Supporting Information for PRE-GI-020:
"Inadequate Procedures to Address Anticipated Operational Occurrences"

Description of Proposed Issue

On January 3, 2018, the U.S. Nuclear Regulatory Commission (NRC) Region IV staff submitted a proposed generic issue (GI), PRE-GI-020, "Inadequate Procedures to Address Anticipated Operational Occurrences," related to licensees that do not have appropriate operator procedures to address anticipated operational occurrences (AOOs), such as an inadvertent start of engineered safety features actuation system (ESFAS) equipment. Missing or inadequate procedures could adversely affect the operating crew's ability to take appropriate actions to ensure that reactor safety is properly maintained. The proposed GI specifically notes that Region IV inspectors received information that other nuclear plants are missing off-normal (abnormal) procedures to address similar AOOs.

The NRC staff entered PRE-GI-020 into the Generic Issues Management Control System. Details associated with PRE-GI-020 can be viewed on the NRC's public Web site through the Generic Issues Dashboard at http://gid.nrc.gov/Planning. The staff also entered the submittal into the Agencywide Documents Access and Management System (ADAMS) under Accession No. ML18019A703. The GI Program staff acknowledged receipt of the proposed GI in a memorandum dated February 16, 2018 (ADAMS Accession No. ML18044A398).

An NRC inspection of River Bend Station initially identified the proposed issue. On October 17, 2016, the NRC Region IV staff issued NRC Examination Report 05000458/2016301 (ADAMS Accession No. ML16291A546), giving River Bend Station a noncited violation of Criterion V, "Instructions, Procedures, and Drawings," in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The inspection report cited 10 examples of the licensee's failure to provide appropriate qualitative and quantitative criteria in alarm response procedures and abnormal operating procedures. The NRC Region IV staff assessed this finding as more than minor because it was associated with the procedure quality attribute of the mitigating systems cornerstone and adversely affected the cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to initiating events to prevent undesired consequences. Specifically, inadequate procedures could adversely affect the ability of the operators to take the appropriate actions to ensure that reactor safety is being maintained. The NRC Region IV staff believed that this type of event should require entry into an appropriate abnormal operating procedure in place. However, the NRC Region IV staff found that licensees do not have such a specific procedure in place. Some licensees believe that their operators are well trained to respond using skill of the craft, relying on their knowledge and skills instead of specific abnormal operating procedures. Some licensees do not train on these type events and discourage their inclusion on NRC examinations.

Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the NRC staff determined that the finding was of very low safety significance (green) because of the following:

- The finding was not a deficiency affecting the design and qualification of a mitigating structure, system, or component and did not result in a loss of operability or functionality.

- The finding did not represent a loss of a system or function.

- The finding did not represent an actual loss of function of at least a single train for longer than its outage time allowed by the technical specification or two separate safety systems out of service for longer than the outage time allowed by the technical specification.

- The finding did not represent an actual loss of function of one or more nontechnical specification trains of equipment designated as high safety significance in accordance with the licensee's maintenance rule program for greater than 24 hours.

Literature Search

The GI Program staff conducted a literature search to assess any related work and to determine whether the issue had been previously addressed. The GI Program staff's search revealed several reported industry events related to inadvertent emergency core cooling system (ECCS) injections but did not find any actions that would specifically resolve the primary concerns identified in this proposed GI.

Previous Event Notification Submittals

- Event Notification (EN) 53365, "Inadvertent Injection of High-Pressure Core Spray," dated April 26, 2018 (River Bend Station) (available at https://www.nrc.gov/reading-rm/doc-collections/event-status/event/2018/20180427en.html#en53365)

- Licensee Event Report (LER) 98-001, "RCS Outside Design Basis for Inadvertent ECCS Actuation at Power, Diablo Canyon," dated October 22, 1998, for which the reactor coolant system (RCS) was outside the design basis for inadvertent ECCS actuation at power because of nonconservative assumptions for pressurizer safety valve (PSV) operation (ADAMS Accession No. ML16342A553)

- NRC Examination Report 05000458/2016301, which documents operator examination results for River Bend Station, dated October 17, 2006 (ADAMS Accession No. ML16291A546)

- Safety Evaluation Report (SER) 39-82, "Inadvertent ECCS Actuation and Reactor Power Excursion," dated June 19, 1982 (Peach Bottom Atomic Power Station) (ADAMS Accession No. ML12233A474 (not publicly available))

- Operating Experience (OpE) COMM: Browns Ferry Nuclear Plant, 2011–2017, High-Pressure Core Injection (HPCI) Performance (ADAMS Accession No. ML17165A182 (not publicly available))

- Special Investigation Team (SIT) Report, "Browns Ferry Nuclear Plant, Units 2 and 3—NRC Special Inspection Report 05000260/2017008 and 05000296/2017008," dated November 29, 2017 (ADAMS Accession No. ML17333A173)

Previous Relevant GIs Submitted

The staff's search of previous GIs in NUREG-0933, "Resolution of Generic Issues," issued December 2011, identifies the following GIs that dealt with crediting operator actions to mitigate accidents and transients. However, the staff did not find any previous GIs that would specifically address the concerns identified in this proposed GI.

- Generic Safety Issue 8, "Inadvertent Actuation of Safety Injection in PWRs." The Advisory Committee on Reactor Safeguards (ACRS) identified a concern that a high rate of spurious or inadvertent safety injections (SIs) were occurring. Actuation of the SI system injects cold borated water into the reactor, which subjects injection nozzles to thermal stresses. Repeated inadvertent SI injections may induce future failures. After the Three Mile Island accident, the NRC greatly emphasized the need to verify the safe status of the plant thoroughly before terminating SI. All licensees were required to include a specific set of SI termination criteria in their emergency procedures.

  Thermal fatigue of the SI nozzle may be only a matter of economics. The safety concern with inadvertent SI actuation involves the possible operator unacceptable response that could result in the termination of SI injection in cases where it is required. Repeated exposure to spurious SIs may improperly condition operators to terminate the SI injection and reset the SI injection signal without carefully assessing the status of the reactor plant and the real need for emergency cooling.

  Subtask I.C of NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident," issued May 1980, addresses problems of this nature. The Subtask I.C.1 objective is to improve operating procedures to provide greater assurance that operator and staff actions are technically correct, explicit, and easily understood for normal, transient, and accident conditions. A principal part of Subtask I.C.1 is to improve procedures for dealing with abnormal conditions and emergencies by better delineating the symptoms, events, and plant conditions that identify emergency and off-normal situations that confront the operators.

- Generic Safety Issue, Item A-47, "Safety Implications of Control Systems." The purpose of Item A-47 was to perform an indepth review of the nonsafety-related control systems and to assess the effect of control system failures on plant safety. To this end, the NRC staff established tasks to identify potential control system failures that, either singly or in selected combinations, could cause overpressure, overcooling, overheating, overfilling, or reactivity events. The staff recommended that plants provide systems to protect against reactor vessel/steam generator overfill events and to prevent steam generator dry out.

- Generic Safety Issue, Item B-17, "Criteria for Safety Related Operator Actions." NUREG-0471, "Generic Task Problem Descriptions (Category B, C, and D Tasks)," issued June 1978, involves the development of a time criterion for safety-related operator actions (SROAs), including a determination of whether automatic actuation was required. However, this item dealt only with the switchover of ECCS injection from the injection mode to the recirculation mode following a loss-of-coolant accident (LOCA). When this issue was identified in 1978, existing plant designs relied on operators to take manual actions in response to certain transients as necessary. In addition, some existing pressurized-water reactor (PWR) designs required manual operations to accomplish the switchover from the injection mode to the recirculation mode following a LOCA. Development and implementation of criteria for SROAs would result in the automation of some actions that operators were performing. The use of automated redundant safety-grade controls in lieu of operator actions was expected to reduce the frequency of improper action during the response to or recovery from transients and accidents by removing the potential for operator error. Plants would be required to perform task analyses, simulator studies, and analyses and evaluations of operational data to assess existing engineered safety features (ESFs) and safety-related control system designs for conformance to new criteria. Where nonconformance was identified, modification to existing designs and hardware would be required.

In resolving the issue, the staff concluded that the following actions taken by licensees in response to regulatory requirements imposed since the issue was identified address the safety concern:

- enhancement of operator training and licensing requirements, including plant-specific simulators

- improvement of training based on the "Systems Approach to Training" for all covered staff

- implementation of symptom-based emergency operating procedures

- completion of the Individual Plant Examination (IPE) program at all operating plants

- <u>Generic Safety Issue 70, "PORV and Block Valve Reliability</u>." Power-operated relief valves (PORVs) and block valves were originally designed as nonsafety components in the reactor pressure control system. The block valves were installed because of expected leakage from the PORVs. Neither the PORVs nor the block valves were required to safely shut down a plant or mitigate the consequences of accidents. In 1983, the staff found that plants were relying on PORVs to mitigate a design-basis steam generator tube rupture (SGTR). Generic Safety Issue 70 addressed the acceptability of relying on nonsafety-grade PORVs to mitigate a design-basis accident and the need for improving the reliability of PORVs and block valves. These valves are sometimes used to mitigate certain design-basis accidents and transients, to reduce safety valve challenges, and to potentially help mitigate the effects of an anticipated transient without SCRAM (ATWS). A small-break LOCA through this system and the resulting challenges to safety systems appeared to be of sufficient frequency that, based on the above evaluation, improved reliability of the PORV/block-valve system might reduce a potential public risk.

<u>Applicable Regulations and Guidance</u>

- Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes minimum requirements for the principal design criteria for those water-cooled nuclear power plants that were licensed under Appendix A general design criteria (GDC). Plants licensed before the NRC incorporated the GDC into 10 CFR Part 50 may have similar plant-specific minimum requirements for the principal design criteria specified in their licensing bases.

- GDC 15, "Reactor Coolant System Design," of Appendix A to 10 CFR Part 50 requires that the RCS and its associated auxiliary control and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operations, including AOOs.

- GDC 21, "Protection System Reliability and Testability," of Appendix A to 10 CFR Part 50 requires, in part, that the protection system be designed to ensure that no single failure results in a loss of the protection function.

- GDC 29, "Protection against Anticipated Operational Occurrences," of Appendix A to 10 CFR Part 50 requires that the protection and reactivity control systems be designed to ensure an extremely high probability of accomplishing their safety functions in the event of AOOs.

- 10 CFR 50.34(b), "Final Safety Analysis Report," requires a final safety analysis report to include the following:

  > A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

- Chapter 15, "Accident Analyses," of Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," provides guidance as to which AOOs should be included in the safety analysis reports. The NRC issued similar guidance for the type of information that combined licenses (COLs) should include under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," and in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), is intended to be a comprehensive and integrated document that provides the NRC staff with guidance that describes methods or approaches that the staff has found acceptable for meeting NRC requirements.

- SRP Section 15.0, "Introduction—Transient and Accident Analyses," describes the categorization of accidents and transients used by the NRC staff to review Chapter 15 safety analyses in final safety analysis reports or updated final safety analysis reports (UFSARs).

- SRP Section 15.0, Item I.1.A, states that incidents of moderate frequency and infrequent events are also known as Condition II and Condition III events, respectively, as defined in American Nuclear Society (ANS) 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," and ANS 52.1-1978, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants."

- SRP Section 15.0, Item 2.A(iii)(1)(c), states that "by itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV category without other incidents occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers." This is otherwise referred to as the "nonescalation criterion."

- SRP Sections 15.5.1–15.5.2, “Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction That Increases Reactor Inventory,” states that certain AOOs can cause an unplanned increase in the RCS system inventory that could lead to an increase in the power level and to fuel damage or overpressurization of the RCS without adequate controls.

- RG 1.33, “Quality Assurance Program Requirements (Operation),” Appendix A, “Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors,” lists typical safety-related activities that a licensee should address in its procedures. Most licensees incorporated a commitment to RG 1.33, Revision 1, issued January 1977, into their plant technical specifications during initial plant licensing.

- RG 1.33, Revision 2, issued February 1978, endorsed American Nuclear Society (ANS) 3.2/American National Standards Institute (ANSI) N18.7-1976, “Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants.”

- ANSI/ANS 3.2-2012, “Managerial, Administrative, and Quality Assurance Controls for Operational Phase of Nuclear Power Plants,” replaced ANS 3.2/ANSI 18.7-1976. It incorporated operational experience since the original standard was developed and removed information related to design and construction.

- ANS 51.1-1983, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants,” replaced ANSI N18.2-1973 of the same title. It stated that incidents of moderate frequency and infrequent events are also known as Condition II and Condition III events, respectively, as defined in American Nuclear Society.

- ANSI/ANS 52.1-1978, “Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants”, 1978, January 4, 2018 provide similar criteria as ANS 51.1 but for boiling water reactors (BWRs).

- RG 1.33, Revision 3, issued June 2013, endorses ANSI/ANS 3.2-2012, consistent with American Society of Mechanical Engineers Nuclear Quality Assurance 1, “Quality Assurance Requirement for Nuclear Facility Applications,” and incorporated alternate positions approved by the NRC since the issuance of ANS 3.2/ANSI 18.7-1976. RG 1.33, Revision 3, no longer includes Appendix A, which listed activities that a licensee should address in its procedures.

- RG 1.203, Revision 0, “Transient and Accident Analysis Methods,” issued December 2005, describes a process that the staff considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

Applicable Regulatory Actions

The recent regulatory actions and requirements associated with the issues identified in the proposed GI are summarized below.

The main issue of the proposed GI is determining whether a licensee should have adequate procedures and training to address AOOs that were not directly specified in RG 1.33, Appendix A. When a plant does not have automatic actuations to respond to a spurious start of plant equipment, the plant has to rely on operators to take the appropriate and timely manual actions to prevent the AOO from escalating to a Condition III or IV event. The NRC has provided specific guidance for licensees on when they can credit operator actions and on the requirements for operators to correctly respond in a timely manner.

Licensees' UFSARs describe the original design of nuclear power plant safety systems and the systems' ability to respond to design-basis accidents. The NRC reviewed and approved these UFSARs. Most safety systems were designed to rely solely on automatic actuations to ensure that the safety systems were capable of carrying out their intended functions. In a few cases, limited operator actions were appropriately justified and approved. The NRC must approve proposed changes that substitute manual action for automatic system actuation or that modify existing operator actions, including operator response times that the agency previously reviewed and approved during the original licensing review of the plant. Licensed plants are expected to have the appropriate measures (i.e., procedures, trainings, job performance measures) to validate operator actions that they credit to mitigate transients and accidents to prevent the escalation of events. Licensees use symptom-based guidance in the formulation of their procedures and may not provide specific instructions on how their plants perform an override. Licensees' "Conduct of Operations" procedures may include guidance for overriding systems in the event of an inadvertent safety system actuation.

As part of the Reactor Oversight Process (ROP), the NRC reviews licensees' analyses involving credit for operator actions. Inspectors use the criteria in SECY-99-007A, "Recommendations for Reactor Oversight Process Improvements (Follow-Up to SECY-99-007)," dated March 22, 1999. SECY-99-007A states that if operator actions are required to credit placing mitigation equipment in service or to perform recovery actions, such credit should be given only if the following criteria are met:

- Sufficient time is available to implement these actions.
- Environmental conditions allow access where needed.
- Procedures exist.
- Training is conducted on the existing procedures under conditions similar to the scenario assumed.
- Any equipment needed to complete these actions is available and ready for use.

The NRC Inspection Manual contains Inspection Manual Chapters (IMCs) numbered 0000 through 1999 that are used for policy statements on inspection programs. Specific requirements are placed in an inspection procedure to verify that licensees continue to satisfy licensing commitments identified in their response to a generic communication, such as a bulletin, or when policy changes are implemented (e.g., through the Commission's direction or in response to an audit by the Office of the Inspector General). The NRC's Division of Reactor Inspection (DIRS) and Regional Support are responsible for ensuring that licensees adhere to NRC policies and licensing commitments.

Regional NRC inspectors conduct routine inspections at licensee facilities on how operators respond to accidents and transients. The literature search pointed out that a Region IV inspector issued a violation to River Bend Station on October 17, 2016, for lack of procedure quality (e.g., lack of a procedure to respond to an AOO). When the NRC staff evaluated the risk under the risk significance determination process, it determined that the finding had very low safety significance (i.e., green). Because of the very low safety significance, the finding was treated as a noncited violation. Elsewhere, on October 3, 2017, the NRC staff conducted a special inspection at Browns Ferry Nuclear Plant to review the circumstances surrounding the failure of an HPCI system discharge valve that resulted in an unintentional injection into the reactor vessel during a normal inservice testing (IST) surveillance. These examples represent only a very few instances in which NRC inspectors are taking actions necessary to ensure that licensees are meeting commitments within their licenses and are conducting additional inspections when unusual circumstances arise.

During the review of several regulatory actions, the staff identified several instances where inadvertent ECCS actuations may have the potential to escalate a Condition II event into a Condition III or IV event. The most significant relevant actions involve Westinghouse Nuclear Safety Advisory Letter (NSAL)-93-013, “Inadvertent ECCS Actuation at Power,” dated June 30, 1993; NRC Regulatory Information Summary (RIS) 2005-29, “Anticipated Transients That Could Develop into More Serious Events,” dated December 14, 2005 (ADAMS Accession No. ML051890212); and the appeal of the Byron Station and Braidwood Station PSV backfit. However, none of the regulatory actions specifically addressed whether there were existing procedures, whether the existing procedures were adequate, whether additional procedures were required, or a verification of adequate training. The sections below summarize the related regulatory actions to draw a correlation with the proposed issue questioning imposing requirements on licensees to have procedures to address certain AOOs.

Recent Regulatory Actions

Westinghouse NSAL-93-013 and NRC RIS 2005-29

On June 30, 1993, Westinghouse issued NSAL-93-013 describing the underlying technical issue on PSV performance following a water discharge at PWRs. NSAL-93-013 explains that most PSVs in PWRs are designed for only steam service and that water discharge through such valves is undesirable and should be minimized or avoided as a matter of conservative engineering and prudent operations. This perspective is reinforced by several industry positions and through testing, as well as operator training, and control room procedures intended to terminate a potential pressurizer overfill event before the pressurizer goes water solid. Westinghouse issued NSAL-93-013 to recommend a variety of methods used to achieve compliance. However, the NRC staff found certain Westinghouse recommendations unacceptable. The NRC identified this concern in RIS 2005-29, as described below.

The NRC issued RIS 2005-29 to notify licensees of a concern that it had identified during recent reviews of power uprate license amendment requests. RIS 2005-29 stated that the licensing bases of certain licensees, as described in the plants’ UFSARs, failed to demonstrate that their plant designs and emergency operating procedures effectively preclude the initiation of any Condition III or IV events resulting from relatively frequent Condition II events (e.g., the inadvertent ECCS actuation event). The intent of RIS 2005-29 was to ensure that licensees properly considered this concern to help facilitate the NRC staff’s review of future amendment requests, such as power uprates and fuel transitions.

During subsequent reviews of license application requests, the NRC staff observed that applications involving certain safety analyses, specifically in Chapter 15 of UFSARs, failed to meet the nonescalation criterion for the following three mass addition Condition II events: (1) the chemical and volume control system (CVCS) malfunction, (2) the inadvertent operation of the ECCS during normal operation, and (3) the inadvertent opening of a pressurizer PORV or a PSV. The NRC staff was concerned that these Condition II events (AOOs) could escalate to a small-break LOCA, which is a Condition III event that would not meet the nonescalation criterion. In July 2005, the NRC staff issued draft Revision 1 to RIS 2005-29 for public comment (ADAMS Accession No. ML15014A469). The staff received numerous comments; however, it did not finalized Revision 1 for issuance.

Exelon Appeal of Byron Station and Braidwood Station Backfit for Pressurizer Safety Valves

In October 2015, during a review of the design basis for the performance of PSVs at Byron Station and Braidwood Station, the NRC staff issued a safety evaluation attempting to impose a compliance backfit (ADAMS Accession No. ML14225A871). In December 2015, the licensee appealed the decision (ADAMS Accession No. ML15342A112). In May 2016, the NRC first responded to the licensee's appeal, stating that after forming a panel the staff still believed Braidwood and Byron were not in compliance with 10 CFR 50.34(b) (ADAMS Accession No. ML16095A204). In June 2016, the licensee again appealed the NRC staff's decision (ADAMS Accession No. ML16154A254). The Executive Director of Operations (EDO) formed another review panel; and based upon its conclusions, in September 2016, the EDO issued a memorandum finding in favor of the licensee (ADAMS Accession No. ML16243A067). In addition, the EDO instructed the staff to reassess RIS 2005-29 and proposed Revision 1 (ADAMS Accession No. ML16246A247). The staff responded to the EDO that it would consider issuing a new RIS that would provide updated staff positions on mass addition events (ADAMS Accession No. ML17219A035). The relationship of the proposed GI to the backfit appeal is discussed in further detail below.

The proposed GI centers on the consequences of a failure by operators to take appropriate actions during such events as an unintended mass addition like a spurious ECCS injection. The recent activities involving the NRC staff's issuance of a compliance backfit on Byron Station and Braidwood Station and the subsequent reversal are relevant to the circumstances described in the proposed GI. The backfit appeal evaluation centered on whether the staff can impose a new requirement on the licensee to prevent the plant from filling the pressurizer water solid and subsequently relieving water through the PSVs causing them to fail open.

Upon receiving an appeal letter from the licensee, the EDO established a backfit appeal review panel. The panel published its findings, titled "Backfit Appeal Review Panel Findings Associated with Byron and Braidwood Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis," on August 24, 2016 (ADAMS Accession No. ML16236A198). The following relevant statements from the panel's report are applicable to the scenarios described in the proposed GI:

> Based on its review, the Backfit Panel concluded that the NRC staff positions taken to support the compliance backfit finding represent new and different staff views on how to address potential pressurizer safety valve failures following water discharge. Even though the conservative approach could provide additional safety margin, it does not provide a basis for a compliance backfit. The panel did note that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to Byron and Braidwood.
>
> The Panel supports the NRC staff's view that non-escalation (from Condition II to Condition III or IV, as defined in American Nuclear Society

> Standard 51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," dated August 6, 1973) is a known and established standard applicable to Byron and Braidwood. However, this event progression standard does not establish specific standards for valve qualification. Therefore, it is not the basis for a compliance backfit given this set of facts.
>
> Staff PRA analysis for the backfit Panel provided insights on the risk significance of the sequence at issue. This PRA analysis suggests that an inadvertent ECCS actuation sequence, assuming that pressurizer overfill would lead to a small loss-of-coolant accident, contributing approximately 1 percent of the total internal events core damage frequency (CDF). If the backfit were implemented such that the pressurizer safety valves would always reclose properly, the CDF reduction is estimated at 1.5E-07 per year.

Following the issuance of the findings from the backfit appeal panel, the EDO issued a memorandum, titled "Result of Appeal to the Executive Director for Operations of Backfit Imposed on Byron and Braidwood Stations Regarding Compliance with 10 CFR 50.34(B), GDC 15, GDC 21, GDC 29, and the Licensing Basis," dated September 15, 2016 (ADAMS Accession No. ML16246A247). The EDO stated that an adequate basis must exist for backfitting in order for the staff to use the compliance exception in 10 CFR 50.109(a) 4(i). The EDO stated that the staff's positions in 2001 and 2004 were consistent with multiple NRC approvals both before and since; and licensees do not need to assume that the subsequent failures of PSVs occur following a water discharge if the likelihood of failure is sufficiently small and based on well-informed staff engineering judgment. Without the presumption of a PSV failure to reseat, the concerns in the backfit safety evaluation related to compliance with 10 CFR 50.34(b) and GDC 15, 21, and 29 were no longer an issue. Therefore, the EDO did not support the staff's use of the compliance exception to impose the subject backfit and stated that the current licensing bases for Byron Station and Braidwood Station comply with the applicable regulations and provide adequate protection of public health and safety.

Following the issuance of the EDO's backfit appeal memorandum, the Office of Nuclear Reactor Regulation (NRR) staff were instructed to develop a plan to assess the safety significance of the underlying technical issue described in Westinghouse NSAL-93-013 and the positions in RIS 2005-029 and the proposed revision. On September 6, 2017, the NRR staff responded to the EDO's decision in a memorandum, titled "Supporting Information for Staff Recommendations in Response to Executive Director for Operations Tasking in September 15, 2016, Exelon Backfit Appeal Decision" (ADAMS Accession No. ML17237C035 (not available in public ADAMS)). The following important statements from the staff's memorandum are relevant to this proposed issue:

> In the event the PSVs are not demonstrated to reliably close following liquid discharge, the reactor coolant system (RCS) mass addition event could transition into a small break loss-of-coolant accident (LOCA). This represents a transition from a Condition II event to a Condition III event, which violates non-escalation language included in the current licensing bases for many plants. However, the consequences of a failure of a PSV to close are bounded by small break LOCA analyses, and licensees have demonstrated that offsite doses as a result of a small break LOCA are within the regulatory limits and do not adversely impact the health and safety of the public. Although failure to prevent this escalation would increase the frequency of a small break LOCA to beyond that assumed for

a Condition III event, the consequences of the event, i.e., offsite dose, should not change.

In evaluating the safety significance of the issue, the staff considered whether the compliance exemption, or adequate protection backfit rule were applicable. The staff found that a compliance exception was not applicable because the various NRC decisions and communications associated with relief valve qualification for liquid discharge in the past were not consistent enough to represent a known and established standard.

An assessment of risk was also performed in support of the Byron/Braidwood backfit appeal (ADAMS Accession No. ML16214A199). The results showed that risk benefit from implementing precautionary measures was very small, 1.5E-7. This insight led the staff to believe that a cost-benefit analysis would demonstrate that costs of requiring PSV liquid qualification associated with imposing a backfit would not be justifiable in light of the safety benefits. This insight along with the fact that the consequences of transient are bounded by an existing analyzed event, i.e., small break LOCA, led the staff to determine that the adequate protection exception could not be applied.

In the backfit appeal, the staff could not show that an established known NRC position existed; therefore, the compliance exception in 10 CFR 50.109, "Backfitting" (the Backfit Rule), could not be applied. Additionally, the results of the backfit appeal show that this mass addition event had a minimal impact on safety and was bounded by the small-break LOCA analyses in the licensee's design basis. After considering the costs of the qualification and testing of PSVs to credit operability following a liquid discharge compared to the increase safety benefit, the costs could not justify the safety benefit. The panel concluded that these types of issues that have a small risk benefit would not be candidates for backfit because a significant increase in adequate protection is not provided. The GI staff sees a strong correlation between the proposed GI and this recent staff evaluation involving the Byron Station and Braidwood Station compliance backfit. The proposed GI describes several AOOs that were not initially captured during the initial phase of developing procedures to address AOOs, as described in RG 1.33, Appendix A. However, based on recent operational experiences of inadvertent ECCS injections and on observations of NRC regional inspectors during reactor operator simulator examinations, the Region IV inspector who submitted the GI believes that an operator response to these type events should be procedurally controlled.

References

American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.2-2012, "Managerial, Administrative, and Quality Assurance Controls for Operational Phase of Nuclear Power Plants" (replaces ANS 3.2/ANSI 18.7-1976; Agencywide Documents Access and Management System (ADAMS) Accession No. ML12313A054 (not available in public ADAMS)).

ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

ANSI/ANS 52.1-1978, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants."

Event Notification 53365, "Inadvertent Injection of High-Pressure Core Spray," April 26, 2018 (River Bend Station) (available at https://www.nrc.gov/reading-rm/doc-collections/event-status/event/2018/20180427en.html).

Licensee Event Report: LER 98-001, "RCS Outside Design Basis for Inadvertent ECCS Actuation at Power, Diablo Canyon," October 22, 1998 (ADAMS Accession No. ML16342A553).

Management Directive (MD) 6.4, "Generic Issues Program," January 2, 2015 (ADAMS Accession No. ML18073A162).

Memorandum of initial submittal, "Receipt of Proposed Generic Issue on Inadequate Licensee Procedures for Inadvertent Emergency Core Cooling System Injection," January 3, 2018 (ADAMS Accession No. ML18019A703).

Memorandum from the Office of Nuclear Regulatory Research (RES), "Request for Review of Proposed Generic Issue PRE-GI-020, Inadequate Procedures to Address Anticipated Operational Occurrences (Condition II events)," dated February 28, 2018, ADAMS Accession No. ML18044A369 (not available in public ADAMS)).

Memorandum from NRR to RES, "Request for Review of Proposed Generic Issue PRE-GI-020, Inadequate Procedures to Address Anticipated Operational Occurrences," dated March 23, 2018 (ADAMS Accession No. ML18081A936 (not available in public ADAMS)).

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