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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.

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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

Summary of Risk Analysis

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)	$1.2\text{x}10^{-6}$	$8.4\text{Ex}10^{-5}$	$3.4\text{x}10^{-5}$	$1.1\text{x}10^{-5}$
Plant #2: Combustion Engineering PWR	$2.6\text{x}10^{-6}$	$2.2\text{Ex}10^{-6}$	$1.1\text{x}10^{-6}$	$6.1\text{x}10^{-7}$
Plant #3: Boiling-water reactor (BWR)4 with Mark I containment	$3.5\text{x}10^{-6}$	$7.1\text{x}10^{-7}$	$5.2\text{x}10^{-7}$	$4.3\text{x}10^{-7}$
Plant#4: Westinghouse PWR	$9\text{x}10^{-7}$	$3\text{x}10^{-6}$	$1.8\text{x}10^{-6}$	$1.5\text{x}10^{-6}$
Plant#5: Westinghouse PWR	$6.6\text{x}10^{-6}$	$1.9\text{x}10^{-5}$	$8.4\text{x}10^{-6}$	$3.5\text{x}10^{-6}$
Plant#6: BWR6 with Mark III containment	$1.1\text{x}10^{-6}$	$3.8\text{x}10^{-6}$	$2.4\text{x}10^{-6}$	$1.8\text{x}10^{-6}$
Plant#7: BWR4 with Mark I containment	$2.0\text{x}10^{-7}$	$2.4\text{x}10^{-6}$	$1.0\text{x}10^{-6}$	$4.1\text{x}10^{-7}$
Plant#8: BWR4 with Mark I containment	$4.8\text{x}10^{-6}$	$1.4\text{x}10^{-5}$	$1.2\text{x}10^{-5}$	$1.1\text{x}10^{-5}$

<u>Summary of Risk Insights</u>

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 ($6\text{x}10^{-3}$/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 ∆P across strainers	Early detection and procedures that instruct operators to monitor ∆P 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

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SUMMARY OF RISK ANALYSIS RESULTS AND INSIGHTS

To support the LIC-504 analysis, staff in the Division of Risk Assessment performed risk estimates using the NRC's standardized plant risk analysis (SPAR) models for eight plants of diverse designs. These risk estimates represent the increases to the base core damage frequency (CDF) for an event similar to the one at the Duane Arnold Energy Center (DAEC) on August 10, 2020, when that plant experienced a derecho and a weather-related loss of offsite power (WRLOOP). The plant designs studied included boiling-water reactor (BWR) 4 plants with a Mark-1 containment design, BWR-6 with a Mark 3 containment, and pressurized-water reactors (PWRs) including designs by Westinghouse and Combustion Engineering (CE).

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

This risk analysis in support of LIC-504 differs from an Accident Sequence Precursor Program analysis and a Management Directive 8.3 analysis in the treatment of the initiating event frequency (IEF) and the risk metric used. For initiating event assessments when performing ASP or MD 8.3 analyses, the IEF is set to 1.0 (i.e., the event happened) and the metric used is conditional core damage probability (CCDP). This LIC-504 analysis uses the normal frequency of a WRLOOP (i.e., 5.99×10^{-6}), and the metric used is a change in CDF (ΔCDF).

This analysis refers to each plant's emergency service water (ESW) system. However, it is recognized that this system's name and features may vary from plant to plant. For the purpose of this analysis, ESW refers to the safety-related portion of the service water system that provides a source of cooling water to the emergency diesel generators (EDGs) and other loads during a loss of offsite power.

Modifications to the Simplified Plant Analysis Risk Models

In order to model the impact of an extended WRLOOP challenging the plant's ESW system strainers, the following model changes were applied to the analyzed plants:

- *Offsite Power Recovery*. To model an event similar to that which occurred at DAEC, the analysis assumed that offsite power is lost and cannot be recovered for 24 hours. This assumption captures the difficulty of recovering from the potential effects of significant damage to switchyard and transmission components in a severe weather event.

- *Crediting FLEX Strategies*. FLEX refers to the plant's diverse and flexible mitigation capabilities that are used during extended loss of AC power (ELAP) scenarios. Since the initiating event for these estimates is a WRLOOP, it is appropriate to provide credit for flexible coping strategies or FLEX.

Enclosure 1

FLEX hardware reliability parameters suitable for inclusion in the NRC SPAR models are not yet available.

Therefore, the base SPAR models currently use the reliability of permanently installed equipment for FLEX equipment as the default, which is nonconservative and inconsistent with the limited operating experience with FLEX equipment. As part of an NRC audit of preliminary FLEX hardware data provided by the Pressurized-Water Reactor Owners Group, Idaho National Laboratory reviewed the FLEX hardware parameters estimated by the owners group. This review revealed that the FLEX diesel generator failure-to-start (FTS) probability is 3 to10 times higher and failure-to-run rate (FTR) is 2 to 5 times higher than permanently installed EDGs. The portable engine-driven centrifugal pump FTS probability is at least 8 times higher and FTR rate is at least 6 times higher than permanently installed pumps. See Table 1 in INL/EXT-20-58327, "Evaluation of Weakly Informed Priors for FLEX Data," issued May 2020 (ADAMS Accession No. ML20155K834), for additional information. Therefore, to provide a more representative estimate of the FLEX hardware reliability parameters, this analysis increased the hardware failure rates by a factor of 3.

- *Removal of EDG Repair Credit for ELAP Scenarios*: The SPAR models provide credit for repair of postulated EDG failures for SBO scenarios. However, this potential credit may not apply in scenarios where an ELAP will be declared because (1) operators will be focused on implementing the FLEX mitigation strategies, (2) the DC load shedding activities could preclude recovery of EDGs, and (3) the failure of EDG due to degraded ESW (a likely outcome for this scenario) may be difficult to recover. As a result, the decision was made to not credit EDG repair. This may be a conservative approach since some scenarios could exist where diesel recovery could be possible.

- *Removal of 72-Hour AC Power Recovery Requirement*: The base SPAR model requires AC power recovery within 72 hours for a safe/stable end state for ELAP scenarios with successful FLEX implementation. If AC power is not recovered in these scenarios, the SPAR models assume core damage. The probabilistic risk assessment (PRA) standard definition for safe/stable end state does not require AC power recovery. Because of the large uncertainty in modeling assumptions related to availability and reliability of components and strategies for mission times that are well beyond 24 hours, the 72-hour AC power requirement was eliminated in this analysis. As part of this change, the FTR events for FLEX diesel generators and pumps have a 72-hour mission time in the base SPAR model. These mission times were changed to 24 hours to be consistent with the 24-hour mission time used in the SPAR model.

- *Adjusting the Common Cause Failure (CCF) Parameters for ESW Strainers*: This change constitutes an approach to model the increased potential for the ESW system strainers to fail due to the increased debris loading experienced during events similar to what occurred at DAEC. The event that occurred at DAEC demonstrated the potential for a derecho to cause a failure of redundant trains of ESW.

  The CCF failure rate of the ESW strainers was recognized as a key source of uncertainty in the analysis. Strainer design can vary. Many plants (like DAEC) have strainers with automatic backwash capability, other plants have different designs. As a result of the uncertainty associated with choosing a CCF failure rate, the analysis used three different CCF values of .1, 0.038, and .01. The value of 0.1 was chosen since it had been used as the screening analysis in the first step of the LIC-504 analysis documented in the memo dated November 25, 2020, ADAMS Accession No. ML20315A117.

The CCF value of 0.038 was chosen based on the following. A review of operating experience found 12 WRLOOPs from 1997 to 2019. None of the documentation reviewed captured a concurrent degradation of ESW. For the DAEC event, the conditional probability of ESW failure given a WRLOOP would be 1 over 13, or 0.077.

Since the DAEC was an ESW degradation rather than a complete failure, the conditional probability could be better represented by 0.5 over 13, which is 0.038.

Sensitivity analyses were performed at different failure rates as shown in Table 1. The Uncertainties portion of this enclosure discusses this further.

- *Ability to Bypass Clogged ESW Strainers:* The ability for operators to bypass the clogged ESW strainers is inconsistently applied in the base SPAR models. However, when readily available information was found for each plant that supported bypass of the strainers, the model was confirmed to include the action or modified to include the operator action. All plants that were found to have the ability to bypass a clogged strainer have been credited in this analysis. It is noted as a potential nonconservatism in the analysis for crediting a bypassed strainer. Once a strainer is bypassed and unstrained water is provided to downstream heat exchangers there is an increased potential for obstructing or fouling the heat exchangers complicating a potential recovery of the EDGs affected. This analysis does not quantify this risk; it is simply noted as an uncertainty. There have been examples of plants that have operated for short periods with service water strainers bypassed without issues.

- *Additional Model Changes:* During the development of these risk estimates staff from the Division of Risk Assessment modified some SPAR models to provide a more accurate estimate using readily available information.

Uncertainties

The following uncertainties were considered when deriving risk insights during the integrated risk-informed decisionmaking process discussed in LIC-504. The analysis is intended to support NRC decisionmakers in evaluating appropriate regulatory actions and should not be considered an exhaustive risk assessment of the issue. Therefore, the analysis has some known conservatisms, non-conservatisms, and key uncertainties in the analysis. For instance, the NRC staff did not review plant-specific procedures, engage with licensees for more details about their PRA models, conduct operator and staff interviews, perform plant walkdowns, etc.

Also, use of an IEF for a WRLOOP is likely more conservative than for an extreme LOOP caused by a derecho, which would both prevent both offsite power recovery for 24 hours and pose a significant challenge to ESW strainers. Further details on these items are included below:

- *Uncertainties in the nature of the event itself and the effect on each plant's ESW system:* Because of the number of unknowns associated with this type of event, this analysis does not assume a complete blockage of the intake to traveling screens that would prevent sufficient water from entering the ESW system. This analysis assumes ESW supply itself is not lost, and that the traveling screens are not completely blocked.

On the contrary, the analysis assumes that smaller debris is able to reach and clog the ESW strainers. This is similar to what happened at DAEC on August 10, 2020. DAEC had traveling screens that remained powered during a LOOP, but not all plants have that same design feature (i.e. some plants have traveling screens that do not have a safety-related power supply and would not continue to rotate during a LOOP).

- *Use of the WRLOOP IEF*: The standard SPAR models contain initiating event frequencies for various LOOP events. Of those events, a WRLOOP provides the closest estimate to the type of derecho event that Duane Arnold experienced. It is likely that the actual frequency for a WRLOOP that would simultaneously cause a sustained loss of offsite power and an in-rush of debris that would challenge a plant's service water system is likely to be lower than the standard IEF used for a WRLOOP. However, not enough data exist to derive a more accurate frequency estimate, and the standard frequency for a WRLOOP was used while noting that its use could be conservative.

- *Crediting Strategies That Use Fire Protection Water as an Alternate Cooling Strategy for Diesel Generators*: Several plants studied in this analysis use water from the fire protection system as an alternate cooling source to their EDGs in the event that the normal supply of cooling water (ESW) becomes unavailable. For this analysis there would be potential that the same in-rush of debris that could cause an ESW strainer to clog could also impact a fire protection system in a similar way. However, there were too many variables to consider (different plant designs, differences in suction sources, relative distance between ESW intakes and fire water suction intakes) for the sake of this study. As a result, it was noted as a potential non-conservatism and uncertainty to assume fire water would not be impacted by the debris.

- *Modeling of FLEX Strategies:* The crediting of FLEX mitigation strategies has a significant impact on these analysis results. Considerable uncertainty is associated with various aspects of FLEX modeling in PRA. Factors such as the failure rates for FLEX components, how to credit operator actions, and modeling how the FLEX strategies will affect the plant during an accident sequence are key uncertainties for this analysis.

- *Plant Effects When a Strainer Is Bypassed:* It was not possible to accurately model the effect on the plant when a strainer is bypassed and unstrained cooling water is flowing through downstream heat exchangers, so this is another uncertainty. It is noted that during the derecho at DAEC, when operators bypassed one of the ESW strainers they found no degradation in service water flow to downstream components.

- *Differences in Individual Plants' Ultimate Heat Sink (UHS):* It was noted during the analysis that some plants were likely to be more susceptible to an in-rush of storm created debris than others. This because of the different characteristics of each plant. Some plants have rivers as their source for their UHS, while others use different types of reservoirs, lakes, or service water ponds. Although it is not possible to evaluate this factor quantifiably, some plants are likely more susceptible than others to this type of event.

- *Adjusting the CCF Parameters for ESW Strainers:* Often, since LIC-504 analyzes emerging issues for which the information on the magnitude of the degradation is limited, significant uncertainties may be associated with the estimated failure probabilities of potentially affected structures, systems, and components. In such cases, the best tools available for decisionmakers are sensitivity analyses. In Table 1, sensitivities have been performed at different CCF probabilities for the ESW strainers to fail since this was identified as a key source of uncertainty in this analysis.

- *Failure Probability Assigned to Operators Deciding Whether to Declare an Extended Loss of AC Power:* In scenarios where FLEX is considered, there is a basic event representing the probability that operators will fail to declare an ELAP when beneficial. Within the PRA model, this represents a key decision point for whether operators will choose to implement FLEX strategies during an SBO or pursue restoration of power. Recent experience studying this human error probability has indicated that a failure probability of between 1.1×10^{-3} to 1.6×10^{-2} would apply to this event in cases in which the plant procedures required operators to use judgment in that decision. The current SPAR models for this analysis used a value of 1×10^{-2} for each plant. A sensitivity was performed in which this value was lowered to 1×10^{-3}, however, the results for each plant did not change significantly, and this event was found not to influence the risk results.

Once the SPAR models were modified to reflect these changes, the increase in risk for a WRLOOP for each plant was estimated using ΔCDF as the metric. Changes to large early release frequency (ΔLERF) were not estimated, since earlier reviews had indicated that ΔCDF and not ΔLERF was the metric of concern.

Table 1 - ∆CDF for Each Plant with Various ESW Strainer CCF Probabilities

Plant	WRLOOP Baseline CDF	∆CDF at CCF of 0.1	∆CDF at CCF of 0.038	∆CDF at CCF of 0.01
Plant# 1: Westinghouse PWR	1.2×10^{-6}	8.4×10^{-5}	3.4×10^{-5}	1.1×10^{-5}
Plant# 2: CE PWR	2.6×10^{-7}	2.2×10^{-6}	1.1×10^{-6}	6.1×10^{-7}
Plant# 3: BWR-4	3.5×10^{-7}	7.1×10^{-7}	5.2×10^{-7}	4.3×10^{-7}
Plant# 4: Westinghouse PWR	9×10^{-7}	3×10^{-6}	1.8×10^{-6}	1.5×10^{-6}
Plant# 5: Westinghouse PWR	6.6×10^{-7}	1.9×10^{-5}	8.4×10^{-6}	3.5×10^{-6}
Plant# 6: BWR-6 with Mark III Containment	1.1×10^{-6}	3.8×10^{-6}	2.4×10^{-6}	1.8×10^{-6}
Plant# 7: BWR4 with Mark 1 Containment	2×10^{-7}	2.4×10^{-6}	1×10^{-6}	4.1×10^{-7}
Plant# 8: BWR-4 with Mark 1 Containment	4.8×10^{-6}	1.4×10^{-5}	1.2×10^{-5}	1.1×10^{-5}

FLEX

10 CFR 50.155 “Mitigation in Beyond-Design Basis Accident,” issued in response to Fukushima Daiichi event has further strengthened the ability of operating nuclear plants to withstand the impacts of natural hazards, such as a derecho which could cause an extended LOOP concurrent with challenges accessing the UHS. Specifically, FLEX strategies and equipment were significant in lowering the risk to each plant in this analysis.

Since the initiating event of concern is a WRLOOP without the possibility for recovering offsite power for 24 hours, combined with a challenge to the ESW system, the majority of accident sequences were SBO sequences that FLEX strategies were designed to mitigate. As a result, all plants studied had a substantial risk reduction from FLEX. Crediting FLEX remains a key source of uncertainty in this analysis. Table 2 presents the results of a sensitivity study showing the difference in CDF (risk reduction) for each plant crediting FLEX.

Table 2 – Sensitivity Study ∆CDF for Each Plant Showing the amount of Risk Reduction Provided by FLEX. (using the strainer CCF value of 0.038)

Plant	**∆CDF (FLEX strategies credited)**	**∆CDF (FLEX strategies not credited)**	**Difference in CDF made by FLEX (factor of reduction)**
Plant 1	3.4×10^{-5}	2.7×10^{-4}	2.3×10^{-4} (7.9)
Plant 2	1.1×10^{-6}	4.4×10^{-6}	3.3×10^{-6} (4.1)
Plant 3	5.2×10^{-7}	7.3×10^{-7}	2×10^{-7} (1.4)
Plant 4	1.8×10^{-6}	2×10^{-5}	1.8×10^{-5} (11.4)
Plant 5	8.4×10^{-6}	9.5×10^{-5}	8.7×10^{-5} (11.4)
Plant 6	2.4×10^{-6}	1.3×10^{-5}	1.1×10^{-5} (5.4)
Plant 7	1×10^{-6}	2.9×10^{-6}	1.9×10^{-6} (2.9)
Plant 8	1.2×10^{-5}	3.5×10^{-5}	2.2×10^{-5} (2.8)

Summary

After estimating the ∆CDFs, the staff reviewed the risk results and dominant PRA cutsets for each plant.

During the review of the results, it became clear that there were three mitigation features that significantly influenced plant risk for this event. Those were: (1) the ability of operators to bypass a clogged strainer if needed; (2) the ability to align fire protection water or another source of water to provide cooling to diesel generators; and (3) having additional diesel generators (not including FLEX diesels) that were not dependent on service water for cooling. Of the eight plants studied, the three plants that had the most of these mitigating features, had the lowest risk.

An additional beneficial feature that could reduce plant risk would be having a safety related power supply to traveling screens for the ESW system.
Table 3 provides additional details.

Table 3

Plant	Notes
1 Westinghouse PWR	This plant showed the highest risk of the eight plants analyzed with a ΔCDF of 8.4×10^{-5} when the strainer CCF was assumed to be 0.1. The ΔCDF lowered to 3.4×10^{-5} at 0.038 and 1.1×10^{-5} when the strainer CCF was lowered to 0.01. **Factors that increase risk**: The reason for the risk of this plant being higher compared to other plants is that fewer systems are available to help mitigate the combined WRLOOP and challenge to the service water system. As stated above, the features that could reduce the risk are the ability for operators to bypass strainers, an additional diesel generator (not including FLEX diesels), or the ability to align an alternate cooling source to their EDGs, and this plant does not have those features **Factors that reduce risk**: This plant has ESW traveling screens that have safety related power supplies and remain in service following a LOOP. A review of the cutsets (when strainer CCF is at 0.038) finds that 26% of the risk is from a dominant cutset in which the ESW strainers fail from CCF. This prevents cooling water from reaching the EDGs and they fail causing an SBO. The turbine driven auxiliary feedwater pump (TDAFW) fails to run, which leads to core damage before FLEX strategies can be implemented.
2 CE PWR	This plant had a ΔCDF of 2.2×10^{-6} when the strainer CCF was assumed to be 0.1. The ΔCDF was 1.1×10^{-6} at 0.038 and $6. \times 10^{-7}$ when the strainer CCF was lowered to 0.01. **Factors that increase risk**: This plant has service water strainers that cannot be bypassed. The top 3 cutsets, which collectively make up about 16% of the risk are non-SBO scenarios. The top cutset is due to a failure of the ESW strainers combined with Reactor Coolant Pump (RCP) seal failure, and failure of Reactor Coolant System (RCS) injection leading to core damage. In the next two cutsets the strainers fail, and then auxiliary feedwater subsequently fails leading to core damage **Factors that reduce risk**: This plant has multiple systems that would help in the event of interest. Including a nonsafety diesel that is air cooled and therefore doesn’t rely on service water for cooling. Operators can place this diesel in service during an SBO event. The availability of this additional diesel without a dependency on service water lowers the plant risk for this event. Additionally, this plant has multiple backup sources of cooling water for its EDGs if the service water system fails. These additional sources of cooling have only been partially credited in the SPAR model for this analysis, and additional modeling would only lower the risk further for this plant.

3 **BWR**-4 with Mark-I containment	This plant had a ΔCDF of 7.1×10^{-7} when the strainer CCF was assumed to be 0.1. The ΔCDF lowered to 5.2×10^{-7} at 0.038 and 4.3×10^{-7} when the strainer CCF was assumed to be 0.01. **Factors that reduce risk:** This plant has a small nonsafety supplemental diesel that does not rely on service water for cooling. Also, this plant has service water strainers that can be manually bypassed by operators if the strainers fail. That ability was already modeled in the SPAR model and helps lower the risk for the event. The top cutsets which account for 15% of the risk are not due to failures of the service water strainers, but result from the following: • A WRLOOP initiating event followed by failures of reactor core isolation cooling (RCIC) and combined with operators failing to initiate a manual reactor depressurization • A WRLOOP initiating event followed by a failure of RCIC and combined with operators failing to initiate a manual reactor depressurization, and a stuck-open solenoid operated relief valve (SRV)
4 **Westinghouse** PWR	This plant had a ΔCDF of 3×10^{-6} when strainer CCF was assumed to be 0.1. The ΔCDF lowered to 1.8×10^{-6} at 0.038 and 1.5×10^{-6} when strainer CCF was assumed to be 0.01. **Factors that reduce risk**: This plant has the ability for fire protection water to automatically (without operator action required) realign to supply cooling water to their diesels. This feature reduces the risk significantly in the event their service water system fails. This plant's service water system differs from the others in our study in that it does not have strainers. Instead each pump has an independent traveling screen and screen wash system with automatic controls. For the purpose of this study the decision was made to adjust the traveling screen CCF as a surrogate instead of adjusting the CCF for a strainer failure. The traveling screens have a smaller screen mesh than most other plants. Qualitative considerations show that the configuration of this plant's service water intake makes an in-rush of debris from a derecho less likely than at some other plants. The PRA cutsets reveal that the top 25% of the risk is due to scenarios in which a WRLOOP occurs, a the TDAFW pump fails, and subsequent random failures of the EDGs occur, causing an SBO. Without a TDAFW pump, the plant does not have time to implement FLEX, and core damage occurs. The CCF of the traveling screens (used as a surrogate for the ESW strainers) is not in the top 25% of the cutsets.

5 Westinghouse PWR	This plant had a ΔCDF of 1.9×10^{-5} when strainer CCF was assumed to be 0.1. The ΔCDF lowered to 8.4×10^{-6} at 0.038 and 3.5×10^{-6} when strainer CCF was lowered to 0.01. **Factors that increase risk**: This plant has strainers on their service water system that cannot be bypassed. **Factors that reduce risk**: This plant can provide an alternate cooling source to at least one of their EDGs. The process for this requires manual actions that could take some time to align. For the purpose of this analysis some credit for these manual actions were provided to credit this additional source which did reduce the risk for this plant. The PRA cutsets reveal that roughly the top 50% of the risk was due to the following scenario: A WRLOOP occurs, the ESW strainers fail which causes the EDGs to fail, and the plant enters an SBO. The alternate cooling source to the EDGs fails, and while FLEX is being deployed part of the FLEX strategy fails (either due to reactor makeup pumps, FLEX diesels, or operator errors) that leads to core damage.
6 **BWR**-6 with Mark III containment	This plant had a ΔCDF of 3.8×10^{-6} when the strainer CCF was assumed to be 0.1. The ΔCDF lowered to 2.4×10^{-6} at 0.038 and 1.8×10^{-6} when the strainer CCF was assumed to be 0.01. **Factors that reduce risk**: Operator action can manually bypass a clogged strainer if required, and the analysis gives credit for this. The top 30% of the risk came from the following PRA cutsets: • 17% of the risk was from a scenario in which a WRLOOP occurs, the ESW strainers fail from CCF, and operators fail to bypass the strainers which causes the EDGs to fail from lack of cooling. An SBO results and RCIC fails which leads to core damage before FLEX can be deployed. • 13% of the risk was from a scenario in which a WRLOOP occurs, the ESW strainers fail from CCF, and operators fail to bypass the strainers which causes the EDGs to fail from lack of cooling. An SBO results and SRV fails to shut.
7 **BWR**-4 with Mark-1 containment	This plant had a ΔCDF of 2.4×10^{-6} when the strainer CCF was assumed to be 0.1. The ΔCDF lowered to 1×10^{-6} at 0.038 and 4.1×10^{-7} when the strainer CCF was assumed to be 0.01. **Factors that reduce risk:** This plant has more EDGs than the other plants studied and can use the fire protection water as a backup cooling source for its EDGs in case service water fails. Both these features lowered risk significantly. The top PRA cutset accounts for 6% of the risk. In this scenario a WRLOOP occurs and the EDGs fail from CCF unrelated to service water. The failed EDGs cause an SBO, and RCIC fails to run. With the failure of RCIC FLEX cannot be implemented and core damage occurs.

8 BWR-4 with a Mark-1 containment	This plant had a ΔCDF of 1.4×10^{-5} when the strainer CCF was assumed to be 0.1. The ΔCDF lowered to 1.2×10^{-5} at 0.038 and 1.1×10^{-5} when the strainer CCF was assumed to be 0.01. **Factors that increase risk**: This plant doesn't have an alternate cooling strategy for their EDGs or additional diesel generators. **Factors that reduce risk**: This plant does have the ability for operators to bypass the strainers. The SPAR model provided credit for this. Cutsets show that failure of the ESW strainers is not in the top 10 cutsets. Of the risk, 8% is from a dominant cutset that is a WRLOOP with random failures of the EDGs causing an SBO. RCIC fails to run, and the failure of RCIC leads to core damage. The next dominant cutset (6% of risk) is from a WRLOOP, with the EDGs failing to run, causing an SBO. An SRV fails to close, which causes core damage before FLEX can be deployed.

ENCLOSURE 2

Summary of the Evaluation of Recommendations

Table C-1 of LIC 504: Decision Options

#	Option[a]	Analysis Approach[b]	Affected Principles or Factors[c]	Criteria Used to Evaluate Options[d]	Evaluation[e]
1	Take no follow-up actions.	Used quantitative and qualitative guidance in 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).	Not analyzed since this option is not recommended.	Guidance in LIC-504.	Not recommended because risk analysis for eight plants highlighted potential opportunities to enhance plant safety and enhance the quality of the U.S. Nuclear Regulatory Commission's (NRC's) risk-informed decisionmaking activities.
2	Issue orders to promptly shut down or implement compensatory measures at nuclear power plants (NPPs) that may be vulnerable to events similar to the derecho that occurred at the Duane Arnold Energy Center (DAEC) NPP.	Used conservative upper bound results generated using the NRC's Standardized Plant Analysis Risk (SPAR) models.	Not analyzed since this option is not recommended.	LIC-504 criteria on conditional core damage frequency (CCDF), conditional large early release frequency (CLERF), and guidance on defense in depth (DID) and plantwide safety (SM).	Not recommended because conservative upper bound evaluations demonstrated that CCDF was less that $1x10^{-3}$/year and CLERF was less than $1x10^{-4}$/year. Also, the issue does not contribute to significant degradation of DID plantwide and SM (see ADAMS Accession No. ML20315A117 for details).
3.	Issue orders requiring licensees to comply with design requirements.	Used input from DAEC inspection report to determine whether insights	Not analyzed since this option is not recommended.	Design- and licensing-basis requirements for DAEC.	Not recommended because follow-up inspections did not identify violations of design requirements, procedure noncompliances, or

		obtained from DAEC derecho event demonstrate potential noncompliances with system design requirements.			inadequate quality assurance or maintenance (see ADAMS Accession No. ML20314A150).
4	Proceed with a cost-benefit portion of a regulatory analysis in accordance with guidance in 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).	Used maximum change in core damage frequency (ΔCDF) gains that can be achieved at various plants analyzed, considering uncertainties, conservatisms, and nonconservatisms.	Not analyzed since this option is not recommended.	Information provided in Figure 2-2 of NUREG\BR-0058, Revision 5.	Not recommended since the ΔCDFs calculated using readily available information have high uncertainties, conservatisms, and nonconservatisms, and implementation of recommendations #7, 8, 11, 12, and 13 could be sufficient to achieve the desired potential safety gains. Furthermore, these ΔCDFs represent safety gains in the event a modification is able to completely eliminate the potential increase in risk. A cost-benefit portion of a regulatory analysis should be based on the subset of this ΔCDF that can be eliminated by a proposed modification. Such an analysis is outside the scope of an LIC-504 analysis. Management could revisit the need to perform the cost-benefit portion of the regulatory analysis based on insights obtained from implementing the five recommendations (items 7, 8. 11, 12, and 13).
5	Issue bulletin requiring licensees to determine their potential vulnerability to events such as the DAEC derecho event and implement potential safety enhancements to	Used refined ΔCDF results generated using SPAR models for eight NPPs and potential impacts on DID and plantwide SM and NRC Office of	Not analyzed since this option is not recommended.	NRR generic communication guidance on circumstances that should prompt issuance of bulletins and LIC-504 guidance (Figure 3).	Not recommended because the estimates of risk significance do not warrant requiring prompt actions especially in light of significant uncertainties in risk estimates and availability of alternative means (e.g., issuance of an information notice) that can be employed to

	reduce risks from events that could cause loss of power (LOOP) and concurrently degrade emergency service water (ESW) performance.	Nuclear Reactor Regulation (NRR) guidance on generic communications.			achieve desired outcomes.
6	Issue generic letter requesting licensees to provide information that the NRC staff could use to determine whether follow-up regulatory actions are warranted.	Used refined CCDF and CLERF results generated using SPAR models for eight NPPs and potential impacts on DID and plantwide SM and NRR guidance on generic communications.	Not analyzed since this option is not recommended.	NRR guidance on circumstances that should prompt issuance of generic letters and LIC-504 guidance (Figure 3).	Not recommended since circumstances do not warrant issuance of a generic letter. For example, analysis performed from readily available information demonstrates that the issue may be of significance to a handful of operating units and, as such, issuance of a generic letter is unnecessary. Furthermore, the NRC can use alternative means (e.g., information notice) to achieve desired outcomes.
7	Issue an information notice informing licensees about the event and factors that influence the risk significance based on insight gained from the NRC's LIC-504 analysis.	Used refined ΔCDF results generated using SPAR models for eight NPPs and potential impacts on DID and plantwide SM and NRR guidance on generic communications.	Availability of information would enable licensees to examine whether there are opportunities to enhance safety at their plants.	NRR guidance on circumstances that should prompt issuance of information notices and LIC-504 guidance relating to issuance of generic communications.	Recommended since issuance of an information notice may prompt licensees to explore potential safety enhancements. For example, a site that has several factors that could increase risk (e.g., located in a region with relative high propensity to weather-related LOOPs, not equipped with the capability to bypass ESW strainers) may find opportunities to enhance safety and, if appropriate, update the probabilistic risk assessment (PRA) model.
8	Examine how industry response to Fukushima-related orders on extended loss of alternative current power and loss of ultimate heat sink	Used refined ΔCDF results generated using SPAR models for eight NPPs with and without credit for FLEX strategies.	Insights from such an examination may demonstrate means to maximize risk	Differences in ΔCDF with and without credit for FLEX strategies (see Enclosure 1 for details).	Recommended since it demonstrates safety gains attributed to FLEX strategies and the means to maximize those benefits and provides insights on opportunities for improvement.

	(EA-12-049) impacted the risk significance of this issue and identify opportunities to further enhance reliability of diverse and flexible mitigation capability (FLEX) strategies.		reductions that could be achieved from FLEX strategies.		
9	Perform additional performance monitoring.	Used risk analysis results to identify key uncertainties that may significantly impact estimated risks.	Not analyzed since this option is not recommended.	NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking, Final Report,” Revision 1, issued March 2017 (ADAMS Accession No. ML17062A466), on how performance monitoring could be used as a means to address key uncertainties associated with risk assessment.	Not recommended because the risk assessment did not identify any issues whose uncertainties could be addressed by additional performance monitoring.
10	Perform additional inspections at more plants.	Used risk analysis and related operating experience to identify benefit of additional inspections.	Not analyzed since this option is not recommended.	Operating experience with respect to derechos and regulatory requirements.	Not recommended since the team did not identify issues of potential noncompliances (violations of regulations, technical specifications).
11	Communicate risk insights gleaned from the DAEC LIC-504 with	Discussed options available to NRR’s Division of Reactor	Availability of information would enable	NRR/DRO/IOEB guidance on selecting information	Recommended because availability of risk insight will enable regions to enhance use of budgeted inspection

	regional staff and NRR staff.	Oversight (DRO), Operating Experience Branch (IOEB), to share information with regional staff.	inspectors to examine whether there are opportunities to enhance safety at their plants.	that must be shared with regional staff.	resources in a risk-informed manner at a few plants where there may be opportunities to enhance safety.
12	Share risk insights gained from the DAEC accident sequence precursor and the LIC-504 analysis with the regulated community.	Considered benefits that can be obtained by sharing risk insights with regulated communities who have the ability to influence risk-informed decision making in the regulated community.			Recommended because it would enhance the regulated community's awareness of the importance of a combined loss of ESW and LOOP for a few plants.
13	During fiscal years 2021 and 2022, update two SPAR models that possess multiple design characteristics that yield relatively higher risk estimates.	NRR's Division of Risk Assessment (DRA), PRA Oversight Branch (APOB), has an ongoing action to identify and update several SPAR models. NRR\DRA\APOB has identified two plants for updating in fiscal years 2021 and 2022.	Improving accuracy of SPAR models enhances the NRC staff's risk-informed decisionmaking capabilities.	General guidance used by NRR\DRA to identify a limited number of SPAR models for future updates.	Recommended because it would enable the staff to further enhance its understanding of the risks associated with derechos.

a. Define each decision option (e.g., shut down plant immediately, shut down in specified time period, or disallow a plant from restarting).
b. Identify available analytical tools (quantitative or qualitative), such as risk analysis tools or engineering models.
c. Identify potential impact on the principles of risk-informed decisionmaking or other factors being analyzed or evaluated to differentiate the options.
d. Define the basis or standard for accepting or rejecting each decision option.
e. Compare the options and justify the option recommended for implementation.