Response to Action Item 5-6 Section 5.4.2.2

MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #1 (AI 5-6.19)

(Editorial) The staff notes that Final Safety Analysis Report (FSAR) Table 1.9-2 (Page 9 of 33) lists FSAR Subsection 5.4.2.6 as the applicable FSAR section addressing NRC Standard Review Plan (SRP) Section 5.4.2.2. This table entry should be changed from 5.4.2.6 to 5.4.2.2 to be consistent with the FSAR.

Response

KHNP will change the typo in Table 1.9-2 from 5.4.2.6 to 5.4.2.2.

Impact on DCD

DCD Table 1.9-2 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

Table 1.9-2 (9 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
5.3.3 – Reactor Vessel Integrity	Rev. 2 03/2007	The APR1400 conforms with this SRP.	5.3.3
5.4 – Reactor Coolant System Component and Subsystem Design	Rev. 2 03/2007	The APR1400 conforms with this SRP.	5.4
5.4.1.1 – Pump Flywheel Integrity (PWR)	Rev. 3 05/2010	The APR1400 conforms with this SRP with the following exception: • Design stress criteria.	5.4.1.1
5.4.2.1 – Steam Generator Materials	Rev. 3 03/2007	The APR1400 conforms with this SRP.	5.4.2.1
5.4.2.2 – Steam Generator Program	Rev. 2 03/2007	The APR1400 conforms with this SRP.	~~5.4.2.6~~ 5.4.2.2
5.4.6 – Reactor Core Isolation Cooling System (BWR)	Rev. 4 03/2007	Not applicable (BWRs only)	N/A
5.4.7 – Residual Heat Removal (RHR) System	Rev. 5 05/2010	The APR1400 conforms with this SRP.	5.4.7
5.4.8 – Reactor Water Cleanup System (BWR)	Rev. 3 03/2007	Not applicable (BWRs only)	N/A
5.4.11 – Pressurizer Relief Tank	Rev. 4 05/2010	The APR1400 conforms with this SRP.	5.4.11
5.4.12 – Reactor Coolant System High Point Vents	Rev. 1 03/2007	The APR1400 conforms with this SRP.	5.4.12
5.4.13 – Isolation Condenser System (BWR)	03/2007	Not applicable (BWRs only)	N/A
BTP 5-1 – Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	Rev. 3 03/2007	The APR1400 conforms with this BTP.	10.4.8.3

Response to Action Item 5-6 Section 5.4.2.2

MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #2 (AI 5-6.20)

In accordance with NEI 97-06, "Steam Generator Program Guidelines," implementation of an acceptable steam generator program requires that all tubes be accessible from the primary side, that every tube can be inspected over its full length, and that every tube can be removed from service and stabilized if necessary. This is part of meeting GDC 32, since steam generator tubes are part of the reactor coolant pressure boundary and have a potential for degradation over their full length.

Revise FSAR Subsection 5.4.2.2 to state that these criteria are met for the APR1400 steam generators.

Response

KHNP will add the following sentence in Section 5.4.2.2.1 Design Description:

> "The steam generator tubes of the APR1400 can be accessed from the primary side of the steam generator for the full length inspection."

Impact on DCD

DCD Section 5.4.2.2.1 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

b. 10 CFR 50.55a(g) requires that inservice inspection (ISI) programs meet the applicable inspection requirements in ASME Section XI. The SGP is a portion of the ISI program. In addition, 10 CFR 50.55a(b)(2)(iii) addresses SG tubes and states that if the plant Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern.

c. 10 CFR 50.36 applies to the SGP in the Technical Specifications.

d. Appendix B to 10 CFR 50 applies to the implementation of the SGP. Of particular note are Criteria IX, XI, and XVI. Criterion IX requires, in part, that measures be established to ensure that special processes, including nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures. Criterion XI requires, in part, the establishment of a test program to ensure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits in applicable design documents. Criterion XVI requires, in part, that measures be established to ensure the prompt identification and correction of conditions that are adverse to quality.

e. 10 CFR 50.65 requires that licensees monitor the performance or condition of SSCs against goals to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.

5.4.2.2.1 Design Description

The SG tubes of the APR1400 can be accessed from the primary side of the SG for full length inspection.

The SGs are designed to permit access required for tube inspections, testing, plugging, and repairs. The design is described further in Subsection 5.4.2.1.6.

5.4.2.2.2 Implementation of SGP

KHNP has been operating the SGP comprehensive integration procedure, which includes the constitution and application of SGP documents, the roles and responsibilities of organizations, and also the technical program including the QA process. The SGP includes degradation assessment, inspection, integrity assessment, tube plugging and

Response to Action Item 5-6 Section 5.4.2.2

MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #3 (AI 5-6.21)

FSAR Subsection 5.4.2.2 refers to steam generator tube repair in several places. The first is in FSAR Subsection 5.4.2.2.2.1, which lists sleeves as a feature of reactor coolant pressure boundary (RCPB) integrity related to degradation assessments.

As stated in NEI 97-06, repair methods (e.g., sleeving) shall be reviewed by the NRC prior to implementation. Since the APR1400 does not propose sleeving as a tube repair method, revise FSAR Section 5.4.2.2 to delete all statements about sleeves and tube repairs (e.g., from FSAR Subsections 5.4.2.2.2.1, 5.4.2.2.2.4, 5.4.2.2.2.12, etc.). The staff notes that the references to tube repairs in FSAR Subsection 5.4.2.2.2.12 ("Reporting") make that information inconsistent with Section 5.6.7 of the proposed APR1400 Technical Specifications.

Response

KHNP will delete repair methods such as sleeving in section 5.4.2.2.2.1, 5.4.2.2.2.4, and 5.4.2.2.2.12.

Impact on DCD

DCD Sections 5.4.2.2.2.1, 5.4.2.2.2.4 and 5.4.2.2.2.12 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

repairs, primary-to-secondary leak monitoring, maintenance of secondary-side integrity, secondary-side water chemistry, primary-side water chemistry, foreign material exclusion, contractor oversight, self-assessment, and reporting.

5.4.2.2.2.1 Degradation Assessment

A degradation assessment is performed prior to the preservice inspection (PSI) and planned ISI for SGs during commercial operation to address the RCPB integrity within the SG (e.g., plugs, ~~sleeves,~~ tubes, and components that support the pressure boundary such as secondary-side components).

deleted

The assessment determines the size and location of existing and potential degradations that are likely to become harmful cracks.

The assessment considers operating experience and provides reasonable assurance that appropriate inspections are performed during the upcoming outage and that the requisite information for integrity assessment is provided.

Some of the important features of the degradation assessment are:

a. Identifying existing and potential degradation mechanisms

b. Choosing techniques to test for degradation based on the probability of detection and sizing capability

c. Establishing the number of tubes to be inspected

d. Establishing the tube integrity limits for condition monitoring and operational assessment

5.4.2.2.2.2 Inspection

SG tube inspections based on degradation assessments are conducted and follow the inspection guidance in the EPRI PWR Steam Generator Examination Guidelines (Reference 12).

Some of the important features of SG tube inspections are:

a. Sampling as supported by the degradation assessment

b. Obtaining the information necessary to develop degradation, condition monitoring, and operational assessments

c. Qualifying the inspection program by determining the accuracy and defining the elements for enhancing nondestructive examination (NDE) system performance, including technique, analysis, field analysis feedback, human performance and process controls

5.4.2.2.2.3 Integrity Assessment

SG tube integrity is assessed after each SG tube inspection. The assessment includes:

a. Condition Monitoring (CM): A backward-looking assessment that confirms that adequate SG tube integrity has been maintained during the previous inspection interval.

b. Operational Assessment (OA): A forward-looking assessment that demonstrates that tube integrity performance criteria will be met throughout the next inspection interval.

5.4.2.2.2.4 Tube Plugging and Repairs

Plugging and repair methods are qualified and implemented in accordance with industry standards (i.e., ASME Section XI, IWA-4700). The qualification of the plugging and repair techniques considers the SG conditions and mockup testing. The EPRI guidance document, "Pressurized Water Reactor Steam Generator Examination Guidelines" (Reference 12), provides a pre-service inspection of the plugging or repair. The EPRI document ~~documents~~, "PWR Steam Generator Tube Plug Assessment" deleted ~~and "PWR Steam Generator Sleeving Assessment,~~" provides ~~provide~~ further guidance on tubing maintenance and repair.

b. Review and approval of the degradation assessment

c. Review and approval of the contractor's examination procedures

d. Monitoring of the contractor's examination work progress

e. Review and approval of the contractor's deliverables

f. Review and approval of the tube integrity assessment (CM/OA) and associated support documents

5.4.2.2.2.11 Self-Assessment

Self-assessment of the SGP is performed by the SG expert team and independent peer review by knowledgeable personnel in the power stations on a periodic basis. The self-assessment identifies areas for program improvement along with program strengths.

5.4.2.2.2.12 Reporting

The following reports will be submitted to the NRC within 6 month after completion of an inspection performed under the SGP:

a. The scope of inspection performed on each SG

b. Active degradation mechanisms found,

c. NDE techniques utilized for each degradation mechanism

d. Location, orientation(if linear), and measured sizes (if available) of service-induced ~~indications~~ degradation

e. Number of tubes plugged (or repaired) during the inspection outage for each ~~active~~ deleted degradation mechanism

f. Total number and percentage of tubes plugged (or repaired) to date

Response to Action Item 5-6 Section 5.4.2.2

MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #4 (AI 5-6.22)

The second paragraph of FSAR Subsection 5.4.2.2.2.1 implies that the degradation assessment focuses only on cracking. Since the purpose of the Steam Generator Program is to maintain tube integrity, the degradation assessment must consider all potential forms of degradation.

Revise FSAR Subsection 5.4.2.2.2.1 by deleting the second paragraph to make it clear that steam generator tube degradation assessment is not limited to cracking.

Response

KHNP will delete the 2nd paragraph of section 5.4.2.2.2.1.

Impact on DCD

DCD Section 5.4.2.2.2.1 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

repairs, primary-to-secondary leak monitoring, maintenance of secondary-side integrity, secondary-side water chemistry, primary-side water chemistry, foreign material exclusion, contractor oversight, self-assessment, and reporting.

5.4.2.2.2.1 Degradation Assessment

A degradation assessment is performed prior to the preservice inspection (PSI) and planned ISI for SGs during commercial operation to address the RCPB integrity within the SG (e.g., plugs, sleeves, tubes, and components that support the pressure boundary such as secondary-side components).

deleted

~~The assessment determines the size and location of existing and potential degradations that are likely to become harmful cracks.~~

The assessment considers operating experience and provides reasonable assurance that appropriate inspections are performed during the upcoming outage and that the requisite information for integrity assessment is provided.

Some of the important features of the degradation assessment are:

a. Identifying existing and potential degradation mechanisms

b. Choosing techniques to test for degradation based on the probability of detection and sizing capability

c. Establishing the number of tubes to be inspected

d. Establishing the tube integrity limits for condition monitoring and operational assessment

5.4.2.2.2.2 Inspection

SG tube inspections based on degradation assessments are conducted and follow the inspection guidance in the EPRI PWR Steam Generator Examination Guidelines (Reference 12).

Response to Action Item 5-6 Section 5.4.2.2

MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #5 (AI 5-6.23)

FSAR Subsections 5.4.2.2.2.7 and 5.4.2.2.2.8, respectively, describe the secondary-side and primary-side water chemistry programs to limit corrosion-related degradation. These programs are described and reviewed primarily in other sections of the FSAR (9.3.4, 10.3.5, Technical Specifications 5.5.10).

Revise FSAR Subsection 5.4.2.2 to identify the subsections where these programs are more fully described.

Response

KHNP will add the following sentence in 5.4.2.2.2.7:
"More details are described in Section 10.3.5 and the Technical Specification 5.5.10 of the APR1400 DCD."

KHNP will add following sentence in 5.4.2.2.2.8:
"More details are described in Section 9.3.4.2.7 of the APR1400 DCD."

Impact on DCD
DCD Sections 5.4.2.2.2.7 and 5.4.2.2.2.8 will be revised as indicated in the Attachment.

Impact on PRA
There is no impact on the PRA.

Impact on Technical Specifications
There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports
There is no impact on any Technical, Topical or Environmental Reports.

5.4.2.2.2.5 Primary-to-Secondary Leak Monitoring

In order to prevent the leakage of primary coolant to the environment through SGs, primary-to-secondary leak monitoring procedures are established in accordance with the Technical Specifications and the EPRI "PWR Primary-to-Secondary Leak Guidelines" (Reference 13) provides further guidance. Monitoring gives operators information that is needed to safely respond to situations in which tube integrity becomes impaired and significant leakage or tube failure occurs. There are three action levels of reactor operation according to the leak rate in SGs.

5.4.2.2.2.6 Maintenance of Steam Generator Secondary-Side Integrity

Secondary-side SG components that are susceptible to degradation are monitored if their failure could prevent the SG from fulfilling its intended safety-related function. The monitoring includes design reviews, assessment of potential degradation mechanisms, industry experience for applicability, and inspections, as necessary, to provide reasonable assurance that degradation of these components does not threaten tube structural and leakage integrity or the ability of the plant to achieve and maintain safe shutdown.

The secondary-side visual inspection program defines the scope of inspection, and the inspection procedures and methodology to be used, when secondary-side visual inspections are performed. Additional guidance is provided in the EPRI "Steam Generator Integrity Assessment Guidelines" (Reference 14).

5.4.2.2.2.7 Secondary-Side Water Chemistry

Procedures are prepared for monitoring and controlling secondary-side water chemistry to inhibit secondary-side corrosion-induced degradation such as outside diameter stress corrosion cracking in accordance with the EPRI "PWR Secondary Water Chemistry Guidelines" (Reference 15).

More details are described in Section 10.3.5 and the Technical Specification 5.5.10 of the APR1400 DCD.

5.4.2.2.2.8 Primary-Side Water Chemistry

Procedures are prepared for monitoring and controlling primary-side water chemistry to inhibit primary-side corrosion-induced degradation such as primary water stress corrosion cracking in accordance with the EPRI "PWR Primary Water Chemistry Guidelines" (Reference 16).

More details are described in Section 9.3.4.2.7 of the APR1400 DCD.

5.4.2.2.2.9 Foreign Material Exclusion

Procedures are prepared for control and monitoring of foreign objects and loose parts to prevent fretting and wear degradation of the tubing The SG program includes secondary-side visual inspections and procedures to preclude the introduction of foreign objects into either the primary or secondary side of the SG whenever it is opened for inspections, maintenance, repairs, modifications, or other reasons.

Such procedures include the following as a minimum:

a. Detailed accountability for all tools and equipment used during any activity when the primary or secondary side is open

b. Appropriate controls and accountability for foreign objects such as eyeglasses and personal dosimetry

c. Cleanliness requirements

d. Accountability for components and parts removed from the internals of major components (e.g., reassembly of cut and removed components)

The potential for introduction of loose parts or foreign objects from secondary-side systems is also considered.

5.4.2.2.2.10 Contractor Oversight

Procedures are established for oversight of contractor work for the SG inspections and cleaning work that will be performed during the refueling outage. Critical aspects of the oversight includethe following as a minimum:

a. Review and approval of the scope of work to be performed by a contractor

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MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #6 (AI 5-6.24)

FSAR Subsection 5.4.2.2.2.12 ("Reporting") contains three paragraphs at the end that are not related to reporting. These paragraphs include a description of the tube surveillance program, preservice inspection (PSI), and differences between the Technical Specifications and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

Revise FSAR Subsection 5.4.2.2.2.12 to delete the third of these paragraphs and relocate the information in the first two of these paragraphs to more appropriately titled parts of Subsection 5.4.2.2.

Response

The first two paragraphs in the Subsection 5.4.2.2.2.12 will be moved to the end of Section 5.4.2.2.2.3 (Integrity Assessment) and the third paragraph will be deleted.

Impact on DCD

DCD Sections 5.4.2.2.2.3 and 5.4.2.2.2.12 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

Some of the important features of SG tube inspections are:

a. Sampling as supported by the degradation assessment

b. Obtaining the information necessary to develop degradation, condition monitoring, and operational assessments

c. Qualifying the inspection program by determining the accuracy and defining the elements for enhancing nondestructive examination (NDE) system performance, including technique, analysis, field analysis feedback, human performance and process controls

5.4.2.2.2.3 <u>Integrity Assessment</u>

SG tube integrity is assessed after each SG tube inspection. The assessment includes:

The SG tube surveillance program, including performance criteria for tube integrity, tube repair criteria, and tube inspections, is described in the Technical Specifications (Chapter 16) Subsection 5.5.9. The repair criteria are determined based on NRC RG 1.121 and the EPRI guidelines. Limiting conditions for operation and reactor coolant system operational leakage limits, including primary-to-secondary leakage limits, are described in the Technical Specifications (Chapter 16) Subsection 3.4.12 and 3.4.17.

b. Operational Assessment (OA): A forward-looking assessment that demonstrates that tube integrity performance criteria will be met throughout the next inspection interval.

5.4.2.2.2.4 <u>Tube Plugging and Repairs</u>

Plugging and repair methods are qualified and implemented in accordance with industry standards (i.e., ASME Section XI, IWA-4700). The qualification of the plugging and repair techniques considers the SG conditions and mockup testing. The EPRI guidance document, "Pressurized Water Reactor Steam Generator Examination Guidelines" (Reference 12), provides a pre-service inspection of the plugging or repair. The EPRI documents, "PWR Steam Generator Tube Plug Assessment" and "PWR Steam Generator Sleeving Assessment," provide further guidance on tubing maintenance and repair.

g. The results of condition monitoring, including the results of tube pulls and in situ testing

h. The effective plugging percentage for all repaired tubes in each SG

i. Repair method utilized and the number of tubes repaired by each repair method

deleted here and moved to the end of the subsection 5.4.2.2.2.3

~~The SG tube surveillance program, including performance criteria for tube integrity, tube repair criteria, and tube inspections, is described in the Technical Specifications (Chapter 16), Subsection 5.5.9. The repair criteria are determined based on NRC RG 1.121 and the EPRI guidelines. Limiting conditions for operation and reactor coolant system operational leakage limits, including primary-to-secondary leakage limits, are described in the Technical Specifications (Chapter 16), Subsections 3.4.12 and 3.4.17.~~

deleted

~~Preservice inspection of all tubes in accordance with ASME Section XI and the EPRI PWR Steam Generator Examination Guidelines described in NEI 97-06 is performed using techniques that will also be used during subsequent inspections.~~

deleted

~~If Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern.~~

5.4.2.3 Tests and Inspections

Prior to, during, and after fabrication of the steam generator, nondestructive tests based on Section III of the ASME Code are performed.

Initial hydrostatic tests of the primary and secondary sides of the steam generator are conducted in accordance with ASME Section III. Leak tests are also performed. Following satisfactory performance of the hydrostatic tests, magnetic-particle inspections are made on all accessible welds.

Inservice inspections of the steam generator are performed in accordance with ASME Section XI, including automatic ultrasonic for SG transition region.

Response to Action Item 5-6 Section 5.4.2.2

MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #7 (AI 5-6.25)

The next-to-last paragraph in FSAR Subsection 5.4.2.2.2.12 states that PSI of all tubes is performed in accordance with the EPRI guidelines, NEI 97-06, and Section XI of the ASME Code.

Revise FSAR Subsection 5.4.2.2.2.12 to clarify the requirements using language consistent with the Technical Specification (TS) requirements for inservice inspections. This is necessary because the PSI for steam generator tubing is not addressed by the TS, and the requirements are not clear in Section III of the ASME Code. Revising FSAR Subsection 5.4.2.2.2.12 can be accomplished with a statement such as:

“Preservice inspection is performed on the full length of 100% of the tubes in each steam generator using techniques capable of detecting degradation and fabrication abnormalities along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet. The preservice inspection will be performed after field hydrostatic testing and before operation.”

Response

The paragraph provided will be added to section 5.4.2.2.2.2.

Impact on DCD

DCD Sections 5.4.2.2.2.2 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

Some of the important features of SG tube inspections are:

a. Sampling as supported by the degradation assessment

b. Obtaining the
and operation

c. Qualifying th
elements for
including technique, analysis, field analysis feedback, human performance and process controls

Preservice inspection is performed on the full length of 100% of the tubes in each steam generator using techniques capable of detecting degradation and fabrication abnormalities along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet. The preservice inspection will be performed after field hydrostatic testing and before operation.

5.4.2.2.2.3 Integrity Assessment

SG tube integrity is assessed after each SG tube inspection. The assessment includes:

a. Condition Monitoring (CM): A backward-looking assessment that confirms that adequate SG tube integrity has been maintained during the previous inspection interval.

b. Operational Assessment (OA): A forward-looking assessment that demonstrates that tube integrity performance criteria will be met throughout the next inspection interval.

5.4.2.2.2.4 Tube Plugging and Repairs

Plugging and repair methods are qualified and implemented in accordance with industry standards (i.e., ASME Section XI, IWA-4700). The qualification of the plugging and repair techniques considers the SG conditions and mockup testing. The EPRI guidance document, "Pressurized Water Reactor Steam Generator Examination Guidelines" (Reference 12), provides a pre-service inspection of the plugging or repair. The EPRI documents, "PWR Steam Generator Tube Plug Assessment" and "PWR Steam Generator Sleeving Assessment," provide further guidance on tubing maintenance and repair.

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Issue #8 (AI 5-6.26)

Paragraph b in Subsection 5.4.2.2 and the last paragraph in Subsection 5.4.2.2.2.12 have statements, such as the following, that address potential differences between the TS and Section XI of the ASME Code:
If the plant Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern.

These statements are based on earlier versions of the regulations in 10 CFR 50.55a. That statement is no longer part of 10 CFR 50.55a.

Revise Paragraph b in Subsection 5.4.2.2 and the last paragraph in Subsection 5.4.2.2.2.12 to delete these statements from the FSAR. If the applicant is aware of any potential conflicts/differences between the ASME Code and the TS such that the requirements of both cannot be met, revise the FSAR to address those conflicts/differences.

Response

The following sentence in paragraph b of Section 5.4.2.2 will be deleted.

"In addition, 10 CFR 50.55a(b)(2)(iii) addresses SG tubes and states that if the plant Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern."

The last paragraph in Section 5.4.2.2.2.12 will also be deleted.

Impact on DCD

DCD Sections 5.4.2.2 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

b. 10 CFR 50.55a(g) requires that inservice inspection (ISI) programs meet the applicable inspection requirements in ASME Section XI. The SGP is a portion of the ISI program. ~~In addition, 10 CFR 50.55a(b)(2)(iii) addresses SG tubes and states that if the plant Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern.~~ deleted

c. 10 CFR 50.36 applies to the SGP in the Technical Specifications.

d. Appendix B to 10 CFR 50 applies to the implementation of the SGP. Of particular note are Criteria IX, XI, and XVI. Criterion IX requires, in part, that measures be established to ensure that special processes, including nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures. Criterion XI requires, in part, the establishment of a test program to ensure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits in applicable design documents. Criterion XVI requires, in part, that measures be established to ensure the prompt identification and correction of conditions that are adverse to quality.

e. 10 CFR 50.65 requires that licensees monitor the performance or condition of SSCs against goals to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.

5.4.2.2.1 Design Description

The SGs are designed to permit access required for tube inspections, testing, plugging, and repairs. The design is described further in Subsection 5.4.2.1.6.

5.4.2.2.2 Implementation of SGP

KHNP has been operating the SGP comprehensive integration procedure, which includes the constitution and application of SGP documents, the roles and responsibilities of organizations, and also the technical program including the QA process. The SGP includes degradation assessment, inspection, integrity assessment, tube plugging and

g. The results of condition monitoring, including the results of tube pulls and in situ testing

h. The effective plugging percentage for all repaired tubes in each SG

i. Repair method utilized and the number of tubes repaired by each repair method

The SG tube surveillance program, including performance criteria for tube integrity, tube repair criteria, and tube inspections, is described in the Technical Specifications (Chapter 16), Subsection 5.5.9. The repair criteria are determined based on NRC RG 1.121 and the EPRI guidelines. Limiting conditions for operation and reactor coolant system operational leakage limits, including primary-to-secondary leakage limits, are described in the Technical Specifications (Chapter 16), Subsections 3.4.12 and 3.4.17.

Preservice inspection of all tubes in accordance with ASME Section XI and the EPRI PWR Steam Generator Examination Guidelines described in NEI 97-06 is performed using techniques that will also be used during subsequent inspections.

deleted

~~If Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern.~~

5.4.2.3 Tests and Inspections

Prior to, during, and after fabrication of the steam generator, nondestructive tests based on Section III of the ASME Code are performed.

Initial hydrostatic tests of the primary and secondary sides of the steam generator are conducted in accordance with ASME Section III. Leak tests are also performed. Following satisfactory performance of the hydrostatic tests, magnetic-particle inspections are made on all accessible welds.

Inservice inspections of the steam generator are performed in accordance with ASME Section XI, including automatic ultrasonic for SG transition region.

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MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #9 (AI 5-6.27)

Several issues need to be addressed for the APR 1400 Technical Specifications because they are inconsistent with the latest Standard Technical Specifications. The STS provide for the establishment and implementation of steam generator program to ensure that tube integrity is maintained, which is part of meeting GDC 32. The relevant STS are comprised of Revision 4 of NUREG-1432 (for Combustion Engineering plants) plus the changes incorporated by Technical Specification Task Force Traveler 510 ("Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," Revision 2, ML110610350). Examples of the differences between the APR1400 TS and the STS are listed below and should be revised, or the applicant should provide justification for not using the STS.

- Revise the definition of Identified LEAKAGE (a.3.) in TS 1.1 to include "(primary to secondary LEAKAGE)".

  Response

  KHNP will add the definition of "primary to secondary leakage" to TS 1.1 as shown in Attachment 1.

- Revise the definition of Pressure Boundary Leakage (c) in TS 1.1 to change "except SG LEAKAGE" to "except primary to secondary leakage".

  Response

  KHNP will revise "except SG leakage" to "except primary to secondary leakage" in TS 1.1 as shown in Attachment 1.

- Revise TS 3.4.17, "Steam Generator (SG) Tube Integrity," in three places to be consistent with the STS. TSTF-510 changed the terminology from "tube repair criteria" to "tube plugging [or repair] criteria." (Plants without an approved tube repair method (sleeving) exclude the bracketed part.)

  Response

  KHNP will change "tube repair criteria" to "tube plugging criteria" in TS 3.4.17 in accordance with the STS as shown in Attachments 2 and 3.

- Revise the terminology in the steam generator TS and Bases to replace "repair criteria" with "plugging criteria," for consistency with TSTF-510. There are also corresponding changes in Subsection 5.4.2.2 of the FSAR.

  Response

  KHNP will change "repair criteria" to "plugging criteria" in the TS and Bases in accordance with TSTF-510 as shown in Attachments 4.

- Revise the requirements in for the frequency of verifying tube integrity in TS 5.5.9, "Steam Generator (SG) Program." These requirements were changed in TSTF-510 and are explained and shown in the TSTF-510 documentation (see ML110610350 or the

"Consolidated Line Item Improvement Process" link on the NRC's Technical Specifications web page).

Response

KHNP will change TS 5.5.9.d.2 as shown in Attachment 5.

- Revise the one instance of "tube repair" that was not changed (by mistake) in TSTF-510 to "tube plugging" (TS 5.5.9.d.2, 3rd sentence).

Response

KHNP will change TS 5.5.9.d.2 as shown in Attachment 5.

- Revise TS 5.6.7 ("Steam Generator Tube Inspection Report") and Bases section B 3.4.18 for consistency with TSTF-510.

Response

KHNP will change TS 5.6.7 in accordance with TSTF-510 as shown in Attachment 6.

- In TS 5.5.9.a, in the last sentence, change "inspected, plugged" to "inspected or plugged" to clarify the meaning.

Response

KHNP will change "inspected, plugged" to "inspected or plugged" in TS 5.5.9 as shown in Attachment 7.

- In the Background of the TS Bases, Section B 3.4.12, at the end of the fifth paragraph, "to not interfere with the function of RCS leakage detection system." The corresponding phrase in the STS is, "to not interfere with RCS leakage detection." Revise the wording for consistency with the STS.

Response

KHNP will revise the words "to not interfere with the function of RCS leakage detection system" to "to not interfere with RCS leakage detection" as shown in Attachment 8.

- At the end of the last sentence of Applicable Safety Analyses, Bases Section B.3.4.12, change "LCO SELECTION CRITERION 2" to "Criterion 2 of 10 CFR 50.36(c)(2)(ii)," for clarity and consistency with the STS.

Response

KHNP will revise the words "LCO SELECTION CRITERION 2" to "Criterion 2 of 10 CFR 50.36(c)(2)(ii)" as shown in Attachment 8.

- In the ACTION Section A.1 of TS Bases Section B.3.4.12, delete primary-to-secondary LEAKAGE from the first sentence. Note 2 in the associated SURVEILLANCE REQUIREMENTS states that this is not applicable to primary-to-secondary LEAKAGE.

Response

KHNP will delete "Except for primary to secondary LEAKAGE" in the first sentence of the first paragraph of page B3.4-54 as shown in Attachment 8.

Response to Action Item 5-6 Section 5.4.2.2

- In the SURVEILLANCE REQUIREMENTS section in TS Bases Section B.3.4.12, at the end of the second paragraph under SR 3.4.12.2, for consistency with the STS, add the sentence, "If it is not practical to assign the leakage to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG."

  Response

  KHNP will add following sentence at the end of the second paragraph of SR 3.4.12.2 as shown in Attachment 8:

  > "If it is not practical to assign the leakage to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG."

- In the APPLICABLE SAFETY ANALYSES section of TS Bases Section b.3.4.17, in the first paragraph on page B 3.4.17-2, the APR 1400 bases read, "… released to the atmosphere via safety valves and relief valves." Corresponding wording in the STS is, "… released to the atmosphere via safety valves and the majority is discharged to the main condenser." Revise the APR1400 bases for consistency with the STS.

  Response

  KHNP will revise the words, "… released to the atmosphere via safety valves and relief valves" to "… released to the atmosphere via safety valves and the majority is discharged to the main condenser" as shown in Attachment 9.

- In the APPLICABLE SAFETY ANALYSES section of TS Bases Section b.3.4.17, in the second sentence of the second paragraph on page B 3.4.17-2, the APR 1400 bases omit the phrase, "or is assumed to increase to [1.13 L/min (0.3 gpm)] as a result of accident induced conditions." (The brackets indicate this is a plant-specific leak rate). Revise the APR 1400 bases for consistency with the STS.

  Response

  KHNP will add following the following phrase in the second sentence of the second paragraph on page B 3.4.17-2 as shown in Attachment 9:

  > "or is assumed to increase to [3.785 L/min (1.0 gpm)] as a result of accident induced conditions."

- Delete the two paragraphs at the top of page B 3.4.17-3 because they are duplicates from the bottom of the previous page.

  Response

  KHNP will delete the two paragraphs at the top of page B 3.4.17-3 as shown in Attachment 9.

- Since the APR 1400 design certification is not requesting approval of tube repair methods (e.g., sleeving), delete the three references to tube repairs from the TS Bases Section SR 3.4.17.2. The three references are, "[repaired or], [or repaired], and the sentence, "Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program."

Response to Action Item 5-6 Section 5.4.2.2

Response

KHNP will delete the three references from the TS Bases Section SR 3.4.17.2 as shown in Attachment 9.

- Change the first reference for TS Bases Section 3.4.17 from IEEE Standard 603 to NEI 97-06.

  Response

  KHNP will change the first reference "IEEE Standard 603" in the TS Bases Section 3.4.17 to "NEI 97-06" as shown in Attachment 9.

- Revise TS Bases Section 3.4.17 as needed for consistency with TSTF-510 Bases Section 3.4.18.

  Response

  KHNP will revise TS Bases Section 3.4.17 by using the following terminologies: plugging, degradation, growth, etc. as shown in Attachment 9.

Impact on DCD

DCD Section 5.4.2.2 will be revised as indicated in the Attachments.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

Technical Specifications 1.1, 3.4.17, 5.5.9, and 5.6.7 and Bases B.3.4.12 and B.3.4.17 will be revised as indicated in the Attachments.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

1.1 Definitions

LEAKAGE (continued)

(Primary to Secondary LEAKAGE)

3. Reactor coolant system (RCS) LEAKAGE through a steam generator (SG) to the secondary system.

b. Unidentified LEAKAGE

All leakage (except RCP seal water injection or leakoff) which is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

primary to secondary leakage

LEAKAGE (except ~~SG LEAKAGE~~) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

MAXIMUM ALLOWABLE CONTAINMENT LEAKAGE RATE (L_a)

MAXIMUM ALLOWABLE CONTAINMENT LEAKAGE RATE (L_a) shall be 0.1 % of containment air volume, per day at the calculated peak containment pressure (P_a).

MID-LOOP

MID-LOOP is defined as the plant condition with the fuel in the reactor vessel and the reactor coolant level below the top of the hot legs at their junction with the reactor vessel.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, reactor coolant cold leg temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in reactor vessel.

OPERABLE - OPERABILITY

A system, subsystem, division, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, train, component, or device to perform its specified function(s) are also capable of performing their related support function(s).

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

plugging

All SG tubes satisfying the tube ~~repair~~ criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

plugging

-- NOTE --
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube ~~repair~~ criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of affected tube(s) is maintained until next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug affected tube(s) in accordance with Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A or B not met.	B.1 Be in MODE 3.	6 hours
OR	AND	
SG tube integrity not maintained.	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR	3.4.17.1	Verify SG tube integrity in accordance with Steam Generator Program.	In accordance with Steam Generator Program
SR	3.4.17.2	Verify each inspected SG tube that satisfies tube ~~repair~~ plugging criteria is plugged in accordance with Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.4.2.1.6 Provisions for Accessing the Primary and Secondary Sides of the Steam Generator

The steam generators have 533.4 mm (21 in) manways (Figure 5.4.2-1). On the primary side, there is one manway for the cold leg side and another for the hot leg side. Manway locations are optimized for use of remote manipulators for inspection and maintenance. Access for eddy current testing is through the primary-side manways.

On the secondary side, two manways are provided to allow access to the separator and dryer area. In addition, an internal hatch provides access to the top of the tube bundle. These openings allow inspection, which provides information on the condition of separation equipment, feedwater ring, and top of the tube bundle. Two 203.2 mm (8 in) handholes, at the tubesheet elevation, are included to provide access for tubesheet sludge lancing as well as for inspection of the downcomer annulus. These handholes can be used to remotely inspect for and retrieve loose parts. In order to enhance the steam generator integrity, the feedwater box is designed to limit the introduction of foreign objects greater than the 6.35 mm (0.25 in) tube gap through economizer feedwater region, which is a major path of foreign object inflow. Two 127.0 mm (5 in) inspection holes are provided to remove the foreign objects trapped in the feedwater box.

5.4.2.2 Steam Generator Program

The purpose of a steam generator program (SGP) is to maintain the structural and leakage integrity of steam generator (SG) tubes. An SGP provides effective monitoring and management of tube degradation and degradation precursors for prompt preventive and corrective actions. The SGP contains a balance of prevention, inspection, ~~evaluation and repair,~~ and evaluation and leakage monitoring measures. The SGP is established and maintained based on the requirements of NEI 97-06 (Reference 11) and its referenced EPRI guidelines.

The SGP complies with the relevant requirements of the following NRC regulations:

a. GDC 32 of Appendix A to 10 CFR 50 requires, in part, that the designs of all components that are part of the RCPB permit periodic inspection and testing of critical areas and features to assess their structural and leak-tight integrity.

b. 10 CFR 50.55a(g) requires that inservice inspection (ISI) programs meet the applicable inspection requirements in ASME Section XI. The SGP is a portion of the ISI program. ~~In addition, 10 CFR 50.55a(b)(2)(iii) addresses SG tubes and states that if the plant Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern.~~ deleted

c. 10 CFR 50.36 applies to the SGP in the Technical Specifications.

d. Appendix B to 10 CFR 50 applies to the implementation of the SGP. Of particular note are Criteria IX, XI, and XVI. Criterion IX requires, in part, that measures be established to ensure that special processes, including nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures. Criterion XI requires, in part, the establishment of a test program to ensure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits in applicable design documents. Criterion XVI requires, in part, that measures be established to ensure the prompt identification and correction of conditions that are adverse to quality.

e. 10 CFR 50.65 requires that licensees monitor the performance or condition of SSCs against goals to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.

5.4.2.2.1 Design Description

The SGs are designed to permit access required for tube inspections, testing, ~~plugging, and repairs~~ and plugging. The SG tubes of the APR1400 can be accessed from the primary side of the SG for full length inspection. The design is described further in Subsection 5.4.2.1.6.

5.4.2.2.2 Implementation of SGP

KHNP has been operating the SGP comprehensive integration procedure, which includes the constitution and application of SGP documents, the roles and responsibilities of organizations, and also the technical program including the QA process. The SGP includes degradation assessment, inspection, integrity assessment, ~~tube plugging and~~ and tube plugging

~~repairs,~~ primary-to-secondary leak monitoring, maintenance of secondary-side integrity, secondary-side water chemistry, primary-side water chemistry, foreign material exclusion, contractor oversight, self-assessment, and reporting.

5.4.2.2.2.1 Degradation Assessment

A degradation assessment is performed prior to the preservice inspection (PSI) and planned ISI for SGs during commercial operation to address the RCPB integrity within the SG (e.g., plugs, ~~sleeves,~~ tubes, and components that support the pressure boundary such as secondary-side components).

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~~The assessment determines the size and location of existing and potential degradations that are likely to become harmful cracks.~~

The assessment considers operating experience and provides reasonable assurance that appropriate inspections are performed during the upcoming outage and that the requisite information for integrity assessment is provided.

Some of the important features of the degradation assessment are:

a. Identifying existing and potential degradation mechanisms

b. Choosing techniques to test for degradation based on the probability of detection and sizing capability

c. Establishing the number of tubes to be inspected

d. Establishing the tube integrity limits for condition monitoring and operational assessment

5.4.2.2.2.2 Inspection

SG tube inspections based on degradation assessments are conducted and follow the inspection guidance in the EPRI PWR Steam Generator Examination Guidelines (Reference 12).

The COL applicant is to prepare PSI and ISI program of the SG tubes.

AI 5-6.27_5.4.2.2_#9 Attachment 3 (4/7)

Some of the important features of SG tube inspections are:

a. Sampling as supported by the degradation assessment

b. Obtaining the
and operation

Preservice inspection is performed on the full length of 100% of the tubes in each steam generator using techniques capable of detecting degradation and fabrication abnormalities along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet. The preservice inspection will be performed after field hydrostatic testing and before operation.

c. Qualifying th
elements for
including technique, analysis, field analysis feedback, human performance and process controls

5.4.2.2.2.3 Integrity Assessment

SG tube integrity is assessed after each SG tube inspection. The assessment includes:

The SG tube surveillance program, including performance criteria for tube integrity, tube repair criteria, and tube inspections, is described in the Technical Specifications (Chapter 16) Subsection 5.5.9. The repair criteria are determined based on NRC RG 1.121 and the EPRI guidelines. Limiting conditions for operation and reactor coolant system operational leakage limits, including primary-to-secondary leakage limits, are described in the Technical Specifications (Chapter 16) Subsection 3.4.12 and 3.4.17.

at
on

b. Operational Assessment (OA): A forward-looking assessment that demonstrates that tube integrity performance criteria will be met throughout the next inspection interval.

deleted

5.4.2.2.2.4 Tube Plugging ~~and Repairs~~

Plugging ~~and repair~~ methods are qualified and implemented in accordance with industry standards (i.e., ASME Section XI, IWA-4700). The qualification of the plugging ~~and repair~~ techniques considers the SG conditions and mockup testing. The EPRI guidance document, "Pressurized Water Reactor Steam Generator Examination Guidelines" (Reference 12), provides a pre-service inspection of the plugging ~~or repair~~. The EPRI ~~documents~~, "PWR Steam Generator Tube Plug Assessment" ~~and "PWR Steam Generator Sleeving Assessment,"~~ ~~provide~~ further guidance on tubing maintenance ~~and repair~~.

document

provides

deleted

cracking in accordance with the EPRI "PWR Primary Water Chemistry Guidelines" (Reference 16). More details are described in Section 9.3.4.2.7 of the APR1400 DCD.

5.4.2.2.2.9 Foreign Material Exclusion

Procedures are prepared for control and monitoring of foreign objects and loose parts to prevent fretting and wear degradation of the tubing The SG program includes secondary-side visual inspections and procedures to preclude the introduction of foreign objects into either the primary or secondary side of the SG whenever it is opened for inspections, maintenance, ~~repairs,~~ modifications, or other reasons.

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Such procedures include the following as a minimum:

a. Detailed accountability for all tools and equipment used during any activity when the primary or secondary side is open

b. Appropriate controls and accountability for foreign objects such as eyeglasses and personal dosimetry

c. Cleanliness requirements

d. Accountability for components and parts removed from the internals of major components (e.g., reassembly of cut and removed components)

The potential for introduction of loose parts or foreign objects from secondary-side systems is also considered.

5.4.2.2.2.10 Contractor Oversight

Procedures are established for oversight of contractor work for the SG inspections and cleaning work that will be performed during the refueling outage. Critical aspects of the oversight includethe following as a minimum:

a. Review and approval of the scope of work to be performed by a contractor

b. Review and approval of the degradation assessment

c. Review and approval of the contractor's examination procedures

d. Monitoring of the contractor's examination work progress

e. Review and approval of the contractor's deliverables

f. Review and approval of the tube integrity assessment (CM/OA) and associated support documents

5.4.2.2.2.11 Self-Assessment

Self-assessment of the SGP is performed by the SG expert team and independent peer review by knowledgeable personnel in the power stations on a periodic basis. The self-assessment identifies areas for program improvement along with program strengths.

5.4.2.2.2.12 Reporting

The following reports will be submitted to the NRC within 6 month after completion of an inspection performed under the SGP:

a. The scope of inspection performed on each SG

b. Active degradation mechanisms found,

c. NDE techniques utilized for each degradation mechanism

d. Location, orientation(if linear), and measured sizes (if available) of service-induced ~~indications~~ degradation

e. Number of tubes plugged (or repaired) during the inspection outage for each ~~active~~ deleted degradation mechanism

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in situ testing

h. Repair method utilized and the number of tubes repaired by each repair method.

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~~i. Repair method utilized and the number of tubes repaired by each repair method~~

deleted here and moved to the end of the subsection 5.4.2.2.2.3

~~The SG tube surveillance program, including performance criteria for tube integrity, tube repair criteria, and tube inspections, is described in the Technical Specifications (Chapter 16), Subsection 5.5.9. The repair criteria are determined based on NRC RG 1.121 and the EPRI guidelines. Limiting conditions for operation and reactor coolant system operational leakage limits, including primary-to-secondary leakage limits, are described in the Technical Specifications (Chapter 16), Subsections 3.4.12 and 3.4.17.~~

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~~Preservice inspection of all tubes in accordance with ASME Section XI and the EPRI PWR Steam Generator Examination Guidelines described in NEI 97-06 is performed using techniques that will also be used during subsequent inspections.~~

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~~If Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern.~~

5.4.2.3 Tests and Inspections

Prior to, during, and after fabrication of the steam generator, nondestructive tests based on Section III of the ASME Code are performed.

Initial hydrostatic tests of the primary and secondary sides of the steam generator are conducted in accordance with ASME Section III. Leak tests are also performed. Following satisfactory performance of the hydrostatic tests, magnetic-particle inspections are made on all accessible welds.

Inservice inspections of the steam generator are performed in accordance with ASME Section XI, including automatic ultrasonic for SG transition region.

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.12, “RCS Operational Leakage,” plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and relief valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., do not rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary leakage from all SGs of 1.13 L/min (0.3 gpm). For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.15, “RCS Specific Activity” limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR Part 100 (Reference 3), or the NRC-approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the ~~repair~~ plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ plugging criteria is removed from service by plugging. If a tube was determined to satisfy the ~~repair~~ plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, “Steam Generator Program,” and describe acceptable SG tube performance.

BASES

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube ~~repair~~ criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Tube Surveillance Program. The SG ~~repair~~ criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

plugging

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Reference 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the “as found” condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

plugging

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube ~~repair~~ criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

plugging

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ criteria is ~~[repaired or]~~ removed from service by plugging. The tube ~~repair~~ criteria delineated in Specification 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube ~~repair~~ criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

plugging

deleted

Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.

plugging

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the ~~repair~~ criteria are plugged ~~[or repaired]~~ prior to subjecting the SG tubes to significant primary to secondary pressure differential.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary-to-secondary accident-induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1.14 L/min (0.3 gpm) per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.12, “RCS Operational LEAKAGE.”

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40 % of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integ
degra
flaws
asses
empl

1.

2.

After the first refueling outage following SG installation, inspect each SG every 72 effective full power months or every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If the degradation assessment indicates potential degradation at the next scheduled inspection, the number of inspections for each degradation mechanism in that inspection period may be prorated such that the fraction of tubes/locations inspected at the end of the inspection period is at least equal to the ratio of the number of SG inspection outages performed subsequent to the determination that a new degradation mechanism may occur or that new locations may be susceptible to degradation mechanisms divided by the total number of SG inspection outages performed in that inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

5.6 Reporting Requirements

5.6.5 Accident Monitoring Report

When a report is required by Condition B or F LCO 3.3.11, "Accident Monitoring Instrumentation (AMI)," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.6 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

degradation

d. Location, orientation (if linear), and measured sizes (if available) of service induced ~~indications~~,

deleted

e. Number of tubes plugged during the inspection outage for each ~~active~~ degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. Repair method utilized and the number of tubes repaired by each repair method.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are ~~inspected, plugged~~ inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

GDC 30 (Reference 1) requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. NRC RG 1.45 (Reference 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of LEAKAGE inside containment is expected from auxiliary systems that cannot be made 100 % leak tight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with ~~the function of~~ RCS leakage detection ~~system~~.

deleted

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

dleted The

APPLICABLE SAFETY ANALYSES

~~Except for primary to secondary LEAKAGE~~, ~~the~~ safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of LEAKAGE can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1.13 L/min (0.3 gpm) primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The LEAKAGE contaminates the secondary fluid.

The DCD Tier 2 (Reference 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to condenser. The 1.13 L/min (0.3 gpm) primary to secondary LEAKAGE is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes the entire 1.13 L/min (0.3 gpm) primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB are well within the limits defined in 10 CFR 50.34.

RCS operational LEAKAGE satisfies ~~LCO SELECTION CRITERION 2~~.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gasket is not pressure boundary LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity or containment sump level. These leakage detection systems are specified in LCO 3.4.14, “RCS Leakage Detection Instrumentation.”

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 0.39 L/min (150 gpd) cannot be measured accurately by an RCS water inventory balance.

The 72-hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leak detection in the prevention of accidents.

SR 3.4.12.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 0.39 L/min (150 gpd) through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, “Steam Generator Tube Integrity,” should be evaluated.

The 0.39 L/min (150 gpd) limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG.

If it is not practical to assign the leakage to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The SR is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state ope[illegible]teady sta[illegible] pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The 72-hour Frequency is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Reference 5).

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.12, “RCS Operational Leakage,” plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves ~~and relief valves~~. the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., do not rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary leakage from all SGs of ~~1.13 L/min (0.3 gpm)~~. 3.785 L/min (1.0 gpm) or is assumed to increase to 3.785 L/min (1.0 gpm) as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the L… For accidents that assume f… function of the amount of act… dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR Part 100 (Reference 3), or the NRC-approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the ~~repair~~ criteria be plugged in accordance with the Steam Generator Program.

plugging

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ criteria is removed from service by plugging. If a tube was determined to satisfy the ~~repair~~ criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, “Steam Generator Program,” and describe acceptable SG tube performance.

BASES

LCO (continued)

deleted

~~In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.~~

~~A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "Steam Generator Program," and describe acceptable SG tube performance.~~

The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

BASES

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube ~~repair~~ plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Tube Surveillance Program. The SG ~~repair~~ plugging criteria define limits on SG tube degradation that allow for ~~flaw~~ degradation growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Reference 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the “as found” condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain ~~flaws~~ degradation satisfying the tube ~~repair~~ plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

plugging

degradation

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ criteria is ~~[repaired or]~~ removed from service by plugging. The tube ~~repair~~ criteria delineated in Specification 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the ~~flaw~~ size measurement and for future ~~flaw~~ growth. In addition, the tube ~~repair~~ criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

plugging

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~~Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.~~

plugging

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the ~~repair~~ criteria are plugged ~~[or repaired]~~ prior to subjecting the SG tubes to significant primary to secondary pressure differential.

NEI 97-06, "Steam Generator Program Guidelines."

BASES

REFERENCES

1. ~~IEEE Standard 603-1991~~
2. 10 CFR Part 50, Appendix A, GDC 19.
3. 10 CFR Part 100.
4. ASME Section III, Subsection NB.
5. NRC RG 1.121, August 1976.
6. EPRI 1022832, “Pressurized Water Reactor Steam Generator Examination Guidelines,” November 2011.

Response to Action Item 5-6 Section 5.4.2.2

MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #10 (AI 5-6.28)

In 2013, the U.S. NRC issued Information Notice (IN) 2013-20, "Steam Generator Channel Head and Tubesheet Degradation." IN 2013-20 describes operating experience with exposure of the carbon steel base material in the steam generator channel head and primary side of the tubesheet. The base material in those cases was originally protected by corrosion-resistant weld material (stainless steel or nickel-base alloy).

Revise FSAR Subsection 5.4.2.2 to describe:

> The controls in place to prevent damage to cladding during fabrication and maintenance activities that could expose the underlying carbon steel to primary coolant
> The inspections that will be performed to evaluate the integrity of corrosion-resistant cladding and monitor the integrity of locations where exposure of the base metal has occurred.

Response

KHNP requests that this issue be an RAI since additional evaluation is needed to determine the appropriate controls and inspections that are necessary to respond to this request.

Impact on DCD
There is no impact on the DCD.

Impact on PRA
There is no impact on the PRA.

Impact on Technical Specifications
There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports
There is no impact on any Technical, Topical or Environmental Reports.

Response to Action Item 5-6 Section 5.4.2.2

MEB Issue List Regarding APR-1400, FSAR Section 5.4.2.2

Issue #11 (AI 5-6.29)

Steam generator ISI and PSI are required by 10 CFR 50.55a and are identified in SECY-05-0197 as Operational Programs to be implemented by licensees. However, since steam generators have distinct ISI and PSI requirements, it is appropriate to list the requirements and milestones separately in the COLA Table of Operational Programs under ISI and PSI.

Revise FSAR Subsections 5.4.2.2, 5.4.16, and 1.8 to add a Combined License item to show the implementation milestones for Steam Generator inservice inspection (ISI) and preservice inspection (PSI) in the table of Operational Programs (Chapter 13) for APR1400 Combined License Applications.

Response

A COL item for the SGMP to include steam generator tube PSI and ISI will be described in Sections 5.4.2.2.2, 5.4.16, 13.4.1 (Combined License Information), and Table 1.8-2.

Impact on DCD

DCD Sections 5.4.2.2.2, 5.4.16, 13.4.1, and Table 1.8-2 will be revised as indicated in Attachments 1, 2, 3, and 4.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

repairs, primary-to-secondary leak monitoring, maintenance of secondary-side integrity, secondary-side water chemistry, primary-side water chemistry, foreign material exclusion, contractor oversight, self-assessment, and reporting.

5.4.2.2.2.1 Degradation Assessment

A degradation assessment is performed prior to the preservice inspection (PSI) and planned ISI for SGs during commercial operation to address the RCPB integrity within the SG (e.g., plugs, ~~sleeves,~~ tubes, and components that support the pressure boundary such as secondary-side components).

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~~The assessment determines the size and location of existing and potential degradations that are likely to become harmful cracks.~~

The assessment considers operating experience and provides reasonable assurance that appropriate inspections are performed during the upcoming outage and that the requisite information for integrity assessment is provided.

Some of the important features of the degradation assessment are:

a. Identifying existing and potential degradation mechanisms

b. Choosing techniques to test for degradation based on the probability of detection and sizing capability

c. Establishing the number of tubes to be inspected

d. Establishing the tube integrity limits for condition monitoring and operational assessment

5.4.2.2.2.2 Inspection

SG tube inspections based on degradation assessments are conducted and follow the inspection guidance in the EPRI PWR Steam Generator Examination Guidelines (Reference 12).

The COL applicant is to prepare PSI and ISI program of the SG tubes.

COL 5.4(5) The COL applicant is to verify the as-built RV support material properties and 60-year neutron fluence.

COL 5.4(6) The COL applicant is to prepare PSI and ISI programs of the SG tubes.

5.4.17 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." U.S. Nuclear Regulatory Commission.

3. Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Rev. 1, U.S. Nuclear Regulatory Commission, August 1975.

4. APR1400-A-M-NR-14001-P, "KHNP APR1400 Flywheel Integrity Report," KHNP, November 2014.

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components, The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

6. ASME PTC 8.2, "Centrifugal Pumps," The American Society of Mechanical Engineers, 1990.

7. NEMA MG-1, "Motors and Generators," National Electrical Manufacturers Association, 2009 (with 2010 Revision 1).

8. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," Revision 0, U.S. Nuclear Regulatory Commission, August 1976.

9. ASME Section III, Appendix N, "Dynamic Analysis Methods," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

10. Bulletin 79-13, "Cracking in Feedwater System Piping," Rev. 1, U.S. Nuclear Regulatory Commission, August 30, 1979.

11. NEI 97-06, "Steam Generator Program Guidelines," Rev. 3, Nuclear Energy Institute, January 2011.

13.4 Operational Program Implementation

The COL applicant is to develop a list of operational programs, a description of the operational programs, and the associated implementation milestones (COL 13.4(1)).

13.4.1 Combined License Information

COL 13.4(1) The COL applicant is to develop operational programs and provide schedules for implementation of the programs, as defined in SECY-05-0197 (Reference 1). The COL applicant is to provide commitments for the implementation of operational programs that are required by regulation. In some instances, the programs may be implemented in phases, where practical, and the applicant is to include the phased implementation milestones.

COL 13.4(2) The COL applicant is responsible for developing a leakage monitoring and prevention program for the systems, as specified in Subsection 5.5.2 in Chapter 16 Technical Specifications. The leakage monitoring and prevention program is to provide suitable methods and acceptance criteria as defined in NUREG-0737 Item III.D.1.1 (Reference 2).

COL 13.4(3) The COL applicant is to develop an implementation plan of PSI and ISI programs of the steam generator tubes. The COL applicant is to provide a commitment for the implementation plan of PSI and ISI programs of the steam generator tubes.

13.4.2 Refere

1. Staff Requirements Memorandum to SECY-05-0197, “Staff Requirements-SECY-05-0197-Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,” U.S. Nuclear Regulatory Commission, February 2006.

2. NUREG-0737, “Clarification of TMI Action Plan Requirements,” U.S. Nuclear Regulatory Commission, November 2006.

Table 1.8-2 (8 of 29)

Item No.	Description
COL 5.2(8)	The COL applicant is to provide and develop the implementation milestones for the inservice inspection and testing program for the RCPB, in accordance with ASME Code Section XI and 10 CFR 50.55a.
COL 5.2(9)	The COL applicant is to address the provisions to accessibility of Class 1 components for ISI if the design of the APR1400 Class 1 component is changed from the DCD design.
COL 5.2(10)	The COL applicant is to provide the list of Code exemptions in the ISI program of the specific plants, if it exists.
COL 5.2(11)	The COL applicant is to prepare and provide any requests for relief from the ASME Code requirements that are impracticable as a result of limitations of component design, geometry, or materials of construction for the specific plants, if necessary. The request will contain the information on applicable Code requirements, alternative ISI method, and justification.
COL 5.2(12)	The COL applicant may invoke ASME Code Cases listed in NRC RG 1.147 for the ISI program.
COL 5.2(13)	The COL applicant is to prepare and implement a boric acid corrosion (BAC) prevention program compliant with Generic Letter 88-05.
COL 5.2(14)	The COL applicant is to prepare the preservice inspection and testing program.
COL 5.2(15)	The COL applicant is to address and develop milestones for preparation and implementation of the procedure for operator responses to prolonged low level leakage.
COL 5.3(1)	The COL applicant is to provide a reactor vessel material surveillance program for a specific plant.
COL 5.3(2)	The COL applicant is to develop P-T limit curves based on plant-specific data.
COL 5.3(3)	The COL applicant is to verify the RT_{PTS} value and the USE at EOL based on plant-specific material property and neutron fluences.
COL 5.3(4)	The COL applicant is to provide and develop the inservice inspection and testing program for the RCPB, in accordance with ASME Section XI and 10 CFR 50.55a.
COL 5.4(1)	The COL applicant is to prepare operational procedures and maintenance programs as related to leak detection and contamination control of RCS.
COL 5.4(2)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of RCS.
COL 5.4(3)	The COL applicant is to prepare operational procedures and maintenance programs as related to leak detection and contamination control of SCS.
COL 5.4(4)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of SCS.
COL 5.4(5)	The COL applicant is to verify the as-built RV support material properties and 60-year neutron fluence.
COL 5.4(6)	The COL applicant is to prepare PSI and ISI programs of the steam generator tubes.