FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER

TSTF-577, REVISION 1

"REVISED FREQUENCIES FOR STEAM GENERATOR TUBE INSPECTIONS"

USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

(EPID L-2020-PMP-0005)

1.0 INTRODUCTION

By letter dated March 1, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21060B434), the Technical Specifications Task Force (TSTF) submitted Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections" (TSTF-577), to the U.S. Nuclear Regulatory Commission (NRC). TSTF-577 proposed changes to the Standard Technical Specifications (STSs) for pressurized-water reactor (PWR) designs under the consolidated line item improvement process (CLIIP). Upon approval, these changes would be incorporated into future revisions of NUREG-1430, NUREG-1431, and NUREG-1432, and this traveler would be available to licensees for adoption through the CLIIP.[1] TSTF-577 also proposed changes to NUREG-2194,[2] which will be evaluated in a separate safety evaluation (SE).

The Electric Power Research Institute (EPRI) submitted additional technical information related to extending the inspection interval for thermally treated Alloy 600 (Alloy 600TT) steam generator (SG) tubing in support of TSTF-577. The EPRI reports were submitted via letters dated November 18, 2020 (ADAMS Accession No. ML20335A173), and March 2, 2021 (ADAMS Accession No. ML21070A197), and are non-publicly available.

The proposed changes would revise the technical specifications (TSs) related to SG tube inspection and reporting requirements in the administration controls section of the TS.

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[1] U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Babcock and Wilcox Plants," NUREG-1430, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4, April 2012 (ADAMS Accession Nos. ML12100A177 and ML12100A178, respectively).

U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Westinghouse Plants," NUREG-1431, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4, April 2012 (ADAMS Accession Nos. ML12100A222 and ML12100A228, respectively).

U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Combustion Engineering Plants," NUREG-1432, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4, April 2012 (ADAMS Accession Nos. ML12102A165 and ML12102A169, respectively).

[2] U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Westinghouse Advanced Passive 1000 (AP1000) Plants," NUREG-2194, Volume 1 "Specifications," and Volume 2, "Bases," Revision 0, dated April 2016 (ADAMS Accession Nos. ML16110A277 and ML16110A369, respectively).

2.0 REGULATORY EVALUATION

2.1 Steam Generator Description

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, isolate fission products in the primary coolant from the secondary coolant and the environment. SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis.

2.2 Proposed Changes to the Standard Technical Specifications

TSTF-577 proposes to revise STS 5.5.9, "Steam Generator (SG) Program," and STS 5.6.7, "Steam Generator Tube Inspection Report," related to SG tube inspections and SG tube inspection reporting requirements. Mark-ups of the proposed changes are provided in TSTF-577.[3] Although multiple versions of marked-up STSs were provided in the traveler, the NRC staff is evaluating those in the enclosure, "Changes to the Technical Specifications Based on TSTF-510" (also referred to as the "Markup Based on TSTF-510"), in this SE. The proposed changes are applicable to the following:

- NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants";
- NUREG-1431, "Standard Technical Specifications, Westinghouse Plants"; and
- NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants."

2.2.1 Standard Technical Specification 5.5.9, "Steam Generator (SG) Program"

The proposed changes to NUREG-1430, NUREG-1431, and NUREG-1432 with respect to STS 5.5.9 are as follows:

- STS 5.5.9.d would be revised by adding a REVIEWER'S NOTE and a bracketed phrase regarding portions of the tube that are exempt from inspection by alternate repair criteria. The REVIEWER'S NOTE makes it clear that the bracketed phrase applies only to Alloy 600TT SG tubing.

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[3] The current STS requirements were established by TSTF-449, Revision 4, "Steam Generator Tube Integrity" (ADAMS Accession No. ML051090200), which was approved on May 2, 2005 (ADAMS Accession No. ML051160106), and was adopted by all operating PWR plants. TSTF-449, Revision 4 was incorporated into the STS, Revision 4. The NRC staff approved TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" (ADAMS Accession No. ML110610350), on October 19, 2011 (ADAMS Accession No. ML112101604), and it has been adopted by most of the applicable PWR plants. TSTF-510, Revision 2, was approved after the publication of the STS, Revision 4. TSTF-577 included mark-ups showing this change but they are not reviewed in this SE because they were already reviewed in NRC-approved TSTF-510, Revision 2.

- STS 5.5.9.d.2 would be revised by deleting the requirement to base inspection frequency on the more restrictive metric between either the effective full power months (EFPM) or refueling outage and to use just the EFPM metric.

- For mill-annealed Alloy 600 (Alloy 600MA) SG tubing, STS 5.5.9.d.2 would be revised by changing the requirement to inspect 100 percent of the tubes from every 60 EFPM to every 24 EFPM.

- STS 5.5.9.d.2 would be revised by deleting the allowance to extend the inspection period by 3 months and by deleting the discussion of prorating inspections.

- For Alloy 600TT SG tubing, STS 5.5.9.d.2 would be revised by changing the requirement to inspect 100 percent of the tubes at periods of 120, 90, and 60 EFPM to 54 EFPM. A 72 EFPM inspection period would be permitted if SG tubing has never experienced cracking (not including regions exempt from inspection by alternate repair criteria) and the SG inspection was performed with enhanced probes. A description of the enhanced probe inspection would be added.

- For thermally treated Alloy 690 (Alloy 690TT) SG tubing, STS 5.5.9.d.2 would be revised by deleting the requirement to inspect 100 percent of the tubes during each period in paragraphs d.2.a, d.2.b, d.2.c, and d.2.d (144, 120, 96, and 72 EFPM, respectively) and by adding the requirement to inspect 100 percent of the tubes every 96 EFPM.

- STS 5.5.9.d.3 would be revised by adding a REVIEWER'S NOTE and a bracketed phrase regarding portions of the tube that are exempt from inspection by alternate repair criteria. An additional bracketed phrase would be added that permits deferring SG inspections after cracking indications are found if the 100 percent inspection was performed with enhanced probes. The REVIEWER'S NOTE makes it clear that both bracketed phrases apply only to Alloy 600TT SG tubing.

- STS 5.5.9.d.3 would be revised by adding that each SG inspected at the next inspection after crack indications are found, includes each "affected and potentially affected" SG. The next inspection after crack indications are found would be changed from "shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections)" to "shall be at the next refueling outage."

- Other editorial and punctuation changes (i.e., revising "steam generator" to "SG").

2.2.2 Standard Technical Specification 5.6.7, "Steam Generator Tube Inspection Report"

The proposed changes to NUREG-1430, NUREG-1431, and NUREG-1432 with respect to STS 5.6.7 are as follows:

- Existing reporting requirement b. would be renumbered as c. and be revised by editorial and punctuation changes.

- New reporting requirement b. would be added to require the nondestructive examination (NDE) techniques utilized for tubes with increased degradation susceptibility be reported.

- Existing reporting requirement c. would be renumbered as c.1. and be revised by editorial and punctuation changes.

- Existing reporting requirement d. would be renumbered as c.2. and be revised to note that the location, orientation (if linear), measured size (if available), and voltage response do not need to be reported for tube wear indications at support structures that are less than 20 percent through-wall. However, the total number of tube wear indications at support structures that are less than 20 percent through-wall would be reported.

- New reporting requirement d. would be added to require an analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection relative to the applicable performance criteria, including the analysis, methodology, inputs, and results.

- Existing reporting requirement e. would be renumbered as c.4. and be revised by editorial and punctuation changes.

- Existing reporting requirements f. and h. would be combined, be renumbered as e., and be revised by editorial and punctuation changes.

- New reporting requirement f. would be added to require the results of any SG secondary side inspections be reported.

- Existing reporting requirement g. would be renumbered as c.3. and be revised to add the requirements to report the margin to the tube integrity performance criteria and a comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment. In addition, the requirement to report the results of tube pulls and in-situ testing would be deleted.

- New reporting requirement g. would require any plant-specific reporting requirements, if applicable.

- Existing reporting requirement h. would be renumbered as c.5. and be revised by editorial changes.

2.3 Applicable Regulatory Requirements and Guidance

As described in the Commission's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132, dated July 22, 1993), the NRC and industry task groups for new STSs recommended that improvements include greater emphasis on human factors principles in order to add clarity and understanding to the text of the STSs, and provide improvements to the Bases of the STSs, which provide the purpose for each requirement in the STSs. The improved vendor-specific STSs were developed and issued by the NRC in September 1992.

Section IV, "The Commission Policy," of the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors states, in part:

> The purpose of Technical Specifications is to impose those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise

> to an immediate threat to the public health and safety by identifying those features that are of controlling importance to safety and establishing on them certain conditions of operation which cannot be changed without prior Commission approval.
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> …[T]he Commission will also entertain requests to adopt portions of the improved STS [(e.g., TSTF-577)], even if the licensee does not adopt all STS improvements. …The Commission encourages all licensees who submit Technical Specification related submittals based on this Policy Statement to emphasize human factors principles.
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> …In accordance with this Policy Statement, improved STS have been developed and will be maintained for each NSSS [nuclear steam supply system] owners group. The Commission encourages licensees to use the improved STS as the basis for plant-specific Technical Specifications. …[I]t is the Commission intent that the wording and Bases of the improved STS be used … to the extent practicable.

The Summary section of the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors states, in part:

> Implementation of the Policy Statement through implementation of the improved STS is expected to produce an improvement in the safety of nuclear power plants through the use of more operator-oriented Technical Specifications, improved Technical Specification Bases, reduced action statement induced plant transients, and more efficient use of NRC and industry resources.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.36(a)(1) require that:

> Each applicant for a license authorizing operation of a … utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.

The regulations in 10 CFR 50.36(b) require that:

> Each license authorizing operation of a … utilization facility … will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to [10 CFR] 50.34 ["Contents of applications; technical information"]. The Commission may include such additional technical specifications as the Commission finds appropriate.

The categories of items required to be in the TS are listed in 10 CFR 50.36(c).

The regulations in 10 CFR 50.36(c)(5), “Administrative controls,” state that “[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in [10 CFR] 50.4.” Technical Specification Section 5.0, “Administrative Controls,” for all current PWR licenses requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Programs established by the licensee, including the SG Program, are listed in the administrative controls section of the TS to operate the facility in a safe manner.

The regulations in 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” establish, in part, requirements with respect to the integrity of the SG tubing. The following criteria are applicable:

Criterion 14, “Reactor coolant pressure boundary,” states that the RCPB “shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.”

Criterion 15, “Reactor coolant system design,” states that “[t]he reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.”

Criterion 30, “Quality of reactor coolant pressure boundary,” states, in part, that “[c]omponents which are part of the [RCPB] shall be designed, fabricated, erected, and tested to the highest quality standards practical.”

Criterion 31, “Fracture prevention of reactor coolant pressure boundary,” states, in part, that the RCPB “shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.”

Criterion 32, “Inspection of reactor coolant pressure boundary,” states, in part, that RCPB components “shall be designed to permit … periodic inspection and testing of important areas and features to assess their structural and leak tight integrity….”

The regulations in 10 CFR 50.55a specify that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), except as provided in 10 CFR 50.55a(c)(2), (3), and (4). Section 50.55a further requires that throughout the service life of PWR facilities, ASME Code Class 1 components must meet the Section XI requirements of the ASME Code to the extent practical, except for design and access provisions, and pre-service examination requirements. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to in-service inspection of SG tubing is augmented by additional requirements in the TS.

The NRC staff’s guidance for the review of TSs is in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP), Chapter 16.0, “Technical Specifications,” Revision 3, dated March 2010

(ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared STSs for each of the LWR nuclear designs. Accordingly, the NRC staff's review includes consideration of whether the proposed changes are consistent with the applicable referenced STS, as modified by NRC-approved travelers. In addition, the SRP states that comparing the change to previous STSs can help clarify the TS intent.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed TSTF-577, Revision 1, which proposed, in part, changes to NUREG-1430, NUREG-1431, and NUREG-1432. The regulatory framework the NRC staff used to determine the acceptability of the proposed changes consists of the requirements and guidance listed in Section 2.3 of this SE. The NRC staff reviewed the changes to determine whether the proposed changes to the STS meet the standards for TS in 10 CFR 50.36(c)(5), as well as conform to the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors. Notably, the NRC staff evaluation focused on the efficacy of the proposed STS changes continuing to ensure SG tube integrity, which is the stated goal of the SG Program. In particular, the evaluation assessed whether the structural integrity performance criterion and accident-induced leakage performance criterion will continue to be met with the proposed revised inspection intervals (maximum allowable time between SG inspections) and inspection periods (maximum allowable time between 100 percent of SG tubes inspections). The structural integrity performance criterion and accident-induced leakage performance criterion are explained in STS 5.5.9.b, items 1 and 2, respectively.

3.1 Standard Technical Specification 5.5.9, "Steam Generator (SG) Program"

Alloy 600MA SG Tubing

Alloy 600MA is widely recognized in U.S. PWRs as the SG tubing most susceptible to degradation, such as stress-corrosion cracking (SCC). Most U.S. plants have replaced their original SGs containing Alloy 600MA tubing such that only three units currently operate with Alloy 600MA tubing. Under TSTF-577, the maximum inspection interval for each SG and for each SG tube would be 24 EFPM. This is comparable to the current STS maximum inspection interval of 24 EFPM or one refueling outage (whichever is less). TSTF-577 also proposes that 100 percent of the tubes in each SG must be inspected every 24 EFPM, which is more often than the current STS interval of every 60 EFPM. Once adopted, the proposed changes would essentially require plants with Alloy 600MA tubing to inspect 100 percent of the tubes each refueling outage. While the current STS allow less than 100 percent inspection of Alloy 600MA SG tubing at each refueling outage, the current industry operating practice is to inspect 100 percent of the SG tubes each refueling outage. Therefore, the proposed STS changes would align with current industry practice. The NRC staff finds these proposed changes acceptable since the STS requirement for inspecting 100 percent of the Alloy 600MA tubes would be reduced from 60 EFPM to 24 EFPM, thereby requiring more tube inspections relative to the current STS, and operating experience has demonstrated that plants with Alloy 600MA tubing are maintaining tube integrity with the current STS inspection interval.

Alloy 600TT SG Tubing

The thermal treatment imparted to Alloy 600TT tubing results in significantly improved resistance to SCC in comparison to Alloy 600MA tubing, as evidenced by 40 years of operating experience. Excluding tube cracking that occurs near the bottom portion of the tube sheet (i.e., below the H* alternate repair criteria (ARC) distance), U.S. PWRs have reported 205 SCC indications in 135 Alloy 600TT tubes to date, including axial and circumferential outside diameter SCC (ODSCC) and axial and circumferential primary water SCC, with most of these indications having been ODSCC.[4] In contrast to Alloy 600MA tubing, which had increasing numbers of detected cracks per inspection with time, cracking in Alloy 600TT tubing has been intermittent. No U.S. PWRs have replaced Alloy 600TT SGs due to tube degradation. Operating experience has demonstrated that Alloy 600TT tubing has significantly greater resistance to SCC than Alloy 600MA, but significantly less resistance to SCC than the higher chromium containing Alloy 690TT tubing. For PWR plants with Alloy 600TT tubing, the maximum interval between inspections of each SG is currently limited to 48 EFPM or two refueling outages (whichever is less). In addition, 100 percent of the tubes must be inspected at sequential periods of 120, 96, and, thereafter, 72 EFPM. Licensees of PWR plants with Alloy 600TT tubing have maintained SG tube integrity against the SG Program tube integrity performance criteria under these requirements.

The NRC staff evaluated the maximum time between inspections for plants with Alloy 600TT tubing. The EPRI report, "Feasibility Study for Multi-Cycle 600TT Operational Assessments," submitted via letter dated November 18, 2020, concluded that 72 EFPM between inspections was feasible. In general, the longer the time interval that current tube conditions are projected forward, the greater the uncertainty associated with those projections. The NRC staff considers SCC to be the limiting degradation mechanism with respect to maximum time between inspections for Alloy 600TT tubing, since detection of cracking is more challenging compared to detection of volumetric degradation, such as tube wear. Also, tube wear at structures has been effectively managed by plants with Alloy 600TT tubing to maintain tube integrity over many operating cycles.

The NRC staff reached its conclusion about the maximum time between Alloy 600TT tubing inspections based on multiple factors, including: (1) plant operating experience such as location and frequency of cracking; (2) the Alloy 600TT crack database with details concerning crack detection and estimated crack growth; (3) the analyses detailed in the EPRI feasibility study that modeled various intervals between SG inspections; and (4) more extensive tube inspection reporting requirements that will provide the NRC staff with additional information from plant SG inspections. The NRC staff conclusion is for the maximum permissible time between inspections and, as such, is only applicable if the unit-specific operational assessment (OA) justifies the interval. Therefore, the NRC staff concludes that there is reasonable assurance that all Alloy 600TT plants can maintain tube integrity for up to 54 EFPM. In addition, the NRC staff concludes that there is reasonable assurance that tube integrity is maintained for Alloy 600TT plants up to 72 EFPM between tube inspections, provided that the following two conditions are met:

1. SCC cracking has not been detected during tube inspections (excluding tube end cracking that is already covered by an ARC) **AND**

[4] Audit Summary for the Regulatory Audit of Electric Power Research Institute for Steam Generator Task Force Information Related to TSTF-577, "Revised Frequencies for Steam Generator Tube Inspections," dated December 15, 2020 (ADAMS Accession No. ML20345A008).

2. An enhanced probe inspection method[5] is performed at the 100 percent tube inspection entering each 72 EFPM inspection interval.

Condition 1 recognizes that Alloy 600TT plants with many years of operation that have not detected cracking (excluding that already covered by an ARC) should be eligible for the longest inspection interval. The NRC staff recognizes that cracking could occur and be detected in some of these plants during future inspections, at which time the maximum interval between inspections for that plant would decrease to 54 EFPM. Condition 2 recognizes that widespread advanced probe inspections will achieve a greater probability of crack detection compared to a bobbin probe inspection. This will also provide for earlier detection should cracking occur in unanticipated locations or in tubes not currently inspected with enhanced probes. The NRC staff concludes that the combination of these two conditions provides reasonable assurance that tube integrity will be maintained during a longer 72 EFPM inspection interval. The NRC staff also notes that the 54 EFPM and 72 EFPM intervals are the maximum permissible times between inspections and are only applicable if the unit-specific OA justifies the interval.

Alloy 690TT SG Tubing

For PWR plants with Alloy 690TT tubing, the maximum interval between inspections of each SG is currently limited to 72 EFPM or three refueling outages (whichever is less). In addition, 100 percent of the tubes must be inspected at sequential periods of 144, 120, 96 and, thereafter, 72 EFPM. Licensees of PWR plants with Alloy 690TT tubing have managed SG tube integrity against the SG Program tube integrity performance criteria under these requirements.

For Alloy 690TT tubing, TSTF-577 proposes that both the maximum time between inspections and the time to inspect 100 percent of the tubes be 96 EFPM. TSTF-577 would modify the STS to state, in part: "After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period." The nuclear industry's Steam Generator Task Force (SGTF) presented a technical basis supporting the 96 EFPM Alloy 690TT inspection interval during the February 13, 2019 (ADAMS Package Accession No. ML19044A416), and February 24, 2020 (ADAMS Package Accession No. ML20066E421), public meetings with the NRC staff.

As discussed in the February 13, 2019, public meeting and documented in SG tube inspection reports, no SCC has been detected in U.S. PWR plants with Alloy 690TT SG tubing. The NRC staff is also unaware of SCC in any operating Alloy 690TT SGs. The first units with Alloy 690TT SG tubing have been in service for more than 30 years. This operating experience is a key consideration for the NRC staff in evaluating the proposed inspection interval. Therefore, the staff's review for Alloy 690TT tubing focused on active tube degradation due to wear at structures such as tube support plates, anti-vibration bars, and flow distribution baffle plates. Tube wear from foreign objects is discussed in Section 3.2 of this SE.

The number of tubes with wear and the extent of tube wear varies at plants with Alloy 690TT tubing. Currently, there are 44 PWRs in the U.S. with Alloy 690TT tubing. Of these 44 PWRs, 15 units have few wear indications and have not plugged any tubes due to tube wear at

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[5] The enhanced probe inspection method is defined as performing 100 percent SG tube inspections from tube end to tube end (except for any portions exempt from inspection by an ARC) with eddy current probes equivalent to or better than array probe technology. For regions where enhanced probes cannot be used, tube inspection techniques must be capable of detecting all forms of existing and potential degradation in that region.

structures. In contrast, 11 units have experienced more aggressive wear in terms of either large numbers of tubes with wear indications (thousands) or greater wear growth rates. From an inspection perspective, tube wear is both easier to detect at shallow depths and is more reliably sized compared to SCC. No Alloy 690TT plant has failed to meet the TS tube integrity performance criteria due to wear at structures. Individual plant data shows that tube wear growth rates typically decrease over time. Tube wear at structures has been effectively managed at plants with Alloy 690TT tubing to maintain tube integrity over many operating cycles. Therefore, the NRC staff has reasonable assurance that the proposed 96 EFPM maximum time between inspections is acceptable. The NRC staff notes that the 96 EFPM between inspections is the maximum permissible time between inspections and, as such, is only applicable if the unit-specific OA justifies the interval. Some Alloy 690TT tubing plants with more aggressive wear rates or large numbers of wear indications will not be able to achieve the maximum proposed time between inspections. Units experiencing more aggressive wear are limited to an inspection interval determined by their OAs that consider plant-specific wear rates, as required by the SG Program.

Detection of Cracking

Although the NRC staff finds acceptable the proposed inspection intervals and periods in STS 5.5.9.d.2, the staff notes that projecting future tubing conditions for long periods of operation based on the existing condition and anticipated flaw growth can involve significant uncertainty that may be difficult to reliably bound. For this reason, the proposed STS 5.5.9.d.3 would supplement the performance-based requirement concerning inspection intervals and periods with a prescriptive requirement that provides added assurance that tube integrity will be maintained. Based on the knowledge gained during the NRC's audit of EPRI[6] for information related to Alloy 600TT, including estimated crack growth rates from historical review of eddy current crack indications present for more than one inspection, and recognizing the improved probability of detection of cracks inherent with the enhanced probe inspection method, the NRC staff concludes that a conditional extension to the time to the next inspection is also acceptable. Therefore, following detection of crack indications, the NRC staff concludes that it is acceptable to perform the next inspection at the second refueling outage (for Alloy 600TT only), if the enhanced probe inspection method was performed and plant-specific OA justifies the interval. The NRC staff notes that the conditional opportunity to inspect at the second refueling outage after detecting crack indications does not apply to units with Alloy 600MA SG tubing or Alloy 690TT SG tubing (if it occurs).

3.2 Standard Technical Specification 5.6.7, "Steam Generator Tube Inspection Report"

Licensees are required to submit an SG tube inspection report to the NRC, in accordance with their TS, typically within 180 days of the completion of the SG inspection. The NRC staff reviews each SG tube inspection report to ensure that the report includes the information required by the licensee's TS, ensures that the inspections performed appear to be capable of detecting potential SG tube degradation, ensures that SG tube integrity is being effectively managed, and determines whether the inspection results appear to be consistent with the operating experience at similarly designed and operated units.

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[6] Audit Summary for the Regulatory Audit of Electric Power Research Institute for Steam Generator Task Force Information Related to TSTF-577, "Revised Frequencies for Steam Generator Tube Inspections," dated December 15, 2020 (ADAMS Accession No. ML20345A008).

The NRC staff reviewed the proposed changes to the current STS SG tube inspection reporting requirements and determined that they are acceptable because they will provide additional detailed information to allow the staff to better understand the overall condition of the SGs. For example, proposed new reporting requirement “d” would be added to require an analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection relative to the applicable performance criteria, including the analysis, methodology, inputs, and results, to be reported. Proposed new reporting requirement “f” would be added to require the results of any SG secondary side inspections to be reported. The NRC staff notes that the current STS implicitly require secondary side inspections, if failure of secondary side components can result in a loss of tube integrity. Section 2.4.3 of TSTF-577 states that a template that will provide guidance to the nuclear industry for complying with the revised reporting requirements is being developed and will be added to the EPRI Steam Generator Integrity Assessment Guidelines.[7] The NRC staff concludes that the proposed changes to the reporting requirements, including any supporting templates, will provide more detailed and consistent information to the NRC.

Foreign Object Wear

The NRC staff acknowledges that projecting future loose part wear is usually not possible since past fleet-wide operating experience has shown that new foreign object generation, transport to the SG tube bundle, and interactions with the tubes cannot be reliably predicted. However, plants can reduce the probability of loose parts with feedwater loose part strainers, by maintaining robust foreign material exclusion programs, and through application of lessons learned from previous industry operating experience with loose parts. Plants have demonstrated the ability to conservatively manage loose parts once they are detected by eddy current examinations or by secondary side foreign object search and retrieval inspections. TSTF-577 states that upgrades to the management of foreign objects will be included in an upcoming revision to the EPRI Steam Generator Integrity Assessment Guidelines. The nuclear industry’s SGTF discussed the additional upgrades to the management of foreign objects during the January 26, 2021 (ADAMS Package Accession No. ML21025A072), public meeting with the NRC staff. If unanticipated aggressive tube wear from new foreign objects was to occur, operating experience has shown that a primary-to-secondary leak will probably occur, rather than a loss of tube integrity. In the event of a primary-to-secondary leak, the NRC staff will interact with the licensee in accordance with established procedures in NRC Inspection Manual Chapter (IMC) 0327, “Steam Generator Tube Primary-to-Secondary Leakage,” dated January 1, 2019 (ADAMS Accession No. ML18093B067). Therefore, the NRC staff determined that foreign object wear does not affect the staff’s conclusion with respect to the proposed changes in TSTF-577.

4.0 CONCLUSION

The NRC staff concludes that the proposed changes in TSTF-577 to STS 5.5.9 and STS 5.6.7 are acceptable because, as discussed above, they continue to ensure SG tube integrity and, therefore, protect the public health and safety. In particular, the structural integrity performance criterion and accident-induced leakage performance criterion (explained in STS 5.5.9.b, items 1 and 2, respectively) will continue to be met with the proposed revised SG inspection intervals

[7] Electric Power Research Institute, Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines, Revision 4, dated June 2016 (non-publicly available), submitted via letter dated July 18, 2016 (ADAMS Accession No. ML16208A272).

(maximum allowable time between SG inspections) and inspection periods (maximum allowable time between 100 percent of SG tubes inspections). Additionally, the proposed changes to the reporting requirements will provide more detailed and consistent information to the NRC. Therefore, the NRC staff finds that the proposed changes to the SG program and inspection reporting requirements are acceptable because they continue to meet the requirements of 10 CFR 50.36(c)(5) by providing administrative controls necessary to assure operation of the facility in a safe manner.

The NRC staff finds that there is reasonable assurance that plants adopting TSTF-577, with respect to the proposed changes to NUREG-1430, NUREG-1431, and NUREG-1432, will maintain SG tube integrity. Therefore, the NRC staff concludes that the proposed changes to NUREG-1430, NUREG-1431, and NUREG-1432 in TSTF-577 are acceptable.

Principal Contributors: C. Tilton, NRR/DSS/STSB
P. Klein, NRR/DNRL/NCSG
L. Terry, NRR/DNRL/NCSG

Date: April 14, 2021