one RHR train failed and one CS train failed, has the same impact on sump strainer accumulation as the high-likelihood configuration with one ECCS train failed due to loss of nuclear service cooling water (NSCW). Therefore, the CFP calculated for one ECCS train failure was applied to the configuration with one RHR and one CS train failed.

    iii. If there was no similar high-likelihood configuration for a low-likelihood configuration, such as failure of one RHR train and both CS trains, a CFP of 1.0 was assumed.

2. For each low-likelihood configuration identified for each LOCA size, the bounding annual core damage frequency (CDF) was calculated based on the associated initiating event frequency, probability of the equipment failure configuration, and the selected representative or bounding CFP. The bounding CDF values from the low-likelihood configurations for each LOCA size were then summed. The base case CDF (assuming no GSI-191 failures) was neglected, so the sum of the bounding CDF from the low-likelihood configurations was assumed to be the risk increase (ΔCDF).

3. The bounding LERF due to low-likelihood configurations was calculated by multiplying the bounding CDF by the conditional large early release probability (CLERP) given core damage. The CLERP was obtained by dividing the base case (i.e., the case with no GSI-191 failures) LERF by the base case CDF. (Note that the CLERP for the high-likelihood configurations is identical to that for the base

case with no GSI-191 failures, 2.9E-03, so this approach to calculate bounding LERF for the low-likelihood configurations is reasonable.) The bounding LERF from the low-likelihood configurations was assumed to be the risk increase (ΔLERF).

c. Secondary side breaks inside containment (SSBI) that result in a consequential LOCA upon failure to terminate safety injection or a stuck open PORV may generate debris inside containment and require sump recirculation resulting in the transport of debris to the containment sump. A bounding evaluation was performed for the SSBI risk contribution to GSI-191 failures by quantifying the SSBI accident sequences in the Vogtle internal events PRA with a failure probability of 1.0 assigned to the sump strainers. The resulting risk increase was 1.18E-7 for CDF and 6.89E-9 for LERF. This was considered overly conservative to include in the determination of the overall GSI-191 risk impact.

   Therefore, a conservative evaluation of the SSBI risk contribution was performed based on CFPs calculated in NARWHAL for secondary side breaks. The NARWHAL evaluation assumed all secondary side breaks (feedwater or main steam line breaks inside containment) were double-ended guillotine breaks, and that one or both trains of containment spray (CS) had failed. Sump strainer failure due to GSI-191 phenomena occurred only for main steam line breaks when both trains of CS fail. Using the CFP from NARWHAL to quantify the SSBI risk contribution to GSI-191 failures produced a risk increase of 1.39E-9 for CDF and 8.25E-11 for LERF, nearly two orders of magnitude below the bounding evaluation results and well within the Regulatory Guide 1.174 Region III risk acceptance guidelines (Reference 2).

d. As described in the response to RAI 38.a, secondary side breaks outside containment do not have the potential to generate debris inside containment, and therefore were screened from further analysis.

NRC RAI 39

Section 2.3.1 of RG 1.174, Rev. 2, states that:

> ... the scope of a PRA is defined in terms of the causes of initiating events and the plant operating modes it addresses. Typical hazard groups considered in a nuclear power plant PRA include internal events, internal floods, seismic events, internal fires, high winds, external flooding, etc.

It is not apparent that the impacts of internal fire and external hazards, other than seismic, have been addressed in the submittal. These other external hazards may affect the total change in risk for this application of the PRA.

a. Please provide a justification (e.g., qualitative arguments or bounding analyses) that demonstrates that the risk contributions from internal fire would not affect this application of the PRA.

b. Please provide a justification (e.g., qualitative arguments or bounding analyses) that demonstrates that the risk contributions from external events other than seismic events would not affect this application of the PRA.

SNC Response to RAI 39

a. Consistent with the guidance in NUREG/CR-6850 (Reference 3), internal fire hazards were not assumed to result in pipe breaks. However, fire-induced LOCAs can occur, including spurious opening of a pressurizer PORV or safety valve, spurious reactor head vent, continuous letdown, spurious interfacing system LOCA, or reactor coolant pump (RCP) seal LOCA due to loss of seal cooling. Of these, only an RCP seal LOCA has the potential to generate debris inside containment. A spurious opening of a pressurizer PORV or safety valve, or spurious reactor head vent is discharged to the PRT, which has negligible sources of debris near the rupture disk. Spurious interfacing system LOCAs or continuous letdown all discharge outside containment. Therefore, these scenarios were all screened from the analysis. The quantity of debris generated by an RCP seal LOCA is equivalent to the quantity generated by a small or medium LOCA, which was found to not challenge the sump strainers; therefore, fire-induced RCP seal LOCAs were also screened from the analysis.

b. An evaluation of external hazards conducted for Vogtle concluded that in addition to internal flood, internal fire and seismic events, the only external hazards applicable to Vogtle are:

   - aircraft impact,
   - extreme winds and tornadoes,
   - external flooding including intense local precipitation,
   - industrial and military facility accidents,
   - pipeline accidents,
   - transportation accidents, and
   - turbine-generated missiles

   None of these external hazards listed have the potential to generate debris inside containment and were screened from the GSI-191 analysis. Therefore, seismic events are the only external hazard that affect this application of the Vogtle PRA.

References

1. **ML17116A098.** Vogtle Electric Generating Plant - Units 1 & 2 Supplemental Response to NRC Generic Letter 2004-02. April 21, 2017.
2. **Regulatory Guide 1.174.** An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis. Revision 2.
3. **NUREG/CR-6850.** Fire PRA Methodology for Nuclear Power Facilities. September 2005.