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

March 30, 2021

TO: Michael X. Franovich, Director
Division of Risk Assessment
Office of Nuclear Reactor Regulation

Joseph Donoghue, Director
Division of Safety Systems
Office of Nuclear Reactor Regulation

Chris Miller, Director
Division of Reactor Oversight
Office of Nuclear Reactor Regulation

FROM: Sunil Weerakkody, Senior Level Advisor
Division of Risk Assessment
Office of Nuclear Reactor Regulation

Signed by Weerakkody, Sunil
on 03/30/21

SUBJECT: DUANE ARNOLD ENERGY CENTER LIC-504 TEAM RECOMMENDATIONS

In accordance with Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-504, "Integrated Risk-Informed Decisionmaking Process for Emergent Issues," Revision 5, dated March 4, 2020 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML19253D401), the U.S. Nuclear Regulatory Commission (NRC) staff assessed the derecho event that occurred at the Duane Arnold Energy Center (DAEC) Nuclear Power Plant on August 10, 2020, to evaluate potential safety impacts to other nuclear power plant licensees. This memorandum presents the staff's recommendations for followup agency actions based on risk insights and assessment of readily available information necessary to develop risk insights for the purposes of the LIC-504 analysis.

Issue Description

A derecho is a widespread, long-lived, straight-line windstorm associated with a band of rapidly moving thunderstorms. Derechos typically occur during the warm season (May–August). A key distinction between a derecho and a tornado is the widespread damage swath. A tornado's width is generally less than a mile, with the widest around 2.5 miles. For a storm to be classified as a derecho, it must travel at least 240 miles and move at speeds of at least 58 miles per hour (mph), although the winds are often more powerful.

CONTACT: Sunil Weerakkody, NRR/DRA
301-415-2870

The August 2020 Midwest derecho had winds up to 112 mph, which is equivalent to an Enhanced Fujita (EF) 1 tornado.[1] Tornadoes can also be embedded within derechos and produce concentrated areas of even more intense damage. The derecho at DAEC demonstrated that the plant's design is adequate to withstand the impacts of high winds and resulting debris-generated missiles related to the derecho, albeit with significant damage to nonsafety-related systems and degradations to the functionality of the emergency service water (ESW) system. Specifically, during the DAEC derecho event, all safety-related systems remained functional throughout the event and enabled successful shutdown of the plant even with offsite power lost for more than 24 hours. The challenge to the ESW system during the derecho was excessive debris at the intake structure entering the system and clogging ESW pump strainers. ESW provides cooling to the emergency diesel generators (EDGs), and excessive strainer differential pressure (ΔP) and subsequent reduced service water flow required operators to manually bypass one train of strainers and monitor EDG operation while in the bypass configuration. The second ESW train also experienced some clogging; however, the ΔP across the strainer stabilized without reaching the threshold that prompts operators to take recovery actions.

In summary, the DAEC event posed concurrent challenges to offsite power supplies and the functionality of the ESW system due to a sudden inrush of debris to the intake structure (i.e., a combined event). This emergent issue points to a component of the completeness uncertainty[2] of probabilistic risk assessment (PRA) models; generally, PRA models used by licensees and the NRC do not model weather-related loss of offsite power (WRLOOP) events that are concurrent with a loss of ESW caused by the same initiating event (in this case, a derecho). Considering the analysis for eight sample nuclear power plants with different design characteristics, the LIC-504 team estimated risk increases due to the combined event and concluded that the safety implications can significantly vary based on site, plant design, and plant operating characteristics. Risk analyses for the group of sample plants confirmed that the potential increases in risk associated with the issue is below the value for which the NRC would consider taking immediate regulatory action. However, based on risk insights and insights from NRC staff with expertise on the design and operation of the system, the LIC-504 team concluded that based on certain site, design, and operating characteristics discussed below, the potential for safety enhancements may be present.

The following site characteristics influence the magnitude of this safety issue:

- likelihood of events such as derechos that could cause extended loss of offsite power (LOOP)
- likelihood of such events to add significant debris to the facilities' ultimate heat sink (UHS)
- likelihood of high debris transport rates

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1 The EF scale is the guideline used to define tornado intensity, relating tornado damage and estimated maximum wind speed. An EF1 tornado is defined by wind speed of 86–110 mph (see NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," Revision 2, issued February 2007 (ADAMS Accession No. ML070810400)).

2 PRAs have two types of uncertainty: aleatory and epistemic. Epistemic uncertainties consist of three components: parametric, modeling, and completeness uncertainty. Completeness uncertainty captures a limitation of the PRA model—that not all failures and events are modeled. Risk-Informed decisionmaking relies on other attributes, such as defense in depth, to address concerns associated with these uncertainties in risk-informed decisionmaking.

- spatial proximity of cooling water (e.g., ESW, circulation water, fire water) intake locations to the UHS.

The following design characteristics influence the magnitude of this safety issue:

- ability of traveling screens at intake structure to mitigate impacts of sudden increases in debris loading

- availability of ESW pump strainers and capability to back wash strainers

- ability for operators to bypass strainers and operate EDGs with strainers bypassed

- capability to detect and mitigate strainer blockage (e.g., bypass strainers)

- availability of strategies to provide cooling to EDGs and other critical systems using systems that do not rely on ESW (e.g., fire protection water)

- availability of additional sources of alternating current (AC) power, such as a station blackout diesel or other supplemental diesels

The following operating characteristics influence the magnitude of the safety issue:

- plant response to warnings for impending severe weather (e.g., severe weather preparedness procedures)

- adequacy of maintenance activities of the above equipment

- procedures and training on additional mitigating components and strategies (e.g., diverse and flexible mitigation capability (FLEX))

- ability for operators to promptly diagnose and bypass a clogged strainer

<u>Summary of Risk Analysis</u>

To obtain risk insights, NRR's Division of Risk Assessment (DRA) staff used Standardized Plant Analysis Risk (SPAR) models of eight power plants with different design characteristics. DRA staff estimated the magnitude of the potential increases to the sample power plant core damage frequencies (CDFs) resulting from a combined event where a common hazard imposes a potential for a LOOP concurrent with a sudden inrush of debris to intake structures, ultimately affecting the cooling capability of the EDGs.

The eight plants chosen for this analysis include six of the seven plants that were previously chosen for the first step of the LIC-504 process, as documented in the memorandum dated November 25, 2020 (ADAMS Accession No. ML20315A117). These plants were chosen because they have similar features to DAEC. Specifically, the plants selected were either (1) single-unit sites that do not have a dedicated alternate AC source, such as a station blackout diesel, that could mitigate the risk from such events or (2) plants that may have service water systems more susceptible to the effects of high winds.

Table 1 summarizes the results of the risk assessments.[3] In a number of situations, the staff used conservative and nonconservative assumptions and readily available information for the sample plants (i.e., some design and operational details may not have been factored into the analysis). Enclosure 1 to this memorandum provides additional detail on assumptions, uncertainties, conservatisms, and potential nonconservatisms for the analyses.

The DAEC event demonstrates the potential for a derecho to cause redundant trains of an ESW system to fail. A review of operating experience documentation found that 12 WRLOOP events occurred during critical operation of U.S. nuclear power plants from 1997 to 2019. The documentation did not reveal any degradation of ESW systems. Considering the DAEC event, the conditional probability of ESW failure during a WRLOOP would be 1 failure over 13, or 0.077. However, since the DAEC experienced an ESW degradation rather than a complete failure, the conditional probability would be 0.5 failure over 13, or about 0.038. Therefore, the staff assumed a common-cause failure (CCF) probability for ESW of 0.038 as one of the sensitivities in the analysis. This is a key uncertainty in the analysis.

Table 1 Increases in CDF (∆CDF) Per Year

Plant	Baseline CDF for WRLOOP	∆CDF of Strainer at 0.1 CCF	∆CDF at 0.038 CCF	∆CDF of 0.01 CCF
Plant #1: Westinghouse pressurized-water reactor (PWR)	$1.2x10^{-6}$	$8.4Ex10^{-5}$	$3.4x10^{-5}$	$1.1x10^{-5}$
Plant #2: Combustion Engineering PWR	$2.6x10^{-6}$	$2.2Ex10^{-6}$	$1.1x10^{-6}$	$6.1x10^{-7}$
Plant #3: Boiling-water reactor (BWR)4 with Mark I containment	$3.5x10^{-6}$	$7.1x10^{-7}$	$5.2x10^{-7}$	$4.3x10^{-7}$
Plant#4: Westinghouse PWR	$9x10^{-7}$	$3x10^{-6}$	$1.8x10^{-6}$	$1.5x10^{-6}$
Plant#5: Westinghouse PWR	$6.6x10^{-6}$	$1.9x10^{-5}$	$8.4x10^{-6}$	$3.5x10^{-6}$
Plant#6: BWR6 with Mark III containment	$1.1x10^{-6}$	$3.8x10^{-6}$	$2.4x10^{-6}$	$1.8x10^{-6}$
Plant#7: BWR4 with Mark I containment	$2.0x10^{-7}$	$2.4x10^{-6}$	$1.0x10^{-6}$	$4.1x10^{-7}$
Plant#8: BWR4 with Mark I containment	$4.8x10^{-6}$	$1.4x10^{-5}$	$1.2x10^{-5}$	$1.1x10^{-5}$

Summary of Risk Insights

In addition to generating changes to CDFs, the team generated risk insights by reviewing and comparing dominant cutsets for each plant. In addition to risk analyses performed by DRA staff, the team also considered the insights provided by NRR's Division of Safety Systems staff, as captured in Table 2. Together with the information in Table 1, these insights enabled PRA analysts to make an informed judgment on the magnitude of the risk associated with the postulated combined event. In addition to insights obtained from the risk analyses of the eight sample plants, the team generated risk insights by reviewing NRC Region III's inspection report on the DAEC event (ADAMS Accession No. ML20314A150) and holding discussions with the

[3] This LIC-504 analysis used ∆CDF as a metric instead of conditional core damage probability, which was used in the accident sequence precursor (ASP) (ADAMS Accession No. ML210222A415) and Management Directive (MD) 8.3, "NRC Incident Investigation Program," dated June 25, 2014 (ADAMS Accession No. ML21022A415), analyses for this event. The analyses differ in that the ∆CDF estimates derived for the LIC-504 analysis use the initiating event frequency of a WRLOOP ($6x10^{-3}$/year) versus setting the initiating event frequency to 1.0 when performing an initiating event analysis under the ASP, MD 8.3, or significance determination process.

DAEC resident inspector. The team also benefited significantly from risk insights gleaned from the accident sequence precursor (ASP) analysis performed by the NRC Office of Nuclear Regulatory Research (RES) (ADAMS Accession No. ML21056A382) and the evaluation completed by Region 3 under Management Directive (MD) 8.3, "NRC Incident Evaluation Program" (ADAMS Accession No. ML21022A415).

As identified in Table 1 above and Enclosure 1 to this memorandum, the team performed a variety of sensitivity studies to examine vulnerabilities at the subject plants. Although FLEX strategies were not deployed for the DAEC derecho event, the team was interested in the degree of risk reduction provided by the implementation of FLEX strategies since a derecho could potentially lead to an extended loss of AC power. Based on the sensitivity analysis (Table 2 of Enclosure 1), the staff found that the degree of risk reduction due to FLEX strategies ranged from 11.4 to 1.4.

Table 2 Summary of Risk Insights

Site and Design Characteristics	
Characteristic	**Impact of Characteristic on Risk**
Frequency of the combined event that causes a LOOP and a concurrent challenge to the functionality of ESW and fire protection water due to debris	Sites located in areas that have lower likelihood of events such as derechos are at reduced risk.
Susceptibility of the water source for ESW for debris accumulation during a derecho	Sites that have UHS sources that are not prone to accumulation of debris have reduced risk.
Relative location of the intake to redundant ESW trains and the location of suction for fire pump suction at plants that use fire protection water as a diverse capability for EDG cooling	Plants with suction sources that are spatially significantly apart are at reduced risk because concurrent blockage of redundant and diverse suction capabilities is reduced.
Availability of additional diesels that do not rely on ESW in addition to availability of diesels procured and installed as part of FLEX strategies	Plants with additional AC power sources (which are often not dependent upon ESW for cooling) and have the ability to provide motive power to essential loads are at reduced risk.
Availability of alternative strategies to provide cooling water to EDGs (including water from fire protection system or other source)	Plants with alternative strategies to provide cooling water to EDGs are at reduced risk.
Ability to promptly recognize the increased ΔP across strainers	Plants that have alarms or annunciators that inform operators of increasing ΔP across the ESW strainer and intake structure screens are at reduced risk.
Ability to bypass strainers and ability of EDGs to successfully operate in the bypass mode	Plants that have the capability to bypass strainers decreases risk since the EDGs may operate successfully in that temporary configuration. However, long-term bypass of unstrained water can result in increased risk to downstream components.
Source of AC power to traveling screens	Plants whose traveling screens are powered by emergency AC are at reduced risk.
Operating Characteristics	

Ability to promptly recognize the increased ΔP across strainers	Early detection and procedures that instruct operators to monitor ΔP across strainer and intake structure screens upon receipt of warnings for severe weather may decrease risk.
Use of FLEX strategies	Procedures, testing, and training that maximize reliability and identify risk reductions from FLEX strategies could reduce potential risk increases attributed to this event.
Procedures and abnormal operating procedures relating to severe weather warnings	Severe weather preparedness procedures and abnormal operating procedures that recognize the potential to increase the likelihood of blockage of intake structure and strainers decrease risk (e.g., running the intake screen debris cleaning pumps at full speed).

Recommendations

As demonstrated by the ASP analysis and the MD 8.3 evaluation, the DAEC event was of high risk significance. Considering the above, the LIC-504 team examined a number of activities and developed 4 recommendations that may reduce future risks from events such as the DAEC derecho using guidance offered in LIC-504, Revision 5.

Enclosure 2 to this memorandum summarizes the recommendations considered in the team's evaluation. The team considered information provided in various NRC documents (e.g., NUREG\BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Revision 5, draft report for comment, issued April 2017 (ADAMS Accession No. ML17100A480); Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, issued January 2018 (ADAMS Accession No. ML17317A256) and guidance in Figure 3 of LIC-504 to develop recommendations. The team also consulted with staff in the NRR's Division of Reactor Oversight, Operating Experience Branch.

The LIC-504 team recommends the following:

- Issue an information notice to licensees about the risk insights gained through the NRC's analysis of the DAEC derecho event.
- Share risk insights obtained from the LIC-504 analysis with the NRC's regional staff.
- Identify opportunities to engage with external stakeholders (e.g., PRA practitioners, owners groups) about the insights gained during this evaluation.
- Update two SPAR models during fiscal year 2021 and 2022 as part of the normal update process to further enhance staff's understanding of risk insghts gained from LIC-504.

Issuance of this memorandum concludes the LIC-504 process.

SUBJECT: DUANE ARNOLD ENERGY CENTER LIC-504 TEAM RECOMMENDATIONS

DATED:

ENCLOSURES:
1. Summary of Risk Analysis (ML21078A178)
2. Analysis of Options (ML21078A186)

ADAMS Accession No.: ML21078A127 NRR-106

OFFICE	NRR/DRA	NRR/DRA/APOB	NRR/DRA/APOB	NRR/DORL/LPL3
NAME	SWeerakkody	MLeech	AZoulis	JWiebe
DATE	3/30/2021	3/29/2021	3/30/2021	3/26/2021

OFFICE	RIII/DRP/B2/PERO	NRR/DSS/SCPB	NRR/DSS/SCPB
NAME	JSteffes	NKaripineni	GCurran
DATE	3/29/2021	3/29/2021	3/29/2021

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