Summary of Results of Seven Screening Criteria
PRE-GI-020: "Inadequate Procedures to Address Anticipated Operational Occurrences"

Management Directive (MD) 6.4, "Generic Issues Program," specifies seven screening criteria that must be met for the proposed issue, PRE-GI-020, "Inadequate Procedures to Address Anticipated Operational Occurrences," to continue in the generic issue (GI) process. The GI Program staff's assessment as to whether the proposed GI met each of the criteria is summarized below. Enclosures 2 and 3 provide a more in depth evaluation.

1. The issue affects public health and safety, the common defense and security, or the environment.

   Of the several scenarios identified in the GI submission, the staff determined that the most risk-significant scenario is an inadvertent emergency core cooling system (ECCS) safety injection (SI) that fills the pressurizer water solid and challenges the pressurizer safety valves (PSVs).

   The GI Program staff reviewed the information from a recent Byron Station and Braidwood Station appeal of a U.S. Nuclear Regulatory Commission (NRC) backfit determination to assess the risk associated with anticipated operational occurrences (AOOs) resulting from the inadvertent actuation of ECCS SI pumps. The GI program staff concluded that such scenarios identified in the GI would not qualify as a compliance backfit unless the plant's licensing basis reflected a commitment to mitigate such an AOO.

   In addition, the GI Program staff reviewed the probabilistic risk assessment (PRA) study, "An Assessment of Core Damage Frequency for Byron/Braidwood Nuclear Power Plants Supporting Backfit Appeal Review Panel," dated August 11, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16214A199), in which the NRC staff concluded that the PSV failure scenario can be bounded by the small-break loss-of-coolant accident (SBLOCA) analysis in the plant's design basis. Therefore, the NRC staff concluded that an adequate protection backfit was not justified in the PSV failure scenario.

   In addition, the PRA study calculated the risk benefit to be only a 1.5×10^{-7} change in core damage frequency (CDF) if all additional requirements, including training and procedures, are imposed to preclude such an event. Office of Nuclear Regulatory Research (RES) Office Instruction TEC-002, Revision 2, "Procedure for Processing Generic Issues," dated September 26, 2011, provides a threshold value to use in determining whether the change in CDF is a significant safety concern for an issue to continue in the GI Program. Figures A1 through A3 in TEC-002 illustrate several graphs for the staff to use in determining whether the increase in risk meets the threshold to continue. Based on the findings from the NRC's evaluation of a PSV failure, the resultant point on the graph in Figure A2 would fall within the zone to exclude a reactor issue from further consideration in the GI Program. Therefore, the inadvertent ECCS SI that leads to a PSV failure scenario does not result in a significant enough risk to meet the threshold in TEC-002. Therefore, the GI staff concluded that the proposed GI does not adversely affect public health and safety.

   Hence, the proposed issue <u>does not</u> satisfy Criterion 1.

2. The issue applies to two or more facilities and/or licensee/certificate holders, or holders of other regulatory approvals.

   Based on information provided by licensees to NRC regional staff, many nuclear power plants do not have specific procedures or training to address the inadvertent operation of ECCS injection into the reactor coolant system (RCS) or a spurious auxiliary feedwater (AFW) injection into the steam generators (ADAMS Accession No. ML18158A138 (not publicly available)).

   Hence, the proposed issue does satisfy Criterion 2.

3. The issue is not being addressed using other regulatory programs and processes; existing regulations, policies, or guidance.

   As a result of the backfit appeal of the Byron Station and Braidwood Station PSV decision, the Office of the Executive Director for Operations (OEDO) directed the Office of Nuclear Reactor Regulation (NRR) to revisit Regulatory Information Summary (RIS) 2005-29, "Anticipated Transients That Could Develop into More Serious Events," dated December 14, 2005 (ADAMS Accession No. ML15014A469), and the proposed Revision 1 to RIS 2005-29 (ADAMS Accession No. ML15014A469), which had been issued for public comment. When this GI was proposed, the NRR staff was considering developing a new version of a RIS to replace RIS 2005-29 and proposed Revision 1. While this new RIS would discuss mass addition events, it was never intended to address operating procedures. Recent discussions with NRR staff indicate that NRR has discontinued development of Revision 1 to the RIS and is planning to withdraw both RIS 2005-29 and Revision 1.

   Therefore, the issue of ensuring that licensees have adequate procedures and training to prevent an AOO from escalating from a Condition II event to a Condition III or IV event is currently not being addressed through existing regulatory programs and processes.

   Hence, the proposed issue does satisfy Criterion 3.

4. The issue can be resolved by new or revised regulation, policy, or guidance.

   The licensing basis for most operating plants contains a commitment in their technical specifications stating that written procedures shall be written, implemented, and maintained using the applicable procedures in Regulatory Guide (RG) 1.33, Revision 2, "Quality Assurance Program Requirements (Operation)," issued February 1978. Specifically, Appendix A, "Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors," to RG 1.33 lists typical safety-related activities that a licensee should address in its procedures. The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. The staff periodically reviews RGs for updates to reflect changes noted in industry best practices, operational experience, and technological advances.

   The NRC also issues Regulatory Information Summaries (RIS) as necessary, to inform licensees of recurring issues and operating experience issues that the NRC believes licensees should address. The NRC issued RIS 2005-29 to inform licensees of a concern identified during reviews of power uprate requests in which licensees failed to demonstrate that anticipated transients will not progress to more serious events.

If the NRC staff finds that additional guidance to licensees is necessary, the staff can update or issue new guidance to licensees using a RIS or RG (with consideration for backfitting).

Hence, the proposed issue <u>does</u> satisfy Criterion 4.

5. The issue's risk or safety significance can be adequately determined in a timely manner.

The proposed GI describes a scenario of an inadvertent ECCS SI into the reactor vessel, which was the most significant scenario of the AOOs. Such an event in a pressurized-water reactor (PWR) may lead to an overpressurization event that causes a pressurizer safety valve (PSV) to relieve water and subsequently fail open. In the document, "An Assessment of Core Damage Frequency For Byron/Braidwood Nuclear Power Plants Supporting Backfit Appeal Review Panel," (ADAMS Accession number ML16214A199), the probabilistic risk assessment (PRA) staff evaluated the increase in risk of an overpressurization condition that leads to a water discharge through the PSVs, which subsequently challenges the ability of the PSV to subsequently close.

If the PSV fails open, it would produce consequences similar to an SBLOCA. The PRA staff calculated the risk of two cases. The Case A plant operated assuming realistic values for human actions. The Case B plant operated with a "perfect backfit" of adequate procedures and operator actions that will always prevent pressurizer overfill and a subsequent challenge to the SRVs. The model for this case assumes that operator actions taken to unblock a blocked pressurizer power-operated relief valve (PORV) and to terminate SI are always successful. The delta CDF can be considered as the "best estimate" benefit from a "perfect backfit" that would reduce the concern for pressurizer overfill and PSVs that are stuck in the open position.

The PRA concluded the following:

> The base CDF (Case-A) is 1.4E-05/year. The CDF difference between Cases A and B (CDFA - CDFB) is calculated to be 1.5E-07/year. This is a measure of maximum benefit that may be attained with a "perfect" backfit that avoids the issue (e.g. consequential small LOCA due to SRV failure after pressurizer overfill). This value is below the "very small" CDF delta risk threshold of 1E-06/year.

The PRA staff's evaluation of the PSV failure illustrates the ability of the staff to calculate the risk of scenarios mentioned in the proposed GI.

Hence, the proposed issue <u>does</u> satisfy Criterion 5.

6. The issue is well-defined, discrete, and technical.

The proposed issue gives examples of when a correct operator response would be required to respond to an AOO to prevent the escalation from a Condition II event to a Condition III or IV event. Operational experience indicates that inadvertent ECCS events have and continue to occur. If operators do not make the correct and timely response, then the event may escalate into a more severe condition, which would conflict with the criteria in ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," that prohibit the escalation of an AOO to a more serious condition without other incidents occurring independently and the criteria for AOOs in Section 15, "Introduction—Transient and Accident Analysis," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," in which an AOO cannot generate a postulated accident without other incidents occurring independently.

Hence, the proposed issue <u>does</u> satisfy Criterion 6.

7. Resolution of the issue may potentially involve review, analysis, or action by the affected licensees, certificate holders, or holders of other regulatory approvals.

   The resolution of this issue may involve the identification of applicable event sequences, additional training, an assessment of needed operator actions, and development of plant procedures.

   The proposed GI provided many of applicable event scenarios. An assessment of operator actions uses categories of human error probabilities (HEPs) to credit major actions by operators during accident and transient scenarios. If a licensee credits operator actions in Chapter 15 of its updated final safety analysis report (UFSAR) or safety analyses, then the licensee would be required to evaluate how successful operators are in performing this action and would assign a probability of success in its risk and safety analyses. If operating experience continues to show that operators are not being successful as credited, then the licensee may have to reevaluate the HEPs used in its PRA to assess the risk to the plant.

   Plant procedures can be developed based on the event sequences and assessment of operator actions. Operators can be trained to respond accordingly.

   Hence, the proposed issue <u>does</u> satisfy Criterion 7.