

## REFERENCE No. 1

### REFERENCES SUPPORTING STEAM GENERATOR INSPECTION USING INSPECTION PROCEDURE 71111.08

(Sorted by Inspection Requirement)

#### **In-situ Pressure Testing** (Item 02.03.c.1)

EPRI TR-107620-R1, "Steam Generator In Situ Pressure Test Guidelines" June 1999

- *Provided to regional staff under separate cover in September 2001*
- Focus on Section 4 and Appendix B

Letter from David J. Modeen, Nuclear Energy Institute, to Dr. Brian W. Sheron, NRC, "Interim Guidance for In Situ Pressure Testing," dated November 8, 2000

- *Provided to regional staff under separate cover in September 2001*
- ADAMS ML 003770571

Regulatory Issue Summary 2000-22, "Issues Stemming from NRC Staff Review of Recent Difficulties Experienced in Maintaining Steam Generator Tube Integrity"

- *Available on NRC External Website in the "Reference Library"*
- ADAMS ML003758988

#### **Projections versus Actual Inspection Results** (Item 02.03.c.2)

Operational Assessment/Condition Monitoring - provided by licensee

EPRI TR-107621, "Steam Generator Integrity Assessment Guidelines"

- *Provided to regional staff under separate cover in September 2001*
- Focus on Section 6 and Appendices D, E, F and G

Regulatory Issue Summary 2000-22, "Issues Stemming from NRC Staff Review of Recent Difficulties Experienced in Maintaining Steam Generator Tube Integrity"

- *Available on NRC External Website in the "Reference Library"*
- ADAMS ML003758988

#### **Examinations** (Item 02.03.c.3, Item 02.03.c.4, Item 02.03.c.5, Item 02.03.c.8, Item 02.03.c.10 )

EPRI TR-107569, "PWR Steam Generator Examination Guidelines"

- *Provided to regional staff under separate cover in September 2001*
- *Revision 5 is the latest version available, however, a few licensees may be utilizing the draft Revision 6 for performing an activity, not covered in Rev. 5, called Degradation Assessment. If this is the case, the licensee should be able to provide access to Rev. 6.*

- Item 02.03.c.3, focus on Section 3 and 4 of Rev 5

- Item 02.03.c.4, focus on Degradation Assessment in Rev 6
- Item 02.03.c.5, focus on Degradation Assessment in Rev 6
- Item 02.03.c.8, focus on Supplements H1 and H2 in Rev 5

Regulatory Issue Summary 2000-22, "Issues Stemming from NRC Staff Review of Recent Difficulties Experienced in Maintaining Steam Generator Tube Integrity"

- *Available on NRC External Website* in the "Reference Library"
- ADAMS ML003758988

Examination Technique Specification Sheet

- *Available through licensee on-site*

Site Specific Analysis Guidelines

- *Available through licensee on-site*

### **Repair Criteria** (Item 02.03.c.6)

Technical Specifications

- *Available through licensee on-site*

U.S. NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995.

- *Available on NRC External Website* in the "Reference Library"

U.S. NRC Generic Letter 97-05, "Steam Generator Tube Inspection Techniques," December 17, 1997.

- *Available on NRC External Website* in the "Reference Library"

### **Leakage** (Item 02.03.c.7)

EPRI TR-104788, Rev 2, "PWR Primary-to-Secondary Leak Guidelines"

- *Provided to regional staff under separate cover in September 2001*
- Focus on Appendix A

9900 Technical Guidance, "Steam Generator Tube Primary-to-Secondary Leakage"

- To be issued with final version of 71111.08

### **Loose Parts** (Item 02.03.c.9)

U.S. NRC Generic Letter 97-06, "Degradation of Steam Generator Internals," December 30, 1997.

- *Available on NRC External Website* in the "Reference Library"

EPRI TR-107569, "PWR Steam Generator Examination Guidelines"

- *Provided to regional staff under separate cover in September 2001*

## **REFERENCE No. 2**

## STEAM GENERATOR REFERENCES

The following references are readily available through the NRC Website, ADAMS or NUDOCS. The few that are not have been provided under separate cover to the regional office staff.

1. U.S. NRC Regulatory Issue Summary 2000-22, "Issues Stemming from NRC Staff Review of Recent Difficulties Experienced in Maintaining Steam Generator Tube Integrity," November 3, 2000.
  - *Discusses lessons learned from Indian Point 2 tube rupture event and ANO-2 in-situ pressure test results. The RIS discusses ten issues related to tube integrity including the need to consider all relevant operating experience, the need to assess the root cause of all degradation observed, the importance of data quality, the importance of qualifying inspection techniques with flaws representing those observed in the field, the importance of site specific qualification for generically qualified techniques, the consideration of flaw size measurement error in selecting tubes for in-situ pressure testing, the need for rigorously analyzing results of incomplete in-situ pressure tests, the effects of pressurization rate on burst pressure, the potential non-conservatism associated with using the fractional flaw method in performing condition monitoring/operational assessments, and the importance of benchmarking operational assessment results against actual operating experience.*
2. U.S. NRC Generic Letter 97-06, "Degradation of Steam Generator Internals," December 30, 1997.
  - *Highlights degradation observed on secondary side of steam generators and the importance of monitoring secondary side structures. Details provided in IN's 96-09 and 96-09 supplement 1. Requests licensee's to describe their programs for inspecting secondary side structures.*
3. U.S. NRC Generic Letter 97-05, "Steam Generator Tube Inspection Techniques," December 17, 1997.
  - *Emphasizes the importance of only using qualified techniques to size degradation. Requests licensees to provide technical basis for any indications which they leave in service based on sizing.*
4. U.S. NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995.
  - *Discusses voltage-based tube repair criteria methodology. Applicable only to Westinghouse plants with 3/4" or 7/8" outside diameter, mill annealed alloy 600 tubes supported by drilled hole tube support plates.*
5. U.S. NRC Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes," April 28, 1995.
  - *Discusses inspection findings at Maine Yankee related to circumferential cracking at the expansion transition (ID initiated). Discusses factors affecting detection of circumferential cracking including the scope of inspection, NDE*

- methods, and plant specific factors. Requests licensee's to discuss inspections performed and plans for inspecting locations susceptible to circumferential cracking.*
6. U.S. NRC Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," April 2, 1991.
    - *Discusses actions necessary to extend interval for conducting SG tube inspections when extending fuel cycle. Emphasizes that the provisions of Specification 4.0.2 do not apply for extending SG inspection intervals (i.e., the TS limits of 24 and 40 calendar months can not be extended 25% through the use of TS 4.0.2)*
  7. U.S. NRC Bulletin 89-01, Supplement 2, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs," June 28, 1991.
    - *Expands Supplement 1 to include all Westinghouse mechanical plugs fabricated from thermally treated Inconel 600.*
  8. U.S. NRC Bulletin 89-01, Supplement 1, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs," November 14, 1990.
    - *Discusses additional failures of plug heats not addressed in the first issuance of Bulletin 89-01. Expected to result in reduced lifetime estimates.*
  9. U.S. NRC Bulletin 89-01, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs," May 15, 1989.
    - *Requested addressees to develop an "action plan" for addressing degradation of certain heats of Westinghouse mechanical plugs.*
  10. U.S. NRC Bulletin 88-02, "Rapidly Propagating Cracks in Steam Generator Tubes," February 5, 1988.
    - *Describes North Anna 1 tube rupture due to high cycle fatigue, denting, and lack of effective anti-vibration bar support.*
  11. U.S. NRC Information Notice 98-27, "Steam Generator Tube End Cracking," July 24, 1998.
    - *Describes experience with tube end cracking primarily at B&W plants (i.e., ANO 1 and Davis Besse) although experience and applicability to other designs is discussed. Discusses need to perform supplemental inspections in this area and the need to qualify these techniques.*
  12. U.S. NRC Information Notice 97-88, "Experiences During Recent Steam Generator Inspections," December 16, 1997.
    - *Discusses degradation of carbon steel ribs on the feedwater impingement plate at Harris, and erosion corrosion of tube supports at San Onofre 3. Degradation of secondary side components could lead to loss of tube support and/or loose parts. Discusses qualification of techniques for sizing IGA in OTSGs at ANO-1 and CR3 and the importance of considering plant-specific and time-dependent circumstances. Discusses tube leak at McGuire 2 and the importance of properly determining the positioning of probes within a tube (e.g., RPC position) - techniques for minimizing probe positioning inaccuracies are also discussed. Discusses possibility of ID and OD initiated axial and circumferential cracking occurring at dents less than 5 volts - these flaws may not be detected with the bobbin coil probe). Discusses need to*

*inspect welded tubesheet sleeves with visual, UT, and ECT methods to detect flaws (Zion 1 and 2 and Prairie Island 1). The degradation in these sleeves was attributed to improper surface preparation during sleeve installation. Discusses high voltage growth rates at ODS/CC at tube support plate elevations at Farley 1 at Braidwood 1 (i.e., GL 95-05 implementation) - results indicate importance of evaluating "accuracy" of projections. Discusses likelihood that degradation continues to grow even after plugging the tube (e.g. McGuire 1 and Braidwood 1).*

13. U.S. NRC Information Notice 97-79, "Potential Inconsistency in the Assessment of the Radiological Consequences of a Main Steam Line Break Associated with the Implementation of Steam Generator Tube Voltage-Based Repair Criteria," November 20, 1997.
  - *Describes the need to account for density differences between the measured volumetric leak rate, which is recorded at room temperature, and the allowable volumetric leak rate for assessing radiological consequences, which is determined at operating temperature. Affects leakage measurements made in support of GL 95-05.*
14. U.S. NRC Information Notice 97-49, "B&W Once-Through Steam Generator Tube Inspection Findings," July 10, 1997.
  - *Discusses service induced degradation observed in B&W once through steam generators at dented locations, expansion transitions, the freespan region, the sludge pile region, and at sleeve joints. Discusses need to monitor locations potentially susceptible to degradation with qualified probes. Also discusses importance of tube pulls and in-situ testing in assessing tube integrity.*
15. U.S. NRC Information Notice 97-26, "Degradation in Small-Radius U-Bend Regions of Steam Generator Tubes," May 19, 1997.
  - *Discusses experience with cracking in the U-bend region of small radius tubes (e.g., Rows 1 and 2). Discusses experience at Zion 2, Sequoyah 2, Diablo Canyon 1, and Braidwood 2. Also discusses experience with identifying OD degradation in the U-bend region of small radius tubes at Palo Verde and St. Lucie 1. Of particular note is that Braidwood 2 has thermally treated alloy 600 tubes (subsequent to issuance of the IN, questions regarding the nature of the indication were raised and there is no consensus on whether this is SCC or not). Discusses inspection techniques to detect U-bend degradation and the lack of pulled tube data for this mechanism. Discusses importance of assessing the integrity of the tubes.*
16. U.S. NRC Information Notice 96-38, "Results of Steam Generator Tube Examinations," June 21, 1996.
  - *Discusses experiences with new steam generator tube examination techniques and issues arising from them including classifying inspection results, periodicity of examinations, and expanding the initial inspection scope. Discusses identification of ID and OD axial and circumferential indications at dented support plate elevations at Sequoyah, Diablo Canyon and Salem. Discusses circumferential indications at expansion transitions including roll expanded, explosively expanded, and hydraulically expanded tubes. Discusses free span degradation at McGuire, Farley,*

*Palo Verde, ANO 2, and Point Beach 1. Discusses importance of inspecting all portions of the tube to ensure new degradation mechanisms are detected. Discusses importance of inspecting tubes using techniques and equipment capable of reliably detecting degradation to which the tube is susceptible to and factoring in plant-specific circumstances). Discusses importance of tube pulls for assessing inspection reliability, developing additional confidence in the ability to size indications, determining the root cause of the indication, and possibly identifying corrective actions. Discusses actions necessary when no qualified sizing technique exists. Results are applicable to all tube materials.*

17. U.S. NRC Information Notice 96-09, Supplement 1, "Damage in Foreign Steam Generator Internals," July 10, 1996.
  - *This information provides additional experience related to wastage of tube support plates not associated with chemical cleaning (which was discussed in IN 96-09). This IN lends insights into the cause of the "wrapper drop event" detailed in IN 96-09. Discusses importance of reviewing support plate signal anomalies and performance of secondary side inspections.*
18. U.S. NRC Information Notice 96-09, "Damage in Foreign Steam Generator Internals," February 12, 1996.
  - *Discusses degradation of support plate and wrapper at a foreign facility. Anomalous support plate signals found during routine eddy current exam of tubes led to a visual inspection which indicated portion of the plate had wasted away and was resting on the next lower tube support plate (attributed to chemical cleaning). Tube support is important for lateral support (i.e., vibrational stability and ability to withstand earthquake and LOCA loadings). Another foreign PWR identified tube support plate ligaments which were cracked at the uppermost tube support plate. In another foreign PWR, the wrapper welds at the vertical supports failed resulting in the wrapper dropping down in the steam generator.*
19. U.S. NRC Information Notice 95-40, "Supplemental Information to Generic Letter 95-03, 'Circumferential Cracking of Steam Generator Tubes'," September 20, 1995.
  - *Supplements information supplied in Generic Letter 95-03 related to Maine Yankee inspection findings. Discusses tube pull results which indicated the flaws were not coplanar and compares sizing estimates from the high frequency pancake coil to the destructive examination results. The high frequency pancake coil is more sensitive inside diameter initiated flaws.*
20. U.S. NRC Information Notice 94-88, "Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes," December 23, 1994.
  - *Discusses inspection findings at Maine Yankee related to circumferential cracking at the top of the tubesheet which indicated tubes may not have adequate structural margin (related to IN 92-80). Discusses limitations of pancake type coils (i.e., previous IN's indicated that licensees should use specialized probes, but this IN discusses weaknesses even when using "specialized" probes) as a result of electrical noise, interfering signals such as from probe liftoff and deposits. Also*

*discusses importance of the training and performance demonstration testing of data analysts and the importance of tube pulls in validating inspection methods.*

21. U.S. NRC Information Notice 94-87, "Unanticipated Crack in a Particular Heat of Alloy 600 Used for Westinghouse Mechanical Plugs for Steam Generator Tubes," December 22, 1994.
  - *Discusses alloy 600 mechanical plug cracking experience at St. Lucie 1. This information notice is related to Bulletin 89-01 and calls into question the algorithm used for determining replacement of Westinghouse alloy 600 mechanical plugs.*
22. U.S. NRC Information Notice 94-62, "Operational Experience on Steam Generator Tube Leaks and Tube Ruptures," August 30, 1994.
  - *Summarizes many recent tube leakage events: Braidwood 1 (crack near anti-vibration bar); Palo Verde 2 (IN 93-56 and 94-43) ANO 2 (IN 92-80); McGuire 1 (IN 94-05); Indian Point 3 (IN 88-99); and North Anna 1 (Bulletin 88-02). Discusses importance of leak rate monitoring including real time monitoring, establishing appropriate alarm setpoints and operational limits, and providing clear guidance to operators on how to respond to increased leakage.*
23. U.S. NRC Information Notice 94-43, "Determination of Primary-to-Secondary Steam Generator Leak Rate," June 10, 1994.
  - *Describes problems associated with various methods used to determine steam generator primary-to-secondary leak rates. Specifically identifies a problem with using the blowdown line samples to estimate steam generator primary-to-secondary leakage in CE plants. Discusses Palo Verde tube rupture event and McGuire 1993 tube leak. Discusses importance of using one technique to detect and others to quantify leakage.*
24. U.S. NRC Information Notice 94-05, "Potential Failure of Steam Generator Tubes Sleeved With Kinetically Welded Sleeves," January 19, 1994.
  - *Discusses potential failures of B&W kinetically (explosively) welded sleeves by describing McGuire 1 tube leakage event and tube pull results. Although applicable at its time, sleeves of this design are no longer (i.e., October 2001) in use in any operating domestic steam generators. Nonetheless, the findings illustrate that tube material properties can differ from that listed in Certified Material Tests Reports.*
25. U.S. NRC Information Notice 93-56, "Weaknesses in Emergency Operating Procedures Found as a Result of Steam Generator Tube Rupture," July 22, 1993.
  - *Discusses Palo Verde tube rupture and weaknesses in the emergency operating procedures related to identifying the event as a tube rupture.*
26. U.S. NRC Information Notice 93-52, "Draft NUREG-1477, 'Voltage-Based Interim Plugging Criteria for Steam Generator Tubes'," July 14, 1993.
  - *Discusses the availability of draft NUREG-1477 for public comment. The NUREG describes voltage-based limits for repairing Westinghouse steam generator tubes with axially oriented outside diameter stress corrosion cracking at tube support plate elevations.*

27. NRC Information Notice 92-80, "Operation With Steam Generator Tubes Seriously Degraded," December 7, 1992.
- *Discusses inspection findings at ANO-2 which indicated multiple tubes did not have adequate structural integrity. Staff indicated that primary cause of excessive degradation was that inappropriate probes were used during the prior inspection (i.e., RPC probe was not used for detection of circumferential cracking despite operating experience which indicates that the tubes are susceptible to this mechanism). Emphasizes data analyst training and performance demonstration.*
28. U.S. NRC Information Notice 91-67, "Problems With the Reliable Detection of Intergranular Attack (IGA) of Steam Generator Tubing," October 21, 1991.
- *Discusses reliable detection of SCC in light of inspection findings at Trojan. Complements IN 90-49 and provides tube pull results which indicate that amplitude threshold criteria can be non-conservative. Discusses limitations of reliably detecting SCC and IGA.*
29. U.S. NRC Information Notice 91-43, "Recent Incidents Involving Rapid Increases in Primary-to-Secondary Leak Rate," July 5, 1991.
- *Discusses events (primarily Mihama 2, Maine Yankee, and TMI-1) of rapidly increasing primary to secondary leakage. Mihama 2 leakage was attributed to fatigue similar to North Anna 1 SGTR. The Maine Yankee leakage was attributed to an axial crack in the U-bend of a tube in the "steam blanketed region". The TMI-1 leakage was attributed to a fatigue crack in the lane/wedge region. Related to IN 88-99 and Bulletin 88-02. Discusses need to provide real time monitoring of leakage and the usefulness in minimizing the frequency of SGTRs.*
30. U.S. NRC Information Notice 90-49, "Stress Corrosion Cracking in PWR Steam Generator Tubes," August 6, 1990.
- *Discusses Millstone 2 experience with circumferential SCC at expansion transition and emphasizes that circumferential degradation is only detectable through the use of specialized probes such as the RPC. Also briefly discusses circumferential cracking at expansion transition at Maine Yankee, North Anna 1, Trojan, Sequoyah, and McGuire 1. Discusses circumferential cracking at tube support plates (Palisades, Indian Point 3, and Zion 1). Discusses axial cracking at tube support plates and discusses limitations of amplitude threshold screening criteria (e.g., S/N, voltage). In light of results, discusses need for operational assessments and the need for qualification and performance demonstration for data acquisition equipment and analysis. Discusses that tubes will not necessarily leak in excess of TS limits before rupture.*
31. U.S. NRC Information Notice 89-33, "Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs," March 23, 1989.
- *Discusses Westinghouse tube plug failure event at North Anna 1 due to intergranular cracking where the top portion of the mechanical plug severed and was propelled up the length of the tube by primary system pressure until it impacted the outer curvature of the tube above the seventh tube support plate. Also discusses similar degradation of Millstone 2 plugs.*



32. U.S. NRC Information Notice 89-65, "Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Supplied by Babcock and Wilcox," September 8, 1989.
  - *Discusses degradation of B&W "rolled", "ribbed", and "taper welded" design at McGuire 2 (and other plants) due to primary water stress corrosion cracking. Discusses importance of inspecting alloy 600 plugs.*
33. U.S. NRC Information Notice 88-99, "Detection and Monitoring of Sudden and/or Rapidly Increasing Primary-to-Secondary Leakage," December 20, 1988.
  - *Describes Indian Point 3 leakage event due to a circumferential crack. Discusses need for real time monitoring for primary-to-secondary leakage monitoring and procedures for responding to rapidly increasing leakage. Further analysis (i.e., after the information notice was issued) indicated the tube failed due to fatigue even though it had anti-vibration bar support in contrast to Bulletin 88-02 where the tubes "lacked" AVB support (see NUREG-1604 page 7-16).*
34. U.S. NRC, "Bases for Plugging Degraded PWR Steam Generator Tubes," Regulatory Guide 1.121.
  - *provided under separate cover to regional office staff*
35. U.S. NRC, Regulatory Guide 1.83
  - *provided under separate cover to regional office staff*
36. U.S. NRC, "Steam Generator Tube Integrity" Draft Regulatory Guide DG-1074, December 1998.
37. U.S. NRC, "Circumferential Cracking of Steam Generator Tubes", NUREG-1604, April 1997.
38. U.S. NRC, "Steam Generator Tube Failures," NUREG/CR-6365 (INEL-95/0383), April 1996.
39. U.S. NRC, "Steam Generator Operating Experience, Update for 1989-1990," NUREG/CR-5796, December 1991.
40. U.S. NRC, "Steam Generator Operating Experience, Update for 1987-1988," NUREG/CR-5349 (SAIC-89/1113), June 1989.
41. U.S. NRC, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," NUREG-0844, September 1988.
42. U.S. NRC, "Steam Generator Operating Experience, Update for 1984-1986," NUREG/CR-5150 (SAIC-87/3014), June 1988.
43. U.S. NRC, "Steam Generator Operating Experience Update for 1982-1983," NUREG-1063, June 1984.
44. U.S. NRC, "Steam Generator Tube Experience," NUREG-0886, February 1982.

45. U.S. NRC, "Summary of Tube Integrity Operating Experience with Once-Through Steam Generators," NUREG-0571, March 1980.
46. U.S. NRC, "Summary of Operating Experience with Recirculating Steam Generators," NUREG-0523, January 1979.
47. Memorandum from Brian Sheron, NRR, to John Larkins, ACRS, dated April 6, 1995, "ACRS Review of Generic Letter (GL) 95-xx, 'Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes.'"
  - *provided under separate cover to regional office staff*
48. EPRI TR-107260-R1, "Steam Generator In Situ Pressure Test Guidelines" June 1999
  - *provided under separate cover to regional office staff*
49. NEI 97-06, "Steam Generator Program Guidelines"
  - *provided under separate cover to regional office staff*
50. EPRI TR-107569, "PWR Steam Generator Examination Guidelines"
  - *provided under separate cover to regional office staff*
51. EPRI TR-107621, "Steam Generator Integrity Assessment Guidelines"
  - *provided under separate cover to regional office staff*
52. EPRI TR-104788, "PWR Primary-to-Secondary Leak Guidelines"
  - *provided under separate cover to regional office staff*

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