UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OFFICE OF NEW REACTORS
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
WASHINGTON, DC 20555-0001

May 10, 2016

NRC DRAFT REGULATORY ISSUE SUMMARY YYYY-#### DISPOSITION OF INFORMATION RELATED TO THE TIME PERIOD THAT SAFETY-RELATED STRUCTURES, SYSTEMS, OR COMPONENTS ARE INSTALLED

ADDRESSEES

All holders of and applicants for an operating license or construction permit for a nuclear power reactor under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities."

All holders of and applicants for a power reactor early site permit, combined license, standard design approval, or manufacturing license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." All applicants for a standard design certification, including such applicants after initial issuance of a design certification rule.

All permanently shut down reactors with spent fuel in spent fuel pools (Millstone 1, Kewaunee, Crystal River, San Onofre and Vermont Yankee).

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to reiterate existing requirements related to dispositioning information pertaining to the capability of safety-related structures, systems, and components (SSCs) to perform their safety-related functions in nuclear power plants. This RIS addresses instances where a licensee becomes aware of credible information[1] pertaining to the time period that a safety-related SSC is installed that may impact its ability to perform its safety-related function(s). Licensees must assess this information consistent with their licensing basis and applicable NRC requirements.

In addition, this RIS reinforces the obligations of nuclear power plant licensees to maintain safety-related SSCs in accordance with 10 CFR Part 50, Appendix B to "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," the licensee's NRC-approved quality assurance (QA) program, and the licensee's site-specific operability/functionality determination process. This RIS requires no specific action or written response on the part of an addressee.

[1] Examples of credible information include, but are not limited to: vendor advisories or operating experience, NRC generic communications, and industry operating experience.

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BACKGROUND INFORMATION

In August 1992, the Commission published (57 FR 35455 dated August 10, 1992) a policy statement on “Availability and Adequacy of Design Basis Information at Nuclear Power Plants.” In the policy statement the Commission concluded that:

> … maintaining current and accessible design documentation is important to ensure (1) that plant physical and functional characteristics are maintained and are consistent with the design bases as required by regulation, (2) systems, structures, and components can perform their intended functions, and (3) the plant is operated in a manner consistent with the design bases.

As described in Appendix B to NEI 97-04, “Design Bases Program Guidelines,” which was endorsed by Regulatory Guide 1.186, “Guidance and Examples for Identifying 10 CFR 50.2 Design Bases,” dated December 2000, 10 CFR 50.2 design bases information includes the bounding conditions under which structures, systems, and components must perform their design functions. The 10 CFR 50.2 design bases of a facility are a subset of the current licensing basis and are required pursuant to 10 CFR 50.34 and 10 CFR 50.71(e) to be included in the Updated Final Safety Analysis Report. Underlying this is substantial supporting design information. Supporting design information includes other design inputs, design analyses, and design output documents. This information includes both docketed information and information retained by the licensee. Each Licensee’s NRC-approved QA program, operability/functionality process, and corrective action program encompasses the treatment of 10 CR 50.2 design bases information and substantial supporting design bases information.

Safety-related SSCs installed in a commercial nuclear power plant must conform to the requirements of the licensee’s NRC-approved QA program and other NRC requirements. Appendix B to 10 CFR Part 50 establishes QA requirements for the design, manufacture, construction, and operation of safety-related SSCs. Appendix B to 10 CFR Part 50 applies to activities affecting the safety-related functions of SSCs, including designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

In NUREG-0737, “Clarification of TMI [Three Mile Island] Action Plan Requirements,”[2] the NRC staff states that TMI Task Action Plan I.C.5, “Procedures for Feedback of Operating Experience to Plant Staff” (NUREG-0660), requires “all involved in the assessment of operating experience to review information from a variety of sources.” Licensees must also prioritize such information based on its safety significance. Further, as a result of NRC Generic Letter (GL) 1983-28, Supplement 1, “Required Actions Based on Generic Implications of Salem ATWS Events,”[3] and GL 1990-03, “Relaxation of Staff Position in Generic Letter 83-28, Item 2.2, Part 2, ‘Vendor Interface of Safety-Related Components,’ ”[4] licensees established programs to ensure that vendor information for safety-related SSCs is complete. These programs were established, in part, to ensure that vendor information is properly evaluated for its effect on safety-related equipment.

Additionally, licensees must consider operability of SSCs in accordance with plant technical specifications. When a licensee either becomes aware that a safety-related SSC has been

[2] Available in ADAMS at Accession No. ML051400209
[3] Available in ADAMS at Accession No. ML031210064
[4] Available in ADAMS at Accession No. ML031140578

installed for longer than the amount of time described in the licensing basis, or becomes aware of credible information that challenges the presumption that a safety-related SSC can continue to perform its safety function(s), the licensee must address and document this potential nonconforming condition in accordance with its NRC-approved QA program, operability/functionality determination process, and corrective action program.

NRC Inspection Manual Chapter (IMC) 0326, "Operability Determinations & Functionality Assessments for Conditions Adverse to Quality or Safety,"[5] defines a nonconforming condition as "a condition of an SSC that involves a failure to meet the CLB [current licensing basis] or a situation in which quality has been reduced because of factors such as improper design, testing, construction, or modification." IMC 0326 also describes an acceptable process for a nuclear power plant licensee to make operability/functionality determinations.

Through ongoing inspection and operating experience reviews, NRC staff has identified instances in which licensees did not incorporate relevant operating experience (e.g., vendor information) into plant procedures and programs. The NRC issued Information Notice (IN) 2012-06, "Ineffective Use of Vendor Technical Recommendations,"[6] to inform addressees of operating experience regarding ineffective use of vendor recommendations at U.S. nuclear power plants. One of the events discussed in the IN was determined to be risk significant, resulting in a White inspection finding. This event involved a dual-unit trip and a subsequent emergency diesel generator (EDG) failure to start. The EDG failure was attributed to a time delay relay that had been in service longer than the vendor documentation recommended. This condition had not been adequately evaluated or addressed by the licensee.

NRC staff reviewed five years of operating experience (from 2007-2011) related to the performance of SSCs at nuclear power plants. In its review and analyses, NRC staff observed a notable increase in the number of inspection findings and licensee event reports involving nonconformances with the provisions of Appendix B to 10 CFR Part 50 and failures of safety-related components that had been installed in the plant for longer than the amount of time specified in the plant's licensing basis or vendor documents. The staff also noted multiple instances where no corresponding technical evaluation of the condition had been completed. These and other observations are documented in "IOEB [NRC's Operating Experience Branch] Analysis Team Study on Component Aging—Insights from Inspection Findings and Reportable Events."[7]

There are several requirements to ensure that safety-related SSCs will perform their specified safety-related function(s), including, but not limited to:

- 10 CFR 50.34, "Contents of applications; technical information," 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," and 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

- 10 CFR 50.36, "Technical specifications."

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[5] Available in ADAMS at Accession No. ML13274A578
[6] Available in ADAMS at Accession No. ML112300706
[7] Available in ADAMS at Accession No. ML13044A469

- 10 CFR 50.65; "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." An acceptable approach for complying with the maintenance rule is described in NUMARC 93-01, Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,"[8] which was endorsed through NRC Regulatory Guide (RG) 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."[9]

- Updated final safety analysis report (UFSAR) discussions of conformance with 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," specifically General Design Criterion (GDC) 1, "Quality Standards and Records," and GDC 4, "Environmental and Dynamic Effects Design Bases."

- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power."

- 10 CFR 50.54, "Conditions of licenses."

- Applicable codes and standards that specify construction, inservice inspection, and inservice testing requirements, incorporated by reference in 10 CFR 50.55a, "Codes and standards," with conditions.

- 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," for those plants that have entered the period of extended operation.

SUMMARY OF ISSUE

If an SSC has been installed in a nuclear power plant for longer than the amount of time described by the plant's licensing basis documentation, the licensee must assess whether the SSC can continue to be relied on to perform its intended safety-related function(s) consistent with its licensing basis and applicable NRC requirements. Normally, the licensee should make these determinations before exceeding this documented time period.

Additionally, when a licensee becomes aware of credible information that may impact the ability of a safety-related SSC to continue to perform its safety-related function(s), there are NRC requirements, some of which are listed above, that direct the licensee to evaluate this information to ensure that the SSC can continue to perform its safety-related function(s). These licensee determinations must be documented, as appropriate, in accordance with the licensee's NRC-approved QA program, operability/functionality process, and/or corrective action program. These programs are collectively established to ensure that: (1) a technically defensible determination is made regarding the continued ability of the SSC to perform its specified safety-function (i.e., operability/functionality), (2) corrective actions, if required, are established , and (3) any corrective actions are completed in a timeframe commensurate with their safety significance.

It is also important to note that, while compliance with the provisions of 10 CFR 50.65 (i.e., the "maintenance rule") is required, this regulation in and of itself does not relieve licensees of the need to comply with other applicable regulations, NRC-approved program requirements, and regulatory commitments. The maintenance rule is performance-based and, as a result, does

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[8] Available in ADAMS at Accession No. ML11116A198
[9] Available in ADAMS at Accession No. ML113610098

not require corrective action until the performance or condition of an SSC fails to meet licensee-established goals or criteria.

Examples of several hypothetical situations illustrative of the information previously discussed in this RIS are contained in the appendix.

In summary, when a licensee becomes aware of credible information pertaining to the time period that a safety-related SSC is installed, and the information may impact the SSC's ability to perform its safety-related function, then the licensee must assess the information consistent with their licensing basis and applicable NRC requirements. These instances must be addressed in accordance with a licensee's NRC-approved QA program, operability/functionality determination process, and corrective action program.

BACKFITTING AND ISSUE FINALITY DISCUSSION

This RIS reinforces the requirements of 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" and licensees' NRC-approved QA programs and other existing regulations. The RIS requires no written response or action beyond that already required by NRC regulations. Therefore, this RIS does not represent backfitting as defined in 10 CFR 50.109(a)(1), nor is it otherwise inconsistent with any issue finality provision in 10 CFR Parts 50 or 52. Consequently, the NRC staff did not perform a backfit analysis.

***FEDERAL REGISTER* NOTIFICATION**

[Discussion to be provided in final RIS]

CONGRESSIONAL REVIEW ACT

This RIS is not a rule as defined in the Congressional Review Act (5 U.S.C. §§ 801-808).

Paperwork Reduction Act Statement

This RIS contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). The Office of Management and Budget (OMB) approved these information collections 3150-0011.

PUBLIC PROTECTION NOTIFICATION

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

CONTACT

Please direct any questions about this matter to the technical contacts listed below.

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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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Appendix

Hypothetical Examples of Acceptable Disposition of Information Related to the Time Period That Safety-Related Structures, Systems, or Components Are Installed

Scenario 1

Background:

Component:

Boiling-Water Reactor (BWR) Automatic Depressurization System (ADS) Valve Pilot Solenoid actuator

Safety Classification:

Class 1E, safety-related

Technical Specifications:

ADS function of seven safety relief valves is required when in Mode 1, and when in Mode 2 or 3 above, 150# reactor steam dome pressure. Surveillance requirements: Verify each ADS valve opens when manually actuated (18-month staggered test basis). Technical Specifications (TS) require procedures in accordance with Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation), Revision 2, 1978, Appendix A."

Other Requirements:

American Society of Mechanical Engineers Operations and Maintenance Code Appendix I

Maintenance Rule Applicability:

In scope of the maintenance rule; current status is 50.65(a)(2).

Preventive Maintenance Program Guidance:

Plant procedures define a maintenance interval of 18–24 months (during refueling). Maintenance consists of visual inspection of the solenoid, testing electrical continuity of the circuit, and replacing worn components as needed. The licensee's replacement interval as specified by maintenance procedures based on site operating experience is 5–8 years.

ADS pilot valve vendor manual, which is not part of the plant's supporting licensing basis, recommends replacement every 3–8 years, based on application.

Overview:

No failures at the plant, but there have been several failures of ADS pilot solenoid actuators at other BWRs over several years attributed to degradation of epoxy resin potting on the

solenoid coils. The average age at failure was 5 years. As a result, the manufacturer recently issued a service information letter (SIL) to all BWR licensees recommending that the replacement interval be changed to 3–5 years. Three of the seven ADS solenoid actuators will be 7 years old by the end of the current operating cycle. The remaining valves have been installed for 1–2 years.

Case 1:

Initial Actions:

- For any solenoid operated valve affected by the SIL, perform an operability determination.
  - If inoperable, follow the technical specification required action statements for the affected ADS valves.
  - If operable, the licensee will develop additional appropriate actions.

Licensee Planned Approach:

Licensee reviews the SIL through the corrective action program:

- Licensee plans to follow the SIL recommendation, and replace the affected valves at the next outage of sufficient duration.
- Document evaluation that the three valves, installed for 7 years, will remain operable until the planned replacement.
  - Review determines that maintenance procedures should be revised to reflect a new replacement interval of 3–5 years

Case 2:

Initial Actions:

- Same as Case 1

Licensee Planned Approach:

Licensee enters the SIL into its corrective action program:

- Corrective action program review determines that current maintenance procedures are acceptable.
- Licensee completes an engineering evaluation and determines that the existing maintenance procedure's replacement interval (5–8 years) is acceptable.
- Licensee documents its determination that the SIL recommendation does not apply because of different environmental conditions experienced by the solenoid actuators.

Scenario 2

Background:

Component:

Fuel oil flexible hoses (rubber) used to deliver fuel to the emergency diesel generators

Safety Classification:

Safety-related

Technical Specifications:

Verify the fuel oil transfer system operates to automatically transfer fuel oil from storage tank[s] to the day tank and engine mounted tank. TS require procedures in accordance with Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation), Revision 2, 1978, Appendix A."

Maintenance Rule Applicability:

In scope of the maintenance rule; current status is 10 CFR 50.65(a)(2)

Preventive Maintenance Program Guidance:

Plant procedures define a maintenance replacement interval of 3 years. This is consistent with the manufacturer's recommendation.

Other Requirements:

10 CFR 54.37(b) [applies to Case 4 only]

Overview:

The following scenario and the associated cases are based on a plant that has received a renewed license and is operating in the period of extended operation (i.e., beyond 40 years). During the review of the license renewal application (LRA), the licensee did not address aging effects associated with exposure of the internal surfaces of the hoses to fuel oil. Based on the results of the staff's review of the licensee's LRA, and the licensee's commitments, the hoses should be replaced based on the manufacturer's recommended interval. Subsequent to entering the period of extended operation, the licensee determines that the hoses have not been replaced based on the manufacturer's recommended interval.

Case 1:

The licensee concludes that it will replace the hoses during routine planned maintenance activities within the 3-year interval, and retain the current replacement interval.

Initial Actions:

- Perform an operability determination to determine whether the emergency diesel generators (EDGs) with the installed fuel hoses are operable until the hose replacements are completed.

Licensee Planned Approach:

- Licensee documents its determination to replace the hoses during maintenance activities and retain the current replacement interval.

- The timeliness of the replacement of the hoses is based on the licensee's corrective action and maintenance programs. The requirements of 10 CFR 54.37(b) are not invoked because the hoses would still not be subject to aging management review (i.e., they are periodically replaced).

Case 2:

The licensee concludes that a new replacement interval is appropriate based on the licensee's staff review of the material, and environment and replaces the hoses accordingly.

Initial Actions:

- Perform an operability determination to determine whether the EDGs with the installed fuel hoses are operable consistent with the current licensing basis (CLB) [as defined in 10 CFR 54.3].

Licensee Planned Approach:

- Licensee documents its determination that a new replacement interval is appropriate based on the licensee's staff review of the material and environment.

- The hoses are replaced based on the new replacement interval. The requirements of 10 CFR 54.37(b) are not invoked because the hoses would still not be subject to aging management review (i.e., they are periodically replaced).

Case 3:

The licensee concludes that periodic replacement is not required based on the licensee's staff review of the material and environment.

Initial Actions:

- Perform an operability determination to determine whether the EDGs with the installed fuel hoses are operable, and consistent with the CLB.

Licensee Planned Approach:

- Licensee documents the basis in its corrective action program or engineering processes. This documentation would be subject to staff review during inspection. A final safety analysis report (FSAR) update would not be applicable because there are no effects of aging to be managed in accordance with 10 CFR 54.37(b).

Case 4:

The licensee concludes that it will manage potential aging effects by an alternative method such as periodic visual inspections accompanied by physical manipulation of the hoses.

Initial Actions:

- Perform an operability determination to determine whether the EDGs with the installed fuel hoses are operable consistent with the CLB.

Licensee Planned Approach:

- Licensee documents its determination that it will manage potential aging effects by an alternative method such as periodic visual inspections accompanied by physical manipulation of the hoses.

- Determine a new replacement interval based on the licensee's staff review of the material and environment.

- This disposition would be documented in the next FSAR update. As required by 10 CFR 54.37(b) the FSAR update must describe how the effects of aging will be managed for these hoses such that the intended function(s) in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation.

Scenario 3

Background:

Component:

Pressurized Water Reactor (PWR) direct current (DC) electrical power subsystems (125 VDC Station Battery)

Safety Classification:

Class 1E, safety-related

Technical Specifications:

Two independent trains of 125 VDC batteries are required in MODES 1-4. Surveillance requirements include periodic testing of the following parameters: battery electrolyte level, float voltage, float current, no evidence of leakage, and battery pilot cell temperature greater than or equal to minimum established design limits. Technical Specifications require procedures in accordance with Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation), Revision 2, 1978, Appendix A."

UFSAR/Licensing Basis Considerations:

Batteries are sized to have sufficient capacity to supply the required loads for a LOCA/LOOP duration. The required final (end of duty cycle and end of life) battery cell voltages for each load group have been analyzed to demonstrate that adequate voltage is provided to the loads. The size of the batteries are based on a 20-year expected lifetime, as described in the UFSAR. The battery life calculation assumes that battery room temperature remains at or below its design value. As battery room temperature increases, battery life is reduced based on how much ambient temperature exceeds the design value.

Other Requirements:

None

Maintenance Rule Applicability:

In scope of the maintenance rule, current status is 10 CFR 50.65(a)(2).

Preventive Maintenance Program Guidance:

Plant procedures include requirements to log routine checks of battery room temperatures and calculate how battery life degrades over time based on temperatures.

Overview:

No failures at the plant. An inspector noted that yearly average room temperatures were above the assumed design room temperature limit used to determine battery life.

Case 1:

Licensee calculation determines that all battery cells have a reduced life, and have already exceeded the reduced calculation.

Initial Actions:

- Perform an operability determination incorporating the impact from exceeding the design room temperature limit.

- Follow TS required actions for any cell determined inoperable.

Licensee Planned Approach:

Licensee enters the issue into the corrective action program:

- Licensee completes an engineering evaluation using actual room temperature values and determines that some battery cells have exceeded their calculated life.

- Licensee replaces cells that have exceeded their calculated life and updates data to recalculate new life for remaining cells.

Case 2:

Licensee determines that all battery cells have reduced life based on calculations made using actual room temperatures.

Initial Actions:

- Perform an operability determination addressing the reduced life of the battery cells.

Licensee Planned Approach:

The Licensee determines that all battery cells remain operable and have not yet exceeded their design life based on calculations made using actual room temperatures. This information is entered into the licensee's corrective action program:

- As part of the corrective action program evaluation, the licensee discovers a configuration control issue with the battery room ventilation system that caused elevated room temperatures. Correcting this reduces battery room temperatures back below the design limit.

- After completing an engineering evaluation, the licensee determines that the reduction in room temperature will allow continued operation beyond the calculated reduced lifetime, but will required battery cell replacement prior to the original design lifetime.

Scenario 4

<u>Background</u>:

<u>Component</u>:

Pressurized Water Reactor Main Steam Isolation Valve (MSIV) Limit Switch

<u>Safety Classification</u>:

Safety-related

<u>Technical Specifications</u>:

None

<u>UFSAR/Licensing Basis Considerations</u>:

Where possible, all safety-related systems and components are designed to withstand the maximum expected 40-year integrated radiation dose at their respective locations within the plant. If it cannot be assured that equipment is designed for the 40-year dose, a replacement program for that equipment is established. The replacement program ensures operational integrity of the equipment throughout the life of the plant.

<u>Other Requirements</u>:

None

<u>Maintenance Rule Applicability</u>:

In scope of the maintenance rule, current status is 10 CFR 50.65(a)(2).

<u>Preventive Maintenance Program Guidance</u>:

Plant maintenance procedures require periodic verification that limit switch indication properly matches actual valve position.

<u>Overview</u>:

The MSIV limit switches were designed for 40 years of service in a mild radiation environment. This information is described in the UFSAR. The following cases highlight potential situations that could call into question the reliability of MSIV limit switches based either on actual failures or the availability of operating experience.

<u>Case 1</u>:

Although the limit switches are designed to withstand the plant's radiation environment for 40 years, the switch vendor recently issued a notice stating that additional analysis of the

phenolic material in the switches indicates an increased potential for failure after 30 years of service in a mild radiation environment. This determination was based on industry operating experience and laboratory analysis.

Initial Actions:

- Perform an operability determination based on notification from the vendor.
- If the MSIVs are determined to be inoperable, follow TS required actions for the affected MSIV.
- If the MSIVs are determined to be operable, follow the licensee planned approach below.

Licensee Planned Approach:

Licensee enters the issue into the corrective action program:

- Licensee completes an engineering evaluation using the information provided by the vendor. The results of the engineering evaluation agree with the vendor's determination that the switches are susceptible to failure after 30 years.
- Licensee replaces affected limit switches that have exceeded their revised service life and calculates revised replacement dates for remaining limit switches.
- Licensee revises updated final safety analysis report (UFSAR) information to explain the basis for the new service life for MSIV limit switches.

Case 2:

During MSIV surveillance for valve position indication, the correct valve position is not indicated in the Control Room. The licensee investigates and determines that the limit switch has failed. A root cause evaluation reveals the failure mechanism was likely radiation-induced embrittlement of the limit switch materials.

Initial Actions:

- Licensee replaces the broken limit switch and performs and extent of condition on the remaining MSIV limit switches.
- Licensee performs an operability determination on the affected MSIVs.
- If any MSIVs are determined to be inoperable, the licensee follows the TS required action statements for the affected MSIVs.

Licensee Planned Approach:

Licensee enters the issue into the corrective action program:

- After completing an engineering evaluation, the licensee determines that all MSIV limit switches are subject to the same radiation embrittlement factors that caused the failure of the switch.

- Licensee determines that all MSIV limit switches should be replaced during the next outage of sufficient duration and appropriate conditions.

Case 3:

The plant has been operating for 45 years. During a routine review, a design engineer discovers that MSIV limit switches were not replaced at the end of their 40-year design life.

Initial Actions:

- Licensee performs an operability determination on the affected MSIVs and determines they are operable.

- Licensee initiates an engineering evaluation to determine whether the affected components can remain in service beyond 45 years.

Licensee Planned Approach:

Licensee enters the issue into the corrective action program:

- Perform an extensive operating experience search to determine failure rates and occurrences for this model of limit switch.

- After completing an engineering evaluation, and based on no known age-related failures of this model of limit switch, the licensee determines that the limit switches can remain in service until the next refueling outage.

- Licensee revises UFSAR information for MSIV limit switches, reflecting the revised design life.