UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

January 6, 2022

NRC REGULATORY ISSUE SUMMARY 2021-XX
OPERATIONAL LEAKAGE

ADDRESSEES

All holders of operating licenses and combined licenses for nuclear power reactors, except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) staff is issuing this regulatory issue summary (RIS) to clarify the requirements to address operational leakage. This RIS requires no action or written response on the part of an addressee.

BACKGROUND INFORMATION

Operational leakage from American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code) Class 1, 2, and 3 structures, systems, and components (SSCs) is a special case in technical specifications (TS) that the NRC staff has determined needs to be clarified for nuclear power plant licensees. Operational leakage is a safety concern due to the unknown condition of the component when leakage is identified. An SSC exhibiting operational leakage could be vulnerable to structural concerns. The requirement for maintaining structural integrity is a fundamental assumption used in the development of the TS under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications." TS are derived from safety analyses that assume ASME BPV Code Class 1, 2, and 3 components continue to have structural integrity during operation.

In other TS, for example, a pump failing or a relief valve sticking open, the impact on operability (i.e., the capability of the SSC to perform its specified safety function) is readily evaluated. In contrast, leakage from ASME BPV Code Class 1, 2, and 3 SSCs does not provide a clear indication for understanding the impact on the system's operability. Operational experience has shown that a small leak may be the first and only indication of a significant degradation issue that could challenge the structural integrity of a component. Several degradation mechanisms, such as stress-corrosion cracking, thermal fatigue, flow-accelerated corrosion, and general corrosion, can lead to a system being vulnerable to structural concerns while exhibiting only limited indication of leakage. The NRC staff relies on operational leakage detection as a key defense-in-depth measure to identify potential degradation concerns in these SSCs. For these reasons, licensees are required to effectively evaluate leakage to determine the impact to SSC TS operability.

Through 10 CFR 50.55a(g), the NRC has required the use of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME BPV Code. Section XI provides

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methods to address leakage from these SSCs, and the NRC has found that the methods in Section XI provide reasonable assurance of the structural integrity of SSCs and hence the operability of the system.

The regulatory requirement of 10 CFR 50.55a(g) and TS clarified by this RIS is that structural integrity must be ensured for an ASME BPV Code Class 1, 2, or 3 component that is required to be operable according to the TS, and the only approved methods for doing so are provided in the ASME BPV Code, as incorporated by reference, in 10 CFR 50.55a, "Codes and standards." The ASME Code offers multiple methods to meet this requirement, including a series of ASME BPV Code Cases (e.g., N-513, N-705) and Nonmandatory Appendix U, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Piping and Class 2 or 3 Vessels and Tanks," to ASME BPV Code, Section XI. These provide specific allowances for leakage from Code Class components to demonstrate they continue to meet their intended function provided that structural integrity is maintained. The NRC staff has reviewed and, in most cases, approved these measures.

The NRC staff has communicated these requirements to licensees in Generic Letter 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping," dated June 15, 1990, and through RIS 2005-20, "Revision to NRC Inspection Manual Part 9900 Technical Guidance, Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety," which has been revised twice (Revision 1, dated April 16, 2008, and Revision 2, dated June 5, 2015). In addition, the NRC staff's position is documented in inspection criteria for these requirements in Inspection Manual Chapter 0326, "Operability Determinations," and Inspection Manual Part 9900 Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety." As well, the NRC staff's positions on these issues were thoroughly documented in a series of letters in 1996 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20006H277 and ML20112K084). In these letters, the NRC staff explained that 10 CFR 50.55a(g)(4) provides the regulatory basis to address leakage in components, within the scope of Section 50.55a, during operation. Further, the NRC staff clarified that ASME Code interpretations are not a part of NRC regulations, that the NRC has the responsibility for assuring that the CFR is effectively implemented, and that alternatives must be reviewed and approved by the NRC staff prior to use by a licensee.

Despite these communication efforts and inspection activities, the NRC staff is aware that some confusion exists over differences between the provisions of the ASME BPV Code and the requirements in NRC regulations. ASME has stated through ASME BPV Code Interpretations XI-1-92-03 and XI-1-92-19 (Question 2) that the corrective actions specified in ASME BPV Code, Section XI, paragraph IWA-5250, are not required by the ASME BPV Code to be implemented when leakage is found outside of the performance of an ASME BPV Code-required pressure test and visual examination for leakage (VT-2) examination. As noted above, the requirements of 10 CFR 50.55a, plant TS, and plant licensing bases require that the same ASME BPV Code methods for evaluating structural integrity be used whether the leakage is discovered during an ASME Code-required pressure test and VT-2 examination or by other means (e.g., plant walkdowns) as operational leakage. Essentially, regardless of when and how leakage is detected in these SSCs, the same ASME BPV Code methods for evaluating structural integrity must be used.

SUMMARY OF ISSUE

Given the potential confusion between NRC requirements and ASME Code interpretations, the NRC staff seeks to clarify that 10 CFR 50.55a(g) includes regulatory requirements that apply to operational leakage for all licensees of boiling- and pressurized-water reactors. This RIS clarifies the NRC requirements for evaluation, control, and treatment of operational leakage in systems required to be operable by plant TS.

Operational leakage is leakage through a flaw in the pressure-retaining boundary of an ASME BPV Code Class 1, 2, or 3 SSC discovered during the operational life of the nuclear power plant except when leakage is identified during an ASME BPV Code-required pressure test. The term "through-wall" describes a condition that extends from one surface to another surface in a component. If through-wall operational leakage is observed from an ASME BPV Code Class 1, 2, or 3 SSC and the structural integrity of the SSC must be established to conclude that the system remains operable, then the methods described in the provisions of the applicable inservice inspection requirements, as specified in 10 CFR 50.55a(g), must be used. These methods require analysis in accordance with the original construction code; implementation of an NRC-approved ASME BPV Code Case or Appendix U to ASME BPV Code, Section XI, to verify structural integrity; or performance of a repair/replacement activity.

This RIS emphasizes that operational leakage must be addressed in the same manner as leakage detected during an ASME BPV Code, Section XI, pressure test. Specifically, when operational leakage is found in a system that is within the scope of ASME BPV Code, Section XI, and that system is required to be operable by plant TS, the licensee must evaluate the component for operability. Structural integrity determinations must be conducted in accordance with the applicable provisions of the original construction code, ASME BPV Code, Section XI, or otherwise addressed through authorized methods (Generic Letter 90-05 is an example of guidance on one authorized method). These determinations necessitate an evaluation in accordance with an NRC-approved Code Case; use of Appendix U to ASME BPV Code, Section XI; or a repair/replacement activity.

10 CFR 50.55a(z), "Alternatives to codes and standards requirements," can be used as long as NRC staff authorization is granted prior to its implementation. Implementation is deemed to be the moment that the structural integrity of the component is required to be established (e.g., before expiration of the TS allowed completion time). Licensees can avail themselves of the alternatives process to propose evaluation methods or repair/replacement activities to resolve any circumstances that are not addressed by NRC-approved ASME Code evaluation methods.

For nuclear power plant licensees authorized to implement 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," the discussion in this RIS applies to Risk-Informed Safety Class (RISC)-1 and RISC-2 categorized components that are required to be operable by plant TS because 10 CFR 50.55a(g) continues to apply in full to RISC-1 and RISC-2 components. The NRC regulations in 10 CFR 50.55a(g) also continue to apply to those RISC-3 or RISC-4 categorized components that are classified as ASME BPV Code Class 1 and are required to be operable by plant TS. Operational leakage in other RISC-3 or RISC-4 categorized components that are classified as ASME BPV Code Class 2 and 3 and are required to be operable by plant TS must be treated in accordance with the licensee's approved 10 CFR 50.69 requirements. The NRC regulations in 10 CFR 50.69(d)(2)(i) require that, at a nuclear power plant implementing a 10 CFR 50.69 program, periodic inspection and testing activities must be conducted to ensure RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions. The Commission provides information for the implementation of 10 CFR 50.69 in

Volume 69 of the *Federal Register* (FR), 69 FR 68008, dated November 22, 2004.

BACKFITTING, FORWARD FITTING, AND ISSUE FINALITY DISCUSSION

This RIS clarifies the existing requirements for operational leakage. It does not present a new or changed NRC staff position, nor does it require any action or written response from any addressee. Therefore, issuance of this RIS is not a backfit under 10 CFR 50.109, "Backfitting," nor does it affect any issue finality provision in 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants." Further, the NRC staff does not intend to impose this RIS on applicants in a way that would constitute forward fitting as that term is defined in Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests."

***FEDERAL REGISTER* NOTIFICATION**

The NRC will publish a notice of opportunity for public comment on this draft RIS in the *Federal Register*.

CONGRESSIONAL REVIEW ACT

Discussion to be provided in final RIS.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not contain new or amended information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing requirements were approved by the Office of Management and Budget (OMB) under approval number 3150-0011.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under NRC Library/Document Collections.

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