UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, ILLINOIS 60532-4352

December 16, 2021

EA-21-176

Mr. Terry Brown
Site Vice President
Energy Harbor Nuclear Corp.
Davis-Besse Nuclear Power Station
5501 N. State Rte. 2, Mail Stop A–DB–3080
Oak Harbor, OH 43449–9760

SUBJECT: DAVIS-BESSE - NRC INSPECTION REPORT (05000346/2021091); PRELIMINARY GREATER THAN GREEN FINDING

Dear Mr. Brown:

This letter transmits the NRC's preliminary determination of the safety significance of an inspection finding described in NRC inspection report 05000346/2021050. The finding has preliminarily been determined to be of greater than very low safety significance resulting in the need for further evaluation to determine the final significance. As described in the previous inspection report, the finding involved the failure to develop a preventative maintenance schedule for the inspection of the emergency diesel generator (EDG) field flash selector (FFSS) switch. The switch was not inspected from the time it was installed in the plant in 2006. The failure to inspect the switch contributed to the long-term degradation of the switch electrical contacts and ultimately contributed to an EDG failure during fast start testing on May 27, 2021. This finding was assessed based on the best available information, using the applicable Significance Determination Process (SDP). The final resolution of this finding will be conveyed in separate correspondence.

After the special inspection concluded and during the development of the preliminary significance determination you provided the perspective, based on a vendor analysis, that the EDG FFSS failure during the fast start test was most likely the result of foreign material between the switch electrical contacts as evidenced by the presence of nickel on the contact surface. You concluded that the failure was not caused by the lack of inspection and long-term switch electrical contact degradation. The NRC has preliminarily determined this vendor analysis does not rule out contact degradation due to lack of inspection as a significant contributing cause of the failure. There is sufficient operating experience on electrical contact failure due to contamination to reasonably consider this degradation mechanism to be credible. Therefore, we continued to conduct our significance determination with this assumption.

The basis for the staff's significance determination is provided in the enclosure. Important assumptions used in the staff's significance determination include the period of time the EDG was unavailable to automatically provide emergency power (i.e., the exposure period), the feasibility and reliability of actions to recover the EDG if it failed, and the contribution of fire risk scenarios to the safety significance of the finding.

On November 30, you informed the NRC that you had conducted a risk evaluation of the degraded condition and that while you maintained the failure was not related to the performance deficiency identified by the NRC, you believed the risk of the degraded condition itself was of low to moderate safety significance. The NRC was not able to review your evaluation prior to completing the preliminary determination enclosed in this letter but will review it and consider your position as we make our final decision on this matter.

The NRC significance determination is subject to several uncertainties which could affect the outcome of the significance determination. Therefore, before we make a final decision, we invite you to provide any additional information on the docket to help in our evaluation of the significance of the finding. Specifically, we would like any information or perspectives you can provide in the following areas:

- The assumed exposure period used in the NRC preliminary evaluation.
- Actions that can be taken by operators to recover from a fast start failure of the EDG, including operator training on the FFSS.
- The feasibility and reliability of the operator actions to recover from the fast start failure of the EDG, particularly during the dominant fire risk scenarios.
- Conservatisms or non-conservatisms in your fire Probabilistic risk assessment (PRA) that could affect the outcome of this evaluation.
- Why you believe the contact contamination is not a credible cause of the FFSS that occurred during testing, including how inspection and preventive maintenance would not prevent the failure.

This finding does not represent a current safety concern because the FFSS on both EDGs have been replaced and procedures have been put in place to perform monthly continuity checks on the contacts. However, the finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the Enforcement Policy, which can be found on the NRC's Web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

In accordance with NRC Inspection Manual Chapter 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of November 19, the date of the issuance of the special inspection report that initially documented the finding. The SDP encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination.

Before we make a final decision on this matter, we are providing you with an opportunity to (1) attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing, or (3) if you choose not to submit information or attend a Regulatory Conference, accept the NRC's final significance determination. If you request a Regulatory Conference, it should be held within 40 days of the receipt of this letter, and we encourage you to submit supporting documentation at least one

week prior to the conference in an effort to make the conference more efficient and effective. The focus of the Regulatory Conference is to discuss the significance of the finding and not necessarily the root cause(s) or corrective action(s) associated with the finding. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 40 days of your receipt of this letter. If you decline to request a Regulatory Conference or to submit a written response, you relinquish your right to appeal the final SDP determination, in that by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation sections of Attachment 2 of NRC Inspection Manual Chapter 0609.

If you choose to send a response, it should be clearly marked as a "Response to An Apparent Violation; (EA-21-176)" and should include for the apparent violation: (1) the reason for the apparent violation or, if contested, the basis for disputing the apparent violation; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken; and (4) the date when full compliance will be achieved. Your response should be submitted under oath or affirmation and may reference or include previously docketed correspondence, if the correspondence adequately addresses the required response. Additionally, your response should be sent to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Center, Washington, DC 20555-0001 with a copy to Laura Kozak, Acting Branch Chief, U.S. Nuclear Regulatory Commission, Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532 within 40 days of the date of this letter. If an adequate response is not received within the time specified or an extension of time has not been granted by the NRC, the NRC will proceed with its enforcement decision or schedule a Regulatory Conference.

Please contact Laura Kozak at 630-464-3344 and in writing within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for these inspection findings at this time. In addition, please be advised that the characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room and in the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

Michelle Hayes Signed by Hayes, Michelle on 12/16/21

Michelle Hayes, Acting Deputy Director
Division of Reactor Safety

Docket No. 05000346
License No. NPF-3

Enclosure:
As stated

cc: Distribution via LISTSERV®

Letter to Terry Brown from Michelle Hayes dated December 16, 2021.

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Summary of the Detailed Risk Evaluation for the Davis-Besse Field Flash Selector Switch Issue and Basis for Preliminary Significance Determination

Exposure Time – The last successful test using the Field Flash Selector Switch (FFSS) in the 400-rpm setting, to create a fast start was November 12, 2020. Emergency Diesel Generator-1 (EDG) was tested successfully several times following that date and then the last successful monthly test using the FFSS in the 800-rpm setting was April 27, 2021. At the conclusion of that test the switch was placed in the 400-rpm setting to ready the engine for a valid demand. The contacts on the switch were not operated after that time. The engine failed to start for the fast-start surveillance test on May 27, 2021, and the engine was repaired and returned to service on May 28, 2021, at 17:30.

The NRC analysis used the t + repair time for the window from April 27 to May 28 assuming EDG-1 was unavailable after the last successful month surveillance test on April 27, 2021. The analysis then used the t/2 approach for the window between November 12 and April 27 because there was uncertainty regarding when the contact degradation had become significant to challenge functionality of EDG-1 after the last successful fast-start surveillance test on November 12, 2020. A total exposure time of 83 days + 31 days, or 114 days was used in the analysis. A factor of 114 days/365 days/year, or 0.312 was applied to the annualized results.

A sensitivity evaluation was also performed on the exposure time because it is a key assumption that impacts the significance determination. Assuming an exposure period of 31 days (April 27 to May 28), which represents the time between the last FFSS operation and the surveillance test failure, the delta-CDF estimate is approximately 5.7E-6/yr.

Failure Mechanism – The failure of EDG-1, during a hypothetical demand, would be in the first few seconds of an attempted start of the engine. Consequently, the basic event in the Davis-Besse Standardized Plant Analysis Risk (SPAR) model "Diesel Generator 1-1 Fails to Start" was set to "True." All other basic events were left at their nominal failure probabilities, except for the Mitigating Strategies equipment, which is discussed below.

Common Cause Implications – The two safety-related EDGs at Davis-Besse are similar in all respects, and the performance deficiency affected both engines. The potential for common cause failure was used in the NRC's evaluation. However, it is important to note that the Davis-Besse Station Blackout EDG was sufficiently different in design, vintage, and other material properties such that it was not part of the Common Cause Component Group with the two safety-related engines in the SPAR model.

Mitigating Strategies – The Mitigating Strategies equipment and response procedures (commonly and collectively known as FLEX) were credited in this analysis. Given that the performance deficiency affected the safety-related EDGs, and the accident sequences of concern were station blackout sequences that become *extended* losses of AC power scenarios, FLEX was included in the internal and external events analysis. However, the failure probabilities in the SPAR model for the FLEX basic events were adjusted using a 3x multiplier to more accurately reflect the higher unreliability of portable equipment. The analysis also compared the FLEX unavailability/unreliability values used in the licensee Probabilistic risk assessment (PRA) model with the NRC SPAR model (using the 3x multiplier). No significant differences were identified.

Repair/Recovery of Failed Components – The diagnosis and replacement of the failed speed switch following the surveillance testing failure was one day in duration. Three different options to recover EDG-1 exist, including: 1) taking the FFSS from fast start setting (400 rpm) to the slow start setting (800 rpm), 2) following emergency shutdown of the engine, attempting restart without manipulating the FFSS, and 3) depressing the "F'ld [Field] Flash" pushbutton, which bypasses the 400/800 rpm contacts. Any of these three options may have flashed the field and recovered the failed EDG-1.

For the preliminary determination, potential recovery of the EDG was included in the internal event analysis result but not in the fire risk result. As described below, the overall risk of the performance deficiency was determined using the NRC SPAR model for internal events and the licensee's model for seismic and fire risk. Fire risk was the dominant risk contributor.

The licensee's PRA model of record does not include EDG recoveries, while the NRC SPAR model for Davis-Besse includes EDG recoveries. Therefore, the NRC conducted a sensitivity evaluation to determine the potential impact of adding recoveries to the licensee's fire model. In comparing differences in SPAR model delta-CDF results with and without recovery, the change in delta-CDF was 39 percent of the result when no EDG recovery was credited. The 0.39 factor was applied to the licensee's fire PRA result in the sensitivity evaluation which showed the delta-CDF value would be reduced from 2.2E-5/yr. to 1.1E-5/yr. This result is also yellow but closer to the yellow-white threshold.

The sensitivity analysis indicates that recovery is a key assumption in the significance determination warranting further review on the impact to the risk evaluation. Also, the SPAR model EDG recoveries are based on data collected for many types of EDG failures. The failure that is the subject of this performance deficiency does not require repair for EDG-1 to successfully start and may be more specifically evaluated using human reliability analysis.

Internal Events Risk – The dominant internal event accident sequence is a weather-related loss of offsite power sequence and contributes 32 percent of the total internal events risk. However, the overall results were dominated by fire and internal events only represented 16 percent of the total.

External Events Risk –

- Fire – Fire was the dominant contributor to the overall delta-CDF result and was included quantitatively. The NRC used results from the licensee's fire PRA in the significance determination. The dominant fire sequences are large damaging fires, or high energy arc faults, which cause a loss of offsite power (LOOP) with a subsequent loss of decay heat removal via the once-through steam generators with subsequent failure of makeup/high pressure injection cooling. These scenarios contribute 54 percent of the fire risk. FLEX equipment/strategies were credited in both the base and non-conforming cases. The analyst sampled the top 20 dominant cut-sets and verified that bounding and/or unrealistic assumptions were not being used, however further reviews or discussions with Energy Harbor staff may be needed to either confirm or refute this assumption.

- Tornados/High Winds and External Flooding – External flooding or tornados leading to a LOOP, though credible, was determined to be several orders of magnitude less frequent than the LOOP values used in the internal events model. No further analysis was performed.

- Seismic – Seismic-induced events were quantified using the licensee's all hazard model, though they were not a significant contributor to the overall result.

Uncertainty – With all risk evaluations, there are both aleatory (randomness) and epistemic (lack of knowledge) uncertainties. The aleatory uncertainty was assessed using the Davis-Besse SPAR model. The 5 percent and 95 percent values for the consolidated results (including both internal and external events) were 7.7E-6/year to 5.1E-5/year, respectively. The remaining epistemic uncertainties with the results were centered in the areas of 1) exposure time, 2) recovery of EDG-1 during station blackout/extended loss of AC power scenarios, and 3) conservatisms or non-conservatisms in the fire PRA.

Item of Merit – The risk contribution for the two most commonly used items of merit, delta-CDF and delta-LERF, were quantified in the analysis. Delta-CDF remained the item of merit.

Consolidated Results –

	Base Case	Non-Conforming Case	Delta Risk (change in CDF)
Internal Events (NRC Results)	1E-6	4.4E-6	3.4E-6
Seismic (Licensee Results)	4E-6	5.3E-6	1.3E-6
Fire (Licensee Results)	1.5E-5	3.2E-5	1.7E-5
Total =	**2E-5**	**4.2E-5**	**2.2E-5**

Conclusion and Basis for Decision – The quantitative and qualitative inputs, when considered with the epistemic uncertainties described above, support the treatment of this finding/violation as potentially "Greater-Than-Green" (delta-CDF ≥ 1E-6/year).